ENS RRFM '97

Transactions

Oral Presentations and Posters

1st International Topical Meeting on
Research Reactor Fuel Management
February 5 to 7, 1997
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Organised by the European Nuclear Society

in cooperation with the Belgian Nuclear Society

and the International Atomic Energy Agency

Invited and Contributed Papers

Oral and Poster Presentations

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Session 1:

Research Reactors and Fissile Materials Supply

The New German Research Reactor FRM-II

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ABSTRACT

A new German high-flux research reactor is presently being built in Garching by the Technical University of Munich. The new reactor, called FRM-II, shall replace the existing "Forschungsreaktor München" FRM which has been operating very successfully for about 40 years now. The new reactor has been optimized primarily with respect to beam tube applications of slow neutrons, but will also allow to irradiate samples with thermal neutrons. So the FRM-II has been designed to provide a high flux of thermal neutrons in a large volume outside of the reactor core, where the neutron spectrum can be locally modified by using special spectrum shifters. The goal was further to obtain this high flux at a reactor power being as low as possible since this represents the best choice because of lowest background radiation for the experiments, lowest nuclear risk potential, lowest costs and superior inherent safety features.

The essential design feature of the FRM-II is a very compact reactor core consisting of a single fuel element only, which is cooled by light water and surrounded by a large heavy water moderator tank. The cylindrical fuel element is made up of two concentric tubes, the outer one having a diameter of about 24 cm. A total of 113 fuel plates is welded between the two tubes. They are all indentical and curved to involute shape so that the cooling channels between them have a constant width of 2.2 mm. Each of the plates is 1.36 mm thick and structured as a three layers' sandwich with two cladding layers at the surfaces and the fuel zone in between. Since it is not easy at all to provide sufficient excess reactivity for such a small core, the new high-density uranium silicide fuel has to be used in combination with highly enriched uranium. Fuel density grading and a ring of burnable poison are provided to flatten the power density profile in the fuel element. The fuel element is placed in a vertical core channel tube, which separates the light water of the primary cooling circuit (down flow) from the surrounding heavy water moderator tank which has both 250 cm diameter and height.

More than 50% of the fast fission neutrons immediately leak out of the small core into the large heavy water moderator tank where they slow down and thermalize to build up a high flux of thermal neutrons. The design values of the FRM-II are: a reactor power of 20 MW, an unperturbed thermal flux maximum of $8 \cdot 10^{14}$ n/cm^2 s in the moderator tank, and a cycle length of about 50 full power days. The average power density in the active zone of the core is about 1.1 MW/liter. The ratio of thermal flux (outside of the core) to power is highest of all reactors in the world.

The moderator tank is placed in the center of a big reactor pool filled with light water. 10 horizontal beam tubes penetrate the concrete of the biological shield of this pool and lead the neutrons to the scattering instruments in the experimental hall of the reactor building and in an adjacent neutron guide hall. Some of the beam tube noses are in contact with "spectrum shifters", i.e. with a large "cold source" filled with liquid deuterium or with a "hot source" containing graphite of more than 2000° C temperature, but there is also an uranium converter to produce fission neutrons for one beam tube. Further there are two inclined beam tubes and one vertical guide tube. Several vertical channels allow to insert samples to get irradiated in a high thermal neutron flux. The fields of applications of this "multipurpose neutron source" range from fundamental research in physics, biophysics and chemistry to applied research (e.g. material sciences), medical research and treatments, up to environmental applications (e.g. trace impurities detection by activation analysis) and industrial utilization (as e.g. silicon doping and radioisotopes production).

The FRM-II reactor will be controlled by a single hafnium control rod which moves within the inner tube of the fuel element; at its lower end it is connected with a beryllium follower. It can be decoupled from the control rod drive mechanism to fall down and act as a fast shutdown system. A second, redundant and diverse fast shutdown system is provided by five hafnium shutdown rods in the moderator tank which are fully withdrawn during reactor operation, however. Four of the five rods would suffice to shut down the reactor even if the control rod would totally move out of a fresh fuel element. Additionally the "compact core" reactor is characterized by pronounced inherent safety features which would, e.g., make the reactor subcritical under all postulated severe accident conditions.

The primary cooling circuit is a virtually closed loop which however is connected with the reactor pool through a strainer located underneath the core; in this way the large pool water reservoir is made use of as a pressurizer of the primary circuit and for the core cooling after shutdown.There are four primary pumps which are all equipped with flywheels and with check valves against reverse flow. They are mounted - together with the heat exchangers to the secondary circuit - in a water-tight "primary cell" of small volume.

After reactor shutdown core cooling is still provided by the forced flow of the primary pumps. Three independent battery buffered shutdown pumps are installed, one of which would be sufficient to maintain the forced flow through the fuel element if the primary pumps were not available. The shutdown-pumps suck water from the pool, feed this water via check valves into the collector of the primary circuit and, after having passed the fuel element, back into the pool. Three hours after shutdown the fuel element can be cooled by natural convection: the pumps are shut off, with the decreasing pressure in the collector two natural circulation flaps open automatically and the water flow through the fuel element reverses. If the external heat sink were not available, the decay heat could be stored completely in the pool water, no recooling of which would be needed.

The FRM-II reactor building is 30 m high and has the form of a 40 x 40 m^2 square in its lower and of an octagon in its upper part. It provides full protection against earthquakes and - a new feature for research reactors - against an air plane crash. For that the outer walls and the roof of the building consist of reinforced concrete of 1.8 m thickness, and the reactor pool is decoupled from the outer building structure such that the pool water would not be lost. The confinement of the reactor building also represents the ultimate barrier against an uncontrolled release of radioactive fission products to the environment in case of an accident. On one side the reactor building is connected with a neutron guide hall which is about 60 x 46 m^2 wide and 11 m high.

Finally, the project status is as follows. While the conceptual design work for the FRM-II was started by the Technical University of Munich (TUM) as early as 1980, the safety analysis report was completed and the application for nuclear licensing of the FRM-II was submitted to the licensing authority in February 1993. All this has been done by the TUM together with the Siemens company which was nominated to become the general contractor in June 1994. In April 1996 the project obtained the first partial nuclear licence which covered the general safety concept acceptance, the site opening and the construction of the reactor building. Immediately after that the construction work has been started. A dummy fuel element, which is now subject to hydraulic testing, has been fabricated by the company CERCA, France. The final (i.e. third) partial nuclear licence is expected for the year 2001 which will be followed by the nuclear start-up, a fifty days full power test run and, finally, routine operation of the research reactor for the benefit of the user community.

REACTEUR JULES HOROWITZ : A NEW MATERIAL TESTING REACTOR PROJECT

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ABSTRACT

The "REACTEUR JULES HOROWITZ" is a new research reactor project entirely dedicated to materials and nuclear fuels testing, the location of which is foreseen at the CEA-CADARACHE site, and the start-up in 2005.

The launching of this project arises from a double finding :

⇨ the development of nuclear power plants aimed at satisfying the energy needs of the next century cannot be envisaged without the disposal of experimental reactors which are unrivalled for the validation of new concepts of nuclear fuels, materials, and components as well as for their qualification under irradiation.

⇨ the present park of experimentals reactors is 30 to 40 years old and it is advisable to examine henceforth the necessity and the nature of a new reactor to take over and replace, at the beginning of next century, the reactors shut-down in the mean time or at the very end of their lives.

Within this framework, the CEA has undertaken, in the last years, a reflection on the mid and long term irradiations needs, to determine the main features and performances of this new reactor.

The selected reactor project, among several different concepts, is finally a light water open pool concept, with 100 MW thermal power, characterized by an in-core central loop. It could reach neutronic fluxes twice those of present french reactors, and allows many irradiations in the core, in the central loop, and around the core, under high neutron fluxes. But to obtain such performances, it is necessary to settle high forced flow rates and upward flow in the core, in order to preserve the operating flexibility of the reactor ; this leads to design a specific fuel assembly.

The Current Status and Future Trends in the Use of Russian Research Reactors

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INTRODUCTION

The report presents a survey of the current situation in using, modernization and upgrading of operating Russian research reactors. Also the report throws light upon the problems of the development of fuel elements and assemblies for research reactors. The perspectives of upgrading of operating research reactors and constructing of new reactors are discussed in connection with future plans in developing of nuclear science and technology.

MODERN SITUATION IN USING OF RUSSIAN RESEARCH REACTORS

In Russia, the first research reactor was started in 1946. Since then many research reactors, differing as to type, power level and experimental programmes, were developed and constructed. In total, there are 18 civilian steady power research reactors in operation in Russia, not counting several dedicated and prototype reactors and critical assemblies (or zero-power-reactors).

The important peculiarity of Russian research reactors is the large variety of types of fuel assemblies that are used in different reactors. At the present time, more than ten types of fuel assemblies are in use. The fuel elements distinguish on the geometrical types, the enrichment of uranium, the height of the active part, the types of fuel materials.

Clearly, to construct and operate for a long time so many reactors, it was essential to create a national school of designing, construction and operation of research reactors. It is important that an ingeneous domestic fuel fabrication technology was developed for research reactors. This technology has some advantages over the technology for fabricating MTR type fuel because the amount of structural materials uses in the core is minimized and not more than the amount necessary for fuel elements fabrication.

In the former Soviet Union it was developed three generations of fuel elements and assemblies for research reactors on the basis of using of various aluminum materials (alloys and oxides).

Generations	*Years*	*Enrichment [%]*	*U-235 Concentration [g/l]*	*Thickness (diameter) of fuel elements [mm]*	*Specific heat transfer surface [m^2/l]*
First	1954-1970	10-36	50	10	0.098
Second	1963-1985	36-90	58-70	3.2-2.0	0.2-0.362
Third	1972-till now	90	68-130	1.4-1.25	0.45-0.66

Also the Russian Reduced Enrichment Research Reactors Program that was started late in the 70ies, continuing now. The main results of this work would be increase the density of the fuel

meat in the composition on the basis of uranium dioxide and the change of the fuel composition with high uranium density in aluminum matrix.

Russian research reactors are used to perform numerous studies in different areas of science and technology: nuclear physics (NP); neutronic studies on condensed-matter physics (CMP); irradiation material testing for fission and fusion reactors (MT) in experimental loops and rigs; applied studies (AS) such as radioisotope production, irradiation silicone doping, neutron activation analysis, neutron radiography. Every scientific program is supported by big quantity of unique experimental facilities.

With so many research reactors in operation in Russia, it proved possible to enlarge experimental capabilities based on upgrading them rather than constructing new ones. That is why Russian research reactors underwent numerous reconstructions during their lifetime. All reconstructions pursued the objective of increasing the neutron fluxes in experimental facilities. Naturally, this called for improving the core neutronics and thermal hydraulics, updating the cooling system parameters. Core neutronics improvement consisted mainly in increasing uranium-235 content in the core and limiting harmful neutron absorption. This allowed to minimize critical and working core volume, hence increasing the ratio of the neutron flux to the reactor power. So the development of new fuel elements and assemblies is the key problem of every reconstruction.

One recent example of reconstruction of research reactor is the renovation of a very high flux reactor — SM-2 in RIAR (Dimitrovgrad). The upgraded reactor, called now SM-3, is much safer than it was prior to modernization. During upgrading, a new reactor vessel was installed inside the existing one to avoid core dewatering during accidents with leakage in the main vessel. Now Russian design institutes develop the design of the reconstruction of the material testing reactor MIR in RIAR (Dimitrovgrad), pool type reactor — WWR-TS in Obninsk and several another reactors.

In the nearest future it is possible to develop and to construct of new research reactors. Now the design of new pulse type reactor — MIGR is in progress.

FUTURE TRENDS

The general opinion of the Russian specialists and official persons is that the maintaining of research reactors is absolutely necessary for to guarantee the safe operation of NPP and development of new NPP; to develop fundamental and applied research using neutrons; to produce radioisotopes.

The irradiation possibilities of Russian research reactors in general are sufficient for today and for the nearest future but it is necessary to realize some modifications of experimental facilities. The main direction of upgrading of experimental possibilities of Russian research reactors is reconstruction and modernization (SM-3, MIR, WWR-TS, IVV-2M, IR-50). The another reason for the modernization of Russian research reactors is the ageing of the equipment of reactors because the majority of reactors is very old.

The new fuel elements and assemblies for Russian research reactors shall be developed. We also propose to convert the composition using in the fuel elements of Russian research reactors in connection with the problem of the reduction of enrichment of uranium in research reactors.

CONCLUSIONS

During fifty years many research reactors were developed and constructed in Russia.These reactors have different types, power levels and experimental programmes.

Russian research reactors are used to perform numerous studies in different areas of science and technology such as: nuclear physics,neutronic studies on condensed-matter physics, irradiation material testing for fission and fusion reactors in experimental loops and rigs and applied studies (radioisotope production, irradiation silicone doping, neutron activation analysis, neutron radiography).

Russian research reactors underwent numerous reconstructions during their lifetime. All reconstructions pursued the objective of increasing the neutron fluxes in experimental facilities. Now the experimental possibilities of Russian research reactors are sufficient for Russian nuclear power program, fundamental investigations and applied works.
Ageing of the equipment and the problem of insufficient funding of research reactors require the necessity of shutdown of some reactors.

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SUPPLY OF LOW ENRICHED (LEU) and HIGHLY ENRICHED URANIUM (HEU) FOR RESEARCH REACTORS

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Abstract

Enriched uranium for research reactors in the form of LEU (= low enriched uranium at 19.75 % U-235) and HEU (= highly enriched uranium at 90 to 93 % U-235) was and is - due its high U-235 enrichment - a political fuel other than enriched uranium for power reactors.

The sufficient availability of LEU and HEU is a vital question for research reactors, especially in Europe, in order to perform their peaceful research reactor programs.

In the past the USA were in the Western hemisphere sole supplier of LEU and HEU.

Today the USA have de facto stopped the supply of LEU and HEU, for HEU mainly due to political reasons.

This paper deals, among others, with the present availability of LEU and HEU for European research reactors and touches the following topics:

- Historical US supplies
- Influence of the RERTR-Program
- Characteristics of LEU and HEU
- Military HEU enters the civil market
- What is the supply situation for LEU and HEU today?
- Outlook for safe supplies of LEU and HEU

1. Historical US-supplies of LEU and HEU

Following President Eisenhower's "Atoms for Peace" initiative (1953) and subsequent US legislation (Atomic Energy Act), the USA transferred to Europe a "flood" of secret information about the peaceful use of atomic energy, and the first research reactors were constructed in Western Europe. The enriched uranium in the fuel elements was in most cases at the beginning below 20 %, i. e. 19.75 % U-235 (LEU).

In the sixties the U-235 assay in the fuel elements could be easily adapted to the assay of weapons grade material of 93 % U-235 (HEU) and the USA made without difficulties HEU available to research reactors in Western Europe. Until 1974 the HEU could be leased from the former US Atomic Energy Commission (AEC); the research reactor operators had only to pay a relatively low lease charge of less than 5 % per year on the value of the HEU.

In 1974 the former AEC changed its supply policy and research reactor operators had to purchase the HEU. In 1977, however, US-President Carter became concerned about the "wide spread of weapons usable material" in research reactors. He initiated the International Fuel Cycle Evaluation (INFCE) and specially its working group 8 to minimize the traffic of HEU. As a result of the working group 8 it was decided that the ideal U-235 assay in the sense of non-proliferation should be less than 20 %, namely 19.75 % +/- 0.2 %. The development of such a high density fuel was initiated by the Reducing Enrichment in Research and Test Reactors (RERTR) Program of the US-Department of Energy (DOE) with international cooperation.

2. Influence of the RERTR-Programme on LEU and HEU supplies

What influence/success has now the RERTR-Program on the supplies of LEU and HEU outside the USA?

Table 5 contains a breakdown of US-HEU exports from the 50ties until 1991 which was prepared by the United States Regulatory Commission (NRC) to the US Congress in 1993. After checking of these figures I personally came to the conclusion that the report includes under the HEU amounts also 19.75 % U-235 uranium and intermediate assays such as 35 % and 45 % U-235 which have also been exported together with HEU. EURATOM countries are said to have received from the USA around 21 metric tons (mt) of HEU and non-EURATOM countries around 4.6 mt, totally to approx. 26 mt of HEU (which I believe contain the figures for LEU). If you are looking more close to these figures (see Table 6) it can be easily noted that Germany received the biggest portion (around 11.3 mt) followed by France (7.2 mt) and UK (2 mt).

Where can now the influence of the RERTR programme be seen in the present supply situation?

Table 7 shows the annual average exports of LEU (19.75 %), MEU (45 %) and HEU (93 % U-235) by the US-DOE to Europe.

My comments with regard to these figures are as follows:

19.75 % enriched uranium:

Table 7 shows annual exports of only 80 kgs from 1983 to 1993. The relatively small quantities are due to the fact that there were sufficient stocks outside the USA. Consequently there was no need for the export of LEU from the USA. The stocks of LEU outside the USA are due to the availability of LEU which was no longer used at the time of termination of certain research projects.

45 % enriched uranium:

Some countries (especially Japan) elected their own way during the conversion of the enrichment of 93 % to 19.75 %. They used fuel elements with uranium having a U-235 assay of 45 % as an intermediate enrichment for a certain time. The procurement of 45 % enriched uranium was terminated in 1991; the USA exported in the decade 1983 to 1991 approx. 600 kgs of uranium (45 % U-235 enriched).

93 % enriched uranium:

In the period 1983 to 1993 the USA exported approx. 1300 kgs of HEU. In 1994 supplies came to a complete stop due to the implementation of the U.S. Energy Act 1992 which de facto stopped all HEU exports. In addition safeguards problems occurred at the Y-12 plant in Oak Ridge which prevented physically the supply of HEU.

3. Characteristics of LEU and HEU

As the research reactor community is aware, NUKEM developed in collaboration with fabricators of fuel elements for research reactors standard specification form "Commercial grade LEU" as per Tables 8 to10 for LEU. Specification for HEU are similar, take, however, into account higher values for U-232, U-234 and U-236.

4. Military HEU enters the civil market

In the past years there has been a lot of speculation about the amount of stocks of military HEU accumulated by the weapon states USA and Russia.

NUKEM published closer information in its monthly NUKEM Market Report (NMR) in the past. The facts are seemingly as follows:

The USA produced altogether approx. 1000 metric tonnes (mt) HEU and Russia approx. 1500 mt, totalling to approx. 2500 mt HEU (see table 11). U.S Energy Secretary's Hazel R. O'Leary's "new openess" initiative lifted the veil of secrecy on America's stockpile of military HEU - and how much of it would never be used again to build nuclear weapons. Accordingly approx. 483 mt have been produced by the K-25 plant at Oak Ridge and 511 mt at the Portsmouth Plant (Table 12). This year, the U. S. Energy Secretary declared 174.3 mt of the HEU excess (Table 13).

This excess material plus additional declassified material is so far is in the order of 259 mt, leaving the U.S. government with 735 mt. If one were to subtract HEU requirements for past government bomb testing, nuclear navy and research , the amount in the U. S, stockpile would drop to about 500 mt of highly quality material.

In 1993 a government-to-government agreement was concluded between the U. S. and Russian and calling for USD 12 billion worth of LEU from Russian weapons-grade HEU (500 mt) to be converted into 15.000 mt of LWR-fuel over a period of 20 years (Tables 14 and 15). The blended 4.4 % U-235 enriched uranium has a content of 152.000 mt of natural uranium equivalent which corresponds to a factor of 4.4 of the world production of natural uranium (34.200 mt in 1995, Table 16).

The enormous quantity of natural uranium equivalent contained in the downblended 500 mt of HEU of Russian origin will certainly influence the market for natural uranium in the following years.

5. What is the supply situation for LEU and HEU today?

Table 17 shows the suppliers of LEU today.

US-DOE can at the moment not supply due to safeguards problems at the Y-12 plant. Resumption of production of LEU is anticipated in 1998.

UKAEA remains a supplier for LEU and performs processing and recovery services.

Russia, by several suppliers, remains supplier for LEU and performs processing and recovery services.

NUKEM disposes of large stocks of LEU.

China is not in the market.

Cogéma remains a supplier for LEU and performs processing and recovery services.

6. Outlook for safe supplies of LEU and HEU and conclusion

Due to available stocks and production possibilities the supply of LEU for research reactors is safe far beyond the year 2000.

Certain European Research Reactors, however, must furthermore use HEU in order not to stop reactor operation. Since the USA de facto have stopped after 1993 exports of HEU. Ways have to be found to ensure its availability in Europe; i. e. by using available stockpiles, even if the enrichment is less than 93 % and purchases from other sources.

It was a great pleasure to present this paper to you and I thank you for your attention.

URANIUM SUPPLY AND LICENSING PROGRESS AT CERCA

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1- INTRODUCTION

Over the past years, uranium licensing for use in the facility has been increasingly difficult under the pressure of tighter rules for radiation and contamination monitoring.

This contribution intends to summarize the status of licensing in our workshop.

2- AVAILABLE URANIUM SOURCES

2.1. Fresh Uranium

It is very difficult, or almost impossible to find sources of real resh Uranium with no Fission Products or Transuranics, especially if 93% material is needed.

2.2. Uranium from critical experiments

A large quantity of Uranium used at CERCA these past years has been issued from critical experiments, mainly from the KNK SNEAK experiments in KARLSRUHE.

This material comes generally in the form of nickel plated metal plates, or pure metal with various enrichments. The material has either to be diluted with depleted uranium to reach an enrichment of 19.75 %, or may be used "as it is" in the case of higher isotopic composition (around 90 %).

The very low irradiation levels achieved during the experiments, and hence the very low burn-up, are very favourable in term of Fission products, and Transuranic products. Furthermore, when dealing with original enrichments of 30 or 90 %, the dilution factor adds a safety margin to the allowable limits.

2.3. Reprocessed and reenriched Uranium

Other sources of Uranium are from reprocessed and/or reenriched material.

In most cases the spent fuel in a MTR reactor has an enrichment in the range of 50 to 65%. After reprocessing, three choices are possible for further use :

- The reprocessed Uranium may be used "as it is" for element fabrication with high TU content resulting from The life history of the Uranium, which may exceed certain allowable limits. Some other problems may occur during the storage of fresh fuel elements.

- The reprocessed uranium may be reenriched. This possibility is not used now, because of the non availability of dedicated enrichment lines, which should be otherwise polluted by all by-products. Furthermore, this leads to high U-232 content (as it is lighter, U2 is even more easily separated than U5). However, it should be noted that 93 % Uranium coming from the US in the 80' and early 90' was probably reenriched from rather clean material, if we examine carefully its isotopic composition.

- The reprocessed Uranium may be diluted down to an enrichment of 19.75 %. In that case the dilution factor has a benefic influence on the U-232, TU and FP content.

Due to the very high content of radioisotope, great care must be taken when examining the possibility to use Uranium coming from reprocessing to avoid licensing problems.

3- LICENSING EXPERIENCE OF CERCA

The Safety Anlysis Report applicable to the MTR workshop was amended some years ago to consider the possibility to use 19.75% Uranium coming from critical experiments.

4- FUTURE DEVELOPMENTS

Since the mid 80', and in the progress of the RERTR program, CERCA has used thousands of kilograms of LEU. Some of this material was provided by US DOE, some came later from critical experiments as explained before. Some are now coming from Russian plants.

The trend is now to use the available material, CERCA decided to ask for a large increase in the limits presently authorised for U-232, Transuranic and Fission products.

A new specification, so called "Uranium Commercial" has been issued and submitted to French Safety Authorities. This specification takes into consideration acceptable limits for CERCA's fabrication unit without any significant increase in prices.

At the time of the writing of this presentation, we are expecting a positive issue on this subject which will ease the acceptance of available Uranium batches in CERCA's workshop.

As soon as this specification will be accepted and put into force, it will be possible for CERCA to work with a new set of characteristics concerning the fissile materials, wider than previously allowed. This specification shall become a standard for the short term future.

The new values will open the way to new safety working conditions, and at the same time to softer acceptance criteria.

RECOVERY OF URANIUM FROM MANUFACTURING SCRAPS AND BLENDING OF RECYCLED MTR URANIUM

Dr Donald Skea, UKAEA, Dounreay, Thurso
Mr Philip Cartwright, UKAEA, Dounreay, Thurso

1 INTRODUCTION

1.1 To ensure the continued operation of some research reactors requires the careful use of the High Enriched Uranium (HEU) resource currently within the civil nuclear cycle. For these reactors, either a suitable fuel cannot be made with Low Enriched Uranium (LEU) or conversion to LEU would lead to sufficient operational and financial penalties to render the reactor economically unviable. To maximise the use of the current civil stockpile of HEU, there is a need for a uranium recovery plant, able to recover uranium suitable for reuse in reactor fuel, from a wide variety of feedstocks. Such a plant has been operated at Dounreay for more than 30 years.

1.2 The uranium recovery plant (D1203) was historically associated with the MTR fuel cycle in the United Kingdom, and to a lesser extent with MTR reactors in other parts of the world. The announcement of the closure of the Harwell MTR reactors in 1990 left the Dounreay recovery plants with a reduction in fuel scrap recovery, and consequently led to spare capacity. Some of this spare capacity has been filled with uranium recovery and supply work from overseas, the remainder has been programmed to be used for the recovery of the United Kingdom uranium stockpile.

1.3 The uranium recovery plant has successfully recovered and processed uranium for re-use in the MTR cycle from many different feedstocks;

- UF_6
- U/Al alloys
- U/Al/Si alloys
- U/Th oxide in graphite
- impure uranium metal and uranium oxide
- recycled MTR uranium
- metal reduction slags
- other uranium alloys.

1.4 The uranium recovery plant is the focal point for the Dounreay uranium product. Not only does it convert its own uranyl nitrate product, but it also handles the uranium products from the mixed oxide reprocessing plant and the MTR fuel reprocessing plant.

2 THE DOUNREAY URANIUM RECOVERY PLANT (D1203)

2.1 Modular construction has given this plant considerable versatility. The plant consists of a number of separate, but interconnected, facilities which can be used in many combinations. There are three main uranyl nitrate precipitation/uranium oxide production lines and two metal production lines, which enable work to proceed on

different enrichments simultaneously. A recent period of plant operation saw the following simultaneous activities:

- recovery of 90% enriched uranium through dissolver/solvent extraction/billet line B.
- dry blending to 20% enriched uranium metal on billet line A (part).
- wet blending to 20% enriched U_3O_8 through blending dissolver/billet line A (part)/ oxidation furnace.
- receipt/evaporation of 80% enriched uranyl liquor from irradiated MTR fuel reprocessing, followed by precipitation on billet line A (part).
- receipt/evaporation of depleted uranyl liquor from the MOX reprocessing plant.

Technical description

Dissolution

2.2 The plant has twin stainless steel dissolvers which feed a single solvent extraction line. The dissolver conditions are varied markedly depending on the type of feedstock. Dissolver liquor can be clarified by the use of filters or centrifuge. The dissolvers will be replaced during 1997 to provide a more flexible system for processing low uranium content feedstocks.

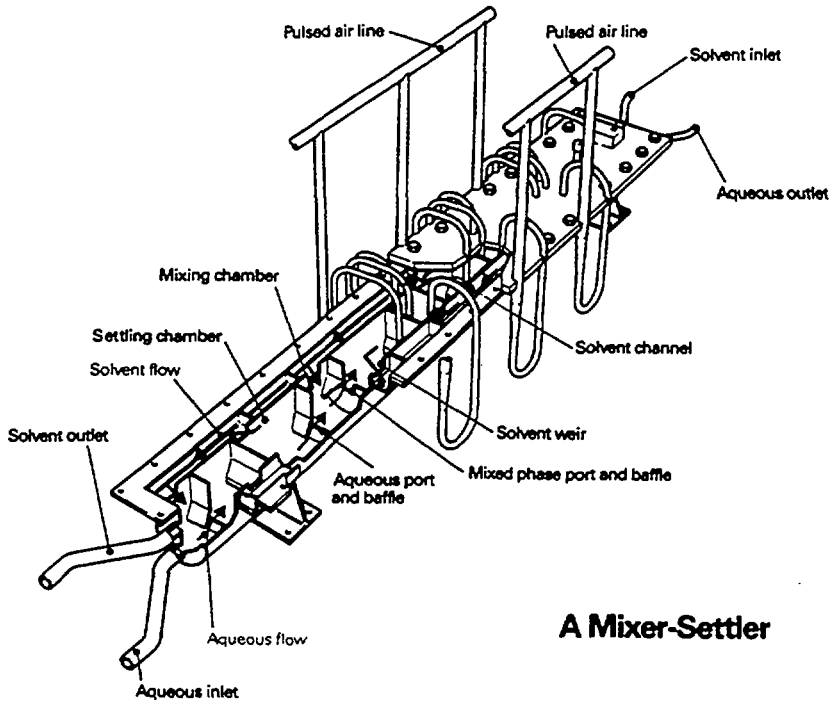
Solvent extraction

2.3 The solvent extraction system utilises the successful 100 mm channel geometry mixer settlers (Figure 1) used in the other processing plants on the Dounreay site. The solvent extraction system, in the recovery plant can be used as either a single or a dual cycle system to separate one or two products from the feedstock. The flowsheets used to achieve the necessary separations are all based on the TBP in odourless kerosene, but the concentrations of TBP and the inactive feeds used are varied, to suit the feedstock.

Where there is the need for the addition of mercury to assist in the dissolution of an aluminium rich feed, the raffinate is treated to remove the mercury.

The inactive feed system to the solvent extraction system permits the addition of several different feeds to enhance product purity. An example of this is ferrous sulphamate, which enables the removal of trace level of plutonium.

Figure 1



Evaporation

2.4 There are four evaporators in the uranium recovery plant, one of which is used solely for the processing of Fast Reactor depleted uranium, and another is used for the specific task of evaporating high thorium containing liquor from the processing of THTR uranium. There is a titanium evaporator for the uranium product from the solvent extraction system, and a glass evaporator for the uranium product from the reprocessing of irradiated MTR fuel. The duties of these last two evaporators can be interchanged through their internal pipework connections. All of the evaporators are used for the concentration of the dilute products from the solvent extraction systems.

Precipitation/drying

2.5 The concentrated uranyl nitrate from the evaporator systems are fed to one of three precipitators. There, the uranium is precipitated, by the addition of ammonia to produce Ammonium Diuranate (ADU). The ADU is filtered from the solution and the supernate is sampled and analysed before being discharged to drain. The ADU is dried at 320 °C in an oven under an air atmosphere, to produce UO_3. The depleted uranium from the Fast Reactor reprocessing is drummed for storage at this stage. MTR cycle uranium may be processed further to produce a high specification product on one of the two billet lines.

Reduction/fluorination

2.6 There are two reduction furnaces on each of the two uranium billet lines. First stage reduction uses hydrogen at high temperature. If cooling is allowed under hydrogen the end product is UO_2. The introduction of air at furnace temperatures after the reduction stage yields U_3O_8. Second stage reduction utilises anhydrous hydrogen fluoride to produce UF_4, the intermediate product for the metal production process used. When the required product is uranium oxide, whether UO_2, UO_3 or U_3O_8, with a specific particle size distribution, equipment is available for the grading and analysis of the particles to produce the required product.

Uranium metal reduction

2.7 There is a single metal reduction furnace on each of the billet lines. The UF_4 is mixed with calcium and packed in magnesium oxide crucibles. The mixture is then heated in a vacuum furnace to produce a pure uranium metal product, suitable for use in MTR fuel.

Uranium size reduction

2.8 To meet the requirements of MTR fuel fabricators it is necessary to produce metal in pieces of uniform size and of a weight range normally between 150 and 300 grammes. To achieve this, the billets from metal reduction are melted in an induction furnace and poured into 20 mm diameter moulds. The resulting bars are cropped into short pieces to meet the necessary size and weight requirements.

Blending

2.9 Blending of uranium is undertaken to produce the necessary enrichment. This has been done successfully by three methods. The first is using a small stand-alone dissolver designed specifically for blending purposes. Enriched and depleted uranium oxides are dissolved in calculated proportions to give the required enrichment. Fine tuning of the blending process is often accomplished by the subsequent addition of measured volumes of depleted uranyl nitrate liquor. The second method used is the dry blending of enriched and depleted UF_4 powder. There must be efficient mixing of the powder if the process is to run smoothly. Fine enrichment adjustments are often made at the size reduction stage, during vacuum melting/casting. Thirdly, uranium enrichment can be adjusted by direct wet blending of liquor product from scrap recovery or irradiated MTR reprocessing.

Wastes

2.10 As all of the material processed in the recovery plant has low plutonium content (less than 10 ppm) and low fission product content (less than 37000 Bq/gm U), the waste streams are all low active. Liquid waste is discharged to sea under the Dounreay site

liquid waste discharge authorisation and solid waste is compacted and disposed of on the Dounreay site.

Plant capacities

2.11 Plant capacities are dependent on the different combinations of facilities being used. The following are indicative values:

- recovery (dissolution/solvent extraction/metal production) 250-1200 kg U per year depending on type of feedstock.
- blending to produce 19.75% enriched uranium, 900 kg U per year.

Interaction with other Dounreay plants and programmes

2.12 The following are the major interactions with other plants:

- *MTR Element Fabrication Plant* - the uranium recovery plant recovers scraps from the fabrication process.
- *Irradiated MTR Fuel Reprocessing Plant* - purified uranyl liquor is passed to the uranium recovery plant for precipitation, or blending then conversion to oxide and metal product.
- *MOX Reprocessing Plant* - purified uranyl liquor is passed to the uranium recovery plant for precipitation to oxide product. This route provides the option for the recovery of enriched uranium from HEU MOX. The uranium product is converted to the necessary form for use in the MTR fuel cycle.
- the uranium recovery plant is also recovering material from early United Kingdom uranium fuel development programmes, from zero energy reactors, and from spallation targets. This allows the recovery of product for sale and ensures that the waste streams are consistent and do not require special consideration.

3 PRODUCTS

3.1 The uranium recovery plant at Dounreay has the following products:

- form - U metal, UO_2, U_3O_8, UO_3
- enrichment - up to 90%+
- origin - manufacturing scraps, uranium from irradiated MTR reprocessing, customers uranium/U oxide, uranium from MOX reprocessing.

- the products are all produced to specifications which are agreed in advance with the customer. Typical values of the main analysis are;

  impurities - less than 1500 ppm
  plutonium content - less than 250 Bq/gm U
  fission products - less than 600 Bq/gm U.

3.2 The products are all checked for their compliance with the agreed specification by chemical analysis undertaken in the analytical laboratories at Dounreay. These laboratories are furnished with the equipment and analytical techniques to allow confirmation of all of the specification values.

4 DISCUSSION

4.1 The uranium recovery plant at Dounreay has a number of standard products relevant to the MTR fuel cycle. It is, however, in a unique position to recycle/blend reprocessed uranium due to its close link with other plants on site.

4.2 MTR fuel element manufacturers have specific requirements for uranium feedstocks, depending on their individual production facilities. There are, therefore, strict limits for impurities, transuranic elements, fission product content and the uranium isotopes U_{232}, U_{234} and U_{236}.

4.3 The uranium recovery plant at Dounreay can produce the uranium oxide and metal products from a large variety of feedstocks to allow continued fabrication of MTR fuel for both LEU and HEU fuels.

4.4 The uranium recovery plant at Dounreay can also provide essential support services to fuel manufacturers, permitting the recovery of fabrication scraps and with its connection to the mixed oxide and MTR fuel reprocessing plants, permits the recovery of most irradiated and unirradiated fuels allowing a secure future for MTRs within the current civil nuclear fuel cycle.

4.5 For national programmes to clear old fuel development wastes, the uranium recovery plant at Dounreay offers the opportunity to recover the uranium for re-use and to produce waste streams which are suitable for either discharge or disposal.

HIGHLY ENRICHED URANIUM RECYCLE AND CONVERSION FACILITY AT COGEMA PIERRELATTE

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ABSTRACT

Enriched uranium inventories (over 5 % U 235) exist under various chemical forms and enrichments. The reuse of these materials for Research Reactor Fuel, Experimental Reactors or as raw material for Research Laboratories requires a change in the chemical form and/or a change in the enrichment level and/or an improvement in the chemical purity. The COGEMA Pierrelatte facility (Recycle and Conversion Facility) is well suited to perform these tasks. Its experience in this field has been acquired over the last 30 years.

This paper presents the capabilities of this facility.

INTRODUCTION

The Pierrelatte gaseous diffusion plant which was commissioned about thirty years ago to produce highly enriched uranium metal for the French Defense Department has definitively stopped production in June 1996.

The adjacent facility (Recycle and Conversion Unit, named « URE » in French) designed from the outset to convert enriched UF6 into metal and recycle fabrication scraps from the Defense Department is also active since over fifteen years in the civil sector : recycling metal from the fabrication scraps received from MTR fuel fabrication plants, chemical conversion to the chemical forms of oxide - or metal - irrespective of the isotopic uranium content, for various Research Laboratories, Experimental Reactors or Fuel Fabricators.

This facility will stay in operation. The functions, the equipment and the input/output products of this versatile facility will be described.

FUNCTIONS OF THE FACILITY

This versatile facility is able to convert uranium from one chemical form into another, to purify the material if required and to perform isotopic adjustment.

a) Conversion from one chemical form into another.

Basically, all the classical transformations are possible :

- UF6 into UF4, UF4 into U metal, U metal into U308
- U oxides or U alloys or U metal into uranyl nitrate,
- Uranyl nitrate into U308 or into UO2 (sinterable powder) or into UF4

b) Chemical purification of the material.

Under the uranyl nitrate form, uranium is purified using solvent extraction treatment of the PUREX process. After one or two purification cycles the required purity is obtained.

c) Isotopic adjustment

Isotopic adjustment is performed in liquid phase : uranyl nitrate solution or melted uranium metal. Clean natural or depleted uranium is used to obtain the desired enrichment level.

Typically, material with enrichment levels higher than 20 % are downgraded into 19.75 % material. But other isotopic levels can also be obtained.

THE EXISTING EQUIPMENT

Figure 1 presents a simplified flowsheet of this facility. It is structured along two main process lines : the Dry Process (gas and solids) and the Wet Process (uranyl nitrate in aqueous solution).

a) Dry process

This process leads to the production of U metal from UF6. Highly enriched UF6 is first reduced to UF4. By calcium reduction and remelting the metal is cast into 5 kg ingots.

In an independent melting furnace, isotopic adjustment is performed under molten metal form. Each batch represents approximately 30 kg U, cast into small ingots of 300 g U.

b) Wet process

The input products are first dissolved into uranyl nitrate, after possible roasting. Other uranium bearing chemical forms may demand more complex treatment.

The obtained solution undergoes, if necessary, one or two solvent extraction purification cycles in Mixer-Settlers using TBP.
Isotopic adjustment can be performed on this purified solution.

The following steps of the process are :

- precipitation in the form of ADU and calcination in the form of U308
- reduction in the form of UO2 (a sinterable product which can be used to fabricate fuel pellets).
- fluoridation into UF4 (the UF4 thus produced returns to the dry process for metal production by calcium reduction).

c) Licenses

The facility is licensed to process uranium on the entire isotopic content range, from depleted uranium up to 99 % U 235. In terms of radioisotopic polluants, the facility is authorised to process uranium with :

Transuranic elements	<	250	Bq/gU
Fission Products	<	9250	Bq/gU
U 232	<	5	ppb/U.

INPUT PRODUCTS

A great variety of products are processed in this facility. They originate from defabrication of unirradiated fuels or are scraps from fuel fabrication.

From fuel plates : UAl, UAl_x, $U_3 Si_2$, U/Nickel

From fuel pellets : UO_2, UO_2/Steel

Reprocessed uranium as uranyl nitrate has also been processed.

Processing of special fuels is under development : UZr, UC_2

OUTPUT PRODUCTS

Most of the production is under the form of U metal, generally 19.75 % enriched, for reuse by Fuel Fabricators.

Large quantities of U308 have also been produced. U308 is a convenient chemical form for transportation and this easy to dissolve chemical form is generally preferred for later use.

Sinterable UO2, for pellets production is required by some reactors.

On request, U metal of any isotopic level can be obtained.

CONCLUSIONS

The Recycle and Conversion Facility of COGEMA Pierrelatte is a modular, versatile facility, well suited to recycle existing stockpiles or fuel fabrication scraps. Its long past experience of more than 30 years will be now fully made available to the civil sector. Specific needs can be examined and if necessary development work undertaken by a Research and Development team working on the Site.

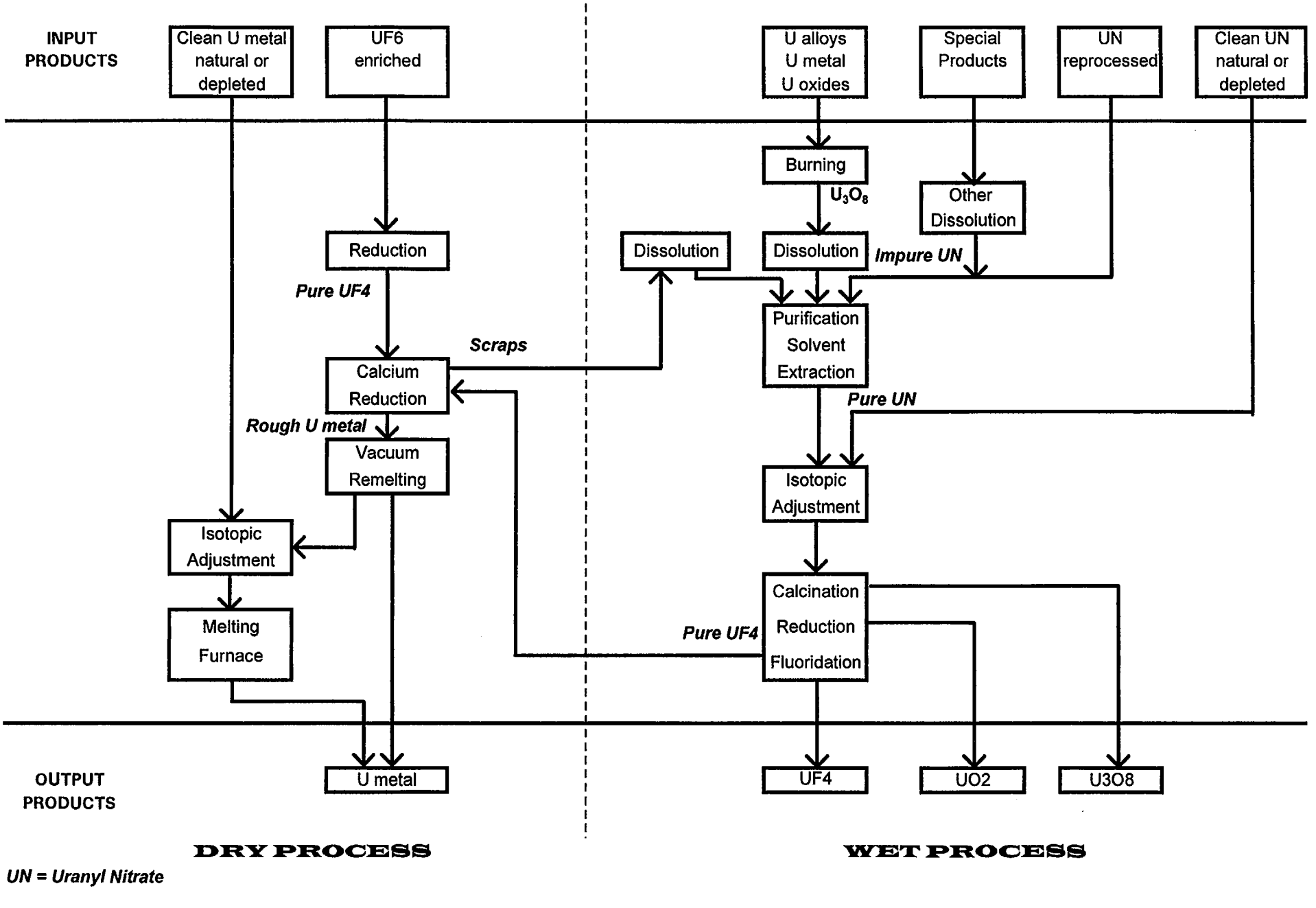

Fig.1 COGEMA PIERRELATTE HEU RECYCLE AND CONVERSION FACILITY

NOVOSIBIRSK CHEMICAL CONCENTRATES PLANT, INC. IS A MANUFACTURER OF NUCLEAR FUEL AND FISSILE MATERIALS FOR RESEARCH REACTORS

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NCCP, Novosibirsk

Novosibirsk Chemical Concentrates Plant, Inc (NCCP) was founded in 1948. During this period the plant worked the way from concentrates reprocessing and uranium and its compositions production to compounds fabrication of fuel assemblies (FAs) for power, research reactors and other types of reactors. The accumulated experience of handling nuclear materials, fuel element (FE) design, developed technologies, such traditions as to place Customers interests over the own ones, to work providing quality assuarance and high engineering level of items are realized in the production of FAs for research reactors.

In 1973 NCCP began to deliver FEs and Fas for majority of research reactors built according to the Russian projects both on the former USSR territory and beyond it (See Appendix, fig.1).

In contrast to the plate type FEs used in most foreign research reactors, the FE design of Russian origin was based on three-layer seamless tubular design where middle layer is a fuel composition and outlying layers are claddings. It is possible to meet any and all Customers requirements to the FA operational characteristics by means of changing the profile and the tube wall thickness, the active layer and cladding thickness. The technological basis of such design principles realization is considered to be the development of technologies of hot coextrusion and gas reduction which were patented and implemented in Russia.

With the aim to improve FA technical level and quality, NCCP in cooperation with Institutes developing FEs and FAs, such as Research and Developement Institute of Power Engineering (RDIPE), All-Russian Institute of Inorganic materials (VNIINM), Kurchatov Institute (RNC KI) are constantly improving the FE and FA design. In the course of our items modification the FAs with EK-10, S-36 rod type fuel elements have developed in to FAs with tubular FEs with double-side heat-bearing agent cooling. These are VVR-M2, IRT- 2M, MR FAs of second generation with ca. 150 g/dm^3 uranium concentration in FA volume and ca. 3,5 cm^2 /cm^3 specific heat-pickup surface. Further the conversion was made to the third generation FAs with thin- wall tubular FEs of the VVR-M 5, IRT-3M, IVV-10 and other types with up to 350 g/dm^3 uranium concentration in the reactor core volume unit and up to 6,5 cm^2/cm^3 specific heat-pickup surface.

Uranium-aluminium alloy that made it possible to increase the uranium concentration up to 1,3 g/cm^3 was used as a fuel component of the first generation FEs. The further uranium concentration increase became possible thanks to the conversion to ceramic uranium-aluminium fuel core.

It was decided at the NCCP in cooperation with the VNIINM to use as the fuel core component the uranium dioxide that makes it possible to provide for practically wasteless technology. Such choice simplifies resolution of tecnological problems pertaining to the spent nuclear fuel utilization.

In the 70 ies when starting to use the ceramic fuel core, the uranium concentration was increased up to 1.7 g/cm^3. In the 80 ies the 2.5 g/cm^3 concentration was mastered.

The improvement of FA technical level proceeds with the requirement of maintaining FA high quality level. The certified quality assuarance system being in effect at the plant conforms to the international ISO 9000-9004 standard and guarantees high quality of FAs produced at the NCCP. The available quality control program is based on the multi-parameter control of FEs and FAs quality at different production stages. Thorough technological control of FE state during the intermediate operations is performed. FE routine control consists of the assessment of visual appearance, cladding material structural state, control of fuel layer length and its continuity, as well as fuel layer homogeneity, control of cladding thickness and fuel, control of cladding-to-core adjoining quality. All these types of control are performed using non-destructive methods. In addition to non-destructive methods the periodical quality control is carried out by means of destructive methods.

The created control system demonstrated full correspondence of FEs to the specification requirements and provided their operational capability. Since 1973 till the present time our plant has produced more than 25000 FAs with 56 modifications for 31 reactors; and with pride we state the fact that the plant received no claims from the Cuctomers for replacement of defective goods.

NCCP production complex producing FAs for research reactors includes the following productions : chemical-metallurgical production, production of alluminium component parts, instrumentation production, central plant laboratory, engineering scientific-technical departments.

- Chemical-metallurgical production includes uranium hexafluoride processing, removing different impurities from uranium containing mixtures, production of uranium monoxide-oxide, dioxide, tetrafluoride, uranium metal and its alloys. Typical quality of uranium compounds produced at the plant is presented in the table. (See Appendix, fig. 2.) It is possible to provide for individual terms of delivery and characteristics of uranium compounds according to users requirements.

- Production of FAs for research reactors includes FEs fabrication (See Appendix, fig. 3) and their assembling. At present NCCP, Inc. manufactures the following types of research reactor FAs. (See Appendix, fig.4).

- The production of aluminium component parts where the component parts for FEs and FAs are fabricated is available at the plant. It includes extrusion of required profile billets conforming to component parts profile; and subsequently they are machined to size by means of metal-cutting equipment. (See Appendix, fig. 5).

- Tool making production meets demands in high-precision technological & measuring instruments of all manufacturing divisions of the NCCP.

- Central Plant Laboratory is a testing centre certified by GOSSTANDART of the Russian Federation in the National System of Testing Laboratories GOST R; it fulfills certification of all items of chemical-metallurgical production.

- Engineering scientific-technical departments staffed with high-skilled specialists provide development and maintaining of a high level FAs design and technology and carry out research works in regard to putting into production prospective achievements of science and engineering.

Nowadays, NCCP, Inc. in cooperation with VNIINM is developing and mastering production process of FEs based on uranium dioxide with fuel concentration of up to 4 g/cm^3 in terms of total uranium, with up to 20% U_{235} enrichment; it is fulfilled within the framework of Russian Research Reactors Conversion Program. Fabricated pilot batches of FAs are delivered for performing in-pile tests.

In addition to FAs for research reactors our plant produces FAs for power, pulse, industrial reactors, FEs and micro-FEs for gas-cooled reactors. On the basis of technologies developed at our enterprise the targets for Mo_{99}, Co_{60}, Ir_{192} production, irradiation-alloyed silicon and dozens of other radionuclides are turned out. Moreover, our plant carries out works on usage of more uranium-consuming compositions on the basis of uranium silicide and UZrNb, and produces different in-pile instrumentation with or without uranium.

The production process is equipped with up-to-date installations for chemical, thermal treatment and machining of uranium materials with up to 91 % U_{235} enrichment, which guarantees nuclear safety and full biological protection of nuclear personnel.

NCCP, Inc. welcomes all domestic and foreign enterprises of nuclear fuel cycle to mutually benefitial cooperation and is ready to satisfy the additional Customers requirements to its produce.

THE RERTR PROGRAM

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INTRODUCTION

The Reduced Enrichment Research and Test Reactor (RERTR) Program was established in 1978 at the Argonne National Laboratory (ANL) by the Department of Energy (DOE), which continues to fund the program and to manage it in coordination with the Department of State (DOS), the Arms Control and Disarmament Agency (ACDA), and the Nuclear Regulatory Commission (NRC). The primary objective of the program is to develop the technology needed to use Low-Enrichment Uranium (LEU) instead of High-Enrichment Uranium (HEU) in research and test reactors, and to do so without significant penalties in experiment performance, economic, or safety aspects of the reactors. Research and test reactors utilize nearly all the HEU that is used in civil nuclear programs, either in their fuel or in irradiation targets for the production of medical radioisotopes. Eliminating the continuing need of research and test reactors for HEU supplies and usage would remove a serious nuclear proliferation concern, and has been for many years an integral part of U.S. nonproliferation policy. This paper reviews the main accomplishments of the program through the years, emphasizing last year's progress and the next planned activities.

DISCUSSION

1. Fuel Development

An important goal of the RERTR program is to develop new LEU fuels with much higher uranium density than those used in the past with HEU. In the course of this work, the qualified uranium densities of the three main fuels which were in operation with HEU in research reactors when the program began (UAl_x-Al with up to 1.7 g U/cm^3; U_3O_8-Al with up to 1.3 g U/cm^3; and $UZrH_x$ with 0.5 g U/ cm^3) have been increased significantly. The new qualified uranium densities extend up to 2.3 g U/cm^3 for UAl_x-Al, 3.2 g U/cm^3 for U_3O_8-Al, and 3.7 g U/cm^3 for $UZrH_x$. Each fuel has been tested extensively up to these densities and, in some cases, beyond them. All the data needed to qualify these fuel types with LEU and with the higher uranium densities have been collected.

New important fuel types have also been developed. For U_3Si_2-Al, which is the most important fuel so far developed by the program, the U.S. Nuclear Regulatory Commission (NRC) has issued a formal approval[1] of its use in research and test reactors with uranium densities up to 4.8 g/cm^3. A whole-core demonstration using this fuel was successfully completed in the ORR using a mixed-core approach. Plates with uranium densities of up to 6.0 g/cm^3 have been fabricated by CERCA with a proprietary process, are being tested in SILOE, and will soon be tested in OSIRIS. For another new fuel, U_3Si-Al, miniplates with up to 6.1 g U/cm^3 have been fabricated by ANL and the CNEA and irradiated to 84-96% in the Oak Ridge Research Reactor (ORR). PIE of these miniplates have given good results, but have shown that burnup limits would need to be imposed for the higher densities. Four full-

size plates fabricated by CERCA with up to 6.0 g U/cm^3 have been successfully irradiated to 53-54% burnup in SILOE, and a full-size U_3Si-Al (6.0 g U/cm^3) element, also fabricated by CERCA, has been successfully irradiated in SILOE to 55% burnup. However, conclusive evidence indicating that U_3Si becomes amorphous under irradiation has convinced the RERTR Program that this material could not be used safely in plates beyond the limits established by the SILOE irradiations.

RERTR activities related to the development of fuel plates with much higher effective uranium loadings were interrupted in 1989 by DOE guidance to concentrate on the implementation of the fuels which had already been developed by that time. This guidance, however, was reversed during the past year. Near the end of March 1996, after a pause of over six years during which no fuel development could be pursued by the RERTR program, the U.S. Department of Energy provided both the funding and the guidance needed by the program to resume the development of advanced LEU fuels to improve the options and performance of research reactors undergoing conversions. The DOE action, which provided $1.4 million for fiscal year 1996, was supplemented by a $1.5 million contribution from the US Department of State. DOE plans to support this effort with $3.0 million in fiscal year 1997 and has committed itself, in written testimony to Congress, to continue to support the fuel development effort for several years until its objectives are met.

As a result of these events, the RERTR program is now deeply involved, for the first time in many years, in fuel development activities. The first and foremost task is to reestablish a cadre of skilled fuel developers building on the expertise of the previous fuel development effort. Two groups with complementary strengths have been formed for this purpose, one at the Argonne-East site in Illinois and the other at Argonne-West in Idaho. Orders have been placed to procure new equipment, several uranium-bearing materials have been evaluated for their potential application in high-density fuels, and a preliminary plan has been developed for the initial phase of the fuel development effort. Initial efforts will be concentrated on the production of microplates containing dispersion fuels formed with U-Mo and U-Zr-Nb alloys, in addition to other uranium compounds, in combination with various matrix materials. Irradiation of the first microplates is planned to begin in the Advanced Test Reactor (ATR), in Idaho, during April 1997.

2. Reactor Analysis

The RERTR program has upgraded, modified, and developed many methods and computer codes to assess the performance and safety aspects of research reactors using the LEU fuels developed by the program. These methods and codes address many aspects of research reactor operation, including neutronics, fuel cycle, thermal-hydraulics, transient analysis, and radiological consequences. Many generic and specific analyses have demonstrated the validity of these methods. The results were published in three IAEA Guidebooks (TECDOC-233 for light-water research reactors, TECDOC-324 for heavy-water reactors, and TECDOC-643 for safety and licensing). The program's computational and design capabilities have created a standard which is internationally recognized.

Extensive studies have been conducted, with favorable results, on the performance, safety, and economic characteristics of LEU conversions. These studies include many joint study programs, which have been in progress for about 29 reactors from 18 different countries. A study to assess the feasibility of using LEU in the fuel of a modified version of

the FRM-II reactor, which is being designed with HEU at the Technical University of Munich, has identified an alternative LEU design which could provide the same experiment performance and the same fuel lifetime as the HEU core currently planned for the FRM-II.

The RERTR program has also provided coordination of the safety calculations and evaluations for the US university reactors planning to convert to LEU as required by the 1986 NRC rule. In addition, during the past year the RERTR program was tasked by the Department of Energy to assess the feasibility of converting to LEU each of the DOE facilities which currently use HEU.

3. ^{99}Mo Production from LEU.

The RERTR program is pursuing an analytical/experimental program to determine the feasibility of using LEU instead of HEU in fission targets dedicated to the production of ^{99}Mo for medical applications. ^{99}Mo is by far the most important medical radioisotope which is currently produced in research reactors through the use of HEU targets. The goal of the program is to develop and demonstrate during the next few years one or more viable technologies compatible with the processes currently in use with HEU at various production sites throughout the world. This activity is conducted in cooperation with several other laboratories including the University of Illinois and the Indonesian National Atomic Energy Agency (BATAN). Procedures have been developed for dissolution and processing of both LEU silicide targets and LEU metal foil targets. These procedures are ready for demonstrations on full-size targets with prototypic burnups.

Significant progress was achieved in this area during the past year. For the first time, LEU metal-foil target prototypes were irradiated and, after irradiation, were easily extracted from other target materials for separate dissolution and processing. This constituted a very important milestone in the development of a viable process based on the use of LEU metal-foil targets.

4. Russian RERTR Program

The scope and main technical activities of a plan for the equivalent of a Russian RERTR program have been agreed upon by the RERTR Program and several major Russian institutes led by the Research and Development Institute for Power Engineering (RDIPE). The objective of this program is to develop and demonstrate within the next five years the technical means needed to convert from HEU to LEU fuels approximately 26 research reactors designed and supplied by institutes of the Russian Federation. The main Russian institutes taking part in this cooperative undertaking, beside RDIPE, are the All-Russia Research and Development Institute of Inorganic Material (VNIINM), the Novosibirsk Chemical Concentrates Plant (NZChK), and the Yekaterinburg Branch of RDIPE. Both Mrs. Hazel O'Leary, Secretary of the U.S. Department of Energy (DOE), and Mr. Viktor N. Michailov, Minister of Atomic Energy of the Russian Federation (MINATOM), have expressed strong support for this initiative which has expanded the scope of the RERTR program and enabled it to address the problems created by use of HEU in civil nuclear programs nearly everywhere in the world.

5. Reactor Conversions.

Twelve research reactors outside the U.S., which used to require HEU supplies of US origin when the program began, have been fully converted to the use of LEU fuels. These reactors include ASTRA (Austria), DR-3 (Denmark), FRG-1 (Germany), JMTR (Japan), NRCRR (Iran), NRU (Canada), OSIRIS (France), PARR (Pakistan), PRR-1 (Philippines), RA-3 (Argentina), R-2 (Sweden), and THOR (Taiwan).

Nine research reactors in the U.S. have also been fully converted to the use of LEU fuels. These reactors reactors include FNR (Michigan), RPI (New York), OSUR (Ohio), WPIR (Massachusetts), ISUR (Iowa), MCZPR (New York), UMR-R (Missouri), RINSC (Rhode Island), and UVAR (Virginia).

Three foreign reactors, including IEA-R1 (Brazil), SSR (Romania), and TR-2 (Turkey), have been partially converted, and two more, GRR-1 (Greece) and HOR (Netherlands), have fabricated LEU cores. Safety evaluations for four additional domestic reactors have been completed, and calculations for four more reactors are in progress. Approximately 60% of the work required to eliminate use of HEU in US-supplied research reactors has been accomplished

6. Spent Fuel Disposition

The need to ensure that the fuel irradiated in research and test reactors can be properly disposed of has always ranked high among RERTR program priorities. In 1983, early reprocessing studies at the Savannah River Laboratory concluded that the fuels then developed by the RERTR program could be successfully reprocessed at the Savannah River Plant, and DOE had defined the terms and conditions under which those fuels would be accepted for reprocessing. These results were rendered moot, however, by DOE's decision to phase out reprocessing at the Savannah River Plant and by the expiration of the Off-site Fuel Policy at the end of 1988.

