UNDERWATER INSPECTION, REPAIR AND RECONSTITUTION OF WATER REACTOR FUEL

PROCEEDINGS OF A TECHNICAL COMMITTEE MEETING
ORGANIZED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
AND HELD IN PARIS, 3-6 NOVEMBER 1987
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INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 1988
FOREWORD

The present meeting was scheduled by the International Atomic Energy Agency, upon proposal of the Members of the International Working Group on Water Reactor Fuel Performance and Technology.

At the invitation by the International Union of Producers and Distributors of Electrical Energy (UNIPEDE), the meeting was hosted by the Government of France and organized by Electricité de France, in Paris. A technical visit was organized by FRAGEMA.

Forty-one participants representing eleven Member States and two International Organizations attended the meeting. Twenty papers were presented during three technical sessions, followed by panel discussions which allowed to formulate the conclusions of the meeting and recommendations to the Agency.

This meeting resulted in an excellent success, providing a forum for the exchange of information between utilities, fuel designers and other authorities and specialists, on a topic which is a very current and real concern to the industries of many Member States.

The local organizing committee, the session chairmen and all the contributors are responsible for this success.
EDIT ORIAL NOTE

In preparing this material for the press, staff of the International Atomic Energy Agency have mounted and paginated the original manuscripts as submitted by the authors and given some attention to the presentation.

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SUMMARY

1. SESSION 1. INSPECTION DIAGNOSTIC
   (Chairmen : Mr. V.V. Novikov, Mr. K. Knecht)

1.1 GENERAL
   All the partners, including utilities, fuel vendors and manufacturers, research and development and other specialists, agreed on the necessity to have available the appropriate inspection devices and methods for on-site fuel control and failure diagnostic.

1.2 SUMMARY
   It was agreed that there may be different reasons for inspection:
   a. To provide information about the general operational performance of the fuel assembly
      - leaktightness test (sipping, ultrasonic detection)
      - visual inspection
   b. To learn the operational behaviour of lead test assemblies.

   The following determinations are normally carried out:
   - Oxide thickness measurements on peripheral rods
   - Fuel rod growth
   - Fuel assembly dimensioning
   - Hold down spring force measurements
   - Extended inspection can also be carried out on individual rods, after disassembling fuel assemblies.

   N.B. Philosophy on extended inspection differs with utilities. Some utilities do not want extended inspection because of disturbance in the normal work, while other ones support these inspections for the following reasons:
   - Expectations to increase the power rate and the burn up
   - They have limited information from the fuel supplier.
1.3 CONCLUSIONS
- On site inspection can avoid high costs related to hot cell examinations;
- Control checking of a part of the fuel assemblies provides information about the operational behaviour and gives confidence to the utilities and authorities;
- Feedback of the inspection leads to design improvements of fuel assembly;
- Early information prevents the utilities running into problems (eventually leading to loss in production) during subsequent operation;
- Whenever possible long time-consuming inspections should be separated and carried out during reactor operation;
- The inspection carried out should be limited to actual new features, i.e. old inspection methods which delivered a lot of data should be cut back.

1.4 RECOMMENDATIONS
To have sufficient permanent facilities available in every plant. For example:
- leak tightness devices (sipping test, ultrasonic)
- Materials for visual inspection

All methods used during shut down periods should be designed to meet the short time schedule.

Materials devices should be designed to be easily decontaminated.

Materials used for inspection should be independent of the facilities utilized for other purposes in the power plant. (For instance, main crane of the reactor building etc.) and conversely, the systems for assembly manipulation should not be used for other purposes.
2. SESSIONS 2 AND 3

Repair and Reconstitution Techniques and Power Plant Experience with these techniques.

(Chairmen) : Session 2 : Mr L. Van Swan, Mr. M.A.C. Coquerelle
Session 3 : Mr P. Bournay, Mr. P. Duflou

2.1 SUMMARY

The general techniques, methods, tools and work procedures for the repair and the reconstitution of failed BWR and PWR fuel assemblies were described in three papers.

The application of those techniques in specific powerplants was given in five papers; 2 other papers described the techniques used in hotcells.

The utility point of view regarding repair, reconstitution and continued use of failed assemblies was presented in two papers. A panel discussion resulted in consideration of additional items such as economic and licensing aspects related to repair, reconstitution and reuse of failed assemblies.

2.2 BROAD CONCERNS EXIST ON THE FOLLOWING POINTS:

1. Proven techniques for the repairs and reconstitution of failed assemblies are available and widely used;
2. Repaired and reconstituted assemblies have typically performed without problems during subsequent irradiations;
3. Fuel assemblies should be readily repairable. Such assemblies are currently being provided by all major fuel vendors.
4. No licensing problems were identified in using repaired and/or reconstituted fuel assemblies.

2.3 ISSUES IDENTIFIED AS NEEDING FURTHER DISCUSSION:

1. Economics of assembly repair and/or reconstitution.
In addition to a consideration of the balance between direct repair costs and the loss of remaining assembly energy the following
considerations need to be addressed:
- The storage and/or disposal of failed fuel rod and/or other hardware parts;
- The increased irradiation exposure of plant personnel when not repairing leaking assemblies.

2. Advantages and disadvantages of repair and/or reconstitution during reactor outages.
The advantages seem less obvious for reconstitution than for repairs.

3. The need for the permanent installation at poolside of repair and/or reconstitution equipment. The installation of equipment facilitating assembly repair appears in general to be justified. The need for the permanent installation of constitution equipment is less clear.

4. Problems associated with fuel rod reshuffling. For example, neutronic and accountability considerations.

5. Possible problems associated with the inversion of assemblies during reconstitution work.

6. The need to investigate the cause of fuel failures and its impact on inspection and repair procedures.

7. Techniques of fuel assembly consolidation for longterm storage.

2.4 RECOMMENDATIONS
Following subjects for further meetings are recommended by the participants:
- Analysis of the factors determining the economics of repair and reconstitution techniques;
- Analysis of the nature, courses and consequences of failures;
- Study of possible ways of failed fuel rods conditioning (reprocessing, intermediate or final repository).
C'est avec plaisir que je viens, en tant que Président du Comité d'Etudes de la Production Nucléaire de l'UNIPEDE, ouvrir cette réunion consacrée à l'inspection, la réparation et la reconstitution des éléments combustibles des réacteurs de la filière à eau légère.

C'est l'Agence Internationale de l'Energie Atomique qui, à l'origine, avait prévu cette réunion. Puis, très vite, l'AIEA a proposé à l'UNIPEDE de participer à l'organisation de cette rencontre. L'UNIPEDE est, peut être certains d'entre vous l'ignorent-ils, une organisation regroupant les producteurs d'électricité, d'Europe Occidentale notamment. Beaucoup d'entre eux sont des exploitants nucléaires. Il y avait dès lors deux raisons pour que nous acceptions cette "co-sponsorisation". Première raison, l'UNIPEDE et l'AIEA entretiennent des relations régulières et confiantes ; c'est ainsi que l'Agence invite des représentants de l'UNIPEDE à ses réunions de travail, nous permettant ainsi de donner notre avis sur les recommandations, normes ou standards qu'élabore l'Agence. La seconde raison est, que bien évidemment, le combustible nucléaire est un élément essentiel de l'économie et de la fiabilité des parcs nucléaires que nous exploitons.

Deux chiffres situent bien cette importance. Chaque élément combustible représente un potentiel énergétique élevé : 100 à 150 millions de kWh. Il représente aussi un capital très lourd puisque chaque élément coûte de l'ordre du million de dollars. Il est donc normal que tout exploitant cherche à utiliser son combustible au maximum de ses possibilités. Une fois l'investissement initial fait, c'est en optimisant l'usage du combustible que le producteur d'électricité nucléaire peut le plus facilement peser sur le coût du kWh.

Au-delà de cet enjeu économique fondamental, on peut souligner que le combustible représente, du point de vue de la sûreté, la première barrière vis-à-vis des relâchements des produits de fission. Par ailleurs, sa bonne tenue permet de maintenir un très bas niveau d'activité du circuit primaire contribuant à minimiser l'irradiation des opérateurs.

Pour toutes ces raisons, le combustible est l'objet de contrôles particuliers et d'une surveillance attentive pendant toutes les phases de son élaboration, de son utilisation en réacteur et de son stockage.

La surveillance en fonctionnement passe essentiellement par un contrôle de l'activité de fluide primaire qui permet de déceler les défauts d'échancrété, pratiquement dès leur origine, permettant déjà un premier diagnostic.
A l'arrêt de la centrale, les dispositifs d'examen permettront d'établir un diagnostic précis ; d'où l'intérêt de développer une panoplie de dispositifs de manipulation et d'examen, et des appareillages de mesure plus ou moins sophistiqués pour y parvenir.

Il convient de souligner que notre expérience à tous est très clairement que la bonne tenue du combustible contraint très rarement le producteur d'électricité à arrêter l'exploitation de la centrale avant l'arrêt programmé pour rechargement. A ce jour, pour la France, deux arrêts seulement ont eu lieu, pour raison de combustible, avant l'arrêt normal, sur 200 cycles de combustible.

Il n'en demeure pas moins important, en cas de défaut ou de simple examen, de pouvoir continuer à utiliser le combustible jusqu'au terme de ses possibilités. C'est ce qui justifie les développements de matériel de manipulation, de démontage et de remontage d'un élément combustible pour permettre sa remise en service après élimination des crayons ou des structures défaillantes. Une concertation étroite est nécessaire pour adapter conception du combustible, appareillage de démontage et souci de l'exploitant de ne pas être confronté à des opérations trop délicates et trop longues.

Aujourd'hui, le combustible, son cycle de façon générale et plus particulièrement son fonctionnement en réacteur demeurent un champ de progrès tout à fait considérable. Il suffit de citer à ce propos les développements en cours dans tous les pays nucléaires : augmentation de l'irradiation, allongement des campagnes, recyclage du plutonium, améliorations technologiques diverses. Ces efforts ne peuvent se concrétiser qu'après un certain nombre d'essais en vraie grandeur donc après passage du combustible prototype dans un cœur de réacteur industriel. Le combustible doit être suivi d'une manière appropriée afin de perturber le moins possible le fonctionnement de l'installation. Apparaît ainsi l'intérêt de pouvoir procéder à certains examens technologiques particuliers sur site, à des démontages en vue d'examen en laboratoire, à des remontages de crayons examinés pour confirmer la fiabilité des améliorations étudiées.

Même si nos centrales fonctionnent bien, il est clair que de tels développements vont se poursuivre pendant plusieurs décennies et nous permettre d'améliorer encore les performances de nos centrales.

Cette rencontre vient donc bien à son heure. Elle nous en présence des équipes de développement de dispositifs d'examen et de manipulation, des équipes de conception de combustible et des exploitants. Les trois sessions reflètent bien les différents aspects de ce problème.

La représentation internationale très large montre que ces questions suscitent un grand intérêt dans de nombreux pays. Je voudrais saluer tout particulièrement nos amis étrangers qui ont fait le voyage à Paris pour participer à ces travaux et leur souhaiter un bon séjour parmi nous.

Au nom de l'UNIPEDE, je voudrais souhaiter à tous des séances studieuses, donnant l'occasion d'échanges d'expérience et des discussions fructueuses. Vous contribuerez ainsi à l'amélioration de l'utilisation du combustible, à en diminuer le coût d'exploitation, en définitive à assurer une meilleure fiabilité et une meilleure sûreté de nos centrales nucléaires.
OPENING SPEECH

R. CARLE
Deputy Director General, Electricité de France,
Chairman of the Nuclear Generation Study Committee
of UNIPEDE

As chairman of the Generation Study Committee of UNIPEDE, it is my pleasure to open this meeting on the inspection, repair and reconstitution of fuel assemblies for pressurized water reactor plants.

It was the International Atomic Energy Agency that first decided on this meeting. They contacted UNIPEDE at an early stage and suggested that we participate in the meeting organization. Most of you are familiar with UNIPEDE; this organization is a forum for electrical utilities, especially those from Western Europe, and many of these utilities utilize nuclear energy. There are thus two reasons for UNIPEDE's co-sponsorship of this meeting. First, IAEA and UNIPEDE enjoy very good, regular relations; the agency invites UNIPEDE representatives to their working meetings, which allows us to give our opinions about the recommendations, standards and codes they are drawing up. The second reason is that nuclear fuel, obviously, is an essential element in nuclear power plant economics and reliability. Two figures illustrate this point. Each fuel element has a high energy potential, i.e. some 100 to 150 million kWh, and costs about a million dollars. Accordingly, all operators try to get the maximum from fuel. Once the initial investment has been made, the operator must optimize its use to influence kWh cost.

Apart from this purely financial aspect, the fuel is - in safety terms - the first barrier against the release of fission products. In addition, its correct behaviour allows the level of radioactivity in the reactor coolant system to be kept very low, and it thus contributes to the minimization of operating personnel irradiation.

For all of these reasons, fuel is subjected to specific verifications and careful surveillance throughout all phases of its production, in-reactor utilization, and storage.

In-service surveillance basically consists of monitoring the level of radioactivity in the reactor coolant. This allows leaktightness faults to be detected almost immediately and a first diagnosis to be made.
When the unit is shutdown, examination means are used to allow a precise diagnosis to be made. This shows the interest in developing a wide range of handling and examination tools, as well as sophisticated measuring devices, so as to best accomplish this task.

It should be noted that the experience of all of us clearly shows that, thanks to good fuel behaviour, a utility very rarely has to stop plant operation prior to the scheduled refuelling outage. Up to now in France, there have only been two shutdowns, other than normal shutdowns, for reasons related to fuel; and this for 200 fuel cycles.

Nevertheless, it is still important, in the event of a failure or even a simple examination, to be able to continue using fuel up to the end of its potential. This justifies the development of equipment for handling, dismantling and re-assemblying fuel elements so as to allow it to be put back into service after defective rods/structures have been removed. Close concertation is needed to take into account fuel element design, dismantling equipment, and the operator’s worry of being faced with having to perform operations which are too complex and time-consuming.

Today, fuel, generally speaking its cycle but particularly its in-reactor functioning, is an area of great progress. This is demonstrated by the developments underway in of all the countries that utilize nuclear power. These developments include radiation increases, cycle extension, plutonium recycling as well as other various technological improvements. These developments can only be made concrete after a certain number of full-scale tests, thus after the prototype fuel has been used in a commercial reactor. The fuel must be monitored in such a way that it has a minimum impact on the operation of the installation as a whole. This shows the interest of being able to perform certain specific technological examinations on site, of dismantling to allow in-laboratory examinations, and of reassembling the the rods examined in order to confirm the reliability of the improvements studied.

Even though our plants are currently operating well, it is clear that such developments will continue for tens of years to come and will allow us to further improve plant performances.

So, this meeting is at the right time. It brings together the teams who are developing examination and handling tools, fuel design teams, and plant operators. The three sessions clearly reflect the different aspects of the question.

Our very international audience shows that the subject is of great interest in many countries. I would especially like to thank those who have come from abroad to participate to this meeting in Paris and extend them a very warm welcome among us.
On behalf of UNIPEDE, I wish to all of you that you will find here an opportunity to exchange ideas and experience during the study sessions. In this way, you will contribute to improving the use of fuel, decreasing its operating costs, and finally, to increasing the safety and reliability of our nuclear power plants.
INSPECTION DIAGNOSTIC:
FACILITIES, TECHNIQUES AND PROGRAMMES

(Session I)

Chairmen

V.V. NOVIKOV
Union of Soviet Socialist Republics

K. KNECHT
Federal Republic of Germany
FRAGEMA FUEL SURVEILLANCE PROGRAMS IN PWRs

G. RAVIER
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B. HOUDAILLE
Commissariat à l'énergie atomique, Centre d'études nucléaires de Saclay, Gif-sur-Yvette

B. GAUTIER
Service Etudes et projets thermiques et nucléaires, Electricité de France, Villeurbanne, France

Abstract

For several years, extensive FRAGEMA fuel surveillance programs have been implemented to constitute a large statistical data base. This large data base is useful for evaluating FRAGEMA fuel performance under a variety of operating conditions (load follow, frequency control, high burnup...) and to provide a feedback to fuel designer and manufacturer.

In this paper, surveillance programs are presented and the main results briefly given. They cover fuel rod dimensional changes under irradiation and fuel rod waterside corrosion.

Two examples of the application of these results are given:

- Rod bow: based on recorded values, a design curve has been plotted and used to permit a reduction of DNBR penalty. This is an example of feedback to design.

- Cladding growth: different rod behavior patterns have been closely related to manufacturing batches and ingot chemical analysis. This is an example of feedback to fabrication.

1. INTRODUCTION

The enhancement of fuel performance under increasingly severe PWR operating conditions has led FRAGEMA, in close cooperation with its clients, to develop new products with the aim of satisfying their needs.

Working with the CEA and EDF, FRAGEMA has set up surveillance programs for these products, aimed at:

- checking the suitability of the methods used to meet the specific concerns of the utility and to meet the concerns of the safety authorities, where appropriate,
- acquiring data for incorporation into the product licensing file,
- extending the data base needed to improve product design and fabrication.

These surveillance programs, tailored to each product, are tools for providing feedback of in-reactor irradiation experience to the designer and to the manufacturer.

2. TYPES OF SURVEILLANCE PROGRAM

To meet the above requirements, FRAGEMA has implemented a surveillance program in two parts:

- a standard surveillance program consisting of on-site examinations,
- an additional examination program specific to a given product and/or to particular irradiation conditions.

2.1 STANDARD PROGRAM

The aim of this program is to evaluate as rapidly as possible fuel assembly behavior during irradiation.
It involves visual and dimensional examinations in the spent fuel pit during refuelling outages. The examinations, which are detailed in Table 1, are used to characterize the mechanical behavior of the fuel. The assemblies were characterized before being loaded, so the data obtained is easy to process. These inspections were conducted by means of an examination stand with underwater cameras [1].

If necessary, they are backed up by a sipping test.

2.2 ADDITIONAL PROGRAMS

These are dependent on the specific information targeted. They are either on-site examinations using special-purpose devices or hot cell examinations of fuel rods (after withdrawal) or of the assembly structure. Table 2 lists the on-site examinations conducted as part of the FRAGEMA fuel surveillance programs.

3. SURVEILLANCE PROGRAMS

As soon as a new type of fuel is loaded into the power reactor or whenever more severe reactor operating conditions may lead to a change in fuel behavior, a surveillance program is implemented.

At first (1975-1983), FRAGEMA-supplied fuel featuring Inconel grids was irradiated in base-load reactors with standard fuel management (1/3 core reloads). Extensive 17 x 17 fuel surveillance programs were implemented, mainly in France with the aim of acquiring a large data base on the mechanical behavior of the fuel assembly and its components and on the thermal/mechanical behavior of the fuel rod [2].

As of 1983, the more severe reactor operating conditions, the extension of French nuclear installed capacity and the search for optimized fuel managements led to surveillance programs being applied to the following areas:

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**TABLE 1**

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**TABLE 2**

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<td>Crud sampling on rods</td>
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<td>Measurement of zircaloy grid width</td>
<td>Grid irradiation-induced growth</td>
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<td>Leaking rod detection</td>
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<td>Thimble tube visual inspection</td>
<td>Thimble tube wear</td>
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a) increase of discharge burnup: five assemblies of the 17 x 17 type, all characterized before irradiation, were irradiated for five cycles and reached a burnup of about 56,000 MWd/tU.

In addition to on-site dimensional examinations, fuel rods were examined in a hot cell after 4 and 5 cycles and one of the assemblies is undergoing hot laboratory examination. A further 4 assemblies with a 4.5 % feed enrichment were loaded; two of them have just completed their fourth cycle of irradiation. The assemblies were examined on-site and the rods were routed to a hot cell.

An on-site examination program covering a large number of assemblies irradiated over four cycles in different 900 MW reactors is now under way.

b) use of burnable poisons. Assemblies containing characterized gadolinium-poisoned rods were loaded in 1983 and completed their third cycle of irradiation this summer. Some rods were withdrawn after each cycle for hot cell examinations.

c) surveillance of zircaloy-grid AFA fuel. Five assemblies from the first AFA 17 x 17 reload were characterized and examined after one and two cycles of irradiation. These assemblies are now in their third cycle. Likewise, some characterized assemblies of the AFA 14 x 14 type were examined after one cycle.

As a general rule, there will be a surveillance program for each AFA type which will be loaded over the coming years.

d) surveillance of 17 x 17 XL for 1300 MW reactors.

e) the use of reprocessed fuel. Characterized 17 x 17 assemblies containing fuel using reprocessed uranium are now in their first cycle of irradiation.

f) plutonium recycle: 16 PuO$_2$/UO$_2$ MOX assemblies will soon be loaded.

Table 3 lists the main FRAGENA product surveillance programs.
4. MAIN RESULTS

All the on-site examinations demonstrated the excellent behavior of the fuel.

Irradiation-induced rod bow moderately increased with burnup and values remained well below design limits, as shown in Figure 1.

Rod growth (Figure 2) was two to three times faster than assembly growth. In each case, a narrowing of the rod/bottom nozzle gap was first observed. Despite the scatter of certain results, the total rod/nozzle gap was nonetheless sufficient to accommodate this differential growth at the end of irradiation.

Fuel rod crud deposits were slight and samples showed that in 900 MW PWR's their thickness was only about a few microns.

Cladding oxidation remained moderate. A large number of measurements were taken on site; the results compared favorably with those yielded in hot cells, particularly by metallographic examination. The results on-site and in the hot cells were yielded by fuel irradiated over five cycles up to burnups of 55,000 MWD/MTU; the maximum measured oxide thickness was close to 60 microns, which attests to the satisfactory behavior of the cladding up to these burnup levels (Figure 3). As Table 3 shows, the ongoing surveillance programs will soon be extending the data base, especially for high burnups and increasingly severe reactor operating conditions.

Grid spring relaxation increased with irradiation but rod restraining forces still remained sufficient to prevent any cladding damage resulting from rod vibration.

The results of the AFA 17 x 17 fuel visual and dimensional examinations after one and two cycles of irradiation show a behavior pattern similar to that of inconel-grid assemblies at equivalent burnups, as shown in Figures 1 and 2.
5. TWO EXAMPLES OF APPLICATION OF RESULTS

5.1 ROD BOW LIMIT CURVE

The data acquired in power reactors from more than fifty 17 x 17 assemblies irradiated over one to five cycles were used to plot an irradiation-induced rod bow limit curve. The measured assemblies were arranged in burnup groups; the fractional channel closure only exceeded by 5% of the rod-to-rod channels was determined for the worst-case grid span. The points obtained are shown in Figure 1. The limit curve was plotted by applying a multiplier to the best-estimate curve passing through the points.

By using this curve, it was possible to reduce the DNBR penalty relative to the design methodology previously applied. The results yielded to date justify its use for AFA 17 x 17 assemblies (Figure 1).

This is an example of experience feedback to the designer.

5.2 ROD GROWTH

The results obtained on-site, mainly with 17 x 17 fuel, show some data scatter. The impact of TREX origin was demonstrated, as shown in Figure 4. This Figure shows that, at equal burnup, the rod growth in TREX group A is different from that in TREX group B.

Analysis of some of the results showed that there were differences in the pre-irradiation mechanical properties of these two groups [3]. Differences in the chemical composition of the zircaloy 4 ingots were also observed.

One of the difficulties in the analysis of these results arises from the fact that the TREX origin and therefore the manufacturing process are not the same.

Figure 3: Fuel cladding rod corrosion versus burnup

Figure 4: Fuel rod growth - Influence of TREX origin (17 x 17 fuel type)
In addition, significant growth differences were even observed in cladding tubes produced from the same ingot and irradiated in the same assembly. A test program on archived tubes was launched and showed the influence of Zircaloy carbon content on clad creepdown and irradiation-induced fuel rod growth.

This is an example of experience feedback to the manufacturer and the designer.

6. CONCLUSION

The surveillance programs implemented by FRAGEMA in close cooperation with EDF and CEA have demonstrated the excellent behavior of FRAGEMA fuel under irradiation conditions.

The results have led to the generation of an extensive data base which can now be used to upgrade the design and fabrication methods.

7. REFERENCES

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EXPERIENCE AND DEVELOPMENT OF POOL SIDE INSPECTION FACILITIES OF NUCLEAR FUEL AT LOVIISA NUCLEAR POWER STATION

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Abstract

The first examination of a spent fuel assembly in Loviisa NPS, Finland, took place in May-June 1985. By November 1987 three fuel assemblies, two shield assemblies (non-uranium bearing assemblies at the periphery of the core) and five absorber assemblies have been examined with the equipment. In addition the lengths of 15 irradiated fuel assemblies and their rods have been measured during annual refuelling outages.

The equipment has been developed for removing, inspecting and reinserting individual fuel rods. Thus, in addition to dimensional and visual inspections of the fuel assemblies with and without the shroud tube surrounding the rod bundle it is now possible to measure individual rods. During the latest examination campaign in 1987 seven rods were removed and subjected to gamma scanning and spectroscopy, profilometry and diametral gap measurements. Further irradiation of extensively examined assemblies is not planned.

In spite of some difficulties experienced in the initial phase of measuring the individual rods, the pool side inspection stand at Loviisa NPS now provides satisfactory means to observe the most important parameters in spent assemblies reflecting the fuel behaviour. The pool side inspections will play a central role in the high burnup program which has been started to demonstrate the reliable behavior of the fuel in enhanced maximum burnups foreseen in the near future.

1 INTRODUCTION

The Finnish power company Imatran Voima Oy (IVO) operates two nuclear power units with pressurised water reactors of Soviet type (VVER-440). The fuel examination techniques and experience at Finnish power plants were reported in the IAEA's Specialists' meeting in 1984 in Tokyo [1] and in IAEA's Research coordination meeting in 1985 in Vienna [2]. At that time the construction of the pool side inspection stand had just been completed. The first examination of a spent assembly took place in May 1985.
Since 1985, several different examinations have been successfully performed. The equipment has been developed for removing, inspecting and reinserting individual fuel rods. Thus, in addition to dimensional and visual inspections of the fuel assemblies with and without the shroud tube surrounding the rod bundle it is now possible to inspect and measure individual rods. During the latest examination campaign in 1987 seven rods were removed and subjected to gamma scanning and spectroscopy, profilometry and diametral gap measurements.

2 EQUIPMENT FOR POOL SIDE EXAMINATIONS

2.1 Basic equipment

Due to the closed structure of the VVER-440 fuel assembly, see figure 1, its examination involves a number of special measures in addition to actual measurements or visual examination. Following basic operations, measurements and inspections were initially available with the pool side examination stand:

(a) Visual inspection of the assemblies.
(b) Length, diameter, bowing and twisting measurements of the assemblies.
(c) Fuel rod length measurements for rods in the assembly.
(d) Measurement of spring force of the support fingers at the top of the assembly.
(e) Disassembly of the shroud tube.
(f) Visual inspection of fuel rod bundle (assembly without the shroud tube).
(g) Reassembly of the shroud tube.

All facilities for the above measures were ready for operation by the time of the first spent fuel examination in 1985.

A description of the basic equipment for the inspections above has been presented in [1] and [2].

2.2 Equipment for examining individual fuel rods

Shortly after the completion of the stand a decision was made to enhance its capabilities to allow for making the following operations and measurements to individual rods:

(a) removing individual rods from the assembly,
(b) visual inspection,
(c) profilometry,
(d) diametral gap measurements,
(e) $\gamma$-scanning measurements and
(f) reinsertion of the rod in the bundle

FIG. 1 General view of VVER-440 fuel assembly
2.2.1 Removal of rods from the assembly

A grip rod was designed and fabricated for handling individual rods.

The fuel rods in a VVER-440 fuel assembly are fixed to the bottom plate by fixing pins going through the end plugs of the rods. Thus, before pulling the rods off they have to be loosened from the bottom plate. The method to loosen the rods adopted in Loviisa is to cut off the end plugs just above the fixing pins.

In the old construction of the assembly there was also a top plate which prevented the pulling of individual rods from the bundle. This obstacle was removed by cutting off a piece of the plate around the rods to be removed.

Equipment and method was also developed to diminish the axial friction of the spacer grids when removing the rods.

2.2.2 Profilometry and diametral gap measurements

A measuring device has been developed to measure the diameter profile of individual fuel rods. The device consists of two parallel measuring fingers between which the fuel rod is placed. The distance between the fingers is measured using a LVD transducer. The measuring device can be moved along the fuel rod and the fuel rod can be rotated during the measurement.

The diametral gap between the fuel and the cladding is measured by pressing the rod between two parallel flat edges and recording the pressing force as a function of clad deformation. The pressing force is accomplished pneumatically. The so called "free" and "pressed" gaps are estimated from the shape of the curve.

2.2.3 γ-measurements

A device for γ-scanning measurements of individual fuel rods has been designed and fabricated for IVO by Asea-Atom Ab, Sweden. The mechanical part of the equipment consists of a collimator housing, collimator and a carriage for moving the rod in front of the collimator. The mechanical part, with the exception of the motors and controls for the rod carriage movement, is placed under water into the fuel pool. Schematic presentation of the equipment is shown in figure 2. The measurement system (detector, analysators, etc.) used in the first application in May 1937 of this device was owned and operated by Asea-Atom. The work is underway to furnish the apparatus with IVO's own measurement system.

FIG. 2 Schematic presentation of the pool side gamma measurement equipment
3 OPERATIONAL EXPERIENCE OF
THE POOL-SIDE INSPECTION STAND

The first examination of a spent fuel assembly took place in May-June 1985. By November 1987 three fuel assemblies, two dummy assemblies (non-uranium bearing assemblies at the periphery of the core) and five absorber assemblies have been examined with the equipment. In addition the lengths of 15 irradiated fuel assemblies and their rods have been measured during annual refuelling outages.

3.1 Operation of the inspection stand

The examinations are usually carried out by a team of 2-4 specialists, one of which is responsible for the inspections of the fuel and the others for the operation of the stand.

There has been no major problems in machining the irradiated material, disassembling and reassembling the shroud tube. Dimensions of the reassembled fuel assembly and the strength of its joints have fulfilled all requirements of handling, storing and transportation.

The fact that the equipment is used a limited time periodically, once or twice a year, leads to rather extensive overhaul and testing prior to actual examination campaigns. Failures of gaskets and rubber tubes of the pneumatically driven devices are the most common reasons for the malfunctioning of the equipment.

3.2 Examination of fuel assemblies

Essentially two types of "standard" fuel assembly examination have been performed so far.

Lengths of assemblies as well as lengths of individual rods in the assembly have been measured and will continue for some time to be measured in the fuel pool of Loviisa-2 reactor building during refuelling shutdowns.

Extensive examinations, which include removing of the shroud tube around the assembly, have been performed and are planned to be performed in the near future approximately to one assembly once a year. These examinations are carried out in the fuel pool of a spent fuel storage located in the auxiliary building of Loviisa-2.

3.2.1 Length measurements

Out of each reloading of Loviisa-2 two assemblies are chosen to fuel assembly and rod length measurements. Length of chosen assemblies and their fuel rods is measured before irradiation and once a year during the irradiation. As a result, the elongation of the assembly and the rods from year to year is obtained. This gives a rough integral picture of the mechanical behaviour of the fuel. Possible influence of design changes might also be seen comparing the elongation behaviour of two different rod design versions provided that the power histories are comparable.

Length measurements using the equipment of the pool side inspection stand have been carried out in Loviisa-2 during the refuelling outages in 1986 and 1987. Nearly 2000 fuel rod length measurements have been in this connection. The work is done simultaneously with the shuffling of the fuel assemblies in the reactor with no effect on the outage time.

3.2.2 Extensive examinations

Due to the closed structure of the VVER-440 assembly and the Finnish regularity requirement that it shall be possible to some extent to inspect irradiated fuel assemblies the pool side inspection stand was constructed. The first target was to achieve a possibility to make visual inspection of the fuel rod bundle. Later on, possibility of dimensional and gamma measurements of individual fuel rods was set as goal.

Three spent fuel assemblies have been subjected to extensive pool side examination during 1985 - 1987. The first two of these examination campaigns consisted of the basic operations, inspections and measurements enlisted in section 2.1. The latest examination included also removing of seven fuel rods from the bundle and their visual inspection, profilometry, diametral gap measurement and axial gamma-scanning and gamma-spectrum measurement in the plenum region.

3.2.2.1 Visual inspections of fuel assemblies

Visual inspection of the assemblies has been performed with the help of a periscope. All six faces of the hexagonal assembly have been systematically surveyed prior and after the removal of the shroud tube. Photographs have been taken of typical appearance of the assembly as well as of places where indications of any abnormalities could be suspected. A typical picture of a rod bundle photographed through the periscope is presented in figure 3.

3.2.2.2 Dimensional measurements of fuel assemblies

In addition to length measurements mentioned in section 3.2.1 above, length, diameter, twist and bow as well as lengths of the rods in the assembly are routinely measured for fuel assemblies subjected to extensive examination.
3.2.2.3 Removal of rods from the assembly

In May 1987 seven fuel rods were removed from a fuel assembly for rod measurements by method described in section 2.2.1. The rods were located in one row in the periphery of the assembly. Before pulling off the rods the axial force of the spacer grids was diminished for some of the rods. The rods were pulled off without difficulties.

3.2.2.4 Profilometry and diametral gap measurements

The removed rods were subjected to profilometry and diametral gap measurements. However, due to delays in the fabrication of the measurement system, the testing and tuning of the equipment prior to actual measurements was left very scarce. This led to a prolonged measurement campaign, interrupted by the holiday season and the yearly maintenance shutdown of both of the Loviisa units. In fact, the tuning of the profilometry and diametral gap measurement system has not yet been completed and is expected to last to the end of this year.

