Proceedings of the 1990 International Meeting on
REDUCED ENRICHMENT FOR RESEARCH AND TEST REACTORS

Newport, Rhode Island
September 23-27, 1990

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Program Chairman

Administrative Arrangements

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Proceedings Publication

Stacy Rest

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July 1993
The previous Reduced Enrichment for Research and Test Reactor meetings were held at:

Argonne National Laboratory - November 1978
Saclay, France - December 1979
Argonne National Laboratory - November 1980
Juelich, FRG - September 1981
Argonne National Laboratory - November 1982
Tokai, Japan - October 1983
Argonne National Laboratory - October 1984
Petten, The Netherlands - October 1985
Gatlinburg, Tennessee - November 1986
Buenos Aires, Argentina - September 1987
San Diego, California - September 1988
Berlin, Germany - September 1989
Newport, Rhode Island - September 1990
PREFACE

The global effort to reduce, and possibly, eliminate the international traffic in highly-enriched uranium caused by its use in research reactors requires extensive cooperation and free exchange of information among all participants. To foster this free exchange of information, the Reduced Enrichment Research and Test Reactor (RERTR) Program, at Argonne National Laboratory, sponsored this meeting as the thirteenth of a series which began in 1978. The previous meetings of this series are listed on the facing page.

The common effort brought together, as several other times in the past, a large number of specialists from many countries. One hundred twenty-three participants from 26 countries, including scientists, reactor operators, and personnel from commercial fuel suppliers, research centers, and government organizations, convened in Newport to discuss their results, their activities, and their plans relative to converting research reactors to low-enriched fuels. As more and more reactors convert to the use of low-enriched uranium, the emphasis of our effort has begun to shift from research and development to tasks more directly related to implementation of the new fuels and technologies that have been developed, and to refinements of those fuels and technologies. It is appropriate, for this reason, that the emphasis of this meeting was placed on safety and on conversion experiences.

The Rhode Island Nuclear Science Center, which is at an advanced stage of their plans to convert the RINSC reactor to low-enriched silicide fuel, was the focus of this meeting and a most gracious host. RINSC has availed itself of the opportunity to enhance its capabilities during the process of conversion by designing a much-more-compact core. On behalf of all attendees, I extend thanks to Dr. Frank DiMeglio and his colleagues from RINSC for the informative tour of their facility and for their hospitality. We shall always remember the beautiful setting among the former summer mansions of the wealthy in Newport and the delightful New England clambake, hosted by RINSC.

I would like to thank several members of the RERTR Program who contributed to editing of these proceedings. In particular, James L. Snelgrove and James E. Matos who corrected and edited several papers in consultation with the authors. Compilation of the completed papers and assembly of the proceedings in their final form was smoothly accomplished through the combined efforts of Helen Weber, of Sharon Richmond, and especially of Stacy Rest, a student at the University of Illinois who worked with us this summer to prepare these and other proceedings.

Finally, I thank all attendees for their many contributions and for the spirit of enthusiasm and friendship that they brought to the meeting.

Armando Travelli
Program Chairman
It is with deep regret that I have to report that Robert F. Domagala, who for many years spearheaded the RERTR fuel development effort and was much respected and loved by all the members of the program, died of a heart attack on December 5, 1989 during a technical visit to BATAN, Indonesia.
1990 International Meeting on Reduced Enrichment for Research and Test Reactors
September 23-27, 1990
Hotel Viking, Newport, Rhode Island U.S.A.
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September 24, 1990

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SESSION I

September 24, 1990

NATIONAL PROGRAMS

Chairmen:

G. D. Thamm
(KFA, Federal Republic of Germany)

A. Travelli
(ANL, USA)
STATUS OF THE RERTR PROGRAM

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ABSTRACT

The progress of the Reduced Enrichment Research and Test Reactor (RERTR) Program is described. After a brief summary of the results which the RERTR Program, in collaboration with its many international partners, had achieved by the end of 1989, the major events, findings, and activities of 1990 are reviewed.

The past year has seen an intense debate within DOE and Congress about the goals, achievements, and future funding needs of RERTR Program. The debate persists, but consensus has been reached that the program should be extended for four more years, and that transfer of already developed LEU technology and assistance to research reactors undergoing conversions should continue. The funding provided for FY 1990 and 1991 is adequate for these tasks.

Consistent with this guidance, the RERTR Program has placed on hold any further fuel development efforts aimed at fuels with uranium densities in excess of 4.8 g/cm$^3$, and has concentrated its efforts on technology transfer and implementation activities.

Postirradiation fuel data were analyzed and interpreted, and new conclusions and models for the behavior of research reactor fuels under irradiation were derived. Final reports on the ORR Whole-Core Demonstration and on RERTR fuel development activities are almost ready for publication. A series of computer codes for the analysis of research reactors was successfully converted to run on microcomputers. Visits, consultations, and lectures were provided to potential new fabricators of LEU fuels and to international IAEA courses on the application of such fuels. Analyses, calculations, and safety evaluations were conducted to support both US and foreign research reactors in converting to LEU fuels.

An approximate quantitative evaluation has revealed that nearly half of the conversion work required to eliminate the need for further HEU exports has been collectively accomplished. A major program goal for the next several years will be to work even more
closely than in the past with the various reactor and fuel fabrication organizations, so that implementation of the developed LEU technologies is completed during the next four years. More than ever before, we are looking forward to continued international cooperation in the pursuit of this important goal.

INTRODUCTION

The Reduced Enrichment Research and Test Reactor (RERTR) Program was established in 1978 by the Department of Energy (DOE). It was managed and funded by DOE through 1986. It was funded by the Arms Control and Disarmament Agency (ACDA) between 1987 and 1989, with management responsibility shared between ACDA and DOE. Both funding and management responsibilities for the program were reassumed by DOE in 1990. The primary objective of the program is to develop the technology needed to use Low-Enrichment Uranium (LEU) instead of High-Enrichment Uranium in research and test reactors, and to do so without significant penalties in experiment performance, economics, or safety aspects.

Excellent progress has been made toward the achievement of this objective through the close cooperation which has existed since the beginning between the program and the many organizations represented at this meeting.

The Rhode Island Atomic Energy Commission (RIAEC), operator of the Rhode Island Nuclear Science Center (RINSC), was among the first organizations to cooperate with the RERTR Program and deserves special mention. The joint study agreement between RIAEC and the RERTR Program dates back to more than ten years ago. During this period, RIAEC and RERTR personnel have studied together many aspects of the excellent RINSC facility. In particular, they have investigated some very promising options for RINSC to convert to low-enrichment fuel in compliance with the 1986 NRC rule while, at the same time, ensuring enhanced operational flexibility and experiment performance. We have always been much impressed by the aggressiveness and creativity with which RIAEC personnel have faced this challenge.

It was with great pleasure and anticipation that we decided to hold this meeting in Newport, not only for the natural beauty and charm of the surroundings but also because this will give us an opportunity to visit the RINSC facilities and to hear about their plans and preparations for conversion.

OVERVIEW OF THE SEPTEMBER 1989 PROGRAM STATUS

By September 1989, when the last International RERTR Meeting was held\(^{(1)}\), the main results achieved in the fuel development area were:

(a) The qualified uranium densities of the three main fuels which were in operation with HEU in research reactors when the program began (UA\(_{1x}\)-Al with up to 1.7 g U/cm\(^3\); U\(_{3}O_{8}\)-Al with up to 1.3 g U/cm\(^3\); and UZrH\(_x\) with 0.5 g U/cm\(^3\)) had been significantly increased. The new uranium densities extended up to 2.3 g U/cm\(^3\) for UA\(_{1x}\)-Al, 3.2 g U/cm\(^3\) for U\(_{3}O_{8}\)-Al, and 3.7 g U/cm\(^3\) for UZrH\(_x\). Each fuel had 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 had been collected.
For $\mathrm{U}_3\mathrm{Si}_2$-$\mathrm{Al}$, after reviewing the data collected by the program, the U.S. Nuclear Regulatory Commission had issued a formal and generic approval of the use of $\mathrm{U}_3\mathrm{Si}_2$-$\mathrm{Al}$ fuel in research and test reactors, with uranium densities up to 4.8 g/cm$^3$. In addition, a whole-core demonstration using this fuel had been successfully completed in the ORR using a mixed-core approach.

For $\mathrm{U}_3\mathrm{Si}$-$\mathrm{Al}$, miniplates with up to 7.1 g U/cm$^3$ had been fabricated by ANL and the CNEA, and irradiated to 84-96% in the ORR. PIE of these miniplates had given good results, but had shown that some burnup limits might need to be imposed for the higher densities. Four full-size plates, fabricated by CERCA with up to 6.0 g U/cm$^3$ had been successfully irradiated to 53-54% burnup in SILOE, and a full-size $\mathrm{U}_3\mathrm{Si}$-$\mathrm{Al}$ (6.0 g U/cm$^3$) element, also fabricated by CERCA, had been successfully irradiated in SILOE with 55% burnup. However, conclusive evidence indicating that $\mathrm{U}_3\mathrm{Si}$ became amorphous under irradiation had convinced the RERTR Program that this material as then developed could not be safely used beyond the limits established by the SILOE irradiations.

Two concepts based on hot-isostatic pressing (HIP) procedures had been developed for LEU silicide fuels with the potential for effective uranium densities much greater than 4.8 g U/cm$^3$. One of the concepts was based on a composite structure of $\mathrm{U}_3\mathrm{Si}$ wires and aluminum (up to 12.9 g U/cm$^3$), while the other was based on a $\mathrm{U}_3\mathrm{Si}_2$-$\mathrm{Al}$ dispersion structure (up to 10.2 g U/cm$^3$). Sample miniplates had been produced for both concepts.

In other important program areas, reprocessing studies at the Savannah River Laboratory had concluded that the RERTR fuels could be successfully reprocessed at the Savannah River Plant and DOE had defined the terms and conditions under which these fuels will be accepted for reprocessing.

A new analytical/experimental program had begun to determine the feasibility of using LEU, instead of HEU, in fission targets dedicated to the production of $^{99}$Mo for medical applications. A procedure for basic dissolution and processing of LEU silicide targets had been developed and was ready for demonstration on a full-size target with prototypic burnup. Tests had been initiated to investigate the applicability of Hot Isostatic Pressing procedures to the bonding of LEU metal foils to a Zircaloy base.

Extensive studies had been conducted, with favorable results, on the performance, safety, and economic characteristics of LEU conversions. These studies included many joint study programs, which were in progress for 28 reactors from 17 different countries.

Coordination of the safety calculations and evaluations was continuing for the US university reactors planning to convert to LEU as required by the 1986 NRC rule. Three of these reactors had already been converted, three other safety evaluations had been completed, and calculations for six more reactors were in progress.

PROGRESS OF THE RERTR PROGRAM IN 1990

The past year has seen much uncertainty and debate about the funding of the RERTR Program, and much discussion about the achievements and goals of the program. The increasing constraints on the overall US budget have increased the pressure on the program budget, while the significant RERTR achievements have reduced the perception of the
importance of the steps that remain to be done. Transfer of the program's funding responsibility from ACDA to DOE did not seem to have the desired effect. Thus, for several months it appeared almost certain that the program would be terminated at the end of September 1990, and that many research reactors currently using HEU fuels would be required to convert to LEU fuels without the assistance which the program was able to provide in the past.

The difficulties inherent in this approach became increasingly apparent during the spring of 1990. The debate about what goals should be pursued by the program, and for what budget, still continues. However, it is now generally agreed that the RERTR Program will continue to be funded through the end of September 1994 and that the program will continue to provide analytical and computational support to research reactors preparing for conversion. In the words of Admiral Watkins, US Secretary of Energy,

"We believe that our future efforts to limit the use of high-enriched uranium...should focus on the conversion of research reactors using low-enriched fuels technology already developed. The FY 1990 funding of $1.2 million is focused on this technology transfer and reactor conversion assistance component of the program. We feel it would be appropriate to continue this type of assistance for the period FY 1991-1994."[3]

The funding level which was available to the program during fiscal year 1990 was adequate for the purpose stated by Admiral Watkins, and so is the funding level projected for fiscal year 1991, which was approved by the House on June 19 and by the Senate on July 19. What is not included in either budget is the development of advanced research reactor fuels along the lines described last year at the 1989 International RERTR Meeting in Berlin. The major objections which have been raised to the development of the advanced fuels are related to the high costs involved and to the small number of reactors for which those fuels are needed.

The activities pursued by the RERTR Program during 1990 have been consistent with this guidance and with the priorities outlined at the Berlin meeting.

1. The results of postirradiation examinations of silicide plates were further studied to derive a better understanding of the irradiation behavior, safety characteristics, and applicability of these fuels[4].

2. Improved analytical models were developed to explain and predict the behavior of silicide fuels under irradiation for a variety of different conditions[5].

3. A final, comprehensive report of the ORR whole-core demonstration with silicide fuel is nearing completion[6,7] to assist in the planning and safety evaluations of reactors considering mixed-core conversions.

4. Significant progress was made in the preparation of a report that will condense in a single document the significant information accumulated by the RERTR Program on the fabrication of new LEU fuels, especially uranium silicide[8]. Work on final reports on irradiation of full-size elements has begun[9].

5. A series of computer codes which can be used to analyze the performance and safety characteristics of research reactors operating with LEU fuels were successfully converted to run on
microcomputers\textsuperscript{[10]}. This work is expected to improve the applicability and usefulness of the codes.

6. Visits, consultations, and lectures on LEU fuel fabrication technology were provided to new potential producers of LEU silicide fuels.

7. Visits, consultations, and lectures were provided to international IAEA courses on concepts, methods, computer codes, and safety considerations with regard to LEU core conversions of research reactors.

8. Analyses, calculations, and safety evaluations were conducted to support US research reactors in their efforts to convert to LEU fuels as required by the US Nuclear Regulatory Commission. The results of this work are included in some of the papers presented at this meeting\textsuperscript{[11,12,13]}.

9. Analyses, calculations, and safety evaluations were performed also for reactors undergoing or considering conversions outside the US, within the joint study agreements which are in effect between the RERTR Program and several international research reactor organizations.

10. Calculations were performed for advanced compact-core designs and concepts, to define their limits and to investigate the feasibility of using reduced enrichment fuels in their cores. The results of these studies are very preliminary and are not reported at this meeting.

As our focus shifts from the development of increasingly advanced fuels to the implementation of the fuels which have already been developed, it is of special interest to assess how much the international RERTR effort has progressed in resolving the main problem which motivated the launching of the RERTR Program twelve years ago. Stated in its simplest terms, that problem was the amount of HEU which needed to be exported every year from the United States to fuel research and test reactors, and the number and vulnerability of the international shipments caused by those exports.

Forty-three research reactors with power no less than 1 MW used to require HEU exports and are still in operation without imminent plans for being shut down. Only three of these reactors (the OSIRIS reactor in France, the THOR reactor in Taiwan and the PARR-1 reactor in the Philippines) have been fully converted to LEU fuel, and this might give the impression that progress towards elimination or reduction of the HEU exports has been minimal. This impression would be wrong. As is well known to all of us, many steps need to be accomplished before a conversion is complete, even after the appropriate fuels have been developed, tested, and commercialized, and many reactors have already completed at least some of these steps.

Six important steps which many reactors may be expected to take on their way toward conversion are listed below:

Step 1. Determine feasibility of conversion.
Step 2. Develop conversion plan.
Step 4. Order LEU elements for conversion.
Step 5. Load the first LEU elements in the core.

Step 6. Unload the last HEU elements from the core.

Not all of these steps need to be taken by all reactors—many reactors, for instance, may bypass the irradiation of prototypes. Nevertheless, these steps correspond to different stages of the conversion process and may be used to gauge progress in an approximate manner. The forty-three reactors requiring HEU exports have been subdivided in seven categories according to the most advanced of these steps which they have achieved, and the results are displayed graphically in Figure 1. The two graphs of Fig.1 illustrate the distributions of the numbers of the reactors, and of the average number of kilograms of $^{235}\text{U}$ yearly exported for those reactors, among the various categories. If each step is assumed to require approximately the same amount of effort, the percentage of each diagram which is shadowed corresponds to the percentage of the overall work which has been accomplished, either in terms of reducing the number of reactors requiring HEU exports or in terms of reducing the yearly exports of $^{235}\text{U}$. These percentages are $43.8\%$ for the number of reactors and $47.4\%$ for the yearly $^{235}\text{U}$ exports. Therefore, the progress which has been achieved collectively by the international RERTR effort corresponds approximately to one half of that required to achieve the ultimate goal.

Nearly five years have passed since $\text{U}_3\text{Si}_2\text{Al}$ LEU fuel was fully qualified for use in research reactors. It seems realistic to assume that most of the remaining conversion work can be accomplished during the next four years. This is a collective goal that deserves our best effort. The reactors, their operation, and their safety are the responsibility of the individual reactor organizations, which therefore will have to shoulder the biggest share of the work required for the conversions. However, we intend to work diligently with those organizations and to spare no effort to ensure that this goal is achieved.

PLANNED ACTIVITIES

The activities which the RERTR Program proposes to undertake during the coming years are consistent with the plan outlined at last years' international meeting and with the new budget guidance which has been provided to the program. The major elements of this plan are described below.

1. Complete testing, analysis, and documentation of the fuels which have already been developed, and support their implementation.

2. Transfer LEU fuel fabrication technology to countries and organizations which require such assistance.

3. Perform calculations and evaluations for reactors planning to undergo conversion, to assist in improving performance and in resolving safety issues.

4. Within the available budget, continue development of a viable process, based on LEU, for the production of fission $^{99}\text{Mo}$ in research reactors.
SUMMARY AND CONCLUSION

An intense debate about the goals, achievements, and future funding needs of the RERTR Program has taken place within DOE and within Congress during 1990. The debate persists, but already consensus has been reached that the program should

1. continue for four more years,
2. continue to transfer already developed LEU technology, and
3. continue to assist research reactors in the implementation of such technology.

Consistent with this guidance, the RERTR Program has placed on hold any further fuel development effort aimed at LEU fuels with uranium densities greater than 4.8 g U/cm$^3$, and has concentrated its efforts on the following activities:

1. Existing fuel data were analyzed and interpreted. New conclusions and models for the behavior of research reactor fuels under irradiation were derived. These conclusions and models offer new insights into the conditions under which the new LEU fuels may be applicable.
2. Final reports of experiments and analyses are being prepared. A final report of the ORR Whole-Core Demonstration is nearing completion, and will assist most research reactors planning to convert to LEU fuel through a succession of mixed-cores. A final report on RERTR fuel development activities at ANL is also nearing completion. In addition, work on final reports on irradiation of full-size elements has begun.
3. A series of computer codes for the analysis of research reactors operating with LEU fuels was successfully converted to run on microcomputers. This work is expected to improve the applicability and usefulness of the codes.
4. Visits, consultations, and lectures were provided to potential new fabricators of LEU fuels and to international IAEA courses on the application of such fuels.
5. Analyses, calculations, and safety evaluations were conducted to support both US and foreign research reactors in their activities related to core conversions to LEU fuels.

An approximate quantitative evaluation of the LEU conversion progress which has been collectively achieved has revealed that nearly half of the work required to eliminate the need for further HEU exports has been accomplished. A primary program goal for the next several years will be to work even more closely than in the past with the various reactor and fuel fabrication organizations, so that the process of implementing the technologies which have already been developed, and the permanent reduction in HEU exports, can be completed during the next four years. More than ever before, we are looking forward to continued international cooperation in the pursuit of this important goal.
REFERENCES


Fig. 1  PROGRESS TOWARD CONVERSION OF RESEARCH REACTORS REQUIRING HEU EXPORTS

NUMBER OF REACTORS

REQUIRED 235U EXPORTS, KG/YR

CONVERSION STEP REACHED

CONVERSION STEP REACHED
PRESENT STATUS OF THE RERTR IN FRANCE: MAIN PROBLEMS IN THE FIELD OF FUEL STORAGE, TRANSPORTATION AND REPROCESSING

J. J. Graf and A. Ballagny
CEA/CEN
Saclay, France

ABSTRACT

In the very near future, reactor operators will have to find additional interim storage capabilities in order to cope with the suspension of shipments for reprocessing.

While a possible short-term solution for French reactors may be the use of PEGASE (CEA-Cadarache Interim Storage Facility), the DOE is expected to make a point decision shortly.

In the event the DOE no longer performs reprocessing, especially for 93% - enriched uranium, an alternative industrial solution could be implemented in Europe.

Introduction

The supply of uranium by DOE and its reprocessing of irradiated fuels were very satisfactory through 1988.

When the DOE announced it would suspend shipments for reprocessing as of December 1988 for reasons of environmental assessment, none of the research reactor operators felt that this decision could pose a real threat for the operation of the reactors and accordingly impede the scientific research programs for which they are responsible.

Since then operators have lived in the hope, fanned by various rumors, that solution would soon be found within the next six months. They firmly believed the DOE was striving to find a lasting solution and prevent a crisis.

Today the situation is quite different. We are in the midst of the crisis. With interim storage capacity at a saturation point, operators have finally come to realize the seriousness of the situation and are desperately trying to find immediate solutions.
The French context

In France, there are three main experimental reactors with a significant fuel consumption: OSIRIS (70 MW) and SILOE (35 MW), plus the RHF (57 MW), since it is also subject to French regulations.

Annual consumption of all reactors is approximately 85 kg 93% enriched uranium and 600 kg of 7.5%.

After scheduled conversion to silicide fuel of the OSIRIS and SILOE reactors, consumption of 93% enriched uranium will drop to approximately 65 kg; consumption of 20% enriched uranium will amount to 300 kg.

The disposal of irradiated fuels is a problem only for ORPHEE, SILOE and RHF, for which reprocessing was previously performed by the DOE. For some years now, fuels from OSIRIS have been stored in PEGASE.

Two interim storage units are operational at the present time:

PEGASE in pound interim storage

This unit is located at CEA-Cadarache. In its present configuration, PEGASE is virtually saturated by the short-term needs of French reactors, especially OSIRIS. However, additional capacity could be made available within a period of one or two months; however, the safety documents must be prepared to obtain permits. Moreover, capital and operating cost are very high since storage conditions are subject to very stringent safety restrictions.

CASCAD for dry storage

This unit, commissioned in the spring of 1990, was designed to store fuel for 50 years. Its capacity is largely reserved for the needs of French reactors, and especially for fuels from the EL.4 reactor (heavy-water gas cooled reactor shut down in 1983).

The French Safety Authorities are very strict on all matters pertaining to the transportation, handling and storage of irradiated fuels. The most constraining aspect is surface contamination of the elements, which may require all elements to be placed in sealed containers. In addition, a very high earthquake level (SMHV 8) is prescribed for CEA-Cadarache.
Reactor's status as of 20.09.90

<table>
<thead>
<tr>
<th>Reactors</th>
<th>Max. reactor storage capacity</th>
<th>Deadline for fuel disposal</th>
<th>Permit application</th>
<th>Status as of September 1990</th>
</tr>
</thead>
<tbody>
<tr>
<td>ORPHEE</td>
<td>72 FE</td>
<td>November 1990</td>
<td>January 1990</td>
<td>PEGASE 84 EC June 1990</td>
</tr>
<tr>
<td>SILOE</td>
<td>186 FE</td>
<td>March 1991</td>
<td>Pending</td>
<td>Pending permit</td>
</tr>
<tr>
<td>RHF</td>
<td>15 FE</td>
<td>December 1990</td>
<td>Pending</td>
<td>Under study</td>
</tr>
</tbody>
</table>

It is clear that temporary solutions can be found to resolve the issues posed by the suspension of reprocessing. However, the fundamental question remains.

- First, safety authorities are opposed to the accumulation of irradiated fuel whose ultimate destination is not known (open cycle).

- Second, the operators are not prepared to give up the value contained in irradiated fuels. Such a waiver will increase the cost of joint scientific research, usually shared by the universities.

While for 93% enriched fuels, the reprocessing solution continues to appear the most realistic, especially in view of their residual UTS and uranium value, the same is not true for 20% enriched fuels, for which reprocessing is not economically justifiable. For the latter, operators could conduct research on the future of these fuels in order to study, for example, the feasibility of deep storage without prior reprocessing.

Solutions for the future

All operators are hopeful that the DOE will announce the resumption of reprocessing.

If this does not happen, industry will have to propose a substitute solution. Given the uncertainties in the last two years, industry has not been able to map out a real strategy since it has been approached for localized operations only.
It is clear that the Europeans can handle all stages on the cycle. AEA Technology has submitted proposals for interim storage and reprocessing. COGEMA has also stated that it has the appropriate facilities to meet all the needs of the cycle, including the supply of highly enriched uranium.

Among the possible solutions, given the needs for both 93% and 20% enriched uranium, careful consideration should be given to the supply of 20% enriched uranium from a mixture of ex-93% reprocessed uranium and natural uranium. This solution would require an agreement between the reactor operators. In France, for example, where several reactors use 20% and 93% enriched uranium, this solution could be considered, but to optimize the cycle costs, it should be used on a widespread basis.

CONCLUSION

While makeshift solutions are available to offset the effects of a suspension of the reprocessing, the operations of such solutions pose technical problems and are costly.

If the present situation were to persist, industry should move to propose an alternative strategy aimed at limiting the accumulation of irradiated fuels in storage ponds and at cutting down on cycle costs.
STATUS OF REDUCED ENRICHMENT PROGRAM FOR RESEARCH REACTORS IN JAPAN

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ABSTRACT

The reduced enrichment programs for the JRR-2, JRR-3, JRR-4 and JMTR of Japan Atomic Energy Research Institute (JAERI), and the KUR of Kyoto University Research Reactor Institute (KURRI) are in progress under the Joint Study Programs with Argonne National Laboratory (ANL). Recently, the Japanese Government decided to cancel the KUHFR Project.

The JRR-2 and JMTR have already been converted to the use of MEU aluminide fuels in 1987 and 1986, respectively. The operation of the upgraded JRR-3 was started in March 1990 with the LEU aluminide fuels.

The preparation was started for the application of the JMTR and KUR for safety review of core conversion to LEU silicide fuel. The safety review application for two LEU silicide elements in the KUR was accepted in May 1990 and will be approved shortly.

INTRODUCTION

Among eighteen research reactors and critical assemblies in Japan (Tables 1 and 2), those which are relevant to the RERTR program are the JRR-2, JRR-3, JRR-4 and JMTR of JAERI and the KUR of KURRI (Table 3). The RERTR program in Japan has been pursued extensively under the direction of 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, JAERI and KURRI. It has played a remarkable role in deciding policies related to the program, and the 53rd Committee was held on September 12, 1990.
The program in JAERI for the first step, in which the JRR-2 and JMTR are to be converted to the use of 45% enriched uranium (MEU) aluminide fuels and the JRR-3 to the use of 20% enriched uranium (LEU) aluminide fuels, has been completed. The first criticality of the new JRR-3 was achieved on March 22, 1990. After that, the reactor power was increased step by step to high power level and the maximum power of 20 MW was established. Further core conversion of the JMTR for use of LEU silicide fuel with burnable absorber has been studied since 1984 in accordance with the Joint Study with ANL. The preparation is now underway for the safety review application of the conversion, which is scheduled in 1993. Studies on silicide fuel plate behavior under accidental conditions are progressing by utilizing the NSRR. The JRR-2 has been remodeled to realize the neutron field for the boron neutron capture therapy, and one patient was treated in August 1990.

On the other hand, in KURRI the same efforts as in JAERI to reduce the enrichment of the KUR are in progress. KURRI is planning to demonstrate to use two full size LEU silicide fuel elements among the current HEU elements, prior to the safety review for full core conversion with LEU silicide fuel. The safety analysis report for the two-element demonstration was submitted to the government in May, 1990, and it is to be approved shortly. The irradiation of these LEU elements will start in mid-1991. This year, five patients were treated with the boron neutron capture therapy.

JAERI

JRR-2

After the fuel of the core was converted from 93% enriched uranium (HEU) to 45% enriched uranium (MEU) the reactor has been operated for 30 operation cycles (12 days/cycle) of more than 81,000 MWh. No evidence for fuel failure has been observed.

An increased demand for in-core irradiation is satisfied by the new reactor core which consists of all MEU cylindrical fuel elements with a central irradiation hole. Although the reactivity loss was increased by the irradiation materials, the utilization performance such as neutron fluxes for irradiation and for beam experiments is almost the same as those of the HEU fuel core.

Recently, the remodeling of the thermal column facilities was carried out for medical use. After that, one patient with brain tumor was successfully treated by means of boron-neutron capture therapy in August 1990.

The reactor has been operated for 30 years since the first criticality was achieved in 1960. The decommissioning of the reactor is now under consideration.

JRR-3

The JRR-3 was upgraded. The construction and installation works of the JRR-3 had been completed. The first criticality of the new JRR-3 was achieved
on March 22, 1990 using the LEU UA$_1$–Al (2.2gU/cm$^3$) fuel elements. The minimum critical mass of the new core corresponded to 14 standard fuel elements and 6 follower fuel elements. This value was approximately equal to the calculated result. Finally, the 26 standard fuel elements and 6 follower fuel elements were loaded in the core as designed.

The measurements of characteristics such as control rod reactivity worths, reactivity coefficients and neutron flux distributions were carried out on full core under zero power operation. After these experiments, the reactor power was increased step by step to high power level and the maximum power of 20 MW was established in middle of August 1990.

### JRR-4

Some feasibility studies on the low enriched UA$_1$–Al dispersion fuel for the JRR-4 were finished in 1988. The core design using the U$_3$Si$_2$–Al dispersion fuel has been carried out considering the present status of the development on the silicide fuel, and some advances has been made from the view point the core neutronics, thermohydraulic characteristics etc.

### JMTR

The JMTR has been operated with MEU fuels since July 1986. Preparations are now underway to apply for safety review of the core conversion. The safety analysis report for the use of LEU fuel is scheduled to be submitted to the Science and Technology Agency in autumn of 1990 and the JMTR operation with LEU fuels will be started in 1993.

The LEU fuel for the JMTR is determined to be silicide (U$_3$Si$_2$) fuel with 4.8 gU/cm$^3$ and burnable absorbers of aluminum sheathed Cd wires are located in each side plate. The use of this fuel gives the following advantages to the JMTR operation; JMTR can operate 25 consecutive days without refueling, which is currently carried out after 12 day operation, and the operating characteristics remain unchanged.

Several R&D works on silicide fuel have been successfully conducted in 1989 through 1990, such as measurements of release/born ratios of FPs for high burnup fuel samples, hydraulic tests using dummy fuel elements with depleted uranium, measurements of thermal conductivity. Fabrication and inspection technique on silicide fuel elements with burnable absorbers have been checked through fabrication of the dummy fuel elements.

Concerning review of safety evaluation, possibility of stress corrosion cracking on primary pipings was investigated and non-destructive inspection of primary piping was conducted. Experimental study on DNB heat fluxes correlation was conducted to obtain necessary information and data for safety analysis. The safety analysis report are currently being finalized.

### Pulse Irradiation of Silicide Fuels in the NSRR

Information on silicide fuel behaviors under accidental conditions is
requested as a data base for the safety evaluation in the application of silicide fuels to the research reactors.

Thermal and mechanical response, fuel failure initiation and fuel fragmentation criteria of silicide fuel are to be studied by subjecting silicide test fuel plates to pulse irradiation in the NSRR.

Mini-plate type fresh silicide fuels are used as the test samples. In the irradiation, test fuels are supported in stagnant water under atmospheric pressure and room temperature conditions in the test capsule. Pulse irradiation will be conducted by increasing the energy deposition level gradually from below 100 cal/g fuel to above 240 cal/g fuel.

Fabrication of the test fuel plates have been completed and that of irradiation capsules is under way. Tests will be started in November 1990 and 5 tests are planned in Fiscal 1990.

KURRI

KURRI has had the Joint Study Program with since May 1977. The KUR has been successful in operation with HEU alloy fuel since 1964. This year the boron neutron capture therapy was applied to treatments for 5 patients: 4 brain tumor patients including 2 Germans and one melanoma patient.

The KUR will be converted to use LEU silicide fuel. Neutronics calculations and thermal hydraulic analysis for use of LEU fuel in the KUR have been almost completed, and it has been confirmed that the KUR core can be converted to use LEU silicide fuel without major design change.

Before the full core conversion with LEU silicide fuel, two elements will be demonstrated in the current HEU core. The safety analysis report for two-element irradiation test was submitted to the Science and Technology Agency and it is be approved shortly. The irradiation test with these elements will start in mid-1991. The full core conversion will be anticipated in 1993.

Recently the Japanese Government suggested to cancel the KUHFR project, for which the related tasks are under consideration.
REFERENCES


2. K. Sato, "Opening Statement to the Internation Meeting on Reduced Enrichment for Research and Test Reactors", in Proceedings of the Internation Meeting on Reduced Enrichment for Research and Test Reactors, Tokai, Japan, October 24-27, 1983, JAERI-M 84-073, pp.8-10 (May 1984).


<table>
<thead>
<tr>
<th>Name</th>
<th>Owner</th>
<th>Site</th>
<th>Type and enrichment</th>
<th>Max. power</th>
<th>Start-up data</th>
</tr>
</thead>
<tbody>
<tr>
<td>JRR-2</td>
<td>JAERI</td>
<td>Tokai</td>
<td>D$_2$O (CP-5)</td>
<td>10 MW</td>
<td>1960.10</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>U-Al</td>
<td>93%</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>UAl$_x$</td>
<td>45%</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>10 MW</td>
<td>1987.11</td>
</tr>
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<td>UTK-KINKI</td>
<td>Kinki Univ.</td>
<td>Higashiosaka</td>
<td>H$_2$O (UTR)</td>
<td>1 W</td>
<td>1961.11</td>
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<tr>
<td>TRIGA-II</td>
<td>Rikkyo Univ.</td>
<td>Yokosuka</td>
<td>H$_2$O (TRIGA)</td>
<td>100 kW</td>
<td>1961.12</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>U-ZrH</td>
<td>20%</td>
<td></td>
</tr>
<tr>
<td>TTR</td>
<td>Toshiba</td>
<td>Yokosuka</td>
<td>H$_2$O (pool)</td>
<td>100 kW</td>
<td>1962.3</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>U-Al</td>
<td>20%</td>
<td></td>
</tr>
<tr>
<td>JRR-3</td>
<td>JAERI</td>
<td>Tokai</td>
<td>D$_2$O (tank)</td>
<td>10 MW</td>
<td>1962.9</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>NU</td>
<td>20%</td>
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<td></td>
<td></td>
<td></td>
<td>U$_O_2$</td>
<td>1.5%</td>
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</tr>
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<td></td>
<td></td>
<td>H$_2$O (pool)</td>
<td>20 MW</td>
<td>1990</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>UAl$_x$-Al</td>
<td>20%</td>
<td></td>
</tr>
<tr>
<td>TRIGA-II</td>
<td>Musashi Inst. Tech.</td>
<td>Kawasaki</td>
<td>H$_2$O (TRIGA)</td>
<td>100 kW</td>
<td>1963.3</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>U-ZrH</td>
<td>20%</td>
<td></td>
</tr>
<tr>
<td>KUR</td>
<td>KURRI</td>
<td>Kumatorii</td>
<td>H$_2$O (tank)</td>
<td>5 MW</td>
<td>1964.6</td>
</tr>
<tr>
<td>JRR-4</td>
<td>JAERI</td>
<td>Tokai</td>
<td>H$_2$O (pool)</td>
<td>3.5 MW</td>
<td>1965.1</td>
</tr>
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<td></td>
<td>U-Al</td>
<td>93%</td>
<td></td>
</tr>
<tr>
<td>JMTR</td>
<td>JAERI</td>
<td>Oarai</td>
<td>H$_2$O (MTR)</td>
<td>50 MW</td>
<td>1968.3</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>U-Al</td>
<td>93%</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>UAl$_x$</td>
<td>45%</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>50 MW</td>
<td>1986.8</td>
</tr>
<tr>
<td>YAYOI</td>
<td>Univ. of Tokyo</td>
<td>Tokai</td>
<td>fast U (horizontally movable)</td>
<td>2 kW</td>
<td>1971.4</td>
</tr>
<tr>
<td>NSRR</td>
<td>JAERI</td>
<td>Tokai</td>
<td>H$_2$O (TRIGA)</td>
<td>300 kW</td>
<td>1975.6</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>U-ZrH</td>
<td>20%</td>
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<td>-------------</td>
<td>----------------------</td>
<td>------------</td>
<td>---------------</td>
</tr>
<tr>
<td>VHTRC</td>
<td>JAERI</td>
<td>Tokai</td>
<td>Graphite U 20%</td>
<td>10 W</td>
<td>1961.1</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>horizontally split</td>
<td></td>
<td></td>
</tr>
<tr>
<td>TCA</td>
<td>JAERI</td>
<td>Tokai</td>
<td>H₂O(tank) UO₂ 2.6%</td>
<td>200 W</td>
<td>1962.8</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>UO₂-PuO₂ 2.6%</td>
<td></td>
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</tr>
<tr>
<td>NCA</td>
<td>NAIG</td>
<td>Kawasaki</td>
<td>H₂O(tank) UO₂ 1-4.9%</td>
<td>200 W</td>
<td>1963.12</td>
</tr>
<tr>
<td>JMTRC</td>
<td>JAERI</td>
<td>Oarai</td>
<td>H₂O(pool) U-Al 93%</td>
<td>100 W</td>
<td>1965.10</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Uₐₓ 45%</td>
<td></td>
<td>1983.8</td>
</tr>
<tr>
<td>FCA</td>
<td>JAERI</td>
<td>Tokai</td>
<td>fast U 93% U 20% Pu</td>
<td>2 kW</td>
<td>1967.4</td>
</tr>
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<td></td>
<td></td>
<td></td>
<td>horizontally split</td>
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<td></td>
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<tr>
<td>DCA</td>
<td>PNC</td>
<td>Oarai</td>
<td>D₂O(tank) UO₂ 0.22%</td>
<td>1 kW</td>
<td>1969.12</td>
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<td></td>
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<td></td>
<td>UO₂-PuO₂ 1.5%</td>
<td></td>
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</tr>
<tr>
<td>KUCA</td>
<td>KURRI</td>
<td>Kumatori</td>
<td>various U-Al 93%</td>
<td>100 W</td>
<td>1974.8</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>45%</td>
<td></td>
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</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(short time)</td>
<td></td>
<td>1981.5</td>
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Table 3. Research Reactors Relevant to RERTR in Japan

<table>
<thead>
<tr>
<th>Name</th>
<th>Power (MW)</th>
<th>First Critical</th>
<th>Fuel Enrichment</th>
<th>Conversion</th>
</tr>
</thead>
<tbody>
<tr>
<td>KUR (KURRI)</td>
<td>5</td>
<td>1964</td>
<td>HEU \rightarrow LEU</td>
<td>1993</td>
</tr>
<tr>
<td>KUHFR (KURRI)</td>
<td>30</td>
<td>to be canceled</td>
<td></td>
<td></td>
</tr>
<tr>
<td>JRR-2 (JAERI)</td>
<td>10</td>
<td>1960</td>
<td>HEU \rightarrow MEU</td>
<td>1987</td>
</tr>
<tr>
<td>JRR-3 (JAERI)</td>
<td>20</td>
<td>1962</td>
<td>LEU \rightarrow LEU</td>
<td>1990</td>
</tr>
<tr>
<td>JRR-4 (JAERI)</td>
<td>3.5</td>
<td>1965</td>
<td>HEU \rightarrow LEU</td>
<td>1990's</td>
</tr>
<tr>
<td>JMTR (JAERI)</td>
<td>50</td>
<td>1968</td>
<td>MEU \rightarrow LEU</td>
<td>1993</td>
</tr>
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**Related Critical Assemblies**

<table>
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<tr>
<th>Name</th>
<th>First Critical</th>
<th>Fuel Enrichment</th>
<th>Conversion</th>
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<tr>
<td>KUCA (KURRI)</td>
<td>0.0001</td>
<td>1974</td>
<td>HEU \rightarrow MEU</td>
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<td>JMTRC (JAERI)</td>
<td>0.0001</td>
<td>1965</td>
<td>HEU \rightarrow MEU</td>
</tr>
<tr>
<td>Year</td>
<td>Event</td>
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<tr>
<td>------</td>
<td>-------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1977</td>
<td>11. Japanese Committee on INFCE WG-8 was started.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1977</td>
<td>11. Joint Study Program was proposed at the time of the application of export license of HEU for the KUHFR.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1978</td>
<td>5. ANL-KURRI Joint Study Phase A was started.</td>
<td></td>
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<tr>
<td>1978</td>
<td>6. Five Agency Committee on Highly Enriched Uranium was organized.</td>
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</tr>
<tr>
<td>1979</td>
<td>2. ANL-KURRI Joint Study Phase A was completed.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1979</td>
<td>5. Project team for RERTR was formed in JAERI.</td>
<td></td>
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<tr>
<td>1979</td>
<td>7. ANL-KURRI Joint Study Phase B was started.</td>
<td></td>
<td></td>
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<tr>
<td>1980</td>
<td>1. ANL-JAERI Joint Study Phase A was started.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1980</td>
<td>8. ANL-JAERI Phase A was completed.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1980</td>
<td>9. ANL-JAERI Phase B was started.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1981</td>
<td>5. MEU UA1-Al full core experiment was started in the KUCA.</td>
<td></td>
<td></td>
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<tr>
<td>1983</td>
<td>3. ANL-KURRI Phase B was completed.</td>
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</tr>
<tr>
<td>1983</td>
<td>8. MEU UA1-Al full core experiment in the JMTRC was started.</td>
<td></td>
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<tr>
<td>1983</td>
<td>11. ANL-KURRI Phase C was started.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1984</td>
<td>3. ANL-JAERI Phase B was completed.</td>
<td></td>
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</tr>
<tr>
<td>1984</td>
<td>4. ANL-JAERI Phase C was started.</td>
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<tr>
<td>1984</td>
<td>4. MEU-HEU mixed core experiment in the KUCA was started.</td>
<td></td>
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</tr>
<tr>
<td>1984</td>
<td>9. Irradiation of 2 MEU and 1 LEU UA1-Al full size elements in the JRR-2 was started.</td>
<td></td>
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<tr>
<td>1984</td>
<td>10. Irradiation of LEU UA1-Al full size element in the JRR-4 was started.</td>
<td></td>
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<tr>
<td>1984</td>
<td>11. Thermal-hydraulic calculations for the KUR core conversion from HEU to LEU was performed.</td>
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<tr>
<td>1985</td>
<td>1. Irradiation of MEU UA1x-Al full size elements in the JMTR was started.</td>
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<td>1985</td>
<td>3. Irradiation of MEU UA1-Al full size elements in the JMTR was completed.</td>
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<td></td>
<td>Irradiation of LEU U Si-Al miniplates in the JMTR was started.</td>
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<td>1985</td>
<td>6. Irradiation of LEU USi y-Al miniplates in the JMTR was completed.</td>
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<td>1985</td>
<td>10. Neutronics calculations for the KUR core conversion from HEU to LEU was performed.</td>
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<tr>
<td>1986</td>
<td>1. Irradiation of MEU UA1-Al full size elements in the JRR-2 was completed.</td>
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<td>1986</td>
<td>5. Irradiation of MEU UA1-Al full size elements in the JRR-2 was completed.</td>
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<tr>
<td>1986</td>
<td>8. The JMTR was fully converted from HEU to MEU fuels.</td>
<td></td>
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<tr>
<td>1987</td>
<td>11. MEU UA1-Al full core in the JRR-2 was started.</td>
<td></td>
<td></td>
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<tr>
<td>1988</td>
<td>7. PIE of MEU, LEU UA1-Al full size element in the JRR-2 was completed.</td>
<td></td>
<td></td>
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<tr>
<td>1988</td>
<td>12. Irradiation of a LEU UA1-Al full size element in the JRR-4 was completed.</td>
<td></td>
<td></td>
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<tr>
<td>1990</td>
<td>3. LEU UA1-Al full core experiment in the new JRR-3 was started.</td>
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</tbody>
</table>
Brief Notice on the Termination of the German AF-Program

G. Thamm
Forschungszentrum Jülich GmbH
5170 Jülich, Fed. Rep. of Germany

After a decade of joint efforts and work by industry and national laboratories, the German AF-Program was closed at the end of 1989. The main goal to provide all means required to convert the research reactors in the Federal Republic of Germany has been reached. Conversion concepts meeting the essential criteria worked out by INFCE and all licensing and operating specific data have been elaborated so that the reactors in their present design will be either converted from HEU to LEU operation or closed after consumption of the available HEU material until mid of the nineties. The costs of the AF-Program finally amount to 51.5 million German Marks. Approximately 43 million German Marks were provided by the Federal Ministry for Research and Technology, the remainder was procured by industry.

Although some AF-Group representatives will go on to participate in international RERTR meetings, the continuation of international cooperation, even to a very limited extent, seems to be nearly impossible for two main reasons.

First, there will be no budget for such cooperation and second, the majority of the former program partners are engaged in new tasks and other R&D work.

The former AF Steering Committee (comprising all program partners and German research reactor operators) with a slightly reduced number of members will continue to exist and to meet at least once a year, thus forming a successor group of consultants for the Federal Ministry for Research and Technology on MTR fuel cycle issues.

Within this group, of course, information and experiences of German research reactor conversions will be exchanged, but it is obvious that the group will mostly be concerned with the disposal of irradiated research reactor fuel by investigating alternative solutions since the resumption of reprocessing fuel from foreign customers in the USA is very uncertain today.
HEU - LEU CONVERSIONS: STATUS AND APPLICATION CONSIDERATIONS

THEODORE S. MICHAELS
USNRC - NPRDE
U.S.A.

STATUS OF HEU-LEU CONVERSIONS AS OF 9/90

- 25 REACTORS AFFECTED (1986)
- 1 LICENSE TERMINATED (WEST, ZION)
- 3 ARE PLANNING TO DECOMMISSION (U. OF WASH., CINTICHEM, VA. CAV.)
- 5 HEU - LEU CONVERSION ORDERS ISSUED (RPI, WPI, OSU, MANHATTAN, IOWA AND RPI, WPI, AND OSU OPERATING WITH LEU)
- 16 REACTORS STILL AFFECTED
- 7 HAVE FUNDING FROM DOE (MO.ROLLA, U. OF VA., R.I., U. OF LOWELL, PURDUE, U. OF FLA., GA. TECH.)
- 2 REQUESTS FOR UNIQUE PURPOSE EXEMPTION (NIST, MO. COLUMBIA)
- 7 HAVE NO FUNDING FROM DOE (WASH. STATE, TEXAS A&M, WISC., OREG. STATE, MIT, GA, GE)
- SPERT (NUREG 1281), TRIGA (NUREG 1282) AND URANIUM SILICIDE (NUREG 1313) FUEL REVIEWS COMPLETE

CONTENTS OF APPLICATION TO NRC FOR AUTHORIZATION TO CONVERT TO LEU

- SAR - DESCRIPTION AND ANALYSIS OF ALL SAFETY FACTORS AFFECTED BY CHANGE FROM HEU TO LEU (SEE SAR DETAILS)
- PROPOSED CHANGES IN LICENSE, INCLUDING TECH SPECS
- PROPOSED QUANTITIES AND DURATION OF POSSESSING BOTH HEU AND LEU
- PROPOSED CHANGES IN OTHER LICENSE-RELATED PROGRAMS (PLANS) - IF THERE ARE NO CHANGES PROVIDE STATEMENT TO THIS EFFECT WITH EXPLANATION
  - PHYSICAL SECURITY
  - EMERGENCY PLANNING
  - OPERATOR RE-TRAINING AND/OR REQUALIFICATION
OUTLINE OF PLAN FOR RETRAINING OPERATORS AND HOW AND WHEN EFFECTIVENESS WILL BE VERIFIED

- FUEL LOAD PLAN AND REACTOR CALIBRATIONS (BEFORE STARTUP)
  INCLUDE FUEL LOADING SCENARIO, APPROACH TO CRITICAL REACTOR STARTUP, POWER, REACTIVITY, AND CONTROL ROD CALIBRATIONS AND EXPERIENCE HISTORY OF INVOLVED PERSONNEL

- REVISED SCHEDULE IF NECESSARY (SEE "FORMAT FOR SCHEDULE")

SAFETY DETAILS

DISCUSS THE FOLLOWING FOR BOTH HEU AND LEU

- FULL CONSTRUCTION AND GEOMETRY
  COMPOSITION AND PLACEMENT OF FUEL ELEMENTS/RODS
  - GRAMS U-235 PER PLATE/ROD
  - NUMBER OF ELEMENTS/RODS, PARTIAL ELEMENTS/RODS
  - FUEL ELEMENT/ROD CONFIGURATION

- FUEL STORAGE
  - STORAGE OF HEU AND LEU
  - NEW FUEL RACKS

- CRITICAL OPERATING MASS OF U-235
  - CRITICAL OPERATING MASS
  - SENSITIVITY CALCULATIONS OF REACTIVITY FOR DIFFERENT CORE CONFIGURATIONS
  - COMPARISON OF PROMPT NEUTRON LIFETIME AND EFFECTIVE DELAYED NEUTRON FRACTION

- HYDRAULICS AND THERMAL-HYDRAULICS
  MAXIMUM FUEL TEMPERATURES

- POWER DENSITY AND POWER PEAKING
  - POWER DISTRIBUTION AMONG FUEL ELEMENTS
  - POWER DENSITIES IN FUEL PLATES/RODS
  - OVER POWER CONDITION AT WHICH ONSET OF NUCLEATE BOILING OCCURS

- CONTROL ROD WORTHS FOR EACH ROD

- SHUTDOWN MARGIN

- EXCESS REACTIVITY

- TEMPERATURE, VOID AND DOPPLER FEEDBACK COEFFICIENTS

- FISSION PRODUCT INVENTORY AND FUEL CLADDING THICKNESS
- POTENTIAL ACCIDENT SCENARIOS
  DISCUSS ACCIDENT SCENARIOS IN ORIGINAL LICENSE OR RENEWAL APPLICATIONS AFFECTED BY CONVERSION FROM HEU TO LEU

- DESCRIBE RELATED FACILITY CHANGES, IF ANY

OUTLINE OF REACTOR START-UP REPORT AND COMPARISON WITH CALCULATIONS

- MEASURE THE FOLLOWING FOR HEU AND LEU AND COMPARE WITH CALCULATIONS FOR HEU AND LEU
  - CRITICAL MASS
  - EXCESS (OPERATIONAL) REACTIVITY
  - CONTROL AND REGULATING ROD CALIBRATIONS
    - MEASUREMENTS OF DIFFERENTIAL AND TOTAL ROD WORTHS
  - SHUTDOWN MARGIN
  - THERMAL NEUTRON FLUX DISTRIBUTIONS

- REACTOR POWER CALIBRATION
  METHODS AND MEASUREMENTS THAT ASSURE OPERATION WITHIN THE LICENSE LIMIT. COMPARISON BETWEEN HEU AND LEU NUCLEAR INSTRUMENTATION SET POINTS, DETECTOR POSITIONS AND DETECTOR OUTPUT

- PARTIAL FUEL ELEMENT WORTHS FOR LEU
  MEASURED FOR DIFFERENT NUMBERS OF PLATES FOR WHICH THE FUEL IS CAPABLE; COMPARISON WITH CALCULATIONS

- DISCUSS HOW COMPLIANCE WITH VOID AND TEMPERATURE COEFFICIENT VALUES IN TECH SPECIES IS TO BE ASSURED. COMPARE WITH ANY CALCULATIONS

- EXPLAIN ANY SIGNIFICANT DIFFERENCES IN THE VALUES PREVIOUSLY MEASURED OR CALCULATED WHICH HAVE AN IMPACT ON BOTH NORMAL OPERATION AND POTENTIAL ACCIDENTS WITH THE REACTOR

- MEASUREMENTS MADE DURING INITIAL LOADING OF THE LEU FUEL, PRESENTING SUBCRITICAL MULTIPLICATION MEASUREMENTS, PREDICTIONS OF MULTIPLICATION FOR NEXT FUEL ADDITIONS, AND PREDICTION AND VERIFICATION OF FINAL CRITICALITY CONDITIONS

- UPDATE SCHEDULE (SEE "FORMAT FOR SCHEDULE")

FORMAT FOR SCHEDULE

- DATE OF AVAILABILITY OF FUNDS
DATE OF SUBMITTAL TO NRC OF APPLICATION TO CONVERT,  
(INCLUDING ALL NECESSARY DOCUMENTATION)

DATE OF NRC "ORDER TO CONVERT" = "A"

: $A + N_1$; DATE OF COMPLETION OF ALL PLANS AND SCHEDULING

: $A + N_2$; DATE OF RECEIPT OF LEU FUEL

: $A + N_3$; DATE OF COMPLETION OF ANY FINAL TESTS WITH HEU

: $A + N_4$; DATE OF REMOVAL OF HEU AND REPLACEMENT BY LEU

: $A + N_5$; DATE OF SHIPMENT OF HEU

: $A + N_6$; DATE OF COMPLETION OF INITIAL TESTS OF  
OPERATIONAL PARAMETERS WITH LEU

: $A + N_7$; DATE OF REPORTS TO NRC AND DOE, AS APPROPRIATE,  
SUMMARIZING NEW OPERATIONAL CONDITIONS AND COMPARING WITH  
SAR PREDICTIONS AND LICENSE/TECH SPEC CONDITIONS

$N = \text{NUMBER OF MONTHS REASONABLY REQUIRED}$
SESSION II

September 24, 1990

FUEL DEVELOPMENT

Chairmen:

Y. Fanjas
(CERCA, France)
STATUS OF LEU FUEL DEVELOPMENT
AND IRRADIATION TESTING AT CHALK RIVER LABORATORIES

D.F. Sears, K.D. Vaillancourt, D.A. Leach,
E.L. Plaice, E.J. McKee, R.R. Mead SWcroft and W.S. Simmons

AECL Research
Chalk River Laboratories
Chalk River, Ontario, KOJ 1JO

ABSTRACT

The status of the LEU research-reactor fuel development and irradiation testing program at Chalk River Laboratories is reviewed, and the progress made during the past year is reported. Construction of a new fuel-fabrication facility has been completed and LEU fuel manufacturing has begun. Partial conversion of the NRU reactor from HEU to LEU fuel has been demonstrated; the irradiation of 31 prototype Al-61 wt% U₃Si (3.15 gU/cm³) dispersion fuel rods, approximately one third of a full NRU core, was completed in 1990 January. Post-irradiation examinations confirm that Al-U₃Si fuel behaves well, and is suitable for use in NRU. Al-U₃Si₂ dispersion fuel has also been developed to complement our Al-U₃Si capability. The irradiation of mini-elements containing Al-64 wt% U₃Si₂ (3.15 gU/cm³) was completed in 1989 December. Post-irradiation examinations revealed that up to the terminal burnup of 93 at%, the U₃Si₂ behaved similarly to U₃Si, but swelling was lower. This has paved the way for a full-scale demonstration irradiation to qualify U₃Si₂ fuel for use in NRU. Three full-length NRU rods containing Al-U₃Si₂ have been fabricated and the qualification irradiation in NRU commenced 1990 August.
INTRODUCTION

As part of an international effort to reduce the use of highly enriched uranium (HEU, 93% U-235) in research reactors, Chalk River Laboratories has developed low-enriched uranium (LEU, <20% U-235) fuels for the conversion of the NRU reactor. LEU fuel will also be used in the new MAPLE-type research reactors being developed by AECL. The 10 MW prototype, MAPLE-X10, is currently under construction at Chalk River.

In this paper the progress made in the LEU fuel development and irradiation testing program over the past year is reported. The status of manufacturing in the new fuel-fabrication facility at Chalk River is discussed. The partial conversion of NRU from HEU to LEU Al-U₃Si fuel and the Al-U₃Si₂ fuel qualification irradiation are reviewed. Results from post-irradiation examinations (PIE) of mini-elements containing Al-U₃Si₂ dispersion fuel are presented. A strategy for closing the fuel cycle for Mo-99 production is also introduced.

NEW FUEL-FABRICATION FACILITY

During the past year a significant milestone was achieved in the LEU fuel program at Chalk River. Construction of a new fuel-fabrication facility was completed and production of LEU fuel has started. The modern 1150 m² facility was designed to meet the future fuel requirements of NRU and the new MAPLE class of research reactors. It will also be used to manufacture targets for the production of radioisotopes.

All of the equipment developed for the fabrication of Al-U₃Si dispersion fuel cores (melting and casting, powder production, preparation for extrusion, and core extrusion) has been installed in the new facility. The back-end processing equipment (extrusion-cladding through final assembly), which will be transferred from the HEU fuel-production line, has not yet been installed. The operating licence, which was approved in 1989 November, does not allow HEU and LEU material in the facility at the same time. Therefore, to reduce the risk of disrupting the supply of NRU fuel during the conversion from HEU to LEU, the shut-down, decommissioning and relocation of the portion of the HEU line required for manufacturing LEU fuel has been delayed. This will allow sufficient fuel to be manufactured to provide an inventory for the conversion transition period, after which the equipment will be transferred into the new building, and all of the LEU fuel-fabrication operations will be integrated under one roof.

Much of the work at Chalk River over the past year was dedicated to licencing the facility and implementing quality-assurance programs. The process has taken longer than originally anticipated. This is not surprising, considering that this was the first time in many years that an application had been made to licence a fuel-fabrication facility in Canada, and during that period the regulations have been tightened considerably. The approvals are being granted in stages by the Atomic Energy Control Board (AECB) of Canada. Initially, production has to be carried out in a batch mode to demonstrate that all of the controls are in place and the regulations have been met. After this has been demonstrated and audited, approval for continuous production will be granted. Production is in the
former stage, but the licence for continuous production is expected shortly.

To date, fuel elements for 65 NRU rods have been produced on the full-scale equipment, with the equivalent of 24 of these produced in the new facility. NRU requires 90-110 fuel rods per year, depending on the duty cycle.

Uranium Recovery From Scrap

Currently, the scrap from fuel production at Chalk River is collected, and after a sufficient quantity has been stored, it is sent elsewhere for uranium recovery. With the conversion from HEU to LEU fuel, about five times as much uranium has to be processed. The attendant scrap accrual could pose problems for inventory management and control. Therefore, a program has been launched to develop the technology for scrap recovery and recycling in the new facility.

The scrap from LEU fuel production can be categorized into two types: bulk uranium silicide, e.g., from casting (crucible slag) and powder production; and aluminum-based uranium silicide, e.g., from blending, core extrusion and machining. A process has been developed to recover the uranium from the bulk scrap, for recycling with fresh material in the U-Si melting process. A separate process has been developed to recover the uranium silicide particles from the aluminum-based scrap. The advantage here is that the particles can be recycled directly into the fabrication process without re-melting and powder production.

Both processes have been demonstrated in the laboratory using natural uranium and scrap from LEU fuel production. Chemical and X-ray diffraction analyses showed that the product met the specifications for direct recycling. The processes are being scaled-up for installation in the new facility.