Since then, the RERTR program has contributed to the best of its ability to the resolution of this crucial problem affecting many of the reactors with which it was cooperating. After overcoming significant legal obstacles, 153 urgent-relief elements from several European countries were received at the Savannah River Site under the provisions of an Environmental Assessment published in April 1994. A second urgent-relief shipment of 99 spent fuel elements from Greece and Switzerland was received in November 1995, still under the provisions of the Environmental Assessment. Publication of the Final Environmental Impact Statement[2] for the return of spent research reactor fuel occurred in February 1996 and was followed, on May 13, 1996, by publication of the related Record of Decision.[3] These documents, and the favorable conditions which they established for the return of spent research reactor fuel, paved the way for a great number of fuel shipments which are expected to eliminate, over a thirteen-year period, the large inventories of spent fuel which currently fill the pools and storage facilities of many research reactors. This action will resolve urgent operational problems of the reactor sites while, at the same time, eliminating a serious proliferation concern. The first shipments under the provisions of the Record of Decision, containing 280 spent research reactor fuel elements coming from reactors in Germany, Switzerland, Sweden, Chile, and Colombia, were received at the Savannah River Site during September 1996.

PLANNED ACTIVITIES

1. Complete the orders for the new fuel fabrication equipment needed to develop advanced fuels, and set up the fuel fabrication laboratory at ANL-W.
2. Produce a first series of microplates, including samples of the main materials of interest for the advanced fuel development.
3. Conduct out-of-pile tests on some of the fuel materials, to assess their properties and likely performance.
4. Begin irradiation testing of microplates in the ATR.
5. In collaboration with the Russian RERTR program, continue to implement the studies, analyses, fuel development, and fuel tests needed to establish the technical and economic feasibility of converting Russian-supplied research and test reactors to the use of LEU fuels.
6. Continue calculations and evaluations about the technical and economic feasibility of utilizing reduced-enrichment fuels in reactors that require such assistance, and in reactors of special interest.
7. Continue development of one or more viable processes, based on LEU, for the production of fission ^{99}Mo in research reactors.
8. Complete testing, analysis, and documentation of the LEU fuels which have already been developed, support their implementation, and transfer their fabrication technology to countries and organizations which require such assistance.

SUMMARY AND CONCLUSION

The RERTR program has made, and continues to make, excellent progress. Close cooperation with international organizations has been the cornerstone of the program since its inception. This cooperation and the high quality of the technical contributions which many partners have brought to the overall effort are to be credited for much of the progress which the program has achieved.

The past year, in particular, has marked a turning points for several important RERTR program activities, and has left the program well prepared to face the challenges ahead.

(a) In the area of **U.S. acceptance of spent fuel** from foreign research reactors, a second shipment of 99 urgent-relief spent fuel elements was completed. The Final Environmental Impact Statement was published in February 1996, and the Record of Decision was published in May 1996. The first shipments under the Record of Decision were received at the Savannah River Site in September 1996.

(b) In the area of **advanced fuel development**, adequate funding and guidance were received by the RERTR program near the end of March 1996. Fuel development activities are now in progress, including procurement of equipment, screening of candidate materials, and preparations for the production of a first series of microplates. The first irradiations are planned to begin in the Advanced Test Reactor, in Idaho, during April 1997.

(c) In the area of **reactor analyses**, the RERTR program has been tasked by the Department of Energy to assess the feasibility of converting to LEU fuel each of the DOE research reactors which currently use HEU fuel. A preliminary assessment resulting from this study has been prepared. The study of an alternative LEU core for the FRM-II design has been extended to address, with excellent results, several controversial performance and safety questions.

(d) The **Russian RERTR program**, which aims to develop and demonstrate within the next five years the technical means needed to convert Russian-supplied research reactors to LEU fuels, has made significant progress.

(e) Significant progress was made on several aspects of ^{99}**Mo production** from fission targets utilizing LEU instead of HEU. In particular, LEU metal-foil target prototypes were irradiated and, after irradiation, were easily extracted from other target materials for separate dissolution and processing. This was an important milestone in the development of a viable process based on the use of LEU metal-foil targets.

The most important events affecting the program's future are unquestionably those related to the return of the spent fuel and to the advanced fuel development.

The Record of Decision, and the successful completion of the first related shipments, brings to an end a long period of concern for many research reactors. However, it must be noted that the U.S. spent fuel acceptance policy is valid only for ten more years of operation. Reactor operators would be wise to plan now for the final disposition of their spent fuel when the ten years have elapsed.

Resumption of advanced fuel development means that new fuels can be developed to enable conversion of the reactors which cannot be converted today, to ensure better efficiency and performance for all research reactors, and to allow the design of more powerful new advanced LEU reactors. Because of the provision of the Atomic Energy Act, as amended in 1992, it also satisfies one of the conditions that would enable the U.S. to export HEU for the operation of those reactors which need it.

We are beginning our new tasks in earnest. Old agreements are been renewed, and new ones are being forged. Especially in the fuel development area, our success will depend, as in the past, on cooperation and free exchange of ideas and information. Once more, I ask for the international friendship and cooperation that have been a trademark of the RERTR program since its inception, so that we can attain together President Clinton's goal to minimize and eventually eliminate the use of highly-enriched uranium in civil nuclear programs.

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Session 2:

Fuel Fabrication

DIFFERENT TYPES OF RESEARCH REACTOR PLATE FUEL AND THEIR POSSIBLE OPTIMISATION

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1- INTRODUCTION

For new reactors, and of course for existing reactors which need to be upgraded, there is an important variety of design choices to be done on the fuel point of view.
Parameters such as power density, neutron flux distribution, irradiation cycle, reactivity of the fuel during irradiation, can be optimised.

The aim of this paper is to provide an inventory of fuel manufacturing technical possibilities. This inventory cannot be exhaustive but, at least, gives a guideline for conception or optimisation of fuels.

2- DIFFERENT TYPES OF FUEL COMPOUND

Nowadays, mainly two metallurgical compounds are used as uranium fuel, namely UAlx (UAl_3 + UAl_4) or U_3Si_2. Those fuels can be used with enrichment such as 19,75 %, 90 % and 93 % U-235. In the case of uranium coming from reprocessing, an **intermediate enrichment can be considered** leading to an increase in the manufacture cost because of the non standard enrichment which request a specific management from Uranium account point of view .
UAlx fuel compound can be used to obtain uranium densities up to 2,2 g U_T/cm^3 rather than U_3Si_2 allows to reach 6 g U_T/cm^3.

Uranium amount within a plate is adjusted by means of volume fraction of fuel compound in the given meat volume, leading to a corresponding Uranium density. For a same density, U_3Si_2 fuel plates will contain less volume of fuel compound and less porosity as compared to UAlx fuel. Consequently, U_3Si_2 fuel plates are expected to have a better thermal conductivity for a same Uranium content as compared to UAlx plate.

3- DIFFERENT TYPES OF FUEL MEAT

The meat of the plate is in fact a dispersion of Uranium compound (UAlx or U_3Si_2) within an Aluminium matrix. The meat itself is cladded with Aluminium alloys. CERCA has a large background of the use of A5, AG3 NE or AlFeNi cladding alloys. But, for special cases new claddings can be developed or tested.

For the optimisation of the fuel performances, several configurations can be considered concerning the meat.

Increase in Uranium content :

If the aim is to increase the uranium content per plate **without changing the external dimensions**, three technical ways can be used. First, increase in the volume fraction of fuel particles leading to an increase in Uranium density (a maximum of 6 g U_T/cm3 with U_3Si_2 fuel compound can be reached). Second solution, increase in the volume of the meat leading to a reduction of the average cladding thickness. Third solution, the use of U_3Si_2 instead of UAlx (because it contains more Uranium, 92 wt % instead of 71wt %).

Uranium distribution :

If the aim is to adjust the U-235 distribution in the plate, as a function of flux performances, mainly two solutions can also be considered. First, the meat is composed of several areas which have their own Uranium density (g U_T/cm^3). In this case, the fuel meat thickness remains constant. This is the case of the FRM-II fuel reactor design. Second solution : the Uranium density (g U_T/cm^3) remains constant but U-235 distribution is adjusted by means of meat thickness which is varying. This was the case of the ANS fuel reactor design.

4- DIFFERENT TYPE OF FUEL ASSEMBLIES

For optimisation purposes, it can be considered to change the ratio H/U. This is directly connected to the plate (and meat) thickness and the water gap.

From the plate thickness point of view, there are almost no practical upper limits. For the time being, the minimum plate thickness is about 1,27 mm with a meat thickness of 0,51 mm but there are no technical reasons to not reduce slightly furthermore those values which are considered as a standard.

Concerning the water gap, which is a cooling and moderator parameter, there are no practical upper limits whatever fuel design is chosen (square, cylindrical, involute fuel shape).
However, the minimum water gap is related to the assembly technology and the fuel shape. From our background, for square fuel shape, below 2 mm, special tools must be developed and tests must be carried out.
For cylindrical fuel shape, a minimum water gap of 3 mm is considered to be a reasonable limit. In other respects, a diameter of about 30 mm for the inner fuel plate corresponds to a practical limit from the plate and fuel manufacture point of view. Below those values, manufacture test must be carried out.
For involute fuel shape, a minimum water gap of 1.8 mm is expected to be a reasonable limit.

5- DIFFERENT NEUTRON POISONS

In order to adjust the reactivity of fuel assemblies all along irradiation or to decrease locally the fuel reactivity, it is possible to load fuels with neutron absorber or poison in different locations of the fuel such as :

1- Cadmium wires in the side plates all along the fuel plate.

2- Boron or Samarium at the end of the fuel meat, or within the fuel meat itself, or in the side plates.

It must be pointed out that the amount of Boron or Samarium can be adjusted as needed. Nevertheless, in the case of neutron absorbers located within the fuel meat, this solution leads to an extra cost of reprocessing due to an additional purification step needed to get back Uranium available for fuel manufacture.
If Boron is chosen as neutron poison, it can be Boron natural or enriched with Boron 10.

6- INSTRUMENTATION

Most of the time, the design of an improved fuel is followed by irradiation tests of prototypes.
In order to compare calculations forecasts and real behaviour, fuel prototypes can be equipped with measurement devices such as pito tubes for pressure measurements or with thermocouples for temperature measurements.

More particularly, CERCA has developed an original manufacture procedure allowing to locate thermocouple within the plate cladding itself. With that technique, no thermocouples can be seen from the outside of the fuel plate. Consequently, no flow perturbation is introduced.

7- CONCLUSION

This paper gives technological ways available to improve fuel with their manufacture limits. Nevertheless, each case is a particular case, therefore the global limits given as a guideline must be analysed regarding the tolerances requested by the fuel designer.

A partnership study is strongly recommended with the fuel manufacturer in order to reach the most suitable solution on the cost and technical point of view.
In this case, CERCA offers it's 30 years background in the fuel manufacture with a very large variety of fuel (more than 60 types of MTR fuel plate assemblies).

In some cases, if it is needed to go beyond the global given limits, R&D team of CERCA is able to assume specific programs in connection with his customers.

THE MANUFACTURE OF MTR FUEL ELEMENTS AND Mo^{99} PRODUCTION TARGETS AT DOUNREAY

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INTRODUCTION

Uranium/aluminium alloy elements have been produced at Dounreay for nearly 40 years. In April 1990 the two DIDO-type reactors operated by the United Kingdom Atomic Energy Authority (UKAEA) at Harwell were closed, with the result that a large portion of the then current customer base disappeared and, to satisfy the needs of the evolving market, the decision was taken to invest over £1m in new equipment for the manufacture of dispersed fuels and molybdenum production targets.

BACKGROUND

Until 1990, the fuel elements produced by the UKAEA were of the following types:

1. Tubular-type alloy elements for reactors in the United Kingdom, Germany and Australia:

    - with extruded fuel tubes (approximately 4000 elements produced)
    - with the plates electron beam welded to form fuel tubes (approximately 5000 elements produced)
    - with the plates roll swaged into continuous combs to form a tube-like structure (<50 elements produced).

2. Plate-type alloy elements for reactors in the United Kingdom, South Africa and South America:

    - with flat plates assembled into a box section (approximately 200 elements produced)
    - with either flat or curved plates roll swaged into slotted side plates to form a box-like structure (approximately 800 elements produced).

Combined with the decision to invest in the dispersed fuel line, there was a commitment to develop new techniques and products such as:

1. dispersed High Enriched Uranium (HEU) (uranium aluminide) elements
2. dispersed Low Enriched Uranium (LEU) (uranium silicide) elements
3. dispersed HEU targets for Mo^{99} production (uranium aluminide).

The production of dispersed fuel was a new product for the UKAEA and required the investment of considerable time and technical expertise for the development of new manufacturing techniques. This led to the successful completion of a number of projects including:

1. 6 HEU qualification elements (produced from re-cycled HEU) for the BR2 reactor in Mol in Belgium. These elements have successfully undergone the necessary irradiation to qualify the product.

2. 6 HEU qualification elements for irradiation in the High Flux Reactor (HFR) in Petten in Holland.
3. 2 HEU qualification control elements for irradiation in the HFR in Petten.
4. 2 LEU (silicide) qualification elements for irradiation and Post Irradiation Examination in the HFR in Petten. These elements are currently being irradiated.
5. 1700 Mo^{99} production targets (flat plates) produced for ECN, Petten.

UKAEA continue to fabricate fuel elements for customers based on the alloy production route and intend to fabricate further elements using the dispersed fuel line.

The overall market for MTR fuel elements is reducing, however the UKAEA has a good potential baseload for the manufacturing plant for many years and intends to maintain a manufacturing capability to continue to satisfy the needs of the MTR community.

CURRENT PRODUCT DETAILS

The production of dispersed-type and alloy-type fuel plates and targets

The production of both types of fuel consists of the following processes:

Dispersed Fuel	**Alloy Fuel**
Uranium metal casting and size reduction	Uranium metal casting and size reduction
UAl_x/U_3Si_2 button production	U/Al alloy billet casting
UAl_x/U_3Si_2 powder production	Fuel core blanking
Powder sieving, weighing and blending	Fuel pack assembly
Powder compaction	Fuel pack rolling
Fuel pack assembly	Fuel plate blanking
Fuel pack rolling	Fuel plate inspection
Fuel plate blanking	
Fuel plate inspection	

All fuel element and molybdenum target production in the plant begins with the primary casting of uranium billets (typically three billets weighing 2.5 kg each) into rods 20 mm diameter. The billets are loaded into a graphite crucible and induction heated under vacuum, the heating rate being closely controlled to allow the evolution of gas from the billets. When gas evolution is complete, after two to three hours, the furnace temperature is raised until the uranium has melted. The melt is then bottom-poured into a graphite mould. When the mould has cooled, the uranium product, in the form of 20 mm diameter rods, is cropped into short lengths suitable for further processing.

ALLOY-TYPE FUEL PLATE PRODUCTION

The correct proportions of aluminium and uranium, of the required enrichment, are first weighed out and then cast into a uranium/aluminium billet using a similar method to that for primary casting. Gas evolution is not a problem with alloy casting but it is important to obtain complete solution of the uranium in the aluminium to obtain an even distribution of uranium in the alloy billet. This is achieved by superimposing a 50 Hz component on the high frequency induction coil. Eddy currents are induced in the molten metal and good stirring of the alloy is

achieved. The temperature is then raised to approximately 100 Celsius above the melting point of the alloy and the charge is then bottom-poured into a rectangular graphite mould.

After cooling, the billet is removed from the mould, weighed and transferred to an air re-circulating oven where it is heated to 600 Celsius and rolled down to the required thickness on a rolling mill. Rectangular fuel core plates are then blanked out from the rolled strip.

Each fuel core plate is then checked for U^{235} content by gamma counting. The results of the gamma count are compared with those from an accurately calibrated standard core to give the precise U^{235} content of each fuel core plate to an accuracy of 0.01g. One fuel core plate in one hundred produced is dissolved and analysed to confirm the U^{235} content.

Each fuel core plate is then assembled in a pure aluminium frame and between two pure aluminium cladding plates. Prior to assembly, the surfaces of the cladding plates, frame and fuel core plate are scratch brushed and degreased to remove the oxide layer, which forms on the surfaces of aluminium, to ensure that a good metallurgical bond is obtained at the component interfaces during fuel plate rolling. The assembly is then machine argon arc welded along the longitudinal edges.

The welded fuel packs are heated to 600 Celsius and rolled to the required thickness (approximately 1.5 mm) following a rolling schedule that produces a fuel plate containing a fuel core of the required length, width and thickness (around 650 mm x 50 mm x 0.7 mm but depends on customer requirements). Rolled fuel plates are placed in a Core Location Unit where the outline of the fuel core within the cladding is revealed on an x-ray fluoroscope and positioned against an appropriate graticule. If the fuel core outline is in accordance with the manufacturing drawing, locating holes are punched in the surplus aluminium at the ends of the fuel plates. These holes match locating pins in tools on a blanking press where the plates are blanked to final size.

FUEL PLATE INSPECTION

All components undergo rigorous inspection at each stage of the manufacturing process.

After rolling, the fuel plate is subjected to a blister test by heating to 425 Celsius and soaking at this temperature for one hour. After cooling, the surfaces of the plate are visually examined for blisters. The purpose of this test is to check the bond between the fuel core and the aluminium cladding. Any area of dis-bond may contain air which, after heating of the plate, will expand, leading to the formation of a blister on the surface. The integrity of the bond is also checked using ultrasonic inspection equipment. Each fuel plate can be checked against a standard which can be used to ensure that any defects or indications are within specified limits.

Each plate is also radiographed to allow the fuel core dimensions to be checked against the manufacturing drawing and to check that uranium or aluminium segregation in the fuel core is within specification. The radiographs are also assessed for uranium homogeneity. Any suspect area can be checked using a gamma counter to ensure that the uranium concentration is within specified limits. The gamma counter is also used to ensure that the distribution of uranium at specified positions in the fuel plate is within specified limits.

After chemical cleaning, the fuel plates are checked for alpha contamination before they undergo final inspection. This involves a dimensional check and an examination of the plate surfaces for defects such as inclusions, scratches or indentations.

Samples are taken from 1% of fuel plates produced for a destructive examination of the core and cladding thickness. The samples are punched from the fuel plates, mounted in a resin block and then polished and etched to allow examination using a microscope to ensure that the cladding thickness is within specified limits.

DISPERSED-TYPE FUEL PLATES AND Mo^{99} PRODUCTION TARGETS

Targets for molybdenum production are similar to the fuel plates for fuel elements but are much smaller measuring 150 mm x 40 mm x 1.5 mm. Each target plate has a UAl_x fuel core containing about 5.2g of HEU. Targets follow the same manufacturing route as dispersed-type fuel plates.

Production of dispersed-type fuel plates and targets begins with the arc-melting of uranium and aluminium to form UAl_x, for HEU aluminide fuel, and uranium and silicon to form U_3Si_2, for LEU silicide fuel, in the form of buttons.

The buttons are then ground into powder which, after impurity analysis, is sieved to obtain the required range of particle sizes. After sieving, sufficient material to give the required uranium content in one fuel plate is weighed out. Sufficient aluminium powder to make up the volume of the fuel core of one fuel plate is then blended with the fuel powder in a small aluminium container. All operations within the powder processing lines are carried out in an argon atmosphere which is continuously monitored to maintain the oxygen content at less than 1%.

Several of these containers are transferred to a hydraulic compaction press where, again under argon, the powder is carefully poured into precision machined dies and the press activated to follow an automatic sequence to produce fuel compacts of the required dimensions.

To reduce the incidence of blisters on fuel plates after the blister test, it is necessary to vacuum degas the compacts prior to fuel pack assembly. Each compact is then placed inside an aluminium frame which is then assembled between two aluminium cladding plates and welded to form a fuel pack.

The type of cladding material used for dispersed-type fuel plate manufacture requires the rolling temperature to be lower than that for alloy fuel. The fuel packs are heated to 425 Celsius and rolled to the required thickness, as with alloy fuel plate manufacture.

After core location and blanking to size, the fuel plates undergo similar inspection operations to alloy-type plates.

CLADDING MATERIALS

For alloy fuel, type 1050A aluminium (99.5% purity) is used. With dispersed fuel, there can be a tendency for fuel particles to penetrate the cladding. As a result, stronger cladding material is required and type 5251 aluminium alloy is used (1.7 to 2.4 % magnesium). For Mo^{99}

production targets, although a dispersed fuel, 1050A aluminium is used for the cladding. This is necessary to maintain the purity of the product after the molybdenum extraction process.

The rolling characteristics are different for different cladding materials. The bonding of the fuel plate components is more difficult with the aluminium alloys required for dispersed fuel due to the formation of magnesium oxide on the surfaces of the components at elevated temperatures which interferes with the bonding process. Different rolling parameters are required, ie. pre-heat and re-heat temperatures, soak times, number of passes through the rolls, and percentage reduction in roll gap between each pass and much time has been spent in determining the process variables.

The type of cladding material used also affects the shape of the fuel core within the aluminium cladding. The rolling process has the effect of localised thickening of the fuel core at the trailing end of the rolled fuel plate (a phenomenon known as dog-boning), which can result in a reduction of the thickness of the aluminium cladding. Cladding component design and plate rolling procedures must ensure that the effect of dog-boning is accounted for and the fuel plate cladding thickness is within specification

ELEMENT TYPES

The fuel elements produced are, in general, either of the concentric tubular or plate-type design.

- Concentric Tube Fuel Element

  Electron Beam Welded (see Fig 1)

  Three of the roll-bonded fuel plates are formed into 120^0 curves and then electron beam welded along the longitudinal edges to produce a circular fuel tube. Four tubes, of different diameters, are assembled concentrically using support combs to maintain coolant channel gaps between the tubes. This assembly is then enclosed in an aluminium outer tube and top and bottom end fittings to form a complete fuel element. The outer aluminium tube can be alloyed with a small quantity of boron (typically 0.4g).

  Roll-swaged (see Fig 2)

  Three curved fuel plates are assembled into three continuous slotted combs and grooves are cut along the length of the combs thus securing the fuel plates in position. The depth of the grooves is controlled to achieve the required swaging strength.

- Plate-type Fuel Elements (see Fig 3)

  The individual fuel plates (either flat or curved depending on customer requirements) are assembled into slotted side plates and then secured by roll-swaging. Typically up to 21 fuel plates are secured in this fashion to form a rectangular-section fuel box. End fittings are fixed to the box, by welding or riveting, to complete the fuel element assembly.

QUALITY ASSURANCE

To give our customers value we aim to achieve the highest quality and best performance and must out-perform our competitors whilst reducing costs. The MTR Fuel Fabrication Plant continues with the philosophy of providing accurate documentation to assure customers of quality products and compliance with agreed manufacturing methods and parameters.

As a result of these efforts, in July 1994 the Quality Management System of the MTR Fuel Fabrication Plant was approved by Lloyd's Register Quality Assurance Limited to ISO 9002:1994.

Customer audits indicate that we are achieving good results and continue to meet programmed delivery dates.

CONCLUSION

MTR fuel elements have been fabricated at Dounreay for nearly 40 years for customers on five continents. To date approximately 10,000 fuel elements have been manufactured, comprising a total of around 150,000 fuel plates.

The UKAEA has invested over £1m in new plant and equipment in support of dispersed fuel production and have developed the necessary manufacturing techniques to enable quality fuel of this type to be fabricated on a production basis. High levels of investment in recent years have been geared to reducing costs and developing new business as well as improving safety and quality.

The UKAEA will continue to work alongside customers and suppliers, offering technical support and endeavouring to respond appropriately to customer feedback, establishing and maintaining working relationships that ensure that such feedback is absorbed into our planning and quality management strategies. This is achieved by ensuring that our organisational structure enables us to be accessible, agile and responsive to customer needs and that key staff are available.

For the immediate future, three main priorities will shape our efforts - **safety**, **quality** and **cost**. The long term future of the MTR Fuel Fabrication Plant at Dounreay rests firmly in the belief that we have the vision, the people, the capabilities and the commitment to satisfy our customers needs well into the next century.

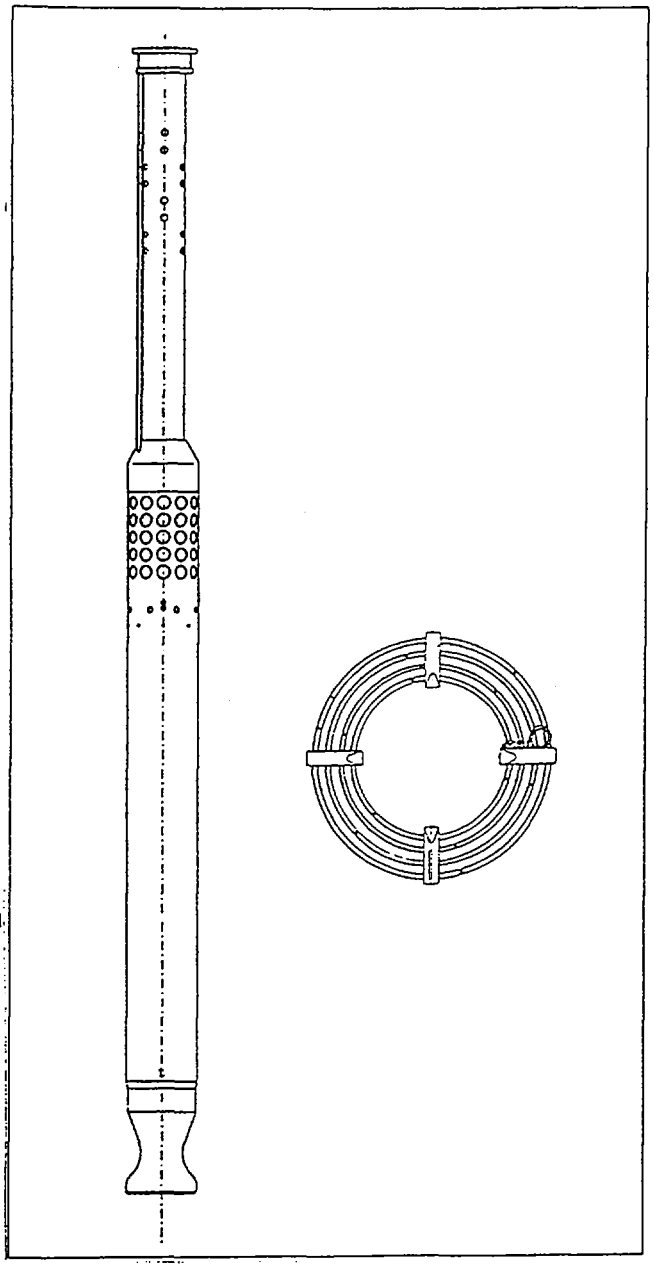

Fig. 1 Electron beam welded tubular type fuel element

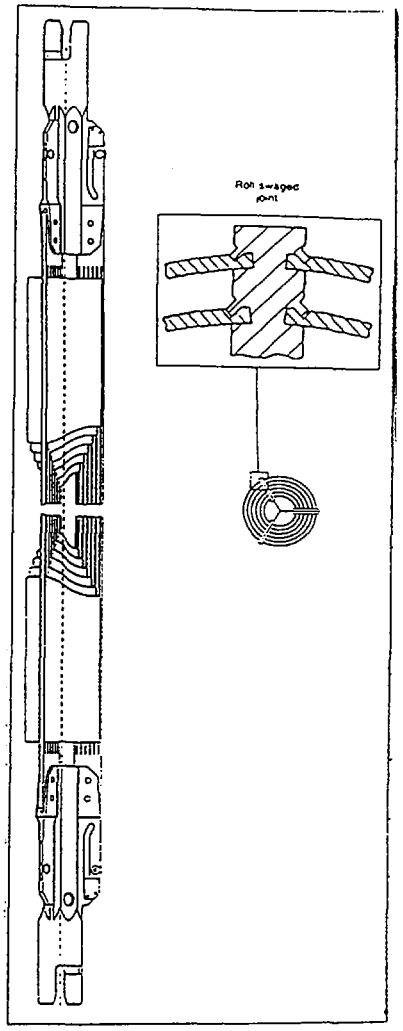

Fig. 2 Roll swaged tubular type fuel element

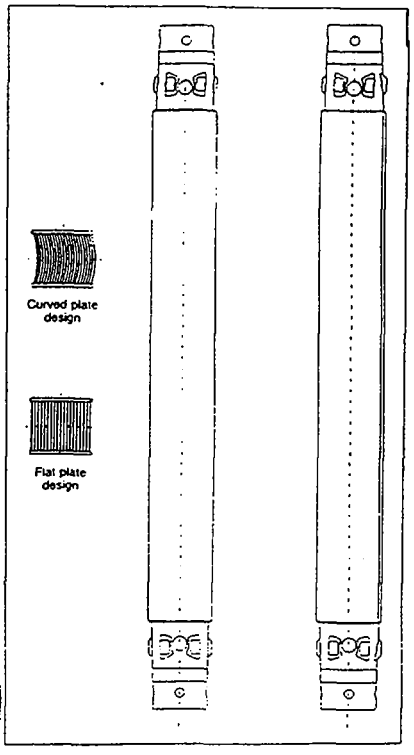

Fig. 3 Roll swaged box type fuel element

DEVELOPMENT OF VERY-HIGH-DENSITY LOW-ENRICHED-URANIUM FUELS

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INTRODUCTION

Fuel development has been a cornerstone of the U.S. Reduced Enrichment for Research and Test Reactors (RERTR) program since its inception in 1978. This development work, performed cooperatively with many partners internationally, resulted by 1987 in the qualification of several dispersion fuels having significantly increased densities: UAl_x-Al at 2.3 g U/cm^3, U_3O_8-Al at 3.2 g U/cm^3, and U_3Si_2-Al at 4.8 g U/cm^3. Approximately 20 reactors have been converted from high-enriched uranium (HEU) to low-enriched uranium (LEU), and several new LEU reactors have been built using these fuels. In most cases the uranium silicide fuel has been used.

Although a fuel with a uranium density of 4.8 g U/cm^3 is sufficient to convert approximately 90% of the research reactors outside the U.S. which used HEU of U.S. origin in 1978, conversion of the remaining reactors, which use a significant quantity of HEU, require fuels having considerably higher densities. Therefore, the RERTR program continued to develop methods to fabricate plates containing higher loadings of uranium silicide fuels until funding constraints forced an end to this work at the end of September 1989. CERCA, one of the RERTR program's key partners in the development of uranium silicide dispersion fuel, continued its development efforts and announced in 1993 that it had developed an advanced fabrication process which allowed the loading of U_3Si_2-Al fuel to be increased to 6.0 g U/cm^3 [1].

Even though development of higher-density fuels was stopped, analysis of the irradiation experiments performed earlier in the RERTR program continued, resulting in a better understanding of the fundamental behavior of these fuels. Also, work continued to better understand and refine dispersion fuel fabrication techniques. Papers reporting these results have been presented annually at the RERTR international meetings. During this period of decreased fuel development activity within the RERTR program, development of uranium silicide dispersion fuel in the U.S. continued under the auspices of Oak Ridge National Laboratory's (ORNL's) Advanced Neutron Source (ANS) project, with Argonne National Laboratory (ANL) designated as the lead fuel development laboratory [2]. A series of three irradiation experiments in the High Flux Isotope Reactor (HFIR) at ORNL provided data on fuel behavior at high temperatures and high fission rates, and revealed that there are limits to the use of U_3Si_2-Al under these conditions. Results from the most recent of these experiments prompted a major new analysis of U_3Si_2-Al fuel irradiation behavior [3]. Irradiation performance modeling was also emphasized since fuels could not be tested at or beyond ANS conditions. The advances made during this interim period have helped provide a good basis for further development of dispersion fuels.

In 1994 the Department of Energy announced its intention to resume the development of very-high-density fuels to address needs of existing and new research and test reactors. However, owing to pressures of a shrinking federal budget, funding for this work could not be made available until last March. This paper describes the plans and schedule of this new development work.

GOAL AND APPROACH

We have set a goal of achieving a uranium density in a dispersion fuel meat of 8 to 9 g U/cm^3. We think that such a density is achievable from the fabrication point of view, and it matches the

RERTR program's assessments of uranium densities needed to convert the reactors not convertible using 4.8 g U/cm^3 U_3Si_2 fuel. Of course, irradiation behavior and fabrication costs will ultimately determine the success of any proposed fuel.

We think that the best chance of successfully developing a fuel with acceptable fabrication cost lies with extension of the current aluminum-based dispersion fuel concept. Since CERCA's experience with highly loaded U_3Si_2-Al [1] and UN-Al [4] fuels indicates that one is not likely to achieve a fuel volume loading greater than 55% in a commercially viable process, fuel dispersants with very-high uranium densities, ≥15 g U/cm^3, must be used. With the exception of U_6Fe and U_6Mn, which were tested earlier in the RERTR program and shown to be subject to breakaway swelling at relatively low burnups [5, 6], and the similar compounds U_6Co and U_6Ni, which are also expected to exhibit poor swelling behavior, no uranium compound meets our density requirement.

Therefore, only pure uranium metal or alloys of uranium and small amounts of other metals can be considered as fuel dispersants. Although pure uranium is a notoriously poor performer under irradiation, a series of alloys which maintain uranium in the metastable γ phase have shown good irradiation performance in bulk form under fast reactor conditions. Examples of such alloys are listed in Table 1.

Table 1. Densities of Representative γ-Phase Uranium Alloys

Alloy*	Density, g/cm^3	U Density, g/cm^3
U-9Mo	17.0	15.5
U-5Mo	17.9	17.0
U-3Zr-9Nb	16.2	14.3
U-4Zr-2Nb	17.3	16.3

*Alloying element amount given in wt.%.

Our experience with the U_3Si_2-Al dispersion fuel taught us that the key issues which must be addressed are the reaction of the fuel alloys with the matrix and the irradiation behavior of the fuel alloys and of any reaction products. Should reaction of the fuel alloys with an aluminum matrix be excessive, we will investigate the use of a magnesium matrix while retaining the aluminum cladding, since magnesium does not form compounds with uranium. We have also considered using zirconium as both matrix and cladding material, but we do not currently plan to pursue this option owing to concerns about fabrication cost.

PLANS AND SCHEDULE

Fabrication

During the early stages of the RERTR program, fabrication development was performed at ANL's Illinois site (ANL-East), at Oak Ridge National Laboratory (ORNL), and at the Idaho National Engineering Laboratory (INEL). However, by early 1996 only a very small development group at ANL-East remained. Therefore, a major increase in our fabrication development capacity was required to support an aggressive fuel development effort. In order to utilize best the resources available within ANL, we decided to augment the fabrication development group at ANL-East and to establish a new fabrication development laboratory and group at ANL's Idaho site (ANL-West). The procurement process for most of the equipment needed at ANL-West is underway, and detailed installation plans are being prepared. In addition, we have reached the currently required staffing levels at both sites. At ANL-West we have already begun to produce fuel alloys and expect to begin

producing fuel powder, by mechanical means, and compacts for small test plates by the end of November 1996. We anticipate that the development laboratory will be ready for full use before the end of CY 1997, when the installation of the longest-lead item, the rolling mill, will have been completed. In the meantime, we will carry out all rolling operations at ANL-East. In particular, the first rolling experiments have already been performed and fuel plates for the initial irradiation tests (discussed below) are scheduled to be rolled during the first quarter of CY 1997.

We plan to follow the development pattern that worked so successfully for the uranium silicide fuel, that is, to limit our fabrication development to the study of basic issues and to the production of small fuel plates for irradiation testing and to depend on the commercial fabricators to adapt and extend our results to the fabrication of full-sized fuel plates or tubes. We have held preliminary discussions with both Babcock and Wilcox (B&W) and CERCA. Both are interested in participating but will need to assess further the commercial potential of the proposed fuels before deciding the amounts of company resources which can be committed to the required development work.

ANL and the Korea Atomic Energy Research Institute (KAERI) are setting up a cooperation agreement under which the use of atomized fuels will be explored. KAERI recently completed a study of fuel-matrix reaction rates in dispersions of U-Mo alloys in aluminum, with encouraging results [7]. KAERI has agreed to provide a small quantity of U-Mo powder for fabrication development work at ANL.

Materials Properties Studies

In order to address the key issues stated above, a number of studies are planned on the phase structure of the uranium alloys and diffusion of the matrix materials into the alloys. We expect to make substantial progress in this work over the next year. ANL is also negotiating a cooperation agreement with the A. A. Bochvar All-Russian Research and Development Institute of Inorganic Materials in Moscow, under which phase studies of interacting materials is one area of possible collaboration.

Irradiation Behavior Experiments

Little is known about the irradiation behavior of the proposed fuel alloys when they are dispersed in aluminum or about the behavior of the fuel itself at the high burnups typical of research and test reactors. Our highest priority is to begin irradiation screening tests of the proposed dispersions. We are planning to perform these irradiations in the Advanced Test Reactor (ATR) at INEL and to perform the postirradiation examinations at ANL-West. All of the preparatory activities required for the irradiations are underway, including design of the experiment rig, safety analysis of the experiment, and coordination with the ATR experiment staff.

Because of space limitations in irradiation holes near the core, the fuel plates (which we are calling microplates) must be much smaller than the miniplates irradiated previously by the RERTR program in the Oak Ridge Research Reactor (ORR). The current design calls for microplates with outside dimensions 76 mm x 22 mm x 1.3 mm, compared to the 114 mm x 51 mm x 1.3 mm dimensions of a typical ORR miniplate. The microplates will be produced using 9.5-mm-diameter cylindrical compacts, resulting in an elliptical-shaped fuel zone with nominal dimensions of 57 mm x 9.5 mm x 0.5 mm and an area of <500 mm^2, whereas the rectangular fuel zone of the typical miniplate measured 102 mm x 46 mm x 0.5 mm and had an area of 4692 mm^2. Nevertheless, mechanical analyses have shown that the fuel zone area of the microplate is large enough to behave in the same manner as that of a much-larger plate.

Since the first tests will focus on fuel particle-matrix interactions and fuel particle swelling, it is not necessary to test plates with uranium densities approaching the 8 to 9 g U/cm^3 goal. In fact, it is much more important to obtain results as quickly as possible. Therefore, in order to minimize heat removal problems in the design of the irradiation rig and to minimize fuel plate fabrication problems, we will limit the uranium density to no more than 4.8 g U/cm^3.

The uranium alloy fuels to be tested are listed in Table 2. We know that both the resistance to reaction with aluminum and the γ-phase stability of these alloys increases with increasing amounts of alloying addition; so we designed this test matrix to establish the types and minimum amounts of alloying additions necessary. We also plan to test a U_2Mo-Al dispersion, since U_2Mo may be formed during the irradiation of the U-Mo alloys, and U_3Si_2-Al, in order to compare the performance of the microplates with that of previously irradiated miniplates and full-sized plates. The current design allows 16 microplates to be irradiated in one module and three modules to be irradiated in one ATR irradiation position. We currently plan to begin the irradiation near the end of April 1997. We will irradiate two sets of plates—one to about 40% burnup and the other to about 80% burnup. The first set will require two 45-day ATR cycles. Assuming that we meet this schedule, postirradiation examinations are expected to start around October 1997.

Table 2. Fuel Alloys Being Prepared for Irradiation Screening Tests
(Alloying element amount given in wt.%)

U-10Mo	U-10Mo-0.05Sn	U-9Nb-3Zr
U-8Mo	U-4Mo-0.1Si	U-2Mo-1Nb-1Zr
U-6Mo	U-5Nb-3Zr	U-2Mo-1Nb-1Zr-1V
U-4Mo	U-6Nb-4Zr	U-6Mo-2Ru

After the screening tests, we will test one or more successful candidate fuels in full-sized plates, irradiated either individually or in full elements. Since the U.S. has no facilities suitable to perform such irradiations, we must look to our international partners for such tests. Very preliminary discussions have been held with representatives of the CEA in France and BATAN in Indonesia. Irradiation of single plates may be feasible in KAERI's HANARO reactor.

Out-of-reactor irradiation tests of U_3Si_2 have provided valuable insight into the changes taking place in the fuel particles in the very early stages of irradiation (see, for example, Ref. 8). These studies have included neutron irradiations at ANL's Intense Pulsed Neutron Source (IPNS) and ion-beam irradiations at ANL's High-Voltage Electron Microscope (HVEM) facility. We plan to continue such work with the new fuels. A recent improvement in the ability to measure aluminum diffusion rates at the HVEM facility was funded by the RERTR program and should prove particularly useful.

Irradiation Behavior Modeling

As mentioned previously, modeling has become a very important part of our development work, since it is impossible to anticipate all conditions under which the fuels we develop might be used. We are currently modifying the DART code [9] to incorporate swelling models for uranium alloy dispersion fuels, based on data available from previous irradiations of bulk fuel. We will use the model as a tool to help us understand the observed irradiation behavior of the new fuels. We will be pursuing improvements to our model, especially in the area of mechanical deformations, under the agreement with the Bochvar Institute in Moscow, mentioned earlier.

OTHER FUELS

The RERTR program maintains an interest in several fuels which do not have sufficiently high uranium densities to qualify as very-high-density fuels, namely, U_3Si_2, UO_2, and UN. Current activities are briefly discussed below.

U_3Si_2-Al

Because U_3Si_2-Al fuel is being used in many research and test reactors today and is being chosen as fuel for new reactors being designed, the RERTR program remains committed to provide support for current and potential users of this fuel. Our current emphasis is on obtaining a better understanding of the limits to the applicability of U_3Si_2-Al fuel in certain high-temperature, high-fission rate situations [3]. More data are needed to further define those limits. The French CEA is currently irradiating some 5.8 and 6.0 g U/cm^3 plates in SILOE, and we have been discussing a collaboration with the CEA in performing postirradiation examinations of these plates. In addition, we are considering the possibility of performing another irradiation in the HFIR.

The ANL-KAERI cooperation agreement also calls for the irradiation in HANARO of miniplates fabricated in the U.S., some containing atomized U_3Si_2 powder produced by KAERI and some containing the traditional comminuted U_3Si_2 powder in order to demonstrate the acceptable irradiation characteristics of the atomized fuel. A miniplate irradiation rig for HANARO is currently being designed, and preliminary thermal-hydraulic safety calculations for the experiment have been completed.

UO_2

Considerable experience in the behavior of U_3O_8-Al dispersion fuel was obtained during the early phases of the RERTR program. Our interest in UO_2-Al fuel has been kindled by our collaboration with the Russian RERTR program, in which that fuel is being tested up to 3.8 g U/cm^3 for use in converting certain Russian-designed research reactors to use LEU. We have revised the DART code to include a model for uranium oxide, based on the U_3O_8 data [10]. We expect to continue to work with our Russian colleagues to evaluate the results of irradiation tests planned to begin within the next year.

UN

In 1994, CERCA reported that uranium densities of about 7.0 g U/cm^3 could be achieved in UN-Al dispersion fuel by a commercially acceptable process [4]. Although this achievement represented a significant advance in uranium density, a study performed at ANL [11] has shown that the rather-large thermal neutron absorption cross section of nitrogen substantially reduces the advantage of UN-Al fuel over U_3Si_2-Al fuel at the same fuel volume loading. We understand, however, that concerns about problems presented by uranium silicide fuels in the back end of the fuel cycle have prompted some consideration of the use of UN as a substitute for U_3Si_2. We have found only one reference to irradiation of UN-Al fuel [12]. The swelling of UN-Al fuel at about 14 vol.% loading was found to be greater than that of U_3O_8-Al fuel at a similar volume loading. Irradiation behavior data at high loadings of UN are not available; however, based on the low-loading data and our evaluation of U_3O_8-Al swelling [10], we think that severe burnup limits will be required for high-density UN-Al fuel. Nevertheless, the RERTR program will be happy to collaborate in evaluating the usefulness of this fuel as an alternative to U_3Si_2.

CONCLUSIONS

The RERTR program has recently begun an aggressive effort to develop dispersion fuels for research and test reactors with uranium densities of 8 to 9 g U/cm^3, based on the use of γ-stabilized uranium alloys. Fabrication development teams and facilities are being put into place, and preparations for the first irradiation test are in progress. The first screening irradiations are expected to begin in late April 1997 and the first results should be available by the end of 1997. Discussions with potential international partners in fabrication development and irradiation testing have begun.

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TRIGA INTERNATIONAL A NEW TRIGA® FUEL FABRICATION FACILITY AT CERCA

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1- INTRODUCTION

Since the beginning of CERCA in the early 60', the company has always be involved in the development and the manufacturing of nearly all kind of nuclear fuel, either for power reactors or for research reactors.

Among these developments, we may note that CERCA has worked and get involved in the industrial development of graphite moderated fuel, PWR fuel, HTGR fuel, UAlx and U_3Si_2 fuels for research reactors.

Under the diversity of reactor worldwide, one type, which was designed by GENERAL ATOMICS in SAN DIEGO (CALIFORNIA, USA), TRIGA®, has known a very special and important development, mainly among universities and also other users.

At the time when GENERAL ATOMICS expressed its intention to cease fuel fabrication on its site of SAN DIEGO, CERCA has been chosen to carry on the fabrication of TRIGA® fuel.

After negotiations in 1994 and 1995, a partnership 50 % - 50 % was decided, and on July 1995, a new company was founded, with the name TRIGA INTERNATIONAL SAS, head office in Paris and fuel fabrication facility at CERCA in Romans (France).

The intent of this presentation is, after a short reminder about TRIGA® fuel design and fabrication and to describe the new facility with special emphasis on the safety features associated with the modification of existing fabrication buildings.

2- BRIEF DESCRIPTION OF TRIGA® FUEL TYPES

The nuclear fuel used in TRIGA® reactors is based on an Uranium/Zirconium alloy with an enrichment in U-235 of 19,75 %, and a mass fraction of U in the alloy varying from 8,5 to 45 %.
Additionally the U/Zr alloy is hydrided (a Zirconium hydride is formed) to be in a position to get internal moderation in the fuel, which gives to TRIGA® reactors their special safe behavior. Also Erbium is eventually added in small quantities as a burnable poison intended to smooth the neutron flux all along the life of the fuel.

Three types of fuel elements, with a 1.5" diameter, are used in the low power reactors which are : "Standard", "Instrumented", and "Fuel Follower Control Rods".

The Standard elements may be used as individuals, or combined in "Fuel Clusters" in case of a TRIGA® converted MTR reactor.

In the case of more powerful reactors (as the Romanian reactor), the fuel element consist of fuel pins (1/2" in diameter) assembled in bundles.

3- TRIGA® FUEL FABRICATION PROCESS

TRIGA® fuel fabrication requires special skills and know-how gained by GENERAL ATOMICS along the years, and which were part of the technology transfer, and special training of CERCA technicians before the closure of SAN DIEGO workshop and during the start-up of the fabrication at CERCA's workshop.

3.1. Fuel meats fabrication

The Uranium/Zirconium alloy is first cast in an induction furnace at high temperature under controlled atmosphere. Prior to casting all components (Uranium, Zirconium, Erbium, recyclable material) have been carefully weighed and checked by Quality department.

After fusion of components in a graphite crucible, the bath is poured in a graphite mold and allowed to cool down to room temperature. A second fusion is then performed to allow for homogenization and structural quality of the alloy in the mold.

After cooling, the casting is removed from the mold, the fuel meats are brought to length and diameter with a lathe, and identified with the casting number. During machining chips are sampled and used for chemical analysis.

The meats are then hydrided ; for this purpose they are placed in an electrically heated furnace at high temperature under a hydrogen atmosphere. After completion, fuel meats are controlled for quality of hydriding, ground down to final diameter in a centerless grinder, controlled for surface defects and dimensions, then approved by Quality and stored before being assembled.
Some meats used for instrumented elements are eventually machined on a surface grinder to allow the routing of thermocouple leads up to the upper fitting of the element.

3.2. Fuel elements fabrication

Fuel elements are assembled by sliding the fuel meats in the cladding. Eventually the graphite reflectors or the thermocouples have to be installed prior to introduce the meats in the element.

The end fittings are welded and a Helium leak test is performed to check the tightness of the welds. In the Fuel Follower Control Rods, the B4C rod and spacers are hold at the right position by swaging the cladding in the spacer with a magnetoforming equipment.

Finally the elements are controlled for their dimension and a contamination measurement is performed.

4-THE NEW TRIGA® WORKSHOP AT CERCA ROMANS

4.1. The workshop in CERCA facility

The new TRIGA® workshop at CERCA has been installed in the same building as the MTR fuel workshop, in a section which was previously used by other fabrications no longer supported in Romans.
Prior to beginning of construction, this part of the facility has been freed of all remaining equipment, which had to be disposed of according to the safety regulations, part of the walls and ground had also to be scraped out and disposed of in containers.

After clearance of Safety Authorities, construction began in August 1995, and Civil Work was completed by January 1996.

On the US side, packing of all equipment began at SAN DIEGO in November 1995 and the containers were delivered at ROMANS in February 1996.

After completion of the building, all machines were installed between May and July 1996, and first tests with depleted Uranium began in July. Finally the last clearance to allow work with enriched Uranium was granted on October 31 1996, and real work began shortly after.

4.2. Safety report

To start construction work, a letter describing the main safety options, which were to be described with more detail in the Safety Report, was sent to authorities who in return listed all rules to be followed during the construction of the building.

A preliminary Safety Report was necessary to get the operational license. This was written between October and December 95, and was officially transmitted to the safety authorities at the beginning of January 96.
This report was completed during the first half of 96 with the seismic and the criticality calculations.
Additionally to the said calculations, the report includes all assessments about contamination, fire hazard and prevention.

5- TRIGA INTERNATIONAL

The workshop is the property of TRIGA INTERNATIONAL, the JV created 50/50 by GENERAL ATOMICS and CERCA.
CERCA was appointed by TRIGA INTERNATIONAL to operate the workshop.

The contacts with TRIGA INTERNATIONAL customers are mastered by a commercial team including experts from GENERAL ATOMICS and CERCA.

We think that this new workshop will ensure to the customers of GENERAL ATOMICS a very good liability for the future and the insurance for delivery on due time of that fuel elements.

Thermal Compatibility of U-2wt.%Mo and U-10wt.Mo Fuel Prepared by Centrifugal Atomization for High Density Research Reactor Fuels

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1. Introduction

Research on the intermetallic compounds of uranium was revived in 1978 with the decision by the international research reactor community to develop proliferation-resistant fuels. The reduction of 93% ^{235}U (High Enrichment Uranium or HEU) to 20% ^{235}U (Low Enrichment Uranium or LEU) necessitates the use of higher U-loading fuels to accommodate the additional ^{238}U in the LEU fuels. While the vast majority of reactors can be satisfied with U_3Si_2-Al dispersion fuel, several high performance reactors require high loadings of up to 8~9 g U cm^{-3}.

Consequently, in the renewed fuel development program of the Reduced Enrichment for Research and Test Reactors (RERTR) Program, attention has shifted to high density uranium alloys. Early irradiation experiments with uranium alloys showed promise of acceptable irradiation behavior, if these alloys can be maintained in their cubic γ-U crystal structure [1]. It has been reported that high density atomized U-Mo powders prepared by rapid cooling have metastable isotropic γ-U phase supersaturated with molybdenium, and good γ-U phase stability, especially in U-10wt.%Mo alloy fuel [2]. If the alloy has good thermal compatibility with aluminum, and this metastable gamma phase can be maintained during irradiation, U-Mo alloy would be a prime candidate for dispersion fuel for research reactors. In this paper, U-2wt.%Mo and U-10wt.%Mo alloy powder which have high density (above 15 g-U/㎤), are prepared by centrifugal atomization. The U-Mo alloy fuel meats are made into rods by extruding the atomized powders. The characteristics related to the thermal compatibility of U-2wt.%Mo and U-10wt.%Mo alloy fuel meat at 400℃ for time up to 2000 hours are examined.

2. Experimental procedure

Depleted uranium lumps (99.9 pct pure) and molybdenium buttons (99.7 pct pure) were used in the preparation of the U-Mo powders by rotating-disk centrifugal atomization. The fuel meats are prepared with atomized powders by extruding at 400℃. U-2wt.%Mo and U-10wt.%Mo alloy fuel meat were annealed at 400℃ at various times. After heat treatment, the dimensional changes in the specimens were measured. The samples then were examined in a scanning electron microscope (SEM) to characterize the morphology and the microstructure of fuel meat. Electron-probe micro-analysis (EPMA), energy dispersive spectrometry X-ray analysis (EDX) and X-ray diffraction analysis (XRD) were applied to make identification of composition and phase.

3. Results and discussion

Table 1 shows the dimensional changes of the Al-24vol.% U-2wt.%Mo and U-10wt.%Mo alloy samples annealed at 400℃ during various times. U-Mo fuel meats in the as-fabricated condition contain a small amount of porosity. Sintering of fuel meats would explain the ~ 4% densification during the initial time of heat treatment. U-2wt.%Mo fuel meat shows a large volume increase up to 26% after holding at 400℃ for 2000 hours. However, the Al-24vol.% U-10wt.%Mo fuel meat shows only a slight (0.34%) volume change after annealing at 400℃ for 2000 hours.

Backscattered electron images of Al-24vol.% U-2wt.%Mo samples after annealing at 400℃ for 2000 hours are illustrated in Fig. 2. Almost all particles in the U-2wt.%Mo annealed for 2000 hours sample exhibit an irregular interface resulting from diffusion between U-Mo particle and aluminum

matrix. Also, particle cracking, which is known to originate from the formation of brittle intermediate compounds, is observed in the U-2wt.%Mo sample annealed for 2000 hours at the peripherical boundary of particle [3]. Castleman attributes the cracks to stress generated by volume changes during the reaction into the intermetallic compounds [4]. In the current study, it is thought that cracks are formed as a result of the constraints imposed upon the brittle intermediate phase by the geometry of the specimen configuration. On the other hand, this fuel meat shows two aspects of the penetration of aluminum atoms resulting in volume expansion. Fuel particles, illustrated in Fig. 1-(a), are mainly composed of large islands, unreacted regions located in the center of the particle, and small islands, somewhat penetrated regions directed toward the center. It is shown that the new phases developed in the peripherical boundary of the U-Mo particle, but did not develop greatly in the center part of the particle. It implies that the new phase appears to have begun to form from the circumferential rim due to the diffusion routes penetrated along what was unstable paths such as the original grain boundaries of the uranium solid solution. It is thought that large islands are regions unreacted with aluminum, which contain low Mo content according to the EDX results. Fuel particles illustrated in Fig. 1-(b) is mainly composed of small islands, somewhat penetrated regions directed toward the center of U-2wt.%Mo power. It is shown that the penetration of aluminum atoms in the initial step proceeds mainly at the circumferential part of the U-Mo particle, but in the final step the formation of intermediate phases begins toward the center of the particle. The original uranium grain has gradually been reduced to small islands surrounded by the new phase in the matrix of U-Mo particles. In the long run, all regions of fuel particles are filled with U-Al compounds. The developed reactions cause an overall volume increase in the dispersion fuel.

Electron-probe micro-analysis traces with backscattered electron images carried out on these annealed U-2wt.%Mo samples for 100 hours (Fig. 2-(a)) confirms that there is no formation of an intermediate phase region between the U-Mo particle and the Al matrix. The sharp drop in Al composition shown in the probe trace of Fig. 2-(a) shows little penetration of Al atoms through the interface between U-Mo particle and Al matrix. This is regarded as evidence of no volume change in the annealed samples for 100 hours, as shown in Table 1. Electron-probe micro-analysis traces on these annealed samples for 2000 hours (Fig. 2-(b)) confirms little formation of a thick intermediate phase layer between U-Mo particle and Al matrix. The Al composition profile shown in the probe trace of Fig. 2-(b) illustrates a slow decrease around the interface, a considerable penetration of Al atoms through the interface, and little Al penetration in the large island, which is regarded as evidence of a large volume change (26%) in sample annealed for 2000 hours, as shown in Table 1. Area scan analyses of U-2wt.%Mo samples with EDX after annealing at 400℃ for 2000 hours are illustrated in Table 2. It is determined that the matrix regions of the particle mainly consists of UAl_3 with dispersed voids, while the large islands contain few aluminum atoms and the small islands consists of uranium penetrated by lots of aluminum atoms. Fig. 3 shows the XRD patterns for Al-24vol.% U-2wt.%Mo fuel meats after heat treatment at 400℃ for 2000 hours. The phases of U-2wt.%Mo fuel meats consists of Al, UAl_3, γ-U and α-U phase. It is worth noticing that uranium-aluminide, mainly UAl_3, is formed in the original U-Mo particles due to the diffusion of Al atoms. This result also corresponds with area scan analyses with EDX on microstructures.

Electron-probe micro-analysis traces with backscattered electron images carried out on this annealed Al-24vol.% U-10wt.%Mo sample for 2000 hours (Fig. 4-(a)) indicates little formation of a thick reaction layer between U-Mo particle and Al matrix as in the case of U-2wt.%Mo samples. The Al composition profile shown in the probe trace of Fig. 4-(a) illustrates a relatively slow decrease around the interface, but a little diffusion of Al atoms through the interface, which is regarded as evidence of few volume changes in samples annealed for 2000 hours, as shown in Table 1. Micrographs of U-10wt.%Mo fuel meat (Fig. 4-(b)) show no prominent formation of irregular interfaces, cracks and voids in the particles associated with swelling. Micrographs of U-10wt.%Mo fuel meats annealed for 100 and 2000 hours show that the fine cellular structure below 5 ㎛ in size remains without great coarsening of uranium grains despite long annealing. Also, the SEM images of U-10wt.Mo powder like the rapidly solidified microstructure still reveal some Mo segregation, or cored microstructure around the grain boundary [6]. EPMA trace with backscattered electron image and scan electron image carried out on these U-10wt.%Mo samples annealed for 100 hours (Fig. 6) also illustrate that some Mo segregation around the grain boundary still exists. It is supposed that molybdenium atoms cored in the grain boundary inhibit the diffusion of aluminum atoms which proceeds along the grain boundary into the U-Mo particle. Thence, it is thought that the diffusion-controlled swelling resulted from Al penetration can be retarded by a band barrier segregated with Mo atoms around the grain boundary [6~7].

4. Conclusion

The U-2wt.%Mo fuel meat shows great dimensional changes, formations of uranium "islands", voids and new phase, and irregular interfaces after 2000 hours at 400℃, but the U-10wt.%Mo fuel meat doesn't show. The U-2wt.%Mo particles do react with the aluminum and cause great fuel swelling. The amount of reaction and fuel swelling increases with increases in time. The aluminum appears to diffuse into the U-2wt.%Mo particles along unstable pathways such as grain boundaries to form the new U-Al compounds, resulted in an volume increase in the dispersion fuel. In the contrast, the molybdenium segregated around the grain boundary of U-10wt.%Mo particles inhibit the diffusion of aluminum along the grain boundary into the particle, resulted in a slight swelling.

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(Unit: %)

Time (hr)	U-2Mo Alloy			U-10Mo Alloy		
	△ ℓ	△ d	△ V	△ ℓ	△ d	△ V
11	-0.15	-0.17	-0.49	-0.09	-0.18	-0.45
40	-0.03	-0.30	-0.63	-0.02	-0.06	-0.14
107	+0	-0.23	-0.46	-0.14	-0.05	-0.24
350	+0.19	+0.06	+0.31	-0.07	0	-0.07
1000	+1.83	+0.39	+2.61	-1.04	-1.52	-4.08
2000	+4.28	+10.86	+26	0.12	-0.11	-0.34

Table 1. Dimensional changes of Al-24vol.% U-Mo fuel meats after annealing at 400℃ during various times.

