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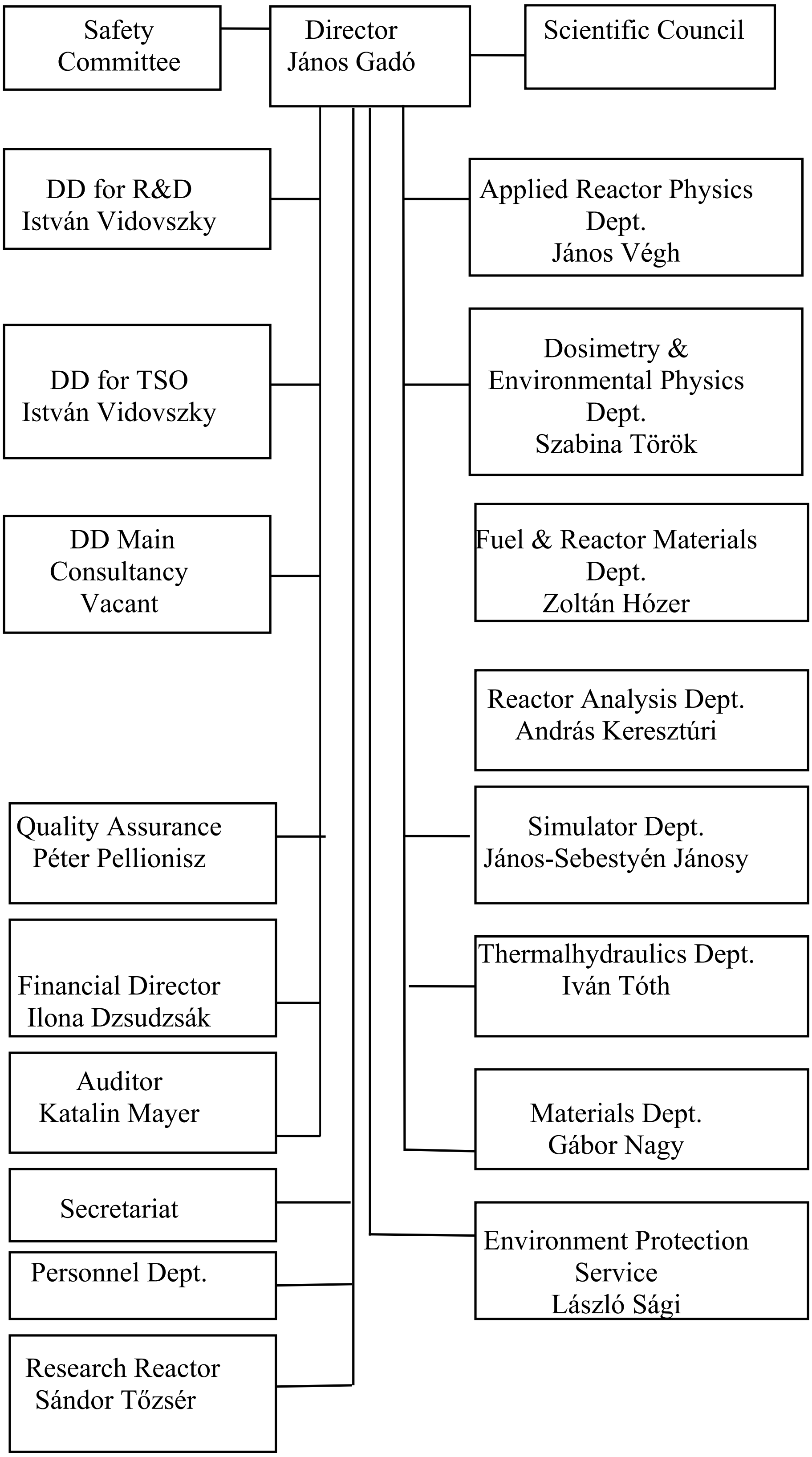
PROGRESS REPORT ON RESEARCH ACTIVITIES IN 2006

BUDAPEST, AUGUST 2007

CONTENTS

Organisation of the KFKI AEKI as of January 1, 2006



Dear Reader,

Welcome to the 2006 Progress Report of the Hungarian Academy of Sciences KFKI Atomic Energy Research Institute!

2006 was a remarkable year in the co-operation of the institute with Paks NPP. The removal of the consequences of the 2003 severe incident was successfully completed and the role of the institute in preparing this action was widely acknowledged. The primary power of Unit 4 was uprated from 1375 MW to 1485 MW, and the contribution of the institute in this result was very significant. The institute prepared a study for the Hungarian electricity company MVM on the main tasks needed for starting the actions aiming at new nuclear units in Hungary.

The institute continued the research works in connection with the development of the Supercritical Water Cooled Reactor, sponsored by the National Research and Technology Administration and also by the EC. Contributions to fusion programmes are also increasing. These research activities on innovative systems attract young scientists and engineers and thus they can significantly contribute to the maintaining of competence in the field of nuclear power.

The institute started the preparations for the conversion of the Budapest Research Reactor from HEU to LEU fuelling. This project that should be completed by 2010, includes the removal of spent fuel from the KFKI Campus to Russia.

The principal objective of our Progress Report is to present the results of the projects. Short thematic introductions summarize the main research directions, providing the reader with a background information.

I hope you will find some interesting results in the Progress Report.

János Gadó
Director

THEMATIC SECTIONS

Safety, regulation

The present section contains research, analysis and investigations to enhance safety or to support regulation. The reported activity includes concrete safety analys of a spent fuel pool, analysis of effects of power up rating and effects of a specific change in the technology. The reported research involves technical developmenet in DBA analisis, investigation of the water hammer phenomemon as well as analysis of the applied statistical methods and uncertainty estimations.
We supported the authority by a guide plan on aging management, furthermore, we made study on the construction of new nuclear power plant units in Hungary.
We studied the Paks cleaning tank incidents by model experiments.

Revision of FSAR for Paks NPP

GYÖRGY ÉZSÖL, TAMÁS FEKETE, JÁNOS GADÓ, ANDRÁS GÁCS, ANDRÁS KERESZTÚRI, LÁSZLÓ MARÓTI, IVÁN TÓTH, JÁNOS VÉGH

Objective

The Final Safety Analysis Report of Paks NPP was completed by 2004. In 2005 a renewed Hungarian Nuclear Safety Regulation was put into force. As a consequence, the FSAR had to be revised. The revision was a complex task since it included the wearly actualisation of FSAR and also the modifications required by the Hungarian Atomic Energy Authority in its resolution on the 2004 version of FSAR. The revision project was organised by the plant as a continuation of the previous works. Thus the various Chapters were prepared by the Hungarian nuclear community, including AEKI experts. AEKI experts played a significant role also in supervising the project, especially the modifications related to the new safety regulation.

Results

AEKI was responsible for or participated in the revision of the following Chapters:
3.5 Initial events
4. The fuel system
5.2 The reactor vessel
5.6 The pressurizer
6.2 The Emergency Core Cooling System
7.6.2 The VERONA system
15. Safety Analysis
16. Technical Specifications

The Supervisory Team with a strong AEKI participation was responsible for the management of FSAR modifications due to the new safety regulations. The fulfilment of the various new paragraphs in the safety regulation was evaluated.

Methods

The methodology is based on the US NRC codes and practices and also on the prescriptions and guides of the Hungarian nuclear regulation.

References

Paks NPP FSAR, version issued in 2006

Remaining work

The remarks of HAEA to the 2006 version of FSAR should be taken into consideration during the next update of the FSAR. After this, the yearly update of the FSAR will be elaborated solely by the Paks NPP.

Safety Analysis of the Paks NPP Spent Fuel Storage Pool

ZOLTÁN HÓZER, JÁNOS GADÓ , ANDRÁS KERESZTÚRI , GÁBOR HORDÓSY, LÁSZLÓ MARÓTI, GÁBOR NAGY , BARBARA SOMFAI, PÉTER VÉRTES, EMESE SZABÓ, VIKTOR KEREKES, IVÁN TÓTH, LÁSZLÓ PERNECZKY, SÁNDOR DEME, EDIT LÁNG , TAMÁS PÁZMÁNDI, LÁSZLÓ SÁGI,

Objective

The main objectives of the project are the development of new models for the simulation of spent fuel pool accidents and the evaluation of the consequences of potential spent fuel pool accidents.

Results

The main results of Paks NPP spent fuel storage pool design basis accident analysis can be summarised as follows:

- A proposal has been developed for the activity limits of spent fuel storage pool (SFSP) water (different technological parameters should be kept to guarantee these values).
- On the basis of accident analysis, the available time for actions was specified.
- It was pointed out that unit No. 2 with the containers of damaged fuel should be operated according to the same requirements as the other units.
- The environmental dose limits were not reached in the calculated cases.
- The drop of water level during accidents leads to high dose rates in the reactor hall.
- The temperature in the reactor hall can be kept at a normal value even in accidents.
- Emergency cooling of SFSP can be provided by the existing systems, but new procedures must be introduced.

Methods

In the frame of the safety analysis of design base accidents, new models have been required. Those models have been developed, and the following types of calculations have been carried out:

- Decay heat and isotope inventory calculations with the KARATE and ORIGEN codes.
- Thermal hydraulic calculations were carried out with the RELAP system code using a detailed nodalisation scheme and the selection of appropriate code options made possible the simulation of SFSP accidents.
- Activity release from fuel was calculated with the algorithm that is used for the simulation of the consequences of LOCA accidents.
- Activity release from containers of damaged fuel was calculated for unit No. 2, where the damaged fuel of the 2003 April incident are stored.
- Effect of direct gamma radiation was estimated for that part of the reactor hall that is located above the spent fuel storage pool.
- Temperature in the reactor hall was calculated for several scenarios and considering different operations of the ventilation system.
- Shielding and subcriticality calculations were carried out with the MCNP4C code.
- Dosimetry calculations in the reactor hall covered the consequences of activity release form the spent fuel storage pool.
- Activity release through the chimney was simulated considering different operation regimes of the ventilation system and the activity concentration in the air of the reactor hall.
- Environmental release calculations were based on the activity release through the chimney.

The calculations performed included many conservative assumptions that must be considered in the evaluation of results.

Reference

GADÓ JÁNOS, HÓZER ZOLTÁN: Final report on spent fuel pool safety analyses and emergency dose map, AEKI-FRL-2006-729-01/18 (in Hungarian)

Remaining work

The planned work has been completed.

Effect of power uprating on emergency operating procedures

ATTILA GUBA, LÁSZLÓ PERNECZKY, IVÁN TÓTH

Objective

A complete Design Basis Analysis was performed in 2002-2003 for the Paks NPP in view of uprating its power to 108%. Since power uprating also affects the Emergency Operating Procedures (EOP) of the plant, additional analyses were required to confirm some of the actions foreseen in the procedures.

Results

With the power increased, the effectiveness of boration had to be investigated for a non-isolated, circumferential main steam header break. The calculations were performed both with and without operation of the cold overpressure mitigation system. In the latter case the conditions for stopping the HPIS pumps are fulfilled in about 15 minutes. The overpressure mitigation system stops the HPIS injection already at appr. 8 min. after transient initiation and it could be shown that the boron injected in the meantime assures the required 2% subcriticality by a sufficient margin.

Analyses focusing on the cool-down process during natural circulation demonstrated that the cool-down rates defined for 100% power are valid also for 108% power: the cooling of the control rod drives and the HPIS injection to the upper head assure subcooled conditions. Also the necessary boron concentration prior to initiation of cool-down was determined. In case of natural circulation without let-down, sufficient margin to the prescribed 2% subcriticality could be demonstrated.

Methods

The RELAP5 code was applied for the analyses.

Reference

[1] A. GUBA, L. PERNECZKY, I. TÓTH: Upper plenum coolability during natural circulation cool-down at 108% power. AEKI-AOKU-2006-702/01/M1.

[2] A. GUBA, L. PERNECZKY, I. TÓTH: Natural circulation cool-down without let-down at 108% power. AEKI-AOKU-2006-702/02/M1.

[3] A. GUBA, L. PERNECZKY, I. TÓTH: Non-isolated, circumferential main steam header break at 108% power. AEKI-AOKU-2006-702/03/M1.

Remaining work

None.

Experimental simulation of the Paks-2 cleaning tank incident

ZOLTÁN HÓZER, PÉTER WINDBERG, IMRE NAGY, ANDRÁS VIMI, NÓRA VÉR, LAJOS MATUS, MIHÁLY KUNSTÁR, TAMÁS NOVOTNY, ERZSÉBET PEREZ-FERÓ, MÁRTON BALASKÓ

Objective

In order to improve the understanding of the phenomena that took place during the Paks-2 incident, series of separate effect tests and integral tests have been carried out. The experiments in the CODEX (Core Degradation Experiment) facility simulated the whole course of the incident using electrically heated fuel rods.

Results

The main conditions of the incident were reproduced in the CODEX-CT experiments. After the tests the fuel bundles showed a view similar to the samples observed at the Paks NPP after the incident. The upper part of the rods was almost fully oxidized, while the bottom part – cooled by water during the experiment – remained intact. The middle and upper sections were broken into several pieces. The material was very brittle, further fragmentation took place during the handling of fuel rods. The post-test examination indicated very high hydrogen content (several thousands ppm) in the Zr components (cladding and shroud).

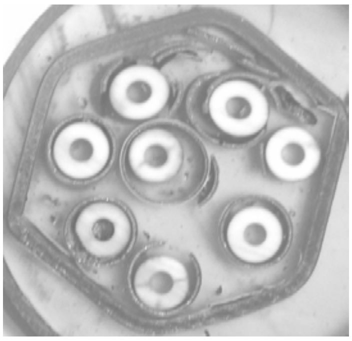


Fig. 1. Cross section of the CODEX-CT-1 bundle

The simulation of the cleaning tank incident provided detailed information on the probable scenario of the real incident. The data will be used for model development purposes to improve the predictive capabilities of fuel behaviour codes.

Methods

In the CODEX-CT experiments the rods were filled with alumina pellets and pressurised inside. The bundles were covered by hexagonal shrouds made of Zr2.5%Nb alloy. Inlet and outlet junctions were connected to the bottom section and perforation was used to facilitate the formation of by-pass flow. The external heater rods surrounded the bundle. The spent fuel storage pool was simulated by an expansion tank that was connected to the test section. Special condenser unit was applied to receive coolant from the test section during the quenching phase.

Reference

HÓZER Z., WINDBERG P., NAGY I. MATUS L.,VÉR N., KUNSTÁR M., VIMI A., HORVÁTH L., PINTÉRNÉ CSORDÁS A., NOVOTNY T., PEREZNÉ FERÓ E., BALASKÓ M.: The CODEX-CT-1 Experiment: Quenching of Fuel Bundle After Long Term Oxidation in Hydrogen Rich Steam, AEKI-FRL-2006-213-01/02 (in Hungarian)

Remaining work

Post-test examination of the CODEX-CT-1 and CT-2 bundles will be carried out in 2007. The data base of the experiments (facility description, on-line measured data and post-examinaiton results) will be established.

OECD-IAEA Paks Fuel Project

ZOLTÁN HÓZER, EMESE SZABÓ

Objective

The OECD-IAEA Paks Fuel Project was launched in order to investigate fuel behaviour in accident conditions on the basis of the Paks-2 event. The first phase of the project (2006-2007) focuses on the numerical analyses of the incident.

Results

AEKI as the Operating Agent of the project has completed the following tasks:

- Database of the Paks-2 event has been completed and the first version has been updated considering the comments and requests of participating organizations.
- Special password protected website has been established for the project.
- The requirements for output parameters have been specified and circulated among the participants.
- Proposal has been developed for the selection of fuel assemblies for hot cell examination (that can be carried out in the second phase of the project).

Methods

Using the database code calculations have been carried out by the project participants in the following areas:

- Integral calculations covering the whole scenario of the incidents using severe accident codes
- Thermal-hydraulic calculations describing the cooling conditions which possibly existed during the incident
- Simulation of fuel behaviour describing the oxidation and degradation mechanisms of fuel assemblies, by means of transient fuel codes
- The release of fission products from the failed fuel rods.

Reference

Z. HÓZER: Proposal on selection of fuel assembly for examination, AEKI-FRL-2006-408-01/01, AEKI January 2006.

E. SZABÓ, Z. HÓZER, CS. GYŐRI, GY. HEGYI: Database for the OECD-IAEA Paks Fuel Project, Preliminary version, AEKI-FRL-2006-408-01/02, AEKI January 2006.

E. SZABÓ, Z. HÓZER, CS. GYŐRI, GY. HEGYI: Database for the OECD-IAEA Paks Fuel Project, Version 1.0, AEKI-FRL-2006-408-01/02-M1, AEKI, April 2006.

Z. HÓZER: Requirements for output parameters, AEKI-FRL-2006-408-01/03, AEKI, April 2006.

E. SZABÓ, Z. HÓZER, I. NAGY: Description of the website for the OECD-IAEA Paks Fuel Project, AEKI-FRL-2006-408-01/04, AEKI, July 2006.

Remaining work

The project will be completed in 2007. AEKI will prepare the Final Report describing the results of numerical analyses of the Paks-2 incident.

Experimental verification of the operation of the TH50-80S201 locking bolt on the hydroaccumulators in the Paks NPP

GYÖRGY ÉZSÖL, GÁBOR BARANYAI

Objective

When emptying the hydroaccumulators, N2 gas can get into the primary circuit and influence the heat transfer disadvantageously. To avoid the admission of gas to the primary system, a PHÖNIX type level measurement has been applied to unit 4 of the Paks NPP. The question is whether the time delay of the level measurement of the PHÖNIX device remains below the prescribed time and/or level limit or not.

Results

Using the PMK-2 infrastructure, we have performed an experimental program on a hydroaccumulator model with the PHÖNIX device. with the same The experiments have been carried out with the same technical conditions as in the plant. The level is measured by a magnetic floating body in the PHÖNIX device, giving an electric signal for a reed-relay. The acceptable level of the time delay is considered to be 1 s. Experiments have been performed modelling the level decrease in the hydroaccumulator model in cases of large, medium and small breaks, altogether in 9 cases. In case of design basis accident (200% break size) the time delay is 5.5 s, what corresponds to 0.3 m level decrease in the hydroaccumulator. The conclusion of the test series is that the time delay in case of a large break is too large. The □P measurement follows the water level with a time delay less than 1.0 s.

Methods

A comprehensive experimental program has been carried out.

Reference

GY. ÉZSÖL, G. BARANYAI: Experimental verification of the TH50-80S201 Locking Bolt on the Hydroaccumulators in the Paks NPP, in Hungarian, AEKI-THL-2006-734/01/M1

Remaining work

The project has been completed.

Unification of the Modes of Operation of the Pressure Reduction Valves (AR) in the Secondary Circuit of the 1-4 Units of the Paks NPP

GYÖRGY ÉZSÖL, ATTILA GUBA

Objective

On the four Paks NPP units, operational modes of the pressure reduction valves (AR) in the secondary circuit were different. The plant personnel initiated the unification of the operational modes. The effects of the unification on the safety of the plant were evaluated by the calculations performed for turbine trip, loss of feedwater and feedwater line break.

Results

The operation of AR valves in phase I. of the development was as follows: the valve opened at 53 bar and closed at 45.6 bar without time delay, without any control action. In phase II. of the development, the valve operation was as follows: when the pressure exceeded the value of 52.6 bar the valve operation was controlled by the control unit in a prescribed way. The difference in the operation lead to different pressure history in the secondary circuit and it could affect the cooldown of the plant in a different way.

The evaluation showed that the unification of the operational mode of the AR valves does not affect the safe cooldown of the plant. Only the pressure history in the secondary side is slightly affected in case of a turbine trip. Consequently, there is no safety risk of the unification.

Methods

Calculations were performed by the RELAP5 and the results were evaluated. In the evaluation, the results of the Final Safety Report (VBJ) were also utilized.

Reference

GY. ÉZSÖL, A. GUBA: Unification of the Modes of Operation of the Pressure Reduction Valves (AR) in the Secondary Circuit of the1-4 Units of the Paks NPP, in Hungarian, AEKI-THL-2006-749/01/M1

Remaining work

The project was completed.

Statistical aspects of safety analysis

MIHÁLY MAKAI

Objective

There are two dominant views concerning the question: how many runs of a deterministic code with random input are needed to estimate the maxima of the outputs. Although the question is a purely technical one, there may be different interests in the background to advocate cheaper analysis. Notwithstanding that problem is tractable in a rigorous way, there are people applying obscure engineering approaches to corroborate a lower number of runs.

Results

In order to find a way out from that situation, we elaborated a simple benchmark [1] to test a statistical procedure. The benchmark assumes that we distinguish 2500 states of the reactor to be analysed, and we know the outputs associated with each input set. The user has to apply his method, carry out a given number of runs by performing selection of outputs in accordance with the statistical method and see the performance of the method. The author's opinion was expressed in connection with Graham B. Wallis' method in Ref. [2].

Methods

In oTheur works, a series of calculations is carried out by using a production code. By statistical inference we deduce the impact of input uncertainty to output variables of the calculation model.

Remaining work

Since no progress can be expected from pursuing the topics, the work will be shifted to other areas of safety analysis. It would be instructive to elaborate a safety measure for a power plant, and to work out a methodology for its measurement.

Reference

[1] M. MAKAI, L. PÁL: Amendment to safety analysis of nuclear reactors, Proc. of Conf. on Safety and Reliability for Managing Risk, Lisbon, September 2006, Taylor and Francis, London, 2006

[2] M. MAKAI: Comment on "Evaluating the probability that the output of a computer code with random inputs will meet a set of evaluation criteria", Reliability Engineering and System Safety, to appear

Regulatory body's guides on ageing management and regulatory review for research reactors

SÁNDOR TŐZSÉR, KÁROLY KÉSMÁRKY AND ISTVÁN VIDOVSZKY

Objective

In 2005, new Nuclear Safety Regulations (NSRs) were issued by the Hungarian Government [1], which include regulations Research and Training Reactors. The Regulatory Body launched a framework program to develop a complex series of safety guides. The primary objective of issuing these safety guides is to provide practical guidance to fulfil the requirements prescribed by the NSRs and to ensure a methodology for both inspectors and operators.

Results

To support the Regulatory Body's initiative, in 2006 two sets of studies were written by the Department of Budapest Research Reactor (BRR) within the framework of this program. They were as follows:

- Regulatory Body's Surveillance of Ageing Management of Research and Training Reactors;
- Nuclear Safety Review of Research and Training Reactors.

Both studies formed the basis of the Regulatory Body's guides that have already been issued on these two subjects.

Methods

The method applied throughout the studies was: to clarify the requirements (according to the study subjects); adopt a broad range of international recommendations and practices gained during the nearly 50-year operation of BRR; and outline a form of practical guidance proposed to be followed by the operation organization of Research and Training Reactors. Although the authority activities are emphasized in the study's titles, the studies – in accordance with the objective – handle the roles and clarify the duties and responsibilities of all parties, be they organizations, regulatory bodies, or individuals (regardless of their position relative to the management hierarchy). Thus, the structures of the studies (according to the concerned study field) are:

- Object oriented clarification and summary of the requirements prescribed by NSRs for Research and Training Reactors (the safety objectives concerning the two study subjects, as well as the tasks that have to be accomplished to meet these objectives);
- Overview of the concerned IAEA documentations (Safety Standards Series, TECDOC Series, as well as drafts currently in production);
- Summary of the BRR's experiences based on the Report on Periodic Safety Review [2] and the mandatory documentation and data logging/event recording practices of the BRR;
- Proposed procedures and/or activities undertaken to perform the tasks and obligations (e.g. in the case of ageing management one of the study's supplements contains a complete list containing the systems, subsystems and components involved in the ageing management program along with monitoring phenomena and boundary conditions);

Remaining work

The work has been completed.

Reference

[1] Nuclear Safety Regulation, Vol. No. 5. Budapest 2005.
[2] Report on Periodic Safety Review of BRR. Budapest 2003.

Preliminary studies to prepare for the construction of new nuclear power plant units in Hungary

JÓZSEF BAJSZ[1], JÁNOS GADÓ, ÁKOS HORVÁTH, ZOLTÁN HÓZER, JÁNOSY J. SEBESTYÉN, JÁNOS VÉGH, ISTVÁN VIDOVSZKY

1 – Paks Nuclear Power Plant Ltd, H-7031 Paks, P.O.Box 71

Objective

In 2006, the Hungarian Electric Power Company (MVM Rt.) contracted with MTA KFKI AEKI to prepare a comprehensive study reviewing the most important foreseeable tasks in connection with the construction of new nuclear power plant units in Hungary. The motivation for the study was that recent analyses of Hungarian energy policy and future electric power production / consumption balance have shown that between 2020 and 2030 new nuclear power plant units must be built in order to ensure safe, cheap, maintainable and CO_2-free electricity production in our country.

The basic objective of the study was to enumerate and analyze the tasks during the preparative phase. The most important items treated in the report were as follows:

- analysis of commercially available G3 and G3+ reactor types to be potentially built in Hungary (e.g. EPR, AP1000, VVER-1000, Mitsubishi APWR),
- overview of development and construction perspectives of various G4 reactor types,
- analysis of options related to nuclear fuel and long-term handling of radioactive waste,
- analysis of the Paks site with respect to the construction of new NPP units,
- applicability of the European Utility Requirements (EUR) document for tender specification,
- proposals (1) for the tasks to be completed in the preparative phase, (2) for the measures to ensure domestic nuclear competence and (3) for the labour division between Hungarian nuclear institutes.

The project was only a first step on the long road leading to the construction of new nuclear power plant units in Hungary. According to recent international practice, it takes about 3-4 years to start the licensing procedure after the announcement of the intention to build a new reactor unit and the construction itself takes another 5-6 years. Therefore it is high time to start preparations if a new NPP unit is to start producing electricity around 2020.

Results

The main conclusions and proposals of the study can be summarized as follows:

- Technically and commercially the only viable solution is to select a G3 or G3+ reactor for construction.
- Reactors of G4 type are not yet mature for construction; it is obvious that a commercial version of any G4 reactor design will not be built before 2020.
- In order to ensure public acceptance of new NPP units Hungary must have a clear and workable concept for the long-term handling of spent fuel and other high-level radioactive waste.
- The Paks site has a number of advantages if the construction of new NPP units is considered, but it is quite likely that the constraints on the Danube water temperature increase can be satisfied only by using cooling towers for the new units.
- Several changes are required in the laws controlling the licensing procedure of the new NPP units, the Nuclear Law, the Electric Power Law and the Environmental Protection Law must be harmonized.
- One of the most important first steps in the preparatory phase is the completion of a Preliminary Environmental Impact Assessment study.

References

[1] National tasks in connection with the establishment of new nuclear power plant units (report prepared for MVM Rt.), AEKI-ARL-2006-766-00/01, MTA KFKI AEKI, December 2006 (in Hungarian)

Remaining work

Preparations for the continuation of the project are in progress.

Reactor physics

Safe and economic operation of nuclear installations require reactor physical analyses even today. Today, however, not basic laws are investigated but testing of existing algorithms, estimation of uncertainties. Algorithms to be applied in safety analysis are validated.
The new reactor generation needs a large amount of research, AEKI, as participant of the NUKENERG project, is interested in supercritical water reactors. Rsearch on reactor physical calculational methods and material candidates are reported in the present section.

Reactor physics activity in the NURESIM project

ANDRÁS KERESZTÚRI, GYÖRGY HEGYI

Objective

Concerning reactor physics, AEKI is involved in SP1 (core physics) and more specifically in the following tasks:

- Specification of data and reference results for VVER benchmarks.
- Verification and validation of the NURESIM core physics methods, solvers and data.

Results

In the last period of the project, KFKI AEKI has elaborated the detailed geometry and material data of a set of asymptotic ZR-6 measurements according to the unified format of specifications and performed sample calculations by the KARATE code system in order to justify the applicability of the data for benchmarking [1].

More detailed specifications of the ZR-6 measurements, focusing on the temperature reactivity coefficients and the heterogeneities caused by burnable poison pins, are elaborated [2,3].