3.2.2.5 γ-measurements

The seven removed rods were axially γ-scanned with the aim of detecting possible migration of Cs to pellet boundaries as sign of fission gas release in excess of a few percents.

γ-spectrum was also measured in the plenum region of the rods in order to evaluate the amount of released $^{85}$Kr, which is used to assess the fission gas release rate quantitatively [3].

The γ-measurements were performed with the equipment and by the staff of Asea-Atom Ab, Sweden, see section 2.2.3. The equipment worked well, the only "difficulty" being low release of fission gases making it difficult to distinguish the $^{85}$Kr-peak from the spectrum.

3.2.2.6 Verification of the rod measurement results

In order verify, or "calibrate" the fuel rod measurements using the pool side inspection stand, especially assessment of fission gas release rate, four fuel rods will be sent in 1988 to Studsvik, Sweden for hot cell examinations.

3.3 Examination of Absorbers

The control assemblies of VVER-440 reactors consist of a fuel assembly on top of which an absorber is placed. During normal operation the absorbers are practically withdrawn from the reactor. When a control assembly is inserted in the reactor, the absorber replaces the fuel assembly in the core. The absorber is made of stainless steel and consists of top and bottom pieces, hexagonal shroud tube and inner tube. Hexagonal boron steel bushes are used as absorbing material.

To investigate the possibilities to prolong the lifetime of the absorbers the dimensions of five irradiated absorbers have been measured. One of the absorbers was then disassembled and inspected visually. Samples consisting of basic structural material and welds were cut and the lowest boron steel bush was removed for further mechanical and corrosion tests in a hot cell.

3.4 Examination of Shield Assemblies

The reactors of both units of Loviisa NPS are furnished with 36 shield assemblies in the peripheral positions of the core to diminish the radiation exposure of the pressure vessel. The shield assemblies are stainless steel structures with outer dimensions identical to ordinary fuel assemblies. The possibility to prolong the life time of the shield assemblies was examined simultaneously with the absorbers. In this connection the dimensions and spring forces of support fingers were measured for two irradiated shield assemblies.
An investigation program has been started to demonstrate the reliable behaviour of the fuel in enhanced maximum burnups foreseen in the near future. This investigation program includes the following pool side examinations:

(a) Three high burnup fuel assemblies (maximum rodwise burnups 43 – 48 MWd/kgU) discharged from Loviisa-2 reactor in 1986 and 1987

(b) Two fuel assemblies including 12 characterized fuel rods each. These assemblies will be removed from Loviisa-2 reactor 1990 and 1991 (with maximum rodwise burnups of ca. 48 MWd/kgU)

The examination of the first of the three assemblies mentioned in (a) above is going on at the moment and the other two will be examined in 1988. The assemblies mentioned in (b) above will be examined in 1991 and 1992. A schedule of IVO's high burnup program is presented in figure 4.

An examination of an assembly which has been removed from a reactor due to leakage of fission gases is planned to be performed during the period between the examinations of the high burnup assemblies (1989-1990).

### Table I

Summary of examinations performed using the pool side inspection stand of Loviisa NPS by October 15, 1987

<table>
<thead>
<tr>
<th>Examination</th>
<th>Number of items examined</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Fuel Assembly</td>
<td></td>
</tr>
<tr>
<td>1.1 Visual inspection</td>
<td>3 assemblies</td>
</tr>
<tr>
<td>1.2 Length of assembly</td>
<td>17 assemblies</td>
</tr>
<tr>
<td>1.3 Diameter, twist and bow of assembly</td>
<td>4 assemblies</td>
</tr>
<tr>
<td>1.4 Length of rods in the assembly</td>
<td>1691 fuel rods</td>
</tr>
<tr>
<td>1.5 Spring force of support fingers</td>
<td>9 assemblies</td>
</tr>
<tr>
<td>2. Individual Fuel Rods</td>
<td></td>
</tr>
<tr>
<td>2.1 Visual inspection</td>
<td>7 rods</td>
</tr>
<tr>
<td>2.2 Profilometry</td>
<td>7 rods</td>
</tr>
<tr>
<td>2.3 Diametral gap</td>
<td>7 rods</td>
</tr>
<tr>
<td>2.4 y-scanning</td>
<td>7 rods</td>
</tr>
<tr>
<td>3. Absorbers</td>
<td></td>
</tr>
<tr>
<td>3.1 Length, diameter, twist and bow</td>
<td>5 absorbers</td>
</tr>
<tr>
<td>3.2 Visual inspection of disassembled absorber</td>
<td>1 absorber</td>
</tr>
<tr>
<td>4. Shield assemblies</td>
<td></td>
</tr>
<tr>
<td>4.1 Length, diameter, twist and bow</td>
<td>2 assemblies</td>
</tr>
<tr>
<td>4.2 Spring force of support fingers</td>
<td>2 assemblies</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Examination</th>
<th>Burnup range (MWd/kgU) or irradiation time [y]</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.1 Visual inspection</td>
<td>22.1 – 38.7</td>
</tr>
<tr>
<td>1.2 Length of assembly</td>
<td>8.4 – 40.0</td>
</tr>
<tr>
<td>1.3 Diameter, twist and bow of assembly</td>
<td>22.1 – 38.7</td>
</tr>
<tr>
<td>1.4 Length of rods in the assembly</td>
<td>8.4 – 40.0</td>
</tr>
<tr>
<td>1.5 Spring force of support fingers</td>
<td>47.9</td>
</tr>
<tr>
<td>2. Individual Fuel Rods</td>
<td></td>
</tr>
<tr>
<td>2.1 Visual inspection</td>
<td>44.0 – 47.7</td>
</tr>
<tr>
<td>2.2 Profilometry</td>
<td>44.0 – 47.7</td>
</tr>
<tr>
<td>2.3 Diametral gap</td>
<td>44.0 – 47.7</td>
</tr>
<tr>
<td>2.4 y-scanning</td>
<td>44.0 – 47.7</td>
</tr>
<tr>
<td>3. Absorbers</td>
<td></td>
</tr>
<tr>
<td>3.1 Length, diameter, twist and bow</td>
<td>4</td>
</tr>
<tr>
<td>3.2 Visual inspection of disassembled absorber</td>
<td>4</td>
</tr>
<tr>
<td>4. Shield assemblies</td>
<td></td>
</tr>
<tr>
<td>4.1 Length, diameter, twist and bow</td>
<td>5</td>
</tr>
<tr>
<td>4.2 Spring force of support fingers</td>
<td>5</td>
</tr>
</tbody>
</table>
ON-SITE UNDERWATER EXAMINATION TECHNIQUES FOR BWR FUEL ASSEMBLIES

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Abstract

On-site fuel examination plays important roles for establishing the data base of fuel irradiation performance under reactor operating conditions. The data base up to extended burnups is essential for the development of future high burnup fuel. On-site underwater examination techniques and devices for visual inspection, dimensional measurement, oxide thickness measurement, and fission gas release measurement have been developed. The paper presents these techniques and devices.

1. INTRODUCTION

The efforts on the improvements of LWR fuel design have been focused on extending discharge burnup to reduce power generation costs. It is essential for implementing the improvement that the assessment of fuel irradiation performance under reactor operational conditions based on the extensive data up to the extended burnups should be done to assure the reliable design and operation of future high burnup fuel design.

The areas of concern with extended burnup include dimensional change, corrosion of cladding and spacer, fission gas release, and overall fuel performance. To establish the data base for these areas, on-site fuel examinations could play important roles. The on-site examination is rather convenient compared to hot laboratory examination and has an advantage to be able to examine same fuel assembly with exposure at every refueling outage. Non-destructive examination techniques have been developed for detailed surveillance of fuel assemblies in a spent fuel pool.

2. ON-SITE UNDERWATER EXAMINATION TECHNIQUES AND DEVICES

The on-site examinations are carried out in the spent fuel pool using the existing Fuel Preparation Machine (FPM) which serves for axial movement and rotation of an assembly, and the devices developed by Toshiba and NAIG.

REFERENCES

The devices are roughly classified into three categories. These are as follows; (1) devices for the examinations of fuel assembly and fuel rod without disassembling, (2) devices for the examinations of individual fuel rod after disassembling, and (3) disassembling and reassembling tools. Table 1 provides a list of the major devices for visual inspection, dimensional measurement, oxide thickness measurement, and fission gas release measurement without disassembling. These techniques and devices are herein described.

2.1 Underwater Color TV System for Visual Inspection

The system consists of a TV camera installed in a watertight housing, a positioning device, a recording unit, and a control unit. A high resolution camera is used with a combination of a TV zoom lens, a teleconverter and a close-up lens. A fuel assembly can be thereby observed in arbitrary magnifications from the full-width of the fuel assembly to the close-up of one fuel rod. Video signal from TV camera is transferred to a video tape recorder and simultaneously fixed on photo film through frame memory and hard copy. Figure 1 shows a schematic view of setup of the device in the spent fuel pool. A BWR 8x8 fuel assembly is placed on the FPM. The positioning device has four rollers which contact pool wall, maintaining constant geometry between itself and the FPM. The TV camera can move horizontally and thus, close-up of each fuel rod at the outside row of an assembly can be easily observed, providing detail informations on the appearance of the fuel rod. The coverage of nodular corrosion can be estimated through the observation.

2.2 Dimensional Measurement Technique

The dimensional measurement includes overall assembly length, individual fuel rod length, and expansion spring length measurements. The definitions of the measured values are given in Figure 2. A steel tape measure attached at one end to a fuel assembly handle provides the vertical position data of an object being observed on the TV monitor. Spacer axial locations and the distance between a lower tie plate (LTP) and an upper tie plate (UTP) (assembly length) are measured by the tape. Fuel rod length is determined through the measurement of the distance between the top of fuel rod upper end plug and the upper surface of the UTP by using a plunge-gage technique. The measuring device is placed onto the UTP. Figure 3 illustrates the measurement method. A small sensor rod moves horizontally to locations to be measured, and then, moves down vertically onto the upper surface of the UTP or the top of the upper end plug. The moving distance is indicated through a LVDT (Linear Variable Differential Transformer). The measured data show relative fuel rod growth. From this data and the assembly length data, the fuel rod length is derived. The lengths of almost all fuel rods can be thus measured without removing the fuel rods from the assembly. The applicability of the method was proven against the depth-micrometer measurement. The plunge-gage measurement is also used to determine the length of an expansion spring.

Table 1: Major Devices for Underwater Fuel Examinations without Disassembling

<table>
<thead>
<tr>
<th>Device</th>
<th>Scope</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Visual Inspection</strong></td>
<td></td>
</tr>
<tr>
<td>* Color TV System</td>
<td>Visual Inspection of Upper Tie Plate</td>
</tr>
<tr>
<td>- high resolution camera</td>
<td>Peripheral Fuel Rod, Expansion Spring</td>
</tr>
<tr>
<td>- positioning device</td>
<td>- General Appearance, Integrity</td>
</tr>
<tr>
<td>- recording unit</td>
<td>- Surface Deposit, Peripheral Rod to Rod Spacing</td>
</tr>
<tr>
<td>- control unit</td>
<td></td>
</tr>
<tr>
<td><strong>Dimensional Measurement</strong></td>
<td>Measurement of Assembly Length, Axial Location of Spacer</td>
</tr>
<tr>
<td>* Tape Measure and Color TV System</td>
<td>Measurement of Lengths of Almost All Fuel Rods in Assembly and Expansion Spring</td>
</tr>
<tr>
<td>* Plunge Gage Device</td>
<td>- plunge gage with LVDT</td>
</tr>
<tr>
<td>- positioning device</td>
<td>- control unit</td>
</tr>
<tr>
<td>* Devices for Cladding and Spacer</td>
<td>Measurement of Oxide Thickness of Peripheral Fuel Cladding and Outside of Spacer</td>
</tr>
<tr>
<td>- eddy-current probe</td>
<td></td>
</tr>
<tr>
<td>- positioning device</td>
<td></td>
</tr>
<tr>
<td>- reference cladding and spacer sample</td>
<td></td>
</tr>
<tr>
<td>- control unit</td>
<td></td>
</tr>
<tr>
<td><strong>Oxide Thickness Measurement</strong></td>
<td></td>
</tr>
<tr>
<td>* Devices for Cladding and Spacer</td>
<td>Gamma-ray Spectrometry of Kr-85 in Plenum Area of Fuel Rod at Corner of Assembly</td>
</tr>
<tr>
<td>- high resolution Ge detector with associated electronics</td>
<td></td>
</tr>
<tr>
<td>- collimator and air-box with adjustable stroke</td>
<td></td>
</tr>
</tbody>
</table>
Figure 1: Schematic View of Visual Inspection Device Setup.

Figure 2: Dimensional Measurements.

Figure 3: Measurement Method of Distance between Top of Upper End Plug and Upper Surface of Upper Tie Plate.
2.3. Corrosion Oxide Thickness Measurement Technique

An eddy-current technique is used to measure the oxide thickness of a fuel cladding and a spacer, based on an eddy-current lift-off principle. Figure 4 illustrates the measurement method. The device is placed to the region to be measured, by holding an assembly with an air-actuated clamp arms in order to set a probe on fuel rod or spacer surface with good geometrical reproducibility.

The device for fuel cladding measurement has two probes of which diameter and excitation frequency are 6 mm and 4 MHz, respectively. The probes can move both horizontally and vertically, and the outside row of fuel rods in the assembly can be thereby scanned axially between spacers. The output voltage of the probes are transferred to electronics and recorded on a tape recorder and a chart recorder. The device has two short standard fuel claddings coated with 30 to 150 microns thick zirconia for calibration. Thus, the axial distributions of the oxide thickness can be obtained for the peripheral fuel rods of the assembly.

The device for spacer measurement has one probe of which diameter and excitation frequency are 3 mm and 3 MHz, respectively. A standard sample piece cut from a fresh spacer is attached to the device as a reference. The surface shape of spacer band is uneven and eddy-current signal is quite sensitive to a surface shape of metal. Both the spacer to be measured and the standard piece are horizontally scanned. The difference between two eddy-current signals corresponds to the oxide thickness of the outside of the spacer.

The verification of the method was made through the comparison between the eddy-current measurement and the metallographical measurement of fuel cladding samples coated with zirconia. The results of the comparison are given in Figure 5, showing good agreement between two measurements.

![Figure 4](image1.png)

**Figure 4** Measurement Method of Corrosion Oxide Thickness of Fuel Rod.

![Figure 5](image2.png)

**Figure 5** Comparison of Oxide Thickness of Fuel Cladding Coated with Zirconia Measured by Eddy-Current Technique and Metallographical Method.

2.4. Plenum Gamma-ray Spectrometry

Gamma-ray spectrometry of Kr-85 in the plenum area of the fuel rod is applied to the determination of fission gas release in the fuel rod. The measurement system consists of a high resolution Ge detector with associated electronics, a heavily shielded detector housing, an adjustable collimator.
and a data acquisition system. Figure 6 shows the schematic drawing of the collimator and the detector housing. The collimator is so designed that a fuel rod at an assembly corner can be measured without removing a channel box and any fuel rods from the assembly. The collimator has an air-box with adjustable stroke. After positioning the assembly on the FPM, the air-box is lengthened to the point that a front shield contacts the channel box of the assembly. Thus, gamma-ray from fuel rods other than the corner fuel rod can be shielded. Since the gamma-ray detection system is operated at high count rates, counting losses occur in the system. To determine the counting losses, the electronics include a random pulse generator which provides a peak of known input count rate.

The concentration of Kr-85 in the plenum area in the fuel rod is derived from the measured 514 keV gamma-ray count rate from the well-defined region of the plenum. The counting efficiency of the system was measured with a well-characterized Kr-85 source. The efficiency is also estimated through calculation based on a mono-energy photon transport theory. The intrinsic efficiency of the Ge detector experimentally determined was used in the calculation. The results are summarized in Table 2, showing that the calculation can estimate the efficiency with high accuracy. The measured Kr-85 concentration is related to the amount of fission gas released by using void volume from fuel rod design or fabrication data and the ratio of (Kr*Xe)/(Kr-85) from nuclear calculations taken into consideration the irradiation history of the fuel rod.

### Table 2 Measured and Calculated 514 keV Gamma-ray Counting Efficiency of the System of Plenum Gamma-ray Spectrometry.

<table>
<thead>
<tr>
<th>Emission Rate of 514 keV Gamma-Ray (1/sec)</th>
<th>Count Rate Efficiency</th>
</tr>
</thead>
<tbody>
<tr>
<td>Measurement (E)</td>
<td>(counts/sec)</td>
</tr>
<tr>
<td>1.52x10^5</td>
<td>1.15x10^{-1}</td>
</tr>
<tr>
<td>(±3 %)</td>
<td>(±0.8 %)</td>
</tr>
<tr>
<td>Calculation (C)</td>
<td>1.74x10^{-7}</td>
</tr>
<tr>
<td>Difference ((C-E)/E x 100)</td>
<td>7.44x10^{-7}</td>
</tr>
</tbody>
</table>

(Note) Value in parenthesis = Uncertainty (1σ)

### CONCLUDING REMARKS

A BWR fuel burnup extension program has been in progress as a joint program of BWR utilities and plant makers in Japan to accumulate fuel data necessary for the development of high burnup fuel through on-site fuel examinations. The on-site fuel examination techniques and devices described in this report have been successfully applied to the examinations.

### REFERENCES


ON-SITE BWR FUEL EXAMINATION TECHNIQUES AND RESULTS

R. SEEPOLT, N. EICKELPASCH
Kernkraftwerke Gundremmingen
Betriebsgesellschaft mbH, Gundremmingen, Federal Republic of Germany

Abstract

Onsite Fuel Examination Programs are performed on a regular basis at the KRB Gundremmingen nuclear power station. The fuel examination programs include: incore sipping testing for leak tightness, visual examination, measurement of the fuel bundle dimensional and corrosion behavior, sampling and analysis of crud deposits. The KRB Gundremmingen nuclear power station consists of two 1250 MW net BWR units, supplied by KWU Kraftwerk Union AG. The reactors are loaded with advanced fuel of 9x9 assembly lattice geometry. For the purpose of onsite inspection of fuel and core components the units are equipped with fuel pond side wall inspection windows. Ongoing fuel examination programs are performed for two main reasons: to comply with the operation license requirement and to follow a lead test assembly irradiation program. The fuel inspection results did not reveal any detrimental effect regarding incore performance of 9x9 fuel. The distinctive behavior of beta quenched cladding with respect to Zircaloy corrosion and irradiation induced growth was again confirmed. A fuel channel dimension surveillance program is established. The reuse of irradiated channels is definitely planned. Channel bulge and channel bow evaluation results for 3.05 mm (20 mil) channels are reported. Crud samples from fuel surfaces were taken and analyzed. Iron input via the reactor feedwater of 4-5 ppb resulted in maximum local crud layer thickness of up to 60 µm without detrimental effect on fuel performance.

Introduction

Onsite fuel examination programs are performed on a regular basis at the KRB Gundremmingen nuclear power station. The examination programs include visual inspection of fuel assemblies and related core components, e.g. fuel channels and control rods. Furthermore fuel assembly performance is surveyed by in-core sipping testing, by measurement of the fuel bundle dimensional and corrosion behavior and by sampling and analysis of corrosion product deposits.

Particular investigation programs are contracted with the fuel suppliers (KRB-B: KWU; KRB-C: ANF) and are conducted in cooperation with them.

The KRB nuclear power station consists of two BWR units, KRB-B and KRB-C, 1250 MW each. The units were designed and constructed by Kraftwerk Union AG and are jointly owned by the utilities RWE AG and Bayernwerk AG.

Both reactors went in commercial operation in 1964. KRB-B is now operating in its 4th reactor cycle and KRB-C in its 3rd cycle.

Provisions for fuel and core component examinations were given attention already during the design phase of the plant. Both fuel pools are equipped with side wall lead glass windows, which allow for visual examination measures in a timely fashion even during reactor refueling shutdowns (Fig. 1).
Systematic inspection of fuel bundles in the plant can result in preventive measures in order to avoid fuel failures.

The identification of fuel failure causes and the repair of defect fuel assemblies are important goals for operational and economic reasons as well:

- Minimization of fission product release into the reactor coolant during operation is the basis for minimizing the contamination of the plant systems and subsequently minimizing the personal radiation exposure and the activity releases - if any - as well.

- The waste disposal costs are an essential part of the fuel cycle costs. This portion is specifically increasing with increasing burnup [1]. Therefore it is economical to repair fuel assemblies in order to reach the planned discharge burnup and in order to reach a discharge burnup as high as possible.

2 Fuel Examination Techniques and Results

The examination techniques which are in use on a regular basis at the KRB plant are listed in Table I.

<p>| TABLE I |</p>
<table>
<thead>
<tr>
<th>FUEL EXAMINATION TECHNIQUES AT KRB</th>
</tr>
</thead>
<tbody>
<tr>
<td>- Sipping Test</td>
</tr>
<tr>
<td>- Visual Examination</td>
</tr>
<tr>
<td>- Dimensional and Geometric Inspection</td>
</tr>
<tr>
<td>- Crud Deposit Measurement and Analysis</td>
</tr>
<tr>
<td>- Zr-Corrosion Measurement*</td>
</tr>
<tr>
<td>- E.C. Leakage Detection*</td>
</tr>
<tr>
<td>- U.S. Inspection*</td>
</tr>
</tbody>
</table>

2.1 Fuel Design

The major fuel design parameters of the KRB fuel are summarized in Table II.

In view of the comparatively high specific core power (28 kW/kg U) and unrestricted load follow capability of the units as a major plant design goal, a 9x9 fuel development program was initiated early. As a result of this the initial cores were already loaded with larger portions of the new fuel type. Now it becomes the standard reload fuel for both reactors.

<p>| TABLE II |</p>
<table>
<thead>
<tr>
<th>DESIGN PARAMETER OF KRB FUEL ASSEMBLIES</th>
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<tbody>
<tr>
<td>Assembly</td>
</tr>
<tr>
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</tr>
<tr>
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</tr>
<tr>
<td>Total Weight</td>
</tr>
<tr>
<td>UO₂ Weight</td>
</tr>
<tr>
<td>Enrichment</td>
</tr>
<tr>
<td>Average Linear Heat Rating</td>
</tr>
<tr>
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</tr>
<tr>
<td>9x9</td>
</tr>
<tr>
<td>8x8</td>
</tr>
<tr>
<td>4.47 m</td>
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<tr>
<td>14.3 mm²</td>
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<tr>
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</tr>
<tr>
<td>10.02 mm²</td>
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<tr>
<td>8.89 mm²</td>
</tr>
<tr>
<td>300 kg</td>
</tr>
<tr>
<td>312 kg</td>
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</tr>
<tr>
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<td>158.4 W/cm²</td>
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</tr>
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</tr>
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</tr>
<tr>
<td>He Pressure</td>
</tr>
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</tr>
<tr>
<td>80</td>
</tr>
<tr>
<td>62</td>
</tr>
<tr>
<td>3710 mm</td>
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<td>318 mm</td>
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<tr>
<td>324 mm</td>
</tr>
<tr>
<td>6.5 bar</td>
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<td>6.5 bar</td>
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<table>
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<table>
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<tbody>
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<td>Length/ Diameter</td>
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<tr>
<td>Material</td>
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<tr>
<td>Density</td>
</tr>
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</tr>
<tr>
<td>9.11 mm</td>
</tr>
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<td>9.21</td>
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<td>10.44 mm</td>
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<td>1.20</td>
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<tr>
<td>UO₂ (UO₂ + 0.62O₂)</td>
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<table>
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<th>Cladding (Water Rod Tube)</th>
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<td>Material</td>
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<td>---------------------------------------------------------------</td>
</tr>
<tr>
<td>13 mm</td>
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<td>0.55 mm</td>
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</tr>
<tr>
<td>7</td>
</tr>
<tr>
<td>7</td>
</tr>
<tr>
<td>Zircaloy-4/Inconel X 718</td>
</tr>
</tbody>
</table>

* Reloading Fuel (9x9 only)
2.2 Fuel Examination Equipment and Examination Results

2.2.1 Sipping

To check the leak tightness of the fuel assemblies, after each completed reactor cycle an in-core sipping test is carried out with the entire core (784 bundles).

The time required for a sipping campaign often influences directly the duration of the refueling outage. Therefore KRB uses a multiple bundle (8-fold) sipping equipment.

A total core sipping (784 fuel bundles) requires approximately 50 hours.

A computerised evaluation method of the fission product concentration in the water sample, mainly I-131/135, Cs-134/137 and Xe-133, is used in KRB.

The background I-131-activity must not exceed $1 \times 10^4$ Bq/m$^2$ in order to reach an adequate sensitivity.

2.2.2 Visual Examination

The main tool for the onsite fuel examination is the "hot-cell" window, which is located at a side wall at each of the spent fuel storage pools. This window together with the appropriate equipment for fuel handling, e.g. lifting, lowering, rotating, allows for thorough surface visual inspection.

The following pictures which were obtained during recent interim reactor refuelling shutdowns are descriptive examples for major findings.

Fig. 2 shows the top piece of a 9x9 bundle. Interference of a locking tab and the thread of an end plug due to force onto the bundle via the fuel grab device resulted in a faulty elongation of the tie rod. An immediate design modification of the locking tab eliminated the problem.

Fig. 3 illustrates the appearance of the newly designed 9x9 fuel rod spacer with respect to corrosion. Zirconium oxide nodules can be seen distributed over the plate surface, except for the circular areas surrounding the spot welds, which are not corroded at all.

Fig. 4 compares corrosion behaviour of beta-quenched cladding (center three rods) with that of standard cladding material. The beta-quenched rods appear consistently shiny and free of corrosion, while the adjacent standard-clad rods exhibit a uniform greyish appearance resulting from white oxide formation.

Fig. 2 KRB Fuel Bundle Top Piece (9x9)
Fig. 3 Spacer of a KRB Fuel Bundle (9x9)
Fig. 4 Nodular Corrosion at BWR Fuel Bundle (KRB)
Fig. 5 gives an illustration of the distinctive behavior of the beta-quenched fuel cladding material with respect to growth in comparison to standard material. The photograph shows the gaps provided for fuel rod expansion at the upper end of the assembly. Obviously the gaps are largest for the three center rods of this precharacterized bundle, meaning the growth is least for the beta-quenched cladding tubes.

2.2.3 Dimensional Inspection of Fuel Channels

The incore service time of BWR fuel channels can be extended to cover a second fuel assembly lifetime. To attain this goal a channel management program is necessary, which is depending upon both measured fuel channel data and a validated predictive channel deformation model as well.

For this reason fuel channel measurement campaigns are performed in the time between refueling outages. Those irradiated channels are measured which will be reinserted into the core with the next fuel reload batch. The number of fuel channels to be measured is determined by a 1 out of 4 scheme (quarter core loading symmetry). With reload batches of around 200 fuel bundles per year and reactor, approximately 100 channels have to be measured annually at KHS. Therefore a fast measuring device is required. The commercial equipment frequently in use is of AREVA-ATOM design and manufacture and has been described in more detail elsewhere [2].

As of today 155 channels were measured in 3 campaigns. Measured values are quantitatively evaluated for channel bulge, bow, twist, displacement and length.

Results of the bow and bulge evaluation for the 3.05 mm (120 mil) channels of KRB-C are presented in Fig. 6 and Fig. 7.

2.2.4 Corrosion Product Deposits (Crud)

Iron input via the feedwater into the reactor is responsible for the crud deposition at the fuel surface of BWRs and - as a consequence - for the radiation field at the primary system components [3]. Especially for BWR plants with forward pumped heater drains or deep bed condensate deionisers this input can be considerable, however it contributes no detrimental effect on fuel performance.

As measured iron-input concentration of 4-5 ppb in the feedwater resulted in crud layer thicknesses between 20 and 60 µm.

Crud sampling is performed utilizing an in-house designed and constructed, remotely controlled underwater device, Fig. 8.
3 Conclusion

Onsite examination of fuel and related core components can provide the reactor operator and the component supplier with early information regarding potentials for design improvements.

Onsite examination can avoid costly hot cell post irradiation examination.

Onsite examination is contributing to the economic utilisation of core components and thus reducing radioactive waste generation.

References


EXPERIENCE WITH THE BROWN BOVERI FAILED FUEL ROD DETECTION SYSTEM (FFRDS)

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Columbia, Maryland,
United States of America

Abstract

The Brown Boveri ultrasonic system for detection of failed fuel rods is described and data is presented from 57 campaigns where nearly 800,000 fuel rods were inspected. Comparisons with independent eddy current measurements are made for approximately 9500 fuel rods. The database developed from these inspections is used to present failure data for 17 different fuel designs. Projections of potential reductions in radiation levels during subsequent nuclear plant operations are given. Finally an estimate is made of the amount of defective fuel that must be stored in the U.S. and which will eventually go to a repository.

INTRODUCTION

Detection of failures in nuclear reactor fuel assemblies has been accomplished by a number of techniques, including sampling the reactor coolant system during operations, sipping each assembly after discharge, eddy current testing of removed fuel rods, and ultrasonic inspection. An advanced ultrasonic system developed by Brown Boveri, the Failed Fuel Rod Detection System (FFRDS), has been utilized to identify individual failed fuel rods. High reliability has been demonstrated in the rapid inspection of assemblies in the U.S. and overseas utilizing the Brown Boveri system. Since initial operations began in 1979, the Brown Boveri Failed Fuel Rod Detection System (FFRDS) has been used to inspect about 800,000 fuel rods in 57 separate inspection campaigns worldwide. Approximately 75 percent of the inspections have been performed in the United States. Table I provides a summary of the Brown Boveri experience.

THEORY OF FFRDS OPERATION

The Brown Boveri FFRDS quickly inspects the individual rods in a fuel assembly and identifies, with a high degree of accuracy, those rods with through-wall defects. The ultrasonic system is incorporated in a remotely controlled automatic manipulator positioned underwater in a spent fuel pool or reactor cavity. All inspections are accomplished without having to disassemble the fuel bundle. The basic concept utilizes a “pitch and catch” ultrasonic signal technique to detect the presence of water in each defective fuel rod.

---

**TABLE I**

BBC BROWN BOVERI NUCLEAR FUEL INSPECTION EXPERIENCE
United States Experience

<table>
<thead>
<tr>
<th>Date</th>
<th>Nuclear Plant</th>
<th>Fuel Design/ Assys. Rods</th>
<th>Supplier</th>
<th>Inspected</th>
<th>Inspected</th>
</tr>
</thead>
<tbody>
<tr>
<td>2-3/83</td>
<td>Surry</td>
<td>PWR 15x15/W</td>
<td>95</td>
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<td></td>
</tr>
<tr>
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<td>Millstone-2</td>
<td>PWR 14x14/W (CE-type)</td>
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<td>5,632</td>
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<td>2,448</td>
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</tr>
<tr>
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<td>Calvert Cliffs</td>
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<td>D. C. Cook</td>
<td>PWR 17x17/W</td>
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#### OVERSEAS EXPERIENCE

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<td>1(a)</td>
<td>204</td>
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<td>2/83</td>
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<td>PWR 14x14/Fram</td>
<td>104</td>
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### TABLE I (cont.)

<table>
<thead>
<tr>
<th>Date</th>
<th>Nuclear Plant</th>
<th>Fuel Design/ Supplier</th>
<th>Assys. Inspected</th>
<th>Rods Inspected</th>
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<td>11-12/85</td>
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<td>PWR 14x14/Fram</td>
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<td>9/86</td>
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<td>PWR 17x17/BFR</td>
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<td>Doel-2/Belgium</td>
<td>PWR 14x14/Fram</td>
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<td>9/87</td>
<td>Maanshan/Taiwan</td>
<td>PWR 17x17/W</td>
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**OVERSEAS & U.S. TOTAL**

<table>
<thead>
<tr>
<th>Date</th>
<th>Nuclear Plant</th>
<th>Fuel Design/ Supplier</th>
<th>Assys. Inspected</th>
<th>Rods Inspected</th>
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<tr>
<td></td>
<td></td>
<td></td>
<td>Subtotal:</td>
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<td></td>
<td></td>
<td>208,021</td>
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**NOTES:**

(a) out-of-pile qualification  
(b) in-pile qualification  
(c) out-of-pile demonstration
The FFRDS (Figure 1) has a flexible two-finger ultrasonic probe with separated transmitter and receiver piezoelectric crystals. This probe is remotely moved along a row of fuel rods by a watertight manipulator, while a recording system provides a real-time hard copy recording of the ultrasonic signals received through each rod.

The probe is usually inserted near the bottom of the assembly above the lower spacer grid (or "skirt"). If necessary, the inspection may also be conducted at other vertical positions along the fuel assembly.

When only water is present between transmitter and receiver, this is indicated by a certain amplitude and signal transit time of the received ultrasonic pulse (see IT and TT of Figure 2). However, when a fuel rod is encountered the signal transit time will be reduced due to the increased propagation velocity of the ultrasonic waves in the metal rod compared with only water. In addition, the amplitude of that signal is lowered due to scattering of ultrasonic waves at the fuel rod/water interfaces. If the rod is defective i.e., if it contains some water, then the amplitude of the signal is reduced still further.

a) Gas-filled, defect-free rod
b) Water-filled, leaking rod

Figure 1 - Ultrasonic Probe Traverses a Row of Fuel Rods

Figure 2 - Ultrasonic Fuel Inspection Technique

IT = initial transmitter pulse
RW = reflected through-wall pulse
TT = through-transmission pulse
T = transmitter
R = receiver

Evaluation range

---

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T = transmitter
R = receiver

Evaluation range

---
Further by the additional scattering caused by water inside the rod. Thus, failed fuel rods are readily detected by simply recording the reduced amplitude of the transmitted signal.