CONVERSION OF NRU TO Al-U₃Si DISPERSION FUEL

The second significant milestone achieved in the LEU program over the past year was the completion of a partial-core conversion demonstration in NRU. In 1990 January, the irradiation of 31 prototype NRU rods containing Al-61 wt% U₃Si dispersion fuel, approximately one third of a full NRU core, was completed successfully. The prototype fuel rods were manufactured using the LEU silicide dispersion fuel-manufacturing equipment to demonstrate that NRU's annual fuel requirements could be met with the scaled-up process. The NRU fuel-rod design and manufacturing process is described elsewhere [1].

Starting in 1988 September, the rods were installed in NRU in place of the normal HEU driver fuel, at a fuelling rate of about two rods per week. They were irradiated under the normal fuel-management scheme, whereby the fuel is initially installed in a low-flux site, subsequently moved to higher-flux sites, then back to a low-flux site as it reaches the exit burnup of 80 at% U-235. The partial LEU core has allowed the reactor engineers and physicists to evaluate the bulk effects of the LEU conversion on reactor
operations. No difference in fuel-rod behaviour or handling, compared with that of HEU fuel, was observed.

PIE confirmed that the Al-U₃Si fuel rods, fabricated under typical manufacturing conditions, behaved well [1]. This conclusively demonstrated that the Al-U₃Si fuel is suitable for use in NRU, and paves the way for the full conversion of NRU. The safety assessments and licencing documents were prepared during the past year, and have been submitted to the AECB. Approval for the full conversion of NRU is expected shortly.

Al-U₃Si₂ FUEL DEVELOPMENT AND IRRADIATION TESTING

As reported previously [2], Al-U₃Si₂ fuel-fabrication technology has been developed at Chalk River to complement our Al-Si capability. The irradiation of 12 mini-elements containing Al-64 wt% U₃Si₂ was completed successfully in 1989 December. Now a demonstration irradiation of prototype full-length Al-U₃Si₂ fuel rods is underway to qualify the fuel for use in NRU. The mini-element and full-scale demonstration irradiations are described below.

Al-U₃Si₂ Mini-Element Irradiation

The primary objective of the mini-element irradiation, experiment EXP-FZZ-921, was to compare the high-burnup behaviour of U₃Si₂ to that of U₃Si, to evaluate its suitability for NRU. The secondary objective was to evaluate the effect of U₃Si₂ particle-size distribution on the fuel behaviour.

The mini-element design has been described elsewhere [2]. Twelve mini-elements were fabricated with a variety of U₃Si particle sizes, to establish limits for the size distribution to be used in manufacturing specifications. Three groups of four mini-elements were tested. The mini-elements contained Al-64 wt% U₃Si₂ (3.15 gU/cm³), but each group contained a distinct size distribution with different proportions of fine (<44 μm) particles. The U₃Si₂ was slightly hypo-stoichiometric; it contained 7.0 wt% Si (stoichiometric U₃Si₂ contains 7.3 wt% Si). At this composition approximately 4 wt% of the as-cast material consists of free uranium metal (at equilibrium, ignoring impurity phases). This represents the lowest level of Si, and thus the highest concentration of free uranium, that may be expected from local non-uniformity in U₃Si₂ billets produced by vacuum-induction melting. Thus, this test irradiation also provided the opportunity to evaluate the effect of free uranium, at the highest levels expected in production, on Al-U₃Si₂ fuel behaviour.

The mini-elements were irradiated to 93 at% burnup in NRU from 1988 June to 1989 December. The initial linear-power rating was approximately 92 kW/m, decreasing as burnup increased, to 52 kW/m. Three mini-elements were removed after 60 at% and 80 at% burnup, respectively, for interim PIE. The remaining six mini-elements were removed in 1989 December after the assembly achieved 93 at% burnup. PIE consisted of visual examinations in the NRU bays and hot cells, immersion-density measurements, neutron radiography, optical metallography, SEM examinations with WDX analysis, image analysis of the fission-gas bubble-size distribution, and burnup
The results of the interim examinations have been described elsewhere [2]. In this section, the results from the high-burnup examination, and the WDX and image analyses, are presented.

Visual and neutron radiographic examinations showed that the mini-elements were in good condition after 93 at% burnup. It was previously reported that, up to 80 at% burnup, Al-U₃Si₂ fuel swelling was less than that of Al-U₃Si fuel. Immersion-density measurements confirmed that the trend continued to high burnup. After 93 at% burnup, fuel-core swelling ranged from 4.2 to 4.7 vol%, compared to 5.8 to 6.8 vol% for mini-elements containing Al-U₃Si from experiment EXP-F2Z-918 [3]. The swelling results are shown in Figure 1 (both fuel types contained 3.15 gU/cm³ and the power histories were similar).

As expected, metallographic examinations revealed that the U₃Si₂ dispersion fuel behaved in a manner similar to the U₃Si dispersions previously tested at Chalk River [2]. Figure 2 shows the typical microstructure in a U₃Si₂ mini-element after 93 at% burnup. The uranium silicide particles reacted with the aluminum matrix to form an interfacial layer around each particle. The majority of particles contained fission-gas bubbles ranging in size up to 18 μm in diameter. Measurements of the fission-gas bubble size distribution were made using a computerized image analyzing system. A typical particle is shown at the top of Figure 3. It can be seen that the bubbles are randomly distributed, unlike the uniform fine distribution reported for flat plate-type fuel [4], and that few fission-gas bubbles are retained in the interfacial layers. Figure 4 shows the typical fission-gas bubble size distribution in a U₃Si₂ particle. Note, however, that some particles (less than 1%) were devoid of fission-gas bubbles, as shown in the bottom of Figure 3. SEM examinations using WDX analysis showed the particles without gas bubbles contained U, Si and Al; however, it appeared that the Si peak heights were lower than those from particles with fission-gas bubbles. The preliminary interpretation is that the particles were not U₃Si₂, but were probably particles initially consisting mostly of free uranium. This is consistent with the starting U₃Si₂ material being hypo-stoichiometric.

The results show that U₃Si₂ is suitable for use in NRU. The high-burnup behaviour is similar to that of previously tested Al-U₃Si dispersion fuel, indicating that the same basic mechanisms control the fuel behaviour. No detrimental effects were observed from free uranium - the fuel had as much as 4 wt% U in the as-cast material (U-7.0 wt% Si). This indicates that U₃Si₂ can be used without heat treatment, i.e., as-cast, with free uranium. Also, there seems to be no detrimental effect from having a high fraction of fine (<44 μm) particles in the fuel.

The excellent results from the U₃Si₂ mini-element irradiation paved the way for a full-scale demonstration irradiation in NRU. This is described below.

Demonstration Irradiation Of Prototype Al-U₃Si₂ Fuel Rods

The primary objective of the prototype irradiation is to qualify Al-U₃Si₂ fuel for use in NRU. A secondary objective is to qualify the fuel-manufacturing process, since it is slightly different from the Al-U₃Si
Figure 1. Swelling of LEU silicide dispersion fuel containing 3.15 gU/cm$^3$, as a function of burnup. Mini-elements from Exp-FZZ-918 contained Al-61 wt% U$_3$Si and those from Exp-FZZ-921 contained Al-64 wt% U$_3$Si$_2$. 
Figure 2  Typical cross-section of Al-64% U$_3$Si$_2$ mini-element after 93 at% burnup.

Figure 3  Micrograph of Al-64 wt% U$_3$Si$_2$ fuel after 80 at% burnup showing particle devoid of fission-gas bubble (H) between typical particles with randomly distributed fission-gas bubbles (I).
Figure 4. Size distribution of fission-gas bubbles in Al-64 wt% U₃Si₂ fuel particles after 80 at% burnup, via image analysis.
Three prototype rods were fabricated for the demonstration irradiation. The uranium silicide dispersion fuel manufacturing process at Chalk River has been described elsewhere [1]. Most of the Al-U₃Si equipment was used to manufacture the Al-U₃Si₂ fuel; however, the U₃Si₂ powder was produced on new equipment in a separate glovebox. The stoichiometric composition was used for melting and casting, and since the mini-element irradiations showed no adverse effects from free uranium in the core, the U₃Si₂ was not heat treated. While the as-cast material was brittle and easier to crush into powders than U₃Si, the extrusion process had to be modified to accommodate the increased volume loading (due to the lower density of U₃Si₂, 12.2 vs 15.4 g/cm³ for U₃Si) and the abrasiveness of the material.

The three full-length rods were inserted in NRU in 1990 August, to begin the first phase of the irradiation. They will be irradiated for approximately one year to reach the design burnup of 80 at%. The irradiation is progressing smoothly. After discharge and sufficient cooling, PIE will be carried out to qualify the fuel (and the manufacturing process) for NRU.

TARGETS FOR ISOTOPE PRODUCTION

The long-term objective of the radioisotopes operation at Chalk River has been to close the fuel cycle for Mo-99 production. The needed steps include recovering the enriched uranium inventory from previous Mo-99 target dissolutions, recycling the material, and further depleting it. The program will have to satisfy environmental and safeguards criteria and regulations.

Consequently, a program has been launched to recover the uranium collected from previous Mo-99 target dissolutions, and to develop and qualify targets for use with this recovered enriched uranium (REU). LEU targets will be developed in parallel with REU targets since the long-term plan is to switch to LEU for Mo-99 production after the REU has been depleted. This will avoid the need for separate test irradiations and chemical processing development. Candidate metal and oxide targets designs are under consideration, and preliminary scoping studies and out-reactor tests are underway.

CONCLUSIONS

1. The LEU conversion program at Chalk River is progressing well. Construction of a new fuel-fabrication facility has been completed and fuel production for NRU and AECL's new MAPLE-type reactors has begun.

2. Partial conversion of NRU to LEU, using a mixed-core approach, has been demonstrated. Thirty-one prototype Al-61 wt% U₃Si rods, one third of a full NRU core, have been successfully irradiated.

3. Al-U₃Si₂ dispersion fuel manufacturing technology has been developed to complement our Al-U₃Si capability.
4. The irradiation of mini-elements containing Al-64 wt% U$_3$Si$_2$ to 93 at% burnup has been successfully completed. PIE confirmed that U$_3$Si$_2$ dispersions swell less than U$_3$Si, and the behaviour is consistent with the excellent behaviour of U$_3$Si fuel tested in the development program.

5. The irradiation of three prototype full-length NRU rods containing Al-64 wt% U$_3$Si$_2$ fuel has begun to qualify this fuel for use in NRU, and to qualify the manufacturing process.

6. A program has been launched to close the fuel cycle for Mo-99 production at Chalk River; the uranium stored from previous target dissolutions will be recovered and recycled. The development and testing of new REU and LEU targets is underway.

ACKNOWLEDGEMENTS

The following are gratefully acknowledged for their contributions: R.J. Chenier and the Chalk River Laboratories' hot-cell staff for the metallography work, and R. Behnke, D. Bruneau and the Whiteshell Laboratories' hot-cell staff for the image analysis and SEM work.

REFERENCES


ATOMIZATION OF U, Si NUCLEAR FUEL

I.H. Kuk, J.B. Lee, C.T. Lee, S.J. Chang
Korea Atomic Energy Research Institute

ABSTRACT

Kaeri has been developing fuel fabrication technology for KMRR since 1986. Papers were presented in respect to the comminution of the U, Si heat-treated blank in the RERTR meetings at San Diego and Berlin in 1987 and 1989. The following points were made in the papers(1) (2):

(1) The fuel meat of KMRR consists of U, Si1+x particles dispersed in Al matrix.
(2) Comminution of the as-cast ingot before heat-treatment is easier than comminution after heat-treatment because the heat-treated U, Si is tougher than the as-cast U, Si2. The former process is called "powder heat-treatment."
(3) For powder heat-treatment, the particle size of primary U, Si2 should be reduced to a certain size depending on the powder size.
(4) The particle size of primary U, Si2 depends on the cooling rate of the molten alloy.
(5) It was proposed that the primary U, Si2 particle size might be reduced further by the rapid solidification process.

Uranium silicide molten alloy is cooled very rapidly by atomizing up to the rate of 10⁶K/sec. Atomizing is performed in two ways, by splat quenching and by disk rotating.

Alloy powders produced by atomizing are supersaturated with high Si content in the matrix and the primary U, Si2 particle is very fine. Atomized powders have homogeneous distribution of Si content, so that the powder heat-treatment is applicable even for very fine powders. Further, the fabrication process can be simplified by eliminating the comminution process.

It is assumed that the thermal conductivity may increase as the particle gets finer. A few measurements were made but were not sufficient even to be described here.

EXPERIMENT

The metallic uranium used in this experiment has a chemical composition as shown in Table 1.

Table 1. Chemical Composition of Uranium

<table>
<thead>
<tr>
<th>U (wt%)</th>
<th>C</th>
<th>Fe</th>
<th>Si</th>
<th>Ni</th>
<th>Cu</th>
<th>H</th>
<th>C</th>
<th>Mg</th>
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<td>99.9</td>
<td>20</td>
<td>54</td>
<td>10</td>
<td>37</td>
<td>27</td>
<td>7</td>
<td>2</td>
<td>12</td>
</tr>
</tbody>
</table>

(impurities in ppm)

Alloying element of silicon is of purity 99.9% and the addition of Si is 3.9 wt% in an alloy.
In splat quenching, molten alloy pool in an arc melting furnace is hammered by a chill block. Most of the splats are spread at the surroundings and a little amount is left at the pool site. Remainders at the pool site are in ribbon type, and those at the surroundings are spread into powders.

In disk rotating, molten alloys in the vacuum induction furnace are poured on a rotating disk with the injection of He/Ar gas mixture and are spread by centrifugal force. Atomized powders are sieved into grades from 800 \( \mu \text{m} \) to 10 \( \mu \text{m} \). Powder size could be controlled by the inlet rate of the molten alloy, rotating disk speed and atmosphere. Parametric study in controlling powder size is not completed yet and is not included in this paper.

Atomized powders are examined in a scanning electron microscope to observe the primary \( \text{U}_3\text{Si}_2 \). Si contents are measured by EDX to find the supersaturated Si in the U matrix in terms of the primary \( \text{U}_3\text{Si}_2 \) particle size. Various size powders are mounted in an epoxy mount and the microstructure are analyzed by the optical microscope. Each different size powder is analyzed by X-ray diffractometry in order to identify the product phases.

RESULT AND DISCUSSION

Conventional Process

The \( \text{U}_3\text{Si}_2 \) alloys are usually cast in an induction furnace and they are heat-treated at 800 degrees celsius for 72 hours. The heat-treated blank has a \( \text{U}_3\text{Si}_{1+x} \) phase, which is so tough to break into pieces and comminution of the heat-treated blank is very hard and laborious. The established processes in fabrication of KMRR fuel meat are shown in Fig. 1.
The long complicated processes on the left is reduced to several steps on the right by obtaining powders directly from melt and introducing the powder-heat treatment as was reported at the 1988 RERTR meeting in San Diego.\(^1\)

**Microstructure of the quenched splats**

The central part under a chill block produces ribbons of slightly dendritic structure as is shown in a microstructure of Fig. 2(a). The thickness of the splats is about 200-300 \(\mu\)m and the size of the primary \(U_3Si_2\) is about 1-4 \(\mu\)m. In periphery of a chill block splat are produces having fine isotropic \(U_3Si_2\) particles as is shown in a microstructure of Fig. 2(b). The thickness of the splats is about 50-100 \(\mu\)m and the size of the primary \(U_3Si_2\) is about 1 \(\mu\)m. It is found that the primary \(U_3Si_2\) size decreases to near 1 \(\mu\)m as the splat size decreases.

![Microstructure SEM](image)

**Fig. 2**
SEM microstructure of splat quenched specimens

(a) central part

(b) periphery

**Microstructure of the Powder Atomized by Rotating Disk**

The similar results are obtained in disk rotating with splat quenching, as the primary \(U_3Si_2\) size decreases with decreasing powder size. It means that the nucleation sites of the primary \(U_3Si_2\) increases ad the growth of the phase is reduced by increasing cooling rate. A calculation of cooling rate is made in relation to the powder size, resulting in that a 100 \(\mu\)m powder is atomized at a cooling rate of \(10^4\)K/sec and that a 20 \(\mu\)m powder is atomized at \(10^6\)K/sec.
This indicates that the finer the powder, the finer the primary $\text{U}_3\text{Si}_2$ particle is formed by the more rapid cooling rate. Powder size is controlled by the inlet rate of molten alloy, rotating speed of a disk, and atmosphere. Product powders range from 800 $\mu$m to 10 $\mu$m. Typical shapes of powders are shown in Fig. 3, 40 $\mu$m in (a) and 100 $\mu$m in (b). It is noted that the particles are spherical.

![Fig. 3. Powders atomized in a rotating disk](image)

(a) 40 $\mu$m

(b) 100 $\mu$m

The primary $\text{U}_3\text{Si}_2$ increases in size with the increase of the powder size. One interesting thing is that some powder has one or two open recess holes in which internal structures are seen. Primary $\text{U}_3\text{Si}_2$ forms in different shapes depending on the powder size. Three different groups are categorized in terms of size and morphology of $\text{U}_3\text{Si}_2$: the powders smaller than 140 $\mu$m having very fine equiaxed $\text{U}_3\text{Si}_2$ particles; the powders between 140 $\mu$m and 180 $\mu$m facet-like grown $\text{U}_3\text{Si}_2$; and the powders larger than 180 $\mu$m dendritically grown $\text{U}_3\text{Si}_2$. They are shown in Figure 4 (a), (b), and (c and d), respectively. The dependence of morphology on the powder size demonstrates that the cooling rate increases with decreasing powder size. This agrees well with the results of the calculations.
Fig. 4. Changes of $\text{U}_3\text{Si}_2$ morphology dependent on the powder size

(a) Structures in a hole of the powder smaller than 140 $\mu$m. Fine and equiaxed $\text{U}_3\text{Si}_2$.

(b) Structures in a hole of the powder between 140 $\mu$m and 180 $\mu$m. Facet-like grown $\text{U}_3\text{Si}_2$.

(c) Structures in a hole of the powder larger than 180 $\mu$m.
Homogeneous Distribution of Si

Inhomogeneous distribution of Si is usually built-up during solidification. Since molten alloy is, however, spread directly into pieces and solidified at a rapid cooling rate, every different atomized powder has nearly equal composition.

Micro-segregation is reduced by the rapid solidification. It is found that Si is homogeneously distributed in a powder. It leads to the supersaturation of Si in the uranium matrix and the degree of supersaturation increases as a powder gets smaller. This means that the smaller the particle, the higher the supersaturation of Si with the higher cooling rate. Si content in the uranium matrix measured by EDX shows as Fig. 5 that solubility of Si increases with decreasing the size of primary $U_3Si_2$. This may enhance heat-treatment of peritectoid reaction.

\[
\text{solubility of Si (wt\%)} \uparrow
\]

![Fig. 5. Result of EDX in the Uranium matrix for the splat quenched powders of different $U_3Si_2$ sizes.](image)
Phase Stability

Ribbons under the central part of a chill block show x-ray diffractometry results as in Fig. 4(a) where it is demonstrated that U₃Si₂'s are formed distinctly in a matrix of uranium. On the other hand, the results of the splats from the periphery of a chill block display an indication of disturbance of the crystal structure as is shown in Fig. 4(a). This phenomenon becomes more predominant as the particle gets smaller.

![Diffractometry Results](image)

Fig. 6. The results of X-ray diffractometry of splat quenched specimens (a) central part (b) periphery

CONCLUSION

Following the presentation of the paper, "The Powder Heat-Treatment," in the RERTR meeting at San Diego, atomization of uranium silicide was made this year. It was found that silicon is homogeneously distributed among the rapidly atomized powders. The fabrication process is simplified by eliminating comminution and introducing powder heat-treatment.
Atomization is carried out in two ways, one by splat quenching and the other by disk rotating. By atomization, rapid solidification is made in the product powders in which very fine primary $U_3Si_2$'s are formed. The conclusions are the following:

(1) Powder size is controlled varying the inlet rate of molten alloys, rotating speed of a disk, and atmosphere.

(2) As a powder gets smaller, the powder is cooled more rapidly, resulting in the finer primary $U_3Si_2$ structures. The morphology of $U_3Si_2$ in atomized powders shows the dependence of cooling rate in powders of different size.

(3) When the rate of solidification increases, producing smaller $U_3Si_2$ particles, solubility of Si in the uranium matrix increases, which may enhance the heat-treatment of the peritectoid reaction.

(4) As cooling rate increases with finer $U_3Si_2$, a slight indication of crystal disturbance is found in X-ray diffractometry.

REFERENCES

(1) I.H. Kuk, "Efforts to Reduce the Difficulties in Comminution of $U_3Si$ Alloys," 9'th RERTR meeting, San Diego, 1987


ABSTRACT

The objective of this paper is to present the changes which occurred at CERCA's manufacturing facility, related to the use of silicide LEU fuel. Today, CERCA is producing routinely more than twice as many plates as in the year 1988. These plates contain either UAlx or U3Si2 fuel. We will examine how CERCA managed to do so in a short period of time and succeeded to stabilize its production.

Also, we will take the opportunity to present our status of silicide irradiation program. Finally, the increase from 1986 to 1990 in the use of LEU 19,75 % U235 enriched uranium, to produce MTR fuel at CERCA will be exposed.

PART I
STABILIZATION OF A DOUBLED PRODUCTION

As can be seen in slide 1, since 1989, CERCA has been producing more than twice as many plates as in 1988, though the number of U enrichments used in production and the types of fuel plates designs became much more diverse. To do so, CERCA increased its production capacity means:

1. The Team

The number of engineers directly involved in the MTR fuel production increased by a factor of two since 1988. This allows as well closer contact with our clients: some of those engineers, are project engineers, in charge of setting up the technical specifications and following the production of fuel elements of the customers.
In the same period of time, the production and control team has been doubled in size. These well-trained people are now fully operational.

2. Increase of the Workshop Area

Since 1989, 2 successive extensions of the workshop area have been completed. The total increase in area reaches 60%. A part of these extensions has been devoted to the control of the fuel plates and elements, representing a 55% increase of the initial control space area. During these transformations, the production has been kept going.

3. Investment in Fabrication and Inspection Equipment

New equipments have been acquired for the production and the inspection of fuel plates, as can be seen in slide 2. For each type of fuel (UA1x, U3Si2), whatever the enrichment is, CERCA owns complete powder lines, which gives a high grade of flexibility. Important equipments such as rolling mills, power presses, welding machines etc..., have been installed.

Moreover, the use of our recently installed computerized weighing system allows to record all the cores' characteristics (chemical analysis and weights of fuel and aluminum powders) which improves our manufacturing and control processes. The use of a denser fuel (U3Si2) led CERCA to develop special apparatus, such as a new UT control machine. Today also, CERCA owns 2 UT and 2 X Rays' machines used to control respectively the bonding quality and the homogeneity of the uranium distribution in the fuel plate.

PART II
LEU FUEL U3SI2

The use of LEU fuel U3Si2 19.75% U235 enriched has been fully qualified by NRC, German TUEV and other European Authorities, up to a density of 4.8 g U/cc.
1. Status of CERCA's U₃Si₂ Irradiation Program

This is an opportunity to expose the experimental limits that CERCA has been able to reach (slide 3):

- Densities up to 5.4 for U₃Si₂, with a 75 % average burn up
- Average burn up of 82 % for U₃Si₂ 4.8 g Ut/cc (98 % peak burn up)
- Densities up to 5.5 for a mixture of U₃Si₂ and U₃Si, with a 55 % average burn up
- Densities up to 6.0 for U₃Si, with a 54 % average burn up.

This is our experimental program. Since U₃Si₂ 4.8 g Ut/cc has been fully qualified, CERCA focusses today its attention on the use of this fuel on an industrial scale, which explains that no further irradiation test has been carried out with U₃Si₂.

2. Production of U₃Si₂

To ensure its clients a steady and reliable production of U₃Si₂ fuel elements, CERCA had to solve each of the problems related to the change of fuel (slide 4):

First of all, the melting of the ingots has been adapted to U₃Si₂, to get an homogeneous compound, and the production of powders has been optimized.
Specific processes of chemical analysis have been developped in our laboratory and adapted to the standard line.

U₃Si₂ is much denser than UAlₓ. Therefore, when mixing silicide with aluminium powder, the tendency to segregate is more important for U₃Si₂. This would result in the production of heterogeneities in fuel cores, but we have developed techniques allowing to master the homogeneous mixing of the U₃Si₂ and aluminium powders before and during the pressing of the fuel cores.
The homogeneity of the uranium distribution is checked accurately, using CERCA's X Ray machines called HOM RX and examining X Ray films for each plate.

The rolling of the sandwiches induces a defect called dog bone which is more sensitive with high U densities. The design and constitution of the cores as well as cladding and frame and the rolling process have been adapted to reduce this effect.
Specific techniques have been developed to limit the number of stray particles (white spots).

Finally, CERCA is using a UT machine, specially designed for high density fuels to control the absence of decohesion defect. The design of this machine was imperative to be able to inspect high density fuel.
The processes involved in the manufacturing and control of the silicide fuel plates are rather similar to those for aluminides. Nevertheless, the specificities related to U3Si2 imply to solve all the particular problems. Thanks to its constant efforts, in particular in the R & D field, CERCA has been able rapidly to master the whole processes of production.

PART III
THE INCREASE OF LEU FUEL PRODUCTION AT CERCA

1. Quantities of U used

Slide 5 gives the mass percentage of uranium used at CERCA from 1986 to 1990, according to the type of enrichment. It takes into account the raise of production since the end of 1988.

General tendancies may be observed:

- During the last 4 years, we used roughly as much 93 % enriched U as in the year 1986
- In opposite, the proportion of 20 % and 45 % enriched U increased drastically
- Since 1989, the mass of 20 % enriched U is larger than the mass of 93 % enriched U which has been used in 1986.

Slide 5 figures out also 2 important points that CERCA overcame rapidly:

- The production increase by a factor of nearly 3 in U mass
- The diversity of the output increase. The different types of fuels, kinds of plates and elements designs, leads to a much more diverse production. CERCA succeeds to manage simultaneously the manufacturing of very diverse products as well as acceptance inspections by several customers during the same period.

2. Plates Production

Slide 6 presents the same results, through the number of fuel plates produced from 1986 to 1990. It does not show as drastically as slide 5 the increasing use of LEU:

The U235 content for a U3Si2 or UAlx plate is almost the same, but in mass of uranium the difference is quite large.
Nevertheless, slide 6 shows the increasing importance of production of LEU fuel and the same general remarks as for slide 5 can be drawn:

- The doubling of production in 1989
- The importance of LEU and MEU fuels increased
- The diversity of the products increased.

PART IV
COMPARISON OF U3Si2 AND UAlx PRODUCTION COSTS

As soon as CERCA was entering into the production of U3Si2 LEU fuel element on an industrial scale, we tried to estimate the fabrication cost variation when moving from HEU UAlx fuel to LEU U3Si2 fuel. Because the designs, the specifications and the uranium loading are very different from one reactor to the other, it is not so easy to take a particular reactor as an example to get general trends for the others. Nevertheless, examining a specific reactor core in 1987, we estimated that the surplus fabrication cost was around 30% with respect to an highly enriched aluminide fuel. We were just taking into account the cost which are completly under our control: fabrication of plates and elements, excluding U material cost, transportation and scraps recycling costs.

Changing from HEU to LEU, in the case of study, the U235 content was raised in order to overcome the influence of U238. As a counterpart, this allows the element to stay longer in the reactor.

As shown previously, we had to solve the successive technical problems which we were facing, and from time to time, the evaluation of the manufacturing cost surplus seemed to be optimistic.

But finally, our efforts permitted to level down the technical difficulties and to settle a smooth production with yields which are for U3Si2 at least as good as those for UAlx. Of course, this allows us to keep our costs as low as possible. Thanks to our today experience, (since 1986 we manufactured several thousands of silicide plates), we checked that this forecast estimation of 1987 is more or less verified.

This 30% increase is mainly due to:

- The total U content which is roughly 6 times higher with LEU than HEU
- The specificities we exposed previously, concerning the manufacturing and inspection (See chapter II 3).
CONCLUSION

Only two years later, CERCA's production is twice as much as in 1988. All the new means of production and inspection are fully operational and our technical efforts allow us to master the whole process for silicide fuel production in a large scale.

As shown previously, the use of LEU and MEU at CERCA increased drastically in a two years' span.

Today, our R & D efforts are pursued to improve the existing production and inspection technical means, to minimize the production time and to keep the costs as low as possible, for example, improving further the yields of production and installing a scrap recovery facility.
Introduction

I: Stabilization of a doubled production

II: LEU fuel $U_3Si_2$

III: Increase of LEU fuel Production

IV: Production Cost Comparison $U_3Si_2$ vs $UA_{1x}$

Conclusion
Increase of Production of MTR fuel plates produced by CERCA

Level 100 corresponds to the number of plates produced in 1986
Main Additional Equipments

I : Fabrication Equipments :
- Melting Furnace
- Powder Line for each enrichment
- Core Press
- Power-Press
- Hot and Cold Rolling Mills
- Annealing Furnace
- Planing Machine
- EB Welding Machine
- Lathes ...

II : Control Equipments :
- Computarized Weighing System
- X Ray Machine
- Ultrasonic Control Machine
# CERCA U₃Si₂ and U₃Si IRRADIATION PROGRAM STATUS

## ENRICHMENT 20%<sub>c</sub>

<table>
<thead>
<tr>
<th>FUEL</th>
<th>REACTOR</th>
<th>URANIUM DENSITY (g/cc)</th>
<th>NUMBER OF Elt</th>
<th>IRRADIATION START</th>
<th>END</th>
<th>AVERAGE BURN-UP (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>ORR</td>
<td>4.8</td>
<td>1 Element, 1 Element</td>
<td>Apr. 83, Apr. 83</td>
<td>Oct. 83, Aug. 84</td>
<td>56 (54) * 82</td>
<td></td>
</tr>
<tr>
<td>U₃Si₂</td>
<td>2.0</td>
<td>1 Plate</td>
<td></td>
<td></td>
<td></td>
<td>78</td>
</tr>
<tr>
<td></td>
<td>3.7</td>
<td>1 Plate</td>
<td>Jul. 84, Jan. 86</td>
<td></td>
<td>75</td>
<td></td>
</tr>
<tr>
<td>SILOE</td>
<td>5.2</td>
<td>1 Plate</td>
<td>Feb. 86</td>
<td></td>
<td>54</td>
<td></td>
</tr>
<tr>
<td>R2</td>
<td>4.8</td>
<td>1 Element</td>
<td>Nov. 86, Sept. 89</td>
<td></td>
<td>78</td>
<td></td>
</tr>
<tr>
<td>ORR</td>
<td>4.8</td>
<td>20 Elements</td>
<td>Jan. 86, Mar. 87</td>
<td></td>
<td>Up to 52</td>
<td></td>
</tr>
<tr>
<td>HFR Petten</td>
<td>4.8</td>
<td>1 Element</td>
<td>Oct. 88</td>
<td></td>
<td>60</td>
<td></td>
</tr>
<tr>
<td>U₃Si₂</td>
<td>4.7</td>
<td>1 Element</td>
<td>Nov. 86, Jun. 88</td>
<td></td>
<td>83</td>
<td></td>
</tr>
<tr>
<td>OSIRIS</td>
<td>5.5</td>
<td>1 Element</td>
<td>Oct. 88</td>
<td></td>
<td>55</td>
<td></td>
</tr>
<tr>
<td>HFR Petten</td>
<td>5.5</td>
<td>1 Element</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>U₃Si</td>
<td>5.5</td>
<td>2 Plates, 2 Plates</td>
<td>Jun. 82, Nov. 83</td>
<td></td>
<td>56 (53) * 58 (54) *</td>
<td></td>
</tr>
<tr>
<td>SILOE</td>
<td>6.0</td>
<td>2 Plates, 1 Element</td>
<td>Sept. 84, Oct. 85</td>
<td></td>
<td>55</td>
<td></td>
</tr>
</tbody>
</table>

<sup>*</sup> Measured value

*slide 3*
$U_3\text{Si}_2$ Limits Experienced by CERCA:

- Densities up to 5.4 for $U_3\text{Si}_2$, with a 75% burn-up.

- Burn-up of 82% for $U_3\text{Si}_2$ 4.8 g $U^T/$cc

- Densities up to 5.5 for a mixture $U_3\text{Si}_2$ - $U_3\text{Si}$, with a 55% burn-up.

- Densities up to 6.0 for $U_3\text{Si}$, with a 54% burn-up.

(Densities in g$U^T/$cc)
- Compound Homogeneity
- Powder Production
- $U_3Si_2$ - Al Powders Segregation
- Dog Bone Effect
- Control of High Density Fuel
Mass % of U used at CERCA according to U$^{235}$ enrichment

Level 100 corresponds to the total mass of U used in 1986
Number of MTR fuel plates produced by CERCA

Level 100 corresponds to the number of plates produced in 1986.
Cost Comparison between LEU and HEU

130

100

LEU

HEU

1990
SESSION III

September 24, 1990

FUEL IRRADIATION TESTING AND EVALUATION

Chairman:

J. Snelgrove
(ANL, USA)
ABSTRACT

Within the framework of the RERTR-programme four low-enriched MTR-type fuel elements are being irradiated in the High Flux Reactor (HFR) of the Joint Research Centre of the European Commission (JRC) at Petten, The Netherlands. Two of the elements contain U3Si2-Al and two contain U3Si1.6-Al as fuel matrix. The elements will be irradiated up to 75% burn-up. Three of the elements have reached a burnup of 55% to 60% without any appreciable problem. Due to mechanical damage on the outer fuel plate one element (LB02) was temporarily withdrawn from the test irradiations after having reached a burn-up of 20%. The target burn-up of 75% for the remaining three elements will be reached at the end of 1990.

Prior to the actual irradiation a number of reference measurements with respect to reactivity, cooling channel geometry and neutron fluence rate were carried out. Also hydraulic measurements and extensive nuclear and thermal-hydraulic calculations were performed in order to show the thermal safety of the test irradiations. For the determination of the relevant characteristics during irradiation the following periodical investigations are carried out:

- Cooling channel width measurements using an ultra-sonic probe.
- Neutron metrology.
- Reactivity measurements.
- Nuclear calculations (power, burn-up, neutron fluence, safety).
- Visual inspection.

After irradiation up to the specified burn-up level at least one of the elements containing U3Si1.6-Al will probably be subjected to PIE. The first PIE activities will start the second half of 1991.

INTRODUCTION.

The LEU-test irradiations described in this paper are being carried out in the context of the Reduced Enrichment Fuel for Research and Test Reactors (RERTR) programme. The Petten-programme is based upon an agreement between ANL (providing fuel material and taking care of fuel reprocessing) and JRC-Petten (coordinating and financing all irradiation and post irradiation examination activities). The project is managed and executed by the Netherland Energy Research Foundation, while the test fuel elements have been manufactured by CERCA (France) and B&W (USA). The test irradiations will be completed at the end of 1990, while the subsequent post irradiation examination of at least one element will start late 1991 and be finished in the middle of 1992.

2. OBJECTIVE AND SCOPE.

The objectives of the test programme are:
1. To verify the compatibility of the (already qualified) U3Si2-Al fuel applied in two of the test elements (LC03 and LB01), with typical HFR-irradiation conditions.

2. To determine the characteristics and demonstrate the reliability and safety of the more advanced U3Si1.6-Al fuel type, applied in the other two test elements (LC04 and LB02) under HFR irradiation conditions. This fuel type may be applied in medium and high powered research reactors with a demand for higher fuel loadings.

The present scope comprises the fabrication, pre-irradiation characterization and irradiation testing of all four LEU test elements and probably the post irradiation examination of at least one element containing U3Si1.6-Al fuel (LC04/LB02). Test irradiation is carried out in the core of the 45 MWth Petten High Flux Reactor [1] up to a burnup level of 75%.

3. TECHNICAL DESCRIPTION OF THE LEU-TEST ELEMENTS.

The specifications for the four HFR-LEU test elements are partly based on the specifications of the present HFR-HEU elements and partly on the specifications for the ORR-LEU-demonstration core. They have been described in [2].

Figure 1 gives a crosssectional view of the test elements while table 1 shows the main characteristics of the LEU elements as compared with a standard HEU element. Some major features of the test elements are:

- All external dimensions, which assure the proper positioning of the element in the core grid of the HFR, comply with the present standard fuel element dimensions. For technical reasons the north and south pads at the upper side are omitted. The positioning in this direction is assured by the side plates which have been modified.

- In order to obtain the required U-235 loadings, the fuel meat thickness has been increased to 0.76 mm (HFR standard fuel elements: 0.51 mm) for both the inner and outer fuel plates.

- The thickness of the Al fuel cladding remains unchanged i.e 0.38 mm for inner fuel plates and 0.57 mm for the two outer plates.

- The cooling channel width is 2.46 mm (+-0.25 mm) which is slightly larger than that of the present HEU fuel elements (2.18 +/- 0.25 mm).

- To achieve the required minimum cooling channel dimensions and optimal uranium-metal ratio with the thicker plates, the number of plates per element has been reduced to from 23 to 20 plates.
Curved fuel plates (R=140 mm) are applied, similar to the present HEU elements. To facilitate measurements of the cooling channel gap, the cylindrical end piece and the comb, which is normally applied for fuel plate fixation, have been omitted at the upper side of the test elements.

As burnable poison uncladded cadmium wires are installed in longitudinally machined slots in the side plates adjacent to each fuel plate (40 wires per element). The diameter of the wires is 0.5 mm and 0.6 mm for the Cerca and B&W elements respectively.

Provisions for neutron dosimetry consists of four non-reloadable flux monitors installed in longitudinal slots in the side plates. The flux wires have been installed during assembly.

Figure 2 gives a cross sectional view for the LEU test element as compared with that of the present HEU fuel element. In figure 3 the as-built test elements LB01/LB02 are shown. The materials applied in the four test elements are specified in table 2.

4. PRE-IRRADIATION CHARACTERIZATION.

Apart from checks and measurements required for normal fuel element quality control, certain additional measurements and calculations were performed prior to the actual test irradiations:

- accurate fuel plate thickness measurements (manufacturer).
- cooling channel measurements using an ultra-sonic device.
- pressure-drop versus coolant flow measurements in an external loop inclusive dimensional measurements before and after the hydraulic tests (see figure 4).
- detailed neutron fluence rate measurements in the planned irradiation positions. (see figure 5)
- reactivity measurements with the fresh test elements.
- determination of calculation model for the HFR-TEDDI core-analysis code (fig 6 and table 3)
- calculations with respect to burn-up of cadmium and uranium. (fig 7,8)
- nuclear and thermal hydraulic core-calculations. (figure 9 and tables 4,5,6, and 7)

In addition, several representative fuel plates were put aside during the manufacturing for metallurgical characterization of the non-irradiated fuel.
5. IRRADIATION HISTORY AND CHARACTERISTICS.

Based on thermal safety considerations and in connection with the standard HFR core loading pattern the peripheral core positions A2, A8, G2 and G8 have been selected for the irradiation of the LEU test elements (see figure 9). The average power production in these positions varies between 0.5 and 1.0 MW per element, corresponding with maximum heat fluxes of 90 to 150 W/cm² (see tables 4, 5, 6, 7). In order to compensate for power and flux gradients the test elements are periodically switched between north/south and east/west. After each irradiation cycle (25.7 days) cooling channel measurements are carried out in at least three preselected cooling channels of each element. On a regular basis reactivity measurements are performed as well. Detailed neutron fluence measurements are carried out initially and when the cadmium is depleted. Visual inspection of the elements is carried out during each reactor stop period. The complete irradiation history supplemented with burn-up data is presented in the tables 8 up to 11.

6. TEST IRRADIATION SURVEILLANCE RESULTS.

6.1 Visual observation.

Careful, visual inspection of outer appearance of the elements is carried out during each reactor stop period. As confirmed by the cooling channel measurements some minor scratches and deformations on the concave outer plates of all elements could be observed. These scratches which are introduced during normal handling operations do not influence the thermal safety of the test elements. After some 20% burn-up a sharp pit in the outer plate of element LB02 was detected. Visual observation with an underwater camera revealed that the damage was caused by external mechanical forces, presumably due to the various handling operations. Local contact between the two outer plates seems possible. A dark coloured region around the pit confirmed the occurrence of elevated temperatures. It was decided to stop the irradiations of this element until further analyses are carried out. The damaged element has been at full power during at least one reactor cycle without any indication of cladding failure due to overheating. Figures 10, 11 and 12 give an impression of the local plate deformation in element LB02.

6.2 Cooling channel width measurements.

Cooling channel measurements are carried out after each irradiation period for two reasons. First, unforeseen changes in the cooling channel geometry which might influence the coolability of the test elements should be detected as soon as possible. Secondly the minor swelling of the fuel plates as expected from earlier test programmes should be verified by these measurements. A special ultra-sonic measuring device which had been developed for an earlier similar test irradiation programme [3] was applied.
It had to be adapted for curved plates and further the measuring crystal was used in double echo mode (see fig 13) in order to improve signal processing. The measuring method consists of the two-sided transmission of an ultra-sonic wave (8Mc) by a crystal positioned in the middle of the water containing channel. The propagating waves are reflected by the opposite fuel plates, sent again to the opposite plates and finally the reflected waves are received by the same crystal. The two opposite propagation paths have the same physical length, so the two receiving waves arrive at the same time at the crystal element, independent of its position. The measured propagation time is proportional to the cooling channel width.

Three of such crystals are mounted on a traveling probe, which can be moved up- and downwards through the particular cooling channel. In this way three simultaneous scans within one channel can be performed. Provisions have been made for automatic recording of three sensor signals, of the vertical position within the channel and of the water temperature. An aluminium block containing two accurate dimensioned gaps serves as a calibration facility for the actual measurements (see fig 14).

By the use of a mechanical displacement system, the test element can be positioned very precisely with respect to the travelling probe. In this manner each channel can be selected remotely. In figure 15 some details of the measuring device are shown. After each irradiation period the two outer gaps and some preselected inner cooling gaps of each element are scanned. The results are compared with the initial measurements, which cover all cooling channels. Some typical channel width profiles are shown in figures 16 and 17. From these figures it can be observed that the resolution is extremely good (+- 5 um). From the measuring results obtained so far it can be concluded that the width profile for each channel remains very constant with increasing fuel burn-up. Some local deformations in the outer channel can be observed, which are caused by external mechanical point-forces exercised during the frequent handling operations. These local deformations however do not effect the integrity of the fuel plate and have no influence on the coolability of adjacent fuel plates.

The element-averaged decrease in cooling channel width as a function of burn-up is depicted in figure 18. It seems that during the first three irradiation periods an oxide layer of about 50 um is formed on the cladding. Further irradiation has not yielded any significant further change in averaged cooling channel width. Some actual fuel plate dimensions after irradiation will be measured as part of the P.I.E.-programme and will be compared with initial thickness measurements, performed during assembly in the factory.

6.3 Reactivity measurements.

Periodical reactivity measurements have been performed for two reasons:

(I) Determination of the reactivity behaviour of the LEU test elements as a function of burn-up. With these data an
estimate of the core reactivity behaviour in case of LEU can be made.

(II) Determination of the effectiveness of cadmium as a burnable poison.

The measurements have been performed by comparing the reactivity of the LEU-elements with that of a fresh HEU-element (420 g U235 and 1000 mg B10). Central core position D5 was selected for this purpose. From the initial measuring results (see figure 19) it appears that the reactivity of element LB01/LB02 is almost equal to that of the reference HEU-element. The elements LC03 and LC04 have initially a reactivity which is 100 pcm and 220 pcm higher respectively. The difference in absorption of cadmium wires with diameters of 0.5 and 0.6 mm is 120 pcm. Further it can be seen from figure 19 that the point of maximum reactivity for LB01/LB02 is reached somewhat later due to thicker cadmium wires applied. Due to combined burn-up of cadmium and uranium a net reactivity increase of about 300 pcm occurs during the first irradiation phase.

The cadmium wires are depleted after a U235 burnup of about 23%. After this point a more or less linear reactivity behaviour remains with an average decrease of 6 pcm/gr U235. From figure 20 it can be seen that the 40 cadmium wires with 0.5 mm diameter in element LC03 represent a reactivity effect of -800 pcm in position D5. After the cadmium wires have been depleted a more or less linear reactivity behaviour remains due to the burn-up of U235. The reactivity measurements have been carried out with an on line measuring instrument, coupled to the linear channel of the reactor instrumentation. The effects of xenon, samarium and bulk fission products are accounted for in the processing of the measuring data.

6.4 Neutron metrology.

In order to determine the nuclear irradiation characteristics of the LEU-assemblies a neutron metrology programme is carried out. It can be divided into two main parts:

(I) Determination of the total thermal and fast neutron fluence in order to calculate the total burn-up at the end of irradiation. For this purpose four permanent flux monitors have been installed in machined slots in the side plates (see fig. 1). The unloading of the activated detectors will take place during post irradiation examination.

(II) Determination of the detailed flux and power distribution within the fuel region. For this purpose low power flux runs have been performed before and after the cadmium depletion. The first measurements were used for verification of the computer codes applied during the actual irradiations. Figure 5 gives a comparison of the measured and calculated thermal neutron fluence rate distribution. From this figure it can be concluded that there is a good agreement between both results (± 2%). The second flux run has been performed
after some 20% burn-up and was meant to determine the change in local power distribution due to the relative fast depletion of cadmium in the side plates. From these measurements it appears that the effects of power redistribution are largely compensated by the decrease in average power due to U235 burn-up.

The results of the neutron fluence measurements for the total irradiation period will be compared with the results of gamma scanning and chemical burn-up analyses, which may be performed as part of the PIE-programme.

6.5 Nuclear calculations.

Before the start of each irradiation cycle the relevant characteristics of the core loading are calculated with use of a two dimensional diffusion code (HFR-TEDDI). The computed characteristics include o.a. neutron fluence rates, power densities, burn-up and thermal-hydraulic data. For each of the LEU test elements a representative calculation model consisting of a fuel region with a side plate region on the left and right side (see fig.6) has been defined. The nuclear constants of the three regions have been determined with aid of the computer codes MICRO-FLUX-2 (thermal) and GGC-IV (epi-thermal + fast). For the thermal group constants several submodels such as for cadmium wires and for fuel plates have been used. The group constants for the fresh elements are compiled in table 3. For several depletion grades of the U235 and Cd113 contents the nuclear data of this calculation model have been determined and stored on a library file. For the actual U235/Cd113 combination an interpolation procedure is applied.

The depletion of the uranium contents of the LEU elements is being calculated accounting for the burn-up chains of U235 and U238, inclusive the production of plutonium isotopes. The burn-up of the cadmium wires has been determined by performing several MICRO-FLUX runs with a detailed calculation model (see fig.7). The macroscopic absorption crossection of the side plate regions as a function of burn-up (thermal fluence) is presented in figure 8. From the burn-up calculations performed sofar (60%) it appears that about 15% of the energy is produced by Pu239 (see fig.21). Using the same thermohydraulic safety criterium as applied for the HEU elements it appears that the LEU elements in A2,A8,G2 and G8 are less severe than the most critical HEU-element (see tables 5,6). It has also been demonstrated by measurements and calculations that the relatively fast burn-up of the cadmium wires does not lead to more critical thermal-hydraulic conditions as compared to the initial state.

7. CONCLUSIONS.

From the practical experiences, measurements and calculations sofar carried out on the four LEU test elements (50%-60% burn-up) the following main conclusions can be drawn:

75
From cooling channel measurements no significant fission induced swelling effects can be derived. A 50 um thick oxyde layer is formed during the first three irradiation periods. On the concave outer plates of all elements some minor scratches and deformations have been detected. These local deformations which are caused by mechanical forces during the frequent handling operations do not significantly influence the cooling conditions. After having reached a burn-up of about 20% a sharp pit on the concave outer plate of element LB02 was detected. From visual observations it could be concluded that there might be local contact between the two fuel plates and that elevated temperatures have occured around the pit during the last irradiation period. No release of fission products has been detected. It was decided to interrupt the irradiation of this element.

From detailed neutron fluence rate measurements it was found that there is a good agreement between the measured and calculated thermal neutron fluence rate distribution in the fresh LEU elements in their planned irradiation positions. Measurements performed after the cadmium was depleted revealed no significant worsening of the thermal hydraulic conditions.

Reactivity measurements indicated that depletion of the cadmium wires is completed at a thermal neutron fluence of about 0.5 10e25 m-2. The 0.5 mm cadmium wires represent a negative reactivity value of 800 pcm in position D5. The thicker wires applied in LB01/LB02 have a reactivity value of -950 pcm. The relative fast depletion of the cadmium wires leads to an increase in power peaking, which is however compensated by the decrease in average power due to burn-up of U235 and by the flattening of the power profile due to inhomogeneous burn-up. The reactivity measurements are performed on line and are corrected for the influences of xenon, samarium and bulk fission products.

Burn-up calculations indicate an energy contribution of about 15% due to the formation of Pu239 and Pu241 at a burn-up level of 60%. About 20 g. Pu239 and 3 g. Pu241 is present in LEU elements having a burn-up of 60%.

With exception of element LB02 the test elements have withstood the various handling operations without problems. The mechanical and thermal loads during irradiation in the HFR did not yield any major problem for all four elements. The presence of the square upper inlet section is essential for a good execution of the cooling channel measurements and the neutron metrology.
8. REFERENCES.

High Flux Materials Testing Reactor HFR Petten
Characteristics of facilities and standard irradiation devices
EUR 5700eN
1986

for the High Flux Reactor (HFR) Petten.
LEU.T. 0886
Petten, August 1986

Final report on the irradiation testing and post-irradiation
examination of low enriched U308-Al and UA1x-Al fuel elements
by the Netherlands Energy Research Foundation. (ECN)
ECN-85-191
Petten, December 1985
### Table 1: Main Characteristics of Leo Sampling Test Elements (Simons) as Compared with a Standard Leo Element

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<td>CERCA (PS)</td>
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<td>20%</td>
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### Table 2: Material Specifications of Leo Test Elements

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<td>- volume density</td>
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<td>- enrich.</td>
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<td>U-235 MASS</td>
<td>550 &lt;1</td>
<td>625 ±12.5</td>
<td>550 &lt;11</td>
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<td>- per element</td>
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<td>61.5 ±12.5</td>
<td>72.4 ±7.26</td>
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<tr>
<td>- per plate</td>
<td>72.4 ±7.26</td>
<td>61.5 ±12.5</td>
<td>72.4 ±7.26</td>
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<td>DIAMETER (mm)</td>
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<td>BASIS PER ELEMENT</td>
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<td>CO-133 MASS</td>
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<td>AGS H</td>
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<td>- lower end piece</td>
<td>AGS600 H</td>
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<td>- welding additive</td>
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### Table 3. Nuclear Group Constants of Fresh Sodium Test Elements

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<th>Elem. ID:</th>
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#### A. Fuel Part

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<th>SIGL</th>
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#### B. Side Plate Part

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### Table 4. Thermophysical Characteristics of Fresh Sodium Test Elements (625/0.5) as Compared with a Standard NUD Core at Initial State (20/40/2015)

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### Table 5. Nuclear Group Constants of Fresh Sodium Test Elements (625/0.5)

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#### B. Side Plate Part

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### Table 6. Nuclear Group Constants of Fresh Sodium Test Elements (625/0.5)

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#### B. Side Plate Part

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<th>TABLE 6: THERMONUCLEAR CHARACTERISTICS OF SILICON TST ELEMENTS AFTER CADMIUM BURNUP (150/0) AS COMPARED TO A STANDARD HEU CORE AT INITIAL STATE (2010/2019)</th>
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<td>FUEL TYPE</td>
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<td>U-235 MASS (g)</td>
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<td>IN FUEL PART (10^-16 m -2.1-1)</td>
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<td>SIGMA(4) IN FUEL PART (n=1)</td>
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<td>POWER IN FUEL VOLUME (MW)</td>
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<td>TOTAL HEAT TRANSFER SURFACE (m^2)</td>
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<td>AVG. COOLANT VELOCITY (m/s)</td>
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<td>COOLANT CHANNEL WIDTH (mm)</td>
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<td>TABLE 7: THERMODYNAMIC CHARACTERISTICS OF Zirconium Test Elements After Continuous Burnup (550/4) AS COMPARED WITH A STANDARD FEU CORE AT EQUILIBRIUM STATE. (20100/2015)</td>
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CORRELATION OF IRRADIATION-INDUCED MICROSTRUCTURAL CHANGES AND FISSION GAS SWELLING IN URANIUM COMPOUNDS

G. L. Hofman, J. Rest, R. C. Birtcher, and J. L. Snelgrove
Argonne National Laboratory
Argonne, IL USA

ABSTRACT

Intermetallic compounds of uranium, in particular uranium silicides, dispersed in aluminum, are the preferred choice as fuel for a new generation of very-high-flux research reactors because of their high uranium density and expected radiation stability. However, these fuels have never experienced the high fission rates that may be encountered in some reactor designs. This paper addresses the effect of high fission rates on swelling behavior of intermetallic uranium compounds. High fission rates have two opposite major effects on the swelling behavior. A positive effect, i.e., a lowering of the swelling rate at higher fission rates, has been identified in \( \text{U}_3\text{Si}_2 \).

An opposite effect may arise from a previously discovered phenomenon of greatly enhanced swelling upon irradiation-induced amorphization of these compounds. There are indications that compounds that were heretofore not susceptible to amorphization may become so at higher fission rates.

The present paper discusses the results of a multifaceted research effort that includes irradiation tests in a high-flux reactor, neutron irradiation and neutron diffraction studies at a pulsed neutron source, and in-situ charged particle irradiation studies in a high-voltage electron microscope. In addition, theoretical models are being analyzed which will be incorporated in a new computer code that describes the behavior of dispersion reactor fuels under varying conditions.

1. Introduction

Research on the irradiation behavior of intermetallic compounds of uranium was revived several years ago with the decision, by the international research reactor community, to develop weapons-proliferation-resistant fuels. The desired reduction in \( ^{235}\text{U} \) from 93% (High Enrichment Uranium or HEU) to 20% (Low Enrichment Uranium or LEU) necessitates the use of higher-uranium-density fuels to accommodate the additional \( ^{238}\text{U} \) in the LEU fuels.
Intermetallic compounds are well suited for use in research reactor fuel elements, which typically consist of small-particulate fuel dispersed in aluminum. These uranium compounds are generally brittle, facilitating powder fabrication, several have high densities, and their non-fissile elements have good neutronic properties.

This recent research effort has yielded successful low-enriched fuel designs for research and test reactors with high total uranium loadings. Because the anticipate operating conditions in ultra-high-flux reactors now being designed, i.e., high temperature and high fission rates, lie beyond the ranges tested in the reduced enrichment program, additional research has been undertaken to ensure successful operation of these fuels under more severe conditions. The present paper deals with this effort, specifically as it relates to irradiation-induced swelling of the fuel due to fission gas.

2. Irradiation Behavior in Current High-Flux Reactors

The irradiations were carried out in the Oak Ridge Research Reactor (ORR) at a peak thermal neutron flux density of $2 \times 10^{14} \text{ cm}^{-2} \text{ s}^{-1}$. All fuels consisted of particles which passed through a 150-μm sieve. The particles were dispersed in pure aluminum or rolled into aluminum 6061-clad plates. The maximum fuel meat temperature during irradiation was estimated to be approximately 120°C, based on the model of homogeneous fuel meat having the thermal conductivity measured for the bulk heterogeneous meat.

Fuel swelling was determined from postirradiation immersion volume measurements on the plates using the Archimedes method. In calculating the fuel particle swelling from these measurements, appropriate corrections for corrosion of the cladding and the presence of as-fabricated porosity in the dispersion were made. It was also verified that the contributions to plate volume changes due to aluminum swelling and fuel-aluminum interaction were negligibly small.

Swelling of several intermetallic compounds of uranium is shown in Fig. 1 as a function of the irradiation dose expressed as fission density. Clearly the higher-density compounds, $\text{U}_6\text{Fe} @ 17.5 \text{ g cm}^{-3}$ and $\text{U}_3\text{Si} @ 15.4 \text{ g cm}^{-3}$, are much less capable of accommodating large volume fractions of generated fission gas than the lower-density compounds, $\text{U}_3\text{Si}_2 @ 12.2 \text{ g cm}^{-3}$ and $\text{USi} @ 10.8 \text{ g cm}^{-3}$.

The two higher-density compounds, $\text{U}_6\text{Fe}$ and $\text{U}_3\text{Si}$, exhibit a sharp increase in swelling rate (breakaway swelling) at, respectively, $\sim 3 \times 10^{27}$ and $5 \times 10^{27}$ fissions m$^{-3}$, ($\sim 40\%$ and $\sim 75\%$ $^{235}\text{U}$ burnup) while, in contrast, no breakaway swelling has occurred in $\text{U}_3\text{Si}_2$ up to an exposure of $5.3 \times 10^{27}$ fissions m$^{-3}$ (95% $^{235}\text{U}$ burnup). In addition, the swelling of $\text{U}_3\text{Si}_2$ and, also, USi is lower throughout the whole burnup range.
As shown in Fig. 2, the differences in swelling behavior of these compounds can be traced to the development of large fission gas bubbles in U₆Fe and U₃Si. At first glance, there appears to be an (unfortunate) correlation between higher compound density and the occurrence of breakaway swelling. Although it is true that the structure of the compounds which determines their density also may control the behavior of fission gas in them, further analysis of the data indicates that a more-detailed mechanistic picture of the irradiation behavior may be developed. Analyses of the fission gas bubble growth suggest that the compounds can be divided into two classes: one consisting of compounds that exhibit breakaway swelling (which we shall call unstable swelling compounds) and another containing compounds with stable swelling characteristics. Direct and circumstantial evidence indicates that different irradiation-induced changes in the crystal structure of the compounds cause their different behavior.

2.1 Bubble Formation

In order to gain more insight into the formation of fission gas bubbles, the various compounds were examined with a scanning electron microscope. Samples were obtained by fracturing small sections of irradiated fuel plates through the fuel core. This yielded many suitable transgranular fracture surfaces of the fuel particles.

It is at once clear from the comparison of U₃Si and U₃Si₂, shown in Fig. 3, why the swelling of U₃Si₂ is both lower and more stable compared to U₃Si. The small gas bubbles in U₃Si₂, which are not visible in the optical image shown in Fig. 2, form a dense and rather regularly spaced population without evidence of interlinkage up to virtually complete burnup of the ²³⁵U in the LEU fuel.

2.1.1 Unstable Swelling Compounds

In searching for a possible explanation for the unstable behavior of U₃Si and U₆Fe, it was discovered that these compounds become amorphous during irradiation and subsequently exhibit very plastic, viscous, irradiation-induced behavior at low temperatures, where they are normally hard and brittle [1]. This effect was verified for U₃Si by in-situ Kr ion bombardment in a high-voltage electron microscope (HVEM) [2]. As shown in Fig. 4, U₃Si can be rendered amorphous with relatively low radiation-damage dosage below a certain temperature (the glass transition temperature).