The APOLLO2 code has been implemented in KFKI AEKI. The know-how for running the code was learned by taking part in training courses and by personal communications. There are still problems, slowing down the progress, in case of certain tasks due to the lack of effective communication between the code developers and users.

The specified new set of the above mentioned asymptotic benchmark problems (see Ref. [1]) has been solved by the APOLLO2 code and compared to the experimental results [4]. The agreement is good.

The input decks for the tasks concerning the reactivity coefficients and burnable poison (see [2] and [3]) has been elaborated, and the calculations have been started.

Methods

Model development and validation.

Remaining work

Concerning the reactivity coefficients and burnable poison, the calculations are to be finished.

References

[1] GY. HEGYI, A. KERESZTÚRI: Specifications of the ASYMPT.ZR-6_MEAS Benchmark of NURESIM-SP1, KFKI AEKI Hungary, 2006, uploaded to the NURESIM web page.

[2] GY. HEGYI, A. KERESZTÚRI: Specifications of the DRDT.ZR-6_MEAS Benchmark of NURESIM-SP1, KFKI AEKI Hungary, 2006, uploaded to the NURESIM web page.

[3] GY. HEGYI, A. KERESZTÚRI: Specifications of the GDABS.ZR-6_MEAS Benchmark of NURESIM-SP1, KFKI AEKI Hungary, 2006, uploaded to the NURESIM web page.

[4] GY. HEGYI, A. KERESZTÚRI: Validation of the APOLLO2.7 code by Asymptotic ZR-6 Measurements (Solution of the ASYMPT.ZR-6-MEAS Benchmark of NURESIM-SP1), KFKI AEKI Hungary, 2006, uploaded to the NURESIM web page.

Preparatory activity for the safety analysis of the Budapest Research Reactor

ANDRÁS KERESZTÚRI, JÓZSEF VIGASSY, CSABA MARÁCZY, GYÖRGY HEGYI, EMESE TEMESVÁRI, ISTVÁN TROSZTEL, ISTVÁN BENKOVICS

Objective

The renewal of the safety analyses of the Budapest Research Reactor FSAR for the present fuel has been initiated by the Hungarian Atomic Energy Authority. Additional analyses are necessary for licensing the new low enriched fuel.

Results

The activity in 2006 was devoted to the preparatory works with the following results:

1. The validation report of the KIKO3D 3D dynamic code for the given application, based on new and previous calculations, was prepared [1].
2. Local power peaking factors, which are important in the environment of the water traps, were determined by full core Monte Carlo calculations [2] for both the present and the low enriched fuel.
3. The handbooks, containing the reactor specific but transient independent input data for the KIKO3D, ATHLET, coupled KIKO3D-ATHLET, LOCASYM codes, have been prepared [3-6]. The group constants of KIKO3D were generated by the KARATE code system.
4. The appropriate equilibrium reloading scheme and core containing only low enriched fuel was determined.
5. The reactor physics frame parameters (reactivity coefficients, control rod worth, power peaking factors) assuring the conservative core state for both the present and the low enriched core were determined [7].
6. The validation of the ATHLET code for the given application was performed by using energetic startup measurements [8].
7. A thermohidraulic benchmark problem has been defined, solved and compared to the ANL results.

Methods

Model development and validation.

Remaining work

Safety analyses

References

[1] GY. HEGYI, ET AL: The validation of the KIKO3D code for the core design calculations of the Budapest Research Reactor, 2006, in Hungarian.

[2] G. HORDÓSY: Local power peaking factors inside the fuel assemblies of the Budapest research reactor, KFKI-AEKI report, 2006, in Hungarian.

[3] A. KERESZTÚRI, ET AL.: The KIKO3D code handbook for the core design and kinetic calculations of the Budapest Research Reactor, 2006, in Hungarian.

[4] I. TROSZTEL: ATHLET mod 2.0 – cycle A input handbook for the safety analyses of the Budapest Research Reactor, 2006, in Hungarian.

[5] A. KERESZTÚRI, ET AL.: The ATHLET-KIKO3D code handbook for the core design and kinetic calculations of the Budapest Research Reactor, 2006, in Hungarian.

[6] J. VIGASSY: The LOCASYM code handbook for the calculations of the Budapest Research Reactor, 2006, in Hungarian.

[7] A. KERESZTÚRI, ET AL.: Reactor physics frame parameters for the safety analysis of the Budapest Research Reactor, 2006, in Hungarian.

[8] I. TROSZTEL: Validation of ATHLET mod 2.0 – cycle for the safety analyses of the Budapest Research Reactor, 2006, in Hungarian.

Uncertainty analysis of hot channel calculations of the safety analyses

ISTVÁN PANKA, ANDRÁS KERESZTÚRI

Objective

The fulfillment of the safety analysis acceptance criteria is usually evaluated by separate hot channel calculations using the results of neutronic or/and thermo-hydraulic system calculations. Traditionally, the evaluation is based on several conservative assumptions taking into account the uncertainty of input parameters and the large number of fuel rods in the core. In order to get more realistic results, an uncertainty analysis methodology has been elaborated combining the response surface method with the one-sided tolerance limits method of Wilks. The elaborated modules were applied to an ATWS event (inadvertent withdrawal of control assemblies) in a VVER-440 reactor core. The calculations were performed by the TRABCO single hot channel code using the radial averaged group constants generated by KARATE. The validity of the single channel approximation against multi-channel calculations is discussed in [1].

Results

In the selected ATWS transient, according to the conservative analysis, a number of fuel rods are experiencing DNB for a longer time and must be regarded as failed. Their number must be determined for a further evaluation of the radiological consequences.

For this reason, the results of conservative and uncertainty hot channel calculations were compared with regard to the number of failed fuel rods in the reactor core and some other relevant output parameters (max. clad surface temperature and the maximum fuel temperature). The global calculations were performed with the SMATRA code in a conservative manner. In the hot channel calculations all fuel rods were modeled by using the response surface method and the uncertainty of the following input parameters were considered: pin power distribution, mass flow, critical heat flux, inlet temperature of the coolant, pin average burnup, initial gap size, selection of power history influencing the gap conductance value. The statistical evaluation was based on the one-sided tolerance limits of Wilks in 95 % / 95 % sense.

In case of conservative analysis we investigated more approaches for the uncertainty of the pin power distribution (best-estimate pin powers, Monte-Carlo analysis concerning the pin powers, increased pin powers with 2 or 3 sigma).

It was concluded that the estimated number of failed fuel rods depends to great extent on the applied uncertainty approach of the hot channel calculations. Simple use of the best estimate power distribution is slightly non-conservative concerning the failed fuel rods. Whereas, full hot channel uncertainty analysis leads to the minimum number of failed fuel rods in 0.95/0.95 sense. Concerning the temperatures, in case of uncertainty analysis we get higher values (the acceptance criteria were fulfilled) than in the conservative analysis, due the detailed model and some best-estimate parameters applied in the conservative analysis.

Methods

Safety analysis, application of the statistical methods

Remaining work

The planned work has been completed. Further on the full uncertainty analyses should be complemented with uncertainties coming from the global calculation.

Reference

[1] I. PANKA, M. TELBISZ: Sensitivity analysis of hot channel calculation methods, Progress in Nuclear Energy **49** (2007), 27-36

[3] I. PANKA, A. KERESZTÚRI: Uncertainty analysis for hot channel, Proceedings of the sixteenth Symposium of AER, Bratislava, Slovakia, 2006

Validation of the reactivity coefficient calculations through the ZR-6 experiments

GYÖRGY HEGYI

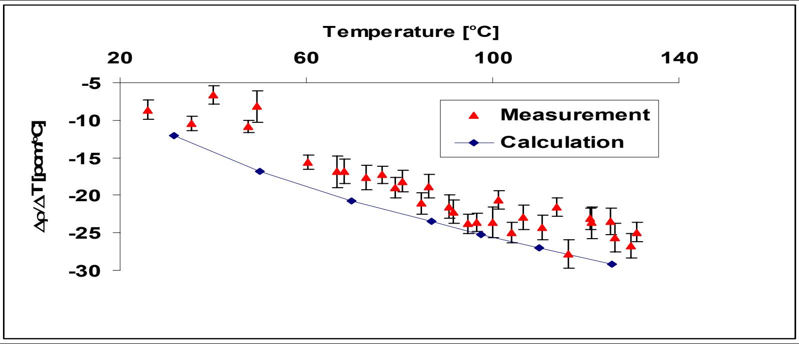
Objective

The reactivity coefficients are very important in design, control and safety, particularly in the light water reactors. For safety considerations, the temperature reactivity coefficient is desired to be negative at normal conditions, throughout the core life. For an accurate prediction of this parameter it is very important to validate any newly processed library.

In a previous work a series of experiments available on ZR-6 had been analyzed. Our calculations, using modules of the KARATE program package with our own-developed few group library based on ENDF/B-VI evaluation, have shown that the error of our calculations are similar to the target accuracy known from the literature. In the present work additional ZR-6 experiments, published in the new IRPhE handbook revised by Z. Szatmáry have been investigated.

Results

The results of the experiments are the partial derivatives of the reactivity at H=Hcr (T). 2D calculations at fixed axial buckling are required, because there are no suitable measurements for the temperature dependence of radial reflector saving. The simulation of the experiments was performed by the COREMICRO code using the few group constant data evaluated by the MULTICELL code and the measured axial bucklings. In the COREMICRO code the group constants are prescribed hexagon by hexagon. In case of the simulation of the experiments the grid corrected water levels have to be used. The figure below shows a typical simulation of the measured $\partial\rho/\partial T$ as a function of temperature. Generally, good agreements with the measurements were observed, taking into account the uncertainty of the input data. The summary of the work will be presented on the ICNC 2007 conference [1-2]



Comparison of the measured and calculated $\partial\rho/\partial T$ as a function of temperature

Methods

Deterministic neutron transport calculations

Remaining work

The finalization of the calculations is completed. Publication is under preparation and some more are planned for 2007. Experiences obtained within the project are used in the NURESIM EU project.

References

[1] GY. HEGYI, G. HORDÓSY, A. KERESZTÚRI, CS. MARÁCZY: „Results of Experiments Carried out on VVER Type Lattices Containing Gd Absorber Rods and Their Application for Code Validation”, Proc. of The 8th International Conference on Nuclear Criticality, St. Petersburg, Russia, May 28-June 1, 2007

[2] GY. HEGYI: “Analysis of Experiments on the Reactivity Temperature Coefficient for ZR-6 Lattices Using the KARATE Code System“, Proc. of The 8th International Conference on Nuclear

Criticality, St. Petersburg, Russia, May 28-June 1, 2007

Statistical evaluation of the on line core monitoring effectiveness for limiting the consequences of the fuel assembly misloading event

ANDRÁS KERESZTÚRI, ATTILA MOLNÁR, LAJOS KORPÁS, EMESE TEMESVÁRI

Objective

In VVER type reactors, on line core monitoring is used for early detection of such abnormal events as fuel assembly misloading, inadvertent misalignment of Control Assemblies, blockage of coolant channels. The questions to be answered are as follows.

- How to detect the abnormal event of misloading, what combination of the online measured (and calculated) quantities can be applied as „frame parameters" for this purpose.
- How to evaluate the uncertainties of the measured data, determining the above "frame parameter" uncertainties.
- How to built the above uncertainties into the determination of the appropriate operational level of the indication of the abnormal event - in order to indicate it at a high confidence level but to avoid the erroneous indication in normal operation at high probability, too.
- What are those satisfactory configurations of the measurements (numbers and positions of the detectors) at which the above, to some extent contradictory requirements are met, or with other words, the given safety related function of the online monitoring is fulfilled.

Results

The investigations proved the satisfactory effectiveness of the online core monitoring down to 55 % power even in case when only the 75 % of the temperature measurements is available if the indication is based on the measured asymmetry factor.

Methods

To answer the above questions, a Monte Carlo method was developed, namely the deviations of measurements from the "true" values are sampled. The true values are simulated by KARATE realistic reactor physics calculations both for the normal and abnormal states. The sampling parameters have been determined by matching the statistical behavior of the real measurements. The other source of the uncertainties, also modeled by the Monte Carlo method, was the availability of the individual detectors, because in spite of the very great variability of the detector failures, only minimum number of the available measurements and very general rules can be prescribed in the Technical Specification. These possible prescriptions have been taken into account in the sampling procedure as constraints.

Remaining work

There is no remaining work.

Reference

A. MOLNÁR, L. KORPÁS, E .TEMESVÁRI, GY HEGYI: Investigation of the VERONA system functions related to the core nuclear characteristics, KFKI report, AEKI-RAL-2006/705/03/M0.

Safety analyses of the C-30 fuel cask at upgraded burnup

ANDRÁS KERESZTÚRI, GÁBOR HORDÓSY, PÉTER VÉRTES

Objective

The new analyses aim at the fulfillment of the safety requirements at increased maximum assembly burnup of 56 GWd/tU and also at increased lattice pitch. The new maximum burnup corresponds to the expected value of the foreseen introduction of 4.2 % enriched fuel. The higher burnup and the increased lattice pitch concerns potentially only the biological shielding and the hydrogen production. (Subcriticality analyses at the increased lattice pitch have been performed already in an earlier study.) Because of the technological circumstances, the shielding analyses had to be carried out both for wet and dry conditions.

Results

Although the maximum allowed heat production was unchanged, the enveloping conservative value of the gamma dose rate increased by 15 %. The reason was the steeper azimuthal distribution of the gamma source. The neutron dose rate was increased significantly, 1.5 times, due to the higher average burnup, nevertheless it remained below the limits. The hydrogen production depends on the gamma dose absorbed in the water, but its increase due to the increased lattice pitch is not significant if the heat production limitation is kept. Neither the hydrogen nor the oxygen production can reach the critical value necessary for the explosion.

Methods

The gamma and neutron sources were calculated by the ORIGEN code assuming constant power time dependence conservatively. The used VVER-440 specific one group cross sections were evaluated by using the KARATE code system. The gamma source originated from the fission production had to be supplemented by the contribution of the activated ^{59}Co isotopes. This latter part was calculated also by the KARATE program. The gamma and neutron shielding were calculated by the MARMER and MCNP codes.

Remaining work

There is no remaining work.

Reference

G. HORDÓSY, A. KERESZTÚRI, P. VÉRTES: Biological shielding and hydrogen production analyses of the C30 cask at increased burnup, , KFKI report AEKI-G-1097, June 2006.

Fuel and materials

The safe and economic operation of nuclear installations requires sound knowledge of material properties. The clad chemical reactions with water, its hydrogen uptake and oxidation, the interaction of rhuthrnium with other metals are one line of the research. Corrosion issues, the mechanical effects of irradiations are the other line. These problems are investigated in a wide international co-operation.

The aim of the research is to work out a reliable fuel behaviour code. We participate in the development of the FUROM code. We report also on the application of a novel investigation method, the neutron tomography.

Evaluation of cladding ductility of E110 on the base of one side steam oxidation

ERZSÉBET PEREZ-FERÓ, TAMÁS NOVOTNY

Objective

Investigation of ductile-brittle transition of the Zr1%Nb cladding pre-treated with one side steam oxidation at temperatures from 900 to 1100°C was the primary objective of the experiments.

Results

Altogether 11 oxidation tests were performed. Only the outer surface of the samples was oxidized, since both ends of the tubes were plugged. The specimens were used for mechanical tests. According to our earlier studies with oxidized E110 rings, 50 mJ/mm specific energy was found as a boundary of ductility.

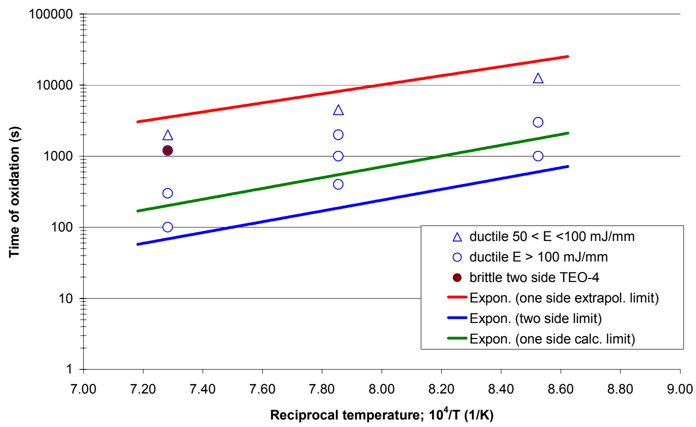


Figure 1. Ductile-brittle transition of E110 cladding

Representing the time of oxidation as a function of the temperature and distinguishing the specimens on the basis of the specific energy of ring compression, a ductility limit (τ- oxidation time till cladding embrittlement) could be estimated for one side oxidation:

$$\tau_{E110} = 0{,}0803 \exp(14678/T)$$

The experimental data demonstrated that the cladding embrittlement occurs later in case of one side oxidation than in case of two side oxidation.

Methods

A high temperature tube furnace was used for the oxidation of the samples. The ductile-brittle transition of Zr1%Nb cladding was assessed through the analysis of ring compression tests' data. The ring compression tests were performed at room temperature with pre-oxidized specimens. The specific energy at failure was used for the evaluation of the cladding ductility.

Reference

E. PEREZ-FERÓ, T. NOVOTNY: Evaluation of cladding ductility of E110 on the base of one side steam oxidation, Report AEKI-FRL-2006-272-01/03, in Hungarian

Remaining work

In the future, the very long time oxidation of few samples will be carried out in order to get brittle samples for ring compression test.

Oxidation of E110 cladding in hydrogen rich steam atmosphere

ERZSÉBET PEREZ-FERÓ, TAMÁS NOVOTNY

Objective

The purpose of the experiments was to investigate the high temperature oxidation of Zr1%Nb in hydrogen rich steam atmosphere with different hydrogen contents at an extended interval; furthermore to map the role of steam's hydrogen content on the oxidation kinetics and the effects of oxidation and hydrogen uptake on the mechanical properties.

Results

Altogether 25 oxidation tests were performed at temperatures from 900 to 1100 ^{o}C. At each temperature several samples were oxidized during different times in order to achieve different extents of oxidation. The tests were carried out at low (5 vol.%) and at relatively high hydrogen content (65 vol.%) in the steam. The oxidation mass gain rates measured in high vol. % H-steam mixture have no significant difference as compared to the results in low vol. % H-steam mixture. Comparing the oxidation rate constants measured in pure steam and in hydrogen-rich steam atmosphere, the hydrogen content in the steam is likely to decelerate the cladding oxidation.

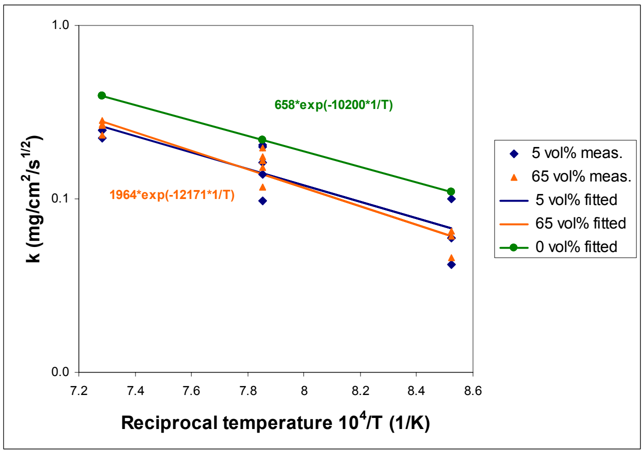


Figure 1. Oxidation rate constants for E110 as a function of reciprocal temperature in steam-hydrogen mixture

Methods

A high temperature tube furnace was used for the oxidation. The experimental set-up consisted of a steam generator, a three-zone furnace and a condensing system. The oxidation was performed in steam + hydrogen mixture. The outlet hydrogen rate, the condensed water and the mass gain of the cladding rings were measured. The samples were characterized by their oxygen content.

Reference

PEREZNÉ FERÓ ERZSÉBET, NOVOTNY TAMÁS: Oxidation of E110 cladding in hydrogen rich steam atmosphere, Report AEKI-FRL-2006-258-01/01, in Hungarian

PEREZNÉ FERÓ ERZSÉBET, NOVOTNY TAMÁS: Photo catalogue of E110 cladding oxidized in steam atmosphere, Report AEKI-FRL-2006-123-01/01, in Hungarian

Remaining work

In the future the mechanical testing (ring compression tests) of the oxidized cladding rings and the measurement of the amount of absorbed hydrogen will be carried out. The metallographic analysis of the oxidized samples is planned.

Hydrogenisation of Zr specimens at the temperature of 600 °C

TAMÁS NOVOTNY, ERZSÉBET PEREZ-FERÓ

Objective

Hidrogenization of two types of Zr cladding (Zr1%Nb and Zircaloy-4S) samples received from UJP PRAHA was the main objective of this work. Furthermore to study the hydrogen solubility of these Zr alloys at temperature of 600°C.

Results

In accordance with the specification of UJP PRAHA, hydrogenization of 10 samples was performed at low temperature. The target and the achieved hydrogen concentrations are summarized in the table below. The hydrogen content of the samples was calculated on the basis of the mass gain.

Target (ppm H_2)	Zircaloy-4S (ppm H_2)	E-110 (ppm H_2)
600	588	636
1000	1047	961
1500	1562	1433
2000-2500	2111	1972

The hydrogenization at 600 ^{o}C took much longer time (several days) than in our earlier tests performed at and above 900 ^{o}C. The first metallographic investigations showed that the metallic part of Zr cladding hydrogenized at low temperature had much finer structure than the samples treated above the phase transition temperature.

Methods

The experiments were carried out in a quartz tube attached to a vacuum system. The equipment was heated by a three-zone furnace.
Before the absorption, we measured the mass of the samples and calculated the initial pressure of hydrogen. The system was uploaded with >99,999 % hydrogen and completed with >99,999 % argon gas. The pressure of the system was measured with a Vacuubrand DVR 5 manometer and was monitored online with PC. The end of the H_2-uptake was evaluated on the basis of the pressure change. At the end of the H_2-uptake the sample was withdrawn to the cold part of the quartz tube. The mass of the sample was measured and the hydrogen concentration of the sample was calculated.

Figure 1. Picture of the new vacuum system

Remaining work

The measurement of hydrogen content of the sample by hot extraction method is planned.

An accurate synthetic algorithm of the time dependent migration leading to fission gas release from the fuel grain

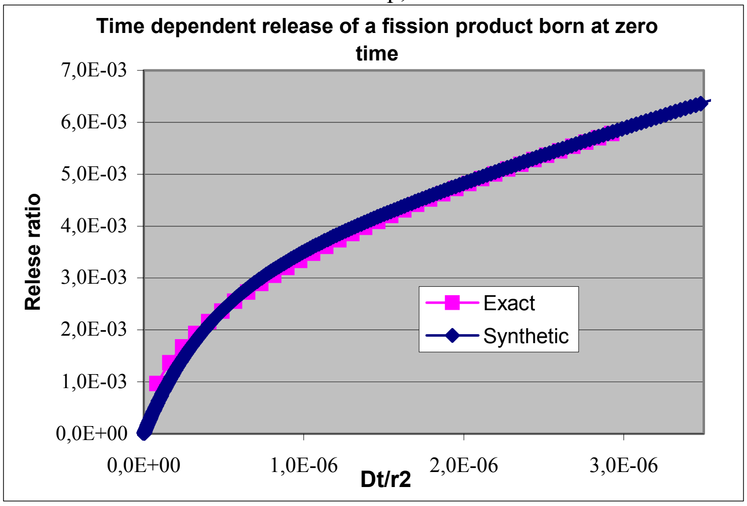
ANDRÁS KERESZTÚRI

Objective

The main process leading to fission gas release from a grain is the diffusion which is influenced by the resolution and the recoil. Even in case of time independent source of fission products, the exact analytical solution of the diffusion problem leads to slowly converging series, which is practically not applicable in the fuel behavior codes. The objective was to elaborate an algorithm which on the one hand is applicable in the practice but on the other hand gives back the exact solution with high accuracy.

Results

The elaborated method is based on the Green function technique. A synthetic step-wise algorithm, having only very few unknowns and to be refreshed in each time step, is derived from the Green function, which is on the one hand easy to built into the existing fuel behavior codes, on the other hand gives the exact solution of the infinite series back with high accuracy even at arbitrary time dependence of the fission product and resolution source, recoil rate, diffusion constant, and effective depth of recoil and resolution. The special Green functions and consequently built-in constants are "universal" in the sense, that they are not depending on the parameters of the above mentioned processes. Such a "universal" Green function, corresponding to fission source, is shown in the Figure. The method is generalized for taking into account the chain of fission product decays and a recipe of the slight modification of the reactor physics burnup codes is given.



Methods

Model development.

Remaining work

The proposed method is focusing only on the processes in the grain, the potential further models of the migration outside the grains are not treated, which can be unchanged according to the given fuel behaviour code.

Reference

A. KERESZTÚRI, Release of isotopes from the fuel grain due to the diffusion process, KFKI report, AEKI-FRL-2006-719-01/03.

Development of the FUROM code

JÁNOS GADÓ, ÁGNES GRIGER, KATALIN KULACSY

Objective

Improvement of the new fuel performance code FUROM was the task in 2006. For this sake the code was equipped with new models and functions, and a new version of the code called FUROM-1.3 was issued. In order to make the calculations of the gas inventory in the fuel-to-clad gap more accurate, a new, two-step fission gas release (FGR) model was developed as well. The new model is capable of following ramp tests. To verify and validate the new version, some of its important and critical models were analysed for reliability and appropriateness.

Results

High burn-up leads to increased porosity. That was modelled and its effect was taken into account in the fuel heat conductivity correlations.

The code was supplied with a new model to estimate the thickness of the oxide layer on the water side surface of the Zr1%Nb, that model was based on former AEKI's results. Another new model was elaborated for the Zircaloy cladding, as well as to determine the hydrogen uptake. The effect of the oxide-layer was taken into account in the heat transfer calculations.

A new subroutine was implemented to determine the inventory of all fission products during the irradiation process, and another new model for the calculation of the release of the radioactive isotopes and for the determination of the activity in the gap.

A new FGR model was elaborated (see in this Yearbook under a separate title), which includes a stationary gas release model valid when the reactor operates at more or less constant power, and a transient model to be used when the linear heat generation rate increases rapidly. This latter yields release rates higher than calculated by means of one-step release models. This model is not yet included in FUROM-1.3.

The verification/validation process established that

- the model for the irradiation growth of the cladding and the anisotropy factors used by FUROM are absolutely authentic.
- the value of the minimum gap size used in the code affects the forming of the temperatures in fuel elements, at the same time the measure of the fission gas release is practically independent of it.
- the determination of the heat transfer coefficient of the gas mixture in the gap is perfect.
- the gas release model of FUROM can be used very well below 2000 °C maximum center line temperature.
- the better estimation of the gas release under fast power change (ramp test) which is controlled not only by the temperature requires a more sophisticated model (it is in progress).

Methods

Comparisons between the predictions of FUROM and measurements.

Reference

Á. GRIGER, J. GADÓ: ”*The physical models of FUROM code*”, (2007) AEKI-FRL-2007-719-1.3-01 (in Hungarian)

Á. GRIGER: ”*The program guide of FUROM code*”, (2007) AEKI-FRL-2007-719-1.2-02 (in Hungarian)

Á. GRIGER: ”*The user guide of FUROM code*”, (2007) AEKI-FRL-2007-719-1.2-03 (in Hungarian)

Á. GRIGER: ”*The improvement of the results of FUROM code* ”, (2006) AEKI-FRL-2006-719-01-01 (in Hungarian)

K. KULACSY: “*A two-step model for the calculation of the activity of the fuel-to-clad gap*”, (2006) AEKI-FRL-2006-719-01-02, (in Hungarian)

K. KULACSY, A. KERESZTÚRI: "*Calculation of the activity content of the fuel-to-clad gap in nuclear fuels*", (2006) AEKI-ARL-2006-207/01, (in Hungarian)

Remaining work

Incorporation of a two-step gas release model in the FUROM code and its validation, as well as the revision of the creep models are the next tasks.

