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Poolside inspection, repair and reconstitution of LWR fuel elements

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FOREWORD

The Technical Committee Meeting on Poolside Inspection, Repair and Reconstitution of LWR Fuel Elements was organized by the International Atomic Energy Agency upon the recommendations of the International Working Group on Fuel Performance Technology. At the invitation of the Government of Switzerland, the meeting was held in Bad Zurzach in October 1997 and hosted by Nordostschweizerische Kraftwerke (NOK).

The purpose of this meeting was to review the state of the art in the area of poolside inspection, repair and reconstitution of light water fuel elements and to evaluate the progress achieved in this area since previous IAEA meetings on this topic in Tokyo in 1981 and 1984, in Paris in 1987 and in Lyon in 1991.

The meeting provided a forum for the exchange of information between utilities, fuel designers and other authorities and specialists on a topic of current interest and real concern to industries in many Member States. The respective technologies are widely used or planned to be used in order to identify elementary major causes of fuel failure and to improve fuel utilization by repair and subsequent reuse of fuel elements.

The IAEA wishes to express its gratitude to the local organizing committee, the Chairman, W.R. Stratton, Head of the Nuclear Fuel Branch in NOK, the session chairmen and all the participants who contributed to the success of this meeting. The officer of the IAEA responsible for the organization of the meeting and for the completion of this publication was P. Menut from the Division of Nuclear Fuel Cycle and Waste Technology.

EDITORIAL NOTE

CONTENTS

SUMMARY

1. EQUIPMENT AND TECHNIQUES (SESSION 1)

1.1. Summary

1.1.1. Equipment

The inspection methodologies have not changed significantly since the last meeting held in Lyon, 1991. The largest change has been in the full scale automation of much of the equipment.

Several papers were presented describing new automated equipment for inspection. It is quite common now that fuel sipping to detect fail fuel assemblies is performed simultaneously with the fuel unloading. This has resulted in significant time savings which allows a facility to come back online sooner or allow for additional work in the same outage.

The automated inspection systems incorporate (often in modular form) the full range of inspection techniques (eddy current, ultrasonic, profilometry, visual, etc.) to verify fuel integrity and performance.

The presentations also indicated that the wide variety in fuel designs (PWR, BWR, CANDU, WWER) result in different equipment to perform the same inspections. Argentina which operates two different types of reactor has designed and built two completely different systems to accommodate the physical differences in the fuel assemblies, others are aiming to develop common systems for two or more fuel assembly types.

1.1.2. Techniques

Due to regulatory environments there is in many cases a need to develop new inspection techniques. One paper on this issue describes the use of an image analyzer to inspect critical dimensional parameters in fuel assemblies because of the regulator's demand for the complete inspection of all fuel assemblies. In another case, the potential problem with the deformation of fuel assemblies and the possible incomplete RCC insertion has caused the development of new inspection techniques. A paper was presented on a novel method to measure the deformation of a RCC guide thimble without unloading the fuel.

The need to non-destructively monitor fuel integrity and internal gas pressures has resulted in the development of an instrument to monitor these parameters based on thermophysical properties. This equipment has been successfully demonstrated on all types of Russian fuel and has possible application to PWR and other types of fuel.

1.2. Recommendations

The development and application of poolside examination equipment and techniques must take into consideration the requirements of the fuel designer, producer, utility, and regulator.

The equipment and technique must ensure that the fuel is not damaged during the inspection. This will allow the fuel assembly to be used for further reloads and ensures the safety of personnel.

The equipment and technique must provide meaningful and accurate data and not increase the length of an outage significantly.

Additional work should be done on expanding the nondestructive method of determining internal gas pressure to PWR and BWR rods since internal gas pressure, an important indicator of fuel performance, may become the limiting factor for higher burnups.

1.3. Areas for new development

Development of nondestructive techniques for hydrogen determination in zirconium alloys is still a requirement for some reactor types.

1.4. Repair and reconstitution of fuel assemblies

Automated equipment has been introduced for the repair and reconstitution of fuel assemblies. The time at which fuel assemblies are repaired is governed by regulators and the utility critical path requirements rather than by technical limitations.

Examples are given of the repair of five or more fuel assemblies during an outage, whereby most countries delay any fuel assemblies repair activities until after outage.

Past recommendations on designers to modify fuel designs to facilitate fuel repair and reconstitution have to some extent already been carried out. Future designs need to continue to take this into account.

2. EXPERIMENT AND NEEDS (SESSION 2)

2.1. Summary

The session showed a wide range of approaches to Poolside Examination, repair and reconstitution depending on the stage of nuclear development reached in a country but it also showed that in all countries with nuclear power plants there is a growing awareness of the need for rapid, accurate and efficient methods of examining and repairing fuel at the plant itself. In some cases, equipment was being developed and installed "in-house", while in others the experience and equipment of a major fuel vendor was being sought, or a combination of the two.

There was found to be little need to repair CANDU type fuel due to the lower unit value of these fuel assemblies[1]. There still remains a requirement to inspect CANDU type fuel. Designs to allow repair and reconstitution of WWER (1000) fuel assemblies are now being introduced.

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[1] Currently spent CANDU fuel is not re-loadable in the standard station designs.

In addition to examining and repairing defect fuel assemblies, the use of poolside equipment also for extraction of rods for determination of performance (modeling, design verification, PIE, etc.) and for consolidation of rods in compacted fuel storage was considered. In all cases, the economic aspects have to be evaluated.

The most important aspects of an effective poolside service were found to be;

- Good cooperation between plant and fuel vendor supplying much of the specialised equipment;
- Experienced and well trained teams with good knowledge of the plant and the fuel design;
- Support of the licensing/safety authorities;
- Availability of the equipment for the specific tasks when needed.