Also, enlargement of perforations in the sample shown in Fig. 5 under continued bombardment in the metallic glass state indicated a large plastic flow in the amorphous material. No such flow was detected in samples irradiated above the glass transition temperature, i.e., in the crystalline state. Figure 6 shows the Dispersion Analysis Research Tool (DART) code-calculated gas bubble size distribution in irradiated amorphous U₃Si at a fission density of about 6 x 10²⁷ fissions m⁻³. Also shown in Fig. 6 is the DART-calculated bubble distribution in U₃Si assuming it remains crystalline.
DART is a mechanistic model for the prediction of fission product-induced swelling in dispersion fuels. DART calculates the irradiation-induced fission gas bubble size distributions as a function of fuel morphology, as well as solid fission product swelling. The gas-driven swelling models in DART are based on those in the GRASS-SST code [3]. The DART mechanical/stress model consists of spherical fuel particles surrounded by matrix aluminum. The model treats the inner fuel sphere as an elastically deformable body and the surrounding shell of Al-matrix material as perfectly plastic. During the irradiation, the generation of fission products induces elastic deformation of the fuel particles. First, the as-fabricated pores in the matrix material close in response to fuel particle volume expansion; then, plastic flow/swelling of the aluminum occurs. The calculation of swelling for U$_3$Si shown in Fig. 6 assumed that plate delamination occurred allowing unrestrained swelling of the core. The calculations show that as the irradiation proceeds the U$_3$Si bubble size distribution evolves to larger sizes, eventually leading to breakaway swelling. The kinetics of swelling in crystalline U$_3$Si results in the gas occupying very small bulk bubbles only. The analysis of fission gas behavior in irradiated amorphous U$_3$Si suggests that irradiation softening combined with bubble over-pressurization effects results in very large plastic flow. The effect of plastic flow has been modeled by an effective diffusivity. Enhanced diffusion leads to very rapid nucleation of gas bubbles and a rapid coarsening of the bubble size distribution.

2.1.2 Stable Swelling Compounds

The swelling of the densest of the low-swelling compounds, U$_3$Si$_2$, is illustrated in more detail in Fig. 7. The swelling curve of this material exhibits a distinct "knee" that shifts to a higher fission density with increased fission rate due to higher enrichments (MEU is 40% enriched). Below the "knee" no gas bubbles can be detected by the SEM and if present they must be below the resolution limit of the instrument, i.e., smaller than -0.04 μm in diameter. At the "knee" gas bubbles form in a heterogeneous fashion, as shown in Fig. 8. Above the knee, the bubble population rapidly multiplies and the bubble size increases linearly with fission density for all three enrichments (see Fig. 9).

The bubble morphology retains its stable character up to the highest fission densities, i.e., rather uniform size and bubble spacing with no sign of interlinking of bubbles, as illustrated in Fig. 10. The bubbles also appear to form short, linear, intersecting patterns suggestive of an association with an underlying microstructural feature.

A bubble population with the observed bubble diameter can only be calculated if microstructural features such as grain boundaries or dislocation networks are introduced in the code. However, the original grain size of the fuel is large, on the order of the fuel particle size; thus some microstructural modification (other than amorphization) must be invoked to explain the observation. At this time, there is no direct evidence of such a restructuring of crystalline U$_3$Si$_2$. However, some indirect evidence from other experiments leads us to speculate that a dense network of subgrain boundaries forms at
a dose corresponding to the "knee" in the swelling curve, upon which the previously "invisible" gas bubbles form and then grow at an accelerated rate. Formation of subgrains was observed by Bleiberg [4] in UO$_2$. He showed that original 10-20 µm grains subdivided into ~1 µm subgrains at approximately $2 \times 10^{27}$ fissions m$^{-3}$ while retaining their crystalline structure. UO$_2$ swelling also increased in rate at this point, and, most interestingly, the bubble morphology is strikingly similar to that observed in U$_3$Si$_2$ and USi. One of the present authors has also observed transformation of irradiated uranium oxide to very small grains, as shown in a SEM fractograph in Fig. 11. Further evidence of this phenomenon was found in our ion bombardment studies on U$_3$Si above the glass transition temperature, as shown in Fig. 12. It seems plausible that formation of small subgrains, on the order of 1 µm in diameter, also takes place in U$_3$Si$_2$. Figure 13 shows DART-calculated results for fuel particle swelling of low-enriched U$_3$Si$_2$-Al fuel plates as a function of fission density. The calculations shown in Fig. 13 were made assuming that fuel restructuring occurred at about $3 \times 10^{27}$ fissions m$^{-3}$ with subsequent subgrain boundary formation characterized by an average grain size of 1 µm. Also shown in Fig. 13 are the measured swelling values. As in be seen in Fig. 13, the assumption of subgrain boundary formation provides a calculated swelling consistent with observation.

The changes in swelling rate, observed bubble morphology, and overall magnitude of swelling can thus be explained. Why the "knee" shifts to higher dose with increasing enrichment (dose rate) remains unclear at this point. If the microstructural transformation were dose dependent, one would expect the "knee" to occur at the same dose for all enrichments. The fact that it does not indicates a more complex kinetic behavior. It is interesting to note that the "knee" occurs at approximately the same irradiation time for all enrichments. Work on this aspect of the swelling is continuing.

3. Assessment of Swelling Behavior at Very High Fission Rates

The highest density compounds, U$_6$Fe and U$_3$Si, are clearly not suitable for use in ultra-high flux reactors that require thin plate-type elements with high loadings of $^{235}$U. Only with designs employing strong mechanical restraint, such as thick-wall tubular elements, can the swelling of these fuels be kept to acceptable levels. The obvious choice then, is the most dense of the stable compounds, viz. U$_3$Si$_2$. However, the anticipated operating conditions of one ultra-high flux reactor now being designed will lie well beyond the range at which U$_3$Si$_2$ has been tested. The present data were obtained from irradiations in the ORR at a peak fission rate of $8 \times 10^{20}$ fissions m$^{-3}$ s$^{-1}$ and a maximum fuel temperature of approximately 120°C, whereas the fuel in the new reactor would operate at up to 400°C, with a peak fission rate of $\sim 3 \times 10^{22}$ fissions m$^{-3}$ s$^{-1}$.

Although we have not found any indication of instability in the swelling behavior of U$_3$Si$_2$, it is not prudent to base a reactor fuel design on an extrapolation of this behavior
so far beyond the existing database. It is for this reason that we are carrying out the modeling work and have extended the experimental work in several areas.

Because of the unstable swelling behavior of amorphous U₂Si, the question arises as to whether U₃Si₂ might undergo a similar transformation at higher damage rates and higher temperatures. Ion bombardment experiments in the HVEM were performed to determine if amorphization occurs and, if so, what the glass transition temperature of this compound is. Results indicate that like U₃Si, U₃Si₂ becomes amorphous during 1.5-Mev Kr irradiation at approximately the same damage dose (0.1 dpa) at a damage rate of 7 x 10⁻³ dpa s⁻¹ and that the glass transition temperature is similar (~250°C). The data points for U₃Si₂ are included in Fig. 4.

Thus, the Kr bombardment results raise the concern about the possible amorphization of U₃Si₂ at high fission rates encountered in the peak power regions of an ultra-high-flux reactor, e.g., 3 x 10²² fissions m⁻³ s⁻¹ or 5 x 10⁻² dpa s⁻¹. In these high power regions the fuel might operate at temperatures of up to 400°C, well above the measured glass transition temperature, and, therefore, the fuel in those regions will likely remain crystalline. Because of the apparent importance of subgrain formation (or grain refinement) on the swelling behavior, further Kr bombardment studies are being performed on U₃Si₂ in the crystalline state (above 250°C) to characterize the grain restructuring in this material.

Ion bombardment alone cannot provide all the answers to the complex question of irradiation damage because fission damage may be different in detail from ion-beam damage. In order to further characterize the irradiation behavior of the fuel under more typical conditions in an ultra-high-flux reactor, an irradiation experiment has been run in the Oak Ridge High Flux Isotope Reactor (HIFR). This experiment consisted of small U₃Si₂- and U₂Si-aluminum dispersion samples irradiated to the projected fission density of an ultra-high-flux reactor, at typical temperatures as well as below the glass transition temperature as measured by the Kr bombardment experiment. The maximum fission rate was approximately one-half that of the peak fission rate of the reactor currently being designed, but above the equivalent damage rate of the Kr irradiations. The irradiation of the HIFR experiment was completed in September 1990, and the postirradiation examinations will yield information on fission gas bubble morphology, delineating possible stable and unstable behavior.

In addition, a third experiment is in progress with both U₃Si and U₃Si₂ in the ANL Intense Pulsed Neutron Source (IPNS). The samples are irradiated in small-dose steps, after which neutron diffraction is performed. The purpose of this experiment is to determine structural changes in the compounds during the initial stages of the fission damage process.
Rietveld analysis of the neutron diffraction patterns yields detailed measurements of changes in lattice parameters as a function of fission density, as shown in Fig. 14. The unit cell of U₃Si is enlarging while that of U₃Si₂ is contracting. Although these initial results do not yet lead to any firm conclusions on the mechanism of radiation damage in U₃Si and U₃Si₂, it is hoped that the diffraction data coupled with HVEM-Kr bombardment results will lead to a satisfactory explanation. Changes in the diffraction pattern of U₃Si, also shown in Fig. 14, indicate some of the damage processes involved. Two orthogonal reflections from the original tetragonal unit cell are merging into a single, cubic, reflection while a broad diffuse background is building up. This suggests that amorphous regions are accumulating, presumably at the core of the damage cascades while the surrounding crystalline material is transforming from tetragonal to a high-temperature cubic phase. The diffraction patterns of U₃Si₂ have not as yet yielded a clear picture. It appears, however, that the fission-induced crystallographic changes in U₃Si₂ are quite different from those in U₃Si.

Finally, the swelling stability of U₃Si might not be so critical if the fuel loading can be kept below about 15 vol%. Such a low loading results in principle in much greater mechanical restraint of fuel particle swelling due to the surrounding aluminum matrix and in elimination of inter-particle fission gas bubble growth (which can lead to breakaway swelling of the fuel core) due to relatively large particle separation.

Results of DART calculations indicate that, as expected, mechanical restraint is very important in controlling the swelling of amorphous U₃Si. For the case shown in Fig. 15, the yield strength of the aluminum matrix was arbitrarily increased by a factor of two to simulate irradiation hardening and illustrate the restraining effect of the matrix. The DART code is being refined to include irradiation-enhanced creep and swelling of aluminum because these deformation mechanisms will likely be important in the hard neutron spectrum of a typical ultra-high-flux reactor. Some limited ORR irradiation data of highly enriched U₃Si and U₃Si₂ at 14 vol% dispersant loadings indicate that the difference in fuel plate core swelling is indeed not large for the two fuel compounds (see Fig. 15).

4. **Conclusions**

High density compounds such as U₆Fe and U₃Si exhibit breakaway swelling behavior and are not suitable for use as dispersants in plate type fuel requiring high-fissile loading. The breakaway swelling of these compounds is attributed to an irradiation-induced crystalline-to-amorphous transformation.

The medium-high density compound, U₃Si₂, was found to have moderate and stable swelling. It is characterized by a two-stage behavior, possibly associated with irradiation-induced grain refinement of the crystalline compound. The swelling of U₃Si₂ appears to be dose-rate dependent.
A fuel plate behavior model, DART, was developed. The code calculations are in reasonable agreement with measured data when conventional fission gas models are modified to include amorphization and grain refinement. Kr ion bombardment experiments indicate that U₃Si₂ also transforms to an amorphous structure at high dose rates, raising concern for high-flux applications. Neutron irradiation and neutron diffraction experiments are being performed to gain better understanding of the differences or similarities between irradiation damage in U₃Si and U₃Si₂ in order to confidently extrapolate the lower dose rate behavior to ultra-high-flux reactor conditions.

References


Acknowledgements

The authors are thankful for the services of the Argonne Hot Cell personnel under L. A. Neimark, the test fuel fabrication work of T. Wieneck and the late R. F. Domagala, and the coordination of irradiation experiments by G. L. Copeland, M. Mueller, and J. Richardson.
Fig. 1. Swelling of various LEU intermetallic fuel compounds as a function of accumulated fissions in fuel.
Fig. 2. Optical metallography of various intermetallic fuel compounds, showing unstable and stable swelling behavior.

$\text{U}_3\text{Si}_2$: $> 4.9 \times 10^{27} \text{fiss. m}^{-3}$ (> 90% Burnup)

$\text{U}_3\text{Si}$: $4.8 - 5.5 \times 10^{27} \text{fiss. m}^{-3}$ (70 - 80% Burnup)

$\text{U}_5\text{Fe}$: $2.2 - 3.0 \times 10^{27} \text{fiss. m}^{-3}$ (30 - 40% Burnup)
Fig. 3. Bubble morphology in $U_3Si$ (upper) and $U_3Si_2$ at $5 \times 10^{27}$ fiss. m$^{-3}$.
Fig. 4. Damage dose to render U₃Si amorphous as a function of temperature, showing glass transition temperature at ~250°C, 1.5-MeV Kr ion bombardment (U₃Si₂ data discussed later in text).
Fig. 5. Evidence of plastic flow during Kr ion bombardment of amorphous U₃Si.
Fig. 6. DART calculation of fission gas bubbles in crystalline and amorphous U₃Si.
Fig. 7. Swelling of $\text{U}_3\text{Si}_2$ of various enrichments and fuel dispersion loadings as a function of fission density (USi data included).
Fig. 8. First evidence of fission gas bubbles in LEU $U_3Si_2$ just at the "knee" in the swelling curve.
Fig. 9. Average fission gas bubble diameter as a function of fission density in LEU, MEU, and HEU U$_3$Si$_2$. 
Fig. 10. Fission Gas bubble morphology in LEU, MEU and HEU $\text{U}_3\text{Si}_2$ irradiated to various fission densities.
HEU @ $9.5 \times 10^{27}$ fis. m$^{-3}$

HEU @ $14 \times 10^{27}$ fis. m$^{-3}$

Fig. 10. (Continued)
Fig. 11. Fracture surface of uranium oxide particles before and after irradiation, showing grain refinement.
Fig. 12. TEM images of U₃Si₂ before and after Kr ion bombardment showing grain refinement.
Fig. 13. Swelling of U$_3$Si$_2$ calculated with DART (dashed line without grain refinement).
Fig. 14. Neutron diffraction data of neutron-irradiated $U_3\text{Si}$ and $U_3\text{Si}_2$. 
Fig. 15. Comparison of core swelling in 14 vol% U₃Si- and U₃Si₂-aluminum dispersion fuel plates, ORR irradiation data and DART calculations for U₃Si (solid lines).
MECHANISTIC MODEL FOR THE IRRADIATION
BEHAVIOR OF DISPERSION FUEL
FOR RESEARCH REACTORS*

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ABSTRACT

Indirect evidence from various experiments leads the authors to speculate that a dense network of subgrain boundaries forms at a dose corresponding to the "knee" in the irradiated uranium silicide dispersion fuel swelling curve; fission gas bubbles nucleate at the boundaries and then grow at an accelerated rate relative to that in the bulk material. Compositional changes induced by the burnup of uranium could provide a basis for the formation of relatively small grained crystals. In this paper, an alternative formulation is presented wherein the stored energy in the material is concentrated in a network of "recrystallization" sites that diminish with dose due to interaction with radiation-produced defects (vacancy-impurity pairs). Recrystallization is induced when the energy per site is high enough that the creation of grain boundary surfaces is offset by the creation of strain-free volumes with a resultant net decrease in the free energy of the material. This formulation is shown to provide a reasonable interpretation of the observed phenomena.

The DART thermal conductivity model has been improved to better account for the dependence of the thermal conductivity on both the as-fabricated and the irradiation-induced porosity. The original model smeared the irradiation-induced porosity throughout the meat, which resulted in a stronger dependence than was physically reasonable. The modification described in this report removes this objection, while at the same time retaining the strong dependence on the as-fabricated porosity characteristic of the unirradiated U$_3$Si$_2$ data. For completeness, and to establish a proper introduction to the modified calculation, the derivation of the fundamental mixing formulas used in the analysis is provided.

Introduction

Recent experimental observations on low-temperature swelling of irradiated uranium silicide dispersion fuels have indicated that the growth of fission gas bubbles appears to be affected by fission rate. The swelling behavior of U3Si2 is illustrated in detail in Fig. 1. The swelling curve of this material exhibits a "knee" that shifts to a higher fission density with increased fission rate. Below the "knee," no gas bubbles can be detected by SEM, and if present they must be below the resolution limit of the instrument, i.e., smaller than \( \approx 0.04 \) mm in diameter. Just at the "knee," gas bubbles are first seen to form in a heterogeneous fashion, as shown in Fig. 2. Above the knee, the bubble population rapidly multiplies and the bubble size increases linearly with fission density for all three fission rates (see Fig. 3). The knee for the highest fission rate data shown in Fig. 1 was estimated based on the results for the lower fission rate data, as well as the gas bubble size data shown in Fig. 3.

The bubble morphology retains its stable character up to the highest fission densities, i.e., rather uniform size and bubble spacing with no sign of interlinking of bubbles, as illustrated in Fig. 4. The bubbles also form short, linear, intersecting patterns suggestive of an association with an underlying microstructural feature.

Current state-of-the-art models for fission gas behavior do not predict a dependence of bubble growth on fission rate. Calculations with GRASS-SST\(^1\) have interpreted the swelling as due to the combined effects of a population of bubbles below the limits of experimental resolution and a distribution of larger, visible bubbles pinned to dislocations and subgrain boundary networks.\(^2\) The position of a peak in the bubble size distribution is determined by the offset between the growth of the bubbles due to diffusion of gas atoms and shrinkage due to fission-fragment-induced re-solution. Both irradiation-enhanced diffusion and gas atom re-solution have an approximately linear dependence on the fission rate. Therefore, an increase in the rate alone would not significantly affect the position of a bubble peak and thus gas-bubble swelling. Sensitivity studies have also indicated that a large change (hypothetically many orders of magnitude instead of approximately linear) in bubble nucleation rate at higher dose rates would not affect fuel swelling appreciably.
A bubble population with the observed bubble diameter can only be calculated if microstructural features such as subgrain boundaries, phase boundaries, or dislocation networks are introduced (the original grain size of the fuel is large, i.e., on the order of the fuel particle size). Indeed, the short, linear intersecting pattern of bubbles shown in Fig. 2 is suggestive of bubbles associated with microstructural features such as grain boundaries. In addition, to provide for an interpretation of the observed rate dependence of swelling, the formation of these defect structures should be a function of dose rate as well as dose. At this time, there is no direct evidence for such a restructuring of crystalline U₃Si₂. However, some indirect evidence from other experiments leads us to speculate that a dense network of subgrain boundaries forms at a dose corresponding to the “knee” in the swelling curve, upon which gas bubbles nucleate and then grow at an accelerated rate relative to that in the bulk material. Formation of subgrains was observed by Bleiberg in uranium oxide. He showed that in relatively low power UO₂ at very high burnup original 10⁻²⁰ mm grains subdivided into ≈1 mm subgrains at approximately 2 x 10²⁷ fissions m⁻³ (≈10 at.% U atoms fissioned) while retaining their crystalline structure. UO₂ swelling also increased in rate at this point (i.e., a “knee” in the swelling curve!) and, most interestingly, the bubble morphology is strikingly similar to that observed in U₃Si₂. One of the present authors (G. L. Hofman) has also observed transformation of irradiated UO₂ to very small grains, as shown in the SEM fractograph in Fig. 5. Further evidence of this phenomenon has been found in ion bombardment studies on U₃Si above the glass transition temperature. It seems plausible that formation of small subgrains, on the order of 1 mm in diameter, also occurs in U₃Si₂.

U₃Si₂ specimens irradiated with 1.5 MeV Kr ions in a high voltage electron microscope were observed to go amorphous at temperatures lower than 200°C. In addition, the amorphization of U₃Si₂ at low temperature was inhibited when the material was initially irradiated at temperatures above the glass transition experiment. The resolution of the apparent anomaly between the high electron microscope irradiations, and the in-reactor irradiations is thought to be the existence at the onset of the in-reactor irradiation of high fuel particle temperatures (>200°C) due to the existence of a gap between the U₃Si₂ particles and the matrix material, with a subsequent decrease in temperatures to around 100°C upon gap closure.
**Theory**

The dispersion analysis research tool (DART) is a mechanistic model for the prediction of fission-product-induced swelling in dispersion fuels. DART calculates the size distributions of the irradiation-induced fission gas bubble as a function of fuel morphology, as well as solid fission product swelling. The gas-driven swelling models in DART are based on those in the GRASS-SST code. The DART mechanical/stress model consists of spherical fuel particles surrounded by matrix aluminum. The model treats the inner fuel sphere as an elastically deformable body and the surrounding shell of Al-matrix material as perfectly plastic. During irradiation, the generation of fission products induces elastic deformation of the fuel particles. First, the as-fabricated pores in the matrix material close in response to fuel particle volume expansion; then plastic flow/swelling of the aluminum occurs.

In searching for an interpretation of the phenomena shown in Figs. 1-4, it is important to point out that significant compositional changes are occurring in the material as the irradiation proceeds. For example, as the uranium is burned up, the composition of the fuel moves into the $\text{U}_3\text{Si}_2$-$\text{USi}$ two-phase region of the equilibrium phase diagram. The possible formation of relatively small grained USi can provide boundaries upon which enhanced gas bubble growth and swelling can occur. The viability of compositional changes as the mechanism for the “knee” as well as for providing a rate dependence of the swelling is currently being assessed. The following model for grain recrystallization is being proposed as one possibility for the interpretation of the phenomena. The results of future experiments and analyses will be used to examine differing descriptions of the phenomena, as well as uncertainties in selected materials properties.

Recrystallization is a common phenomena in the cold working of steels and many other types of materials. The initiation of recrystallization usually (but not always) may be traced to the abnormal growth of some particular cell formed by deformation and/or recovery. Thus, potential sites where recrystallization can occur are not formed by statistical fluctuations, but are in fact already present in the structure of the material. In general, the recrystallization process has as its driving force the increased energy of a matrix that has been plastically deformed. The newly evolving grains are strain-free and the migrating boundaries can absorb dislocations as they move through the strained material.
The concept that recrystallization nuclei could originate from small pre-formed blocks of "strain-free" crystal within the microstructure of a deformed state was proposed by Cahn and Beck. These "strain-free" blocks within a crystal are formed by the spatial rearrangement of dislocations into lower energy arrays by polygonization. This model for recrystallization nuclei was further modified by Cottrell who suggested that an additional necessity for the formation of a successful recrystallization nucleus was likely to be the development of a highly mobile and thus, in general, high angle bounding interface. These ideas have led to the current understanding that a viable recrystallization nucleus will have a size advantage compared to the neighboring subgrain structure and will be surrounded, at least in part, by a highly mobile interface. Doherty and Cahn analyzed a two-dimensional situation where a "subgrain" becomes bounded by a high angle (mobile) interface but lacks a necessary size advantage. By assuming that the surface energy of the high angle boundary was three times that of the surrounding subgrain boundaries, these authors noted that, for a situation involving equilibrated triple junction angles, the linear size of the potential nucleus would have to be approximately three times that of the rest of the population of the subgrains.

Based on the above discussion, assume the existence of $c_s$ recrystallization sites per unit volume of material. These sites are further assumed to form relatively early in the irradiation period and to be associated with triple points formed from a low angle boundary substructure. The recrystallization sites act as sinks for irradiation-produced defects. Radiation-induced compositional changes in the vicinity of voids, dislocation loops, and external surfaces have been observed in a number of irradiated metals and alloys. Frequently, the segregation leads to the precipitation of a second phase in voids and dislocation loops. It is assumed here that the impurity atoms require considerable energy to become part of a dumbbell interstitial and therefore do not migrate via an interstitial mechanism. It is further assumed here that long-range diffusion of vacancy-impurity pairs to the immobile sites eliminates the sites (e.g., removes the site as a viable recrystallization nucleus by destroying the mobility of the interface at a rate given by

$$\frac{dc_s}{dt} = -K_{sm} c_s c_m.$$  \hspace{1cm} (1)

where $K_{sm}$ is the reaction rate for the immobilization of recrystallization sites by vacancy-impurity pairs,
\[ K_{\text{sm}} = 4\pi r_{\text{sm}} D_{v} / \Omega, \tag{2} \]

\( r_{\text{sm}} \) is the annihilation radius of a recrystallization site/vacancy-impurity pair, and \( c_{m} \) is the pair concentration. Assuming that the concentration of vacancy-solute pairs is in steady state with the concentrations of vacancies, interstitials, and solute atoms leads to \(^{14}\)

\[ c_{m} = R_{v} c_{v} c_{i}. \tag{3} \]

where \( c_{v} \) is the vacancy concentration, \( c_{i} \) the impurity concentration, and \( R_{v} \) is the reaction rate for the formation of vacancy-solute pairs in thermal equilibrium. An equilibrium concentration of mobile defects is reached relatively early in the irradiation.\(^{15}\) Studies made by the present authors utilizing a chemical rate theory of solute segregation\(^{17}\) for an impurity concentration which is dependent on the fission (or dpa) rate indicate that the above conclusion for a fixed impurity concentration also holds for the irradiation conditions discussed in this paper. The equilibrium concentration of mobile defects within the bulk material, \( c_{v} \), can be determined from the rate equations describing point-defect behavior. At the low irradiation temperature of the \( U_{2}Si_{2} \) fuels, mutual recombination will be dominant over annihilation at internal sinks, and the appropriate solution of the rate equations is \(^{16}\)

\[ c_{v} = \left( \frac{\Omega}{4\pi r_{iv} D_{v}} \right)^{1/2} K^{1/2}. \tag{4} \]

where \( D_{v} \) is the random-walk diffusion coefficient of vacancies, \( \Omega \) is the atomic volume, \( K \) is the production rate of Frenkel pairs, and \( r_{iv} \) is the radius of the recombination volume.

Using Eqs. 3 and 4 in Eq. 1 gives

\[ \frac{dc_{s}}{dt} = -K_{\text{sm}} f K^{1/2} c_{s}. \tag{5} \]

where

\[ f = \left( \frac{\Omega}{4\pi r_{iv} D_{v}} \right)^{1/2} R_{v} c_{i}. \tag{6} \]

Upon integrating Eq. 5,
where \( C_s \) is the concentration of nucleation sites at \( t = 0 \).

The concentration of viable recrystallization sites represented by Eq. 7 is quite different from that given by classical nucleation theory in that the concentration decreases with fluence instead of increasing with irradiation until the nucleation barrier is surmounted and the higher-energy state of the crystal forms. In the present case, the sites are formed early on in the irradiation by the damage process at relatively low values of the strain energy. Subsequently, the available stored energy in the material is limited to the relatively plastic nature of the irradiated material and is not available to increase the density of sites. As the irradiation proceeds and the sites are eliminated by interaction with the vacancy-solute pairs, the available stored energy is concentrated on fewer sites (i.e., one can consider the sites as holes in the material acting as stress concentrators), thus increasing the energy per site.

For a given value of stored energy, \( E_s \), the nucleation of a new crystal of material results in a net change in free energy given by

\[
\Delta G = -E_s V + 8\pi \gamma \left( \frac{3}{4\pi} V^{2/3} \right),
\]

where \( V \) is the volume of the newly nucleated crystal (a spherical crystal shape has been assumed for simplicity). The first term on the right side of Eq. 8 is the decrease in free energy due to the creation of a strain-free volume, \( V \), and the second term is the work required to create the boundary surface with surface energy density \( \gamma \). \( \Delta G \) has a maximum at a value of \( V \) given by

\[
V_{\text{max}} = \frac{4\pi}{3} \left( \frac{4\gamma}{E_s} \right)^3.
\]

where the value of \( \Delta G \) at \( V_{\text{max}} \) is

\[
(\Delta G)_{\text{max}} = \frac{128\pi \gamma^3}{3 E_s^2}.
\]

From Eqs. 9 and 10, it is seen that as \( E_s \) increases, \( V_{\text{max}} \) and \( (\Delta G)_{\text{max}} \) shift to smaller values. When \( (\Delta G)_{\text{max}} \) decreases to a value on the order of
the thermal energy, kT, a relatively small energy fluctuation can allow the
system to jump over the "barrier" (DG)_{max}, and a new crystal of material will
be created.

As (ΔG)_{max} approaches kT, the stored energy, which is taken to be
concentrated in the network of nucleation sites with density c_s, is assumed to
have a rate of change with respect to a change in c_s given by Boltzmann's law,
i.e.,

\[ \frac{dE_s}{dc_s} = -\frac{kT}{c_s}. \tag{11} \]

Integrating Eq. 11 results in

\[ c_s = e^{-E_F/kT}. \tag{12} \]

where E_s has been formally identified with the formation energy of the new
crystal, E_F. In principle, irradiation creep processes can provide additional
contributions to the stored energy term in Eq. 8. However, materials such as
U3Si2 and UO2 have been observed to have a high degree of plasticity under
irradiation. Plastic flow processes during irradiation will inhibit the
buildup of significant strain energy densities.

Equating the expression obtained for c_s by a kinetic analysis, Eq. 7, to
the expression obtained by a thermodynamic analysis, Eq. 12, results in an
expression for the fission density at which the "knee" occurs (onset of
subgrain boundary formation) in terms of the fission rate:

\[ (Nt)_{knee} = \frac{E_F^p K^{1/2} f^{1/2}}{K_{sm} f kT}, \tag{13} \]

where b is a conversion factor = 6 x 10^{23} fissions m^{-3} dpa^{-1}, and
E_F^p = E_F + E_p^s; E_p^s is the formation energy of a recrystallization site. Eq. 13
is the basic result of the above analysis. In principle, the material
constants E_F, K_{sm}, and f can be determined by experiment.

Fig. 6 shows a comparison between Eq. 13, using the values of the
material constants listed in Table 1, and the data obtained from Fig. 1. Due
to the unavailability of many of the material constants for U3Si2, estimates
of various parameters were taken from the references listed in Table 1. E_F^p
was adjusted to obtain agreement between the theory and data for a fission
rate of 2 x 10^{20} fissions m^{-3}s^{-1}. A steady-state impurity concentration of

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10^{-3} is consistent with estimates of concentrations of various rare earth fission products. As is evident from Fig. 6, the above analysis provides a plausible interpretation of the data. Additional confirmation of the ideas presented in this paper await the results of future experiments.

Figure 7 shows DART-calculated results for fuel particle swelling of low-enriched U_3Si_2-Al fuel plates as a function of fission density. The solid-line calculations were made in the spirit of the theory presented in this paper by assuming that fuel recrystallization occurred at about 3 \times 10^{27} fissions m^{-3} (Fig. 6) and that subgrain boundary formation is characterized by an average grain size of 1 mm. Subsequently, gas-atom diffusion to the grain boundaries, bubble nucleation, and accelerated growth (relative to that of bubbles in the bulk material) result in an increased calculated swelling rate as shown in Fig. 7. In general, bubbles on grain boundaries are thought to grow to larger sizes than those in the bulk due to a reduced effect of gas-atom re-solution. (At the relatively low temperatures characteristic of the U_3Si_2 irradiations, the effects of re-solution dominate bubble growth as compared to enhanced gas-atom diffusion along the grain boundaries.) Physically, this result is interpreted as due to a small value of gas-atom penetration depth. On average, ejected gas atoms remain within the influence of the boundary and are quickly recaptured. The calculated intergranular bubble sizes are consistent with those that have been observed. The dotted-line fuel swelling calculations in Fig. 7 were made assuming that no grain refinement occurred. Also shown in Fig. 7 are the measured swelling values. As can be seen, the assumption of subgrain boundary formation provides calculated swelling consistent with observation.

<table>
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<tr>
<th>Table 1. Values of various material constants used in the calculation</th>
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<tr>
<td><strong>Parameter</strong></td>
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<tr>
<td>( E_F )</td>
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<tr>
<td>( D_v^0 )</td>
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<tr>
<td>( e_v )</td>
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\[ * D_v = D_v^0 \exp(-e_v / kT) \]
An Improved DART Model for Porosity Effects on Thermal Conductivity

The DART thermal conductivity model has been improved to better account for the dependence of the thermal conductivity on both the as-fabricated, and the irradiation induced porosity. The original model\(^2\) smeared the irradiation-induced porosity throughout the meat which resulted in a stronger dependence than was physically reasonable. The modification described in this report removes this objection, while at the same time retaining the strong dependence on the as-fabricated porosity characteristic of the unirradiated U\(_3\)Si\(_2\) data. For completeness, and to establish a proper introduction to the modified calculation, the derivation of the fundamental mixing formulae used in the analysis is provided below.

For dispersion fuels, two distinct classes of pores exist, namely fission gas microbubbles generated within the U\(_3\)Si\(_2\) fuel particles and as-fabricated voids contained within the Al matrix. The analytical treatment of such pores is however similar, where pore geometry and physical properties are of prime importance. Pore geometry is defined by its size, shape, and orientation with respect to the direction of heat flow. Physical properties of importance are the emissivity of the pore surface and the thermal conductivity of the gas trapped within the pore.

Figure 8 can be used to establish a geometric relationship for the thermal conductivity of a solid material containing a gas pore. The unit cell can be represented as a cube of solid material surrounding a spherical pore. The most important variable considered and the one that appears in all theoretical models\(^19\text{-}21\) is the volume porosity (\(P\)), defined as:

\[
P = \frac{\text{Pore Volume}}{\text{Pore Volume} + \text{Volume of solid}}
\]  

Assuming heat flows in the \(y\)-direction only, the effective thermal conductivity (\(k_e\)) of the unit cell in the \(y\)-direction is given by the expression:\(^{21}\)

\[
k_e = P_c k_{ap} + (1 - P_c) k_s,
\]

where

- \(k_e\) = effective thermal conductivity
- \(k_s\) = thermal conductivity of solid material
- \(k_{ap}\) = apparent thermal conductivity of the pore tube
and $P_c$ is the fraction of the cross-sectional area of the $x$-$z$ face of the unit cell which is occupied by the pore. To assess the conductance over the path length in the $y$ direction a pore tube length ($L$) is defined, which considers both the pore and the solid material occupying the remaining tube length. The apparent thermal conductance of the pore tube (pore plus solid material in path length) can be evaluated considering the thermal resistances of these two regions, which can be expressed as:

\[
\frac{1}{k_{ap}} = \frac{P_L}{k_p} + \frac{1-P_L}{k_s}
\]

(16)

where

\[
P_L = \text{fractional length of pore}
\]

\[
k_p = \text{effective pore thermal conductivity}
\]

\[
\text{(gas conductivity plus radiation)}
\]

Eliminating $k_{ap}$ between Eqs. (15) and (16) yields the following expression:

\[
\frac{k_e}{k_s} = 1 - P_c \left[ 1 - \frac{k_p}{k_s} \left( \frac{P_L}{P_L} \right) \right]
\]

(17)

In the above expression the effect of volume porosity on the thermal conductance is contained in the quantities of $P_c$ and $P_L$. For the case of fission gas bubbles within U3Si2 fuel particles, it is assumed that the bubbles are spherical of radius ($R$) and are distributed uniformly in the material. Under these assumptions the following expressions apply:

\[
P_c = \pi R_g^2 / d_g^2 = \pi R_g^2 \rho_g^{2/3}
\]

(18)

\[
P_L = 2 R_g^2 / d_g = 2 R_g \rho_g^{1/3}
\]

(19)

where $r_g$ is the bubble density (bubbles/cc) and $d_g$ is the interbubble spacing $\left(\text{i.e.}\left[\rho_g^{-1}\right]^{1/3}\right)$. The thermal conductivity of a pore filled with gas is given by the expression:

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\[ k_p = k_g + 4\varepsilon \sigma RT^3. \quad (20) \]

where

- \( k_g \) = bulk thermal conductivity of the gas
- \( \varepsilon \) = emissivity
- \( s \) = Stefan-Boltzmann constant
- \( T \) = temperature

For the size pores and the range of temperatures under consideration, the second term in Eq. (20) representing radiative heat transfer can be neglected. Thus Eq. (17) applied to gas bubbles thus reduces to the form:

\[ \frac{k_e^g}{k_f} = 1 - \left( \frac{\pi R_g^2 \rho_g^{2/3}}{2k_f} \right) \left[ 1 - k_g / \left( 2k_f \rho_{g}^{1/3} \right) \right] \quad (21) \]

where

- \( k_e^g \) = effective thermal conductivity of fuel particles containing fission gas
- \( k_f \) = thermal conductivity of bulk U$_3$Si$_2$

Eq. (21) can be written in the form:

\[ \frac{k_e^g}{k_f} = 1 - \pi \left( \frac{R_g}{d_g} \right)^2 + \frac{\pi k_g}{2k_f} \left( \frac{R_g}{d_g} \right) \quad (22) \]

where the ratio of the bubble radius \( R_g \) to interbubble spacing \( d_g \) can be expressed as \( R_g / d_g = R_g / \rho_{g}^{1/3} \).

The geometrical condition for bubble interconnection is given by \( d_g = 2R_g \). Eqs. (21) and (22) represent expressions for \( k_e^g / k_f \) in terms of average properties of the fission-gas-bubble size distribution which evolves within the U$_3$Si$_2$ fuel particles during irradiation. The expression for \( k_g \) used to evaluate Eqs. (21) and (22) has been obtained from Ref. 21 and is given by:

\[ k_g(xenon) = \left( 4.0288 \times 10^5 \right) T^{0.872}(W/mK). \quad (23) \]

In order to include the effects of both as-fabricated and irradiation induced porosity on the thermal conductivity of U$_3$Si$_2$ particles dispersed in an aluminum matrix, successive applications of mixing formula of the form...
given by Eq. (21) have been utilized. For example, the thermal conductivity of aluminum containing a dispersion of U$_3$Si$_2$ particles (containing fission gas) with no as-fabricated porosity is given by

$$K_g(xenon) = \left(4.0288 \times 10^5\right)T^{0.872}(W/mK). \quad (24)$$

where

- $k_e^p =$ effective thermal conductivity of dispersion with no as-fabricated porosity
- $R_f =$ radius of particle
- $k_e^f =$ effective thermal conductivity of U$_3$O$_8$ particles containing fission gas (given by Eq. (22))
- $\rho_f =$ density of U$_3$O$_8$ particles
- $k_{al} =$ thermal conductivity of pure aluminum

If as-fabricated porosity is now introduced into the material

$$\frac{k_e^m}{k_e^p} = 1 - \pi R_f^2 P_f^{2/3}. \quad (25)$$

where

- $k_e^m =$ thermal conductivity of dispersion containing both as-fabricated and irradiation induced porosity
- $R_p =$ radius of as-fabricated pores
- $\rho_p =$ density of as-fabricated porosity

In deriving Eq. (25), the thermal conductivity of an as-fabricated pore has been assumed to be zero.

Utilizing Eqs. (24) and (25) $K_e^m$ can be expressed as

$$k_e^m = k_{al}\left[Z_1 + Z_2 F_v^{2/3} + \right.$$

$$Z_3\left(k_e^f / k_{al}\right)F_v^{1/3}\left(1 - \pi\left(3F_p / 4\pi\right)^2\right) + \left.$$ 

$$Z_4 F_p^{2/3} + Z_5 (F_v F_p)^{2/3} \right]. \quad (26)$$

where the fuel and pore volume fractions, $F_v$ and $F_p$, are given by
\[ F_v = \frac{4}{3} \pi R_f^3 \rho_f, \]
\[ F_p = \frac{4}{3} \pi R_p^3 \rho_p, \]

and where \( Z_1 - Z_5 \) are constants.

Implicit in the derivation of Eq. (26) is the assumption of a homogeneous distribution of spherical particles (i.e., fission gas, as-fabricated pores, \( \text{U}_3\text{Si}_2 \) particles) within the respective host material. In order to quantify the effects of deviations from this idealized geometry such as nonspherical pores and particles, as-fabricated and irradiation induced pores preferentially distributed along the particle-matrix interface (e.g., fission gas released from the grain interior can collect at the particle/matrix interface), the constants \( Z_1 - Z_5 \) have been determined by regression analysis based on the unirradiated dispersion fuel data taken from Ref. 25 and shown in Table 2. The results of the analysis yield:

\[
\begin{align*}
Z_1 &= 0.99717 \\
Z_2 &= -0.10169 \\
Z_3 &= 3.9696 \\
Z_4 &= 0.36262 \\
Z_5 &= 0.0531
\end{align*}
\]

Eq. (26) provides a physically based expression for the evolution of thermal conductivity in irradiated dispersion fuels. As such, even through the geometric constants \( Z_1 - Z_5 \) were determined based on unirradiated data, Eq. (26) embodies physically realistic functional dependencies (e.g., the dependence of the thermal conductivity on fuel and matrix volume fractions and on the as-fabricated and irradiation induced porosity which allow for a reasonable extrapolation to irradiated materials.

Figure 9 shows DART calculated total fuel swelling strain (\( V/VO \)), gas bubble swelling strain (\( VG/VO \)), aluminum matrix volume fraction (\( MVF \)), fuel volume fraction (\( FVF \)), change in meat thermal conductivity (\( K/KO \)), and change in can thickness (\( D/DO \)) as a function of fission density for fuel plates fabricated with 35% fuel volume fraction and 5% as-fabricated porosity. As shown in Fig. 9, the meat thermal conductivity initially increases due to the closure of the as-fabricated porosity with a subsequent decrease to a limiting
value given by the thermal conductivity of the gas in the bubbles. That the thermal conductivity decreases to the extent shown in Fig. 9 is not surprising given the following observations:

1. The change in thermal conductivity is steeper than that predicted by standard mixing formulas because the geometric constants Z1-Z5 (in particular, Z1-Z3) were determined based on the unirradiated data which displays a change with respect to the as-fabricated porosity and fuel volume fractions steeper than that predicted by standard mixing formula. Presumably, this behavior reflects the presence of non-spherical porosity and/or cracks along the particle matrix interface. It is important to note that although the constants Z1-Z5 were determined based on the unirradiated data in Table 2, their applicability to the irradiated material may have a physical basis in that the irradiation-induced porosity eventually accumulates at the fuel particle-matrix interface. Indeed, recent work on the possible recrystallization of U3Si2 points at an accelerated accumulation of gas at the particle surface.

2. The decrease in the thermal conductivity displayed in Fig. 9 is primarily due to the increase in the fuel volume fraction; the fuel particles are almost completely in contact having squeezed the aluminum matrix into the fuel meat-aluminum cladding interfacial region.

Fig. 10 shows the results of an analogous calculation as that shown in Fig. 9 but for fuel plates fabricated with 15% fuel volume fraction and 1% as-fabricated porosity. The decrease in the thermal conductivity shown in Fig. 10 is much more gradual over the irradiation period than that shown in Fig. 29. The reason for this behavior is the relatively low fuel volume fractions characteristic of the irradiation shown in Fig. 10. The fuel volume fraction approaches 30% as the fuel approaches 100% burnup, and even at this level the fuel particles are not fully interconnected; the high thermal conductivity matrix aluminum is still dominating heat transfer through the material.

Fig. 11 shows a comparison of the DART calculated thermal conductivity for U3Si fuel (solid line) as a function of irradiation-induced porosity compared to measured values in irradiated and subsequently annealed bulk U3Si. The effect of swelling porosity on the thermal conductivity of irradiated U3Si was deduced from electrical conductivity measurements on samples with pore volume fractions of 0.01 to 0.27. The authors of Ref. 28
state that the porosity in the irradiated and annealed U₃Si consists of fission gas as compared to predominantly empty voids in the irradiated only case. As shown in Fig. 11, the DART calculated change in thermal conductivity is in good agreement with the measured values.

Table 2. Unirradiated Dispersion Fuel Data Taken From Reference 25

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<th>(k_e^m) / kal</th>
<th>(F_v)</th>
<th>(F_p)</th>
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<td>39.0</td>
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Conclusions

One possible interpretation of the observed fission-rate dependence of low-temperature swelling of irradiated uranium silicide dispersion fuels is ascribed to the formation of subgrain boundaries in the material. Subsequently, fission gas atoms diffuse to the boundaries and nucleate fission gas bubbles that grow at an accelerated rate relative to that in the bulk material. The existence of "recrystallization" sites is proposed upon which the stored energy density is concentrated. The available stored energy in the material is limited due to the relatively plastic nature of the irradiated material. Recrystallization, occurs when the density of "recrystallization" sites decreases to a level where the net change in free energy due to the creation of strain-free volumes (decrease in free energy) and the creation of boundary surfaces (increase in free energy) is on the order of the thermal energy, kT. The decrease in "recrystallization" site density is proposed to be due to interaction with mobile irradiation-produced defects (vacancy-impurity pairs). The theory provides a plausible interpretation of the observed phenomena.

The DART thermal conductivity model has been improved to better account for the dependence of the thermal conductivity on both the as-fabricated and the irradiation-induced porosity. The original model smeared the irradiation-induced porosity throughout the meat, which resulted in a stronger dependence than was physically reasonable. The modification described in this report removes this objection, while at the same time retaining the strong dependence on the as-fabricated porosity characteristic of the unirradiated U$_3$Si$_2$ data.
References


Fig. 1. Swelling of U₃Si₂ for three fission rates as a function of fission density.
Fig. 2. First evidence of fission gas bubbles in U$_3$Si$_2$, irradiated at $3 \times 10^{27}$ fissions m$^{-3}$s$^{-1}$ just at the "knee" in the swelling curve.
Fig. 3. Average fission gas bubble diameter in U₃Si₂ as a function of fission density for three values of the fission rate.
Fig. 4. Fission gas bubble morphology in U₃Si₂ irradiated to various fission densities at three different fission rates.

(5 x 10²⁷ fts m⁻³, 2 x 10²⁰ fts m⁻³ s⁻¹)  
(7 x 10²⁷ fts m⁻³, 3 x 10²⁰ fts m⁻³ s⁻¹)
Fig. 4. (Continued)

(9.5 \times 10^{27} \text{ fts m}^{-3}, 7 \times 10^{20} \text{ fts m}^{-3} \text{s}^{-1})

(14 \times 10^{27} \text{ fts m}^{-3}, 3 \times 10^{20} \text{ fts m}^{-3} \text{s}^{-1})
Fig. 5. Fracture surface of UO$_2$ particles before and after irradiation, showing grain refinement.
Figure 6. Comparison between Eq. 13 and data obtained from Fig. 1.
Fig. 7. Swelling of U₃Si₂ calculated with DART (dashed line without grain refinement).
Fig. 8. Unit cell representation of a porous body for which the effective thermal conductivity is assessed.
Fig. 9. DART-calculated total fuel swelling strain (V/VO), gas bubble swelling strain (VG/VO), aluminum matrix volume fraction (MVF), fuel volume fraction (FVF), change in meat thermal conductivity (K/KO), and change in can thickness (D/DO) as a function of fission density for fuel plates fabricated with 35% fuel volume fraction and 5% as-fabricated porosity.
Fig. 10. DART-calculated total fuel swelling strain (V/VO), gas bubble swelling strain (VG/VO), aluminum matrix volume fraction (MVF), fuel volume fraction (FVF), change in meat thermal conductivity (K/KO), and change in can thickness (D/DO) as a function of fission density for fuel plates fabricated with 14% fuel volume fraction and 1% as-fabricated porosity.
Fig. 11. A comparison of the DART-calculated thermal conductivity for U₃Si fuel (solid line) as a function of irradiation-induced porosity compared to measured values in irradiated and subsequently annealed bulk U₃Si.
THE FOAMING OF U-Al FUEL UNDER SIMULATED REACTOR ACCIDENT CONDITIONS

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ABSTRACT

Postirradiation heating tests were conducted on segments of UA14/Al dispersion fuel plates clad with Al to scope the foaming behavior of such fuels during beyond-design-basis accident scenarios. Four tests investigated maximum temperature, ramp rate, and duration with a liquid phase as parameters in foam formation and stability. Real-time fission-gas release was also determined during the foaming process. Ramp-rate had the most noticeable effect on foam formation and collapse.

INTRODUCTION

The reference fuel for numerous research and test reactors is a dispersion of UA14 in an aluminum matrix and is clad with an aluminum alloy. Such fuel systems have been in use for many years and a significant body of information exists for the behavior of these fuels in design-basis reactor accidents. However, there has been recent interest in the behavior of such fuels under accident conditions that are beyond design-basis scenarios and that could involve core disassembly.

Accident scenarios that are being assessed postulate rapid swelling, or foaming, of the fuel as a significant mechanism for core disruption during such accidents. In-cell phenomenological behavior experiments have been conducted on irradiated fuel to investigate U-Al fuel foaming and to sufficiently characterize foaming so as to provide core behavior modelers with a relevant and broad data base. Issues of specific interest are foam growth kinetics, foam stability, and fuel relocation after foam collapse. The in-cell experiments are conducted on small fuel coupons or longer fuel strips.

This report covers four in-cell tests that were conducted on fuel coupons of aluminum-clad Al-UA14 dispersion fuel. The purpose of these tests was to establish whether sufficient foaming occurs in the U-Al system to warrant in-depth characterization. These tests, designated NP-1, -2, and -3, and MA-1, covered what were believed to be the

principal parameters of interest: maximum temperature, ramp rate, and duration with a liquid phase. Once the feasibility of foaming was established, these tests could be used as a basis to scope a follow-on experimental program to more fully characterize fuel behavior in order to generate behavior models. These tests have indeed established fuel foaming as a phenomenon to be considered in accident scenarios, and they have also provided a basis for planning future tests.

EXPERIMENT DESCRIPTION

A. Fuel and Specimen Description

The fuel used in these four experiments was obtained from the Reduced-Enrichment Research and Test Reactor (RERTR) fuel development program. Specimens for the NP tests came from a "miniplate," A132, that was fabricated with UAl₂ particles in a 6061 Al matrix, yielding 64 wt.% U in the fuel region, or "meat," which was nominally 0.022 in. thick. The cladding was 0.014-in.-thick 6061 Al, giving a fabricated plate thickness of 0.050 in. The nominally 45% enriched U achieved a depletion of 65% during irradiation for a fission density of 2.2 x 10²¹ fiss/cm³ in the fuel meat. Figure 1 shows a cross section of the irradiated, but untested, fuel. In unirradiated fuel, the uranium would constitute 19.7 wt.% of this section.

The specimen used in the MA-1 test came from an RERTR demonstration fuel plate, NLE-451, which was fabricated with UAlₓ powder in an aluminum matrix to yield 43 wt.% U in the meat. The meat thickness was nominally 0.023 in. and the cladding was 0.022 in. thick. The cladding material was Al-0.85 wt.% Mg alloy. The 45% enriched U was 89% depleted for a fission density of 1.57 x 10²¹ fiss/cm³. Figure 2 shows a cross section of the irradiated, but untested fuel. In unirradiated fuel, the uranium would constitute 9.5 wt.% of this section.

The specimens used in the tests were nominally 0.5 in. x 0.25 in. and the same thickness as the source fuel plate. The fuel was exposed on the four edges of the specimens; cladding was present only on the faces. Only the cladding of the A-132 specimens (NP) had been cleaned to remove oxide scale formed during irradiation. The NLE specimen was used "as-irradiated."

It should be noted that the solidus temperature for the 6061 alloy is 582°C, whereas that for the Al-Mg alloy is 630°C.

B. Fuel Behavior Test Apparatus

The fuel foaming tests were conducted in the Fuel Behavior Test Apparatus (FBTA) located in the Alpha-Gamma Hot Cell Facility. The FBTA consists of an infrared radiant furnace to provide specimen heating; a thermocouple to measure test temperature; a helium-gas purge system to provide an inert atmosphere and remove released fission gas from the apparatus; a flow-through detector to detect the fission gas (⁸⁵Kr) in the helium stream; and a microcomputer system to perform
furnace control, trip protection, and data acquisition functions. The FBTA is also equipped with a small window in the side of the furnace to permit specimen viewing, and a gas collection system for quantitative measurement of the fission gas released from the fuel. Neither the viewing nor gas collection capabilities were used in these initial scoping tests.

Figure 3 shows a schematic diagram of the FBTA. The fuel specimen was placed upright (on edge, long dimension as height) in a tantalum cup at the center of the furnace heating zone. Solid-sided tantalum cups were used because direct visual observation and photography of foaming was not required in these tests. The cup dimensions (0.260-in. diam and the same height as the specimen cavity) were chosen so that the ratio of the void volume for foam expansion to the total internal volume of the cup (0.78 in the NP tests and 0.71 in the MA-1 test) would closely approximate the relative coolant channel fraction (0.78) in a nominal fuel assembly design. The test temperature was measured with an unsheathed Pt/Pt-10Rh (Type S) thermocouple welded onto the outside of the tantalum cup. A new cup/thermocouple assembly was used in each test. Owing to the nature of external heating, the specimen temperature during furnace heatup generally lagged the measured cup temperature, and the lag was dependent on the heating rate. A correlation between specimen and cup temperatures was established in a series of temperature calibrations using aluminum standards in the tantalum cups. During rapid heating, e.g., 35°C/s, the specimen temperature lagged the cup temperature by =20°C. At a heating rate of 10°C/s, the temperature lag was only =3°C.

The tantalum cup was contained within two concentric quartz tubes in the FBTA. A high-purity helium purge flow in the inner quartz tube provided an inert atmosphere for the test specimen. The fission gas released from the fuel was carried away by the helium purge gas to an out-of-cell flow-through detector for monitoring of the 85Kr activity. The transit time from the source to the detector, at a nominal helium flow rate of 10 L/min, was estimated to be =20 s. The gas was then passed back into the hot-cell. (Alternatively, the gas could be passed through liquid-nitrogen-cooled charcoal where the Kr and Xe could be trapped for subsequent quantitative measurement.) Because fission-gas release was not planned to be a significant figure of merit for these tests, only qualitative gas-release data were obtained.

C. Test Conditions

The planned thermal conditions of the foaming tests (Table 1) were designed to investigate the macroscopic physical behavior of the fuel under potential decay heating rates during an accident. Based on early perceptions of accident scenarios in a nominal reactor design, a relevant heating rate range of 10-35°C/s was selected, with the higher rate likely to be of greater interest. Because foaming in the classic sense requires the presence of a liquid phase, the duration of the tests was either 30 or 60 s longer than the time needed to reach the 640° eutectic temperature in the U-Al system. The test temperatures were selected to be greater than the L + UA14 → L + UA13 transition temperature of 730°C for the NP tests, and just below that temperature for the MA-1 test to ensure the presence of a liquid phase.
Prior to each test, calibration runs were made to establish the computer-controlled parameters that were needed to achieve the desired heating rate, peak temperature, and duration above the eutectic temperature. As a result of these calibration runs, the planned thermal conditions were close to those realized in the actual tests (see Figs. 4 to 7). The cooling rates for all tests were not controlled and were simply the result of cooling by the helium purge gas with the furnace shut down.

D. Posttest Examination

After each test, the cup containing the specimen was mounted for longitudinal metallography with the intent of examining successive surfaces as the specimen was ground to the midplane. The flat plane of the fuel specimen was perpendicular to the plane of polishing so that the examined surface would progressively move from the specimen's edge to the midplane. Microphotography was used to document the foamed fuel structures found at the various elevations.

EXPERIMENTAL RESULTS

A. Test NP-1

The heating and cooling curve for the NP-1 test is shown in Fig. 4. Because of the desired short duration (30 s) above 640° and the rapid heating rate, the test temperature only peaked at 820° and immediately came down. The thermal arrest, beginning at 620° on the cooldown, indicates the fuel temperature lagged that of the tantalum cup by ~20°C. The fission-gas release profile was initially erratic and the activity level was significantly greater than expected. Hence, the recorded data did not capture the peak release. The time delay for release is consistent with initiation of release at the eutectic temperature. However, the sensitivity of the system is not sufficient to reach this conclusion with great certainty.

The series of metallographic cross sections beginning near the I.D. of the cup and progressing to the specimen is shown in Figs. 8-10. Fission-gas bubble expansion, coalescence, collapse, and also surface breakthrough can be seen in the sections. Cladding runoff did not occur. Rather, the cladding apparently reacted with the fuel meat during the heatup. Because the upper part of the specimen extended
above the top of the cup, that portion of the specimen was cooler than
the part within the cup. Complete reaction between the cladding and
the meat nonetheless occurred at the top, and gas-driven eruptions in
the surface are evident.

B. Test NP-2

The heating and cooling curve for the NP-2 test is shown in
Fig. 5. This test had the same 35°C/s nominal heating rate as NP-1 but
the duration was =60 s longer than the time needed to reach the 640°C
eutectic temperature instead of 30 s. Apparently, unlike the NP-1
test, a secondary release of fission gas occurred after the specimen
had cooled to essentially ambient temperature. This was probably the
result of surface contraction and the breakthrough of surface gas
bubbles. Fuel cross sections are shown in Figs. 11-14. Similar to the
NP-1 test, there was no cladding runoff. Bubble coalescence resulted
in the near-formation of a single principal cavity at the centerline,
with vented bubbles being compressed to the periphery. Volume
expansion brought the specimen into greater contact with the cup, which
accounts for the apparent thermal arrest in the cooling curve. This
specimen, too, extended above the top of the cup and was therefore
cooler at the top. There was considerably more volume change than in
the NP-1 specimen, indicating that the longer duration in the liquid
state resulted in more foaming.

C. Test NP-3

This test was conducted with the same duration as NP-2, 60 s
longer than the time needed to reach the 640°C eutectic temperature,
but the heating rate was only 10°C/s. The time/temperature and
time/fission-gas-release profiles are given in Fig. 6. Similar to the
NP-2 test, there was a secondary release of fission gas after cooldown.
It would appear, from the areas under the truncated gas release curves
for NP-2 and -3, that the slower temperature ramp rate effected less
gas release.

The metallographic cross sections are shown in Figs. 15-18. As
in tests NP-1 and 2, there was no cladding runoff. The principal
difference between NP-3 and NP-2 is that bubble coalescence and
expansion were less extensive and that evidence of structure collapse
is present, i.e., movement of ligament structure into the space of
former bubbles is present. Both of these observations are inconsistent
with the apparently greater retention of gas in this specimen. Some of
the observed structural movement could be the result of the collapse
and reformation of bubbles in the new "matrix." It appears that the
fission gas did not move from the solid particles into the liquid phase
and then into the gas phase at a fast enough rate to support the foam
before collapse.

D. Test MA-1

The MA-1 test differed in two ways from its essentially companion
test, NP-1. First, the fuel's uranium content was lower (9.5 wt.% vs
19.7 wt.% in the meat/cladding combination). Second, the maximum test
temperature was 730°C. The time/temperature and time/fission-gas-release profiles are given in Fig. 7.

The metallographic cross sections are shown in Figs. 19-22. The specimen was originally mounted incorrectly and the plane of polishing was not the desired orientation. The specimen was remounted after partial polishing and the resulting planar sections do not represent complete cross sections of the specimen. Nevertheless, the sections are valuable for evaluating relative bubble size and coalescence and cladding/fuel interactions for comparison with the higher-uranium-content NP tests. In none of the sections was cladding runoff found. However, a greater abundance of aluminum appears to be evident at the bottom of the specimen and this could be responsible for the lesser tendency of this specimen to foam than its NP-1 counterpart. The MA-1 specimen shows indications of foam collapse, perhaps including what appears to be a lower bubble density in the bottom of the specimen. But the “retarded” foam formation and subsequent greater tendency to collapse can also be explained, perhaps most simply, by the lower temperature achieved in MA-1. Temperature would affect both internal bubble pressure and matrix fluidity.