Model developments for the hydrogen uptake and the embrittlement of E110 cladding

CSABA GYŐRI*, EMESE SZABÓ, ZOLTÁN HÓZER, LAJOS MATUS, ERZSÉBET PEREZ-FERÓ
*JRC Institute for Transuranium Elements, Karlsruhe (ITU)

Objective

A co-operation project between ITU and AEKI aimed at increasing the predictive capabilities of the TRANSURANUS fuel code for LOCA-related phenomena. More precisely, the main objectives were to simulate the hydrogen uptake by the Zr1%Nb alloy and its effect on the cladding performance under accident conditions.

Results

Based on experimental data of AEKI, an empirical model was developed to calculate the hydrogen absorption in Zr1%Nb cladding during steam oxidation and in a gaseous atmosphere.
Deformation and the burst of pre-oxidised and hydrided Zr1%Nb tube specimens tested in two AEKI experimental series were simulated by the present version of the TRANSURANUS code and the results were compared to the corresponding experimental data (Figure 1).
Pre-test computations for the first Halden LOCA test with VVER fuel were performed by means of the extended TRANSURANUS fuel code, in order to analyse the principle combination of the initial and boundary conditions leading to fuel rod failure and to support the optimisation of the test parameters.

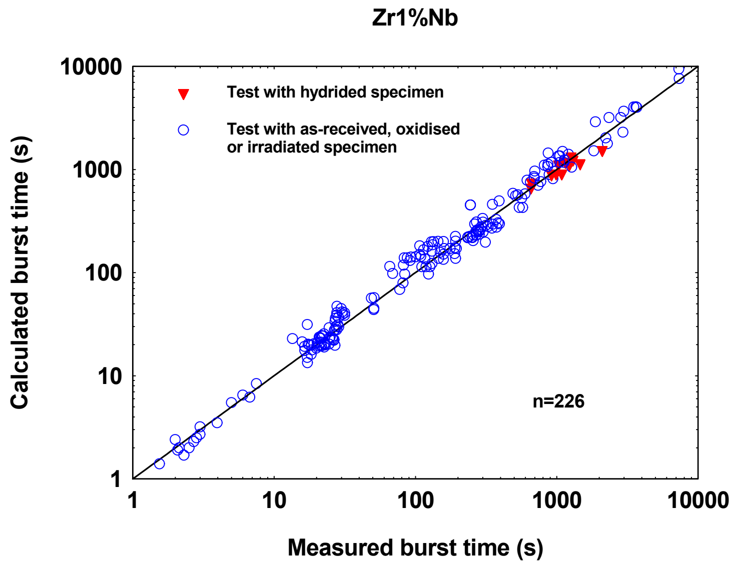


Figure 1. Calculated versus measured time of burst for hydrided

Specimens are hydrided (triangle) and as-received, oxidised and irradiated(o) Zr1%Nb specimens. Comparison of TRANSURANUS analyses and new ballooning test data.

Methods

The hydrogen absorption model is based on a quasi steady-state approach which is similar to that applied for the simulation of high-temperature cladding oxidation. The reaction rate is defined as a product of the solubility limit and an appropriate kinetics constant fitted separately to experimental data.

Reference

E. SZABÓ: TRANSURANUS analyses of cladding ballooning tests with pre-hydrided samples, AEKI-FRL-2006-479-01/01

CS. GYŐRI, Z. HÓZER, L. MATUS, E. PEREZ-FERÓ: Modelling of hydrogen absorption and embrittlement of Zr1%Nb cladding under postulated LOCA conditions, AEKI-FRL-2006-479-01/02

CSABA GYŐRI, ZOLTÁN HÓZER, EMESE SZABÓ: TRANSURANUS analyses for the Halden LOCA test IFA-650 with VVER fuel, AEKI-FRL-2006-479-01/03

Remaining work

The application of the TRANSURANUS code for the simulation of CODEX-CT tests is planned.

Oxidation and release of ruthenium from white inclusions

NÓRA VÉR, LAJOS MATUS, MIHÁLY KUNSTÁR, ANNA PINTÉR, JÁNOS OSÁN, ZOLTÁN HÓZER

Objective

Ruthenium in irradiated UO_2 appears in form of small alloy precipitations together with Mo, Rh, Pd and Tc (white inclusions). The main objective of this work was to study the release of ruthenium from white inclusions and compare the results to pure ruthenium data.

Results

The results proved that the oxidation rate of Ru from Mo-Ru-Rh-Pd alloy is about 50-60% of the evaporation rate of pure Ru at 1000 and 1100oC as well. This slower reaction rate could be partly resulted from the larger grain size of the alloy compared to the commercial ruthenium powder. There were no significant differences in the partial pressures of RuOx in the high temperature reaction chamber between pure alloy and alloy with other fission product components neither at 1100oC nor at 1000oC temperature of investigation.

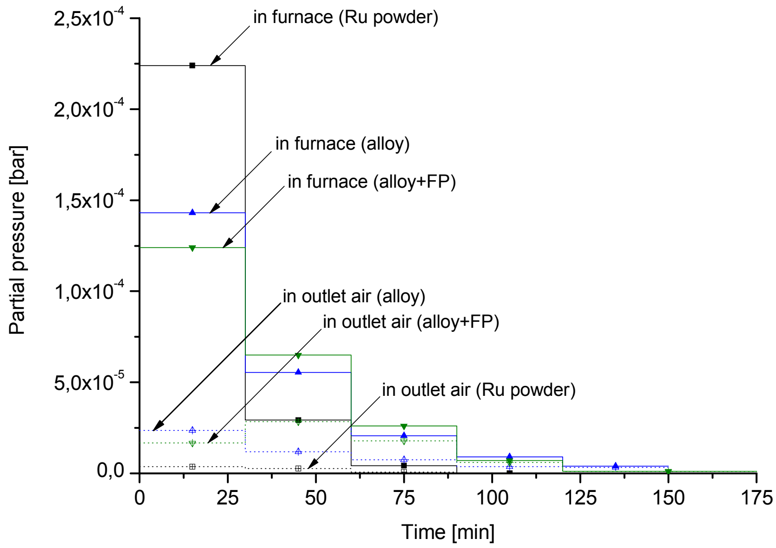


Figure 1. Partial pressures of ruthenium-oxides in furnace and outlet air (T = 1100oC)

Methods

The experiments were conducted in a high temperature furnace using ZrO_2 matrix with fine grains of Mo-Ru-Rh-Pd alloy, moreover with other inactive fission product components as well. Reference measurements were made with pure Ru powder in ZrO_2 at 1000 and 1100^{o}C. The applied concentrations of the alloy (chemical composition: Mo 47% / Ru 27% / Rh 7% / Pd 19% by weight) and other inactive fission product components (Cs, I, Se, Sn, Ag, Nd, Sb, Cd, Te, Ce, Ba, Zr) were representative for medium burnup fuel.

Reference

N. VÉR, L. MATUS, M. KUNSTÁR, A. PINTÉR, J. OSÁN, Z. HÓZER:Oxidation and release of ruthenium from white inclusions, AEKI-FRL-2006-409-01/01

Remaining work

In 2007 the delay of ruthenium release caused by different fission products will be studied.

Susceptibility of fuel cladding to localized corrosion and stress corrosion cracking

ANDRÁS SOMOGYI, ÁKOS HORVÁTH, RÉKA RÉPÁNSZKI, MÁRTA HORVÁTH, ANNA PINTÉR-CSORDÁS, ZSOLT KERNER, GÁBOR NAGY AND ROBERT SCHILLER

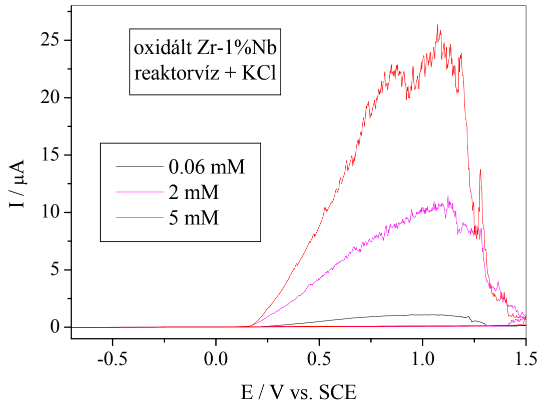
Objective

The overall objective of the project, supported by the Hungarian Atomic Energy Authority, is to determine the susceptibility of fuel cladding to localized corrosion and stress corrosion cracking. Both processes may have deteriorative effect: the fuel elements may become inhermetic resulting in the appearance of fission product in the primary circuits of nuclear power plants.

The input variables are not unique: rather, points in a set are candidates for a real input. To resolve this situation, a series of calculations is carried out and the variables to be limited are regarded as stochastic values. The question is, how should one to determine the number of runs to ensure assuring safety at a given (say 95%) level.

Results

During the second year of the project pitting corrosion susceptibility of the Zr-1%Nb cladding material were studied in experiments at room and high temperatures. The environmental conditions at high temperature were designed to simulate those of the primary circuit. The occurrence of pits were studied as a function of halide ion concentration and oxygen content in the solutions containing boric acid.



The effect of chloride addition on the stability of the passive layer of oxidised Zr-1%Nb material. The solution contains boric acid and potassium-hydroxide.

In addition to the solution content, the effect of stress on the surface oxide of Zr-1%Nb was also investigated in autoclave experiments. Constant stress was applied by inserting tapered Ti samples in the cladding tubes, and the results were evaluated by optical microscopy.

Methods

Electrochemical techniques: polarization experiments, open circuit potential measurements, electrochemical noise measurements. Surface analytical techniques: scanning tunneling microscopy, scanning electron microscopy, optical microscopy.

Remaining work

The work is going to be continued by developing a model for stress corrosion cracking on the basis of the experimental results.

AEKI contribution to the co-ordination of the VVER safety research (COVERS) project

FERENC GILLEMOT, MÁRTA HORVÁTH, ÁKOS HORVÁTH, LEVENTE TATÁR, TAMÁS FEKETE

Objective

The purpose of the COVERS project is to enhance the cooperation and information exchange among the VVER operating EU countries. 27 organisations of Finnland, Czech Republic, Slovakia, Ukraine, Hungary, Bulgaria, Germany and Spain are participating in the project, and Russia is a joint member. The project is co-financed by the members and by the DG Research of EU. The project has two technical working groups: safety and material ageing. AEKI is the WG leader of the ageing WP and subgroup leader in the safety WP.

Results

Started in January 2005, COVERS organised 4 meetings, and one very successful training seminar. During the meetings several lectures on VVER specific problems were presented. The ageing WP further developed the VERLIFE guide. VERLIFE is a guide of life management of the VVER NPP-s. This guide is the first common code of several EU countries, and widely used in life time management, it became a basic document for the national authorities. COVERS has a portal providing many important information for the members and for the public. COVERS also participated in the elaboration of two new EOI to initiate further framework project of the VVER users.

Methods

COVERS organises a main meeting every 6 months, where participants present their newly obtained results, runs a knowledge management and conservation system, organises training courses, maintains a common portal for the members, and continually develops the VERLIFE guide.

Remaining work

In the last year of the COVERS, issue of several important documents is planned; organisation of two meetings, two training seminars, elaboration of the future cooperation is planned.

AEKI contribution to the integrated project PERFECT

FERENC GILLEMOT, MÁRTA HORVÁTH, TAMÁS FEKETE ÁKOS HORVÁTH, LEVENTE TATÁR, TAMÁS FEKETE, ERIK HOUNDEFFO

Objective

The purpose of the PERFECT project is to model the effect of irradiation. The project has 3 main work packages: physical modelling, reactor in-core structures, and reactor pressure vessel. The physical model is a molecule-dynamic one describing the occurrence of precipitations, segregations and changes in the dislocations structure. The in-core structures mainly deal with irradiation assisted stress corrosion of austenitic steel in-vessel devices. The third one deals with the integrity of the reactor pressure vessel. The different models are connected by using SALOME platform and written in Fortran 99. Eleven core members (including KFKI AEKI) and more than 30 associated partners (institutes, universities, etc.) from EU are participating in the project. US is a joint member.

Results

Started in January 2003, PERFECT made remarkable efforts in modelling. Two physical reactor models simulate the irradiation effects on the microstructure. The input temperature, flux, spectra can be changed in a limited range, and a few alloying and pollution element can be modelled in low alloyed ferritic steels. Another model tries to describe the irradiation assisted stress corrosion cracking, but the inclusion of several affecting factors is still missing. The reactor pressure vessel integrity models try to apply Beremin's model, local approach and empirical formulas to calculate the fracture toughness of the irradiated steel from the output of the physical models. All the models are operating, but due to the present limitations they are good only for presentations, and they model properly only the ideal model materials.

Methods

The different models have been elaborated as Fortran 99 routines and connected to each other. Database has been elaborated and research data are collected to validate the models. KFKI AEKI mainly participated in research data testing, collection and model validation.

Remaining work

The project will be finished in 2007. Because of the very ambitious goal of the work, it is necessary to continue the project beyond the EU FP6. It is expected that work of another 10-15 years is necessary to get an irradiation simulator, which will take into account many more environmental effects, describe real reactor pressure vessel steels and will supply validated results for engineering purposes.

Study of industrial object by discrete neutron tomography

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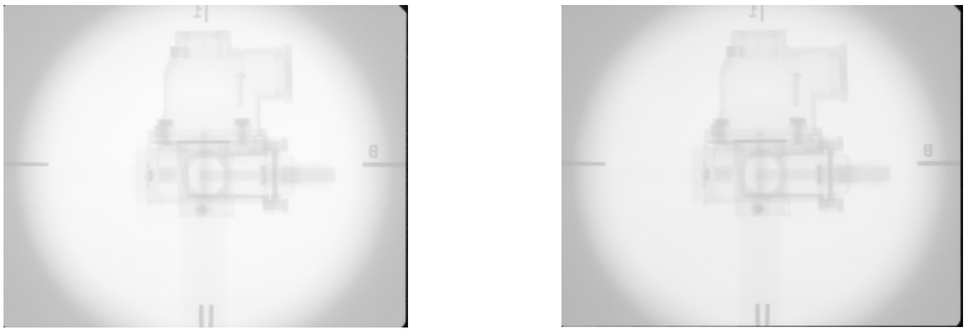
Objective

A gas pressure regulator was the first industrial object which was studied by Discrete Neutron Tomography (DNT) at the Budapest Research Reactor.

Results

The Radiography Station projections were obtained by an imaging plate detector. This detector was shifted and rotated during the measurement (see .a).



a.) unregistered projection   b.) registered projection

Figure 1. **The unregistered and registered projections.**

Consecutive images were taken so that geometric differences were introduced between consecutive projections. Hence the same projection directions from the source might pass the imaging plate at different pixel locations. Such geometric differences could degrade the result of the reconstruction. An image registration algorithm was applied to recover these geometric differences.

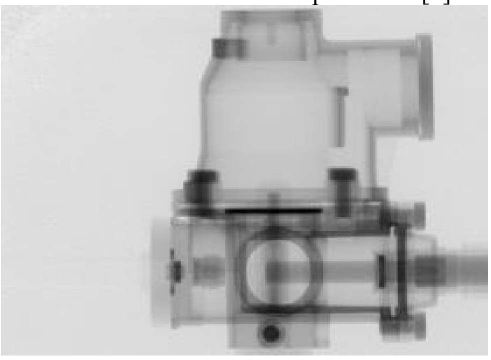
It was assumed that a rigid-body transformation (translation along the axes and rotation about the center of the image) could align the images. Our automatic registration algorithm developed previously, utilized the normalized version of mutual information as pixel similarity measure[1]. Although the algorithm was fully automatic, a preprocessing step was necessary. The center region of the images, where the projection of the object was visible must be masked out and the similarity measure was evaluated only outside of it. To eliminate the rotational invariance, artificial markers were affixed in front of the imaging plate, the projections of which were visible in the top, bottom, left and right hand side of the images. One of the projection images is selected as the reference image and the others, including the open beam projection image, were registered against it one by one. Visual inspection confirmed that the registration provided acceptable results (see .b).

We have cropped the area which contains the objects in all projections. We do did not want to reconstruct the empty area. After cropping the relevant part of the projection we have re-sampled the projections in order to reduce the reconstruction time for DT.

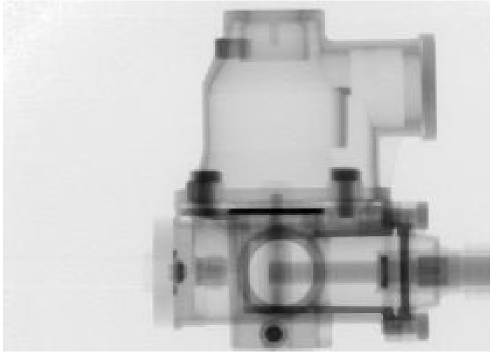
The projections produced by neutron rays suffer from distortions. Some of these distortions come from the image acquisition system itself. In order to reduce these effects we have to correct them. Correcting the uniformity distortion we need the open beam projection. For each projection we have to multiply all pixels with the corresponding open beam projection reciprocal value (see .a). If the projection intensity changes during the measurement, we have to correct it in such a way that the average intensity should be constant in the background area (see .b).

After these preprocessing methods we have to find the initial beam energy to perform a logarithmic transform. We have taken the average mean of that part of the projections where no object can be found. This method gives a good approximation if we do not know the value precisely.

Classical Filtered Back Projection method is the most general reconstruction algorithm. Further details can be found on this topic in Ref.[1].



a.) Uniformity



b.) Intensity

Figure 2. Projections after uniformity correction

In case of the discrete tomography method, the reconstruction problem is equivalent to finding a solution of the linear equation system

$$Ax = b\text{, where } x \in D = \{0,\ldots,d_n\}. \tag{1}$$

The A matrix can be calculated knowing the geometry of the acquisition system. If we have not so many projections then the number equations is less than the number of unknowns. It means that Eq. (1) can have several solutions, even discrete valued ones. Furthermore, due to the measurement errors, we are looking for the solution which is close to the precise one. In accordance with this, we can reformulate the solution of equation system (1) to the following minimization problem:

$$C(x) = \|Ax - b\|^2 + \gamma \cdot \Phi(x), \tag{2}$$

where the first term assures x to be close to the solution of Eq. (1). The second terms allows us to include a prior information x into the optimization problem. We tried to find such an x, which has large coherent regions.

$$\Phi(x) = \sum_{j=0}^{m} \sum_{l \in Q_j^k} g_{l,j} \left|\xi_j - \xi_l\right|, \tag{3}$$

where in the Q environment of the current position we penalize the first term of the Eq. (2) with the difference in the distance between the two positions.
Since we are looking for discrete x values, numerical optimization methods are not suitable here. We have chosen the Simulated Annealing (SA) optimization procedure.
We have removed the inner part of the gas pressure regulator in order to have binary valued slices. We have tried to determine the attenuation coefficient for these slices performing reconstruction with different attenuation values. This heuristic method has accounted for the first term of the target function in Eq. (2) and the smoothness (see Eq. (3)) of the reconstructed object. After that we have moved to a slice such, which contains more then two values inside and tried to find the next attenuation coefficient. The result for two and three valued slices can be seen in . The result of three valued slices has errors coming from other projection distortions. That can be seen on the classical FBP results as well and the fact that this slice is actually not three valued but four valued.

Two kinds of reconstruction method are applied on the corrected projection data. The classical filtered back projection gives good results at high number of projections. Naturally we get a better result if we have larger number of projections in case of a classical reconstruction. The discrete tomography method can be applied when the object consists of two materials and the number of projection is low.

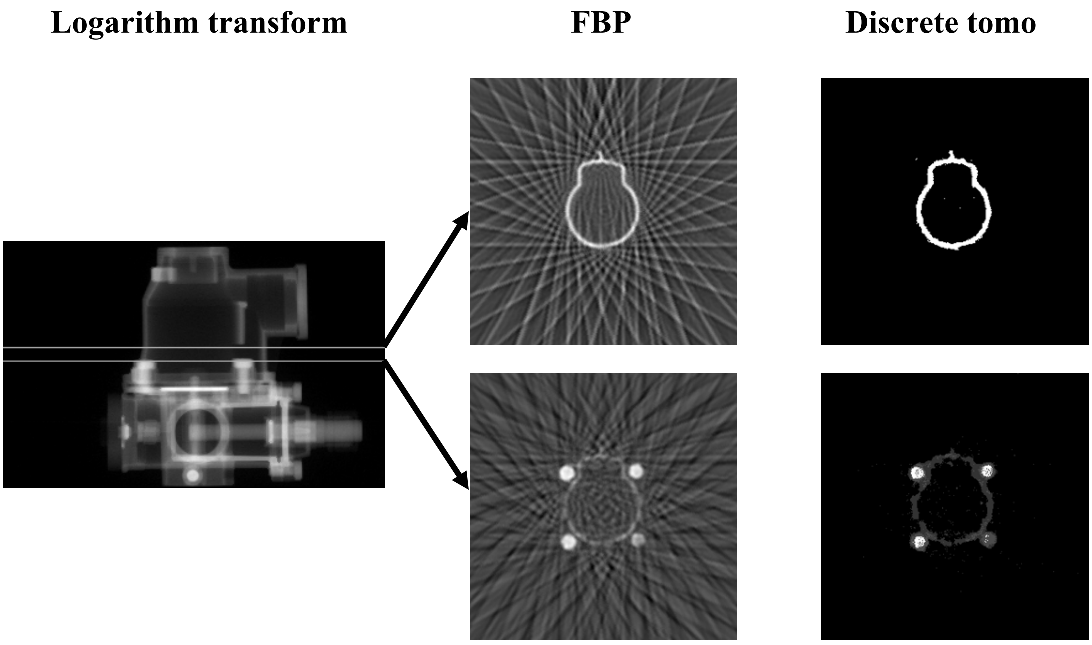


Figure 3. Results of the classical FBP and DT method on Radiography Station (18 projections)

References

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Remaining work

The project is finished. Its results are published in Book of Abstracts of WCNR-8. 4p.

Thermal hydraulics

AEKI's activity in thermal hydraulics is traditionally storng. In the subject period, extended code testing (WAHA3, CATHARE, the NEPTUNE CFD code in the NURESIM project), uncertatinty analysis of the LOFT experiment, development of new codes (boiling in a subchannel, and a new code for magnetohydrodynamic simulation are reported. The extended numerical modeling made it necessary to build an effective computational tool:a PC cluster.

Investigation of condensation-induced water hammer in pipelines

GÁBOR BARANYAI, IMRE BARNA, GYÖRGY ÉZSÖL

Objective

The main objective of the project is to further develop and extensively validate the WAHA3 computer code, with the aim of evaluating the pipelines of the existing nuclear power plants with regard to the water hammer phenomenon.

Results

To perform validation experiments for studying the water hammer phenomenon in pipelines, the PMK-2 facility has been completed by a test section instrumented with special high speed pressure transducers and combined local void-temperature probes. The flooding is provided by a special pump. The data collecting and data acquisition system is able to receive, convert and store the high speed signals.

To test the set-up and instrumentation, a test program has been performed on a smaller test section. The results show that the instrumentation is capable of measuring the water hammer phenomenon. Turther development of WAHA3 has been continued.

Methods

Development and construction of an experimental set-up, on the basis of the PMK-2 facility.

Reference

G. BARANYAI, I. F. BARNA, GY. ÉZSÖL: WAHA-EMK02, in Hungarian, GVOP-3.1.1.-2004-05-0025/3.0

Remaining work

In accordance with the schedule of the project, the experiments will be performed in 2007. Results will be used to validate the WAHA3 code, and then the code will be suitable for calculations of pipelines of plants.

Application of CATHARE code to DBA analysis

ANTAL TAKÁCS, ISTVÁNNÉ FARKAS

Objective

In 2006, we worked on two tasks for the Hungarian authority HAEA/NSI. In 2005, a plant transient initiated by the opening of a 90 mm break on the pressurizer top was recalculated by the French thermohydraulic system code CATHARE and results were compared with a reference RELAP calculation [1]. The same transient was also calculated by the Nuclear Technical Institute of the Technical University Budapest (BME-NTI) using the Finnish APROS thermohydraulic system code. The first task of 2006 was to compare the calculation results of BME-NTI and KFKI-AEKI.

The second task in the project was to calculate and analyze a transient of 200% cold leg break six hours after reactor shut-down in unit 4 of Paks NPP, when the hydroaccumulators are already isolated from the system.

Results

The CATHARE and the APROS calculations similarly modelled the relevant physical processes of the transient. The main difference between them occurred in the operation of the hydroaccumulators. The APROS calculation anticipated injection in ten steps while CATHARE indicated four of them. The loop seal behaviours showed a good agreement [2].

Regarding task two this was the first time to calculate an LB LOCA with using the plant scheme. The CATHARE simulated reasonably well the transient concerned [3]. After core dry-out, a heat-up of fuel claddings was predicted. Then the available LPIS system rewets the core and cooling is achieved before the end of the 600 seconds transient time. The maximum cladding temperature was 550 ^{o}C which was much below the 1200 ^{o}C safety limit.

Methods

The CATHARE results have been compared with the APROS results and the differences have been explained in the first task. In the second task CATHARE results have been compared with the correspondent ATHLET calculation.

Remaining work

Further application of CATHARE code is envisaged in cooperation with HAEA.

Reference

[1] A. TAKÁCS: 90 mm Break of the Tube Between the PRZ and the PRZ Safety Valve, Application of CATHARE Code for the Assessment of the Final Safety Report Chapter 15, OAH/NBI-ABA-28/06, Budapest, January, 2005 (in Hungarian)

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Uncertainty analysis of the LOFT L2-5 experiment

ATTILA GUBA, PÉTER SARKADI, IVÁN TÓTH, ISTVÁN TROSZTEL

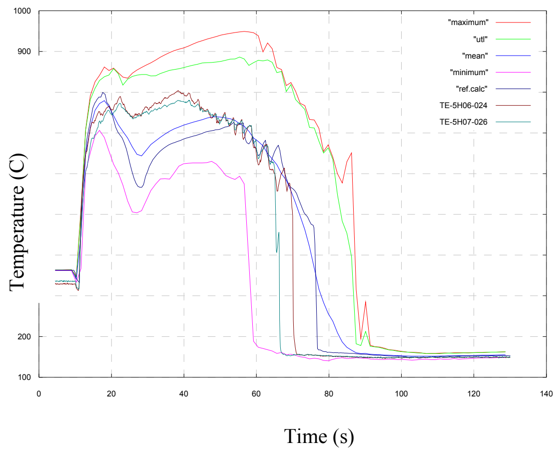
Objective

The institute participates in the OECD BEMUSE project that focuses on uncertainty analysis of best-estimate calculations in case of a large break LOCA. In the first part of the project the participants performed a base case calculation, sensitivity studies and the uncertainty analysis for the LOFT L2-5 test.

Results

In the base case calculation best-estimate values were applied for all input and available model parameters. The primary pressure evolution during the transient calculated by AEKI was one of the best among the results of the project participants [1] that witnessed the correct prediction of the break flow rate and the calculated peak cladding temperature (PCT) was within the uncertainty band of the test result.

For the uncertainty analysis 35 parameters have been selected: they did not include non-input model parameters due to difficulties in varying them in the ATHLET code. In order to calculate the upper tolerance limit for the PCT at 95% probability and 95% confidence level, 105 ATHLET runs were carried out that – according to the Wilks formula – allowed to discard the calculation with the highest PCT. The 105 calculations resulted in an uncertainty band of 151 and 305 K for the first and second PCT, respectively, and the uncertainty band completely enveloped the highest measured PCT curve for the whole transient. The uncertainty analysis was supported by the HAEA.



Methods

The ATHLET mod2.0 code version was used for the LOCA analysis, while the uncertainty analysis was performed by the GRS method that is based on the probabilistic approach with propagation of input parameter uncertainties. The SUSA code package was applied for this latter step.

Reference

[1] BEMUSE Phase II Report. Re-analysis of the ISP-13 Exercise, Post Test Analysis of the LOFT L2-5 Test Calculation. NEA/CSNI/R(2006)2 OECD NEA report, May 2006.

[2] A. Guba, P. Sarkadi, I. Trosztel: Analysis of code uncertainties within the OECD BEMUSE Project. AEKI-THL-2006-252/02/M0. (In Hungarian)

Remaining work

The next step in the BEMUSE project is the uncertainty evaluation for a large break LOCA event in a reference power plant, the Zion PWR.