Given these conditions and careful planning it was shown for example to be possible to cover even major inspection and evaluation campaigns on 29 fuel assemblies during a one month outage period and to achieve licensing approval based on the results.

In future there could be a need to provide simple but effective monitoring of the integrity of fuel assemblies (including non-standard research reactor fuel) in long-term storage.

2.2. Recommendations

to fuel designers

Provide spacer designs which are more resistant to damage during reconstitution or repair, i.e. reducing the risk of damage to mixing vanes, corner pieces, etc.

to plant owners

Make adequate space available in the fuel pool and surrounding work area for poolside activities and aiming to reduce working time over the pool to reduce operator dose;

Build up a good working relationship with the safety authorities;

Have knowledgeable expertise on fuel available.

to equipment suppliers

Introduce micro-probe devices for visual examination in internal fuel assembly regions;

Improve oxide thickness measuring devices to overcome problems for example of zinc dosed plants (crud induced interference with oxide measurements);

Reduce the use of large heavy equipment (which may have a return to simpler, semi-manual operation of some equipment);

For reconstitution work provide equipment for removing and collecting broken fuel rods;

Develop devices for accurate determination of internal rod pressure.

3. GENERAL RECOMMENDATIONS

- Users and suppliers of equipment should make available data showing the experience with and the reliability of re-inserted, reconstituted fuel assemblies.

- Where possible, equipment suppliers/utilities should devote some time to anticipating future problem areas (e.g. with the trend to higher burnups).

EQUIPMENT AND TECHNIQUES
(Session 1)

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APPLIED MONITORING METHODS FOR THE CONTROL OF FUEL ELEMENTS AND REACTOR INTERNALS IN ARGENTINE NUCLEAR POWER PLANTS POOLSIDE FACILITIES

XA9848392

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Abstract

Two pressurized heavy water reactors are being operated in Argentina: Atucha 1, a Siemens KWU designed and built pressure vessel 345 MWe reactor, started in 1974, and Embalse, a CANDU-600 MWe, started in 1983. Post-irradiation tests have been performed in accordance with the needs of fuel development and control programs for internal parts, such as the coolant channels (CC) where the fuel elements (FE) dwelled. The bays built in both NPP offer a good place to perform a large variety of pool-side inspection and testing activities. During the last 20 years more than 2000 CANDU-type FE, more than 1000 Atucha FE and above 100 Atucha CC , have been inspected in these bays. The usual inspection activities consist of: visual inspection , sipping tests, disassembling of FE , metrology, gamma scanning and eddy current testing. In this paper all of these techniques are briefly described and the obtained experience is shown.

1. INTRODUCTION

Since the beginning of operation of the Argentine Nuclear Power Plants, pool-site inspections of fuel elements have been performed regularly. According to several criteria a routine examination is carried out to evaluate the overall fuel performance [1],[2]. Specific inspection and measurements are performed to characterize the prototype fuel elements which have modifications from the original design. An additional testing methodology is applied when a fuel element is suspected of being defective, which is indicated by the fission product leak monitoring system.

Both Argentine NPPs are fueled with natural uranium and cooled and moderated with heavy water. The design of Atucha 1 reactor core involves 253 coolant channels where the fuel elements dwell during operation and there exists mechanical interaction between both components. So, an important task was the implementation of the pool-side facilities for inspection and testing of replaced CCs. [3].

The purpose of this paper is to summarize the inspection system commonly used in Argentine NPPs pool-side facilities and to show the gained experience and future needs.

2. DESCRIPTION OF POOL-SIDE FACILITIES

The first step of pool-side inspection always consists of a completed visual examination using well-known optical devices such as a periscope and a underwater TV camera. However, both NPPs involve two different concepts, and so the inspection systems and the methodologies are slightly different.

2.1. Embalse examination systems

Two stations have been installed; the **first one** is a simple site that allows to perform an inspection immediately after the fuel elements are discharged from the reactor. It is located in the maneuver bay over the transfer rail conveyor and consists of a modified standard tray with a fuel element rotation device and four easily removable lamps. The observation is normally done directly

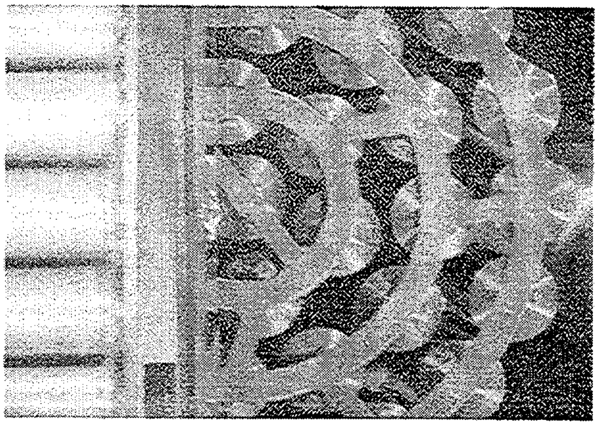

FIG. 1. Typical overview of Embalse fuel element at minimum magnification

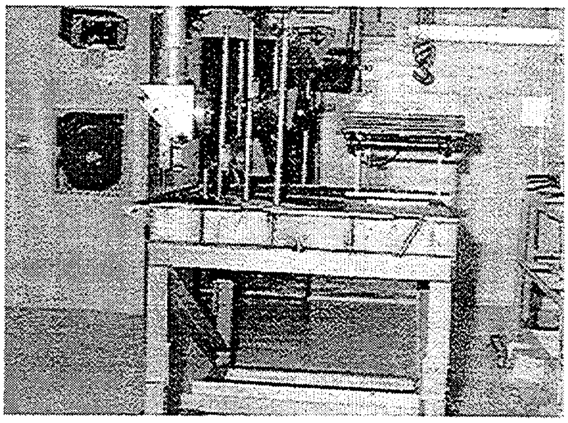

FIG. 2. Underwater table with modules for inspection and destructive test at Embalse during installation