When the ultrasonic probe is moved along a fuel rod row, two discrete echoes are received for every fuel rod, the first from the front half of the fuel rod (see NW1 of Figure 2), and the second from the back half (see NW2 of Figure 2). This is not the case with other ultrasonic system designs which utilize a transmitter and receiver mounted on a single probe arm. With these other systems only a single through-wall echo is used for evaluation at each rod, with no redundant signal as a backup verification. Also, only a limited circumferential inspection is possible with these single probe arm systems, and much more accurate alignment of the probe is required. Alignment becomes even more crucial for those systems with multiple arms employing a transmitter and receiver mounted on each arm, especially when guide tubes much larger than the fuel rods are encountered.

The results of the inspection are plotted on an X-Y recorder and simultaneously displayed on a monitor for the operator. In addition, the analog ultrasonic signals are permanently recorded utilizing the X-Y plotter as the inspections are performed, and real-time analysis of data permits immediate, positive identification of failed rods. Two discrete pairs of signals are recorded in each direction as the probe passes in and out of the row of rods. In the outward movement of the probe, the signal is traced back over the data previously recorded during the inward movement, thus providing four signals for each rod and increasing the reliability of the results.

Figure 3 shows a typical ultrasonic signal plot for a 14 x 14 PWR fuel assembly. Defective rods can be clearly seen at positions B-12 and D-8. Similar data are obtained for BWR fuel.

The ultrasonic probe is flexible enough to pass by the thicker guide tubes inside the fuel rod bundles of some fuel assemblies. This insures that the fuel assemblies being inspected are not damaged. As additional protection, an overload protection device has been installed in the mechanical system. The motion of the probe and the inspection operations are electronically controlled in both the automatic and manual modes. During inspection of one row of fuel rods, any attempted manual movement of the ultrasonic probe to another row is prevented by a fail-safe interlock system. In the automatic mode, the probe will be automatically indexed to each successive row, returning to the starting position after completing inspection of the entire assembly. Measurements across one face of a PWR fuel assembly are completed in 4 to 6 minutes. Smaller BWR assemblies are measured proportionately faster. If time permits, measurements of two faces at 90 degrees provide the highest assurance of detecting defective rods.

Further details on the FFRDS are provided in References 1 and 2.
of detecting even partial through-wall failures (although to do so requires removal of the rods from the assembly), this technique is extremely accurate. In every discrepancy noted, the FFRDS was shown to be correct, and the conventional sipping was wrong. The sipping not only missed leaking assemblies but also incorrectly identified sound assemblies as leakers.

In addition to the question of accuracy, conventional sipping systems have inherent disadvantages compared to the FFRDS ultrasonic technique. These disadvantages arise in sipping because active leakage of fission products is required. As a result, sipping cannot be used reliably to detect failures in either low burnup or long-cooled fuel. Sipping is also susceptible to the detection of “false positives” when tramp uranium is present on the fuel assembly, or when inadequate sipping container flushing results in cross-contamination. In addition, a fundamental shortcoming of sipping is that with bulk measurements, no specific identification can be made of individual leaking rods, as with the Brown Boveri FFRDS. By only detecting the presence of water in fuel rods, independent of any radioactivity, the FFRDS avoids all of the disadvantages of sipping.

In a laboratory sensitivity measurement, Brown Boveri tests have shown the capability of the FFRDS to detect as little as 1/2 gram of water in a single fuel rod. Leakers are also routinely detected in fuel as it is being off-loaded from a reactor core. Finally, since the FFRDS can only detect water and not radioactivity, the results are independent of burnup, external contamination and the amount of time the fuel may have been in storage.

4. Eddy Current Comparison

A number of direct comparisons have been made between FFRDS and eddy current on individual fuel rods (cf. Ref. 3). Time-consuming eddy current examinations were performed by organizations other than Brown Boveri on fuel rods that were being reconstituted. Table II summarizes the results of five separate comparison campaigns of fuel examinations, with overall agreement being 99.96 percent. No eddy current measurements are possible without removal of rods from the fuel assembly, whereas essentially equivalent accuracy can be achieved with the FFRDS with no disassembly required. The distinct advantage of not disassembling the fuel assemblies, beyond the obvious savings in time and costs, is that fuel assembly repair (reconstitution) can be done independently from inspection. If spare fuel assemblies are available to substitute for defective ones, then repairs can be performed off the critical path of a refueling outage.

5. Experience with the FFRDS

Table I lists worldwide usage of the FFRDS. A total of 3739 fuel assemblies comprising 793,947 fuel rods have been inspected to date, starting in 1979. A total of 984 defective fuel rods have been identified, representing 0.02% of those inspected, with an average of nearly 2 defective rods per defective fuel assembly. About 13% of all fuel assemblies inspected are defective. Inspections have been performed for essentially all types of PWR designs in the U.S. and overseas (West Germany, France, Belgium, Japan, Taiwan, and Korea). BWR fuel has also been inspected. Fuel assemblies of 17 different designs by 5 different manufacturers have been inspected.

Each of the following fuel characteristics has been successfully inspected by the FFRDS:
- Long cooling (up to 15 years)
- Low burnup (less than 30 MWD/MTU)
- High burnup (30-40,000 MWD/MTU)
- Hairline cracks or pinhole leaks
- Missing end caps with all fission gases escaping
- Significant crud levels on cladding
- Pellet-clad interaction
- Externally contaminated with fission products or tramp uranium

TABLE II

<table>
<thead>
<tr>
<th>Date</th>
<th>Plant</th>
<th>Fuel Type</th>
<th>Both UT &amp; E.C.</th>
<th>No. Rods</th>
<th>No. Rods With UT &amp; E.C.</th>
<th>Discrepancies Agreement</th>
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<tr>
<td>10/83</td>
<td>Millstone-2</td>
<td>CE/14x14</td>
<td>5000</td>
<td>2</td>
<td>100%</td>
<td></td>
</tr>
<tr>
<td>03/85</td>
<td>Millstone-2</td>
<td>CE/14x14</td>
<td>171</td>
<td>0</td>
<td>100%</td>
<td></td>
</tr>
<tr>
<td>10/85</td>
<td>San Onofre-3</td>
<td>CE/16x16</td>
<td>335</td>
<td>0</td>
<td>100%</td>
<td></td>
</tr>
<tr>
<td>09/85</td>
<td>Surry 1 &amp; 2</td>
<td>W/15x15</td>
<td>4000</td>
<td>2</td>
<td>99.95%</td>
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<td>07/86</td>
<td>ANO-2</td>
<td>CE/16x16</td>
<td>12</td>
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<td>TOTALS</td>
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<td></td>
<td>9518</td>
<td>4</td>
<td>99.96%</td>
<td></td>
</tr>
</tbody>
</table>

* All independent measurements performed by organization other than Brown Boveri under separate contract with utility.
** All ultrasonic measurements performed by Brown Boveri with FFRDS
Table III summarizes the results of the FFRDS inspections which involved a significant number of fuel assemblies of each type in use. Where insufficient data were available on a given fuel type, these were omitted from Table III.

### TABLE III

**Integrity of Nuclear Fuel**

**Results of FFRDS Inspections by Fuel Design Type**

<table>
<thead>
<tr>
<th>Fuel Design Type</th>
<th>Radiation Level</th>
<th>Avg. Defective Rods/Defective Assemblies</th>
<th>Number of Assemblies Inspected</th>
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<tr>
<td>KWU/7x7 (a)</td>
<td>7.3</td>
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<td>-</td>
<td>2.8</td>
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<tr>
<td>Fram/17x17</td>
<td>1.3</td>
<td>-</td>
<td>4.6</td>
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<tr>
<td>KWU/15x15</td>
<td>0.98</td>
<td>-</td>
<td>2.0</td>
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<tr>
<td>CE/14x14 (Exxon) (b)</td>
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<td>KWU/16x16</td>
<td>0.72</td>
<td>3100</td>
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<td>1900</td>
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<td>CE/16x16</td>
<td>0.15</td>
<td>650</td>
<td>2.2</td>
</tr>
<tr>
<td>B&amp;W/15x15</td>
<td>0.15</td>
<td>650</td>
<td>1.8</td>
</tr>
<tr>
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<td>0.14</td>
<td>600</td>
<td>1.7</td>
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<td>W/14x14</td>
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<td>CE/16x16 (Exxon) (b)</td>
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<td>0.016</td>
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<td>1.5</td>
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B&W - Babcock & Wilcox, U.S.
CE - Combustion Engineering, U.S.
Fram - Framatome, France
KWU - Kraftwerk Union, West Germany
W - Westinghouse, U.S.
Exxon - Advanced Nuclear Fuel, U.S.

**Notes for Table III**

(a) BWR, all others PWR
(b) CE design, Exxon fabrication
(c) Westinghouse design, CE fabrication
(d) Westinghouse design, B&W fabrication; Stainless steel clad, all others Zirc-4 (PWR) or Zirc-2 (BWR)
(e) Radiation levels could be reduced by this percentage, using the results of Reference 7. Five fuel types not considered as representative sample (see text).

The first five fuel types listed in Table III with relatively high percentages of fuel rod defects, were inspections where the fuel was preselected as being defective. This high percentage of defective rods is not obtained from a representative sample of these fuel types.

### 6. Spent Fuel Considerations

Proper long-term management of spent fuel also requires inspection for failed fuel. ALARA considerations dictate that spent fuel pools be kept as free of fission product activity as practicable. This can be achieved by a systematic program of inspection of all discharged fuel and isolation of defective assemblies. The most efficient method of doing this is with the Brown Boveri FFRDS which can identify individual failed rods, even for low burnup and/or long-cooled spent fuel. Even if no action other than the isolation of the failed assembly is taken in the short term, the utility will know which individual fuel rods must eventually be removed from the assembly prior to consolidation and disposition of the spent fuel. Removal or isolation of these rods will probably be mandated in the future by U.S. NRC future requirements that preclude reconstitution, consolidation, dry storage or off-site shipment without specific precautions for any fuel with known significant failures. Restrictions for dry storage will likely apply either at the reactor or for eventual off-site storage at a U.S. facility such as the proposed Monitored Retrievable Storage facility.

Previously reported fuel integrity data (Ref.4,5) were based on radiochemical analyses of reactor coolant and provide no direct measure of the number of fuel rod failures as is done with the FFRDS. Information on fuel failures, such as available in the Brown Boveri data base and summarized in Table III, can be used to improve the estimates of defective fuel that must be handled at the proposed U.S. Monitored Retrievable Storage facility and/or spent fuel repository. Using the projected cumulative discharged fuel from U.S. reactor given in Reference 6, estimates are provided in Table IV.

### 7. Costs and Benefits from Fuel Inspections

The costs of fuel inspection with the Brown Boveri FFRDS is insignificant relative to the value of the fuel. For a typical PWR fuel assembly which is inspected with the FFRDS and found to have failed fuel rods after one cycle, the cost of inspection is less than 0.2 percent of the residual value of the unburned fuel. To this must be added the costs of reconstitution.

The financial benefits from inspecting fuel, isolating leaking rods and reconstituting sound fuel for reinsertion can be very substantial. Reference 3 details the experience of one U.S. utility with tests that savings from reinsertion of repaired assemblies from two cycles of plant operations resulted in net savings of 12 million dollars.

Radiological benefits to an operating plant from locating and removing defective fuel can also be achieved. In an U.S. NRC study (Ref. 7),
TABLE IV

Estimate of Defective Fuel Assemblies in U.S. Spent Fuel

<table>
<thead>
<tr>
<th>Year</th>
<th>No. Spent Fuel Assemblies(a)</th>
<th>Estimated Cumulative No. Defective F/A(b)</th>
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<tr>
<td>1987</td>
<td>60,100</td>
<td>8,288</td>
</tr>
<tr>
<td>1992</td>
<td>103,862</td>
<td>14,322</td>
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<tr>
<td>1997</td>
<td>150,213</td>
<td>20,713</td>
</tr>
<tr>
<td>2000</td>
<td>177,887</td>
<td>24,529</td>
</tr>
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</table>

(a) Data from Ref. 6

(b) Extrapolated for both PWR and BWR assemblies, based on overall averages from Brown Boveri FFRDS Data Base

Estimates are provided that a PWR operating at full power with 0.125% failed fuel can experience an increase of 540% in radiation exposure rates, as compared to a PWR operating with intact fuel. Using this estimate, the potential reduction in radiation levels that may be realized by locating and eliminating defective fuel rods is given in Table T. These reductions are based on the assumption that the FFRDS measured failure rates are actually representative of each fuel type.

A prudent nuclear utility operator will want to maintain reactor coolant system fission product activity levels well below any limits imposed by regulations for safety reasons. Utilities have found significant economic advantages in locating and removing from service any failed fuel. Economies are realized while following good ALARA practices to achieve lower radiation levels for maintenance and normal plant operations. By routinely inspecting every fuel assembly to be reinserted for further burnup after each refueling, a plant can be assured of achieving the lowest feasible fission product contamination in the primary system, thus minimizing worker exposures from this system.

Experience at U.S. nuclear plants has shown that significant problems result when operating with a relatively small number of leaking fuel rods, i.e., even when reactor coolant activities are well below license limits. These problems include:

- Increased reactor coolant system radioactivity of fission products particularly cesium, iodine, and strontium, plus possible transuranics
- Increased worker radiation exposures
- Increased purification system letdown flow producing an increased volume of liquid radioactive waste
- Increased volume of solid radioactive waste due to increased frequency of demineralizer bed replacement
- Possible need for workers to wear respirators with attendant loss of productivity and overall increase in person-rem accumulation
- Delayed access to containment building at end of plant operating cycle which extends plant outage
- Primary system long-term contamination with tramp uranium could cause high coolant activities in subsequent cycles
- Potential for need to derate power generation
- Increased handling and costs for eventual disposal of failed fuel assemblies

Although it is infrequent that a plant will exceed its licensed (i.e. Technical Specification) limits, increased regulatory and management attention is being given to reducing reactor coolant activities by identifying and removing leaking fuel rods. The single largest concern is the resultant increase in worker radiation exposures. The U.S. NRC study reported in Reference 7 also illustrates that in specific plant areas, the degraded fuel may elevate radiation exposure rates even more. Given these type of results, it is clear why both regulatory and industry organizations are giving serious consideration to establishing reactor coolant activity goals as nuclear plant performance indicators. With the demonstrated capability of the Brown Boveri FFRDS, these organizations can reasonably expect reduced coolant activities.

References


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ON-SITE FUEL EXAMINATION EQUIPMENT IN EDF PWRs

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Abstract

This paper describes the equipment used in EDF Nuclear Plants to examine irradiated fuel assemblies.

Techniques and equipment are:

(a) Visual inspection performed with equipment installed on most of the 900 and 1300 MW EDF units.

(b) Dimensional inspection performed with equipment also using a movable underwater video camera, furthermore, the camera movements are accurately measured.

(c) Fuel rod waterside corrosion measurements using eddy current techniques; these measurements provide information about oxide thickness on fuel rods and on spacers.

(d) Crud scraping, allowing crud deposit sampling from the fuel rod surface in order to perform in-laboratory analysis.

(e) Gamma scanning equipment which is installed on the PEMCI 900 and 1300, allowing calculation of the burn-up and the local power on all the peripheral rods of a fuel assembly and on one corner rod or one rod row.

Other gamma measurements can be made on individual rods after their extraction from special fuel assemblies.
Failed fuel identification using several successive test devices:
The sipping test in the mast of the refueling machine which identifies gaseous fission product releases during the normal core unloading sequence.
The sipping test cells which are permanently installed in the fuel storage pit; they can determine the size of the fuel defects to evaluate if the fuel assembly is reloadable.
The failed fuel localization system which determines the number and the position of the failed fuel rods for replacement purposes.

**INTRODUCTION**

The in-reactor behavior of nuclear fuel and the qualification of new fuel designs are both closely monitored by EDF and have been the object of extensive research and development work particularly in conjunction with the CEA and FRAGEMA.

This R & D work covers both fuel assembly skeleton mechanical behavior (growth, deformation) and fuel rod mechanical and physical/chemical behavior (growth, oxide layer and crud deposit measurement, neutron radiography, fission gas analysis, mechanical tests etc...); some of these inspections call for on-site extraction of fuel rods for hot cell examination.

The data collected on-site from the fuel assemblies and in hot cells from the fuel rods are the basis for improved knowledge of fuel behavior, modeling of physical phenomena and qualification of computer codes with the aim of optimum fuel utilization.

To carry out these R & D programs, it was necessary to design and build on-site examination and servicing equipment. More sophisticated hardware has been installed in the 900 MW and 1300 MW lead units but all the units are equipped with self-contained examination facilities; multi-site examination stations are also available. The purpose of this paper is to briefly describe the equipment in use in EDF units.

**TECHNIQUES AND EQUIPMENT**

**2. VISUAL INSPECTION**

2.1. VISUAL INSPECTION

Visual inspection involves straightforward, rapid checkout of fuel assembly appearance; to do this, a purpose-built visual inspection system has been developed: the PECU "Poste d'Examens du Combustible Usé" (spent fuel examination facility). This facility is installed in all the French 900 MW and 1300 MW units (i.e. thirty 900 MW units and a dozen 1300 MW units), except for Paluel 1 and 2, Bugey and Fessenheim which have more sophisticated hardware.

The facility is set up in the spent fuel pit and consists of:

(a) A fuel storage rack with a circular upper guide and a square lower guide joined to a rotating plate.
(b) A fuel support frame.
(c) Two hand-operated mirrors with their mechanical stand.
(d) A camera fitted to a camera carriage
(e) A control/command and power supply desk.

Visual inspections are conducted under water by means of a video camera; the facility keeps the fuel assembly in position and guides the camera in three perpendicular directions.

These examinations provide images of the lateral faces of the assembly and of the top and bottom nozzles. They provide information on the condition of the fuel rods, grids and nozzles and allow detection of loose parts, debris and deposits.

**2.2. VISUAL AND DIMENSIONAL INSPECTION**

The reactors at the Fessenheim site were the first commercial pressurized water reactors to be commissioned in France. An intensive R & D program was therefore launched at this site (fuel surveillance, rod bow analysis, increased discharge burn-up), for which EDF designed a fuel assembly visual and dimensional examination facility. This was named the PSEC "Poste Simplifié d'Examens du Combustible" (simplified fuel examination facility) to contrast with the PEMCI "Poste d'Examens Multiples du Combustible Irradié" (irradiated fuel multipurpose examination facilities) which can perform gamma spectrometries in addition to the usual visual and dimensional inspection.

2.2.1. PSEC

A single model of this facility is available at Fessenheim and can be moved from unit to unit. It is firmly anchored in the shipping cask loading pit and is used to perform visual examinations and take measurements as follows:

(a) Distances between nozzles.
(b) Peripheral rod/nozzle distances.
(c) Pin-to-pin spacing at each span.
(d) Assembly twist.
(e) Assembly bow.
(f) Holddown spring height.

The PSEC consists of:

(a) An underwater stand anchored to the pool wall and providing fuel assembly restraint and camera carriage guiding (2 motion).
(b) A carriage supporting a high-precision camera and its lighting system and allowing X and Y motions.
(c) Electrical equipment inside control racks.
A video channel linked to the camera, allowing the operator to scan the fuel assembly image on the video monitor and make dimensional measurements; fuel assembly close-ups and visual examination recording are also available. Camera positioning is accurate to 1/10 mm; optical accuracy is 1/100 mm on the video monitor. The operator controls carriage and camera motions, and lighting and scan adjustment. Measurements are taken manually; there is no automatic data acquisition or data processing and operators are solely responsible for measurement. A complete dimensional measurement on one fuel assembly lasts about 5 hours.

2.2.2. PEMCI 900 (Figure 1)

The irradiated fuel multi-purpose examination facility is available in two models at units 2 and 3 of the Bugey plant, where it is secured to the spent fuel pit wall. It is used to perform visual examination of six assembly faces, to take the same measurements as the PSEC (to the same accuracy) and to conduct gamma spectrometry examinations (whose principle is described below).

PEMCI 900 consists of a structure in the form of a frame, a carriage, an assembly stand and an instrumentation/control desk. The carriage can accommodate either the visual and dimensional examination facility or the gamma spectrometry station.

The visual and dimensional examination facility features:

(a) A camera/lighting assembly.
(b) A set of mirrors.
(c) An instrumentation/control desk.

The camera/lighting assembly can examine either all the rods of one assembly face or only three rods; the mirrors provide viewing of the nozzle upper and lower faces.

The gamma spectrometry station features:

(a) An underwater section mainly containing the measurement apparatus (a liquid nitrogen-cooled hyperpure germanium-lithium detector, a collimation system and shielding).
(b) An above-water section containing the liquid nitrogen tank and the instrumentation/control desk.

2.2.3. PEMCI 1300

Units 1 and 2 of Paluel were the first 1300 MW reactors to come on line in France and the behavior of their fuel (17 x 17 XL) is being closely monitored. It was therefore decided to equip Paluel 1 and 2 with an irradiated fuel multi-purpose examination facility (PEMCI 1300) secured to the spent fuel pit wall. The tasks accomplished by this apparatus are similar to those of the PEMCI 900: visual examinations, measurements and gamma spectrometry. The main design difference between PEMCI 900 and PEMCI 1300 lies in the spectrometry station; it is conveyed on a carriage inside the pool for the 900 model whereas it is floor-mounted for the 1300; the assembly is moved past a window in the pool wall. The gamma spectrometry collimation unit is located in the window.
PENCI 1300 features the following equipment items:

(a) A subassembly for guiding the fuel assembly and the camera carriage along axis Z.
(b) The main carriage supporting the fuel carriage and the camera carriage.
(c) A floor-mounted gamma spectrometry system.
(d) An instrumentation/control system for manual or automatic control of station functions.

2.2.4. PADEC

The PADEC "Poste Amovible D'Examen du Combustible" (removable fuel examination facility) combines the main features of the PSEC (visual and dimensional examinations - underwater mobile camera) with the added advantage of removability. It can be used in the 28 units of the 900 MW CP1 and CP2 plant series. It is secured to the wall between the cask loading pit and the washing pit.

2.2.5. FRAGEMA PEMS "Poste d'Examens Multi-Sites" (multi-site facility) is based on the same principle as the others; the only difference is that it separates the camera movement/measurement function from the fuel assembly restraint function. This means that, depending on pool layout, it is possible to mount the carriage opposite an assembly stand, opposite an assembly suspended from a handling tool or opposite an assembly on a special support.

2.3. MEASUREMENT OF ROD OXIDE THICKNESS

The principle is based on thickness measurement of a layer (zirconia) covering a metal mass (zircaloy) by eddy current examination using a probe brought in contact with the cladding surface by a mechanical device.

The equipment developed by FRAGEMA consists of:

(a) The eddy current channel (probe, cable, electronics).
(b) The probe positioning device which keeps the probe radial and perpendicular to the rod axis.
(c) A telescopic pole for conveying the positioning device and moving it up to the assembly whose peripheral rod oxide layer is to be measured.
(d) A control cabinet.

The fuel assembly is moved up to the measurement head suspended from the handling tool.

The measurement probe is fitted on the selected rod by moving the telescopic pole, using a video camera to monitor operations. It is possible to brush the rods using a rotating metal brush before measurement to dislodge loose deposits.

Fig. 2: Multisite Examination Stand, Carriage Assembly

A measurement accuracy of about ± 3 µm is obtained by calibrating the measurement channel on rod cladding sections with known oxide thicknesses.

The same equipment in tandem with a slightly different positioning device is used to measure oxide thicknesses on the assembly grid outside faces.

2.4. PERIPHERAL ROD CRUD SAMPLING

Any loose crud deposits on the peripheral rods are scraped off by means of a tool equipped with a suction device for collecting the dislodged
2.5. GAMMA SCANNING

γ scanning measurements on fuel assemblies can be taken in EDF plant storage pools. These measurements are taken throughout rod length either in the form of γ counts giving an image of fuel column activity or in the form of γ spectrometries supplying the axial distribution of the main radionuclides measured, so that burn-up and end of cycle power distributions can be deduced.

2.5.1. PEMCI 900

A device can be installed in Bugey 2 and 3, combined with the visual and dimensional examination facility (PEMCI PRSAME). In this application, the GeLi detector with its shield, cryogenic device and collimator unit is mounted directly on the mobile carriage so that the detector can be moved past the assembly.

2.5.2. PEMCI 1300

Specialised γ scanning facilities were later designed and built by the CEA/IRDI (DMG and SPC) on the Paluel 1 and 2 sites. In these facilities, the fuel assembly is moved past the γ collimator. A channel has been built into the storage pool wall for this purpose.

The collimation line is enclosed inside the channel behind the liner. The Ge HP detector is located at the rear of the wall (source-detector distance 1.5 m) at the core of the shield, restoring the biological shielding impaired by the channel.

The collimation line is extended into the pool by one of the two movable devices, depending on the examination. The devices have two different analysis geometries:

(a) One, the "wide slit", allows simultaneous examination of the entire width of an assembly face
(b) The other, the "narrow slit", allows row-wise scanning of fuel rods

In both cases (wide and narrow slit), a system of auxiliary collimators and post-collimators (all interchangeable) and an attenuator screen, whose thickness can be adjusted in steps, are used to modify by a known factor the activity level received by the detector.

The lateral deformation of the scanned peripheral rod are monitored by an ultrasonic measurement system specially built by CEA/DTECH. The U.T transducers are precisely positioned relative to the collimator axis. Closed-loop control of assembly X motion and of U.T measurement allows the rod axis to be positioned ± 0.1 mm from the collimator axis at any time.

Calculation of the relative fission product concentrations and plotting of their distribution are performed on the spot by the system.

Burn-ups and power levels are calculated in the laboratory from the measurements, with due allowance for the respective contributions of
each rod in the examined column (BAOBAB code), for the irradiation history and for reactor core physics data.

2.5.3. Removable fuel rod examination stand

There is also a γ scanning device combined with a "removable fuel rod examination stand". In this application, the fuel rod is withdrawn from the fuel assembly and placed in a support frame moving up and down just in front of the collimation line of the detector. It is also possible to rotate the fuel rod around its vertical axis in order to detect the influence of radial/power gradient.

2.6. FAILED FUEL IDENTIFICATION

A number of methods for identification of non-tightness of irradiated fuel assemblies are used in EDF power plants:

(a) Detection and evaluation of number of failures by monitoring primary coolant activity during reactor operation.
(b) Identification of leaking fuel assemblies by subjecting each assembly to a sipping test in the fuel handling mast during the refuelling period.
(c) Evaluation of the size of the defects by subjecting each leaking assembly to "quantitative sipping" in the fuel building.
(d) Localization of individual failed rods in each leaking fuel assembly.

This last operation allows the fuel assemblies to be repaired if necessary in order to make them reusable in reactors.

2.6.1. On-line sipping test system (figure 4)

The on-line sipping test system which is installed inside the refuelling machine is designed to identify irradiated leaking fuel assemblies during the routine performance of core unloading operations and without significant time losses.

This on-line system uses a gas sipping method which provides inherently greater sensitivity than conventional water sipping and is obviously unaffected by reactor pool water pollution. The system is self-contained and necessitates neither special personnel nor additional fuel handling (unlike the same test in a sipping cell).

Moreover, this test is performed only a few days after the reactor is shut down. At this time the quantity of fission products is still large because they have not yet significantly decayed (the half-life of Xe 133 is about 5 days) and because it is the first time that the fuel assembly is moved.

The system allows to:

(a) Single out defective fuel assemblies during discharging.

(b) Provide priority routing of these assemblies to the Fuel Building Sipping Test Systems for more thorough leak testing.

During Reactor Building fuel handling between the core and the transfer system, the assembly is raised from the core inside the refuelling machine mast. It is held in this raised position throughout refuelling machine travel between the core and the transfer canal.
When one or more rods are defective, the differential pressure caused by raising the assembly out of the core promotes the release of the fission products outside the cladding. The role of the system is to retain these products by means of a gas stream which routes them to a gamma activity measurement unit.

Air is continuously injected at the base of the refuelling machine mast, underneath the assembly nozzle leg. The air stream provides coverage of the entire fuel assembly surface when the assembly is in raised position. It entrains the gaseous fission products (mainly Xenon 133) released by leaking fuel assemblies due to the depressurisation caused by the difference in pressure head between the core and the raised position of the assembly inside the mast.

The activity of the air in the gamma unit is continuously measured by a Gamma Activity Measurement Channel. Counting is performed on the gamma peak of the Xenon 133 isotope or on the total spectrum above a low threshold.

This system was installed in 1982 in Bugey (four units) and Fessenheim (two units) power plants, and in all French Power plants by 1986.

2.6.2. In-cell quantitative sipping (figure 5)

After being detected leaky in the refuelling machine mast, the fuel assembly is sent to a quantitative sipping cell to determine its leak size so that a decision can be made on its reloadability.

The role of the test is to classify the leaking fuel assemblies as a function of the leak size, in order to decide if the assemblies are reloadable or not, given primary coolant fission product limits and the overall leaking fuel assembly population.

The leak size is characterized by an "equivalent diameter" determined by the fission product release kinetics. The fission product release is initiated by the water temperature increase.

In this test, the assembly is enclosed in the sipping cell, the water temperature inside the cell is quickly increased and then stabilized. During this temperature variation, the internal pressure in the rod increases (inert gas and vapor pressure). When the rods are failed, a release of matter (liquid and/or gaseous phase) together with the fission products in the cell water occurs.

The gaseous fission products are collected by a gaseous flow in a closed loop and continuously monitored by a gamma monitoring channel. The cumulated fission product release rate from the start of the temperature rise increases in the loop to a maximum, and then stabilizes due to the pressure equilibrium caused by temperature stabilization. This means that the gas stream activity also stabilizes.

The stabilization time directly depends on the size of the leak; the determination of the "equivalent diameter" is done by comparison between the experimental kinetics (supplemented by cell gas and water samples) and theoretical curves given by the computer code HARIBA which calculates the thermodynamic conditions of the quantitative sipping test by means of calibrated cylindrical holes.

The HARIBA code describes the variation with time of quantities of fluid (liquid or gas) passing through a hole of known geometry and size.
when the thermodynamic conditions (pressure and/or temperature) on one side of the hole are varied. The results of this code were initially qualified in a CEA laboratory by sipping irradiated or non-irradiated rods with calibrated holes.

According to EDF policy, leaking fuel rods with an equivalent diameter bigger than 35 micrometers are not reloaded, and the reloading of the others can be delayed if there is a risk of violating the primary coolant activity limits for the next cycle.

2.6.3. Individual failed fuel rod localization (Figure 6)

The main features of the FRAGEMA localization method are the following:

(a) It is based on the use of ultrasonic techniques.
(b) No motion of the rod during the test is required; the rods are tested in position, inside the assembly skeletons.
(c) A "go-no-go" type diagnosis is generated in real time for each tested fuel rod; the failure criterion used for the diagnosis is the presence of water inside the rods.

An ultrasonic probe is placed at the end of each rod undergoing testing. The wave train generated by the probe is propagated over the entire cladding length, then returns to the probe after reflection from the opposite end of the rod.

A through-wall defect causes water leakage; the water creates an acoustic couplant between the cladding and the fuel pellets, which disperses part of the energy inside the cladding. As a result, the energy which returns towards the probe is lower for a leaky rod than for a sound rod.

The method involves analysis of echo amplitude to single out leaking rods; one of the advantages of this method is that it provides overall rather than local scanning and is therefore insensitive to the water level or axial location inside the cladding.

The equipment used to apply the localization method described above is as follows:

(a) A probe, small enough to be inserted between the fuel assembly top nozzle and the rods undergoing testing.
(b) An underwater probe positioning mechanism mounted on one of the spent fuel pit stoppates or secured to the pit edge under 4 m of water; the 3 orthogonal movements of the probe are produced by carriages driven by step-by-step motors. Two pneumatic jacks are used to position the fuel assembly to be tested in front of the mechanism.
(c) A set of cabinets containing the electronic equipment associated to the probe and to its positioning mechanism, a computer to control the movement, process the signals and provide the diagnosis.
(d) A video system for visual surveillance of underwater operations.

Fig. 6: Single Rod Leak Detection In Irradiated Fuel Assembly
This equipment of the movable type has been operated successfully on EDF plants since October 1984. The test procedure consists of:

(a) The assembly to be tested, hanging from the site fuel handling tool, is positioned in front of the probe positioning mechanism under a water depth of 4 m.

(b) The probe is inserted diagonally in the fuel assembly under the top nozzle. Testing and diagnosis display (in the form of a map similar to a fuel assembly cross-section) are performed automatically. It is necessary to rotate the fuel assembly on its longitudinal axis in order to test all the fuel rods.

3. CONCLUSION

The great variety of hardware and methods used in EDF Nuclear power plants for the fuel surveillance programs have yielded extensive data on the behavior of the different fuel designs. These data have been used to validate the computer code models. Further, they have provided plant utilities with useful data banks for fuel management (reloadability of failed fuel assemblies, information needed for assembly repair).

Moreover, the development and the operation of the necessary sophisticated on-site examination equipment have allowed EDF, CEA and FRAGEMA to gain a considerable experience which will be particularly useful for the follow-up of future fuel evolutions.