CONCLUSIONS

The early scoping tests that are reported here have demonstrated that U-Al fuels can indeed foam to significant proportions and maintain the foam state for at least 1 min.

Peak temperature, duration in a liquid phase, and temperature ramp rate all affected foaming, gas release, and foam collapse in some way. The effects of ramp rate in the range from 10 to 35°C/s were most notable. Both fission-gas release and gas-bubble coalescence were greatest at the higher rate. The combination of less bubble growth even though there was more retained gas (NP-3 compared to NP-2) suggests a rate effect in how the gas was released from the liquid matrix. Higher strain rates on the ligament structure apparently resulted in greater breakage, i.e., coalescence, which then proceeded to greater “swelling” (volume change of the specimen envelope) and gas release from surface bubbles. In none of the tests did the cladding drain away from the meat. Rather, the cladding reacted with the meat in-situ.

The FBTA demonstrated that it is a viable tool for conducting phenomenological tests on fuel foaming under the controlled conditions necessary to obtain a good understanding of foam formation and stability. The capability to clearly define real-time fission-gas release, however, will require additional effort at the higher temperature ramp rates.
Fig. 1. Cross section of irradiated miniplate A132 used in NP tests.
110X

MCT 276100
Fig. 2. Cross section of irradiated fuel plate NLE-451 used in test MA-1.

MCT 276836

50X reduced to 34X
Fig. 3. Partial schematic diagram of Fuel Behavior Test Apparatus.
Fig. 4. Time/temperature and time/gas release histories of test NP-1. Fission-gas release (FGR) has no vertical scale.

Fig. 5. Time/temperature and time/gas release histories of test NP-2. Fission-gas release (FGR) has no vertical scale.
Fig. 6. Time/temperature and time/gas release histories of test NP-3. Fission-gas release (FGR) has no vertical scale.

Fig. 7. Time/temperature and time/gas release histories of test MA-1. Fission-gas release (FGR) has no vertical scale.
Fig. 8. NP-1 specimen 0.105 in. from centerline. Original at 25X, reduced to 14X.
Fig. 9. NP-1 specimen 0.045 in. from centerline. Original at 25X, reduced to 14X.
Fig. 10. NP-1 specimen at centerline. Original at 25X, reduced to 14X. MCT 275077.
Fig. 11. NP-2 specimen 0.117 in. from centerline. Original at 25X, reduced to 14X.

MCT 275385
Fig. 12. NP-2 specimen 0.103 in. from centerline. Original at 25X, reduced to 14X.
Fig. 13. NP-2 specimen 0.058 in. from centerline. Original at 25X, reduced to 14X.

MCT 276098
Fig. 14. NP-2 specimen at centerline. Original at 25X, reduced to 14X.
MCT 276099
Fig. 15

NP-3 specimen
0.104 in. from centerline,
Original at 25X,
reduced to 14X.
MCT 276652
Fig. 16

NP-3 specimen 0.068 in. from centerline.
Original at 25X, reduced to 14X.
MCT 276653
Fig. 17

NP-3 specimen
0.023 in. from centerline.
Original at 25X,
reduced to 14X.
MCT 276654
Fig. 18. NP-3 specimen at centerline. Original at 25X, reduced to 14X. MCT 276655

160
Fig. 19. MA-1 specimen ground at angle to centerline at top and =0.12 in. from centerline at bottom. Original at 25X, reduced to 14X.
Fig. 20. MA-1 specimen after remounting. (Original ground surface on right.) Original at 25X, reduced to 14X. MCT 277550
Fig. 21. MA-1 specimen =0.02 in. from centerline. Original at 25X, reduced to 14X.
Fig. 22. MA-1 specimen slightly past centerline. Original at 25X, reduced to 14X.
SESSION IV

September 25, 1990

MEASUREMENTS AND CALCULATIONS FOR CONVERTED REACTORS

Chairmen:

W. Krull  
(GKSS, Federal Republic of Germany)

K. Kanda  
(KURRI, Japan)
THE ORR WHOLE-CORE LEU FUEL DEMONSTRATION
FINAL REPORT

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ABSTRACT

The ORR Whole-Core LEU Fuel Demonstration, conducted as part of the U.S. Reduced Enrichment Research and Test Reactor Program, has been successfully completed. Using commercially-fabricated U$_3$Si$_2$-Al 20%-enriched fuel elements (4.8 g U/cc) and fuel followers (3.5 g U/cc), the 30-MW Oak Ridge Research Reactor was safely converted from an all-HEU core, through a series of HEU/LEU mixed transition cores, to an all-LEU core. There were no fuel element failures and average discharge burnups were measured to be as high as 50% for the standard elements and 75% for the fuel followers. Experimental results for burnup-dependent critical configurations, cycle-averaged fuel element powers, and fuel-element-averaged $^{235}$U burnups validated predictions based on three-dimensional depletion calculations. Calculated values for plutonium production and isotopic mass ratios as functions of $^{235}$U burnup support the corresponding measured quantities. In general, calculations for reaction rate distributions, control rod worths, prompt neutron decay constants, and isothermal temperature coefficients were found to agree with corresponding measured values. Experimentally determined critical configurations for fresh HEU and LEU cores radially reflected with water and with beryllium are well-predicted by both Monte Carlo and diffusion calculations.

INTRODUCTION

Early in the history of the U.S. Reduced Enrichment Research and Test Reactor (RERTR) Program the need for a whole-core demonstration of high uranium density LEU fuel was recognized. Plate-type U$_3$Si$_2$-Al dispersion fuel was chosen for the demonstration because of its excellent behavior under irradiation (Ref's. 1 & 2), its high maximum practical uranium density (4.8 g U/cc), and its ease of fabrication by commercial suppliers. The primary objectives of the demonstration were:
1. to demonstrate the safe and acceptable behavior of commercially-fabricated 20%-enriched U\(_3\)Si\(_2\)-Al dispersion fuel (4.8 g U/cc in fuel meat) to \(^{235}\)U burnups greater than 50% in a relatively high power research reactor.

2. to demonstrate the safe transition process from an all-HEU equilibrium core, through a series of mixed HEU/LEU cores, to an all-LEU equilibrium core, and

3. to provide an abundance of core physics data for validating analytical methods, codes and burnup predictions.

The 30-MW Oak Ridge Research Reactor (ORR) was chosen for the demonstration because its high power would provide data for the validation of fuel cycle calculations in relatively short times and because analyses showed that the demonstration would cause only minor changes in the performance of standard ORR experiments. Three international fuel vendors fabricated the U\(_3\)Si\(_2\)-Al LEU elements for the ORR demonstration. Sixty fuel elements and twelve shim rod assemblies were fabricated by Babcock and Wilcox (USA), and twenty elements each were supplied by CERCA (France) and NUKEM (FRG). Each 19-plate 20%-enriched fuel element had a meat density of 4.8 g U/cc and contained 340 g \(^{235}\)U. The 15-plate fuel followers had a density of 3.5 g U/cc and contained 200 g \(^{235}\)U. HEU (U\(_3\)O\(_8\)-Al) and LEU fuel elements are of identical geometry.

The demonstration began with an all-HEU reference core (174C) in December 1985. The first three LEU elements were phased into the ORR core at the beginning of January 1986. Following every cycle thereafter, three or four additional LEU elements were inserted into the core while an equal number of HEU elements were discharged. Normally the ORR operated with two nearly identical cores which were alternated between the reactor and the pool to allow for xenon decay. The first all-LEU core (178C) operated in December 1986. With the completion of cycle 179A, the ORR was permanently shut down in March 1987 for reasons entirely unrelated to the demonstration. Table 1 provides a summary of the 30-MW cores used in the demonstration.

Because of this unexpected shutdown, not as many LEU fuel elements were fully irradiated as had been planned initially. Nevertheless, the primary objectives of the demonstration, as stated above, were successfully carried out. Measurements made during the demonstration included core maps of reaction rates, control rod worths, isothermal temperature coefficients, prompt neutron decay constants, and cycle-averaged fuel element powers and \(^{235}\)U discharge burnups. Typical approach-to-critical measurements were also made for fresh HEU and LEU core configurations radially-reflected with both water and beryllium. This report summarizes comparisons of measured and calculated...
results obtained from the demonstration. However, more details than can be given here are provided in Ref. (3). In addition, results of postirradiation examinations of demonstration fuel elements are given in Ref. (4).

CRITICAL ASSEMBLIES WITH FRESH FUEL

Standard approach-to-critical methods were used to determine critical configurations for cores with unirradiated HEU (93% enriched) and LEU (20% enriched) fuel and radially-reflected with water and with beryllium. This data was used to test analytical models and multigroup cross section generation methods with some results given in Ref. (5).

Figure 1 shows core maps for the ORR fresh fuel critical assemblies. Three of these assemblies (HEU-1, LEU-1, and 179-AX5) were reflected in the radial direction with water. For the HEU-1 core, water occupied grid positions C4 and C6. Core 179-AX5 was an LEU core with the magnetic fusion experiments (MFE) replaced with water. Three assemblies (HEU-2, LEU-2, and 179-AX6) were radially-reflected with beryllium. For the LEU core 179-AX6 the MFE's were replaced with beryllium elements.

The ORR shim rods (SR) consist of an upper cadmium poison section and a lower 15-plate fuel follower section. For these measurements the four shim rods located in positions D4, D6, F4, and F6 were banked together and moved as a unit to achieve criticality. For shim rods withdrawn 15.25 inches the bottom of the cadmium poison section lies on the axial midplane of the core.

Table 2 shows the eigenvalues calculated for each of these experimentally determined critical configurations. Small corrections have been applied to account for temperature differences between the experimental conditions and the temperature at which the cross sections were generated. Neutron absorption in impurities and in minor elements used in the aluminum alloys has been expressed in terms of an equivalent boron concentration. The DIF3D code was used to perform the diffusion calculations where the cadmium poison was treated by a set of effective diffusion parameters. The Monte Carlo results are based on 300,000 neutron histories and were obtained from the continuous energy VIM code. Cross sections for both types of calculations were obtained from ENDF/B Version IV. Table 2 shows that the calculated eigenvalues are in good agreement with the experimentally determined critical configurations, for which $k_{\text{eff}}$ is unity.
REACTION RATE DISTRIBUTIONS

Activity distributions were measured by activating thin wires located in water channels between fuel plates. For the fresh fuel criticals gold wires were used for this purpose since the large activation cross section of gold allowed the measurements to be made without significantly activating the fuel. For all the other cores, however, cobalt-vanadium (2 wt% Co) wires were activated for six hours at 300 kw power levels. In general, several wires were located in each fuel element. These cobalt activity measurements had the dual purpose of allowing prediction of maximum fuel power density prior to full power operation of the reactor and of providing reaction rate distributions to compare with calculations. The maximum power densities were used to show that adequate safety margins would be preserved at full power.9

Measured and calculated \(^{198}\text{Au}\) and \(^{60}\text{Co}\) activity distributions averaged over the fuel element have been compared by R. J. Cornella. Some early results are given in Ref. (10). Figure (2) shows the calculated-to-experiment (C/E) ratios for gold wires in the water-reflected fresh fuel cores. The RMS DEV shown in this figure is the root-mean-square deviation of the C/E ratios from unity. Some typical results obtained from Co-V wire data are given in Fig. 3. Note that only those wires irradiated in standard 19-plate fuel elements were used to evaluate the RMS DEV values. In general, measured and calculated \(^{60}\text{Co}\) activities, averaged over each fuel element, agree reasonably well.

SHIM ROD WORTHS

Differential shim rod worths were measured in the ORR by the positive period method. Because of intense gamma-ray fields from highly active fuel elements and associated photoneutron sources, however, the reactor had to operate at high enough powers so that temperature-related feedback effects during the reactivity transient were not negligible. Under these circumstances the transient flux never acquires a purely exponential shape characterized by an asymptotic period. Therefore, differential shim rod worths were evaluated from a careful analysis of the measured shape of the initial portion of the time-dependent flux following a positive reactivity insertion. In this region of the curve temperature change effects are still negligible. To further complicate the analysis, delayed photoneutrons contribute to the kinetic response of the reactor and to the value of the effective delayed neutron fraction. The methods used to determine ORR differential shim rod worths from measured time-dependent fluxes were presented at the 1987 RERTR meeting11 and so will not be repeated here.
Table 3 shows some differential worths measured in several ORR cores using techniques just described. Although the errors are relatively large, most of the C/E ratios are within about 5% of unity. More results may be found in Ref's (3) and (11). Most of the differential shim rod worth measurements were conducted with a coolant flow rate of 1200 gpm. For core 179-AX7, however, measurements were made at both 1200 and 18,000 gpm. These results are reported in Ref. (3) and show that somewhat lower C/E ratios were obtained for the high flow rate conditions. This suggests that the data analysis methods described above did not completely remove temperature-related negative feedback effects and probably accounts for the fact that most of the C/E ratios in Table 3 are somewhat greater than unity.

The total or integral rod worth is obtained by integrating the differential worths from the lower limit (LL) to the upper limit (UL) of rod movement. To carry out these integrations the measured and calculated differential worths were fit to sixth degree polynomials by the least squares process. Results for the D6 shim rod in core 179-AX5 are summarized in Table 4. Also shown in this table are the DIF3D and VIM evaluations of the total D6 rod worth based on eigenvalue calculations for the rod-in and rod-out configurations.

The VIM-Monte Carlo and the DIF3D-diffusion results are in very good agreement. They are also less than 1% larger than the integral worth obtained by integrating the calculated differential worths. However, these integral and total worths are not expected to be exactly the same because of differences in the rod bank positions.

**PROMPT NEUTRON DECAY CONSTANT**

The prompt neutron decay constant is just the ratio of the effective delayed neutron fraction to the prompt neutron lifetime. Reactor noise methods were used to measure this ratio in several ORR cores. Signals from two fission chambers located on different sides of the core were processed by a Fourier analyzer to obtain the cross-power spectral density as a function of frequency. A least squares analysis of this frequency spectrum determines the break frequency which when multiplied by $2\pi$ gives the prompt neutron decay constant. Some preliminary results for the fresh-fueled water-reflected criticals are given in Ref. (13).

Table 5 compares measured and calculated values for the prompt neutron decay constant. Unlike the other cores, 179A used mostly previously irradiated fuel and so had a background contribution from delayed photoneutrons. These delayed photoneutrons must be included in the evaluation of the effective delayed neutron fraction. Table 5 shows the calculated values for the delayed neutron fraction, with and without photoneutrons.
This same value for the total delayed neutron fraction (with photoneutrons) was also used in the evaluation of differential shim rod worths in cores 179A and 179-AX7 (Table 3). As can be seen from Table 5, measured and calculated values for the prompt neutron decay constant are in good agreement.

**ISOTHERMAL TEMPERATURE COEFFICIENT**

The isothermal temperature coefficient was measured in core 179-AX7, which was identical to 179A (Fig. 4) except that the MFE's and irradiation (Eu,Ir) experiments were removed. With the coolant at its lowest temperature, shim rods F4 and F6 were withdrawn to their upper limits, B6, D4 and D6 were banked together at a fixed position (12.76"), and B4 was withdrawn to achieve criticality. Because of friction heating by the pumping system, the coolant temperature slowly increased. With about every 3 degree C temperature increase the B4 critical rod position (CRP) was redetermined. Thus, the B4 CRP was measured as a function of coolant temperature in the range from 25 to 45 degrees C. The differential worth of the B4 shim rod was measured over this same rod displacement interval. Since these temperature changes occurred very slowly and because the reactor was subcritical most of the time, the measurements were made under isothermal conditions.

Table 6 summarizes the isothermal temperature coefficient evaluations. The first column shows the differential worth of the B4 shim rod measured over the interval of interest while dL/dT is the slope of the critical rod position versus temperature data. Cross section sets, generated at 23 and 77 degrees C, were used to calculate the temperature coefficient. Within experimental errors, the calculated isothermal temperature coefficient is consistent with the measured one.

**BURNUP CALCULATIONS**

One of the primary purposes of the whole-core demonstration was to provide data for the validation of fuel cycle calculations. The REBUS-3 code\textsuperscript{14} was used to perform three-dimensional non-equilibrium burnup calculations for each of the 22 full power cores used in the demonstration. For each of these calculations the cycle length, expressed in full power days (FPD's), was divided into several subintervals or time nodes (TN's). In order to make use of the control rod movement capability of the REBUS-3 code, shim rod positions at the boundaries of each time node were determined from control rod position histories recorded throughout each burn cycle.

Table 7 gives the calculated eigenvalues corresponding to the experimentally-determined critical configurations for each
time node and for each operating core. These eigenvalues have
been temperature-corrected to account for small changes between
the operating core temperature and the 296K used for cross
section generation. The configuration for each of these cores is
given in Ref. (3). Table 7 shows that the REBUS-3 code
adequately predicts the influence of burnup on the eigenvalues
throughout the burn cycle. For these calculations the cadmium
poison section of the shim rods was treated by the methods
described in Ref. (7).

The REBUS-3 calculations were also used to determine
cycle-averaged fuel element powers and fuel-element-averaged $^{235}U$
burnups. These results will be compared with corresponding
measured values in the following sections. Together with the
eigenvalues given in Table 7, these comparisons serve to validate
the fuel cycle calculations.

**CYCLE-AVERAGED FUEL ELEMENT POWERS**

After each burn cycle during the whole-core demonstration
the fuel elements were removed from the 30-MW ORR to allow for
xenon decay while a second core was loaded into the assembly.
During these intercycle periods the removed fuel elements were
gamma-scanned axially along their centerlines to measure the
distribution of the $^{140}\text{La}$ fission product activity. Because of
the relatively short half lives of $^{140}\text{Ba}$ and $^{140}\text{La}$, this
information gives a measure of the fission rate densities and so
the power densities, that occurred during the previous burn
cycle. Thus, the $^{140}\text{La}$ data can be used to determine
cycle-averaged fuel element powers. The gamma-scanning apparatus
is described in Ref's (9) and (15) while methods used to analyze
the data are given in Ref's (3), (16), and (17). As described in
Ref. (17), transverse gradient corrections have been applied to
the gamma-scanning data.

Table 8 gives the cycle-averaged measured power $P(E)$ and the
corresponding C/E ratio for each fuel element in each of the
operating cores used in the demonstration. No gamma-scanning
data was taken for the fuel elements in the all-HEU reference
core 174C. The root-mean-square deviation (RMS D/LV) of the
departure of the C/E ratios from unity is shown at the bottom of
Table 8 for each of the cores. Of the 524 C/E ratios given in
this table about 75% differ from unity by 5% or less. Fig. 4
shows some typical core maps of C/E fuel element power ratios.

A careful examination of Table 8 reveals several anomalies.
For cores 174D through 175C the C/E power ratios are unusually
large in the A-row, especially at location A5. However, this
trend tends to disappear for the remaining cores in the
demonstration. The Heavy Section Steel Technology (HSST)
Experiment was located just outside the core box on the west side
of the core (see Fig. 4) for 174D through 175C, but was removed
for all the remaining cores. The C/E data suggests that the HSST was not modeled very well in the diffusion calculations even though good eigenvalues (Table 7) were obtained. A number of core pairs with nearly identical configurations (176B-176C, 176D-177A, 177B-177C) show several low C/E ratios in column 5 for one member of the pair but not the other. In almost all of these cases the low C/E ratio corresponds to an HEU fuel element which was not gamma-scanned for $^{137}$Cs. As will be discussed in the next section, the $^{235}$U mass for these fuel elements is quite uncertain, which may account for this strange C/E behavior. Finally, all the cores beginning with 178C contained only LEU fuel and had the same configuration (see Fig. 4, core 179A). For each of these cases a large C/E value was obtained at position B3 but very normal ratios at the symmetric position B7. The reason for this behavior is not understood.

**FUEL-ELEMENT-AVERAGED $^{235}$U BURNUPS**

During the demonstration fuel elements discharged from the ORR were gamma-scanned to determine the $^{137}$Cs activity distribution. Because of the 30-year half life of $^{137}$Cs, this measurement integrates the activity over all previous cycles of operation and so gives count rates proportional to the total fission density within the fuel element. The $^{235}$U burnup is directly related to the total fission density. Mathematical details for analyzing the $^{137}$Cs gamma-scanning data to determine final $^{235}$U fuel element masses and burnups are given in Ref.'s (3) and (16).

Table 9 gives the experimental values for the $^{235}$U masses and burnups for all 68 LEU fuel elements used in the demonstration and the corresponding C/E ratios. Similar information is given at the end of this table for the LEU fuel followers. Of the 132 HEU fuel elements used in the demonstration only 3 were cycled into the reactor as fresh fuel. Table 10 summarizes the burnup results for these three HEU fuel elements.

Nearly all of the HEU elements were previously irradiated before the beginning of the whole-core demonstration. Thus, their $^{235}$U masses were uncertain at the start of the demonstration. By combining the $^{140}$La and $^{137}$Cs gamma-scanning data, an experimental value was obtained for the $^{235}$U content of the HEU elements at the start of the demonstration. These initial mass values were used in the REBUS-3 burnup calculations. $^{235}$U fuel element masses based on the gamma-scanning of HEU fuel elements are given in Ref. (3).

Table 11 gives the average burnup status for each of the LEU fuel elements at the end of the demonstration. Seven of the standard 19-plate fuel elements achieved average burnups in excess of 50% while two of the 15-plate fuel followers had
average burnups of nearly 75%. Because of the early shutdown of the ORR, however, 32 Babcock and Wilcox fuel elements and 4 fuel followers remained unirradiated.

Calculated and measured axial distributions of $^{235}$U burnups were obtained by dividing the fuel column into six segments of equal height (10.0 cm). Table 12 shows these axial distributions for those LEU fuel elements and fuel followers having average burnups of 50% or greater. Segment A is at the bottom of the fuel column and segment F at the top. For the 19-plate elements the maximum burnup is about 65% and occurs in segment C. The maximum burnup for the fuel followers is somewhat greater that 90% and occurs in segment F. Since there are only a few data points available in each axial segment, errors in the numerical integrations are relatively large and contribute to an appreciable scatter in the C/E ratios. Segment A in the fuel followers is normally located deep in the axial reflector below the core where both cross sections and neutron fluxes are quite uncertain. This contributes to the large C/E ratios in this region.

At the conclusion of the whole-core demonstration a number of plates were removed from selected fuel elements and fuel followers and gamma-scanned to measure $^{137}$Cs activity distributions. In addition, small samples for mass spectrometry analyses were cut from a number of these plates. Analysis of this post-irradiation data provides fuel-element-averaged $^{235}$U burnups which are independent of those obtained from the gamma-scanning of full-sized fuel elements (see Table 9). Basically, the plate gamma-scanning data is used to determine the fuel-element-averaged $^{137}$Cs activity relative to the activity at the location of the mass spectrometry sample. The measured uranium mass spectrum determines the localized $^{235}$U burnup. Thus, the fuel-element-average burnup is the product of these two values. More details of this method are given in Ref. (3).

Results from these burnup analyses are compared with those discussed earlier in Table 13. REBUS-3 calculated values are also included in this table. Generally speaking, $^{235}$U burnups for the 19-plate LEU fuel elements obtained by the two independent experimental methods are self-consistent. Likewise, the REBUS-3 calculated values agree reasonably well with these experimental results. For the fuel followers, however, the burnup results based on mass spectrometry are somewhat smaller than those obtained from the earlier evaluations. Uncertainties in the experimental values are in the 2-3% range.

**URANIUM AND PLUTONIUM ISOTOPIC MASS RATIOS AS FUNCTIONS OF $^{235}$U BURNUP**

From the mass spectrometry measurements values for uranium and plutonium isotopic mass ratios were obtained for various $^{235}$U
burnups. These results are compared with REBUS-3 depletion calculations in Figs. 5-7. The ORNL data were obtained from Ref. 1, appendix G, and refer to the U$_3$Si$_2$-Al test elements irradiated in the ORR prior to the whole-core demonstration. The ANL-W data were obtained from mass spectrometry measurements made at the Argonne National Laboratory in Idaho using samples taken from LEU fuel elements used in the whole-core demonstration. Isotopic dilution methods were used to measure the mg Pu/g U for samples with varying degrees of burnup.

Figures 5-7 show that the REBUS-3 calculations follow the measurements remarkably well. However, it does appear that the $^{240}$Pu/$^{239}$Pu ratio is over-calculated by about 10% in the 30%-70% $^{235}$U burnup range. These calculations are based on ENDF/B-IV data. Changes in the resonance capture data for the plutonium isotopes in ENDF/B-VI are in the direction of improving the $^{240}$Pu/$^{239}$Pu ratio without significantly changing the other plutonium mass ratios.

CONCLUSIONS

Although the Oak Ridge Research Reactor was permanently shut down before the planned completion of the Whole-Core LEU Fuel Demonstration, the primary objectives of the program were met.

1. All 68 of the commercially-fabricated U$_3$Si$_2$-Al LEU fuel elements (4.8 g/cc U in fuel meat) as well as the 8 LEU fuel followers used in the demonstration performed in a completely safe and acceptable manner without any fuel failures. Seven standard elements and four fuel followers achieved average burnups of 50% or greater. In fact, two of the followers had average $^{235}$U burnups of nearly 75% with peak values greater than 90%.

2. The gradual and safe transition from an all-HEU core, through a series of mixed HEU/LEU cores, to an all-LEU core was clearly demonstrated for the 30-MW Oak Ridge Research Reactor.

3. Numerous experimental measurements validated REBUS-3 fuel cycle predictions. Calculations supported experimentally-determined criticality conditions throughout the burn cycle for each of the 22-full power cores used in the demonstration. REBUS-3 cycle-averaged fuel element powers agreed (usually within 5%) with the measured values. Calculated fuel-element-averaged $^{235}$U burnups are in good agreement with results obtained from two independent experimental methods. Measurements of uranium and plutonium mass spectra in discharged fuel elements support the calculations.

4. Standard methods, models and codes successfully accounted for a wide variety of experimental measurements. These included criticality conditions for unirradiated HEU and LEU
cores radially-reflected with water and beryllium, differential and integral shim rod worth determinations, prompt neutron lifetime evaluations, reaction rate distributions, and isothermal temperature coefficient measurements.

5. The interpretation of differential shim rod worth measurements in the ORR had to take into account the combined effects of temperature changes during the reactivity transient and delayed photoneutron contributions to the total delayed neutron fraction. Changing temperature effects were eliminated by analyzing the initial shape of the measured time-dependent flux. However, only a rough treatment of delayed photoneutron contributions was possible. Plans to extract an effective set of kinetic parameters from an analysis of the shape of the flux die-away curve following a rod drop had to be abandoned because no such measurements were made before the unexpected shutdown of the reactor. Nevertheless, the rough photoneutron treatment resulted in C/E ratios close to unity for differential shim rod worths and for the prompt neutron decay constant in core 179A.

In view of the above remarks, it is concluded that the goals of the Whole-Core LEU Fuel Demonstration have been successfully achieved.

REFERENCES


Table 1. SUMMARY OF THE 30-MW CORES USED IN THE ORR WHOLE-CORE DEMONSTRATION

<table>
<thead>
<tr>
<th>CORE</th>
<th>HEU&lt;sup&gt;a&lt;/sup&gt;</th>
<th>LEU&lt;sup&gt;a&lt;/sup&gt;</th>
<th>CYCLE LENGTH&lt;sup&gt;b&lt;/sup&gt;</th>
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<tr>
<td>177A</td>
<td>8+4</td>
<td>17+2</td>
<td>14.773</td>
</tr>
<tr>
<td>177B</td>
<td>4+2</td>
<td>21+4</td>
<td>18.516</td>
</tr>
<tr>
<td>177C</td>
<td>4+2</td>
<td>21+4</td>
<td>18.411</td>
</tr>
<tr>
<td>177D</td>
<td>0+2</td>
<td>24+4</td>
<td>15.334</td>
</tr>
<tr>
<td>178A</td>
<td>0+2</td>
<td>24+4</td>
<td>12.101</td>
</tr>
<tr>
<td>178B</td>
<td>0+2</td>
<td>25+4</td>
<td>0.644</td>
</tr>
<tr>
<td>178C</td>
<td>0+0</td>
<td>26+6</td>
<td>11.138</td>
</tr>
<tr>
<td>178D</td>
<td>0+0</td>
<td>26+6</td>
<td>16.356</td>
</tr>
<tr>
<td>178H</td>
<td>0+0</td>
<td>26+6</td>
<td>20.277</td>
</tr>
<tr>
<td>178J</td>
<td>0+0</td>
<td>26+6</td>
<td>16.502</td>
</tr>
<tr>
<td>179A</td>
<td>0+0</td>
<td>26+6</td>
<td>20.169</td>
</tr>
</tbody>
</table>

Table 2. EIGENVALUE CALCULATIONS FOR ORR FRESH FUEL CRITICAL CONFIGURATIONS

<table>
<thead>
<tr>
<th>CORE</th>
<th>REFLECTOR</th>
<th>MFE's</th>
<th>SHIM ROD</th>
<th>EIGENVALUES</th>
</tr>
</thead>
<tbody>
<tr>
<td>HEU-1</td>
<td>H&lt;sub&gt;2&lt;/sub&gt;O</td>
<td>Yes</td>
<td>17.21</td>
<td>1.0027±0.0022</td>
</tr>
<tr>
<td>LEU-1</td>
<td>H&lt;sub&gt;2&lt;/sub&gt;O</td>
<td>Yes</td>
<td>15.46</td>
<td>0.9966±0.0019</td>
</tr>
<tr>
<td>179-AX5</td>
<td>H&lt;sub&gt;2&lt;/sub&gt;O</td>
<td>No</td>
<td>15.48</td>
<td>1.0027±0.0019</td>
</tr>
<tr>
<td>HEU-2</td>
<td>Be</td>
<td>Yes</td>
<td>17.34</td>
<td>1.0032±0.0020</td>
</tr>
<tr>
<td>LEU-2</td>
<td>Be</td>
<td>Yes</td>
<td>18.41</td>
<td>1.0018±0.0020</td>
</tr>
<tr>
<td>179-AX6</td>
<td>Be</td>
<td>No</td>
<td>18.22</td>
<td>1.0022±0.0019</td>
</tr>
</tbody>
</table>

<sup>a</sup> 27+6 means 27 19-plate fuel elements and 6 15-plate fuel followers.
<sup>b</sup> Cycle length in full power (30-MW) days.
<sup>c</sup>The Magnetic Fusion Experiments (MFE's) were designed for irradiation-testing of materials for use in magnetic fusion devices.
<sup>d</sup>Shim rod withdrawal position is relative to the fully inserted reference position.
Table 3. DIFFERENTIAL SHIM ROD WORHTS IN THE ORR\textsuperscript{a}

<table>
<thead>
<tr>
<th>CORE</th>
<th>SHIM ROD</th>
<th>R\textsubscript{L}-In.</th>
<th>R\textsubscript{F}-In.</th>
<th>% DELTA K/K PER IN.</th>
<th>CALC.</th>
<th>C/E</th>
</tr>
</thead>
<tbody>
<tr>
<td>179-AX5</td>
<td>D4</td>
<td>12.00</td>
<td>12.23</td>
<td>0.6098</td>
<td>1.045+0.052</td>
<td></td>
</tr>
<tr>
<td>179-AX5</td>
<td>D6</td>
<td>12.00</td>
<td>12.19</td>
<td>0.6120</td>
<td>1.054+0.073</td>
<td></td>
</tr>
<tr>
<td>179-AX5</td>
<td>F4</td>
<td>12.00</td>
<td>12.29</td>
<td>0.3588</td>
<td>1.060+0.086</td>
<td></td>
</tr>
<tr>
<td>179-AX5</td>
<td>F6</td>
<td>12.00</td>
<td>12.36</td>
<td>0.3583</td>
<td>1.066+0.064</td>
<td></td>
</tr>
<tr>
<td>179-AX6</td>
<td>D4</td>
<td>12.00</td>
<td>12.31</td>
<td>0.5769</td>
<td>1.014+0.025</td>
<td></td>
</tr>
<tr>
<td>179-AX6</td>
<td>D6</td>
<td>12.00</td>
<td>12.19</td>
<td>0.5727</td>
<td>0.986+0.038</td>
<td></td>
</tr>
<tr>
<td>179-AX6</td>
<td>F4</td>
<td>12.00</td>
<td>12.31</td>
<td>0.5682</td>
<td>1.044+0.027</td>
<td></td>
</tr>
<tr>
<td>179-AX6</td>
<td>F6</td>
<td>12.00</td>
<td>12.28</td>
<td>0.5717</td>
<td>1.027+0.031</td>
<td></td>
</tr>
<tr>
<td>179-AX7</td>
<td>D4</td>
<td>12.00</td>
<td>12.39</td>
<td>0.3958</td>
<td>1.006+0.039</td>
<td></td>
</tr>
<tr>
<td>179-AX7</td>
<td>D6</td>
<td>12.00</td>
<td>12.36</td>
<td>0.4287</td>
<td>1.030+0.040</td>
<td></td>
</tr>
<tr>
<td>179-AX7</td>
<td>B4</td>
<td>12.00</td>
<td>12.52</td>
<td>0.2455</td>
<td>0.978+0.052</td>
<td></td>
</tr>
<tr>
<td>179-AX7</td>
<td>B6</td>
<td>12.00</td>
<td>12.47</td>
<td>0.2718</td>
<td>1.022+0.062</td>
<td></td>
</tr>
<tr>
<td>179-AX7</td>
<td>F4</td>
<td>12.00</td>
<td>12.81</td>
<td>0.1476</td>
<td>1.146+0.098</td>
<td></td>
</tr>
<tr>
<td>179-AX7</td>
<td>F6</td>
<td>12.00</td>
<td>12.92</td>
<td>0.1656</td>
<td>1.053+0.041</td>
<td></td>
</tr>
</tbody>
</table>

\textsuperscript{a}These measurements were made with a coolant flow rate of 1200 gpm.

Table 4. D6 INTEGRAL ROD WORTH FOR ORR CORE 179-AX5

<p>| INTEGRATION LIMITS, In.\textsuperscript{b} | INTEGRAL WORTH, % DELTA K/K |</p>
<table>
<thead>
<tr>
<th>LL=0.0</th>
<th>UL=26.56</th>
<th>Calc.</th>
<th>Exp.</th>
<th>C/E</th>
</tr>
</thead>
<tbody>
<tr>
<td>7.239</td>
<td>6.855</td>
<td>1.056</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>VIM</td>
<td>26.56</td>
<td>0.0</td>
<td>17.72</td>
<td>1.0400</td>
<td>0.9666</td>
<td>7.299, ±0.0018</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>±0.0020</td>
</tr>
<tr>
<td>DIF3D</td>
<td>26.56</td>
<td>0.0</td>
<td>17.72</td>
<td>1.0371</td>
<td>0.9641</td>
<td>7.309, ±0.0273</td>
</tr>
</tbody>
</table>

\textsuperscript{b}Integration of the differential rod worth from the lower to the upper limit gives the total rod worth.
Table 5. PROMPT NEUTRON DECAY CONSTANT

<table>
<thead>
<tr>
<th>CORE</th>
<th>FRACTION</th>
<th>LIFETIME, SEC</th>
<th>CALC.-SEC⁻¹</th>
<th>EXP.-SEC⁻¹</th>
<th>C/E</th>
</tr>
</thead>
<tbody>
<tr>
<td>HEU-1</td>
<td>8.052E-3</td>
<td>47.87E-6</td>
<td>168.2</td>
<td>167.8±0.6</td>
<td>1.0024±0.0036</td>
</tr>
<tr>
<td>LEU-1</td>
<td>7.980E-3</td>
<td>41.55E-6</td>
<td>192.0</td>
<td>192.3±1.2</td>
<td>0.9984±0.0062</td>
</tr>
<tr>
<td>179A</td>
<td>7.255E-3</td>
<td>55.54E-6</td>
<td>130.6</td>
<td>140.5±0.9</td>
<td>0.9297±0.0060</td>
</tr>
<tr>
<td></td>
<td>7.915E-3</td>
<td>55.54E-6</td>
<td>142.5</td>
<td>140.5±0.9</td>
<td>1.0143±0.0065</td>
</tr>
</tbody>
</table>

Table 6. ISOTHERMAL TEMPERATURE COEFFICIENT

ORR CORE 179-AX7

<p>| ISO-HEMATIC TEMPERATURE COEFFICIENT^c |</p>
<table>
<thead>
<tr>
<th>DELTA K/K/°C</th>
<th>dL/dT, °C</th>
<th>EXP., °C</th>
<th>CALC., °C</th>
<th>C/E</th>
</tr>
</thead>
<tbody>
<tr>
<td>-(1.980±0.091)E-3</td>
<td>(6.78±0.21)</td>
<td>-(1.341±0.075)E-4</td>
<td>-1.301E-4</td>
<td>0.970±0.054</td>
</tr>
</tbody>
</table>

^a The prompt neutron decay constant is the ratio of the effective delayed neutron fraction to the prompt neutron lifetime.

^b Includes estimate of delayed photoneutron contributions.

^c The measured isothermal temperature coefficient is the product of the differential shim rod worth (DELTA K/K/°C) and the slope (dL/dT) of the critical rod position versus temperature curve.
### Table 7. CALCULATED EIGENVALUES CORRESPONDING TO MEASURED CRITICAL CONFIGURATIONS FOR ORR DEMONSTRATION CORES

<table>
<thead>
<tr>
<th>CORE</th>
<th>CYCLE LENGTH, CL</th>
<th>TEMPERATURE-ADJUSTED EIGENVALUE</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>FPD's</td>
<td>BOC 1/3 CL MOC 2/3 CL EOC</td>
</tr>
<tr>
<td>174C</td>
<td>16.8402</td>
<td>1.0024 0.9999 0.9985 0.9992</td>
</tr>
<tr>
<td>174D</td>
<td>12.8554</td>
<td>1.0006 0.9970 0.9960</td>
</tr>
<tr>
<td>174E</td>
<td>10.6228</td>
<td>1.0035 0.9966 0.9963</td>
</tr>
<tr>
<td>174F</td>
<td>15.4282</td>
<td>0.9918 0.9916 0.9935 0.9936</td>
</tr>
<tr>
<td>175A</td>
<td>18.5181</td>
<td>0.9963 0.9974 0.9970 0.9967</td>
</tr>
<tr>
<td>175B</td>
<td>20.3049</td>
<td>1.0058 1.0050 1.0040 1.0006</td>
</tr>
<tr>
<td>175C</td>
<td>17.3891</td>
<td>1.0001 1.0003 1.0009 1.0019</td>
</tr>
<tr>
<td>176A</td>
<td>17.2444</td>
<td>1.0041 1.0034 1.0032 1.0042</td>
</tr>
<tr>
<td>176B</td>
<td>21.8645</td>
<td>1.0000 0.9995 0.9992 0.9996</td>
</tr>
<tr>
<td>176C</td>
<td>19.4357</td>
<td>0.9952 0.9946 0.9950 0.9936</td>
</tr>
<tr>
<td>176D</td>
<td>19.4463</td>
<td>1.0042 1.0049 1.0048 1.0084</td>
</tr>
<tr>
<td>177A</td>
<td>14.7731</td>
<td>1.0018 1.0033 1.0030 1.0054</td>
</tr>
<tr>
<td>177B</td>
<td>18.5160</td>
<td>1.0018 1.0004 1.0006 0.9996</td>
</tr>
<tr>
<td>177C</td>
<td>18.4107</td>
<td>1.0038 1.0028 1.0038 1.0038</td>
</tr>
<tr>
<td>177D</td>
<td>15.3341</td>
<td>1.0065 1.0068 1.0070 1.0090</td>
</tr>
<tr>
<td>178A</td>
<td>12.1006</td>
<td>1.0006 1.0012 1.0053</td>
</tr>
<tr>
<td>178Ba</td>
<td>0.6445</td>
<td>0.9946 0.9869a</td>
</tr>
<tr>
<td>178C</td>
<td>11.1377</td>
<td>1.0062 0.9999 0.9996</td>
</tr>
<tr>
<td>178D</td>
<td>16.3556</td>
<td>1.0030 1.0006 0.9986 1.0002</td>
</tr>
<tr>
<td>178H</td>
<td>20.2765</td>
<td>1.0130b 0.9969 0.9975 0.9959</td>
</tr>
<tr>
<td>178J</td>
<td>16.5022</td>
<td>0.9981 0.9983 0.9982 0.9978</td>
</tr>
<tr>
<td>179A</td>
<td>20.1687</td>
<td>0.9969 0.9990 0.9995 0.9991</td>
</tr>
</tbody>
</table>

---

*a* Insufficient excess reactivity. EOC control rod positions were not recorded, only estimated.

*b* Calculation neglects $^{135}$Xe buildup from the just previous experimental core, 178-EX1.
Table 8. SUMMARY OF ORR FUEL ELEMENT POWER C/E RATIOS

<table>
<thead>
<tr>
<th>CORE:</th>
<th>174D</th>
<th>174E</th>
<th>174F</th>
<th>175A</th>
<th>175B</th>
</tr>
</thead>
<tbody>
<tr>
<td>A2</td>
<td>T519 0.670 1.009</td>
<td>T527 0.630 1.096</td>
<td>T547 0.683 1.057</td>
<td>T521 0.589 1.104</td>
<td>T534 0.663 1.049</td>
</tr>
<tr>
<td>A3</td>
<td>T530 0.859 0.991</td>
<td>T521 0.766 1.086</td>
<td>T549 0.768 1.122</td>
<td>T537 0.798 1.055</td>
<td>T531 0.802 1.061</td>
</tr>
<tr>
<td>A4</td>
<td>T554 1.038 1.006</td>
<td>T557 0.975 1.078</td>
<td>T544 1.014 1.056</td>
<td>T525 0.971 1.047</td>
<td>T552 0.976 1.051</td>
</tr>
<tr>
<td>A5</td>
<td>C021 1.055 1.097</td>
<td>C021 1.049 0.993</td>
<td>C024 1.065 1.127</td>
<td>C027 1.025 1.136</td>
<td>N001 1.022 1.151</td>
</tr>
<tr>
<td>A6</td>
<td>T555 0.995 1.029</td>
<td>T545 0.984 1.071</td>
<td>T541 0.996 1.065</td>
<td>T561 0.990 1.041</td>
<td>T556 0.967 1.063</td>
</tr>
<tr>
<td>A7</td>
<td>T540 0.823 0.950</td>
<td>T534 0.780 1.104</td>
<td>T515 0.759 1.103</td>
<td>T538 0.813 1.051</td>
<td>T546 0.787 1.067</td>
</tr>
<tr>
<td>A8</td>
<td>T503 0.644 0.984</td>
<td>T500 0.565 1.095</td>
<td>T519 0.564 1.137</td>
<td>T527 0.599 1.076</td>
<td>T547 0.608 1.061</td>
</tr>
<tr>
<td>B3</td>
<td>T501 0.980 1.003</td>
<td>T465 0.927 0.994</td>
<td>T503 0.996 1.006</td>
<td>T500 0.890 0.998</td>
<td>T491 0.944 0.984</td>
</tr>
<tr>
<td>B5</td>
<td>T455 1.007 1.046</td>
<td>T458 1.040 0.988</td>
<td>T416 1.043 0.983</td>
<td>T454 0.969 1.000</td>
<td>T417 1.004 1.006</td>
</tr>
<tr>
<td>B7</td>
<td>T497 0.889 0.962</td>
<td>T505 0.891 1.023</td>
<td>T522 0.945 0.965</td>
<td>T535 0.921 0.957</td>
<td>T484 0.901 0.979</td>
</tr>
<tr>
<td>C2</td>
<td>C022 1.170 1.038</td>
<td>C022 1.177 0.969</td>
<td>C025 1.194 1.033</td>
<td>C028 1.154 1.011</td>
<td>N002 1.211 0.986</td>
</tr>
<tr>
<td>C4</td>
<td>T526 1.225 0.987</td>
<td>T507 1.142 1.012</td>
<td>T508 1.150 1.016</td>
<td>T528 1.111 0.989</td>
<td>T530 1.154 0.978</td>
</tr>
<tr>
<td>C5</td>
<td>T419 1.161 0.956</td>
<td>T420 1.107 1.010</td>
<td>T453 1.049 1.055</td>
<td>T473 1.068 1.003</td>
<td>T456 1.128 1.003</td>
</tr>
<tr>
<td>C6</td>
<td>T535 1.214 0.945</td>
<td>T528 1.206 0.973</td>
<td>T530 1.129 0.941</td>
<td>T540 1.163 0.974</td>
<td>T515 1.167 0.974</td>
</tr>
<tr>
<td>C8</td>
<td>C023 1.135 0.915</td>
<td>C023 1.134 0.952</td>
<td>C026 1.127 0.999</td>
<td>C029 1.076 1.010</td>
<td>N003 1.113 0.976</td>
</tr>
<tr>
<td>D2</td>
<td>T379 0.771 0.993</td>
<td>T368 0.750 0.981</td>
<td>T408 0.769 0.979</td>
<td>T328 0.739 1.032</td>
<td>T439 0.709 1.020</td>
</tr>
<tr>
<td>D3</td>
<td>T548 1.262 0.984</td>
<td>T529 1.282 0.965</td>
<td>T533 1.249 0.991</td>
<td>C021 1.215 1.006</td>
<td>C024 1.294 0.973</td>
</tr>
<tr>
<td>D5</td>
<td>T434 1.073 0.965</td>
<td>T396 1.161 0.924</td>
<td>T426 1.049 0.988</td>
<td>T364 1.041 0.974</td>
<td>T460 1.186 0.955</td>
</tr>
<tr>
<td>D7</td>
<td>T539 1.214 0.921</td>
<td>T559 1.138 0.937</td>
<td>T560 1.219 0.970</td>
<td>C022 1.183 1.002</td>
<td>C025 1.231 0.976</td>
</tr>
<tr>
<td>D8</td>
<td>T469 0.784 0.920</td>
<td>T475 0.748 1.016</td>
<td>T430 0.755 0.992</td>
<td>T418 0.724 0.971</td>
<td>T523 0.742 0.986</td>
</tr>
<tr>
<td>E2</td>
<td>T515 0.883 0.989</td>
<td>T517 0.887 0.946</td>
<td>T532 0.909 0.986</td>
<td>T023 0.921 1.001</td>
<td>C026 0.949 0.992</td>
</tr>
<tr>
<td>E4</td>
<td>T537 1.174 0.980</td>
<td>T531 1.180 0.969</td>
<td>T518 1.164 0.993</td>
<td>T558 1.235 0.947</td>
<td>T553 1.251 0.929</td>
</tr>
<tr>
<td>E5</td>
<td>Ir</td>
<td>Ir</td>
<td>Ir</td>
<td>Ir</td>
<td>Ir</td>
</tr>
<tr>
<td>E6</td>
<td>T549 1.200 0.950</td>
<td>T536 1.257 0.961</td>
<td>T542 1.182 0.992</td>
<td>T548 1.223 0.954</td>
<td>T554 1.205 0.953</td>
</tr>
<tr>
<td>E8</td>
<td>T556 1.037 0.924</td>
<td>T561 1.058 0.948</td>
<td>T562 0.996 0.977</td>
<td>C030 0.980 0.998</td>
<td>N004 0.983 0.985</td>
</tr>
<tr>
<td>F3</td>
<td>T516 0.771 0.939</td>
<td>T464 0.731 0.877</td>
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RMS-DEV: 0.048 0.054 0.055 0.050 0.047

Note: HEU fuel elements (FE) are identified with the letter T. LEU fuel elements are identified with the letters C (CERCA), N (NUKEM) and B (Babcock and Wilcox).
### Table 8. SUMMARY OF ORR FUEL ELEMENT POWER C/E RATIOS  (Continued)

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**RMS-DEV:** 0.051 0.029 0.029 0.062 0.054

**Note:** HEU fuel elements (FE) are identified with the letter T. LEU fuel elements are identified with the letters C (CERCA), N (NUKEM) and B (Babcock and Wilcox).
Table 8. SUMMARY OF ORR FUEL ELEMENT POWER C/E RATIOS  (Continued)

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RMS-DEV: 0.052 0.064 0.045 0.039 0.033

Note: HEU fuel elements (FE) are identified with the letter T. LEU fuel elements are identified with the letters C (CERCA), N (NUKEM) and B (Babcock and Wilcox).
### Table 8. SUMMARY OF ORR FUEL ELEMENT POWER C/E RATIOS (Continued)

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**RMS-DEV:**
- **0.079**  0.062  0.058  0.045  0.030

**Note:** HEU fuel elements (FE) are identified with the letter T. LEU fuel elements are identified with the letters C (CERCA), N (NUKEM) and B (Babcock and Wilcox).
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<th>IRRADIATION HISTORY</th>
<th>FINAL MASS (GM) &amp; BURNUP (%)</th>
<th>MASS (E)</th>
<th>C/E</th>
<th>BU(E)</th>
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Table 9. MEASURED LEU FUEL ELEMENT $^{235}$U MASSES AND BURNUPS (Continued)

NUKEM FUEL ELEMENTS

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<th>IRRADIATION HISTORY</th>
<th>FINAL MASS (GM) &amp; BURNUP (%)</th>
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Table 9. MEASURED LEU FUEL ELEMENT $^{235}$U MASSES AND BURNUPS  (Continued)

BABCOCK AND WILCOX FUEL ELEMENTS

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### BABCOCK AND WILCOX FUEL FOLLOWER ELEMENTS

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</tr>
<tr>
<td>UB005</td>
<td>179A-B4 178J-D4 178H-D4 178D-D4 178C-D4</td>
<td>MASS(E)  C/E  BU(E)  C/E</td>
</tr>
<tr>
<td></td>
<td></td>
<td>119.9  1.013  40.04  0.982</td>
</tr>
<tr>
<td>UB006</td>
<td>179A-B6 179J-D6 178H-D6 178D-D6 178C-D6</td>
<td>MASS(E)  C/E  BU(E)  C/E</td>
</tr>
<tr>
<td></td>
<td></td>
<td>120.5  1.006  39.76  0.991</td>
</tr>
<tr>
<td>UB007</td>
<td>179A-D4</td>
<td>MASS(E)  C/E  BU(E)  C/E</td>
</tr>
<tr>
<td></td>
<td></td>
<td>175.9  1.006  12.07  0.957</td>
</tr>
<tr>
<td>UB008</td>
<td>179A-D6</td>
<td>MASS(E)  C/E  BU(E)  C/E</td>
</tr>
<tr>
<td></td>
<td></td>
<td>175.0  1.009  12.51  0.939</td>
</tr>
</tbody>
</table>

Note: FF's UB001, UB002, UB003, and UB004 were gamma-scanned with the Ge(Li) detector. Results for the UB005, UB006, UB007, and UB008 FF's are based on gamma scans obtained with the NaI detector.
### Table 10. MEASURED HEU FUEL ELEMENT $^{235}$U MASSES AND BURNUPS

<table>
<thead>
<tr>
<th>FUEL ELEMENT</th>
<th>IRRADIATION HISTORY</th>
<th>FINAL MASS (gm) AND BURNUP (%)</th>
<th>( \text{MASS(E)} )</th>
<th>C/E</th>
<th>( \text{BU(E)} )</th>
<th>C/E</th>
</tr>
</thead>
<tbody>
<tr>
<td>T556</td>
<td>175B-A6 174D-E8</td>
<td>242.0</td>
<td>1.002</td>
<td>15.10</td>
<td>0.988</td>
<td></td>
</tr>
<tr>
<td>T561</td>
<td>175A-A6 174E-E8</td>
<td>247.4</td>
<td>0.998</td>
<td>13.18</td>
<td>1.014</td>
<td></td>
</tr>
<tr>
<td>T562</td>
<td>175C-A4 174F-E8</td>
<td>241.6</td>
<td>1.001</td>
<td>15.22</td>
<td>0.998</td>
<td></td>
</tr>
</tbody>
</table>

### Table 11. AVERAGE $^{235}$U BURNUP STATUS OF ORR LEU FUEL ELEMENTS$^a$

<table>
<thead>
<tr>
<th>RANGE</th>
<th>19-PLATE STANDARD FUEL ELEMENTS</th>
<th>15-PLATE B&amp;W</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>CERCA</td>
<td>NUKEM</td>
</tr>
<tr>
<td>%</td>
<td>CERCA</td>
<td>NUKEM</td>
</tr>
<tr>
<td>70-75</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>50-60</td>
<td>2</td>
<td>2</td>
</tr>
<tr>
<td>45-50</td>
<td>5</td>
<td>3</td>
</tr>
<tr>
<td>40-45</td>
<td>3</td>
<td>6</td>
</tr>
<tr>
<td>35-40</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>30-35</td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td>25-30</td>
<td>1</td>
<td>2</td>
</tr>
<tr>
<td>20-25</td>
<td>3</td>
<td>1</td>
</tr>
<tr>
<td>15-20</td>
<td>3</td>
<td>4</td>
</tr>
<tr>
<td>10-15</td>
<td>1</td>
<td>0</td>
</tr>
<tr>
<td>5-10</td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td>0</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>Total:</td>
<td>20</td>
<td>20</td>
</tr>
</tbody>
</table>

$^a$Based on results from the gamma-scanning of full-sized fuel elements.
<table>
<thead>
<tr>
<th>FUEL ELEMENT</th>
<th>LAST CORE-POSITION</th>
<th>QUANTITY</th>
<th>AXIAL SEGMENT&lt;sup&gt;a&lt;/sup&gt;</th>
<th>TOTAL&lt;sup&gt;b&lt;/sup&gt;</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>A</td>
<td>B</td>
<td>C</td>
</tr>
<tr>
<td>C021</td>
<td>177D-C5</td>
<td>BU(E)-%: 48.28</td>
<td>59.26</td>
<td>62.94</td>
</tr>
<tr>
<td></td>
<td></td>
<td>C/E:     1.040</td>
<td>1.031</td>
<td>1.016</td>
</tr>
<tr>
<td>C024</td>
<td>178D-D5</td>
<td>BU(E)-%: 49.98</td>
<td>60.57</td>
<td>63.56</td>
</tr>
<tr>
<td></td>
<td></td>
<td>C/E:     1.036</td>
<td>1.037</td>
<td>1.030</td>
</tr>
<tr>
<td>N007</td>
<td>178H-C5</td>
<td>BU(E)-%: 48.55</td>
<td>59.82</td>
<td>64.33</td>
</tr>
<tr>
<td></td>
<td></td>
<td>C/E:     1.021</td>
<td>1.011</td>
<td>0.985</td>
</tr>
<tr>
<td>N008</td>
<td>178H-D5</td>
<td>BU(E)-%: 50.87</td>
<td>62.77</td>
<td>66.48</td>
</tr>
<tr>
<td></td>
<td></td>
<td>C/E:     1.006</td>
<td>1.005</td>
<td>0.994</td>
</tr>
<tr>
<td>B042</td>
<td>179A-C5</td>
<td>BU(E)-%: 47.98</td>
<td>61.43</td>
<td>64.84</td>
</tr>
<tr>
<td></td>
<td></td>
<td>C/E:     1.031</td>
<td>0.979</td>
<td>0.969</td>
</tr>
<tr>
<td>B043</td>
<td>179A-D5</td>
<td>BU(E)-%: 48.35</td>
<td>63.35</td>
<td>65.77</td>
</tr>
<tr>
<td></td>
<td></td>
<td>C/E:     1.048</td>
<td>0.966</td>
<td>0.970</td>
</tr>
<tr>
<td>B044</td>
<td>179A-E5</td>
<td>BU(E)-%: 47.00</td>
<td>59.95</td>
<td>63.21</td>
</tr>
<tr>
<td></td>
<td></td>
<td>C/E:     1.055</td>
<td>0.990</td>
<td>0.979</td>
</tr>
<tr>
<td>UB001&lt;sup&gt;c&lt;/sup&gt;</td>
<td>178J-F4</td>
<td>BU(E)-%: 40.10</td>
<td>67.74</td>
<td>82.87</td>
</tr>
<tr>
<td></td>
<td></td>
<td>C/E:     1.097</td>
<td>0.956</td>
<td>0.954</td>
</tr>
<tr>
<td>UB002&lt;sup&gt;c&lt;/sup&gt;</td>
<td>178J-F6</td>
<td>BU(E)-%: 37.20</td>
<td>65.18</td>
<td>82.78</td>
</tr>
<tr>
<td></td>
<td></td>
<td>C/E:     1.215</td>
<td>1.004</td>
<td>0.955</td>
</tr>
<tr>
<td>UB003&lt;sup&gt;c&lt;/sup&gt;</td>
<td>179A-F4</td>
<td>BU(E)-%: 23.57</td>
<td>47.81</td>
<td>66.89</td>
</tr>
<tr>
<td></td>
<td></td>
<td>C/E:     1.103</td>
<td>0.974</td>
<td>0.936</td>
</tr>
<tr>
<td>UB004&lt;sup&gt;c&lt;/sup&gt;</td>
<td>179A-F6</td>
<td>BU(E)-%: 21.77</td>
<td>46.33</td>
<td>65.33</td>
</tr>
<tr>
<td></td>
<td></td>
<td>C/E:     1.224</td>
<td>1.015</td>
<td>0.956</td>
</tr>
</tbody>
</table>

<sup>a</sup>Each fuel segment is 10.0 cm in height. Segment A is located at the bottom of the core.

<sup>b</sup>Fuel-element-averaged burnup.

<sup>c</sup>LEU 15-plate fuel follower element.
Table 13. SUMMARY OF MASS SPECTROMETRY - $^{137}$Cs GAMMA SCAN
ANALYSES FOR ORR FUEL ELEMENTS AND FUEL FOLLOWERS

<table>
<thead>
<tr>
<th>FUEL ELEMENT</th>
<th>PLATE</th>
<th>MASS SPEC. SAMPLE LOC. FROM TOP OF PLATE, In.</th>
<th>235U BURNUP (%), METHOD</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>CALC.</td>
<td>MASS SPEC.</td>
</tr>
<tr>
<td>B043</td>
<td>2</td>
<td>16.05</td>
<td>50.93</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>4.05</td>
<td>46.82±1.17</td>
</tr>
<tr>
<td></td>
<td>10</td>
<td>16.05</td>
<td>51.97±1.30</td>
</tr>
<tr>
<td>C024</td>
<td>2</td>
<td>16.05</td>
<td>52.02</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>4.05</td>
<td>53.47±1.34</td>
</tr>
<tr>
<td></td>
<td>10</td>
<td>16.05</td>
<td>50.87±1.27</td>
</tr>
<tr>
<td>N007</td>
<td>2</td>
<td>16.05</td>
<td>50.07</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>4.05</td>
<td>48.81±1.22</td>
</tr>
<tr>
<td></td>
<td>10</td>
<td>16.05</td>
<td>50.60±1.26</td>
</tr>
<tr>
<td>B041</td>
<td>2</td>
<td>16.05</td>
<td>48.55</td>
</tr>
<tr>
<td>C025</td>
<td>2</td>
<td>16.05</td>
<td>46.02</td>
</tr>
<tr>
<td>N006</td>
<td>2</td>
<td>16.05</td>
<td>48.73</td>
</tr>
<tr>
<td>T490</td>
<td>2</td>
<td>4.05</td>
<td>45.3$^b$</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>9.05</td>
<td>49.36±1.23</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>16.05</td>
<td>50.40±1.26</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>21.05</td>
<td>48.97±1.22</td>
</tr>
<tr>
<td>UB002$^c$</td>
<td>2</td>
<td>6.55</td>
<td>73.79</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>20.55</td>
<td>76.89±1.92</td>
</tr>
<tr>
<td></td>
<td>8</td>
<td>6.55</td>
<td>68.21±1.71</td>
</tr>
<tr>
<td>UB005$^c$</td>
<td>2</td>
<td>5.05</td>
<td>39.32</td>
</tr>
<tr>
<td></td>
<td>8</td>
<td>5.05</td>
<td>37.01±1.30</td>
</tr>
</tbody>
</table>

$^a$Based on gamma-scanning of full-sized fuel elements (see Table 9).
$^b$This result is an ORR estimate. It depends on burnups in pre-demonstration cores for which no calculations are available.
$^c$This is a 15-plate fuel follower element.
Water-reflected cores HEU-1
[no fuel elements (FE) in C-4 and C-6],
LEU-1, and 179-AX5 (without MFE's).