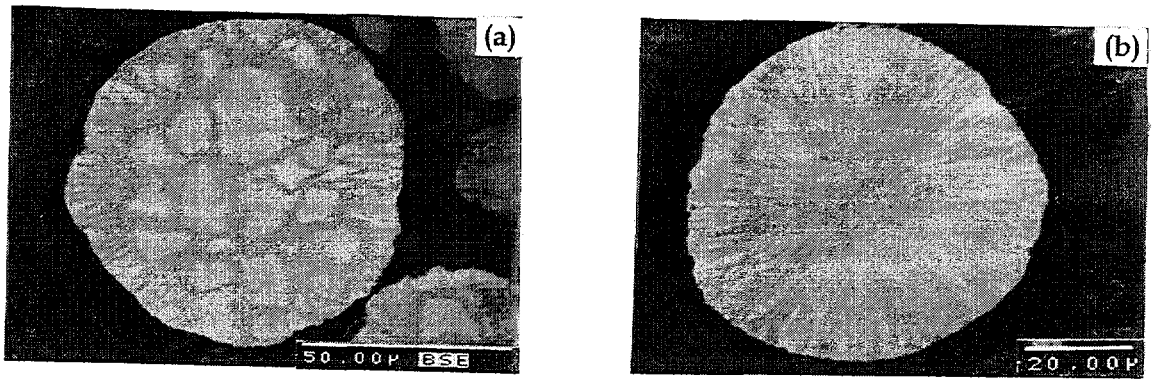

Fig. 1. Backscattered electron images of Al-24vol.% U-2wt.%Mo fuel meat after annealing at 400℃ for 2000 hours; (a) large and small islands, (B) only small islands.

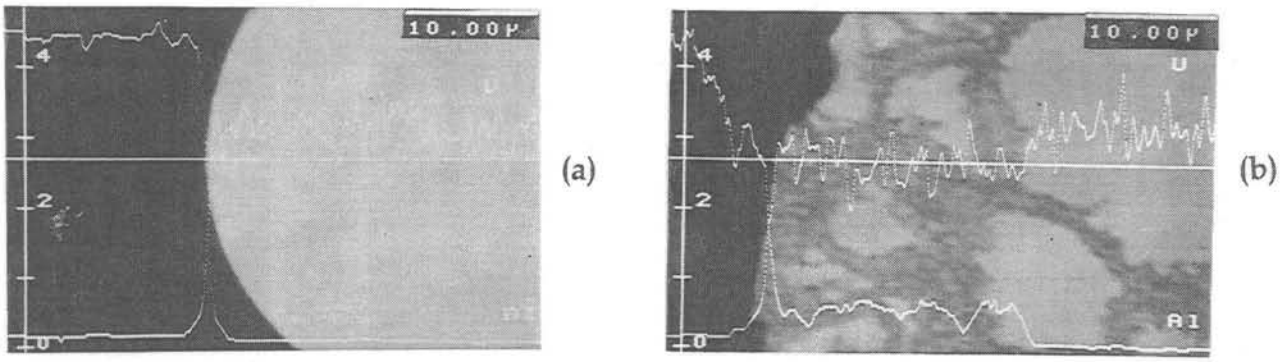

Fig. 2. Electron-probe micro-analysis traces with backscattered electron images carried out on the annealed U-2wt%Mo fuel meats after annealing at 400℃; (a) 100 hours, (b) 2000 hours.

Locations	Composition(at.%)		
	Uranium	Molybdenium	Aluminum
Matrix	23.1	0.2	76.7
Small Islands	67.8	3.2	29.1
Large Islands	93.1	6.9	0

Table 2. Area scan analyses of U-2wt.%Mo fuel meat after 400℃ for 2000 hours.

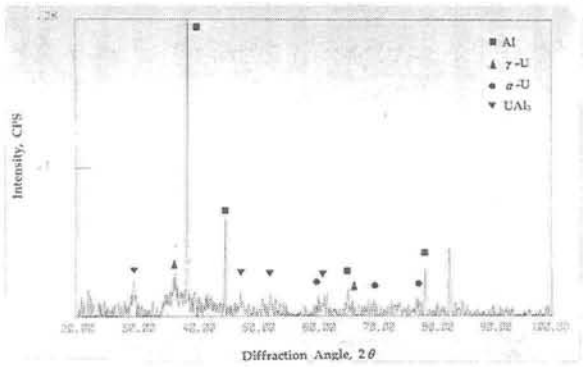

Fig. 3. X-ray diffraction patterns for U-2wt.% Mo fuel meat after heat treatment at 400℃ for 2000 hours.

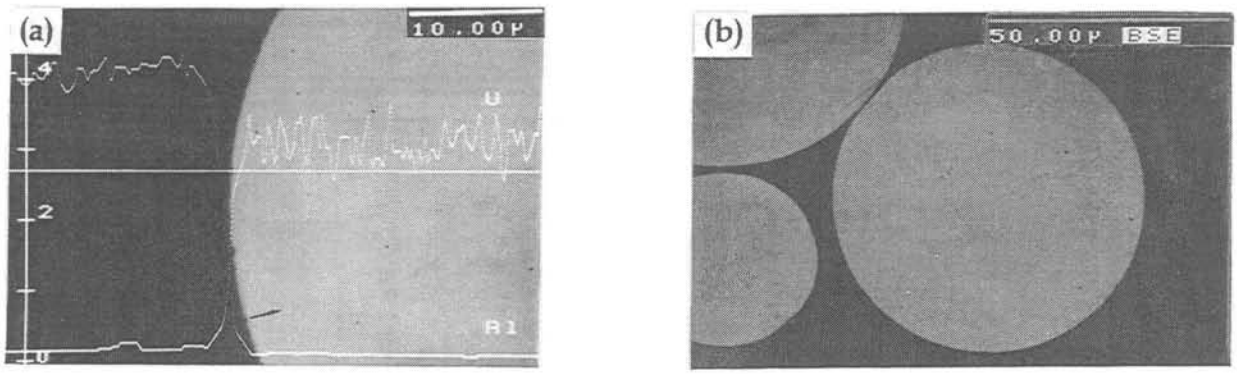

Fig. 4. Electron-probe micro-analysis traces (a) with backscattered electron image (b) carried out on the annealed U-10wt%Mo fuel meat after annealing at 400℃ for 2000 hours.

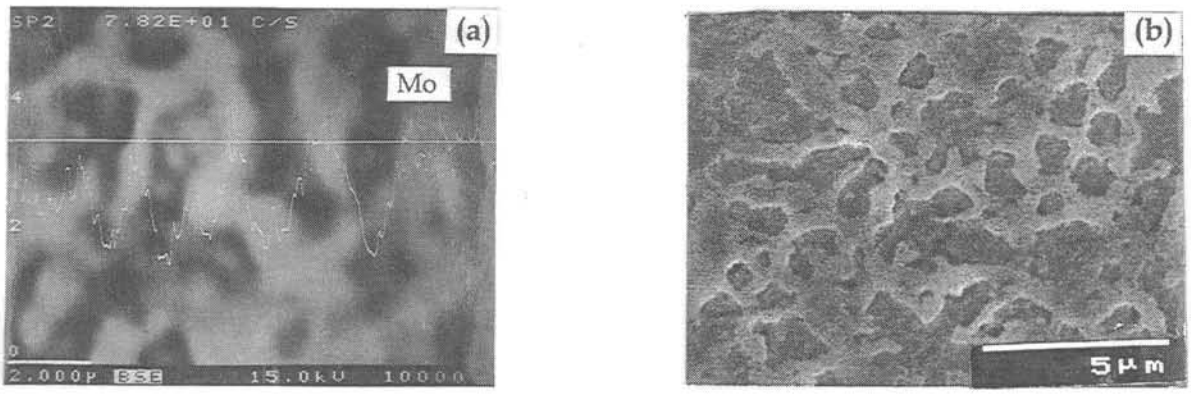

Fig. 5. Electron-probe micro-analysis trace (a) and scanning electron image (b) carried out on the annealed U-10wt.% fuel meat at 400℃ for 100 hours.

Session 3:

Reactor Operation and Fuel Safety

Requirements on Fuel Management for a Safe and Optimum Operation of the German Research Reactor FRJ-2

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Abstract

In case of a failure of a coarse control arm (CCAs) at FRJ-2, reactivity is added to the reactor. The amount of this reactivity depends on the efficiency of the individual CCAs which has been measured as 180 % of the average reactivity of the six arms for the central arm. For this design basis accident, it is required that only 4 out of 5 residual arms must be capable of shutting down the reactor.

This minimum shutdown reactivity is provided by an optimum fuel management including an experimental reactivity determination. Calculation of fuel burnup and material densities is performed by the depletion code SUSAN, which has been verified by separate calculations using ORIGEN. The difference in the reactivity values (between calculation and measurement) is mainly a consequence of the limitation of the of the inverse kinetic method, which is not capable of covering the effects of the flux deformation and interaction of the CCAs and core in the process of reactor scram.

Introduction

According to an official safety review of the German research reactor FRJ-2, a certain shutdown reactivity must be ensured at the beginning of every operating cycle/1/. This results from the fact that Coarse Control arms (CCA) used for reactivity control are held by a supporting mechanism. A failure of the supporting mechanism would allow an absorber arm to swing out of the core and to cause a power excursion as a result of a positive reactivity/2/, if the two independent scram systems (the residual coarse control arms and the fast scram system) fail to operate in time. This design basis accident is extremely unlikely, but it is required that only 4 out of 5 residual arms must be capable of controlling the accident and shutting down the reactor. The reactivity absorption of the fast scram system which is inserted in such a situation is not allowed to be taken into account for formal reasons. The guarantee of this minimum shutdown reactivity on the one hand and the provision of sufficient excess reactivity for the whole period of the operating cycle on the other hand require a fuel management that consists of a theoretical method with an experimental verification.

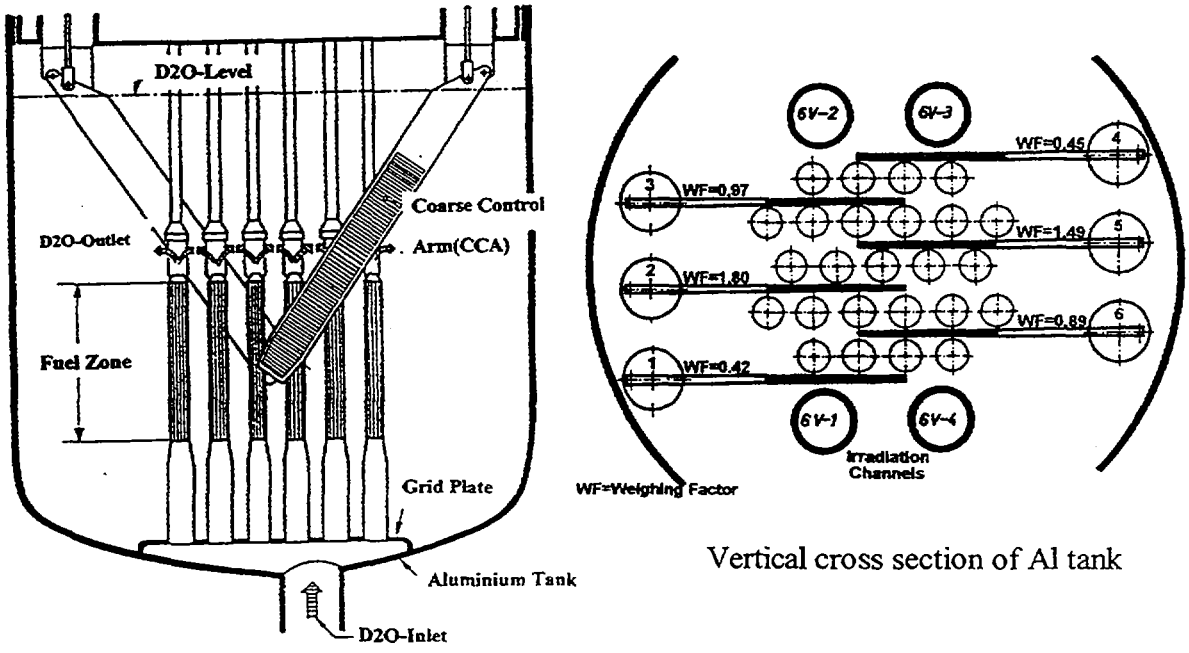
At first burnup calculations are performed for all fuel elements by the application of the SUSAN code which calculates the density of the most important nuclides and absorber material contained in the fuel element. The calculation of reactivity values resulting from the

fuel burnup and composition changes is performed on the basis of reactivity coefficients and flux distribution in accordance with the perturbation model.

To check the theoretical value of the calculated total shutdown and excess reactivity, a measurement is performed after completing the loading steps using the inverse kinetic method. Due to the space-independent character of the method, the results depend on the location of the measuring device, so that the final value is to be corrected regarding spatial effects. Due to the uncertainties in the determination of reactivity, a criticality test is required at the beginning of the operating cycle to experimentally verify shutdown of the reactor by 4 out of 5 residual CCAs in case of accident. For this aim, the most efficient CCA is withdrawn from the active zone and the 2nd one is moved to the critical position(angle).

Description of the FRJ-2

The FRJ-2 is a DIDO-class tank-type research reactor cooled and moderated by heavy water passing through the core in the upward direction. The core consists of 25 so-called tubular MTR fuel elements accommodated within an aluminium tank (Fig. 1). The tank is closed at the top by a solid shield plug and is surrounded by a graphite reflector enclosed within a double-walled steel tank as a second containment.



Vertical cross section of Al tank

Fig. 1: Arrangement of fuel elements and coarse control arms inside the reactor tank

The active part of the tubular fuel elements is formed by four concentric tubes having a wall thickness of 1.5 mm and a length of 0.63 m. Each tube is formed by three material testing fuel plates containing high-enriched uranium in a UAl_x-matrix clad with pure aluminum. The tubes are accommodated in a shroud tube of 103 mm diameter, to which they are attached by four combs at either end. The annular water gap between the tubes has a width of about 3 mm around a central hole of 50 mm diameter.

The reactor is equipped with two independent and diverse shutdown systems, the coarse control arms (CCAs) and the rapid shutdown rods (RSRs). In case of demand, the six CCAs are released from their electromagnets and drop into the shutdown position by gravity, whereas the three RSRs are shot in by their pneumatic actuators.

Calculational basis of fuel management

The burnup for each individual fuel element in the core configuration is calculated by the application of the SUSAN code, which solves the differential equations for the most important nuclides in the fuel composition including uranium, plutonium and neptunium isotopes as well as for the absorber material contained in the fuel element as a function of the irradiation time. The nuclide concentration is determined at each time step by application of a two-dimensional shape function which is measured by the activation method. For this purpose, cobalt wires are fixed along the active length of the fuel elements (axially) and irradiated at the nominal power. The activity of Co-60 and the individual count rate is a measure of the neutron flux distribution in the axial direction of each fuel element. Because of the averaging with the fission rates, the ratio of the fuel element count rate to that of the core represents the flux shape function (factors) of the respective fuel element. The reactivity changes resulting from fuel and absorber burnup are calculated by applying the normalized squares of the flux shape factors in accordance with perturbation model.

For the verification of the mathematical model of the SUSAN code, comparative calculations were carried out with the ORIGEN code/3/. The calculation with SUSAN and ORIGEN was performed for a HEU (80 %w enrichment) fuel element for a power of 0.92 MW and burnup of 60 %w. According to Tab. 1 the results of the calculation with the two codes are in very good agreement for U-235 and U-238. The maximum deviation amounts to 5.3 % for the case of Pu-239. The reason is to be found in the differences between the models for handling and solving the burnup equations as well as in the consideration of a large number of subsequent nuclides in the case of the ORIGEN code.

Table 1: Nuclide densities for the fuel element HEU 170

	SUSAN	ORIGEN	SUSAN	ORIGEN	Diff %
Fluence	0.0		20.148E+20		
U-235(g)	170.0	170.0	68.020	68.000	0.03
U-236(g)	-	-	14.535	14.355	1.24
U-238(g)	42.5	42.5	40.442	40.435	0.02
Pu-239(g)	-	-	0.1113	0.1067	4.13
NP-237(g)	-	-	0.3378	0.3200	5.27
Pu-241(g)	-	-	0.09044	0.0884	2.26
Flux(nv)	1.727	1.725	4.2440	4.1740	1.65

Reactivity balance and kinetic measurements

After calculation of the neutron flux distribution and material densities, the total reactivity loss is determined for the whole period of the operating cycle. On the basis of the limiting values for burnup and the minimum shutdown reactivity for the cold and xenon-free core, the number and positions of the fuel elements for replacement are fixed. The total reactivity change of the core is determined by the calculation of the reactivity values for each individual replacement resulting from changes in the density of U-235 and absorber. On the basis of the reactivity value of the core at the end of each operating cycle, the shutdown reactivity of the next core is determined with the consideration of all replacements.

To check the final reactivity value and to determine the shutdown value of the whole bank under normal operating conditions, a measurement is carried out using the inverse kinetic method. For this purpose the CCAs are dropped from the critical angle after starting up and stabilizing the power at 2 MW. Normally, an acceptable agreement is found with the result of the measurement at a particular location of the measuring detector. In the case of measurements using differently located ionization chambers, the difference in the reactivity values amounts to a maximum of approx. 1 % dk/k for the same core and configuration.The amount of the deviations depends on the location of the detector, burnup state of the CCAs and fuel elements (including loading inhomogeneity). The reason for the discrepancies is that the point kinetic approach for the reactivity determination does not cover the effect of flux deformation taking place in the course of dropping the CCAs. By inserting CCAs, the neutron flux is redistributed in such a way that the flux reduction at the detector position does not represent the same ratio as the average flux of the core. Because of spatial effects, a more sophisticated method is needed for the correction of the measured reactivity values.

Experimental verification of minimum shutdown reactivity

In the case of a loss of the most efficient CCA, a fraction of the total excess reactivity of all 6 CCAs will be released which is determined by the excess reactivity of an average CCA and the weighting factor of the CCA in question. The worth and also the weighting factor of the individual CCAs depends primarily on the position, burnup and core condition. Due to the complexity of the calculational procedure and related uncertainties, the determination of the individual worth is performed by measurement. To this end, the individual CCA is successively kept at the shutdown position (fully down) and the remaining five are raised to the critical position (100 KW). The difference of the measured angle from the critical angle of all 6 CCAs is considered to be proportional to the worth of the CCA kept at the inserted position. The individual worths are determined by adjusting the efficiency curve of all CCAs to the 5 CCAs on the assumption that for the criticality of the reactor in any combination of the CCAs the same reactivity value is needed. The individual worth and consequently the individual weighting factor results from the difference of the two efficiency curves for a given angle. The results of the measurement indicated in Fig. 1 show that the efficiency of the central CCA with 1.80 (above the average) is higher than that of the peripheral one by a factor of 4.14.

On the basis of a weighting factor of 1.80 for the central CCA and a total reactivity worth of -17 %dk/k for all 6 CCAs, the reactivity insertion in case of accident will amount to +2.94 % dk/k(for a shutdown reactivity of -7.2 % dk/k). By comparison, the insertion of 4 out of 5 CCAs will compensate the positive reactivity by absorbing -3.25% dk/k.

Due to the uncertainties in the determination of the worth of the CCAs and reactivity coefficients as well as in the reactivity values resulting from the burnup calculation, a criticality test is performed after completing the loading procedure and before starting the power operation. The undercriticality of the reactor for the accident is proved by withdrawing the most efficient CCA from the reactor core and raising the 2nd one to the angle corresponding to the criticality value of the whole bank. By this test, the configuration of the CCAs under accident conditions is simulated and the reactor shutdown by 4 out of 5 CCA is proved.

Conclusion

A comparison with the ORIGEN code shows that the depletion model of SUSAN is capable of precisely calculating the burnup behavior of the fuel elements of FRJ-2. The accuracy of the calculation depends on the other hand on the accuracy and influence of the flux distribution, which is measured once in an operating cycle and is kept constant in SUSAN during the whole irradiation time. This also influences the reactivity values which are needed for an optimum and safe operation with sufficient shutdown and excess reactivity. Verification of the reactivity values by the inverse kinetic method represents an approximate procedure and needs further correction due to the spatial effects.

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FUEL SHUFFLING OPTIMIZATION FOR THE DELFT RESEARCH REACTOR

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ABSTRACT

A fuel shuffling optimization procedure is proposed for the Hoger Onderwijs Reactor (HOR) in Delft, the Netherlands, a 2MWth swimming-pool type research reactor. In order to cope with the fluctuatory behaviour of objective functions in loading pattern optimization, the proposed cyclic permutation optimization procedure features a gradual transition from global to local search behaviour via the introduction of stochastic tests for the number of fuel assemblies involved in a cyclic permutation. The possible objectives and the safety and operation constraints, as well as the optimization procedure, are discussed, followed by some optimization results for the HOR.

INTRODUCTION

The Hoger Onderwijs Reactor (HOR) is a 2MWth pool-type research reactor situated at the Interfaculty Reactor Institute in Delft, the Netherlands. Its main purpose is to serve as a scientific facility for reactorphysical experiments and to supply neutron beams for use in neutron scattering experiments and neutron activation analysis. It contains highly enriched MTR-type fuel elements, and features a core dimension of approximately 47 cm x 57 cm x 60 cm. The core grid plate has 42 positions, normally loaded with fuel elements including control elements and several reflector elements, containing Be-metal. The reactor is operated continuously 5 days a week. The licensed excess-reactivity is 6%, which requires replacement of a few elements and reshuffling at a three-month interval. The reshuffling operation usually consists of discharging the fuel element with the highest assembly-averaged burnup, followed by a permutation of a limited number of fuel elements such that the vacancy in the core created by discharging the highly-burnt fuel element travels to a position somewhat nearer to the central region in the core, where it is filled with a fresh fuel element.

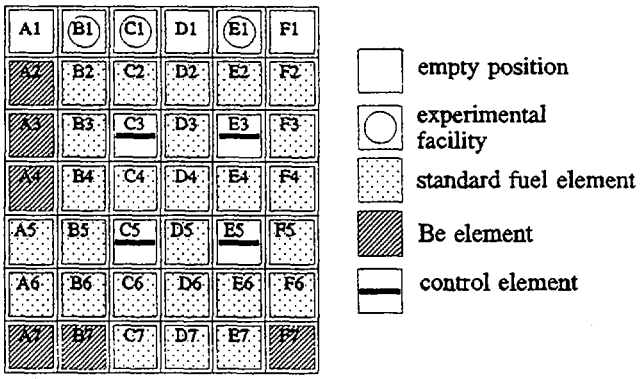

Fig.1 Schematic view of the HOR core

THE OPTIMIZATION PROBLEM

We are interested in optimizing the trajectory along which the fuel element vacancy travels to a position near or in the central region, and find both the loading scheme associated with the highest average out-of-core leakage at the positions of the neutron tubes and the loading scheme associated with the highest value of the effective multiplication factor of the uncontrolled core $k_{eff}^{(uc)}$(EOC) (that is, the core with all control rods fully withdrawn) at End-Of-Cycle. The *safety constraints* are the maximum core reactivity constraint, the shutdown margin constraint and the power peaking constraint. The maximum core reactivity constraint dictates that the effective multiplication factor of the uncontrolled core (that is, the core with all control rods fully withdrawn) $k_{eff}^{(uc)}$(BOC) at Begin-Of-Cycle should remain below 1.06. The shutdown margin constraint requires that it should at all times be possible to shut the reactor down by dropping the two control rods with the least worth, with the other two rods fully withdrawn. This last mentioned constraint usually requires that a fresh fuel element be placed in the near vicinity of a control rod. The power peaking constraint can be derived from thermal-hydraulic analysis. The *operation constraints* are first of all related to a target cycle length of about three months, which requires a minimum core reactivity at BOC. For convenience reasons the number of fuel elements involved in the shuffling operation should be limited. This number n can be added as a penalty to the objective function, and a hard constraint can be implemented according to which n may not exceed a specified value (for example : 12).

THE SEARCH ALGORITHM

The search method proposed by us consists of a stochastic multiple cyclic interchange (simulated annealing) approach[1,2,3], according to which the search for the refueling scheme associated with the highest objective function value can be thought of as divided in multiple stages. The transition from the initial to the final stage is characterized by an increase in the degree of locality of the search procedure. The general idea is that, during the first

stages, the "elite" cluster containing the group of best schemes must be located, after which the solution space is sampled in a more and more local sense to find the local optimum in this cluster. The transition from global to local search behaviour can be made gradual by introducing stochastic tests for the number of fuel elements involved in a cyclic interchange.

The method is initiated with the definition of an initial reference refueling scheme H_1, after which one randomly permutes a stochastic number $\mathbb{N}$ of fuel elements and assesses the effect of the permutation on the objective function value. The number $\mathbb{N}$ of fuel elements involved in the permutation is stochastic in the sense that it obeys a Boltzmann-like probability distribution which tends to gradually favour lower values of $\mathbb{N}$ as the search procedure proceeds. For each search step i, $\mathbb{N}_i$ is determined by the following function :

$$\mathbb{N}_i(r_i) = 2 + \text{int}\left[\left(N_{max} - 1 \right) \cdot \min\left\{ \exp\left(- \frac{f_{best,i-1} - f_1}{T_{i-1}} \right), r_i \right\} \right] \qquad (1)$$

The input variable r_i is obtained from a random number generator that generates random numbers which are uniformly distributed on the interval [0,1) ; $f_{best,i-1}$-f_1 is the best function value improvement found sofar with respect to the function value associated with the initial, "trial" refueling scheme. The annealing temperatures T_i are determined by the initial annealing temperature T_1 and by the recurrence relation $T_i = \alpha \cdot T_{i-1}$, in which the "cooling parameter" α is a number between 0 and 1 (typically, $0.95 < \alpha < 1$). It is obvious that in an early stage of the search procedure, when the annealing temperature is still high, the $\mathbb{N}_i$ will be almost uniformly distributed between 2 and $\mathbb{N}_{max}$ (such as is the case in Fig 2a for $\mathbb{N}_{max} = 4$) ; as the search procedure proceeds, the probability that more than two fuel elements are involved in the cyclic interchanges decreases (Fig 2b).

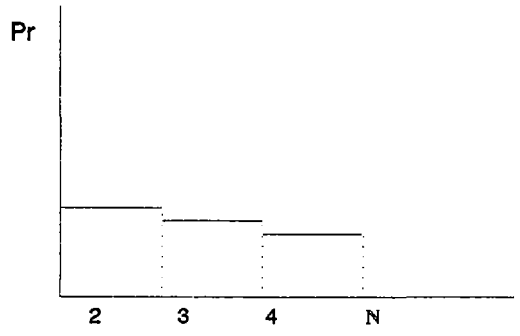

Fig. 2a. Probability distribution for $\mathbb{N}$ at a relatively high annealing temperature

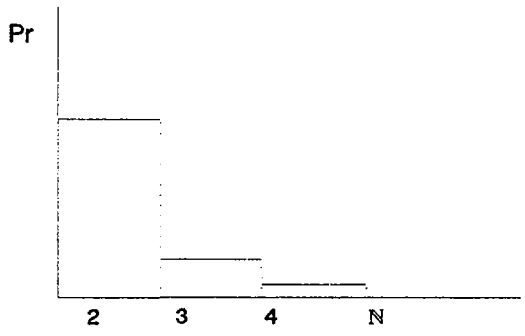

Fig. 2b. Probability distribution for $\mathbb{N}$ at a relatively low annealing temperature

The newly generated loading scheme is accepted as the next reference scheme with acceptance probability

$$Pr_+ = \min\{\exp(\frac{f_i - f_{i-1}}{T_i}), 1\} \quad (2)$$

This means that, if the permutation results in an increase of the objective function value, the newly generated scheme automatically becomes the next reference scheme. In the case of a decrease, a random number a_i is generated, and a stochastic acceptance decision follows in which the newly generated scheme is rejected if $a_i > \exp[(f_i-f_{i-1})/T_i]$. A convergence criterion can be specified according to which this stochastic search procedure ends when either the last fractional improvement falls below a certain specified value, or a certain maximum number of search steps (usually referred to as the annealing chain length) is exceeded.

OPTIMIZATION RESULTS FOR THE HOR

Starting from a fixed initial, mediocre-performance yielding loading scheme we have searched for the loading scheme associated with the best cycle behaviour with respect to two different objectives. Objective 1 was to find the pattern yielding the highest $k_{eff}^{(uc)}$(EOC) while satisfying the shutdown margin constraint, the power peaking constraint and the Begin-Of-Cycle core reactivity constraint. Objective 2 was to find the pattern yielding a maximal out-of-core leakage while satisfying the same set of constraints. We implemented the hard engineering constraint that the total number of fuel shufflings should not exceed 12. The annealing chain length applied was 2000, with $T_1=1.0$, $\alpha=0.996$ and $N_{max}=4$. In Figs. 3 and 4, graphic representations are given of the loading schemes found for objectives 1 and 2, in which the darkness of the fuel assemblies indicates their burnup. In Fig. 5, the optimization trajectory for objective 1 is plotted in the two-dimensional performance plane spanned by $k_{eff}^{(uc)}$(EOC) and the core leakage indicator, along with a random sample of refueling schemes which satisfy the shutdown margin.

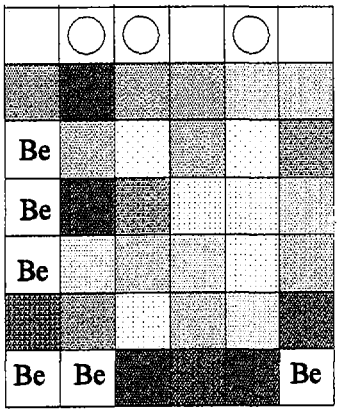

Fig.3 Optimal fuel loading scheme for objective 1

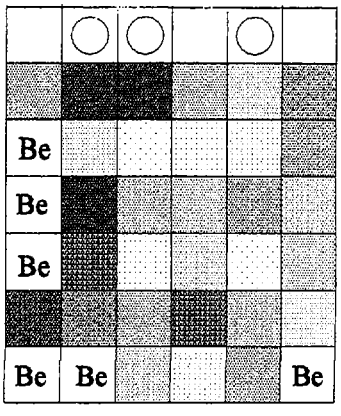

Fig.4 Optimal fuel loading scheme for objective 2

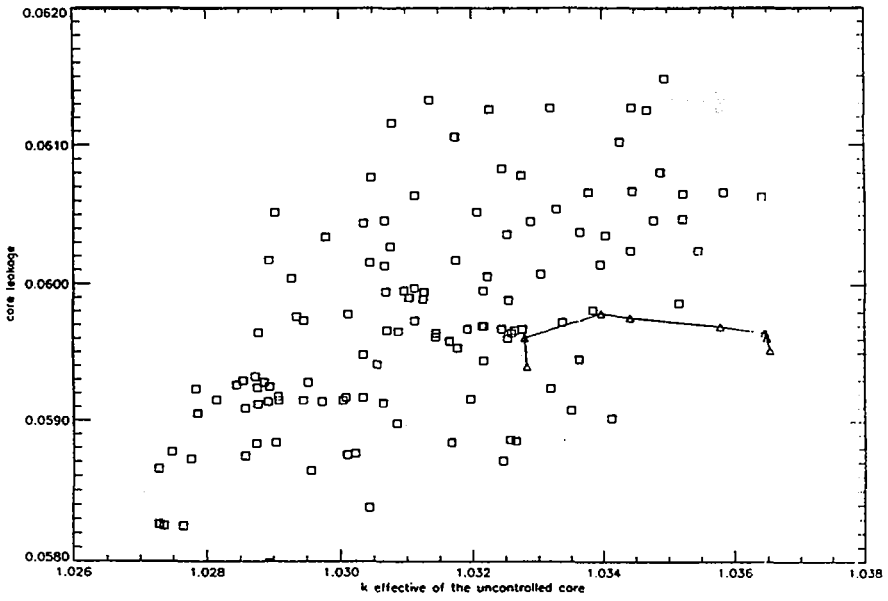

Fig.5 Combined plot of a random sample of refueling schemes and the improved schemes encountered in the optimization trajectory for objective 1.

CONCLUSIONS

We conclude that the stochastic search method that was applied here yields good solutions of the fuel shuffling optimization problem for a research reactor like the HOR. The refueling schemes found comply with the engineering rule for a highly-enriched fuel research reactor like the HOR that the shutdown margin constraint requires that the low-burnt fuel elements be placed close to the control rods. As is generally known, genuine global optimality of the pattern found can hardly be guaranteed[4], but the quality of the results can be visualized by plotting the obtained refueling schemes along with a plot of a random sample of refueling schemes in a performance plane.

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Advanced fuel in the Budapest Research Reactor

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Extended Abstract

Introduction

The Budapest Research Reactor [1], the first nuclear facility of Hungary started to operate in 1959. The main goal of the reactor is to serve neutron research, but applications as neutron radiography, radioisotope production, pressure vessel surveillance test, etc. are important as well. The neutron research will get a much improved tool, when the cold neutron source will be put into operation. The Budapest Research Reactor is a tank type reactor, moderated and cooled by light water. The reactor core is in a cylindrical reactor tank, made of a special aluminium alloy. The diameter of the tank is 2300 mm, the height is 5685 mm. The heavy concrete reactor block is situated in a rectangular semi-hermetically sealed reactor hall. The area of the reactor hall is approximately 600 m^2. It is ventilated individually.

After a reconstruction and upgrading in 1967 the VVR-SM type fuel elements were used in it. These fuel elements provided a thermal power of 5MW in the period 1967-1986 and 10 MW after the reconstruction [2] from 1992.

In the late eighties the Russian vendor changed the fuel elements slightly, i.e. the main parameters of the fuel remained unchanged, however a higher uranium content was reached. This new fuel is called VVR-M2. The geometry of VVR-SM and VVR-M2 are identical, allowing the user to load old and new fuel assemblies together to the active core. The first new type fuel assemblies were loaded to the Budapest Research Reactor in 1996.

The present paper describes the operational experience with the new type of fuel elements in Hungary.

Old and new fuel

As it was said above, the VVR-SM and VVR-M2 fuel elements are rather similar to each other. However some basic parameters of the two types differ. These differences are specified in Table 1.

Table 1. ***Fuel specification***

Data	Old (VVR-SM)	New (VVR-M2)
Cladding thickness [mm]	0.9	0.75
Thickness of fuel [mm]	2.5	2.5
Fissile material composition	Al-UAl_4 eutectic, in Al matrix	UO_2 - Al mixture
Thickness of fissile material in the fuel ("meat" thickness) [mm]	0.7	1.0
Number of fuel elements in the assembly	3	3
^{235}U concentration in the active core [g/assembly-litre]	61.2	69.2
^{235}U concentration in the fissile material [g/cm]	0.5	0.57
Pressure drop in initial core [bar]	0.33	0.33
Average ^{235}U content of fuel assembly [g]	38.9	44.0
Maximal clad temperature [°C]	104	104
Volumetric water content of the elementary cell [%]	54.2	54.2
H/U in elementary cell	235	208
k_∞	1.637*	1.654*
Maximal allowed burn-up [%]	60	60

* according to recent (unpublished) calculations

The geometry of the two assemblies (old and new) are identical, i.e the elementary cell, the outer diameter, the active length, the assembly heads are identical. Consequently there is no difference in the cooling surfaces of the assemblies. The cladding material is in both cases Al of reactor purity (SZAV-1), the fuel enrichment is 36% ^{235}U for both fuel element types. The data identic for both fuel element types are not given in Table 1.

As it can be seen from Table 1, the difference between the two fuel types is slight, but however mainly due to the reduction of the cladding thickness the fissile material content of the fuel assemblies is significantly higher in the new fuel, than it was in the old one.

The licensing of the new fuel elements required reactor physics calculations for normal

operation and for accident conditions as well. The input data for these calculations are the number densities, given in Table 2.

Table 2. *Number densities for the elementary cell*

Type	Isotope	Number density [10^{24}/cm^3]	
		Old	New
"Meat"	^{235}U	1.2912×10^{-3}	0.94058×10^{-3}
	^{238}U	2.2955×10^{-3}	1.64457×10^{-3}
	^{16}O	-	5.17030×10^{-3}
	^{27}Al	4.8430×10^{-2}	4.91997×10^{-2}
Clad	^{27}Al	6.0226×10^{-2}	6.02260×10^{-2}

Analysis of accidents

All possible accidents were analyzed during the reconstruction period, before the new start-up procedure began in 1992. These analyses, described in the Safety Analysis Report of the reactor, served as a basis for the licensing. The analyses were performed for the VVR-SM fuel, but the results of these analyses are valid for the VVR-M2 fuel as well. Only the following assumptions had to be considered.

As a basic point of the safe operation of the research reactor is the protection of the fuel cladding. To protect the cladding, the boiling of the coolant is not allowed, because boiling could lead to the formation of bubbles. The permanent presence of bubbles could cause cavitation type corrosion in the cladding material.

Based on the data in the literature and on the results of our own research, boiling on fuel surface can be avoided, if the cladding temperature does not reach 114 °C. This conservative value for a given power and for given coolant conditions is determined by two parameters. These two parameters are the allowed maximum of the power non-uniformity factor and the heat transfer value. These values obviously do not depend on the material properties, neither of the fuel nor of the cladding.

Operation

The operation of the research reactor has to fulfil the various demands by research and other applications. The reactor was operated according to the pre-determined timetable throughout its operation period. These timetables have always been revised and modified according to the changes in the demand. The timetable was last modified for the year 1995. There were reactor operation periods of 112, 270 and 450 hours of continuous operation and one cycle

consists of 900 MWdays. It means, that one cycle was about 90 effective days, which was followed by a refuelling period of two weeks. The 1995 timetable was planned for 4476 operational hours for the year and this amount was exactly fulfilled. The total energy production was 1883 MW days. The reliability of the reactor was very good as in the 1995 operation period only two unexpected shut downs occurred. The timetable for 1996 has the same structure as that for 1995, the operation period will be somewhat shorter, i.e 3894 hours are planned.

Operational experience with the new fuel

Operational experience in the Budapest Research Reactor is generally excellent, e.g. no fuel failure has been detected since the last reconstruction (1992). The average burn-up levels (55%) reached in the Budapest Research Reactor are relatively high in comparison to burn-up levels reached in similar reactors.

There were only two cladding failures from 1967 up till the last reconstruction. These two failures were slight leakages, they happened both at the beginning of a fuel cycle. This indicates, that the failures were caused by production failures, and are not related to operation.

The first group of VVR-M2 (new) fuel elements (38 of the entire 229) were loaded in January 1966 into the core. The second group of VVR-M2 fuel elements consists of 46 elements, so there are now altogether 84 new elements in the core. The reactivity coefficient measurements in both cases resulted in the same values as in start-up measurements for cores loaded with VVR-SM (old) elements. Two differences have been detected anyhow. The first one is in the power distribution due to the increased fuel amount, the second one is the increase of the cladding temperature, due to the decreased cladding thickness. The measured cladding temperatures were about 98 °C, the warning level is set to 104 °C (see Table 1.).

Conclusion

Experience (almost one year) has proved, that new (VVR-M2) and old (VVR-SM) type of fuel can be used together, without any problem.

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BR2 Mixed Core Management

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1. Introduction

The BR2 reactor is a high flux materials testing reactor, still using 93% enriched uranium in the form of aluminium cladded cylindrical plates. It is moderated by light water and beryllium. The reactor core is composed of beryllium hexagons with central irradiation channels of 200, 84, 50 or 33 mm diameter. The cooling water circuit is pressurized at 12 bar. The pressure vessel is localized in a pool filled with demineralized water. The operating power is routinely between 50 and 80 MWth so that unperturbed thermal neutron fluxes of 10^{15} n/cm^2.s can be achieved in the central channel of the core.

In order to optimize the utilization of the available HEU inventory, the CEN.SCK considers the possibility to elaborate a 'Mixed Core Strategy' based on the irradiation of standard 93% ^{235}U fuel elements together with 72% ^{235}U fuel elements using uranium recovered from the reprocessing of BR2 spent fuel. The exclusive use of 72% ^{235}U fuel elements has also been considered.

2. Core configuration requirements

The BR2 core configuration management has to satisfy the following aspects:

- the experimental programme,
- the safety criteria,
- the economical use.

The experimental programme requires specific irradiation conditions: available volume, thermal and fast neutron fluxes, gamma heating, temperatures, irradiation time, ...

The safety criteria are fulfilled by the cooling capability, the negative reactivity available in the control rods at any moment of the cycle (operation and shutdown), the regulation of the reactor, the hot spot on the fuel plates, ...

The main optimization parameters regarding the operating costs of the core are:

- the definition of a minimal core configuration satisfying the irradiation conditions,
- the location of the control rods in the core (efficiency taking into account the antireactivity of the experiments, the length of the cycle, ...),
- the choice of the right fuel element type (mass, density, burnable poisons, ...),
- the definition of the operating regime taking into account the irradiation programme, the ^{3}He poisoning level of the beryllium matrix, ...

3. Standard BR2 fuel elements

The standard cermet BR2 fuel elements of the type VIn G, presently manufactured by CERCA (France), consist of an assembly of 6 concentric cylindrical plates and are characterized by:

- total mass of ^{235}U: 400 g; enrichment in $^{235}U > 90\%$,
- thickness of the fuel meat: 0.51 mm; density of 0.060 g $^{235}U/cm^2$ or 1.31 g U_{tot}/cm^3,
- burnable poisons: 3.8 g B_{nat} in the form of B_4C and 1.4 g Sm_{nat} in the form of Sm_2O_3,
- thickness of the plates: 1.27 mm,
- water gap thickness between the fuel plates: 3 mm.

The BR2 core configurations used with the second beryllium matrix from 1980 till 1995 were often centred around the central channel H1. They were characterized by the loading of 28 to 39 fuel elements, 6 or 7 control rods, one regulating rod and have produced a total energy of 178 920 MWd.

The standard fuel elements allow a reactor operation at a power of about 60 MW during 21 days with 5 or 6 batches of fuel elements (fresh and partially burnt). The theoretical evaluation of the consumption of fresh fuel elements for the production of 1 000 MWd and a mean burnup of 50% at elimination is 6.2; the corresponding maximum burnup at mid-plane is about 62% or 1.6 e+21 fission/cm³. The experimental value for the exploitation of the second BR2 beryllium matrix gives a value of 5.7 fresh fuel elements loaded per 1 000 MWd for a mean burnup of 53% at elimination. This value depends on the operating regime, the configuration and the experimental loading.

4. AEA test fuel elements

In the frame of a 'Qualification Programme', 6 fuel elements were fabricated in 1994 by AEA-Technology (UK) with uranium recovered from the reprocessing of BR2 spent fuel at UKAEA-Dounreay. The programme had to:

- define the fuel specifications (^{235}U mass and density, burnable poisons, ...),
- define the technical specifications for fabrication,
- define the neutronic specifications (reactivity of the fuel element, ...),
- follow the recommendations of the 'Safety Advisory Committee',
- follow the QA/QC manufacture procedures,
- perform reactivity effects measurements of the fuel elements before each BR2 start-up,
- perform measurements during the irradiation (neutron fluxes, gamma heating, ...),
- analyse the final results.

The AEA test fuel elements of the type VIn E have the same geometry than the VIn G fuel elements and are characterized by:

- total mass of ^{235}U: 330 g; enrichment in $^{235}U = 72\%$,
- thickness of the fuel meat: 0.51 mm; density of 0.050 g $^{235}U/cm^2$ or 1.31 g U_{tot}/cm^3,
- burnable poisons: 1.8 g B_{nat} in the form of B_4C and 1.3 g Sm_{nat} in the form of Sm_2O_3.

They were irradiated during 5 cycles of 21 days in the BR2 reactor till a maximum mean burnup of 43% to 48% (about 1.3 e+21 fission/cm³), without release of fission products.

5. Comparison between VIn G and VIn E fuel elements

In the typical configuration 10U, illustrated in the figure 1, the standard **VIn G** fuel elements (93% ^{235}U) are routinely irradiated during 5 cycles of 21 days at a power of 60 MW, as follows:

- mean burnup of β=12% after 1 cycle in a C-channel,
- mean burnup of β=35% after 2 additional cycles in a A- or B-channel,
- mean burnup of β=45% after the fourth cycle in a D-channel,
- mean burnup of β=52% after a last cycle in a F- or G-channel.

This "theoretical" irradiation profile is in reality perturbed by the requirements of the experimental programmes. For example, the extension of the configuration 10 to satisfy the irradiation conditions of the LWR-CALLISTO loop required the loading of 3 additional fuel elements in G-channels, each for 1 or 2 more cycles; the mean burnup at elimination reached sometimes β=60%.

The **VIn E** test fuel elements underwent a typical irradiation profile of 5 cycles of 21 days at a power of about 60 MW in the 10U configuration, as follow:

- mean burnup of β=12% after 1 cycle in a C-channel,
- mean burnup of β=25% after a second cycle in a A- or B-channel,
- mean burnup of β=35% after the third cycle in a D-channel,
- mean burnup of β=48% after 2 additional cycles in a F- or G-channel.

The comparison of the measured reactivity curves of both types of fuel elements as a function of their mean burnup allows the definition of equivalences between 'batches of burnups':

BATCH	VIn G (93% ^{235}U)	VIn E (72% ^{235}U)
1	0 %	0 %
2	12 %	------
3	25 %	12 %
4	35 %	25 %
5	45 %	35 %
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6 = β elimination	52 %	42 %

Three important conclusions can be drawn from the table above:

- there is no equivalence between a β=12% VIn G (93% ^{235}U) fuel element, characterized by the maximum of reactivity, and a VIn E (72% ^{235}U) fuel element; this creates an unbalanced inventory of the partially burnt fuel elements,
- the mean burnup at elimination is β=52% for the VIn G (93% ^{235}U) fuel elements and β=42% for the VIn E (72% ^{235}U),
- the inventory of VIn E (72% ^{235}U) fuel elements is limited to 4 'batches of burnups' in place of 5 for the VIn G (93% ^{235}U) fuel elements.

6. Mixed Core Management

The preliminary study made with the help of the calculation code GEXBR2-TRPT3-REAC2 showed the possibility to elaborate a strategy based on the **exclusive utilization** of VIn E (72% ^{235}U) fuel elements. Nevertheless this core management is not optimal because:

- the length of the cycle is mostly shorter than 21 full power days,
- the mean consumption of fresh fuel elements is about 9 fuel elements per 1 000 MWd,
- there is a build-up of β=35% fuel elements in the partially burnt fuel elements inventory; this results in a reduction of the mean burnup at elimination from β=42% to 35%<β<42%.

A further study showed that the **mixed utilization** of both types of fuel elements VIn G (93% ^{235}U) and VIn E (72% ^{235}U) can optimize the fuel utilization. Indeed this alternate strategy allows:

- a well balanced management of the irradiated fuel elements inventory,
- a mean consumption of 6.2 fresh fuel elements per 1 000 MWd,
- the possibility to alternate the loading of 6 VIn G (93% ^{235}U) fresh fuel elements for a cycle with 6 VIn E (72% ^{235}U) fresh fuel elements for the next cycle.

7. Conclusion

The BR2 fuel cycle management can be optimized by the fabrication and the irradiation of fuel elements with uranium recovered from the reprocessing of BR2 spent fuel. The VIn E fuel performances could be upgraded by increasing the amount of burnable poisons, the fuel mass, the fuel density, ... in order to obtain a higher reactivity effect at a burnup of about β=12% and a longer cycle duration. The preliminary results of the calculations need however to be confirmed by measurements on effective reactor loads.

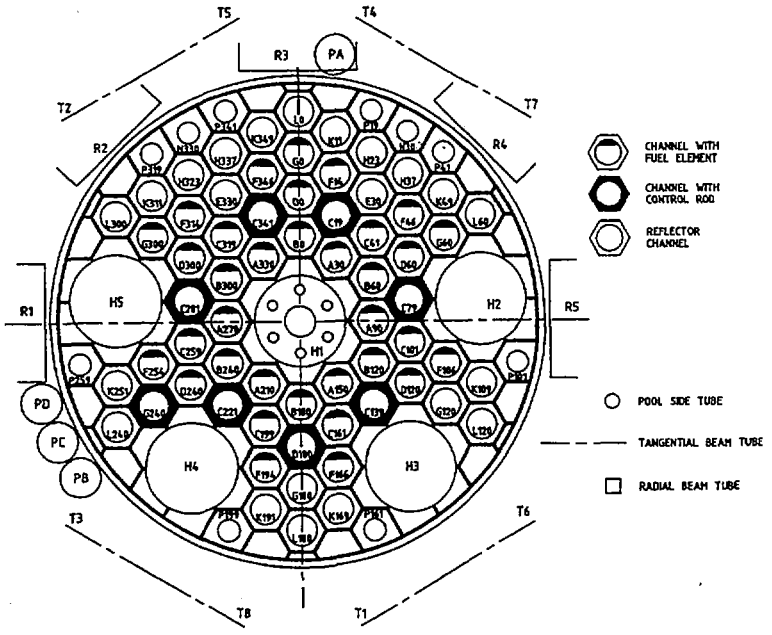

Figure 1 : Typical BR2 Configuration (10U)

FUEL MANAGEMENT STRATEGY FOR THE NEW EQUILIBRIUM SILICIDE CORE DESIGN OF RSG GAS (MPR-30)

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ABSTRACT

The design procedure and fuel management strategy were proposed for converting the oxide core of RSG GAS (MPR-30) to the new equilibrium silicide core using higher uranium loading. The obtained silicide core gave significant extension of the core cycle length and thus increasing the reactor availability and utilisation.

INTRODUCTION

One research activity with high priority in the Center for Multipurpose Reactor, National Atomic Energy Agency, is the RSG GAS (MPR-30) core conversion program from oxide to silicide fuel. Among other advantages of silicide fuel, its high loading of fissile material is expected to increase the operation cycle, hence, higher reactor availability and utilization can be achieved. RSG GAS is a Be reflected, light-water moderated and cooled, 30 MWth (max.) multipurpose reactor. Presently RSG GAS uses MTR-type LEU (19.75 w/o) oxide fuel elements (FEs). As shown in Fig. 1, on the 10 x 10 core grid positions there are 40 standard FEs (each consists of 21 fuel plates), 8 control elements (CEs, each consists of 15 fuel plates) initially loaded with 250 and 178.6 g ^{235}U respectively, Be reflector elements and other irradiation facilities. These fuel loading corresponds to uranium meat density of 2.96 gU/cc. With the original nominal core cycle of 25 days, 7 burn-up classes with burn-up step of approximately 8 % loss of ^{235}U the core produces energy of 750 MWD per cycle.

This research activity has been coordinated with research and development programs in Nuclear Fuel Element Development Center of the Agency, and three experimental silicide fuel elements, with 250 g ^{235}U loading per FE produced by the Center, have been irradiated in the oxide core of RSG GAS. From the economic and licensing point of view, the core conversion program puts rather strict limitation on the new silicide core design, that is, no major modification should occur on the reactor balance of plant and core main structural components. Consequently, several useful free design parameters can not be manipulated to obtain the optimal silicide core design, however, many works can still be done to extend the core cycle and increase the core performance. This paper discusses some part of the work in the neutronic design of the new silicide core, especially, from the fuel management aspects.

DESIGN OBJECTIVE AND CONSTRAINTS

Our design objective for the new silicide core of RSG GAS is to maximize the core cycle length without violating the following constraints:

1. No major modification may occur on the reactor balance of plant, shielding, core main structural components and civil buildings.
2. The number as well as the performance of irradiation positions and facilities must be maintained.
3. The existing one-stuck-rod margin reactivity and thermal-hydraulic safety requirements must be fulfilled.
4. Maximum discharge burn-up is limited to 70 % for licensing purpose.

The first constraint includes the use of the existing Ag-In-Cd control rods. In the present oxide core, the eight (fresh) control rods have a total worth of around 14.5 % while our preliminary investigation showed that the worth decreased as the neutron spectrum in the silicide core becomes slightly harder.

The second constraint prevents the reduction of the number of central irradiation position (CIP), irradiation position (IP), and on top of that, the position of other irradiation facilities (PRTF, rabbit system) must be maintained. Although as a consequence the total number of FEs and CEs are identical with the present oxide core, several free parameters of fuel management are still open to be changed: initial loading of FE/CE, number of burn-up classes and refuelling/reshuffling strategy. These three parameters will be manipulated in the present work to maximise the core cycle length.

The third constraint, in fact, is not imposed solely on the new silicide core design only but also for the present oxide core design, therefore, same design procedure will be adopted. One problem associated with the lower control rod worth previously mentioned is the one-stuck-rod reactivity margin. Fortunately, our preliminary investigation showed that the radial power peaking factor (PPF) for the new silicide core was not a significant problem.

SILICIDE CORE DESIGN PROCEDURE & RESULTS

Design Codes

A fuel management code for obtaining directly the equilibrium core of a research reactor was developed for the present work (Liem, 1996, Bakri *et al.*, 1996). Few group neutron diffusion theory in 2-D, XY and RZ reactor geometry is adopted as the framework of the code. The burn-up dependent fuel cross section library for the code was prepared with WIMS/D4 cell calculation code (Askew *et al.*, 1966). The validation of the code for the present work was checked through benchmark calculations defined in the IAEA-TECDOC-233 (IAEA, 1980) and IAEA-TECDOC-643 (IAEA, 1992). Since most of the calculations were done in 2-D XY reactor geometry accurate axial buckling values must be found. These values were determined through rigorous 3-D XYZ diffusion calculations with Batan-3DIFF code (Liem, 1995).

Fuel Management Strategy

First, the 40 FEs and 8 CEs are grouped into 8 burn-up classes (batches or zoning). Consequently, at BOC 5 FEs and 1 CE are loaded after discharging the same number of old FEs and CE from the core. The refuelling/reshuffling scheme proposed can be categorised as scatter loading, as shown in Fig.1 and indicated by Table 1. The zoning shown is selected to flatten the PPF. Increasing the number of burn-up classes from 7 (for the oxide core) to 8 for the new silicide core is a mean to suppress the excess reactivity at BOC in the one hand, and to increase the fuel discharge burn-up on the other hand. The second step is to determine the optimal initial ^{235}U loading of the fresh FE and CE.

Parametric survey works on the initial ^{235}U loading up to 350 g /FE (fuel meat density of 3.74 gU/cc) have been done to estimate the possible range of core cycle length by first neglecting the constraints previously mentioned (Fig. 2). In general, increasing the initial fuel loading indeed increasing both the cycle length and BOC excess reactivity of the equilibrium core. However, for a same core cycle length, the discharged burn-up tends to decrease with higher initial fuel loading. This is a logical consequence since we use no burnable poison.

The three horizontal lines in the figures representing the imposed constraints, i.e. criticality, maximum discharge burn-up and maximum BOC excess reactivity under one-stuck-rod (OSR) condition. By imposing these constraints the feasible core cycle lengths were found to be 30 - 32.5 days and 35 - 37.5 days for U meat density of 3.55 gU/cc and 4.74 gU/cc, respectively. Compared with the present oxide core cycle length of 25 days a significant extension can be attained, especially for U meat density of 4.74 gU/cc. To select the optimal core cycle length for a certain U meat density, some trade-off must be made. For a shorter core cycle length the reserved EOC excess reactivity (for experiment, irradiation and partial Xe override) becomes larger but the safety reactivity margin for OSR condition decreases.

Although it is not discussed in this work, the direct conversion from 2.96 to 4.74 gU/cc will face many problems in the transition cores. Therefore, the first step of the conversion is to achieve the silicide equilibrium core with 3.55 gU/cc meat density and followed with 4.74 gU/cc. For the former equilibrium core the fuel burn-up distributions at BOC and EOC is shown in Fig. 3

while the radial PPF distribution at BOC is shown in Fig. 4. The scatter loading strategy and higher number of burn-up classes were effective in minimising the radial PPF so that there was no need for modification of the primary cooling system. Table 2 shows the comparison of the reactivity balances between the present oxide typical working core and the equilibrium silicide core. It can be observed that the equilibrium silicide core is suffered from smaller excess reactivity at EOC. However, in the calculation the Be positive feedback reactivity was not taken into account so that the EOC excess reactivity might be higher than the value predicted here.

CONCLUDING REMARKS

The fuel management strategy for converting the oxide core of the RSG GAS to silicide core with higher fissile loading has been proposed to extend the core cycle length so that the reactor availability and utilisation can be enhanced. The analyses showed that the conversion program can be done successfully without major modification except for the fuel meat density and still the obtained equilibrium silicide core met the imposed design constraints and safety requirements.

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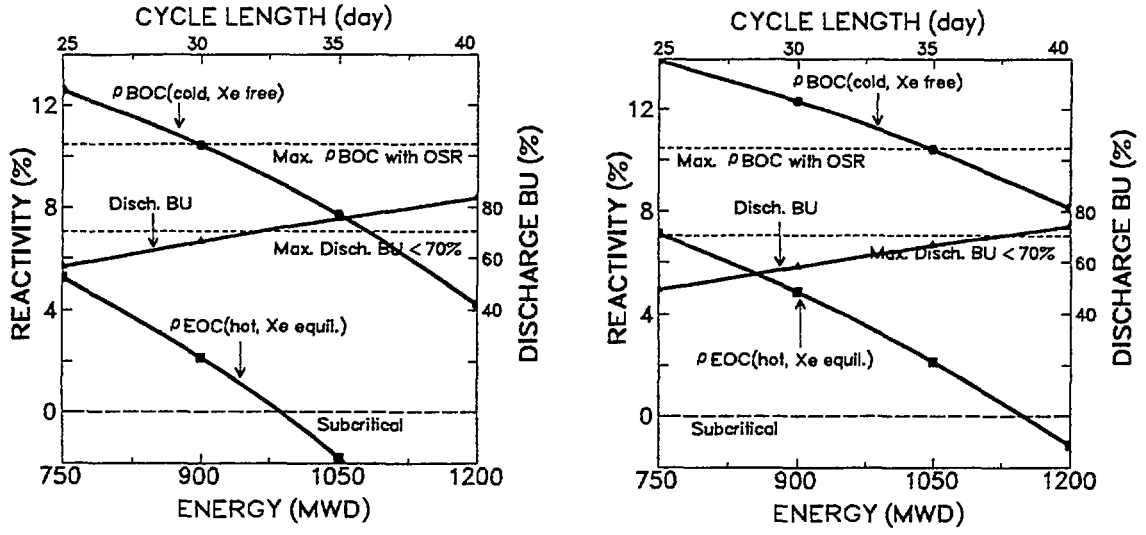

Figure 3. Some design parameters of the equilibrium silicide core of the RSG GAS as function of core cycle length for uranium meat density of 3.55 gU/cc (left) and 4.74 gU/cc (right).

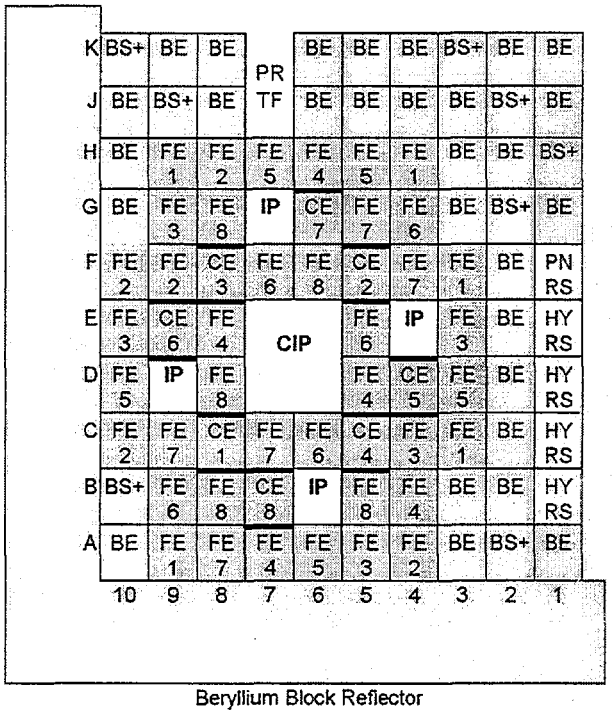

Figure 1. Core-reflector configuration of RSG-GAS with burn-up class in the second rows (FE:fuel element, CE:control element, BE:Be reflector element, BS+:Be reflector element with plug, IP:irradiation position, CIP:central irradiation position, PNRS:pneumatic rabbit, HYRS:hydraulic rabbit, PRTF:power ramp test facility).