DEVELOPMENT OF METHODS AND EQUIPMENT FOR UNDERWATER INSPECTION AND REPAIR OF WATER REACTOR FUEL ASSEMBLIES IN THE SOVIET UNION

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Abstract

Installations for pool-site post-irradiation examination for the VVER-1000 and RBMK-1500 reactors are described in detail. Work programme for the development of repairable fuel assemblies and for the manufacture of repair facilities is discussed, together with the factors which make the interest of this technical policy.

1. Introduction

Research material science complexes, existing in the USSR and "hot" chambers at head blocks of power stations[1] ensure at present the necessary bulk of irradiated fuel investigations for nuclear power. Increasing efficiency and safety reactor work and the problems of fuel assemblies and rods construction development required increasing buks
of nondestructive investigations with simultaneous decreasing their time. Orientation at the application of fuel assemblies without jackets in water-cooled reactors solves the problem of fuel assemblies repair with individual failed fuel rods. The development of methods and means of pool inspection and repair contemplated must solve these problems.

The pool-side facilities development concept consists in the following:
- provision of each reactor of power station with a stand for inspection of irradiated fuel (leakage detection, visual examination, geometric dimension inspection, fuel burn-up identification),
- provision of main blocks of power station with multifunctional research stands for examination of fuel assemblies as a whole and removed pins using a larger amount of nondestructive methods (pressure measurements, eddy current defectoscopy and others),
- creation of repairable fuel assemblies and provision of each power station with a stand for fuel assembly repair,
- provision with stands for underwater examination of some research reactors.

Advisability of such technical policy is due to the following factors [2]:
- decrease of research time and the time of development of new reactor fuel types,
- increase in efficiency of use of the existing fuel types (recommendations on the improvement of operation parameters, fuel assembly repair),
- solution safety problems.

At present:
- the equipment complex for testing of methods, main constructional and measuring stand elements has been put into operation,
- testing of these elements, experimental inspection stands for the VVER-1000 and RBMK-1500 reactors as well as experimental stands for research nuclear reactors have been made.
- work on creation of repairable fuel assemblies and repair facilities is being carried out.

The main problems being solved in this program and the equipment required are given in fig. 1. Within the scope of the program there are developed and manufactured the main elements of equipment: research complex, experimental stands for reactors VVER-1000 and RBMK-1500.

Repair stands and stands for removed fuel assemblies for power stations are being carried out.

2. Equipment for mastering procedures and individual gauges

The main basic to master procedures and gauges for full-scale mock-up fuel assembly poolside inspection is stand called methodical which is located in a "pure" pool in the SPIAK (fig. 2) as well as a part in research reactor pool with gamma-scanning unit and stand for investigation of irradiated fuel from disassembled bundles.

The methodical stand is a framework structure 9 m high. Four height-regulated supports of the stand rest on the bottom of pool and the top of it is fixed to the pool walls.

In the stand there is a carriage with a mobile table which travels along two internal vertical guides. There is also a mobile rotating support for fuel assembly which moves along the same guides. The travel of the carriage and (or) the support is driven by an electric motor located at the top of the stand. The mobile table on the carriage is driven with an immersion step-by-step motor. On the table there is a U-shaped gauge with transducers. Crude measurements of the vertical motion of the carriage (\( \pm 3 \) mm) are performed by a seismy coupled to the carriage through a roller-fine cable pair. The precise measurements of vertical and horizontal travels are performed with the help of ultrasonic transducers.
FIG. 1. Program problems on the pool stands.

FIG. 2. Complex for mastering poolside inspection methods.
1) pool; 2) control room with MCS; 3) methodic stand;
4) water purification system; 5) VVER-1000 stand at tentative mastering; 6) and 7) computer and electronic apparatus;
8) movable carriage; 9) movable support; 10) mock-up fuel assembly.
The stand can accommodate mock-up fuel assembly and fuel rods practically of all reactor types. The electronic measuring and controlling devices of the stand are located in a special cabin near the pool.

In fig. 3 there is a gamma-scanning unit installed in the poodside of research reactor. This unit is intended to master procedures based on measurements of activity (evaluation of burn-up, measurement of pressure upon the fission products activity). It consists of a tube to accommodate an immersion germanium-lithium detector, a lead shielding with a regulated height of a slit (0-30 mm) and object-moving lift (fuel assembly or fuel rod).

Stand for inspection of the expended fuel rods is located in the immediate vicinity from these unit. Stand (fig. 4) is a suspended structure consisted of a device for lifting and/or rotating fuel rod, a bar with fuel rod grip, a gauge with transducers for nondestructive test (eddy-current flaw detection, measurement of oxide film thickness, ultrasonic identification of leaky fuel rods, profilometry). A TV-camera is fixed on a single bar (with provision for substitution by a periscope). In additional to mastering procedures the stand is intended for in-site research of power reactor undersized fuel rods irradiated in transient power conditions.

3. Inspection stands

Inspection stands (fig. 5, 6) are intended for checking design verifications, evaluating the information procedure capability, measuring device optimization. Significant distinguishes of the metal construction stands are due to those in-pile location as well as WER-1000 fuel assembly design and size differences (hexagonal shape of fuel assembly in cross-sectional size 234 mm and 3840 mm in length) and RBMK (a pair of fuel assemblies connected to the bar have the total length 7300 mm and 79 mm in cross-sectional size).

VVER-1000 inspection stand (fig. 5) is a framework structure screwed up of three sections and rested on the bottom of pool. On a rotating support there is a fuel assembly. The carriage travels along a pair of vertical guides with the help of a cable. On the carriage there is dual

![Diagram](image)
FIG. 4. Single fuel rod inspection stand: 1) device for fuel rod removal and rotation; 2) bar with grip; 3) transducer unit; 4) TV camera.

FIG. 5. VVER-1000 fuel assembly inspection stand: 1) X-, Y-, Z-, ω-drives; 2) frame; 3) carriage travelling cable; 4) measuring gauge; 5) double co-ordinate table; 6) carriage; 7) guide column; 8) fuel assembly; 9) bar for fuel assembly rotation.

FIG. 6. Inspection stand for RBMK-1500 fuel assembly: 1) operating floor; 2) drive rod for fuel assembly revolution; 3) fuel assembly suspension; 4) carriage motion screw; 5) fuel assembly; 6) gauge; 7) support; 8) cap; 9) columns; 10) table with double co-ordinate motion; 11) carriage; 12) and 13) X and Y drive motion bars.
A co-ordinate table; its motion along X and Y-direction is effected through reductor with the help of two metal rotating bars. All electric drives (for X,Y,Z) occupy an area above water. On the table there fixed a special U-shaped gauge with measuring transducers. Gauge design and transducer nomenclature are identical with that of RBMK stand. VVER-1000 stand is mastered in the SRIAR where its preliminary tests take place.

RBMK-1500 inspection stand (fig.6) is suspended structure carriers of which are pair of tubes of 300 mm in diameter. Along tubes on guides the carriage is traveled by screw. On the carriage there is dual co-ordinated table with U-shaped gauge fixed on it. X,Y,Z-motion is performed with rotatable bars by means of electric drives on the upper stand area as for VVER-1000 one. Crude measurement of co-ordinate motion is performed with seisyn transducers and the use is made of ultrasonic that of precise measurement. In stand there hung fuel assembly on a proper bar; that of lateral motions are limited with a special cup on a lower support. A part of support has the form of fuel assembly region and provides a gauge test.

4. Main procedures of pool side inspection

Procedures and measuring equipment of stands.

Nomenclature of measuring operations for stands of power reactor inspections includes:
- visual inspections;
- measurement of fuel assembly cross-sectional dimensions in different regions;
- measurement of gap between outer row fuel rods;
- profilometry of outer row fuel rods;
- eddy-current flaw detection;
- measurement of oxide film thickness;
- measurement of crude thickness on outer fuel rods with provision of sampling crudes;
- detection of leaky fuel rods in fuel assembly.

To accomodate transducers realizing the indicated operations special measuring U-shaped gauge was developed (fig.7) It is screwed of sections. Each section has three shelves with longitudinal grooves on which transducers are fixed. Thanks to this design one can promptly replace them, alter the number of procedures performed. The sectional structure makes it possible to modify gauge for fuel assembly of different size.

The measuring and computing system (MCS) based on micro-computer (fig.8) controls the measuring process, collects and documents information. Transducers signals go through a communication line or through intermediate units at MCS that is located together with the TV-display and the stand control board in a special operators’room not far from the pool.

Below consideration is given to individual procedures and measuring equipment tested in methodical stand.

4.1. Visual inspection

Visual inspection means are designed to evaluate the condition of fuel assembly and fuel rods, to measure the individual geometric parameters and control the operation of different measuring transducers. The main means is a TV-system having an irradiation-resistant optics /3/. Use of the TV-cameras with the built-in and additional illuminators makes it possible to inspect both the whole fuel assembly and its elements at the magnification up to 20 X. The X,Y,Z motion of the carriage and revolution of fuel assembly about the longitudinal axes provides for the focusing of TV camera at any fuel assembly part of interest. The use is made of individually regulated light sources for selecting the best illumination.
FIG. 7. U-shape gauge: (a and b — side and front view).
1) illuminators; 2) TV camera; 3) block for measuring diameter and gap between fuel rods; 4) USD for leaky fuel rod identification; 5) eddy current transducer; 6) US transducers for cross-sectional dimensional measuring; 7) device for crud scrap.
FIG. 8. Measuring computer system: 1) board for electronic apparatus; 2) display console; 3) TV monitor; 4) and 5) printer and plotter; 6) microcomputer.

The IV signal arrives at a TV-display; there is also provision for digital processing of TV images. Digital processing system is a part of MCS of inspection and realizes different types of filtration of images, measures the distances between mobile markers, isolates zones of interest, evaluates distribution of image brilliancy and so on.

A periscope is an additional means. For operator’s protection from irradiation it is L-shaped. The horizontal portion is 1 m long, the vertical that is screwed up of sections and can be 10 m long. Instead of the eye-piece of the periscope photographic or TV cameras can be mounted.

4.2. Measurement of sizes

To measure the diameter of the outer row fuel rods as well as fuel rod gaps a special unit was designed with a pair of differential transformer transducers (fig. 9).

FIG. 9. Block for measuring diameter and gap

DLD – differential transformer transducer;
EM – electromagnetic HP-normalizing converter;
AUT – digitizer, IBM – computer, PC – air cylinder;
OF – information output.
Spring loaded probes of the transducer are introduced via a pneumatic drive in the inter rods gaps and pressed to a fuel rod from different sides. The fuel diameter is determined by the sum of the signals from a pair of transducers. To measure the gaps between fuel rods one probe is moved by an electromagnet. The accuracy of the unit measurement is not less than 10 mcm.

To measure sizes the extensive use is made of ultrasonic (US) pulso-echo methods based on the time of US-pulses travelling through water from ultrasonic transducer (UST) to object and in the opposite direction as well as those based on evaluating an amplitude of reflected signals. The advantage of the US methods is in their contact-free operation and the wide range of measurements of the high accuracy. For the realization of US methods the use is made of both commercial UST and devices (flaw detectors, thickness transducers) and the specially designed instruments of higher accuracy and automatic correction of the results when measuring the temperature of water for the time of signal transmission to benchmark.

Fig. 10 shows the scheme of measuring the geometric parameters of fuel assembly (fuel assembly cross-sectional size = \( \pi (\Delta_1 + \Delta_2) \)) on cross-sectional scanning with a pair of UST. In test experiments using an electrically heated mock-up fuel assembly the accuracy of measuring the cross-sectional size was 100 mcm.

Pulso-echo UST are also employed to determine the coordinates of the table with snap gauge. The information on the amplitude of US echo-signals is used for the accurate focusing of measuring transducers on a fuel rod.

4.3. Eddy-current methods

Eddy-current methods of inspection include flaw detection and measurement of oxide film thickness with passing or applied transducers. It is long since those methods were widely employed in shielding cells and their usage in poolside inspection is connected only with some changing the transducer design to ensure its leak-tightness as well as analogy part of the equipment for long coupling lines operation. The use is made of transformer eddy-current transducers (ECT) of passing type and applied that with saddle coils. The sensitivity limit of flaw detector is: longitudinal cracks with 80-100 mcm opening and 3 mm long at the outer surface and 50-200 mcm at the inner fuel clad surface.
4.4. Leak-tightness detection

All the Soviet reactors are equipped with sipping test system.

Search of leaky rods of fuel rods is one of the most important operations prior to repair of fuel assembly. Instruments developed for this purpose are based on the methods described in /4/. Therefore the use is made of US surface waves sensitivity for water presentation under clad. Negligible distinguishing features of instruments are related to fuel assembly geometry of Soviet reactors. For realizing procedure the transmission of different types of US waves in thin Zr cladding was investigated. There wave types were selected providing for maximum sensitivity and US structures were optimized. In recent developments the use is made of alone probe for two pairs of 'emitter-receiver' providing for measuring two rows of fuel rods in one probe passage (Fig. 11).

CONCLUSION

In summary, realization of new methods for measuring fuel to cladding gap, for non-destructive determination of gas pressure and other measurements are presented. Complete nomenclature of realized methods at standard stands will be determined with the results of experimental stands investigations at power stations and with reference to problems to be solved.

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EXPERIENCE AT THE CNA PLANT WITH UNDERWATER INSPECTION AND MODIFICATION OF FUEL ASSEMBLIES

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Abstract

1. The inspection of irradiated fuel has been carried out for a variety of reasons, the main one being the detection of defective fuel assemblies and failed pins. Indeed, if the concentration of fission products in the reactor coolant during operation or in the pit before transportation to the reprocessing plant exceeds the established standards, it may force the operator either to decrease the power, to shut down the plant or to keep in spent fuel pit the defective assemblies. For these reasons, the detection and location of the failed fuel is vital.

The CNA (Centrale Nucléaire des Ardennes, Chooz, France) new sipping unit designed by BELGONUCLEAIRE is an example of a wet sipping system which provides for leak indication either by water soluble fission products (closed circulation of demineralized water) or by fission products (gas bubbles circulation in demineralized water). The demineralized water is used to reduce the activity background and thus to improve the leakage detection efficiency. The sipping box is isolated by a thermal barrier (gas gap) and is equipped with an additional electric heating system which increases the expulsion of fission products and increases the leakage detection efficiency (T +0°C to facilitate the sipping of fuel assemblies with very low burnup or long decay times). A dry sipping possibility also exists in this unit.

After a qualification period, the unit has now been working satisfactorily as a routine procedure during the refuelling periods. More than one hundred assemblies have already been sipped.

2. Reprocessing of the CNA fuel assemblies requires the withdrawal of the 8b-Be secondary neutron sources out of the fuel assemblies before this reprocessing. As the CNA fuel assemblies are not dismountable, a specific equipment has been designed, based on the utilization of the electro-erosion technique to perforate under water the adapter plate of the fuel assembly top nozzle right at the position of the secondary source rod.

This equipment has been operated very effectively: high accuracy in the control of the perforation positioning and speed and in the control of the water contamination by nozzles material during perforation: the secondary source has been removed from more than 80 assemblies out of 10 reloads.

3. The same equipment has been used for the extraction of about 15 fuel rods out of 2 demonstration MOX fuel assemblies for postirradiation examination and reprocessing tests on MOX rods.

1. INTRODUCTION

The present paper intends to describe briefly the experience gained at the CNA plant with underwater inspection and modification of fuel assemblies.

Due to the undismountable design of the CNA fuel assemblies, this experience is mainly focused on

- the sipping of irradiated fuel assemblies for what concerns the underwater inspection experience
2. FUEL ASSEMBLY DESIGN

The CNA fuel assembly contains 208 rods, cladded with 304 stainless steel, arranged in a $15 \times 15$ type lattice, loaded in a skeleton consisting of perforated and bent shrouds assembled and welded to 9 spacer grids and 5 mixing grids.

The 9 spacer grids maintain the rod to rod distances while the 5 mixing grids improve the water mixing and fuel rod heat extraction in the highly rated zone of the fuel columns.

The shrouds are made of 304 stainless steel while the grids are made of 316 stainless steel.

304 stainless steel top and bottom nozzles are welded to the extremities of the shrouds; in addition to the water holes, the adapter plate of both nozzles are provided for with small diameter holes in which the extremities of the rod end plugs are introduced so that the adapter plates are also acting as spacers. Fuel assembly and rods are represented schematically in Figure 1 and Figure 2 respectively.

3. ORIGINAL DESIGN SIPPING EQUIPMENT

3.1. The original design sipping equipment consisted of two cells located in the spent fuel storage pool. These cells were working alternatively, one cell being operated for the sipping of one fuel assembly contained in it, while the second one was being loaded with an other fuel assembly to the sipped.

3.2. The sipping was based on the gas bubble circulation in stagnant water from the spent fuel storage pool. The operating flow sheet is represented in Figure 3.

Nitrogen was supplied to the circuit up to the time water in the cell reached the minimum level inherently fixed by the design of the cover of the cell and in the insulation jacket.

Then gas was forced to circulate through a counting chamber where a NaI detector and a multichannel analyser determine the total gamma counting and the gamma counting corrected from background, under Xe 133 gamma ray (81 kev) [Figure 4].

The Xe 133 gamma ray was counted during the first four minutes of gas circulation and during four minutes once the water temperature increases in the cell had reached 20 °C.

Basis for failed/non failed assemblies decision is given in Figure 5.

3.3. The operation of such an equipment was very simple and inherently fails safe with regard to cooling of the fuel assembly contained in the cell in case of some malfunctions of gas circulation.

However the Xe 133 gamma ray activity measurement accuracy was significantly affected by a background depending on
FIG. 3. Main parts of the wet-sipping flow-sheet based on the gas bubble circulation in stagnant water.

FIG. 4.
- the fuel assembly residual energy becomes very low, after several years in the spent fuel storage pool.

4. NEW DESIGN SIPPING EQUIPMENT

4.1. A new sipping equipment has been therefore installed in the spent fuel storage pool, meeting the following design objectives:

- once in operation, the equipment shall provide a closed circuit with regard to the water of the spent fuel storage pool.
- this circuit shall be fed with demineralized water to reduce the background of the activity measurements.
- the design shall allow rinsing of the open cell with or without a fuel assembly inside, reducing or removing crud deposits on the fuel assembly and in the circuit.
- sipping shall be made possible by closed circulation of demineralized water and radiochemical analysis of water samples for the detection of water soluble fission products, namely Cs134 + Cs137.
- sipping shall be made possible by gas (N2) bubbles circulation in demineralized water and on line counting of gas activity, namely Xe133 gamma ray.
- an electrical heating system shall be provided for in the insulating jacket, against the walls of the cell, able to increase the circuit water temperature by up to 30°C.

The flow sheet of the new design equipment is presented in Figure 6.

4.2. Figure 7 and 8 give the operational sequences and the present acceptance criteria respectively, when the equipment is being operated based on the gas bubble circulation in stagnant demineralized water.

Figure 9 illustrates the results obtained during one campaign of sipping:

In addition, the equipment did not allow to meet COGEMA Reprocessing Branch requirements for what concerns the sipping of the fuel assemblies definitively discharged from the core and to be transported to the reprocessing plant:

- the spent fuel pool water contamination
- the contamination of the cell due to preceding tested assembly, mainly if this latter was failed.

- the fuel assemblies shall be wet sipped if tested more than 2 months after the reactor shutdown;

FIG. 5. Original design sipping equipment basis for failure / non failure decision.
FIG. 6. New design sipping equipment — flow sheet.

- LOAD THE ASSEMBLY
- CLEAR THE ASSEMBLY WITH THE POOL FEED WATER DURING 10 MIN.
- CLOSE THE COVER
- DRIVE OUT THE POOL FEED WATER OF THE CAN BY DEMINERALIZED WATER INJECTION
- INJECT THE CAS IN THE CIRCUIT AND SET UP THE CAS PLUG
- HEAT UP THE ASSEMBLY (ASSISTED) CIRCULATE THE GAS (800 L N₂/HR)
- RECORD THE GAS ACTIVITY DURING 4 MIN

FIG. 7. Wet-sipping based on the gas bubble circulation in stagnant demineralized water — operational sequences.
**THE SECOND AND THE FIRST COUNTINGS ARE COMPARED AS FOLLOWS:**

\[ S_1 = T_{16} + 3 \sqrt{T_{16}} \quad (\text{Xe133 peak + Compton}) \]

\[ S_2 = T_{26} - 3 \sqrt{T_{26}} \]

\[ R = \frac{S_2}{S_1} \]

- \( R \geq 3 \) LEAKY ASSEMBLY
- \( 3 > R > 2 \) SUSPECT ASSEMBLY
- \( R < 2 \) TIGHT ASSEMBLY

* FOR SUSPECT ASSEMBLIES, THE NET INTEGRAL OF THE XE133 ACTIVITY COUNTING IS THEN USED:

\[ R \geq 2 \] SUSPECT ASSEMBLY

\[ R < 2 \] TIGHT ASSEMBLY

* COMPARISON OF THE XE133 COUNTING WITH THE BACKGROUND

* COMPARISON OF THE Cs ACTIVITY OF THE WATER SAMPLE WITH THE POOL ACTIVITY

**FIG. 8.** Wet-sipping based on the gas bubble circulation in stagnant demineralized water — acceptance criteria.

**FIG. 9.** Gas sipping — CNA cycle 14.
2 assemblies (A and C) are failed, based both on the Xe133 activity ratio and the Cs activity of the water sample taken at the end of the operation.

Although assembly B presents Xe133 activity slightly higher than the background, the ratio is meeting the acceptance criteria and the Cs activity of the water sample is far below the water pool Cs activity; both facts indicate that assembly B is not to be declared failed.

4.3. Figure 10 and 11 give the operational sequences and the present acceptance criteria respectively, when the equipment is being operated, based on the closed circulation of demineralized water.

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**LOAD THE ASSEMBLY**

**CLEAN THE ASSEMBLY WITH THE POOL FEED WATER DURING 10 MIN**

**CLOSE THE COVER**

**DEMINERALIZED WATER CLOSED CIRCULATION DURING 15 MIN**

**TAKE A WATER SAMPLE**

\[
\text{Cs FIRST COUNTING} = \frac{T_{ICS}}{TICS} = \frac{(C_{134} + 3 C_{134}) + (C_{137} + 3 C_{137})}{(C_{134} - 3 C_{134}) + (C_{137} - 3 C_{137})}
\]

**HEAT UP THE ASSEMBLY**

(NATURAL ASSISTED)

\[
\Delta T = 20^\circ C
\]

**TAKE A WATER SAMPLE**

**OPEN THE COVER**

---

**Cs SECOND COUNTING**

\[
\frac{T_{ICS}}{TICS} = (C_{134} + 3 C_{134}) + (C_{137} + 3 C_{137})
\]

**FIG. 10.** Wet-sipping based on the closed circulation of demineralized water — operational sequences.

**FIG. 11.** Wet-sipping based on the closed circulation of demineralized water — acceptance criteria.

Figure 12 illustrates the results obtained during one campaign of sipping:

- Assemblies A and B seem to be suspected based on a Cs ratio between 2 and 4.

- However assembly A can be considered as acceptable as the Cs activity of first measurement is close to the average value measured for the other assemblies, and the Cs activity of the second measurement is well below the Cs activity of the spent fuel water pool.

- However assembly B seems more suspected as the activity of the first measurement is significantly above the average measured for the other assemblies, while the activity of the second measurement is slightly above that of the spent fuel water pool. This assembly should be retested.
5. WITHDRAWAL OF SPECIFIC RODS OUT OF THE CNA FUEL ASSEMBLIES

5.1. Each region of CNA fuel assemblies includes 8 assemblies containing each one secondary source rod. This secondary source rod is of the same design as a fuel rod except that the fuel pellets are replaced by Sb-Be oxide pellets. The secondary source rod replaces a fuel rod at one specific predetermined location.

Reprocessing of the CNA assemblies requires the withdrawal of the secondary source rod out of the assemblies before the reprocessing.

42 concerned assemblies from regions 3 to 7 were stored in the La Hague reprocessing plant.

56 concerned assemblies from regions 8 to 14 were stored in the CNA spent fuel pool.

5.2. Within the frame of a plutonium recycle demonstration programme in the CNA reactor, 6 MOX assemblies were irradiated from 1974 to 1978 for 4 assemblies, and from 1976 to 1979 for 2 assemblies.

11 MOX rods and 3 uranium rods have to be removed out of 2 of these demo MOX assemblies for the following purpose:

- 8 MOX rods and 3 Uranium rods for postirradiation examinations
- 3 MOX rods for reprocessing tests.

5.3. As the CNA assemblies are not dismountable, specific transportable equipments have been designed, with the utilization of the electro erosion technique to perforate underwater the adapter plate of the top nozzle at the position of the rods to be extrated.

The equipments include the followings:

- the electro-erosion perforation equipment itself;
. a positioning tool to position the preceding equipment at the right rod position;
. a rod extraction tool;
. a rod storage can selected to be identical to the internal structure of the TN6 highly irradiated fuel shipping container;
. a container for the storage of the wastes and contaminated filters and electrodes before their conditioning according to the CNA usual procedures.

The following documents were first elaborated and approved before the extraction operations were allowed to take place:

. design drawings;
. as built drawings;
. operation and instruction procedures;
. safety evaluation report of the operations;
. equipment acceptance reports based on performance tests performed in the manufacturer workshop.

The extraction operations were conducted according to the following sequences:

. positioning of the perforation equipment;
. perforation;
. extraction of the rod;
. storage of the rod;
. intermediate storage of the wastes produced, and the contaminated filters and electrodes and their conditioning.

Figure 13 presents the flowsheet of the operations with in particular:

- the perforation by electro-erosion;
- the rinsing of the electrode by water pressure during perforation;

FIG. 13. Water rinsing of electrode by pressure and filtration of radioactive wastes.
- the aspiration of the radioactive wastes due to the perforation thru
an immersed filter. The aspiration flow rate, compared to the water-
rinsing flow rate, was such that the operations cause no specific
contamination of the pools in which the operations took place.

The total weight of the electro-erosion equipment was about 26 kg. The
maximum displacement of the electrode was about 250 mm, and the per-
foration might take place up to a water depth of 30 meters. The deplo-
cement of the electrode was reported to the control panel with an
accuracy of 0.01 mm.

Figure 14 represents the positioning tool. This equipment is position-
ed with high accuracy on the fuel assembly top nozzle taking benefit
of the surfaces of the top nozzles used as reference for the position-
ing and guiding of the cruciform control rods.

Figures 15 and 16 show the configuration of the top nozzle after the
extraction of respectively one secondary source rod, and 11 fuel rods.

5.4. The equipments have been operated very effectively, both in the CNA
spent fuel storage pool and in the La Hague storage pool:

- high accuracy in the control of the perforation positioning;
- high accuracy in the control of the perforation speed;
- effective control of water contamination due to perforation.

Typical operation data are as follows:

- volume of stainless steel to be removed per rod extraction: about
  5000 mm³
- perforate rate 100 mm³/min.
- perforation duration: about 53 min.
- electrode consumption: about 1.5 mm per rod extraction
- quantity of perforations: 4 per electrode

5.5. Electro-erosion equipment has been qualified as alternative to more
standard dismantling or cutting tools in the CNA reactor.
FIG. 15. Configuration of top nozzle after perforation and secondary source withdrawal.

FIG. 16. Configuration of top nozzle after perforation and 11 fuel rods withdrawal.
REPAIR AND RECONSTITUTION TECHNIQUES
(Session II)

Chairmen

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Abstract

The state of the technology in fuel assembly repair area has evolved to a high degree of industrial maturity.

But we should point out the importance of technical problems associated to the fuel repair economy: aptitude of the fuel design for repair, underwater irradiated fuel transportation, operator ability to repair fuels which are not its design, diagnostic of the damage, restoration radioactive wastes and leaking rod replacement.

1 - INTRODUCTION

Operational experience gives satisfactory fuel behaviour results. The proportion of assemblies showing such an important damage of their rods or their supporting structure, so that they cannot be reloaded remains extremely low. Nevertheless, by re-using those few assemblies after repair, a better specific burn-up of the initial uranium charge is obtained.

Fuel assembly repair are economically very attractive for operating nuclear utilities. During the past 5 years, the state of the technology in this area has evolved to a high degree of industrial maturity. Damaged fuels are currently repaired, reloaded and then meet their performance and design requirements over the life time of the fuel rods without problems.

In this paper, the status of the process and equipment used is not reviewed. But strategic considerations importance of the diagnostic of the damages and treatment of the expected wastes normally produced during operation, conditions in which repair process take place are summarized.

2 - NUCLEAR FUEL BEHAVIOUR OVERVIEW

Increase in nuclear fuel supplies induced by the growing number of nuclear power plants and accumulation of cycles, brings about increase in damaged fuel assemblies.

On december 31, 1986, for 9,920 irradiated fuel assemblies in EDF's nuclear plants, which have achieved at least one cycle, 178 of them, present a damage:

- 27% by external cause: handling injury, baffle jetting, loose parts.
- 73% by internal cause: fabrication, design or abnormal operating conditions.

EDF, as other european utilities, is allowed to reload leaking fuel assemblies within definite limits, so repair needs are essentially focused on damaged assemblies by external cause.

So, if each fuel assembly stays in reactor during three cycles (third-core refueling management), the energy loss is 89% by external cause.

The repair process provides a suitable reply to the occurred damage. Generally two kinds of repair take place:

- often, complete exchange of assembly skeleton by transferring all the sound rods from the damaged assembly into new skeleton,
- occasionally, extraction of leaked or failed rods and replacement by dummy or uranium fuel rod,
- and, of course, a combination of the two previous.

3 - TECHNICAL PROBLEMS ASSOCIATED TO THE FUEL ASSEMBLY REPAIR ECONOMY

3.1 - Aptitude of the fuel assembly design for repair

New fuel generation design offered by all fuel suppliers takes into account the ability to repair easily irradiated
assemblies. Advanced fuel assembly features the possibility of removing the top nozzle to permit direct access to the rod bundle. The special shape of the rod’s end c-psi allows the rod prehension. So the time needed to repair the fuel assembly decreases greatly and the work station does not take up a lot of space on spent fuel pond.

On the other hand fuel suppliers have developed, in parallel with the new fuel design, a set of equipment for testing and repairing irradiated assemblies. From a contractual clause, fuel supplier has in charge their own assembly’s repair if the damage has fuel design or fabrication cause.

3.2 - Underwater irradiated fuel transportation

Damaged irradiated fuels are scattered about many spent fuel ponds and sites. From a safety point of view and at first analysis, it is more suitable to move the repair device than irradiated fuels from pond to pond. But spent fuel transportation experience, using dry casks is so important that underwater transportation to prevent high fuel temperature, does not raise difficulties.

Now, a specific cask (ROBATEL R62) designed for an underwater transportation of one leaked fuel and a normal cask like TRANSNUCLEAIRE TN 12 to hold one or more damaged PWR fuel assembly, are qualified. It is more attractive to gather together in the same pond all the damaged fuels from the same site (2 or 6 reactors).

Then, fuel repair campaign can be carried out with efficiency and more safe conditions. The repaired assemblies can them be reloaded into the reactor who is in communication with the pond where the repair has been done.

3.3 - Operator ability to repair fuels which are not it’s design

It is becoming usual to have in the same reactor or at the same site damaged fuel assemblies from different suppliers which are also operators in fuel repairs. Sometimes, the utility confides his projects in fuel repair to a different supplier. If, from a commercial point of view, this is a comprehensive attitude, on the other hand, in the strong fresh fuel market competition it is advisable to be prudent to prevent transfer to the competitors of confidential informations on fuel behaviour.

For this reason, some utilities prefer to keep the fuel supplier for repairing his own supply. Moreover utility has technical assurance that tools and equipments have compatible design, condition for safe operation and quality work on the assembly.

3.4 - Diagnostic of the damage

The evaluation of the fuel damage with a minimum of uncertainty is an important phase for the on site repairing campaign organization and the enchainment of the campaigns in other sites. Only few months by year are free for an operation : outside outage period for fuel refuelling and not during, fresh fuel receiving time and spent fuel evacuation. If repair takes too much time for an unexpected event, then it can be necessary to dismantle the repair work station before the end of the campaign and to calculate again a loading core map because the number of restored assemblies that can reloaded into the reactor will be less than expected.

To avoid this situation, the fuel inspection will generally be performed several months before the repair operations. For that purpose, many equipments have been designed : for example visual examination device, an overall sipping test cell to determine whether fuel assemblies are leaking or not and, particularly rod by rod leak detection probe. Ultrasonic and eddy current failed fuel rod detections are now widely used and the detection of the exact position of the leaked rod can be determined quite easily. But it does not exist a simple device to look possible damage inside the rod bundle.

3.5 - Restoration radioactive wastes

For the utility the repair campaign is not finished with the reloading of the restored assemblies. It has in the pond :

- damaged irradiated skeleton
- damaged fuel rods in a basket storage
- dustbins with irradiated metallic pieces, scraps of metal, rod pieces or unanium pellets.

These components occupy significant cells in pond which was, in some countries, designed for storing only the spent fuel during two years before evacuation to other storage site. Specific treatment or conditioning has to be developed to take into account constraints connected with transport, disposal or reprocessing.
For example, damaged fuel rods are stored in a basket similar to a fuel skeleton who has individual cells to confine solid radioactive waste and to cool them. Basket is also designed to be compatible with the fuel handling tools, cask for transportation and the shears used in front of the reprocessing plant to cut spent fuel. On the other hand, damaged skeleton will be cleaned to eliminate, if necessary, alpha contamination, cut to reduce volume, stocked in the pond and conveyed to the low and medium waste disposal after Co60 decrease activity.