Beryllium-reflected cores HEU-2, LEU-2, and 179-AX6 (without MFE's)

Fig. 1 Critical Configurations for Fresh Fuel.
ORR CORE HEU-1
AVERAGE C/E RATIOS FOR IRRADIATED WIRES

RMS DEV = 0.063

ORR CORE LEU-1
AVERAGE C/E RATIOS FOR IRRADIATED WIRES

RMS DEV = 0.051

Fig. 2 Results from Gold Wire Activations.
### ORR CORE 177-AX1
**AVERAGE C/E RATIOS FOR IRRADIATED WIRES**

<table>
<thead>
<tr>
<th></th>
<th>Be</th>
<th>Be</th>
<th>0.82</th>
<th>0.94</th>
<th>0.98</th>
<th>1.02</th>
<th>1.00</th>
<th>Be</th>
<th>Be</th>
</tr>
</thead>
<tbody>
<tr>
<td>B</td>
<td>Be</td>
<td>Be</td>
<td></td>
<td>0.94</td>
<td>SR</td>
<td>0.98</td>
<td>SR</td>
<td>1.26</td>
<td>Be</td>
</tr>
<tr>
<td>C</td>
<td>Be</td>
<td>0.96</td>
<td>1.02</td>
<td>MFE</td>
<td>1.00</td>
<td>0.95</td>
<td>1.02</td>
<td>MFE</td>
<td>0.98</td>
</tr>
<tr>
<td>D</td>
<td>Be</td>
<td>0.97</td>
<td>1.01</td>
<td>SR</td>
<td>1.05</td>
<td>0.97</td>
<td>SR</td>
<td>1.55</td>
<td>0.99</td>
</tr>
<tr>
<td>E</td>
<td>Be</td>
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<td>HFED</td>
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<td>AF</td>
<td>1.02</td>
<td>AF</td>
<td>1.00</td>
<td>Be</td>
</tr>
<tr>
<td>F</td>
<td>Be</td>
<td>Be</td>
<td>1.13</td>
<td>SR</td>
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<td>1.08</td>
<td>SR</td>
<td>1.03</td>
<td>Be</td>
</tr>
<tr>
<td>G</td>
<td>Be</td>
<td>Be</td>
<td>Be</td>
<td>Be</td>
<td>Be</td>
<td>Be</td>
<td>Be</td>
<td>Be</td>
<td>Be</td>
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</table>

**RMS DEV = 0.058**

### ORR CORE 179-A
**Average C/E Ratios for Irradiated Wires**

<table>
<thead>
<tr>
<th></th>
<th>Be</th>
<th>0.97</th>
<th>0.98</th>
<th>0.94</th>
<th>0.95</th>
<th>0.94</th>
<th>Be</th>
<th>0.94</th>
<th>Be</th>
<th>DF</th>
</tr>
</thead>
<tbody>
<tr>
<td>B</td>
<td>Fe</td>
<td>0.91</td>
<td>SR</td>
<td>0.90</td>
<td>SR</td>
<td>0.91</td>
<td>Fe</td>
<td>DF</td>
<td>Fe</td>
<td>Fe</td>
</tr>
<tr>
<td>C</td>
<td>Be</td>
<td>1.01</td>
<td>MFE</td>
<td>0.98</td>
<td>0.97</td>
<td>0.98</td>
<td>MFE</td>
<td>0.90</td>
<td>Be</td>
<td>Be</td>
</tr>
<tr>
<td>D</td>
<td>Fe</td>
<td>1.04</td>
<td>1.04</td>
<td>SR</td>
<td>0.99</td>
<td>SR</td>
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<td>0.98</td>
<td>Fe</td>
<td>Fe</td>
</tr>
<tr>
<td>E</td>
<td>Be</td>
<td>1.02</td>
<td>AF</td>
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<td>1.06</td>
<td>1.02</td>
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<td>Be</td>
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</tr>
<tr>
<td>F</td>
<td>Fe</td>
<td>1.03</td>
<td>SR</td>
<td>1.01</td>
<td>SR</td>
<td>1.08</td>
<td>Be</td>
<td>Be</td>
<td>Fe</td>
<td>Fe</td>
</tr>
<tr>
<td>G</td>
<td>DF</td>
<td>Be</td>
<td>Be</td>
<td>Be</td>
<td>Be</td>
<td>Be</td>
<td>Be</td>
<td>Be</td>
<td>Be</td>
<td>DF</td>
</tr>
</tbody>
</table>

**RMS DEV = 0.050**

Fig. 3 Results from Cobalt-Vanadium Wire Activations

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Fig. 4 Cycle-Averaged Power C/E Ratios for Several ORR Cores.

SR=Shim Rod Assemblies, MFE=Magnetic Fusion Experiments, Ir and Eu=Iridium and Europium Irradiation Facilities, HFED=Mini-Plate Irradiation Facilities, Be=Beryllium Reflector Element, and DFE=Dummy Fuel Element. LEU fuel elements are enclosed with thick black lines except for 179A which is an all LEU core.
Fig. 5 Uranium and Plutonium Mass Ratios for ORR LEU Fuel Elements
Fig. 6 Plutonium Mass Ratios for ORR LEU Fuel Elements
Fig. 7 Plutonium Mass Ratios for ORR LEU Fuel Elements
MEASUREMENTS AND EXPERIENCES DURING CONVERSION
OF DR 3 TO LOW ENRICHED SILICIDE FUEL

Karsten Haack, M.Sc.
Deputy Head of Research Reactor DR 3
Risø National Laboratory
DK-4000, Roskilde, Denmark

ABSTRACT

DR 3 has been operated with 8 U$_3$Si$_2$ 19.75% enriched 180g U235 fuel elements in the core from October 14, 1988 to May 5, 1990, the remaining part of the core being 150g U235 HEU fuel elements. From May 5, 1990, LEU fuel elements, only, have been loaded for fuel replacement. Full LEU-core will be reached December 1990. During every mixed core operation cycle predictions of flux-, power- and burn-up distributions have been performed by means of the 3-dimensional DR3-SIM code and the calculations have been verified by measurements of thermal and fast flux distributions in each fuel element and reactivity changes by each fuel element replacement. Statistics and trends deducted from this comprehensive material is presented in the report. No major problem connected to the conversion has been experienced so far.

INTRODUCTION

At the RERTR-meeting in San Diego, Oct. 1988 (5), I informed about our License Approval and the many and exhausting discussions, calculations, elucidations and evaluations acquired by the authorities, The Danish Nuclear Inspectorate (NI).

We got the permission to start the conversion, and the first 8 LEU fuel elements were loaded into the core Oct. 1988. The permission was given as a "Framework Approval", i.e. on the conditions that we supplied sets of additional information papers before the actual start of the conversion, after ½ years of mixed core operation and after completion of the conversion. The latter should be in the shape of a report on all measurements undertaken during the conversion period, supplemented with our own evaluation of the reactor performance and safety using LEU fuel and our recommendations concerning the future operation conditions.

Meanwhile the NI was moved from the Ministry of Environment to the Ministry of Internal Affairs and all NI employers were replaced. The new NI decided to leave the evaluations of the running cases by the Safety Committee of DR 3 in order to enable NI to make the decisions based on the
recommendations of the Safety Committee.

The significance of this change is an easier case-treatment and we foresee no major problems during the remaining part of the conversion period. The final approval of up to 12 MW LEU operation is expected after the evaluation of our documentary report.

CONVERSION SCHEDULE

The original schedule presented at the San Diego meeting has been followed in principle, but we have been able to utilize the remaining HEU fuel elements more effective, which has extended the conversion period some months (see Figure 1)


From Oct. 1988 through March 1990 DR 3 has been operated with 8 LEU (180g U235, 19.75% enriched, U₃Si₂/Al) fuel elements in selected positions uniformly distributed over the core. The last new HEU fuel element was loaded before start-up to cycle no. 372 at 30 March 1990. Since then new LEU fuel elements, only, have been loaded for replacement of the used HEU fuel elements, as seen on Figure 2.

Full LEU-core may be established at start-up to cycle no. 381: December 13, 1990.

LEU FUEL ELEMENTS PERFORMANCE

All LEU fuel elements loaded in the DR 3 core have performed excellent without troubles of any kind (Sep. 1990). A survey is given in Table 1.
be nearly unaffected by the HEU-LEU conversion. According to the generic calculations in IAEA-TECDOC-324 (1) thermal flux reductions of 9–13% were expected in the core; the fast fluxes should be nearly unaffected by the HEU-LEU conversion.

Table 1. LEU fuel elements loaded in DR 3 during the mixed core operation period. Only LEU/LEU reactivity measurements are stated.

Individual flux profiles and reactivity values have been calculated and measured and reasonable good agreement has been experienced between calculations and measurements.

According to the generic calculations in IAEA-TECDOC-324 (1) thermal flux reductions of 9–13% were expected in the core; the fast fluxes should be nearly unaffected by the HEU-LEU conversion.
Typical thermal and fast neutron flux profiles in the core are shown on Figure 2 - 5 for centre-D3, intermediate-C5) and edge-A2) positions.

In the core position A2 (Figure 2) a very distinct drop of the thermal flux is noticed by the change from HEU to LEU in that position. However, a part of the drop is caused by the LEU fuel element. The drops by fuels replacements in A2 (cycle no. 362 and 370) show this effect. As the LEU fuel element burns up, the thermal flux increases. According to the change and stabilizes at about 9% loss, relative to the HEU flux level. The graphs of C5 and D3 (Figure 3 and 4) show similar characteristics of the thermal flux in D3. Though the mean thermal flux in D3 seems to be unaffected by the change.
from HEU to LEU. A general feature is the large dispersion of the flux values owing to the influence of neighbour fuel elements, which is great in a heavy water reactor.

Figure 5 is an example of the change of the fast neutron flux in an edge position (D6). The change from a HEU to a LEU fuel between cycle no. 373 and 374 is not much different from the HEU-/HEU-changes between cycle nos. 346-347 and 357-358.

In the heavy water reflector 3-7% thermal flux reductions were expected by the generic calculations. This is in rather good agreement with the calculations and measurements shown in Figure 6 for the 4'' vertical experiment tube 4V3. The fast fluxes of 4V3 are shown in Figure 7.
Reactivity predictions during the mixed-core period have been more troublesome than usual, because the fuel weight factors of the LEU fuel elements are different from those of the HEU fuel elements, and the Coarse Control Arm (CCA) calibration curve—which is our reactivity reference—also changes from HEU to LEU operation.

The fuel weight factors (% dk/k change per g U235 added) were measured during the irradiation of the 3 LEU test elements in 1985–86 (3), and all LEU fuel weight factors have been calculated. However, the LEU fuel weight factors are different for a single LEU fuel element in a pure HEU core compared with a LEU fuel element in a pure LEU core.

A compari-
son between the HEU and LEU fuel weight factors is shown in Figure 8. In spite of these troubles we have succeeded in designing the cores within the prescribed limits, see Figure 9 which shows the excess reactivity values at begin-of-cycle and at end-of-cycle.

As very low and very high CCA-angles tilt the vertical thermal flux profiles, we aim at a BOC reactivity value around 14% dk/k. This has been obtained in most cases, as shown on the figure.

MIXED CORE OPERATION

The main aspects or problems concerning the mixed core operation have been:

Troublesome fuel management

A discharged HEU 150g-fuel element contains less than 75g U235. By the replacement by a new LEU 180g-fuel element more than 105g U235 is
added which may increase the reactivity by up to 2.2% dk/k. As the core burn-up per cycle is around 4.5% dk/k it is obvious that it can be difficult to compensate for the burn-up within 1% dk/k by replacing burned-up fuel elements, only. It has been necessary to undertake a lot of "shufflings" (= exchanging edge elements by centre elements) in order to adjust the excess reactivity properly.

**Increasing burn-up by discharge**

Calculations by the DR3-SIM 3D code showed that a U235 mass of 164g in a new LEU fuel element would be equivalent to 150g U235 in a new HEU fuel elements with respect to core excess reactivity. However, it was decided to manufacture the LEU elements with 180g U235 in order to allow for a prospective increasing rig load. Actually, the rig load has decreased because the fuel rod testing programme was terminated in 1989 by political reasons. The fuel rods were tested in high-pressure rigs in the fuel elements, 6-9-rigs in the core at the same time, with a total rig load of 2.4 - 3.6% dk/k.

Consequently, we were obliged to reduce the U235-mass in the core by extending the fuel burn-up to 60-70%.

**Uneven power distribution patterns**

The increased fuel element burn-up influences the fuel management as described earlier. It increases the differences in the power load between new and old fuel elements in the core as well. During the K1-period the 8 LEU elements in the core carried a substantial part of the core power, see Figure 10.

![Figure 10](image-url)

*Sum of Power for eight LEU-positions.*

In cycle no. 354 the 8 new LEU elements produced 42% of the core power although they make up for 31% of the total number of fuel elements in the core, only.

The maximum power of a LEU fuel element was 657 kW, measured in position D3 in cycle no. 367, for below the limit: 836 kW, set by the authorities for the mixed-core period.
Greater neutron flux variations

The greater flux variations owing to the mixed-core operation are demonstrated in the figures 2-5. However, they had only minor influence on the experimenters rigs during the Kl-period, because the LEU-positions were selected as positions usually not used for irradiation rigs.

Change of the kinetic data of the core

The Inhour Equation:

\[ \rho(T) = \frac{1}{K_{\text{eff}} \cdot T} + \sum_{i} \frac{\beta_i}{1 + \lambda_i T} \]

is used for calculation of the reactivity worth of the Coarse Control Arm (CCA) group by "Inverse Kinetics" measurements at shut-down from 10 MW. The delayed neutron data \( \beta_i \) and \( \lambda_i \) used for HEU fuel are those of U235.

For LEU fuel we need to take into account the delayed neutron data of Pu239 as well, because they are quite different from those of U235:

<table>
<thead>
<tr>
<th>Fraction ( \beta ) (%)</th>
<th>U235</th>
<th>Pu239</th>
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<tbody>
<tr>
<td>Decay constant ( \lambda ) (s(^{-1}))</td>
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<td>0,23</td>
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<tr>
<td>Average lifetime ( \frac{\beta}{\lambda} ) (s)</td>
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<tr>
<td>Average lifetime ( \frac{\beta}{\lambda} ) (s)</td>
<td>0,106</td>
<td>0,033</td>
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</table>

Table 2. Delayed neutron data

However, the influence of Pu239 on the Kinetic behaviour of the reactor is small because the Pu239 mass in each fuel element is small: About 3g in mean, compared with the mean U235-mass: 127g. Weighted by these masses the average lifetime for LEU becomes 0,104s, close to that of U235.

Consequently, the influence of Pu239 is insignificant in the daily operation and possibly for accident analysis as well. But it has to be taken in consideration by the "Inverse Kinetic" calculations.

TERMINATION OF THE CONVERSION

When the full LEU core is reached by the end of 1990, a final whole core scan of thermal and fast flux will be made, similar to those made in every cycle of the 2 years conversion period: 8 Co- and 8 Ni-wires in every vacant core position + exp. tube 4V3, summing up to about 10.000 wires irradiated and counted during the conversion period.

All flux measurement results and the corresponding flux and fuel element power calculation results are required in a report to the authorities primo 1991. The report shall furthermore contain all reactivity calculations and measurement results during the conversion period, measurements of the Coarse Control Arm (CCA) calibration curve and measurements of tempe-
ature coefficients in a pure LEU core together with our own evaluation of
the safety and operational aspects of 12 MW LEU operation, based on the
experiences obtained during the conversion period.

We expect that the response of the authorities will be a license for
12 MW LEU operation of DR 3.

ACKNOWLEDGEMENTS

The author acknowledges the enthusiastic, skilful and engaged work of
mr. Povl Wiig on the flux and reactivity calculations and measurements
during the conversion period, done on top of his daily duties.

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Burnup Determination of LEU Fuel at the SAPHIR Reactor

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CH-5232 Villigen PSI, Switzerland

Abstract
The burnup of 20 irradiated LEU fuel elements has been determined as part of the annual series of reactivity-based burnup measurements at the SAPHIR reactor facility. It has been shown that the target discharge burnup value of >60%, attained earlier with HEU fuel elements, can be easily reached with LEU fuel. This confirms - considering the much larger $^{235}\text{U}$ loading of the LEU elements - that a considerably greater, integral energy production (MWD) is being achieved with LEU fuel elements in SAPHIR than with HEU. The accuracy of the reactivity method for burnup determination is estimated as $\pm 5\%$, and high-resolution gamma-spectroscopy measurements are being planned to provide an independent check on the validity of the current results.

1. Introduction

The research reactor SAPHIR, at the Paul Scherrer Institute in Switzerland, is a 10MW$_{th}$ facility of the swimming-pool type employing MTR fuel elements [1]. The usual operational cycle is 3 weeks at full power (24 hours per day with 3 shifts, 7 days per week), followed by 1 week for low-power operation, core loading changes and maintenance work. A total of upto $\sim 6500$ hours of full-power operation is thus achieved annually.

The reactor core, which rests on a grid plate suspended from a movable bridge, usually has an L-shaped beryllium reflector on the two sides accessible to the various beam tubes (used mainly for neutron scattering research). Apart from the 4 control rods, the fine regulating rod and the reflector elements, a typical core loading consists of 30 to 32 MTR fuel elements at different stages of burnup, with two empty in-core positions for isotope production purposes. Outer core irradiation positions can be used for a range of other applications, e.g. irradiation damage studies, nuclear chemistry and silicon transmutation doping.
A standard SAPHIR fuel element has 23 plates. In terms of $^{235}\text{U}$ enrichment, however, there are 3 fuel element types in current use, viz. HEU with 93% (~280g $^{235}\text{U}$), MEU with 45% (~320g $^{235}\text{U}$) and LEU with 20% (~410g $^{235}\text{U}$) enrichment. The reactor, during the last few years, has usually been operated with a "mixed" core, i.e. containing fuel elements of all 3 types. The refuelling strategy, developed over the many years of operational experience available, has been optimized such that a discharge fuel burnup of >60% can be achieved, and this has now been shown to be the case for all 3 fuel element types.

An important aspect of the operational strategy at SAPHIR is an annual series of burnup measurements for the various individual fuel elements which have been in use. This is carried out to check and correct, if necessary, values estimated numerically from the fuel element specifications and its operational history. Results from a series of burnup measurements with LEU fuel elements are reported and discussed in the present paper.

2. Methods for Burnup Determination

2.1 Reactivity Worth Measurements

The basic principle here is the measurement of the reactivity worth of a fuel element of unknown burnup in a well-defined core configuration and the comparison of this value with the worths of at least two reference elements, viz. a fresh fuel element and another with a known burnup.

The core configuration used for these measurements is usually one consisting of fresh fuel elements only, i.e. without any reflector elements and without any empty in-core positions. (Fig. 1 shows details of the SAPHIR core configuration, LDG 576, used in the currently reported series of measurements.) A core-edge position (MP), with fresh fuel elements on 3 sides, is reserved for insertion of the fuel element to be measured, and a neutron source (NS) is located on another side of the core, as is also a special fission chamber (FC) with a high count-rate capability.

The reactor is made critical at low power (1kW), with 3 out of the 4 control rods fully withdrawn and the 4th control rod (GR1, located away from the measurement position MP) adjusted such that the fine regulating rod (FR) acquires a given, desired intermediate position. It is assumed that the distortion of the neutron flux at MP, due to the partly inserted control rod GR1 (as also the regulating rod), is not significant. A rod drop measurement for GR1 (with the
other rods "frozen" in their withdrawn positions) gives the "shutdown reactivity" of the core configuration and can thus serve as a measure of the reactivity worth of the fuel element under investigation, located at MP. A comparison of this value, with shutdown reactivity values obtained with reference fuel elements (i.e. of known burnup) located at MP, leads to an estimate of the burnup of the investigated fuel element.

<p>| | | | | | |</p>
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Fig. 1: SAPHIR Core Configuration LDG 576
(Clockwise from upper left, for each grid position, are given: El.No., enrichment, burnup and description)

Description:

FE  standard MTR fuel element
GRn main control element, n=1,2,3,4
FR  regulating element (fine)
NS  neutron source
MP  measurement position
FC  fission chamber
2.2 High-Resolution Gamma-Spectroscopy

Another effective method for burnup determination is high-resolution gamma-spectroscopy of the fuel [2]. Although the installation of an appropriate detector system for "viewing" the shielded fuel, e.g. under water, requires a significant investment of effort initially, the method has several advantages as compared to the reactivity method. Thus, for example, there is no need for a critical core configuration and burnup measurements can, in fact, be carried out at a suitable position in the reactor pool without effecting normal reactor operation. An experimental arrangement for carrying out such measurements at SAPHIR is currently under construction and will be used in the near future to obtain an independent check on the LEU burnup values reported here on the basis of reactivity measurements.

<table>
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Table 1: Reactivity-Based Burnup Measurements (BU) for LEU Fuel and Comparison with Results (B\textsubscript{Uh}) Deduced from Operational History
3. Results Obtained

Table 1 gives the results of the various LEU fuel element burnup measurements carried out using the reactivity method (Section 2.1). Criticality was achieved (for each investigated fuel element at location MP, Fig. 1) with the main control rods GR2, 3 and 4 fully withdrawn, the rod GR1 and the fine regulating rod FR acquiring the positions given in the table. The shutdown reactivity (rho) was determined by the rod drop method with GR1, only a small correction (~0.06 cent/mm) being necessary for the slight variation (between the individual measurements) in the position of FR. A statistical accuracy of about ±1% was achieved in the rod drop measurements. As regards reproducibility of the GR1 position for criticality, this corresponded to ~±0.2% in burnup (absolute), as exemplified in Table 2 by the results for a fresh (reference) LEU fuel element which was measured repeatedly during the course of the present series.

<table>
<thead>
<tr>
<th>El. No.</th>
<th>GR1 [mm]</th>
<th>FR [mm]</th>
<th>rho [$]</th>
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<td>-0.129</td>
<td>-0.23</td>
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</table>

Table 2: Reproducibility of Reactivity Measurements for a Zero-Burnup (Reference) LEU Fuel Element

Since no destructive post-irradiation analysis of LEU fuel elements has been possible to date, only HEU reference fuel elements with non-zero burnup are currently available in SAPHIR. On the basis of reactor physics calculations carried out for core configurations with both types of fuel elements in the measurement position, it has been shown that – apart from the fact that the reactivity worth of fresh fuel elements are practically the same for HEU and LEU – the gradients of reactivity-vs.-burnup curves are almost identical as well. These calculations were carried out using a 2-dimensioned (X-Y) model with constant axial buckling [3], so that the accuracy of absolute reactivity predictions is questionable. For a statement on relative effects, however (such as a comparison of the predicted gradients), the model may be considered quite adequate. Thus, the fact that only HEU fuel elements could currently be used for "calibrating"
the reactivity scale for the burnup measurements is not expected to have significantly effected the accuracy of the experimental method, viz. about ±5%.

The LEU burnup results obtained in the present series of measurements (BU, in Table 1) have been deduced assuming a linear variation of reactivity with burnup. This seems to be a reasonable simplification, even though the 2-D model calculations indicate some non-linearity (see Fig. 2). The predicted absolute reactivity worths for the non-zero burnup reference fuel elements are considerably lower than the measured values, and it is felt that a more detailed (3-D) model would be needed to effect any significant improvement in the currently used linear interpolation procedure for deducing intermediate burnup.

The highest LEU burnup value obtained in the current measurements is >70% (El.No. 469), while 3 other fuel elements have values >60%. This is ample confirmation that the target discharge burnup values of >60%, attained earlier with HEU and MEU fuel elements in SAPHIR, can be reached with LEU fuel and that the target can, in fact, even be slightly
increased. Also given in Table 1 are burnup values (BUh) estimated from the operational history of the individual LEU fuel elements. This is a rather simple procedure, based on the global energy produced during full-power operation with the given fuel element located in the reactor core. While empirical factors for the different fuel element types (HEU, MEU, LEU) are applied, no explicit consideration of the neutron flux distribution over the core is made, and it is, therefore, not surprising that differences between the BU and BUh values in Table 1 are considerable (>10%, relative) in several cases.

4. Conclusions

Burnup measurements, based on the reactivity method, have been carried out for 20 irradiated LEU fuel elements in SAPHIR, and it has been shown that a discharge burnup of as high as ~70% can be achieved. This is slightly higher than values obtained earlier with HEU fuel in SAPHIR and implies – in view of the much larger $^{235}$U loading of the LEU elements (~410g, cf.~280g) – that a considerably greater integral energy production (MWD) is being achieved with LEU fuel elements.

The accuracy of the reactivity method is estimated as ~±5%. It is planned to carry out an independent set of measurements applying high-resolution gamma-spectroscopy in the near future, and this should provide useful confirmation of the currently reported results.

Acknowledgement:

The authors are grateful to the SAPHIR operational team for the efficient execution of the numerous core manipulations and reactivity measurements.

References


COMMISSIONING TEST OF LOW ENRICHED UAI FUEL CORE
ON UPGRADED JRR-3

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ABSTRACT

The first criticality of the upgraded Japan Research Reactor No.3 with low-enriched uranium (LEU) fuel was achieved on March 22, 1990, and succeeding commissioning test is satisfactorily progressed. The reactor power reached the maximum designed power level, 20MW, on August 22. The measured nuclear and thermal hydraulic characteristics of the reactor gave good agreement with the designed predictions. This paper presents a summary of results of the core physics measurements and hydraulic tests that have been conducted during the commissioning test period.

INTRODUCTION

The construction and installation works of the upgraded Japan Research Reactor No.3 (JRR-3) has been progressed since 1985.[1] The upgraded JRR-3 is the first research reactor to perform the whole core LEU fuel demonstration in Japan. A commissioning test started in March 1990, succeeding the completion of functional tests of all reactor systems.

The commissioning test consists of three phase. Figure 1 shows the commissioning test schedule.

In the first phase, primary coolant flow distribution of the core was measured at non-nuclear stage. The first fuel element was loaded to the core on March 15, and the first criticality of upgraded JRR-3 was achieved on March 22. In this phase, core nuclear characteristics, such as excess reactivity, control rod worth, shut down margin, reactivity coefficients, neutron flux distributions, etc. have been measured mainly at zero-power.

The second phase was the power-up test phase in which reactor power was raised step by step, and reached the maximum designed power level, 20MW, on August 22. In this phase, measurement of thermal power, neutron flux monitor calibration, examination of automatical reactor control system, check of radiation shielding, etc. have been performed.

In the final phase, reactor was continuously operated at 20MW for 100 hours, in order to examine the stability of all the reactor system.
Figure 1 Schedule of Commissioning Test Of Upgraded JRR-3

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<td>-</td>
</tr>
<tr>
<td>TERNAL HYDRAULICS</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>CHARACTERISTICS of EX. FACILITIES</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>OTHERS(C.R. SYSTEM, SHIELDING, etc)</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>REACTOR POWER</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>TOPICS</td>
<td>○ FIRST CRITICALITY (Mar. 22)</td>
<td>○ 20MW ATTAINMENT (Aug. 22)</td>
</tr>
</tbody>
</table>
Figure 2 shows the reactor components of the upgraded JRR-3 and surrounding installations. The upgraded JRR-3 is a light water moderated and cooled pool type reactor. The core components consist of core, heavy water reflector tank, plenum and structural components. They are set up the bottom of the reactor pool.

The core is cylindrical in shape, 0.6 m in diameter and 0.75 m in height. It is composed of 26 standard fuel elements (SFEs), 6 control rod elements, 5 irradiation elements and 12 pieces of beryllium reflector. Each control rod element consists of neutron absorber, box shape hafnium, and follower fuel element (FFE) connected with the absorber. They are driven through the core by CRDM installed beneath the core.

Beryllium reflectors are installed between the fuel region and the inner wall of the heavy water reflector tank. The heavy water reflector tank is a double cylindrical type aluminium vessel, with height of 1.6 m and an outer diameter of 2 m. Nine vertical irradiation thimbles, nine horizontal beam tubes and CNS facility are installed in this tank.

Primary coolant flows downward through the core at nominal flow rate of 2,400 m³/hr.
FUEL DESCRIPTION

Specifications of SFE and FFE are listed in Table 1. The fuel element is MTR type, with 19.75 wt% enriched UA1 _x−Al dispersion fuel and the uranium density of fuel core is 2.2 g/cc. Fuel element geometry is shown in Figure 3. A SFE contains 20 fuel plates and a FFE contain 16. Both elements have the same active length, 750 mm. Nominal U-235 loadings of each element is 300 g and 190 g for SFE and FFE, respectively.

The upgraded JRR-3 is planned to be operated continuously for 28 days per one operational cycle and 9 cycles per one year. Five or six fuel elements will be refueled after every operational cycle, so that fuel elements of nearly 2 cores will be necessary for one year reactor operation. The maximum burn-up of spent fuel is designed 50% on the average of element.

Table 1 Summary of Fuel Specification

<table>
<thead>
<tr>
<th>Item</th>
<th>Standard Fuel Element</th>
<th>Follower Fuel Element</th>
</tr>
</thead>
<tbody>
<tr>
<td>Size of Fuel</td>
<td>76×76×150mm</td>
<td>64×64×880mm</td>
</tr>
<tr>
<td>U-235 Enrichment</td>
<td>20%</td>
<td>20%</td>
</tr>
<tr>
<td>U-235 Contents</td>
<td>300g</td>
<td>190g</td>
</tr>
<tr>
<td>Size of Fuel Plate</td>
<td>1.52′×71′×770′mm</td>
<td>1.52′×60′×770′mm</td>
</tr>
<tr>
<td>Fuel Plate Number</td>
<td>20/Element</td>
<td>16/Element</td>
</tr>
<tr>
<td>Fuel Meat Material</td>
<td>Dispersed UA1−Al</td>
<td>Aluminium Alloy</td>
</tr>
<tr>
<td>Cladding Material</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Maximum Burn-up</td>
<td>50% (on the average)</td>
<td></td>
</tr>
</tbody>
</table>

Figure 3 Cross Section of Fuel Elements
CORE HYDRAULIC MEASUREMENT

Preceding the fuel loading test, distributions of primary coolant flow velocity in the core was measured under the condition of various total primary coolant flow rates, using non-fueled mock-up fuel elements (both of SFE and FFE). These aluminum alloy elements had the same geometrical specifications and were fabricated the same fabricator as the real fuel elements.

Two particular elements (one for each type) with Pitot tubes were used to measure the coolant velocity at the several coolant channels. Measurements were performed for almost all the fuel loading position in the core.

Figure 4 shows the measured minimum and averaged coolant velocities inside the each element at the total coolant flow rate of 2,400 m³/hr. The measured coolant velocities of all the fuel loading position satisfactorily exceeded the minimum limitation required by the thermal hydraulic design.[2]

APPROACH TO CRITICALITY

Four BF₃ counter channels and two CIC channels were prepared as extra neutron instrument channels for the approach to criticality. The BF₃ counters and the start-up source (a Sb gamma-ray source) were positioned as shown in Figure 5, and the CIC chambers were positioned inside the irradiation thimbles in the heavy water reflector tank.

The fuel loading procedure of the upgraded JRR-3 required the FFES to be loaded first as the part of control rod elements. They were loaded in position C1, E1, B4, F4, C6 and E6 connected with each neutron absorber. Initial count rates were measured at this core configuration with the control rods half and fully withdrawn.
SFEs were added in amounts predicted from the inverse multiplication curves, in order shown in Figure 5. The minimum critical core was achieved with 14 SFEs. The measured excess reactivity of the minimum critical core was 1.9 %Δk/k, and critical U-235 loading was estimated 4,970 g.

**REACTIVITY PARAMETERS AND REACTIVITY COEFFICIENTS**

Main reactivity parameters of the core were measured by the inverse kinetic method utilizing personal computers.[3]

The excess reactivity of the fully loaded core was obtained to sum up the increased reactivity for every additional SFE from minimum critical core. The shutdown margin was measured under the one rod stuck condition which had largest reactivity worth. Control rod worths were measured for two different control rod patterns. Reactivity worth of the emergency reactor shut down system, heavy water dumping system, was also measured.

The measurement of the moderator temperature coefficient of reactivity was performed by increasing the moderator (primary coolant) temperature with Joule’s heat of primary pumps. The moderator void coefficient of reactivity was measured by a SFE which had air-filled plastic plates of known volume in all coolant channels, being located several positions in the core.

Table 2 summarizes the measurement results, and Figure 6 represents the measured space dependent reactivity worths of SFEs.

Table 2 Reactivity Parameters

<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>MEASURED VALUE</th>
</tr>
</thead>
<tbody>
<tr>
<td>Excess Reactivity (%Δk/k)</td>
<td>15.9</td>
</tr>
<tr>
<td>One Rod Stuck Margin (%Δk/k)</td>
<td>4.1</td>
</tr>
<tr>
<td>Total Control Rod Worth (%Δk/k)</td>
<td>29.2</td>
</tr>
<tr>
<td>Max. Control Rod Worth (%Δk/k/rod)</td>
<td>5.7</td>
</tr>
<tr>
<td>Temperature Coefficient (%Δk/k°C)</td>
<td>-0.013</td>
</tr>
<tr>
<td>Void Coefficient (%Δk/k/%void)</td>
<td>-0.66</td>
</tr>
<tr>
<td>Heavy Water Dump Worth (%Δk/k)</td>
<td>-1.4</td>
</tr>
<tr>
<td>Xe Worth (at Start-up) (Δk/k)</td>
<td>4.6</td>
</tr>
<tr>
<td>Xe Worth (at Shut down) (Δk/k)</td>
<td>12.8</td>
</tr>
</tbody>
</table>

Figure 6 Space Dependent Reactivity Worth of Standard Fuel Elements
Measurements of thermal neutron flux distributions in the core were performed at low power condition using gold foil activation method. These measurements have purposes of confirming the reactor power calibration at zero-power level, allowing prediction of power distributions prior to full power operation, providing data for comparison with results of calculations. [4] 50 μm-thick small gold foils were mounted onto the surface of fuel plates of all SFEs; 9 vertical positions per plate or 5 plates per element in maximum.

Nickel wire activation measurements are performed to obtain fast neutron flux distributions at another low power condition. Axial distributions of thermal and fast neutron fluxes at irradiation holes in the beryllium and the heavy water reflector region are also measured.

Estimated average values at each SFE loaded position of thermal flux based on the measurements are shown in Figure 7. The average thermal flux value of whole fuel region are estimated $8 \times 10^{13}$ n/cm²·sec. Typical axial distributions are shown in Figure 8. These results give good agreement with the design predictions.

![Figure 7](image1.png)  
**Figure 7** Averaged Thermal Neutron Fluxes in SFEs Based on Measurements

![Figure 8](image2.png)  
**Figure 8** Axial Distributions of Thermal Flux In SFE (C4) and in A Irradiation Thimble in Heavy Water Reflector (DR-1)
CONCLUSIONS AND STATUS

The commissioning test of the upgraded JRR-3 is being successfully progressed, including full power continuous operation of over 100 hours at the beginning of September. During this period, whole reactor systems have worked well with no troubles or malfunctions. A lot of experiments and tests have been carried out to perform the measurements of nuclear and thermal hydraulic characteristics of reactor core. Moreover, many performance tests have been conducted on irradiational and beam experimental facilities, including CNS facility. The final evaluations of the results are being progressed.

The regular operation of the upgraded JRR-3 (28 days operation at 20 MW and 7 days shut down) will be started in this November. The equilibrium core will be achieved at the beginning of April in 1991.

ACKNOWLEDGMENT

The authors would like to acknowledge the cooperation of HITACHI, Ltd. and NKK Cooperation, in conducting the commissioning test of the upgraded JRR-3.

REFERENCES

PREDICTION AND MEASUREMENT OF THE NEUTRON ENVIRONMENT IN OSURR EXPERIMENTAL FACILITIES FOLLOWING CONVERSION TO LEU

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ABSTRACT

Neutron energy spectra were measured for two OSURR experimental facilities. The method of multiple foil activation was used with the SAND-II neutron spectrum unfolding code to obtain spectral data for the Central Irradiation Facility and Rabbit tube. Limited data was obtained for the thermal column using only gold foils. Calculations were made of the neutron environment in various locations using the MORSE code. A shift towards higher neutron energies was observed comparing results for LEU and HEU cores. A slight loss in total thermal neutron flux was measured for all positions. Calculational results show reasonable agreement with measured data.

INTRODUCTION

The Ohio State University Research Reactor (OSURR) is an open pool, MTR-type research and training reactor, licensed to operate at a steady-state thermal power of 10 kW. Construction of the OSURR facility began in mid-1959 and was completed in late 1960. Initial criticality of the OSURR was attained in March, 1961. The OSURR system was built by the Lockheed Nuclear Products Division of the Lockheed Georgia Company.

The OSURR was initially fueled with high-enrichment uranium (HEU) fuel, similar to the MTR fuel design, utilizing flat-plate, picture-frame elements. Nominal uranium enrichment was 93%. All fuel elements had a nominal 3 X 3 in cross section, with cylindrical lower end boxes for insertion into a grid plate which has a total of 30 possible element positions arranged in a 5 by 6 array. An air-filled central tube, denoted the Central Irradiation Facility (CIF), is the standard high-flux experimental irradiation facility of the OSURR. All components were arranged in a 5 by 5 array on the grid plate. A row of graphite-bearing irradiation elements, denoted as Graphite Isotope Irradiation Elements (GIIEs), were positioned along one edge of the core in one complete grid plate row, thus allowing all grid plate positions to be occupied. OSURR control rods include three shim safety rods and one regulating rod. The shim safety rods are grooved stainless steel, oval-shaped solid rods having a nominal 1.5%-weight natural boron concentration, while the regulating rod is solid, ungrooved stainless steel. Shim safety rod poison section nominal length is 61 centimeters.

The OSURR core is located in an open reactor pool containing 5800 gallons of filtered, demineralized light water. The core is cooled by natural convection, and operates at room temperature and atmospheric pressure. Pool depth is 20 feet, with a minimum depth of 15 feet from the top of the core to the pool surface. Experimental facilities, in addition to the CIF and GIIEs noted above, include two 6-inch I.D. beam ports, a pneumatic transfer facility (Rabbit), two graphite-loaded thermal columns, and several movable irradiation tubes.
The University began to study conversion of the OSURR to low enrichment uranium (LEU) fuel in 1984, when it was clear that the U.S. Nuclear Regulatory Commission (NRC) would pass a rule requiring use of LEU in domestic, NRC-licensed non-power reactor (NPR) facilities, where feasible. A proposal was submitted to the U.S. Department of Energy (DOE) in March, 1985, to fund conversion of the OSURR from HEU to LEU fuel. It was proposed to utilize DOE funding to accomplish the fuel conversion, while University cost-sharing would fund concurrent studies and efforts to upgrade the OSURR operating power to as much as 500 kW steady-state thermal power. DOE funding was approved in July, 1985, with work beginning in August, 1985.

OSURR conversion efforts completed to date have been described in previous publications and presentations [1]. These involved analysis of the LEU core and prediction of safety-related parameters, as well as other analyses and design efforts related to the power upgrade phase of the project and possible changes in experimental facilities. The OSURR SAR was submitted to the NRC in October, 1987. The NRC issued an order dated September 27, 1988, to convert the OSURR to LEU fuel. This order appeared in the October 4, 1988 Federal Register and became effective 30 days later. The LEU fuel elements were received at the OSURR facility on 11/8/88. The HEU core was unloaded and placed in a secured storage pool beginning on 11/9/88 and completed in late November, 1988. LEU fuel element loading began on 12/7/88, with criticality for the initial core configuration attained on 12/15/88. A standard core geometry was established which provided reasonable operational margins for control rod worth and excess reactivity. Figure 1 shows the current OSURR core geometry, denoted as standard core geometry LEU-2. The fuel loading of the element at position 5B is the same as that of a control element. A 4 in Pb wall acts as a gamma shield for Thermal Column 1. The shield is approximately 14 inches away from the edge of the core. Thermal Column 1 then extends 60 in back from the lead.

Fig.1: OSURR LEU-2 Core Configuration

Results of core parameter testing were presented in a prior publication [2]. Good agreement between predicted and measured values was obtained for parameters such as excess reactivity, shutdown margin, and control rod worths. Notice of NRC approval to upgrade the OSURR operating power appeared in the March 30, 1990 edition of the Federal Register. The 30-day period for leave to intervene expired on April 30, 1990, with no petitions to intervene received. Final approval of the change in operating power is pending issuance of OSURR Technical Specifications by the NRC.

Conversion to LEU results in unavoidable changes to the neutron environment in and around the reactor core. Several reactor operators have expressed concern over the possible impact of these changes on the capabilities of the reactor. In particular, the decrease in thermal neutron flux and the increase in higher-energy neutrons can be detrimental to the performance of various classes of research reactors. For example, with relatively
low-power research reactors, reduction in thermal flux worsens an already limited capability for activation analysis [3]. Conversely, for higher-power research reactors, where large sums have been expended for optimizing the neutron environment in the core or in certain experimental facilities, changes in the neutron energy spectrum can negate the gains made in the optimization process [4]. Thus, measurement of the changes induced in the OSURR neutron spectrum can aid in assessing these potential impacts for reactor facilities planning conversion to low enrichment U$_{3}$Si$_{2}$ fuel.

The main objective of this paper is to report neutron spectrum measurements within various OSURR experimental facilities following conversion to LEU fuel. The experimental results are compared against predictions using the MORSE code. The changes in the neutron environment following conversion are assessed using both the experimental and computational results.

OSURR NEUTRON SPECTRUM MEASUREMENTS

Methodology

For measurement of neutron energy spectra, the OSURR staff has traditionally utilized the method of multiple foil activation combined with a neutron spectrum unfolding code. Such an approach is typical for research reactors, given the availability of appropriate neutron dosimeters (foils) and the acceptance of various unfolding codes based on neutron-induced nuclear reaction cross section libraries. Several graduate students have performed such measurements for various OSURR experimental facilities as part of their degree programs [5-7].

Selection of appropriate foils and cover materials depends partly on the activation properties of the target material and decay characteristics of the product radionuclides. Also, foil selection may depend on the type of neutron spectrum unfolding desired. For example, in power reactors, where material damage induced by high-energy neutron is of concern, accurate spectrum unfolding in the fast neutron energy range is desirable. Thus, dosimeters are selected whose sensitivities are high in this energy region. Research reactors operators, however, are generally interested in an overall neutron energy spectrum determination for their facilities, including specification of the energy-dependent neutron flux from thermal energies on the order of fractions of eV through fission neutron energies of tens of MeV. Thus, a so-called "broad spectrum" foil selection is appropriate in this application. Table 1 lists the foil elements and reaction data for neutron dosimeters used in OSURR neutron spectrum measurements. In general, foils were 0.5 in in diameter, and 0.005 in thick. Cadmium covers were 0.04 in thick, and large enough to completely cover the foils.

<table>
<thead>
<tr>
<th>Foil Composition</th>
<th>Target Isotope</th>
<th>Reaction Type</th>
<th>Cover Material</th>
</tr>
</thead>
<tbody>
<tr>
<td>Au</td>
<td>$^{197}$Au</td>
<td>n-γ</td>
<td>none</td>
</tr>
<tr>
<td>Au</td>
<td>$^{197}$Au</td>
<td>n-γ</td>
<td>Cd</td>
</tr>
<tr>
<td>Cu</td>
<td>$^{63}$Cu</td>
<td>n-γ</td>
<td>none</td>
</tr>
<tr>
<td>Cu</td>
<td>$^{63}$Cu</td>
<td>n-γ</td>
<td>Cd</td>
</tr>
<tr>
<td>Mn-Cu</td>
<td>$^{55}$Mn</td>
<td>n-γ</td>
<td>none</td>
</tr>
<tr>
<td>Mn-Cu</td>
<td>$^{55}$Mn</td>
<td>n-γ</td>
<td>Cd</td>
</tr>
<tr>
<td>Co</td>
<td>$^{60}$Co</td>
<td>n-γ</td>
<td>Cd</td>
</tr>
<tr>
<td>Fe</td>
<td>$^{56}$Fe</td>
<td>n-γ</td>
<td>none</td>
</tr>
<tr>
<td>Fe</td>
<td>$^{56}$Fe</td>
<td>n-γ</td>
<td>Cd</td>
</tr>
<tr>
<td>Al</td>
<td>$^{27}$Al</td>
<td>n-γ</td>
<td>Cd</td>
</tr>
<tr>
<td>Al</td>
<td>$^{27}$Al</td>
<td>n-γ</td>
<td>Cd</td>
</tr>
<tr>
<td>Ni</td>
<td>$^{58}$Ni</td>
<td>n-γ</td>
<td>Cd</td>
</tr>
<tr>
<td>Mg</td>
<td>$^{24}$Mg</td>
<td>n-γ</td>
<td>Cd</td>
</tr>
</tbody>
</table>

The OSURR staff has used the SAND-II neutron spectrum unfolding code for previous measurement of the neutron environment at various locations. SAND-II was used for the present measurements because of its availability and relative ease of use. SAND-II is an iterative foil activation method developed by McElroy [8] in 1965, and is generally applicable to all types of neutron energy spectra in the range of $10^{-10}$ to 18 MeV. The code requires specification of an initial spectral approximation based on the user's knowledge of the physical system under study. SAND-II uses iterative spectral perturbation to obtain a "best fit" solution for a system of simultaneous integral equations for saturated foil activity, based on infinitely-dilute foil activities, which are in turn derived from the measured values of decay-corrected foil activity. SAND-II contains an internal reference...
library of atom densities and cross sections for three cover materials (cadmium, boron, and gold).

OSURR personnel have modified the original SAND-II code obtained from the Radiation Shielding Information Center (RSIC) to execute on an 80386-based personal computer. Further modifications have allowed the use of updated multigroup neutron cross section libraries, such as DOSDAM81 and DOSDAM84 [9].

In previous studies, the OSURR neutron energy spectrum has been represented by three components: a Maxwellian thermal component assuming thermal equilibrium at 20°C, a 1/E or "resonance" matched to the thermal component at or about 1.85 x 10^{-7} MeV and a Watt-form fission spectrum matched to the 1/E form at 0.5 MeV, extending through 18 MeV. The three component form was used as initial input to the SAND-II calculation, and convergence was attained fairly rapidly in most cases. A variable match point feature was added to the SAND-II internal library of spectrum functions, which allowed for differences in the magnitude of the thermal component in various locations. This feature allows specification of the match point energy between the thermal and 1/E components of the approximation spectrum. The upper match point energy is fixed at 0.5 MeV.

Locations in and around the OSURR core selected for neutron spectrum measurement included the CIF, Rabbit facility, main graphite thermal column, both beam ports, and selected positions between fuel plates within the boundaries of given fuel elements. To date, only the CIF and Rabbit neutron spectra have been fully characterized using a full range of foil types, while limited measurements have been done in the thermal column. Other measurements are planned as facility and personnel scheduling permits. Foil placement between fuel plates will also require fabrication of a special holder assembly.

The CIF, Rabbit, and thermal column locations were selected for initial measurement since they are the most heavily utilized OSURR experimental facilities. The CIF and Rabbit are used mostly in neutron activation experiments, while Thermal Column 1 in Fig. 1 has been utilized in boron neutron capture therapy (BNCT) research. It was judged important to characterize the impact of the fuel conversion on these research areas, since many research reactors have or are planning similar programs.

Foil irradiations were designed to either achieve saturation activity for certain reaction products, or induce enough activity to allow relatively simple determination of end-of-irradiation activity, from which saturation activity could be calculated. Generally, irradiation times of a few minutes up to several hours were used, depending on the desired reaction. Irradiation were carried out at full power. Isotope activities were measured with a computer-based Ge(Li) spectrometer system. Measured activities were corrected for decay, saturation time, and foil/cover perturbations, then normalized to disintegrations/target nucleus, prior to input to the SAND-II code. SAND-II output was directed to various data files for later processing, including hardcopy output, further numerical analysis, and graphical plotting of differential and integral neutron energy spectra.

Results

Table 2 shows the results of the neutron spectrum measurements in various experimental facilities for core geometry LEU-2, as well as other measurements which will be discussed in later sections. Figure 2 plots the neutron spectral data for the CIF. The overall spectrum shape shown in Fig. 2 is characteristic of all OSURR experimental facilities, with differences in the magnitude of the flux in various neutron energy regions. Table 2 shows that the total neutron flux in the CIF of the OSURR operating at 10 kW is estimated to be 7.48 x 10^{11} neutrons per square centimeter per second (nv). Of this, about 4.22 x 10^{11} nv falls within the subcadmium energy levels, i.e., less than about 0.5 eV. For the OSURR CIF, the neutron flux above 0.5 MeV is estimated by SAND-II as about 1.23 x 10^{11} nv, which is normally taken to be fission spectrum. Thus, about 2.03 x 10^{11} nv falls within the 1/E, or resonance region. Another means of characterizing the spectral distribution is by subcadmium-epicadmium totals. As noted above, the subcadmium flux totals about 4.22 x 10^{11} nv. The epicadmium flux (i.e., all other flux up to and through fission energies) is about 3.26 x 10^{11} nv, yielding a subcadmium/epicadmium ratio of about 1.29 and indicating a fairly "hard" neutron environment.

For the Rabbit facility, SAND-II estimated a total neutron flux of 6.15 x 10^{10} nv, with a spectrum shape similar to that shown in Figure 2. The subcadmium flux is 4.4 x 10^{10} nv, while the epicadmium flux is 1.75 x 10^{10} nv, leading to a subcadmium/epicadmium ratio of about 3.51. This indicates a "softer" neutron environment than the CIF, which is expected given the ex-core location of the Rabbit facility.

A limited series of measurements using only bare and Cd-covered gold foils have been completed for Thermal Column 1. Foils were placed at various distances from the Pb shield at the core end of the thermal column. Initial experimental estimates of thermal flux at these positions were obtained by subtracting Cd-covered foil specific activity from the total specific activity of the bare gold foils, and using a 2200 meter/second capture cross section for ^{197}Au to calculate the subcadmium flux. Table 3 shows the results of these measurements. As expected, the subcadmium flux decays more slowly than the epicadmium flux as distance away from the core increases, due to neutron thermalization in graphite. The flux is essentially completely thermalized at 24 in away from the Pb shield.
Table 2
Neutron Spectrum Measurements in Various Experimental Facilities For the HEU and LEU-2 Cores

<table>
<thead>
<tr>
<th>Neutron Energy</th>
<th>Core Composition and Facility</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>HEU Core Flux (nv)</td>
</tr>
<tr>
<td></td>
<td>CIF</td>
</tr>
<tr>
<td>0.0001 eV - 18 MeV (Total Flux)</td>
<td>6.86E+11</td>
</tr>
<tr>
<td>0.0001 eV - 0.5 eV (Subcadmium Flux)</td>
<td>4.30E+11</td>
</tr>
<tr>
<td>0.5 eV - 18 MeV (Epicadmium Flux)</td>
<td>2.56E+11</td>
</tr>
<tr>
<td>0.5 eV - 0.5 MeV (Resonance Flux)</td>
<td>1.73E+11</td>
</tr>
<tr>
<td>0.5 MeV - 18 MeV (Fission Flux)</td>
<td>8.34E+10</td>
</tr>
<tr>
<td>Subcadmium to Epicadmium Flux Ratio</td>
<td>1.68</td>
</tr>
<tr>
<td>Subcadmium to Fission Flux Ratio</td>
<td>5.16</td>
</tr>
</tbody>
</table>

![Neutron Flux Spectrum](image)

Fig. 2: Measured Neutron Flux Spectrum in the Central Irradiation Facility

Table 3
Results of Gold Foil Measurements at Various Positions
In the OSURRE Thermal Column 1 For LEU-2 Core

<table>
<thead>
<tr>
<th>Location From Pb Shield in Thermal Column (in)</th>
<th>Subcadmium Flux (neutrons/cm²)</th>
<th>Epicadmium Flux (neutrons/cm²)</th>
<th>Epicadmium/Subcadmium Ratio</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>7.0E+09</td>
<td>2.8E+09</td>
<td>4.0E-01</td>
</tr>
<tr>
<td>12</td>
<td>2.3E+09</td>
<td>1.4E+08</td>
<td>6.1E-02</td>
</tr>
<tr>
<td>24</td>
<td>5.9E+08</td>
<td>3.7E+06</td>
<td>6.3E-03</td>
</tr>
</tbody>
</table>
Core and Facility Modeling

One of the main objectives of the OSURR power upgrade is to improve the utilization of the facility in BNCT research. In that respect, new computational work concentrated on the assessment of the changes in the neutron environment in Thermal Column 1. Computer modeling of both HEU and LEU-2 core compositions was done using the MORSE code [10]. MORSE is a three-dimensional, multipurpose, multi-group transport code generally used for shielding calculations. For the computational work, the fuel elements were described as homogenized mixtures, with the control rods completely withdrawn. For the modeling of the rest of the facility, exact geometry and compositions were used. For neutron generation probability distributions, the core power distribution obtained from previous computational work [1] and $^{239}$U fission spectrum were used. The cross-section library was BUGLE-80. BUGLE-80 has a coupled, 47 neutron group, 20 gamma-ray energy group structure, with $P_3$ approximations for angular distributions. Neutron energy group upper limits are from 0.1 eV to 17.3 MeV. Fast neutrons were defined as those falling in groups 1-34, corresponding to an energy range of 7.11 keV to 17.3 MeV. Epithermal neutrons were taken to be those within groups 35-45, or 0.414 eV to 7.11 keV. The upper thermal neutron energy was 0.414 eV (or groups 46 and 47).

For the CIF, computational results obtained in earlier studies used in preparation of the OSURR SAR served as a basis for calculational data. Methods and results of these studies were described in previous publications [1].

Validation and Results

The computational work consisted of MORSE runs on a CRAY YMP with the computational detectors placed at the locations shown in Table 3. Different types of detectors were investigated for suitability. A point detector led to unphysical results. Boundary crossing detectors and track length detectors were found to be more suitable. The parametric work indicated that a large volume track length detector was needed to obtain good statistics with reasonable CPU times. However, since the flux is averaged over the volume of the detector with track length detectors, the thermal flux was found to be underestimated at locations near the core where the flux gradient is high. Boundary crossing detectors were found to yield good results while providing good statistics. Thus, a boundary crossing detector with a 1 in radius was used for all calculations.

To benchmark the code, Table 3 flux ratios were compared to MORSE results from the boundary crossing detector. The MORSE predictions for locations at 0 in and 12 in behind the Pb shield are $0.499\pm0.119$ and $0.555\pm0.023$, respectively. The third location (i.e. 24 in behind the Pb shield) produced inconclusive results because of poor statistics due to deep penetration. Comparison of the MORSE results to the results given in Table 3 shows agreement within 10% - 25% which is reasonable in view of the statistical uncertainty on the MORSE predictions.

In the runs for HEU and LEU-2 cores, the only changes in the code input were core composition and geometry. Results for several locations including the edge of the core, and 0, 12 and 24 inch behind the Pb shield in Thermal Column 1 are shown in Table 4, normalized to neutrons detected/source neutron.

<table>
<thead>
<tr>
<th>Location</th>
<th>Core Composition and Three Group Neutron Spectrum (neutron/source neutron)</th>
<th>LEU Core</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>HEU Core</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Thermal</td>
<td>Epithermal</td>
</tr>
<tr>
<td>Core Edge</td>
<td>1.18E-04</td>
<td>1.09E-04</td>
</tr>
<tr>
<td>0 in. From Pb in Thermal Column</td>
<td>2.32E-05</td>
<td>4.63E-06</td>
</tr>
<tr>
<td>12 in. From Pb in Thermal Column</td>
<td>7.86E-06</td>
<td>1.35E-07</td>
</tr>
<tr>
<td>24 in. From Pb in Thermal Column</td>
<td>3.54E-06</td>
<td>0</td>
</tr>
</tbody>
</table>
Results of neutronic studies of the OSURR core for the SAR [1] indicate a CIF thermal (0 eV to 0.625 eV) flux of about $3.3 \times 10^{11}$ n/s, with an epithermal (0.625 eV to 10 MeV) of about $2.25 \times 10^{11}$ n/s. This leads to a thermal/epithermal ratio of 1.47 for the CIF.

DISCUSSION OF RESULTS

Comparison of HEU and LEU Measurements

As noted earlier, several graduate students measured neutron spectra in various experimental facilities for the HEU core. Table 2 summarizes these measurements for the CIF and Rabbit, and compares them with results presented earlier for the LEU core.

Comparing Table 2 total flux in the CIF for the LEU core to that of the HEU core, there has been an increase of about 9%. This increase results from the change in core geometry. The LEU core has fewer total elements than the HEU core. Table 2 results for various spectral components for these two cases also shows that the gain is in the epithermal energy region. The subcadmium flux in the CIF for the LEU core is reduced about 1.9% from that of the HEU core, while the epithermal flux for the LEU core has increased about 27% over that for the HEU core. Thus, for experiments requiring primarily thermal neutron irradiation (e.g., activation analysis), the LEU core performance is slightly degraded compared with the HEU core for irradiations conducted in the CIF.

For the Rabbit facility, the total flux for the LEU core is reduced about 3.6% from that obtained for the HEU core. This small change can be attributed to fewer fuel elements positioned along the face of the core adjacent to the Rabbit facility. Reduction in the subcadmium flux component of the spectrum is about 2.9% comparing HEU to LEU cores. Again, it can be concluded that performance has been slightly degraded for thermal neutron irradiations conducted in the Rabbit facility.

In general, while thermal flux degradation has occurred following the LEU conversion, epithermal flux has been enhanced. For certain irradiations (e.g., materials damage studies, or inducement of defects), such changes are beneficial. For example, the fission flux (above 0.5 MeV) component of the OSURR CIF LEU-2 neutron spectrum for the LEU core increased about 47.5% over that for the HEU core. These changes are reflected in the flux ratios shown in Table 2.