Numerical Simulation of Magnetohydrodynamics

PÉTER KÁVRÁN, GÁBOR HÁZI

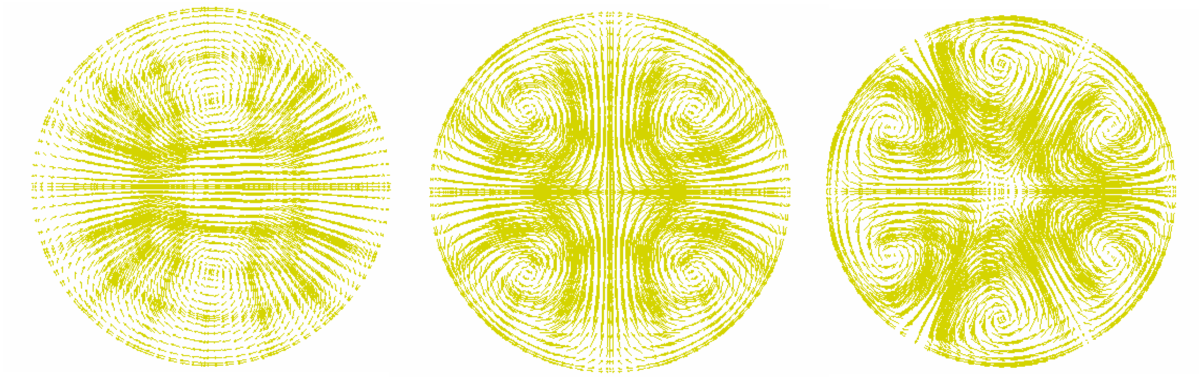
Objective

In fusion research, measurements reflecting the state of the plasma have been performed in tokamaks and stellarators. The measurement data are available at KFKI RMKI (Research Institute for Particle and Nuclear Physics). A numerical code SPECMHD, based on magnetohydrodynamic (MHD) description of plasmas, is being developed in our institute, in order to help the interpretation of measurement data.

Results

Our code SPECMHD solves the nonlinear magnetohydrodynamic equations [1] by spectral element method [2,3] in arbitrary three dimensional geometry. The geometry is generated by the GAMBIT mesh-generator software. Geometry generation for realistic tokamak or stellarator equilibrium needs sophisticated algorithms that have not been developed yet. The parallelized version of the code can be run on distributed memory cluster architectures. The solver algorithm has been tested against the standard Orszag-Tang vortex, Brio-Wu shock tube, Kelvin-Helmholtz instability and Harmann-flow test problems [4]. A magnetohydrostatic test has been used to verify the exponential spatial convergence property of the spectral element method [4]. We have started to study fixed boundary circular cylinder instabilities with the code using perfectly conducting boundary conditions (see Fig. 1). Initial perturbations for velocities are of the form $v_1(r,\theta,z) = v_r(r)\cos(m\theta - \frac{2\pi}{L}z)$.

Fig. 1. Velocities for fixed boundary instabilities in a circular cylinder (m=1, m=2, m=3 internal modes [1]).



Method

The code used to investigate instabilities is based on spectral element method in standard Galerkin framework. For shock problems an extension of the code based on discontinuous Galerkin method is used [4].

Remaining work

Growing rates of MHD modes need to be determined from the fixed boundary circular cylinder simulations and the simulation results have to be compared with published results. By calculating a simple analytical tokamak equilibrium, comparison of measuement data and simulation result for instabilities could be performed.

References

[1] GLENN, BATEMAN: MHD Instabilities, The MIT Press, 1978,
[2] M.O. DEVILLE, P.F. FISCHER, E.H. MUND: High Order Methods for Incompressible Fluid Flows, 2002,
[3] GEORGE EM KARNIADAKIS, S.J. SHERWIN: Spectral/hp Element Methods for CFD, Oxford University Press, 1999,
[4] G.HÁZI, P.KÁVRÁN: A maghasadás és a magfúzió határterülete 2., MTA TTF-40.34/2-8/2006, AEKI-G-1040/2006.

Participation in the NURESIM project

GÁBOR HÁZI, ATTILA MÁRKUS, GUSZTÁV MAYER

Objective

The objective of the NURESIM (European platform for NUclear REactor SIMulations) project is to develop a common European standard software platform for numerical modeling of nuclear reactors. As a specific objective of the project, the improvement of the modeling capabilities of the current models is envisaged. Considering thermal-hydraulics, the focus is on the formation of two key phenomena: Pressurized Thermal Shock and Critical Heat Flux (CHF). Both phenomena are studied at various levels, including macroscopic and mesoscopic modeling methods.

In the framework of the NURESIM project, boiling in a subchannel of a rod bundle is simulated in our institute. The objective of the calculations is to get insight into the development of CHF in case of rod bundles and to improve the interfacial transport models. Parallel with this activity we also validate the CFD code of the NURESIM project (NEPTUNE CFD) simulating rod bundle flows [1].

Results

We have recently shown that the numerical approach we are using for numerical simulations, namely the lattice Boltzmann method with the pseudopotential extension [2], can be thermodynamically consistent, as far as a suitable pseudopotential is chosen [3].

The vitality of our numerical approach has been demonstrated by presenting the simulation results of nucleate boiling in a simple geometry (see Fig. 1 left) [4]. Comparing our results with measurement data or the results of other numerical simulations we obtained good qualitative agreement, motivating a further step towards simulation of boiling in rod bundles. Nucleation has been simulated in a subchannel of a rod bundle with triangular rod arrangement (see Fig. 1 right). Preliminary results have been published and discussed in order to establish the future work [4].

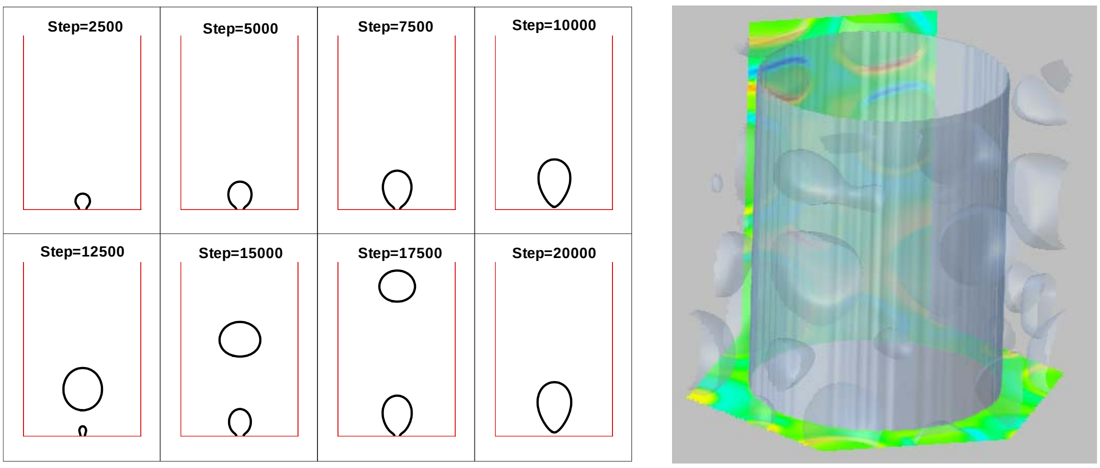


Fig. 1. Bubble growing and detachment from a heated wall (left) and simulated bubbles in a subchannel (right).

Methods

For the calculations we have been using the lattice Boltzmann method. This method is an innovative approach for modeling complex systems and it is especially true in the field of two phase flow modeling where the mesoscopic nature of this method can be fully exploited.

Publications

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2. G. HÁZI, A. MÁRKUS: "On the thermodynamic consistency of the lattice Boltzmann method with pseudopotential", submitted *Physical Review E* (2007)
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PC cluster building

JÓZSEF PÁLES, GÁBOR HÁZI

Objective

In the last few years our fine-scale thermohydraulics investigations increased significantly the need for computing power and resources. In order to satisfy the high computational demand cost effectively, we decided to build a PC cluster using cheap hardware components and open source software packages.

Results

The cluster building was started at the beginning of 2006. All hardware and software installations needed only two months and then the application of the PC cluster could be started. To test the performance of the cluster, a SKAMPI (benchmark for MPI implementations) run was performed and the benchmark results have been reported[1,2].

During the year large eddy simulation of subchannels [3],[4], spectral element simulations of magnetohydrodynamic instabilities and bubble flow simuations have been performed on the cluster. Many of those calculations could not have been carried out using single PC-s, because of the huge memory demand. On the top of that, linear speed up (processor - requested time) could be achieved in many applications as it has been expected.

Methods

The cluster is composed of 32 nodes connected by a high-speed switched Gigabit Ethernet network. Since the switch has 48 ports, the cluster can be extended in the future by connecting further nodes to the switch. Each node is a simple PC with an Intel P4 processor, 2GByte memory and 120GByte hard disk, but the server has a DVD writer and higher hard disk capacity (for more details see [1], [2]).

Besides the hardware, the other important system component of the cluster is the software platform, which makes it possible to use the computer network as a single computer. Currently we use Gentoo Linux operating system on each node, and the MPD job management system is used to organize the users, jobs etc. The administration, maintenance and the use of the system have been facilitated by the development of home-made procedures and shell scripts.

In order to exploit the performance of the cluster we had to make our numerical codes parallel. For these purposes we used the MPI parallel communication programming library, and the PETSc parallel mathematical programming library.

Publications

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[2] PÁLES J., HÁZI G.: The AEKI PC cluster, AEKI-SZL-2006-112-01/01 (in Hungarian)

[3] MAYER G., PÁLES J., HÁZI G., Large eddy simulation of subchannel using the lattice Boltzmann method, Annals of Nuclear Energy, 34, 140-149, (2007)

[4] G. MAYER, G. HÁZI: "Direct numerical and large eddy simulation of longitudinal flow along triangular array of rods using the lattice Boltzmann method", *Mathematics and Computers in Simulation*, **72**, pp. 173-178, (2006)

Advanced nuclear technologies

An intensive work to develop reliable, more economic reactor reactor types can be observed world wide. These efforst are usually summarized under the Generation Four (G-IV) acronym. AEKI participates in a number of international co-operations and national research projects to develop well defined aspects of the G-IV reactors. Our efforst are focused mainly to the supercritical water reactos, the present section summarizes the research on that area.

NUKENERG project: supercritical water reactor core design

CSABA MARÁCZY, GYÖRGY HEGYI, ÁRON BROLLY, GÁBOR HORDÓSY, PÉTER VÉRTES, SÁNDOR FEHÉR*, TIBOR REISS*

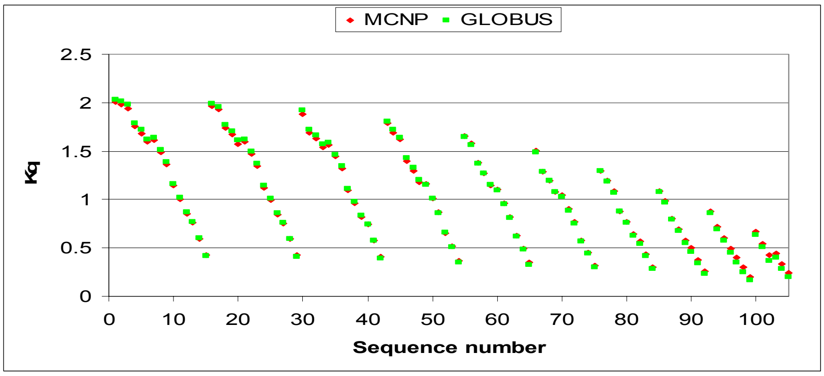
**Budapest University of Technology and Economics Institute of Nuclear Techniques, Budapest Pf. 91. H-1521 Hungary*

Objective

The European version of Supercritical Water Reactors (SCWR), the High Performance Light Water Reactor (HPLWR) operates in the thermodynamically supercritical region of water. The basic objective of the work is to design the core of HPLWR on the basis of an existing square fuel assembly proposal. The more heterogeneous than usual fuel assembly structure, the smaller than usual node size, the steep axial water density variation and last but not least the lack of experimental results require a thorough verification of the GLOBUS nodal code with the help of Monte Carlo calculations.

Results

The step by step verification procedure included the test of few group cross section generation, the test of axial streaming and axial reflector description method, the test of interaction of assemblies and radial reflector description method. The last test was the realistic full core calculation at prescribed thermohydraulic distributions with reflector. The test of the power distribution and multiplication factor of the GLOBUS code was carried out in case of a core built up by 3x3 clusters of assemblies having different enrichments. The relative deviation of the GLOBUS multiplication factor from MCNP Monte Carlo code is 0.5%. The radial power distributions affecting the heatup of coolant were evaluated in the 45 degree symmetry sector. The figure below shows the applicability of the GLOBUS code to HPLWR core calculations.



The radial power distribution of fuel assemblies in the 45 degree symmetry sector

Methods

Deterministic and Monte Carlo neutron transport calculations.

Remaining work

Improvement, further validation and application of the developed code system for HPLWR reactor cores.

References

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Preparation for the safety evaluation of a G-IV reactor cooled by supercritical water

ANDRÁS KERESZTÚRI, MIHÁLY MAKAI, GYÖRGY HEGYI, ISTVÁN TROSZTEL

Objective

The final goal of the 3 year project is to evaluate the safety of a G-IV reactor cooled by supercritical water. The generic question to be answered is if there is any essential safety related problem making the concept not feasible. The evaluation has to be based on own calculations and on literature data. As not all the details of the reactor have been specified yet, a specific task is the design of the protection system, which - in the given case - can be based only on movable absorber rods. Consequently, special attention is to be paid for the reactivity initiated abnormal events and their modeling.

Results

The following preparatory activities were carried out in the first year.

- Specification of the appropriate acceptance criteria of the safety analyses for the specific reactor type [1]. The general basic safety criteria in the NRC and IAEA documents were taken into account as a starting point. Unfortunately, the basic document related to the acceptance criteria, namely the "Standard Review Plan, Section 4.2" is focusing only on Zr clad fuel, which is not applicable at the given high temperature. Therefore some important acceptance criteria had to be established by using the "ASME Boiler & Pressure Vessel Code, Section III" document for the given stainless steel cladding. It was found that the temperature dependent ultimate and yield stresses should be used also as acceptance criteria which underlines the importance of using fuel behavior codes in the safety analyses.

- The planned safety analyses will be performed by the modified version of the coupled KIKO3D-ATHLET code. Detailed software specifications of the two modified codes were elaborated [2,3]. The appropriate thermal hydraulic heat transfer and friction correlations depending on several parameters were selected after a careful literature study.

Methods

Code design, literature study

Remaining work

Concerning the work plan, there is no remaining work of the first year.

References

[1] M. MAKAI, A. KERESZTÚRI: The basic principles of safety evaluation and the acceptance criteria of the safey analyses for the reactor cooled by supercritical water, KFKI report, August 2006.

[2] GY. HEGYI, Á. BROLLY: Software design of the reactor physics module for the calculation of the reactor cooled by supercritical water, KFKI report, August 2006.

[3] I. TROSZTEL: Software design of the thermal hydraulics module for the calculation of the reactor cooled by supercritical water, KFKI report, August 2006.

High performance light water reactor - Phase 2

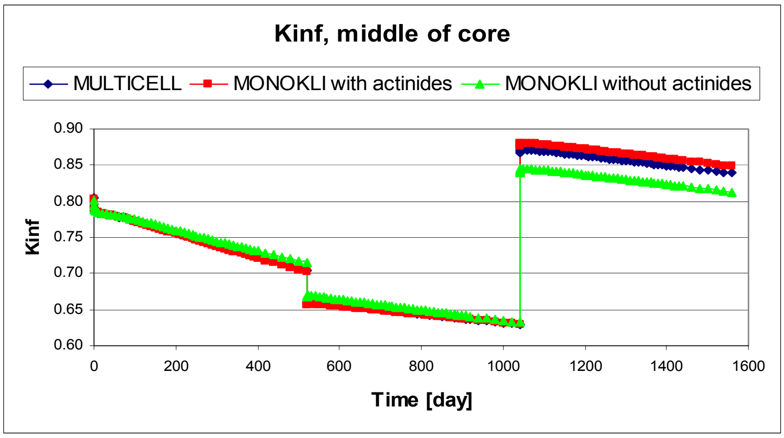
CSABA MARÁCZY, GYÖRGY HEGYI, GÁBOR HORDÓSY, EMESE TEMESVÁRI

Objective

The High Performance Light Water Reactor (HPLWR) belongs to one of the six reactor types currently being investigated within the framework of the Generation IV International Forum: the Supercritical Water Cooled Reactor. The objective of the HPLWR project is to assess the feasibility of a high efficiency light water reactor operating in a thermodynamically supercritical region. In September 2006, the "HPLWR Phase 2" FP 6 project had its Information Exchange Meeting and Kick-Off Meeting in our institute. Seven research institutes, two universities and an industrial partner work on the concept. The main task of KFKI-AEKI is the core design. Besides burnable absorbers, cluster type control rods are applied to compensate the excess reactivity at the beginning of the cycle as no boric acid reactivity control is allowed. The presence of absorber clusters and the high axial water density variation in the core requires verifying the few group cross section generation methodology.

Results

For a preliminary fuel assembly configuration, few group diffusion type cross sections were mathematically prepared as polynomial functions of technological parameters. In the first case, the actinide inventory of fuel was simply described with burnup and in the second case the most important actinides were calculated with the MONOKLI code during the burnup process. The MULTICELL deterministic transport code was used to provide the reference infinite multiplication factor (k_{inf}) of a fuel assembly node at the middle of the core for three cycles with different technological parameters. The superiority of the second method with small calculational burden is apparent in the figure below.



Comparison of burnup calculations for a HPLWR fuel assembly slice.

Methods

- Deterministic neutron transport calculations.
- Isotopic inventory calculations.

Remaining work

Application of the developed codes and data for HPLWR reactor cores.

References

[1] GY. HEGYI, CS. MARÁCZY, Á. BROLLY, P. VÉRTES, G. HORDÓSY: Application of KARATE to Supercritical Water Reactors. Proceedings of the.16th Symposium of AER, September 25 – 29, 2006, Bratislava, Slovakia, ISBN-978-963-372-633-4 p.745-750

Fuel cycle investigations of G-IV reactors

PÉTER VÉRTES, ÁRON BROLLY

Objective

The final goal of this project is a comparative analysis of different reactor types from the point of view of transmutation capability. The coupled systems, i.e. the utilization of processed material of spent fuels of one reactor in another one, represent particular interest.

Results

A system, called NOTRADAT, for neutronic calculations of reactors has been elaborated. By means of this system, the following tasks have been done:

- test calculations of Sq2 assembly of HPLWR reactor
- a benchmark exercise for MYRRHA subcritical (ADS) reactor [1]
- benchmark exercise for a sodium-cooled fast reactor on the base of BN-800 model

Methods

The NOTRADAT system includes a multigroup micro- and macro- library generation, the multigroup transport calculation with S_N code, burnup library generation and burnup calculation. These are accomplished by means of the well-known NJOY, TRANSX, DANTSYS and TIBSO codes and by some newly developed auxiliary programs.

Remaining work

Fuel cycle analysis of coupled system: from waste of reactor of one type producing new fuel and burning it in a reactor of other type.

Reference

[1] P. VÉRTES: Some neutronic calculations of MYRRHA assembly, to be presented on Third Research Coordination Meeting) of the Coordinated Research Project (CRP) on “Studies of Innovative Reactor Technology Options for Effective Incineration of Radioactive Waste”, Indira Gandhi Centre for Atomic Research (IGCAR), Chennai, India, 15 –19 January 2007

Thermohydraulics of supercritical water reactors

GÁBOR HÁZI, ISTVÁN FARKAS, ATTILA MÁRKUS

Objective

The supercritical water reactor (SCWR) is a Generation IV concept that uses supercritical water as coolant. Recent investigations have shown that using of long standing thermohydraulics correlations (heat transfer, pressure loss) is not a viable way for SCWR, because of their poor predictive capabilities near the critical point. Accordingly, heat transfer in supercritical fluid has been studied in our institute to get more insight into the underlying transfer mechanisms.

Results

A new lattice Boltzmann model has been developed to simulate heat transfer in supercritical fluid. The model can be used to study heat transfer near the critical point. Due to the diverging isothermal compressibility, it has been shown that the so-called piston effect is the major heat transfer mechanism in this region. Using the new method numerical examples were presented [1], simulating heat transfer near the pseudocritical temperature in a rectangular cavity. It was demonstrated that our model can predict well the influence of the piston effect on the heat transfer. We have also studied the onset of convection in a Rayleigh-Benard configuration. It was shown that our model can predict qualitatively well the onset of convection near the critical point, where there is a crossover between the Rayleigh and Schwartzchild criterions.

Methods

A new lattice Boltzmann method has been developed to simulate heat transfer in supercritical fluid.

Publications

4. G. HÁZI, A. MÁRKUS: "Modeling heat transfer in supercritical water using the lattice Boltzmann method", submitted *Physical Review E* (2007)

Structural materials for next generation nuclear reactors

ÁKOS HORVÁTH, FERENC GILLEMOT, GÁBOR URI, FERENC GAJDOS, ATTILA IMRE, GYÖRGY JÁKLI, ZOLTÁN HÓZER

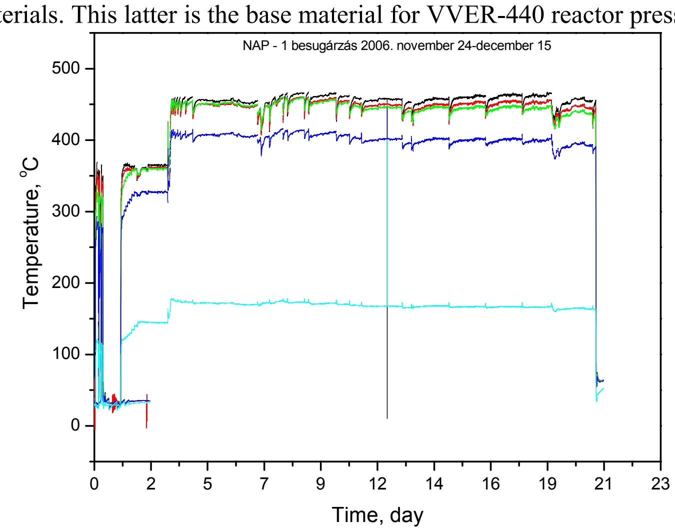
Objective

The Structural Materials subtask of the NAP NUKENERG project aimed at investigating the candidate materials for the supercritical water reactor, and materials of high heat resistant in fusion energy systems.

The increased temperature – as well as the higher neutron flux – means higher load on the structural materials. Development of new fuels and material is mandatory for these reactor concepts, which raises challenging scientific and technical issues. The input variables are not unique: rather, points in a set are candidates for a real input. To resolve this situation, a series of calculations is carried out and the variables to be limited are regarded as stochastic values. The question is, how should one to determine the number of runs to ensure assuring safety at a given (say 95%) level.

Results

During the second year of the project, the high temperature dry irradiation rig, BAGIRA, has been prepared for irradiations at temperatures up to 500°C. The original rig structure has been partially replaced by Ti alloy, the first irradiation cycle has been carried out using Eurofer, Tungsten and 15Ch2MFA materials. This latter is the base material for VVER-440 reactor pressure vessel.



Irradiation temperature vs. time in the upgraded dry irradiation rig

A supercritical pressure autoclave for corrosion studies is constructed. The fuel rod performance code, TRANSURANUS, is being prepared for new cladding materials, chromium steels or inconels.

Methods

During irradiations, up to $3*10^{13}$ n/cm$^{2}\cdot$ fast fluence rates – in dry conditions – can be reached in the Budapest Research Reactor. The fracture toughness of the irradiated samples is measured by using the Master Curve method. Reconstitution of the standard and mini sized Charpy specimens is made by Stud Welding Technology.

Corroded samples are characterised by standard electrochemical methods, SEM and metallography.

The project is also aimed at running the experiments in co-operation with partner European laboratories.

Remaining work

The project will be continued with the evaluation of the mechanical properties of the irradiated specimens. The samples corroded in high pressure and high temperature water will be analysed.

Hydrogen production by plasma chemical methods and in supercritical water

ROBERT SCHILLER AND GABOR NAGY

Objective

As a continuation of our studies reported last year, a critical overview on the present status of hydrogen production by plasma chemical methods and in supercritical water was performed (partially supported by the European Fusion Development Agreement)..

Results

Plasma chemical methods comprise both “cold” and “hot” plasma conditions.

The main advantage of “cold” plasma reactors is that small, mobile units can be constructed for waste gas treatment and hydrocarbon reforming. At present, this technique is not aimed at primary, large scale, non-fossil hydrogen production.

The methane $+H_2O \rightarrow$ methanol reaction, a „cold” plasma process for chemical energy storage, seems energetically favourable because it applies low temperature (~100 ^oC) heat source while produces high quality chemical energy.

„Hot” plasmas enable one to use much shorter in-reactor residence times than those needed for conventional units. As an example the $CH_4+H_2O \rightarrow CO+3H_2$ reaction proceeds in a „hot” plasma nine orders of magnitude faster than in a thermal converter.

Pyrolysis of waste gases or of biomass for hydrogen production can be effected in a hybrid argon-water stabilized plasma torch at a wall temperature of 1700 ^oC.

The most pressing problem with plasma chemical processes is their high energy consumption.

Supercritical states offer possibilities to influence thermodynamic and/or kinetic conditions of chemical reactions in a number desired ways. Variations of pressure and temperature within extended ranges allow optimizing reaction conditions with respect to chemical kinetics, transport properties and phase behaviour. Some examples of the important processes follow:

Hydrogen is produced from biomass by hydrothermal gasification. Here, in an excess of water, the reaction at temperatures up to 700 °C and pressures around 30 MPa directly leads to valuable hydrogen instead of synthetic gas.

Gasification of glucose in supercritical water at 600 ^oC, 34.5 MPa, and 30 s residence time results in the formation of hydrogen, carbon dioxide, carbon monoxide, and methane.

A recent, most promising method is irradiation; at around 400 ^oC the hydrogen yield is an order of magnitude higher than at room temperature and corresponds to some 30% of energy efficiency.

Methods

Plasma generation, high pressure – high temperature chemistry, gas analysis

Remaining work

Theoretical work on reaction kinetics under critical and supercritical condition.

Assessment of energetics of plasma chemical reactions with a particular view to low-temperature heat sources.

Publication

R.SCHILLER: Hydrogen Production by Radiolysis, Pune University Workshop on Radiation & Photochemistry (PUWORP-2006), Book of Abstracts

R. SCHILLER, G. NAGY, J. HAYWARD AND D. MAISONNIER: Radiation Chemical and Plasma Chemical Processes for Hydrogen Production from Water, First Budapest International Hydrogen Energy Forum 2006. 10. 9-10 (CD)

Material testing reactor innovations

ÁKOS HORVÁTH, ÁRON BROLLY, ZSOLT KERNER

Objective

The acronym MTR+I3 covers an Integrated Infrastructure Initiative to strengthen European experimental capabilities for testing material and fuel by:

- building a lasting cooperation between Material Testing Reactor (MTR) operators and relevant laboratories,
- maintaining the European leadership with up-dated capabilities and competences,
- improving and structuring services with coordinated developments and uses of existing MTRs,
- preparing the future European landscape by implementing Jules Horowitz Reactor (JHR) and subsequent complementary research reactors.

The main outcome of the Joint Research activities, by building the MTR community, is the preparation of the implementation of future joint irradiation programs with common tools and practices, in an European MTR of second generation.

AEKI is the leader of the Supercritical Water loop work package, participates in the development of in-core corrosion instrumentation, in neutron screen development and transmutation studies.The input variables are not unique: rather, points in a set are candidates for a real input. To resolve this situation, a series of calculations is carried out and the variables to be limited are regarded as stochastic values. The question is, how should one to determine the number of runs to ensure assuring safety at a given (say 95%) level.