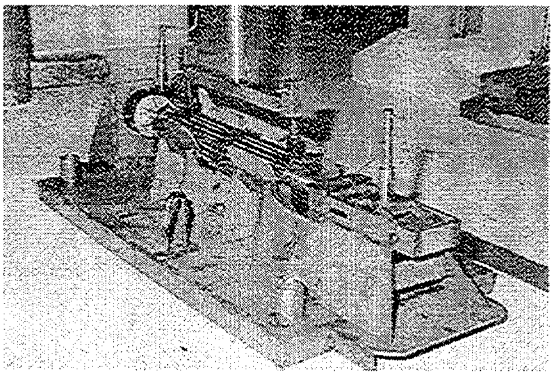

FIG. 3. Module to disassemble the outer fuel rod

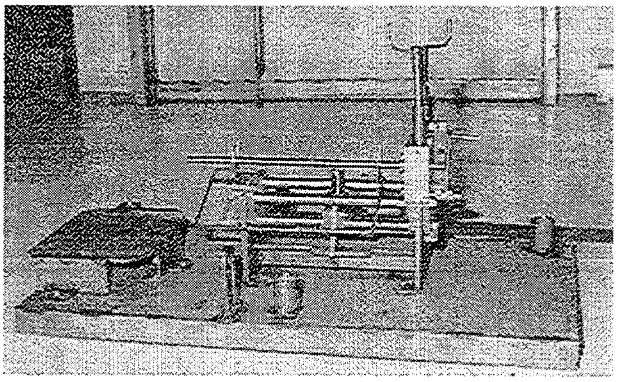

FIG 5. Underwater lathe to cut small pieces of the cladding

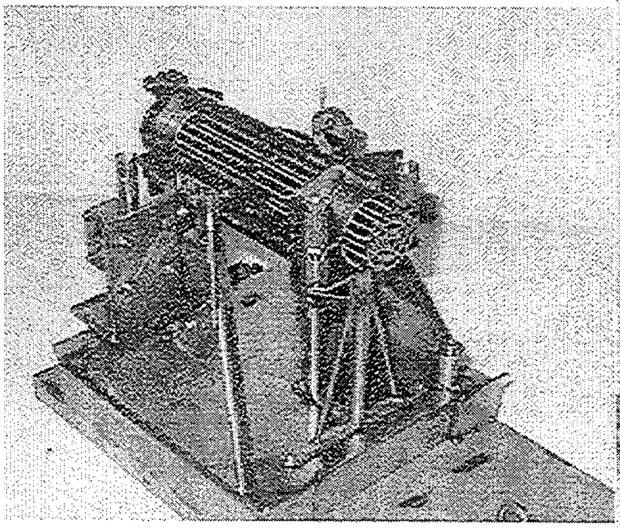

FIG. 4. Module to tilt and fit the bundle for disassembling the inner fuel rods

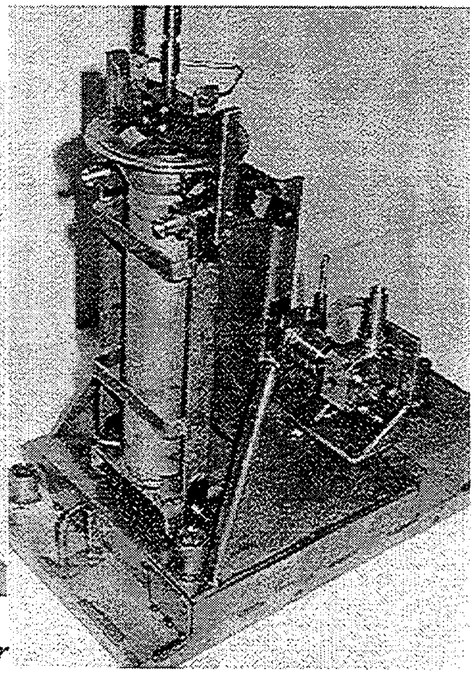

FIG. 6. Module to can the remains parts

through a telescope mounted on a 70 cm slide rail for axial displacement along the whole fuel element, which allows also a front view of both end plates. (see Fig 1). A CCD camera has been tried successfully and is being used nowadays to take photographs more easily than before.

The **second one** is a more complete station where non-destructive and destructive tests are performed, and it is located in a defective fuel storage bay (see Fig 2). It consists of a large underwater table, standing on the bay floor at 4.5 meter deep, and includes several removable modules to perform testing:

- One module consists of a rotating device, similar to the one mentioned above, but it allows to inspect, either the fuel bundle or a disassembled rod. This module is always in place, as well as several mobile columns with lighter.

- One module to disassemble only the outer fuel rods, by torsioning the end cap-end plate weld (see Fig 3).

- One module to disassemble the inner fuel rods, by cutting the end plate. It tilts and fits the bundle vertically, then a special clipper cuts any desired part of the grid. The clipper is handled manually through a set of bars hanging from an x-y table on the surface for precise positioning (see Fig 4).

- One module to perform non-destructive test (eddy current) of individual fuel rods..

- One module to cut small pieces of cladding. It consists of an ad-hoc underwater lathe which is driven by a motor or manually through a bar from surface. Another bar adjusts the advancing of the cutting tool (see Fig 5).

- One module to can tightly, in dry condition, an entire fuel rod, in order to transport it to the hot cell facility.

- One module to can the remaining parts of the cut fuel bundles, using a standard can (see Fig 6).