3.6 - Leaking rods replacement.

In the future, the increase of the fresh fuel enrichment reload about to 4.5% U 235, requires a stock containing varying enrichment for the substitution rods if assembly repair takes place after one or two cycles.

Moreover design of mixed plutonium-uranium oxide fuel, who has on the same assembly three zones of plutonium enrichment, request a fine analyse, to establish the nature of the replacement rod.

5 - CONCLUSION

Post-operational checks on different countries demonstrate that repairing and reconstitution of damaged fuel assemblies is a success operation.

The utility point of view is not to encourage repair but to develop staff training for fuel handling, to correct internal structure to remove baffle jetting, to demand high quality level for fuel design and fabrication and to keep watch over cleanliness during maintenance works to avoid presence of unidentified piece in the primary circuit.

DISASSEMBLING AND REBUILDING 900 MW UNIT FUEL ASSEMBLIES IN CELIMENE

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Abstract

The Celimene high activity laboratory, in the Nuclear Research Centre of Saclay, has equipment for and experience of disassembling and rebuilding fuel assemblies from 900 MW light water reactors. These operations have been performed for R & D purposes; they allow removal for investigation of some of the fuel rods and examination of the skeleton. The rebuilt assemblies are sent to the fuel reprocessing plant. Reirradiation of these assemblies has not been considered so far and would require modifications of the procedure and of parts of the new skeleton. Disassembling and rebuilding have already been performed on three assemblies and a fourth one will be rebuilt in the coming months.

In view of the scale of the French nuclear power generating program and the increasing proportion of PWR facilities, it was decided in 1975 to provide CEA with the technical capability of carrying out monitoring and development programs on fuels used in this technology, in collaboration with the manufacturer Framatome and the operator EDF. Therefore, in 1976, arrangements for diverting individual fuel assemblies from the normal reprocessing cycle were implemented (Figure 1). This necessitated the design and construction of a special transport cask (IL 42) suitable for loading a fuel assembly in the spent fuel pit and unloading it in air within the Celimene cell at Saclay, as well as the transformation and fitting out of the cell so as to enable manipulation, examination, dismantling and rebuilding of PWR fuel assemblies (Figure 2).

There are two requirements for laboratory examination of an irradiated fuel assembly: firstly, it must be possible to guarantee that the temperature reached by the fuel assembly during transport and manipulation does not change its mechanical properties, and secondly it must be certain that the fuel assembly can be returned to the normal reprocessing cycle.

To make sure that the first requirement is met, and in view of the thermal properties of the cask used for transport under inert gas, it is necessary to impose a cooling period of approximately one year in the spent fuel pit. Also, the temperatures reached by the various parts of the cask are measured during transport using non-invasive thermocouples. Finally, during the measuring operation, it is possible to read the surface temperature of the rods within the fuel assembly bundle using a device with a blade bearing a thermocouple.
To meet the second requirement, it is necessary to recondition the fuel assembly in a manner compatible with the technical requirements and approved by the Mechanical Departments of the reprocessing plant as, once rebuilt, the fuel assembly must not have any particularities requiring the operator to modify its procedures or carry out additional operations. It is for this reason that, depending on the implementation arrangements, we have developed, in close collaboration with the Mechanical Department of COGEMA, a number of techniques which make it possible to either attach a new top nozzle to the irradiated skeleton or to re-install the rods in a completely new structure.

Several measurements are performed on the assembly before dismantling. Visual examination is carried out with a periscope and records are taken with a camera mounted on an X-Y table to avoid parallax. The height of the assembly, the length of the peripheral rods, the axial positions of all the grids and various other dimensions are also measured. Finally the width of the water gaps can be determined at any level. This is achieved with a device consisting of a set of mirrors, mounted on a pair of blades, which reflect a laser signal to a receiving diode. This operation is entirely computerized.

After having carried out all the non-destructive tests, the fuel assembly is placed in a frame which holds it perfectly rigid by means of clamps securing the grid assemblies. As standard fuel assemblies can only be disassembled from the bottom and in view of the difficulties involved in rotation in the cell, we have developed a number of methods of machining the guide tubes at the top. Cutting can be carried out either using a circular saw with a diameter of 410 mm made of hardened steel which is either diamond studded or provided with cutting teeth (Figure 3), or by internal machining of the guide tubes with a miniature diamond

![Diagram of fuel cycle](image1)

![Diagram of hot cell Célimène](image2)

![Diagram of machining equipments](image3)
studded grinding wheel turning at high speed with an epicyclic cutting movement. Finally, a tool with three cutting bits which are expanded by an internal cone has been developed and is under construction. This equipment makes it possible to machine the guide tubes below the upper nozzle and also, by extending the transmission shafts, to cut the guide tube from the inside at any level. It should therefore be possible, by making two cuts at levels Z1 and Z2, to extract a limited number of peripheral rods to gain access to sections of the guide tubes for expert examination. After separation of the top from the guide tubes, the rods are extracted while measuring the necessary extraction force and are then placed in special racks (Figure 4).

There are then two possibilities depending on whether the program includes examination of the skeleton or not. If destructive tests are not to be carried out on components of the skeleton, the necessary pins are extracted and mechanical re-assembly of a new top on the guide tubes of the irradiated fuel assembly is carried out. The top nozzle is machined in accordance with the original drawings, in particular as concerns the external dimensions and the dimensions concerned in gripping by the manipulating tools. It is mechanically secured, by expansion of an internal pin, to the guide tube which is maintained in contact with a solid part connected to the nozzle (Figure 5). Pull tests have shown that a limited number of joints of this type (8) are necessary for manipulation of the fuel assembly. This technique has successfully been implemented with a Borselle fuel assembly.

In view of the considerable differences in the dimensions of the top of a standard fuel assembly (Westinghouse/Framatome), the principle of attachment by expansion has been retained, but we have re-designed the upper section of the attachment part to ensure that there is no overthickness with regard to the top support plate. Furthermore, a means of attachment with a long tie rod making it possible to obtain expansion just above the upper grid assembly has been developed (Figure 3). Finally, to obviate any risk of separation at the top, especially after storage under water for a long period, it has been planned to install a central tie rod capable of meeting handling requirements alone. This last feature is undergoing qualification and shall be presented to COGEMA for approval in the near future.

If destructive tests, in particular mechanical tests or metallographic examinations, are to be carried out on structural parts, guide tubes, grids or nozzles, it is necessary to remove all the rods and subsequently to reinstall them one-by-one, by pressing them into a new structure (Figure 6). This structure, which is in all respects identical to a standard structure, is equipped alternatively with standard grid and modified grid in which all the springs are specially treated to reduce the force necessary for reinstalling the rods. The rods are then secured in place with the springs of four of the eight grid assemblies. The system used for securing the guide tubes to the top nozzle is of a simplified screwed type. The new structure is equipped on assembly with 132 dummy rods introduced into every other recess, which forms a pattern making it possible to ensure guidance of the irradiated rods during their re-insertion. After loading 132 irradiated rods, the 132 dummy rods are removed and loading of the irradiated rods is continued. All these operations are carried out using a rod guide box and an X-Y indexing table localizing the channel to be loaded.

A number of operations involve advanced technology which has been developed either within the scope of a reprocessing research and development assistance contract with COGEMA, or within the scope of multilateral programs (Westinghouse, Framatome, EDF and CEA) relating to studies of behaviour of the fuel assemblies and structural components of pressurized water reactors. The removal of some fifty rods has been carried out with a Borselle fuel assembly as part of the reprocessing research and development and a dummy top has been re-installed on the structure. The FEB 34 fuel assembly of FSH2 was entirely disassembled by mechanical machining for expert examination of the structural parts of the skeleton. The FCEA 03 CX fuel assembly of CEA design, i.e., with sliding grid assemblies, was disassembled using a process analogous to that used for assembling. In both cases, the rods not involved in the examination programs were reloaded into a new structure and returned to the normal reprocessing cycle. An operation of the same type is in progress on the FEC 52 fuel assembly of FSH2 which has been irradiated for five cycles.

It can therefore be concluded that, providing the acceptance criteria relating to the reprocessing workshop manipulation procedures are respected, there is no difficulty in diverting a fuel assembly to a laboratory for a limited or complex program.

![Rods Manipulation Diagram](image-url)
KWU TECHNIQUES FOR LWR FUEL ASSEMBLY REPAIR AND RECONSTITUTION

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Abstract

Although KWU fuel assemblies (FA) have shown an excellent operation record since the early beginnings of commercial nuclear power generation, techniques for FA inspection, FA repair and reconstitution were developed at a very early stage, both for PWR's and BWR's. In all KWU built plant reconstitution equipment is installed in the spent fuel storage pool.

For other utilities, KWU can provide its reconstitution technique for PWR assemblies by means of transportable equipment. In addition, an electric discharge machining technique (EDM) is available to be applied to non-KWU fuel assemblies originally not intended to be repairable.

The techniques applied for a variety of repair and reconstitution tasks describe such as removal of defective rods, including EC testing, re-conditioning of damaged spacers, and transfer of fuel rods to another skeleton. Some more recent developments are sketched in the following.

Spacer springs broken by baffle jetting are a common reason for transfer of rods into a new skeleton. To avoid cost for disposal of the old skeleton, KWU has developed special dummy rods with retractable springs for filling up the damaged rod positions in-situ.

The repair and reconstitution equipment not only serves for repair of defective fuel assemblies but is also the key to fuel performance evaluation and future developments by providing access to individual fuel rods. Measurements of fuel rod diameter and oxide thickness are performed in many plants to provide the data base for design and further development. To extend the burnup of pathfinder fuel rods, they are transferred from their original assembly to another host assembly. Selected fuel rods are removed at intermediate or final burnup for detailed examinations in hot cells.

All techniques have been applied to a large number of PWR and BWR fuel assemblies which were subject to further investigations.

1. INTRODUCTION

Light water reactor fuel assemblies have reached a remarkable level of manufacturing quality and operational reliability. The presence of leaking fuel rods during operation is recognized by continuous monitoring of the coolant fission product activity. During refuelling, the assemblies containing leaking fuel rods are identified by a sipping test. Visual inspection of the fuel assemblies reveals possible damage of structural components. Further irradiation of such assemblies would result in continued fission product release which may impede the operational flexibility within the plants technical specifications or may be impossible due to the mechanical damage.

Reconstituting fuel assemblies with defective rods provides significant economic benefits since the considerable cost of replacing a defective but not fully expended fuel assembly with a new one can be avoided. The limited spent fuel storage capacity in many plants and the high costs for reprocessing is another incentive not to waste storage positions on fuel assemblies that have not yielded the designed amount of energy.

The extraction of defective rods from fully burned fuel assemblies is desirable for better flexibility of transportation to and storage in the reprocessing plant, too.

All KWU built BWR and PWR plants have permanently installed reconstitution equipment available. The methods in use have been standardized since 1976. For other PWR plants reconstitution services can be offered with a transportable reconstitution equipment developed in 1985 and used in some plants.

It is the aim of this paper to describe the equipment as well as the standard methods being used and the operational experience with reconstituted fuel assemblies.

2. EQUIPMENT

In KWU plants, sufficient space is available in spent fuel pools to accommodate the reconstitution equipment. As a rule, the main PWR reconstitution components, such as the tilting station, which is used to invert the fuel assembly, and the working brackets, which take up the fuel assemblies, are permanently installed. The tilting station consists of a tilting basket which is a rigid and precise container with removable lids at both ends. It is supported by two bearings in a frame structure positioned on the floor of the spent fuel pool. This design of the tilting station allows the removal of the tilting basket together with the fuel assembly from the frame structure in vertical position. The long tools for remote manipulations are in the pool only during the reconstitution campaign.

All these tools are designed for easy remote-controlled handling. Heavier tools are equipped with floats or are suspended from balancers. One of the most important long handling tools is the "fuel rod exchange device" used to withdraw and re-insert the fuel rods individually. To test the fuel rod cladding by eddy current and transfer defective rods into a canister. See Fig.1.

2. EQUIPMENT

The tool consists of a long tube which houses the fuel rod gripper. At the upper end, an electric motor serves for force-monitored axial movement of the gripper.

An eddy current coil is mounted at the lower end of the tool. The whole tool is precisely positioned above the fuel assembly according to the fuel rod pitch by means of a positioning carriage at the bridge railing and an indexing plate placed directly on the flange of the tilting basket and guided on the fuel assembly.
In BWR plants, the fuel preparation machines are used to take up the reconstitution candidates. Fig. 2 depicts the reconstitution equipment for BWR assemblies in the spent fuel pool of a KMU plant. One of the main components the guide channel, which is used to provide radial guidance for the rods after removal of the upper tie plate, is placed over the upper part of the fuel assembly and supported on the upper guide of the fuel preparation machine. This guide channel can take up an indicator plate with movable and fixable pins which allows to carry out length measurements of all rods or an indexing plate which is used to guide the mouthpiece of the fuel rod exchange device. The other long handling tools are similar to the ones used on PWR fuel assemblies. When the equipment is not in use, the long bar tools are disassembled into part lengths and stored in special aluminum boxes.

For third party PWR plants, advanced transportable equipment is available for reconstitution services /3/. The schematic arrangement in the spent fuel pool of a power plant is shown in Fig. 3. It was first used in 1986 in a power plant in the US. After that it was also used in a French power plant. Fig. 4 depicts the main part of the equipment, which is called the fuel assembly reconstitution unit (FARU), placed on the floor of the...
spent fuel cask area. Adjustable foot pads serve for leveling the FARU. The base frame structure is formed like parallel bars and carries the tilting basket, which takes up and inverts the fuel assembly. A fuel rod canister into which the defective rods are inserted is placed in a second tilting basket. The FARU also takes up auxiliary equipment such as the single rod inspection position and the tilting basket lids. The long bar tools are similar to the ones described earlier.

3. RECONSTITUTION TECHNIQUES and APPLICATION

Based on the design of PWR and BWR fuel assemblies the reconstitution method for exchanging defective fuel rods is very similar in both cases. For this description the general reconstitution procedures will be limited to PWR fuel assemblies.

General procedure

KWU fuel assemblies are designed for easy reconstitution. Traditionally, PWR fuel assemblies had a welded or screwed connection of the guide thimbles to the upper end fitting and a removable screwed one to the lower end fitting. To obtain access to the fuel rods, the standard reconstitution procedure includes the inversion of the assembly. Recently, modifications of the spacers and easily removable upper end fittings were introduced to also allow reconstitution without inversion. The main steps of the standard working procedure are explained in the sequence (1) through (5) illustrated in Fig. 5.

All underwater manipulation is performed by skilled operators standing on the fuel handling bridge positioned above the reconstitution unit. After removal of the mechanically locked nuts from the guide thimble bolts, the lower end fitting is removed. The fuel rod exchange device is used to grip the individual rods and to simultaneously pull and test them by eddy current over their full length. This technique provides detection not only of through-wall defects but also cladding wear which has not yet resulted in complete penetration. This method can be performed rod by rod for the whole fuel assembly or on single predetermined fuel rods. The monitored force during axial movement of the
rods allows to check the spacer spring friction force of every spacer cell thus assuring the proper fit of the fuel rod. Defective fuel rods are first transferred to an inspection position for detailed visual characterization before inserted into a fuel rod canister. The empty position in the fuel assembly is checked with a friction force mandrel for sufficient spring load in the spacer cell. This position is then filled either by a Zircaloy dummy rod or by a ceramic fuel rod.

Special Procedures

Besides the exchange of defective fuel rods several special procedures have been developed in response to various tasks which evolved during the long time reconstitution experience. Table 1 contains a summary of reconstitution tasks which are described as in the following.

Table 1: Special Reconstitution Tasks on Light Water Fuel Assemblies

<table>
<thead>
<tr>
<th>Task Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>Removal of weld-locked screws</td>
</tr>
<tr>
<td>Removal of broken rods</td>
</tr>
<tr>
<td>Removal of debris</td>
</tr>
<tr>
<td>Readjustment of spacer springs</td>
</tr>
<tr>
<td>Upgrading features for the fuel assembly structure</td>
</tr>
<tr>
<td>Insertion of dummy rods with retractable springs</td>
</tr>
<tr>
<td>Transfer of fuel rods into another skeleton</td>
</tr>
<tr>
<td>Exchange of structural parts (end fittings)</td>
</tr>
</tbody>
</table>

Removal of weld-locked screws

Other vendors often use welded pins to lock the screws connecting the lower end fitting to the guide thimbles in the countersink of the end fitting. These pins have to be removed to enable the lower end fitting to be unscrewed. The design of our replacement screws is such that they can be locked mechanically. Electro-discharge machining (EDM) equipment, see Fig. 6, using the spark erosion process has been developed for this purpose. Removal of all lock-pins and preparation of the lower end fitting tie plate for mechanical locking of all screw positions is achieved by a manifold electrode in one working step. The electrode is also shown in Fig. 6. The application of this technique enables re-use of the original end fittings, thus reducing radioactive waste to a negligible amount.

Removal of severed fuel rod parts (broken rods)

Cause of severed fuel rods could be heavy vibration due to cross flow at baffle joint positions or secondary hydriding of a long-time-operated defective fuel rod. The first step is the removal of those rod parts from the fuel assembly which could be gripped from the open end of the assembly. Then the adjacent rods must be removed to achieve sufficient space for special pliers. After that, the fuel assembly is brought back into its original orientation and suspended from the fuel assembly gripper. The remaining fuel rod part is either pulled sideways out to the periphery of the fuel assembly or is pushed step by step through the spacers up to a position at which it can be gripped after the fuel assembly has been re-opened.

Readjustment of spacer springs

A frequent occurrence is the deformation of the springs in spacer cells when defective fuel rods with locally increased diameters have to be withdrawn. These springs can be readjusted with a tool which grips the spring from the inside of the spacer cell and pulls it to the cell center (see Fig. 7). The success of readjustment is checked by measuring the friction force with a calibrated mandrel passing between spring and opposite dimples of the spacer cell.

Use of dummy rods containing retractable springs

Normally, defective rods are replaced by donor fuel rods or solid Zry rods. In case where the spacer cells are so extensively damaged e.g. by baffle jetting that springs or dimples are missing, the standard method cannot be used. The replacement rod would not be held sufficiently firm in this case and the exchange of the fuel rods to a new skeleton was necessary. To avoid cost for waste disposal of the old irradiated skeleton would be and the effort of fuel rod transfer, a special dummy rod has been
Fig. 7: Reallocation of spacer springs (cross-section through a spacer cell), a) pliers in open position for gripping behind deformed leaf spring; b) pliers in closed and backward position with readjusted leaf spring. (1) tool spindle; (2) guide piece; (3) pliers; (4) deformed leaf spring; (5) spacer strip.

Developed. The special dummy rod is equipped with retractable springs and/or dimples at the appropriate axial spacer positions. The retractable springs are actuated after the special dummy rod is correctly placed. The functioning of the retractable spring is schematically shown in Fig. 8. Such special dummy rods were inserted into several PWR fuel assemblies during 1987.

Upgrading Measures for the Fuel Assembly Structure

Specific procedures are applied to improve or repair the structure of the fuel assembly. For instance, when peripheral spacer strips are damaged by baffle jetting, additional spacer segments or fixation parts were used to provide sufficient hold for the rods in this region. There are also tools available which allow the readjustment of bent guide vanes on peripheral spacer strips. The exchange of the end fittings can be carried out if needed. Hold-down springs of PWR fuel assemblies are easily exchangeable.

Skeleton exchange

In cases where the spacers or guide tubes of a skeleton are severely damaged, the transfer of the irradiated fuel rods to a sound skeleton is the only way of reconstitution. This work can be performed with the standard reconstitution equipment supplemented by components which take up the replacement skeleton.

4. APPLICATION OF RECONSTITUTION TECHNIQUE FOR OTHER PURPOSES

Of special advantage to utilities is the availability of reconstitution equipment when a sudden problem such as debris within a fuel assembly or baffle jetting occurs during a shutdown period. Debris within a fuel assembly may damage the fuel rods during further operation; in some cases, they must be removed by opening the fuel assembly.

Fuel rods were replaced by solid steel rods as a preventative measure to ensure that no fuel rod will be damaged during further operation in a region of the core influenced by baffle jetting. This allows the utility to operate the reactor without the risk of fuel rods failing until such time as appropriate equipment is available to repair the core baffle.

The versatility of the reconstitution equipment allows it to be used for fuel assembly examination programs. Availability of reconstitution equipment makes it possible to perform routine inspections on individual fuel rods (see Table 2). A multiple measuring device can be mounted to the reconstitution equipment. It is equipped with an LVDT diameter gauge, an oxide thickness probe, an encircling eddy current coil and a TV camera. Helical or linear scans can be obtained by passing the fuel rod through the measuring device, with or without simultaneous rotation. Outer diameter, oxide thickness and inhomogeneities of the cladding tube can be measured and recorded. A simultaneous visual observation of the rod surface helps to perform routine inspections on individual fuel rods.

Table 2: Possible Inspections on Individual Fuel Rods with the Aid of the Reconstitution Equipment

- Visual inspection
- Eddy current testing
- Length measurement
- Diameter profilometry
- Oxide layer thickness profilometry
- Crud sampling
interpret the findings /5/. The eddy current system can also be used with a fixed reference coil to determine the metal loss of cladding tubes e.g. by nodular corrosion (BWR).

PWR fuel rod corrosion shows the maximum layer thickness in the upper third. A more uniform layer can be achieved by inverting the rods at an intermediate burnup. This has been initially investigated on several individual rods which have by now reached a rod average burnup of up to 59 GWd/t(U), see Fig. 9 /6/. Based on the excellent performance of these rods, the rod inversion test program was continued on a demonstration scale. The fuel rods of 4 test assemblies were inverted after two cycles, corresponding to half their target burnup. The inversion of fuel rods of complete fuel assemblies has been performed by interchanging the fuel rods between two assemblies with the first one in upright orientation and the second one inverted (see Fig. 10). This technique required the removal of the upper end fitting of one fuel assembly and the removal of the lower end fitting of the second fuel assembly.

5. EXPERIENCE

Reconstitution equipment and techniques are standardized and successfully applied since 1976. Up to now, 230 fuel assemblies have been reconstituted for different reasons. The majority of these assemblies have already been reinserted and have shown excellent performance during subsequent irradiation. The assemblies have been in operation for up to 3 cycles. Table 3 summarizes the experience to date. The operation time of PWR fuel assemblies before reconstitution was between one and two operational cycles. All these fuel assemblies and a large number of assemblies were inverted for examination purposes and no defects were observed due to this operation.

Some of these fuel assemblies were pretested with an ultrasonic testing system for leaking rods. An advantage of this system is the reduction of the overall reconstitution time, since only the indicated rods have to be eddy current tested. The system, which indicates the presence of water in defective rods, operates with sufficient accuracy and a tendency towards slight overprediction. After subsequent operation, leaking fuel
Table 3: KWU Experience with Fuel Assemblies Reconstituted for Re-use since 1976 (Status 1987)

<table>
<thead>
<tr>
<th>Type of Fuel Assembly</th>
<th>Number of Fuel Assemblies Reconstituted</th>
<th>Operating Cycles before Reconstitution</th>
<th>Fuel Assemblies Reconstituted to Date</th>
<th>Operating Cycles since Reconstitution</th>
</tr>
</thead>
<tbody>
<tr>
<td>PWR (KWU)</td>
<td>137</td>
<td>1 - 2</td>
<td>112</td>
<td>1 - 3</td>
</tr>
<tr>
<td>PWR (Non-KWU)</td>
<td>35</td>
<td>1 - 2</td>
<td>7</td>
<td>1 - 2</td>
</tr>
<tr>
<td>BWR (KWU)</td>
<td>58</td>
<td>1 - 2</td>
<td>53</td>
<td>1 - 3</td>
</tr>
<tr>
<td>All Types</td>
<td>210</td>
<td>1 - 2</td>
<td>192</td>
<td>1 - 3</td>
</tr>
</tbody>
</table>

rods were recognized in a few cases where fretting due to insufficient spacer hold force or debris were the main failure mechanisms. According to this experience it is recommended to perform 100% eddy current testing when fretting is the main failure mechanism. Otherwise one has to consider whether or not some defects during subsequent operation can be accepted.

The design of BWR fuel assemblies permits access to fuel rods from the upper end only. All the rods have been tested by eddy current. Only one rod was found to be defective after subsequent operation, due to misinterpretation of only one eddy current signal which indicated a small inside cladding defect during the reconstitution campaign.

6. CONCLUSION

Favorable operational experience with reconstituted fuel assemblies confirms the applicability of the techniques used. Reconstituting defective fuel assemblies provides significant economic benefits since the considerable cost of replacing a defective but not fully expended fuel assembly with a new one can be avoided. Continued use of such assemblies could increase fission product release to the coolant and thus the radiation exposure of personnel during maintenance periods.

The concentration of defective rods into one unit (fuel rod canister) limits the distribution of fission products during intermediate storage in the spent fuel storage pool or at the reprocessing plant.

Routine fuel inspections and non-destructive examination on a large scale provide continuous documentation of fuel performance and reliability. These data form the basis for design verification and further fuel assembly development.

Further refinement of the reconstitution technique is underway to reduce working time and exposure of the operating personnel. This is achievable through computer-controlled processes and an adequate on-line data acquisition.

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1.0 INTRODUCTION

The General Electric Company has an active and aggressive fuel inspection program in place with qualified personnel, equipment and procedures to disassemble, inspect, repair and retrieve components from irradiated BWR fuel assemblies. These activities have a 25 year experience base at GE and continue to provide an integral part of fuel technology and engineering's fuel performance data base and makes available a fuel repair service to the BWR customers. To date, over 5,000 production fuel assemblies have been inspected (not including fuel sipping) and over 55,000 individual fuel rods have been visually inspected or nondestructively examined. Over 100 LTAs and special fuel assemblies have been inspected and over 50 are presently undergoing irradiation, with new designs being proposed for the future. Inspections have been performed at 21 domestic plants and 10 foreign plants, from BWR 1 through BWR 6.

The inspection capabilities include visual examinations with photographic and video documentation, fuel bundle disassembly and reassembly, rod to rod spacing, fuel bundle growth, individual rod length and diameter changes, individual rod flaw detection and oxide thickness, instrumented segmented rod interrogation, fuel assembly and rod gamma scanning, rod fission gas extraction/retrieval, cladding crud scraping and the retrieval of fueled and unfueled assembly components for hot cell examination. Many of these functions have been reported on in detail elsewhere and utilize industry accepted methods. This paper will address equipment/techniques which are relatively new or under development at GE and have not been previously reported.

2.0 LEAD TEST ASSEMBLY SURVEILLANCE

A key element in the development of new fuel designs and materials is the irradiation and outage surveillance of prototype lead test assemblies (LTAs). GE's evolutionary introduction of new concepts is predicated on successful demonstration of the product in a typical BWR environment. LTAs are thoroughly characterized prior to reactor insertion and specified components are either inspected/measured during outages or retrieved to hot cells (i.e., Vellecitos Nuclear Center) for destructive examination.

2.1 Visual Examination

The cornerstone of GE's visual inspections has been a high resolution underwater periscope which permits detailed viewing of an assembly in the fuel prep machine/elevator (FPM). Individual rods which have been removed are translated in front of the periscope optics. The periscope offers single operator viewing through an eyepiece, a 35mm camera or polaroid still photography access port and a TV camera interface (see Figure 1). GE has been using a high resolution (600 lines) color TV camera with a monitor of similar resolution (650 lines) color TV camera is being developed. The zoom lens system will have sufficient magnification (30x) to allow close inspection while maintaining the sensitive camera components at a sufficient distance from the high energy gamma sources to preclude image degradation. New color chip technology has a demonstrated high sensitivity to low radiation fields and therefore, 3-tube Saticon camera electronics will be utilized. The camera envelope will be a cylinder 20 cm in diameter and 60 cm long with a near zero weight in water (16 kg dry).
The eventual replacement of periscopes with underwater color cameras will substantially reduce equipment shipment costs, setup time, maintenance and replacement costs. Still photographs can be taken of the high resolution monitor during the inspection (straight from the high resolution cameras) and the quality will rival still photographs taken through the periscope. Images recorded on cassette tape can also be photographed from the monitor or with a freeze frame device; however, the quality will be somewhat reduced due to recorder resolution limitations.

2.2 Cladding Oxide Thickness

The determination of oxide thickness on the waterside surface of the fuel cladding is accomplished with an eddy current probe device which measures the distance (lift-off) of the EC probe from the curved surface of the cladding. The EC probe is 2.5 mm in diameter. The probe electronics utilize a 3 MHz excitation frequency and can accommodate barrier and nonbarrier cladding with the proper application of standards. A schematic of the system is shown in Figure 2.

Recent developments include a device which spring loads two opposing EC probes onto the cladding generating a continuous axial scan of oxide thickness at 0° and 180° azimuthal orientations simultaneously. The probes pass over a standard with known thicknesses of oxide (gage lab characterized) prior to engaging the cladding surface so each trace incorporates the standard thicknesses for comparison (see Figure 3).

A data acquisition system (DAS) has been developed to digitize the analog signal generated for ease of data manipulation and reduction. The system records a reading at every 1.3 mm axially, and is able to compare the reading against the standards for quantification and subsequent generation of an axial plot of oxide thickness (see Figure 4). The DAS can axially amplify any region of interest for a more detailed presentation (see Figure 5).

2.3 Instrumented Segments

GE began irradiating segmented rods in BWR power reactors in the mid-1970s to facilitate post irradiation shipments, subsequent test reactor irradiations, and hot cell PIE. A segmented rod has the same envelope dimensions as a standard rod but consists of four equal sections which are screwed together and are readily disassembled. A recent development with segmented rods has been the incorporation of internal instrumentation which permits obtaining internal fission gas pressure, fuel column stack length change, and cladding elongation during reactor outage inspections.
OXIDE THICKNESS STANDARDS: .001, .002, .003, .004 INCH

FIG. 3. FOXI analog strip chart trace with integral standards.

FIG. 4. ROXI D.A.S. processed oxide thickness for full rod length.

FIG. 5. ROXI D.A.S. processed oxide thickness for reduced length (10 inches).
The technology used for the instrumented segment measurements was developed by the Institute for Energiteknikk, OFD Halden Reactor Project in Halden, Norway. Halden developed the pressure bellows application and the Linear Variable Differential Transformer (LVDT) Zero Crossing Detection unit essential to performing the measurements. Each instrumented segment contains four ferritic metal slugs which can be axially located by passing a detector head containing three encircling coils over the surface of the cladding. The coils are electrically connected to LVDT electronics as shown in Figure 6. Figure 7 shows a schematic of the axial placement and function of the ferritic cores. Prior to irradiation, each of the core locations was established for comparison to subsequent measurements.

The bottom core is fixed at the lower end plug and becomes the reference location for the fuel column stack change and the overall cladding length change. The second core from the bottom sits on top of the fuel column and is free to move with the fuel pellet column. The axial location of the second core with respect to the bottom reference core establishes fuel column length changes. The third core from the bottom is attached to the free end of a collapsible stainless steel bellows in the plenum region. The fixed end of the bellows is attached to the upper end plug (UEP). As the internal pressure changes, the bellows flexes and moves the third core with respect to the fourth core which is fixed to the UEP. The bellows deflection is calibrated to external bellows pressure and changes in axial position of the third core with respect to the reference fourth core translates into internal gas pressure. Figure 8 shows a typical ferritic core axial location plot for a segmented rod from which the computer accurately fixes their axial positions.

The measurement requirement was one of accurately determining the axial position of the LVDT coil between two core zero crossing points. This was achieved with an axial position indicator utilizing a magnetic strip called a SONY gage. For the bellows core location which required the greatest precision, an accuracy of ±10 microns was achieved. Creep characteristics of the stainless steel bellows in the hard neutron flux spectrum of a BWR are being evaluated after the first cycle of irradiation of these devices was completed in March 1987. The previous in-reactor experience with these instruments and the creep data base had been limited to the low power density Halden test reactor.
2.4 Laser Puncture and Reseal

The release of fission gas from the fuel matrix during irradiation and the buildup of internal rod pressure is a characteristic of interest to fuel rod designers. The retrieval and subsequent disposal of irradiated fuel rods from power reactors is very costly and has provided only a limited fission gas release database. It has proven more practical to puncture the plenum of sound discharged fuel rods at reactor sites, measure the gas pressure and retrieve a gas sample for analysis. Previous techniques utilized a hardened punch to make a hole in the cladding and a mechanical seal over the hole to preclude subsequent outgassing during storage; however, a long-term hermetic seal was not obtained. Laser techniques can replace the mechanical punch and a defocused laser beam will reseal (hermetically seal) the hole.

The rod puncture and gas retrieval system which will utilize the laser is similar to a previous device which utilized the mechanical punch and is shown schematically in Figure 9. The fuel rod is captured vertically and a leak-tight clamshell puncture head is secured in the plenum region about 2 meters under water. All water is flushed and evacuated out of the puncture head using an inert gas. The laser system is located above water and the beam is aimed vertically downward through an aluminum beam pathway tube with appropriate optics to redirect the beam horizontally through a quartz window onto the rod’s plenum region as shown in Figure 10.