Results

The project has been launched in October 2006, thus it is in the preparation phase. The outcome of the supercritical water loop work package will be material performance data from corrosion tests in supercritical (high temperature and high pressure) water.

The preparation of high temperature electrochemical noise measurements has been started.

The task aimed at developing neutron screens optimised for given applications has been started with collecting the existing experience with neutron screens. The work represents AEKI’s contribution to the European MTR-I3 project

Methods

The manufacturing practices of irradiation devices in different MTRs will be collected and used for implementing transnational access to irradiation facilities.

The corrosion tests will be evaluated by electrochemical and surface analytical techniques. Monte-Carlo and deterministic neutron transport calculations will be used for contributing to neutron screen development and transmutation studies.

The project is also aimed at running the experiments and calculations in co-operation with partner European laboratories.

Remaining work

The project is just started, all of the above activities will be performed during the next three years.

Irradiation effects on tungsten alloys

FERENC GILLEMOT, MÁRTA HORVÁTH, ÁKOS HORVÁTH, GÁBOR ÚRI

Objective

The present and future fusion devices will operate at high temperatures. The diverter surface of these devices has to withstand the heat of a beam carrying out the pollution from the hot plasma. The load on the diverter surface can be as high as 10 MW/m^2, and the diverter should withstand the neutron irradiation and erosion by the pollution flow from the plasma. One group of the candidate materials for diverters are the tungsten alloys. The pure tungsten is a very brittle material. Lanthanum alloying increases the toughness and reduces the erosion rate of the tungsten. The objective of the present study is to collect information on the irradiation toughness of the tungsten.

Results

The unirradiated tungsten-lanthanum alloy has considerable toughness. After 0.3 dpa irradiation the toughness values widely scattered. Some samples have shown still considerable toughness (but much less than in as received conditions) others behaved in brittle manner. Scanning electron microscopy results have shown that the brittle fracture initiated at polluted zones (supposed tungsten oxide inclusions) in the tungsten alloy. Irradiation also reduced the electrical conductivity. The 0.3 dpa irradiation is much less than the end-of-life service irradiation of a future fusion device, but most of the damaging mechanism (segregations, precipitations etc.) occurs at the beginning of the irradiation, consequently low fluence irradiations can provide very useful information for material development and for material ageing models.

Methods

2 mm diameter tungsten rods have been tested in as received and irradiated state. Irradiation has been performed in the Budapest research reactor in the BAGIRA device, in nitrogen-helium atmosphere. The irradiation temperature is 65 and 280 °C. In as received conditions and after irradiation the electrical conductivity and mechanical properties (tensile and bending tests) have been measured. The fractured surfaces have been tested by scanning electron microscope.

Remaining work

The present project is finished. Final report has been sent to EFDA. AEKI is planning to continue the study of irradiation effects on tungsten alloys: high temperature irradiation and the effect of production technology will be the subject of the further research.

IFMIF material properties handbook

FERENC GILLEMOT

Objective

The neutron irradiation fluence of future fusion devices at the end of life will be in the range of 30-150 dpa, which is two orders higher than the fluences of present fission reactors. (Similar load is expected at the fourth generation fission reactors). At present, only a few high flux material testing reactors can irradiate material samples to this high fluence range, but the irradiation spectra, the occurrence of helium bubbles, and effect of transmutations differ considerably between in research reactors and fusion devices. To test materials in realistic fusion environments, a special spallation source IFMIF is designed. Material Properties Handbook (MPH) for the designers and regulators of ITER (the first real fusion device which is under construction) has been elaborated during the previous years by a group of material specialists. The purpose of the project was to survey the applicability of the ITER MPH to IFMIF design.

Results

Several structural materials of ITER and IFMIF will be the same, and the environmental conditions are also similar. The chapters of ITER MPH dealing with 316 LN stainless steel, Euroferr, lithium can directly be used at the design work of IFMIF.

Methods

The design of IFMIF has been studied and the ageing mechanism of the IFMIF structural materials identified. The description of the relevant ageing mechanisms has been checked in the ITER MPH and the list of missing data have been reported.

Remaining work

The project is finished. Final report has been sent to EFDA.

Environment, radioactive waste, health physics

The mission of AEKI underlines the importance of research of the environment. In this spirit, our activity involves development of monitoring and analytical methods. Modern analytical methods are applicable to analyse environment polution in air, water, and soil.

As to health physics, health consequences of radiation, analysis of the coolant in the power plant primary circuit, the adsorption of sission products, testing of atmospheric dispersion codes. In accordance with the principle of preparedness, we model the environment consequences of a nuclear accident. Furthermore, refinement of analitical methods, research of conseqences of inhaled aerosols are among the topics to be reported below.

Air quality measurements at Budapest Airport

VERONIKA GROMA, BÁLINT ALFÖLDY, JÁNOS OSÁN AND SZABINA TÖRÖK

Objective

Air transport is a very important sector of today's economy, boosting it and creating a great number of jobs, mainly in the immediate environment of the airports. Consequently, many European and American airports develop their infrastructure, build new runways, passenger and cargo terminals. The volume of traffic of Budapest Ferihegy Airport expanded by 26 % from 2003 to 2004, more than half a million airplanes crossed the Hungarian air space every year. Landing airplanes carried more than 8 million passengers in 2006. As a consequence, the yearly emission of carbon monoxide and nitrogen oxide is equal to or ,depending on the traffic load, exceeds several thousand tons. Notwithstanding, the environmental impact of airport traffic in Budapest is monitored continuously only in the field of noise pollution.

The main objective of the project is the establishment of a system for monitoring the airport air quality for gaseous and particulate pollutants.

Results

During winter and spring measurement campaignes the concentration values for all measured pollutants (CO, NOX, SO2, O3, PM10, PM2.5) were significantly lower at the airport than concentrations measured in downtown and suburban areas of Budapest. In addition, since the temporal variations are different in the dimension of hour averages, we could distinguish between airport and non-airport related traffic sources. The best correlation with aircraft traffic data was found for NOX concentration. Usually this is the only component which is monitored at large international airports (Frankfurt, Heathrow). Limiting value exceedence was observed only in PM10 concentrations around handling and terminal areas, where service activity is most remarkable. Accordingly, special aerosol measurements were performed. First, high resolution size fractioned (30 clusters between 0,25 and 32 μm) sampling was performed next to the runway. With X-ray analysis of the samples, the origin of the particles could be identified; the aerosols originating either from the engine, either suspended from the ground or being result of the wearing away of wheels during the touch down. The chemical composition of size fractioned aerosol samples were analyzed, whereby high Zn, S and Cu concentrations were detected in some particles - also next to the runway - denoting the presence of tire debris and particles eroded from the brake.

Method

The pollutants were mostly measured by non-intrusive optical methods. Particulate mass and number distribution with very high temporal resolution, ionic and other inorganic compounds as well as soot particles were determined by laboratory instruments.

References

G. SCHUREMANN, K. SCHAEFER, S. EMEIS, S. TOROK, V. GROMA, Application of Bayesian Inverse Methods to Determine Emission Strengths on the Airport Budapest, Proceedings of CEM conference, Zurich 2007.

B. ALFÖLDY, J. OSÁN, Z. TÓTH, S. EMEIS, K. SHAEFER, Aerosol optical depth, aerosol composition and air pollution during summer and winter conditions in Budapest, Athmospheric Environment, in press

Remaining work

After full evaluation of the measurement data the sites of permanent monitoring stations will be selected. In 2007 a research aircraft will be commissioned for vertical profile measurement of the same pollutants.

Geographical extension of the impact pathway approach for assessment of environmental damages in the energy sector

JÁNOS OSÁN AND SZABINA TÖRÖK

Objective

The accurate assessment of external costs related to environmental effects of electricity generation is very important for energy strategies of EU countries. A standardized *EcoSense* model was developed in the '90s for calculation of external costs of electricity generation that is based on the impact pathway approach and is capable of determining the monetary values of damages. This model is currently used evaluation of the energy sector in the EU15 countries. This approach requires multidisciplinary expertise and detailed data for pollution sources and receptors. The *NEEDS* Integrated Project (New Energy Externalities Development for Sustainability) aims to evaluate the full costs and benefits (i.e. direct and external) of energy policies and of future energy systems, both at the level of individual countries and for the enlarged EU as a whole. For this reason a geographical extension of the model is needed in order to cover Newly Associated Countries, other Central and Eastern European Countries as well as Mediterranean Partner Countries. Calculation of negative energy externalities has already been carried out in some of the countries under examination, although in limited scope, with various methods and high variability in quality. In order to apply the impact pathway approach for these countries and to take into account the damages occurring in countries other than the country of the emission source, the database for environmental receptors had to be extended and updated.

Results

The EcoSense model considers human morbidity and mortality, crops yield loss, as well as damages in materials and in ecosystems as environmental damages that can be taken into account as effects of atmospheric emissions to receptors. Based on calculations for the EU15 countries and for Hungary, more than 90 % of the environmental external costs are related to human health effects; therefore the most important receptor data to be updated is the population density. Since the EcoSense model uses a receptor grid and population as well as agricultural data, which are usually available on the basis of administrative units, the data collection was not an easy task. Due to the association to EU, the statistical and administrative structure of several new member states has been recently changed. Based on several years of experience, our research group could advise the Central Eastern European and Mediterranean Partner Countries in checking availability of population, agricultural and materials data. In order to obtain a standardized database, a data template was prepared, filled with Hungarian data as an example and circulated among the partners. Digital maps of the administrative units collected from the partners were verified according to the new EU nomenclature, and combined to one map covering the whole region (Europe and Northern Africa) under study allowing of the conversion of data to the required receptor grid.

Method

The population and agricultural data for Hungary were obtained from the Hungarian Central Statistical Office. The codes of the new administrative regions in the enlarged EU, as well as some crops data were obtained from the EUROSTAT REGIO database.

References

J. OSÁN, S. TÖRÖK: Environmental external costs of the Hungarian electricity generation sector, Submitted to MVM Közlemények

Remaining work

After the extension and update of the receptor database, the assessment of externalities of the Hungarian electricity generation sector will be performed using the impact pathway approach through case studies. Fossil, nuclear, biomass and wind fuel cycles will be modeled using emission data for year 2005. Cost-benefit analysis of emission reduction technologies as well as fuel change will also be addressed.

SINAC: simulator software for interactive modeling of environmental consequences of nuclear accidents

ISTVÁN NÉMETH

Objective

The SINAC program system was developed to follow the consequences of radioactive releases of a (hypothetical) nuclear accident. Atmospheric dispersion, plume depletion by dry-out and wash-out, cloudshine and groundshine doses, dose commitments from inhalation and ingestion, early and late health effects are computed. Effects of the introduction of countermeasures are taken into account.

Results

The SINAC Version 6 program system was installed at the Hungarian Atomic Energy Authority in 2000. Further developments were carried out in 2006.

The program works under WINDOWS operation system. Results (path of the plume, contaminated area, doses, health effects) are displayed on maps with spectacular symbols, and/or in tables showing the computed values. Quantities are computed for about 500 segments (defined by polar or Cartesian coordinates) and for 64 settlements located in the environment of Paks nuclear power plant.

The latest version of the program computes the gamma dose rate, the surface contamination and the air concentration at the measuring points of the national monitoring system. The time dependence of the measured and the computed values can be displayed and compared.

The emission data can be specified by the user, the source term from the CERTA nuclear emergency centre can be specified in every simulation hour. Twenty built-in scenarios can be defined for the different types of nuclear accident. Automatic runs can be defined to compare the results with and without countermeasures.

The system contains the EURO DISPERSE long distance atmospheric dispersion computing module, which can be used for calculating the dispersion of the radioactive release originating from a source outside the country.

The computed data can be displayed on OTAB maps (version 2.5) by ARCVIEW GIS 3.3 geographic information system. The results can be saved on disk in text file format and can be read by Microsoft Word and Excel.

The SINAC Version 10 can read the stack data from the Paks NPP. Input data for different types of source terms (e.g. RODOS data) take into account the precipitation type; and the results are compiled with regard to the new foodchain model calculations.

Intercomparison of atmospheric dispersion codes and possibilities of their validation

SÁNDOR DEME, EDIT LÁNG, ISTVÁN NÉMETH, TAMÁS PÁZMÁNDI, LÁSZLÓ SÁGI

Objective

There are several computer codes developed in the frame of various projects. Those codes are used to determine the environmental doses due to atmospheric release of radionuclides. PC CREAM,which is used for estimating the consequences of routine releases, PC COSYMA, which is suitable for performing calculations for safety analysis reports have been developed in the frame of EU projects. The code InterRAS is used widespread at US NRC to estimate doses in accidental situations. The aim of this project was the intercomparison of the results of the mentioned codes.

Methods

Since comparison of the results is essential, we have elaborated method for the intercomparison of different codes using test cases. The basic test case for the intercomparison was atmospheric release of ^{88}Kr, ^{137}Cs and ^{131}I to the environment at the level of 100 m and at Pasquill D dispersion category, dry and rainy weather, at a wind speed of 9 m/s at release height, and investigations for sensitivity study were carried out as well.

The calculations were focused on estimating air and ground contamination and on doses for different exposure pathways (cloud shine, ground shine and inhalation) for two exposure periods (7 days and 50 years) and for receptor points at distances of 1, 2, 3, 5 and 10 km.

Requirements for and possibilities of the validation of results for any model parameters were worked out. It was pointed out, that faultless, experimental validation requests well known release of radionuclides at definite meteorological conditions and monitoring the environmental radiation parameters for time periods up to one hour, however comparison of the calculations could be advantageous as well.

Results

Some results of the calculations for 1 GBq 131I release in case of dry weather at 1 km distance from the stack are given in table. The figures show the calculation results to be in good agreement with PC COSYMA and InterRAS codes. As the PC CREAM code is for the calculation of long-term release (using one year meteorology data and the results averaged for 16 sectors), estimated consequences here are slightly moderated.

Table 1. Effective doses in Sv for 50 years of exposure time

Dose type	Dispersion code and dose calculation		
	PC CREAM	PC COSYMA	InterRAS
Cloud shine	$1{,}4\bullet10^{-11}$	$7{,}3\bullet10^{-11}$	$5{,}4\bullet10^{-11}$
Ground shine	$9{,}1\bullet10^{-10}$	$4{,}2\bullet10^{-09}$	$4{,}2\bullet10^{-09}$
Inhalation	$1{,}5\bullet10^{-09}$	$3{,}5\bullet10^{-08}$	$2{,}9\bullet10^{-08}$

Minimum detectable releases of several radionuclides using telemetric environmental monitoring system of Paks NPP were estimated. It was pointed out, that detection limits of the measured radiation parameters are higher with 3-4 orders of magnitude than data expected for emission due to normal operation.

Remaining work

As the next step the comparison will be completed using the new code MACCS 2 (developed by US NRC).

Evaluation of activities during the dismantling and reconstruction of the Budapest Research Reactor in 1986-1991 from clearance levels point of view

TAMÁS PÁZMÁNDI

Objective

Radioactive waste is produced during the generation of nuclear power and the use of radioactive materials in industry, research and medicine. The European Commission estimates that about one third of the approximately 150 power reactors currently operating in the European Union will need to be shut down by 2025. This will result in the need to dismantle, decontaminate and demolish these nuclear facilities as well as to undertake processing, conditioning and disposal of nuclear waste and spent fuel.

The importance of the safe management of radioactive waste for the protection of human health and the environment has long been recognized. Large amounts of very low level metal and concrete waste from nuclear facilities, especially in the decommissioning stage, makes clearance an interesting option for reducing the requirement of final disposal facility volume.

Methods

No decommissioning has taken place in Hungary so far and the earliest year when a decommissioning project will start in Hungary is 2023, but this date will probably be extended further on. The legal framework for decommissioning originates from the Act on Atomic Energy, passed by the Hungarian Parliament in 1996 as Act CXVI. From that time on, a structure of governmental and ministerial decrees and regulatory documents has been elaborated. Detailed decommissioning plans are available for NPP Paks and preliminary plans have been submitted to the authority by the two research reactors. Detailed cost estimation for the decommissioning of NPP Paks has been elaborated; the other nuclear installations have only rough cost estimates based on international experiences without taking into account the specific situation of the respective nuclear facilities (technical, economic, legal and social circumstances).

The course of revising, updating and extending these plans goes on continuously, as a part of the Periodic Safety Reviews of the installations.

Results

The option of clearance in different countries is presently applied in accordance with the national legislation and practice. The need for internationally recommended nuclide specific clearance levels has been increasingly evident, as the decommissioning of nuclear installations has become more extensive, generating large amounts of metal and concrete waste with very low activity. Recently, under the auspices of the International Atomic Energy Agency, a set of international recommendations on waste management has been published. Principles and requirements have been set out and principles and standards of the European Union shall also be applied.

Large amounts of very low level waste should be excluded from the regulatory process, because they present such a low risk that control by regulatory processes would be a waste of resources.

The authority and the licensee have to be prepared technically and legally for this task. When defining and optimizing the clearance levels, not only the radiation protection of the workers and the population, and the general environmental protection has to be taken to account, but the amount of radioactive waste and the expenses also have to be considered. The domestic regulations also have to be taken into consideration.

The research in the field of preparation of the regulation and implementation of clearance is the first part of a long-term project. In this report the experiences gained during the reconstruction of the Budapest Research Reactor in 1986-1991 are reviewed.

During the elaboration of the clearance process, it is recommended to take the experiences of the reconstruction of the Budapest Research Reactor into consideration, and to collate the concepts with the practice established there.

Remaining work

Task remaining for 2007 and 2008 are further evaluation of the methods and making recommendations and suggestions for the authorities.

Comparison among different decommissioning funds methodologies for nuclear installations

TAMÁS PÁZMÁNDI

Objective

Hungary has a transparent and declared policy for the further application of nuclear energy. It will play a key role in the energy production of Hungary in the 21st century. No figures have been announced so far on the exact percentage of nuclear energy in electricity production; however, it is likely that its present share of approximately 40 % will not decrease. Decommissioning is considered as an inherent part of nuclear activities. It is of paramount importance that the funding of these decommissioning activities will be adequate and available when needed in order to avoid affecting negatively the safety of EU citizens. Nuclear operators are expected to accumulate all the necessary funds during the operating life of facilities.
International survey has been performed. The conclusion is that there are significant differences in the operation, governance, investment and accessibility of the existing funds across the EU. Member States oversee different regimes for estimating, collecting and managing decommissioning costs.

Methods

In Hungary, the sole financial basis for all future decommissioning activities is the state-owned Central Nuclear Financial Fund (CNFF). The Hungarian Atomic Energy Authority (HAEA) is the authentic institution responsible for managing the CNFF. The CNFF is a segregated asset of the Hungarian State Treasury. Its increase is based on two main factors: the annual contribution of NPP Paks from its income (growth of the capital) and the annual accrual (yield of the capital) assigned by the annual budget and prescribed by the Act on Atomic Energy. In order to provide sufficient information a series of HAEA and Public Agency for Radioactive Waste Management (PURAM) reports and publications were studied and additional information was obtained from interviews with HAEA experts.

Results

A comprehensive assessment of the financial consequences and risks of the decommissioning funds from governance, accounting, valuation and investment perspectives has been undertaken in the course of this study. The report arrived at the following conclusions:

- The *Polluter pays principle* for decommissioning is widely accepted and has to be the fundamental basis of the granting an operating license.
- The discussions on decommissioning funds focused on nuclear power plants.
- Cost estimates are subject to high degree of risks and uncertainties, differences in reported cost estimates occur due to varying discounting mechanisms and the timing of dismantling.
- In most countries there are only limited rights for the public to access information on decommissioning costs and funds.
- Many operating companies and governments are satisfied with the current situation and have concerns towards an EU harmonization process of nuclear decommissioning financing.

Based on the findings of the report a number of recommendations are made on how to ensure that adequate funds are available when necessary. These recommendations are made to Member States and for actions that could be undertaken now on the European level. Furthermore, the report makes suggestions on how further harmonization could be achieved on the EU level if necessary. Along with these recommendations there are suggestions for information sharing and reporting that should be undertaken across the EU to increase transparency.

Reference

Comparison among different decommissioning funds methodologies for nuclear installations, on behalf of the European Commission Directorate-General Energy and Transport, H2, Service Contract TREN/05/NUCL/S07.55436

Experimental analysis of the biological effects of low-dose alpha and neutron radiations

JULIANNA SZABÓ, JÓZSEF PÁLFALVI, IMRE BALÁSHÁZY, ISTVÁN FEHÉR

Objective

This work is part of the low dose program of the Institute. One way of studying the health effects of low dose ionising radiation is to carry out biological experiments on monolayers of living cell cultures. The main objective of this work is to contribute to the understanding of the relation between the macroscopically determinable ″low dose″ of high LET (linear energy transfer) radiation and the biological damage of cells caused by the ionising particles. The experimental work is focused on two radiation fields: (i) alpha and (ii) neutron radiations. The investigated biological endpoints were cell mutation, bystander effect and adaptive response. Preparation of alpha-sources was also an objective.

Results

New, high activity 210Po alpha sources were prepared for studying the biological effects of alpha irradiation on living cell cultures.

The neutron dosimetrical studies which accompanied the cell culture irradiation experiments performed at the Biological Irradiation Facility of the Budapest Research Reactor demonstrated that a significant part of the dose (16%) comes from the wall of the cell holder flasks. 1 mGy neutron dose means that every 8th cell is hit by neutron generated protons.

In collaboration with the Division of Cell Biology of the "Frédéric Joliot-Curie" National Research Institute for Radiobiology and Radiohygiene, the adaptive response induced by low doses of alpha and neutron radiation was investigated by the determination of mutation frequencies. The adaptive response could be detected in both cases. The bystander effect and its role in the formation of mutagenic adaptive response were studied by the irradiated cell conditioned medium (ICCM) technique. Based on the results, adaptive response was also observed in bystander cells (see Fig.1.).

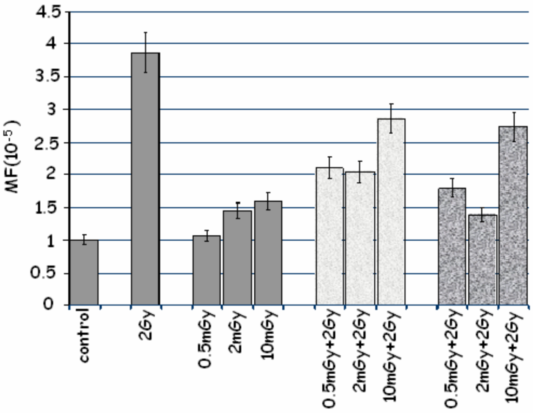


Fig.1. *Mutation induction in CHO cells by conditioned medium from neutron irradiated cells. Mutagenic adaptive response was detected when cells were incubated for 3 h ( ), and 5 h ( ) after priming doses from irradiated medium ( ) and challenge of 2Gy gamma rays.*

Methods

The ^{210}Po alpha sources were produced by the $^{209}Bi\ (n,\gamma)\ ^{210}Bi \rightarrow {}^{210}Po$ reaction by irradiating natural Bi in the Budapest Research Reactor. (It must be mentioned that the misuse of the ^{210}Po isotope was completely excluded.)

The microdosimetrical studies related to the neutron irradiation experiments were performed by mounting solid-state nuclear track detectors simulating the cells on the wall of the cell holder flasks. After irradiation the track detectors were etched in the standard way and then evaluated by an image analyzer coupled to an optical microscope.

In the cell irradiation experiments, the studied biological endpoints were mutation induction, bystander effect and adaptation. To assess the radiation sensitivity of the directly irradiated and bystander cells, the clonogenic survivals and the mutation frequencies were determined. The bystander effect was measured by irradiated cell-conditioned medium (ICCM) technique. Adaptation was measured by applying at first low dose priming dose and a few hours later 2 Gy challenge dose.

Reference

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Analysis of primary cellular consequences of low doses of radon inhalation by a CFD based microdosimetric model

IMRE BALÁSHÁZY, ISTVÁN SZŐKE, ÁRPÁD FARKAS

Objective

This work is part of the low dose program of the Institute. Based on epidemiological data, a linear dependency can be detected between the health effects of ionizing radiation and the radiation dose in case of exposures to high doses up to 1 Sv. Cell biology experiments executed in in vitro circumstances demonstrated that low dose radiation may also cause biological mutations. Statistical surveys were unable to draw a reliable dose-effect curve below 100 mSv. During the year of 2006, our efforts were focused on the application of the biophysical mechanism based microdosimetric model developed in the previous years. This complex CFD based model was applied for realistic exposure conditions to investigate the primary cellular consequences of low doses of inhaled radon progenies. The results were then compared to analyse the exposure – biological outcome dependency.

Results

- • The variation of the microdosimetric parameters that have a key role in tumour formation were studied as a function of time spent in the atmosphere of the New Mexico uranium mines.
- • Cell inactivation and cell transformation probabilities of all types of exposed tracheo-bronchial cells in the central airways were calculated at different exposure levels of the New Mexico uranium mines atmosphere.
- • The curves between exposure levels and cell inactivation/transformation probabilities were determined in the range of 1 to 100 inhalations of a worker in the New Mexico uranium mines.

Methods

The complex, biophysical mechanism based microdosimetric model described in the previous report was utilized to calculate the primary cellular consequences of radon inhalation. Based on the concentration of radio aerosols in the atmosphere and the time of exposure, this mathematical approach computes the distribution of hit probabilities, the distribution of cell death and cell transformation probabilities along the epithelium of the central airways. Calculations were made for 1, 10 and 100 inhalations in the athmosphere of the New Mexico uranium mines. The data of aerosol composition, the breathing parameters and the parameters describing the cell structure of the epithelium were obtained from the literature.

References

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BALÁSHÁZY I., FARKAS Á., SZŐKE I.: Biophysical mechanism based modelling of the biological effects of radon inhalation. 3rd Hungarian Radon Forum, Veszprém, Hungary, 26 April 2006. Book of Proceedings 14-20, Pannon Univ. Publisher, Veszprém, Hungary, ISBN 963 6420 99 8, (In Hungarian).

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FARKAS Á., HOFMANN W., BALÁSHÁZY I., SZŐKE I.: CFD as a tool in risk assessment of inhaled radon progenies. Rad. Prot. Dos. Vol. 122, Nos. 1-4, 537-539. (2006)

Remaining work

A new extrapolation technique is under development to allow much longer exposure times.

Modelling the health effects of aerosols

IMRE BALÁSHÁZY, ÁRPÁD FARKAS, ISTVÁN SZŐKE, LORÁND GERGELY TATÁR, BALÁZS GERGELY MADAS, ERIK DOBOS*, KINGA KARLINGER**, TIBOR KERÉNYI**, ALPÁR HORVÁTH**

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Objective

State-of-the-art numerical modelling techniques have been developed for the characterisation of the health consequences of inhaled aerosols. Our specific objectives were: numerical generation of the cellular structure of the bronchial epithelium, computation of radiation burden in the central airways of the radon progenies which are carried by the mucus from the deeper regions of the bronchial airways, characterisation of the effects of different airway and lung diseases on the distribution of particle deposition within the human respiratory tract, and, computation of the deposition probabilities and distributions of inspirable bacteria and viruses.

Results

- The cellular structure of the central airway epithelium was constructed numerically. The cell size distributions and shapes of the four different cell types were realistic.
- The contribution of the radon progenies, carried from the deeper regions, to the dose absorbed by the central airways may be a few times higher than the burden of the primary deposition.
- Airway constrictions, blockages and tumours may significantly increase local deposition.
- Asthma, bronchitis and emphysema may considerably modify the deposition distributions of inhaled particles. In case of these diseases, the extrathoracic and the tracheo-bronchial depositions are extensively higher than in healthy subject.
- Description of the deposition mechanism of inspirable micro organisms in the respiratory system may help in the elaboration of prevention strategies against respiratory infections.

Methods

The FLUENT CFD code was applied to describe regional airflows and deposition patterns in healthy and diseased airways e.g. in airway constrictions, and around airway blockages and tumours.

The latest version of the stochastic lung deposition model was improved to be able to describe deposition in lungs with airway diseases e.g. asthma, bronchitis, emphysema.