In examining the Table 4 results, we see that core spectrum hardening due to LEU conversion leads to a substantial increase in the epithermal flux in Thermal Column 1 (i.e. 65% to 400%), while the increase in the fast flux is only 5%. The thermal flux levels are slightly less than that of the HEU except at the 12 in position where the increase in the flux is 24%. The increase possibly occurs due to differences in the neutron thermalization distances originating from the differences in the HEU and LEU core neutron spectra at the core edge.

Comparing Table 3 results to the data obtained for the HEU core in previous experiments [6], again a decrease in thermal flux is observed. At the 0 in position, the loss of subcadmium flux for the LEU core is about 48% from that obtained for the HEU core. Such a relatively large change can be attributed to both LEU conversion and the removal of several fuel elements adjacent to the thermal column extension. The use of gold foils instead of a full set of foil detectors may also contribute this difference. The observed difference may change when more detailed investigations of the thermal column neutron environment are completed.

Comparison of Measured and Calculated Results

Results of the CIF spectrum measurements compared to predicted environment used in the SAR show reasonable agreement. The calculated subcadmium flux is about 22% less than that measured. Similarly, the predicted epithermal flux is about 31% less than that measured. These differences may be the result of differences between the core model used in the calculations and the final core configuration obtained in the initial LEU fuel loading. Also, the measured results obtained from SAND-II unfolding may vary from the actual spectral environment since such methods have inherent uncertainties. SAND-II results may vary from the actual physical environment by as much as 30% [8]. In general, agreement within 30% between computed neutron spectra and measurements made with activated foils and a spectrum unfolding code is considered reasonable.

Since the MORSE code provides flux information in normalized form (neutrons detected/source neutron), comparisons with measured data can be made for spectral component ratios only. At 0 in from the Pb shield in Thermal Column 1, MORSE estimates a thermal/(epithermal+fast) (or subcadmium/epicadmium) flux ratio of about 2. This ratio compares reasonably well with a measured bare/Cd-covered Au foil activity ratio of 2.5. Similarly, at 12 in from the lead shield, the predicted thermal/(epithermal+fast) ratio is about 18, while the
measured Au foil activity ratio is 16.

From Table 3 and Table 4, we see that both the experimental and computational results indicate the same decrease in the epicadmium flux from 0 in to 12 in (i.e. by a factor of 20). For subcadmium flux, decreases by factors of 3.1 and 3.86 are observed in Table 3 from 0 in to 12 in and from 12 in to 24 in, however, the corresponding decreases predicted by MORSE (i.e. Table 4 results) are by factors of 2.2 and 3.1. While the discrepancy between experimental and predicted subcadmium fluxes at 24 in can be attributed to statistical uncertainty on the computational results, the discrepancy at 12 in could not be explained in view of the good agreement between the computed and experimental subcadmium/epicadmium flux ratios at this point (i.e. 18 and 16).

SUMMARY AND CONCLUSIONS

Measurement of the neutron environment in the CIF and Rabbit facility of the LEU-fueled OSURR have been completed. A general hardening of the neutron spectrum in each facility has been observed as a result of conversion to LEU. The loss of total subcadmium neutron flux is on the order of a few percent from that available with the HEU core. Epicadmium fluxes are enhanced 20 to 30 percent over those measured for the HEU core.

Computational models used in the present study as well as in earlier studies related to the SAR preparation were found to be reasonably accurate representations of the measured physical system. Predicted neutron spectra agreed with measured values within 30%.

The impact of LEU conversion on a given research reactor facility will depend on the nature of the facility and its major areas of utilization. For those reactors primarily engaged in thermal neutron irradiations using access facilities within or near the core, the loss of thermal neutron flux can likely be managed by increasing irradiation time, although the increase in non-thermal flux may result in undesirable activation products being produced at concentrations higher than previously observed.

For reactors using ex-core experimental facilities, changes in core geometry related to LEU conversion will likely result in some loss of total neutron flux, with concurrent changes in the thermal/non-thermal flux ratio. Such losses could be offset by operating power adjustment, or by modifying core geometry. Core geometry changes, such as loading more fuel elements near these ex-core facilities, may result in other changes in core parameters (e.g. control rod worth or excess reactivity), requiring further safety analyses to be done for the facility and possibly require license amendments. The OSURR experience also indicates that LEU conversion may lead to a substantial increase in the epithermal flux in a thermal column facility, with negligible increase in the fast flux. Such a change in the neutron environment is desirable in view of the recently increased activity in BNCT research using epithermal neutrons.

For research facilities utilizing optimized cores and/or experimental facilities, the impacts of changes in neutron spectra are potentially severe. If "contamination" of a radiation environment with non-thermal neutrons is sought to be minimized, LEU conversion can be contrary to such a goal. A 20 to 30 percent increase in the non-thermal neutron component in such facilities can negate the considerable effort that may have been invested in the optimization process. Recovery of such losses may require additional studies to re-optimize the environment, engineering changes to add or modify existing shielding and/or filtering, and possible changes in operating power to offset the effects of additional shields and filters. Facilities impacted in this way may wish to seek "unique purpose" exemption from the NRC, or financial support for necessary facility modifications from DOE, citing the purported basis of the LEU conversion rule as a national policy issue which recognizes the role of the Federal Government in providing financial support for conversion-related costs.

REFERENCES

4. Carter, R. S., letter to N.R.C. Secretary (10/26/84).
SESSION V

September 25, 1990

SAFETY TESTS AND ANALYSES FOR CONVERSIONS

Chairmen:

Y. Futamura
(JAERI, Japan)

T. Aldemir
(Ohio State University, USA)
As to the safety review of two fuel elements of LEU silicide fuel to be used in the existing KUR, the hearing in the Science and Technology Agency started in August 1989\(^1\), and the application was accepted in May 1990. Since then, the Technical Advisory Committee for Safety Review\(^2\) have been held more than ten times, and the final committee is anticipated to be held on September 28, 1990. Therefore, we can not talk about the official result of the safety review yet.

The main reasons that the hearing and the committee have taken so long are: (1) the full core conversion to use LEU silicide fuel has been partially examined and (2) several amendments according to the new regulations have been required.

The following items were examined:

(1) Core characteristics\(^3,4\)
- neutron flux distributions
- hot channel factors
- temperature coefficient

(2) LEU silicide fuel
- integrity, e.g. fission product gas release
- homogeneity

(3) Accident analysis
- fuel element drop
- core of vacant plug
- leakage of heavy water from the heavy water tank
- control rod withdrawal
- cold water injection
- coolant channel blockage
- loss of primary coolant
- loss of electricity supply
- failure of primary pumps

(4) Environment
- population distribution

The safety and the fuel fabrication procedure will be approved shortly. The demonstration of two LEU silicide elements will be performed in 1991.
REFERENCES


EXPERIMENTAL STUDY ON
DNB HEAT FLUX CORRELATIONS FOR JMTR SAFETY ANALYSIS

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Japan Atomic Energy Research Institute
Oarai-machi, Ibaraki-ken, Japan

ABSTRACT

Conversion of the JMTR core from MEU fuel to LEU fuel is scheduled to be made in 1993. LEU fuel for the JMTR is determined to be the silicide fuel with 4.8 gU/cm$^3$ and burnable absorbers of aluminum sheathed cadmium wires are located in each side plate. Thermohydraulic analysis of the JMTR has been made to revise the safety analysis report for the application of license on the use of the silicide fuels. In the safety analysis, DNB (Departure from Nucleate Boiling) heat flux correlations for the JMTR downflow condition were reconsidered because it was pointed out that DNB heat fluxes from experiments with thin rectangular channels under low flow rate and low pressure conditions are much lower than predicted values by conventional correlations. Relevant DNB data, however, are very limited for the JMTR operational range, so that DNB experiments were conducted simulating the JMTR fuel subchannel.

INTRODUCTION

The fuel element of the JMTR has 19 flat fuel plates arranged in parallel, with 2.67 mm water gap. The core thermohydraulic design of the JMTR has been made for the normal operation condition at 50MWt, and primary coolant is pressurized at about 13 kg/cm$^2$abs (reactor core). Fuel plates are cooled by forced convection of downflow at the velocity of 10 m/s in order to avoid the nucleate boiling on the surface of fuel plates. Safety analysis for the normal operation, anticipated operational transients and postulated accidents are made in the pressure range of 1~13 kg/cm$^2$abs and the coolant velocity range of 0~10 m/s.

The minimum DNB ratio is specified as one of safety criteria and it should not exceed 1.5 for the reactor normal operation and anticipated operational transients, so DNB heat flux correlation is of great importance for evaluating results of thermohydraulic analysis.

Recent studies by Mishima[1] and Sudo[2], however, pointed out that most of conventional DNB heat flux correlations might be applicable only to high flow.
rate conditions and positive subcooling at the burnout position. They also suggested that experimental DNB heat fluxes with thin rectangular channels under low flow rate and low pressure conditions are much lower than predicted by those correlations particularly for downward flow, due to the buoyancy effects.

Therefore, DNB experiments were conducted to provide DNB heat flux data for reconsidering DNB correlations applicable to JMTR thermohydraulic conditions, because available data are very scarce except for ones around atmospheric pressure.

REVIEW OF DNB HEAT FLUX CORRELATIONS

Existing DNB heat flux correlations were reviewed to consider applicability to the JMTR downflow condition. Since existing correlations are generally supported by many data in other studies, it is considered that selection of the existing correlation which is satisfactorily applicable to the JMTR safety analysis would be more reliable than development of new correlations. Reviewed correlations are listed in Table 1. Five round tube correlations (Lowdermilk, Macbeth, Labuntsov, Thorgerson, Katto) and Mirshak correlation in Table 2 were reviewed in IAEA[3]. Bernath correlation[4] was applied to the JMTR original safety analysis which was revised in 1983. Zenkevich-subbotin correlations were applied to the HFIR thermohydraulic design with an uncertainty factor of 1.3 based on comparison with DNB data by Gambill[4]. Sudo correlations scheme[2] was proposed in recent experimental study for the JRR-3 safety analysis. Considering ranges of thermohydraulic conditions for each correlation, Labuntsov, Katto, Mirshak, Bernath and Sudo correlations appear to be applicable to the JMTR safety analysis.

Table 1 Thermohydraulic Parameters' Ranges of DNB Heat Flux Correlations

<table>
<thead>
<tr>
<th>Correlation</th>
<th>Hydraulic Dia. (mm)</th>
<th>Pressure (kg/cm² abs.)</th>
<th>Velocity (m/s)</th>
<th>Subcooling (°C)</th>
<th>Steam quality</th>
</tr>
</thead>
<tbody>
<tr>
<td>Lowdermilk</td>
<td>1.3 ~ 4.8</td>
<td>Atmospheric</td>
<td>0.03 ~ 30</td>
<td>~24 (inlet)</td>
<td>positive</td>
</tr>
<tr>
<td>Macbeth</td>
<td>Low Velocity</td>
<td>3 ~ 10</td>
<td>1 ~ 138</td>
<td>14 ~ 841 (kg/m²s)</td>
<td>positive</td>
</tr>
<tr>
<td></td>
<td>High Velocity</td>
<td>1 ~ 23.9</td>
<td>14 ~ 5750 (kg/m²s)</td>
<td>65 ~ 1400 (kJ/kg, inlet)</td>
<td>positive</td>
</tr>
<tr>
<td>Labuntsov</td>
<td></td>
<td>1 ~ 200</td>
<td>0.7 ~ 45</td>
<td>0 ~ 240 (burnout point)</td>
<td>negative-0</td>
</tr>
<tr>
<td>Thorgerson</td>
<td></td>
<td>1.7 ~ 13.3</td>
<td>3.05 ~ 18.3</td>
<td>&gt;25 (burnout point)</td>
<td>negative</td>
</tr>
<tr>
<td>Katto</td>
<td></td>
<td>1 ~ 200</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Mirshak</td>
<td>5.3 ~ 11.7</td>
<td>1.7 ~ 5.9</td>
<td>1.5 ~ 13.7</td>
<td>5 ~ 75 (burnout point)</td>
<td>negative</td>
</tr>
<tr>
<td>Bernath</td>
<td>&lt; 30.5</td>
<td>1 ~ 210</td>
<td>0.1 ~ 47.6</td>
<td>0 ~ 182 (burnout point)</td>
<td>negative-0</td>
</tr>
<tr>
<td>Zenkevich</td>
<td>4.1 ~ 11.9</td>
<td>105 ~ 217</td>
<td>&gt;270 (kg/m²s)</td>
<td>&gt;10 (burnout point)</td>
<td>negative</td>
</tr>
<tr>
<td>Sudo</td>
<td></td>
<td>Atmospheric</td>
<td>&lt;3000 (kg/m²s)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>JMTR Fuel Subchannel</td>
<td>5.1</td>
<td>1 ~ 13</td>
<td>0 ~ 10</td>
<td>51 ~ 142 (inlet)</td>
<td></td>
</tr>
</tbody>
</table>
1. Labuntsov Correlation[3]

\[ q_{\text{DNB}} = 347.5 \theta(P)(1+2.5 U^2/\theta(P))^{0.25}(1 + 15.1 C_p \Delta T_{\text{SUB}}/\lambda P^{0.5}) \]  
\[ \theta(P) = 0.99531 P^{1/3}(1 - P/P_c)^{4/3} \]  

2. Katto Correlation[3]

\[ q_{\text{DNB}} = q_{\text{DNB}0} [1 + \kappa(\Delta h_i/h_{f_e})] \]  
\[ q_{\text{DNB}0} = 0.1 C_1 G h_{f_e} \] ( \( q_{\text{DNB}0} \) : DNB heat flux for \( \Delta h_i = 0 \) )

\( \kappa \) and \( C_1 \) are constants depending on thermohydraulic ranges of \( L, H, N \) and \( N'P \) specified by Katto.

3. Mirshak Correlation[3]

\[ q_{\text{DNB}} = 360 (1 + 0.12 U)(1+0.009 \Delta T_{\text{SUB}})(1+0.19 P) \]  


\[ q_{\text{DNB}} = \left[ 8.21 D/(D+P_h/\pi) + 0.058 U/D^{0.6} \right] \]  
\[ \times \left[ 272.9+102.6 \ln P-97.2 P/(P +1.05) - U/0.68 - 1.8 T_B \right] \]

5. Sudo Correlations Scheme[2]

Sudo constituted the correlations scheme both for upflow and downflow as shown below with correlation(5) and (6) originally proposed by Mishima[1] and correlation(7) proposed by himself. He made clear the difference between upflow and downflow and reliability of the scheme with experimental data obtained previously and by himself.

For downflow, lower \( q_{\text{DNB}}^* \) of correlations(6) and (7) is selected. If the selected \( q_{\text{DNB}}^* \) is less than the \( q_{\text{DNB}}^* \) calculated by correlation(5), then the correlation(5) is applied. For upflow, higher \( q_{\text{DNB}}^* \) of correlations(5) and (7) is selected.

\[ q_{\text{DNB}}^* = 0.7(A/A_HH)(W (\tau / \sigma)^{1/2})^{1/2} / (1 + (\tau_{f_e}/\tau)^{1/4})^2 \]  
\[ q_{\text{DNB}}^* = (A/A_HH)(\Delta h_i/h_{f_e}) G^* \]  
\[ q_{\text{DNB}}^* = 0.005(G^*)^{0.611} \]

These correlations were compared in the pressure range for the JMTR safety analysis(1~13 kg/cm² abs), and a comparison at the pressure of 13 kg/cm² abs is shown in Figure 1. Non-dimensional heat flux is taken on the ordinate and non-dimensional mass velocity on the abscissa and the coolant inlet temperature is 49°C. The comparison shows that Sudo correlations scheme predicts the lowest DNB heat fluxes of all correlations, and similar results were observed in the pressure range of 1~12 kg/cm² abs in the present study. Since Sudo correlations scheme has been supported by many round and rectangular channel DNB heat flux data including low flow rate region, the scheme is considered to be most applicable to the JMTR thermohydraulic conditions. However, the applicability of the scheme under the pressurized condition has not been proved yet. DNB heat flux data with rectangular channels at low and moderate pressures are sparse, so that DNB experiments were conducted.
**DNB EXPERIMENT**

**EXPERIMENTAL LOOP**

The experimental loop consists of a test section, a pressurizer, a heat exchanger, a circulating pump, a resin tank, an electric heater, a flow control valve, stop valves and pipings connecting these components. A flow diagram of the experimental loop is shown in Figure 2. The flow orientation in the test section can be set either upward or downward with stop valves. Loop water can be pressurized up to 18 kg/cm² abs by helium gas. The loop is filled with distilled water and the resin tank is installed to maintain water quality. The electric heater is employed to control the inlet water temperature. The test section consists of a pair of heating plates, alumina insulators, glass windows and frames as shown in Figure 3. Each heating plate, which is 40 mm wide and 760 mm long, is made of Inconel 600 and placed on the two opposite sides of the flow channel with a gap of 2.67 mm, simulating a rectangular subchannel of a fuel element. During experiments, flow patterns can be observed through glass windows on both sides of the flow channel. To eliminate premature burnout by corner effect, edges of the flow channel are extended 5 mm beyond edges of heating plates. The test section is enclosed in a containment tube for protecting personnel from flashing steam when a test section is failed. The power to the test section was applied by a 80 V, 2600 A, ac power supply. The flow rate of coolant is measured with a turbine flowmeter and a mass flowmeter. The temperature and pressure of coolant are measured in the upper and lower plena. Sheathed Chromel-Alumel thermocouples are silver-soldered on the back surface of each heating plate to detect DNB.
Table 2 Experimental Conditions

<table>
<thead>
<tr>
<th>Flow Direction</th>
<th>Downward</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pressure</td>
<td>1 ~ 13 kg/cm² abs</td>
</tr>
<tr>
<td>Flow Rate</td>
<td>0 ~ 2800 kg/m² s</td>
</tr>
<tr>
<td>Coolant Inlet Temperature</td>
<td>49°C</td>
</tr>
</tbody>
</table>

Figure 2 Flow Diagram of DNB Experiments Loop

Figure 3 Schematic of Test Section

EXPERIMENTAL PROCEDURE AND CONDITIONS

The loop is evacuated by a vacuum pump, then filled with distilled water and the pump is operated for purification until electric conductivity reaches below 0.1 $\mu$S/cm. Then, coolant is heated up to about 80 °C with the electric
heater and heating plates for deaeration. After deaeration, coolant velocity and pressure are adjusted to the determined values. The coolant inlet temperature is kept at 49 °C by feedback control. The electric power to heating plates is increased stepwisely until burnout occurs. When burnout occurs and any of thermocouples attached on heating plates detects a steep rise of temperature beyond a preset value which is 400 ~ 450°C in most cases, the power to heating plates is instantaneously cut off in order to prevent failure of heating plates. Experimental conditions are shown in Table 2.

EXPERIMENTAL RESULTS

Periodic flow instabilities of large amplitude, which resulted in premature burnout, were observed in the early experiments. Flow rate and temperature of heating plates were oscillated with a period of 2 ~ 3 seconds. Since this was eliminated by reducing volume of the upper plenum, the flow instability was presumably thought to be a pressure-drop oscillation type.

All burnouts occurred near the lower end of the flow channel, as would be expected with uniform heating experiments. Heating plates locally deformed, contacted each other and failed averagely after 10 ~ 20 burnout experiments. DNB heat flux data are compared with Sudo correlations scheme for low, intermediate and high flow rate regions as shown in Figure 4 (a)~(c).

For low flow rate region, present DNB data and other data are compared with correlation(5) as shown in Figure 4(a). Experimental data are in reasonable agreement with predicted value, and it is clear that these DNB data are much lower than conventional pool-boiling DNB heat flux correlation by Zuber and Kutateladze as suggested by Mishima[1].

\[ q_{\text{DNB}}' = 0.16 \]  \hspace{1cm} (8)

For medium flow rate region, present experimental data are well agreed with correlation(6) as shown in Figure 4(b). Coolant outlet temperatures reached near saturation point in most cases, so that the thermohydraulic condition for correlation(6), in which exit quality is zero, was satisfied. DNB data in the range of \( G^* = 300 \sim 900 \) (Velocity = about 3 ~10 m/s) are currently lacking, but DNB heat flux is expected to increase linearly as flow rate increases in the intermediate flow rate range because DNB mechanism would remain basically the same. DNB experiments are now underway with test sections by reducing the width of heating plates to obtain DNB data in the range of \( G^* = 300 \sim 900 \).

For high flow rate region, downflow DNB data with a rectangular subchannel by Gambill[4] and Mirshak[6] were compared with correlation(7) because the present experiments could not cover the region due to limitation on electric power capacity for heating plates. These data are larger than predicted values by correlation(7) and this difference might be due to effect of subcooling, because correlation(7) is empirically determined without considering any effect of subcooling. Downflow DNB data in the high flow rate region are well correlated empirically by modifying correlation(7) with a factor of outlet subcooling.
\[ q_{\text{DNB}}^* = 0.005 ( G^* )^{0.611} ( 3.75 C_v \Delta T_{\text{SUB}} / h_f + 1 ) \]  \hspace{1cm} (9)

DNB data are in good agreement with the correlation(9) as shown in Figure 4(c). Predicted DNB heat fluxes by the correlation(9) are close to predicted values by conventional DNB heat flux correlations such as by Mirshak or Labuntsov.

From the above, Sudo correlations scheme would be selected for the JMTR safety analysis with the outlet subcooling factor in the high flow rate region. Figure 4(a)~(c) show that Sudo correlations scheme gives good predictions, allowing the error of 33% to the lower limit of the experimental data, so that minimum DNBR of 1.5 is considered to be conservative criteria for the occurrence of DNB.

Figure 4
Comparison of Measured DNB Heat Fluxes with Predicted by Correlations(5), (6) and (9)
CONCLUSION

DNB experiments with the thin rectangular channel, simulated the JMTR fuel subchannel, were carried out to consider applicable DNB heat flux correlations to the JMTR safety analysis for conversion to LEU fuel. Present experiments show that DNB heat fluxes in low and intermediate flow rate regions under moderate pressure are much lower than predicted value by conventional correlations, as suggested by Mishima and Sudo. Comparison of Sudo correlations scheme with present DNB data under pressurized conditions up to 13 kg/cm² abs and other data shows that the correlations scheme with the outlet subcooling factor for high flow rate region gives good predictions, and minimum DNBR of 1.5 is considered to be conservative criteria for the occurrence of DNB.

NOMENCLATURE

<table>
<thead>
<tr>
<th>Symbol</th>
<th>Description</th>
<th>Unit</th>
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<tbody>
<tr>
<td>A</td>
<td>Flow Area (m²)</td>
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</tr>
<tr>
<td>A_h</td>
<td>Heated Area (m²)</td>
<td></td>
</tr>
<tr>
<td>C_s</td>
<td>Specific Heat (kcal/kg)</td>
<td></td>
</tr>
<tr>
<td>D</td>
<td>Equivalent Hydraulic Diameter (m)</td>
<td></td>
</tr>
<tr>
<td>G</td>
<td>Mass Velocity (kg/m²s)</td>
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<tr>
<td>G*</td>
<td>Non-dimensional Mass Velocity</td>
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<tr>
<td>h_v</td>
<td>Latent Heat of Evaporation (kcal/kg)</td>
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</tr>
<tr>
<td>Δh_i</td>
<td>Inlet Subcooled Enthalpy (kcal/kg)</td>
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<tr>
<td>P</td>
<td>Pressure (kg/cm²abs.)</td>
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<td>P_h</td>
<td>Heated Perimeter (m)</td>
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<tr>
<td>q_DNB</td>
<td>DNB Heat Flux (kcal/m²s)</td>
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<tr>
<td>q_DNB*</td>
<td>Non-dimensional DNB Heat Flux</td>
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<tr>
<td>T_b</td>
<td>Bulk Temperature (°C)</td>
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</tr>
<tr>
<td>U</td>
<td>Velocity (m/s)</td>
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</tr>
<tr>
<td>W</td>
<td>Width of Channel (m)</td>
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</tr>
<tr>
<td>γ_v</td>
<td>Specific Weight of Vapor (kg/m³)</td>
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</tr>
<tr>
<td>γ_l</td>
<td>Specific Weight of Liquid (kg/m³)</td>
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<tr>
<td>λ</td>
<td>Characteristic Length (m)</td>
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<tr>
<td>σ</td>
<td>Surface Tension (kg/m)</td>
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ACKNOWLEDGEMENTS

The authors would like to express their hearty gratitudes to Professor K. Mishima of Kyoto University for his suggestions and discussions.

REFERENCES

UPDATE ON WORLD-WIDE USE OF TRIGA-LEU FUEL INCLUDING LOSS OF FLOW TESTS

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ABSTRACT

Higher uranium loaded TRIGA-LEU fuel was developed in the late 1970's and tested extensively in the GA TRIGA reactors and in the ORR as part of the RERTR program between 1979 and 1984.

The higher loaded TRIGA-LEU fuel has been in commercial use since 1980. Reload segments were inserted in several reactors, including the THOR reactor in Taiwan which was converted in a step-wise fashion from HEU plate-type fuel to TRIGA LEU 4-rod clusters.

A 3 MW TRIGA with forced flow cooling began operation in Bangladesh in 1986 with a full core loading of fuel with 20 wt-% U. The expected core life is 1000 MW days before the initial reload segment is required. A 3 MW TRIGA conversion reactor, the PRR-1, achieved criticality in the Philippines in March 1988. It was loaded with 4-rod clusters of fuel containing 20 wt-% U. As part of the start-up of the converted PRR-1, loss of flow tests were conducted to demonstrate the safe operation of the core when the coolant pump is shut down and no reactor scram is initiated.

The 14 MW TRIGA in Romania, in operation since 1979, has achieved a burnup of about 13,300 MW days on its initial core with HEU fuel and is awaiting shipment of reload fuel. Five reload clusters of TRIGA-LEU fuel with similar burnup capabilities, have been available for shipment for several years. This TRIGA-LEU fuel, with 45 wt-% U, has about 870 gm of U-235 when configured in a 16-rod cluster designed for use in MTR cores. This compares with a loading of about 140 gms of U-235 contained in presently offered silicide fuel elements.
Introduction

The development of higher uranium loaded TRIGA fuel with low enriched uranium (TRIGA-LEU fuel) started in 1978. It is characterized as higher loaded fuel because the standard, basic TRIGA fuel has always utilized low enriched uranium. It contains 8.5 wt-% U, with an enrichment of about 19.7%. The higher loaded fuels contain either 20, 30, or 45 wt-% U, each with 19.7% enrichment. Also, an important aspect is the fact that the presently manufactured highest loaded fuel with 45 wt-% U has a uranium volume fraction of only about 20%. This compares with the near-limiting value of about 50 vol-% uranium in the present LEU silicide fuels.

General Atomics has participated in the RERTR program since its inception and tested the above mentioned range of fuel loadings in the ORR to burnup values reaching about 65% averaged over the entire fuel rod. These tests were conducted between December 1979 and August 1984. A great deal of testing on the LEU fuel was also done in the TRIGA Mark F reactor in San Diego. A Safety Evaluation Report, NUREG-1282, on the TRIGA-LEU fuel was issued by the NRC in 1987. However, sales of the fuel began in 1979 and deliveries in 1980 when reload fuel with 20 wt-% U was delivered for reactors in Taiwan and Thailand.

Figure 1 presents a summary of TRIGA-LEU fuel usage to date.

Taiwan

The THOR - Tsing Hua Open Pool reactor achieved initial criticality in April 1962. It was fueled with High Enriched Uranium (HEU) plate-type fuel and subsequently operated at 1 MW with about 21 fuel elements surrounded by one row of graphite reflector elements. The system had a typical 7 x 9 grid plate, 4 control blades, a thermal column and eight radial beam ports. The forced flow cooling system operated at about 1500 GPM (340 M³/hr). The reactor is used mainly as a teaching tool for research in neutron physics, reactor engineering, chemistry, isotope production and training.

For the purposes of decreasing the fuel cycle costs and increasing the nuclear safety of the facility, the step-wise conversion to the use of TRIGA fuel (U-ZrH₁₆) with low enriched uranium was begun in 1977. Initially, five unshrouded clusters of TRIGA fuel, each containing four fuel rods with a 20-inch fueled length, as shown in Fig. 2, were introduced in the outer portion of the core. The facility continued to operate with a mixed core, with an increasing fraction of TRIGA fuel, as several more fuel cluster additions were made over the following several years. The first nine clusters contained U-ZrH₁₆ fuel with 8.5
wt-% U (19.7% enriched) and then the following clusters contained higher loaded fuel with 20 wt-% U (19.7% enriched). Each 4-rod unshrouded cluster of fuel with 20-wt% U contains about 450 gms of U 235. The decreasing fraction of plate fuel was generally kept in the central part of the core. A typical mixed core configuration is presented in Fig. 3.

By 1986, over 25 TRIGA fuel clusters (with over 100 fuel rods) had been acquired and in the summer of 1987 the THOR reactor began operating with a complete TRIGA core. The initial all-TRIGA core configuration had a central dry tube irradiation facility (approximately 2.5 inch [63.5 mm] I.D.) surrounded by 34 fuel rods with 8.5 wt-% U and two additional smaller irradiation facilities (approximately 1.25 inch [38.1 mm] I.D.), consisting of one dry tube and one water-filled tube. The outer 64 fuel rods contained 20 wt-% U. There was a single row of graphite reflector elements adjacent to three sides of the core and 5 movable dry tubes for irradiations (approximately 2.5 inch [63.5 mm] I.D.) were adjacent to the fourth side. The initial all-TRIGA core had an excess reactivity of about 3.5%, the maximum measured fuel temperature was about 270 degrees Celsius at 1 MW and the measured thermal neutron flux in the smaller dry tube near the center of the core was about $1.3 \times 10^{17} \text{ n/M}^2\text{-sec}$. The initial all TRIGA fuel core configuration is given in Fig. 4.

**Power upgrade** - During 1988, the decision was made to upgrade the reactor power to 3 MW. A contract was awarded to General Atomics in 1989 for the engineering and evaluation of the primary and the secondary cooling systems, and the supply of a new primary pump and plate-type heat exchanger. The basic principles for the upgrade were to retain the existing 8-inch (203 mm) primary piping, and that all other components (such as cooling towers, additional pumps and piping), and all installation would be provided locally and under the direction of the University. The 3 MW cooling system will have a primary loop flow rate of about 2500 GPM (570 M³/hr) and will not require any change from the present core configuration.

**Thailand**

The Thailand Research Reactor, TRR-1, was initially put into operation in October, 1962. It utilized plate-type fuel with fully enriched uranium and operated at power levels up to 1 MW.

In 1975, the TRR-1 was shut down in order to convert the high enriched uranium plate core to low enriched rod-type TRIGA fuel. At that time, the grid plate, control systems and safety systems were also replaced. The converted reactor has been operating since November 1977, mostly at steady-state power levels up to 1 MW, and occasionally at power levels up to the maximum rate power of 2 MW. Additionally, the reactor is designed for routine operation in both a pulsed mode and a
The initial loading of TRIGA fuel contained 8.5 wt-% U. In 1980, reload fuel with 20 wt-% U was introduced into the core to obtain an enhanced burnup lifeline from the core. A recent core configuration, showing 25 elements with 20 wt-% U in the D and E rings, is presented as Figure 5. A significant inventory of 20 wt-% fuel is available for additional reloading of the core.

The Philippines

The original Philippine Research Reactor (PRR-1) was commissioned as a 1 MW reactor in 1963 and initially used plate-type LEU fuel. Replacement plate fuel used HEU (93% enriched). The upgrade to 3 MW with TRIGA-LEU fuel gave this multipurpose facility an expanded steady-state operating range, added pulsing capability, and satisfied international requirements for proliferation resistant fuel.

The reactor upgrade started in January 1985 with extensive modifications to the cooling system. This included installation of a larger diameter piping external to the shield structure, a new primary pump, heat exchanger, and cooling towers. The upgraded cooling system has a design flow rate of 2500 GPM (vs the original 1700 GPM) and includes a new PNRI-supplied 20-channel instrumentation system. A new GA-supplied reactor instrumentation and control system was also installed. All of the modification and installation work was done by or under the direction of Philippine Nuclear Research Institute (PNRI) personnel.

Core Description - An isometric view of the shrouded core and core components is shown in Figure 6 and a core configuration for the TRIGA conversion PRR-1 is presented in Figure 7. The TRIGA-LEU fuel is contained in 3-inch square shrouded 4-rod clusters (Figure 8) which fit into the original grid plate. The long life Er-U-ZrH₁₆ fuel rods are shown in Figure 9 and the fuel specifications are given in Table 1.

Each 4-rod fuel cluster contains about 337 gms of U-235 and the operational core contains 30 fuel clusters with 115 fuel rods, 2 control rods, and 3 experimental locations. The four control blades from the original system were retained, but new drives were provided during the upgrade. Two new control rods were added to provide pulsing capability and automatic, servo control.

Criticality and Initial Operation - Initial core loading and criticality of the TRIGA conversion PRR-1 took place on March 11, 1988. Criticality was achieved in the loading step from 78 to 82 fuel rods. The core loading was increased to operational size and detailed measurements of the worth of each rod and blade were
performed by the period method utilizing a GA reactivity computer to rapidly deduce the worth of the segment being measured. Final calibration values are shown in Table 2.

Power calibrations were performed by utilizing a refined calorimetric procedure where the reactor operated in the neutral convection cooling mode and heated the known volume of water in the tank. Power was determined by measuring the rate of temperature rise of this well stirred, known water volume. The calorimetric method was used rather than the flow-temperature change method using forced-flow cooling because of the difficulty in instrumenting for, and obtaining, good average inlet and exit core coolant temperatures. Interpretation of the flow-change in temperature measurements is also complicated by the lack of complete isolation of primary coolant from the bulk pool water. There is some mixing of primary cooling and bulk pool water at the top of the core. Final power calibrations were performed at a reactor power of about 1.3 to 1.5 MW as this is about a limiting power value for natural convection operation in this configuration.

After about two hours of operation with forced-flow cooling at 3 MW, when the entire cooling system was approaching equilibrium conditions, power, as determined directly by the measured flow-change in temperature values was very close (within about 6%) to the calorimetric calibrated nuclear channel value. The initial flow-change in temperature power was about 80% of the nuclear channel value.

A short summary of 3 MW operating characteristics is given in Table 3.

As will be noted, TRIGA reactors have a large power coefficient when compared to research reactors with other types of fuel; in this case, requiring over 3 dollars in reactivity to achieve the full 3 MW operating power. This is a result of the large prompt negative temperature coefficient (approximately $1 \times 10^{-4}$ change in kelvin per degrees Celsius averaged between 20 degrees Celsius and 700 degrees Celsius) associated with the Er-U-ZrH$_x$ fuel. This large temperature coefficient results in a much more linear response of the reactor to reactivity changes as opposed to the more exponential response of most reactors. This characteristic contributes very greatly to the overall safety of the reactor, including the response to loss of forced flow cooling to be discussed later.

Operation with Natural Convection Cooling - The 4-rod cluster core can be cooled by natural convection cooling to power levels up to about 1.5 MW without any observable power fluctuations. Between 1.5 MW and 1.75 MW, depending upon the bulk water temperature, power fluctuations begin to appear. This "power chugging" is caused by the negative void coefficient

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acting through voids caused by local boiling in the region of minimum spacing between fuel rods at the high power locations in the core. "Power chugging" has been studied extensive in TRIGA cores. For a given average volume fraction of water in the core, its onset is influenced by the minimum dimension between fuel rods (power at onset is decreased for smaller spacing) and the bulk coolant temperature. The voids cause the power to decrease, and then as the voids disappear, the power level increases to its original value, where voids are again formed and the process repeats itself. The magnitude of the "power chugging" or negative power oscillations increases with increasing power levels and above a threshold value, which is about 1.5 MW in the PRR-1. Power "chugging" is a positive safety feature since it occurs prior to departure from nucleate boiling (DNB) which could cause very severe problems. "Power chugging" is such a strong, negative reactivity producing phenomenon that it may, in fact, prevent DNB from causing core damage.; its high visibility, seen through the reactor power instrumentation, gives the operator an unmistakable warning prior to conditions being reached where damage can occur. There is more very interesting work to be done in this area.

The present occupational limit on power level under natural convection cooling conditions has been set at 1.25 MW by PRNT. With natural convection cooling, the reactivity loss in going to 1.5 MW is $2.40 ($2.10 to 1.25 MW). The measured fuel temperatures are 350 degrees Celsius (T1) and 400 degrees Celsius (T2) at 1.5 MW (325 degrees Celsius and 365 degrees Celsius at 1.25 MW).

Pump Run Down Demonstration - As a safety test and demonstration at the conclusion of all of the start-up and full power demonstration tests, the effects of loss of forced flow cooling were demonstrated from two different power levels, 1 MW and 2.5 MW. The power level was established with the coolant pump on. The pump was then shut down, but the reactor remained in operation with no changes in any control rod or blade settings. After the pump was shut off, the reactor power and fuel temperature were monitored during the pump run down and rapid transition to natural convection cooling. The entire process took about 1.5 to 2 minutes for pump run down, flow reversal through the core from downward flow in forced cooling to upward flow in natural convection cooling, and establishment of a new, lower equilibrium power level.

The forced flow primary cooling system delivers water to the region just above the top of the core. Upon exciting the core at the bottom, the water enters a closed bottom plenum and then flows into the piping system and into a delay tank, pump and heat exchanger. When forced flow is stopped, a "flapper" in the wall of the outlet pipe, still within the reactor pool, falls open and allows a convective coolant flow path from the pool into the
bottom plenum and up through the core.

An identical test has been performed on another TRIGA reactor with forced down-flow cooling but no entrance for pool water into the bottom plenum under natural convection conditions. In this case also, the transition from forced down-flow cooling to natural convection up-flow cooling was very rapid and smooth. Fuel and water temperatures and reactor power re-equilibrated very smoothly as a natural convection coolant flow path was established where the cooler water flowed downward past the outermost fuel elements, reversed direction in the bottom plenum, and then flowed upward past the hotter, central fuel elements.

The ability of TRIGA reactors in this power range to respond safely and smoothly to a loss of forced flow cooling without a reactor scram is due primarily to the large, prompt negative temperature coefficient of reactivity, characteristic of U-ZrH$_{1.6}$ fuel. Significant contributions also come from the water temperature and density coefficients.

For the PRR-1 at 1 MW with forced cooling, the maximum measured fuel temperature was 300 degrees Celsius. When the pump was turned off, the temperature increased slightly to 310 degrees Celsius and then fell to 245 degrees Celsius as the reactor power decreased and equilibrated at a natural convection cooled power of about 0.45 MW, as shown in Figure 10. The average core water temperature increased significantly in the natural convection mode because the coolant flow rate is greatly decreased and the change in temperature over the length of the core is much greater under natural convection flow conditions (spanning the range from inlet temperatures of about 30 degrees Celsius to outlet temperatures of 100 degrees Celsius in many channels at higher powers). The change in water temperature and pressure produced negative reactivity effects which were balanced by the increase in reactivity resulting from a decrease in average core fuel temperature. Thus, the core operates with an equilibrated range of fuel and core water temperatures (and associated core water density) under forced flow conditions - where the change in temperature over the core length is relatively small (approximately 4.5 degrees Celsius with a flow rate of 2500 GPM). When the primary pump is shut off, the range of fuel and water temperatures (and associated core water density) smoothly re-equilibrates, producing a loss in reactivity and reactor power because of significantly increased core average coolant temperature and associated loss in density. The smoothness of the power transition is directly related to the large, prompt negative temperature coefficient.

At 2.5 MW with forced cooling, the maximum measured fuel temperature was 500 degrees celsius and this value did not increase after the pump was turned off, but rather, decreased smoothly to about 405 degrees Celsius as the power decreased to
about 1.7 MW with natural convection cooling. This is a power level where "power chugging" occurs under natural convection cooling and thus the 1.7 MW was not a stable power, but an average power. The power trace is shown in Figure 11. It is estimated that if the pump were shut off at a power level of 3 MW, and no scram occurred, the reactor power would decrease to a natural convection cooled average power of just over 2 MW with significant "power chugging" but with absolutely no damage to the fuel, core or any system component.

**Yugoslavia**

The TRIGA Mark II reactor in Yugoslavia went into initial operation in 1966. It is a steady-state reactor and operates at power levels up to 250 KW. Several different models of TRIGA fuel have been used in the core, including FLIP fuel with 70% enrichment. TRIGA-LEU fuel with 20 wt-% U was shipped to Yugoslavia in 1983.

**Malaysia**

The TRIGA Mark II reactor in Kuala Lumpur is a 1 MW steady-state reactor with pulsing capability. It first went into operation in 1982 with TRIGA fuel containing 8.5 wt-% U. Reload fuel with 20 wt-% U, to enhance the core burnout lifetime, was shipped to Malaysia in late 1982.

**Bangladesh**

The 3 MW TRIGA Mark II reactor in Dhaka achieved initial criticality in 1986. It has pulsing capability and is forced flow cooled at 3 MW. The initial core loading was with fuel containing 20 wt-% U giving a loading of 99 gms of U-235 per element. The calculated burnup to initial reload segment for this core is 1000 MW days.

**General Atomics**

The GA TRIGA Mark F reactor has been loaded with FLIP fuel (8.5 wt-%, 70% enriched U) for nearly 15 years. As reload fuel is needed, TRIGA-LEU fuel with 30 wt-% U is being used. Each of these elements contains 163 gms of U-235. The 30 wt-% fuel is being used because of the very high duty cycle of the reactor - 24 hours/day, 7 days a week at 1.5 MW - to support research on thermionic reactors. The reactor currently contains 14 LEU fuel elements with a 30 wt-% U loading.

**Romania**

The Romania 14 MW TRIGA began full power operation in late 1979. The initial core loading utilized fuel with 10 wt-% U, 93% enriched. This initial core was operated for over 8000 MW days.
before any core reconfiguration was performed. With the addition of only 3 spare clusters, the core achieved a burnup of about 13,300 MW days. The reactor has been shut down since September 1989 awaiting government agreements to ship reload fuel from the USA to Romania. Five clusters of TRIGA-LEU reload fuel with 45 wt-% U have been awaiting shipment for several years. The TRIGA-LEU fuel is designed to have a burnup life similar to that of the HEU fuel, and the LEU fuel was tested extensively in the ORR as part of the RERTR program.

It is of interest to note that 16-rod TRIGA-LEU fuel clusters, designed to fit in an MTR grid plate, contain about 870 gms of U-235. This compares to the 340 gms contained in presently offered silicide fuel elements.

CONCLUSION

TRIGA-LEU fuel with uranium loadings of 20, 30, and 45 weight-% has been in operation since 1978. Seven TRIGA reactors or reactor conversions are currently operating with partial or full core loadings of LEU fuel and five LEU clusters have been fabricated for the Romanian 14 MW SSR. To date, no fuel failures or anomalies have been experienced and the performance of the LEU fuel compares favorably with the HEU "FLIP" fuel which the LEU basically replaces.
<table>
<thead>
<tr>
<th>LOCATION</th>
<th>REACTOR (LEU START DATE)</th>
<th>WT%-ENRICHME</th>
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<tr>
<td>SAN DIEGO</td>
<td>MARK F (1978) TESTING</td>
<td>1/2-INCH 45-</td>
</tr>
<tr>
<td></td>
<td></td>
<td>1 1/2-INCH 20-</td>
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<td></td>
<td>MARK F (1989) RELOAD</td>
<td>45-30-</td>
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<tr>
<td>OAK RIDGE</td>
<td>ORR (1979) TESTING</td>
<td>20-30-45-</td>
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<td>TAIWAN</td>
<td>GE REACTOR MTR + TRIGA FUEL (1980) RELOAD</td>
<td>20-</td>
</tr>
<tr>
<td></td>
<td>(FULL CORE INVENTORY ON HAND)</td>
<td></td>
</tr>
<tr>
<td>THAILAND</td>
<td>C-W CONVERSION (1980) RELOAD</td>
<td>20-</td>
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<td>YUGOSLAVIA</td>
<td>TRIGA MARK II (1983) RELOAD</td>
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<td>MALAYSIA</td>
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<td>PHILIPPINES</td>
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<td>14 MW RELOAD FUEL AVAILABLE</td>
<td>45-</td>
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Figure 1. Operations with TRIGA LEU Fuel
UEL 

UZrH1.6

20 WT-% U (19.7% ENRICHED)

452 GM U235/4-ROD CLUSTER

ERBIUM BURNABLE POISON

FUEL ROD DIAM: 1.375 INCHES
34.9 mm

FUEL HEIGHT: 20 INCHES
508 mm

LAD

304 STAINLESS STEEL

CORE LIFE ~1000 MW DAYS

Figure 2. TRIGA 4-Rod Fuel Cluster
Figure 3. THOR Typical Mixed-Core Configuration
Figure 4. THOR Initial All-TRIGA Core Configuration
Figure 5. Thailand TRR-1 TRIGA-LEU Core Configuration with Mixed Core Loading
### TABLE 1
Summary of Fuel Specifications

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<th>NOMINAL VALUE</th>
<th>FUEL MODERATOR MATERIAL</th>
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<td>ENRICHMENT (U-235)</td>
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<td>ERBIUM CONTENT</td>
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<tr>
<td></td>
<td>DIAMETER</td>
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<tr>
<td></td>
<td>LENGTH</td>
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<td>U-235 LOADING/ROD</td>
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### TABLE 2
Measured Worth of Control Rods and Blades

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<th>Blade/Type</th>
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<td>BLADE 2</td>
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<tr>
<td>BLADE 4</td>
<td>$2.60</td>
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<tr>
<td>TRANS ROD</td>
<td>$2.13*</td>
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<tr>
<td>REG ROD</td>
<td>$2.82</td>
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</table>

*AFTER SHORTENING CONNECTING ROD BY 4.75 INCHES
FULL INSERTION WORTH WAS $2.61

$\beta_{eff} = 0.0071$ IS USED TO CONVERT WORTH FROM DOLLARS TO REACTIVITY UNITS
Figure 6. Isometric View of PRR-1 Core and Components in Grid Box
Figure 7. PRR-1 TRIGA LEU Operational Core 6.1% Excess Reactivity
Figure 3. Fuel Cluster Assembly
Figure 9. TRIGA Incoloy 800 Clad Fuel Rod with End Fittings
<table>
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<tr>
<th>Parameter</th>
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<td>$k_{EXCESS}$ (COLD)</td>
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<td>$\Delta k$ (0-3 MW)</td>
<td>$3.10$</td>
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<tr>
<td>$\Delta k$ (EQUILIBRIUM XENON)</td>
<td>$3.50$ (CALCULATED)</td>
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<tr>
<td>MEASURED FUEL TEMPERATURE</td>
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<tr>
<td>T1 (LOCATION 43)</td>
<td>$430^\circ$C</td>
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<tr>
<td>T2 (LOCATION 65)</td>
<td>$540^\circ$C</td>
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Figure 10. PRR-1 LOFC from 1 MW
Figure 11. PRR-1 LOFC from 2.5 MW
SOME NEUTRONICS AND THERMAL-HYDRAULICS CODES
FOR REACTOR ANALYSIS USING PERSONAL COMPUTERS

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ABSTRACT

Some neutronics and thermal-hydraulics codes formerly available only for main frame computers may now be run on personal computers. Brief descriptions of the codes are provided. Running times for some of the codes are compared for an assortment of personal and main frame computers. With some limitations in detail, personal computer versions of the codes can be used to solve many problems of interest in reactor analyses at very modest costs.

INTRODUCTION

It has been recognized that many reactor operators have limited computational capabilities in terms of available codes and access to main frame computers. The original collection of codes for main frame computers was distributed to various U.S. reactor operators and some foreign countries that have joint agreements with the RERTR program. These codes have also been used for several IAEA sponsored training courses; at first only on main frame computers, then both on a main frame and as a backup in personal computer (PC) versions at the Australian course, and finally exclusively as PC versions at the Argonne National Laboratory (ANL) course. The PC versions of the codes will be addressed in this paper. A brief history and description of the codes in this package are provided in the next section. Some timing comparisons are presented for various generations of PCs. Finally, the advantages and limitations of the codes are discussed.

DESCRIPTION OF THE CODES

This collection of codes consists of the LEOPARD/LINX/UM2DB package for neutronics applications, the codes NATCON and PLTEMP for steady-state thermal-hydraulics analysis, and the PARET code for both steady-state thermal-hydraulics and transient analysis. The LEOPARD/LINX/UM2DB package was adapted for research reactor applications by the University of Michigan for the analysis of
the first LEU demonstration core\textsuperscript{1} and upgraded under the RERTR program at ANL. The NATCON code was developed at ANL for the analysis of plate type reactors cooled by natural convection. The PLTEMP code was developed by Mishima\textsuperscript{2} while at ANL as part of the Japanese and ANL/RERTR program joint study, and later the code was upgraded at ANL. The PARET code was adapted for research reactor applications at ANL. The PC versions of these codes were generated at the Australian Nuclear Science and Technology Organization\textsuperscript{3} as a backup option for an IAEA training course. The codes have now been upgraded at ANL to match their current main frame counter parts.

The LEOPARD\textsuperscript{4} code provides burnup-dependent cross-section data in 2 and 4 energy groups based on ENDF/B-IV data. This non-dimensional lattice code supports a simple unit/super cell in either pin or plate geometry, and uses the MUFT/SUFCATE scheme to produce a spectrum for the collapse of the 172 group thermal/54 group epi-thermal library data to broad group data. The burnup dependence is provided by limited but adequate decay chains with fitted correlations for fission products that now have been modified for research reactor applications. The binary file of burnup-dependent cross-section data produced by LEOPARD interfaces with the LINX code.

The LINX code provides a link between the LEOPARD and the UM2DB and UM3DB codes by processing the binary burnup-dependent data produced by LEOPARD into a binary library in either 2 or 4 energy groups. This library can then be read by either UM2DB or UM3DB.

The UM2DB (UM3DB) code\textsuperscript{5} provides a 2 (3) dimensional finite difference solution of the neutron diffusion equations with depletion and shuffling of the fuel with burnup. The code reads burnup-dependent data generated by the LINX code from a binary file, while non-depleting cross-section data are input as card image data. The finite difference solution in the code is based on center of mesh differencing, but the code has now been modified to provide edge of mesh extrapolation for power peaking estimates. The code now also includes modifications to support internal boundary conditions for control rod modeling, etc. UM3DB has not been implemented for the PC at ANL.

The NATCON code\textsuperscript{6} provides a capability for the analysis of the thermal-hydraulics of plate-type research reactors cooled by natural convection. The code includes steady-state estimates of what are often considered safety margins. These estimates may include power peaking factors, hot channel factor components, and laminar flow entrance effects. The code now includes only light water properties, but heavy water properties could easily be substituted.

The PLTEMP code\textsuperscript{2} was developed for the thermal-hydraulic analyses of the Japanese KUHFR and KUR reactors. The code can model the entire active core including bypass flow, a single
element, or now, after modifications at ANL, a single plate with it adjacent flow channels. The code provides steady-state estimates for departure from nucleate boiling (DNB) based on a selection of correlations. The code allows the application of separate hot channel components, radial power peaking factors, and an axial peaking factor. The ANL version of the code now includes an option for imposing an axial distribution on each channel and providing estimates for the heat flux and temperatures for selected channels. The code is well suited to providing steady-state estimates for safety margins in plate-type research reactors with forced flow cooling. The ANL version of the code now shares the use of the PARET code properties libraries.

The PARET code provides a steady-state capability for thermal-hydraulic analysis, and a coupled point kinetics, hydynamics, and heat transfer capability for transient analysis. The code was originally written for the analyses of the SPERT III series of reactivity insertion transients which included pin-type geometry and pressures more typical of power reactors. The ANL version of the code now contains correlations more appropriate for research reactors, and the properties library now has revised data to cover a range of pressures more typical of research reactors. Properties libraries may be generated for either light water or heavy water applications. Results from the code have been compared to the experimental results from the SPERT I transients and the heavy water SPERT II transients with a reasonably high level of agreement. In addition to the analysis of reactivity insertion transients, the code is capable of analyzing loss of flow transients including flow reversal and the establishment of natural convection.

The ANL version of these codes have been compiled with a Microsoft Fortran compiler (version 4.1) with a math co-processor under DOS 3.2. It was assumed that the PC would have 640 Kbytes of memory, but the dimensions on UM2DB were reduced to allow the executable code to fit on a 360 K floppy disk for transportability. The other codes have fixed dimensions and fall below this 360 K limit.

**PC AND MAIN FRAME COMPARISONS**

Computing times have been determined for some of the codes in this collection for an assortment of PCs and main frame computers. These should serve as a reference for other computers. The main frame computers in this comparison are a VAX 8700, IBM 3033, and IBM 3084, and the PCs included are IBM-XT, IBM-AT, and IBM PS/2 model 70. Some of the PC characteristics are provided in Table I. These IBM PC models were chosen as representative but not the fastest available. Later models of the IBM-AT have clock speeds of 8 MHz, and the IBM PS/2 is available in 20 and 25 MHz models. Most IBM compatibles are driven at clock speeds higher than those quoted in Table I. A
comparison with an IBM-AT compatible with a clock speed of 8 MHz is provided in a later table.

Table I. IBM Personal Computer Characteristics

<table>
<thead>
<tr>
<th>Type</th>
<th>CPU/Co-processor</th>
<th>Clock Speed, MHz</th>
<th>Word Length, bytes</th>
</tr>
</thead>
<tbody>
<tr>
<td>XT</td>
<td>8088/8087</td>
<td>4.77</td>
<td>8</td>
</tr>
<tr>
<td>AT</td>
<td>80286/80287</td>
<td>6.0</td>
<td>16</td>
</tr>
<tr>
<td>PS/2</td>
<td>80386/80387</td>
<td>16.0</td>
<td>32</td>
</tr>
</tbody>
</table>

Since the codes LINX, NATCON, and PLTEMP do not have timing routines included, computing times for these codes will not be included in this comparison. It suffices to say that running times for these codes are relatively short even on the slowest PC. Table II provides a comparison of CPU times for the various PCs and codes. The IBM-AT is about a factor of two times faster than the IBM-XT due largely to word length differences. The IBM PS/2 is about 3.5 to over 4.6 times faster than the IBM-AT due to differences in both word length and clock speed. The CPU times quoted in Table II for both LEOPARD and UM2DB are per burn step. The full LEOPARD run with 8 burn steps ran 9.25 minutes on the IBM-AT. The same case on the XT stopped with a run time error after four burn steps. The PARET loss of flow case also failed on the XT with a run time error.

Table II. PC CPU Timing Comparison

<table>
<thead>
<tr>
<th>Code Case</th>
<th>CPU Running Time, minutes</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>IBM-XT</td>
</tr>
<tr>
<td>LEOPARD\textsuperscript{a}</td>
<td>2.40</td>
</tr>
<tr>
<td>UM2DB\textsuperscript{a}</td>
<td></td>
</tr>
<tr>
<td>Test Problem</td>
<td>4.93</td>
</tr>
<tr>
<td>Benchmark</td>
<td>2.89</td>
</tr>
<tr>
<td>PARET Benchmark $1.5$</td>
<td>23.05</td>
</tr>
<tr>
<td>Loss of Flow</td>
<td>---</td>
</tr>
</tbody>
</table>

\textsuperscript{a} CPU time per burn step

It may be of some interest to compare the PC run times to typical run times on main frame computers. Table III provides a comparison of three main frame computers with the IBM-AT running.
times for the PARET cases in Table II. The VAX 8700 runs about 38 times faster than the IBM-AT. The IBM 3033 runs about 1.2 times faster than the VAX. The IBM 3084 ran about 1.6 times faster than the 3033 machine and more than 77 times faster than the IBM-AT.

Table III. CPU Running Times for Main Frame Computers and PC

<table>
<thead>
<tr>
<th>Code Case</th>
<th>VAX 8700</th>
<th>IBM 3033</th>
<th>IBM 3084</th>
<th>IBM-AT</th>
</tr>
</thead>
<tbody>
<tr>
<td>Benchmark $1.5</td>
<td>0.38</td>
<td>0.31</td>
<td>0.19</td>
<td>14.08</td>
</tr>
<tr>
<td>Loss of Flow</td>
<td>4.61</td>
<td>3.66</td>
<td>2.27</td>
<td>176.5</td>
</tr>
</tbody>
</table>

Typically IBM compatibles run as fast or more often faster than the true IBM PC. Table IV provides a comparison of an IBM-AT at 6 MHz and an IBM compatible running at 8 MHz. The CPU running time for the compatible is just a little better than the 8/6 difference in clock speeds.

Table IV. Comparison of IBM-AT and IBM Compatible

<table>
<thead>
<tr>
<th>Code Case</th>
<th>CPU Running Time, minutes</th>
</tr>
</thead>
<tbody>
<tr>
<td>PARET Benchmark $1.5</td>
<td>IBM-AT at 6 MHz</td>
</tr>
<tr>
<td>Benchmark $1.5</td>
<td>14.08</td>
</tr>
<tr>
<td>Loss of Flow</td>
<td>176.5</td>
</tr>
</tbody>
</table>

It is also important that the codes give reasonably accurate answers to the problem assigned. The test problem for UM2DB is the same problem referenced in the 2DB User's Manual, and the eigenvalue that one should expect to compute is given as 0.986701. The IBM PCs each give the same eigenvalue for each iteration and converge to a value of 0.9867171. The VAX 8700 converges to a value of 0.9867166, the IBM main frames each give a value of 0.9867009. The convergence criterion on the eigenvalue in each case was 0.000001. While the agreement is quit good in each case, the IBM PCs agree better with the VAX main frame than the IBM main frame values, but the IBM main frame values agree best with the predicted value in the manual. The other PC results agree well with the main frame results, and there appears to be no significant loss of accuracy in using the PC version of any of the codes.
PC CODE LIMITATIONS

In addition to the obvious slower running times on a PC, some of the codes have further limitations. Some of the limitations may be machine dependent, some may be code dependent, and some may be dependent on the compiler used to generate the executable code (including the math co-processor). The long running time for some problems may be a real limitation.

One limitation common to all of the codes is the form of the output from the codes. The output file from each code is still set up for printing on a line printer not for display on the screen of a PC. Each line of output may extend over the full 132 columns allowed by most line printer plus a carriage control character in some cases. Most PC displays are limited to 80 characters per line, and some editors or word processors may either truncate longer lines or display the line in wrap around mode. A simple file lister is available that displays about 80 columns without wrap around and allows the user to shift the viewing window to the right (and back left) to view the remaining columns of data. This file lister also allows the user to search on selected key words or phrases in the output. Most of the codes in this package display some useful information on the screen as the code is executing on the PC.

The current versions of the PC codes are machine limited to the extent that the codes were each compiled under Microsoft Fortran 4.1 which in turn requires a level of DOS of 3.0 (or higher) and with a math co-processor installed. The codes will not run without a co-processor. The codes were all run on machines with at least 640 kbytes of memory, but this may not be a limitation. The IBM-XT failed to successfully execute for two of the problems in this mix of cases. These same problems executed successfully on the other PCs. It is not at all clear what caused these failures. These failures could even be due to limitations of the compiler used to generate the code. The Australians highly recommended the Lahey F77L compiler, and judging by the amount of modification that was required to successfully compile their source code with the Microsoft compiler this compiler is at least much more forgiving.