Observations in this site are normally done through an inverted periscope mounted in another slide rail for axial displacement. An underwater TV camera is used to monitor all the operations during the test. A digital height gage attached to the slide rail is used to perform axial length measurements. The precise mechanism of the slide rail allows to measure length with an accuracy of less than 0.2 mm.

The sipping equipment developed at Embalse is a standard one, of wet type. The equipment consists of an insulating container standing on the floor of the maneuver bay where bundle is tested The water inside the close circuit flows through manifold and a pump, and following an appropriate procedure the leak of fission products is detected with a NaI gamma detector. It is also possible to take a sample of water to analyze it in the laboratory with a multichannel.

The testing methodology applied when a channel is suspected of having a defective fuel elements is as follows: First of all, as soon as possible the discharged bundles are visually inspected at the **first site**. Secondly, sipping testing is done for those bundles without a visible failure, and thirdly, only in some cases, bundles are transferred to the **second site** to perform careful inspection, disassembling of fuel rods, and so on.

The eddy current testing has shown adequate sensitivity to detect abnormal hydride (deuteride) zones which are not observed during visual inspection. Cylindrical coils with a very low lift-off are used at a frequency over 700 KHz.

The visual inspection at the first site and sipping test are performed regularly, until now more than 10 % of the discharged fuel elements has been evaluated. Four campaigns at 1985, '91, '92 and '95 have been performed using the facilities of the second site in which about 30 bundles were disassembled and more than 50 fuel rods were tested.

2.2. Atucha 1 examination systems

Atucha 1 fuel element is similar to a typical LWR one, but it is longer and it has a circular configuration of 36 active rods and one supporting tube, all of them hanging from the upper end plate. The rods have three lines of fifteen welded bearing pads which slide over fifteen rigid spacers to hold the rods together. The average refueling frequency is ≈ 1.3 FA per full power day.

There are two inverted periscopes placed permanently in both bays at Atucha 1 NPP to perform visual inspection. Both systems consist of a 12-meter-deep tube solidly mounted to the border of the bay, with a 45° internal mirror to inspect the outer ring of the fuel assembly and another external folding mirror to inspect the bottom of the fuel. Six high-power spotlights are included , they are easily removable for maintenance (see FIG 7). Documentation is obtained by means of either a TV camera, or a photo camera attached to the telescope.

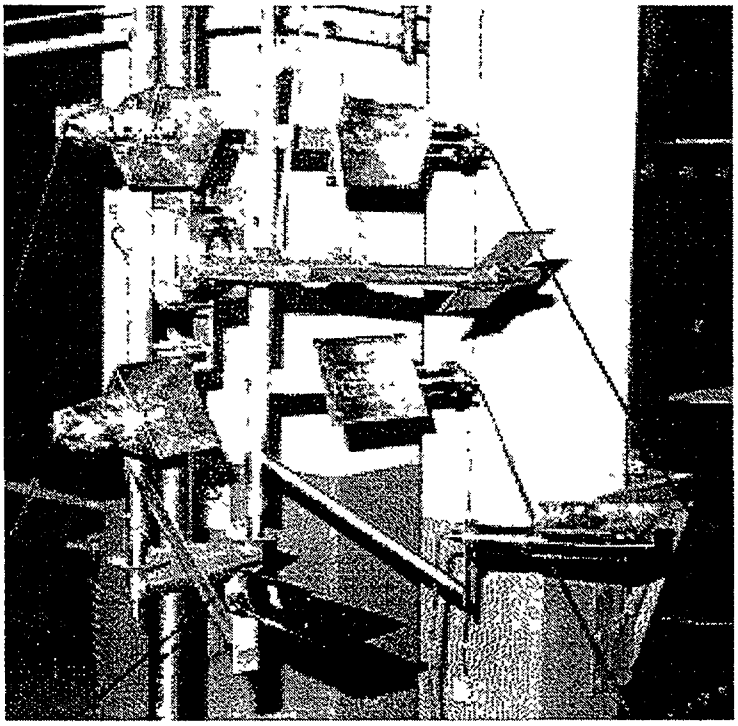

FIG. 7. View of the bottom of the periscope at Atucha 1 during installation.

2.2.1. *Fuel assembly inspection*

Nowadays, all the fuel elements discharged from the reactor are inspected daily. This activity is an essential part of the Early Warning Program which has been carried out since 1990. Before that date an incident had occurred inside the reactor core, when a peripheral CC broke because of a mechanical interaction with other internal parts. In this way the state of the spent fuel elements is monitored and indications of the in-service behavior of the CC are indirectly also obtained.

The fuel element is positioned in front of the periscope window through the fuel-storage maneuver bridge, which allows to scan all the outer rod visually.

It was not necessary to develop the sipping test equipment in bay, because in Atucha 1 the refueling machine is used as a natural sipping testing. This system has given optimum result and allows to check all fuel elements during, their discharging and also their refueling [4].

There are not facilities at Atucha 1 for the disassembling of FA in order to perform testing of each fuel rod. It will be one of the future activities.

The length change of the fuel rod in the outer ring is determined by the measurement of the distance between the sharp end of two of fifteen-on-line bearing pads. In this way, either the total or the partial elongation along of the fuel rod is measured. Both of the congruent extremes of the bearing pad are aligned, lifting the mast, one at each time an existing horizontal line in a TV monitor; comparing it with a pre-calibrated ruler which is situated in a special clamp fixed to the upper part of the mast. The difference in length between both the ruler and the segment of the rod is measured through a digital gage indicator . The accuracy of measurement is less than 0.3 mm.

2.2.2. *Coolant channels inspection.*

The external visual inspection of CC is performed in same way as the fuel assembly inspection. A special column with an underwater TV camera is used for internal visual inspection.