References

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Study of the oxidation state of uranium in individual particles of abandoned mine tailings

ANITA ALSECZ, ZSOLT KERNER, JÁNOS OSÁN, SZABINA TÖRÖK

Objective

The uranium production in the Mecsek Mountains was abandoned in 1997. After that, a project started in order to decrease and eliminate the damages caused mostly by the waste ore tailings and mill tailings. The majority of the tailings ponds were recultivated and covered with clay and loess. The recultivation of tailings pond No. I is not yet finished. Nuclear spectroscopic methods are well established in order to study the problem concerning radionuclides. However, very limited information is available on the distribution and chemical form of uranium in the tailings sludge. The aim of this work was to determine the oxidation state of uranium in the tailings and examine the correlation with environmental parameters. The original ore was also analyzed.

Results

The pH of the mine tailings was around 7.6 for all samples, the redox potential was between 70 and 430 mV vs. saturated calomel electrode. This variance of the values indicates that the chemical state of the tailings is inhomogeneous.

Uranium in the mine tailings particles were 50–80 % in the less mobile U(IV) form and 20–50 % in the more mobile U(VI) form. In the original ore, the oxidation state of the uranium was found to vary in a wide range in the measured individual particles. Uranium in the tailings particles was found to be in more reduced form than main forms in the original ore, which was expected from the uranium mining technology. Depleted uranium particles separated from soil contained uranium in even more reduced form, mostly as U(IV), resulting in lower environmental risk than in case of mine tailings particles.

Methods

Tailings samples were collected from the uncovered part of tailings pond No. I, dried in air and prepared on Nuclepore polycarbonate filter. The pH and redox potential of the tailings was measured in the sampling hole. Fragments of the uranium ore provided by the mine remediation company were prepared similarly.

In order to identify the uranium containing particles in tailings samples efficiently, the particle-containing sample substrate was attached to a Solid State Nuclear Track Detector (SSNTD). The sample–detector system was irradiated with fast neutrons in the Budapest Research Reactor. Based on the tracks on the detector, the localization of U-rich particles was possible with a precision of ± 100 μm.

The oxidation state of uranium in the pre-selected individual U-rich particles was investigated at the micro-fluorescence beam line L at HASYLAB using μ-XANES. UO2, UO3 and U3O8 particles prepared and tested in house were used as standards. The spectra of the particles separated from tailings and ore samples were evaluated using linear combination of standard spectra of UO2, UO3 and U3O8.

References

A. ALSECZ, J. OSÁN, J. PÁLFALVI, S. TÖRÖK, G. FALKENBERG: Non-destructive x-ray and neutron techniques for assessing environmental impact of uranium at the premises of remediated Hungarian uranium mine, European Conference on X-Ray Spectrometry, EXRS 2006, Paris, France, 19-23 June, Book of abstract, P4-22, 171

Remaining work

For reliable correlation between the redox potential and the oxidation state more sample analyzes is needed.

Aerosol sampling in the QUENCH-11 experiment

ANNA PINTÉR CSORDÁS, IMRE NAGY, PÉTER WINDBERG, LAJOS MATUS, MIHÁLY KUNSTÁR, NÓRA VÉR, ILDIKÓ PUMMER, ZOLTÁN HÓZER

Objective

The experiment QUENCH-11 on boil-off and subsequent quenching was dedicated to investigate degraded core reflood situations with rather low mass flow rate, which may occur if pumps cease and/or if low make-up systems were activated in the course of accident management. The test conditions simulated a depressurised plant sequence in which the core would be essentially dried-out and with a limited steam flow due to boiling of residual water in contact with the hot structures in the lower plenum. AEKI experts participated in QUENCH-11 experiment at FZK, Karlsruhe with aerosol sampling in order to characterise the release of aerosols during such an accident. The mean objective of this project was the examination of the role of oxidising atmosphere on the release of aerosols during the early phase of severe accidents comparing the QUENCH-11 test with boil-off in steam to QUENCH-10 test with high temperature air ingress.

Results

The largest amount of aerosols in QUENCH-11 was found on the last impactor that was activated at the maximum temperature of ~2000 ^{o}C. It indicated that the aerosol release correlated with the oxidation process. In accordance with the results of the SEM+EDX and that of the SSMS, the escaped aerosol contained different elements depending on the phase of the experiment. However it was typical for the samples of this experiment, that more amounts of "impurity elements" such as Al, Si, Ca, Mg, K, Na were present than in the QUENCH-10 experiment. The elemental analysis pointed out most of the released aerosol to originate from the structural elements of the facility and not from the fuel bundle. Contrary to the QUENCH-10 test no Zr was found by SSMS on the impactor samples, only in the pocket of the Ni plate.

The most important conclusions based on the comparison of QUENCH-10 test carried out in air atmosphere and QUENCH-11 performed in steam from the point of view of aerosol release are the followings:

- reduced amount of Zr containing aerosols was observed in steam conditions (much less in QUENCH-11 than in QUENCH-10),
- in QUENCH-10 the steel components were found most frequently in the aerosols,
- in QUENCH-11 W, Mo and other impurities were the dominating elements in the released aerosols.

Methods

Aerosol sampling was carried out during the QUENCH-11 test using a valve operated impactor system and a Ni plate placed in the outlet section of the facility. Six impactors were activated during the oxidation period of the test at low water level. Four impactors were used in the quenching phase. The Ni plate with a small pocket collected aerosol particles continuously during the test duration. After the test the samples were delivered to AEKI from FZK, and the post-test examination was carried out in Hungary. The examination included

- Mass gain measurements,
- SEM and EDX analysis,
- Application mass spectrometric methods.

Reference

ANNA PINTÉR CSORDÁS, IMRE NAGY, PÉTER WINDBERG, MIHÁLY KUNSTÁR, LAJOS MATUS, NÓRA VÉR, ILDIKÓ PUMMER, ZOLTÁN HÓZER: Post test examinations of aerosols formed in QUENCH-11 experiment, AEKI-FRL-2006-113-01/01

Remaining work

In the year 2007, a new QUENCH test will be carried out at FZK with silver-indium-cadmium control rod. Aerosol sampling will be performed again by AEKI experts.

Adsorption of uranium and fission products on the structural materials of the primary circuit

ZSOLT KERNER, MIHÁLY KUNSTÁR, GABOR NAGY, RÉKA RÉPÁNSZKI, IBOLYA SZIKLAI-LÁSZLÓ, NÓRA VÉR

Objective

The research aimed at studying the accumulation of some possible contaminating ions on the surfaces of structural materials (zirconium alloys, 08H18N10T stainless steel and magnetite) of the Paks NPP in the 20-60 °C temperature range. Iodide and cesium adsorption on three different surfaces (vacuum deposited zirconium and magnetite, sputtered 08H18N10T stainless steel) was investigated by electrochemical quartz crystal microbalance. Experiments were carried out in electrolyte solution similar to the cooling water and the adsorbing ion was added stepwise. Cerium and uranium accumulation and desorption were measured on Zr2.5%Nb and 08H18N10T surfaces by immersing the metal to cooling water containing the adsorbing material. ^{141}Ce isotope prepared in the Budapest Research Reactor was used and measured by gamma spectrometry. The amount of uranium was measured by mass spectroscopy.

Results

Quartz crystal microbalance measurements show that the maximum coverage of the iodide and cesium ions on zirconium, stainless steel and magnetite surfaces are a monolayer. Langmuir isotherm can be used to describe the concentration dependence of the adsorbed quantity. The temperature dependence in the 20-60 °C range is not significant. Iodide ions adsorb stronger than cesium ions and lower solution concentration is needed to reach the maximum coverage on steel surface than on zirconium.

Mass spectrometry results show that the quantity of adsorbed uranium decreases by increasing temperature. Stainless steel adsorbs 2-3 times more uranium than the zirconium alloy at room temperature. Most of the uranium desorbs in boric acid solution within one week.

Cerium ions adsorb strongly on zirconium alloy and stainless steel and only a small portion desorbs during 3 days in distilled water. The adsorbed quantity increases by increasing temperature.

Methods

Electrochemical quartz crystal microbalance (EQCM), mass spectrometry, gamma-spectrometry

References

G. NAGY, A. SOMOGYI., G. PATEK, T. PINTÉR, R. SCHILLER: Estimation of the amount of surface contamination of a water cooled nuclear reactor by cooling water analysis, Annals of Nuclear Energy, 34, 496-500 (2007)

G. NAGY, Á. HORVÁTH, ZS. KERNER, R. SCHILLER, T. PINTÉR, G. PATEK: The amount of active deposit on the surfaces of the primary circuit of a water cooled nuclear reactor after a non-nuclear incident, Trombay Symposium on Radiation & Photochemistry (TSRP-2006) Volume I. Invited Talks, pp.1-4 ISBN No: 81-88513-19-9

R. RÉPÁNSZKI, ZS. KERNER , G. NAGY: Adsorption of fission products on stainless steel and zirconium, Proc. of 9th international conference of Fundamentals of Adsorption, Giardini Naxos, Italy, May 20-25, 2007

Remaining work

Further experiments are planned to investigate the temperature dependence of the adsorption of ionic species up to 300 °C and the accumulation of their colloid forms.

Scanning electron microscopy applied for various particles

ANNA CSORDÁS PINTÉR AND JÁNOS OSÁN

Objective

Special specimen preparation method was developed for extracting nano-sized secondary phases of Zr1Nb alloys and studying them by highly developed SEM + EDX. Conventional SEM + EDX were applied for various aerosol samples and sediments.

Results

Identification and elemental analysis of secondary phase particles of the Zr1Nb alloys are important tasks in several projects. Their small size and the presence of the Zr-Nb containing matrix make difficult the elemental analysis by energy dispersive microanalysis (EDX). Therefore extraction of these phases would be desirable. We adapt and slightly modify an extraction replica producing method from the literature having several steps of specimen preparation, which need time and skill. Nano-sized particles and their aggregates were successfully extracted from the metallic samples. They could be revealed and analysed mainly with an up-to-date SEM + EDX, due to their nano-size. The size of these secondary phase particles and aggregates was found to be depending on the producing technology of the samples. Larger sized (100-250 nm) nano-particles were detected in the raw material, while smaller ones (30-80 nm) were found in a sample heat treatment at 1000 oC for 300 sec (Figure 1). EDX analysis of various nano-particles has shown that -Zr phase particles were extracted and their Nb content was 5-8 wt % for the raw material, while 8-12 wt % for the sample heat treated at 1000 oC.

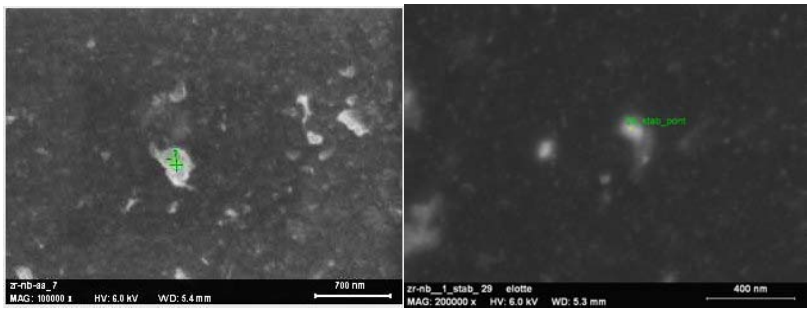

Figure 1 Nano-sized β-Zr particles extracted from Zr1Nb raw material (left side image) and from a sample hea- treated at 1000^{o}C (right side image)

Uranium-rich particles were successfully identified by SEM-EDX in the tailing material of the Hungarian uranium mine, at an abundance of about1 1 %. The typical size of the U-rich particles was found to be 2-3 m. An average uranium content of 8 m % was calculated by a Monte Carlo simulation method developed earlier. Similar method was applied for individual particles containing Mn and Zn at elevated concentrations (on the average 19.1 and 14.9 wt %, respectively), identified in sediment samples originating from a polluted Hungarian river2.

Methods

State of the art SEM with Schottky field emission gun and ultra-thin-window X-ray detector
Energy dispersive X-ray analysis (EDX) with a system suitable for light element analysis

Reference

A. PINTÉR CSORDÁS, A. ALSECZ AND J. OSÁN: Application of Thin-Window Electron Probe Microanalysis Combined with Iterative Simulations for Small Sized Secondary Phase Particles and Sediments, Proc. of the 16th International Microscopy Congress, Sapporo, Japan, Vol. 3, 2006, No. IMC16-00000106

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Epiboron neutron activation analysis: an option to analyze unfavorable matrices

RÉKA SZŐKE, IBOLYA SZIKLAI-LÁSZLÓ

Objective

When applying instrumental NAA, induced activities of major components can occasionally dominate the resulting spectrum hampering the detection of trace elements in a sample to be investigated. A solution to the above mentioned problem is the application of selective irradiation using boron filters. The objective of this study was to measure the boron activation ratios and to calculate the improvement factors of trace and minor elements in glass wool.

Results

In an epithermal irradiation, the activities of all nuclides having high resonances in the epithermal cross-sections are strongly enhanced with respect to the activities of those nuclides having an 1/v cross section. The activation ratios were calculated for individual (n,γ) reactions using the specific activities of samples induced in bare and filtered irradiations:

$$R_B = A_{spec}(bare) / A_{spec}(filtered)$$

The improvement achieved by filtered irradiations may be characterized by the so-called improvement factor IFB:

$$IF_B = \sqrt{R_B(2) / R_B(1)}$$

where RB (1) and RB (2) refers to the determinant and dominant interfering element, respectively. Table 1 lists the nuclear data RB, along with its standard deviation, for the radionuclides used, and the calculated IFB for interferences of 24Na and 46Sc for some trace and minor elements in glass fibers. Na and Sc were chosen as typical interferences always present in geological samples.

Table 1. Nuclear data, B ratios and improvement

Element	Nuclear reaction	Nuclear data		Activation ratio	Improvement factor	
		$\bar{E}_r$, eV	I_0/σ_0	RB ± SD (%)	IFB(Na)	IFB(Sc)
As	$^{75}As(n,\gamma)^{76}As$	106	13.6	6.2 (2.2)	2.38	3.91
Co	$^{59}Co(n,\gamma)^{60}Co$	136	1.99	44.9 (3.7)	0.34	0.54
Cs	$^{133}Cs(n,\gamma)^{134}Cs$	9.27	13.2	15.0 (3.5)	1.06	1.73
Eu	$^{151}Eu(n,\gamma)^{152}Eu$	0.448	0.87	440 (4.5)	0.03	0.06
	$^{153}Eu(n,\gamma)^{154}Eu$	5.8	5.66	39.6 (0.3)	0.37	0.61
La	$^{139}La(n,\gamma)^{140}La$	76	1.24	69.0 (2.0)	0.22	0.35
Rb	$^{85}Rb(n,\gamma)^{86}Rb$	839	14.8	4.7 (2.9)	3.15	5.16
Sb	$^{121}Sb(n,\gamma)^{122}Sb$	13.1	33.0	7.6 (5)	1.94	3.20
	$^{123}Sb(n,\gamma)^{124}Sb$	28.2	28.8	6.0 (3.1)	2.46	4.08
Th	$^{232}Th(n,\gamma)^{233}Th/^{233}Pa$	54.4	11.5	8.8 (1.8)	1.68	2.75
Zn	$^{64}Zn(n,\gamma)^{65}Zn$	2560	1.91	30.4 (6.0)	0.49	0.79
	$^{68}Zn(n,\gamma)^{69m}Zn$	590	3.19	18.1	0.81	1.34
Zr	$^{94}Zr(n,\gamma)^{95}Zr$	6260	5.31	9.3 (1.7)	1.60	2.63
	$^{96}Zr(n,\gamma)^{97}Zr/^{97m}Nb$	338	251.6	1.4 (1.9)	10.6	17.3
U	$^{238}U(n,\gamma)^{239}U/^{239}Np$	16.9	103.4	4.4 (1.2)	3.36	5.51

Methods

k_0-based NAA using a boron carbide (B4C) filter and a high-rate gamma-ray spectrometry with Loss-Free Counting. Following boron-covered irradiations all samples were measured after a minimum delay (~1 hour), following bare irradiations the first measurements of the samples were carried out only after 4-5 days.

Conclusions

To test the applicability of this technique standard reference materials were analyzed by ENAA, and for comparison, parallel samples were measured by INAA. The concentrations of most elements agreed within 3%. The reliability of epiboron NAA is comparable to conventional thermal neutron activation analysis, and the number of the elements that can be determined instrumentally in biological and geological materials is considerably extended.

Reference

R. SZŐKE, I. SZIKLAI-LÁSZLÓ: Epiboron NAA: an option to analyze unfavorable matrices. J Radioanal. Nucl Chem., accepted.

SEM and microanalytical studies of various particles originated from NPP Paks

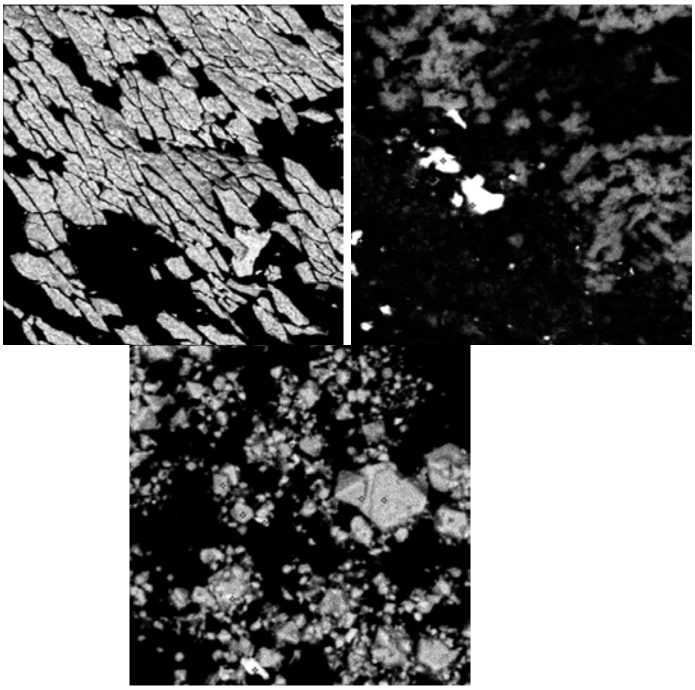
ANNA PINTÉR CSORDÁS

Objective

The objective of this work was to study the elemental composition and the morphology of various samples taken partly by filtering of the primary coolant of NPP Units 1 and 3 and partly taken from some steam generator tubes of Units 1, 3 and 4. It was also aimed to compare the surface of a stainless steel material with the one treated by the coolant of the cooling pond.

Results

Samples filtered from the primary coolant of Units 1 and 3 contained mainly iron-nickel oxide and iron oxide particles with rather small amount of chromium. Besides some zirconium oxide grains were also detected. *Differences in the elemental composition and in the morphology of the corrosion particles were found among some samples from steam generator tubes of various reactor units.* The sample 4B2G originated from Unit 4 contained well developed magnetite crystallites (see Figure 1). In Unit 4, there was no problem with corrosion (no decontamination). At other Units like Unit 1 and 3 several decontamination procedures were performed. Samples taken from some steam generator tubes of Units 1 and 3 contained *chromium-iron-nickel - and iron-nickel-chromium oxides in form of layers and particles* [1]. Layers enriched in chromium might originate from the inner corrosion layer.



240 μm x 240 μm sample area 60 μm x 60 μm sample area

1BL56 sample - Unit 1 3B2G sample - Unit 3 4B2G sample - Unit 4

Figure 1. Digital SEM images of steam generator tubes of Units 1, 3 and 4, respectively

On the surface of a stainless steel (SS) sample treated by the coolant of the cooling pond, corrosion particles were found and a few tenth of mass % uranium could be detected at two corroded area. It seemed that uanium could be bound to the SS container of the pond [2].

Methods

Energy dispersive X-ray microanalysis (EDX) was the main method for studying the elemental composition of various samples. Their morphology were investigated by scanning electron microscopy (SEM).

Remaining work

The objective of the present project was fulfilled.

Reference

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[2] N. VAJDA, ZS. MOLNÁR, T. PINTÉR, A. PINTÉR CSORDÁS, ZS. STEFÁNKA, K. VARGA: Analysis of radioactive particles of NPP origin, lecture in: NATO Advanced Research Workshop, Yalta, 7-11 May, 2007.

Water chemistry at the Budapest Research Reactor

IBOLYA SZIKLAI LÁSZLÓ, RÉKA SZŐKE

Objective

The control of reactor water parameters such as conductivity, pH, radioactivity and their interrelationship with reactor circuit materials are important factors in controlling the corrosion. Another parameter that is carefully monitored and controlled in most nuclear facilities is chloride content. The reason for maintaining the chloride ion concentration at the minimum level is that several forms of corrosion are accelerated by chloride ion, and, of greatest concern is chloride stress corrosion. The aim of the study was to monitor the water quality through sampling and measuring the concentrations of selected halogens, alkali and alkali earth metals, corrosion products and other impurities in the water samples of the Budapest Research Reactor.

Results

Water samples were collected at six different sampling points of the reactor: primary and secondary water, fuel storage “at-reactor-pool”, fuel storage “away-from-reactor”, auxiliary water, water purification system. The following water chemistry parameters were measured: corrosion causing species like chloride and other halides like Br and I, Na, K and alkaline earth metals: Mg, Ca and Ba, soluble and insoluble corrosion products: Al, Fe, Cr, Mn, Cu, Co and other impurities: Mg, La, Sb, Ta, V and Zn. Water chemistry data were stored in PC base-data storage system along with other technological parameters such as temperature, pH, conductivity, water purification system operation and flow rate. Figure 1. summarizes the annual variations of chloride concentration in the primary coolant. Comparison of the values to the authority’s limit [1] shows that all data, except in one sample, were below the limit.

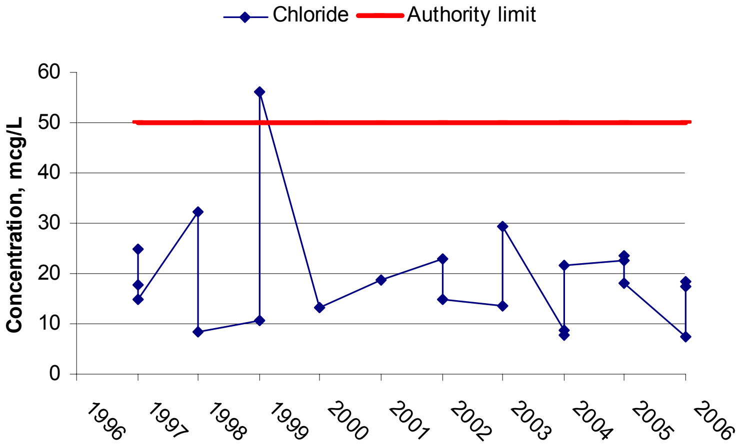

***Fig. 1**. Maximum concentrations of chloride in the primary coolant at the Budapest Research Reactor, 1997-2006*

Method

INAA, high resolution gamma-spectrometry with HPGe detector, Hypermet-PC spectrum deconvolution software and INAACNC program for concentration computation. The chloride content was determined by a pre-concentration procedure.

References

[1] Final Safety Report of the Budapest Research Reactor, Budapest, 2004.

Remaining work

To continue data acquisition, processing and evaluation, to build up a data base for corrosion characteristics of the specimens and the measured water parameters in order to establish a reliable trend analysis.

Space Activities

One of Hungary's earliest and most significant workshops in space research was the KFKI Atomic Energy Research Institute (KFKI AEKI). For the last four decades more than 40, Hungarian made measuring apparatus developed in KFKI AEKI have been flown on board geophysical rockets, satellites, deep space probes, manned spacecraft and space stations.

There were a number of main projects in the field of space research during 2006 dealt with in our institute. The first project was to maintain the *Pille* dosemeter systems on board the International Space Station (ISS); another one was to follow the flight and test on orbit the plasma (SPM) and dust (DIM) detectors developed for *Philae*, the lander of the *Rosetta* comet probe. Important programs were to carry out measurements on board the ISS *(BRADOZ* and *SORD*/*MATROSHKA* experiments) by neutron track detectors and to participate in ESA's *EuTEF-DOSTEL* experiment on the ISS *Columbus* module. As the first benefit of KFKI AEKI's experience in space technics was a commercial portable TL dosemeter system *(PorTL)* developed on the base of the *Pille* system. Equipment made with AEKI participation and launched to space are summarized below:

Date of start (Y.M.D.)	Space vehicle	Equipment	Name
1970.11.28.	Vertical-1	plastic foil micrometeorite trap	Tanja
1971.08.20.	Vertical-2	plastic foil micrometeorite trap	Tanja
1972.04.07.	Intercosmos-6	plastic foil micrometeorite trap	Tanja
1974.10.31.	Intercosmos-12	micrometeorite detector	K-1-3
1975.12.11.	Intercosmos-14	micrometeorite detector	K-1-3
1977.09.24.	Intercosmos-17	micrometeorite detector	K-1-4
1977.10.25.	Vertical-6	retarding potential analysers	LAM-1, -2
1978.10.30.	Prognoz-7	solar wind analyser	D-173 B
1978.11.03.	Vertical-7	retarding potential analyser	LAM-1
1979.03.12.	Progress-5 / Salyut-6	TLD (thermoluminescent dosimeter) capsules	Integral
1979.05.13.	Progress-6 / Salyut-6	TLD capsules	Integral
1980.05.26.	Soyuz-36 / Salyut-6	TLD capsules	Integral
1980.05.26.	Soyuz-36 / Salyut-6	TLD reader + dosimeters	Pille
1980.09.18.	Soyuz-38	TLD capsules	Integral
1981.12.21.	Vertical-10	retarding potential analysers	LAM-2
1983.03.02.	Cosmos-1443 / Salyut-7	TLD reader + dosimeters	Pille
1984.10.06.	Space Shuttle STS-41G	TLD reader + dosimeters	Pille-S
1984.12.15.	VEGA-1	particle analyser	Plasmag
1984.12.21.	VEGA-2	particle analyser	Plasmag
1986.06.19.	Cosmos-1760	ultrathin TLDs	-
1987.03.03.	Progress-28 / Mir	TLD dosimeters	Pille
1988.07.07.	Phobos-1	particle spectrometer	HARP
1988.07.07.	Phobos-1	particle analyser	TAUS
1988.07.12.	Phobos-2	particle spectrometer	HARP
1988.07.12.	Phobos-2	particle analyser	TAUS
1995.07.22.	Progress-M28 / MIR	TLD reader + dosimeters	Pille'95
1995.10.08.	Progress-M29 / MIR	TLD reader + dosimeters	Pille'95
1997.01.12.	Space Shuttle STS-81 / MIR	TLD reader + dosimeters	Pille'96
2001.02.24.	Progress-M44 / ISS	neutron dosimeter	BRADOS-1

2001.10.31.	back: Soyuz-TM32	track detectors	
2001.03.08.	Space Shuttle STS-102 / ISS	TLD reader + dosimeters	Pille'97
2003.02.02. *2003.10.28.*	Progress-M47 / ISS back: Soyuz-TMA2	neutron dosimeter track detectors	BRADOS-3
2003.09.28.	Progress-M48 / ISS	TLD reader + dosimeters	Pille-MKS
2004.01.29. 2005.10.11.	Progress-M1-11 / ISS back: Soyuz-TMA6	neutron dosimeter track detectors	Matroshka-I
2004.03.02.	Rosetta / Philae	simple plasma monitor dust impact monitor	ROMAP / SPM SESAME / DIM
2005.02.28. 2005.10.11.	Progress-M52 / ISS back: Soyuz-TMA6	neutron dosimeter track detectors	BRADOS-5
2005.05.31. 2005.06.16.	Foton-M2 / Szojuz U	neutron dosimeter track detectors, TLDs	BIOPAN-5
2005.12.21. 2006.12.22.	Progress-M55 / ISS back: Space Shuttle STS-106	neutron dosimeter track detectors	Matroshka-2A

Cosmic ray detection in the space: the BRADOS-5 project

JULIANNA SZABÓ, JÓZSEF PÁLFALVI, BEÁTA DUDÁS

Objective

The complex radiation field inside the International Space Station (ISS), as well as the dose received by its crew was studied for several years in the BRADOS (1-5) projects organised by the Institute for Biomedical Problems (IBMP, Moscow) with the participation of different laboratories. Based on the experiences gained during the previous BRADOS (1 and 3) experiments a 3 axes solid state nuclear track detector (SSNTD) stack was designed for the BRADOS 5 project. By this new assembly the directional distribution of the primary galactic cosmic rays and of the secondary neutron radiation inside the ISS can be studied. The detectors were flying on the ISS between 28 February and 11 October 2005 for 225 days.