Table 1. Reshuffling and refueling strategy.

From	To	From	To	From	To
H-9	F-10	F-5	F-8	C-7	B-8
H-8	C-4	F-4	F-6	C-6	G-5
H-7	F-7	F-3	C-10	C-5	D-4
H-6	D-10	E-10	B-4	C-4	D-5
H-5	E-5	E-9	G-6	C-3	H-8
H-4	F-9	E-8	D-3	B-9	C-9
G-9	E-8	E-5	A-8	B-8	**out**
G-8	**out**	E-3	A-7	B-7	**out**
G-6	B-7	D-10	G-4	B-5	**out**
G-5	G-8	D-8	**out**	B-4	A-6
G-4	C-7	D-5	H-5	A-9	A-4
F-10	G-9	D-4	E-9	A-8	B-5
F-9	A-5	D-3	C-6	A-7	H-7
F-8	C-5	C-10	E-3	A-6	B-9
F-7	F-4	C-9	D-8	A-5	H-6
F-6	**out**	C-8	F-5	A-4	E-10

H	BE	0.0 10.31	10.63 19.72	36.92 44.51	28.77 36.63	39.13 45.83	0.0 9.47	BE
G	BE	21.13 30.81	62.10 68.34	IP	58.46 65.25	56.08 62.09	45.00 51.80	BE
F	10.31 21.13	9.47 20.45	22.03 32.54	44.50 53.24	59.96 66.44	11.31 22.03	53.25 59.96	0.0 10.17
E	18.00 27.81	50.44 58.46	30.81 40.74	CIP		45.83 54.44	IP	19.71 29.03
D	36.63 45.00	IP	57.41 64.68			29.34 39.12	41.71 50.44	40.75 47.99
C	10.17 19.71	50.64 57.43	0.0 11.31	51.81 58.99	47.99 56.08	32.54 41.71	19.72 29.34	0.0 10.63
B	BS+	44.11 50.65	58.99 64.50	65.25 71.02	IP	59.72 65.95	27.80 36.31	BE
A	BE	0.0 8.94	54.43 59.72	29.03 36.91	36.31 44.10	20.45 28.76	8.94 17.99	BE
	10	9	8	7	6	5	4	3

Figure 3. BOC (upper row) and EOC (lower row) fuel burn-up distributions of the equilibrium silicide core of the RSG-GAS.

H	BE	1.14 1.73	1.03 1.48	0.92 1.16	0.93 0.98	0.81 1.02	1.04 1.42	BE
G	BE	1.13 1.50	0.83 0.87	IP	0.89 0.56	0.78 0.67	0.84 1.02	BE
F	1.24 1.55	1.26 1.17	1.24 0.76	1.09 0.91	0.85 0.58	1.23 0.69	0.86 0.86	1.12 1.38
E	1.14 1.23	1.02 0.54	1.20 0.79	CIP		1.08 0.75	IP	1.08 1.16
D	1.02 1.12	IP	0.96 0.67			1.18 0.76	1.08 0.55	0.89 0.90
C	1.09 1.34	0.86 0.85	1.26 0.70	0.92 0.61	1.02 0.82	1.11 0.63	1.13 0.97	1.18 1.38
B	BS+	0.81 0.95	0.72 0.59	0.78 0.46	IP	0.82 0.81	1.02 1.25	BE
A	BE	0.99 1.30	0.67 0.81	0.94 0.96	0.94 1.15	0.97 1.32	1.03 1.47	BE
	10	9	8	7	6	5	4	3

Figure 4. Radial PPF distributions of the equilibrium silicide core of RSG GAS (first and second rows represent all control rods up and down conditions, respectively).

THERMAL HYDRAULIC MODEL VALIDATION FOR HOR MIXED CORE FUEL MANAGEMENT

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INTRODUCTION

A thermal-hydraulic core model has been developed for the Hoger Onderwijsreactor (HOR), a 2 MW pool-type university research reactor. The model was adopted for safety analysis purposes in the framework of HEU/LEU core conversion studies. It is applied in the thermal-hydraulic computer code SHORT (Steady-state HOR Thermal-hydraulics) which is presently in use in designing core configurations and for in-core fuel management. An elaborate measurement program was performed for establishing the core hydraulic characteristics for a variety of conditions. The hydraulic data were obtained with a dummy fuel element with special equipment allowing a.o. direct measurement of the true core coolant flow rate. Using these data the thermal-hydraulic model was validated experimentally. The model, experimental tests, and model validation are discussed.

STEADY-STATE HOR THERMAL-HYDRAULICS (SHORT)

thermal hydraulic model

The HOR is cooled by forced convection downward flow of pool water sucked into the core region and circulated by the primary pump through the heat exchanger. During forced flow conditions a vertically movable suction header is seated to the bottom of the grid plate (Fig. 1). The core hydraulics are modelled in the SHORT code system.

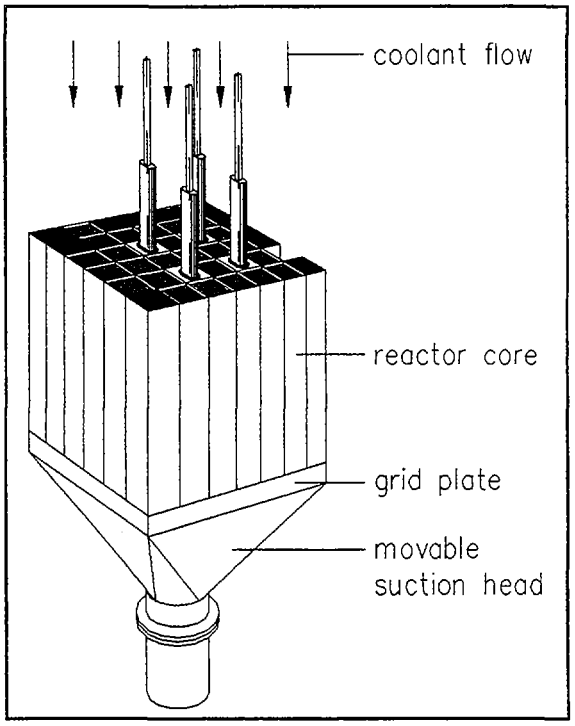

Figure 1. HOR core cooling system

The SHORT program [1] is composed of the two established computer program modules FLAC [2] and COBRA-IIIC [3]. FLAC solves the equations for the flow and pressure distribution in arbitrarily arranged hydraulic resistance networks and COBRA-IIIC carries out thermal-hydraulic calculations in an array of heated channels with or without links for cross-flow. To make the application for the HOR user-friendly, FLAC and COBRA-IIIC are coupled in SHORT. Appropriate empirical functions for heat tranfer and fluid friction are used on a preselected basis, as well as basic geometry data. Only core operating conditions and the core configuration are to be specified as input data by the user. From these the SHORT program then calculates the data required for running FLAC and COBRA-IIIC.

Once the input data are provided SHORT starts to generate a flow resistance network which consists of resistances in the axial and transverse direction simulating the flow of pool water into the interconnected system of gaps between core components which is then sucked through the small holes in the grid plate. The flow through the core components is also modelled. Leakage flow of pool water into the suction head plenum through small gaps is accounted for by an ad-

ditional resistance connecting the plenum below the grid plate with the pool. FLAC then iteratively calculates the pressure and flow distribution in the network and gives a breakdown of the flow through the small holes in the grid plate, the flow through the interior of the core components and the leakage flow.

For the thermal-hydraulic calculations to be performed by COBRA-IIIC only the relevant flow in the heated section of the fuel element cooling channels is taken from the FLAC analysis. On the basis of the channel axial power profiles that follow from the core neutronic calculations, COBRA-IIIC then evaluates the flow distribution in the array of heated cooling channels. These axial power profiles are calculated with the 3D Monte Carlo code KENO V.a [4]. If hot channels are specified, hot channel and hot spot factors are applied to the temperature rise in the coolant, the velocity and the heat flux. These modified values are then used to calculate the safety margin with respect to the occurence of flow instabilities.

Finally SHORT allows for an analysis of the outer fuel plate temperatures at the fuel element boundary. For an unshrouded core like the HOR, the heat removal conditions of such outer fuel plates are quite different from those of the inner plates. This is mainly due to the deviating coolant velocity profile for the channels in between fuel assemblies. For these channels there is a significant increase in coolant velocity downwards from top to bottom.

Code installation and verification

The SHORT code system was developed in the framework of safety analysis studies performed for the HOR [5]. It had been designed to run on the developer's platform for performing this kind of computer calculations. After completion the code system was ported to IRI's computing environment, so adequate testing and verification of the installed modules was necessary. Test input and output was supplied by the authors of the code system. For managing the programming environment different software tools are in use at IRI. First: for quality asssurance purposes the QA FORTRAN package is used for checking the source code. It is a toolset for analyzing, improving and maintaining the quality of FORTRAN programs. Second: a software engineering tool (DECset) is available. It can be used to develop, install, and to maintain a code system in a systematic way.

To verify the correct installation of SHORT these tools were applied to the system and a number of test runs were performed. Some minor problems were detected and subsequently corrections were made. Finally for validating purposes, the results of test cases which had been run on the designers platform were reproduced successfully.

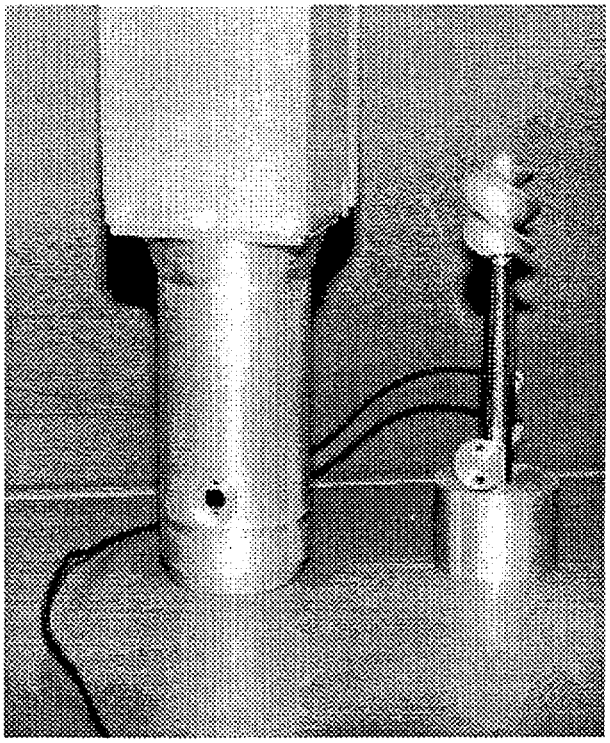

Figure 2. Dummy fuel element with flow meter dismounted from bottom nozzle

EXPERIMENTAL TESTS

Method

The hydraulic data were obtained with a special device, designed for the purpose of in-core measurements, e.i. a dummy fuel element assembly was equipped with a turbine-type flow meter in the lower end fitting as shown in Figure 2. The device was calibrated and certified [6] independently off-site in a dedi-

cated test and calibration loop, owned and operated by the Dutch national calibration institute responsible for maintaining the national measurement standards. The influence of the turbine flow meter was assessed separately by taking pressure difference data of the dummy fuel assembly with and without the turbine.

Measurement program
With the reactor shutdown, in-core measurements were performed under steady-state conditions for two different core types at a large number of grid positions for nominal settings of 67, 80 and 89 dm^3/s for the primary loop flow rate. For determining the influence of the suction head gap bypass flow on incore coolant velocity, measurements were taken with and without temporarily sealing of the suction head gap. Additional to these steady-state measurements transient measurements have been performed in the framework of contributions to the safety analysis report [5]. They consisted of determining the influence of deliberately and instantaneously forcing down the suction head while the primary pump kept running and recording flow coast down after shutting-off the primary pump.

Results
From the measurement data, a.o. the core flow rate distribution and the true core coolant flow rate were determined. Figure 3 displays the measured in-core coolant velocity distribution for a nominal primary flow rate of 80 dm^3/s for the (mixed) compact core. The average coolant velocity was 0.961 m/s. It can be seen that in general, the coolant velocity is highest in the central core region, whereas at the core boundary velocities are somewhat lower. The spread of the relative flow rate is from 0.978 to 1.025. Figure 3 also displays the relative increase of the incore flow rate when sealing of the suction head gap. It can be seen that the influence is only marginal.

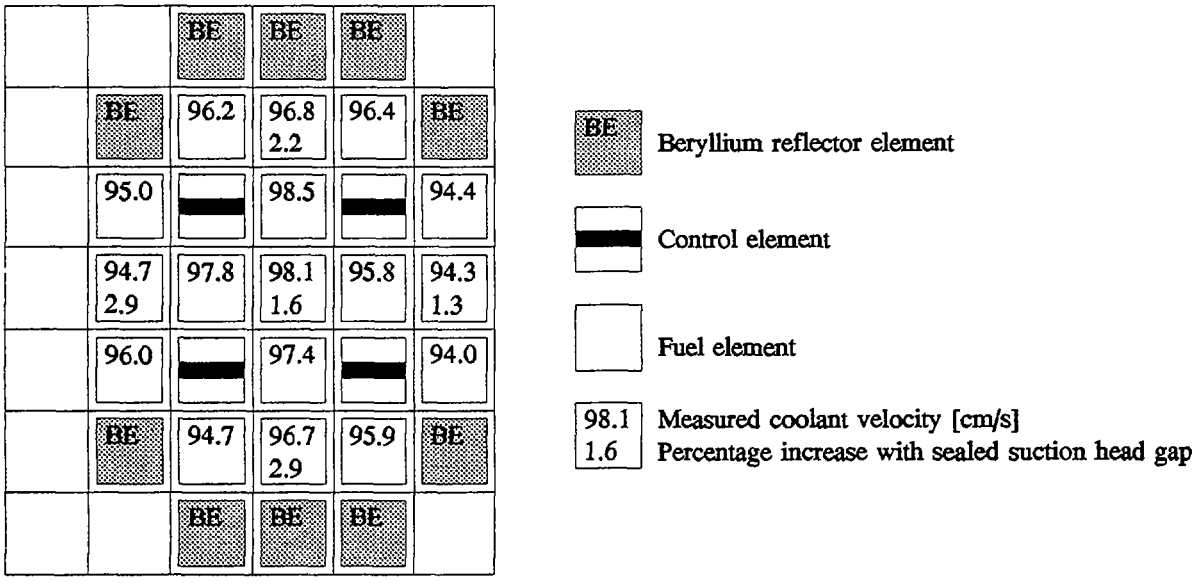

Figure 3. Coolant velocity distribution

EVALUATION OF MEASUREMENTS AND IMPACT ON MODEL

Fuel element flow rate
To evaluate the validity of the SHORT program, calculations were performed for the experimental test conditions of the simulated compact core. Compared to the experimental test results, the suction head bypass flow rate was overpredicted substantially with underprediction of the active core flow. According to this an adjustment of the flow network in SHORT was made. The suction head bypass of originally 5.5% of total flow for the compact core model was reduced by adjusting

the flow resistance of the specific flow path of the resistance network to meet the measured bypass. No other changes in the input or modelling were made. The calculated fuel element active flow rate increased due to the reduction of the suction head bypass by 3.4 %. It is concluded that after the adjustment the true active coolant velocity is still conservatively underpredicted by SHORT by 3.9 %. The results are presented in Table 1.

Table 1. Comparison between test results and apriori and posteriori SHORT-calculation for the flow paths in the simulated compact core.

Flow path	Flow [kg/s]		
	Test	SHORT apriori	SHORT posteriori
Active cooling channels	68.43 (86.2%)	63.57 (80.1 %)	65.73 (82.8 %)
Absorbers and small holes in grid plate	9.26 (11.7 %)	11.46 (14.4 %)	11.85 (14.9 %)
Suction head bypass	1.7 (2.1 %)	4.37 (5.5 %)	1.80 (2.3 %)

Flow distribution across the core

The fuel element flow rate was measured at 17 grid positions in the simulated mixed core. These results were compared to an apriori assumed design value of 1.03 for a nuclear hot channel subfactor used in the SHORT program. This subfactor relates to the nonuniform mass velocity at the core inlet affecting individual fuel elements. A value of 1.03 corresponds to an assumed flow reduction of 3% at the inlet of the fuel element containing the hot channel. Since the radial nuclear hot channel factor occurs in the central region of the core the apriori design value is conservative for the planned core configurations and loading patterns. Thus this factor could be set to 1.0 thus increasing the design value of the fuel element flow rate by 3 %.

CONCLUSIONS

The true core coolant flow rate for different core conditions and flow distribution was assessed by experimental methods for validating the thermal-hydraulic code SHORT. This code is used for core design and in-core fuel management purposes for the HOR. The in-core fuel management of mixed HEU/LEU cores requires special attention with respect to power peaking constraints. Based on the results mentioned above and the experiences so far, the code SHORT is considered to be a valuable asset as a design tool for in-core fuel management, serving proper guidance during the HOR conversion process.

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Reactivity variations associated with the core expansion of the MARIA research reactor after modernisation.

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ABSTRACT.

Polish high flux research reactor MARIA is a pool type reactor moderated with beryllium and water and cooled with water. The fuel is 80% enriched uranium, in the shape of multitube fuel elements, each tube made up of UAl_x alloy in aluminium cladding.

MARIA reactor has been operated in the year of 1977 - 85 and then it was modernised and again put into operation in December of 1992. The modernisation as regarded the reactor core comprised a beryllium matrix expansion from 20 to 48 blocks.

Within the frame of the power start-up and trial operation the reactor has been extended from 12 to 18 fuel channels.

On that stage of reactor operation the power of mostly loaded fuel channels was constrained to 1,6 MW.

Reactor has been operated within the 100-hrs campaigns for an irradiation of target materials and for performing measurements at the horizontal channels outlets.

In the previous time it has been noticed substantial differences in reactivity changes of the core in similar campaigns of reactor operation. It concerns the reactivity losses during poisoning period of the reactor within the first 30 - 40 hrs of operation as well as in the fuel burning up process.

An analysis of the reactivity variations during the core extension will made possible the fuel management optimisation in further reactor operation system.

1. INTRODUCTION.

Modernisation of the MARIA research reactor as related to the core was connected with extension of beryllium matrix from 20 to 48 blocks which was accomplished by the substitution of the retired graphite reflector blocks with the new beryllium ones.

In the first stage of the power start-up and the trial operation the work of reactor has been based on the 12 fuel rods core configuration. After that the reactor core was extended until the aimed configuration has been achieved which enabled to produce the radioisotopes and perform the neutron beams investigation at the horizontal channels' outlets.

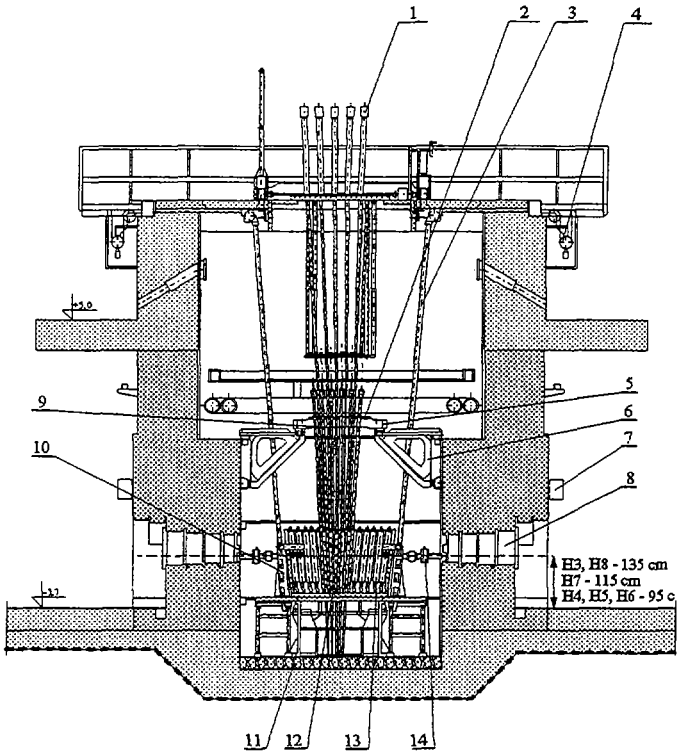
The overall reactor power resulted from the fuel channels' thermal power of which the most thermal loaded channel was operated with a power which did not exceed 1,6 MW.

Reactivity excess of the core during the irradiation process had to be contained within the limits of ρ min providing the continuity of the reactor operation duration of 100, 120 or 260-hrs, and ρ max. being imposed by the operational constraints of Safety Report.

The conclusions following from the presented work, will allow to optimise the fuel management during the further reactor operation.

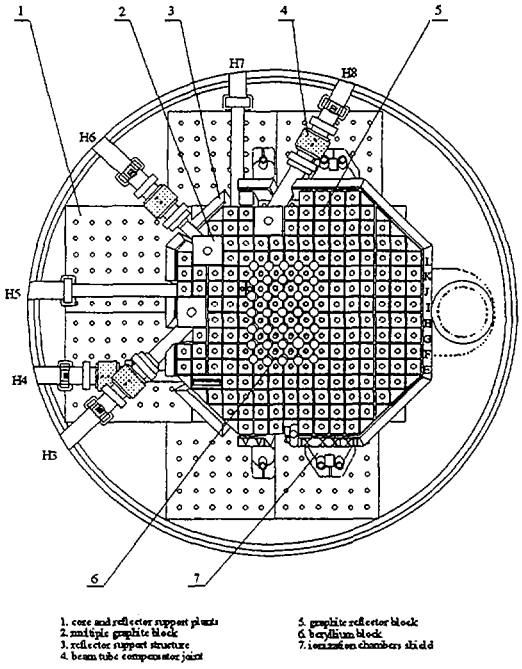
2. DESCRIPTION OF REACTOR MARIA.

A vertical cross section across the reactor pool is shown in Fig.1. The reactor core, fuel and loop channels, headers, and connections between the headers and fuel channels are all submerged in the pool under a layer of water ensuring sufficient radiation shielding above the core.



1. control rod drive mechanism
2. mounting plate
3. ionization chamber channel
4. ionization chamber drive mechanism
5. fuel and loop channels support plate
6. plate support console
7. horizontal beam tube shutter drive mechanism
8. beam tube shutter
9. fuel channel
10. ionization chambers shield
11. core and support structure
12. core and reflector support plants
13. reflector blocks
14. beam tube compensator joint

Fig. 1. Vertical cross-section of the MARIA reactor



1. core and reflector support plants
2. multiple graphite block
3. reflector support structure
4. beam tube compensator joint
5. graphite reflector block
6. beryllium block
7. ionization chambers shield

Fig. 2.Horizontal cross-section of the MARIA reactor

The characteristic design feature of the core is a conical arrangement of fuel channels. The fuel channels are situated in beryllium matrix made of blocks of 110 cm height and enclosed by a lateral reflector made of graphite blocks in aluminium cans. Some of the blocks contain horizontal holes for extraction of neutron beams from the reflector to the horizontal beam tubes penetrating the reactor lateral shield. A view of the reactor core and reflector from above is shown in Fig.2.

In area of reactor core and reflector there is a number of vertical irradiation channels, made of aluminium tubes. The reactor core layout, shown in Fig. 3, gives an example of core management, which fulfils the needs of radioisotope production and several experimental programs.

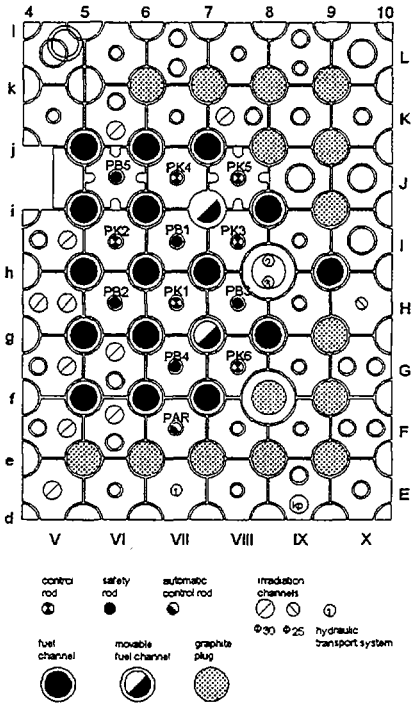


Fig. 3 Reactor MARIA core layout

3. REACTIVITY VARIATIONS DURING THE REACTOR CORE LAY-OUT DEVELOPMENT.

The subject of this analysis is relevant to the typical reactor operation cycles (No. 1, 2, 3 and 4) in the consecutive stages of the core extension during power start-up and the trial operation. A reactivity margin enabling the accomplishment of the assumed reactor operational cycles must cover reactivity losses caused by the poisoning effects and the fuel burn-up.

The data concerning the fuel channel position inside the reactor core, specific fuel burn-up and thermal power are contained in Table 1.

Table 1
[MW/%]

	Cycle No 1				Cycle No 2				Cycle No 3				Cycle No 4				
	5	6	7	8	5	6	7	8	5	6	7	8	5	6	7	8	9
j		0,69	0,65			0,64	0,56		0,81	0,89	0,74		0,59	1,08	0,66		
		0,18	0			6,87	6,31		0	27,7	27,0		15,6	4,0	39,2		
i	0,98	1,11	0,88	0,50	0,98	1,27	1,04	0,64	1,17	1,54	1,11	0,71	0,90	1,37	0,97	0,52	
	0,50	0,55	0,39	0	9,15	1,04	8,52	4,76	25,3	15,8	25,6	28,1	24,5	15,3	26,2	37,1	
h	1,23	1,32	1,01		1,35	1,53	1,31		1,4	1,45	1,5		1,15	1,40	1,21		0,39
	0,99	0,96	0,68		11,1	11,6	9,43		18,6	24,8	17,7		18,4	14,0	24,3		2,8
g	0,70	0,74	0,51		1,23	1,33	1,11	0,59	1,33	1,50	1,16	0,59	1,07	1,18	1,17	0,54	
	0,58	0,56	0,40		8,9	8,96	6,78	0,22	21,8	20,5	26,2	26,0	22,3	25,5	14,8	32,6	
f						0,88	0,71		0,92	1,13	0,73		0,74	0,84	0,76		
						0,81	0,02		29,3	20,0	29,5		21,6	25,9	8,6		

3.1. BASIC CHARACTERISTICS OF THE CONSECUTIVE CONFIGURATION OF REACTOR CORE.

The first operational configuration of reactor core comprised 12 practically new fuel elements. In the considered first cycle (No 1) the reactor has been operated for 4 days with an overall power of 12 MW, and the power of the most loaded fuel channel was 1,32 MW.

This core had a large built-in reactivity which in the initial state was ρ_1 = 8 $. Reactivity loss caused by the reactor poisoning achieved its steady state after 40 hrs from the start-up time. In this time the poisoning effects and the fuel burn-up caused the negative reactivity effect $\Delta\rho$ = 5,9 $. Reactor was shut down after 97 hrs of operation when the reactivity excess was ρ_2 = 1,9 $. Due to that the fuel burn-up coefficient was equal to 0,007 $/MWd.

The second configuration of reactor core consisted of 15 fuel channels featured with low burn-up (<12%) because in the meantime the reactor was operated with a variable power of 5÷15 MW within 430 hrs.

In the example being considered cycle (No 2) the reactor was operated with an overall power of about 19 MW, and the power of the most loader fuel channel was 1,53 MW.

In the initial state reactivity excess of the core was ρ_1 = 5,4 $. The reactivity drop caused by the phenomenon of reactor poisoning reached the study state after 32 hours of the reactor operation and the combined effect together with fuel burn-up was 3,8 $.

The final reactivity margin was ρ_2 = 1,47 $ when the reactor has been shut down after 82 hours of its operation.

Bearing in mind aforementioned data one can determine the fuel burn-up coefficient which was equal to 0,005 $/MWd.

The third reactor core configuration contained 17 fuel channels with a fuel burn-up of about 20%. In the mean time the reactor was being operated with a power of ca.23MW in 100, 120 and 264-hours cycles within about 925 hrs.

In the considered working cycle (No 3) the reactor has been operated with power of about 24 MW and the power of the most loaded fuel channel was 1,54 MW. In the initial state reactivity margin was 5,7 $. After 30 hrs of reactors operation the reactivity margin due to the fuel poisoning approached the saturation state and the total reactivity loss (due to fuel poisoning and burn-up) was equal to $\Delta\rho$ = 3,8 $. In the further period of reactor operation for about 70 hrs reactivity loss was equal to about 1 $. From the above data one could concludes that the burn-up coefficient was 0,014 $/MWd.

The fourth reactor core configuration was set up with 18 fuel channels and an average fuel burn-up level was similar to that one previously considered. In the mean time reactor has been operated for about 2500 hrs with a power of 20÷23 MW. In the approaching time for the actual configuration, 18 spent fuel channels with an average burn-up of about 42% have been gradually unloaded from the reactor core and they were substituted by the new ones.

In the considered cycle (No 4) reactor was being operated at a power of about 23 MW and the power of the most loaded fuel channel was equal to 1,4 MW. Initial reactivity margin was 5,4 $ and the

reactivity loss approached a steady state due to fuel poisoning after ca. 30 hours and was equal to $\Delta\rho$ = 3,7 \$. The further reactivity drop was caused by the fuel burn-up within the next consecutive 75 hours and was equal to 0,9 \$.

3.2. COMPARATIVE ANALYSIS OF REACTIVITY VARIATIONS DURING THE CORE EXPANSION.

In the first working stage the reactor core has been characterised by a large reactivity margin which followed from the state of fuel being practically fresh. This core was also featured by the largest reactivity efficiencies of the control rods. For this configuration within the first operation hours the reactor poisoning was mainly created by the production of Xe-135 which quantity approached the saturation state after 40 hours, and the magnitude of the poisoning was equal to about 5,5 \$.

The further reactor operation resulted in the overall reactor poisoning in which a bigger share was constituted by the built-up of Sm-149. Due to that the reactivity loss for the considered cycle No 4 within the first several dozen hours of reactor operation was substantially smaller than for cycle No 1 (3,5 \$ in comparison with 5,5 \$) which is illustrated in Fig.4.

The core expansion was accompanied by the time diminished from 40 to 30 hours in which the poisoning phenomenon reached the steady state.

In the first trial operation period the reactivity loss was significantly lower, after the reactor saturation poisoning was achieved (Fig. 4).

This effect has been caused by burning-up of the poisons' (He-3) being accumulated in beryllium blocks which were used in the reactor before its modernisation, i.e. up to 1985.

The lowest reactivity losses connected with reactor operation have been noticed for 15-fuel channels' configuration. In that system the reactor core constituted relatively compact lay-out.

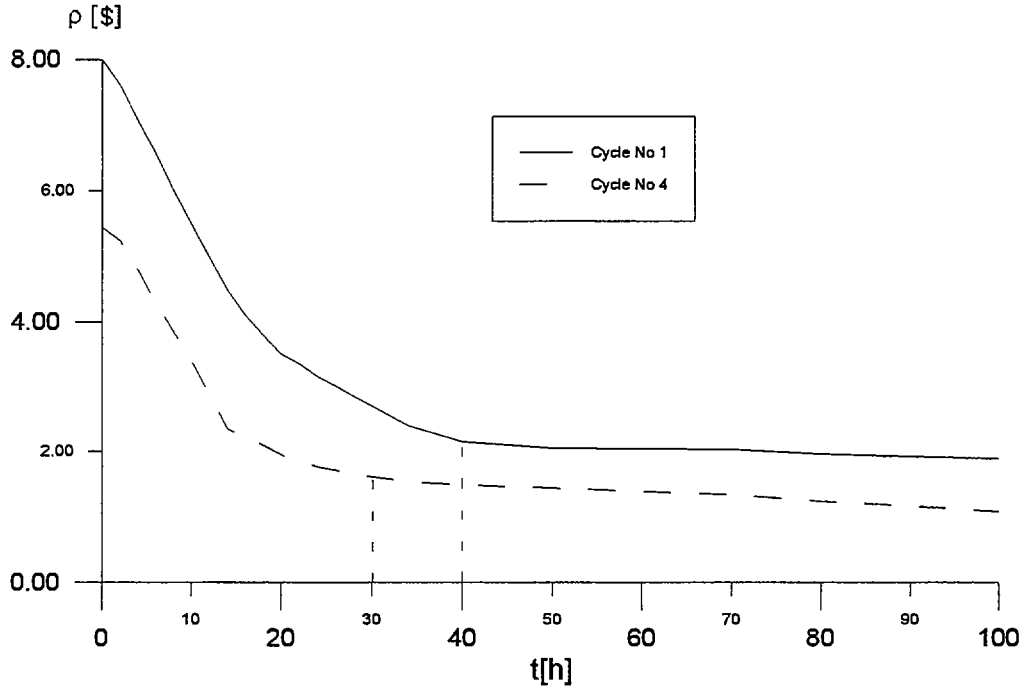


Fig. 4. The reactivity variations during reactor operation.

4. SUMMARY.

A beryllium matrix expansion by means of applying the new beryllium blocks instead of the graphite reflector improved the reactivity balance of the reactor core.

During the core expansion one has to endeavour to achieve the compact lay-out of the fuel channels for which the reactivity losses are getting to be lower.

The basic constraint for optimisation of fuel management in the MARIA reactor core is connected with providing the opportune conditions for isotopes' irradiation in vertical channels and for neutrons physics investigation at the outlets of horizontal channels.

For the bigger core configurations it is possible to achieve large specific fuel burn-up by applying the principle that the fresh fuel channels are installed in central area of the core, whereas the fuel channels with a higher burn-up on the core boundary.

OSIRIS: AN EXAMPLE OF GRADUAL CONVERSION

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Mr. Guidez, Mr. Beylot and Mr. Joly

SRO - CEA - Saclay - FRANCE

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Paper presented at the RRFM conference in Bruges, 5 to 7 February 1997

1 - RECAP OF CONVERSION OF OSIRIS TO SILICIDE

The conversion of the Osiris core from "Caramel" to silicide forms part of the move towards standardising a well-known fuel, developed in the context of the RERTR programme, and used by the majority of reactors. This standardisation makes it possible to anticipate reduced manufacturing and reprocessing costs in the long term. When the reactor went critical in 1966, the core was loaded 93% in UAl and an initial conversion was made in 1985 towards Caramel fuel (7.5%). This conversion was carried out in one go, and involved certain reactor modifications (increase of core flowrate etc.).

For the conversion to silicide, two modes have been considered:

- either the conversion in one go, by creating a core entirely composed of new U_3Si_2 elements. For this initial core, the uranium masses of the various elements must be different, so as to best simulate the burnup (expressed in megawatt days per tonne of total uranium, MWd/t) of the elements constituting a core in the steady state (in Osiris the most reactive elements are placed around the edges of the core).

- or gradual conversion, consisting of introducing an increasing number of U_3Si_2 elements in the core as the spent Caramels are unloaded. Therefore, with this system Caramel elements and silicide elements co-habit until the core has been completely converted.

Following financial analysis, the second mode was adopted, as it allows the complete stock of Caramel elements to be gradually used up.

The saving corresponds to the cost of around thirty Caramel-type fuel elements (which it would have been necessary to burn incompletely) and around twenty "exotic" silicide elements, with a lower ^{235}U load (which would have needed to be specially manufactured for a limited lifetime). The corresponding financial gain was estimated in 1990 as being around 20 million French Francs.

2 - OVERVIEW OF PREPARATORY WORK REQUIRED TO OBTAIN SAFETY AUTHORITY APPROVAL AND TO START THE CONVERSION

2.1 - Reactor compatibility

An identical outside geometry, a similar flowrate and line-loss, and a comparable hafnium basket make the whole silicide core compatible with the reactor in its current state. Furthermore, this enables the core to be converted gradually. Therefore, the following consequences can be reiterated:

- compatibility with the existing shell,
- no need to modify main and secondary coolant systems,
- identical control rod drop drive mechanism and control rod drop time (after counter-weight has been modified),
- no modification to instrumentation and control,
- identical operating mode and rules,
- fuel storage and handling rules retained.

With regard to the differences, the following points are to be noted:

- the weight of the element is lower. The core overall is thus lighter. Change from around 1.3 to 0.8 t,
- the control rod counter-weight needs to be modified,
- the body of the element is no longer zircaloy, but AG3. With regard to the materials, this constitutes a return to a familiar situation for the operating organisation (UAl).

2.2 - Neutron studies

We adopted the "conventional" solution, already used at Osiris, with the highly-enriched U-Al fuel, of placing boron in the side plates of the standard fuel elements and the followers of control elements.

This studie revealed:

- an increase in the number of elements consumed of around 30%,
- an increase in γ heating (to be taken into account for the experimental devices),
- little variation in thermal flux over the outside surfaces (+9% to +7%),
- a very advantageous increase in thermal flux inside the core (+20% to 36%).

2.3 - Hydraulic tests

The basic principle adopted when the silicide fuel characteristics were being adopted was to retain a circulation flowrate in the core and thus a line loss comparable to that of the Caramel core.

Full scale hydraulic tests were then conducted on a standard element to confirm its line loss characteristics and to check its behaviour under forced circulation.

These tests made it possible to confirm the final positioning adopted with 22 plates. The line loss measured for the element is slightly lower than that of the Caramel element.

2.4 - Thermal hydraulic studies

2.4.1 - Core in steady-state thermal hydraulic regime

The thermal hydraulic calculation of the core was carried out with the FLICA 3 code for the whole core, and for a hot channel of each type of fuel present in the core (standard element and control element).

The calculations were carried out for several mixed cores.

It can be seen that the removable thermal power on the silicide element is systematically greater than that calculated on the standard Caramel elements.

2.4.2 - Core in transient thermal hydraulics regimes

The calculations modelling the whole core plus the main coolant system were made using the SIRENE code, so as to characterise the thermal hydraulic behaviour of the fuel.

The calculations were carried out in three transient situations:

- main coolant system pumps coasted down,
- jamming of an impeller of one of the pumps,
- guillotine break of a pin header

Five mixed cores were studied in these situations :

It thus appears that the core plates are correctly cooled in all situations.

From the 2/3 silicide core, the behaviour of the all-silicide core is better than that of the all-Caramel core with better cooling characteristics.

This is due to the fact that the heat capacities of the cores studied decrease during the conversion. The silicide core thus discharges the accumulated energy faster than the Caramel core.

2.5 - Studies relating to control rods

The principle adopted for the follower elements is to retain the hafnium baskets of the Caramel fuel rods with a slightly different mass and design.

In practice, the "neutron weight" of these two follower elements is comparable, which explains the possibility of using the Caramel follower elements until the end of conversion to silicide.

By taking the weight of the Hafnium basket into account, the whole weighs around 10 kg less. The flowrate and the hydraulic thrust are comparable. This leads to a corresponding modification of the control rod counter-weights, to compensate for this variation. Tests were carried out with the new control rod, in particular to qualify the drop time.

The neutron characteristics of the new control rod were also qualified during specific tests on ISIS.

2.6 - Radiological consequences of accidents

The radiological consequences of a Borax type accident and an accident involving fusion of a fuel element in air were studied.

The radiological consequences of the fusion of a plate underwater are similar to those of the Borax accident divided by 10^3 (the fuel consisting of around 919 fissile plates).

These studies show that the upper-bound accident consists of:

- fusion in air as regards internal exposure to the thyroid,
- the Borax accident for the other cases of exposure.

2.7 - Other studies

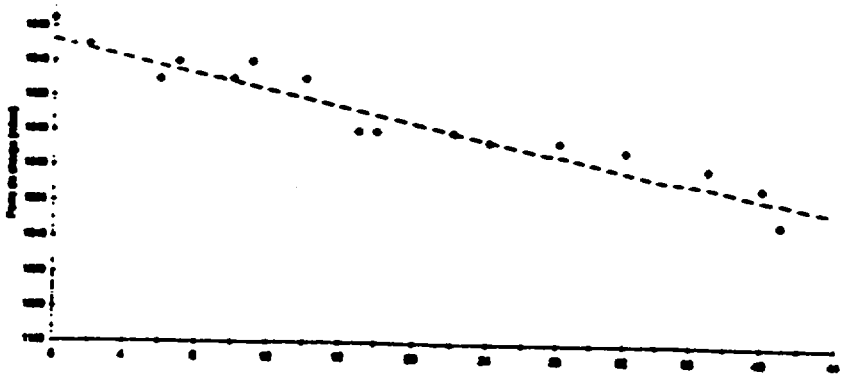
Other studies have been conducted on the following points:

- storage and handling,
- reprocessing,
- reactivity accidents,
- behaviour in event of cladding rupture (Epsilon tests at Siloë),
- tests in the reactor on fuel assemblies with modified manufacturing specifications,
- etc..

3 - PROGRESSION OF CONVERSION

- The conversion authorised in 1994 began in January 1995 to be ended in November 96 (it remains only one control rod in Caramel).

- The ΔP measured for the core changed from one cycle to the next, as predicted in the calculation.



- The neutron parameters changed as predicted. In particular, the γ flux measured by calorimeter changed from one cycle to the next to reach the calculated value. This led to gradual adjustments throughout the year of conversion, of the thermal adjustments of the test devices.

- A fuel assembly was specially equipped with a plate containing five thermocouples set into the cladding.

This will make it possible, during the tests planned for May 1997, to validate the plate temperature changes predicted in the thermal hydraulic calculations for several positions in the core, several power levels and several transients.

- In conclusion, the gradual conversion took place as expected and complied with the initial predictions.

4 - OVERVIEW AND CONCLUSION

Progressive conversion presents a certain number of disadvantages:

- safety dossier more unwieldy to manage (calculation of a great number of cores),

- more restricted design of new fuel because it must meet certain imposed hydraulic (line loss) or neutron parameters,

- difficulties throughout the transient period (around 1 year) for experiments whose parameters are changed (in particular, gradual increase of around 80% of γ heating in the core in the case of the Osiris conversion).

However, this formula, when it is possible, presents a definite financial advantage (minimisation of reactor modifications and fuel saving) which largely justifies the disadvantages mentioned above.

Session 4:

Back-end Options and Transportation

MTR SPENT FUEL REPROCESSING THE COGEMA ANSWER

by
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Both the biggest nuclear fuel reprocessing facility and the most modern one, the La Hague plant is know world-wide by all the actors involved in the nuclear industry. High technology solutions which are implemented in the plant's two reprocessing lines combined with very good level of performance have led to an indisputable leadership. The first line, named UP2, entered industrial service in 1966. Since then, its design annual capacity has been increased from 400 to 800 tons. The new, improved unit, which is now known as UP2-800, has been operating for 2 years. Another reprocessing line, UP3, was commissioned in 1990 and is now delivering its nominal 800 t/y capacity after only five years of service. However, despite these tremendous achievements, some of La Hague activities, mainly in the research reactors spent fuels reprocessing, remain less known.

COGEMA has gained a great experience in material testing reactor (MTR) fuels reprocessing. The UP2 plant, for instance, has processed several different types of fuel since it begun its operations. Although most of them were Light Water Reactors fuels from BWRs and PWRs, some « exotics » fuels like those coming from RNR were also reprocessed.
Also variable are the weights of the elements which can be handled by La Hague. They range from 150 to 550 kg. The same flexibility could be observed in the burn-up field as the reprocessing plant is able to manage elements from 10 000 to 45 000 Mwd/t.
The plant is already available for aluminides spent fuels coming from MTRs. Research and development works are under way to reprocess silicides, Metallic Natural Uranium, UO2 and some other special fuels. COGEMA has also a significant experience in Highly Enriched Uranium fuels as it has already reprocessed for many years those coming from the Commissariat à l'Energie Atomique's (CEA) Orphee and Siloe MTRs in COGEMA's UP1 plant in Marcoule.

Ideally located near the European countries which operate MTRs , the La Hague reprocessing plant is the best solution for the reprocessing of their spent fuel. The facility could offer the same benefit than those already offered to the European utilities' nuclear power plant:

- Reduction of the ultimate waste overall toxicity and volume
- Reprocessing of the depleted uranium which could be re-enriched
- Recovering of the plutonium. Current reprocessing operations on PWR spent fuels recover 99,88% of the Pu contained.
- Safe and reliable confinement of the final waste through a proven, efficient vitrification process. Glass is the solution retained for High Level Waste storage. This process, which has been developed by France for more than 30 years, is today internationally recognised as the best solution for High Level Waste storage as it confines over 99% of the activity present in the spent fuel.

After being vitrified, the ultimate residues are conditioned in a standardised container called the « Universal-Canister » (UC). This canister could accommodate either vitrified residues or compacted wastes. Its weight is around 500 kg for glass products and 800 kg for compacted residues. The UC is 1,34 m high and has a diameter of 43 cm.
COGEMA's efforts to reduce the final volume of ultimate waste makes it possible today to generate only two Universal Canister (one for vitrified fission products and one for compacted residues) for every ton of aluminides spent fuel reprocessed. Compared to the direct disposal option, the reprocessing solution, as offered by COGEMA to the MTR operators, represents a reduction by a factor of 50 of the waste's volume which will have to be disposed in a geological repository.

MTR's operators can rely on a mature industrial process. From a production stand-point, 9 645 t of LWR fuel have been reprocessed at La Hague by July 31, 1996 among which 5 987 t where reprocessed at UP2-800 and 3 658 t at UP3. Since last year, both plants have reached their full capacity of 1600 t/yr. However, thanks to innovative design and technical solutions, the La Hague facilities benefit from outstanding adaptability and flexible capacities.

La Hague facilities are under EURATOM and AIEA's control. In 1995, almost 1 700/ man/day of international inspections have been performed at La Hague thus making this facility one of the most, not to say the most, safeguarded civil nuclear place in the world.

The environmental issue is as satisfactory. Environment protection is regularly monitored by COGEMA. More than 20 000 samples of water, soil, air, sand, meat and vegetables are collected annually around the La Hague plants which lead to 80 000 analysis each year. The occupational doses to workers are, in average, a hundred times less than those currently authorised in France and about 10 times less than natural exposure.

Both the experience gained throughout the years and the large scale operations done at UP2-800 and UP3 enable COGEMA to offer its customers the most efficient solution for their spent fuels.

The paper will focus on the technology used at La Hague and its application to the reprocessing of MTRs spent fuels.

Handling of Spent Fuel from Research Reactors in Japan

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ABSTRACT

In Japan eleven research reactors are in operation as shown in Table 1. After the 19th International Meeting on Reduced Enrichment for Research and Test Reactors (RERTR) on October 6-10, 1996, Seoul, Korea[1], the Five Agency Committee on Highly Enriched Uranium, which consists of the Science and Technology Agency, the Ministry of Education, Science and Culture, the Ministry of Foreign Affairs, Japan Atomic Energy Research Institute (JAERI) and Kyoto University Research Reactor Institute (KURRI), was held on November 7, 1996, where the handling of spent fuel from research reactors in Japan was discussed as follows:

1. So far, the number of spent fuel elements in JAERI and KURRI to be returned to the US are shown in Table 2. There are some more spent fuel from other institutes, but it was not discussed in details in this committee.

2. Advantages and disadvantages to return them to the US in comparison with that to Europe are also discussed.

3. The first shipment to the US is planned for 60 HEU elements from JMTR in 1997. The shipment from KURRI is planned to start in 1999.

Table 1. Japanese Research Reactors in Operation

Name	Owner	Site	Type and enrichment			Max. Power	Start-up date
JRR-2	JAERI	Tokai	D_2O (CP-5)	U-Al UAl_x-Al	93% 45%	10MW 10MW	1960. 10 1987. 11
UTR KINKI	Kinki University	Higashi-osaka	H_2O (UTR)	U-Al	90%	1W	1961. 11
TRIGA-II RIKKYO	Rikkyo University	Yokosuka	H_2O (TRIGA)	U-ZrH	20%	100kW	1961. 12
TTR-1	Toshiba	Kawasaki	H_2O (pool)	U-Al	20%	100kW	1962. 3
JRR-3	JAERI	Tokai	D_2O (tank) H_2O (pool)	U UO_2 UAl_x-Al	Natural 1.5% 20%	10MW 10MW 20MW	1963. 9 1972. 1 1990. 3
MuITR	Musashi Inst. Tech.	Kawasaki	H_2O (TRIGA)	U-ZrH	20%	100kW	1962. 3
KUR	KURRI	Kumatori	H_2O (tank)	U-Al	93%	5MW	1964. 6
JRR-4	JAERI	Tokai	H_2O (pool)	U-Al	93%	3.5MW	1965. 1
JMTR	JAERI	Oarai	H_2O (MTR)	U-Al UAl_x-Al U_3Si_2-Al	93% 45% 20%	50MW 50MW 50MW	1968. 3 1986. 7 1994. 1
YAYOI	University of Tokyo	Tokai	fast (horizontally movable)	U	93%	2kW	1971. 4
NSRR	JAERI	Tokai	H_2O (TRIGA)	U-ZrH	20%	300kW	1975. 6

Table 2. Spent Fuel Elements to Be Returned to the US

Enrichment	Reactor Name	Number of Elements
HEU or MEU	JRR-2	155
	JRR-4	44
	JMTR	738
	KUR	232
Subtotal		1,169
LEU	JRR-3M	237
	JRR-4	1
	JMTR	184
Subtotal		422
Total		1,591

Reference

1. K. Kanda et al., "Status of Reduced Enrichment Program for Research Reactors in Japan", presented at the 19th RERTR Meeting, Oct. 6-10, 1996, Seoul, Korea.

Fuel transportation for research reactors.

S. Hérin, H. Libon, H. Sannen
TRANSNUBEL

Abstract

The requests of transporters are not always well understood by those people who are responsible for sending nuclear material on the road. This is also true for the operators of research reactors. This presentation will try to clear up the origins and the object of such requests.

A first part is devoted to the regulatory requirements imposed on the transportation of nuclear fuel.The regulatory requirements applicable to the transportation of radioactive material find their origin in the socalled United Nations Orange Book. This is the basis of the present IAEA regulations, which in turn are put into force by the member states end by the modal organisations (ADR, IMO, ICAO, ...) in their own regulations or laws. The packages themselves are the center of the safety of transports. The packages must fulfill a set of requirements such that they retain their characteristics of leak tightness, shielding, heat dissipation and, in the case of fissile material, the absence of criticality risk, even in accident conditions. The way this can be proved and achieved (tests, calculations, analogies, ...) is described. The requirements for the performance of the transport itself are analysed as well.
A particular emphasis is made on the necessity of Quality Assurance on the design, fabrication, maintenance and use of packagings as well as on the preparation and the performance of the transports themselves.
As we are dealing with such sensitive material as Uranium (sometimes highly enriched) or Mixed Oxide Fuel very strict rules must be applied for the protection of that material. The minimum requirements for the physical protection of nuclear material are set forth in the New York and Vienne Conventions signed by most of the States all over the world. The application of those rules are described.

The second part of this paper outlines the main features (dimensions, weight, contents and their limitations) of commonly used packagings for the transport of research reactors fuel, a distinction being of course made between the transport of fresh fuel (packagings such as MTRD, GB1612, TNB145, FS13, ...) and of irradiated fuel (packagings such as Unifetch, Pegase, TN6, TN7, ...).

In the last part, some words are given over the various modes of transportation (mainly road, sea and air) and their limitations.

WORLD-WIDE FRENCH EXPERIENCE IN RESEARCH REACTOR FUEL CYCLE TRANSPORTATION

Danièle Raisonnier - Transnucléaire

ABSTRACT

Since 1963 Transnucléaire has safely performed a large number of national and international transports of radioactive material. Transnucléaire has also designed and supplied suitable packagings for all types of nuclear fuel cycle radioactive material from front-end and back-end products and for power or for research reactors.

Transportation of the nuclear fuel material for power reactors is made on a regular and industrial basis. The transportation of material for the research reactor fuel cycle is quite different due to the small quantities involved, the categorisation of material and the numerous places of delivery world-wide.

Adapted solutions exist, which require a reactive organisation dealing with all the transportation issues for LEU (Low Enriched Uranium) and HEU (Highly Enriched Uranium) products as metal, oxide, fresh fuel elements, spent fuel elements including supply of necessary transport packaging and equipment.

This presentation will :

- explain the choices made by Transnucléaire and its associates to provide and optimise the corresponding services,

- demonstrate the capability to achieve, through reliable partnerships, transport operations involving new routes, specific equipment and new political constraints while respecting sophisticated Safety and Security regulations.

INTRODUCTION

Leading supplier of containers for the nuclear industry, Transnucléaire has a broad experience in the design, licensing and manufacture of all the packagings required. Transnucléaire provides « door to door » transport services for all types of radioactive material for research reactors.

CURRENT ISSUES ON TRANSPORTATION OF NUCLEAR MATERIAL FOR RESEARCH REACTOR

The transport of fresh and irradiated material from Light Water Reactors (LWR) is made on an industrial scale in Europe and from Japan to Europe. The French experience gained in this field has demonstrated its maturity in terms of safety, reliability, cost-efficiency and environment impact.

The transportation of fresh MTR or Triga material (metal or oxide fuel elements) is subject to the same basic regulations regarding safety with more stringent requirements with regard to security (physical protection aspects). The transportation of MTR spent fuel is subject to the same regulations as LWR spent fuel.

To be able to provide and optimise the transports services in full compliance with applicable regulations and customers requirements it is our role to pay a special attention to the following issues :

Transport Regulations

The transport of all radioactive materials is regulated by the IAEA (International Atomic Energy Agency) Regulations (Safety series n° 6) which are generally incorporated in national and international regulations. Following a comprehensive review by panels of experts a revised version of the IAEA Regulations for the Safe Transport of Radioactive Material was approved by the Board of Governors in October 1996. This updated regulation will be in force at the turn of the century and incorporated in national and international regulations at the same time.

The basic principle of the IAEA Regulations is that the safety is vested in the package permitting the transport by all conventional modes (road, rail, air and water). It is nevertheless important to take now into consideration two key amendments of this 1996 revision regarding **air transport** :

- a more robustly designed packages (type C package) will be required for high activity material
- additional subcritical conditions are imposed for packages containing fissile material.

With regard to **marine transport** the International Maritime Dangerous Goods (IMDG) Code has been supplemented in early 1995 by the INF Code laying down stringent requirements for ships transporting Irradiated Nuclear Fuel, Plutonium and High-Level Radioactive Waste. This code is not a mandatory instrument but is now implemented in numerous countries. The INF Code should be kept under review and be amended as necessary to cover other issues such as : route planning, notifications, tracking and third-party liability.

Cask licensing

A special attention has also to be paid to the renewal of certificates of approval for packages in use for several decades. Most of the competent authorities are taking this opportunity to review more deeply the safety assessments (particularly criticality aspect and drop accident conditions) under full Quality Assurance program, keeping in mind their enhanced Compliance Assurance obligations.

TRANSNUCLEAIRE ANSWERS

Cask developments

- **Transport of uranium components (fresh and reprocessed material)**

For material enriched to 20 % or more we have developed the TN BGC-1. The TN BGC-1 is a type B(U)F package designed for the transport and storage of fissile materials such as highly enriched uranium. It is licensed in France -F/313/B(U)F- to accommodate 13 different types of contents. Validations and or approval certificates have been obtained in the United Kingdom, in Spain and in the CIS. Approval is presently under review by the U.S. Department of Transportation (DOT) and should be issued shortly. This package is in full compliance with the 1985 IAEA Regulations and will meet the requirements of the 1996 IAEA Regulations for the transport of Uranium components by all modes of transport.

The packaging consists of a cylindrical body made of stainless steel and resin fitted with a special closure system. The body is surrounded by a protective cage in aluminium. During transport, the top part of the packaging is protected by a shock absorbing cover.

The main features are the following :
- overall dimensions : 600 x 600 x 1821 mm
- empty weight : 280 kg
- maximum loaded weight : 396 kg.

The currently approved contents, as concerns Uranium, are :
- a maximum of 40 kg of uranium enriched in U 235 to any degree, in the form of uranium oxide powder (UO2 or U308).
- ingots or compact assemblies of non-cladded plates with a maximum weight of 45 kg metallic uranium content
- uranyl nitrate up to 95 % enrichment (capacity 10 liters)
- uranium oxide pellets (about 45 kg of UO2)
- uranium in any solid form (up to 17 kg of U235).

The transport of reprocessed uranium is also authorised by this certificate. Transport of other contents as Triga fuel elements should also be considered with additional safety/criticality analysis.

More than 200 TN BGC 1 packagings have been manufactured and are now in operation.

We are working on a new package design for fresh MTR fuel elements in order to comply with the new regulations and to cover the needs of customers asking for better performances and capacity and to cover new fuel design.

- **Transport of spent fuel**

A new family of casks meeting the last requirements of the IAEA Regulation is under development. A cask to replace the IU04 cask with a better capacity will be made available in 1998 to meet several needs : French programs (CEA, ILL) and international programs such as MTR spent fuel return to the U. S. and delivery to COGEMA reprocessing plant.

TRANSPORTATION SERVICES

The quantities of material to be transported for the whole research reactor fuel cycle are not very important in weight, but the problems to solve are numerous. This activity will probably not become a routine activity. Considering the evolution of the situation in several fields (sources of uranium supply, political constraints, evolution of the regulations) it is important for the reactor operators to rely on transport companies having a good knowledge about all international transportation aspects.

Shipments of fresh uranium (metal or fuel elements) are regularly performed between several places in Europe and to Eastern countries and the U.S. All the transport modes can be used depending on the specific requirements of each country. Details about recent transport developments will be presented during the Conference (transport of Fort St Vrain fuel to France for processing and blending, transport of material from Russia to France).

The new possibilities offered to the reactor operators to send back to the U. S. their US-spent fuel assemblies for storage and or to France for reprocessing at La Hague COGEMA plant will increase the number of ocean transports.

To transport this spent fuel (INF material) in full compliance with IMO A 748 Resolution, Transnucléaire proposes now several INF 2-certified ships. Two of them « Bouguenais » and « Beaulieu », are operated under an exclusive contract with a French ship-owner. These ships are not exclusively dedicated to the transportation of nuclear material in order to keep the ocean transport at a reasonable level of price.

These ships have been used for several transports in Europe (Spain, Sweden, France), from Japan to the U.S. and from Europe to the U. S. for delivery of MTR spent fuel to the U. S. Department of Energy Savannah River Site through the military port of Sunny Point in 1995, and Charleston Naval Weapons Station in September 1996.

In conclusion, Transnucléaire and its affiliates demonstrate their capabilities to achieve through reliable partnerships transport programs involving new routes, dedicated equipment and new political constraints while meeting sophisticated regulations.

US Foreign Research Reactor Spent Nuclear Fuel Return Policy - status quo

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The Back-end of the Research Reactor (RR) fuel cycle used to be a matter of great concern since the beginning of 1989 when the **U.S. Department of Energy (DOE)** had suspended its Spent Nuclear Fuel (SNF) return policy, the former so-called "Off-Site Fuels Policy". Under this policy the U.S. accepted, temporarily stored and reprocessed the SNF of U.S. origin from overseas and returned uranium credits to the RR operator for new uranium procurements with DOE.

In Mai 1996, DOE established a new SNF return policy covering SNF of defined RR which is already stored or generated over the next 10 years of the new policy period.

The first shipments, one from Europe with 231 SNF elements and one from South America with 53 SNF elements arrived in the U.S. end of September 1996.

Research Reactor Spent Nuclear Fuel (SNF) Return to the U.S.