2.2.2.1. Dimensional measurements and NDT of coolant channels.

Measurement of the axial length of the Zircaloy part of the operating coolant channels was considered an important task, as the irradiation induced growth is a matter of particular concern. For this purpose several techniques were tested; the most successful, precise and repetitive results were obtained by direct comparison with a calibrated ruler, observing through an underwater TV camera mounted in a 2" column. The ruler was made out from the frame of an empty fuel bundle, having all the spacer and only 6 outer bars, in a so-called *squirrel cage* format. This design allows the perfect fit of the ruler to the inner surface of the shroud tube, avoiding any errors caused by a possible bowing of the channel. As the ruler is made of the same material as the coolant channel, differential thermal dilatation effects are avoided, and so this device can be used in the fuel pool as well as inside the reactor where axial temperature gradients exist. Each end of the ruler has three *verniers*; the reference points are the sharp ends of the shroud tube. Readings are made directly on the TV monitor. The appreciation in each measurement is of less than 0.2 mm, and so the uncertainty in the measured length value is of 0.4 mm.

Internal diameter (ID) measurements of shroud tubes were also performed in the spent fuel bay. A 2" diameter column was used, carrying a specially designed head with an underwater LVDT sensor (see Figure 8). This head adjusts itself automatically to measure the maximum chord, i.e. the diameter. Maximum and minimum values of the ID were measured at each 25 cm along the tube. In this way the ovality of the tube is obtained, and axial profiles of the mean value of the ID can be estimated (see FIG. 9).

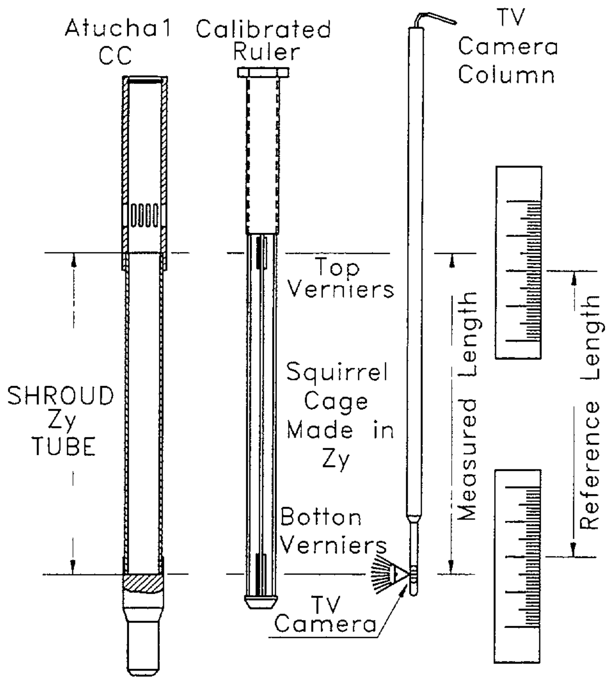

FIG. 8. Scheme of total length measuring of shroud tube of CC

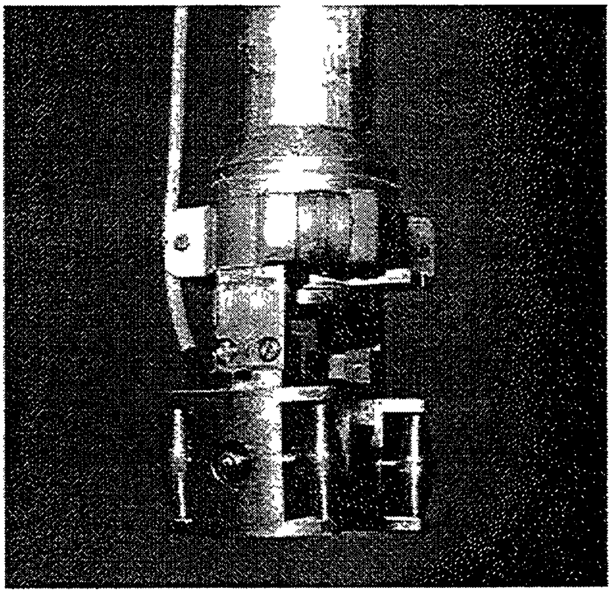

FIG. 9. Head for measuring ID of CC

For a number of channels, the curvature remaining after being removed from the reactor was measured in the bay. The apparatus designed *ad hoc* consists of four parallel columns arranged in a square structure; one of the columns is the coolant channel itself. The channel is adjusted to two guide bearings, in a similar way as it operates in the reactor. A TV camera carrying a normal lens is focused on the central part of the Zircaloy zone; in this way, rotating the channel, its maximum deflection with respect to a reference string can be determined (see FIG. 10).

NEW INSPECTION AND RECONSTITUTION TECHNIQUES FOR FUEL ASSEMBLIES INCREASE POWER PLANT EFFICIENCY

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Abstract

The amount of time required to complete many nuclear fuel service activities can be cut significantly through the use of innovative procedures and equipment. In addition to other benefits, these new approaches lower the radiation exposure of service and plant personnel. These new procedures and hardware can be integrated into the power plant processes and equipment to more effectively complete the required service activities.

1. MAST SIPPING: LEAK TESTING DURING CORE UNLOADING

Siemens has developed a mast sipping technique to identify leaking fuel assemblies. Whereas the sipping test has normally been performed to date in a sipping box located in the fuel pool following removal of the fuel assemblies from the reactor, today the same test can be performed as the fuel assemblies are being removed. This new method can save time and permits the condition of a fuel assembly to be identified at the earliest possible point in time. It has been used successfully in two plants in Germany.

1.1. Procedure

The mast sipping procedure makes use of the reduction in pressure which occurs when fuel assemblies are lifted out of the reactor pressure vessel, the resulting pressure differential allowing water-soluble or gaseous fission products to be released from defective fuel rods. The basic operating principle is illustrated in Figure 1. The lines used for sampling are either temporarily or permanently attached to the refueling machine mast while the control cabinet for the sipping system (Fig. 2) is portable.