Results

Applying the calibration functions (see Methods) the LET spectra were determined. As shown in Fig. 1, the LET spectra measured on the 6 different sides of the BRADOS 5 box are practically the same, which means that no remarkable directional dependence could be identified in the radiation field. The absorbed dose rate and dose equivalent rate values were also calculated: 24 μGy/d and 131 μSv/d, respectively. The latter one is somewhat lower than in case of the previous BRADOS experiments (see References), taking into account the stacks placed on the same panels. (The BRADOS 5 box was placed on the ISS panel 10, near window 6, behind thick shielding.)

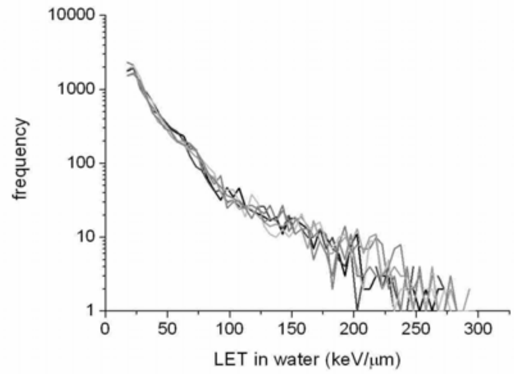

Fig. 4*: The corrected LET spectra measured on the 6 different sides of the BRADOS 5 box.*

Methods

The detector stack was composed of an inner block made of 13 polyallyl-diglicol carbonate (PADC) sheets alternating with Makrofol sheets, which was surrounded on each lateral position by a stack of 2 Makrofol and 2 PADC sheets. The entire assembly was wrapped in a 105 μm thick Al box. To visualize the latent tracks induced on the PADC sheets by various ionizing particles, a multiple etching (2, 6, 15 h etching time) procedure was applied. The evaluation of the detectors was done by a sophisticated image analyzer system, which automatically recognizes the tracks and measures their geometrical and optical parameters which allow to obtain the LET spectra. The dose values were based on these spectra.

For calibration purposes an identical detector stack was exposed at the HIMAC accelerator in Japan. The result of these studies is the calibration function which makes relationship between the track etch rate ratio (V) obtained from the track parameter measurements and the LET of the incident particle obtained either by measurements or by calculations using the SRIM2003 code. Different calibration functions were determined for each etching step. During the evaluation of the unknown space detectors beside the calibration functions a LET dependent combined correction factor was also used.

References

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Matroshka – I project

BEÁTA DUDÁS, JÓZSEF PÁLFALVI, JULIANNA SZABÓ

Objective

This project studies the dose distribution of the cosmic rays inside an anthropomorphic phantom, called Matroshka, placed outside of the International Space Station (ISS) simulating space walk. The phantom consists of several segments made of human tissue equivalent plastic, imitating the astronauts. The dose caused by the radiation field can be investigated by tissue equivalent solid state nuclear track detectors (SSNTD), which can be positioned in any segment of the body, including the major organs. The Matroshka phantom containing our SSNTD stacks was flying altogether for 622 days, staying 539 days outside of the ISS.

Results

Generally, the doses were found to be lower than it was expected. The absorbed dose measured on the poncho of the phantom was only slightly higher than the values inside the organs and within the ISS. The individual dose values of each detector located within a stack were averaged and the standard deviation was calculated. The contribution of the manually evaluated, long range galactic cosmic ray tracks to the absorbed dose and dose equivalent remained always well below 1% of the dose caused by other particles and measured by the automated image analyzer.

Table 1: Absorbed dose, dose equivalent and quality factor values obtained by the Matroshka-I SSNTDs (doses from manually evaluated tracks are not included).

Position	Absorbed Dose, D [mGy]	Dose equivalent, H [mSv]	Quality factor, Q
Reference	17.05 ± 0.59	79.34 ± 1.96	4.66 ± 0.2
Lung	14.38 ± 1.24	76.89 ± 3.33	5.37 ± 0.39
Kidney	13.58 ± 1.7	71.72 ± 4.7	5.32 ± 0.35
Poncho-3	18.31 ± 1.89	108.43 ± 4.5	5.96 ± 0.54
Poncho-4	19.46 ± 0.81	114.64 ± 8.05	5.89 ± 0.26

Methods

The AEKI provided five SSNTD stacks: two assemblies for the organ doses were placed inside the lung and the kidney, two stacks were placed in the pockets of the poncho, and one stack stayed as reference inside the ISS. The stacks were composed of polyallyl-diglicol carbonate (PADC) sheets and several charged particle converters (as Al, Ti, steel and polycarbonate sheets). The system was calibrated at high energy particle accelerators - BNL, USA; HIMAC, Japan -, also using 1 MeV protons (Van de Graaf) and a collimated 210Po alpha source. The detectors, after chemical etching, were investigated by an image analyzer. Those tracks which were not recognized automatically by the image analyzer were investigated by eye. These were long range HZE particles with high incident angle. From the track parameters obtained from the measurements the linear energy transfer (LET) spectra were determined. Based on the LET spectra the absorbed dose, the dose equivalent and the averaged quality factor were deduced for each detector surface within a stack.

Remaining work

The Matroshka project was extended with phases II A & B and our team is participating in these experiments as well, in 2007 and 2008.

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Pille, a portable TLD system on the ISS

ISTVÁN APÁTHY, ANTAL CSŐKE, SÁNDOR DEME, ISTVÁN FEHÉR, PÉTER SZÁNTÓ

Objective

Measurement of dose due to ionizing radiation of cosmic rays during space flights as well as at environmental monitoring on the Earth is mainly based on thermoluminescent dosimetry. This method offers considerable advantages because of its high precision, wide range, rigidity etc. At the same time, its application involves the disadvantage that the readout of dosimeters can be done only in laboratories equipped with relatively large and usually heavy TLD readers. This means that it is not possible to read out the dosimeters during a space mission on board, and an uncertainty occurs at terrestrial environmental measurements caused by the extra dose collected during the transport of the dosimeter from and to the laboratory. Our small, portable and space qualified TLD reader *Pille* (in English, butterfly), originally developed at the end of the 1970's, capable of evaluating the TL dosemeters at the place of exposure (*in situ* TLD reader) eliminates the above mentioned disadvantages.

A new implementation of the dosimetry system helps researchers understand the space radiation environment on board the ISS (International Space Station) and can also be used for personal dosimetry.

Results

The Pille'MKS TLD system for the Russian segment of the ISS (Zvezda) was launched and successfully activated in 2003. Since then, it is continuously applied for dosimetry mapping and the routine and EVA (Extra-vehicular Activity) individual dosimetry of astronauts/cosmonauts as part of the service system. Seven dosemeters are located at different places of the ISS and read out monthly by the cosmonaut. Two dosimeters are dedicated to EVAs and one dosimeter is permanently inserted in the Pille reader and read out automatically every 90 minutes, providing high resolution dosimetric data.

In 2006, during the time of the flight of Expedition 13 and 14 about 4000 measurements were performed, among others the extra doses of astronauts during their EVAs have been detected.

Moreover, a series of calibrations have been carried out on the Pille dosemeters at the CERF Neutron Reference Field (CERN, Switzerland) in the frame of the CERF ICCHIBAN project.

Methods

The TLDs record the total absorbed dose from ionising radiation. As a kind of passive detector, they accumulate a signal over the course of the exposure. This signal is then measured during the readout of the TLD. At readout, the TLD is heated while giving off visible light proportional to the dose, that is amplified, measured and evaluated by our compact, lightweight reader. A PC communicating with the reader via a standard serial line serves as a user interface enabling detailed analysis of the dose values and parameters belonging to them.

Remaining work

Evaluating and interpreting continously on-board data, maintaining the on-board system; continuing terrestrial calibrations by high energy heavy particles; participating in ESA's "Matroshka-2A" experiment on board the ISS by Pille dosemeters to investigate the dose distribution on the surface and inside an antropomorph phantom.

References

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P.SZÁNTÓ, YU.A.AKATOV, I.APÁTHY, V.V.ARKHANGELSKY, L.BODNÁR, S.DEME, I.FEHÉR, A.HIRN, T.PÁZMÁNDI, G.REITZ, M.TYURIN, P.VINOGRADOV: TL dose measurements on board the Russian segment of the ISS by the "Pille" TL system during Expedition 13 and 14, 16th IAA Humans in Space Symposium, Beijing, Cina, accepted for publication in Astra Astronautica, 2007

Developing a 3D silicon detector telescope (TriTel) for several space experiments

ISTVÁN APÁTHY, SÁNDOR DEME, ATTILA HIRN, JÓZSEF PÁLFALVI, TAMÁS PÁZMÁNDI, PÉTER SZÁNTÓ

Objective

Astronauts working and living on spacecrafts are only partly protected from the cosmic radiation by the wall of the spacecraft. The dose of the ionizing radiation they are exposed to has to be measured and taken into account as a source of risk, as it is at least two orders of magnitude over the dose levels typically encountered on the Earth's surface due to natural radiation. Since the radiation field in space is a mixture of different particles differing in energy and varying with time, the dose equivalent significantly differs from the absorbed dose. Recently used equipments are suitable for measuring certain radiation field parameters, but a combination of different measurements and calculations is required to characterize the radiation field in terms of dose equivalent. The objective of this project is to develop a complex dosimetry system capable of measuring the required parameters of the radiation field.

TriTel is a 3D silicon detector telescope and it is capable of determining the absorbed dose and the dose equivalent, which takes into account the radiation quality as well. Hence it gives a measure of the stochastic biological effectiveness of the incoming particles. The three-axis arrangement is going to eliminate mostly the highly anisotropic sensitivity of the recently used one-dimensional silicon telescopes.

Results

Optimization of the parameters – among others the improvement of the signal-to-noise ratio – of the analog signal processing chain (preamplifier – shaping circuit – amplifiers) continued. Measurements started with light emitting diodes (LEDs) later to be mounted into the telescopes of TriTel in order to test the signal processing circuits and the coincidence circuits, too. The pulse shaping circuits were tested with a 210Po alpha source and with test charges produced with a pulse generator. Further optimization and the calibration of the system are under way.

The data processing algorithms have been developed. Monte Carlo simulations have been used in order to study the dead time of the multiplexed analog-digital converter (ADC) of TriTel. In order to give a rough estimation of the expected fluxes and spectra of protons and electrons in orbit, calculations were made with the Space Environment Information System (SPENVIS) online tool.

Methods for future measurements and calibration of TriTel in high energy accelerators have been elaborated, as well.

Remaining work and flight opportunities

The main objective of the following years will be the manufacturing of the engineering and later the flight models.

Several flight opportunities have already come in sight:

- ISS TriTel: In the near future a TriTel will be installed into the International Space Station (ISS) in cooperation with the Institute of Biomedical Problems (IBMP, Moscow). Every 24 hours the multi-channel analyzer provides 12 different primary spectra (for each axis two gated and two full spectra, one taken in the South Atlantic Anomaly and the other in the rest of the orbit). At the end of the one-day-long period all the spectra are sent to the human interface unit of TriTel onboard the Russian module. Spectra will be stored on a memory card which will be returned to Earth every six months for further on-ground evaluation.
- A menu-driven graphical interface of the human interface unit will help the astronauts to change the settings, download data or update the on-board software.
- TriTel SURE: Onboard the Columbus module of the ISS within the framework of the European SURE (ISS: a Unique Research Infrastructure) program, which has priority in Hungarian space research. SURE is a 4-year ESA project funded by the European

Commission under the Sixth Framework Programme with priority access given to the twelve new EU Member States.

- The objective of our project will be to obtain a more accurate description of the radiation environment in terms of the absorbed and dose equivalent inside the European Columbus module of the ISS. Therefore a complex dosimetry system comprising a 3D silicon detector telescope and three passive detector stacks will be assembled in our Institute. The TriTel SURE will be similar to the ISS TriTel but both of its modules will be located inside ISS. The passive detector stacks – comprising several layers of track detectors and a layer of thermoluminescent (TL) detectors – will be mounted on the detector unit of TriTel oriented as the axes of a Cartesian co-ordinate system.
- TriTel-S: Within the framework of the Student Space Exploration and Technology Initiative (SSETI), a more compact version of TriTel (TriTel-S) will be operated onboard the European Student Earth Orbiter (ESEO) in Geostacionary Transfer Orbit (GTO). The orbit will be a highly eccentric one with a perigee of 250 km, an apogee of 35950 km, an inclination of 7° and a period of 10.55 hours, therefore the spacecraft is going to cross the Van Allen belts twice an orbit.

The higher particle fluxes and the more serious mass, volume and power constraints demanded a much smaller and more compact design than in case of the ISS version.

The maximum amount of measurement data that can be produced by TriTel-S and that can be transferred to the ground station per orbit is 120 kB which means that data reduction is inevitable. The memory inside TriTel-S will serve only for temporary storage of the data.

The launch of the orbiter is expected in December 2008 from French Guiana. The device may be a precursor of a subsequent version of TriTel planned for a future Mars probe, too.

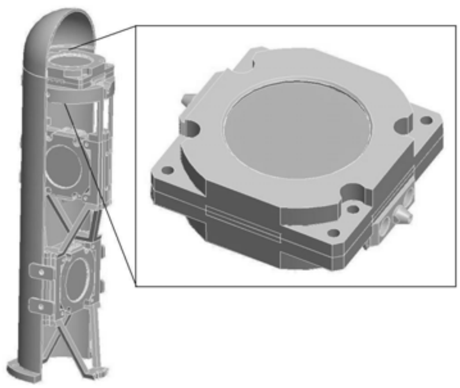

Figure 1.: ISS TriTel

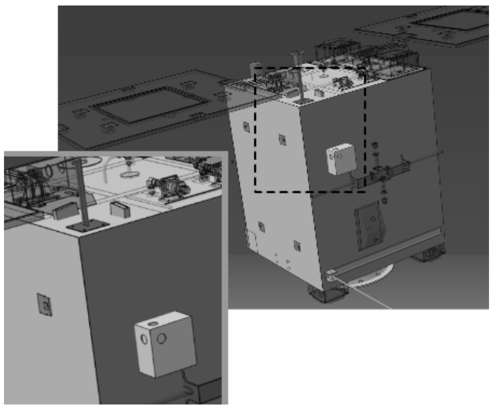

Figure 2.: TriTel-S onboard SSETI ESEO

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PorTL - a commercial portable dosemeter system

ISTVÁN APÁTHY, ANTAL CSŐKE, SÁNDOR DEME, ISTVÁN FEHÉR, PÉTER SZÁNTÓ

Objective

For reducing the radiobiological risk of the civilian population, especially of health workers and the workers in the nuclear industry, their exposure to radiation must be kept below specified limits. For this purpose, frequent and accurate dose measurements are increasingly important. Among the generally accepted tools, thermoluminescent detectors (TLDs) are commonly used for environmental monitoring and for personal dosimetry, for medical dosimetry and for dosimetry in nuclear facilities.

The TLD method involves the disadvantage that the detectors must be transported for evaluation to a laboratory equipped usually by a large, heavy and expensive TLD reader operated by qualified personal, considerably increasing the costs and delaying the achievement of the results. A portable, easy to handle and relatively inexpensive TLD reader suitable for reading out the TL dosemeters at the place of exposure ("in site TLD reader") provides the possibility to overcome the above-mentioned disadvantage. Till now, not a single up to date manufactured type was available on the market.

Results

Our institute - with the contribution of BL Electronics Bt. (Hungary) - has developed a new and unique TLD system (named „PorTL") containing a small, portable, battery powered and moderate price Reader for commercial use and a series of dosemeters fitted to it. The construction was based on our experience achieved by the unique "Pille" TLD system generations which has been successfully applied on board of spacecrafts and space stations since 1980. With regard to laboratory systems, PorTL has a number of advantages: it is small, light, portable and battery powered; all of the measured data and operational parameters are stored in the Reader itself; it is easy to handle, not requiring special qualification; the dosemeters are much more durable than those of most traditional systems; the dosemeters can be read out and evaluated at the place of exposure; the Reader can read out a dosemeter automatically in a preprogrammed cycle or initiated by a PC connected to it; the Reader and the dosemeters can be set and the measured data can be downloaded by a PC via a standard serial interface.

A number of PorTL systems are already in use i.e. at DESY (Hamburg, Germany), at NASA, at the Paks Hungarian Nuclear Power Plant, at DLR (German Aerospace Research Centre).

Methods

The TLD records the total absorbed dose from ionizing radiation. As a kind of passive detector, accumulates a signal over the course of the exposure. This signal is then measured in the readout phase. At readout, the TLD is heated while giving off visible light proportional to the dose, that is amplified, measured and evaluated by our compact, lightweight reader. A PC communicating with the reader via a standard serial line serves as a user interface enabling detailed analysis of the dose values and parameters belonging to them.

Remaining work

Besides the existing dosemeter types using Al2O3:C, 6LiF:Mg,Ti and 7LiF:Mg,Ti TL materials, we intend to develop a CaSO4:Dy dosemeter as well as a combined 6LiF/7LiF one to measure doses in mixed neutron-gamma radiation fields.

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Dust impact monitor, simple plasma monitor

ISTVÁN APÁTHY, ATTILA PÉTER

Objective

ESA's *Rosetta* spacecraft is the first mission designed to both orbit and land on a comet. During its trek to Comet 67P/Churyumov-Gerasimenko, *Rosetta* will make two excursions into the main asteroid belt and fly by two asteroids, Steins and Lutetia. After entering orbit around the comet, in 2014, *Rosetta* will release a small Lander named *Philae* onto the icy nucleus.

The objective of the mission is to study the origin of comets, and the relationship between cometary and interstellar material and its implications with regard to the origin of the Solar System.

KFKI AEKI is participating in two of the nine scientific experiments the Lander carries. The first one, *DIM (Dust Impact Monitor*) is a part of the SIP (Small Instrument Package) SESAME for determining the mechanical and electrical properties of the comet's surface; the second one, *SPM (Simple Plasma Monitor)* is a part of the SIP known as ROMAP which complements the plasma packages on board the *Rosetta Orbiter*. The *DIM* acoustic dust detector will investigate the material that impacts and falls on the nucleus of the comet, the *SPM* sensor is capable of measuring the major solar wind parameters as a function of the distance from the Sun.

Results

Since its launch in 2004 Rosetta is flying on its orbit to comet Churyumov-Gerasimenko, the fly lasts approximately 10 years. During this time, systematic in flight test campaigns for checking the service and scientific instrumentation of the Orbiter and the Lander were and will be fulfilled. All operations on the flying Lander are first tested on the identical Ground Reference Model (GRM). In 2006, the on-board checking campaigns during the so-called Cruise Phase went on. The evaluation of the housekeeping and scientific in flight data of SESAME and ROMAP achieved during the last active checkout proved the proper operation of both instruments. We were taking part in data archiving and planning the operation during descent and the first scientific sequence.

Methods

The piezoelectric arrangement (sensor) of DIM, located outside the Lander and on top of it, with active surfaces looking into three orthogonal directions, will sense the impacts of particles having energies in the range of 10-11 J ... 10-7 J. The sensor's electric output signals of broad dynamic range are amplified by wide-band logarithmic amplifiers. The characteristics of the impact signals (peak amplitudes, contact times, average) are measured by an appropriate electronic circuit, connected to the common Data Processing Unit of SESAME by a digital bus-system.

The SPM sensor is a type of electrostatic, hemispherical analyser having 2 ion channels and 1 electron channel and containing a Faraday cap as well. The energy range of the instrument is 0-12.6 keV for ions with a resolution of 3%, and 0-4.5 keV for electrons with a resolution of 10%; the field of view of the sensor is 140°x150° for ions and 8°x15° for electrons. The sensors of ROMAP are mounted on the end of a short boom and are coupled to a small DPU (Digital Processing Unit) to store data and control the power consumption in modes with reduced data rates.

Remaining work

To participate continuously in the onboard tests during the cruise phase of the flight; performing on-ground calibrations of the sensors and tests of the Ground Reference Model; taking part in data archiving and planning the operation during descent and the first scientific sequence on the cometary surface in the future too.

EuTEF-DOSTEL, a silicon detector telescope for the ISS

ISTVÁN APÁTHY, SÁNDOR DEME, TAMÁS PÁZMÁNDI

Objective

The *EuTEF* (European Technology Exposure Platform) contains 7 standard payloads with common subsystems and is mounted on the European ISS (International Space Station) module *Columbus* to be launched in 2007 depending on the modified flight schedule of the US Space Shuttle. One of the experiments in the first payload group to fly is *EuTEF-DOSTEL,* containing a silicon detector telescope for monitoring the temporal variation of the particle count rate, the dose rate, and the particle and linear energy transfer (LET) spectra in the external environment of the ISS.

The EuTEF-DOSTEL experiment is being developed and realized in the framework of an international co-operation under the leadership of DLR (Germany). The KFKI Atomic Energy Research Institute (KFKI AEKI) has two co-investigators (CoIs) participating in the scientific preparation of the experiment and is committed to following-up and evaluating its operation during flight and subsequently interpreting the data obtained.

Furthermore, KFKI AEKI is contributing to the development and the design of the instrument hardware, the development of the DOSTEL / EuTEF interface software as well as the on-ground software for data evaluation.

Results

In 2006, the software of DOSTEL was updated and the unit was delivered to CGS for integration. The check up procedure was completed successfully. The EuTEF-DOSTEL documentation was updated and the Acceptance Data Package was completed and delivered. A Ground Model (GM) of EuTEF-DOSTEL was built and delivered to ESTEC.

The Columbus module of ESA was transported and delivered to NASA in May, 2006. Its launch to the ISS is foreseen at the autumn of 2007.

Methods

The silicon detector telescope sensor (SDTS) of the instrument uses PIPS detectors and will continuously monitor temporal variation of the particle count rate, the dose rate, particle and LET spectra. Three modes of operation will be performed: a) single detector mode for dose measurements in the individual detectors, b) coincidence mode for the measurement of particle and LET spectra in the range 0.1 to 200 keV µm-1, c) single event mode. While modes a) and b) integrate over software defined time periods resulting in reasonable data reduction, the mode c) measurement will store the conversion results of the four detectors for each individual particle. This mode produces an immense amount of data and will only be activated during limited time periods.

LET spectra will be deduced from the energy deposit of coincident events in the PIPS detectors. Since the incidence angle of the particles is not measured, the energy deposit is converted into LET in silicon, as energy deposit/t , where t is the mean path length in the detector. The energy loss in water (tissue) relative to that in silicon is taken to be 1.23 , independent of the particle energy.

The silicon scintillator sensor of the experiment is designed to separately measure the dose of charged and neutral radiation in a tissue equivalent BC430. PIN photodiodes with a silicon thickness of 300 µm are used to build an almost closed shield of silicon detectors around the sensitive scintillator volume of 10.7x10.7x20 mm^3 in order to detect incoming charged particles. Four detectors are in optical contact with the scintillator and will measure the light proportional to the energy deposit in the scintillator as well as the direct energy deposit in the silicon detector when hit by charged particles. Neutrons are selected by a criterion requiring no signal in the two small anti-coincidence detectors and similar signals in the four light sensitive detectors. Any penetrating charged particle in one of the light sensitive detectors will significantly increase that signal compared to the pure light signals in the other ones. The energy spectrum of recoil protons from neutron interactions will be measured in the range 1-10 MeV. The ionizing part will be measured in the linear energy range from 0.1 to 200 keV μm^{-1}.

Remaining work

Tasks remaining for the year 2007 are to complete and update the specification of all commands, including parameters and possible command responses and telemetry for the EuTEF Instrument Mission Databases; to modify the EuTEF DHPU Application SW to implement the automatic instrument switch-off operation.

Effect of the radiation parameters on the signal of thermoluminescent dosimeters

ISTVÁN APÁTHY, SÁNDOR DEME, TAMÁS PÁZMÁNDI, PÉTER SZÁNTÓ

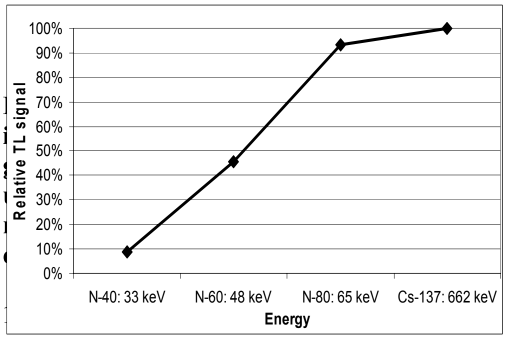
Objective

Dosimeter systems are used in many fields, for example in medical – diagnostic or therapeutic – applications of ionizing radiations and in nuclear industry. To perform accurate measurements, especially in case of neutron and heavy ion radiations, it is necessary to take the properties of the measured radiations into account. These radiation fields are usually very complex, they contain radiations of different energies. Thermoluminescent dosimetry (among other methods) is applicable for measuring the components of the mixed radiation fields, since the different thermoluminescent materials have different sensitivity to different types of radiations. Dosimeters made of different types of thermoluminescent materials are capable of measuring the components of the radiation field. The aim of this project is to analyze the influence of different type and energy of the radiation on the shape of the glow curves and the linearity of thermoluminescent dosimeter systems.

A part of the measurements has been performed under the frames of *TéT Croatian-Hungarian International cooperation programme.*

Methods

Measurements were carried out at several energies with different thermoluminescent materials in the Ruđer Bošković Institute, Zagreb and in the Hungarian Academy of Sciences KFKI Atomic Energy Research Institute. The Al_2O_3:C and LiF thermoluminescentThe dosimeters were irradiated with gamma and X-ray radiation at several energies. We used Al filters of different thickness for correction the energy dependence of the system. The lowest X-ray energy used was 33 keV and mixed neutron-gamma irradiation was also used.



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RFig. 1. Relative TL signals
5 mGy, Al_2O_3:C, 0.5 mm of Al filter

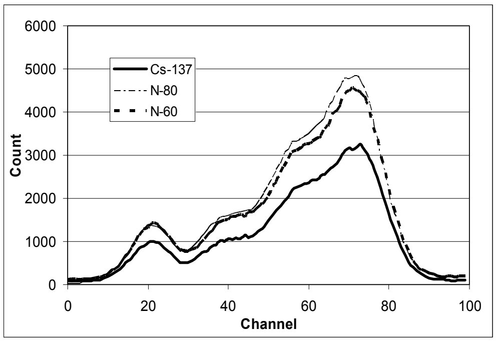

Figure 2: Glow curves at three energies
5 mGy, LiF, 0.5 mm of Al filter

Results

We calculated the angle of incidence and energy dependence of the dosimeters using different thickness of Al filters. The measurements have shown that the investigated PorTL thermoluminescent dosimetry system meets the requirement of the IEC-61066 and the ISO-4037 international standards for thermoluminescent dosimetry systems. With both of the used thermoluminescent materials – Al_2O_3:C and LiF – the system is applicable for environmental and personal dosimetry purposes.

Remaining work

Measurements in mixed radiation fields at several energies are planned in the following years.

Reference

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Process information, noise diagnostics and simulation

In 2006, the modernization of the VERONA core monitoring system was continued at Unit 3 and 4. The full-scope simulator was also equipped with the new VERONA in February 2006 in order to ensure appropriate operators' training. In June 2006, the final version of the new system was installed at Unit 4, previously the new version was running parallel to the old VERONA for a complete fuel cycle. Unit 4 was restarted after refueling in July and the reactor power was increased gradually to the target 108%. The 108% power level was achieved in September 2006; the new VERONA system was running properly during the whole power ascension programme and served reactor operators with reliable core information during this important period. Now Unit 4 is operated steadily at 108% level, producing 500 MW electrical outputs. In September – October 2006 the final version of the new system was also installed at Unit 3, after it was running parallel to the old version for a whole year. However, the power increase at Unit 3 will take place only in 2008. The work will be continued in 2007 by the installation of a new VERONA configuration at Unit 1 and by implementing a so-called remote VERONA-t system specially tailored to serve operation and maintenance personnel.