The UM2DB code has at least two limitations. The first is common also to the main frame version of the code. With a depletion problem where burnup-dependent cross-section data are read from a binary file created by LINX, UM2DB is unable to properly read these data when four energy group data are requested. The code runs successfully with two energy group data, and when data is entered as card images in the input. It is not clear at this time whether the problem is with the LINX code or with the UM2DB code. Most of the use of the code has been with two group data. The second limitation is a limit in the amount of storage available with the dimensions fixed in the code. As noted earlier, the dimension of the container array was set not at its maximum value but such that the executable code
would fit on a low density floppy disk. The container array dimension is now set at 20,000 words and the code occupies almost 347 Kbytes. Thus, with 640 Kbytes of RAM the size of the code could be increased by about 293 Kbytes (less the storage occupied by the resident part of the operating system). The current container array dimension can be increased significantly. Since the memory requirements are different for each problem, it is difficult to define the maximum problem size that can now be run, but perhaps the IAEA benchmark core will give some perspective on size. The benchmark problem with 2 energy groups, 19 X 16 mesh intervals, and 81 zones requires almost 11,000 words of the 20,000 available. This problem assumes symmetry, and only a quarter of the core is modeled. The container array dimension can be increased to match the available storage in RAM, but the complexity of the problem that can be run on a PC is still clearly limited.

The limitations of the LEOPARD code are more fundamental than related to the PC. As noted earlier the geometries that can be represented are limited to simple pin or plate unit cells (fuel, clad, and coolant) with an extraneous region to give a supercell model. The code is also limited to a choice of 2 or 4 energy groups total with only one thermal group, and scattering is limited to from one group to the next adjacent group. The number of materials in the working library is fixed at 35 materials, and Boron-10 is the only control rod material in the library. Although the library contains heavy water as one of the materials, this material should not be used as the coolant in the unit cell model, and its use even as a reflector material is questionable. New materials can be added to the library, but only with great difficulty. The code may be adequate for simple excess reactivity and fuel cycle scoping type computations, but even the four group data with only one resonance group and one thermal group is inadequate for computing reactivity feedback coefficients. Other available codes, such as the WIMS code, should be considered as alternatives.

The limitation of the PARET code on a PC is one more of patience and the stability of the PC and its electrical supply. The loss of flow transient in Table II, for example, ran for almost three hours on an IBM-AT with over 4200 time steps. Some slow ramp reactivity insertion cases have run almost ten times this number of time steps (>30,000). The code could run for more than a day to complete some problems. The most recent version of the code now has a restart capability where a restart file is written after a selected number of time step alternately to two separate units (see ref. 9), and problems can easily be stopped and restarted to run in selected segments of time.
CONCLUSIONS

Some very useful analyses can be done with personal computers where main frame computers are unavailable or too expensive to use. The running times of the same problems on various PCs and some main frame computers were compared. The codes run slower on PCs, and some problems can run for hours on a PC. Where both PCs and main frame computers are available, the user may wish to weigh the loss of the use of the PC against the higher cost of running on the main frame. The relative running times of some of the codes on PCs suggest that a true IBM-XT may be too slow to be practical for some of the cases, but since many XT's are upgraded with accelerator boards or 80286 chips, this may not be of concern. Of greater concern is the failure of the XT to successfully execute two of the problems of interest. The other PCs had no difficulty executing these same problems. This difficulty may be unique to the XT used in this study. The cause of these run time errors was not determined. The reliability of the PC and its source of electricity must be high for the longer running problems.

The accuracy of the answers obtained on the PCs appears to be as good as that obtained with the main frame computers for the same problem. The quality of the answers from the PC with the UM2DB code may be limited by the detail that can be included within the available storage on the PC and by the quality of the cross-section data provided. The quality of the cross-sections provided by the LEOPARD code may not be acceptable for some applications. The LEOPARD code is very limited in the number of materials available in its library, in the geometries allowed, and in the number energy groups and the choices of energy group structure for the broad group library.

The LEOPARD/LINX/UM2DB package can provide some very useful results for some neutronics applications but has some limitations as noted above. The NATCON and PLTEMP codes are very useful for steady-state thermal-hydraulic analysis for both natural convection and forced flow cooling modes. The PARET code can provide both a steady-state thermal-hydraulic capability and a coupled point kinetics, hydrodynamics, and heat transfer capability for transient analysis.

ACKNOWLEDGMENTS

The contributions of Dr. Brian E. Clancy and his associates at the Australian Nuclear Science and Technology Organization in providing the initial PC versions of Fortran source code are gratefully acknowledged.

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SESSION VI

September 26, 1990

CORE CONVERSION STUDIES

Chairmen:

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(Risø, Denmark)

F. DiMeglio
(RINSC, USA)
STATUS REPORT ON THE CORE CONVERSION OF THE OSIRIS REACTOR

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ABSTRACT

Conversion studies - from UO$_2$ 7.5% enriched caramel fuel to U$_3$Si$_2$ 19.75% enriched fuel - of the 70 MWth OSIRIS Reactor were started in 1987. In 1988, the Technical Specifications for the manufacturing of precursor, standard and control elements were written. In 1989, the Thermal-Hydraulic Studies were started and the first fuel element mockups were hydraulically tested. 2 standard elements will be irradiated for 16 to 18 months, starting from June 1990. Since the beginning of this year, several safety related studies have been launched, like Fission Product Release, Criticality study (container and fuel storage shelter), Reactivity Accident. Conversion should be completed in 1993.

Introduction

OSIRIS is a 70 MW pool type reactor which has been in operation since 1966. Up to 1979, the fuel element was made up of 24 fuel plates, U-Al 93% enriched.

In 1975, CEA launched a study program on the reduction of the enrichment of its research reactors fuel element, which resulted in 1979 in the conversion of the OSIRIS reactor to the UO$_2$ 7.5% enriched, 17 plates, caramel fuel element.

Once the qualification of the 19.75% enriched U$_3$Si$_2$-Al fuel element at 4.8 g/cm$^3$ was considered to be established in 1987, detailed technical and economic studies were undertaken, which resulted in the decision of conversion of OSIRIS, as well as SILOE (35 MWth reactor, located at GRENOBLE). Two standard precursor elements U$_3$Si$_2$-Al, 22 plates per elements, have been delivered. Their irradiation in OSIRIS began in June 1990.
THE OSIRIS REACTOR CORE ASSEMBLY

The 70 MW pool-type OSIRIS Reactor Core assembly consists essentially of 4 sections which confine and route the water coolant through 3 identical pumps working in parallel.

- The base which feeds water (at $3.5 \times 10^5$ Pa) to the bottom of the core (upwards flow) and also supports grids placed around the core for the various experimental devices.

- The core tank, containing the core lattice structure, which is machined in a single block of metal and provides 56 channels for standard fuel elements, control fuel elements (of the fuel-follower type) and in-core experiments.

- The water outlet casing and its lateral coolant outlet pipe.

- The chimney whose function is to limit the mixing between the water stream from the core and the pool water.

All the elements inserted in the core are locked in position at both ends.

The drive mechanisms of the control elements are located under the pool.

LEU $\text{U}_3\text{Si}_2\text{Al}$ CORE DESIGN

Safety Requirements:

The standard and control elements belong to the class of Elements of Importance for Safety (E. I. S.). They play a role in the overall safety, through their specific functions:

- cladding integrity
- water coolant channel integrity
- locking and antiejection system
- Heat transfer
- control and shut down of the reactor
Operating Requirements:

- Operating cycle: from 3 to 4 weeks. At each cycle, a rearrangement of the core takes place. Anticipated burnup of 55% leads to an irradiation rate of around 95 000 MWD/tU

- water speed (between fuel plates) around 8.5 m/s
- overall flow in the core: 5 500 m$^3$/h
- Pressure drop in the core: around $1.5 \times 10^5$ Pa.
- Flux at the experimental locations (in the horizontal plane of symmetry, at 70 MW):

  Thermal neutrons
  (0.025 ev) $1.8 \text{ to } 2 \times 10^{14} \frac{n}{cm^2/sec}$

  Fast neutrons
  (> 1 Mev) $2 \times 10^{14} \frac{n}{cm^2/sec}$

Fuel Element Characteristics:

- fuel density U total: 4.8 g/cm$^3$
- Enrichment: 19.75%
- Number of fuel plates: 22
- U total weight per plate: 105.5 g
- $^{235}U$ weight per plate: 20.9 g
- Overall plate thickness: 1.27 mm
- Aluminum alloy cladding thickness: 0.38 mm
- Fissile width: 68.4 mm
- Fissile height: 630 mm
- Percentage of fuel, in volume, in the meat: 43%
- Water channel width: 2.46 mm
- Burnable poison (boron): 0.4 g$^{10B}$/element
Preliminary Results:

By comparison with the corresponding UO₂ performances, the
euronic studies have shown so far the following results for a
continuous passage from UO₂ fuel to U₃Si₂ fuel:

- Decrease in the quantity of U₂³⁵ present in the core
- Slight increase in the U₂³⁵ consumption per MWd
- Increase in the number of fuel elements consumed per
cycle
- Necessity of using a burnable poison, to be located in
the side plates

The conversion scheme is based on temporary unloading of
reusable standard caramel elements, rearrangement of remaining
standard caramel elements, definitive unloading of used caramel
elements and complementing with U₃Si₂ Al standard fresh elements.

This progressive conversion is possible in about one year of
operation, even without resorting to additional fresh caramel fuel
elements.

THERMAL HYDRAULIC STUDIES

The elements have been designed in such a way that the
conversion be as easy as possible, thanks to the thermal hydraulic
similarity between the UO₂ caramel, UA1 and U₃Si₂Al elements in the
core.

The thermal hydraulic studies have taken into consideration
the following steps:

38  UO₂    Standard elements  6 UO₂ Control elements
26  UO₂    Standard elements  6 UO₂ Control elements
12  U₃Si₂  Standard elements
12  UO₂    Standard elements  6 UO₂ Control elements
26  U₃Si₂  Standard elements
38  U₃Si₂  Standard elements  6 UO₂ Control elements
38  U₃Si₂  Standard elements  6 U₃Si₂ Control elements
Flow redistribution is looked at in both natural convection and forced convection conditions.

The computer model of the fuel element is validated by in loop tests of the element, in cold (20 degree) and hot (45 degree) conditions.

In addition, the following configurations are analyzed:
- loss of cooling without pump wheel blocking
- loss of cooling with pump wheel blocking (1 out of 3)
- loss of cooling through the break of the collector

SAFETY RELATED STUDIES

The following studies have been initiated:

. Reactivity Insertion Accident (Borax accident)

The consequences of a fast insertion of reactivity (e. g. 2000 p. c. m. within 0.1 sec) are analyzed.

The energy released during the accident is of the order of 135 Mj.

The most representative items to analyze are the following:

- Local heat flux, Risk of drying of the cladding
- Power oscillations during cooling of the fuel
- $U_3Si_2$ Al - water reaction

. Fusion of a fuel plate underwater

Which may be a consequence of a channel blockage, reactivity ramp, etc.

. Fuel handling accident

Such as leading to the fusion in air of a fuel element.

For these accidents, the radiological consequences of the release of fission products will be determined for both the radiation workers and the public.

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Other studies are concerned with:

- Criticality safety of the U\textsubscript{3}Si\textsubscript{2} Al fuel storage facility under dry and wet conditions.

- Safety analysis of the fuel transport cask

CONCLUSIONS

The effective test phase of U\textsubscript{3}Si\textsubscript{2} Al fuel has already been started with the irradiation of two prototype standard (as opposed to control) elements.

In parallel, detailed safety studies are carried out, in support of the request to the Safety Authorities, for authorization to proceed with the above outlined conversion program.
ABSTRACT

The 2 MW Rhode Island Atomic Energy Commission reactor is required to convert from the use of High Enriched Uranium (HEU) fuel to the use of Low Enriched Uranium (LEU) fuel using a standard LEU fuel plate which is thinner and contains more U-235 than the current HEU plate. These differences, coupled with a desire to upgrade the characteristics and capability of the reactor, have resulted in core design studies and thermal hydraulic studies not only at the current 2 MW but also at the maximum power level of the reactor, 5 MW. In addition, during 25 years of operation, it has become clear that the main uses of the reactor have been neutron scattering and neutron activation analysis. The requirement to convert to LEU presents an opportunity to optimize the core for the utilization and to restudy the thermal hydraulics using modern techniques. This paper presents a status report on the conversion.

INTRODUCTION

The Rhode Island Atomic Energy Commission operates an open pool, MTR type research reactor in Narragansett, Rhode Island. While the reactor has a maximum design power level of 5 MW, current, licensed operation is at a power level of 2 MW.
The reactor was designed by General Electric in the late 1950's with construction beginning in late 1962. The reactor went critical in 1964, to a power level of 1 MW in 1965 and to a power level of 2 MW in 1968.

Before presenting this status report on the conversion of the reactor to LEU, it will be useful to describe those aspects of the utilization, duty cycle, and original design of the facility which have influenced the approach taken for conversion. A detailed description of the reactor was presented at the 1987 RERTR meeting in Buenos Aires and only a synopsis will be presented here/1/.

**REACTOR DESCRIPTION, UTILIZATION AND DUTY CYCLE**

The reactor is a typical swimming pool research reactor utilized primarily for neutron scattering at three beam ports and research programs which require neutron activation analysis as an analytical tool—including small sample analysis—utilizing five irradiation facilities. At one beam port, the University of Rhode Island has installed the only polarized neutron, small angle scattering spectrometer currently operating in the United States.

The normal, equilibrium operating HEU core consists of 30 fuel elements each containing 18 plates and a U-235 content of 124 grams when new. These elements sit on a grid plate in a grid box with permanently installed shrouds in which the boral control rods or blades move. The reactor has been reflected by graphite and the grid contains sufficient spaces for a boral regulating rod and several irradiation baskets. This arrangement is shown in Figure 1.

Note that the four boral control blades move in permanently fixed shrouds and these shrouds cannot easily be relocated. The boral regulation rod is also fixed in the reflector region of the 30 element core but its relocation is possible. For clarity, some of the grid positions are shown vacant. During operation, however, each grid position must contain a fuel element, a reflector piece, an irradiation basket or a plug. Otherwise the coolant flow will by-pass the core through the vacant grid position.
Figure 1: Core Arrangement
Figure 2 is a schematic representation of the HEU core showing 30 fuel elements surrounded by graphite reflectors and a row of irradiation baskets. The control blades are labeled 1, 2, 3 and 4 and the regulating rod is in position D1. This figure also shows the location of the beam tubes and the terminals of the pneumatic systems. Note that the large beam tubes and the pneumatic irradiation systems terminate outside the grid box at row 5, in the center of the grid box. The neutron flux in this position is representative of the flux available to the beam ports. Also note that there is a radiation basket in position D9. Positions A5 and D9 have been used for flux comparisons between HEU and LEU cores.

Another consideration important to the LEU conversion is the decision by the Department of Energy to produce a standard fuel plate for use in all university plate type reactors for which they provide fuel cycle assistance. This fuel plate is slightly thinner than the current plate and contains about 80% more U-235.

The existing core may be characterized as large with a very low power density resulting in a low thermal flux per unit power. It utilizes lightly loaded fuel elements to make the core large enough to encompass the control blades. Even with extraordinary techniques, the maximum burn-up achievable is about 14% and this burn-up is only possible because we are a one shift operation and do not have to contend with equilibrium xenon.

The reactor has operated for 25 years with an 8 hour on - 16 hour off, 5 day per week duty cycle. There are no plans to change this duty cycle. This duty cycle allows for operation with an excess reactivity less than that required for continuous operation. Because of the control blade configuration, this duty cycle also requires special start-up considerations when converting to a compact LEU core.
Figure 2: Typical 30 Element HEU Core
CONVERSION OBJECTIVES

There are six basic objectives of the LEU conversion program. These are:

1. Convert the reactor to the use of LEU without requiring the fabrication of additional HEU fuel.

2. Design a LEU core and an operating scheme to achieve burn-ups greater than the current 14%. This is especially important for anticipated higher power operation.

3. Design a LEU core which will optimize the neutron flux in the beam tubes and will allow for future improvements.

4. Design a reactor core with a flux trap for small sample neutron activation analysis.

5. Design a reactor core which can be operated at power levels up to 5 MW with the appropriate primary coolant flow.

6. Design a LEU core whose initial cost will be about the same as the cost of 30 HEU fuel elements since that is the amount allocated for the core by the Department of Energy.

During extensive scoping studies many core configurations were examined/1/. Incorporating all of the information gathered during these scoping studies and remembering our six objectives, a preferred core design has emerged. The general neutronic and thermal hydraulic characteristics of this LEU core have been presented at the 1989 RERTR meeting in Berlin/2/. This paper will discuss the techniques which will be utilized in achieving an equilibrium core, scheduling, and future improvements.
Figure 3 presents the start-up version of this preferred core which consists of 14 fuel elements. The elements now contain 22 standard plates with a total of 275 grams of U-235 per element. A central beryllium piece with a 38mm hole is incorporated as a flux trap. The regulating rod has been changed to stainless steel and moved one grid position so as to be adjacent to this smaller core.

The core is graphite and beryllium reflected, with an excess reactivity of about 3 \% Δk/k, a regulating rod worth of .44\%, a shutdown margin with blade 3 stuck out of 6.4\% and a total power peaking factor of 2.64. This design allows the use of the existing 24 graphite reflector pieces and requires the acquisition of 14 fuel elements and 16 beryllium reflector pieces for the equilibrium core. This meets one of the conversion objectives since the cost of a beryllium reflector piece is less than the cost of an LEU fuel element.

Because of the one shift operation, the xenon behavior is cyclical and this core can be operated as long as it is possible to operate on Friday morning. Using computer simulation, this core has been "run down" until a Friday morning start-up is no longer possible. The reactivity balance is shown in Table 1.

Table 1. Reactivity Balances on the Friday Morning of the Last Week of Operation for the Startup and Transition Cores.

<table>
<thead>
<tr>
<th></th>
<th>Startup % Δk/k</th>
<th>Core 2 % Δk/k</th>
<th>Core 3 % Δk/k</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fresh Cold Clean,</td>
<td>3.0</td>
<td>5.1</td>
<td>6.9</td>
</tr>
<tr>
<td>Reactivity Losses,</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Burnup</td>
<td>0.3</td>
<td>1.8</td>
<td>3.2</td>
</tr>
<tr>
<td>Xe</td>
<td>1.5</td>
<td>1.5</td>
<td>1.5</td>
</tr>
<tr>
<td>Sm</td>
<td>0.6</td>
<td>0.7</td>
<td>0.7</td>
</tr>
<tr>
<td>Long-Lived F.P.</td>
<td>0.1</td>
<td>0.6</td>
<td>1.0</td>
</tr>
<tr>
<td>Cold-Hot Swing</td>
<td>0.3</td>
<td>0.3</td>
<td>0.3</td>
</tr>
<tr>
<td>Control</td>
<td>0.2</td>
<td>0.2</td>
<td>0.2</td>
</tr>
<tr>
<td></td>
<td>3.0</td>
<td>5.1</td>
<td>6.9</td>
</tr>
</tbody>
</table>
Fig. 3. Startup, Transition, and Equilibrium Cores
(Lifetimes Based on Operation for 8 Hr/D, 5 D/Wk)

STARTUP CORE
Excess Reactivity = 3.0 % $\Delta k/k$
Core Lifetime: ~ 14 Wks (~ 560 Full Power Hours)

CORE 2
Excess Reactivity = 4.1 % $\Delta k/k$
Core Lifetime: ~ 70 Wks (~ 2800 Full Power Hours)

CORE 3
Excess Reactivity = 3.7 % $\Delta k/k$
Core Lifetime: ~ 60 Wks (~ 2400 Full Power Hours)

EQUILIBRIUM CORE
Core Lifetime: ~ 57 Wks (~ 2300 Full Power Hours)
The reactivity requirements for Xe, Sm, long lived fission products, control, and the cold-hot swing is approximately 3% which will allow for approximately 14 weeks of operation before it will not be possible to start up on Friday morning.

After this initial operation, ten beryllium and ten graphite reflector pieces will be reconfigured to provide additional reactivity. Figure 3 also presents this second core showing the fuel remaining in each fuel element after the initial 14 weeks of operation. The reactivity balance is shown in Table 1 and it allows for an additional 70 weeks of operation.

Following this second phase of operation, the graphite and beryllium reflectors will again be reconfigured. This third core is shown in Figure 3 which also shows the fuel in each element at the start of this phase. Table 1 again presents the reactivity balance which now allows for an additional 60 weeks of operation.

Note that the core is now almost completely beryllium reflected. The core has operated for about 3 years and refueling is now required.

Refueling consists of removing the four elements with the most burn-up, placing four fresh elements in the core corner positions, and placing the remaining used fuel elements in the remaining positions with those elements containing the least fuel nearest the center of the core. This process provides the flatest flux and greatest neutron leakage. Eventually an equilibrium core will be reached.

Figure 3 presents this eventual equilibrium core where the four elements with the most burn-up have been discharged and four fresh elements have been added to the edge of the core. The average discharge burn-up for this equilibrium core is about 21%, which is 50% more burn-up than in the current HEU core.

The cyclical behavior of the xenon receives considerable attention when a reactor of significant power is operated regularly for a single shift. If, however, the reactivity requirements for Friday morning start-up are examined, the samarium and long-lived fission products as well as the xenon are important. Figure 4 presents the individual Friday morning reactivity losses as a function of time.
Figure 4. Friday Morning Reactivity Losses

Note that while the xenon loss is the dominant factor for relatively fresh fuel, it is always about 1.5%. In the long term, burnup provides the greatest reactivity loss, followed by xenon, long-lived fission products, and samarium. The behavior shown is for the time period before refueling.

The thermal flux available to the beam tubes has previously been examined for the cores presented above/2/. This examination has shown that while the total of the 7 group fluxes are only somewhat increased over the thirty element HEU core, the sum of the three thermal group fluxes shows a 40% increase in the LEU cores. In addition, it has been shown that as these changes in graphite and beryllium reflectors are made, the thermal fluxes at the beam tubes do not substantially change/2/.

THERMAL HYDRAULIC STUDIES

The thermal hydraulic characteristics of these cores have been studied and previously reported/2/. These studies have shown that operation at 2 MW with these LEU cores is acceptable with the current primary coolant flow of 386 M$^3$/hr. These studies have also shown that operation at 3 and 5 MW will require a primary flow rate greater than previously expected.
DESIGN BASIS ACCIDENT

The design basis accident for this reactor has been a loss of coolant accident with the water draining through a beam port containing no plugs. Recall that the core sits in a grid box and draining of this box is through a 1.25 cm hole drilled in the bottom. Because of this, about 35 minutes is required to complete the draining, after which, the bottom 18 cm of fuel remains in water. It has been possible to show that the low power density HEU core will not melt after this hypothetical loss of coolant accident.

The LEU core has a higher power density than the HEU core. Using the same accident sequence and calculations which were used for the HEU core, it is not possible to conclude that the LEU core will not suffer some melting following a loss of coolant accident. Studies are in progress to refine the design basis accident sequence and calculations. In addition, the design of an emergency core cooling system is proceeding. Such a system is simplified because of the grid box.

FUTURE IMPROVEMENTS

All work on LEU conversion has proceeded so as not to preclude future improvements after LEU conversion has been achieved. These improvements include operation at the maximum design power level of 5 MW and further modifications in the core region to enhance the beam port fluxes.

Because of the grid box design, the beam tubes cannot easily be extended to the LEU compact core to take full advantage of the flux increase. In the future, it will be possible to discard the graphite reflectors and relocate the existing beryllium reflectors from in front of the active beam tubes. In front of the active beam tubes, new beryllium reflectors which contain an air void in the center will be installed. This will be equivalent to moving the end of the beam tube to a position closer to the core resulting in an increased flux available to experimenters. Eventually, this process will also lead to a fully beryllium reflected core.
**SCHEDULE**

The fuel element re-design, and the beryllium reflector and flux trap design are completed. Design of an emergency core cooling system is underway. The safety analysis report is scheduled for submission to the Nuclear Regulatory Commission by January, 1991.

Because of funding limitations, fuel fabrication is scheduled for 1992 allowing almost one year for NRC review. If the schedule is maintained, the conversion should be completed by 1993. Fabrication of additional HEU fuel is not required.

**CONCLUSION**

In conclusion, the redesign of the Rhode Island Atomic Energy Commission research reactor is nearing completion and the preparation of the safety analysis report for conversion to LEU is progressing. The redesign will not only accomplish conversion but will also improve the reactor characteristics for the utilization.

**REFERENCES**


Georgia Tech Research Reactor Conversion to Low Enrichment—Calculational Capabilities Assessment

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Georgia Institute of Technology
Atlanta, Georgia 30332

ABSTRACT

The Georgia Tech Research Reactor (GTRR) is required to convert to low enrichment in accordance with 10 CFR 50.64(c). The U.S. Department of Energy funded in July 1989 a program to analyze the differences and similarities in performance of the high and the low enrichment fuels. The goal of the program is to amend the SAR and the technical specifications of the GTRR so that low enrichment \(\text{U}_3\text{Si}_2\)-Al dispersion plate type fuels can replace the current fuel in use.

Since the GTRR is a heavy water reactor, and because experimental data on heavy water reactors are limited, it was decided to test how well the codes recommended by the RERTR program would predict the GTRR operating parameters with the HEU fuels; that is, to compare the results of calculations with measurements. The purpose of the initial work was to demonstrate the capability to calculate the various reactor parameters related to performance and safety. Thus, the initial effort served as a built-in benchmarking process for validating the computational methods. It also served to produce biasing factors to allow better prediction of the LEU fueled reactor.

Since Georgia Tech was responsible for the defense of changes in the Safety Analysis Report and the Technical Specifications, it was necessary that the Georgia Tech team become involved in the computational effort which was closely coordinated with the RERTR Group at ANL.

In this paper computational results using various codes are reported and where possible compared to measurements. Final concluding remarks and recommendations are made.

Computational Results

Initial attempts to calculate the existing GTRR configuration with the LEOPARD and UM2DB codes have shown that these codes have difficulty with heavy water systems.

Specifically, calculations on a 2-D R-Z simulation of the actual reactor showed a severe discrepancy between the \(k\) values obtained from a two-group and four-group values of \(k\). The modelled loading, having 17 elements, has a measured \(k\) of 1.09, which falls below the two-group and four-group results.)

Table 1 shows the results obtained. When the full reactor is modelled with heavy water, there is a discrepancy of almost 30%
between the two-group and four-group values of $k$. When the model is simplified to include only the bare core, the discrepancy is reduced to 12% with heavy water, but only 0.5% with light water. When the outside boundary conditions on the bare core are changed from vacuum to zero current, thus modelling an infinite core, the discrepancy practically disappears for both types of water. We found later that the UM2DB code had a bug—sums of cross sections for mixed material were not being initialized correctly.

Calculations were done at Argonne National Laboratory with the EPRICELL and DIF3D codes, using 2, 4 or 7 groups in heavy water only. Results are shown in Table 2. Calculations were also done with these codes for a detailed 3-D XYZ model of the 17-element GTRR core. The cross sections were established with five EPRICELL calculations—-one for the fueled portion of the fuel element, one for the sideplates of the fuel element, one for the axial extension of the fuel element, one for the rest of the moderator, and one for the graphite. The DIF3D calculation used a detailed geometry in which all of these different regions were specifically delineated. Results are also shown in Table 2.

In all these cases, there is a notable discrepancy between the 7-group results and the results with 2 or 4 groups. The discrepancy is less than that obtained from LEOPARD and UM2DB and is of the opposite sign. Furthermore, the discrepancy is much less for the infinite core.

Further calculations were done to explore the performance of LEOPARD and UM2DB. Calculations were done on a 9-element core whose available excess reactivity was measured to be less than 1%.

Total $k_{eff}$ was calculated as 0.887 in 4 groups and 1.320 in 2 groups. The void coefficient was calculated by assuming a 10% lower heavy water density in the space between fuel plates. For 4 groups, this gave a $k_{e_{\text{ff}}}$ of 0.861, equivalent to a reactivity of $-1.91 \times 10^{-3} \Delta k/k$ per cm$^3$. Two groups gave a $k_{e_{\text{ff}}}$ of 1.313, and a reactivity of $-3.44 \times 10^{-3} \Delta k/k$ per cm$^3$. These are to be compared with a measured value of $-2.6 \times 10^{-3} \Delta k/k$ per cm$^3$ reported in the SAR as an average over an 11-element core.

In an attempt to get better results, a separate LEOPARD calculation was done to get reflector cross sections, analogous to the way EPRICELL was used at Argonne. This calculation employed a pitch equal to twice the heavy water reflector thickness. This made little difference. It gave $k_{e_{\text{ff}}}$ of 0.885 for 4 groups and 1.266 for 2 groups.

This configuration was used to calculate Doppler coefficient by changing the LEOPARD resonance temperature from 20°C to 520°C. This gave a Doppler temperature coefficient of $-1.35 \times 10^{-3} \Delta k/k/°C$ for 4 groups and $-2.95 \times 10^{-3} \Delta k/k/°C$ for 2 groups.
LEOPARD gives the choice of MND or Wigner-Wilkins thermal spectra. All of the above calculations were done with MND. A change to Wigner-Wilkins gave $k_{\text{eff}}$ of 0.857 for 4 groups and 1.232 for 2 groups.

Meanwhile at Argonne, J. Matos and Ken Freese were able to resolve the discrepancies in their EPRICELL-DIF3D calculations between different numbers of groups by doing a separate EPRICELL calculation for the heavy water surrounding the fuel elements. A comparison between some of the cross sections they obtained from EPRICELL and corresponding values obtained from Leopard is shown in Tables III, IV and V. Note that the EPRICELL transport cross section of D2O for group 4 in the reflector is 16% greater than the value obtained by LEOPARD. Also, the fast group in the two group comparison shows that LEOPARD gives higher transport cross section for D2O. Further evaluations regarding the difference in $k$ values between the two- and 4- groups are planned.

Ken Freese at Argonne was able to use VIM to do a detailed Monte-Carlo calculation of the 9-element core, obtaining a $k_{\text{eff}}$ of 1.005 +/- 0.005, which is in excellent agreement with the experiment.

An earlier Argonne DIF3D calculation for the 17-element core was replicated by putting Argonne's EPRICELL-derived cross sections into UM3DB. The $k_{\text{eff}}$ obtained was 1.0484, in excellent agreement with the DIF3D value of 1.0482. However, DIF3D ran this problem in just 1.6 minutes on Argonne's Cray. The speed ratio between that machine and our Cyber 990 is 3.48, so the same problem should have taken 5.7 minutes on the 990. However, UM3DB took 70.7 minutes to run that problem. UM3DB is obviously an old code not endowed with the best numerical methods for fast convergence.

The MCNP Monte-Carlo code, version 3B, has been put into full operation. To begin with, a $k$-infinity calculation has been done with the unit cell. In the calculation, the geometry of the fuel meat, cladding and side plates was accurately represented. The results obtained were 1.8852 +/- 0.0031 for high-enrichment fuel (HEU) and 1.7829 +/- 0.0038 for low-enrichment fuel (LEU). These are to be compared with values reported previously from LEOPARD-UM2DB calculations for HEU of 1.6015 for 4 groups and 1.5998 for 2 groups.

Calculations were done using MCNP on 9, 14, 17 and 19-element core configurations. The results are shown in Table VI. The LEU calculations assume 200 grams of U-235 per fuel element. The calculations were done using the repeated-structures capability of MCNP-3B. This feature apparently has bugs, because some of the problems failed to run, throwing the computer into an endless loop.
In particular, the 19-element void calculations and all of the 17-element calculations failed to run. These difficulties have been reported to the MCNP group at Los Alamos.

Mr. John S. Hendricks of the MCNP Group at Los Alamos acknowledged in a letter that we have uncovered a bug in MCNP. He said that normally the code corrects the overflow of the geometry description array, but in some cores with repeated structures or lattices, there is a catastrophic failure—the code may choose the cell in the wrong universe level and causes particles to get lost. This problem has now been fixed and MCNP is running with no known problems.

Table VI compares MCNP results obtained for three different cores—9-, 14-, and 19-element cores. The measured $k_{eff}$ value for a nine-element core is $1.00 \pm 0.01$. It is seen that about $3\%$ delta $k/k$ is the overprediction of the $k$ value by the MCNP code.

The void coefficient for the 9- and 14-element cores are (in % delta $k/k/cc$):

- 9-element HEU: $(3.63 +/- 1.30) \times 10^{-4}$
- 9-element LEU: $(3.25 +/- 0.87) \times 10^{-4}$
- 14-element HEU: $(3.71 +/- 0.65) \times 10^{-4}$
- 14-element LEU: $(2.08 +/- 0.75) \times 10^{-4}$

The values for the void coefficient compare with a measured value of $3.4 \times 10^{-4}\%$ delta $k/k/cc$ at the core center and an average value of $2.6 \times 10^{-4}\%$ delta $k/k/cc$ for the whole core with HEU fuel.

Some calculations on the 9-element core were made with LEOPARD and UM2DB (after UM2DB was debugged). The results are shown in Table VII. Here, the "cold" cases had all temperatures at 20 degrees centigrade and the "hot" cases had the temperature at 520°C.

The deviation of the calculations using LEOPARD and UM2DB from the actual $k$ for this core is quite large, being round $17\%$. The reason for that discrepancy is not known at present. However, the difference between 2-group and 4-group calculations has been reduced to $1/2\%$. The Doppler coefficients, shown in Table VIII also agree well between 2 and 4 groups.

Currently, it is believed that MCNP is perhaps the best code for $k$ calculations. However, temperature and Doppler coefficient calculations cannot be made with MCNP. It is also expensive to run. Consequently, it was decided to use the EPRICELL to generate cross sections at different temperatures and use these cross
sections in UM2DB to obtain the temperature and Doppler coefficients.

Efforts to get Epricell running on the Georgia Tech computers have encountered major obstacles. An attempt to convert the code to the Cyber 990 ran into the problem of Fortran compatibility. Epricell was programmed in an older version of Fortran. It references arrays as subprogram arguments in a way that is not acceptable to the Fortran-77 used on the Cyber. (That is, Fortran-77 doesn't like to see an array argument in a subprogram being referenced to one element of a larger array in the calling program.) This construction is used extensively throughout Epricell, so a large amount of rewriting would be needed.

Attempts to run Epricell under MVS on the IBM 4381 were equally unsuccessful. The load module provided by Argonne would not run because it referenced some library routines of unknown nature, probably available only at Argonne. An attempt to recompile the program indicated that the compiler currently in use is just as restrictive as the one on the Cyber.

However, it is still believed that Epricell could be gotten to run, either on the Cyber or on a large PC workstation, if the reprogramming effort were made. The IBM MVS is a cumbersome system, and it is not felt to be worthwhile to use that system if it would involve just as much reconversion as using the other two.

The more than 3% delta k/d difference between HEU and LEU cores is of concern. In order for LEU cores to give comparable reactivity performance as HEU cores, it appears that 22-plate fuel elements would be needed. Figure 1, supplied by Dr. James Matos of Argonne National Laboratory, shows excess reactivity vs. $^{235}\text{U}$ per element. An element containing 22 plates (275 grams $^{235}\text{U}$) would have about 3.5% delta k/k more than an 16-plate element with 200g $^{235}\text{U}$.

It is obvious that at this late stage of reactor calculations, the ability to predict core parameters accurately is not there! We strongly recommend that the LEOPARD code be scrapped.

We also recommend that a temperature dependent group cross set generation capability be added to either the MCNP or the VIM codes. This would simplify the process considerably.
Table I

K Calculations of the GTRR using LEOPARD and UM2DB

<table>
<thead>
<tr>
<th>Configuration</th>
<th>Water</th>
<th>k(2-grp)</th>
<th>k(4-grp)</th>
<th>Discrepancy</th>
</tr>
</thead>
<tbody>
<tr>
<td>Full reactor</td>
<td>Heavy</td>
<td>1.425685</td>
<td>1.101724</td>
<td>29.4%</td>
</tr>
<tr>
<td>Full reactor</td>
<td>Light</td>
<td>0.934967</td>
<td>0.910043</td>
<td>2.7%</td>
</tr>
<tr>
<td>Bare core</td>
<td>Heavy</td>
<td>0.527530</td>
<td>0.468167</td>
<td>12.7%</td>
</tr>
<tr>
<td>Bare core</td>
<td>Light</td>
<td>0.813175</td>
<td>0.809247</td>
<td>0.5%</td>
</tr>
<tr>
<td>Infinite core</td>
<td>Heavy</td>
<td>1.599848</td>
<td>1.601473</td>
<td>-0.1%</td>
</tr>
</tbody>
</table>

* \([k(2\text{-grp}) - k(4\text{-grp})]/k(4\text{-grp})\)
### Table II

**K Calculations at ANL using EPRICELL and DIF3D**

<table>
<thead>
<tr>
<th>Configuration</th>
<th>k(7 gps.)</th>
<th>k(4 gps.)</th>
<th>k(2 gps.)</th>
<th>Discrepancies</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>(7-4)</td>
</tr>
<tr>
<td>RZ model</td>
<td>1.068456</td>
<td>0.972927</td>
<td>0.971253</td>
<td>9.8%</td>
</tr>
<tr>
<td>Infinite core</td>
<td>1.780948</td>
<td>1.747660</td>
<td>1.752178</td>
<td>1.9% -0.3%</td>
</tr>
<tr>
<td>Detailed XYZ model</td>
<td>1.153757</td>
<td>1.057196</td>
<td>1.042042</td>
<td>9.1% 1.5%</td>
</tr>
</tbody>
</table>
Table III

Comparison of Some 2-Group Uranium Cross Sections

<table>
<thead>
<tr>
<th>Group</th>
<th>U-235 Fission</th>
<th></th>
<th>U-235 Capture</th>
<th></th>
<th>U-238 Capture</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>LEOPARD</td>
<td>EPRICELL</td>
<td>LEOPARD</td>
<td>EPRICELL</td>
<td>LEOPARD</td>
<td>EPRICELL</td>
</tr>
<tr>
<td>Thermal</td>
<td>384.0156</td>
<td>357.277</td>
<td>450.3732</td>
<td>419.217</td>
<td>1.875627</td>
<td>1.7597</td>
</tr>
<tr>
<td></td>
<td>558.1779 (MND)</td>
<td>654.6304 (MND)</td>
<td>2.726278 (MND)</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
Table IV

Comparison of Some 4-Group Uranium Cross Sections

<table>
<thead>
<tr>
<th>Group</th>
<th>U-235 Fission</th>
<th>U-235 Capture</th>
<th>U-238 Capture</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>LEOPARD</td>
<td>EPRICELL</td>
<td>LEOPARD</td>
</tr>
<tr>
<td>1</td>
<td>1.2389</td>
<td>1.2471</td>
<td>1.3099</td>
</tr>
<tr>
<td>2</td>
<td>1.8592</td>
<td>1.9373</td>
<td>2.4395</td>
</tr>
<tr>
<td>Thermal</td>
<td>384.0156</td>
<td>357.277</td>
<td>450.3732</td>
</tr>
</tbody>
</table>
## Table V

Comparison of Deuterium Transport Cross Sections

<table>
<thead>
<tr>
<th>Group</th>
<th>LEOPARD</th>
<th>EPRICELL:</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 of 4</td>
<td>1.8322</td>
<td>1.7956</td>
</tr>
<tr>
<td>Fuel Region</td>
<td>1.7110</td>
<td>Reflector</td>
</tr>
<tr>
<td>2 of 4</td>
<td>2.3296</td>
<td>2.4057</td>
</tr>
<tr>
<td>3 of 4</td>
<td>2.2959</td>
<td>2.2633</td>
</tr>
<tr>
<td>1 of 2</td>
<td>2.2425</td>
<td>2.2416</td>
</tr>
<tr>
<td>Thermal</td>
<td>3.7401</td>
<td>3.6267</td>
</tr>
<tr>
<td>3.7108 (MND)</td>
<td>4.3278</td>
<td></td>
</tr>
</tbody>
</table>
### Table VI

**RESULTS OF MCNP K CALCULATIONS**

<table>
<thead>
<tr>
<th>No. of Elements</th>
<th>9</th>
<th>14</th>
<th>19</th>
</tr>
</thead>
<tbody>
<tr>
<td>HEU</td>
<td>1.0348±0.0156</td>
<td>1.1557±0.0092</td>
<td>1.2126±0.0072</td>
</tr>
<tr>
<td>LEU</td>
<td>1.0006±0.0112</td>
<td>1.1160±0.0100</td>
<td>1.1786±0.0090</td>
</tr>
<tr>
<td>HEU, fuel elements voided</td>
<td>0.9761±0.0139</td>
<td>1.0625±0.0134</td>
<td></td>
</tr>
<tr>
<td>LEU, fuel elements voided</td>
<td>0.9524±0.0084</td>
<td>1.0638±0.0160</td>
<td></td>
</tr>
<tr>
<td>HEU, center element removed</td>
<td>1.0088±0.0101</td>
<td></td>
<td></td>
</tr>
<tr>
<td>LEU, center element removed</td>
<td>0.9944±0.0100</td>
<td></td>
<td></td>
</tr>
<tr>
<td>HEU, top reflector voided</td>
<td>0.8982±0.0161</td>
<td></td>
<td></td>
</tr>
<tr>
<td>LEU, top reflector voided</td>
<td>0.9877±0.0096</td>
<td></td>
<td></td>
</tr>
<tr>
<td>HEU, graphite reflector voided</td>
<td>1.0019±0.0135</td>
<td></td>
<td></td>
</tr>
<tr>
<td>LEU, graphite reflector voided</td>
<td>0.9993±0.0135</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
Table VII

VALUES OF K FOR LEOPARD-UM2DB

CALCULATIONS ON 9 ELEMENTS

<table>
<thead>
<tr>
<th>Case</th>
<th>2-group</th>
<th>4-group</th>
<th>Difference (4-2)</th>
</tr>
</thead>
<tbody>
<tr>
<td>HEU Cold</td>
<td>1.172021</td>
<td>1.176737</td>
<td>0.004665</td>
</tr>
<tr>
<td>HEU Hot</td>
<td>1.171649</td>
<td>1.176315</td>
<td>0.004666</td>
</tr>
<tr>
<td>LEU Cold</td>
<td>1.149347</td>
<td>1.154292</td>
<td>0.004945</td>
</tr>
<tr>
<td>LEU Hot</td>
<td>1.146479</td>
<td>1.151507</td>
<td>0.005028</td>
</tr>
</tbody>
</table>
Table VIII

DOPPLER COEFFICIENTS, %(DELTA K/K)

PER DEGREE C

<table>
<thead>
<tr>
<th>Case</th>
<th>2-group</th>
<th>4-group</th>
</tr>
</thead>
<tbody>
<tr>
<td>HEU</td>
<td>-7.22E-5</td>
<td>-7.17D-5</td>
</tr>
<tr>
<td>LEU</td>
<td>-5.00E-4</td>
<td>-4.84E-4</td>
</tr>
</tbody>
</table>
GTRR 9 Element Coil

LEU: DIF3D Calculations

Fig. 1 Excess Reactivity versus amount of U$^{235}$ per fuel element
NORMAL OPERATION (5 MW): Case 6
NUCLEAR POWER EMERGENCY SIMULATION

NORMAL OPERATION (5 MW): Case 6

Diagram showing temperature of clad over time.
NUCLEAR POWER EMERGENCY SIMULATION

NORMAL OPERATION (5 MW): Case 6

TEMP. OF COOLANT

TIME (SEC)
NUCLEAR POWER EMERGENCY SIMULATION

NORMAL OPERATION (5 MW): Case 6
NORMAL OPERATION (5 MW): Case 6
NUCLEAR POWER EMERGENCY SIMULATION

NORMAL OPERATION (5 MW): Case 6

![Graph showing power output over time for normal operation.](image-url)
Nuclear Power Emergency Simulation

2.5% Reactivity (5 MW): Case 5

TEMPERATURE OF CLAD (°C)

TIME (SEC)
Nuclear Power Emergency Simulation

2.5% Reactivity (5 MW): Case 5
Nuclear Power Emergency Simulation

2.5% Reactivity (5 MW): Case 5

TIME (SEC)

MFLOW (OUT)
Nuclear Power Emergency Simulation

2.5% Reactivity (5 MW): Case 5
Nuclear Power Emergency Simulation

2.5% Reactivity (5 MW): Case 5

POWER

TIME (SEC)
Nuclear Power Emergency Simulation

2.5% Reactivity (5 MW): Case 5
Nuclear Power Emergency Simulation

Pump Loss (5 MW): Case 4

TEMP OF FUEL vs TIME (SEC)
Nuclear Power Emergency Simulation

Pump Loss (5 MW): Case 4
Nuclear Power Emergency Simulation

Pump Loss (5 MW): Case 4

TIME (SEC)

TEMP. OF COOLANT

0 0.5 1 1.5 2

0 20 40 60 80 100 120
Nuclear Power Emergency Simulation

Pump Loss (5 MW): Case 4

MFLOW (OUT)

TIME (SEC)
Nuclear Power Emergency Simulation

Pump Loss (5 MW): Case 4
Nuclear Power Emergency Simulation

Pump Loss (5 MW): Case 4

![Graph showing the decay of power over time for a pump loss scenario. The x-axis represents time in seconds, ranging from 0 to 2, and the y-axis represents power in MW, ranging from 0 to 6. The graph shows a linear decrease in power as time progresses.]
Nuclear Power Emergency Simulation

Pump Loss (5 MW): Case 3

![Graph showing power loss over time](image)
Nuclear Power Emergency Simulation

Pump Loss (5 MW): Case 3

TEMPERATURE OF FUEL (°C)

TIME (SEC)
Nuclear Power Emergency Simulation

Pump Loss (5 MW): Case 3

TIME (SEC)

TEMPERATURE OF CLAD (°C)
Nuclear Power Emergency Simulation

Pump Loss (5 MW): Case 3

TEMPERATURE OF COOLANT (°C)

TIME (SEC)
Nuclear Power Emergency Simulation

Pump Loss (5 MW): Case 3

MFLOW (OUT)

TIME (SEC)
Nuclear Power Emergency Simulation

Pump Loss (5 MW): Case 3

![Graph showing reactivity over time for Pump Loss (5 MW): Case 3]
Nuclear Power Emergency Simulation

Normal Operation (5 MW): Case 2
Nuclear Power Emergency Simulation

Normal Operation (5 MW): Case 2

TIME (SEC)

TEMPERATURE OF CLAD (°C)
Nuclear Power Emergency Simulation

Normal Operation (5 MW): Case 2

![Graph showing temperature of coolant (°C) over time (sec).]
Nuclear Power Emergency Simulation

Normal Operation (5 MW): Case 2

MFLOW (OUT)

TIME (SEC)
Nuclear Power Emergency Simulation

Normal Operation (5 MW): Case 2

TIME (SEC)

REACTIVITY
Nuclear Power Emergency Simulation

Normal Operation (5 MW): Case 2

![Graph showing normal operation at 5 MW for Case 2. The graph plots power (MW) against time (sec) with a steady line at 5 MW for the duration of the graph (4 seconds).]
NORMAL OPERATION (5 MW): Case 1
NUCLEAR POWER EMERGENCY SIMULATION

NORMAL OPERATION (5 MW): Case 1

TEMP. OF CLAD

TIME (SEC)
NUCLEAR POWER EMERGENCY SIMULATION

NORMAL OPERATION (5 MW): Case 1

TIME (SEC)

TEMP OF COOLANT

0 10 20 30 40

0 1 2 3 4
HEU-LEU CONVERSION OF FRJ-2 DIDO
AT KFA JÜLICH

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ABSTRACT

The studies and preparations to convert the FRJ-2 DIDO reactor at KFA Jülich from HEU to LEU were being carried out in a joint effort with industry (Interatom, Nukem) in the framework of the German AF-Program and started in the early eighties. It turned out very soon that the fuel element design has to be modified from the present HEU-EB element (electron beam welded fuel tubes) to the LEU-RS element (roll swaged fuel tubes). The basic data for LEU elements and cores and the essential requirements for the conversion process were determined mid of the eighties. The conversion procedure comprising several steps was agreed upon by the licensing authority nearly at the same time. Reactorphysics und thermohydraulic calculations proved that the RS design is superior to the EB design and will provide greater safety margins as well in HEU as in LEU operation. Due to significant delays in performing the first steps of the conversion procedure and because of the necessity to repeat test irradiations with the new RS type elements in HEU and LEU manufactured by CERCA it is expected that the HEU-LEU transition phase will not begin before the 3rd quarter of 1995. Since FRJ-2 at that time will have reached a considerably high age it could not be excluded that it is closed rather than converted to LEU operation mid of the nineties.

INTRODUCTION

The FRJ-2 operated at KFA Jülich for nearly 30 years is a DIDO-type heavy water moderated, reflected and cooled reactor with a closed tank housing the core, the safety and control absorbers and several beam tubes.
The reactor is run at a nominal power of 23 MW. The core is composed of 25 tubular fuel elements comprising four inner tubes containing the fuel and one outer Al-tube containing boron as burnable poison.

Two elements with different U-235 loadings of 150 g and 170 g and an enrichment of 80% U-235 are used. The studies and preparations to convert FRJ-2 from HEU to LEU were being carried out in a joint effort of Interatom, KFA Jülich and Nukem in the framework of the German AF-Program and started in the early eighties.

Since the high-density LEU fuels require 'harder' cladding materials than HEU fuel plates aluminium alloys such as AlMg or AlMgSi must be employed rather than Al-99.5 as used to date. When using the current electron beam welding technique (EB-design) to form fuel tubes from the pre-curved fuel plates the vaporization of the alloy constituents (Mg, Si) results in extraordinarily high failure rates so that a 'new' element design was proposed by Nukem and tested by dummy element fabrication. According to the new design the pre-curved fuel plates are swaged into 3 plates provided with special grooves then forming the final fuel tube (roll swaged element; RS-design). Figure 1 illustrates the differences between the EB- and RS-element design.

![FRJ-2 fuel element (schematic)](image)

**Figure 1:** Comparison of HEU and LEU fuel elements for FRJ-2 DIDO
All essential data for LEU-elements and cores were determined in close cooperation between KFA and INTERATOM and presented at the International RERTR Meeting at Petten in 1985 /1/. In Table 1 a comparison of these data for HEU and LEU elements/cores is given.

Table 1: Comparison of essential data for HEU and LEU elements/cores of FRJ-2 DIDO

<table>
<thead>
<tr>
<th></th>
<th>HEU</th>
<th>LEU</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of fuel elements</td>
<td>25</td>
<td>25</td>
</tr>
<tr>
<td>element design</td>
<td>EW¹</td>
<td>RS²</td>
</tr>
<tr>
<td>fuel</td>
<td>UAlₓ-Al</td>
<td>U₃Si₂-Al</td>
</tr>
<tr>
<td>U-235 enrichment (%)</td>
<td>80</td>
<td>19.75</td>
</tr>
<tr>
<td>uranium density (g/cm³)</td>
<td>0.5/0.6</td>
<td>2.8/3.5</td>
</tr>
<tr>
<td>U-235 content/element (g)</td>
<td>150/170</td>
<td>180/225</td>
</tr>
<tr>
<td>length of HEU-LEU</td>
<td>./.</td>
<td>(min.) 70</td>
</tr>
<tr>
<td>transion phase (FPD)³</td>
<td>45-50</td>
<td>45-50</td>
</tr>
<tr>
<td>av. U-235 burnup (%)</td>
<td>23</td>
<td>23</td>
</tr>
<tr>
<td>av. length of operation</td>
<td>23</td>
<td>23</td>
</tr>
<tr>
<td>cycles (FPD)³</td>
<td>./.</td>
<td>5-10</td>
</tr>
<tr>
<td>reduction of neutron flux</td>
<td>./.</td>
<td>10-15</td>
</tr>
<tr>
<td>(LEU/HEU)</td>
<td>./.</td>
<td>5-10</td>
</tr>
<tr>
<td>- core area fast (%)</td>
<td>./.</td>
<td>5-10</td>
</tr>
<tr>
<td>- core area thermal (%)</td>
<td>./.</td>
<td>10-15</td>
</tr>
<tr>
<td>- reflector area thermal (%)</td>
<td>./.</td>
<td>5-10</td>
</tr>
<tr>
<td>reactivity loss per FPD (% ΔK/K)</td>
<td>0.21</td>
<td>0.17</td>
</tr>
</tbody>
</table>

¹ EW = Electron welded fuel tubes
² RS = Roll swaged elements
³ FPD = Full power days

Since today FRJ-2 mainly is used for beam experiments in the field of solid state and nuclear physics the loss of the reflector thermal flux is of particular importance. This loss has been determined to be well below 10% in LEU standard cores compared to the current HEU version.

Figure 2. shows the radial distribution of the thermal flux in the core and reflector region.
Figure 2: Radial distribution of the thermal flux in the core and reflector region (C-row)

STEPWISE PROGRAM TO CONVERT FRJ-2

At a very early stage planning of the core conversion from the use of HEU to LEU was discussed in detail with the licensing authority and German TÜV. As the result several steps to convert FRJ-2 were defined and agreed upon.

These steps are in detail:

I. Test irradiation of 3 RS elements with HEU (170 g U-235/el.).

II. Test irradiation of 3 RS elements with LEU (U₃Si₂-Al; 200 g U-235/el.).

III. Conversion of the whole core to RS elements in HEU (U-235 content 150 g and 170 g per element as at present).

IV. Operation with HEU RS cores for a period of 6 to 9 months.

V. HEU-LEU transition phase.

VI. Completion of the HEU-LEU conversion and reactor operation with LEU-
RS cores.

The first two steps each had to be started with one test element and subsequent operation of the remaining after the first one had achieved its final burn-up and had been inspected successfully. Apart from the main objective to confirm the reliable and safe behavior of the new element design in HEU and LEU both steps also serve the important goal to qualify the manufacturer in RS element and LEU fuel fabrication. By steps III. und IV. operational experience with HEU RS cores should be achieved mainly with respect to the thermalhydraulic behavior of RS cores. Steps V. and VI. then will complete the HEU-LEU conversion of FRJ-2.

Type and extent of licensing for the individual steps were requested by the authority as follows:

I. and II.:
For each step the authority's approval simply was obtained after notification of the detailed irradiation test procedure.

III. and IV.:
The authority considers a license according to the German Atomic Energy Act to be necessary for both steps in one procedure. An important prerequisite for this license are the results of thermohydraulic calculations for RS cores (see next chapter).

V. and VI.:
These steps also will be treated in one licensing procedure. Essential topics for the application are changes in the core fission product inventory and their consequences for design and beyond design basis accidents (including fuel melting), the kinetic behavior of LEU cores and the increased absorption of $\gamma$-radiation leading to higher after heat generations in LEU fuel elements.

THRMODYNAMICAL CONSEQUENCES OF THE CORE CONVERSION

For the optimum conversion of FRJ-2 reactor to operation with fuel element of the RS type (roll-swaged fuel plates), an investigation of the thermodynamic consequences of fuelling the DIDO core with these fuel elements has been carried out. The following study of the flow conditions will show to what extent an increase in the flow resistance of the RS core due to the design differences will affect the safety margin to the flow instability condition.

Due to continuously roll-swaged webs in the ring gaps, HEU fuel elements of the RS type exhibit a 6.8% smaller free cross-sectional area as compared to the present EB design (Figure 3.), which leads to a higher resistance to heavy water flow and thus also increases the contribution of the core to the total pressure loss of the primary circuit. An increase of the total pressure loss also means that the operating points of the main pumps as the intersection between the characteristic of the pumps and the pressure loss curve of the circuit is shifted downwards and adjusted to a slightly lower flow rate. The decrease of the total flow rate depends on the increase of the pressure loss in the RS core and thus on the change in $D_2O$ velocity as compared to the EB core.
Because of the simple channel geometry of the EB-element, the pressure drop has been determined for this core by applying the steady state thermohydraulic model of the PARET code /2, 3/. This code accounts for pressure drop resulting from friction, elevation and in/outlet of the coolant. The hydraulic model of PARET was verified by recalculation of the pressure drop measurements performed at FRJ-2 /4/. The measurement shows that the pressure loss of the coolant in an EB-element amounts to 39.82 kpa for a flow rate of 18.49 kg/s. By comparison, a model calculation using PARET results in a total pressure loss of 40.81 that represents a good agreement between calculation and measurement. Because of the verified model of the PARET code it was used to calculate a pressure drop characteristic for the EB-elements by varying the flow rate. A polynomial fitting has resulted in the following expression as a function of mass flow through the element (flow rate in m³/s and pressure in kpa):

\[ \Delta P_{EB} = 3.420 + 847.79 m + 83661.68 m^2 \]
For the optimum flow distribution, the fuel elements of FRJ-2 have been equipped with a perforated cone resulting in further pressure drop. On the basis of extended measurements and polynomial fitting the following expression has been derived for the total coolant pressure drop in an EB-element /4/:

\[ \Delta P_{\text{EB}} = -4.308 + 1084.95 \ \text{m} + (1.3446 \times 10^5) \ \text{m}^2 \]

Because of the flow condition, the flow resistance of the RS fuel element can be determined using the Balsius equation /5/, i.e.:

\[ \lambda = \frac{0.3164}{\text{Re}^{0.25}} \]

In conjunction with \( \Delta P = \lambda \cdot L/d_H/2 \cdot v^2 \) (where \( v \), \( L \) and \( d_H \) are referring to as heavy water velocity, the length and hydraulic diameter), the pressure drop in a RS fuel element can be calculated as a function of velocity in term of \( P_{\text{EB}} \) as follows:

\[ \Delta P_{\text{RS}} = \Delta P_{\text{EB}} \left( \frac{v_{\text{RS}}}{v_{\text{EB}}} \right)^{1.75} \]

Under consideration of the geometry data of the RS elements and the resultant velocity ratio, a value of 67.3 kpa is obtained for the pressure loss in the RS core. This means 6.1 kpa increase of the pressure drop as compared to the EB core and consequently an upward shift of the total pressure drop of the primary circuit.

In order to define the \( D_2O \) flow rate through a reactor core completely equipped with RS elements, the intersection between the pump characteristic and circuit pressure loss curve was iteratively calculated as a new operating point of the main pumps. The total pressure drop in the primary circuit consists of the contributions of the core and piping system. The latter is represented by a common quadratic function of the total flow rate as follows /4/:

\[ \Delta P_p = 796.96 \ \text{m}^2 \]

According to the existing specification, the head characteristic of the main pumps (converted into kilo pascal) can be also represented by the following polynomial:

\[ H_p = 193.72 + 315.21 \ \text{m} - 516.91 \ \text{m}^2 \]

The graphic representation of hydraulic condition of the primary circuit in terms of pressure drop for the case of the EB and RS core, respectively, have been given in Figure 4.
The result of iterative calculation shows that the flow rate decreases by 1.37% from 0.460 m³/s to 0.454 m³/s. Despite this reduction in flow rate, the coolant velocity will effectively increase by approx. 5.5% in the case of the RS fuel elements due to the 6.8% smaller free cross-sectional area of the cooling channels. This will improve finally the thermohydraulic safety behaviour of a core completely equipped with the RS elements and increase the safety margin to flow instability.