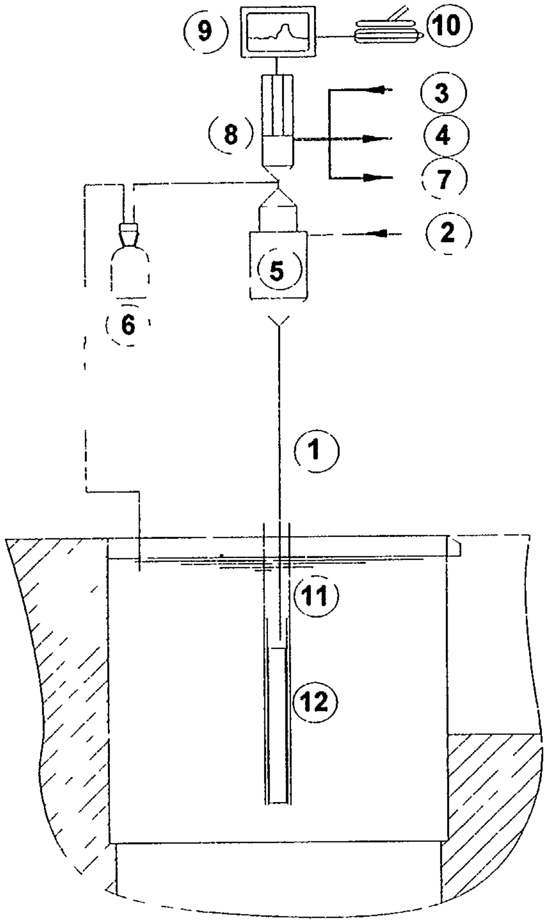
When a fuel assembly has reached its highest position during removal by the refueling machine, water is extracted from the refueling machine mast in the vicinity of the fuel assembly and analyzed for the presence of fission products. Gaseous fission products (xenon and krypton) are separated from the water, dried and routed to a scintillation detector, while the noble gas activity leaving the mast is continuously monitored. The measured data are analyzed immediately after sampling. A water sample is also taken which can be analyzed in a radiochemical laboratory to provide a reference measurement, if necessary. Figure 3 shows the procedure for a mast sipping test.

1.2. Evaluation

The data obtained from the sample measurements are immediately evaluated. The results are displayed on-screen and the data stored in a text file. Figure 4 shows the mast sipping test results of a qualification campaign.

2. MULTI-INSPECTION ENABLES PARALLEL WORK IN THE FUEL POOL

The MULTI-INSPECTION system was developed by Siemens to minimize the time and expense during visual inspections of fuel assemblies and control rod assemblies required during the course of a refueling outage. In the past, these components were held in the refueling machine for the



1 Water sample suction line
2 Demineralized water supply
3 Compressed air supply
4 Air exhaust system
5 Sample conditioning system
6 Water sample extraction
7 Gas sample extraction
8 Gas detector
9 Sample evaluation system
10 Printer for measurement results
11 Mast of refueling machine
12 Fuel assembly