The installation of the new reactor noise diagnostics data acquisition system called PAZAR was started at Unit 3 in 2005. In 2006 Unit 4 was also equipped with a full-function version of the PAZAR system (previously here only a limited scope prototype system was operated). In 2006, the Factory Acceptance Tests of the Unit 1 configuration was performed successfully, as well. It is planned that by 2008 all Paks units will have a new, full-function PAZAR configuration.

The software development work related to the new Instructors' System of the Paks full-scope training simulator was finished by the end of 2005, including a very detailed Factory Acceptance Test. The Site Acceptance Test started in January 2006, and it was completed successfully by the end of February. Then a trial period followed, when simulator instructors were using the new system parallel with the old one. The new Instructors' System was taken into regular use during summer of 2006 and now it is the official system used during all operators' training sessions.

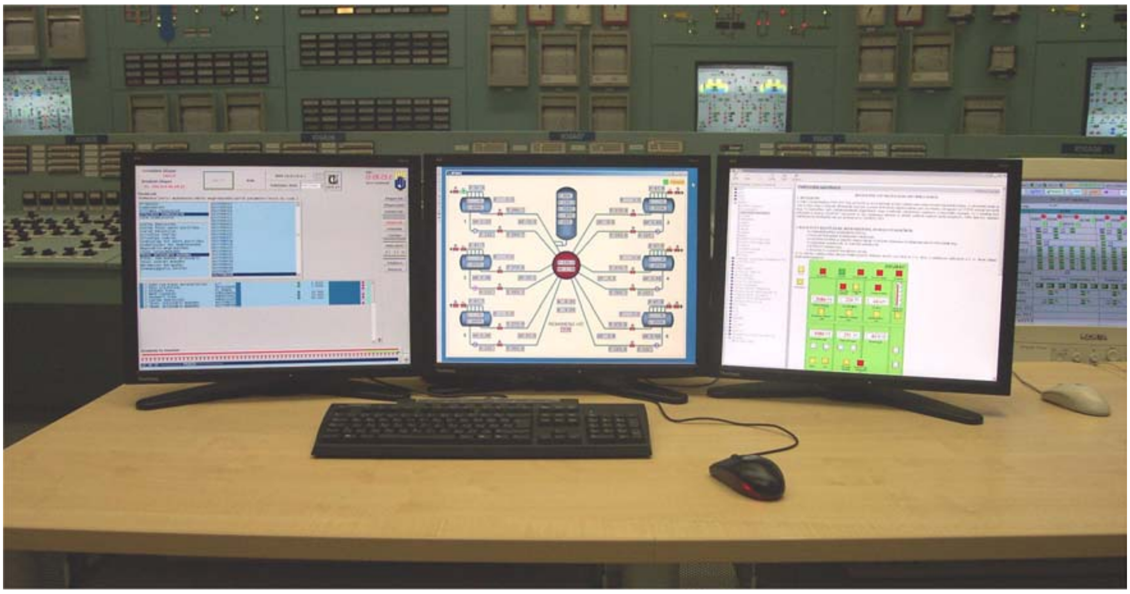

Fig. 1. View of the new instructors' system in the simulator Control Room

Supporting the power increase of the Paks NPP units by the new VERONA core monitoring system

CSABA MAJOR, CSABA HORVÁTH, ZOLTÁN KÁLYA[1], FERENC NÉMETH[1], ISTVÁN PÓS[1], MIKLÓS IGNITS[1], JÁNOS VÉGH

1 – Paks Nuclear Power Plant Ltd, H-7031 Paks, P.O.Box 71

Objective

In 2006, Paks NPP started the implementation of the core thermal power increase from the present 1375 MW to 1485 MW (108%) at Unit 4. Feasibility of the planned power increase had been proven in a series of studies, these were followed by the detailed design and licensing phases. Some changes were needed in the secondary circuit (e.g. in the turbine controller) and a new type of fuel (with slightly increased lattice pitch) was licensed in due time. Changes had to be introduced in other plant components, e.g. modified primary circuit pressure control, decreased hydroaccumulator pressure, etc. Safety margins and operation limits were kept at their previous values, i.e. all previous limits were valid also at 108%. However, increased core power and the new type of fuel required a more accurate and more frequent core analysis; therefore modernization of the VERONA core monitoring system was started in 2002. During the modernization of the software special attention was paid to those functions that were important to achieve a more accurate core monitoring required by the higher core power: more accurate and faster off-line and on-line reactor physics calculations, more reliable system architecture, high capacity and redundant data archive, enhanced human-machine interface. The first new VERONA configurations were installed at Unit 4 and 3 in 2005, then in June 2006 Unit 4 started the new fuel cycle with the new system fully commissioned. The proper operation of the new system was one of the conditions to start the power increase procedure outlined below.

Results

The Paks full-scope simulator was equipped with the new VERONA in February 2006 in order to ensure appropriate operators' training. In June 2006 the final version of the new core monitoring system was installed at Unit 4, previously the new version was running parallel to the old VERONA for a complete fuel cycle. Unit 4 was restarted after refueling in July and the reactor power was increased gradually to the target 108%. The 108% power level was achieved in September 2006; the new VERONA system was running properly during the whole power ascension program and served reactor operators with reliable core information during this important period. Now Unit 4 is operated at 108% level, producing 500 MW electrical outputs. Figure 1 shows the history of the power increase procedure at Unit 4, data were plotted from the VERONA archive. During autumn of 2006 the final version of the new system was also installed at Unit 3, after it was running parallel to the old version for a whole year. According to the plans of the Paks NPP Unit 3 will reach 108% power only in 2008.

Methods

The target 108% (1485 MW) thermal power was achieved at Unit 4 in a long and carefully organized program. The new fuel cycle was started in late June 2006, during the preceding long shutdown period all the required equipment modifications were carried out in the technology. After reaching the 100% (1375 MW) reactor power a detailed evaluation of core conditions was carried out, together with an accurate determination of primary and secondary side heat parameters. Evaluations have shown that the planned intermediate power level (104%) could be achieved safely, without violating operating limits. In the following one month dynamic tests were performed, in order to qualify the modified turbine controller and other equipment (see Figure 1). At the end of July 2006 the 104% intermediate power level was achieved successfully. At 104% a detailed evaluation of plant parameters was performed again to assess core conditions and the possibility to reach the final 108% power level. Results have shown that 108% could be achieved safely, but stationary 104% operation was continued during the next two months in order to collect further data. Finally the target 108% level was reached at the end of September 2006 (see Figure 1). Core conditions and reserves to limits were evaluated again and results have shown that safety limits were not violated even at 108% and Unit 4 could be operated at 1485 MW safely. The new VERONA was running properly during the whole power increase program and served reactor operators with reliable core information.

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Remaining work

In 2007 a new VERONA configuration will be installed at Unit 1, where the power increase to 108% is also planned. A special configuration called VERONA-t will be implemented in 2007, as well: this system functions as software server and supports operation and maintenance personnel by providing various monitoring and management tools.

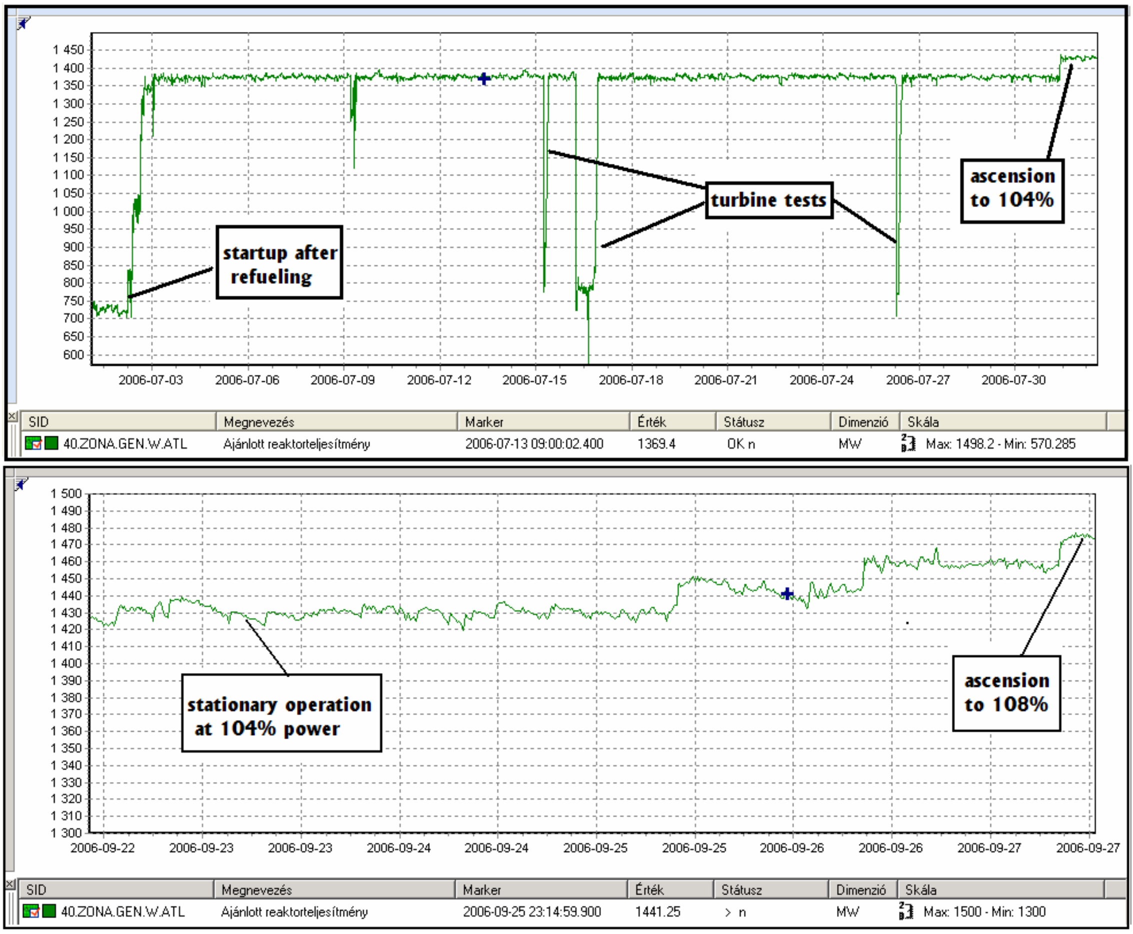

Figure 1. History of reaching the 108% power at Unit 4 as plotted from the VERONA archive

Reconstruction of the reactor noise data acquisition systems at Units 1-4 of Paks NPP

TAMÁS CZIBÓK, ZOLTÁN DEZSŐ, CSABA HORVÁTH, SÁNDOR KISS,
KÁROLY KRINIZS, JÓZSEF LÁZ, SÁNDOR LIPCSEI

Objective

Installation of new neutron noise measurement systems (PAZAR) was started at the end of 2005 at the four VVER-440 reactor units of Paks NPP.

Methods

PAZAR is a distributed autonomous system that is able to sample several hundreds signals simultaneously. The new system is designed in a client-server architecture. Server computers, installed at the plant units, control the signal conditioning hardware and acquire signal data. Client programs, running in a remote central computer, control the measurements on the servers through the local area network.

Results

Two systems are already in operation at Unit 3 and 4 (Fig. 1), they were installed in December 2005 and July 2006, respectively. Now these systems are fully operational and they are used for the regular noise diagnostics measurements performed by AEKI.

Fig. 1. Implementation at Unit 4. Computer and multiplexers are in the smaller rack, while signal conditioning modules are in the larger one

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Remaining work

Further PAZAR configurations will be installed during 2007 and 2008 at Unit 1 and 2, respectively.

Fundamental research in chemistry

It is a tradition that in fundamental chemistry structures of chemical compounds and interactions of atomic groups within a molecule are studied by physical methods. From the subject period, a study of phase equilibrium, experimental investigation of molecular interactions in dimethylethyleneurea solution and a problem from investigating structures of large molecule modelling is reported.

Phase equilibria

ATTILA IMRE

Objective

The aim of this project was to study different kinds of phase transitions and equilibria in pure and multicomponent systems.

Results

Four different types of phase equilibria were studied: liquid-liquid, liquid-vapour, liquid-solid and finally liquid-glassy liquid.

Liquid-liquid equilibria were studied in different binary liquid mixtures. Dielectric measurements were performed under hydrostatic pressure (up to 200 MPa) while theoretical studies were done from the high pressure range down to deep negative pressures (-40 MPa) (Refs. 1-3).

Liquid-vapour equilibrium was studied on a model system by lattice Boltzmann method. This year, the research was focused mainly on the behaviour of the equilibrium curve and the structure of the interface in the immediate vicinity of the critical point. The knowledge of this behaviour will be useful in the future supercritical-subcritical transition studies.

Liquid-solid and liquid-glassy liquid transitions were studied theoretically (with some added dielectric measurements in glassy systems). In case of liquid-solid transition, a maximum on the melting (p-T) diagram was already known for a few materials; it has been shown now that this maximum is not an exception, rather a general phenomenon and it can also be expected in glass transitions (Ref. 4,5). It should be mentioned that one of the studied materials was sodium, an important material in Generation IV studies.

Methods

Dielectric spectroscopy has been used in the liquid-liquid studies in cooperation with the laboratory of Prof. S.J. Rzoska (Silesian University, Katowice).

The liquid-vapour equilibrium data has been calculated by lattice Boltzmann method in cooperation with the group of G. Házi (AEKI).

Remaining work

The analysis of lattice Boltzmann data should be finished.

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Investigation of the structure of condensed phases

GÁBOR JANCSÓ

Objective

The research was aimed at studying the structures and intermolecular interactions in condensed phases. Within the framework of this, the orientation of hydrophobic molecules at the liquid-vapor interface in aqueous solutions and the intermolecular interactions of dimethylethyleneurea dissolved in heavy water were investigated.

Results

The 3-methylpyridine molecules (3MPy) are strongly surface active: the free liquid-vapor surface of its aqueous solution becomes almost saturated with the solute molecules even in solutions of very low concentration. However, the preferential orientation of the interfacial 3MPy molecules, whether they are oriented perpendicular or parallel to the surface, cannot be determined unambiguously from surface tension measurements. The purpose of the present investigation was to clarify this point by computer simulation, and thus demonstrate also the capability of this methodology to analyse such kind of problems successfully. The analysis of the orientation of the surface molecules on the basis of a Monte Carlo simulation of 3 mol% aqueous solution showed that the 3MPy molecules exhibit a dual orientational preference. At the vapor side of the interface they are preferentially aligned perpendicularly to the interface, while at the liquid side the preferred orientation of the 3MPy molecules is close to the parallel alignment with the plane of the interface.

Our previous small-angle neutron scattering (SANS) studies on aqueous solutions of non-electrolytes have demonstrated that SANS can provide information on the solute-solute and solute-solvent interactions. As a continuation of our investigations on aqueous tetramethylurea (TMU) solutions, we decided to study the solutions of dimethylethyleneurea (DMEU), which contains – instead of two methyl groups – an ethylene group and thus has a ring structure. The purpose of the present SANS and volumetric investigations was to study the effect of the structural difference between TMU and DMEU molecules on the intermolecular interactions in their aqueous solutions. The SANS experiments and density measurements were carried out in heavy water in the concentration range of 0.5 – 5 mol% at 298 and 313 K. The results of the density measurements indicated that the water molecules are more structured in the hydration sphere of TMU than in that of DMEU molecule. The SANS study led to the conclusion that the pair-wise solute-solute interactions are weaker in DMEU than in TMU solutions.

Methods

The Monte Carlo simulation of the liquid-vapor interface of the aqueous 3MPy solution was carried out in co-operation with the Department of Colloid Chemistry of Eötvös Loránd University. The small-angle neutron scattering measurements were performed on the „Yellow Submarine" instrument at the Budapest Neutron Centre, in co-operation with the Research Institute for Solid State Physics. The densities were measured using an Anton-Paar DMA 60/602H vibrating-tube densimeter. The reproducibility of the densities was (1 to 2) x 10-5 g-cm-3.

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Trajectory analysis

ATTILA IMRE

Objective

During the computer-assisted analysis of continuous curves, the original images have to be rasterized. This digitalization can cause some error in the later analysis (area, length, etc measurements). The aim of this project was to estimate the extent of these errors. The knowledge of the errors is crucial when a theoretical/simulation result is compared with the measured one.

Results

When the length of a trajectory in a PIV measurement or the area of a corroded spot in a metallography study has to be determined, computer-assisted image analysis has to be used. The originally continuous lines/spots should be digitalized. It is widely known that the digitalization cannot be accurate; ± several pixels are always expected in the length or area measurements. In this project we tried to estimate the extent of these errors and we tried to find some way to minimalize them. Our main focus was the study of the lack of the translational and rotational invariance during rasterization. Systematic errors coming from the different camera positions were presented (Refs. 1 and 2).

Almost all of our theoretical results were applied on biological samples (Refs. 3 and 4); the application on material samples should follow this.

Methods

Analytical methods and computer-assisted image analysis (ImageJ) have been used in these studies.

Remaining work

The results should be applied in the trajectory analysis of PIV measurements as well as in corroded/eroded metal surface analysis.

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International co-operations

This section gives short reports on three international co-operations.

Atomic Energy Research (AER)

ANDRÁS KERESZTURI, ISTVÁN VIDOVSZKY

The international scientific cooperation known as AER was set up in 1991. In 2006, there was no change in the membership list of AER, it continues providing the only forum for organized discussions on VVER reactor problems. AER operates the following working groups: spectral and core calculations, core design, operation and fuel management, core montitoring, surveillance and testing, neutron kinetics and reactor dynamics methods, criticality safety, spent fuel and CFD applications.

Budapest Neutron Centre (BNC)

RÓZSA BARANYAI, MIHÁLY MAKAI

The Budapest Neutron Centre has operated a full international user programme; from the beginning of 2004 the BNC has been taking part in the Integrated Infrastructure Initiative for Neutron Scattering and Muon Spectroscopy (NMI3) programme. NMI3 integrates neutron and muon research in Europe. In the frame of the Access Activities our facilities are available for researchers both from member states and associated states of the European Community. The main objectives of the access programme are to widely utilize BNC facilities and attract more and more scientists. The most important communication channels are the internet, relevant conferences and scientific journals. Budapest Neutron Centre announces call for proposals and collects applications twice a year (mid May and mid October) via internet using of BNC's website (www.BNC.hu). Further to the regular process, a fast track evaluation is also possible in special cases.

Currently, we receive all applications in electronic format. The selection and decision is made via internet at the spring cycle and by the annual meeting of the selection committee in autumn period. On the last meeting the panel members reviewed the selection procedure; they found that the quality of the proposals have been improved the most got good or very good mark.

BNC continues the successful participation in NMI3 Access program. In 2006, BNC has delivered 79 access days (compared to 50 planned) supporting 17 experiments. 25 scientists from 12 countries could benefit from the program. 19 users could be supported by reimbursement of travel and accommodation cost. 10 scientists have been new user at BNC. Almost 50 % of the users were under 35 years of age.

The OECD Halden Reactor Project

ZOLTÁN HÓZER, JÁNOS VÉGH

The OECD Halden Reactor Project (HRP) was established by the OECD Nuclear Energy Agency in 1958, with some OECD countries as member states. The Project's activities are concentrated at Halden (Norway), where a research centre of the Norwegian host institute (Institutt for energiteknikk, IFE) is located. Experimental fuel and structural material irradiations are conducted at the Halden Boiling Water Reactor (HBWR, IFE); post irradiation examinations are carried out at Kjeller (IFE). During the 48 years of research work HRP has accumulated a large amount of experimental and theoretical results, these results have been communicated in ca. 850 research reports and scientific publications. Achievements of HRP are internationally recognized and constitute a great value for research related to the safe utilization of nuclear power. Recently the Project's R&D activities are focused on the following research areas:

- Investigation of nuclear fuel behaviour under various operational conditions and power transients.
- Investigation of corrosion and ageing phenomena in structural materials applied in nuclear plants.
- Experimental and theoretical investigation of human reliability, development of various methods for operators' performance assessment.
- Application of computerised information systems to monitor process data in conventional and nuclear power plants in order to support plant operators in their process supervision work.

The first two topics are referred to as "Fuels and Materials"; the last two is called "Man-Technology-Organisation" (abbreviated as MTO).

The HRP is operated in three-year periods; member states renew their membership for each period. In the 1994-96 period Hungary joined the HRP as an associate member, with KFKI AEKI acting as the Hungarian representative in the Project. AEKI has prolonged its HRP participation for the present three year period, extending from 2006 to 2008. This membership gives AEKI continuous access to the most recent results related to nuclear fuel behaviour studies and computerised information systems.

In the Project's MTO area the most important co-operations in 2006 were as follows:

- Development and application of methods for long-term signal behaviour analysis and equipment condition monitoring, in order to support the safe operation of nuclear power plants.
- Development of reactor noise signal analysis methods applicable in the low and extremely low frequency region.
- Review and assessment of methods, practices applied for the upgrading of nuclear power plant control rooms and other human interfaces.
- Review and assessment of methods, practices applied during large-scale modernization projects of nuclear power plant instrumentation and control (I&C) systems.

Details of activities performed during 2006 in connection with the Project's „Fuel and Materials" area outlined in a separate thematic section of the present Progress Report.

List of abbreviations

AC	Alternating Current
ADC	Analogue Digital Converter
AEKI	KFKI Atomic Energy Research Institute
AER	Atomic Energy Research
AFR	Away From Reactor
AGNES	Advanced General and New Evaluation of Safety - Hungarian program for Paks NPP
ANL	Argonne National Laboratory
AOO	Anticipated Operational Occurrences
AR	Pressure Reduction Valve
ATWS	Anticipated Transient Without SCRAM
BME	Technical University of Budapest
BME NTI	Technical University of Budapest, Institute of Nuclear Techniques
BNC	Budapest Neutron Centre
BNFL	British Nuclear Fuel Ltd.
BOC	Beginning of Cycle
BRR	Budapest Research Reactor
CARD	Computer Aided Reactor Diagnostics
CERN	Conseil Européen pour la Researche Nucleaire
CERTA	Centre for Emergency Response, Training & Analysis
CFD	Computational Fluid Dynamics
CNFF	Central Nuclear Financial Fund
CNRS	Centre National Recherche de Saclay
CNS	Cold Neutron Source
CSFMS	Critical Safety Functions Monitoring System
CTP	CERTA Training Package
DAT	Digital Audio Tape
DBA	Design Base Accident
DC	Direct Current
DIM	Dust Impact Monitor
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DOE	Department of Energy
dpa	Displacement Per Atom
DVD	Digital Video Disk
ECR	Equivalent Cladding Reaction
EDX	Energy Dispersive X-ray microanalysis
ELTE	Eötvös Lóránd University, Budapest

EOC	End of Cycle
EOP	Emergency Operating Procedures
EU	European Union
EVA	Extra Vehicular Activity
FDDI	Fiber Distributed Data Interface
FEM	Finite Element Method
FFT	Fast Fourier Transform
FRM	Fission Gas Release
FSAR	Final Safety Analysis Review
GRM	Ground Reference Model
GRS	Geselschaft für Reaktor Sicherheit
HAEA	Hungarian Atomic Energy Agency
HAEA NSI	Hungarian Atomic Energy Agency Nuclear Safety Inspectorate
HEU	High Enriched Uranium
HMI	Human Machine Interface
HPGe	High Purity Germanium (Detector)
HPIS	High Pressure Injection System
HPLWR	High Performance Light Water Reactor
HRP	Halden Reactor Project
HTML	HyperText Markup Language
HZE	355
IAEA	International Atomic Energy Agency
IBMS	In-Beam Mössbauer Spectrometer
INAA	Instrumental Neutron Activation Analysis
IPSN	Institute de Protection et de Surete Nucleaire
ICCM	irradiated cell conditioned medium
ISS	International Space Station
ITU	Institute for Transuranium Elements
JEDI	Hungarian acronym for “Signal Validation and Noise Diagnostics”
JHR	Jules Horowitz Reactor
JRC	Joint Research Cenntre (Halden)
KFKI AEKI	KFKI Atomic Energy Research Institute
LA	Licencing Authority
LBLOCA	Large Break Loss Of Coolant Accident
LBM	Lattice Boltzmann Method
LED	Light Emitting Diode
LEFE	Low Effort Fast Evaluation
LET	Linear Energy Transfer
LEU	Low Enriched Uranium
LNT	Linear No Threshold
LOCA	Loss Of Coolant Accident
MRI	Magnetic Resonance Imaging

MTR	Material Testing Reactor
MÜSZ	Technical specifications (of Paks NPP)
μ-XRF	Micro X-ray Fluorescence
NACOS	Nuclear Accident Consequence Simulator
NPP	Nuclear Power Plant
NR	Neutron Radiography
NRDP	National Research and Development Programme
NSD	Nuclear Safety Directorate
NSF	Nuclear Spent Fuel
NSR	Nuclear Safety Regulations
OAH	Hungarian Atomic Energy Authority
OECD	Organisation for Economic Co-operation and Development
OG	Operational Guides
OLC	Operational Limits and Conditions
OMFB	National Committee for Technological Development
OTKA	National Foundation for Scientific Research, Hungary
PCS	Plant Computer System
PDR	Paks Diagnostic System
PHARE	Poland Hungary Aid Programme for Reorganising the Economies
PLASMA	Plant Safety Monitoring and Assessment
PLC	Programmable Logic Circuit
PMK	Paks Model Experiment
PSA	Probabilistic Safety Assessment
PSD	Powder Neutron Diffractometer
PSR	Periodic Safety Review
PTS	Pressurized Thermal Shock
PURAM	National Non-profit Company for Radioactive Waste Management
QA	Quality Assurance
PWR	Pressurised Water Reactor
R&D	Research and Development
RISKAUDIT	Joint Venture of the Gesellschaft für Reaktor und Anlagensicherheit mbH (GRS) and IPSN France
RNAA	Reactor Neutron Activation Analysis
RODOS	Real time Online DecisiOn Support
ROMAP	Rosetta Magnetometer and Plasma Monitoring
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RUSET	Ruthenium Separable Effect Test
RWF	Radiation Weighting Factor
SANS	Small Angle Neutron Spectrometer
SBLOCA	Small Break Loss of Coolant Accident

SCADA	Supervisory Control And Data Acquisition
SCRAM	Shutdown Control Rod Actuation Mechanism
SCWR	Supercritical Water Reactor
SEM	Scanning Electron Microscope
SESAME	Surface Electric Sounding and Acoustic Monitoring Experiment
SFSP	Spent Fuel Storage Pool
SG	Steam Generator
SGML	Standard Generalized Markup Language
SIMPLE	Semi Implicit Pressure Linked
SIP	Small Instrument Package
SNF	Spent Nuclear Fuel
SODAR	Sonic Radar
SPDS	Safety Parameter Display System
SPH	Smoothed Particle Hydrodynamics
SPM	Simple Plasma Monitor
SPND	Self Powered Neutron Detector
SQL	Structured Query Language
SS	Stainless Steel
SSMS	Spark Source Mass Spectrometry
SSNTD	Solid State Nuclear Track Detector
SZTAKI	Computer and Automation Research Institute of the Hungarian Academy of Sciences
TAS	Triple Axis Spectrometer
TLD	Thermo-luminescent dosimeter
TOF	Time Of Flight (Spectrometer)
TRU	Trans Uranium Element
TS	Technical Specifications
TSD	Technical Specifications Document
USNRC	US Nuclear Regulatory Commission
TUB	Technical University Budapest
VITA	Vital Information Transfer and Analysis
XML	Extensible Markup Language
XSLT	Extensible Stylesheet Language Transformations
VEIKI	Institute for Electric Power Research
VERONA	VVER ON-line Analysis
VVER	Pressurized Water Reactor of Soviet type
WYSIWYG	What You See Is What You Get
XANES	X-ray Absorption Near Edge Structure

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