Policy

- **'Off-Site Fuels Policy'** (OFP), former return policy
- OFP **expiration:** 1988-12-31 for HEU and 1992-12-31 for LEU

- DOE performed a **'full-blown' EIS** to meet all the NEPA requirements
- EIS result: return of SNF would have **no significant impacts** to environment, workers or public
- 'Nuclear Weapons **Nonproliferation** Policy, Concerning Foreign Research Reactor Spent Nuclear Fuel'
- **1996-05-13:** DOE announcement of 'New RR SNF Return Policy' - **NRP** - date of effectiveness
- return **priorities:** first HEU - than LEU SNF
- all **U.S. origin** RR SNF from overseas covered
- NRP applies to **Al-based** and **TRIGA** (UZrHx) SNF and **target** material
- NRP involves **~19.2 t** HM of SNF (~22,700 elements) and **~0.6 t** HM t target material, all of different U5 enrichments
- NRP covers all SNF **stored** on 1996-05-13 and **discharged** until 2006-05-12 (10 yrs duration)
- **shipments** of concerned quantities can occur until 2009-05-12 (13 yrs period) - 3 yrs cooling
- after that period: full **responsibility** for SNF left with each individual reactor/country
- SNF would be received from **41 countries**
- **types of SNF** from reactors that would be accepted:
  * HEU and/or LEU **operating** or **converting** with/to LEU on 1996-05-13
  * HEU and/or LEU operating on HEU on 1996-05-13, **agree to convert**
  * HEU in case of **life time** cores and will **shut down** during policy period
  * HEU in case **no suitable** LEU is available
  * HEU and/or LEU in case of **shut down**
  * **unirradiated** HEU and/or LEU accepted as SNF
  * LEU only accepted **after all HEU** (reactor related)
  * HEU **target** material after commitment to convert and availability
- HEU and/or LEU not accepted: reactor **start-up** after 1996-05-13

Shipments and Ports of Entry

- reduce shipments/risks by coordination - max. **8 casks/ship**
- DOE estimate: **150-300** ships to Charleston, 5 to Concord
- settlement DOE/INEL: 61 shipments of foreign SNF until 2000
- shipments by **charter** or **regular** container ships
- military ports: **Charleston**, S.C. (~80 %) and **Concord**, Cal. (~5 %)
- military ports for **safety** and **security** reasons (proven infrastructure)
- target material: mainly overland from **Canada**
- transport from U.S. ports to site: by **truck** or **rail**
- DOE **preference** for rail

Management Sites

- **2 sites** prepared for interim storage of SNF
- Al-based SNF: managed in **SRS**, Savannah River, S.C.
- TRIGA SNF: managed in **INEL**, Idaho
- treatment after storage: **packaged** or chemically **separated** and blended down <20 w/o U5

Fees

- HEU and TRIGA SNF: **USD 4,500** per kg tot metal (element related)
- LEU (silicide): **USD 3,750** per kg tot metal (element related)
- fees are subject to **escalation** subsequent **adaptation**
- for signed contracts **only** escalation can apply
- fees only apply for **high-income** economy countries (+ transportation)
- fees for **take over** include: receipt, management, interim storage, chemical separation or geological disposal
- **U.S.** take over the cost (incl. transport) for **all other** concerned countries

Storage Technologies

- **wet or dry** storage in existing facilities (SRS + INEL)
- **new dry storage** concepts/facilities under development
- final disposal: **repository** in Yucca Mountain, Nevada?
- Nuclear **Waste Policy Act of 1982** authorizes geological disposition
- decision on **final disposal:** under NEPA requirements

Contract

- contract conclusion between **DOE** and reactor **operator**
- **title transfer** to the U.S.: at the ports of entry (off-loading) or continental border with exceptions, if appropriate
- **Appendix A:** description of SNF
- **Appendix B:** description of transport cask
- **Appendix C:** description of SNF history

Actual Shipment

- **1st European** shipment: Aug/Sep 1996
- **231** SNF elements in 6 casks from Germany, Sweden, Switzerland
- in parallel: 1st shipment (2 casks) from **S-America** (Chile and Columbia)
- arrival at Charleston/SRS: **1996-09-22**
- further shipments from overseas: not before **Feb 1997**

Legal Process

- S.C. governor filed a **lawsuit** against DOE's NRP
- main argument: EIS is **insufficient** in respect to long term storage in SRS (L-Basin integrity, earthquake)
- **trial** date: 1996-11-25
- judge promised **decision** by 1996-12-01

REPROCESSING OF MTR FUEL AT DOUNREAY

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ABSTRACT

UKAEA at Dounreay has been reprocessing MTR fuel for over 30 years. During that time considerable experience has been gained in the reprocessing of traditional HEU alloy fuel and more recently with dispersed fuel. Latterly a reprocessing route for silicide fuel has been demonstrated. Reprocessing of the fuel results in a recycled uranium product of either high or low enrichment and a liquid waste stream which is suitable for conditioning in a stable form for disposal. A plant to provide this conditioning, the Dounreay Cementation Plant is currently undergoing active commissioning.

This paper details the plants at Dounreay involved in the reprocessing of MTR fuel and the treatment and conditioning of the liquid stream.

INTRODUCTION

The MTR reprocessing plant at Dounreay has operated for over 30 years and has just completed its 61st campaign. During that time nearly thirteen thousand elements from all over the world have been reprocessed. The plant was originally designed to reprocess fuel from the UKAEA MTR reactors at Harwell and Dounreay and from the university research reactors at East Kilbride, Manchester and Ascot. The first reprocessing of overseas fuel took place in 1962 when 25 Danish HEU elements formed part of the 17th reprocessing campaign. Since 1962 fuel from Australia, France, Germany, South Africa, Greece, Japan and Sweden has been reprocessed at Dounreay. A list of MTR elements reprocessed to date is shown in Table 1.

COUNTRY	NO. ELEMENTS	URANIUM (kg)
United Kingdom	9306	1438.3
Belgium	240	59.5
Spain	6	10.8
Denmark	950	105.6
France	289	98.1
Australia	264	32.2
India	83	14.0
Germany	918	135.4
South Africa	216	29.5
Greece	39	29.6
Sweden	168	24.8
Japan	410	82.7
TOTAL	**12889**	**2060.5**

Table 1: Elements Reprocessed 1958-1996

The reprocessing plant at Dounreay has traditionally been run on a campaign basis. A typical operating campaign is between six and nine months in duration and may involve the reprocessing of anything between 300 and 900 elements. The operating mode of the plant is dictated by the workload: for a small campaign the plant can be operated for five days a week, employing a two shift system; for larger campaigns 24 hour, 7 day a week working on a three shift system is used.

As the scope of activities carried out in the Dounreay MTR Reprocessing facility are of a standard nature it has been possible to develop, over the years, a "Standard MTR Reprocessing Contract." The main historical variance from contract to contract concerns the fuel details and transport arrangements.

Commercial Contracts

The UKAEA will help customers, to arrange for the transport of the fuel to Dounreay. Flask receipt at Dounreay is charged at a per flask rate, flask return is at the customers expense. On receipt of the flask the UKAEA will unload the flask and store the fuel pending reprocessing. Prior to reprocessing the UKAEA will, if possible, crop the customers elements to remove excess aluminium, for a fixed charge per element. The activities carried out in reprocessing the fuel include; weighing, element cropping, dissolution, uranium recovery as uranyl nitrate, accountancy and analysis. These reprocessing activities are charged at a fixed rate per kg total metal. On completion of the reprocessing the UKAEA will convert the customer product to either oxide or metal, this is charged at a standard conversion rate per kg of product, depending on the product required. Should the customer wish for his product to be at a lower enrichment, down blending can be carried out as part of the conversion and will not incur an additional charge. The product will be stored at Dounreay for up to 3 months, as part of the reprocessing price; beyond 3 months, storage will be charged at a per kg U235 rate. Also included in the reprocessing price is the storage of liquid ILW, for up to 5 years, and the cementation of the ILW. Storage of the cemented ILW will be charged at a drum rate per year. All UKAEA reprocessing contracts contain a standard clause requiring commitment to Return of Waste as required by the UK Government and as supported by inter-governmental agreements.

THE DOUNREAY MTR REPROCESSING FUEL CYCLE

Fuel Transport

Fuel is transported from the reactor sites to Dounreay under IAEA transport regulations. A number of transport routes are available, with flasks being transported by road and rail to a port, by sea to the United Kingdom and by road to the site.

Once on site the flasks are unloaded to a dedicated storage pond. The pond has the capacity to hold over a thousand elements and this enables fuel to be stockpiled before a reprocessing campaign is commenced. The storage pond is equipped with a 25 tonne crane and is designed to allow the handling of most MTR flasks.

Fuel Transfer for Breakdown

Once a campaign is commenced the elements are transferred for reprocessing using internal transport flasks. For standard elements the fuel break down is carried out in the fuel handling pond of the reprocessing plant. For larger elements a dedicated disassembly cell in one of the other facilities on the Dounreay site is used.

THE MTR REPROCESSING PLANT

The reprocessing plant contains the facilities necessary to cut up the elements, dissolve them and separate the uranium from the actinides and fission products. The process results in two waste streams, a ILW raffinate which contains over 99.9% of the fission products and a low active raffinate which results from the purification of the uranyl nitrate product.

Experience has shown that alloy and dispersed fuels of a variety of enrichments are most successfully processed by tailoring the element handling, cropping, dissolution and extraction processes to suit the different fuel types.

HEU Alloy Fuel

HEU alloy has been the traditional feed-stock of the MTR plant. The elements may be either tube or plate type but are essentially treated in the same way.

The elements are transferred to the plant and loaded into a buffer storage matrix in the fuel handling pond. Depending on the uranium content and the weight of the elements either one, two or three elements are processed as a batch.

Element Dismantling

To prepare a batch of fuel for dissolution the elements are weighed under water using an electronic balance. The elements are then fed through a hydraulic cropper which cuts them into 5 cm pieces. The pieces are then re-weighed to ensure that all of the fuel is charged to the dissolver. The element weights are recorded and corrected to show the true weight of the elements to take account of the buoyancy effect of the water. This weight provides the basis for flowsheet calculations and for charges to the customer.

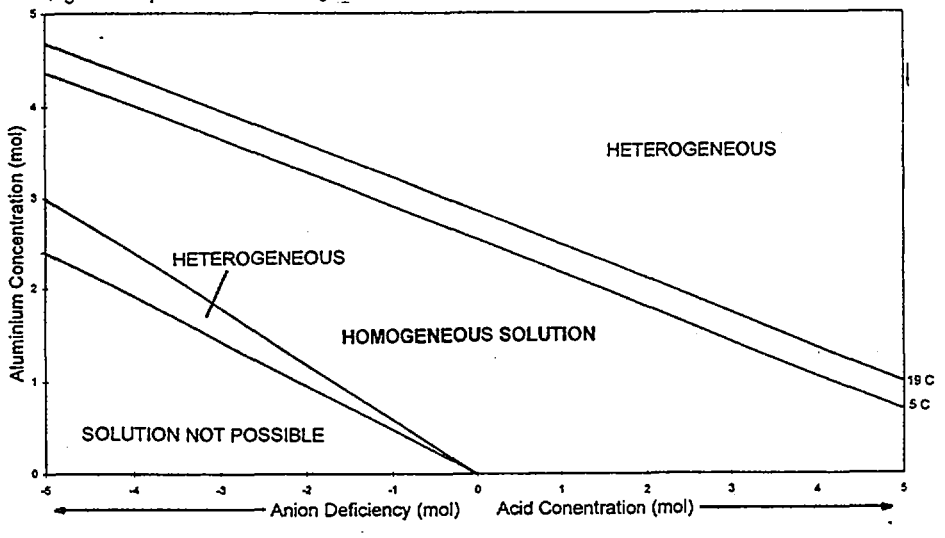
During the cropping process there is some leaching of activity from the elements into the pond water. To ensure that the radiation from the pond is minimised there is aPond Clean-Up Unit. This unit consists of two parallel filters and ion exchange columns. The filters remove any particulate from the water, whilst the ion exchange columns remove dissolved activity. The Pond Clean-Up Unit is able to reduce the pond water activity by a factor of ten on a single pass and is very successful in reducing the radiation levels in the pond water.

Dissolution

The plant contains two dissolvers each with a 160 litre capacity. The cropped pieces of fuel are charged to the dissolver using an elevator which lifts the fuel from the pond and deposits it into a charge chute, through which it enters the dissolver. Uranium dissolves readily in nitric acid, however a mercury catalyst is used to ensure that the aluminium dissolves.

The dissolution conditions are adjusted so the hydrolysis of aluminium nitrate can take place to produce an anion deficient solution which contains the $Al(NO_3)_x(OH)_{3-x}$ complex. There are a number of advantages to using an anion deficient flowsheet: the removal efficiency of fission products is extremely high and the solubility of the aluminium is improved, enabling the volume of raffinate to be reduced. The equilibrium diagram for the acid/aluminium system is shown in Figure 1. This clearly illustrates the increased solubility of aluminium on the acid deficient side.

Figure 1: Equilibirum Phase Diagram - Aluminium/Acid System



The ratio of acid to aluminium added to the dissolver is carefully controlled to ensure that the resultant solution always lies in the homogenous phase of the equilibrium system.

The dissolution reaction takes place over six hours during which time the dissolver is continually refluxed.

Solvent Extraction

The solvent extraction plant operates with two cycles, the first cycle removes over 99.9% of the fission products and the second cycle purifies the product. Mixing is achieved through the use of motor driven paddles.

In Cycle I the dissolver liquor is fed to the extractor box with a ferrous sulphamate solution and contacted with 3% tri-butylphosphate(TBP)/kerosene. The ferrous sulphamate converts any in-bred plutonium to the PuIII oxidation state which prevents it being extracted into the solvent, The loaded solvent is then fed to the strip box where it is contacted with an aluminium nitrate solution to transfer the fission products back to the aqueous phase. The uranium bearing solvent is then back-washed under low acid conditions to give a uranyl nitrate product.

The second cycle of the solvent extraction plant works in the same way as the first but with the percentage of TBP in kerosene increased to 6%.

Dispersed Fuel

Dissolution and solvent extraction operations for dispersed fuel are carried out using standard flowsheets that have been developed at Dounreay. However experience has shown that, when exposed to pond water, dissolution of Cs-137 from the fuel elements takes place, increasing pond activity levels. Alternative procedures are now adopted for the element handling of dispersed fuel to limit the dissolution of caesium. The ion exchange columns of the pond clean up unit are able to remove the dissolved activity from the water.

There is also fission gas release from dispersed fuel elements during cropping. This gas has been analysed and the results are shown in Table 2. The table shows that the gases which are released are not in themselves radioactive and the rise in pond water radiation is purely due to the dissolution of ceasium.

ELEMENT	% COMPOSITION
Krypton	0.68
Xenon	3.16
Hydrogen	19.4
Helium	75.6
Argon	1.13

Table 2: Composition of Evolved Gas from Dispersed Fuel Elements

Element Handling

Element handling procedures are adjusted for dispersed fuel to minimise the time the fuel is exposed to pond water. Normal handling involves weighing the elements for the batch, cropping the batch and re-weighing the pieces. The batch is then charged to the dissolver.

For dispersed fuel the elements are cropped and the pieces charged immediately. This ensures that the time that the fuel is in contact with the water is minimised and the increase in pond water activity is limited.

Silicide Fuel

A considerable amount of research into the reprocessing of silicide fuel has been undertaken by the UKAEA at Harwell. This has indicated that finely divided silica remains as a solid after dissolution and that polysilicic acid is present in solution. Both these factors have the potential to interfere with the solvent extraction process and the presence of finely divided solid material can affect the operation of valves and transfer systems. In addition silicide fuel contains far more plutonium than HEU fuel due to the material in-bred in from the U238.

At Dounreay we have recently undertaken the reprocessing of silicide fuel in the MTR reprocessing plant. The elements reprocessed originated from the RERTR programme to develop silicide fuels in the Netherlands.

The reprocessing of these elements has proved that full scale reprocessing of silicide fuel is possible although the plant would require some modification before a large reprocessing campaign of silicide fuel could be commenced, to enable the removal of siliceous materials prior to solvent extraction.

For the purpose of this trial the dissolution and solvent extraction procedures were adjusted to allow the silicide fuel to be processed through the plant.

Dissolution

Research has shown that the effects of silicic acid on solvent extraction are increased for an anion deficient flowsheet. Dissolution of silicide fuel therefore takes place on an acid flowsheet to give a solution which contains $Al(NO_3)_3$ and an excess of HNO_3. This type of flowsheet has been used frequently in the past in the MTR reprocessing plant and is used as a standard flowsheet in other reprocessing plants. The standard dissolution time of 6 hours is used and sparges are operated until the liquor is transferred from the dissolver to the next process vessel to minimise the settling of silica.

Solvent Extraction

To minimise the effect of silicic acid on the solvent flowsheet the flowsheet is modified. The feed to the plant is diluted by the co-feeding of three molar acid to the extraction box. The feed rate of ferrous sulphamate is also doubled to ensure that all the plutonium in-bred from the U-238 is converted to the Pu III form and rejected in the raffinate.

The Optimised Flowsheet

The flowsheet for silicide reprocessing underwent development during the recent campaign. The optimised flowsheet resulted in a uranium extraction efficiency of 98%, a plutonium content in the product of less than 10 parts per million and a similar raffinate volume per kg of uranium processed as that produced by the reprocessing of HEU alloy fuel.

THE PRODUCT

The solvent extraction process produces a very pure product. The specification for a typical product is shown in Table 3.

1	Boron Equivalence	< 2.5 ppm
2	Uranium Isotopics	as determined
3	Total Impurities	less than 1500 ppm
4	Fission Product Content	< 3.7 E4 Bq/g
5	Plutonium	< 250 Bq/g

Table 3: Typical MTR Uranium Product after Reprocessing

The enrichment of the fuel is maintained throughout reprocessing and the uranium can be re-used in several ways. Uranium as recovered or at a slightly lower enrichment can permit certain reactors to achieve continuity of fuel supply until low enriched fuels of sufficient density are available. For reactors already converted to LEU, facilities are available at Dounreay for down blending recovered uranium to the desired enrichment and to produce uranium in the required form.

WASTE

There are two liquid waste types produced by the solvent extraction plant.

The waste stream produced from Cycle I raffinate contains over 99.9% of the fission products and is held as a liquid in underground storage tanks for at least five years. It is then transferred to the Dounreay Cementation Plant for immobilisation.

A low active waste stream is produced from cycle II raffinate which contains very small quantities of fission products. This waste stream is sampled and is discharged to sea with other plant effluent the site discharge authorisation.

Environmental Discharges

Radioactive discharges to the environment from the reprocessing of irradiated fuel at Dounreay are controlled under an authorisation issued by the UK Government's Scottish Office. In assessment of the permitted discharge from the site, the Scottish Office considers the effect of individual radioactive species on the environment and on critical groups of individuals. The limits that are indicated from this study are then reduced to give a site discharge limit which must be adhered to.

The discharges resulting from the 1996 reprocessing campaign were:

	Alpha	Beta
Liquid Discharges:	1.2 E9 Bq	4.8 E10 Bq
Gaseous Discharges	7.7 E3 Bq	2.7 E6 Bq

These represent a maximum of 1% of the site discharge limits in terms of alpha and beta activity.

ILW WASTE TREATMENT IN THE DOUNREAY CEMENTATION PLANT (DCP)

The Dounreay Cementation plant is currently undergoing active commissioning. The plant is designed to accept MTR raffinate after five years storage and to immobilise it in a cement matrix for disposal.

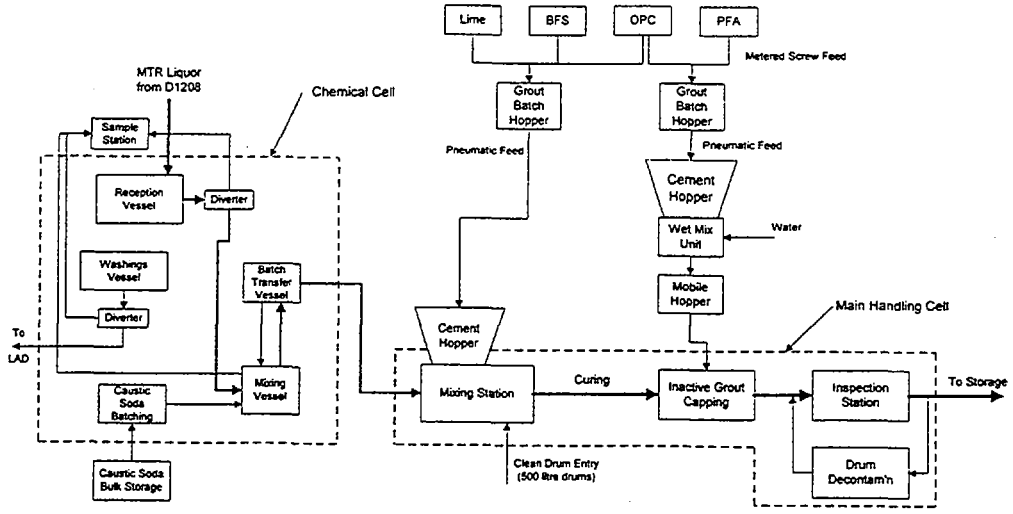


Figure 2 Schematic Diagram of DCP Process

The Cementation process essentially consists of the receipt of MTR raffinate from the underground storage tanks via a dedicated transfer line into a reception vessel. There it is sampled and analysed to determine the volume of sodium hydroxide required to neutralise the solution.

The neutralised liquor is mixed in precise drum volume batches with blast furnace slag, Portland Cement and hydrated lime in a 500 litre drum and allowed to cure for a minimum of 24 hours.

The drum is progressively transferred through the cementation plant until it arrives at the grout capping station. The remaining drum ullage is filled with cement and the drum lid locked. The drum is checked for external contamination before being transferred for storage.

An interim drum store is available at Dounreay for the storage of the cemented waste before transport to the final repository. For UK fuel this will be the NIREX repository and for overseas fuel it will be a repository in the country of origin.

Figure 2 shows a schematic diagram of the Dounreay Cementation Plant.

RADIATION DOSE UPTAKE TO THE WORKFORCE

The monitoring of dose uptake to the workforce is an essential part of the management of fuel reprocessing. Dose restraint objectives are set at the beginning of a reprocessing campaign based on a detailed dose budget. Dose is monitored on a daily basis and is controlled against the dose restraint objectives.

For the reprocessing campaign that has just been completed the total dose to the twelve operators was 42.86 mSv. This is equivalent to an average operator dose of 3.57 mSv and a dose per element reprocessed of 0.2 mSv. This is well within the corporate dose limit of 15 mSv.

Continual improvements are made to the plant and to operating procedures to minimise the dose uptake to personnel.

SAFEGUARDS

Operation of the reprocessing plant at Dounreay is subject to constant surveillance by Euratom Inspectors. Euratom spend approximately 500 mandays on site and are always on hand to witness and record material transfers and sampling procedures.

CONCLUSION

Dounreay has traditionally reprocessed HEU alloy, however, in recent years, the MTR plant has diversified and reprocessed a significant quantity of dispersed fuel. The reprocessing of the two silicide elements in the recent campaign shows a clear commitments to a future for the plant which includes the reprocessing of high density LEU silicide fuel.

CURRENT ACTIVITIES ON IMPROVING STORAGE CONDITIONS OF THE RESEARCH REACTOR "RA" SPENT FUEL

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CH04A0030

1. Introduction

After 25 years of successful operation, the 6.5/10 MW thermal heavy water moderated and cooled research reactor RA at the VINČA Institute was shut down for refurbishment in 1984. Since for a number of reasons this refurbishment has not yet been completed, and having in mind the long shut-down period, the future status of the RA reactor is presently a subject of serious reconsideration.

Closely related to the problem of future use, or eventual decommissioning, of the research reactor RA is the problem of safe and reliable storage of fuel irradiated so far, as well as disposal of the new irradiated fuel if and when the reactor is re-started.

Basic facts about operation, ageing, reconstruction and spent fuel storage of the research reactor RA have been presented and discussed in detail in some earlier papers [1-3]. The following paragraphs describe current activities on improving storage conditions of the research reactor RA spent fuel. Recent results are presented concerning identification and minimization of corrosion processes. Options for permanent resolving of the spent fuel storage problem are discussed and a possible scenario for building a new long term research reactor RA spent fuel storage is proposed.

2. The Present Status of the Spent Fuel Storage Pool at the Research Reactor RA

Research reactor RA fuel element (slug) is an 11.3 cm long cylinder, with 3.72 cm of outer diameter, consisting of an outer tube with 2 mm thick fissionable material and 1 mm thick inner and outer Al cladding, and an 1 mm thick inner Al tube which serves as cooling intensifier. Fuel elements are inserted into a 2 mm thick Al tube (10 or 11 slugs/tube), thus forming a fuel channel. The reactor core consists of up to 82 channels in a square lattice with 13 cm pitch. Until 1976 the reactor was operated with 2% enriched uranium metal fuel, when new fuel with 80% enriched uranium oxide dispersed in aluminium was bought from USSR.

The six meters deep temporary spent fuel storage pool, consists of four connected basins, having thick concrete walls cladded with stainless steel. It is filled with approximately 200 m^3 of stagnant ordinary water. 304 channel-type stainless steel fuel containers, receiving up to 18 spent fuel elements each, are placed vertically in the pool, Fig. 1. Initially, it was planned to transfer spent fuel back to the Soviet supplier, after 4-5 years of cooling in the temporary storage pool. Since this did not happen, in order to increase the storage capacity, the oldest spent fuel was gradually taken out of the original stainless steel containers and repacked into aluminium barrels, each containing 30 aluminium tubes receiving up to 6 irradiated fuel elements per tube. At present, there are 30 such barrels, placed in two layers in the annex of the basin 4, Fig. 2. Cadmium strips were inserted into the barrels to assure necessary subcriticality. Both the barrels and the channel-type fuel holders were filled with demineralized water and hermetically closed.

According to the original design, the RA reactor spent fuel storage had no system for pool water purification. A simple system with a mechanical filter and an ion-exchange column, purchased just several years ago, is neither adequate, nor operable. Monitoring and maintaining of pool water radiochemical parameters were not imposed, and were even considered unnecessary by the operating stuff since the pool water was not supposed to be in direct contact with the spent fuel. For many years nothing has been done to prevent or slow down corrosion of the spent fuel containers, nor to examine the state of the spent fuel inside the containers.

Due to the obvious lack of concern for the conditions of the reactor RA spent fuel storage, it is now in a very bad condition. Water in the pool is dirty and its chemical parameters are such that they foster rather than inhibit corrosion. At the bottom of the pool there is a lot of sludge which had even concealed a lost spent fuel slug since 1970. Simple visual inspection shows that all steel construction elements have heavily corroded, Fig. 3. Corrosion can also be noticed on stainless steel walls of the basins and transport channels, especially at welded parts, Fig. 4.

A serious concern about the conditions of the spent fuel storage pool was first expressed by the author of the present paper in 1994 [1]. A possibility of fuel leakage inside the containers and eventual

fission products release from the containers, was pointed out. Since the VINČA Institute had no expertise nor the material resources for solving the problem of safe long term disposal of irradiated nuclear fuel, an international help and support were asked for through IAEA. Actions taken so far and plans for future activities are presented in the following paragraphs.

3. Activities on Identifying and Improving Conditions in the RA Reactor Spent Fuel Storage Pool

Radiological and chemical analyses of the water from the RA reactor spent fuel storage pool were first made during the visit of IAEA fact finding mission to the VINČA Institute [4]. Radioactivity of the pool water (35.2 to 41.8 Bq/ml ^{137}Cs) was then ascribed to the presumable leaking of the lost fuel element. Repeated analyses of water samples taken at the same points as about a year ago now show an increase of pool water radioactivity approximately by a factor of 3 (93 to 140 Bq/ml ^{137}Cs).

During last year, water samples were also taken on several occasions from channel type stainless steel spent fuel containers selected so as to represent different typical storage conditions: channels containing normally disposed spent fuel from different periods (1969-1979), channels containing fuel with corrosion deposits noticed at the time of disposal, channels containing fuel whose corrosion deposits were removed before the disposal, as well as a channel containing one fuel slug which was leaking already at the time of disposal (1970). Increased specific activity of these samples, ranging from about ten to several hundreds Bq/ml ^{137}Cs in most of the channels, reaching up to several hundred thousand Bq/ml ^{137}Cs in some channels with normally disposed fuel, and even up to two orders of magnitude more in the channel containing the fuel slug which was leaking already at the time of disposal, is a clear indication that at least one fuel element is leaking in most of the channel type spent fuel containers. It is assumed that the situation in aluminium barrels can only be worse.

Strings of spent fuel elements were taken out from some channel-type containers and brought to the hot room in the reactor shield to be visually inspected through the protective glass window. Thick corrosion deposits can be notices on the Al cladding of all fuel slugs, Fig. 5. The lost fuel slug has recently been located and taken out from the bottom of the pool. Visual inspection through the same protective glass window shows that it is heavily corroded, Fig. 6. Yet, in spite of the fact that this fuel element has spent almost twenty years in the corrosively very aggressive sludge at the bottom of the pool, the uranium metal has not been uncovered, which mostly removed the concern expressed earlier by IAEA experts [4] that uranium hydride had been formed in such an amount that it could cause fire if the spent fuel is exposed to air.

Aluminium barrels probably represent the main radiological and safety problem in the RA reactor spent fuel storage pool. The analyses and inspections described above indicate that massive leakage of spent fuel must be taking place in the barrels. On the other hand, oxidation of a large surface of aluminium in contact with water (about 14 m^2) inside hermetically closed barrels could result in high internal hydrogen pressure. A sudden release of this pressure could spread fission products in the pool water and its surrounding and cause unpermitted irradiation of personnel.

Action is therefore initiated in two directions. First, to improve the safety of the existing spent fuel storage. Second, to consider transferring spent fuel into another, presumably dry storage space.

4. A New Project for Safe Disposal of Research Reactor RA Spent Fuel

A new project has recently been formulated, engaging several laboratories of the VINČA Institute (reactor physics, chemistry, radiation protection, nuclear engineering, material sciences), in order to solve the problem of safe disposal of research reactor RA spent fuel. The following management plan for implementation of this project is foreseen.

4.1. Phase 1. Improving conditions in the existing reactor RA spent fuel storage pool

Measuring the pressure inside the aluminium barrels, controlled release of possible overpressure and establishing regular monitoring and control of this pressure will remove direct danger of a radiological accident, caused by a sudden release of a larger amount of gas carrying the long lived fission products from the leaking fuel. These complicated operations can only be performed with the technical assistance of the IAEA.

In order to minimize further corrosion and preserve integrity of the stainless steel channels and aluminium barrels, which now obviously contain leaking spent fuel, the following actions are planned: removing of the sludge from the bottom of the pool (estimated amount 3 m^3) and disposing of this sludge in the cemented form at the VINČA low and intermediate waste disposal site; improving the quality of the water in the pool; isolating the failed lost fuel slug from the pool water and regular

monitoring of the pool water activity in order to identify eventual leakage of the spent fuel containers; starting of the new system for pool water purification; establishing regular monitoring and control of the pool water chemical parameters (pH and conductivity). Minimum duration of these activities is about six month.

4.2. Phase 2. Transfer of the RA reactor spent fuel into another storage space

Prompt actions to minimize further corrosion processes in the existing temporary RA reactor spent fuel storage are mandatory for improving its safety, in the sense that the probability of uncontrolled fission products release into the reactor building, and eventually to the environment is reduced to the minimum possible value. However, the only long-term solution to the interim storage of aluminium cladded spent fuel should be to transfer it to a dry storage facility.

Since transfer of spent fuel back to the supplier is not likely to occur, construction of an independent facility for spent fuel storage is to be considered. The following activities will be necessary for realization of this project: decision-making about the future status of the research reactor RA; decision-making about the most adequate solution for safe disposal of the fuel irradiated so far; producing project documentation and building of a new dry storage facility with the assistance of domestic construction organizations; transfer of the irradiated research reactor RA fuel from the existing temporary storage pool to the newly built longer-term spent fuel storage space in special containers designed for the purpose; adopting regulations, procedures and an adequate organization for long-term supervision, monitoring and control of the newly built facility. Duration of the above activities is estimated to be 2-3 years.

Factors that may cause delays or prevent implementation of the proposed project are lack of expertise and know-how in the field of safe disposal of irradiated nuclear fuel, lack of necessary equipment and material resources, as well as general economic difficulties in the country. International cooperation and support, presumably through the IAEA, or on multilateral and/or bilateral basis, would be highly appreciated and desired.

5. Conclusion

The present status of the research reactor RA spent fuel storage pool at the VINČA Institute presents a serious safety problem. Activities have therefore been initiated to improve conditions in the temporary storage pool in order to minimize further corrosion of the spent fuel containers, as well as to enable transfer of spent fuel into another dry storage space.

Immediate objective these activities are expected to achieve is increased safety of the research reactor facility in the sense that the probability of uncontrolled fission products release will be reduced to the minimum possible value. By storing the previously irradiated fuel of the research reactor RA in a newly built storage space, sufficient free space will be provided in the existing spent fuel storage pool for the newly irradiated fuel when the reactor starts operation again. In the case that it is decided to decommission the research reactor RA, the newly built storage space will provide safe disposal of the fuel irradiated so far.

Solving the problem of safe disposal of research reactor irradiated fuel is a difficult task for a country with no long term nuclear power program, and with limited potentials and resources. International cooperation through the IAEA, or on multilateral or bilateral basis, would be highly recommended and appreciated.

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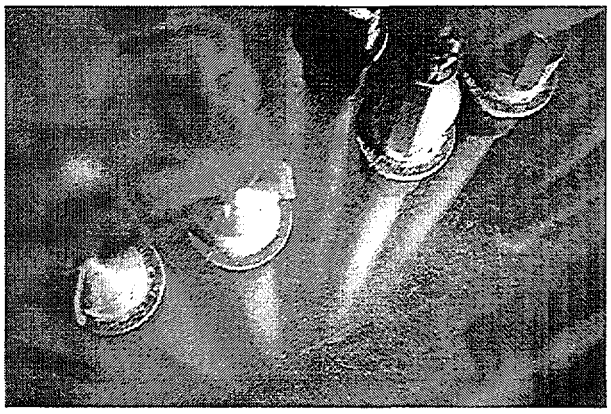

Fig.1 Channel-type spent fuel containers

Fig.2 Barrels with repacked spent fuel

Fig.3 Construction element in the storage pool

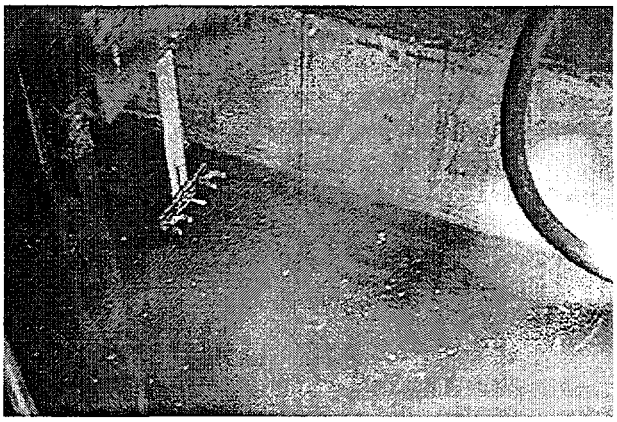

Fig.4 Wall of the storage pool

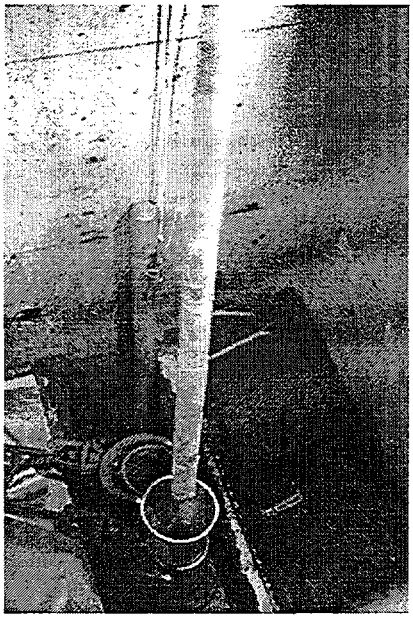

Fig.5 A string of spent fuel elements from a channel-type container

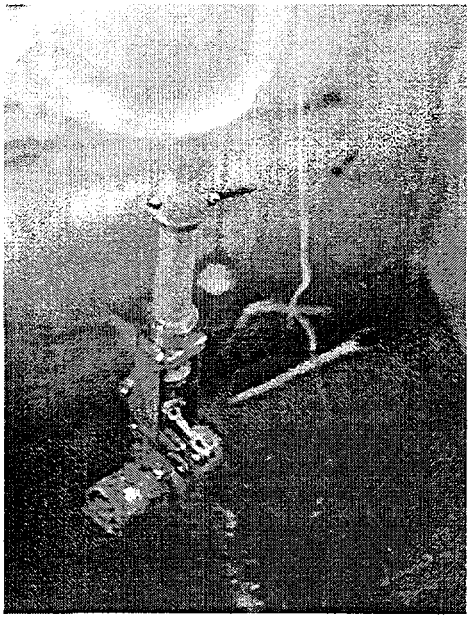

Fig.6 The lost fuel slug taken from the bottom of the pool

Dry Vault For Spent Fuel Depository
Basic Outsets, Operating results and Safety of the CASCAD plant

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I. Introduction

Reprocessing and recycling of fissile materials is the prefered approach to spent fuel management in France. However, a number of spent fuel elements from prototype and experimental nuclear reactors cannot be reprocessed in the existing industrial facilities, either because such facilities are booked to full capacity, or due to technical factors such as non standard nature of fuel or limited series of fuel.

The CEA therefore built a facility in which spent fuel can be stored for a few decades (50 years), until favourable conditions prevail for its disposal. The main features of this project consist in a dry depositary , which presents a low cost of working, against a wet one which is more expensive due to the circulation and the continuous controls of the water. Therefore, this is a fair solution because the experimental fuels will present a rather low residual heat power after decay in the nuclear reactor. At this stage, it becomes possible to cool the fuel elements by a fully passive air circulation. This process allows a good efficiency without mechanical equipement and works all the better as the amount of heat to exhaust is great, in the limits of the design. However, we will see that this concept may be extended to a depository of standard spent fuel elements.

This facility, known as "CASCAD" (shortening for CASemate (=vault) CADarache) started up in 1990, and received its first canister of fuel on May 29th 1990.

This paper reviews the basic design data of the facility, outlines the main techniques used for its construction, draws the safety concepts and presents the first results determined by a looking-back over 4 years of working.

II. Principles and Criteria for Basic Design

The flexibility to deposit a whole range of fuel types with various characteristics is a key feature of the "CASCAD" design. For example, the storage facility enables the fuel assembly to be retrieved during interim depository for reprocessing or final geological disposal :

- containement by tight-canisters (to accept failed cladding too),
- stainless steel wells with tightness and non-contamination controls,
- a windowless storage cell made in concrete for biological shielding which involves 315 storage wells,

- a handling hall with biological shielding in concrete and viewing through a lead-glass window and through a video camera. When any nuclear fuel element is in opération, the staff may go into this cell which is always clean in regard to an eventual radionuclids contamination.
- cask or canisters check-in and inspection areas.

The building in which the facility is housed is 35 meter long, 25 meter wide and 16 meter high. It is a concrete structure erected on a ground slab, 2 meters thick, extending 2 meters beyond the south and north walls. The main part of the building structure consists of the storage vault with wall thickness of 1.25 meter; Stiffness is provided by transverse and axial webs of rod-reinforced concrete. The concrete slab between the vault and hall hold up the entire load of the wells including fuel elements and is itself supported by the webs, since the need to provide free air circulation precludes the use of supporting columns. Then, this slab is subjected to heavy stress due to loading by wells and to the temperature prevailing in the vault.

The "CASCAD" building is designed to receive (at present time), to send (in the future), to handle and to prepare casks up to 400 kN prior or next to the transfert of the canisters to or from the handling hall. The system comprises a wheeled carriage on rail tracks and supports for securing the casks during the motion. Chek-in and check-out for contamination and damaged fuel detection are realized in a special room equipped with a nuclear ventilation to get dynamic containment.

The main item of equipment for canisters transfer is the 50 kN overhead crane installed in the handling hall. This crane is equipped with a multi-purpose handling head which can carry a number of grip modules. Its electrical supply is controlled automatically from the front zone and this programmable unit drives the crane over the wells. The final step of the handling process, the gentle driving down inside the pre-determined well , is conducted in a manual mode with the help of a video camera.

Other items are available in the handling hall such as 2 remote handlers or a 50 kN hook.

The storage vault and the wells (7 meter useful length) are cooled by natural open-convection : a cold air inlet is provided at the bottom part of the vault, the air heated by contact with the wells is discharged through a 42 meter high stack outside the building. A general thermosiphon effect is created, which extracts the heat released. This ventilation system is designed for various conditions, namely :

- thermal loading,
- prevailing winds,
- atmospheric pressure.

Each well accepts 600 Watts maximum in thermal power, and consequently, the maximum heat loading of the vault is about 200 kW. In this case, the flow rate of air may reach 19,000 cubic meter per hour.

III. Safety Concepts

The integrity of the canister, first barrier, is monitored by sampling regularly the internal atmosphere of the wells.

If the contamination control is positive, it is possible to pick up the canister for examination and send it in a specialized facility located nearby "CASCAD" for substitution. If the well is not contaminated, it may be used again. On the contrary, the well is closed waiting for a decontamination operation.

The stack of air cooling extraction is equipped with a continuous air control device. If the air control is positive, a valve avoids a radioactive pollution risk in shutting the stack. In this case, a mechanical ventilation insures the exhaust of the heat through HEPA filter battery. The canisters are picked up from the well and are put in one of the four safe-keeping wells. Then, the leaky well is shut for good.

Concrete biological shielding takes account of fuel elements irradiated at 45,000 MWd per ton and having previously cooled off for five years. The requirements for maximum dose-rate at work station settle the value at 25 micro-Sv/hour.

For these same fuel elements, criticality CEA codes led to a depository tube layout in the form of a triangular pattern with 800 mm spacing. The absence of distortion in the facility stuctures, in the wells and for the canisters, even in case of an earthquake, was demonstrated by calculations using approved codes. About this topic, the facility and its safety-related equipments have been calculated through a dynamic structural analysis based on acceleration diagrams selected as the most representative following a survey of local earthquake history. Such acceleration diagrams are in category 9 of the MSK scale which goes from 1 to 12, for a maximum historically-probable earthquake of intensity 8 on the same scale.

During dry storage, the température of the concrete slab between the vault and hall must be limited to 100 °C for reasons of mechanical strain. Air ducts for natural convection cooling were designed using models and codes developed by the CEA, based on local weather data and on possible disturbances in the close environment. Nevertheless, before deciding to build the "CASCAD" facility, CEA considered as necessary to confirm the calculation by experimentation. Analysis covering heat transfer around a storage well, and the overall heat dissipation process were respectively validated using a full scale model of a complete well and a 1/12 scale model of the vault and cooling system.

Now, a large set of measurement points survey, continuously, the variations of the temperature of slabs, webs and wells.

The 50 kN crane and its auxiliary equipments in the handling cell present a high reliability. In the event of failure of one of the geared motors or their power supplies, the system is so designed that the operation under way can be ended. To that end, axial and transverse drives are equipped with double and separate motor systems and supplies. Even in case of an earthquake, intensity 9, the crane remains available.

Each well owns a shock absorber intended to minimize the consequences of the dropping of a canister during handling. Full scale tests were done by dropping a 350 kg weight into the well.

IV. Operating Results

Today, the "CASCAD" building contains Brennilis EL4 fuels (89 wells) and navy spent fuels coming from French nuclear submarines. The Brennilis nuclear reactor fuel was stored in the plant pools as it was unloaded from 1971 through 1985. It consists in 5300 "clusters" of low-enrichment uranium oxyde rods with zirconium cladding.

The loading of the storage vault reaches 50 Tons shared out on 134 wells, 43% of the total physical capacity.

The natural open-convection exhausts 53 kW, 27% of the maximum heat loading. In regard of this ratio, the flow rate and temperatures recorded in the vault in summer and winter are the following :

DATE	*External temperature (°C)*	*Exhaust temperature (°C)*	*Flow rate (m3/h air)*
1993, July 07 (15 h)	**27**	**28**	**1500**
1994, March 15 (06 h)	**5**	**15**	**7000**

This cooling system demonstrated its ability to ensure low temperatures and, moreover it works with a very important thermal stability. This last point is a real positive effect for the safety study.

During this loading period, no major difficulty occured concerning the good development of the handling flow sheet. However, many little anomalies appeared with the 50 kN crane, especially with the remote control-command. Each time, the manual proceeding allowed to end the motions.

No contamination (< 37 $Bq.cm^{-2}$ in bêta-gamma and < 3.7 $Bq.cm^{-2}$ in alpha) or excess irradiation rate has been detected on the surface of the canisters or inside the casks. The dose-rate measured around the external walls of the vault, inside the building, is included between 5.10^{-4} and 1.10^{-3} $mSv.h^{-1}$. Outside the building, this rate decreases to less than 10^{-5} $mSv.h^{-1}$.

The dose-rate for the entire staff (9 persons) is under 20 mSv per year. Per agent, the maximum was 5.25 mSv and the minimum 0.35 mSv.

V Conclusion

The concept of dry storage in a vault cooled by natural convection is a satisfactory answer to the problem represented by the depository of spent fuel from prototype nuclear reactors for a period of several decades. The depositing of EL4 fuel in the "CASCAD" facility is now proceeded. It provides the CEA group with practical experience of the technical and financial aspects of such undertakings:

- safety,
- passive cooling,
- simplicity,
- cost saving working.

The flexibility to store a whole range of fuel types with various burn-up levels is a key feature in the "CASCAD" design. A lot of studies are underway to demonstrate the feasability of an extension of the "CASCAD" concept to PWR fuel elements. For example, the characteristics of a depositary module for PWR fuels previously stored during 5 years in the pool of the reactor are the following :

- capacity : 1000 fuels assemblies,
- maximum power released : 1 MW,
- maximum temperature on the clads : about 340 °C.

Today most countries are faced with the future of their spent fuels. Long term disposal at the end of which the choice between reprocessing or storage remains open is therefore the natural issue. A concept like "CASCAD" should produce one of the solutions.

Dry storage experiment with spent materials testing reactor fuel elements

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1. Introduction

Irradiated fuel elements from German research reactors have been in the past reprocessed in DOE and UKAEA facilities, respectively. Since this was discontinued in 1988 due to availability reasons, and because no national reprocessing capacity will be installed in Germany, an alternative concept for the management of these fuels was developed including dry intermediate storage on surface followed by permanent disposal in the deep geologic underground. R & D work to support this concept covering both intermediate storage and disposal was initiated in the Forschungszentrum Jülich (KFA-Research Centre Juelich). The results of the dry storage experiment are reported hereafter.

2. Dry storage experiment

2.1 Objectives

For transport and intermediate dry storage of spent research reactor fuel the german GNB company has designed the CASTOR MTR-2 cask, the licensing of which is under way. The cask has a body made from cast iron, an inner nickel plating and double metallic sealing, which can be tested for thightness. It has a mass of 16 metric tons and is designed to hold e.g. 33 box-shaped MTR fuel assemblies.

A literature review was made about the behaviour of research reactor fuel elements at storage conditons /1/. Positive experiences with the wet storage of these elements for periods of up to 30 years as well as with the dry handling or storage for short periods at ambient temperature were reported therein.

For long-term intermediate storage purposes the casks will be vacuum dried (less than 10 grams water after drying) and refilled with helium. These conditions will limit the corrosion attack on the Al cladding substantially and hence prevent a loss of mechanical integrity of the fuel elements for sure.

On the other hand a possible release of airborne radionuclides into the cask's atmosphere at elevated temperature has to be taken into account. Based upon Fick's law of diffusion, the release of gaseous (H-3, Kr-85) and metallic fission products (Cs, Sr) through the cladding was estimated to be insignificantly low at temperatures up to 300°C, which is far below the blistering temperature. Even for the most mobile radionuclide, tritium, an integral release of not more than 35 kBq per year from one MTR fuel assembly was assessed /2/.

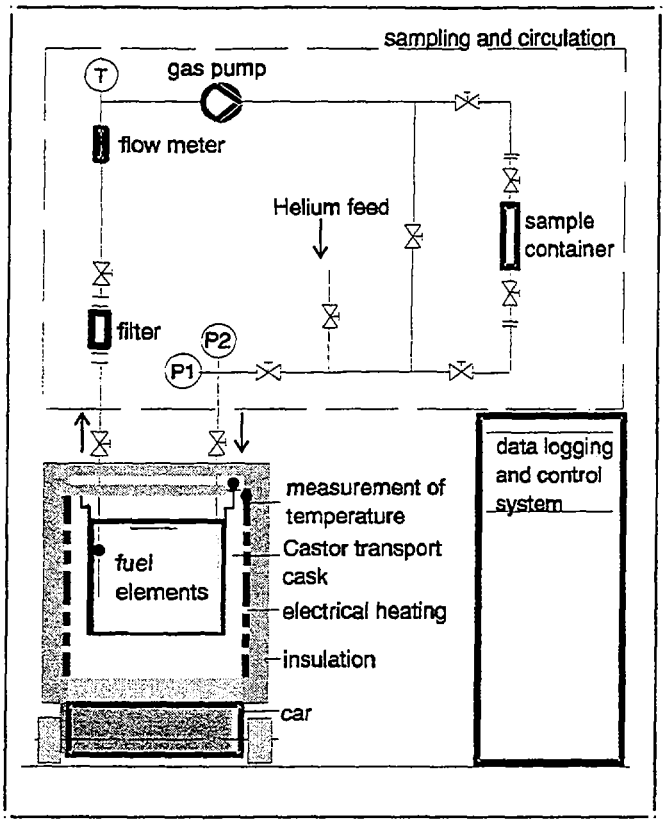

Fig. 1:Dry storage experiment

To support this evaluation, an integral storage experiment was initiated at the KFA, the principle of which is outlined in figure 1.

2.2 Experimental setup

A CASTOR-MTR transport cask, which has a design similar to that foreseen for intermediate storage, is equipped with an outer electrical heating and covered with a thermal insulation. The cask is connected to a gas circuit in order to provide homogenisation of the gas atmosphere as well as sampling and analysing. Tests showed a leakage rate of better than 10^{-6} $cm^3 \cdot s^{-1}$ (NTP) for the entire experimental setup.

Gas samples were taken periodically and analyzed by gas- and radio gas-chromatography, thus showing the chemical composition of the gas as well as ist concentration in H-3, C-14 and Kr-85.

In addition, the concentration of Kr-85 was continuously measured by γ-spectroscopy in the gas circuit. Aerosol-bound radioactivity was filtered from the gas circuit, the filter being periodically checked for e.g. I-129 by γ-spectroscopy.

Moreover, the integral release of H-3 and C-14 was measured by leading the gas through several wash bottles, thus scrubbing the radioactivity from gas circuit. Oxidizable species (HT, CO, CH_3) were oxidized on a copper-oxide bed. The wash solutions were processed and analyzed by β/γ-spectroscopy.

Table 1: Averaged nuclear data of the 15 DIDO fuel assemblies

Before irradiation	
enrichment (U-235)	80 %
U-235 inventory per assembly	157.5 g
Post irradiation [1]	
enrichment (U-235)	60 %
burn-up	50 %
U-235 inventory per assembly	78.4 g
total activity per assembly [2]	90.6 TBq
decay heat per assembly [2]	11.7 W

[1] Discharge from reactor: June to October, 1990
[2] Figures as of June 30, 1994

The fuel under investigation is MTR fuel of the DIDO type, which is being used in the Juelich FR-2 reactor. Table 1 summarizes some details. The fuel contains, on average, 20 wt % U-Al alloy with an initial enrichment of 80 % in U-235. The fuel alloy plates with a thickness of 0.6 mm (meat) are covered on both sides with aluminium claddings of 0.38 mm, the total thickness is then 1.36 mm. Three plates each are welded to form a tube. A fuel element assembly consists of four concentric tubes of approximately 6, 7, 8 and 9 cm diameter, mounted onto structural parts. After cutting off these structural parts, it is about 63 cm long.

On January 31, 1995 fifteen of these spent fuel element assemblies were loaded under water into the CASTOR-MTR transport cask. The cask was covered, the water sucked off, the casks atmos-

phere vacuum dried and then refilled with helium. Gas samples taken during vacuum drying revealed radioactivity levels in general accordance with the contamination of the spent fuel storage pool water.

During the 1.5 year measuring programme, which was running from early May 1995 until October 1996, the temperature at the outer shell of the cask was raised in steps up to 180°C. According to a thermal analysis, the temperature of the fuel element assemblies inside will then be 200°C at the maximum. This is a conservative limit covering all normal and accident conditions during transportation and storage of spent research reactor fuel in a CASTOR-MTR 2 cask, even for short-cooled fuel with high burnup producing a thermal power of 825 W per cask.

3. Results and conclusions

Any airborne particulate activity possibly contained in the gas plenum of the container was filtered off when gas samples were taken. The filter elements were carefully checked for β/γ-active aerosols by γ-spectrometry. None of the filter elements contained any detectable β/γ-activities.

The gas samples periodically taken from the plenum of the container were analysed by gas-chromatography. The gas atmosphere of the container was found to contain more than 90% of inert He, a few percent of N_2 and spurious amounts of H_2, O_2, CO_2 and water vapour.

The gas samples taken from the container were analysed by special radiometric and radiochemical techniques as well as by radio gas-chromatography. Moreover, the samples were checked for any Kr-85 by γ-spectrometry.

In the water vapour pumped off from the container by vacuum drying (20 °C, 1 day, pressure < 1 hPa) only very small amounts of tritiated water and C-14 were found. During the 1.5 years dry storage experiment at temperatures ranging from room temperature to 180 °C some H-3 and C-14 was released from the fuel elements into the gas plenum of the container. The release of both radioactive species seems to rapidly increase when temperature exceeds 160 °C. At temperatures below 180 °C no release of Kr-85 was observed. Table 2 shows a compilation of the gaseous activities contained in the gas plenum of the dry storage cask.

Table 2: Accumulated activities of gaseous radionuclides in the gas plenum of the dry storage cask

experiment no.	temperature [°C]	time of storage [days]	total time of storage [days]	activity in gas plenum [MBq]		
				H-3[1]	**C-14[1]**	**Kr-85**
drying by vacuum	20	-	0	0.13	0.001	-
1	22-28	110	110	0.03	0.09	-
2	90	107	217	2.4	0.06	-
3	130	110	327	1.2	0.25	-
4	160	118	445	1.4	1.1	-
5	180	118	563	19	3.1	1.9

[1] H-3 was found to exist predominantly as tritiated hydrogen, C-14 as $^{14}CO_2$

These results can be interpreted as follows:
The Juelich FRJ-2 is a heavy water reactor whose cooling water is highly contaminated with tritiated water. During reactor operation, the aluminium cladding may chemically bind some of the tritiated water in the outer passivation layer of the fuel elements as $Al_2O_3 * H_2O$ or $Al_2O_3 * 3\ H_2O$. Under dry storage condition at elevated temperatures part of the thus chemically bound tritium may have been released to the gas atmosphere.

C-14 is formed during reactor operation mainly by (n,γ)-reactions with nitrogen impurities in the cooling water and the fuel elements. At elevated temperatures some of the C-14 contained in the outer corrosion layer of the fuel elements may have been released to the gas plenum. At temperatures up to 160 °C, no release of the fission noble gas Kr-85 was observed. The aluminium cladding of the fuel elements must thus have been absolutely gas-tight at temperatures below 180 °C. During the storage test at 180 °C a very small amount of Kr-85 was observed. It can thus not be excluded that some extremely small cracks in the cladding material have opened during the storage test at 180 °C. As an alternative, Kr-85 or blisters may also be produced by fission of uranium impurities of the cladding material.

According to the German radiation protection regulations the maximum permissible activity concentration of air in a controlled area amounts to $8 \cdot 10^9$ Bq/m^3, $3 \cdot 10^6$ Bq/m^3 and $3 \cdot 10^6$ Bq/m^3 for H-3, C-14 and Kr-85, respectively. The gaseous activities found in the dry storage container are extremely small as compared to these maximum concentrations of the breathing air. Radiation hazards should thus not be important, when dry storage containers with fuel elements from the Jülich FRJ-2 reactor have to be opened.

In conclusion, it can be stated that spent fuel element assemblies from the Juelich DIDO reactor can be safely stored in a dry cask with neither impairing their structural integrity, nor imposing an undue risk to the operating personnel and to the general public. Whether this result can be extended to fuel elements from other research reactor types with different burnup history will have to be discussed.

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Behavior of spent aluminum cladded metallic uranium fuel in concentrated salt brines

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I. Abstract

The final disposal of spent material test reactor fuel elements is being discussed in the Federal Republic of Germany as an alternative to reprocessing. A possible repository under consideration is a salt mine and the accident scenario for long-term safety analysis is a hypothetical water ingress. This water ingress involves the formation of highly concentrated salt brines. The design of the disposal casks ensures a lifetime of 500 year in such brines. After this time the brine will penetrate into the cask and corrode the MTR-FE.
Experiments with the different cladding materials show, that aluminum alloys will not resist the attack of brine solution, especially in the presence of iron. Although differences in the corrosion rates of about two orders of magnitudes were observed, the destruction must be considered as instantaneous in geological terms.
Radionuclides are mobilized from spent MTR fuel, when the "meat" becomes accessible to the brine. But after a short period with increasing activity the radionuclide concentration remains in a stable equilibrium or even decreases.

II. Introduction

Spent fuel elements from German Material Test Reactors (MTRs) have in the past been reprocessed in DOE facilities in the USA. As this has been discontinued since 1988, and because no proprietary reprocessing capacity is available in Germany, such material is being considered for final disposal in the deep geologic underground. A possible repository under consideration is a salt mine.
An accident scenario for long-term safety analysis is a hypothetical water ingress through crevices in the main anhydride. Water with a low salt concentration may migrate into the salt dome and solve salts, until the solution is saturated. The disposal casks are designed to resist the corrosive attack of such brines for about 500 years. After the container has failed the brine penetrates the cask wall and comes into contact with the spent fuel elements and finally mobilizes radionuclides.
The subject of our investigations is MTR fuel of the DIDO type, which is used in the Jülich FRJ-2 reactor. It contains, on average, 20 wt% U-Al alloy with an initial enrichment of 80% in ^{235}U. The fuel alloy plates with a thickness of 0.6 mm (meat) are covered on both sides with aluminum claddings of 0.38 mm, the total thickness is then 1.36 mm. Three plates each are welded to form a tube. A fuel element assembly consists of four concentric tubes of approximately 6, 7, 8 and 9 cm diameter, mounted onto structural parts, and after cutting off the structural parts, is about 63 cm long.
Our work is divided into three main parts:
1. Corrosion behavior of the aluminum cladding
2. Leaching of radionuclides from the spent fuel
3. Influence of the aluminum corrosion products on the solubility of radionuclides in the near field

III. Experimental

1. Corrosion of the cladding material

Aluminum alloy plates of Al99.5, AlMg1 and AlMg2 were cut into pieces of 40 x 20 x 2,5 mm and immersed in different salt brines (Table 1) at room temperature and 90°C. Depending on the progress of corrosion, the samples were removed from the brine, freed from corrosion products, dried, weighed and reimmersed in the brine. The experiments were performed in pure brine or with addition of cast iron (GGG40), stainless steel 1.4541 or Hastelloy C4.