FIG 1. Schematic diagram of mast sipping system

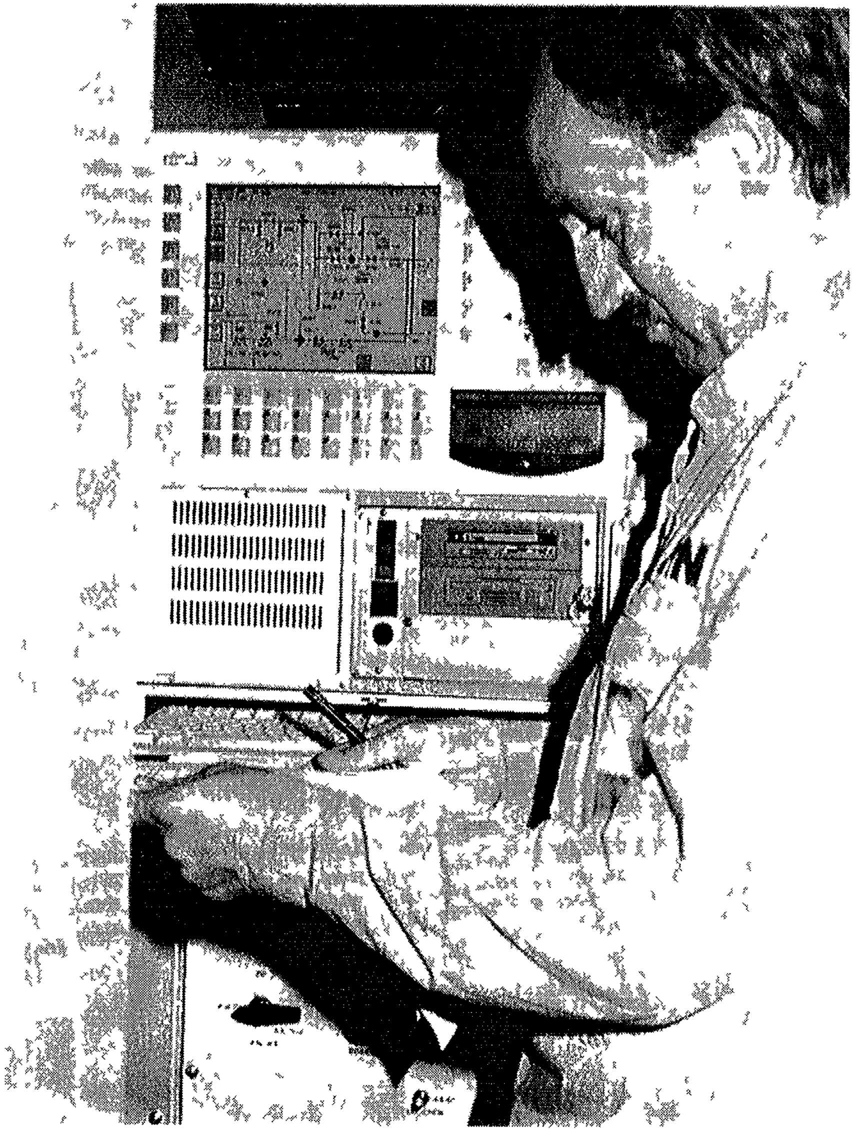


FIG 2 Control cabinet for mast sipping

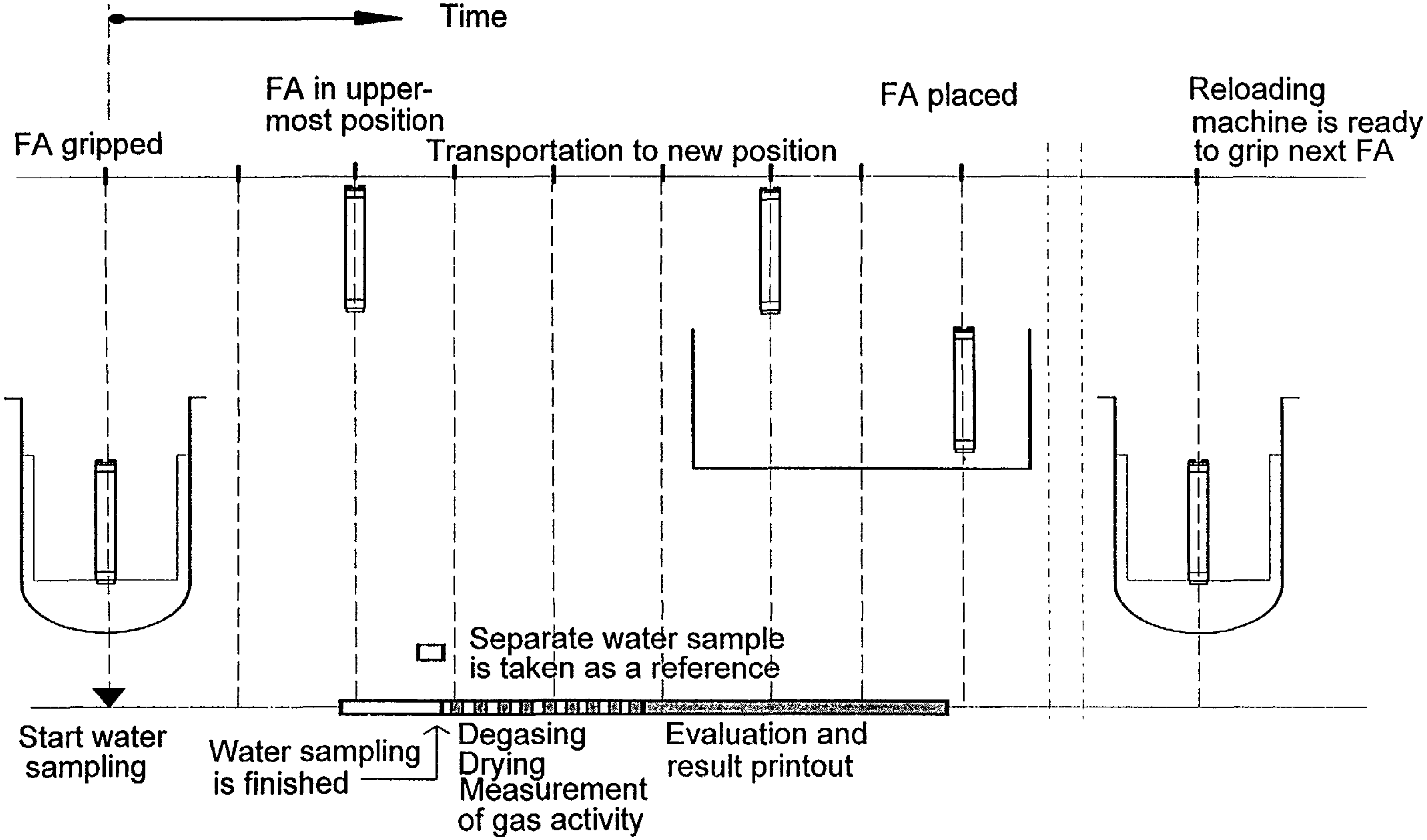


FIG. 3. Mast sipping procedure

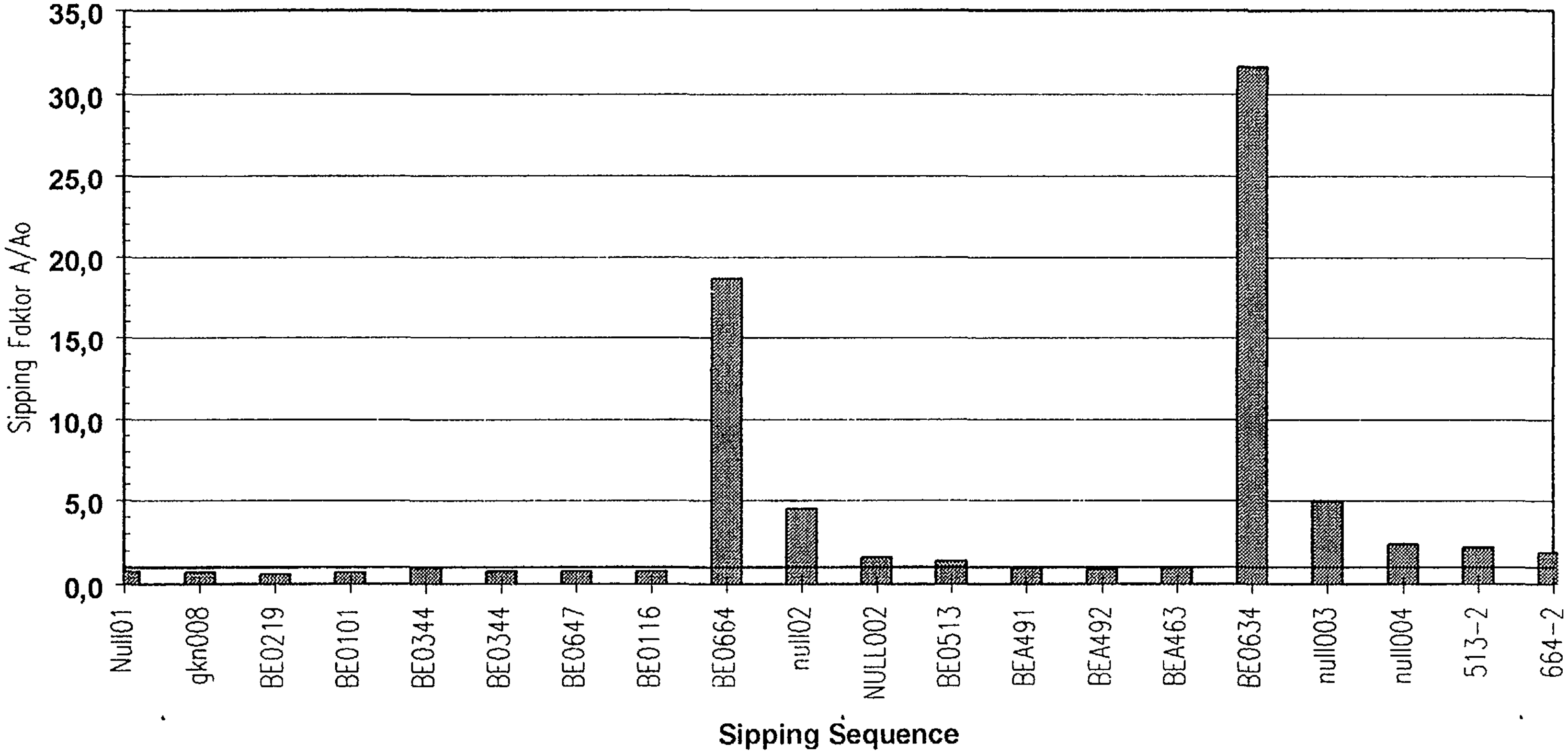


FIG. 4. Mast sipping qualification

duration of the inspection, and inspected using a temporarily installed system. With the new system, the components are placed in receptacles located in the spent fuel pool so that the refueling machine is only required for a short time to position components. While the inspections are being conducted, the refueling machine can be used to insert and remove fuel assemblies, to perform a mast sipping test or to shuffle control assemblies and flow restrictors.

This cuts the time required for refueling activities by as much as three to four days, as was confirmed during several refueling outages at the Gösgen PWR plant in Switzerland, and at Grafenrheinfeld and Philippsburg in Germany.

The MULTIINSPECTION system (see Fig. 5) installed at the Gösgen and Philippsburg plants consists of an inspection manipulator, a control cabinet and two fuel assembly holding boxes (see Fig. 6). Each box is freely accessible from one side, and is open for visual inspection. The fuel assemblies or control assemblies placed into the boxes can be individually rotated to allow inspection from all sides. The compact, portable control cabinet is used to control the overall system from the edge of the fuel pool.

The inspection manipulator is fitted with one or two underwater video cameras which are mounted on a coordinate-controlled traveling table which in turn is mounted on a mast-guided carriage. This arrangement allows the cameras to traverse the entire length of the fuel assembly, including the top and bottom end pieces. The carriage can also be raised to the top edge of the pool, where the cameras can be removed and replaced above the surface of the water.