The dependence of the critical heat flux on the coolant velocity represents the decisive factor for the beginning of flow instability. The theoretical evaluation of the experimental results obtained by Forgan and Whittle [6], which are shown in Fig 5 for different inlet temperatures (T_in), has revealed the following dependence between the critical heat flux, Q_c, and the coolant velocity (v) [7, 8]:

\[ Q_c = -29.35 + (128.15 - 1.104 \, T_{in})v^{0.8} \]
Figure 5: Average Critical Heat Flux at Onset of Flow Instability as a Function of Water Velocity

Any shift in critical heat flux and thus any change in the safety margin as the ratio of $Q_c$ to the actual heat flux can be determined on the basis of this relation. For instance in the case of an inlet temperature of 55°C, the 5.5% increase of the average heavy water velocity (from 4.60 m/s to 4.86 m/s) would entail a 5.1% increase of the safety margin.

The critical heat flux, $Q_c$, at the onset of flow instability was separately calculated for the operating condition of FRJ-2 with 55°C for the coolant inlet temperature and 16.7 kpa for the core inlet pressure using the PARET code in which $Q_c$ is represented by a linear function of the coolant velocity and margin of the inlet temperature to the saturation point. On the basis of the measurements of Forgan and Whittle and in consideration of the geometrical data of the cooling channel, a constant value of 25 was used for the bubble detachment parameter. For a coolant velocity of 4.6 m/s in an EB-element, 216.7 W/cm² was resulted for $Q_c$ which shows a deviation of 8.76% to the value of the above function. In the case of a coolant velocity of 4.86 m/s in a RS-element, the onset of flow instability is shifted to 231.7 W/cm² according to an increase of 6.9%. Accordingly, the cooling condition of the most heated channel of FRJ-2 would be improved by an increase of the safety margin from 1.32 to 1.41 as a result of the core conversion from the use of the EB to RS-
elements. Moreover, a qualitative consideration shows that the thermohydraulic conditions are favourably influenced with respect to the heat removal and heat transport from the fuel plates to the coolant because the roll-swaged webs will additionally contribute to radial heat transport and heat removal due to their function as a heat conducting bridge.

STATUS AND OUTLOOK

Until now the conversion steps I. and II. have been completed using elements delivered by Nukem. For these test irradiations of RS elements in HEU and LEU originally a time period (1985), between 7 and 8 months had been anticipated based on a reactor operation (utilization) of 70 Z - 75 Z per year. Since during the last 4 years the utilization of FRJ-2 reached about 50 Z per year only, the time length of the irradiation test in each case has been accordingly extended to approximately one year. Furthermore, supplementary requirements of the licensing authority (each test had to begin with one element and could be continued for the remaining two elements when the first one after irradiation had passed a TÜV inspection successfully) resulted finally in an extension to about 19 months.

When in March 1988 Nukem decided to stop fabrication of fuel elements for research reactors KFA (and all other German research reactor operators) were forced to find and above all to qualify new fuel element manufacturers. Such a qualification requires that a new fabricator has to prepare all documents necessary for manufacturing (specification, drawings, inspection plan), according to German rules and guidelines and that in addition has to be audited by the customer and German TÜV with respect to the quality assurance system in use and to special fabrication techniques (e.g. fuel powder fabrication, plate manufacturing, RS process, fuel element assembling). Furthermore the authority demanded to repeat steps I. and II. with elements delivered by the new fabricator because a successful irradiation test finally completes the qualification of the fabricator.

Although mid of 1988 an order to fabricate test RS elements (3 each in HEU and LEU) had been placed with the French CERCA actual fabrication began some weeks ago. Therefore at present the time schedule for FRJ-2 conversion can be estimated very roughly only.

We expect that (repetition of ) step I. will begin in the 2nd quarter of 1991 and subsequently step II. in the 3rd quarter of 1992.

Without giving further details the rough estimate reveals that the very important step V. (HEU-LEU transition phase) will not begin before the 3rd quarter of 1995 under optimistic circumstances!

Therefore it is very uncertain today wether this step can be realized. There are two main reasons. In 1995 FRJ-2 will have reached an age of 33 years and if operational troubles would continue to increase as can be observed at present although a "trouble shooting" program was initiated recently KFA may decide to close DIDO rather than to convert it. The second reason also is due to the age of the reactor. There are even
The second reason also is due to the age of the reactor. There are even today indications that a lot of additional backfitting measures and proofs to secure and extend future life of FRJ-2 will be required in combination with the HEU-LEU conversion license and it might turn out that these additional requirements either can not or will not be realized. It also might be of importance that FRJ-2 in 1995/96 will be as old as the Harwell reactors (DIDO and PLUTO) which were closed end of March this year although Harwell repeatedly confirmed that the closure was due to commercial considerations only and not caused by ageing or safety troubles.
REFERENCES


STATE OF THE LEU CONVERSION EFFORT
AT THE UNIVERSITY OF VIRGINIA REACTOR

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ABSTRACT

The University of Virginia has submitted a Safety Analysis Report to the Nuclear Regulatory Commission asking to convert the 2 MWT UVAR reactor from 18-curved plates/element high-enriched fuel to 22-flat plates/element 20% enriched fuel. For the loss-of-coolant accident analysis, we have now chosen to adopt the 450°C softening point of aluminum as the limiting criterion. Calculations were made using a semi-empirical model previously employed for the upgrade from 1 MWT to 2 MWT. A RETRAN model of the reactor and pool was also developed. We conclude that the currently-installed emergency core cooling system is indeed required to prevent damage to the core as a result of a hypothetical-rapid-draining-of-the-pool accident. The actual conversion date of the UVAR to LEU fuel has slipped to late 1991.

STATUS OF THE LEU CONVERSION EFFORT

The UVAR is a 2 MWT swimming pool-type research reactor. It is made up of plate-type MTR fuel elements mounted on an 8-by-8 grid plate that is suspended from a movable bridge. The reactor can only be operated at full power when it is mounted at the South end of the pool, directly above a coolant funnel that provides forced down-flow cooling. In the unlikely event of rapid draining of the pool, the core can also be cooled in this position by a passive emergency core cooling system (ECCS) consisting of reservoir tanks and spray headers.

The UVAR LEU reactor physics design calculations were completed at the end of 1989. It was concluded that a slightly higher fuel loading would give improved UVAR fuel performance as well as a considerable increase in core life. The proposed 22-plate/element LEU fuel design has been accepted by B&W Lynchburg, Va and EG&G Idaho and construction drawings were prepared in November 1989. With this fuel, the UVAR can be operated in a 4-by-4 array with graphite reflectors, or in a 4-by-5 array using a combination of water and graphite reflectors. We
intend to retain the present flexibility of using any number of fuel elements between 16 and 25, subject to meeting our required shutdown margin.

As a result of discussions that took place at the XII RERTR Meeting in Berlin, we decided to reconsider the criterion used for assessing fuel damage that might occur under hypothetical emergency conditions. Instead of using an aluminum melting point condition, as employed in past studies, we have adopted a softening point criterion. Thermal-hydraulics calculations, done by two independent methods, show that under the new criterion the need for an ECCS cannot be ruled out for a range of low-probability pool draining accidents.

The UVAR Safety Analysis Report (SAR) was submitted to the Nuclear Regulatory Commission (NRC) in November 1989. UVA was informed verbally by EG&G in early 1990 that there would be a delay in the conversion schedule caused by budgetary constraints. Although the fuel plates are ready, the new LEU fuel elements have not yet been fabricated. In addition, the NRC has not yet responded to the SAR with followup questions. Therefore, the date of conversion of the UVAR to LEU fuel has slipped, with a new realistic estimate of conversion to take place in mid-to-late 1991. A more serious holdup could be the unavailability of a suitable spent fuel shipping cask, due to the cessation of operations at Centichem, the owner of one such cask.

LOSS OF COOLANT ACCIDENT METHODOLOGY

Introduction

At the time the UVAR was upgraded from 1 MWT to 2 MWT in 1971, a passive ECCS was installed that would spray water onto the core in the unlikely event of a rapid drop in pool water level [1]. For redundancy two independent sprays were installed, each capable of cooling all of the fuel elements in the core. Each contains a large (6500 l) water tank suspended from the pool wall. Spray headers that will spray water onto each of the 64 grid plate locations are attached to the bottoms of the tanks. The tanks are kept full at a level of about 60 mm above the pool level by directing a portion of the demineralizer return water to the tanks. There are no valves in the system; thus water continually flows through the headers by virtue of the 60 mm static head and keeps the spray nozzles clean.

In the event of a very rapid loss of pool water, the spray headers and the core will soon be uncovered and water ultimately will flow through the spray headers driven by the full heads (4.3 m maximum) of the tanks. The tank volumes and the hole sizes in the spray headers are designed to provide a minimum of 0.6 l/s spray flow for 1800 seconds and 0.47 l/s for an additional 3600 seconds following an instantaneous loss of pool water. Fig. 1 shows the design heat removal capacity of the ECCS along with the heat production of the hottest fuel element expected at any time in core life. Although the ECCS specifications do not require any spray flow after 5400 seconds, the tanks are by no means empty at that time since they must still provide sufficient head to maintain 0.47 l/s at 5400 seconds. The spray tanks will continue to direct spray onto the core for several thousand seconds beyond the required time but no credit is taken for any further cooling that this flow may provide.

In the event that the loss of pool water is gradual in nature, the spray tanks will drain slowly along with the pool so that in actual fact, a shorter spray time or even no spray at all might occur. However, for the spray system not to provide any supplementary cooling, the pool must take well over 5400 seconds to drain. Thus no matter how much time it takes to drain the pool, the core will be water cooled for at least 5400 seconds following a low pool level scram before having to rely upon air cooling.
The analysis of the expected peak temperatures in the fuel following a loss of coolant accident (LOCA) for HEU fuel was done at the time of the power upgrade. The analysis showed that for any credible pool draining mechanism, the maximum temperature attained by the fuel was 1122°F. This was below the criterion for damage which was set at the melting point of pure aluminum (1220°F). Nevertheless, to provide an additional margin of safety, the ECCS was installed as part of the power upgrade.

**LEU Conversion**

Due to the larger surface area and greater mass (thermal ballast) associated with the new 22-plate LEU fuel, it was hoped that the core spray system might not be needed after the conversion to LEU. To test this hypothesis and to confirm that the LEU fuel would not melt as a result of a loss of coolant, the LOCA was reanalyzed for LEU fuel.

The first question to be addressed was the criterion for not damaging the fuel. The HEU analysis merely considered the melting point of pure aluminum (1220°F). The alloy that will actually be used for the LEU cladding (6061) has a somewhat lower melting point at 1080°F. It has been reported that the clad will start to blister at 1020°F, while others contend that the softening point at 840°F should be the limiting criterion [2]. Since elevated temperatures following a LOCA may be sustained for several hours, the softening point was decided upon as the minimum temperature at which damage and subsequent release of fission products might occur.

Two independent methods were used for the analysis. One was to use the same semi-empirical method employed at the time of upgrading the UVAR from 1 to 2 MWT, which was
also the method used for the Omega West Reactor (OWR) [3]. The other was to try to construct a RETRAN model for a loss of coolant from an open pool reactor [4].

**Semi-empirical Method**

The Semi-empirical method starts with the heat balance

\[ Q(t) = \frac{dZ}{dt} (MC) + HA Z(t), \]  

where

- \( Q(t) \) = the heat generation rate,
- \( Z(t) \) = difference between the maximum fuel temperature and the average temperature of the surroundings,
- \( MC \) = total heat capacity of a fuel element,
- \( HA \) = over-all heat transfer coefficient,
- \( t \) = time since reactor shut down.

Tests were done at Oak Ridge National Laboratory [5], where irradiated MTR-type fuel elements were allowed to cool in stagnant air. These tests indicate that a correlation of the form

\[ HA = C(aZ^n + b) \]

could reasonably predict the observed maximum fuel temperatures.

Actual loss of coolant tests done at the Low Intensity Test Reactor (LITR) [6], and evaluated by OWR using the Way-Wigner formula for the decay heat source, yielded a set of values for the constants in Eq. 2 that consistently over-predicted the actual temperatures observed in the LITR tests.

Substitution of Eq. 2 and the Way-Wigner formula for decay heat production into Eq. 1 gives an analytical expression for \( Z(t) \). A computer program was written to numerically solve the expression for \( Z(t) \) to generate a curve of \( Z \), and thus maximum fuel temperature vs. time. The program was tested by comparing its results with the results obtained at OWR.

A comparison of the decay heat source tables from the fission of U-235, plus the activation of fission products given in ANSI/ANS Standard 5.1 [7], with the Way-Wigner formula indicates that ANSI/ANS 5.1 will predict a heat source as much as 1.5 times larger than the Way-Wigner formula for the time period shortly after reactor shutdown. Since the heat source is used in the fitting process to find the constants in Eq. 2, the use of a different heat source will result in different constants.

A new value of \( C \) in Eq. 2 was chosen such that when using the ANSI/ANS 5.1 heat source, the fitted results of the LITR experiment were safely enveloped. This gave curves of maximum fuel temperature vs time after shutdown that peaked a bit earlier and higher than the original OWR constants.

Appropriate values of \( MC \), for a 22-plate LEU fuel element were obtained by summing the mass times the heat capacity at 212°F for the various materials in the fuel element [8]. The area, \( A \), in \( HA \) was adjusted to reflect the different numbers of plates between the OWR (18) and the UVAR (22). A starting fuel temperature of 212°F and an ambient temperature of 100°F was assumed for all UVAR analyses. The peak power density, as calculated using the 2DB computer code for a 4-by-4 sixteen element core [9] operating at 2 MWT for 120 hours prior to the loss of coolant, was used to determine the fission product inventory in the hottest element.
RETRAN Method

The LOCA was also modeled using the RETRAN code. This thermal-hydraulic code allowed fictitious pool leaks to be simulated by pipe breaks of various sizes occurring at the bottom of the pool volume. The pool volume was taken to be 13,800 ft$^3$ for the transient. The core modeled here was a 4-by-5 element core with 396 fuel plates and 378 cooling channels. The core was assumed to be operating at 2 MWT continuously up to the time of the pipe break and scram. The axial power distribution was assumed to lie a chopped cosine as an input parameter to the code, while the radial power distribution (also chopped cosine) was applied to the results of the code to find the conditions of the hottest channel.

RESULTS OF THE LOCA CALCULATIONS

Semi-empirical Method

The time to uncover the core following a double-ended shear of the primary piping was calculated to be 1200 seconds [10]. Application of the semi-empirical method to this situation yielded a maximum fuel temperature of 975°F which is, as expected, lower than the temperatures calculated for the HEU core at time of power upgrade. However, since this temperature is well above the softening temperature of the cladding, the ECCS is needed with LEU cores.

The ECCS is designed to keep the core wet and thus cool for 5400 seconds after a low pool level scram. When air cooling is relied upon after this time, the maximum fuel temperature reaches 770°F which is below the limiting criterion.

Figure 2 shows a family of curves of maximum fuel temperature vs. time after shutdown for various times to uncover the fuel. All times to uncover the core that exceed the minimum ECCS cooling criteria result in temperatures that are well below the softening temperature of aluminum at 840°F.

RETRAN Model

Figure 3 shows the calculated cladding temperature in the hot channel at various vertical locations vs. time following a 2 foot diameter pipe break. The maximum temperature reached during the transient in the 4-by-5 core was 555°F and occurred in the Mid-Upper region of the core. This is a result of the combination of the axial peak occurring in the middle of the core and the upward flow of cooling air which gets hotter from the bottom to the top of the core.

The maximum temperature as a function of the diameter of the pipe break closely fits an arctangent which tends asymptotically to a temperature of 650°F at large pipe diameters. The RETRAN code predicted that it would take 800 seconds for a one foot diameter pipe break to uncover the core. This is the slowest transient analyzed and yielded a peak temperature of 505°F.

Comparison Between the Two Methods

To allow the results obtained from the RETRAN model to be compared to those of the semi-empirical method, adjustments to account for the different peak power density between the nominal standard 4-by-5 core used for the RETRAN model and the minimum-size 4-by-4 core used for the semi-empirical method were made to the final differential temperatures. This raised the temperature from the RETRAN model to 620°F for the slowest leak (1 foot pipe) and to 810°F for instantaneously draining the pool.
Figure 2. Peak Fuel Temperature as Calculated Using the Semi-empirical Method for Several Transients. The Dotted Transients Assume No ECCS.

Figure 3. Cladding Temperatures at Different Axial Positions Following a 2 Foot Diameter Pipe Break, as Calculated Using the RETRAN Model.
The scaled RETRAN model consistently yielded lower peak temperatures than the semi-empirical method. For transients that took similar times to drain the pool (800 and 1200 seconds) there was about a 350°F difference between them. The shapes of the temperature vs. time curves were similar between the two methods as shown in Figs. 2 and 4. However, because the correlations used in the semi-empirical method were required to safely envelop the actual data from the LITR experiment, and were therefore higher than the actual data, this method would be expected to be conservative and overpredict the peak temperature. By itself, this is probably not enough to account for the differences observed. A more plausible explanation is that RETRAN has not been used before for such applications, and therefore, much more checking is needed before accepting its results.

![Figure 4. Peak Cladding Temperatures as Calculated by the RETRAN Model for Several Sizes of Simulated Pipe Diameters.](image)

CONCLUSIONS

The results obtained from the semi-empirical method for evaluating fuel temperatures following a LOCA were chosen for the Safety Analysis Report since this method had at least a remote foundation in experimental data and gave more conservative results than the RETRAN model. Based upon the semi-empirical method results, the ECCS as currently installed is adequate to remove the decay heat from the core for the required 5400 seconds following a LOCA. With the softening point of aluminum taken as the safety criterion, use of the ECCS is required to prevent damage as a result of a rapid draining of the pool. The margin of safety between the conditions that lead to exceeding the safety criterion and the conditions following a LOCA, along with the conservatism of the analysis, gives us confidence that the UVAR LEU fuel would not be damaged in such an accident.
REFERENCES


POWER UPGRADE AND CONVERSION OF THE
COLOMBIA R-1 REACTOR TO TRIGA-LEU FUEL.

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ABSTRACT

The IAN-R1 reactor was furnished to the Government of
Colombia under the "Atoms for Peace" program by the U.S. Government. The reactor was constructed by the Lockheed Aircraft
Corporation and achieved initial criticality at the Institute for

The reactor core consists of aluminum clad MTR-type plate
fuel elements containing fully-enriched uranium (HEU). The
reactor was initially designed to operate at a steady-state power
level of 20-kW but has operated at 30-kW for the past several
years. The principal applications of the reactor are radioiso-
tope production, radiochemistry, neutron activation analysis,
training, and neutron beam physics.

Beginning in 1987, a program was initiated to upgrade and
modernize the reactor facility. The modernization program has
included replacement of the original instrumentation and control
system with a new state-of-the-art microprocessor-driven digital
system and addition of a new radiation monitoring system.

As described in this paper, the next phase will consist of
replacement of the original HEU fuel elements with TRIGA LEU fuel
clusters and upgrading the power level to 1000-kW. The power
increase will require revision of the reactor cooling system,
modification of the reactor tank and biological shield, suffi-
cient fuel clusters to permit operation at 1000-kW with natural
convection cooling, and replacement of the control rods to
accommodate the new core configuration.
INTRODUCTION

The IAN-R1 reactor has been operated by the Instituto de Asuntos Nucleares at the Institute's Nuclear Center in Bogota since early 1965. The reactor has been used on a continuous basis for the production of radioisotopes, radiochemistry, neutron activation analysis, personnel and student training, and neutron beam physics. The reactor has operated reliably at an initial power level of 20 kW(t), which has been subsequently increased to 30 kW(t) without difficulty.

The R-1 reactor core currently consists of 13 HEU aluminum clad aluminum-uranium alloy plate-type fuel elements plus three partial elements to accommodate two shim-safety control rods and a regulating rod (Figure 1). Most of these elements have been in the core for over 25 years, so there is some concern regarding potential corrosion of the aluminum cladding, although no evidence of corrosion has been detected to date. Also, the reactor has been operating since start-up with the original instrumentation and control system slightly modified periodically as required.

In 1987, a multi-stage program was established and initiated to upgrade various reactor systems and to ultimately increase the reactor power level to 1000 kW. The program is underway and will be carried out in a series of logical steps, as set forth in Figure 2.

Note that during and following steps 1 through 4, the reactor can continue to operate at 30 kW with natural convection cooling of the core. Prior to the initiation of steps 5 and 6, the reactor will be shut down until the completion of these items.

A brief description of the steps involved in the reactor upgrade and conversion to LEU fuel is provided in the sections that follow.

Instrumentation and Control System Upgrade

The original R-1 reactor instrumentation system was a three-channel system, installed in 1965 by the Lockheed Aircraft Corporation. The system consisted of a start-up (source range) channel, an intermediate range channel, and a power channel. The instrumentation included Keithley picoameters with log N and multi-range power channel capability.

In approaching the upgrade of the nuclear instrumentation system, it appeared desirable to replace the entire system, but in a series of steps which could be completed over a period of several years. The first two steps consisted of acquisition of two NM-1000 microprocessor-based neutron
- Replace Instrumentation and Control System
- Add New Radiation Monitoring System
- Replacement of Control Rod Drives
- Replace HEU Fuel with LEU
- Extend Reactor Tank and Add to Biological Shield
- Modify Reactor Cooling System to Accommodate 1000 kW
- Increase Core Loading of LEU Fuel to Accommodate 1000 kW

Figure 2. Stepwise Upgrade of R-1 Reactor
monitoring channels (including detectors) (Figure 3). This was accomplished in 1987 and 1988. The next two steps consisted of adding the Data Acquisition and Control System (DAC) and the Control System Console (CSC). These sub-systems were acquired in 1989 and 1990 and the complete system will be installed in October, 1990 (Figure 4).

The new IAN R-1 control system is a microprocessor-based I&C system developed by General Atomics. It incorporates a digital wide-range neutron power monitor, analog power safety channels, a variety of state-of-the-art signal conditioners and process controllers, plus a digital data acquisition and control system incorporating IBM PC/AT or compatible computers in industrial versions.

The design and manufacture of this system complies with the guidance given in American Nuclear Society (ANS) and the American National Standards Institute (ANSI) Guide ANSI/ANS 15.15-1978, *Criteria for the Reactor Safety Systems of Research Reactors*. This standard serves the research reactor community in lieu of the ad hoc application of similar standards for power reactors (e.g., IEEE Standard 279-1971).

There are several advantages in a microprocessor-based system which enhance system safety, reliability, and maintainability:

1. The use of powerful microcomputers allows data (operator input as well as output) to be more efficiently and systematically processed and recorded.
2. Several data reductions not previously possible (such as on-line calculation of the prompt period during a pulse) can be done in near-real-time.
3. On-line self-diagnostics can be performed which determines the state of the system at all times.
4. Operational surveillance and operations data are accommodated with all information gathering and processing done routinely and regularly by the console computers.

In addition to the new instrumentation and control system, a new area radiation monitoring system will be installed at the IAN R-1 facility. The monitoring system consists of six detectors, with local displays and alarms, six panel modules with alarms and a panel bin. The range of the system is $10^{-1}$ mR/hr to $10^1$ mR/hr.

Replacement of Reactor HEU Plate-Type Core with TRIGA-TYPE LEU Fuel Cluster

The plate-type fuel elements will be replaced with TRIGA-type 4-rod fuel clusters which are inter-changeable with the plate-type elements. The core configuration will remain essentially as shown in Figure 1. It is
Figure 3. NM-1000 Channel
Figure 4. Operating Console for GA Computer-Based Instrumentation and Control System
possible to replace all of the plate-type fuel at one time or to replace it in several steps with the TRIGA 4-rod clusters. There are no restrictions on the placement of the higher loaded TRIGA fuel clusters in a partially loaded plate core at a power level of 30 kW.

The TRIGA uranium-zirconium hydride fuel will contain 12 wt-% U and will have a nominal H/Zr ratio of 1.6. It is designated as 12 U-ZrH$_{1.6}$ fuel and each fuel rod will contain about 47 grams of U-235. The active fuel height is 381 mm (15 in.). The fuel cladding is stainless steel with an outside diameter of 34.93 mm (1.375 in.) and a thickness of 0.51 mm (0.020 in). The overall fuel element length is about 762 mm (30 in.).

The fuel cluster shown in Figure 5 consists of the four fuel rods and an aluminum bottom adapter and an aluminum top handle with locking plate and locking screw. The bottom adapter is designed to fit into an existing MTR-type grid plate which provides vertical support and spacing. The adapter contains four tapped holes into which the fuel rods are threaded. The bottom end fitting on the fuel rod is provided with a flange at the base of its threads so that the fuel rod seats firmly on the adapter and is rigidly supported in cantilever fashion.

The top handle of the cluster serves as a lifting fixture and a spacer for the upper ends of the fuel rods. A sliding fit is provided between the top handle and the fuel rod upper end fittings to accommodate differential expansion. A 0.94-in.-diameter coolant flow hole is provided in the center of the handle and four cutouts of 1/2 in. radius are provided between fuel rod locations. The corners of the top handles are chamfered to provide additional coolant flow area.

The reloaded core with all 12 U-ZrH$_{1.6}$ TRIGA fuel will require about 45 to 50 fuel rods for the necessary operational excess reactivity (minimum of about $1.00). Thus, it is anticipated that several in-core irradiation locations will be available in the equivalent Fig. 1. core configuration with all TRIGA fuel. In any 4-rod cluster, one fuel rod can be replaced with an aluminum tube which can be used as an experimental location for experiments having an outside diameter up to about 31.7 mm (1.25 in). Thermal flux values of about $1 \times 10^{12}$ n/cm$^2$/sec should be available in the in-core experimental locations at 30 kW.

The presently existing control rods will be used in the converted core at the 30 kW power level. The partial plate elements where the control rods are inserted will be replaced by aluminum billets. Each billet will have a hole shape for the control rod the same as that in the existing partial plate elements.

**Reactor Tank Extension and Concrete Shield Modification**

In order to operate an upgraded R-1 reactor at 1000 kW, significant increases must be made in the shielding surrounding the reactor core. To provide a graphic example of the increased shielding required, Figure 6 presents an isodose line for 1 mR/hr relative to the existing shield and reactor tank.
Figure 5. Fuel Cluster Assembly
Figure 6. Colombia Shield Upgrade Isodose Lines at 1000 kW
From the shielding calculations performed, it appears that a conservative approach would be to increase the existing tank height from 5.28 m to 7.47 m. This can be done in two ways; weld an extension to the existing aluminum tank or provide a new stainless steel tank which could be inserted inside the existing tank. Since the existing tank shows minor corrosion spots in the tank bottom, it might be more desirable to install the new tank, making appropriate provision for the beam-tube penetrations through the tank wall.

The existing concrete shield structure must be supplemented by additional concrete in order to provide reasonable radiation levels at a power level of 1000 kW. Figures 7 and 8 show the amount and geometry of the concrete which must be added to the shield. The weight of the added concrete will be supported by a separate footing, as shown in Figure 7.

**Reactor Cooling System Upgrade**

The current R-1 HEU plate-type core is cooled by natural convection cooling. With the reactor operating at 20-30 kW, no external cooling system is required, since reactor operation is not normally for extended periods of time.

The upgraded reactor, operating with TRIGA LEU fuel clusters, can also be cooled by natural convection cooling, at power levels up to 2000 kW. Thus, provision must be made for removing hot water from the tank, cooling the water, and returning it to the tank. Provision will also be made for a water purification system and water makeup supply.

For the upgraded reactor, a primary cooling system will be provided, consisting of a 350 gpm pump, a plate-type primary-to-secondary heat exchanger, and required piping and valves. The secondary cooling system will also be provided with pump, cooling tower, valves and water treatment system.

The possibility of natural convection cooling of the core greatly simplifies the routing of piping for the cooling system, since no penetrations of the reactor tank or shield are required.

**Increased Core Loading for 1 MW Operation**

At the time of the major shield and cooling system modifications for 1 MW operation, a new grid plate will also be installed. It will have the same 6 cluster width as the present grid plate in order to fit between the
Figure 7. Modified Shield Structure Elevation View
Figure 8. Modified Shield Structure Plan View
existing beam tubes. However, as shown in Figures 9, 10, and 11, it will be either 9 or 11 fuel clusters in length vs the present 7 cluster length. This will allow a great deal of core configuration versatility, including the ability to operate the beam tubes as either tangential or radial to the core. Thus, the gamma and fast neutron components to the beam tube spectrum can be sizably reduced with a tangential beam configuration, as shown in Figures 9 and 10.

Approximately 75 fuel rods are needed for heat removal at a power level of 1 MW. Figure 9 shows a core configuration with 82 fuel rods, 4 control rod locations and 2 in-core irradiation positions. When the core is reloaded for 1 MW operation, the control rods will be changed to new solid B4C rods, clad with aluminum. The control rods are about 31 mm (1.25 in.) outside diameter and are located in a cluster where one of the fuel rods has been removed and replaced with a control rod guide tube. One of the 4 control rods is a pulse rod.

An initial core loading of 82 unburned (or lightly burned) 12 U-ZrH\textsubscript{1.6} fuel rods with a core configuration as shown in Fig. 9 would have more excess reactivity than necessary for 1 MW operation and likely more reactivity than could be controlled with the 4 control rods (and meet the stuck rod criteria). Some of the options available for reducing the excess reactivity at the time of loading the core for 1 MW operation will be:

1. A modified core configuration to spread the core over a larger area and have more in-core experimental locations. An example of this is shown in Fig. 10.

2. Lower the uranium content in the fuel which is added to the original 12 U-ZrH\textsubscript{1.6} TRIGA conversion core when loading for 1 MW operation. This would be 8.5 U-ZrH\textsubscript{1.6} fuel. The 1 MW core with a mixed uranium loading would have the 8.5 U-ZrH\textsubscript{1.6} fuel loaded in the higher power central core region.

3. A combination of the above two options is also available.

If a mixed uranium loading is used for the initial 1 MW core configuration, it would be slowly restored to an all 12 U-ZrH\textsubscript{1.6} core as fuel was replaced due to burnup.

The 1 MW core with all TRIGA fuel will have a pulsing capability of up to 3 dollars, producing pulse peak power levels of up to about 1200 MW.
Figure 9. Proposed IAN Reactor Upgrade Grid Layout and Core Configuration
Figure 10. Proposed IAN Reactor Upgrade Grid Layout and Alternate Core Configuration
Figure 11. Proposed IAN Reactor Upgrade Possible Alternate Grid Layout
CONCLUSION

The IAN R-1 reactor has operated reliably at a power level of 20-30 kW with HEU plate-type fuel and most of the original reactor components for over 25 years, serving the needs of the Institute as a low-cost, reliable neutron source.

In 1987, a modernization program was initiated and the reactor is now equipped with a modern, state-of-the-art digital instrumentation system that will provide for continued reliable reactor control and a greatly expanded capability for data acquisition, storage, and retrieval.

The next step will consist of replacement of all of the original HEU fuel with LEU TRIGA-type fuel clusters that will enhance reactor safety, flexibility of operation, and will meet the non-proliferation objectives of the Republic of Colombia.

The final steps of modernization will be to install a new reactor tank liner, increase the size of the biological shield, and to install a new primary and secondary cooling system. These final steps will provide for the continued safe, reliable operation of the R-1 reactor at an increased power level of 1000 kW, which will permit the greatly enhanced conduct of medical isotope production, activation analysis, student training, neutron radiography and neutron beam physics and ensure that the reactor will serve as a valuable resource for IAN for many years in the future.
SESSION VII

September 26, 1990

FUEL CYCLE

Chairman:

J. Matos
(ANL, USA)
STATUS REPORT OF THE BACK END OF THE FUEL CYCLE FOR RESEARCH REACTORS

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ABSTRACT

US-DOE has suspended its reprocessing policy for Research Reactors by the end of 1988 due to environmental concerns. Most of the Research Reactor operators in the Western hemisphere have had up until now no alternative to close their Fuel Cycle and are worried about this uncertain situation.

UKAEA and NUKEM have established a restricted alternative concept for interim storage and reprocessing under special terms and conditions.

1. Historical Consideration


Only the consumed portion of U-235 had to be paid for.

This policy was to provide financial assistance to owners and operators of Research Reactors under the "ATOM's for Peace" program.

In 1974, US-AEC determined their Research Reactor owners were now financially able to purchase their own fuel and, therefore, the lease policy was determined.

Under this new policy, the HEU had to be bought by the customers under an enrichment contract. For the return of the spent fuel to the U.S., reprocessing contracts were offered by DOE. The fuel was shipped to Savannah River Plant (SRP) or to Idaho Falls Reprocessing Plant (IFRP) and stored for reprocessing. The recovered uranium was credited to the customer's account at DOE and used for new HEU orders.

This policy was valid until 31 December 1988 when the expiring reprocessing contracts were not prolonged as predicted.
In mid January 1989, the public was informed that US-DOE’s new reprocessing policy will not be published until the end of June 1989. In order to fulfill the requirements of the U.S. National Environmental Policy Act (NEPA), DOE had to perform an Environmental Assessment (EA). Potential environmental risks and effects as a result of transporting and processing spent fuel within the U.S. - with the exception of the DOE reactors - had to be studied. This study aimed to consider the impact of a 10-year extension of the proposed new reprocessing policy by DOE before new reprocessing arrangements were made.

That means that no further spent fuel was accepted at SRP or IFRP since the beginning of 1989.

Already three months later we received information that DOE may not meet its deadline of June 1989 for the publication of its new reprocessing policy in the U. S. Federal Register.

Thereafter, further delays occurred and at this time nobody seems to be capable of predicting a time schedule for further proceedings.

2. Consequences for Research Reactor Stations

This situation means for most of the operators of Research Reactors that they cannot close their Fuel Cycle and have no perspectives in that respect for the future.

The consequences are:

a) Storage pools are running full with spent fuel
b) Operation cannot be continued without available storage capacity
c) Licenses for operation are rejected or could even be withdrawn by the competent license authorities.

3. Alternatives for closing the Fuel Cycle

After many investigations, NUKEM could not find an adequate alternative to DOE’s former reprocessing program, so far.

That means that no processor is in a position to:

- reprocess the spent fuel
- credit the recovered uranium on a book account
- accept the credits for new HEU orders
- keep the waste for final disposal
- offer all these services for reasonable terms and conditions.
4. UKAEA's interim storage and option for reprocessing

As a result of close cooperation between UKAEA and NUKEM over more than one year, both parties recently established a concept of how the Fuel Cycle for Research Reactors could be closed under special terms and conditions.

This concept, which was approved by the U. K. Department of Energy (UK-DOE) offers interim storage with a reprocessing option at UKAEA’s Dounreay Establishment, Scotland.

Under this concept, the spent fuel can be stored until September 1994. By then, the customer must decide whether to exercise the reprocessing option under terms and conditions still to be agreed upon at a later date.

If the customer decides against reprocessing, the fuel has to be removed from the U.K by September 1996.

Before signing the storage/reprocessing option agreement, a "Memorandum of Understanding (MoU)" (or Heads of Agreement) has to be signed by the customer.

This MoU describes the form of a reprocessing agreement which the customer would be required to accept before September 1994.

Any reprocessing agreement with UKAEA must include the return of waste clause to the country of origin.

For that reason, each storage/reprocessing option agreement does require a formal agreement between the concerned Governments (by an exchange of diplomatic notes) on return of waste or the spent fuel in case the reprocessing option is not exercised for whatever reasons.

Of course, such a bilateral agreement does require a national waste disposal concept to be realized within a certain time frame.

If the fuel is reprocessed, the actual recovered uranium will be placed to the customer’s disposal at UKAEA.

Customers outside EURATOM will need prior U.S. approval (so called MB 10 procedure) for the transfer and retransfer of the U.S. origin material to and from the U.K..

5. Conclusion

(See attached CHART)
Potential Alternatives: Spent Research Reactor (RR) Fuel Disposal

Alt. I:  -leads to an early shut down
         -dispose FE some day

Alt. II:  -restricted possibilities, for only a few RR available
         -keeps the reactor operating for a certain time

Alt. III:  1. U rec (recovered)
             -no re-enrichment of Urec available
             -down blending to lower enrichment levels

         2. Waste return
            -this is the same problem like Alt. IV
            -final disposal concepts/facilities are not available for most of the reactors concerned

Alt. IV:  Final disposal of the FE

Alt. V:  1. Several uncertainties on the U.S. side
         1.1 Political uncertainties
             -Environmental assessment (EA)
             -Environmental Impact Statement (EIS), might be requested
             -convince the public
         1.2 Economical uncertainties
             -establish a new policy in favor of the potential customers, that means:
               - reprocessing at reasonable prices HEU/MEU/LEU
                 (DOE indicated that the former price of $1,000.-- per kg total metal will be "at least more than doubled")
               - credit the recovered uranium (DOE indicated that Urec < 70 w/o U-5 might not be of use anymore for DOE)
               - Waste taken over included (?)

         2. Uncertainties ion the national side (eg. FRG)

         Waste return might be requested under special terms and conditions from the political point of view.
Big efforts have been taken to successfully bring most of the Research Reactors all over the Western hemisphere on the way to conversion to LEU during more than 10 years.

Now the reactor operators face a new challenge which might even be more sophisticated: The challenge how to close the FUEL CYCLE in case DOE does not renew its reprocessing policy under comparable conditions within a certain period.

I wish that this contribution will open discussions on that essential subject on the Round Table.
POTENTIAL ALTERNATIVES: SPENT RESEARCH REACTOR FUEL DISPOSAL

I. Reactor Pool (expansion)
II. External Storage
III. UKAEA Storage/Re-processing
IV. Final Disposal
V. U.S. - DOE Fuel Cycle

Spent FE

- final shut down?
- dispose FE some day

Urec
Waste return

? ?
SUPPLY OF ENRICHED URANIUM AND SPECIFICATION FOR METAL SUITABLE FOR THE PRODUCTION OF FUEL ELEMENTS FOR RESEARCH REACTORS

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ABSTRACT

Recent changes in U.S. DOE's policy with regard to supply of HEU metal has prompted us to take a new look at the HEU situation. The history and current status of the supply of HEU for the fabrication of research reactor fuel elements is summarized. The enriched uranium metal specifications of DOE, CERCA, and AEA Technologies are compared. Some comments are given on actual impurity values for DOE-supplied HEU. The possible causes of HEU recovered by reprocessing of spent HEU research reactor fuel elements by AEA Technologies are discussed.

Uranium serves as the general starting material for the production of fuel elements for research reactors in the form of metal both for LEU silicide fuels and HEU-U-Al\textsubscript{X} fuels for high performance reactors. The Department of Energy (DOE) supplies generally, under an Enrichment Contract, enriched uranium in the form of UF\textsubscript{6} and provides - upon request - the form of metal as an additional service and at additional charges.

In the middle of 1989, DOE suddenly interrupted the conversion of metal. The reason was the change of policy in the US-Defense program, i.e. DOE could no longer take metal from military stocks which they did over a period of more than three decades.

In order to give you a survey about the quantities which the NUKEM Fuel Cycle Services Division has procured for its customers since the start of the RERTR-program in 1978, picture 1 shows the chronological and accumulated uranium supplies for research and test reactors since 1978 and including the year 1990. The total supplies of 20\% enriched uranium now have reached 2450 kg uranium; the deliveries of a 45\% enriched uranium were totally 530 kg and the supplies of 93\% enriched uranium have summed up to 1660 kg.

This picture also reveals the progress of the RERTR-program; deliveries of 20\% enriched uranium have started in 1985 and have bypassed HEU supplies - expressed in kg - this year.

For more detailed information, I have provided picture 2 which shows the correspondent annual quantities of supplied enriched uranium.

In the meantime, DOE changed its policy again and offered as of mid-summer of this year the supply of 20\% enriched uranium in the form of metal again. For enrichments above 20\% U-235 (i.e. 45 and 93\%) DOE announced that they would supply this metal; on a case to case basis.
We can confirm this since we have - in the meantime - received offers again from DOE both for 20% and 45% enriched metal.

The stop of DOE's policy to provide enriched uranium in the form of metal as of mid 1989 forced us to find other solutions here in Europe in order to fill this gap.

For the supply of 20% enriched uranium in the form of metal we had no difficulties in the past and will not have them in the next five years since we dispose of stocks of enriched metal of US origin stored in Europe and comprised of tons of this material.

For the conversion of 45 and 93% enriched UF6 into metal we contracted in the meantime with the French Cogema who were able to fill the needs.

Also, AEA Technologies, Dounreay has started to offer UF6-metal conversion.

The sudden stop of conversion services by DOE made it necessary for us to investigate on the specifications required for uranium metal since we noted that there is a confusion in general between the specification required by research reactor, the supplier of the enriched uranium, and the manufacturer of fuel-elements. Picture 3 shows the DOE-standard purity of enriched uranium metal. You will note from this table that the metal showing a purity of 99.88% has 1.200 ppm of chemical impurities. Of further interest are the typical analysis of U-234 (1.50%) and U-236 (0.46%). NUKEM's experience as former fuel element manufacturer was that U-234 values in enriched uranium supplied by DOE varied from 0.5% to max. 1.14 and for U-236 from 0.01% to 1.57%.

DOE's specifications do not include a limit on the boron-equivalent which has been limited by European fuel element manufacturers to be max. 3 ppm. HEU received by DOE in the past showed the boron equivalents up to 6 ppm. The materials were nevertheless not rejected due to the fact that a possible return of the material would have created a new export licensing procedure - a time consuming affair.

The specifications of DOE also do not include a numerical value for plutonium and fission and activation products.

The only specification that could be associated with plutonium is the maximum alpha activity from all transuranic elements which is 1500 disintegrations per minute of total uranium. This figure would lead to max. 25 bq tot/g U. For Pu 239 with the specific activity of $6.17 \times 10^{-2}$ Ci/g this would correspond to $10^{-8}$ g Pu/g U or approx. 0.01 ppm = 10 ppb.

We have noted that in the past there was the firm belief of fuel element producers that enriched uranium received from DOE is to be considered as a virgin material. As a result of our investigations, we learned that fuel element producers even discovered Thorium in enriched uranium supplied by DOE.

It is not our intention to criticize the purity of enriched metal delivered by DOE, but to encourage discussions between the research reactor community and the suppliers/producers in order to arrive at more unified/simple specifications.
Together with CERCA, we worked out the following specifications (picture 4). You will note that there exists as a limit for the total chemical impurities a value of 1,500 ppm. Also of importance is the boron equivalence which should not exceed 3 ppm.

Picture no. 5 shows the uranium metal specification of AEA Technologies. AEA has set up a limit for the total boron equivalent of 2.5 ppm.

Of utmost interest in our opinion for the research reactor community for the future is what happens with the supply of enriched uranium if DOE does not offer reprocessing of spent research reactor fuel for a foreseeable period.

You will now certainly ask what has the possible inability of DOE not to reprocess for a certain period to do with the specification of enriched metal.

For us the answer is quite clear:

As is known now to this audience, AEA Technologies is offering to research reactors the storage and reprocessing of spent research reactor fuel. AEA offers the recovery of the spent uranium to metal and does not enrich it again to the original enrichment as DOE offered in the past.

As it is also known to you, an original enriched uranium at 93% U-235 will - after burn up of up to 50% - have a final enrichment between 70 and 80% U-235.

The U-236 content in the spent fuel will be in the range of 6 to 12% depending upon the neutron flux particular to the reactor type.

Since there exists no market for a 70 to 80% enriched uranium, we have investigated what to do with this material. One possibility is to have the reprocessed uranium blended down to 20% to make it suitable again for the manufacturing of LEU fuel elements. This would, of course, mean a loss of separative work. On the other hand, there will be problems especially with the U-236 value even if natural or depleted uranium blending components are being used. Preliminary calculations made by us have shown that the U-236 value in a 20% enriched material might arrive at 2% and above. This would, of course, mean that manufacturers of fuel elements have to agree to higher limits of U-236 contents than before.

For research reactor operators, increased U-236 contents in the fuel elements should not cause significant problems since - according to an investigation which we made - U-236 will behave in the same manner to that of U-238, i.e. a high neutron capture cross section and the resulting parasitic affect on neutrons.

The other possibility to re-use a reprocessed 70 to 80% enriched uranium is to come to an agreement with DOE in the respect that they are ready to take over the reprocessed material as credits for future enrichment campaigns and in a form to be agreed upon.

We have already started discussions with DOE on that subject and shall follow up on this matter.
NUKEM's accumulated Uranium Supplies for MTRs since 1978

- HEU (93 w/o $^{235}$U)
- LEU (20 w/o $^{235}$U)
- MEU (45 w/o $^{235}$U)
NUKEM's Uranium Supplies per Year since 1978

- HEU (93 w/o $^{235}U$)
- MEU (45 w/o $^{235}U$)
- LEU (20 w/o $^{235}U$)


Kgs U

NUKEM
DOE supplies the metal in the form of sheared pieces of casting ranging from 50 to 200 grams with an average of 150 grams per piece.

<table>
<thead>
<tr>
<th>Impurity</th>
<th>Typical analysis (ppm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Al</td>
<td>15.0</td>
</tr>
<tr>
<td>B</td>
<td>&lt; 1.0</td>
</tr>
<tr>
<td>Be</td>
<td>&lt; 0.15</td>
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<tr>
<td>C</td>
<td>400.0</td>
</tr>
<tr>
<td>Ca</td>
<td>&lt; 10.0</td>
</tr>
<tr>
<td>Cd</td>
<td>&lt; 0.10</td>
</tr>
<tr>
<td>Co</td>
<td>2.0</td>
</tr>
<tr>
<td>Cr</td>
<td>15.0</td>
</tr>
<tr>
<td>Cu</td>
<td>10.0</td>
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<tr>
<td>Fe</td>
<td>800.0</td>
</tr>
<tr>
<td>Li</td>
<td>0.4</td>
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<tr>
<td>Mg</td>
<td>10.0</td>
</tr>
<tr>
<td>Mn</td>
<td>10.0</td>
</tr>
<tr>
<td>Mo</td>
<td>30.0</td>
</tr>
<tr>
<td>Na</td>
<td>&lt; 1.0</td>
</tr>
<tr>
<td>Ni</td>
<td>45.0</td>
</tr>
<tr>
<td>P</td>
<td>&lt; 100.0</td>
</tr>
<tr>
<td>Pb</td>
<td>&lt; 5.0</td>
</tr>
<tr>
<td>Si</td>
<td>200.0</td>
</tr>
<tr>
<td>V</td>
<td>&lt; 1.0</td>
</tr>
<tr>
<td>W</td>
<td>&lt; 100.0</td>
</tr>
<tr>
<td>Uranium</td>
<td>99.88 %</td>
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</table>
DOE-Standard Purity - Enriched uranium metal

Physical form

<table>
<thead>
<tr>
<th>Gas Analysis</th>
<th>Typical Analysis % (ppm)</th>
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</thead>
<tbody>
<tr>
<td>H2</td>
<td>1.30</td>
</tr>
<tr>
<td>N2</td>
<td>29.0</td>
</tr>
<tr>
<td>O2</td>
<td>32.0</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Isotopic Analysis</th>
<th>Typical Analysis</th>
</tr>
</thead>
<tbody>
<tr>
<td>U-234</td>
<td>1.50</td>
</tr>
<tr>
<td>U-235</td>
<td>93.15</td>
</tr>
<tr>
<td>U-236</td>
<td>0.46</td>
</tr>
<tr>
<td>U-238</td>
<td>5.39</td>
</tr>
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</table>
Receiving Specification of CERCA, Romans for a fresh 19.75 +/- 2 % U-235 enriched uranium in the form of metal

1. Physical Form

The uranium delivered has to be in form of pieces ranging from 50 to 300 grams uranium.

2. Characteristics of the uranium

The uranium delivered has to have the following isotopic composition (in weight %):

- U-234: - 0.14 %
- U-235: 19.75 % +/- 0.20
- U-236: - 0.14
- U-238: to be measured

The metal has to satisfy the limits of chemical impurities as follows (in ppm max.):

- Al: 150
- B: 1
- C: 800
- Ca: 100
- Cd: 1
- Li: 10
- Si: 300
- Sn: 100
- W: 100
- Zr: 250
- Ag: to be measured
- Mg: Cr: total of
- V: impurities: 1500
- Ni: total of
- Co: Fe: impurities: 1500
- Cu: total of
- Mn: impurities: 1500
- Mo: impurities: 1500
- Pb: impurities: 1500

The metal has to be accompanied by an analytical certificate signed and confirmed by the supplier in which the contents of all chemical elements and isotopes are precised.

The passing over of the chemical composition may be tolerated if the boron equivalent of the impurities does not exceed 3 ppm. The calculation of boron equivalent shall be made from the following elements: Ag-B-C-Cd-Co-Cr-Cu-Fe-Li-Mn-Mo-Pb-V-N and W. As sum for all the other impurities not analysed, it is taken into account a boron equivalent of 0.2 ppm.
AEA FUEL SERVICES: URANIUM METAL SPECIFICATION

1. Physical form

1.1 Billets

1.1.1 Material: Uranium metal enriched with U-235 to a level specified by the customer. It shall be free from slag, inclusions and porosity.

1.1.2 Dimensions: Right cylinders of nominal dimensions:
- Diameter 86.4 mm
- Length 50.8 mm
- Weighing approximately 3.5 kg

1.1.3 Pickling: Billets shall be pickled in conc nitric acid by 40°C to ensure a uniform surface finish.

1.2 Rods

1.2.1 Dimensions: 17.5 mm diameter rods, in lengths specified by the customer up to a maximum of 350 mm. Weights approx. 100 gm ---> 1000 gm.

2. Analysis

2.1.1 The composition of each billet shall be determined by chemical analysis, and shall conform to the following specification:

- Plutonium < 1 ppm with respect to uranium
- Boron < 1
- Cadmium < 1
- Carbon < 100
- Silicon < 100
- Aluminium < 100
- Calcium < 100
- Lithium < 1

Total Boron equivalent < 2.5 ppm with respect to uranium

MTR/RERTR90/10/230990/hmnb
CONTRIBUTED PAPERS
FUEL PLATE DEFECTS FOR RESEARCH REACTORS WITH UAl\textsubscript{x}-Al CORE

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ABSTRACT

In this paper, we present the results of the inspection of 558 fuel plates with UAl\textsubscript{x}-Al cores. The plates were fabricated at the JEN (Spain) and the results of these inspections were part of the activities related to the repairs of the Lo Aguirre (RECH-2) reactor.

INTRODUCTION

In this paper, we present the results of the inspection of 558 fuel plates, whose main characteristics are summarized in Figure 1.

The defects found were classified, and a statistical analysis is provided in order to determine the frequency of occurrence of each defect, according to our classification. The main characteristics are summarized in Table 1.

X-ray plates were taken of each fuel plate using Balteau equipment. The conditions used for the X-rays were the following: 100 Kv, 5.0 mA, and 15 sec at 1 mt. Agfa-Gevaret Structurix D7pb was used for the plates.

Inspection of X-Ray plates as done using a negatoscope and magnifying glass. Prints made under direct contact of the characteristic defects will be presented here.

DEFECTS CLASSIFICATION

The defects were classified under three different categories:

Internal Defects

These are defects appearing inside the "meat." We include in this category all the defects appearing in the zone "completely covered with fissile material" as well as "partially covered with fissile material" area (Figure 2).

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External Defects

These are defects appearing outside the "meat," in the so-called "free from fissile material" (Figure 2).

Geometrical Defects

Here we include geometrical defects of the "meat" such as excessive curvature or lack of perpendicularity at the ends (Figure 3).

INTERNAL DEFECTS

The internal "meat" defects are due to problems with the starting materials (UA1$_x$ or Al powder) as well as due to problems during the mixing, compaction, or lamentation of the sandwich. The classification presented here does not include end defects such as dog-bone or tail-fish, which are not part of this paper.

Point Defects

a) White spots
b) Black spots

Line or Surface Defects

a) Cluster or alignment of white spots
b) Cluster or alignment of black spots

Point Defects

White or black spots correspond to higher or lower point concentrations of fissile material. Normally they are due to a single particle or a cluster of fissile material (white spots) or aluminum (black spots).

White spots are also possible due to gas or trapped humidity, which might be found with a blister test.

Clusters of particle alignments are due to the lamination process. Most of them are oriented in the longitudinal axis of the plates (the rolling direction).

EXTERNAL DEFECTS

Defects outside the "meat" are due to assembly problems of the sandwich and to particles trapped between frame and cover during the assembly stage which are finally trapped outside the meat during rolling.

External defects can be classified into two main groups:
Point Defects

- Remote islands which correspond to individual particles trapped.

Line or Surface Defects

- Clusters which correspond to a set of particles separated less than 5 mm.

RESULTS

The fuel plates examined were fabricated at the JEN and the results reported were part of the repairs of the Lo Aguirre Reactor Fuel (Table 1).

TABLE 1

<table>
<thead>
<tr>
<th>INTERNAL DEFECTS</th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>White Spots and Clusters</td>
<td>52 Plates</td>
<td>9.3%</td>
</tr>
<tr>
<td>Excess of Meat Curvature</td>
<td>15 Plates</td>
<td>2.7%</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>EXTERNAL DEFECTS</th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Remote Islands</td>
<td>130 Plates</td>
<td>23.3%</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>REJECTED PLATES</th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>White Spots</td>
<td>7 Plates</td>
<td>1.3%</td>
</tr>
<tr>
<td>Clusters</td>
<td>10 Plates</td>
<td>1.8%</td>
</tr>
<tr>
<td>Remote Islands</td>
<td>4 Plates</td>
<td>0.7%</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td>21 Plates</td>
<td><strong>3.8%</strong></td>
</tr>
</tbody>
</table>

Considering that the above-mentioned fuel plates were fabricated many years ago and in order to fully characterize them, the length of the active surface was measured using a digitizer by integrating the total meat area divided by the meat width.

The results obtained for a sample of 71 plates were the following:

Meat Width = 57.8 +/- 0.6 mm

Meat Length = 588.3 +/- 6.6 mm

The method used can be recommended for an accurate measurement of the active surface, width, and length of the meat.
Fig. 1 Fuel Plate Characteristics
Fig. 2 Meat Internal and External Defects
Fig. 2 Meat Internal and External Defects
Fig. 2 Meat Internal and External Defects
Fig. 2 Meat Internal and External Defects
Fig. 2 Meat Internal and External Defects
Fig. 3 Geometrical Defects
Fig. 3 Geometrical Defects
ADVANCES IN FUEL FABRICATION AT THE CHILEAN COMMISSION FOR NUCLEAR ENERGY (CCHEN)

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ABSTRACT

A brief description about the progress in MTR fuel fabrication and testing is presented here. Results related to the repairs done for Lo Aguirre Reactor Fuel (RECH-2) and the advances on the construction of a glove box line for enriched uranium fuel fabrication, a hydraulic loop for fuel element tests and a shatter box for silicide powder production will be presented.

More detailed results of the defects found in the fuel plates will be part of another presentation at the same meeting.

INTRODUCTION

As has been previously reported (1) the Lo Aguirre Reactor fuel was damaged during the criticality experiments in 1977. Therefore, 31 fuel elements originally fabricated at the JEN were disassembled, inspected, and reassembled.

For this purpose, different tools and machines were designed and fabricated and some of them will be presented here.

Out of the original 31 fuel elements, only 29 fuel elements were rebuilt due to the rejection of plates according to our specifications.

The reactor went critical in September of 1989 with the reassembled fuel elements. Right now, the reactor is being characterized before the power increase.

Since that core will not last too long, we have designed a line of glove boxes that will be used to fabricate the new core for the reactor. The first box has been satisfactorily tested and the line will be operating next year.

The new core will be fabricated in our facilities. Therefore, we decided to build a hydraulic loop for fuel element tests. Dummy and transparent fuel elements have been fabricated (made out of lucite). Part of the fuels support table (2X2) was also fabricated with the same material in order to be able to visualize the behavior of the cooling water.

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DISASSEMBLING AND REASSEMBLING OF FUEL ELEMENTS

After a few trials with the dummy elements, the disassembling of fuel elements was done as shown in Fig. 1.

Fig. 1 Disassembling of Fuel Elements

Thirty-one fuel elements were disassembled without damaging any of the 558 fuel plates. Therefore, the methodology used was qualified. Moreover, due to our experience, we can recommend this technique for remote operation under hot-cells.

For the assembling of fuel elements, we use the machine and tools shown in Figures 2 and 3.
FUEL ELEMENT ASSEMBLING PLATE SWAGING STAGE

Fig. 2 Assembling of Fuel Elements
Fig. 3 Swaging Tools for Outer and Inner Plates
FABRICATION OF SUPPORT PLATES

Support plates were fabricated using AG3-NE Aluminum. Tolerances and dimensions can be seen in Fig. 4.

Fig. 4 Support Plate, Tolerances and Dimensions
In order to fabricate the plates, a set of circular milling saws with spacers in between was used (see Fig. 5).

Fig. 5 Set for Support Plates Fabrication
Circular Saws Set for the Grouving
Due to the tolerances of our grooving tool and the precise dimensional measurement done with a Coordinate Measuring Machine (CMM) with 1 micron readings, it was impossible to fully comply with our starting tolerances. Total rejection was about 6.4%, which was considered acceptable according to the experience at that time.

The maximum deviation (out of tolerances) on the groove (1.7 mm nominal) was -0.03 mm and for the "pitch" (2.70 mm nominal) was 0.08 mm. Despite thus, the assured minimum coolant channel of 2.46 mm was 100% satisfied.

After measurements of 360 channels (8 fuel elements), the average coolant channel (2.70 mm nominal) was 2.64 +/- 0.06 mm.

Taking into account this experience, a new "Groving Tool" for the fabrication of support plates is under construction.

Rejection during fabrication of spacers was about 6.6%, for holders 10% and for screws 4.5%.

LOOP FOR HYDRAULIC TESTS

The total volume of the circuit including the test section is approximately 4 cubic meters. The pump is capable of reproducing the real conditions of pressure and coolant flow of the reactor in a 2X2 section.

In Fig. 6, a view of the testing section may be seen.

Fig. 6 Test Section of the Hydraulic Loop
Two glove-box lines are under construction (G-5 and G-3). The Glove-Boxes lines will be installed inside a concrete building which will be pressurized to 15 mm of water column above the outside pressure.

The boxes will operate with dry air for U$_3$O$_8$-Al fuel and with nitrogen for the silicides. All the controls will be done using a PLC and message centers will display the required information for the operators. The main control panel will be located at the entrance of the lock gate and manual actions independent of the automatic control will be possible.
SHATTER BOX

A shatter box was built for the comminution of $\text{U}_3\text{Si}_2$, using a "food processor." A stainless steel sealed box with hard steel knives was installed on the shaft. Excellent results have been obtained with it at a very low cost (Fig. 8).

Fig. 8 Schutter Box
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