The laser is a 2.8 KV lamp voltage neodymium: glass (Nd: Glass) pulsing laser for both puncturing and resealing. The diameter of the hole created will be about 0.25 mm. After puncture, the fission gas is expanded into a calibrated volume of approximately the rod’s internal volume, and the pressure is recorded. A valve allows the gas to expand into a second calibrated volume and the second pressure is recorded. The fuel rod fission gas pressure and free volume are calculated using the measured pressures and known calibrated volumes. A sample of the gas can be taken for retrieval and subsequent gas chromatography analysis if required.

It is anticipated that the internal pressure detection bellows contained in the instrumented segments described in section 2.3 can be recalibrated by measuring bellows deflection during known repressurization with this system. Recalibration can yield insight into, and permit compensation for, the irradiation hardening of the bellows material.
After gas flushing of the system, the fuel rod cladding is seal-welded by defocusing the laser beam. The seal-weld is presently not qualified for continued in-pile operation; however, the rod can be considered sound for handling, shipping and storage.

3.0 IRRADIATED FUEL ASSEMBLY IDENTIFICATION AND REPAIR

GE has, on occasion, been requested to repair damaged irradiated fuel assemblies when the value of continued irradiation outweighs the costs of repair. Assembly damage requiring repair has, for example, occurred to spacers during the channeling operation, upper-tie-plate (UTP) bail handles bent during fuel movements, and early in life fuel failures for various reasons. The relatively simple underwater disassembly of the GE BWR fuel assembly allows for removal and replacement of all fuel rods and virtually all components.

3.1 Failed Bundle Identification

Out-of-core vacuum sipping is a method used by GE to detect leaking fuel assemblies which require the removal of individual assemblies from the core into an isolated environment. It is similar to the out-of-core wet sipping method with the exception that a volume of gas, under reduced pressure, is circulated through a closed loop and the gas activity is directly assayed for radioactive fission gas.

3.2 Failed Rod Identification/Characterization

The identification of leaking fuel assemblies for their removal prior to subsequent reactor operating cycles is usually sufficient from a reactor operations standpoint; however, if the specific leaking fuel rods need to be identified for removal, replacement or failure characterization, additional testing is required. At GE, the identification of individual failed rods is accomplished by a combination of three techniques. Each individual rod is removed and electronically interrogated with an encircling coil eddy current searching for cladding defects and an ultrasonic probe (UT) searching for water internal to the cladding. These measurements are complimented with a visual inspection which typically keys on the cladding regions where strong EC signals were observed. Between the EC, UT and visual inspection, individual failed rods are identified and the failure mechanism characterized.

The EC and UT transducers are mounted together in a fixture which attaches to the refuel pool FPM and is translated up and down adjacent to, and with, the fuel assembly being inspected. The individual rods are extracted from the assembly with a hand held collet grapple pole tool, placed in a fixed vertical position and the EC/UT head is passed over the entire length of the rod. A schematic is shown in Figure 11.

The eddy-current system consists of conventional eddy-current instrumentation with multi-frequency and multiple coil capabilities, an encircling coil assembly which contains two independent differential coil pairs and a calibration standard which is fabricated from fuel cladding. The eddy-current instrumentation is arranged so that the input gain, the excitation frequency and the phasing can be independently adjusted for each of the differential coil pairs. In addition, to avoid cross communication between multiple coils operating simultaneously, the eddy-current instrumentation contains a multiplexer which restricts the coil excitation to one coil at any given instance. For cladding flaw detection, the two coil windings are arranged in the encircling coil spool to be adjacent to each other. This arrangement, which is commonly known as the self examining differential coil, utilizes the trailing coil winding as a reference for the leading coil winding. During an axial traverse of the coil over a defective region of the fuel rod cladding, the impedance of the leading coil winding will be affected by the approach of a defect. The impedance imbalance between the two coil windings is registered by the instrumentation as a vector quantity. The resulting signal amplitude and phasing may be employed to analyze the nature and magnitude of the defect.
3.3 Assembly Component/Fuel Rod Replacement

A damaged or failed fuel assembly can be repaired in several ways depending on the circumstances and condition of the assembly. If a fuel rod has failed and can be safely removed or has been inadvertently damaged during inspection, a compatible replacement rod can simply be inserted into the open location. Replacement rods are made available either by manufacturing new fuel at GE's Nuclear Fuel and Component Manufacturing (NF&CM) facility accommodating length and enrichment requirements, or through the use of irradiated fuel of appropriate burnup from a discharged "donor" assembly. A donor assembly is irradiated but has been discharged and contains unfailed candidate replacement rods of suitable enrichment and mechanical dimensions.

The damage sustained by an assembly during operation or handling may preclude continued operation of the basic assembly structural components. Such situations have occurred, for example, when one or several spacers have been inadvertently disengaged from the capture rod and relocated by the channeling operation or when secondary hydride damage has occurred on a failed rod preventing its safe removal and replacement. In these situations, the remaining undamaged fuel rods are relocated into new structural hardware called a "skeleton" assembly utilizing replacement rods, if required, that are either manufactured new or obtained from donor assemblies. The skeleton assembly consists of unfueled assembly components including lower-tie-plate (LTP), eight dummy tie rods, and the upper-tie-plate (UTP). The skeleton components are fabricated at GE's NF&CM facility, are reactor certified/inspected, and assembled at the reactor site.

The GE BWR refuel pool is typically equipped with two fuel prep machines (FFMs). The damaged assemblies are placed in one FFM and the assembled skeleton in the other. Rods are moved individually by hand from the damaged assembly to the skeleton. The dummy tie rods are replaced last by the fueled tie rods. The skeleton assembly, now containing the irradiated and any new fuel rods, can utilize the original UTP (if undamaged) maintaining the original serial number for accountability. The damaged components and/or failed fuel rods can be left in the damaged structure with the dummy tie rods maintaining overall geometry or the components can be disassembled for compact storage and disposal. The skeleton can be used to accumulate failed fuel rods from various assemblies creating a "super leaker" assembly for consolidated storage, shipping and disposal.

The replacement and exchange of fuel rods in irradiated assemblies which will continue irradiation requires both mechanical and nuclear considerations. The end plugs must be compatible with the UTP and LTP. The replacement rod must be of sufficient length to compensate for axial growth of the assembly and provide proper end plug engagement. The nuclear characteristics of a replacement rod, i.e., enrichment, must be compatible with reactivity and burnup characteristics of the rod being replaced. A design analysis is performed to ensure that the repaired assembly has the same design margin as the original assembly at its exposure and no special licensing activity is required for continued irradiation.

4.0 SUMMARY

The GE BWR fuel assembly design exhibits relatively simple underwater disassembly characteristics which facilitates routine fuel surveillance inspections of LTAs and repair of damaged assemblies at reactor sites during refueling outages. All components can be visually inspected in the refuel pool, a variety of physical measurements can be performed on the fuel rods and a damaged fuel assembly can readily be repaired for continued irradiation. Improvements in underwater visual inspection have resulted in high quality photography and color video documentation. Virtually all external mechanical dimensions of a fuel rod can be measured underwater in a reactor's refuel pool and new developments have allowed measurement of some internal rod performance characteristics. Fuel assembly repair activities can be routinely performed during refueling outages as well.
RECONSTRUCTION TECHNIQUE ON THE POWER REACTOR FUEL ASSEMBLIES AFTER POST-IRRADIATION EXAMINATIONS

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Abstract

This paper describes the reconstructions of PWR fuel assemblies after finishing the post irradiation examinations (PIEs) in hot cells of Reactor Fuel Examination Facility (RFEF) in Tokai Research Establishment, Japan Atomic Energy Research Institute (JAERI). Since December 1979, the RFEF has examined 10 fuel assemblies in order to certify the integrity of the currently designed reactor fuel assemblies used in PWR, BWR and HWR, respectively. The PIEs have been programmed by stopping over at the hot laboratory on the way of transportation from the power reactor sites to the reprocessing plant, so that fuel assemblies should be brought to the reprocessing plant after finishing the PIEs. Some out-cell examinations and trial tests had been performed in advance the reconstitution of fuel assemblies, and some manufacturing and inspecting devices were developed to assure the integrity of fuel assemblies.

The results of these works were applied to the procedure as follows:

1) Remaking the punctured fuel rod.
2) Inserting the fuel rod into the assembly.
3) Remaking the top nozzle.

These PIEs procedures were applied to four PWR fuel assemblies and their reconstructions were successfully carried out. Three out of four fuel assemblies were transferred to the reprocessing plant as of July, 1987. Another one is scheduled in December this year. Moreover, two BWR fuel assemblies were transferred to the reprocessing plant after finishing the PIEs and reconstructions, successively as of July, 1987.

1. Introduction

The RFEF has been operated since December 1979, at Tokai Research Establishment, JAERI, and has carried out the PIEs of 10 fuel assemblies for the certification of the integrity of the currently designed reactor fuel assemblies used in PWR, BWR and HWR, respectively.

The PWR type fuel assemblies have been shipped to the RFEF using transfer cask.

The dimensional measurement after the visual inspection was carried out for the fuel assembly. Then, the fuel assembly was disassembled for the investigation of fuel rods. Non destructive Tests (NDTs) such as visual inspection, dimensional measurement, gamma-scanning, X-ray radiography and eddy-current test were conducted on the fuel rods withdrawn from the fuel assembly.

Based on the results of these NDTs, the fuel rods were successively examined on the destructive tests (DTs) including puncture test, ceramography of UO2 pellets, metallography and tensile test of fuel cladding and so on.

On finishing the planned procedures of the post irradiation examinations (PIEs), the fuel assembly was divided into several parts such as the fuel assembly without some fuel rods, the top nozzle, the fuel rods punctured and not punctured and the sectioned fuel rods.

The purposes of the reconstructions are to return these parts of fuel materials to the original positions in the fuel assembly except the sectioned fuel rods and the top nozzle. The several techniques for the reconstructions were designated and some remote operating tools and machines were produced and tested in preceding out-cell testings. Applying the results of these preceding works, the fuel assemblies were reconstituted successfully and transferred to the reprocessing plant. The flow of the disassembly and the reconstitution processes is shown in Fig. 1.

2. Disassembly

2.1 The structure of PWR fuel assembly

The three fuel assemblies, transferred to the reprocessing plant after PIEs, were 15X15 PWR type domestic fuel assemblies irradiated in one, two and three cycles, respectively.
FUEL ASSEMBLY DISASSEMBLY

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RECONSTITUTED FUEL ASSEMBLY

FIG. 1. Flow diagram of the processes in disassembly and reconstitution.

The PWR type fuel rods have been supported by inconel grids that are almost equally spaced intervals along their lengths and are mechanically fixed to thimble tubes. These are attached to the top and bottom nozzles of the fuel assembly. The combination of thimble tubes, grids, top and bottom nozzles is so called the skeleton assembly, and this structure supports the fuel rods. The feature of PWR fuel assembly is shown in Fig. 2.

FIG. 2. Feature of typical PWR fuel assembly.

1.2 Removal of top nozzle

The removal of top nozzle was done by cutting the weld part (3 mm depth) between top nozzle and thimble tubes as shown in Fig. 3. The positioning accuracy of the milling cutter is 1/100 mm. Adjusting the center of fuel assembly needed to cut the weld part was conducted after grasping the assembly tightly by the cramp equipped in the disassembling pit.

2.3 Withdrawal of fuel rods

After removing top nozzle, the fuel rods to be performed NDTs and DTs were withdrawn from the fuel assembly standing upright in the disassembling pit in hot cell. The withdrawal of fuel rod from the assembly was done with a collet set on the top end of fuel rod and pulled up by the hoist located in the upper part of hot cell.

Rod withdrawal forces were measured with load-cell during removal of fuel rod from the fuel assembly to check the relaxation of grid springs. An example of recording chart of rod withdrawal force measurement is shown in Fig. 4. The withdrawn fuel rods were transferred to next stages of PIEs.
2.4 Successive works after PIEs

The fuel rods after finishing NDTs and DTs and only punctured fuel rods were replaced in the fuel assembly. However, the top nozzle and cladding materials were abandoned as radioactive wastes, and the metallography specimens were treated as radioactive wastes because of difficulty of dissolving the resin in the reprocessing process of UO2 pellet.

The fuel assemblies reconstituted after finishing the PIEs have been transferred to the reprocessing plant.

3. Remaking of the fuel rod

3.1 Cutting of the puncturing port

Some fuel rods, which were punctured at the plenum portion to investigate the release behavior of gaseous fission products after NDTs stages, were cut off at the plenum portion (about 50 mm length from the top end of fuel rod) including the puncturing point. Remaking diagram of punctured fuel rod is shown in Fig. 5.
3.2 Inserting of the new plug

The new plug was inserted to the irradiated zircaloy cladding tube with the plenum spring after removing the punctured part of fuel rod. The elimination of oxide of the irradiated tube was done using the brush of stainless steel wire in order to make a sound welded joint of the fuel rod.

The welding techniques between the irradiated zircaloy tube and the non-irradiated zircaloy plug had already been developed in order to make the burst test specimen to check the mechanical strength of cladding tube in the standard course of PIEs in the facility, and the strength of the welding parts had been proven by testing every specimen.

The welding of the part between the plug and the irradiated zircaloy cladding tube was carried out using the non-filler TIG welding process. It was found that the welding part had plenty of mechanical strength. The details of welding torch is shown in Fig. 6, in which the fuel rod is fixed and the welding torch is rotated. The Fig. 7 shows the aspect after circumferential welding.

3.3 Welding for sealing

The small hole (0.8 mm) drilled in the head of the new plug was welded by the seal welding torch, after replacing inner gases of fuel rod with helium (1 atm.), as shown in Fig. 8. This welding uses non-filler TIG process, too.
3.4 Helium leak test

The helium leak test of the welded part of each rod was performed to ensure the leak rate less than $10^{-7}$ atom·cc/sec using the leak detector device with a vacuum system.

4. Inserting of the fuel rods into the assembly

4.1 Processing of the narrow groove

The narrow groove (width 1 mm, depth 8 mm) was machined on the bottom end plug of fuel rod in order to fix the guide cap shaped conically by welding. A diamond wheel cutting machine was used to make the narrow groove.

4.2 Welding of guide cap

A guide cap, as shown in Fig. 9, was prepared for guiding the fuel rod into the appropriate cell not so as to bent the mixing vanes on every grid. The guide cap was circumferentially welded to the bottom end plug of the fuel rod using non-filler TIG welding process, after inserting the teeth of guide cap into the narrow groove.

4.3 Inserting of fuel rods

The rods which were withdrawn from the assembly for the purpose of NDTs and DTs were inserted into the appropriate cell using the crane located near the ceiling of hot cell. The collet connected to hook of the crane was set on the upper end of the rod as pulling up the rod. View of top side after inserting the fuel rods is shown in Fig. 10.

5. Welding of nozzle plate to guide tubes

5.1 Elimination of the oxide on outer/inner surface of thimble tubes

The oxide generated on the outer/inner surface of thimble tubes was eliminated not to influence upon the weld.

5.2 Fixing of the nozzle plate

The guide rods with different length were attached to the scraped top end of the thimble tubes to insert the new adaptor plate easily. After that, the adaptor plate was fixed by the retention tool to make a surface level.

FIG. 9. Attachment of guide cap to the bottom end of the remade fuel rod.

FIG. 10. View of top side after inserting fuel rods.
5.3 Treatment before welding

The diameter of thimble tube was enlarged at the contacting part of the plate to remove the gap between the plate and the tube, and the protruded part of thimble tubes above the plate was cut off by end mill. The chips generated on the cutting surface were eliminated with the brush of stainless steel wire.

5.4 Welding of the adaptor plate to thimble tubes

Previously, mechanical strength tests of welded part between adaptor plate and thimble tubes were carried out to assure the strength of sustaining the whole weight of about 700 kgs.

The adaptor ring was attached to the circular grooves processed on the adapter plate as shown in Fig. 11. The circular welding between plate and tubes was carried out by non-filler TIG process at twenty places after adjusting the welding torch equipped in the disassembly machine.

The visual inspection of weld part was carried out by periscope, and any kinds of unsoundness were not found out. The aspect of welding part between adaptor plate and thimble tubes is shown in Fig. 12.

5.5 Attachment of frame

The frame was fixed to the plate by screws after welding. The reconstituted fuel assembly was pulled up by in-cell crane and transferred to the pool. Final view of the reconstituted fuel assembly is shown in Fig. 13.

6. Conclusion

Since December 1979, the HKKK has examined 10 fuel assemblies in order to certificate the integrity of currently designed fuel assemblies.

The reconstitutions of four fuel assemblies out of ten have been carried out successfully. The techniques and methods concerning the reconstitution of fuel assemblies have been developed. There are three important points, as follows, to execute successfully the reconstitution of PWR fuel assemblies.
1) By the use of the present torch under the remote controlled operation, it was made clear that circumferential welding of the part between new plug and irradiated zircaloy cladding tube had plenty of mechanical strength and that the seal welding of the small hole drilled in the head of the new plug was conducted successfully.

2) As the guide cap was able to be welded to the lower end of fuel rod, the fuel rod was correctly and easily inserted into the individual cell of grid without bending the mixing vanes.

3) Enough strength of welded part between nozzle plate and thimble tubes, for pulling up the reconstituted fuel assembly, was obtained using non-filler TIG circular welding equipment.

These procedures were applied to the four PWR type fuel assemblies and their reconstitutions were successfully completed. Three out of four fuel assemblies were transferred to the reprocessing plant of PNC (Power Reactor and Nuclear Fuel Development Corporation) as of July, 1987. Another one is scheduled in December this year.

7. Acknowledgement

The authors are deeply indebted to Dr. Y. Nishima, Professor Emeritus, University of Tokyo, for his valuable advice on the post irradiation examinations and so on at the RPEP. The authors gratefully acknowledge Nuclear Power Engineering Test Center (NUPEC) for permitting to this presentation, too.

REFERENCES


RECONSTITUTION OF BR3 FUEL ASSEMBLIES

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Abstract

The BR3 reactor is loaded since 1976 (cycle 4) with dismountable assemblies, allowing the extraction and reinsertion of fuel rods at any place in the assembly. The assembly is designed following a square lattice having the 17 x 17 pitch and accepts rods of 9.5 mm outer diameter. A direct access to the rods' upper or lower extremity is obtained by unscrewing bolts maintaining the bottom and top nozzles to the assembly tie-rods.

Rod manipulation, carried out in the reactor storage pool, is performed by means of an handling station allowing the complete reversal of the assembly, for eventual rod extraction by the assembly bottom side. The station is equipped with periscope coupled to TV camera for visual inspection and handling controls; the equipment includes rod handling tools, protection basket and transport basket as well as a sipping cell.

The rod and assembly handling system has been used for assembly repair, core surveillance programme and rod examination. To date, more than 20 assemblies have been opened and closed and about 1,600 fuel rods have been extracted or inserted successfully. Minor difficulties were encountered during these handling campaigns, such as spacer grid spring deterioration by rod insertion. These difficulties are now eliminated by using a new spring design.
Assembly repair has been performed successfully after cycle 4B during which fuel assemblies failed due to cladding outer corrosion at the end of their first irradiation cycle. Defective rods were identified and extracted together with rods susceptible to be near the rupture; sound rods were reassembled in existing skeletons and reirradiated to their target burnup.

The core surveillance programme includes assembly mechanical behaviour follow-up and neutronic evaluation. Mechanical design is mainly related to assembly length increase and spacer grid spring relaxation, the latter being evaluated during the rod handling. Neutronic core follow-up consists in a gamma-spectrometry made on site during special campaigns in order to evaluate the effects related to the gadolinium consumption.

Rod handlings are regularly performed in the frame of intermediate or final postirradiation examinations aiming at assessing the in-reactor behaviour of rods having different designs or fabrication routes. $\text{UO}_2$, $\text{UO}_2\cdot\text{Gd}_2\text{O}_3$ and MOX fuel rods are so examined in the frame of bilateral contracts or international programmes managed by BELGONUCLEAIRE in collaboration with CEN/SCK.

The existence of dismountable assemblies at BR3 offers the possibility of tailor-made irradiation conditions; rods irradiated at high burnup and submitted during reactor shutdown to non-destructive examinations in the CEN/SCK hot cells, have been reinserted in assemblies reconstituted three times and irradiated in the core centrale zone.

1. INTRODUCTION

BR3 is a nuclear power station with a net electricity power of 10.5 MWe, that was first coupled to the grid on Oct. 10, 1962. Originally, the plant was used for training electricity société's personnel. Later, after being taken over by the nuclear research centre SCK/CEN, a start was made with the programme for testing new fuels, such as $\text{UO}_2$ with high enrichment, $\text{UO}_2\cdot\text{Gd}_2\text{O}_3$ (fuel with consumable poison) mixed oxide $\text{UO}_2\cdot\text{PuO}_2$ (in co-operation with BELGONUCLEAIRE), for a number of foreign experimenters. In order to allow the experimenters as much latitude as possible concerning the number of rods and their position in the reactor (flux level), a fuel assembly was chosen that could be dismantled. In this fashion a tailor-made irradiation programme could be offered to the experimenter, with a free choice of the number of cycles and the desired burnup.

The reactor has 91 places of which 73 are occupied by fuel assemblies each with 28 rods, or 2044 positions for irradiation (Fig. 1). It was necessary to develop a facility in order, on the one hand to accommodate in an assembly the rods supplied by various experimenters, and on the other hand to shuffle irradiated rods among the assemblies. The design and installation of this facility was carried out by BR3 personnel. The manufacture of it was entrusted to an outside firm.

It was first used in 1978. Since then there have been five campaigns with dismountable fuel assemblies whereby for instance, between the last two campaigns 50 assemblies were opened and 600 rods shuffled. During the whole exploitation period only two problems have arisen: the misforming of a fuel assembly grille and the dropping of a fuel rod. The cause was not attributable to the facility itself.

Due to the cessation of the BR3 operations, this facility will be used for the last time in 1988.

2. FACILITY POSSIBILITIES

The whole facility was designed according to criteria of reliability, durability and the exclusion of errors during operation, i.e. insertion of rods in the wrong places. The operation's time factor is of secondary importance. All operations can be followed by means of an endoscope coupled to a monitor.
The following manipulations were carried out:
- Removal of the assembly head
- The transfer of a fuel rod in the same assembly between any given position
- The transfer of a fuel rod between different assemblies in any given position
- The insertion of a fresh fuel rod into any given position in an assembly
- The extraction of a fuel rod from any given position in an assembly and its transfer to a storage can
- Visual examination of a fuel rod.

The facility is designed in such a fashion that the assembly can be up-ended, and the same manipulations carried out, but then via the foot of the assembly.

During the very first operation a problem arose during the introduction of a fuel rod into an assembly. The cause was an inadequacy in the design of the grille springs. This was solved by modifying the shape of the spring. This problem will be discussed further on. Afterwards another incident occurred when a fuel rod fell, the cause of which proved to be an un-modified storage can.

Before explaining the operation of the facility, it is necessary to discuss the fuel assembly.

3. DISMOUNTABLE FUEL ASSEMBLY

The fuel assembly (Fig. 2) is of such an original design, that it is possible to withdraw and insert fuel rods in a simple manner via both the head and the foot. This assembly cage consists of 3 grilles and 6 tightening rods of which four run through to the head and foot which are dismountable. The grilles are hexagonally shaped with spaces for 28 fuel rods.
### Dimensional Information of an Assembly:

- Overall length of an assembly: 1466 mm
- Circumscribed circle of a grille: 98 mm
- Overall length of a fuel rod: 1136 mm
- Active length of a fuel rod: 1000 mm
- Fuel rod diameter: 9.5 mm
- Fuel assembly weight: +35 kg
- Fuel rod weight: +800 gr

This information is of a different order of size to the fuel assemblies used in the commercial nuclear power stations.

### 4. Facility, Operation and Tools

Various tools, each with a specific function, were designed for this facility (Fig. 3). By using a representative example, namely the withdrawal of a fuel rod out of an assembly and its insertion into a transport basket, the facility, its accessories and accompanying tools will be discussed. Knowing that all the manipulations are followed by the endoscope and monitor, this will not be explicitly emphasized.

#### 4.1. Insertion of a Fuel Assembly Into a Fuel Assembly Holder

The fuel assembly holder (Fig. 4) has three guides and removable bottom-and-top covers. The bottom guide is hexagonally shaped and precisely fits the lowest fuel assembly grille. The central guide is only functional during the introduction of the fuel assembly. The top guide is partially hinged to make it possible to introduce the assembly horizontally. The fuel assembly is placed in the fuel assembly holder as follows.

- The top cover of the fuel assembly holder is removed and the top guide is opened. With the normal fuel handling tool (F.H.T.) the fuel assembly is placed sideways in the fuel assembly holder and lowered until the foot is supported, the top guide is closed and the foot locked. The assembly is now ready for further manipulations.

#### 4.2. Removal of a Fuel Assembly Head

First of all the four nuts which connect the head to the element are loosened. This is done with a special nut handling tool (Fig. 5) that serves on the one hand to unscrew the nuts and to retain them, and on the other hand to bring on four new nuts and to screw them tight. It is designed in such a fashion that the nuts cannot be dropped. There have already been thousands of faultless manipulations carried out with this tool without losing one nut.

The nut tool support, that assures the alignment of the fuel assembly and the nut tool, is placed on the assembly. The nut tool is guided, in the open position, to the nut via the nut tool support. The three spring steel claws of the nut gripper are open in the "rest" position. By pushing the outside tube downwards the nut gripper claws close under the nut. By turning the whole tool the nut is unscrewed, removed and placed in a receptacle. This is repeated for the four nuts. The nuts are not re-used.

The head tool (Fig. 6) serves both for the removal of the fuel assembly head and for the removal of the head and foot covers of the assembly holder. The locking system works following an analogous principle as that of the nut tool except that closure occurs via a lever manipulation instead of a spring system. The tool is placed on the head, locked and the head removed. After this operation withdrawal of the rods can commence. This last tool has always worked faultlessly.

#### 4.3. Manipulation of a Fuel Rod

To withdraw a fuel rod out of, and insert back into, a fuel assembly, three specially designed handling tools are used, a positioning tool, a protection basket and the rod extraction tool.

In order to exclude all error in choosing a rod, we have at our disposal 28 positioning tools (Fig. 7) that correspond to the 28 positions of the...
rods in the fuel assembly. The desired positioning tool is placed on the assembly holder with the fuel handling tool. The opening in the positioning tool is then immediately above the rod to be extracted.

The protection basket (Fig. 8), that is placed on the positioning plate by means of the fuel handling tool, has a double function: firstly to avoid the rod being bent during the extraction or insertion and secondly to provide protection during transfer in the storage well. The protection basket has two locking devices at the top and bottom, that can be closed to lock-up the fuel and internal springs to clamp the fuel rod.

The rod extraction tool (Fig. 9) is brought immediately above and rests on top of the rod to be withdrawn, by the positioning plate via the protection basket. The rod grippers are closed and locked. During the extraction of the rod the force on the extraction tool is measured (Fig. 10). The fuel rod coming loose from the three grilles can be seen. Once the fuel rod is in the protection basket, the bottom lock is closed, the rod extraction tool is removed and the top lock is closed. The fuel rod is now ready for transport.

4.4. Placing a fuel rod in a transport basket

For the transport of fuel rods away from the power station a transport basket with 12 positions is available, with external dimensions corresponding to a fuel assembly. For insertion of the rod into the transport basket use is made of the same positioning tools as for the assemblies (12 of the 28), seeing that the positions have been made to correspond.

The procedure is as follows. The transport basket is placed in the second fixed assembly holder by means of the fuel handling tool. The head is removed. The positioning tool corresponding to the desired position is put into place. The protection basket is placed on the positioning tool and by means of the rod extraction tool the rod is lowered into the transport basket. Afterwards the transport basket is locked and is ready for transport.

There is a second type of transport basket in use, with 18 positions, which has posed a problem. This will be discussed further.

4.5. Replacing the head on the fuel assembly

A last tool was designed to lock the nuts of the fuel assemblies' head and foot. After replacing the head on the fuel assembly and the nuts have been screwed down tight, the locking tool (Fig. 10) is placed in position to clamp the nuts. Two eccentrically placed cylinders make two horizontal plates slide over one another and clamp simultaneously the two nozzles, an integral part of the nuts. The assembly is now secured.

4.6. Up-ending the assembly

All the manipulations described up to now can also be carried out on the foot of the assembly by up-ending it. After the fuel assembly as been placed in the assembly holder, the head is mounted on the assembly holder. The assembly holder is then turned by hand through 180° until it rests against a buffer-stop. The foot of the assembly holder is removed. The further manipulations are analogous to those for an assembly in the normal position.

5. PROBLEMS

During the ten years this facility has been operated, two problems have arisen the cause of which was found to be outside the facility.

Text continued on p. 121.
5.1. Grille damage

By one of the first insertions of a fuel rod into a fuel assembly, the foot of a rod jammed in a grille. The spring was completely bent in the grille. This was due to the unsuitable shape of the spring and its manner of attachment (Fig. 11). The grille springs received a completely convex shape whose ends finished in the grille (Fig. 12).

5.2. A fallen fuel rod

During the whole of the operation of the facility, only once has a rod fallen, and this when the transport basket with 18 positions (Fig. 13) was being manipulated.

When a badly bent rod was being introduced into the transport basket, the foot of the rod missed its grille position in the middle grille. This was not noticed. When the transport basket was removed the rod fell out. In the future this was avoided by following the insertion of the rod with the periscope.

FIG. 11.
As a general conclusion one can say that this facility has completely fulfilled its objective. During its operation it was ascertained that there were a few cumbersome manipulations that could be considered for simplification. This would be necessary for application in commercial power stations. Especially the manipulations with the 28 positioning tools were long-winded, although they gave complete certainty in the choice of the rod.

All this experience with the manipulating and examination of fuel assemblies will be commercialized in the design of examination stands. In Doel and Tihange (Belgium) nuclear power stations, BR3 (CEN/SCK) has installed stands for the visual inspection and dimensional measuring of irradiated fuel assemblies. These facilities are operational and give complete satisfaction.
POWER PLANT EXPERIENCE
(Session III)

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EDF EXPERIENCE ON FAILED FUEL MANAGEMENT

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Abstract

Up to now, most of the leaking fuel assemblies were reloaded
without greater consequences. In the first part, some details are given
about our methods and means including defects estimation in operation,
sipping tests (qualitative and quantitative) during shut down, and criteria
for failed fuel assemblies reloading. Statistics of failed assemblies and
reloaded ones are also given.

By another way, EDF has a significant program for reconstitution
or repair of "non reloadable" fuel assemblies (out of criteria leaking
assemblies-damaged squeletton for handling reasons - severe damages by baffle
jetting effect...). Those operations are carried out by fuel suppliers and
are described elsewhere. The second part gives the progress status of these
operations and some comments on their management, especially on fuel
assemblies transfers from one pool to another (cost and occupational doses
saving).

1 - INTRODUCTION

Failed fuel management (untightness or handling damaged) was one
of our major concerns since the beginning of PWRs operation. Our leaking fuel
reloading policy was built up in 1979 and 1980. Up to now, our operation
feed back, in this area, reaches about 150 reactor-cycles, 35 of which with
effective leaking fuel reloading.

The devices for damaged fuel assemblies reconstructions and
repairs were designed and manufactured in 1985. The first assemblies
restorations occurred by the end of the same year.

Diagram n° 1 shows a general overview and gives either in
percentage or in number, a status of our failed fuel assemblies, as of the
end of 1986. It also shows some links between the two parts of the failed
fuel management.

Our goal in this paper is to present with more details EDF
experiences in these two parts, namely the leaking fu fuel assemblies reloading
and restoration, focusing not at all on devices descriptions or technical
details, but on our policy, choices, trends and results.
analysis shows that a reloaded leaking assembly with an
equivalent diameter smaller than 35 micrometers doesn't suffer
further large evolution in the following cycles with the present
coreation conditions. This experimental value (35 μ) is our
present limit.

- The second rule concerns the total number of reloaded leaking
assemblies. The activity induced by the reloaded defects has to
be lower than 30 mCi/t (1100 MBq/t) for D.E. Iodine 131. That is
to say, we accept to invest a significant but low level of
activity in operation in order to achieve the nominal burnup of
assemblies.

2.2 - Required means for internal rules implementation

Those required means are the followings:

- Activity surveillance and cladding defects evaluation in
  operation in order to forecast the adequate control tests,
  during shut down for refuelling.

- A leaking assemblies identification, allowing a classification
  between tight and leaking assemblies.

- A defect size measurement in order to implement the first
  criterion, and to reload only the smallest defects in case of
  second criterion potential violation.

Some words about the two last items.

- Leaking fuel assembly identification

Our method and equipment for leaking fuel assembly
identification recently changed compared with the original
ones.

At first, and it is always possible in our units, we used a
combined dry and wet sipping procedure in the sipping test cells
in the spent fuel pool. This method is quite reliable and
efficient, when neither the cladding nor the pool water are
significantly contaminated (tramp uranium or fission products).

In 1981, an other method was tested, using the pressure drop
when the fuel assembly is raised from its core position to its
handling level in reactor pool, also using the closed capacity
formed by the drum of the handling machine mast.

This on-line qualitative sipping test (EDF and CEA patent) is
very few time consuming : about two additional minutes per
assembly on the critical path of the core unloading. It allows
the immediate selection of the leaking fuel assemblies for
further exams. It is not sensitive to cladding contamination.

2.3 - Results of this policy implementation

Diagram n° 2 gives a synthetic overview, by the end of 1986 of our
leaking assemblies reloading policy. Leaking assemblies are
classified according to their occurring cycle and their following
history.