Table 1: Brine compositions

salts	concentration [mol/1000 mol H_2O]		
	brine 1 at 55°C (Q brine)	brine 2	brine 3
NaCl	6.8	1.49	311.57
KCl	17.4	0.4	
$MgSO_4$	3.2		1.95
$MgCl_2$	77.3	97.21	
$CaCl_2$		5.68	
K_2SO_4			2.82
$CaSO_4$		0.01	2.54

2. Leaching of spent MTR fuel element pieces

Pieces form a spent MTR-FE (40 x 20 mm) were immersed in Q-brine (either in pure brine or in the presence of metallic additives) and stored at 90°C. At intervals, varying from days to weeks, the pH value was measured and an aliquot of 2.5 ml was taken from the brine. $^{135/137}Cs$ were measured by γ-spectrometry directly, $^{106}Ru/Rh$, ^{125}Sb, $^{141/144}Ce$, ^{144}Pr, ^{155}Eu and ^{60}Co after separation from Cs with ammonium phosphomolybdate. For α-spectrometry the samples were electrodeposited from 1 M NH_4Cl solution on polished, Ni-coated brass plates. Pu and Am were separated by a TRU•Spec extraction procedure.

3. Influence of the aluminum corrosion products to the solubility of radionuclides in the near field

First orientating experiments were started in the mid of 1996. Aluminumoxide was mixed brine 2 and different radionuclides. In periods of several days an aliquot of the solution was taken and the Americium concentration determined with Liquid Scintilation Counting.

IV. Results

1. Corrosion of the cladding material

The corrosion rate, surface related mass loss divided by the density, depends mainly from the temperature, the brine composition and combination with metallic additives. An increase from room temperature to 90°C causes a grow of the corrosion rate from 0.71 µm a^{-1} at room temperature to 135 µm a^{-1} at 90°C. The first figure shows the corrosion-promoting effect caused by the presence of storage container materials. Hastelloy C4 and SS1.4541 increase the corrosion rate to about 300 µm a^{-1}. The highest corrosion promoting effect was obtained by adding cast iron GGG40. The experiments had to be stopped after 50 - 100 days, because the test specimens had disintegrated.

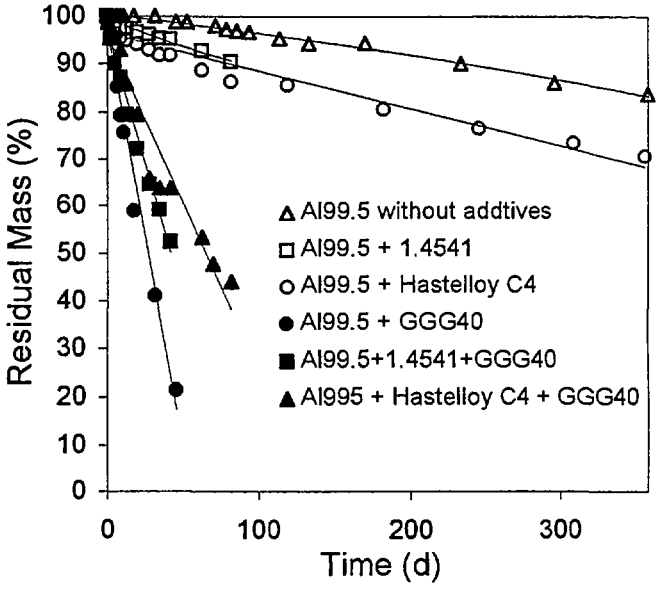

Figure 1: Mass loss depending on metallic additives in Q-brine

Table 2: Corrosion rates

Additives	Corrosion rate ($\mu m\ a^{-1}$)
no additives	135
1.4541	270
Hastelloy C4	320
GGG40	7200
1.4541+GGG40	4500
Hastelloy C4 + GGG40	2900

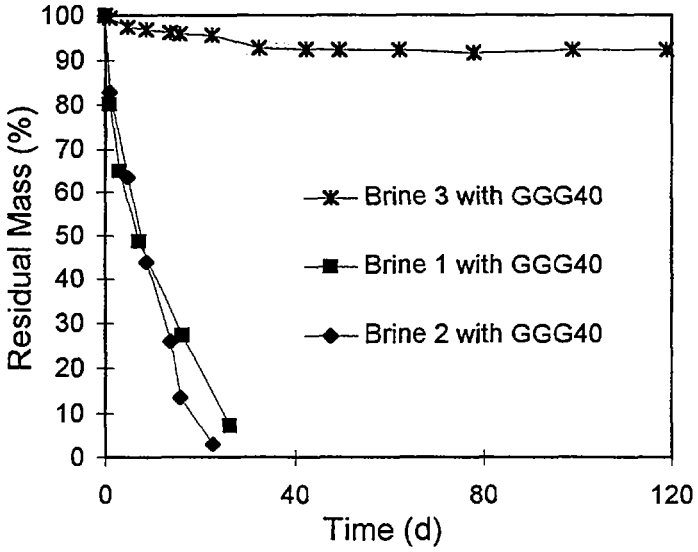

Figure 2: Mass loss depending on brine composition

The corrosion rate (Table 2) increases by about one order of magnitude. Furthermore pit corrosion becomes the main corrosion mechanism. We assume that the iron was oxidized and in a second reaction the iron ions were reduced to metal iron, forming a local element on the aluminum surface.
The influence of the brine composition is shown in figure 2. In brine 3 the corrosion nearly stops after a period of 50 days and the test specimens had lost only about 10% of their mass. The results obtained in brine 2 were compatible with the corrosion in Q-brine. The corrosion rates depend on the magnesium concentration.

2. Mobilization of radionuclides

Figure 3 displays the mobilization of γ-emitters from an MTR-FE in pure Q-brine at 90°C. After a prompt appearance in the leachate, the γ-activities remain almost unchanged or even decrease by up to two orders of magnitude. The shapes of the leaching curves for the α-emitters are quite similar to those obtained for the γ-emitters. The activity value for the α-emitters is in the order of 10^{-1} - 10^{-2} Bq/g fuel. The instantaneously leached radionuclides are mobilized from the surface of the meat or may result from a contamination of the strips during the cutting procedure. In the presence of cast iron a corrosion effect is obvious. Figure 4 and Figure 5 show the influence of metallic additives. The release rate of most of the radioactive nuclides increase by a factor 10^2 - 10^3. Within 24 days, 350 GBq of ^{137}Cs have been released in the presence of GGG40. After this period the activity increases only slowly and reaches a constant level after about 40 days. This equilibrium activity represents a leaching rate of about 6% of the total sample activity. The leaching rate of the other measured γ-emitters is about one order of magnitude lower. The actinide concentration reaches its equilibrium level within 10 days. The leaching rate of the total α-activity is less than 0.1% of the initial inventory. The low leaching rates can be explained by coprecipitation with the aluminum and iron hydroxides formed.

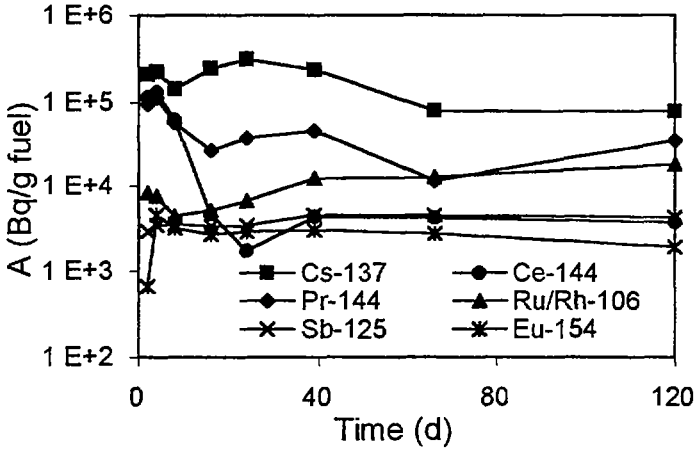

Figure 3: Mobilization of γ-emitters from MTR-FE in pure Q-brine

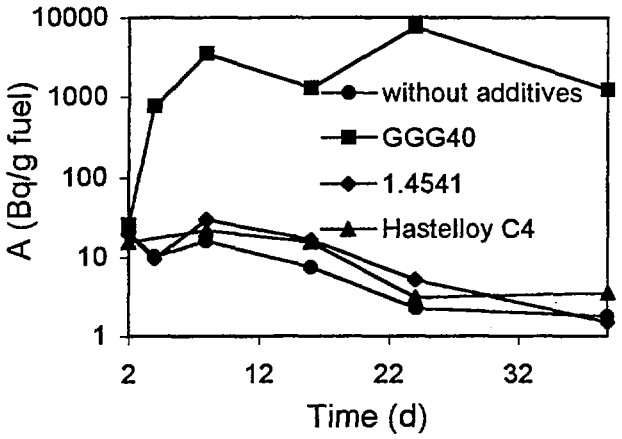

Figure 4: Influence of metallic additives on the mobilization of $^{239/240}Pu$

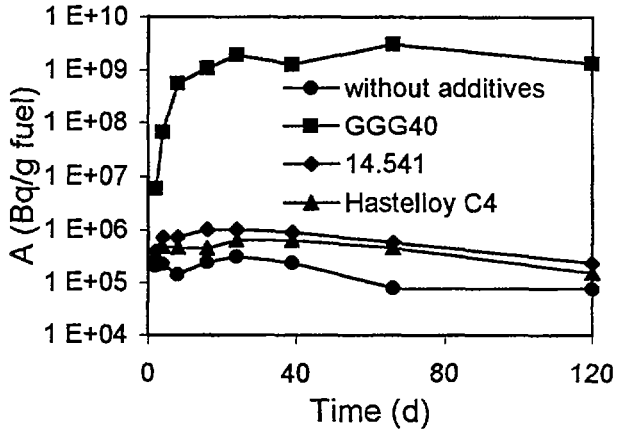

Figure 5: Influence of metallic additives on the mobilization of ^{137}Cs

3. Influence of the aluminum corrosion products to the solubility of radionuclides in the near field

The results obtained in the leaching experiments lead us to assume, that the radionuclides had been adsorbed on the corrosion products of the aluminum cladding. To characterize such solid phases we started some orientation experiments to describe the sorption of radionuclides on aluminumoxide in brine 2. The Figure 6 shows the sorption of ^{237}Np.

The sorption depends on the chemical properties of the different radionuclides, their oxidation state and the composition of the solution. A set of experiments to investigate all these effects had been stared and further results will be available in late 1997.

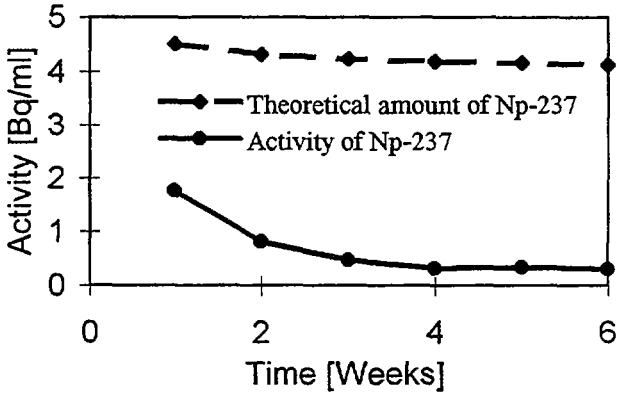

Figure 6: Sorption of ^{237}Np on Al_2O_3

V. Conclusions:

The results obtained from the corrosion experiments show, that the aluminum cladding has no barrier function in the case of an accidental water ingress into the final repository. The MTR fuel elements will be dissolved in the salt brines immediately with respect to the time scale of the long time safety consideration. However the mobilized radionuclides will be immobilized by the corrosion products of the storage cask and the fuel cladding. Therefor the subject of our further work will be to investigate the interactions between solid aluminum phases and the different radionuclides, which is important concerning the long time safety.

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Posters

ISOTOPE $c(^{234}U)/c(^{235}U)$ RELATION IN PIK REACTOR FUEL ELEMENTS

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Research PIK reactor has light water core with central irradiation channel and heavy water reflector in which experimental channels are located [1]. Rod type reactor fuel elements contain 90% enrichment uranium with density of 2.2 g/cm^3 [2]. Uranium-234 contained in fuel elements influences the reactivity. Its content can be changed depending on a method of enrichment and technology of fuel mix preparation. In a natural uranium isotope relation $c(^{234}U)/c(^{235}U) = (0.764 \pm 0.069)\%$ [3]. With the increase of $c(^{234}U)/c(^{235}U)$ from 0 up to 1% PIK reactor reactivity, calculated by MCU code [4], decreases by 0.4%.

In the present work, on the prof. Yu.Petrov's proposal, isotope $c(^{234}U)/c(^{235}U)$ relation in PIK reactor fresh fuel elements is investigated by undestructible γ-spectroscopic method. These data are necessary for increasing the PIK reactor k_{eff} (or critical loading) calculation accuracy .

Energies and the absolute intensities of the strong ^{234}U and ^{235}U γ-lines are listed in the table. The weak ^{235}U γ-lines close in energy to the strong ^{234}U γ-lines are also indicated in the table.

To control the radionuclides impurity from chains of uranium decay (having γ-lines with E_γ ~121 keV), PIK fuel elements spectra were measured with the help of the x-ray spectrometer with the resolution of ~0.4 keV. The analysis of experimental spectra and tabulated data on the chains of decay have shown that there are no radionuclides like that in the chains of uranium decay. The experimental intensity of the ^{235}U γ-line with E_γ=119.88 keV equalls (1.5 ± 0.7)% concerning the ^{234}U γ-line with E_γ=120.912 keV.

Table.

Energies and absolute intensities of the ^{234}U and ^{235}U γ-lines [3]

Uranium-234		Uranium-235	
E_γ, keV	I_γ, %	E_γ, keV	*)I_γ, %
53.23	0.119 (10)	51.179	~0.02
		54.201	<0.03
120.912	0.041 (4)	119.98	~0.026
		143.786	10.5(8)
		185.739	53(7)

*) the systematic error ~10%

To measure the isotope $c(^{234}U)/c(^{235}U)$ relation the ^{234}U γ-line with E_γ=120.9 keV and ^{235}U γ-line with E_γ=143.8 keV have been chosen. The relative content of ^{235}U was supervised on γ-line with E_γ=185.739 keV. The γ-spectrometer with Ge-Li detector with 30 cm^3 in volume having the resolution of 2.5 keV at E_γ=186 keV and 144 keV was used during the measuring process.

As PIK fuel elements have significant thickness on uranium and copper and strongly absorb γ-radiation with energies of 121, 144 and 186 keV (at E_γ=121 keV for PIK fuel element $\mu d \simeq 2$), the factors of absorption μ for these energies were measured by using tablets from bench mark PIK fuel element and exemplary γ-sources. Tablets were 0.2 mm, 0.5 mm and 1.2 mm thick. From measurements of γ-quantum with E_γ=121 keV and E_γ=144 keV accounting speeds coming from the tablets and experimental μd the isotope $c(^{234}U)/c(^{235}U)$ relation in bench mark PIK fuel element was determined. It equalls (0.79 $\pm$ 0.05)%.

The necessary accuracy of the ^{234}U content in fuel elements measurement is assessed from ^{234}U influence on reactivity and accuracy of reactivity calculations and measurements. As the accuracy of reactivity calculations and measurements on PIK critical assembly equalls 0.05% of reactivity, it is necessary to measure $c(^{234}U)/c(^{235}U)$ with an error of <0.0012 (as $c(^{234}U)/c(^{235}U)$ change by 0.0012 results in reactivity changing by 0.05%).

In measuring the content of ^{234}U in PIK fuel elements the relative root-mean-square error equalls $\simeq$8%. Therefore if $c(^{234}U)/c(^{235}U)$=0.008 the error equalls 0.0006. This results in the additional error of 0.025% in defining reactivity.

Isotope $c(^{234}U)/c(^{235}U)$ relation in PIK fuel elements is measured relative to bench mark PIK fuel element, where ^{234}U content was measured. Each fuel element was measured from two sides, turning it to 180° at the second measurement, which reduces a systematic error from absorption γ-radiation in uranium and construction materials. As γ-quanta with E_γ=121 keV and 144 keV have different factors of absorption, the change of the uranium content influences the relation of their accounting speeds.

Changing the uranium content (determined from measurements of ^{235}U γ-quanta with E_γ=186 keV accounting speed) in measured fuel elements equall $\leq$2%. In this cases amendment to changing the absorption (self-shielding) E_γ=121 keV and E_γ=144 keV relation equalls <0.5%. These amendments were not taken into account and were included in the measurement errors.

The $\simeq$60 PIK fuel elements were measured. Isotope $c(^{234}U)/c(^{235}U)$ relation in PIK fuel elements equall (0.82 $\pm$ 0.07)%.

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Non-destructive control of cladding thickness of fuel elements for research reactors

Y. Karlov, Y. Zhukov, S. Chashchin

1. Introduction

For supply of production with non-destructive automation control means the special subdivision - Central scientific-research automation laboratory (CSRAL) was founded at the plant. 150 specialists of CSRAL in the fields of ultrasonic, radiometric, eddy current and optical control develop techniques and equipment of non-destructive items control according to the designers' and product engineers' requirements. Cooperation of laboratory employees and scientists of the Siberian Branch of the Russian Academy of Sciencies yields high results.

Scince 1975 several generations of automatic equipment for non-destructive control of nuclear fuel elements for research and power reactors have been developed at the plant.

2. Technique and equipment

For provision of non-destructive automatic control of internal and external cladding thicknesses of nuclear fuel elements our specialists have developed and produced radiometric equipment on the basis of a beta back scattering method.

Specifications on fuel elements demand measuring of cladding thickness in the range of 0,2 mm up to 0,8 mm with accuracy at 0,95 confidence level. In this case, fuel elements have the diameter (wrench dimension) of 6,0 mm up to 70 mm and the length of 600 mm up to 1400 mm, as well as various thicknesses of fuel layer and uranium amount in it (depending on nuclear reactor type).

Measurement method of cladding thickness of a fuel element is based on recording with detector the ionizing radiation intensity of a beta back scattering by the systems "cladding - fuel layer ". The interrelation between a number of back scattered beta particles (n) and the cladding thickness (d) is described by a well- known formula:

$$n = n_o + (n_n - n_o) [1 - [\exp(- k\rho d)] ,$$ where

n_o - average number of back scattered beta particles in the absence of fuel cladding and by the thickness of the fuel layer exceeding saturation thickness;

n_n - average number of back scatterd beta particles by the cladding thickness axceeding saturation thickness;

k - coefficient of back scattering;

ρ - density.

Elaborators have carried out investigations of the interrelation between the number of back scattered beta particles and the surface density of the fuel layer [g/cm^2] for cladding thickness in the range of 0,2 mm up to 0,8 mm. ^{147}Pm, ^{201}Tl and ^{90}Sr + ^{90}Y radiation sources were used as beta particles sources. In the course of these investigations it appeared that the majority of fuel element types had different reflecting capability of fuel layer beta particles because of the difference in uranium amount.

Therefore, calibration samples with nominal value of surface density of the fuel layer were fabricated for each type of fuel elements.

Scintillation detectors with photoelectron multiplier (PEM) are used for recording of back scattered beta particles.

Measuring sensor with organic scintillator made of stilbene is used for external fuel cladding thickness control. Stilbene crystal has a central hole where the container with colliminator for ^{90}Sr + ^{90}Y beta radiation sources is placed. To control internal cladding thickness measuring sensors are used. They contain a small size source of beta radiation, detector of Bi_4 Ge_3 O_{12} monocrystal or synthetic diamond connected with PEM by means of a fibre-optical or glass light conductor. Such transducer design allows to get an access to the inner cladding of the fuel element with minimum diameter of 6,0 mm.

The simplified block-diagram of radiometric equipment for control of fuel element internal cladding thickness is given in fig.1. The equipment consists of mechanical scanning table with electric drive, measuring sensor with beta-ray source, pulse amplifier, amplitude discriminator and computer. The scanning table has carriage with movable unit for fuel elements fixturing. Removable clamping elements designed for round, square and hexagonal profiles are used for fuel element fixturing. The carriage makes possible to fix one till four fuel elements of the same length. Scanning of internal cladding by means of measuring sensor is performed automatically during linear movement through the preset number of generating lines. Scanning is controlled by computer according to the program selected by operator. The measuring sensor slides on the internal surface of fuel element. The information on cladding thickness presented in the form of recorded back scattered beta particles is input into the computer. Calibration functions for all types of manifactured fuel elements are entered into the computer memory. Taking into consideration the calibration function computer calculates the measured cladding thickness value.

Block-diagram and operation of radiometric equipment for external cladding thickness control are identical to the above described.

Metrological operation of the equipment shall be checked in automatic mode using calibration samples in accordance with the metrological program entered into the computer memory.

Accuracy of cladding thickness measurement is 0,05 mm with 0,95 confidence level.

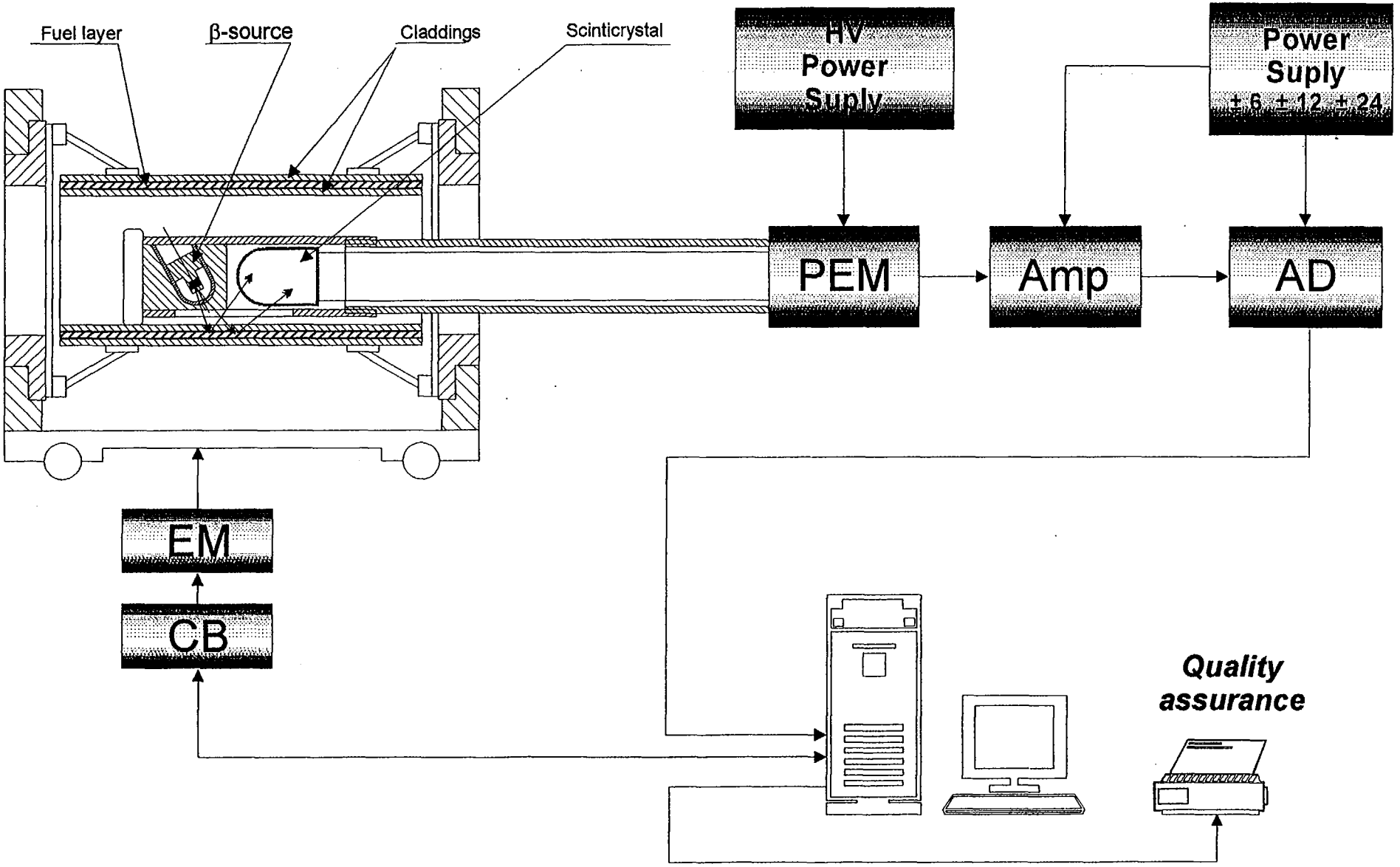
Control rate is 3 mm/s and 6 mm/s.

3. Conclusion

The shortly described method of control of fuel elements for research reactors by means of measuring beta particles back scattering made possible to perform complete automatic non-destructive control of internal and external claddings at our plant.

Availability of this type of control gives high guarantees of the fuel element correspondence to the requirements.

This method can be used to control the three-layer items of different geometry, including plates.



EM - *electronic motor*
CB - *connector block*
PEM - *photo electron multiplier*

Fig. 1

IMPROVED HOR FUEL MANAGENT BY FLUX MEASUREMENT DATA FEEDBACK

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ABSTRACT

Flux distribution in a nuclear reactor can be obtained by utilizing different calculational and experimental methods. The obtained flux distributions are associated with uncertainties and therefore always differ from each other. By combining information from the calculation and experiment using the confluence method[1], it is possible to obtain a more reliable estimate of the flux distribution than exhibited by the calculation or experiment separately. As a feedback, the fuel burnup distribution, which is used as initial data to the calculation can be improved as well. The confluence method is applied to improvement of the burnup distribution estimates for the HOR research reactor of the Delft University of Technology. An integrated code system CONHOR[2] is developed to match the CITATION[3] results of in-core foil activation rate calculations with in-core experimental data through confluence. The system forms the basis for the advanced fuel management of the reactor.

INTRODUCTION

Accurate determination of the power and fuel burnup distributions is important for reactor design, safety inspection and fuel management. Lack of accurate knowledge about these distributions leads to a situation the reliable and safe operation of the research reactor and its fuel elements under the specified constrains is far from optimal. In addition, considerable uncertainties associated with reactor states can propogate to conclusions drawn from research. In case of a power station, underloading of the reactor core leads to a considerable increase of the production costs. The demand of high cost savings, safety and proper operation assurance motivate developments of accurate power and fuel management systems. A modern management system includes experimental devices and computers for measurements and processing of in-core neutron reaction rate data. It also includes computers employing physical models, nuclear and design data to perform flux, power and burnup distribution calculations. A trustworthy information system justifies the corresponding expenses by allowing to operate the reactor optimally.

Informational sources (calculations and experiments) of the control system have associated errors. Different informational sources give different estimates of the same quantity. If the average characteristics of the random uncertainty of each particular informational source are known, then better estimates for the desired quantities can be derived through informational confluence founded on Bayesian conditioning[4]. Properties

of these approaches have been studied recently[1,5,6] for power distribution determination. An integrated CONHOR system is based on this approach and developed for the HOR research reactor of the Delft University of Technology in the Netherlands to match the calculational results with the in-core experimental data through confluence.

BURNUP ADJUSTMENT BY FEEDBACK FROM CALCULATIONAL AND EXPERIMENTAL DATA CONFLUENCE AT THE HOR REACTOR

The HOR (Hoger Onderwijs Reactor) is a 2 MWth pool-type research reactor situated at the Interfaculty Reactor Institute in Delft, The Netherlands. The core grid plate has 42 positions, normally loaded with fuel elements including 4 control ones and several reflector elements. The reactor is operated continuously 5 days a week. The licensed excess-reactivity is 6%. This requires replacements of one or two elements and reshuffling at a three-month interval. After completion of the reshuffling operation the power distribution is calculated. This distribution is used in burnup calculations which are also performed quarterly. An approach used to determine the power distribution employs the product of the neutron flux density and the fission cross section. Foil activities are measured to determine the flux density distributions and power distributions by the PEP[7] code. The code BUPNEW[8], based on approximate relationships between the isotopic composition of the fuel to the total energy produced by fission in each fuel element is used for burnup determination. Neither broadness and accuracy of available nuclear data nor power of modern computer codes is used in the employed methods. The confluence procedure, as described below, supplies a principal possibility of developing a comprehensive informational code system working in a reactor maintenance system. For example, burnup data of the HOR reactor can be significantly improved by flux measurement data feedback through confluence with calculations.

Distributions of the foil activation rates for some reactor state of the HOR reactor are normally available from both calculation and experiment. The experimental activation rates are obtained by converting the measured in-core gold foil activity counts using appropriate cross sections. The calculated activation rates are computed by the CITATION diffusion code using cross sections and nuclide densities as initial data. Fuel burnup values are obtained from PEP-BUPNEW calculations for previous cycles which could introduce considerable errors. The burnup uncertainty is estimated in Ref. 6 to be an increasing function of the burnup.

We denote the error vectors of the calculational and experimental activation rates as **c** and **e**, respectively. The vector **c** is non-zero due to both burnup inaccuracies and calculational model itself, denoted respectively by the vectors **d** and **m**. The vector **c** is approximated by a linear dependence on **d**; the model error **m** is summed with the error propagated from the burnup data **d**:

$$\boldsymbol{c} = \boldsymbol{A}\boldsymbol{d} + \boldsymbol{m} \tag{1}$$

where A is a sensitivity matrix of the calculational activation rates to the burnup data. The model errors **m** are estimated in Ref. 1 as 10% of the absolute values of the calculated activation rates. The experimental activation rates are subject to uncertainty estimated in Ref. 1 as 3% of the absolute values. The vector of statistical differences between the calculational and experimental activation rates is given by the difference in the uncertainties:

$$y = c - e \tag{2}$$

The diagonal variance matrices corresponding to the vectors **c**, **m**, **d**, **e**, and **y** are denoted respectively as C, M, D, E, and Y and describe the uncertainties of the error vectors. The matrix Y describes the properties of the total error uncertainty and can be expressed as

$$Y = ADA^T + M + E \tag{3}$$

where the superscript T denotes transposing the matrix. When both calculation and experiment are performed and we hold the observed value $\mathbf{y}_0$ of the vector of differences **y**, Bayesian conditioning can be employed[1] for the description of this posterior case. For example, for the burnup error vector **d** we secure the posterior non-zero mean

$$d_o = DA^T Y^{-1} y_o \tag{4}$$

and the posterior variance matrix:

$$D_o = D - DA^T Y^{-1} AD \tag{5}$$

where the superscript -1 denotes inversion of the matrix. In spite of the seeming complexity of the cited formulas, the annotation is quite modest. The vector **d** contains the statistically weighted observed differences $\mathbf{y}_0$. The posterior variances D are obtained by subtracting from tho prior variances D_0 some statistically weighted part of it. It means, that by applying the confluence procedure we secure adjustment **d** for the burnup vector. Uncertainties of the posterior values are given by D_0 and appear to be smaller than the prior ones. The CONHOR program system is developed to apply the confluence procedure for determination of the improved burnup for the HOR reactor and to serve as an instrument of fuel management support. The CITATION code is used as a main calculational tool for within the CONHOR system. CONHOR is responsible for creating automatically CITATION input files in accordance with the purpose of the calculation to be executed, for implementation of the confluence procedures and general management of all data flows. The general informational flow of CONHOR is presented in Fig.1. The corresponding calculational flow of the system can be described as follows. CONHOR creates a CITATION input file and performs a calculation of the activation rate distributions at the beginning of cycle (BOC). Then the described above adjustment procedure can be requested. It includes calculations of the sensitivity matrix A by perturbing burnup values. Adjusted burnup values at BOC are obtained using Eq. (4) and Eq. (5) and stored in a library. Afterwards, a depletion calculation based on the adjusted burnup values can be performed. Alternatively a depletion calculation can be done without the burnup adjustment. If burnup is executed, then another burnup library at the end of cycle (EOC) is produced. It can be a library based on either adjusted or unadjusted burnups at BOC, so that all types of comparisons can be made. Such comparisons were made for a number of subsequent HOR cycles[1] and showed convergence of the CONHOR results at EOC of the last cycle. Calculations performed by the CONHOR system for a number of HOR cycles showed the average reduction of the relative burnup uncertainty by a factor of 3. The corresponding relative difference between calculational and experimental activation rates reduced from 10% down to 2-3%.

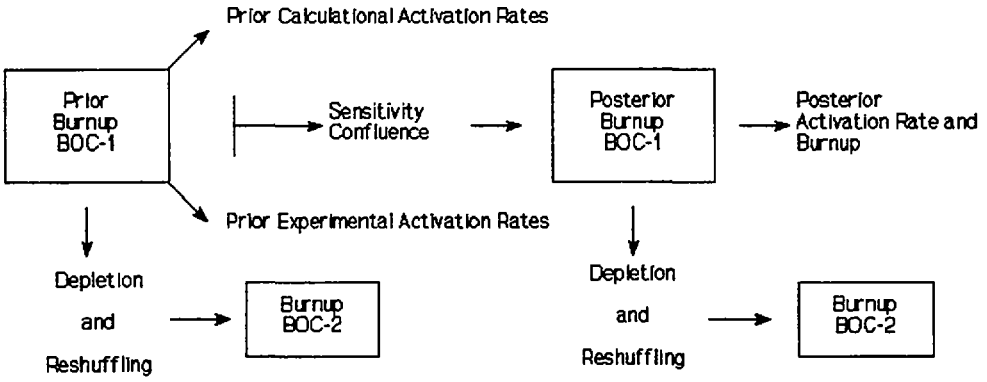

Fig. 1. Informational flow of CONHOR. BOC-1: beginning of first cycle studied; BOC-2: beginning of second cycle studied.

CONCLUSIONS

The confluence technique provides a framework for a comprehensive reactor data system, which can utilize all the available informational sources to offer optimal estimates of the required quantities and their uncertainties. The confluence technique proved to be useful for a burnup adjustment for a small research reactor as HOR. The CONHOR system is expected to become a routine instrument of the HOR reactor fuel management support.

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Core Management, Operational Limits & Conditions and Safety Aspects of The Australian High Flux Reactor (HIFAR)

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ABSTRACT

HIFAR is a DIDO class reactor which commenced routine operation at approximately 10MW in 1960. It is principally used for production of medical radio-isotopes, scientific research using neutron scattering facilities and irradiation of silicon ingots for the electronics industry. The license to operate HIFAR is granted by an independent Government body, the Nuclear Safety Bureau.

The HIFAR reactor uses six neutron absorbing coarse control arms to maintain and control criticality. These move in vertical planes between the rows of 25 fuel elements. A detailed description of the core, including fuel types in use, will be presented. Until 1992, the arms used were cadmium coated stainless steel blades[1]. The cadmium (Cd) coarse control arms use the Cd_{113} isotope, which has a thermal resonance giving an extremely high effective cross section of approximately 27600 barns, thus making it a great candidate for neutron absorption and hence control of criticality.

However, the limitation with these type of control arms is the short lifetime in the core (transmutation of Cd_{113} to Cd_{114} necessates changing frequently) and, more importantly from a safety point of view, the rapid fall off of absorption capacity. Detailed information on these operational limitations and the safety implications will be presented. In order to extend control arm lifetimes and improve the safety implications for HIFAR, a move was made to the use of europium (Eu) tipped control arms [2,3]. Following the preparation of safety submissions [4,5], the first Eu tipped coarse control arm was loaded into the HIFAR core in November 1992. Inverse kinetics measurements were performed in order to assess the differential reactivity worth of the control arm bank prior to and after installation of this (and subsequent) Eu control arms. These measurements, in addition to reactivity measurements, will be presented and discussed in detail. During the period from November 1992 to date, any Cd control arms which have required changing have been replaced with new Eu tipped blades. The last Cd control arm was removed from the core in October this year.

The HIFAR Safety Case was first formulated and reviewed in approximately 1970. As part of the review of the HIFAR Safety Case in 1996, a full analysis of control arm operational delay, drop and insertion time data was performed[6]. This allowed a Safety Case to be developed, providing the basis for HIFAR's operational limits and conditions. Relevant physics related Operational Limits and Conditions have been revised [7] to be consistent with the current HIFAR Safety Case and the use of Eu tipped control arms. These will be discussed in detail.

Details will be given of the current fuel management program, HIFUEL [8], used by the Reactor Analysis Group to predict the reactivity loss due to

burn up of fuel and the power distribution in fuel elements. Reactivity accounting is performed using HIFUEL in addition to maintaining a running reactivity total. This includes allowing for reactivity changes due to at power loads and unloads of targets from within hollow fuel element positions in the core and reactivity loss with burn up of Eu control arms [9]. Details will be provided of reactivity accounting and how this has been computerised, including fitting of reactivity loss data for Eu control arms with varying degree polynomials and computerisation of this to calculate the reactivity loss of the entire control arm bank. HIFAR is currently looking at moving to a new fuel management program HIFAM [10], which includes a 3D flux calculation and more accurate reactivity accounting.

Details of all experimental measurements associated with reactor physics analysis of HIFAR will be discussed, including coarse control arm differential reactivity measurements, rig reactivity measurements and control arm delay, drop and insertion time measurements. Differences between predicted and measured excess reactivity figures for various control arm calibrations (and different U235 core masses) will be presented. Quality System documentation (procedures and instructions) relating to reactor physics will be discussed.

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In-Core Fuel Management Practice in HANARO

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I. Introduction

KAERI(Korea Atomic Energy Research Institute) completed the system performance tests for the HANARO(Hi-flux Advanced Neutron Application Research Reactor) on December, 1994. Its initial criticality was achieved on February 8, 1995. A variety of the reactor physics experiments were performed in parallel with configuring the first cycle core and now HANARO is in the third cycle operation. The in-core fuel management in HANARO is performed on the following strategy[1]:

1) the cycle length of the equilibrium core is at least 4 week FPDs.
2) the maximum linear heat generation rate should be within the design limit.
3) the reactor should have shutdown margin of 1% $\Delta k/k$ at minimum.
4) the available thermal flux should satisfy the users' requirements.

This paper presents the fuel management practice in HANARO. Section II briefly describes the design feature of the HANARO and the method of analysis follows in section III and Section IV describes In-core fuel management practice and the conclusion is remarked in the final section.

II. Brief Description of the HANARO

The HANARO is a tank-in-pool type reactor of 30MW power and is now operated at its third cycle. The fuel meat is 19.75% U_3Si-Al. The fuel elements are clustered into the fuel assemblies. Two types of fuel assemblies are loaded in HANARO: H-type having 36 elements in hexagonal array and C-type having 18 elements in circular array.

The reactor core is separated from the reflector through the reactor inner shell. The core like a honeycomb inside the reactor inner shell is cooled and moderated by H_2O. The core has 31 flow tubes; 23 hexagonal and 8 circular. Eight circular flow tubes in core has rooms for accepting 4.5 mm thick cylinder of natural hafnium to regulate or shutdown the reactor. There are reserved additional eight circular flow tubes surrounding the reactor inner shell near the core to increase core excess reactivity by fuel loading and to irradiate the targets especially requiring the high epi-thermal neutrons.

In the reflector tank, total of 25 vertical holes with different sizes are arranged in reasonable positions, and 7 tangential beam tubes are horizontally allocated in optimal orientation.

III. Method of Analyses

To design the HANARO, KAERI itself set up HANAFMS (HANARO Nuclear Analyses and Fuel Management System). The major component of HANAFMS is WIMS/D4[2] and BOLD-VENTURE[3] and several auxiliary codes such as REGAV-K, WIMPAK, MAPHEX, HEXSHUF support the system. The WIMS nuclear data library was processed using NJOY[4] mainly with ENDF/B-IV and partly with ENDF/B-V. And then the WIMS with its library was benchmarked by comparing some key parameters such as criticality and reaction rate ratios against internationally authorized and recommended critical experiments. The test result for criticality shows 1.00303 ± 0.00519. The reaction rate ratios are also in good agreement with the experimental values[5]. By modelling the local structures of the reactor constituents, the macroscopic cross sections are generated and tabulated as a function of operating condition such as burnup, reactor power, fuel & moderator temperature, moderator density, reflector density and temperature, reflector impurity, etc. Based on the macroscopic cross sections, the BOLD-VENTURE was used to predict the core criticality, flux & power distribution in fuel assemblies, the reaction rates in the target materials, the detector responses. To simulate the neutronics properties of the HANARO, 3-dimensional H-Z model was adopted. The full core is divided into 253 x 253 x 34 grid and each fuel element is described by a single node. This give us the pin power history during depletion.

To validate and verify the HANAFMS prior to applying to the practical design work, the uncertainties of the HANAFMS were analyzed against the results from MCNP4 full core model, which treats the HANARO structure as real as possible[6]. Through this comparison, HANAFMS ensured the prediction capability with reasonable accuracy[7].

IV. In-core fuel management practice with HANAFMS

The major objective of nuclear fuel management lies in ascertaining the reactor safety and performance characteristics along with fuel burnup such as the power distribution of the fuel and excess reactivity, control capability. Based on above objective, the loading pattern is searched and analysed with the HANAFMS. The HANAFMS was benchmarked for criticality calculation with 14 critical experiments. The criticality experiments were performed in the course of construction of the first cycle core. The calculation result is 1.002112 ± 0.003099 for the 14 critical experiments[8]. The power distribution calculation is important to verify the reactor safety. It was compared with the gamma scanning data of 16 H-type fuel assemblies. The fuel assemblies were irradiated in the core for 17 hours at 150kW. The relative power for the fuel assemblies which was calculated by the HANAFMS compared with experimental results. The deviation was less than 6%. Fig.1 shows the operation history of HANARO until the end of its second cycle. The first cycle was operated 35.72 FPD at 15MW. The second cycle has 3 sub cycles, called 2-1, 2-2, 2-3 cycle. Each cycle was operated 42.74 FPD at 15MW, 22.86 FPD at 22MW, 18.23 FPD at 22MW. The criticality calculation of the HANAFMS along with fuel burnup was validated with experiment at BOC of the each cycle.

The criticality was calculated for the critical control rod position which was measured at the BOC of 2-1, 2-2, 2-3 cycle. The result shows 1.005922 ± 0.000794. The calculation of the control rod worth is very important to verify excess reactivity and shutdown margin. The four control rods of the HANARO move individually keeping same height in operation but we

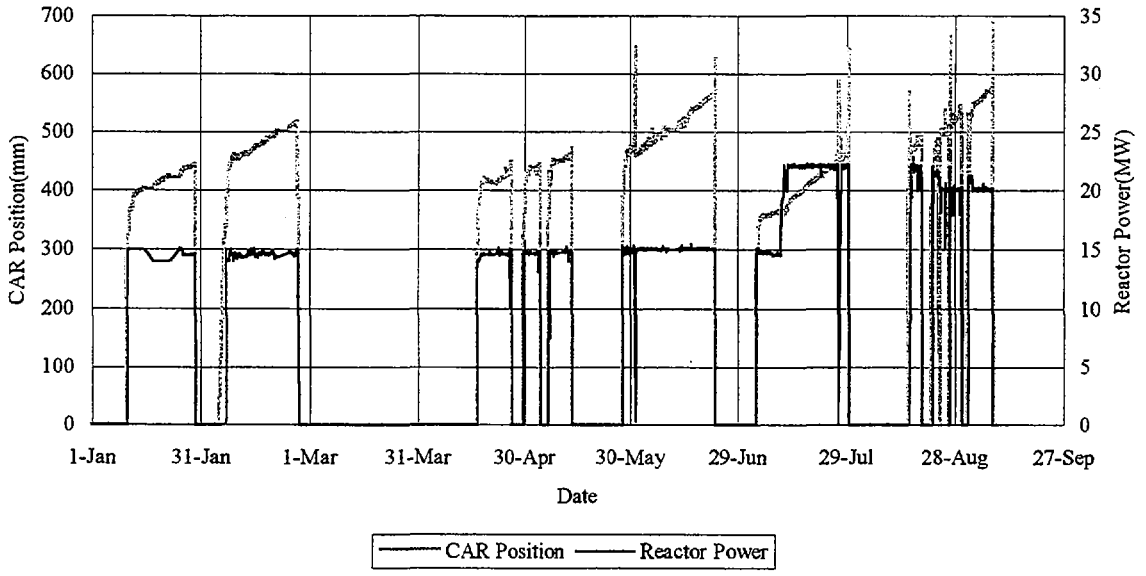

Fig. 1 The operation history of the HANARO

applied the rod swapping method to measure the control rod worth. While two rods are fixed at a designated position, the other two rods are moved in different direction to keep the reactor critical. Fig. 2 shows the measured and calculated control rod worth by the rod swapping method at BOC of the second cycle. From this result, we can produce the control rod worth curve ,which will be used in interpreting the other reactivity worth.

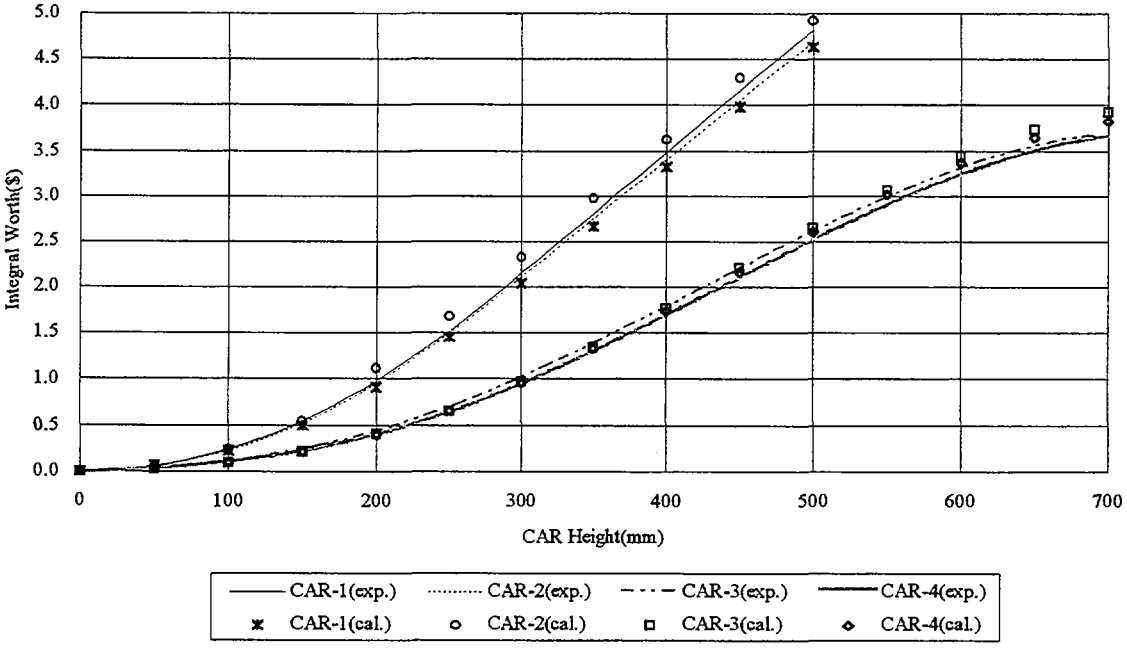

Fig. 2 The control rod worth by calculation and experiment

V. Conclusion

The HANARO is a newly born research reactor in the world. Various kinds of reactor physics tests and experiments have been carried out. To support those experimental activities and to predict the safe operation of the HANARO, the core follow up calculation using HANAFMS is always simulated along with the cycle operation. Up to now, the HANAFMS shows good predictability, but it has some problems. In operation, various test materials and targets for irradiation are loaded in the core. Sometimes they have complex geometry. WIMS/D-4 has a restriction to model in detail to produce cross section data of them. So, we have a plan to use HELIOS[9] to produce the cross section data because HELIOS has a capability to model the irradiation targets regardless of geometry. Since the VENTURE is a finite-difference diffusion theory code and its HANARO model is too large, it can not forsee the neutron behavior in strong absorber or high scattering medium and takes too much time to predict the neutronics behavior of the HANARO. Therefore some adjustments or fudging in cross section is needed to predict the CAR worth more accurately and we endeavor to implement a new time saving system by adopting the nodal method as a core analyser.

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Operation of the FRG-Research Reactors at Geesthacht

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Introduction

Two research reactors have been operated very successfull by the GKSS research centre over decades in a connected pool system.

FRG-1: 5 MW, criticality October 1958
FRG-2: 15 MW, criticality March 1963 and decommissioned February 1995

The FRG-2 was scheduled to stop operation in 1991 for lack of scientific and technical interest for future use [1]. The reactor has been used as Germany`s largest material testing reactor for power reactor fuel and power reactor materials development and safety tests. The FRG-2 has also played an important role in the conversion activities at the GKSS research centre.

The FRG-1 is being used with high availability for beam tube experiments for fundamental and applied research in biology, membrane development, materials research, neutron radiography, neutron activation analyses etc. To enable the sufficient and efficient use of long wave length neutrons a cold neutron source has been installed in one of the beam tubes [2].

The GKSS research centre, the advisory board, the scientific community and other clients are demanding the ongoing operation of the FRG-1 for at least till the year 2010. GKSS has taken many actions to ensure the save operation with high utilization and availability for the next 15 years.

Conversion of the FRG-1 from HEU to LEU

The conversion of the FRG-1 from High Enriched Uranium (HEU) to Low Enriched Uranium (LEU) was made in the year 1991 [3]. For the users the conversion should bring a higher neutron flux at the beam tubes. Therefore the core size was reduced from 46 to 26 fuel elements and an additionel beryllium reflector around the beam tube heads was installed.

For decreasing the operation costs of the reactor it was nessessary to increase the burnup of the fuel elements and to have a longer fuel cycle. Therefore the uranium density was changed from 0.44 to 3.7 U/cc and the weight of U-235 was increased from 180 to 323 g U-235/fuel element. Table 1 shows the comparison of high enriched and low enriched core.

In addition the design of the control rods have been changed from oval B_4C and Cd absorbers to fork absorbers with hafnium blades. So we have more reactivity for a longer operation cycle.

The design and all dimensions of HEU and LEU fuel elements are identical. Only the control fuel elements with fork absorbers has been new designed.

The conversion to LEU elements have been done in one step, without a mixed LEU/HEU core. Some elements have been preburned in our FRG-2 reactor to approx. 12%. The process of conversion has been a standard procedure: measuring of the rod drop time for the control rods; critical experiment; reactivity calibration of the control rods to be aware of conditions for stuck rod criteria, minimum shutdown reactivity speed of control rods, max. excess reactivity; power calibration with foils, minature fission chambers and thermal calibration; flux mapping; measurement of reactivity worth of fuel elements, reflector elements and irradiation samples; measurement of fuel plate temperature with instrumentated fuel elements.

In combination with the installation of the cold neutron source long wave length neutrons >5 Å has been increased by more than a factor of 50.

	HEU-core	LEU-core
U-235 enrichment	90%, 93%	19,75%
meat material	UAl_x	U_3Si_2
U-density	0,44 g/cc	3,7 g/cc
canning material	AlMg1	AlMg2, AG3NE
weight U-235	180 g	323 g
average burnup	40%	55%
max. work / fuel element	60 MWd	145 MWd
control rods	oval sword with B_4C and Cd	fork absorbers with Hf-blades
number of control rods	4	5
core size (equilibrium)		
- standard fuel elements	40	20
- control fuel elements	4	5
- irradiation fuel elements	3	1

Table 1: Comparison of high enriched and low enriched core

Use of the FRG-1

The FRG-1 is used for beam tube experiments and for incore irradiation of small samples for neutron activation analysis. Fig. 1 shows the facilities at the FRG-1.

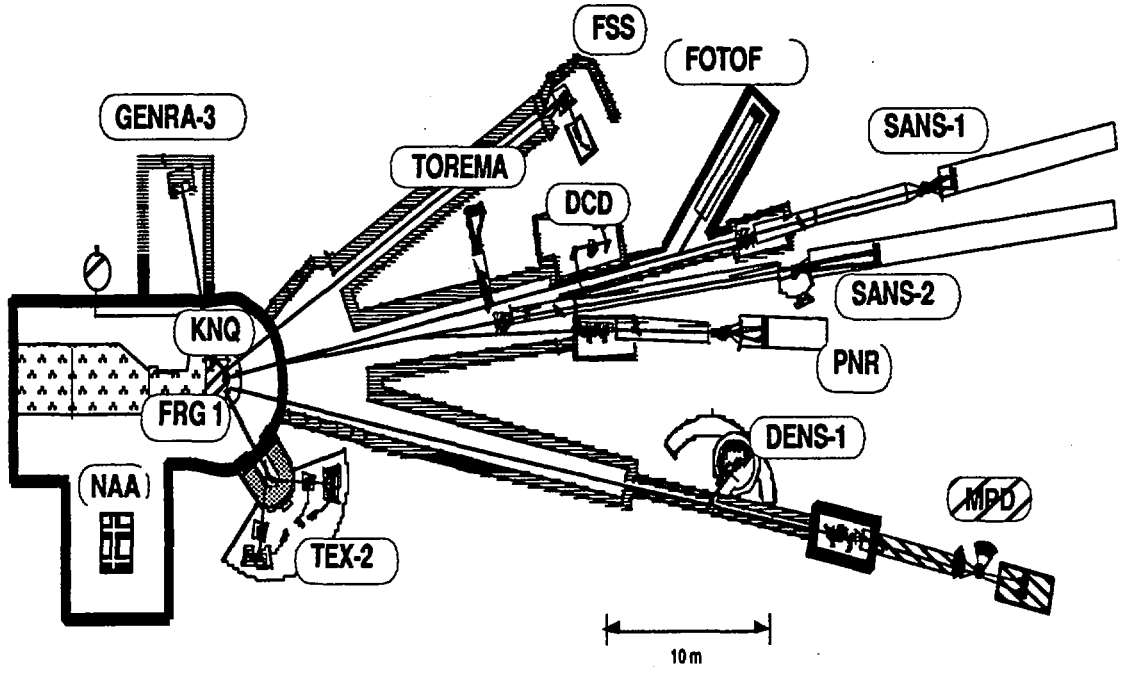

Fig. 1: Facilities at the FRG-1

The instruments are based mainly on elastic neutron scattering. They offer the possibility to study materials, semi-finished products and components with respect to their microstructural constituents, including defects with atomic dimensions as well as microcracks in the micrometer range.

Fig. 2 shows a typical core configuratin with the possibilities for incore irradiation. There are one irradiation fuel element, four irradiation positions in beryllium elements and two positions in alluminium tubes. The maximum thermal neutron flux in the irradiation positions is in the range from 1.5 to 8 E+13 n/cm^{2}s.

The average availability of the FRG-1 during the last 6 years after conversion to LEU was more than 200 full power days/year with increasing trend.

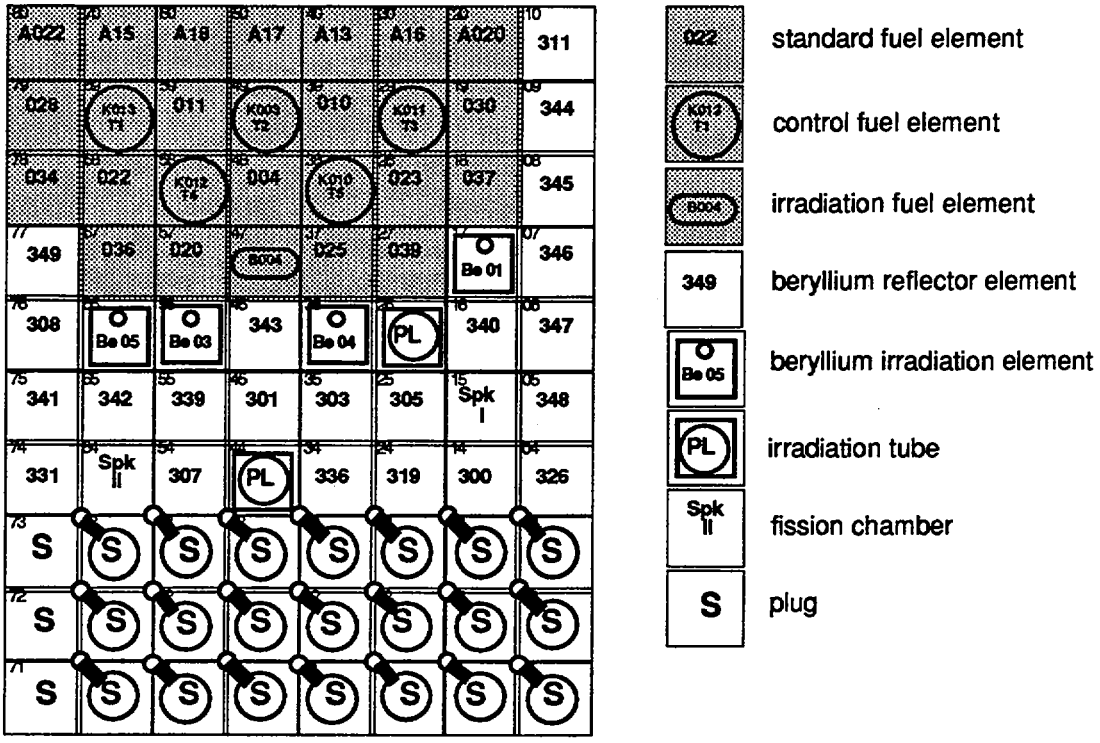

Fig. 2: Typical core configuration

Future of the FRG-1

To prepare the FRG-1 for an efficient future use considerations and first calculations have been performed to reduce the core size again from 26 fuel elements to 12 fuel elements and to take some additional measures for further increase of the neutron flux at the beam tube positions. Table 2 shows the comparison of present and future core.

First calculations indicate that the increase in thermal neutron flux for the beam tubes is between 50% and 160% depending on the position of the beam tube in relation to the present and future reactor core positioning. The maximum thermal neutron flux will be at the position of the cold neutron source - the most used device- and increases there from 9 E+13 to 15 E+13 n/cm^{2}s.

GKSS envisages to operate the FRG-1 at least until the year 2010. To keep this important neutron source at a high level of interest for the scientific community GKSS will make all efforts to improve technical performance of the reactor and to develop the experimental facilities supplied with neutrons from the FRG-1.

	present core	future core
number of fuel elements (FE)	26	12
plates per FE	23	23
number of control rods	5	4
fuel	U_3Si_2	U_3Si_2
fuel density (g U/cc)	3,7	4,8
U-235 (g) content per FE	323	420
reflector	H_2O, Be	Be
tangential beam tube	H_2O	Be
front end of beam tube		optimized

Table 2: Comparison of present and future core

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Calculations of Fuel Element Burn-up for Mixed TRIGA Core

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1. Introduction

One of the most demanding tasks which is normally responsibility of the reactor operation staff is the determination of fuel element burn-up. The task is particularly complicated if the reactor core was mixed and if the loading pattern was frequently changed. The following three methods of fuel element burn-up determination are commonly applied: reactor calculations, reactivity measurements and the gamma scanning method [1, 2, 3]. The first two methods are relatively simple and convenient for determining burn-up of large number of fuel elements. The gamma scanning method is in practice normally used only as a reference method for measuring small numbers of fuel elements as it requires special equipment and high level of expertise.

Reactor calculations, which were not so long ago strictly in domain of reactor manufacturing companies or large national institutes are becoming more and more feasible also to the operators in small research reactor centers due to rapid development of small computers and appropriate software. The purpose of this paper is to describe two computer code packages which were developed at J. Stefan Institute for practical, operator oriented burn-up calculations of TRIGA Mark II cores. Their application for mixed core burn-up calculations is presented as well.

2. Description of Computer Codes

The first is TRIGAC package[4]. It is based on two-group diffusion equation in one-dimensional cylindrical geometry. It is solved in the finite differences approximation by fission density iteration method. The package is supplied with a cross-section library which is calculated with the program WIMSD-4 [5] for all types of TRIGA fuel elements (FLIP, Standard 8.5w% and 12w% uranium concentration, aluminum and stainless-steel cladding, LEU). The cross-sections are tabulated as a function of burn-up, fuel temperature and other operating parameters. The group constants of the fuel elements (and other unit cells, e.g. irradiation channels) in the same radial ring are homogenized by simple volume weighting. The physical model is appropriate for simple uniform loading pattems with only one type of fuel elements in the same ring. It is not appropriate for mixed rings or for regions near control rods and incore irradiation channels. TRIGAC was developed for the first generation of PC computers in 1982 for which any better physical model was not feasible. Running time was around 20 minutes for the first generation of IBM-PCs. It is around 10sec for modern PC (e.g. Pentium 130MHz processor).