Among the 134 identified failed assemblies (global failure rate of
about 1,3 % from the beginning of our PWRs operation) 50 were
normally rejected, 66 were immediately reloaded one or two times,
the reloading of 6 was delayed for different reasons and the last
12 have to be repaired. So 85 % of the leaking assemblies which
didn’t reached their nominal burnup, were (or will be) reloaded.

In another way, the fuel saving, as a result of our policy,
corresponds to the available energy stored in 28 fresh fuel
assemblies (half of a batch).

At last, diagram n° 3 gives the distribution of the end of cycle
activities for D.E. 1 131 and sum of gases. In spite of the
leaking fuel reloading policy which induces a part of the measured
activity, the mean levels remained far from the specified limits
(The highest ones in eq 1 131 were induced by baffle jetting
problems).

2.4 - Influence on the loading pattern

It is important to keep in mind that, for symmetrical power
distribution reasons, the unloading of a failed fuel assembly
(leaking or damaged) often induces the unloading of its three
symmetrical, except if it exists in the spent fuel pool an available
assembly with the same burnup or reactivity. This reveals an
unknown consequence of our leaking fuel reloading policy which is
to minimize the total number of assemblies staying in spent fuel
pool and waiting for further reloading.

All the 900 MW units are now equipped with this device. The
equipment of the 1300 MW units is now in progress.

- Defect size measurement

The defect size evaluation method is based on a kinetic analysis of
the fission gas release during a temperature variation. This
quantitative sipping test is performed in the spent fuel pool
sipping cells. It gives an "equivalent diameter" of the leak in
these experimental conditions, and our reloading criterion is
directly linked with this measurement method.
DIAGRAM 2. Leaking fuel management (internal causes) — Status as of the end of 1986.

3 - DAMAGED FUEL RECONSTITUTION AND REPAIR

Up to now, we can identify 3 main reasons which required damaged fuel restoration:

- leakers which exceed our reloading criterion,
- grid damages as a consequence of handling difficulties,
- and severe damages induced by baffle jetting or debris.

Thus, two kinds of repair will take place in our plants:

- withdrawals of leaking rods from the fuel bundle and replacement by dummy rods or depleted fuel rods,
- complete exchange of assembly skeleton and, if so, simultaneous withdrawal of leaking rods.

3.1 - Our experience of different repair techniques

According to our policy, the repair of a defective irradiated fuel assembly is performed by the supplier itself. This repair takes place in the spent fuel pool and we have now accumulated a good feedback first with FRAGEMA and then with KWU. In the close future, ANF will perform fuel repair in BLAIS.

Up to now, we have a good experience of fuel reconstitution and repair and we can make the following remarks:

- Standard FRAGEMA and KWU fuel assemblies have only a removable bottom nozzle which requires assembly inversion for access to the fuel rods.
- Standard ANF fuel assemblies, advanced FRAGEMA and KWU ones can be easily dismantled from the top nozzle. Such a design will be better for fuel rods handling and should facilitate fuel repair by reducing costs and time to operate.
- FRAGEMA repair device includes a tower of about 12 meters in which the new skeleton is directly above the defective fuel assembly. The FRAGEMA reconstitution concept consists, in fact, in pulling irradiated fuel rods from the assembly through the new skeleton. The major advantage of this operation is to keep the rods in as fabricated stress conditions.
- An other way to repair fuel assemblies is to pull out failed rods and to insert in the bundle a dummy or uranium rod. Such a way works very well when you only have some rods to replace but is more critical for the overall reconstitution of defective assembly. Before the reconstitution, dummy rods should be inserted in the new skeleton to guide irradiated ones and to prevent them to be pulled in the adjacent cells of the spacers.

3.2 - Transfert of irradiated fuel assemblies

Obviously, taking into account the price of enriched uranium, the repair of damaged fuel assemblies will be always an economic way for utilities to recover their investment. Nevertheless, repair and reconstitution induce severe constraints and will lead the utility to order an intervention only when many defective fuel assemblies will be present in the spent fuel pool.

Thus in FRANCE, in order to optimize costs, time operation and occupational doses, we perform under-water transfers of irradiated fuel assemblies to group them on one unit of a plant.

We carried out such transfers:

- two times in TRICASTIN between unit 3 and unit 1 with the R 62 cask,
- once from BUGEY 5 to BUGEY 3 with a TN 12 cask,
- and the last time some weeks ago between GRAVELINES unit 3 and unit 2 with the R 62 for a 4,5 % enriched fuel which have been to be repaired.

The R 62 is a cask designed by ROBATEL to transfer under water, or in dry conditions, one defective fuel assembly as well 12 feet long (PWR 900 MWe) as 14 feet one (PWR 1300 MWe). The TN 12, designed by TRANS NUCLEAR is the normal cask used for shipping irradiated assemblies to reprocessing plant. As a consequence, its use to transfer under water failed fuels required a suitable procedure.

However, as it was mentioned before, when a defective fuel assembly is discharged, core management studies generally require to discharge simultaneously the three symmetrical assemblies in the core. So that, if this defective fuel has been transferred and repaired in an other unit of the plant, it should be transferred back to be reloaded with its symmetrical ones.

So, grouping fuel assemblies to be repaired on the same unit provides benefits but induces sometimes other constraints for the utility.

3.3 - Fuel reconstitution and repair status

As it is shown on diagram 1, 56 defective fuel assemblies could be reloaded without repair or reconstitution, 19 of them have already been repaired as it is detailed in table 1.
TABLE 1. DAMAGED FUEL RECONSTITUTION AND REPAIR STATUS AS OF THE END OF 1986

<table>
<thead>
<tr>
<th>UNIT</th>
<th>NUMBER OF ASSEMBLIES</th>
<th>DEFECTS ORIGIN</th>
<th>SQUELETON REPLACEMENT</th>
<th>RODS REPLACEMENT</th>
<th>OPERATOR</th>
<th>NUMBER RELOADED</th>
</tr>
</thead>
<tbody>
<tr>
<td>FESENHEIM 1</td>
<td>10</td>
<td>HANDLING DAMAGE (8)</td>
<td>8</td>
<td>2</td>
<td>FRAGEMA</td>
<td>5</td>
</tr>
<tr>
<td></td>
<td></td>
<td>BAFFLE JETTING (2)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>CHINON 2</td>
<td>2</td>
<td>HANDLING DAMAGE</td>
<td>2</td>
<td>2</td>
<td>FRAGEMA</td>
<td>1</td>
</tr>
<tr>
<td>TRICASTIN 1</td>
<td>4</td>
<td>HANDLING DAMAGE</td>
<td>2</td>
<td>2</td>
<td>FRAGEMA</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>LARGE DEFECTS (2)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>BUGY 3</td>
<td>3</td>
<td>BAFFLE JETTING</td>
<td>3</td>
<td>2</td>
<td>K.W.U</td>
<td></td>
</tr>
</tbody>
</table>

The next reconstitution campaigns are scheduled as follow:
- with FRAGEMA
  - in GRAVELINES 2 (5 fuel assemblies) on December 87
  - in FESENHEIM 1 (7 fuel assemblies) on Mai 88
  - in BUGY 2-3-4 and 5 (17 fuel assemblies) end of 1988
- with KWU
  - in BUGY 3 (1 fuel assembly) on November 87
  - in DAMPIERRE 1 (7 fuel assemblies) on January 88
- with ANF
  - in BLAYAIS 1 (1 assembly) March 88

Up to now, 6 of the repaired fuel assemblies have been reloaded and reached the end of a new cycle of irradiation without any difficulty.

4 - LEAKING FUEL RELOADING, RECONSTITUTION AND RESEARCH OF FAILURE REASON

Reloading of leaking fuel assemblies and repair are the best ways for the utility, with an acceptable increase of costs, to recover all the energy stored inside uranium. Such a policy really reduces the fuel loss of the utility but cannot be useful, without further investments, in the research of failure cause.

Several actions have been conducted by EDF in order to identify the main reasons of defect occurrence in reactor and to point out for example, with the collaboration of the fuel supplier, if it is necessary to improve either the design or the manufacturing conditions.

We can mention particularly:
- failed fuel rod detection which is performed by our fuel suppliers,
- improvement of visual examination device in spent fuel pool,
- purchase of the R 62 cask to transfer defective fuel rods or assembly between the plant and laboratories,
- modification of hot cells and purchase of special equipments.

Fuel assembly repair is also a very important part in our policy of failure reason research because it allows first the inspection of defective rods in the plant pool and then the selection of those which must be shipped in hot cells laboratories.

4 - CONCLUSION

To conclude we can mention that with the generalization of advanced fuel assemblies (zircaloy spacer grills and removable top nozzle) repairs and reconstitutions will become easier and will require less time to operate and, we hope so, less money.

In such conditions, it will be very attractive to reduce the reloading criterion of leaking fuel assemblies and to repair the major part of them.

Nevertheless, after several years of operation, our leaking fuel reloading policy has been good and never conducted, except in some particular cases (baffle jetting), to exceed our technical specification of primary coolant activity. Obviously, as long as our experience will confirm the non evolution of defect during further cycles in reactor, reloading of leaking fuel assembly will be always the most economic way for the utility.
The paper describes the damage observed on 20 Advanced Nuclear Fuels Corporation (ANF) fabricated PWR fuel assemblies that failed at the Tihange-1 nuclear plant as a result of fretting due to baffle jetting induced fuel rod vibrations. The 20 assemblies were repaired at poolside by transferring sound fuel rods from the damaged assemblies to new cages and by replacing failed rods with inert rods. The tools and work procedures employed during the repair and reconstitution of the assemblies in three different campaigns are detailed. The installation, use and removal of fuel rod clips, developed by ANF to avert baffle jetting damage, is described. The application of clips at Tihange-1 has been highly successful; no more baffle jetting failures were observed after the clips were installed.

1. INTRODUCTION

The Tihange-1 nuclear power plant, which belongs to the French-Belgian company SEMO (50% Electricité de France, 50% Belgian utilities) is a 870 MWe PWR which was put into operation in 1975. Tihange-1, located in Belgium, was the first 3 loop plant of Westinghouse design in Europe. The core comprises 157 fuel assemblies with a 15x15 fuel rod array. The first core and the first 3 reloads were designed and fabricated by a joint venture between FRAMATOME and WESTINGHOUSE. All the subsequent reloads, starting from the reload introduced at the beginning of cycle 5 in 1981, have been designed and fabricated by Advanced Nuclear Fuels Corporation (ANF).

Baffle jetting occurred at Tihange-1 between 1980 and 1986 and resulted in the failure of a large number of assemblies and rods during four cycles of operation.

The baffle jetting phenomenon occurs when a gap opens up at the bolted joint of baffle plates that surround the core. In the core and vessel design of several plants like Tihange-1, the direction of the coolant flow is such that the pressure of the water outside the core baffle is higher than the pressure of the water inside the core baffle, particularly near the top of the fuel assemblies. The pressure differential across the baffle plates and hence across any gap in the plates, creates a high velocity water jet directed at assemblies near the baffle. As shown in Figure 1 there are two ways in which the baffle plates may open up and allow the impinging coolant to impinge on fuel rods. In both cases, the high velocity water jet causes severe vibration of fuel rods in peripheral core positions. The vibrations lead to fuel rod failures and damage to assembly spacers due to fretting of the zircaloy material employed in these parts.

![Figure 1 Cross Section View of Fuel Bundles and Baffle Plates Showing How Gaps May Open in the Baffle Plate Joints.](image-url)
locations. Over the three subsequent reactor cycles a total of 22 fuel assemblies, 20 of them fabricated by Advanced Nuclear Fuels Corporation (ANF), sustained damage directly or indirectly due to baffle jetting. Partially due to an easily removable upper tie plate the ANF assemblies were repaired and/or reconstituted at poolside. Other modifications were made to the assemblies and to the core in efforts to avert baffle jetting damage.

The following observations, repairs, reconstitutions and modifications have been made and are presented in chronological order.

At the end of cycle 4 (EOC 4), in June 1980, 5 assemblies had high sip signals and were not reloaded. A visual examination, conducted after the plant was back in operation, showed that 2 of the assemblies, which had been in peripheral locations, had severely damaged fuel rods on the face adjacent to the core baffle. This was the first indication of a possible baffle-jetting problem at Tihange-1.

High coolant activity developed during cycle 5. At the end of this cycle 5 (EOC 5), in March 1981, all assemblies in the core were sipped and a visual examination of all the peripheral assemblies was conducted. Six assemblies were found to be damaged: 2 Westinghouse assemblies, ending their second cycle, which had been in peripheral locations during the previous cycle (cycle 4), and 4 ANF assemblies, ending their first cycle. A total of 18 fuel rods had breached cladding, 3 other rods were heavily fretted but did not leak. One assembly contained 8 failed rods. The spacers in the upper parts of the assemblies sustained heavy fretting damage.

Figure 2 illustrates the type of damage encountered. Following the outage the 4 ANF assemblies were reconstituted by transferring all sound fuel rods to new cages.

During the EOC 5 outage the baffle joints were peened. ANF modified 4 assemblies with inert rods inserted in positions in the assemblies vulnerable to baffle jetting damage. However, high coolant activity again developed during cycle 6, showing that these measures had not been successful.

A decision was taken to order a large number of inert zircaloy rods to be able to insert them, where appropriate, in fresh fuel assemblies at the next outage.

At EOC 6, in 1982, baffle jetting failures were observed in 8 assemblies including two assemblies that had been modified with inert rods. A ninth assembly contained a single failed rod due to debris induced fretting. The observed damage to fuel rods and spacers was in general less severe during this cycle than during cycle 5. The largest number of failed rods in a single assembly was 4. A total of 18 rods failed and 1 rod was damaged by fretting. During the cycle 6 outage 16 fresh assemblies were modified by replacing from 2 to 4 fuel rods in vulnerable position with inert zircaloy rods.

The 9 assemblies containing failed rods were reconstituted after the outage. Eight of the assembly cages were damaged and were unfit for further use and had to be replaced with new cages. One assembly cage was reused.

At EOC 7, in 1983, seven assemblies were found damaged. Failed fuel rods were found in four of these assemblies. The total number of failed fuel rods was 10 with one assembly containing 5 failed rods. Three of the damaged assemblies did not contain any failed fuel rods. In these assemblies inert rods that had been inserted to prevent damage had fretted through the spacer side plates. The reconstitution of the 7 damaged assemblies was performed after the outage.

During the cycle 7 outage, 20 new assemblies, to be loaded in core positions subject to baffle jetting, were outfitted with so-called fuel rod clips developed by ANF. The fuel rod clips...
are attached to peripheral fuel rods in the assembly and dampen the vibration of the fuel rod system, thus eliminating fretting and damage. A fuel rod clip is shown in Figure 3. The development and use of fuel clips has been described in a previous publication (1). Most of the assemblies containing the clips also contained a solid zircaloy rod with notches used to lock the clips in place.

Clips were installed on 20 new assemblies at the beginning of cycles 8, 9 and 10 in 1983, 1984 and 1985. The clips successfully prevented fretting and baffle jetting damage during each of these cycles. Clips were removed at the end of each of these cycles to allow the assemblies to be moved from the core periphery to interior core locations.

Figure 3 Fuel Rod Clip. The clips are positioned between spacer grids to increase the damping of the fuel rod system.

During the 1986 reactor outage, modifications were made to the vessel internal structure to change the direction of flow of the coolant outside the baffle and thereby reduce or eliminate the pressure differential across any gaps in the baffle joints. No inert fuel rods or clips were installed on any of the assemblies in the current cycle (cycle 11), which is the first cycle since the flow reversal has been implemented. Current indications, after 9 months of cycle 11 operation, are that flow reversal modification has been successful in eliminating baffle jetting induced fuel failures.

An overview of the damages observed and of the measures taken at the end of each cycle is presented in Table 1.

3. ASSEMBLY REPAIR AND RECONSTITUTION.

ANF was the first fuel supplier to make a removable upper tie plate design an integral feature of PWR fuel reload assemblies and has designed fuel rod upper end caps that can easily be grappled with hand held tools. The value of these design features was clearly evident, as they permitted the fairly straightforward and rapid repair and reconstitution of 20 damaged ANF assemblies at Tihange-1.

In addition to the two-position fuel assembly elevator in the Tihange-1 spent fuel pool, only three basic hand held tools were used to repair the failed assemblies. These are an upper tie plate unlocking tool, an upper tie plate removal tool and a fuel rod grapple tool. Associated equipment such as viewing devices, underwater TV cameras, special tools to remove broken rods and guide plates, and eddy current (E.C.) testing equipment were used where required.

The tie plate unlocking tool is a simple castellated tool that fits over the bayonet type lock of the assembly guide tubes and that is used to rotate a collar and thereby lock or unlock the guide tubes from the tie plate.

The tie plate grapple tool is shown in Figure 4, and the fuel rod grapple is depicted in Figure 5. The fuel rod grapple contains a sleeve to protect the withdrawn fuel rod, an eddy current coil for testing the integrity of the fuel rod and may incorporate a load cell to measure rod withdrawal and insertion forces.

The damage to the spacers that form an integral part of the Tihange-1 assembly cage required complete reconstitution of the assembly. Reconstitution consists of the transfer of sound fuel rods to a new cage, the replacement of failed rods with inert zircaloy rods or fuel rods from another assembly...
<table>
<thead>
<tr>
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<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
<th>7</th>
<th>8</th>
<th>9</th>
<th>10</th>
<th>TOTAL</th>
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<td>-</td>
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<td>-</td>
<td>-</td>
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<td>16</td>
<td>10</td>
<td>10</td>
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<td>-</td>
<td>-</td>
<td>4</td>
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</tbody>
</table>

Measures taken

- Inspecting the fuel rod
- Inserting the fuel rod on 16 assemblies
- Installing clips on 20 assemblies
- Installing clips on 20 assemblies

Figure 4 PWR Upper Tie Plate Grapple.

Figure 5 PWR Fuel Rod Grapple Incorporating A Sleeve that Protects the Withdrawn Fuel Rod, an E.C. Coil for Rod Integrity Testing and a Loadcell to Make Withdrawal and Insertion Force Measurements.
and the proper rearrangement of rods within an assembly for neutronic acceptability. The following procedures were employed to reconstitute the four assemblies damaged in cycle 5.

The damaged assembly and a new cage were placed in the fuel assembly elevator. The upper tie plates of both the damaged assembly and the new cage were unlocked and removed with the elevator in its highest position. Tie plates remained attached to the tie plate grapple tools (two were used) and were secured to the fuel pool wall. Tie plate removal is shown in Figure 6.

Each new assembly cage contained inert zircaloy rods in positions predetermined by neutronic calculations. Approximately half of the 200 rod positions in the new assembly cages were filled with full length dummy rods arranged in a checkerboard pattern. The dummy rods, which were later withdrawn, provided support to the assembly cage during shipping and handling at the site and furthermore acted as guides for fuel rods during insertion.

After tie plate removal, stainless steel guide plates were positioned on top of the damaged assembly and the new cage. The guide plates permitted access to one half of the fuel rod positions only, namely those not occupied by dummy rods. At the start of the repair only positions not filled with dummy rods were accessible. On the new cage, a second guide plate was placed on top of the first one, approximately 40 cm above it to ensure that fuel rods being inserted entered the assembly with good alignment to prevent damage to spacer springs, dimples and flow mixing vanes. The guide plate arrangement is shown in Figure 7.

Fuel rods were withdrawn by grappling their upper end caps and lowering the assembly with the fuel elevator. This operation is shown in Figure 8. The protective sleeve containing the eddy current coil rested on the guide plate during rod withdrawal and was lowered at the same time the assembly was lowered. Eddy current indications over the full rod length were thus measured and recorded.

Rods passing the eddy current test were transferred to the new cage; rods with an eddy current signal exceeding that of a predetermined standard were either put back in the damaged assembly or transferred to a failed fuel storage can. Average rod transfer time was 7 to 9 minutes. After all accessible fuel rods had been transferred, the guide plates on both the damaged assembly and the new cage were replaced with a second set that allowed access to the remainder of the rods. The remaining rods were transferred into the cage after removal of the dummy rods from the cage. The reconstituted assemblies were visually inspected by underwater TV and the assemblies

Figure 6 Upper Tie Plate Removal During Fuel Assembly Reconstitution at Tihange-1.

Figure 7 Arrangement of Guide-Plates on Top of New Assembly Cage to Guide Fuel Rod into Correct Position and Prevent Damage to Spacers During Rod Insertion.
were slipped to verify their soundness. The average time to reconstitute an assembly was approximately 70 hours. The work was performed by 2 two-man crews, working 8 hour shifts. Some 830 fuel rods were transferred during this reconstitution campaign not counting dummy rods. Quality control records were maintained during each rod transfer to ensure the proper location of each fuel rod in the reconstituted assembly.

At the end of cycle 6 and again after cycle 7, all the damaged assemblies were reconstituted in the manner described above. Modifications and improvements in the tools were made in the last two of three repair campaigns. One significant improvement was the use of a so-called "spider" instead of individual dummy rods. The spider consists of approximately 100 stainless steel dummy rods attached to a single stainless steel plate. Rather than having to remove dummy rods one at a time, the spider could be inserted and withdrawn in a single maneuver thus saving substantial time during repairs. The average time to reconstitute assemblies in the last two campaigns was approximately 50 hours, with one assembly being repaired in as little as 34 hours.

Details on the number of assemblies repaired, rods transferred and time requirements for the various operations during the 3 campaigns are provided in Table II. As indicated in the table, a total of 4146 fuel rods were transferred during the 3 repair campaigns.

<table>
<thead>
<tr>
<th>Number of reactor cycle</th>
<th>5</th>
<th>4</th>
<th>7</th>
<th>Total</th>
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</thead>
<tbody>
<tr>
<td>Number of assemblies reconstituted</td>
<td>4</td>
<td>9</td>
<td>7</td>
<td>20</td>
</tr>
<tr>
<td>Number of fuel rods transferred</td>
<td>432</td>
<td>1470</td>
<td>1444</td>
<td>4146</td>
</tr>
<tr>
<td>Average time to remove/install tie plate (min.)</td>
<td>19</td>
<td>21</td>
<td>18</td>
<td></td>
</tr>
<tr>
<td>Average time to transfer fuel rod (min.)</td>
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<td>7</td>
<td>7</td>
<td></td>
</tr>
<tr>
<td>Average time to reconstitute assembly (hr.)</td>
<td>70</td>
<td>90</td>
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<tr>
<td>Minimum time to reconstitute assembly (hr.)</td>
<td>55</td>
<td>34</td>
<td>38</td>
<td></td>
</tr>
</tbody>
</table>

It is worthwhile to note that the average dose received by the people involved in the inspection and repair work was on the order of 100 mrem per man during each campaign.

4. FUEL ROD CLIP INSTALLATION AND REMOVAL

Fuel rod clips were installed on 20 fresh fuel assemblies loaded during each of cycles 8, 9 and 10. At cycle 8, eighteen of the initial twenty assemblies contained a solid inert rod with notches to provide axial restraint against clip movement. Two assemblies relied exclusively on friction force to maintain the clips in position. Some of the assemblies contained, in addition, one or two other inert rods in highly vulnerable positions. Later experience has indicated that the clips alone provide adequate protection against baffle jetting in the absence of inert rods. It was also shown that with the particular clip design used, inert axial restraint rods are not required; friction alone was sufficient to ensure restraint against axial movement of the clips.

At the end of their first cycle of irradiation, assemblies with clips are moved from the outside of the core to interior positions. In these interior positions, the clips are no longer needed and they were, therefore, removed from the assemblies during each of the outages following cycles 8, 9 and 10.
A special underwater tool was developed to quickly and safely accomplish the removal of clips during each outage. The operating part of the clip removal tool is shown in Figure 9. Two sets of fingers that are pneumatically activated, enter between the fuel rods, clamp around a clip, and withdraw the clip from the rods. The tool is suspended at the end of a long mast that is used to position and maneuver the tool. A TV camera is incorporated into the tool to provide a close-up view of clip removal and to aid in positioning the tool. After a given clip has been removed from the assembly, it is disposed of in a debris basket. An overall view of the set-up at Tihange-1 is provided in Figure 10.

The sixteen clips on a single assembly could be removed within one hour. Set-up of the tooling required one to two shifts.

Figure 9 Fuel Rod Clip Removal Tool. The fingers Grapple and withdraw the clip from the assembly.

Figure 10 Overview of Fuel Rod Clip Removal Operation at Tihange-1.
5. INERT ROD REMOVAL

As indicated, a large number of the assemblies contained inert rods that were inserted during fabrication or during repairs. The inert rods needed to be removed from the assemblies after final discharge and before shipping the assemblies for fuel reprocessing. The procedures for inert rod removal are similar to the procedures used in assembly repair or reconstitution. After upper tie plate removal the inert rod is withdrawn and placed in a storage basket for later disposal by burial or other means and the tie plate is reinstalled. The open positions are usually not filled with other rods before the assembly is shipped. All tools used are identical to those used in repairs. One to four inert rods can usually be removed from an assembly within one hour.

6. ASSEMBLY INSPECTION AND QUALITY CONTROL

Adequate inspection and detailed quality control procedures are essential after assembly repair to ensure the integrity and good subsequent performance of repaired and reconstituted assemblies.

The inspections and overcheck performed at Tihange consisted of visual examinations of the outside of each assembly to check for possible damage to spacers, rods, misaligned clips, etc.; a careful check of all the locking devices on the guide tubes and the upper tie plate; and a check on the proper orientation of the upper tie plate. The repaired assemblies were also sipped to ensure their soundness.

Detailed quality control procedures included sign off by two persons for every fuel rod transfer or move and signoffs on tie plate orientation, tie plate locking, and final visual inspection.

7. CONCLUSIONS

The repair and reconstitution of assemblies, including the transfer of about 4200 fuel rods, and the installation and removal of inert rods and fuel rod clips has been performed at Tihange-1 without major difficulties. In particular, the inert rods and clips removal operations were carried out during outages without delaying core loading operations.

These achievements show the maturity of fuel underwater repair techniques.

REFERENCE

FUEL ASSEMBLY RECONSTITUTION AT THE PRAIRIE ISLAND PWR USING A REMOTELY CONTROLLED ASSEMBLY REPAIR MACHINE

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* Advanced Nuclear Fuels Corporation, Richland, Washington
** Northern States Power Company, Walch, Minnesota
United States of America

Abstract

The paper gives a description of a fully mechanized assembly repair system recently developed by Advanced Nuclear Fuels Corporation (ANF) for the safe and rapid repair or reconstitution of damaged or failed assemblies. The REPAR (REmote Poolside Assembly Repair) system, consisting of a large X-Y table and a fuel rod extraction mast, is placed on top of the fuel storage racks in the spent fuel pool and repairs or reconstitutions are carried out with the assembly being placed in the storage cell underneath the machine. The underwater equipment is remotely operated from a control console placed on the fuel pool deck. The use of the REPAR system and the work procedures employed to reconstitute 3 fuel assemblies that were damaged during core loading at the Prairie Island Nuclear plant are described. Fuel rods were transferred at a rate of approximately 20 rods per hour. A 14x14 fuel assembly could be completely reconstituted in as little as 13 hours. The work at Prairie Island demonstrated that the REPAR machine can be used during short refueling outages to repair or reconstitute fuel assemblies with minimum impact on plant critical path schedules.

INTRODUCTION

Fuel assemblies are commonly repaired at poolside with long handheld underwater tools to remove tie plates and exchange failed fuel rods with replacement rods. Fuel assembly reconstitutions have been carried out using similar techniques to transfer the sound rods from a damaged assembly to a new assembly cage. A main drawback in these manual repair techniques is the length of time needed to perform the work. A typical PWR assembly reconstitution, when performed with handheld tools, takes from 50 to 70 hours to complete. This length of time makes it impractical to perform assembly reconstitutions during short refueling outages.

To provide a safe and much more rapid means to repair or reconstitute damaged or failed fuel assemblies a fully mechanized repair system has been developed by Advanced Nuclear Fuels Corporation (ANF) titled REPAR (REmote Poolside Assembly Repair). This system has been used at the Prairie Island PWR to reconstitute three assemblies.

The REPAR system consists of a large underwater X-Y indexing table, a fuel rod extraction and insertion mast and a poolside control console. The X-Y table and mast are placed on top of the fuel storage racks in the spent fuel pool and repairs or reconstitutions are carried out with the assembly located in the storage racks in the spent fuel pool underneath the REPAR machine. The underwater equipment is remotely operated from the control console. An overall view of a typical set-up is given in Figure 1. Description of the equipment and its use at Prairie Island is given in the following.

Figure 1 Typical set-up of the REPAR machine on top of fuel storage rack.
REPAR Machine

The REPAR machine consists of 3 equipment subunits that are placed on the spent fuel pool storage racks and a control console that is placed on the fuel pool deck to remotely operate the machine. The 3 parts of the machine, in the order in which they are placed on the storage racks, are:

- A base plate which is firmly anchored to the fuel racks and which provides a level and sturdy platform for mounting the other two parts of the equipment.
- An X-Y table which is placed on the base plate and which provides the positioning mechanism to address all fuel rod positions of two assemblies placed in adjacent fuel rack storage cells.
- A fuel rod extraction mast containing the fuel rod grapple and withdrawal and insertion mechanisms as well as instrumentation to determine the condition of fuel rods withdrawn from an assembly.

BASE PLATE

The base plate is an approximately 100 x 120 cm large platform constructed of anodized aluminum plate. The base plate covers a 3 by 4 fuel storage cell area. A cross-sectional view of the base plate on top of the fuel racks is shown in Figure 2. Four locating posts are provided on the underside of the base plate to position it on the storage racks. Two of the posts are expansion clamps that are used to anchor the base plate to the racks. Clamping and unclamping is done with a handheld tool. Access to the fuel assemblies under repair is provided through holes in the center of the base plate. The holes have generous stainless steel lined lead-ins to avoid hangup during assembly insertion.

X-Y INDEXING TABLES

The X-Y indexing table consists of a base, that fits on the base plate described above, and an X and Y table that can move back and forth in orthogonal directions along stainless steel shafts over the length and width of the base. A top and a side view of the X-Y table is presented in Figure 3.

The X and Y tables are individually driven by computer controlled stepping motors and ball screw drives. The stepping motors can be removed under water to allow operation of the X-Y table by hand in case of motor malfunction.

Precise location of the fuel assemblies with respect to the X-Y table and the rod extraction mast is accomplished through the use of guide plates, Figures 1 and 3. The guide plates, one for each assembly in the repair station, have pins that engage the guide tubes of the assembly and other pins that mate with holes in the X-Y table base. The guide plates prevent lateral movement of the assemblies with respect to the X-Y table. The fuel rods pass through holes in the guide plates during withdrawal or insertion.

During fuel rod movement the X-Y table is locked in a fixed position with hydraulically activated pins.

An eddy current (EC) test coil has been incorporated in the X-Y table for integrity checking of fuel rods. The coil can be removed under water for service or replacement in case of malfunction.

FUEL ROD EXTRACTION MAST

A side view of the fuel rod extraction mast is shown in Figure 4. The mast is approximately 5 meters tall and is bolted onto the X-Y table. Bolting and unbolting can be done with handheld underwater tools. To facilitate shipping the mast consists of two parts that are put together before the unit is put in the fuel pool.

A fuel rod grapple, attached to a carriage, can be moved along the length of the mast. The carriage is hydraulically driven. When fuel rods are removed from the assembly, they are drawn up into a protective sleeve which provides support during rod withdrawal and insertion. A loadcell has been incorporated in the fuel rod grapple to measure fuel rod extraction and insertion forces.
An underwater TV camera is used to view the condition of a fuel rod while it is being withdrawn. Close-up views of the rod surface can be made with the camera. Axial and azimuthal position indicating devices are part of the equipment.

CONTROL CONSOLE

The control console for operating the underwater equipment can be placed at a convenient place on the fuel pool deck. Electrical cables and hydraulic hoses connect the console with the X-Y table and the fuel rod extraction mast. The console contains the electrical, electronic, hydraulic and TV equipment to operate REPAR. A hydraulic pump is incorporated into the console. Only electrical power is needed to operate the REPAR machine.

PRAIRIE ISLAND ASSEMBLY RECONSTITUTION

The REPAR machine was used at Prairie Island PWR to reconstitute three [4x14] fuel assemblies. The assemblies had been damaged during core loading. None of the fuel rods had breached cladding. Deformation of spacer parts was suspected but was not obvious from visual examination. New assembly cages had previously been shipped to the reactor and had been placed in the fuel storage pool.

Equipment set-up, check out and calibration was performed in approximately 2 days.

One of the three damaged assemblies was then placed in one of the storage rack cells and a new cage was placed in the adjacent storage cell addressed by the REPAR machine. The upper tie plate of the fuel assembly and the assembly cage were unlocked and removed with handheld tools. Two upper tie plate grapples were used and the upper tie plates remained suspended from the grapples for temporary storage. Guide plates were then installed on both the assembly and the assembly cage using a special purpose tool, locking both to the X-Y table base.

At the start of the fuel rod transfer the fuel rod extraction mast was indexed to the "home" position (typically the position of a corner fuel rod...
in one of the cells), and a final alignment check was made. The X-Y table was placed in the "locked" condition and it was verified by visual inspection that the locking pins described earlier had been engaged. All parameters that had been entered into the computer for control of the stepping motors such as array size, fuel rod pitch, storage cell pitch, etc., received a final check and it was verified that the fuel rod grapple operated correctly.

Fuel rods extracted from a given position in the damaged assembly were inserted into the same positions in the new cage. The fuel rod moves were logged in and recorded on the computer. At the conclusion of the transfer of the fuel rods from an assembly, a printout of this information was made. This provides a final overcheck that all fuel rods were correctly transferred. During the transfers each fuel rod was automatically scanned along its full length with an EC coil. The EC signals were displayed on an oscilloscope and recorded on a strip chart. These records have also been maintained for QA/QC purposes. Since no fuel rod damage was indicated only a small number of fuel rod visual examinations were performed.

At the completion of rod transfer the upper tie plates were reinstalled on the reconstituted assembly and the old assembly cage, and both were removed from the repair station with the fuel handling crane.

The actual rod transfer time to reconstitute the first assembly was 15 hours. The two remaining assemblies had their rods transferred in 11-1/2 hours and 11-3/4 hours, respectively. Fuel rods were transferred at a rate of approximately 20 per hour. Upper tie plate removal and reinstallation typically required from 15 to 30 minutes per operation, but was occasionally longer due to the large working distance between the deck and the fuel assembly tie plate (~10 meters).

SUMMARY AND CONCLUSIONS

Three damaged assemblies at the Prairie Island PWR were reconstituted with the newly developed REPAR machine. Fuel rods were transferred at a rate of approximately 20 rods per hour. A 14x14 PWR assembly could be completely reconstituted in as little as 13 hours including upper tie plate removal and installation. The computer controlled and stepping motor driven X-Y table and the hydraulically operated fuel rod grapple mechanism performed without problems.

The work at Prairie Island has clearly demonstrated that the REPAR machine or similar sophisticated equipment can be used during refueling outages to safely repair and/or reconstitute several fuel assemblies with minimum impact on plant critical path schedules.

EXPERIENCE OF ON-SITE RESTORATION OF IRRADIATED FUEL AT DOEL 1-2

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Abstract

The first irradiated assembly restoration campaign on the Doel site was performed by FRAGEMA in the Fuel Building of units 1 and 2 during the first half of 1987.

By applying an original process, it was possible to replace the skeletons of 12 fuel assemblies which were not repairable by design, under satisfactory conditions for both restored assembly condition and operator man-rem cost.

1. INTRODUCTION

The fuel assemblies used in Pressurized Water Reactors are high-reliability products whose fabrication and operation are closely monitored. Nevertheless, it sometimes happens that they are damaged during operation. For example, some fuel rods may develop leaks or skeletons may sustain damage during a handling incident.

For DOEL units 1 and 2, the first 7 cycles passed with a very small number of fuel leaks. At the beginning of cycle 8 in October 1982, after an incident involving scattering of chips generated during the September refuelling outage had been identified, a sudden increase in reactor coolant activity made it necessary to shut down the reactor and prematurely discharge the fuel assemblies.

To allow the utilization up to complete burnup of these standard assemblies which are not repairable by design (i.e. without removable nozzles), FRAGEMA developed equipment for replacing the damaged fuel rods and irradiated skeletons; the replacement skeletons were of the Advanced Fuel Assembly repairable type with a 14 x 14 array and an 8-foot active length.

This paper describes the condition of the assemblies undergoing restoration, the process and equipment used and the conditions in which the campaign took place.
2. ASSEMBLIES UNDERGOING RESTORATION

Of the 12 restored assemblies, 9 had been supplied by FFAGEMA and 3 by WESTINGHOUSE; all were of the standard type (non-removable nozzles).

Seven had been prematurely discharged after the chip scattering incident in October 1982 and another had had the bottom of its skeleton severely damaged by being dropped inside the reactor; this was the only leak-tight assembly.

3. RESTORATION PROCESS

Given the likely presence of debris inside the assemblies, the non-removability of the nozzles and the advantages of repairable zircaloy-grid skeletons, the restoration process involved transferring all the sound rods into new AFA type skeletons.

The assembly being restored, which has already been leak-tested and whose leaking rods have been localized, is subjected to the following operations (fig. 1):

(a) The top nozzle is removed
(b) A new skeleton containing the fuel rods for replacing the leakers is placed next to the assembly
(c) The sound rods are transferred into the new skeleton
(d) The leaking rods are transferred into a canister
(e) The top nozzle is mounted on the reconstituted assembly which is then inverted
(f) The bottom nozzle is secured
(g) The reconstituted assembly is inverted again. After a series of inspections, it is stored in the spent fuel pit
(h) The top nozzle is remounted on the irradiated empty skeleton which is then removed.

4. EQUIPMENT DESCRIPTION

The main equipment items are as follows:

(a) A restoration station (fig. 2) in the form of a 12 m high stand seated on the bottom of the spent fuel pit and used for:
   - AFA type assembly inverting
   - Support of the rod gripper and its positioning on each rod
   - Storage of a leaking rod canister
   - Work on assembly nozzles (under 4 m of water).
(b) An automatic tool (fig. 3) for individual rod transfer, equipped with optional systems for sipping testing and evaluation of rod straightness.
(c) Two poolside tool storage fixtures and one tool guide plate underwater storage stand.
(d) One poolside fixture for preparation of the AFA skeleton and its nozzles.
(e) One milling machine (fig. 4) for top nozzle removal.

Fig. 1: Skeleton replacement - Sequence of operations
5. DETAILED DESCRIPTION OF OPERATIONS

(a) The assembly undergoing restoration is placed inside its handling basket.
(b) The basket is mounted on the work stand of the restoration station for top nozzle removal by milling of the skeleton / top nozzle connections.
(c) The basket is positioned for transfer of the rods into the lower part of the station.
(d) A second basket containing the new AFA type skeleton is placed, without nozzles, next to the basket containing the irradiated assembly.
(e) A retractable carriage for positioning the transfer gripper is then set up on the 2 baskets.
(f) The rod transfer gripper is mounted on the cross-cross motion carriages at the top of the station and guided by the retractable carriage to the bottom of the station.
(g) One by one, the sound rods are then transferred from the irradiated assembly to the new AFA type skeleton. The new skeleton already contains the new replacement rods together with short dummy rods used to direct the irradiated rods to the center of the grid cells. These dummy rods are recovered at the end of rod transfer in a box just below the basket containing the new skeleton.
(h) The leaking fuel rods still in the irradiated skeleton are then transferred to a special canister for storage.
(i) The gripper is removed and the carriage is retracted.
(j) The basket containing the reconstituted assembly is then placed on the upper work stand of the station and a new AFA type top nozzle is secured to the skeleton.
(k) The top cover is put on the basket to allow the assembly to be inverted.
(l) When the assembly has been inverted, the dummy rod transfer plate is removed then the bottom nozzle is secured to the reconstituted assembly.
(m) The bottom cover is then put on the basket to allow the assembly to be inverted.
(n) The upright assembly is inspected then moved to a storage rack.
(o) During inspection, the irradiated top nozzle is secured to the damaged skeleton which is then removed.
(p) The dummy rods are recovered then inserted into another new skeleton in the poolsode preparation fixture.

6. INSPECTION OF RESTORED ASSEMBLIES

Each restored assembly was inspected as follows:

(a) Overall visual examination and check on guide thimble / nozzle connection appearance.
(b) Evaluation of the verticality of the upright assembly.
(c) Capacity to accommodate RCCA's and thimble plugs.
(d) Sipping tests.
The sipping test was performed inside the FRAGDNA-supplied removable sipping cell AXCELL. This is a self-contained system which can perform qualitative and/or quantitative inspection (Fig. 5).

Qualitative sipping determines whether fuel assemblies are leaking or not (test lasts less than 20 min.). Quantitative sipping determines fuel assembly equivalent leak site (test lasts 2 hours).

The system consists of a sipping cell inside a frame seated on the floor of the pool. The cell is connected by flexible hoses to the instrumentation and control cabinets. A gamma or beta activity measurement channel is connected to the system.

7. RESTORATION CAMPAIGN

Fuel assembly restoration took place from February 24th to July 27th 1987 with an 8-week break due to fuel building unavailability.

The workforce consisted of a site manager and 2 - 5 operators depending on the work period; the operations were continuously monitored by a plant agent. Only the rod transfer phase, lasting an average of 28 hours, was performed on a shift basis; the other phases took place during normal working hours.

The average restoration time was 65 hours per assembly excluding examinations, unit times varying from 44 hours (i.e. 3.5 working days) to 98 hours (10 working days) for the assembly with a deformed lower part. The latter includes the time needed to test the straightness (before insertion into the new skeleton) of each rod by means of a set of sensors mounted on the transfer tool.

Average accumulated dose per restored assembly was less than 0.3 man-rem.

8. CONCLUSION

Restoration equipment operation was fully satisfactory throughout the campaign; the only notable non-routine maintenance operation was the replacement of an underwater drive line which caused a 4 day stoppage.

Post operational checks demonstrated that the restored assemblies were leaktight and could be reloaded without problems.

Restoration makes it possible to reuse fuel assemblies which had been written off until now, thus allowing utilities to make extensive fissile material savings.
UNDERWATER NUCLEAR FUEL INSPECTION
AND RECONSTITUTION AT VIRGINIA POWER

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Abstract

Virginia Power has experienced fuel cladding defects in three of its four nuclear reactors, dating back to 1981. The first indication of fuel failure occurred at Surry Unit 1 following steam generator replacement. Subsequent examinations indicated that these failures were probably caused by debris-induced fretting. Fuel cladding defects which have occurred at the North Anna reactors are the result of both debris-induced fretting and baffle jetting. Post irradiation examinations to eliminate defective fuel assemblies have included several different techniques. A combination of wet sipping and ultrasonic testing (UT) was first used at Surry in 1983. The North Anna fuel examinations in 1984 were performed using only vacuum sipping. Each leak-detection campaign included underwater video examinations of the defective fuel assemblies and the occasional use of fiber optics and high magnification video. Because the Westinghouse fuel used by Virginia Power is not easily reconstitutable, failed fuel cannot be repaired during a refueling outage but must be replaced through a redesign of the core loading pattern. Reconstitution of a small number of once-burned fuel assemblies from Surry 1 was performed during 1985. This work occurred during a non-outage period and involved fuel assembly inversion and bottom nozzle removal. The repaired assemblies are being re-irradiated in small groups and will be used over a period of years in several fuel cycles.

1. BACKGROUND

Virginia Power is an investor-owned utility which serves most of Virginia and a portion of North Carolina, and has corporate offices located in Richmond, Virginia. The company has more than 13,000 employees and had a peak electricity demand during Summer 1987 of approximately 11,500 megawatts.

Virginia Power owns and operates four Westinghouse, three-loop reactors which contain 157 fuel assemblies each. The two Surry reactors achieved commercial operation in 1972 and 1973, and provide a total of approximately 1600 megawatts. The two North Anna units have a capacity of approximately 2100 megawatts, and were declared commercial in 1978 and 1980. All four reactors are designed for 18-month fuel cycles and use standard Westinghouse fuel. The Surry units use 15X15 fuel assemblies and the North Anna reactors use 17X17 fuel.

2. VIRGINIA POWER EXPERIENCES

The Surry units were troubled by steam generator degradation throughout the 1970's. Tube plugging levels had reached almost 25% when the decision was made to replace the steam generators in Surry Unit 1 during the Cycle 5-6 refueling outage which began in late 1980. At the same time, many design changes were implemented in response to the accident at Three Mile Island.

Soon after the startup for Cycle 6, at Surry Unit 1, increased coolant activity was measured as the result of three defect formation events (Figure 1). The coolant activity increased to 40% of the Technical Specifications limit (i.e. 1.0 μCi/gm dose equivalent 1-131) early in Cycle 6 and gradually decreased during the next 1/4 months to less than 20% of the Technical Specifications limit. During Cycle 6, plans were developed for fuel inspections but the fuel cycle continued until its next schedule: refueling outage date. Because of the large number of defects that was predicted and the limited amount of time available during the Cycle 6-7 outage, fuel was examined only with the intent of eliminating defective fuel from the next cycle. Wet sipping was performed but additional time was not available to identify the failure mechanism during the outage. As in all of Virginia Power's fuel inspection campaigns, testing was conducted during two shifts per day, 10-12 hours per shift.

The wet sipping program processed one fuel assembly approximately every 45 minutes utilizing two testing stations. The testing time was not affected by the time required for the assembly to be off the fuel rack to the testing location, but was greatly affected by the purge and recirculation time of the testing canisters and the acquisition of grab samples. Additional testing was required for grab sample processing on a multi-channel analyzer and evaluation of the data to identify leaking fuel assemblies. Canister contamination became a problem after leaking fuel assemblies were tested. Often, extensive flushing time was required to remove residual fission products from the canister before testing the next assembly. Even with the flushing, the assembly tested after a failed assembly often had to be retested because of misleading results due to fission product carryover.

The wet sipping results for Surry 1, Cycle 6 indicated that 52 fuel assemblies contained defective cladding. These assemblies and some of their symmetric partners were discharged and the Cycle 7 core loading pattern was redesigned using previously irradiated fuel from the storage pool. These assemblies were tested prior to the refueling outage and were verified to be intact. Although a suitable loading pattern was found, the replacement fuel did not have sufficient reactivity to support an 18-month fuel cycle, and the cycle length had to be shortened to 13 months.

Soon after the restart of Cycle 7, detailed inspections of the discharged, defective fuel from Cycle 6 were initiated. These examinations included high magnification (25x) video, fiber optics, and ultrasonic testing to identify the exact fuel rods that were leaking. Significant amounts of hydriding were found and debris was located in the
areas of the lower and fitting and the first (from the bottom) support grid. Much of the debris had the appearance of machine turnings and irregularly shaped wires which could have resulted from the steam generator replacement project. The debris was present in the reactor coolant system (RCS) despite precautions, such as barriers placed in the piping, that were implemented during the steam generator work and modifications made as a result of the TMI-2 accident.

During Cycle 7, debris remained in the RCS and the reloaded fuel assemblies. The result was additional fuel failures as indicated by coolant activity which reached 7 percent of the technical specifications limit (Figure 2). The fuel examination program at the end of Cycle 7 included not only the same wet-slip and ultrasonic testing that followed Cycle 6, but also extensive vacuuming of the reactor vessel to remove debris. Any debris identified, through high magnification video, to be present in the fuel assemblies also was removed.

The full-length video scans of Cycle 7 fuel assemblies indicated that debris was present only in areas of the lower and fitting and first support grid. Debris-induced fretting, which appeared as cuts in the cladding, was often evident on adjacent fuel rods. Debris was not always swept away from the defect site by the coolant flow and could occasionally be seen tightly positioned between a support grid and a fuel rod. Typically, debris removal was accomplished by an automated manipulator using a water jet and hooks.

All of the defective fuel from Cycle 7 was discharged from the reactor and a revised core loading pattern was developed using intact fuel from the storage pool. Fuel assemblies planned for reuse were carefully examined using high magnification video and all debris was removed.

During Cycle 8, additional fuel failures occurred but they apparently were not due to debris being left in the fuel assemblies or reactor coolant system following Cycle 7. Instead, the fuel failures are thought to be the result of partial through-wall cladding damage in fuel reloaded from Cycle 7, and cladding stress induced by a major power transient.

Surry Unit 1 was operating at full power during Cycle 8 in January 1985 when extremely cold weather caused a condenser water icing problem. The power level on both Surry reactors was reduced rapidly using control rods, which resulted in a large negative axial offset approaching -50 percent. Unit 1 demonstrated elevated coolant activity level soon after this transient, but Unit 2, which had no recent history of fuel cladding failure, showed no change in coolant activity. It is suspected that some fuel cladding damage for Unit 1 fuel was not detected during the Cycle 7 examination since the defects were not yet through-wall, and these damaged assemblies were inadvertently reloaded. The cladding stress resulting from the large negative offset caused the partial defects in the lower span of the fuel rods to become through-wall.

Following Cycle 8, fuel inspections were again performed and the defective fuel assemblies were removed from the core. During the fuel inspections conducted at Virginia Power prior to the Cycle 8-9

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**FIG.1. SURRY unit 1 — cycle 6**

**131I activity vs. time.**
examinations, wet-sipping and ultrasonic testing were performed side-by-side to gain confidence in the ability of ultrasonics to detect failed fuel rods. Beginning with the Cycle 8 inspections in 1986, Virginia Power adopted ultrasonic testing as its sole leak detection method. This technique not only determines the leaking fuel assemblies but also identifies the damaged fuel rods in order to facilitate reconstitution.

The Virginia Power fuel inspection program has not been limited to Surry Unit 1. Fuel defects also have occurred at both North Anna reactors. Following Cycle 4 of North Anna Unit 1 in 1984, vacuum sipping was used in lieu of wet sipping in an attempt to expedite the fuel inspections. No ultrasonic testing was performed during this outage.

Previous experience with wet sipping had indicated that significant amounts of time were lost in obtaining grab samples, and in flushing the testing canisters to remove fission products that were left after removing a failed assembly. It was hoped that the testing rate of approximately two assemblies per hour could be doubled. This expectation of a high testing rate was based on the fact that vacuum sipping uses an in-line gas activity detector instead of liquid coolant activity grab samples to ascertain leaking fuel assemblies. In actuality, a higher rate was achieved, but it was only marginally faster than wet sipping at a rate of approximately three assemblies per hour. Since vacuum sipping uses a two water-filled canister arrangement similar to that used for wet sipping, it experienced the same canister contamination problem following removal of a leaking assembly. Extensive liquid flushing of the canisters was required in order to remove residual fission products. Even with this flushing, the results for assemblies tested immediately following the identification of a leaking assembly were questionable and the fuel often had to be retested.

As with Surry Unit 1, debris was again found in the fuel assemblies and reactor coolant system of North Anna Unit 1. However, only a relatively small amount of debris was found and it was not the same type of sharp machine turnings that appeared in Surry Unit 1. Instead, it was short, cylindrical pieces that resembled wire.

During Cycle 6 for North Anna Unit 1, fuel defects were again indicated by the measured coolant activity. Throughout most of Cycle 6, the I-131 level indicated that 1-2 fuel rods were leaking. Toward the end of Cycle 6, a coolant activity increase, coincident with the initiation of a power coastdown to extend the length of the cycle, indicated that additional defects could have occurred. However, it was suspected that some other event, such as a loss of demineralizer flow or efficiency during the power coastdown could have caused a false indication of additional fuel failure.

As in all of our recent refueling outages, the entire core was offloaded to the fuel building for the Cycle 6 refueling. This process not only allows maintenance to be performed on the reactor vessel residual heat removal system but also facilitates the interchange of insert.
components among the reload fuel assemblies. The fuel inspections during this outage included only those assemblies that were being reloaded for Cycle 7 so that the time the core would remain in the fuel building would be minimized. Ultrasonic testing was the only inspection technique used.

The results of the inspection indicated two fuel rod failures in peripheral assemblies. The likely defect mechanism was baffle flow jetting. These failures appeared to have occurred late in the cycle (probably around the time of initiating the power coastdown) since the cladding damage was limited and there was no evidence of secondary hydriding. These two assemblies were removed from the Cycle 7 core loading pattern and replacement irradiated assemblies were obtained from the spent fuel storage pool.

North Anna Unit 2 also had a small number of leaking fuel rods in 1984. The post-irradiation inspections included vacuum sipping and high magnification video of reload fuel assemblies. A small amount of debris was found in the assemblies and the reactor coolant system. The debris was similar to that found in North Anna Unit 1 and was not the same type of machine turnings that were found in Surry Unit 1. A summary of Virginia Power's fuel defect experience is provided in Table I.

An important lesson learned as a result of the fuel cladding defects experienced by Virginia Power is that any maintenance performed on the reactor coolant system is a potential source of debris that could lead to fuel cladding failures. Despite the fact that dams were placed in piping that was cut during the replacement of steam generators at Surry, debris still was present in the coolant system. Virginia Power has implemented a maintenance awareness campaign at both of its nuclear power stations to remind workers that regardless of how small and benign a piece of metallic debris appears, it could result in fuel cladding failure if allowed to enter the reactor coolant system.

3. RECONSTITUTION

The fuel examinations conducted at Surry Unit 1 in 1983 indicated that 86 fuel rods had failed in 52 assemblies. The defects were evenly distributed among the three burnup ranges of fuel that were present in the core. Because of the significant investment involved in having to prematurely discharge this large number of fuel assemblies, Virginia Power initiated a cost/benefit study for the reconstitution of failed fuel.

Virginia Power did not have previous experience with reconstitution and solicited bids from outside vendors. The task was further complicated by the fact that the standard design Westinghouse fuel used by Virginia Power is not easily reconstitutable, and to our knowledge, this type of assembly had not been previously reconstituted in the United States. Bids for the reconstitution were received from both domestic and foreign fuel vendors. After considering the bids in the cost/benefit evaluation, it was decided that the value of the energy remaining in the assemblies exceeded the cost of reconstitution only for once-burned assemblies. A major portion of the total reconstitution cost is the mobilization (i.e., set-up) and demobilization of equipment. Of the 52 failed assemblies from Surry Unit 1, Cycle 6, 29 assemblies were once-burned and were assigned to the reconstitution program.

Since the standard Westinghouse fuel assembly is not easily reconstitutable, the two options for removing and replacing fuel rods involved either cutting off the upper and fitting and developing a new mechanical connection to reattach the end fitting, or inverting the assembly and removing the screws that hold the lower end fitting on the guide tubes. Concerns existed within Virginia Power that neither of these options was desirable. The mechanical integrity of the fuel assembly after having cut off and reattached the upper end fitting was questioned. On the other hand, inverting the assembly to remove the lower end fitting could cause a shift of the pellet stack which could lead to additional cladding stresses upon reirradiation of the fuel. The final decision to reconstitute the assembly through inversion was substantiated by the fact that inversion had already been successfully completed in Europe.
Kraftwerk Union was chosen as the contractor for the reconstitution project, which occurred between June and October 1985. Personnel training and equipment set-up went smoothly. Since this was the first reconstitution experience for Virginia Power, a demonstration project was completed by KWU on an assembly that had been permanently discharged. Minor procedural problems were corrected after this demonstration phase and the repair of the 29 fuel assemblies commenced. As these assemblies were reconstituted, the time per assembly decreased significantly from 7 days to 2 days. No significant problems were encountered during the project. These repaired assemblies are being used in small groups during several subsequent cycles. Thusfar two assemblies have been reinserted and coolant activity measurements indicate that neither of the assemblies has failed. An additional 12 assemblies will be reinserted into Surry Unit 1 next year.

The reconstitution involved replacing failed fuel rods with solid zircaloy rods. Physics analysis performed by Virginia Power prior to the project indicated that groupings of replacement rods should not exceed clusters of four rods. It was determined that no peaking factor or burnup variation problems would exist if the number of replacement rods was limited. During the actual reconstitution of the 29 assemblies, no circumstances were encountered that would have required replacing more than four rods in a fuel assembly.

As the failed rods were removed from the assembly, they were pulled through an encircling eddy current coil to verify the cladding defect. There was only one occurrence of a discrepancy between eddy current and previous ultrasonic indications. For many of the failed rods, a visual examination provided evidence of debris-induced cuts and fretting. A small number of defective rods had experienced significant secondary hydriding and were not examined with eddy current. Hydriding was so severe on one fuel rod that it broke while being pulled through the support grids. Adjacent fuel rods were temporarily removed from the assembly so that the remaining portion of the broken rod could be extracted through the side of the assembly.

No other reconstitution of failed fuel is planned. The cost of contractor mobilization does not justify reconstitution for only a small number of assemblies. If additional fuel failures occur in the future and a decision is made to reconstitute, the repair will be facilitated through a design change of the Surry fuel assembly which will be implemented during 1988 to include a removable upper end fitting.

4. SUMMARY

In summary, during five years of experience with failed fuel and subsequent fuel inspections, Virginia Power has performed wet sipping, vacuum sipping, and ultrasonic testing. The rates of inspection with wet sipping and ultrasonics are comparable but somewhat slower than vacuum sipping. The rate for ultrasonic examination is limited more by the time required to move assemblies around the spent fuel pool than by the testing itself. Ultrasonic testing provides the advantage of identifying not only a leaking fuel assembly but also the defective rod. Thus important information is gained for the potential reconstitution of failed fuel, and has led Virginia Power to adopt ultrasonic testing as its primary fuel cladding leak detection technique.

Reconstitution of failed fuel has been performed by KWU for Virginia Power to regain the reactivity remaining in 29 defective assemblies from Surry Unit 1. Cost/benefit analyses have indicated that reconstitution is economically justifiable only for once-burned assemblies.

Despite the fact that Virginia Power has experienced some amount of fuel failure in seven different fuel cycles, a derating or early shutdown of a reactor never has been necessitated by high coolant activity. Virginia Power is committed to not restarting a new fuel cycle with known leaking assemblies in the reactor core. Therefore, current policy indicates that fuel inspections will be conducted whenever coolant activities higher than background levels are measured.
ON-SITE RESTORATION OF IRRADIATED ASSEMBLIES —
METHODS USED AND EXPERIENCE GAINED BY FRAGEMA

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Abstract
FRAGEMA has developed a complete station for on-site restoration of irradiated fuel with adaptability for work on 14 x 14 8-ft - 17 x 17 12-ft and 14-ft assemblies. Movable and autonomous, it has been used up to October 1987 on 4 different sites to restore 28 assemblies of which 24 had their skeletons completely replaced. This paper describes the methods and equipment used and the experience gained during these first restoration campaigns.

1. INTRODUCTION
For a number of years, fuel suppliers have been offering site-repairable fuel assemblies to cope with the rare but expensive hazards arising from handling operations and fuel assembly irradiation behavior, in exceptional cases. The repair equipment initially developed for these assemblies has often given rise to variants which can also be used for work on assemblies not repairable by design. In addition to fitting a range of fuel assembly arrays, these variants have been used to remedy different types of damage through their adaptability to special environmental conditions (workplace configuration) and operational conditions (assembly accessibility time, reloadability criteria).

In July 1985, FRAGEMA qualified a complete removable station for restoration of fuel in the 900 MWe-type French PWR's; after 3 restoration campaigns in France, it was adapted and used at the Belgian 450 MWe D0F.L 1-2 site. A further adaptation is under way for 1300 MWe type plants.

2. TYPES OF RESTORATION
Two types of restoration were considered for the work station design:

(a) Repair, in which the damaged rods inside sound skeletons are replaced.
(b) Reassembling, in which the sound rods from damaged skeletons are transferred to new AFA-type skeletons (the AFA is a repairable-type fuel assembly with zircaloy grids).

3. RESTORATION PRINCIPLES
Reassembling, which involves a large number of repetitive operations (transfer of about 260 rods for a 17 x 17 assembly) justifies the use of more complex methods and equipment than a straightforward repair operation.

Given that it is frequently necessary to perform both types of operation during a single campaign and it may be necessary to reassemble an assembly initially earmarked only for repair (after rod failure during servicing for example), FRAGEMA has designed a multi-purpose restoration station allowing a range of methods to be applied for each of the two types of restoration. These methods relate to the type of fuel assembly nozzle/skeleton connection and to the shape of the rod ends.

3.1. REPAIR METHODS
3.1.1. 17 x 17 STD assemblies
These assemblies feature welded top nozzle/skeleton connections, the bottom nozzle/skeleton connections are screwed. As a result it is easier to replace the damaged rods after inverting the assembly and removing the bottom nozzle (see fig. 1). Since nozzle removal calls for machining of the screw lock wires, a new AFA-type nozzle is mounted on the skeleton at the end of the operation.

3.1.2. Repairable assemblies (AFA-type)
Since the screwed top nozzle/skeleton connections can be easily mounted and removed under water, the rods are replaced without inverting the assembly, after removal of the top nozzle.

3.2. REASSEMBLING METHODS
3.2.1. 17 x 17 STD or AFA assemblies
To prevent any interference between rods and grids which may be detrimental to the latter, FRAGEMA has adopted a method (see fig. 2) very similar to that used in manufacture for loading rods into new skeletons:
- The damaged assembly is inverted then its bottom nozzle is removed.
- A new nozzleless skeleton is located above the damaged skeleton, precisely aligned with it. A gripper is inserted successively into each grid cell of the new skeleton to transfer the sound rods from the old damaged skeleton; the rod and plugs are grooved to accommodate a gripper whose diameter is less than rod diameter. The bottom nozzle is mounted on the new skeleton, followed by the top nozzle after inversion.

The assembly/skeleton/gripper system is set up vertically in the spent fuel pit.

3.2.2. 14 x 14 STD assemblies
The method described above cannot be used in this case since the assemblies being reassembled have the following features:
- Top connections welded, bottom connections screwed but with 4 corner screws placed underneath the bottom nozzle legs, therefore difficult to dismantle under water; rod end plugs are not grooved and have flat ends.
Fig. 1: 17 x 17 STD assembly repair method: principle

FRAGEMA has therefore chosen a method in which the assembly being reassembled and the new skeleton are placed in an upright side-by-side position. The new nozzleless skeleton is filled with dummy rods whose top ends fit the bottom end plugs of the rods being transferred and whose bottom ends are bullet-shaped for easy passage through the grids (see fig. 3); after the standard top nozzle has been removed (by milling the guide table/nozzle connections), the rods are automatically transferred one by one into the new skeleton. The dummy rods used to guide the rods during their transfer are recovered inside a box underneath the skeleton. The top nozzle is then mounted on the new skeleton followed by the bottom nozzle after reassembled assembly inversion.

Fig. 2: 17 x 17 STD or APA reassembling method: principle

4. EQUIPMENT DESCRIPTION

4.1. OVERALL CONFIGURATION OF RESTORATION STATION

The restoration station features a stand in 3 sections which is set up on the bottom of the cask loading pit (for the French 900 MWe type power plants) or spent fuel pit. It accommodates the fuel assembly support baskets which can be placed under about 4 m of water (for nozzle operations) or under 6-8 m of water (for rod withdrawal); these baskets are hooked one above the other on an elevator in the case of 17...
4.2. TOOLS FOR ASSEMBLY NOZZLE REMOVAL AND INSTALLATION

Assembly nozzles are removed and installed using manual tools in conjunction with guiding plates:

(a) One screwdriver for each type of nozzle/guide thimble junction (see fig. 5).
(b) A swaging tool used for expansion of the locking screws fixing the AFA-type nozzles to the skeletons, in order to prevent them from rotating or dropping in the event of breakage.

Each tool has a single function.

The removal of standard nozzles (on non-repairable fuel assemblies) is performed by using a motor-driven milling machine combined with an underwater vacuum cleaner which collects and retains the filtered chips. All operations are performed under a minimum water depth (about 4 m).
4.3. BASKET INVERTER

The assembly support-baskets are inverted by means of a cable-winch connected to a support structure which can be installed either in an access-gate to the spent fuel pit (for the French 900 MWe type power plants - see fig. 6) or in the restoration work-station itself.

4.4. FUEL ROD TRANSFER TOOLS

During reassembling operations, the rods are transferred one by one by means of 2 different automatic tools:

(a) One ("OSCAR" - see fig 7) capable of pulling rods only and which is used for the process described in fig. 2 (AFA or 17 x 17 fuel type reassembling):

Fig. 6: Assembly support basket inverter

Fig. 7: Fuel reassembling - Fuel rod transfer tool - "OSCAR"
(b) The other ("PIC" - see fig. 8) for rod pulling and insertion in the course of the process described in fig. 3 (14 x 14 STD fuel type reassembling).

The PIC tool is equipped with optional systems for sipping testing and evaluation of rod straightness.

5. INSPECTION OF RESTORED ASSEMBLIES

The following inspections are systematically performed on each restored fuel assembly:

(a) After repair
   - Visual examination of the junctions between guide thimbles and bottom nozzles;
   - Determination of verticality with assemblies mounted upright on their bottom nozzles;
   - Overall tightness.

(b) After reassembling
   - Same as above plus visual examination of the 4 lateral assembly faces;
   - Measurement of assembly bow;
   - Checking of the ability to accommodate thimble plug and rod cluster control assemblies.

These inspections can be made either by site equipment, or by movable equipment provided by FRACEMA consisting of:

(a) An examination stand installed on the restoration station elevator;
(b) An autonomous sipping cell "AXCELL" equipped with a beta or gamma activity measurement channel. This cell can perform a qualitative sipping (i.e. determine if assembly is leaking or not) or a quantitative sipping (i.e. determine assembly equivalent leak size).

6. OPERATIONAL EXPERIENCE

Very few off-routine maintenance operations have been performed on the equipment: it was necessary to replace the pulling stem of the gripper of the OSCAR fuel rod transfer tool after the original one had been bent, and one underwater drive line had also to be replaced during reassembling of a 14 x 14 STD fuel assembly. Except for these 2 events, the equipment operation was very satisfactory.

The operations were conducted with a work force from 3 to 10 people depending on the site, work period and type of restoration; generally, only fuel rod transfer was performed on a shift basis.

The average duration of a reassembling operation, including post-operational checks, was 68 hours for a 17 x 17 - 12 ft STD assembly, 65 hours for a 14 x 14 - 8 ft STD assembly; a repair operation of a 17 x 17 STD assembly took 28 hours including checking of the extracted rods.

Average accumulated dose per restored assembly was low, less than 0.3 man-rem.

7. RESULTS AND CONCLUSION

The systematic checks (mentioned § 5 hereabove) carried out on all the assemblies restored during the 4 campaigns confirmed the validity of the methods and the design of restoration equipment. They demonstrated that restored assemblies were leaktight and could be reloaded without problems in reactors.

The experience accumulated allows FRACEMA to offer a reliable irradiated fuel restoration service to Utilities, which have thus at their disposal an efficient means of making notable fissile material savings.
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