The code was recently redesigned. Our aim was to improve the physical model by using more fine group structure and 2-D geometry. Four group diffusion approximation in r-φ cylindrical geometry was applied. Method of solution are finite differences and fission density iteration. The geometry model represents full TRIGA Mark II cylindrical core with maximum 7 rings and reflector. Every grid location either occupied by the fuel element, control rod, irradiation channel or left empty is treated explicitly as a unit cell. The group constants are calculated with program WIMSD-4, which is integrated in the package. In contrast to the original version of TRIGAC, the group constants are not pre-calculated and stored in the library but calculated for each unit cell in the core separately according to its particular composition, burn-up and temperature. In illustration: WIMS code is run more than 100 times to produce the cross-sections for one diffusion calculation. As the physical model of the code was changed significantly we decided to rename it to TRIGLAV [6]. The running time on Pentum PC is typically 30-60 minutes for one burn-up step.

3. Comparison of Calculated and Measured Burn-up

TRIGLAV was tested on numerical benchmarks and by comparing it to the results of other 2-D codes, e.g. EXTERMINATOR [7]. Numerical tests show that the accuracy better than

10pcm in k_{eff} and 0.1% in fission density distribution can be achieved even with relatively coarse finite differences mesh containing approximately 150x150 mesh points to cover a typical completely heterogeneous TRIGA core with graphite reflector.

The integral test of the code was performed by comparing its results to the results of the measurements for our reactor. Our reactor is 250kW TRIGA Mark II type operating since 1966. The core consists of 6 concentric rings (denoted A-F). The reactor was originally equipped with 3 boron graphite control rods. In 1991 they were replaced by the fueled-follower type control rods made of boron carbide (absorber part) and 12w% uranium fuel (fuel part). Additional pulse rod (boron carbide with air-follower) was installed. Several types of fuel elements were used in 30 years of operation:

1. Aluminum cladding, 8.5w% uranium, 20% enriched standard fuel elements
2. Stainless-steel cladding, 8.5w% uranium, 70% enriched FLIP fuel elements
3. Stainless-steel cladding, 8.5w% uranium, 20% enriched standard fuel elements
4. Stainless-steel cladding, 12w% uranium, 20% enriched standard fuel elements.

FLIP fuel elements were used in combination with 8.5w% uranium standard fuel elements for 15 years. Due to small number of FLIP elements (26 total) mixed rings could not be avoided. In 30 years of operation the core was reshuffled 146 times for fuel management and flux optimization purposes. The burn-up calculation was performed with TRIGAC and TRIGLAV for entire operating history of the reactor. Burn-up of all fuel elements was calculated. The burn-up of selected fuel elements of all types was also measured by the reactivity method.[2, 3] Results for 8.5w%-U standard fuel elements are presented in the Figure.

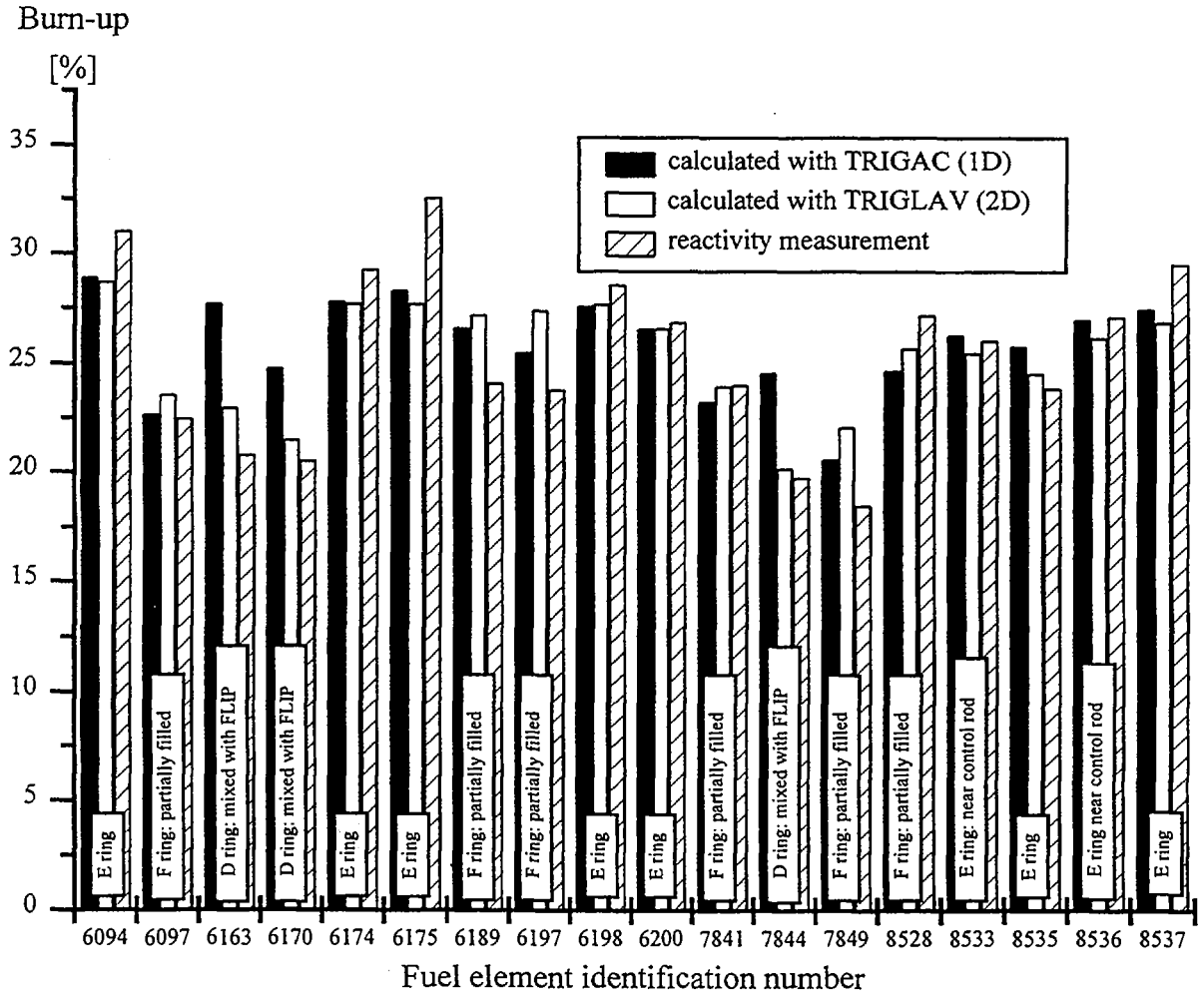

Figure: Calculated and measured burn-up of selected standard 8.5w%-U fuel elements

We selected for the measurement such fuel elements that were expected to be influenced by 2-D flux distribution effects: mixed rings, influence of control rods and irradiation channels, influence of incomplete F-ring. Comparison of measured and calculated results clearly shows that 2-D calculations better predict the burn-up of mixed rings containing highly enriched FLIP and low enriched standard fuel elements. 2-D calculations fail to reduce discrepancy for the fuel elements close to the reflector (E and F rings). Consequently, the reason for the discrepancy should be looked for in the unit cell calculations rather than in the global geometry model. Average deviation between measured and calculated results is estimated to be around +/- 0.05, except for some particular cases discussed above.

Table: Comparison of burn-up calculated in 1-D (TRIGAC) and 2-D (TRIGLAV) approximation for low enriched (standard 8.5w%-U and 12w%-U) and highly enriched (FLIP) fuel elements

Fuel element identification number	Fuel element type	Burn-up calculated with		Difference [%]	Comment
		TRIGAC (1D) [MWd]	TRIGLAV (2D) [MWd]		
6105	stand. 8.5%	4.43	2.42	82.8	D ring: mixed with FLIP
6106	stand. 8.5%	0.24	0.25	-1.9	
6107	stand. 8.5%	1.64	1.85	-11.5	B ring: near control rod
6108	stand. 8.5%	3.28	3.08	6.5	
6109	stand. 8.5%	2.97	2.96	0.4	
6110	stand. 8.5%	4.27	3.93	8.6	
6111	stand. 8.5%	2.11	2.43	-13.2	F ring: partially filled
6112	stand. 8.5%	2.45	2.82	-13.0	B ring: near control rod
6113	stand. 8.5%	2.11	2.40	-12.3	F ring: partially filled
6114	stand. 8.5%	2.11	2.44	-13.4	F ring: partially filled
7214	stand. 12%	1.17	1.16	0.8	
7219	stand. 12%	1.17	1.18	-0.8	
7220	stand. 12%	1.17	1.15	1.7	
7223	stand. 12%	0.93	0.93	0	
7225	stand. 12%	0.93	1.01	-8.0	
7228	stand. 12%	0.93	0.95	-2.1	
7229	stand. 12%	0.93	0.97	-4.3	
7233	stand. 12%	0.93	1.06	-11.8	D ring: asymmetric core
7236	stand. 12%	0.93	1.04	-10.3	D ring: asymmetric core
7245	stand. 12%	0.93	0.96	-2.6	
8285	FLIP	7.26	7.11	2.1	
8290	FLIP	9.53	8.70	9.6	
8302	FLIP	7.26	7.25	0.1	
8303	FLIP	7.26	7.73	-6.1	
8304	FLIP	7.26	8.71	-16.6	D ring: near control rod
8305	FLIP	7.26	7.66	-5.3	
8307	FLIP	7.26	8.20	-11.5	D ring: near control rod
8313	FLIP	9.53	8.41	13.3	C ring: with control rods
8316	FLIP	9.53	8.65	10.2	C ring: with control rods
8318	FLIP	9.53	9.98	-4.5	

4. Analysis of Two-Dimensional Effects

Comparison of fuel element burn-up calculated with TRIGAC (1-D approx.) and TRIGLAV (2-D approx.) is presented in the Table for three groups of 10 elements of different types. The groups were selected randomly out of more than 150 fuel elements that were treated in the calculation. The elements that spent main part of their life at the positions of 2-D effects are denoted in the comments. The difference between 1-D and 2-D results exceeds 10% for such elements. In some particular cases the difference is even higher (e.g. 83%).

5. Conclusions

Comparison of 1-D and 2-D calculations shows that 1-D models fail to predict the burn-up of those fuel elements that spent main part of their lives in mixed rings. The 1-D calculation based on homogenized rings systematically overpredicts the burn-up of low enriched fuel elements and underpredicts the burn-up of the highly enriched if they were used in the same ring for long time. The effect depends on the number of fuel elements of each type but it is typically in the order of 15% relative difference.

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Some Aspects of Accidental Criticality Safety of TRIGA Reactor Spent Fuel Pool

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Criticality safety analysis of a pool type storage for TRIGA spent fuel at "Jožef Stefan" Institute in Ljubljana, Slovenia, is presented. Previous results have shown that subcriticality is not guaranteed for some postulated accidents. To mitigate this deficiency, a study was made about replacing a certain number of fuel elements in the rack with absorber rods.

For this purpose Monte Carlo computer code MCNP4A with ENDF-B/V library and detailed three dimensional fuel rack model was used. At first the analysis was done about the number of uniformly mixed absorber rods in the lattice needed to sustain the subcriticality of the storage when pitch is decreased from rack design pitch of 8cm to contact, assuming that the absorbers retain their proper positions.

Because of supercriticality possibility due to random mixing of the absorber rods among the fuel elements during lattice compaction, a probabilistic study was made, sampling the probability density functions for such random lattice loadings. The results show reasonably low probabilities for supercriticality even for fresh standard TRIGA fuel containing 12 wt % uranium stored in the pool.

I. Introduction

It was shown in [1] for spent fuel storage pool, presented in Fig.1, at 250kW TRIGA reactor operated by "Jožef Stefan" Institute in Ljubljana, Slovenia, that for some accident conditions (e.g. an earthquake followed by total fuel rack disintegration where fuel elements pile together to their contact and water remains in the pool) subcriticality cannot be guaranteed. If the same pool would be used for emergency core unloading, the calculation with the most reactive, fresh 12 wt% standard TRIGA fuel was done to cover the possibility for such unloading to occur in the very beginning of the core life. To mitigate such a risk certain number of fuel elements should be replaced by absorber rods.

The aim of this contribution is to present results of probabilistic criticality safety analysis showing that even if previously uniformly mixed additional absorbers would not remain in their relative positions during the supposed accident, only minimal probability for supercritical configuration can be expected.

II. Description of the pool and the absorber rods

The pool, excavated in the basement of the reactor hall, is 2.6×2.6m wide and 3.6m deep. It is filled with pure demineralised water. The walls are made of reinforced concrete clad with stainless steel. The aluminium fuel rack is attached to the bottom of the pool, consisting of top and bottom support plates connected by vertical props at the sides. Top support plate has 21x10 holes for inserting fuel elements. Pitch is 8cm and hole diameter is 4cm. Holes are arranged in a square array. The rack is divided in three segments of 7x10 positions by aluminium-clad cadmium plates (Fig.1).

Due to its availability cadmium was chosen as absorbing material of the additional absorber rods as well. A short study, regarding the influence of the thickness of cadmium layer in the rod between 0.0001mm and 15mm is presented in Fig.2. It could be seen that its thickness has not much influence on the multiplication factor (k_{eff}) for values greater than 1mm [2]. The final shape of the absorber rod was modeled as with top and bottom plugs made of stainless steel and 6.2mm thick cadmium plate, rolled in the form of hollow cylinder.

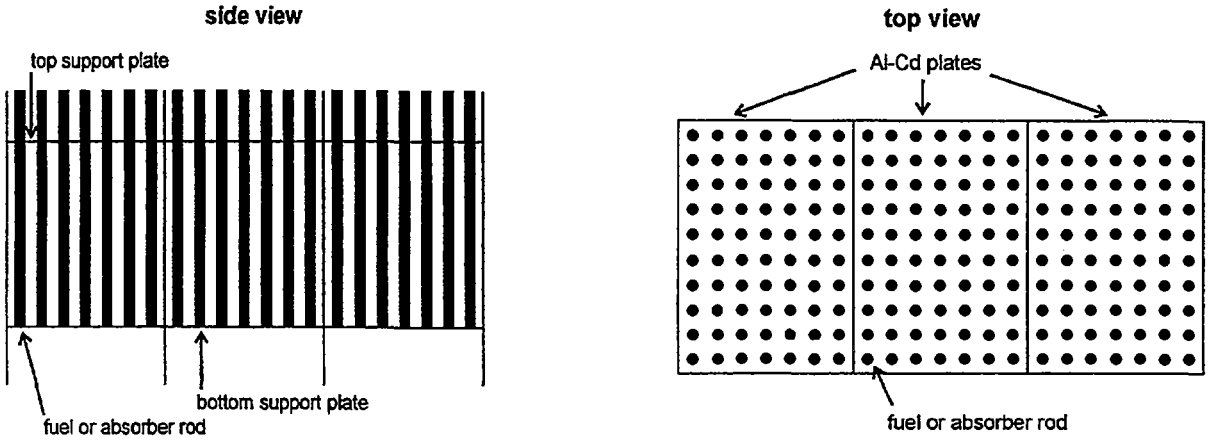

Fig.1: Side and top views of the spent fuel pool rack at TRIGA Ljubljana

III. Calculational method

Monte Carlo computer code MCNP4A [3] was used. A continuous neutron cross section library, evaluated from ENDF/B-V data [4] was applied. The scattering functions for graphite, hydrogen in water molecule and hydrogen in zirconium were taken from the ENDF/B-IV library.

Detailed three-dimensional geometry was used. Both, the fuel elements and the additional absorber rods were exactly modeled, as well as the fuel rack and the walls of the pool, so that both, axial and radial leakage, were correctly described. The multiplication factor of such geometrical model is denoted as k_{eff}.

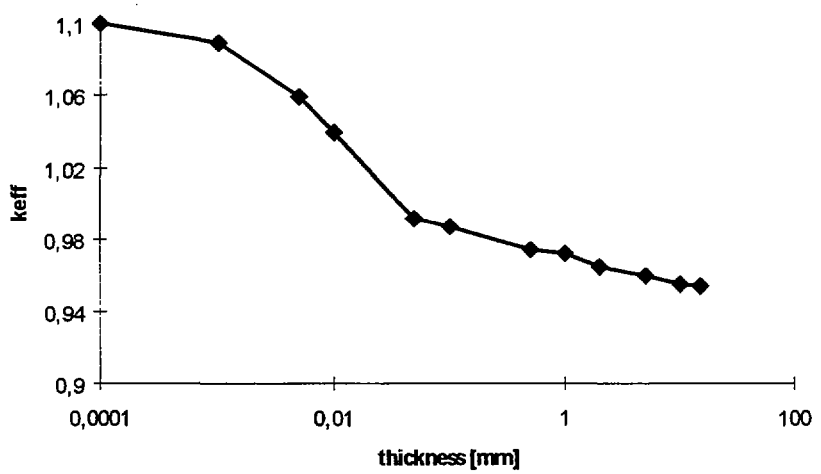

Fig.2: k_{eff} for the pool with 8 cadmium rods in each rack compartment versus thickness of cadmium layer in the rod

IV. Results

Critical number of 12 wt% fuel elements in contact

We started our calculation with a certain subcritical number of fuel elements in arrangement of square lattice. The fuel elements were in contact. The mass of the system was increased by sequentially adding fuel elements. At 35 fuel elements supercriticality was achieved. Other lattice locations contained water.

Criticality for uniformly loaded absorber rods during pitch decrease

To show that uniformly mixed absorbers among the fuel assure subcriticality under postulated accident conditions, there were several absorber rods (8, 10, 12, 15, 17 and 20) mixed among the fresh fuel elements in a more or less regular manner. The grid was considered completely filled. k_{eff} was calculated for different pitches from the design pitch of 8cm down to square contact pitch for every chosen number of absorber rods.

Results show that even only 8 uniformly positioned absorber rods among 62 fuel elements assure subcriticality for all pitches down to contact (Fig. 4). Result for fuel elements in all 70 positions (no absorber rods) is added as an illustration for supercriticality of such loading.

Supercriticality probability estimation

In the case of random mixing of fuel elements and absorber rods as a consequence of the rack disintegration the probability for supercriticality is estimated. There are N possible ways to load n equal absorber rods and 70 - n equal fuel elements into this lattice. N is given by:

$$N = \frac{70!}{n!(70-n)!} \qquad (1)$$

For example, if n = 20, N is approximately $1.62 \cdot 10^{17}$. So the probability for the worst possible loading, e.g. all fuel to one side and absorber rods to another side, would be negligibly low, i.e. 10^{-17}. Although that this is an extreme case, there may be other supercritical loadings, too.

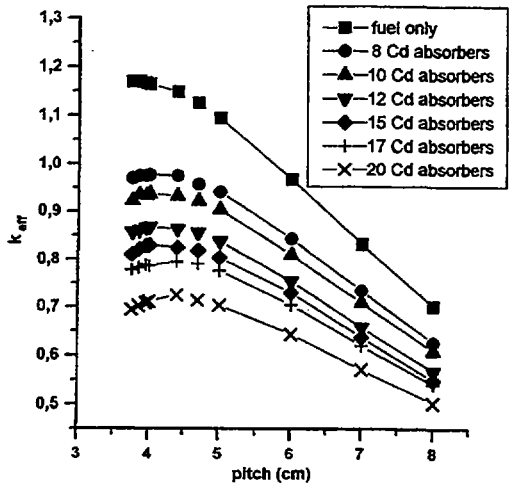

Fig. 4. k_{eff} for various numbers of uniformly loaded absorber rods versus lattice pitch

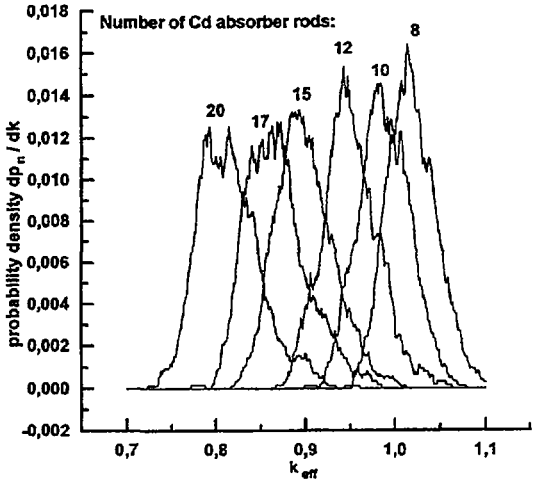

Fig.5. Probability density functions $\partial p_n(k_{eff})/\partial k_{eff}$ for various numbers of randomly loaded absorber rods

To determine the probability for supercriticality the probability density functions $\partial p_n(k_{eff})/\partial k_{eff}$ were sampled. For each selected number of absorber rods n (8, 10, 12, 15, 17 and 20) 500 different MCNP inputs were run with randomly distributed n absorbers and 70-n fuel elements. By retrieving k_{eff} and its standard deviation σ the probability density functions were cumulatively built simply by adding unity to a vector between $k_{eff} - \sigma$ and $k_{eff} + \sigma$ with resolution of 0.0001. The accumulated probability density functions were normalized (Fig. 5), so defining the probability for supercriticality $p_n(k_{eff} \geq 1)$ as:

$$p_n(k_{eff} \geq 1) = \int_{k_{eff} \geq 1} \frac{\partial p_n}{\partial k_{eff}} dk_{eff} \qquad (2)$$

The probabilities for supercriticality estimated with Eq.(2) are given in the second column of Table I. 500 MCNP runs were done for each number of absorber rods. The gaussian and exponential least squares fits were applied to the right hand side tail of the "experimental" probability density functions to avoid the still present scatter of the raw data. For n = 8 and 10 fits were not done because of too high probabilities for supercriticality. Although that gaussian provides better fit [2], both gaussian and exponential supercriticality probability estimations are included in Table I.

Number of randomly loaded Cd absorbers	Probability for $k_{eff} \geq 1$		
n	"experimental" data	gaussian tail	exponential tail
8	0,75	/	/
10	0,35	/	/
12	0,062	0,069	0,12
15	0,0026	0,0036	0,033
17	0	0,0011	0,015
20	0	0,0000013	0,0045

Table I: Probability for supercriticality for various numbers of randomly loaded absorber rods. Pitch: square, fuel elements in contact (3.75cm)

V. Conclusions

Subcriticality of the spent fuel storage pool cannot be guaranteed, if rack disintegration and lattice compaction to square contact pitch would occur and if, at that time, more than 34 fresh 12 wt% standard TRIGA fuel elements would be stored in the pool. The criticality safety study shows that additional cadmium absorber rods have to be used to assure subcriticality. The following conclusions are summarized below:

- Without any absorbers only 34 fuel elements would be maximum number stored in each fuel rack compartment, since any larger number might lead to criticality under severe earthquake conditions.
- For absorber rods uniformly mixed with fuel elements in the case of postulated accident, only 8 absorber rods per 62 fuel elements would assure subcriticality.
- Since random mixing of fuel elements and absorber rods during lattice disintegration can be expected, the probability of supercriticality was estimated for randomly selected fuel / absorber mixture patterns.
- When there are 20 absorber rods per compartment, the probability for supercriticality is below 10^{-6} for postulated accident, resulting in rack disintegration and unfavorable mode of fuel compaction.

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THE IR-8 REACTOR OPERATION

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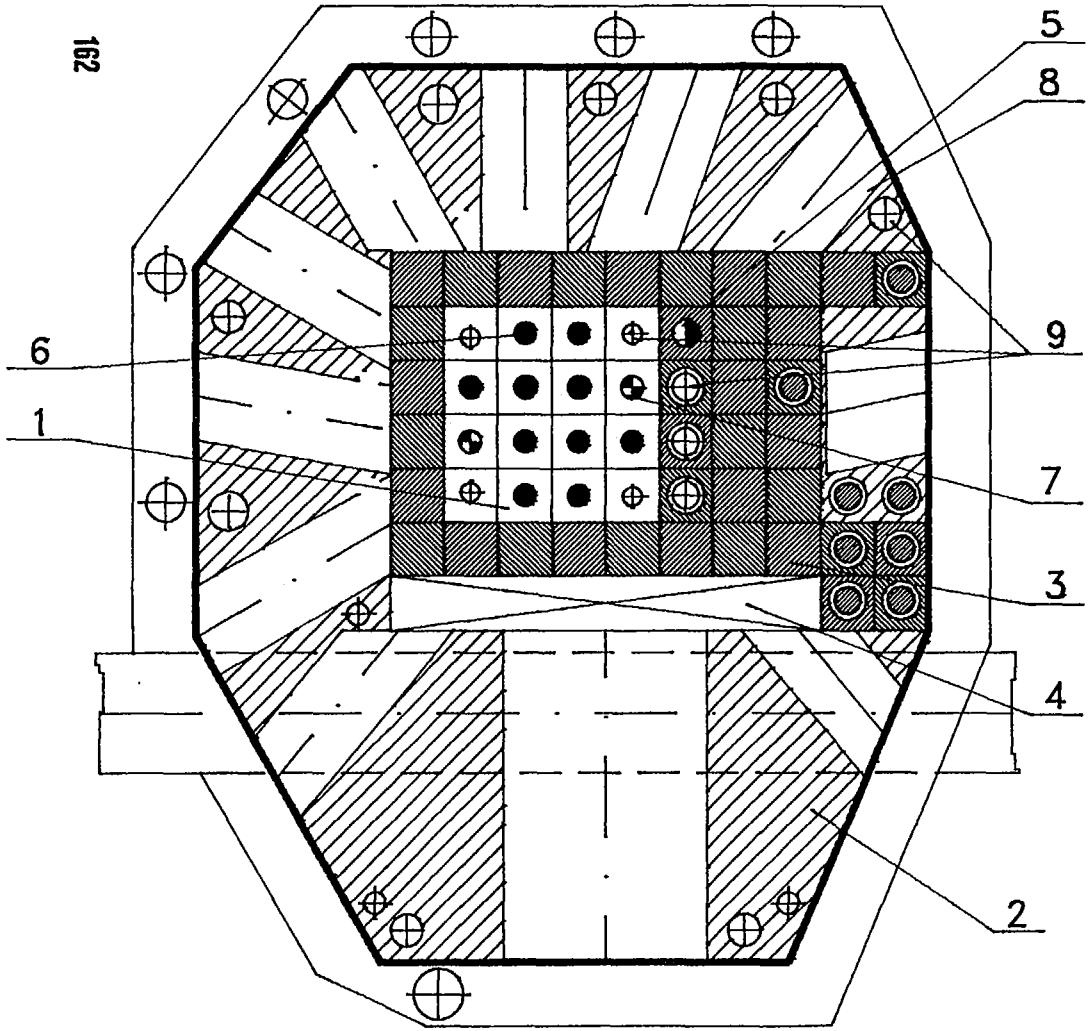
INTRODUCTION

At the Russian Research Center "Kurchatov Institute" (RRC "KI") the IR-8 reactor commissioning was carried out in 1981. [1]. The reactor was developed in return for earlier existing at RRC "KI" of the IRT-M reactor (of modernized the IRT reactor, constructed in 1957).The IRT-M reactor was used for investigations in nuclear physics, solid state physics, radiation chemistry, biology as well as to produce isotops. Under developing the IR-8 reactor the IRT reactor biological shielding with beam tubes and its the process systems were used. The IR-8 reactor creation was founded on application developed by then new fuel assemblies (FA) of IRT-3M type, having two times as great surface of heat transfer and 1.75 times higher U-235 load than the FA of the IRT-2M type, which were used in IRT-M reactor.

FEATURES OF REACTOR PHYSICAL SCHEME AND DESIGN

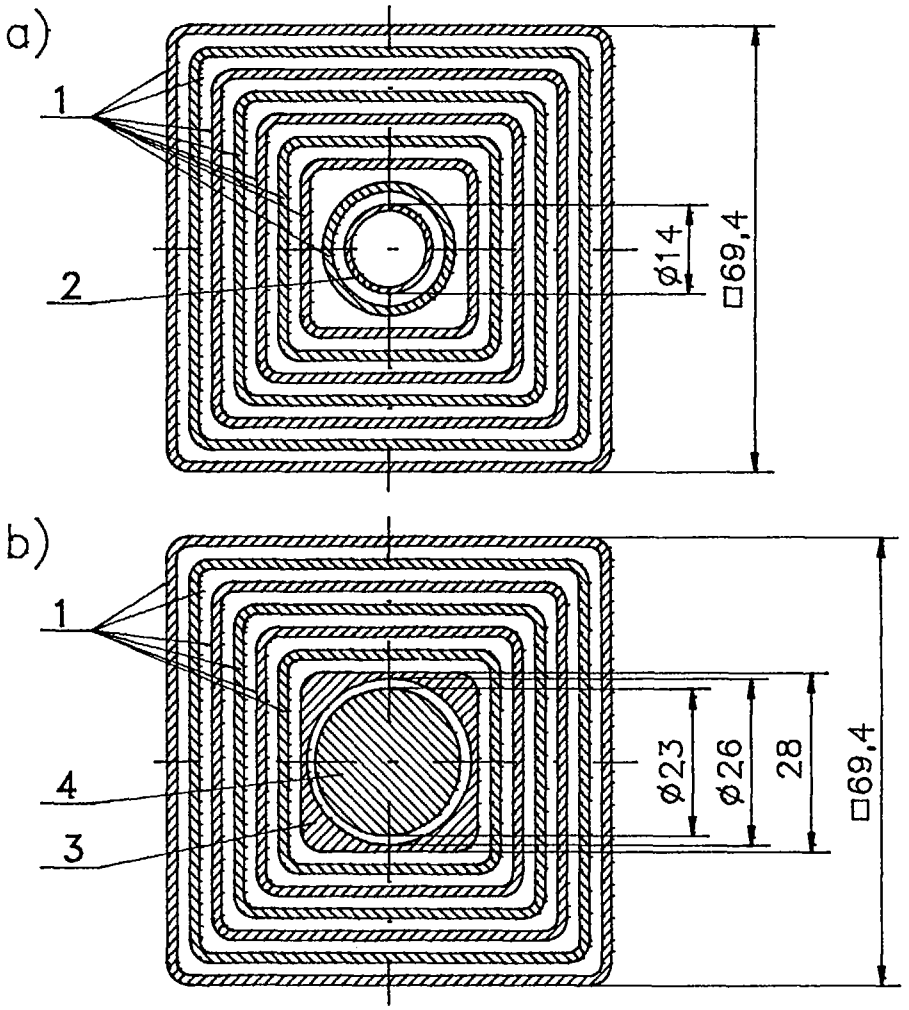
Under developing the IR-8 reactor the task on ensuring thermal neutron flux high density in a geometrically big reflector area and a sufficiently high density of fast neutron flux in the reactor core was posed. In this connection, to use FA of IRT-3M type with a high multiplication factor and small length of neutron migration was decided. It made it possible to obtain a small-sized reactor core and high neutron leaks into the reflector. As it is seen from fig.1, the IR-8 reactor load consists of 16 six-tube FA of the IRT-3M type (fig. 2). Into 12 FA channels with the rods of control and protection system (CPS) are located. Into 4 FA experimental channels are located, in which it is possible to irradiate constructional and fuel materials, individual fuel elements (FE) or isotope targets.

As a reflector on the IR-8 reactor metallic beryllium is used. It was applied also at the IRT-M reactor, but its thickness was insufficient (7 cm) and it surrounded the core not from all sides. In the IR-8 reactor the reflector thickness equals to 30 cm so that it can be considered as physically infinite. The reflector consists of two parts: an internal one formed by removable beryllium blocks and external one formed by prismatic beryllium blocks with holes for beam tubes and vertical experimental channels (a stationary reflector). The reflector is fixed in the vessel made of aluminium.



1 – 6–tube FA; 2 – Blocks of stationary beryllium reflector; 3 – Removable beryllium block; 4 – Lead shield; 5 – Channel with automatic regulating rod of CPS; 6 – Channel with shim–safety rod of CPS; 7 – Channel with safety rod of CPS; 8 – Beam tube; 9 – Vertical experimental channel

Fig.1. Load of the IR–8 reactor core



1 – fuel elements; 2 – displacement tube; 3 – channel; 4 – CPS–rod

Fig.2. Cross section of the IRT–3M fuel assemblies
a) 8–tube FA
b) 6–tube FA

Main parameters of the IR-8 reactor:

Power, MW	8.0
Core volume, l	47.4
U-235 mass in the core, loaded "fresh" FA, kg	4.35
Reactivity margin of the core with "fresh" FA, β_{eff}	29.4
Total reactivity worth of CPS rods, β_{eff}:	
- safety rods	5.7
- shim-safety rods	30.5
Maximum density of neutron flux, $cm^{-2}\ sec^{-1}$:	
- thermal in holes filled with water of removable beryllium blocks of the reflector	$2.3 \cdot 10^{14}$
- fast (E > 3 MeV) in the core	$5.7 \cdot 10^{13}$

All existed at the IRT-M reactor 11 beam tubes are kept at the IR-8 reactor too. In addition a new curvilinear beam tube was designed to extract a beam of ultracold neutrons. Considerable a number of vertical channels can be placed:

- up to 4 channels with an outside diameter of 28 mm or 41 mm installed inside six or four tube FA;
- up to 8 channels with 45 mm outside diameter in removable beryllium blocks;
- up to 11 channels in blocks of stationary reflector with 25, 45 and 54 mm outside diameter;
- up to 6 cahnnels beyond the vessel with 54 mm outside diameter.

THE IRT-3M TYPE FA FEATURES AND RESULTS THEIR USE IN THE IR-8 REACTOR

In the IRT-M reactor from 1968 the IRT-2M type FA with tube of square section were used. FE wall thickness in IRT-2M FA was equal 2mm. In 1972 at RRC "KI" the specification on the IRT-3M type FA were developed . FE wall thickness in IRT-3M type is equal 1.4 mm, FE core thickness of UAl alloy - 0.4 mm, FE cladding - of aluminium alloy CAB-1, gap thickness between FE - 2.05 mm. The FE manufacture technology was developed by the Novosibirsk Chemical Concentrates Plant (NCCP).

Main parameters of the IRT-3M type FA:

- number of FE in the FA, pcs	8	6
- uranium enrichment, %	90	90
- FE core length, cm	58	58
- U-235 loading in FA, g	300	265
- specific loading of the U-235, g/l	100	89
- water volume fraction	0.624	0.548
- specific surface of heat transfer, cm^{-1}	0.525	0.462

The first 16 of IRT-3M type FA were manufactured by NCCP at the end of 1978-begining of 1979. In connection with shut down of the IRT-M reactor at RRC "KI", these FA were loaded into the WWR-CM reactor in Tashkent and into IRT-M reactors in Riga and MEPhI (Moscow). In the IR-8 reactor the IRT-3M FA are used from August of 1981. Before 1986 the IRT-3M FA with FE having core of UAl alloy were used only. In 1985 NCCP ended developing for the IRT-3M FA of FE with core of UO_2 + Al. From 1986 the IR-8 reactor began to operate with mixed core load.

From 1981 before October 1996 in the IR-8 reactor 102 IRT-3M FA were used: 58 FA with FE having UAl alloy and 44 FA - with UO_2 + Al core. In first FA average burnup of U-235 is equal to 42-61% , in second - 44-60%. 64 the IRT-3M FA were sent on a reprocessing and 38 FA are in the pool-storage designed to store up to 120 spent FA.

Now in the IR-8 reactor core 5 the IRT-3M FA with FE having UAl alloy core and 11 the IRT-3M FA with FE having UO_2 + Al core are loaded.
First of operation residence time (lifetime) the IRT-3M FA into the IR-8 reactor amounted 20-21 month, then it was reduced to 13-17 month.

CONCLUSION

Use in the IR-8 reactor of the IRT-3M type FA in course 15 years its operation confirmed their high reliability to average (on volume FA) burnup of U-235 to 60%.

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FUEL AND FUEL CYCLE OF THE SM-2 REACTOR

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I. Brief description

The paper discribes the fuel pin and fuel assembly of the high flux research reactor SM-2. The fuel assembly operating experience is summarised, a brief characteristics of the fuel cycle common for the SM-2 and a complex of the three pool type reactors is given.

2. Introduction

The SM-2 reactor design realized for the first time the concept of an experimental volume with the thermal neutron flux density more than $10^{15} cm^{-2} s^{-1}$ using the neutron trap effect. This concept was successfully embodied through the development of the active core with high energy release density, high fuel content and central neutron trap of the optimal composition and dimensions. This required to resolve some engineering and scientific problems, such as optimization of the fuel pin and fuel assembly manufacturing technology, providing high concentration of uranium-235 in the core, reliable operation at the high (~ 10 m/s) water rate and power density about 10 MW/m^2 as well as the efficient facility operation including the efficient fuel use.

The problem has been solved step by step within several years and mainly during preparation and performance of several reactor reconstructions. The fuel assembly operating experience demonstrated that their burnup potential is higher than it is required for the optimal operating reactor cycle.

Thus, based on this the three pool type reactors-satellites of the Sm-2 have been constructed. These reactors of 6 to 10 MW output use the spent fuel assemblies of the high flux reactor as a fuel. This repot pursues the goal to summarize the operating experience with the highly enriched fuel.

3. Fuel element and assembly

The reactor fuel element was developed by the All-Russian Institute of Inorganic Materials and the fuel assembly was developed by the Scientific Research Design Institute of the Power Technology (Moscow). The manufacturer of the fuel assemblies is the Machinary Plant in Electrostal.

Since 1964 the Sm-2 has been using the fuel assemblies with the selfspacing fuel pins (of rod type) of crucial cross-section (Fig.1). In plan the fuel assembly is a square of side 69 mm, and contains 188 pins placed in the triangle grid junctions at 5.23 mm spacing. The core length of the assembly is 350 mm. Uranium dioxide dispersed in the matrix of the copper-berillium alloy is used as the fuel element core material. The uranium enrichment in uranium-235 is 90% and the uranium-235 mass in one pin is 5 g. The material cladding is stainless steel 0.15mm thick.

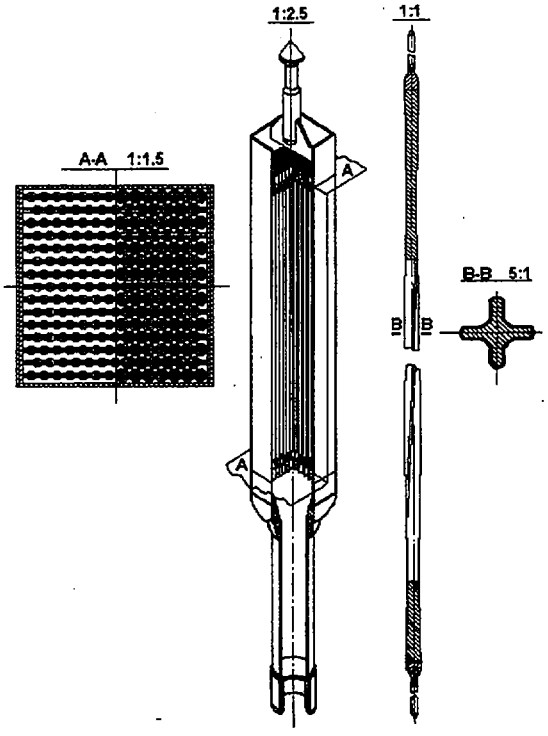

Fig.1. Reactor SM-2 fuel rod and fuel assembly

The pin sircumdiameter is 5.15 mm. In rolling, the pin is given crucial shape simultaneously with twisting around the axis at 300 mm spacing. This provides:

- proper coolant mixing of the "cold" and "hot" cells, decrease of the specific enthalpy in the "hot" spots
- increase of the water flow turbulence and heat-transfer coefficient;
- selfspacing in assembling and bending stress upgrading.

The melting temperature of the cladding material is 1400°C, and of the matrix material - 1000°C. Because of the high core thermal conductivity the temperature drop between its centre and periphery at the nominal rate does not exceed 250°C . The fuel pin surface is cooled by the water under pressure of 5.0 MPa and 50°C at the fuel assembly inlet. The water rate in the most intensive fuel assemblies reaches 12 m/s.

The average uranium-235 burnup in the discharged fuel assemblies reached 30 to 35% and the maximum burnup reached 80%. The microcompaign duration is 10 -12 days. The fuel achieves 30-35% of burnup after irradiation for 8- 9 microcompaigns.

When operating at 100 MW, the average thermal flux density from the fuel pin surface is about 2.6 ± 3.0 MW/m^2.The special tests demonstrated that the heat transfer crisis occurs at the power density equal to 14 - 15 MW/m^2. Moreover it is confirmed that it is possible to use fuel pins at power density up to 11.5 MW/m^2 achieving , in this case, about 33 -40% of burnup.

Based on the information about the failures gained for the last 15 years, the evaluation of the fuel assembly realibility has been made. Within the total operating period of the fuel assemblies equal to 2.6 x 10^6 h. only 5 failures have been registered that corresponds to the failure intensity
2.0 x 10^6 h^{-1}.The material science examination of the fuel assemblies discharged from the reactor in connection with the high parameters of the cladding integrity control system demonstrated that in the most fuel assemblies only one fuel pin was found leaking. The defects were mainly the small cracks (1 -2 mm long) in the socket that is related to the imperfect manufacturing technology. Thus the share of the failed fuel pins does not exceed 5 x 10^{-4}.

4. Fuel cycle

The reactor core / 1 / is composed of 28 stationary and 4 movable fuel assemblies immediately adjacent to one another in two rows forming a cavity of a 140 mm square in the centre (Fig.2). The movable fuel assemblies are placed in the angle cells and together with an absorber placed above them are used as control rods. The central shim rod is displaced in the gaps between the berillium inserts and central berillium block and is removed at the beginning of the campaign to compensate the loss of reactivity for poisoninig.

As the most research reactors the SM-2 operates with partial fuel refuellings. Inside the reactor vessel there are storages to keep the fresh fuel assemblies at the beginning of the campaign.

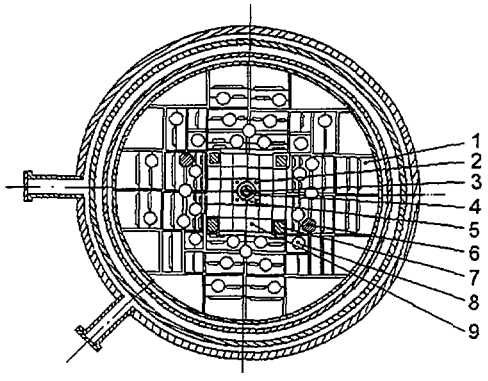

Fig. 2. Sectional view shceme of the SM-2 core.

1.Reflector beryllium block.
2.Safety rod.
3.Central beryllium block.
4.Central shim rod.
5.Beryllium insert.
6.Rod of automatic regulation system.
7.Peripheral control rod.
8. Experimental channel. 9.Fuel assembly.

With use of the fresh fuel assemblies, the spent fuel assemblies are loaded into the storages. During refuelling the in-vessel pressure is not released up to the complete use of the fuel assembly reserve. The amount of the fresh fuel assemblies is sufficient for 5 microcampaigns of 10 -12 days duration. After that the reactor is shut down for 6 days and the core is cooled up to the condition at which it is possible to decrease the pressure in the primary circuit up to the atmosphere pressure, to discharge the spent fuel assemblies from the storages and core and to load the fresh ones. The time capacity factor is 0.765, taking into account the shutdown for the scheduled 30 day repairing. The spent fuel assemblies, the annual amount of which achieves 100 , are stored in the pool in the reactor building and after one year cooling they are used as a fuel in the RBT-6, RBT-10/1 and RBT-10/2 reactors.

A large content of uranium-235 in the spent fuel assemblies and their reliable operation even at the burnup exceeding the average burnup of the discharged fuel assemblies provided prerequisites for construction of a set of pool type reactors using this fuel /2/. These reactors are used for investigations that do not require high rate for neutron fluence accumulation but can require high stability of the testing parameters. The required reactivity margin is provided therefore through the increase of the core volume and location of the fuel assemblies at a 9 mm water gaps that are used also for location of the shim rods loaded from the lateral side. This improves the core access for the experiments. The long continuous reactor operation (up to 200 days) is provided by the low core power density and burnup of samarium-149 accumulated in the SM-2 and acting as an burnable absorber in the RBT reactors. The total amount of the fuel assemblies in the RBT-6 core is 56 and in the RBT-10 - 76. The average burnup of uranium-235 in the loaded fuel assemblies is 30% and in the dischaged fuel assemblies - 47%.

The fuel assemblies discharged from these reactors are stored in the pools for not less than three years and then they are transported to the central storage of the Institute. The chemical reprocessing of the fuel assemblies is performed at the integrated plant "Mayak".

5. Conclusion.

More than 30 years operating experience of the SM-2 fuel assemblies demonstrated their ability to provide the efficient reactor operation. The use of the highly enriched uranium is justified by the necessity to have the high neutron flux density experimental channels in the reactor with rather low power. It would be impossible to locate the required amount of uranium dioxide enriched in 20% in the fuel pin core volume even if it were not required to increase the amount of uranium-235 in the core when lowering the enrichment.

Due to the high rate of neutron fluence accumulation it became possible to conduct the accelerated radiation tests of nuclear material samples, to obtain practically applicable amount of the far transuranium element isotopes and samples of the "light" radionuclide-based preparations having high specific activity. This provided an opportunity to make further improvements of the NPP equipment, to make some technological progress in treatment of the severe diseases, to improve the process monitoring methods in industry, etc.

The unified fuel cycle including the high flux reactor and three reactors of the small power increased the performance of highly enriched fuel.

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Design and experience of HEU and LEU fuel for WWR–M reactor

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A research reactor for providing high neutron fluxes has to have a compact, well breeding core with high specific heat removal. The WWR–M fuel elements meet these demands. They have optimum metal-to-water ratio and the recordly developed specific heat–transfer surface providing in a pool–type reactor at atmospheric pressure the unit heat of (900±100) kW/l [1]. High ^{235}U concentration in fuel elements decreases fuel consumption in carrying out physical experiments. The WWR–M5 assemblies have appeared after long research of jointless tabular fuel elements of the WWR–M reactor which were proposed in 1956. Characteristics of WWR–M assemblies are presented in Table 1 and their performance — in Table 2. (Water temperature at the input of the core is equal to 50°C, its velocity does not exceed 4 m/s). All the WWR–M assemblies are exchangeable. They are collected in a lattice; the lattice–hexagonal sell size being equal to 35 mm. Such size of the assemblies allows one to rearrange the core flexibly what is convenient by the placement of experimental devices in the core. Single WWR–M1 and WWR–M2 assemblies consisted of three coaxial tubular fuel elements. A single WWR–M5 fuel assembly has five tubular fuel elements and a central rod. The end spacing elements of the assemblies have step–type construction for decreasing hydraulic resistance. The active parts of the assemblies have no carrying details, this is an advantage before assemblies of other types (for example, MTR–cassettes). The WWR–M5 assemblies of the last lots have ribs (height=0.5 mm) on the facets of the outward fuel element to guarantee a warrant clearance[2]. The fuel elements jackets have been made of SAV–1 aluminium alloy, meats have been made as of U–Al alloy as well of UO_2 dispersed in an alumoceramic matrix. The WWR–M2 fuel elements are used in Ukraine, fuel elements with an active length of 60 cm (WWR–SM) are used in Hungary, Vietnam and Poland. The core with WWR–M5 fuel elements at 90% enrichment is saturated by fuel sufficiently high, and further increasing fuel concentration raises the value of $K_\infty = 1.79$ relatively insignificantly[3]. The use of meat with higher ^{235}U concentration will not influence practically physical processes

in the core. By using fuel of less enrichment one should increase uranium concentration in the elements for maintaining the reactivity margin. Some experimental WWR–M2 and WWR–M5 assemblies with reduced enrichment and enlarged meat uranium density have been produced and tested to study this problem (see Table 1). We found that characteristics of the WWR–SM (WWR–M2) fuel elements at the ^{235}U concentration in core being equal to 100 g/l have been improved even at the decrease of the enrichment to 20%. During many years the permanent check–up of the fuel elements hermeticity is carried out on the WWR–M reactor. Parameter at unhermeticity is defined as a ratio of the Xe and Kr fragment activity yield into the heat–transfer medium rate to the rate of their equlibrium generation in fuel[4]. This parameter for new assemblies turned out to be equal to $1 \div 3 \cdot 10^{-7}$ mainly for account of surface pollution. After the mean power production per an assembly became $> 20 MW \cdot day/assembly$ it increased up to 10^{-6}. Such yield of Xe and Kr practically does not influence dosimetry situation as the activity of the air over the reactor tank and of the air discharged into the atmosphere (which is determined up to (70÷90)% by the activation of the ^{41}Ar contained in the air) is well below the permissible levels. The fragment yield for the experimental assemblies with reduced enrichment differs from the yield for the operational assemblies insignificantly. In conclusion we note that the WWR–M assemblies allow raising the capacity of a pool–type reactor with an open water table up to 30 MW[5].

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Table 1: Characteristics of WWR–M assemblies

Assembly type	^{235}U enrichment, %	Fuel element wall (meat) thickness, mm	Specific heat-transfer surface, cm^2/cm^3	Composition	Central rod uranium density, g/cm^3	^{235}U concentration in core, g/l
WWR–M1	20	2.3(0.9)	3.67	UO_2+Al	1.55	48.8
WWR–M2 (WWR–SM)	36	2.5(0.7)	3.67	U+Al	1.33	61.2
WWR–M5	90	1.25(0.53)	6.6	U+Al	0.77	125
	90	1.25(0.39)	6.6	UO_2+Al	1.2	125
WWR–M2E*)	36	2.5(0.9)	3.67	UO_2+Al	2.3	122
WWR–M5E*)	36	1.25(0.43)	6.6	UO_2+Al	2.14	102
	21	1.25(0.43)	6.6	UO_2+Al	3	83

*) Number of the tested experimental assemblies is: WWR–M2E till burnup of 43% — 9 assemblies; WWR–M5E (enrichment = 36%)$till burnup of 44\% - -$ $-3 assemblies; WWR--M5E(enrichment$=21%) till burnup of 57% — 9 assemblies.

Table 2: Performance of WWR–M assemblies on the WWR–M reactor of PNPI

Characteristic	WWR–M1	WWR–M2	WWR–M3	WWR–M5
Operation period	1959–63	1963–79	1973–80	1980–1.07.96
Reactor capacity, MW	10	16	18 *)	
Mean (maximum) burnup in unloaded assemblies, %	47(76)	41(91)	28(73)	29(59)
Number of used single assemblies	184	2765	638	2184 **)
Mean power production per assembly, $MW \cdot day/assembly$	9.7	10	7.7	14.7
Total power production, $GW \cdot day$	1.8	28	5	32.1

*) There is a project of increasing the capacity up to 30MW. **) Discounting 156 assemblies with various burnup in the core.

Lead-Shielded Spent Fuel Transport Casks - The Thermal Switch Concept

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INTRODUCTION

There are at least four families of transport casks designs depending on the selected gamma shielding and construction materials : lead casks, depleted uranium casks, forged steel casks, and cast iron casks. These four designs co-exist in the transport cask world market. The purpose of this paper is to provide an historical approach to the lead cask design through ROBATEL 40-year design experience.

IMPROVEMENTS IN LEAD-SHIELDED CASKS DESIGN - THE THERMAL SWITCH CONCEPT

The first spent-fuel transport casks were lead casks. In 1954, ROBATEL manufactured the first lead shielded container for transportation of spent fuel from the French reactor Zoé.

In 1964, the IAEA regulations set design requirements under hypothetical accident conditions and ROBATEL invented the "Thermal Switch" concept to increase the fire resistance of lead casks.

The Thermal Switch is an integrated thermal protection and neutron shielding placed in the lead cask multilayer structure (from the inside out : stainless steel, lead, Thermal Switch, steel). The Thermal Switch properties are : (1) good mechanical resistance to maintain lead geometry under normal conditions, (2) good thermal conductivity to ensure heat evacuation under normal conditions and after hypothetical accident conditions, (3) good resistance to low temperatures, and (4) high thermal capacity to absorb the heat during a regulatory fire.

The first material used as a Thermal Switch was plaster. Increasing thermal payload and neutron shielding requirements led to development along the years of more effective Thermal Switches.

In 1973, ROBATEL performed a fire test on a slice of a 32-ton cask to verify the Thermal Switch thermal properties. The results of this test, validating the concept, were presented to the PATRAM'74 conference in Miami (Bochard, C.M. *Improvements in the field of thermal transfer to type B packagings for transportation of irradiated fuel elements with high residual power*, PATRAM'74 conference).

The next improvement was the development in 1988 of "ROBATEL PNT7" compound. This compound composed of hydrogenated and borated materials binded with a refractory material combines harmoniously thermal protection and neutron shielding functions. The PNT7 compound has the following characteristics : (1) very good stability after a fast aging at normal temperature, (2) 100 to 1000 times better radiation resistance than the best organic materials, (3) incombustibility, (4) good thermal conductivity, (5) no galvanic reaction with steel and lead, (6) very good resistance to freeze/thaw cycles, and (7) very high energy absorption capabilities. Copper fins embedded in the PNT7 compound improve thermal conductivity by a factor of 10, thereby increasing heat evacuation capabilities.

The application of the Thermal Switch concept to lead cask design was validated by extensive testing and analysis during development of the BR-100 cask, a 100-ton rail-barge cask designed for the U.S. Department of Energy in cooperation with Babcock & Wilcox and presented to the PATRAM'88 conference in Washington DC (Mc Guinn, E.J., Childress, P.C., Bochard, C.M. *BR-100 spent fuel shipping cask*, PATRAM'88 conference).

CONCLUSIONS

Lead casks provided with the ROBATEL Thermal Switch are a viable alternative to transportation. The Thermal Switch concept has been validated by testing and analysis. It has been extensively used along the years with 66 type B casks licensed by ROBATEL between 1965 and 1996.

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EXPERIENCES FROM OPERATION, SHUT DOWN AND DISMANTLEMENT OF THE MTR-RESEARCH REACTOR SAPHIR

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1. Introduction

The swimming pool research reactor SAPHIR (MTR type) became operational in 1957 on a power level of 1 MW. For 37 years is was the strongest neutron source within Switzerland for research purposes and industrial applications. The power level of 10 MW was reached in 1983 following the demands of the different facility users.

In the framework of the RERTR program the different steps of reducing the fuel enrichment were practically applied [1,2]. Because of the relative high U-235 content of the elements (HEU: 280 g/element; MEU: 320g/element and LEU: 412 g/element) the core configuration and the cooling conditions have to be considered very carefully. A number of problems could be solved by improving the loading procedures and by modifications of the cooling circuits.

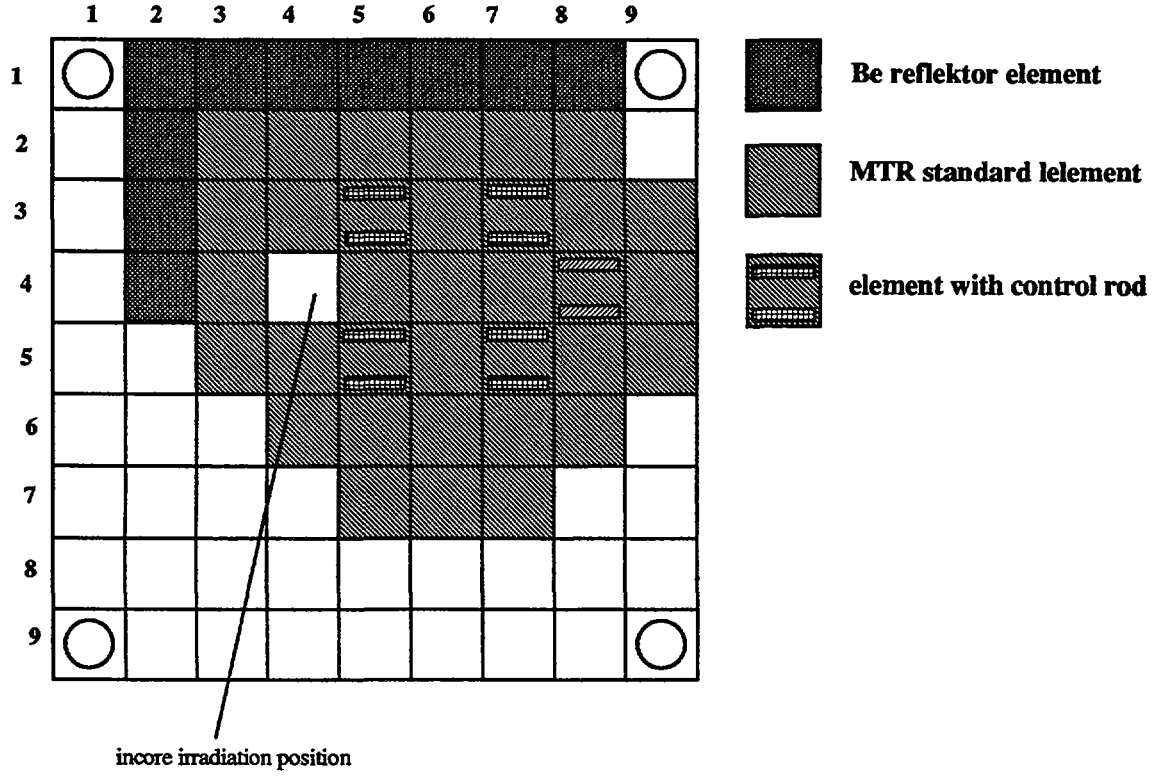

Fig. 1: Core configuration of SAPHIR on a grid plate of 9 * 9 fuel element positions

In 1994 it was decided to shut down the reactor. The owner of the facility (PSI) plans the decommissioning in several steps, starting with shipping all spent fuel elements back to their origin country USA and disposal in the DOE storage facilities. Considerations

regarding criticality safety during storage of the fuel elements in the reactor pool, handling procedures and shipping in suited containers were performed. The decommission of the reactor facility will be done during the next years depending on the demands of the future users of the building and the advances in the licensing procedure.

2. Core design and operation

The operation of the reactor SAPHIR was dedicated to the continuos supply with neutrons on a high flux level for users under the boundary conditions of high fuel burn up and safe operation, regarding radiation protection and emission of activity. In this operation mode, about 6000 hours reactor operation per year could be achieved. The fuel burn up reached between 60 and 70% depending on the type of fuel (HEU, LEU).

A typical SAPHIR core is shown in fig. 1, indicating a partly beryllium reflected core in direction to the neutron beam tubes and a relatively open core to the opposite side. This configuration was not only caused by a lack of sufficient Be reflector elements but mainly due to the possibility of flexibility for irradiation equipments and experiments with fast neutrons. Empty positions inside the core enabled incore irraditions on a high flux level.

2.1. Operation conditions regarding coolant flow

The application of more accurate calculation methods showed unexpected high non uniformities of the power density distribution in LEU-fuel elements next to watergaps due to withdrawn control rod absorbers or water reflected elements. The corresponding heat transfer regime with the given flow rate resulted in a lower safety margin to onset of nucleate boiling and flow instability under conservative assumptions. As a first consequence, the maximum thermal power was reduced from 10 to 6 MW until the implementation of constructive measures to increase the coolant flow velocity in the core.

A higher flow velocity was achieved by increasing the speed of both reactor coolant pumps to their maximum. The higher flow rate in the pool induced stronger turbulences, which destabilised the inactive warm water stratification on the pool surface and caused higher radiation levels. This problem was solved by distributing the coolant flow into the pool via a modified diffusor in a more uniform and laminar regime and by the installation of an electric heater (10 kW) in the warm water outlet at the pool surface.

2.2. Core conversion experiences

Because of the relatively large amount of fuel elements available of all three types of enrichments, the conversion to low enriched reactor cores was not really completed by switching to only one pure core. Indeed, mixed cores were used in routine operation over several years. Having in mind the occurrence of high power densities of new fuel elements (especially LEU), core loading procedures were established based on calculations with the reactor code system ELCOS [3], flux distribution measurements and operational experience in core loadings over many years.

Especially the calculation of the power density distribution provided the possibility to predict „hot positions" in a configuration, which was corrected in most cases by shuffling of some elements. In the last years of SAPHIR operation, nearly all core loadings were precalculated, before the operation on power level started. Most problems occurred near the water filled incore positions and at the core boundary, when fresh fuel was in direct

contact with the water reflector. At these positions the local coolant flow at the plates is reduced additionally because of the missing side plate of an adjacent element.

3. Utilisation of the facility and experiments at SAPHIR

The research reactor SAPHIR was operated as a multipurpose facility for neutron scattering experiments (5 spectrometers at beam lines with thermal neutrons), different irradiation experiments, neutron activation analysis, isotope production, neutron radiography and a radiochemistry facility [4]. Furthermore, it was a tool for education of students and the staff of nuclear facilities (mainly power plants) during the shut down periods.

Irradiation were performed for silicon doping under commercial contracts for ingots up to 4" diameter. The irradiation experiments were carried out for investigations of the material behaviour of structural materials of nuclear power plants and fusion reactor walls as well.

Most of the produced isotopes were : I-131,Co-60, Na-24, Sm-153, Re-186, Y-90, Ag-111 and Ta-182. Techniques and special capsules were developed for the generation of this different materials.

4. Problems after the shut down of the reactor

4.1. Criticality safety

After the decision to shut down the reactor permanently, most of the fuel elements from years of operation were stored in the reactor pool. Whereas a lot of them reached the burn up limits, a total amount of low burned up fuel, suited for about two critical loadings remained in the pool also. Therefore, the storage, movement and reloading had to prescribe on the basis of criticality considerations. It was found that the storage racks at the pool wall could be used without restrictions. To be sure that fuel elements couldn't be placed in a reactive configuration on the reactor grid plate it was locked by a special mechanical construction.

A minimum of about 6 licensed persons, familiar with the reactor and the fuel movement operations is necessary for all procedures with fuel, including the transportation to the final disposal.

4.2. Inventory of fuel and fission products

It is planned to ship all spent fuel elements to the Savannah River repository facility within the framework of the Environmental Impact Statement (EIS) of the Department of Energy of the USA. A first shipment of SAPHIR elements reached the USA in October 1996. The final shipment is scheduled for early in 1997. The specification of the fuel inventory was done also by calculations based on the code ORIGEN-2 under consideration of the fuel burn up history. Measurements of burnup were done in the past by gamma spectroscopy and by reactivity methods [5].

4.3. Transportation casks

For shipments of the spent fuel elements, different types of certified containers are in use world-wide. The residual SAPHIR elements are foreseen to be transported with TN-7/2 containers of the German company Nuclear Cargo Service (NCS). In the past, this casks were used for HEU elements only. To get the licence for using the containers for elements with lower enrichments and mixed loadings with elements of different enrichments, criticality investigations had to be performed. This calculations were done by means of the

2D transport code BOXER of the code system ELCOS, which is validated for such purposes.

It could be confirmed that TN-7/2 containers, allowing transportation of up to 64 MTR fuel elements, do not exceed the prescribed criticality limit (0.95) under standard and accident conditions. Contrary to the behaviour of infinite compositions of MTR elements in light water where HEU configurations show higher k_∞ than LEU, for the transport container geometry the effective eigenvalues of LEU reach the highest values. Furthermore, it could be shown that any change of the distance of the fuel plates (by external or internal forces) will enlarge the margin to criticality.

4.4. Storage facilities

Some investigations were done for compact storage facilities suited for MTR elements. This storage racks with absorbing side plates made of borated steel, could be very flexible and allow infinite storage volumes of fresh and burned MTR elements.

As there is no further demand for SAPHIR fuel storage in the future the storage racks were not manufactured. However, the results of the investigations and the design of the casks are available for other users. The borated steel plates are available for potential users.

5. Strategies of the dismantlement

After all fuel will be removed from the reactor pool, only the activation product of the reactor installations and experimental equipments has to be considered. From this moment on there is no potential of a nuclear excursion but only the radiation protection problem of irradiation by activation products stored in the pool. Some beryllium reflector elements could be used in other reactor facilities, but their radioactivity has to be considered in the case of transportation.

The level of activation and the amount and categories of activated materials will be analyzed by calculations and measurements as well. After those inventory estimations, the path of each individual component will be specified and the disposal procedure proposed to the safety authority. In parallel to the dismantlement of components, the pool water quality has to be maintained continuously in order to avoid corrosion.

6. Conclusions

Practical experiences with mixed MTR cores during operation on a high power level and the handling of the fuel after the shut down were demonstrated supported by calculations of the different fuel assemblies. From this investigations, some general evaluations concerning MTR fuel could be derived.

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Concepts for the Interim Storage of Spent Fuel Elements from Research Reactors in the Federal Republic of Germany

by

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1. Introduction

Research reactors have been operated in the Federal Republic of Germany since the late fifties. These are Material Test Reactors (MTR) and Training, Research and Isotope Facilities of General Atomic (TRIGA). A total of seven research reactors, i.e. three TRIGA and four MTR facilities were still in operation at the beginning of 1996. Provisions to apply to the back-end of the fuel cycle are required for their continued operation and for already decommissioned plants. This was ensured until the end of the eighties by the reprocessing of spent fuel elements abroad.

In view of impending uncertainties in connection with waste management through reprocessing abroad, the development of a national back-end fuel cycle concept was commissioned by the Federal Minister of Education, Science, Research and Technology in early 1990. Development work was oriented along the lines of the disposal concept for irradiated light-water reactor fuel elements from nuclear power plants. Analagously, the fuel elements from research reactors are to be interim-stored on a long-term basis in adequately designed transport and storage casks and then be directly finally disposed without reprocessing after up to forty years of interim storage.

As a first step in the development of a concept for interim storage, several sites with nuclear infrastructures were examined and assessed with respect to their suitability for interim storage. A reasonably feasible reference concept for storing the research reactor fuel elements in CASTOR MTR 2 transport and storage casks at the Ahaus interim storage facility (BZA) was evaluated and the hot cell facility and AVR store of Forschungszentrum Jülich GmbH (KFA) were proposed as an optional contingency concept for casks that cannot be repaired at Ahaus /1/. Development work was continued with detailed studies on these two conceptual variants /2/ and the results are presented in this paper.

2. Reference Concept for the Storage of Research Reactor Fuel Elements at the Ahaus Interim Storage facility (BZA)

The Ahaus interim storage facility completed by Brennelement-Zwischenlager Ahaus (BZA) GmbH in late 1990 was originally intended for the interim storage of spent light-water reactor fuel elements until reprocessing.

After the abandonment of national reprocessing and termination of the precedence of reprocessing over direct final disposal and in view of commitments to accept reprocessing waste, the interim store is today intended to fulfil a temporal and technical buffer function until direct final disposal of the spent fuel elements from light-water reactors and for other spent fuel elements which cannot be reprocessed for technical and economic reasons.

2.1 Use of the BZA Interim Storage Hall

The BZA interim storage hall has a surface area of approx. 7700 m^2 and comprises two storage areas with the reception and maintenance area located in between. Each storage area is divided into 16 standard and 8 edge segments. The capacity of the hall was first designed for the interim storage of a maximum of 420 LWR fuel element casks of the types CASTOR Ia, IIa and Ic (class 1, $P_{th} > 5$ kW).

For the interim storage of spent fuel elements from the high-temperature reactor (THTR) in CASTOR THTR/AVR transport and storage casks (class 2, $P_{th} \leq 5$ kW) it was possible to fill one standard segment with 64 casks stacked on two levels due to the small dimensions and mass of this type of cask.

For the storage of spent research reactor fuel elements in CASTOR MTR 2 transport and storage casks (class 2, $P_{th} \leq 5$ kW), twenty transport aisles with four setdown areas each can be utilized between the standard segments. The CASTOR MTR 2 casks can be stacked on three levels due to their low overall height and mass. This provides a total storage capacity for 240 CASTOR MTR 2 casks, which is clearly more than the required storage capacity estimated in /1/ up to the year 2010 for the spent fuel elements arising from all research reactor facilities (Fig. 1).

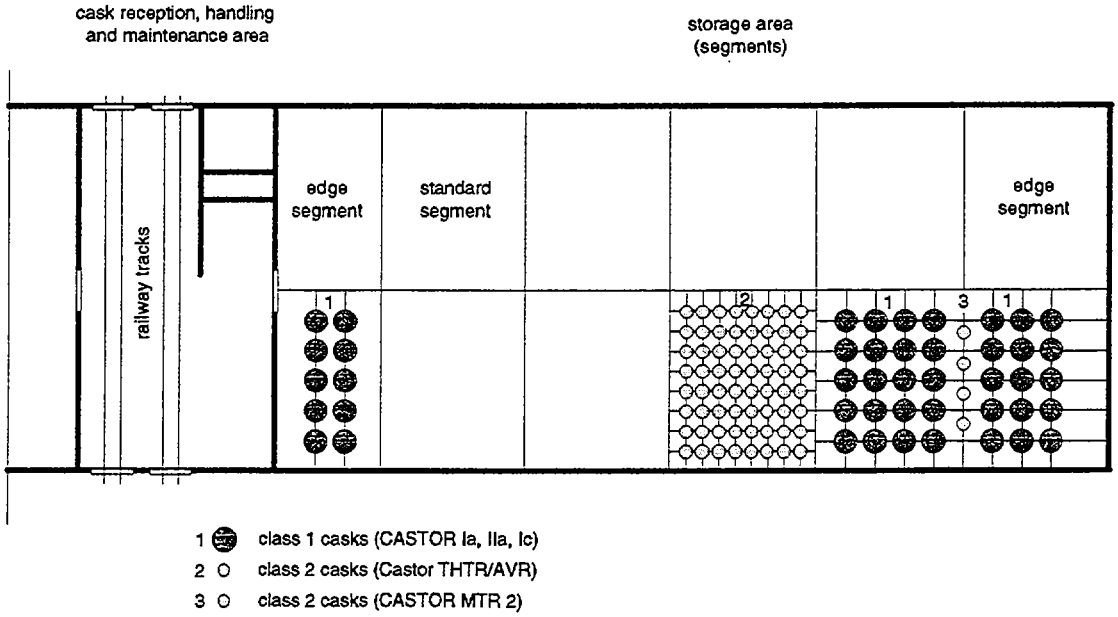

Fig. 1: Simplified ground plan of one storage area of the Ahaus interim storage hall

Licensing under road traffic law of the CASTOR MTR 2 transport and storage cask as a type B (U) package was applied for in December 1993 and the § 6 AtG (Atomic Energy Act) application for the storage of spent research reactor fuel elements in CASTOR MTR 2 casks at BZA was filed in September 1995.

2.2 BZA Safety and Repair Concept

The safety concept for the interim storage of spent fuel elements at BZA is based, in particular, on design requirements for transport and storage casks as a tight enclosure so that any undue release of radionuclides is excluded both in normal operation and under conceivable accident conditions. According to design, the sealing function of both lid barriers is monitored during storage so that any deterioration or failure of a sealing barrier is detected and repair measures for restoring the two-barrier system can be initiated.

Only in the exceptional case /3/ that the functional loss of both sealing barriers is detected and the cask concerned must be opened and emptied for repair can such repair not be carried out at BZA. The incidental conditions of the currently still valid §6 AtG licences therefore contain the demand for removal of a defective cask from BZA and transport to another nuclear facility suitable for repairs. For this reason, the reactor operators are already contractually obliged to retransfer and furnish proof of the transfer to another nuclear facility prior to the shipment of fuel elements /4/.

The aspects of the obligation to retransfer and possible solution approaches are assessed with a view to responsibilities under atomic energy law. Since the incidental provisions of the § 6 AtG notice of approval issued for the Gorleben fuel element store (BLG) in June 1995 no longer contain any requirement to furnish proof of retransfer in the case of repairs, BZA GmbH also expects this regulation for their notice of approval so that the obligation to retransfer in the case of repairs formulated in §4, para. 1a, of the storage contract would become inapplicable.

2.3 Use of the KFA's Hot Cell Facility as a Central Cask Repair Site

In the KFA's hot cell facility, CASTOR THTR/AVR transport and storage casks are currently loaded with canisters containing spent HTR fuel elements (type AVR) and prepared for storage in the AVR store. Both

facilities are part of the KFA's waste treatment and storage building. The installations required for the handling and repair of casks are largely available. Any operating means, adaptation measures and modifications additionally required for research reactor fuel element handling, fuel element support and lifting systems and CASTOR MTR 2 casks can be procured and installed or carried out with reasonable technical expenditure.

Even if it can be expected that the incidental provisions of the BZA licence will no longer contain any requirement to furnish proof of the availability of external repair facilities, a use of the hot cell facility for the exceptional cases described in the BZA repair concept could be maintained as an option. The incorporation of the BZA exceptional repair cases into the KFA repair concept would permit the opening and emptying of MTR 2 casks and would make it possible to avoid such repair measures as the addition of a welded lid at Ahaus.

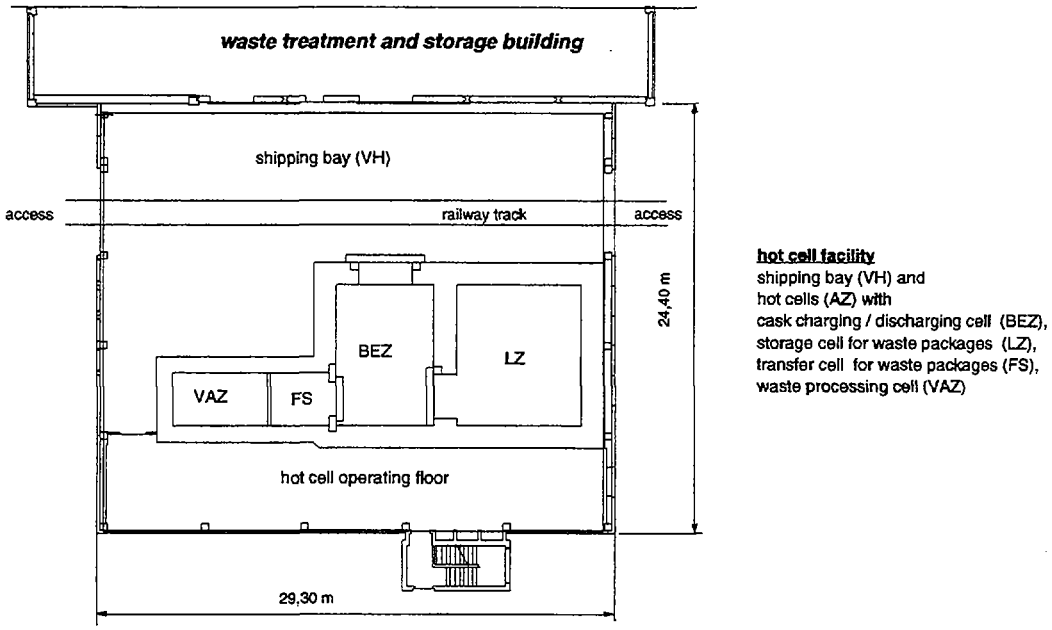

Fig. 2: Simplified ground plan of the hot cell facility as part of the waste treatment and storage building

3. Contingency Concept for the Storage of Research Reactor Fuel Elements in the KFA's Waste Treatment and Storage Building

3.1 Investigation of Alternative Interim Storage Possibilities

In examining and assessing alternative interim storage possibilities it was first checked from the statics aspect whether the available storage areas in the storage hall of the waste treatment and storage building are suitable for the interim storage of CASTOR MTR 2 casks. The result was negative.

Since the CASTOR THTR /AVR casks in the AVR store are alternately stacked on one and two levels and since 56 setdown positions on the second level remain free when the store is completely filled, it was checked again with a view to the permissible floor load whether these setdown positions could be used for the storage of CASTOR MTR 2 casks. This result was positive.

Stacking of the CASTOR MTR 2 casks on three levels causes a floor load approximately comparable to that of the CASTOR THTR/AVR cask stack. The height in the case of stacking the MTR 2 casks on three levels is approx. 5.3 m and thus lower than that of a THTR/AVR double stack of approx. 5.7 m, so that positioning with the existing bridge crane is possible.

In the case of a partial use of the reserve setdown area, which for reasons of store charging and possible repairs is not filled, sufficient storage capacity is available from a technical point of view.
However, complete filling of the second level and filling of the reserve setdown area is not proposed. In that case, the cask monitoring system would have to be modified to comply with the additional number of casks.

Apart from possible regulatory aspects, there are also operational aspects speaking against an extended use of the AVR store for the interim storage of fuel elements from all research reactor facilities.

3.2 Interim Storage of MTR Fuel Elements from the KFA's FRJ-2 Reactor in the AVR Store

According to KFA demand assessments, a maximum of twenty-four CASTOR MTR 2 casks will be needed up to the year 2010 for disposal of the MTR fuel elements from the KFA's FRJ-2 reactor (DIDO). On account of this small number, a use of the AVR store offers itself for the storage of these fuel elements.

The casks could either be set down in free edge positions or on the reserve setdown area in the case of stacking on three levels, involving reasonable extra operating expenditure.

3.3 Interim Storage of the Fuel Elements from all Research Reactor Facilities in an Extended AVR Store

A storage area of 55 setdown positions can be provided by extending the AVR store by two building axes. The setdown area can then be defined and the base statically designed as in the existing store. Should this contingency concept be applied, the storage arrangement of the THTR/AVR casks can be adapted to the constructional modifications to comply with the operation cycles. Moreover, the KFA's own MTR fuel elements can then also be interim-stored there.

4. Conclusions

According to the reference concept, it is intended to temporarily store the fuel elements from German research reactors in CASTOR MTR 2 transport and storage casks in the fuel element interim storage hall at Ahaus. The licensing application according to §6 AtG has been filed by BZA GmbH.

Since the BZA repair concept does not provide for the opening and emptying of casks to restore tight enclosure in the event of a possible loss of the casks' sealing function, the incidental provisions of the §6 AtG licences currently valid for BZA contain the demand for removal of defective casks from BZA and transport to another nuclear facility suitable for repairs. The reactor operators are therefore obliged, in accordance with the currently valid BZA storage contract, to retransfer such casks and to furnish proof of the transfer to another nuclear facility.

The studies deal with the legal aspects of the obligation to retransfer and demonstrate in describing the reference concept that the KFA could assume the function of a central cask repair site. Suitable nuclear installations for handling and repairing CASTOR MTR 2 casks are largely available in the hot cell facility. Installations additionally required for handling the research reactor fuel elements as well as technical modifications to the fuel element baskets of the MTR 2 casks are described in the concept.

The studies for the description of a contingency concept as an alternative to the reference furthermore show that the option for interim storage of the MTR fuel elements from the KFA's FRJ-2 reactor (DIDO) in the existing AVR store is given due to free storage capacity and that the option of an interim storage site as an alternative to Ahaus could be opened up by extending the KFA's AVR store.

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THE REPROCESSING OF IRRADIATED MTR FUEL AND THE NUCLEAR MATERIAL ACCOUNTANCY - DOUNREAY, UKAEA

by

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A. INTRODUCTION

1. MTR irradiated fuel reprocessing has been carried out at Dounreay - since 1958. The most recent campaign was completed as recently as October 1996.

2. The original reprocessing policy arose from the strategic importance of HEU, its high monetary value and supply shortages.

3. In the late 1950's the UKAEA had three HEU fuelled MTRs - two sited at Harwell (DIDO and PLUTO) and one (DMTR) at Dounreay. Similar low power reactors were located at other UK government research and educational establishments. The fuel element manufacturing facilities were sited at Dounreay together with the reprocessing plant and related support service facilities. A co-located MTR fuel cycle system was consequently available from the earliest days. The only external fuel requirement was for a supply of very highly enriched uranium for 'top-up' replacement of the burnt-up U^{235} and process losses. External component and material suppliers have been very much involved in the non-fuel component supply and in the technological developments.

4. Over the last four decades, the Dounreay MTR fuel cycle capability was extended to the commercial contract supply of fuel elements for foreign customers and the provision of a commercial reprocessing service.

5. With the passage of time and the progressive introduction of non-proliferation treaty (NPT) safeguards, the emphasis on the need for reprocessing has shifted. While the original three factors continue to be important, a fourth is that of effective safeguarding - where bringing extraneous, otherwise waste, residues and irradiated fuel together and recovering it through a plant which is Safeguarded to International standards considerably reduces diversion risks. That is so whether the material is placed in secure storage or is reused in new fuel (again safeguarded).

6. An additional issue is the sensible treatment of the radioactive waste - an issue that has been of increasing environmental concern over the years. The essential design features of the MTR fuel are such that some treatment is needed prior to very long-term storage or engineered disposal due to the high fissile content. Reprocessing leads not only to the recovery of reusable fissile material (ie recycling) but thereby leads to waste streams of only trace fissile content. That leads to less restrictive waste disposal/storage requirements.

7. A present strategic issue is a shortage of HEU as the USA have adopted a policy of not releasing HEU onto the commercial market, even to fuel cycles and facilities fully covered by NPT safeguards. Until such time as that policy is amended and until alternative supplies

become available, the HEU fuel requirements are being met in part by the recovery of material from existing stocks.

8. There is scope in the EU to continue to use US origin HEU subject to constraints and end use. This arrangement is embraced within the recent 1996 Euratom : US agreement. That, however, in itself does not lead to a release of new additional material into the market.

9. Consequent upon the international political upheavals and realignments over recent years, there is increasing concern over the security of nuclear materials - particularly HEU - that has become surplus to individual nations' military needs. There are many NPT and diversionary risk advantages in recovering such material and bringing it under full international safeguards. For unirradiated material, the Dounreay HEU recovery plant - the downstream facility associated with the irradiated fuel reprocessing plant - has a role in such an arrangement.

B. THROUGHPUT

10. The initial operations from 1958 were directed at servicing the UK domestic MTRs. Due in part to the excellent performance of the plant, capacity was made available in the early 1960's on a commercial basis to other reactor operators outside the UK.

11. Domestic priorities at Dounreay led to a reduction in foreign reprocessing in the early 1970's. However, due to significant international demands for the service and increased plant availability, foreign reprocessing was pursued more actively again from the late 1970's. Since the closure of the UKAEA MTR project in 1990, there has been increasing emphasis on the provision of this service.

12. From 1973, the Dounreay reprocessing has been carried out under the safeguards surveillance of Euratom who have maintained an on-site inspection presence.

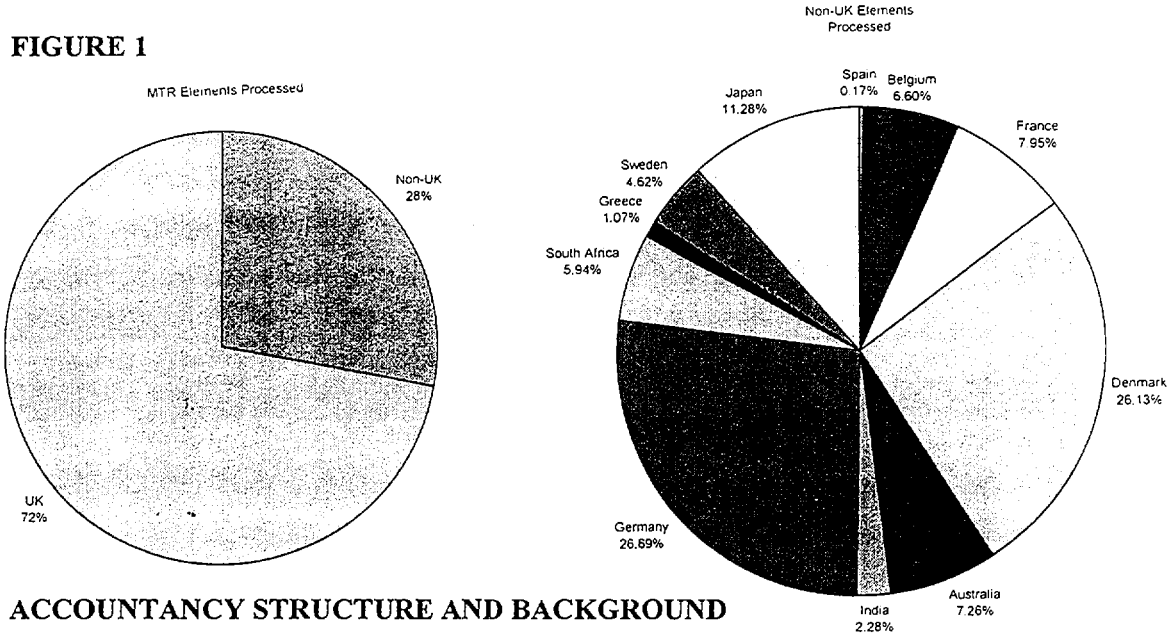
13. The first campaign processed unirradiated reject components from the Fuel Fabrication plant and can be considered to be an active commissioning run. The first irradiated elements were reprocessing in October 1958. Unirradiated material was frequently processed up to 1968. Since 1968 processing of this type of material has taken place in the downstream Billet Production and Recovery Plant.

14. Up to 1962, only fuel from UK reactors was processed. September 1962 saw the first processing (Run 17) of fuel from DR2/3 (Denmark). Fuel from non-UK reactors was processed up to October 1975 (Run 51). This work was carried out for reactors fuelled by US origin HEU either under the EURATOM/US Agreement, for reactors within the EC, or for nominated reactors under the US/UK Memorandum. The recovered uranium was either returned as fuel to reactors, returned to the US, or put to the US/UK Memorandum account.

15. Following closure of the UKAEA MTR project, the first of the second tranche of non-UK fuel was reprocessed in April 1992 (Run 59), which also saw the last UKAEA MTR reactor fuel reprocessing. The run (61) in March 1996 reprocessed UKAEA NESTOR zero energy fuel, the final inventory of the Scottish Universities Research Reactor fuel, fuel from the Australian Hifar reactor, and fuel from the HMI reactor. The latter part of this campaign involved trials on LEU silicide fuel from Petten.

16. In summary, the distribution of the reprocessing from the UK and other countries is illustrated in Figure 1.

FIGURE 1



C. ACCOUNTANCY STRUCTURE AND BACKGROUND

17. Within the Dounreay Site, the unit of accountancy is the 'Plant'. The responsibility for nuclear material, and accounting for it, rests with the 'Plant Manger' with the Analytical Manager and Engineering Manager having responsibility for the analytical techniques and instruments. The Nuclear Material Control Department collates and reports the information generated and provides the Accountancy system structure, training and management. A Computerised Nuclear Material Accountancy System operates to support the control, with the Plant Operators inputting the data. Accountancy and safeguards reports are generated for Euratom (and hence the IAEA), Plant Management, UKAEA Management and UK Government.

18. The accountancy structural nomenclature for the 'Plant' unit has evolved through the years from the initial financial control structure (Cost Centre/Project based) to the present Operational Accountancy Areas (OAAs). An OAA can be considered as a sub Material Balance Area (MBA). At Dounreay, a Safeguards MBA can comprise one or more OAAs. The MTR Reprocessing Plant OAA equates to a single MBA (QDR2).

19. Since 1973, operations at Dounreay have been carried out under the surveillance of Euratom. For the MTR Reprocessing Plant this involves Euratom witnessing transfers from the product tank and the analysis of the liquor. Random samples are also taken for independent analysis at TUI Karlsruhe. Euratom also use the VOPAN 'Blind Analysis' technique for verifying the UK analysis. A recent development has seen Euratom installing data-logger devices and a state of the art Hybrid K-Edge/gamma spectrometer to provide fully independent measurements of product transfer and head-end liquor analysis.

20. In the field of Accountancy and Safeguards, linked to MTR Reprocessing, Dounreay has made contributions to ISO Analytical Standard Methods (Davies and Gray), Tank Weighing (product tank) , Verification of Operator Analysis (VOPAN) and IAEA Inspector training under the UK IAEA Support Programme. These demonstrate Dounreay's commitment to Accountancy and Safeguards.

21. The MTR Reprocessing plant accountancy boundaries are from element receipt at the DMTR Pond (included as part of MTR reprocessing) for element storage, and the issues as the output of the uranium product and the raffinate tanks. The accountancy model is:-

Inputs	**Measurements**
Elements	Shippers data
Excess Analytical Samples	Volume, analysis
Recycled Product Ex BPRP	Volume, analysis
Outputs	**Measurements**
Product Transfers	Volume, analysis
Analytical Samples	Volume, analysis
Waste Liquors	Volume, trace analysis
Solid Waste [1]	NDA
Inventories	**Measurements**
Elements	Shippers data
Tank Heels [2]	Volume, analysis

[1] As the whole element dissolves solid waste is minimal
[2] After wash out.

22. The Dounreay reprocessing plant consists of a head-end above ground pond for fuel handling and preparation for dissolution. There are then two nominal 150ℓ dissolvers operating in parallel. These are followed by conditioning vessels and then a series of transfer vessels leading on to the solvent extraction stages.

23. The uranium aluminium alloy clad fuel is partially dismantled and then cropped into nominal 50 to 100mm lengths. These are charged to the dissolver where they are dissolved in mercuric ion catalysed nitric acid to produce an anion deficient solution - that is a solution containing an excess of aluminium above the stoichometric Aℓ $(NO_3)_3$. The liquor is transferred to the next vessel - the conditioner - where process analysis is carried out. This is particularly important in control accountancy terms.

24. The cropped plates containing not more than a total of 550gms of U^{235} are fed into the dissolver containing dilute boiling nitric acid. 12N nitric acid containing a small amount of mercuric nitrate is fed to the dissolver over a period of about 2.5 hours and the resultant solutions boiled under reflux for about 5 hours. This method of dissolution produces a solution which is deficient in nitrate ion. The solution is transferred to a conditioning vessel where it is sampled and if necessary conditioned to between 2.0 and 3.0 molar aluminium and 0.8 to 1.8N anion deficiency. (AD).

25. A particular issue arises from the higher rate of dissolution of the aluminium than that of the uranium alloy, the latter particularly as the nitric acid becomes depleted due to the aluminium dissolution. That leads to residual uranium being left in the dissolver at each batch. It is for that reason that the subsequent batch is first charged with a 'mop up' volume of nitric acid to dissolve the residual uranium before adding the fresh fuel and the balance of the acid.

26. Process control accountancy of uranium is carried out over the dissolvers and head-end vessels in order to achieve criticality safety. These vessels are not geometrically criticality safe at all foreseeable U^{235} concentrations. It is therefore necessary to maintain a batch-by-batch cumulative assay of material resident in the dissolver in order to avoid an excessive cumulation and an unacceptably high concentration. This is done by sampling and analysing the liquor in the downstream conditioner vessel.

D. ACCOUNTANCY REVIEW

27. The Dounreay accountancy periods are in financial years April - March. There is no direct tie-up between campaigns and accountancy years, however, stocktaking timing is arranged on a flexible basis so that they are carried out at the end of campaigns. In earlier years there were many campaigns in a financial year. Latterly campaigns may straddle accountancy years.

28. In the early days the fissile material accounts were prepared on a quarterly basis, this was relaxed to six monthly in the 1970s in order to better optimise the operational and accountancy programmes.

29. Independently of the formal accountancy and safeguards recording and documented returns, the plant management produce run/campaign accounts to check plant performance and for process and contract control purposes. These employ the same analytical data and are always subject to ratification with the official returns.

30. For all accountancy and campaign returns, particular attention is given to any 'out-of-balance' discrepancy. This feature is identified in safeguards parlance as MUF (Material Unaccounted For) - a positive value is equivalent to a 'gain'. The concept of MUF has been subject to considerable debate - in addition to any unknown losses or gains it embraces error bands associated with assay uncertainties.

31. Reactor calculated data is now taken as the plant input. Therefore, shipper-receiver differences are not reported. The MUF that is reported comprises inaccuracies in fuel manufacture, burn-up, instrumentation measurements, volumes, sampling and analysis. Quality Systems are maintained on all operating features and on all analysis. All instruments and tanks are calibrated and frequently re-checked. Plant operating instructions define mixing times to be used from tank mixing data generated for each tank during commissioning and calibration exercises.

32. Any single MUF is considered in terms of its magnitude and the plant throughput over which it arose. The level of concern is considered against safeguards diversion limits for both the long and short terms. A level of 2% of throughput has been a common 'limit' applied in safeguarding, but attention is now being applied to a new level of concern at a value of 1%. This is in line with the adoption of International Target Values for all measurements and analysis within the Nuclear Industry. (Similar approaches are being carried out in most industries which carry out such measurements). Dounreay is adopting this tighter approach, the first steps being a full review of measurements uncertainties.

33. The cumulative MUF over the life of the plant to date reflects an overall gain (+). This, however, relates to two phases over that time.

Consequent upon this review, the reactor burn-up algorithms were revised to accommodate newer physics data.

35. The situation since April 1970 reflects the revised reactor physics calculations and indicates since then a cumulative MUF significantly less than 1%.

36. These results confirm that up to March 1970, there had been a very significant bias in the input assessment which led to high positive MUFs in the early years. The present MUF levels are consistent with a predominantly random error situation.

37. Occasional process and accountancy difficulties do occur. A 1992/1993 positive MUF was higher than normal and was traced in part to incorrect treatment of the product tank volume calibration data and one abnormal liquor transfer to waste. Insufficient confirmed data was available retrospectively to 'correct' the accounts. A computer program is now in place to provide better control of product transfer recording. A 1993/1994 negative MUF was attributed mainly to the uncertainties in material recycled from the downstream BRP plant. The recycle control arrangements and procedures have been revised to overcome that problem. The most recent campaign saw a low negative MUF within the 1% limit.

38. Overall the MUF assessments to an NMA accountant, suggests there is good control within the plant and the associated measurements of the various input and output streams.

CONCLUSIONS

39. The reprocessing of irradiated HEU MTR fuel is a sensible part of a Safeguards regime. It brings together fuel otherwise scattered around the world into a concerted accountancy and protection arrangement.

40. From a Nuclear Material Accountants view the overall accountancy performance has been excellent. While investigations have been required for a few individual MUFs or trends, very little effort has required to be expended by the Nuclear Materials Control Department. That is a definition of a 'good plant'; it operates, measures and records input and output streams, and then the accountancy falls into place.

41. As identified in this paper, the accountancy of the nuclear material processed in the plant is well founded and sound. The accountancy results over several decades confirm the adequacy of the safeguards arrangements at Dounreay.

42. The processing makes good commercial sense and meets the current philosophy of recycling valuable resource materials. The risks of operating the full fuel cycle are less than those of extended storage of irradiated fuel at disparate diverse locations. The reprocessing at Dounreay accords with all of these philosophies. The assessed risk is at a very low level, well within published UK HSE 'tolerability of risk' regulatory guidelines. The impact of the operations are similarly low within the guidelines, for the operators and for the general public.

An Analysis of Burnup Reactivity Credit for Reactor RA Spent Fuel Storage

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Abstract - The needs for increasing the spent fuel storage capacity and for using higher fuel enrichments have led to the development of a validated methods for assessing the reactivity effects associated with fuel burnup. This paper gives an overview of the criticality safety analysis methodology used to investigate the sensitivity of storage system reactivities to changes in fuel burnup. As an example of the application of this methodology an analysis of the burnup reactivity credit for the three-dimensional model of reactor RA spent fuel storage is described.

I. INTRODUCTION

As nuclear fuel is burned in a reactor, changes in reactivity are arisen from: reduction of fissile content, production of various actinides (some contributing positively toward reactivity, some behaving as neutron poison) and production of fission product poisons. The standard approach for criticality safety analysis of spent fuel transportation and storage problems assumes that all fuel under consideration is unirradiated. This assumption is conservative from a criticality safety point of view (except for some situation of low-irradiated fuel with burnable poisons). As a result of this simplified approach, the treatment needed to evaluate criticality safety to both the fresh and spent fuel storage is the same. Despite providing maximum operator flexibility, this intrinsic conservatism can results in a reduction in the efficiency of these operations, making the enrichment limit found for spent fuel usually more restrictive than the fresh fuel limit. The recent concept[1,2] of allowing reactivity credit for the transmuted state of spent fuel has been incited by the need to reduce the conservativism initially included in the calculations in order to obtain more realistic ("best estimate") results to increase the operational flexibility of the fuel storage facilities. One procedure that have been proposed is to use the major fissile nuclides and actinides without the fission product absorbers.[1] Neglecting fission product poisons has a conservative influence on the calculated k_{eff} (i.e., results in a higher value, approximately for one third of the total burnup reactivity loss). Since the cost of the analysis is proportional to the number of fission product considered, an investigation[2] have been performed to identify nuclides important in criticality analyses and those to be experimentally determined. This resulted in the set of 25 nuclides (referred to as the "burnup credit nuclides") which is recommended for the use in criticality analyses associated with burnup credit.

The initial criticality safety analysis[3] of the spent fuel storage at the Institute 'Vinča' was performed by using the WIMSD4 code,[4,5] which is very suitable for fuel burnup calculations and infinite storage configurations studies. However, this code has been developed for standard geometries, including pin cell arrays and fuel clusters, so that its geometry requirements do not allow for the analysis of some accident configurations, that must be calculated with complicated three-dimensional models. Monte Carlo techniques have merit of dealing with these geometries rather easily. The MCNP continuous-energy Monte Carlo code[6] is one of such codes which is widely used and has been accepted as adequate for criticality assessment. From the calculation point of view, physical phenomenon can be correctly modelled with MCNP, while the consideration of fuel burnup calls for the use of an external code to calculate the evolution with burnup of the fuel isotopic composition. In the current criticality analysis methodology, VEGA-2 two-dimensional transport theory code,[7] based on an extended subgroup method for resolved resonance treatment, is used for this purpose. This paper summarises the burnup credit analysis methodology founded on the use of VEGA-2 code as a complement to the MCNP code and gives some examples of special applications to demonstrate its flexibility.

II. CRITICALITY ANALYSIS METHODOLOGY

The VEGA cell/assembly code, used for burnup calculations is validated for a wide range of fuel enrichment and moderator ratios, including voided configurations.[7-9] An extended subgroup method enables a rigorous resolved resonance treatment in the practical cases where resonant nuclides are located in various media, such as the fuel double heterogeneity problems. A wide variety of situations can be modelled ranging through the thermal reactors, storage systems and items of chemical plant. This include geometry changes such as pin size and pin pitch and the effects of voidage, temperature and fuel burnup, which are an area where the Monte Carlo technique is difficult to use as well as being costly. On the other hand, the MCNP code is very suitable at representing both the spatial and the energy variations. Thus resonance can be accurately represented as well as the intricate detail of the complicated three-dimensional geometry.

Few fission product models for VEGA-2 code were studied and tested with benchmark results. The model containing 45 explicit nuclides and one pseudo nuclide was selected as a reasonably best one to predict the burnup reactivity with high precision for practically all types of fuel and reactor operating conditions. To include the contribution of these fission product absorbers in the criticality calculations based on the MCNP code, a special evaluation represented by the average fission product for U-235 was performed. This evaluation is based on a study of burnup calculations for heavy water reactor RA at the Institute 'Vinča', which indices that only about a dozen fission product absorbers provide 80% of negative reactivity resulting from neutron poison buildup. The calculations also demonstrated that although the relative importance of this poisons varied with time-out-of-reactor (cooling) they continued to be responsible for about 80% of negative reactivity. Since a lower accuracy is permitted at a lower burnup, it was assumed a constant cross section for average fission product throughout the burnup, ignoring its time dependence. The adopted cross section values, composed from explicitly the 10 main nuclides which represent more than 80 % of total poisoning effect (Nd-143, Sm-149, Xe-131, Cs-133, Sm-151, Tc-99, Nd-145, Mo-95, Kr-83, Rh-103) at 10 GWd/t irradiation was chosen as a final values, and corresponding multigroup data for VEGA-2 code and point cross section data for MCNP code were generated with NJOY package.[10] In these burnup credit analyses, the concentration of average fission product absorber, used in the MCNP code, is determined by equating of the total fission product absorption for the VEGA-2 fission product model (with 45 explicit nuclides and one pseudo nuclide) and for effective model with the average fission product. To the date, the current criticality analysis methodology is verified only for the fuel element used in reactor RA. A precise treatment along this line will be continued.

III. RESULTS

As a general rule, input data for the analysis of the reactor RA spent fuel storage are prepared following hypothesis aimed to produce conservative results: geometrical and material composition uncertainties are conservatively considered. Detailed design specifications for the spent fuel storage are modeled as closely as possible.

Research reactor RA (nominal power 6.5 MW) is a heavy water moderated and cooled reactor. The fuel element (slug) is an aluminium cladded hollow cylinder (Figure 1). Temporary spent fuel storage consists of four connected water pools (6 m deep) and one annex pool connected by water passage to the nearest pool. In order to increase the spent fuel storage capacity, some of the oldest metal uranium fuel elements has been taken out of the original stainless steel containers and repacked in 30 sealed aluminium barrels (Figure 2), each containing 30 aluminium tubes receiving up to 6 irradiated fuel elements per a tube. The barrels are filled with demineralised water and placed in two layers in the annex pool. Irradiation of the fuel elements is given in Table I. Cadmium strips were placed in some of the barrels with aim to provide the necessary subcriticality.

In current analysis, the criticality calculations are carried out by MCNP code only for spent fuel elements stored in the barrels. As an conservative approach, it is assumed that each barrel is completely filled with irradiated fuel elements, there are no cadmium strips in the barrels, and
30 barrels are placed in two layers in a hexagonal lattice (pitch 30 cm) that models, as close as possible, an irregular actual position of the barrels in the pool (Figure 3).

Table I Fuel irradiation history

Burnup [GWd/t]	Average burnup [GWd/t]	Number of elements
0 - 2	1.8	100
2 - 5	4.6	1729
5 - 8	7.2	2300
8 - 11	9.9	760
> 11		39

As an example of validation of adopted average fission product model, burnup reactivity credit in single pin cell of the storage barrel, obtained by the MCNP and VEGA-2 codes is shown in Figure 4. All results are in close agreement on the rate of the reactivity lost during fuel burnup.

The results of the MCNP calculation of the reactivity for three-dimensional model of the annex pool of the reactor RA spent fuel storage in function of fuel irradiation (after 30 year of cooling), obtained with the average fission product model and the BMCCS1 nuclear data library are given in Table II. Taking into account the Δk_{eff} margins (due to the applied methodology and geometrical and material uncertainties) it can be concluded that the k_{eff} of the spent fuel in the annex pool of the reactor RA storage has k_{eff} that not exceeded the industry-accepted limit[1,2] of $k_{eff} \leq 0.95$.

IV. CONCLUSION

This paper summarizes main features of the criticality safety analysis methodology used in the Institute 'Vinča' and gives some examples of special application which demonstrate its ability to account burnup reactivity credit for the annular fuel type as used at the RA reactor.

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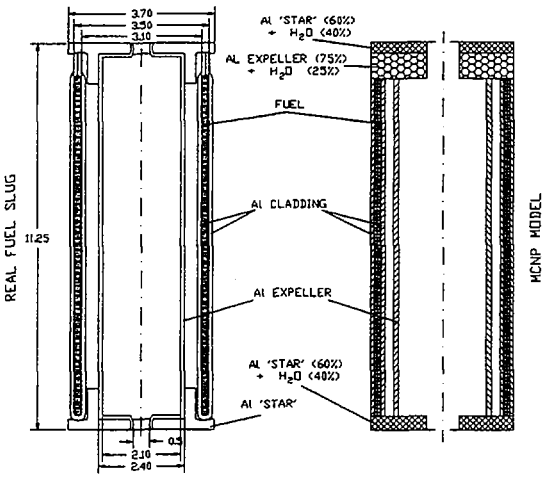

Figure 1 Vertical cross section of the real fuel slug and its MCNP model

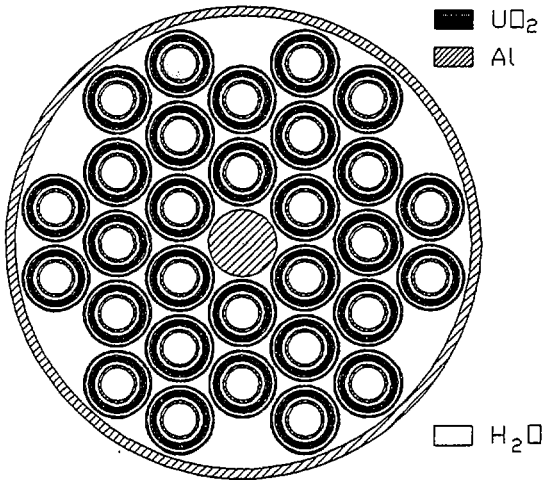

Figure 2 Horizontal cross section of the storage barrel for irradiated fuel

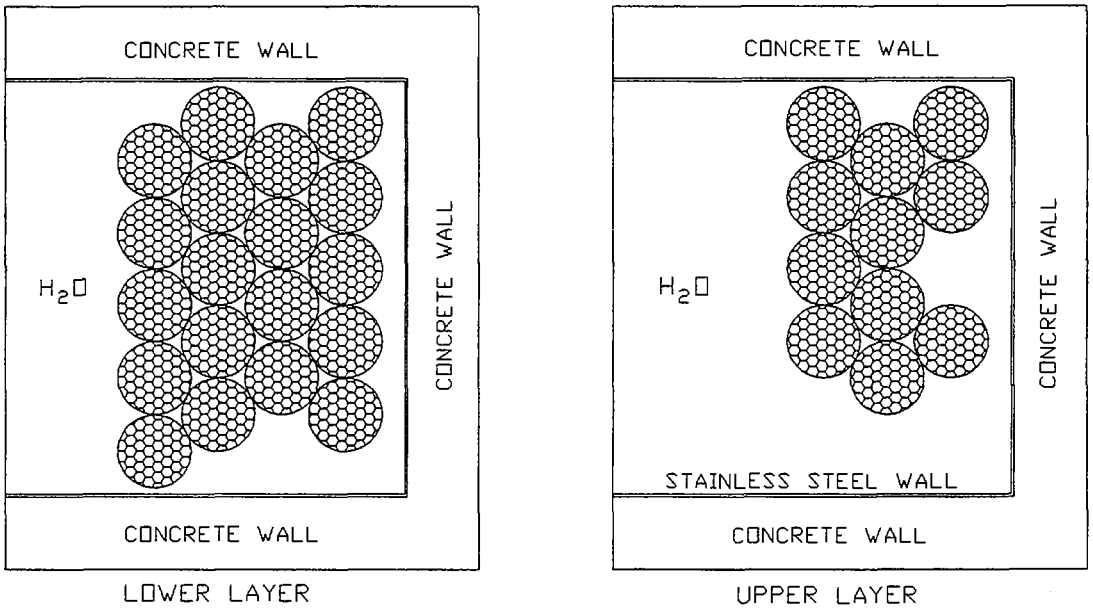

Figure 3 Horizontal cross section of the storage pool with position of the barrels

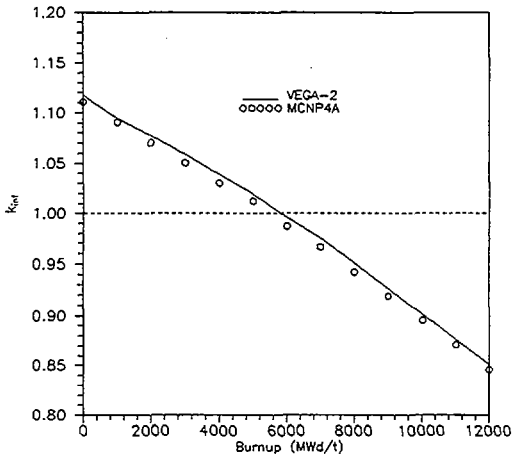

Figure 4 A comparison of the results of the burnup reactivity credit in the single pin cell of the storage barrel

Table II Results of the MCNP k_{eff} calculation

Burnup [GWd/d]	k_{eff}
0	0.82550 ± 0.00114
2	0.79128 ± 0.00132
4	0.76107 ± 0.00106
6	0.72888 ± 0.00110
8	0.69320 ± 0.00112
10	0.65576 ± 0.00107

DEFUELLING OF THE UTR-300 RESEARCH REACTOR

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1. INTRODUCTION

The UTR-300 reactor at the Scottish Universities Research and Reactor Centre was based on the original Argonaut design with two aluminium core tanks set in a graphite reflector each containing six fuel elements cooled and moderated by water flowing up through the tanks in a closed primary circuit. The fuel plates in the original 13-plate elements were uranium oxide-aluminium with a 22g loading of 90% ^{235} U. After 7 years of operation at 100kW (10kW average), the maximum power was increased to 300kW (30kW average) and, in order to maintain the operational excess reactivity, it was necessary to add another plate to each element progressively over the years until they all contained 14 plates. These extra plates were uranium metal-aluminium with 24.5g of 90% ^{235}U. No further modification of the elements was possible and so, with reactivity steadily decreasing, and for a variety of other reasons, a decision was taken to cease operation in September 1995.

This paper describes the procedures whereby the fuel was unloaded from the core into a UNIFETCH flask equipped with a specially designed rotating gamma ray shield and then transported in two separate loads to Dounreay for reprocessing.

2. PRELIMINARIES

The operation was carried out as a plant modification in compliance with an arrangement made against one of the site licence conditions and was assessed in terms of safety as a category 1 modification, having major significance. Such a modification requires a proposal containing a detailed safety case which, after consideration both by the SURRC Nuclear Safety Committee and by an independent consultant, is submitted to the Nuclear Installations Inspectorate (NII). Depending upon the response from NII, work may proceed immediately or must not proceed without an agreement from NII issued in the form of a licence instrument. In the latter case, 'hold points' usually have to be identified beyond which work may not proceed without the further agreement of NII.

During annual maintenance the fuel is routinely transferred into shielded storage pits in the floor of the reactor hall by means of a transfer flask, which holds one element, and it was decided that the loading into the UNIFETCH flask would be carried out in two stages, from the core to the storage pits and from there to the UNIFETCH. This meant that the fuel could be kept in the reactor from shut-down until the transfer with the core ventillation running continuously and the fuel tanks dry. Also, the safety case for transfer to the UNIFETCH from the pits was simpler than it would have been for transfer directly from the core.

2a. FISPIN Calculation

Doses from both shielded and unshielded fuel elements constituted an important aspect of the safety case and the activity of the fuel was therefore calculated at AEA Winfrith by means of the FISPIN code. The calculation was performed with a value of neutron flux modified by the duty cycle of the reactor (18 hours per week) and it is interesting to note that this gives an erroneous value for the activity at shutdown of fission (or activation) products with half-lives much less than the 'on' period of the duty cycle (6 hours) since they will saturate at a value corresponding to the true neutron flux. The effect is evidently a function of the half-life of the species and a detailed calculation shows that for our duty cycle it can be ignored for half-lives in excess of a few days.

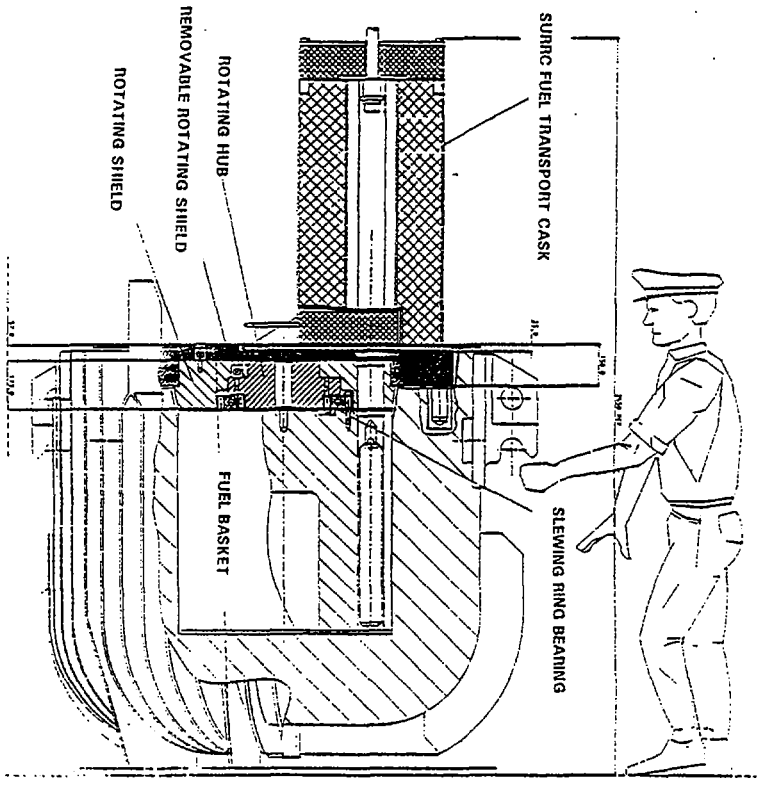

FIGURE 1

2b. Design of the UNIFETCH basket and rotary shield

The UNIFETCH flask is usually loaded and unloaded under water but, since there is no fuel pond on site, it was necessary to do a dry loading through the specially designed shielding arrangement shown in Figure 1. This consisted of a basket with six circumferentially disposed fuel pockets and a blank position. A slewing ring, mounted on a ratchet ring, supported a rotating hub (capable of rotating in one direction only) to which was attached a shield with a single hole in it matching the size of the fuel pocket. The idea was to lower one fuel element at a time from the SURRC transfer flask into each of the six pockets by rotating the shield, ending up with the hole over the blank position and the six elements shielded. To reduce the γ-flash as the element passed from the transfer flask to the UNIFETCH basket, a further shield with a 25mm deep 'footprint' of the transfer flask was mounted on top of the primary rotating shield. The position of a dowel pin on the primary shield relative to the fixed wall of the UNIFETCH flask was marked with adhesive tape for each of the six positions, and the blank, and this same dowel pin served to locate the top

shield. A much more precise alignment was necessary when loading fuel and this was provided by a spring plunger which dropped down through both shields to engage the ratchet ring when the holes were aligned. A locking bar was also engaged in the ratchet ring when a fuel element was being transferred. The shield, carrying the transfer flask, was always rotated to the next position before the flask was removed so that the elements which had already been loaded were shielded at all times (Figure 2).

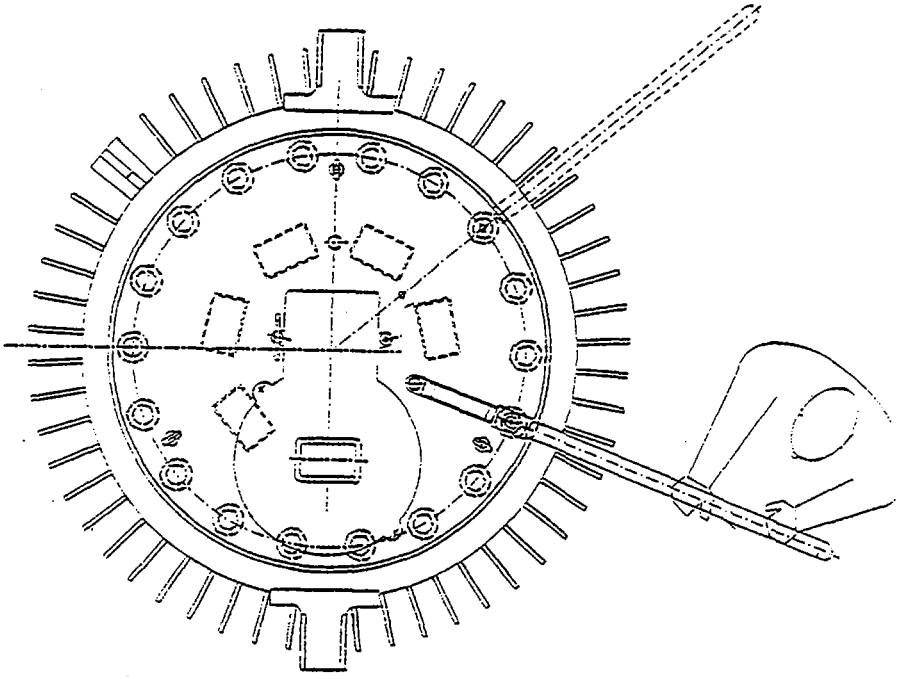

FIGURE 2

2c. The safety case

This was produced by AEA Technology and consisted of a combined preliminary safety report, pre-commissioning safety report and pre-commencement safety report containing a hazard identification (hazops) study. It was backed by a quality plan, detailed operating instructions for the various tasks involved and a criticality calculation.

2d. Precommissioning Trials

The UNIFETCH flask equipped with the basket and shields was brought to SURRC two months before the date proposed for the transfer of active elements and the entire procedure, including the erection of scaffolding around the low loader carrying the UNIFETCH, was tested by transferring two dummy elements from the storage pits to the UNIFETCH. The transfer of fuel elements from the core to the floor was performed routinely by SURRC staff according to documented procedures and lay outside the safety case. The division of labour between SURRC and AEA staff was based upon past experience: all operations involving the SURRC transfer flask were undertaken by SURRC staff whereas the UNIFETCH was handled by AEA staff. It was possible to maintain this practice even when the two flasks were mated during the transfer of a fuel element.

Tests were carried out with a 1GBq ^{60}Co source in the fuel pockets to ensure that there were no shine paths between the basket and the rotating shields. There was no evidence of scattered radiation and calculations of the dose rate in the pockets adjacent to the one containing the source agreed well with the measurements. This in turn gave confidence in the estimated dose rates arising from the fuel elements.

3. HEALTH PHYSICS CONSIDERATIONS AND RADIATION DOSES

The FISPIN output was used in conjunction with shielding calculations to give dose assessments based on the estimated time required to perform each task in the appropriate radiation field. These times were largely derived from the experience gained from fuel manipulations at SURRC over more than 30 years with the same equipment. The maximum calculated whole body dose during the transfer of 6 elements was 239μSv and a dose restraint objective of 250μSv for 6 elements was set. Area dose monitoring was provided by background gamma monitoring equipment in the reactor hall, neutron dose rate monitors, hand held gamma monitors and an air sampler. All operational staff were fitted with whole body thermoluminescent (TLD) and digital dosimeters and finger and head TLDs. The SURRC operator responsible for raising and lowering fuel elements and the AEA operator responsible for rotating the shield were issued with ankle TLDs in order to record any dose arising from movement of the fuel across possible interface gaps. The emergency arrangements required for compliance with the site licence were tested by NII in an inspected exercise involving the local fire brigade. Local HP rules were written for compliance with IRRs. The transfer of fuel from the core to the storage pits was not included in the dose budget and the doses received in this operation (for 6 elements) are shown in Table 1.

	Whole Body Dose (μSv)	Wrist	Ankle
SURRC Op 1	24	110	58
SURRC Op 2	9	-	-
SURRC RPS	11	-	-
SURRC RPA	18	-	-

TABLE 1

Doses recorded during transfer of the same 6 elements from the pit to the UNIFETCH are shown in Table 2.

	Whole Body Dose (μSv)	Calculated (μSv)
SURRC Op 1	9	194
SURRC Op 2	6	-
SURRC RPS	5	194
SURRC RPA	3	-
AEA Op 1	11	163
AEA Op 2	4	163
AEA Project Manager	4	163
AEA HP	4	153
AEA RPA	2	153

TABLE 2

4. CONCLUSIONS

The operation was successfully completed and doses were well within the restraint objective. The difference between the measured and calculated values arose from a longer cooling time than was assumed in the safety case (the activity was roughly half), from overestimates of the time required for execution of the various tasks and from uncertainties in the shielding calculations.