The inspection manipulator can be used for the wide variety of applications by simply attaching the appropriate equipment modules. For example, the system permits optical measurement of bowing and dimensional changes. In addition of the time-saving semi-automatic measurement of the oxide layer thickness on peripheral rods, the same measurements can be made on all interior fuel rods at any height using the INOXIS system.

The MULTIINSPECTION system is available in several different versions, so it is designed as a single inspection position for Grafenrheinfeld. In this case the new fuel elevator was redesigned to take up the fuel assembly or other core components to be inspected (see Fig.7). The fundamental principle of operation, however, remains the same.

3. AUTOMATIC FUEL ROD TESTING AND MANIPULATION EQUIPMENT FOR RECONSTITUTION

All nuclear power plants built by Siemens and some of those supplied by other vendors are provided with standard reconstitution equipment which is used to detect and replace defective fuel rods and to withdraw individual fuel rods for detailed examination. One set of transportable reconstitution equipment is available for use in all other pressurized water reactors for the same tasks.

Siemens has supplemented its reconstitution equipment with a fully automatic system. This equipment is a new design concept capable of high-speed fuel rod transfer from a damaged skeleton into an intact one, sometimes referred to as "reconstruction".

The new concept of the fuel assembly repair process is directed towards carrying out all repair work steps at the fuel storage rack elevation. This equipment can be used for assemblies with removable upper or lower end fittings, depending on the design of the assembly. The tilting device, which is part of the standard equipment, is used for inverting the assemblies for access to the lower end fitting. A baseplate covering three rack positions equipped with pedestals is placed on top of the storage rack. Figure 8 depicts an arrangement in a German nuclear power plant.

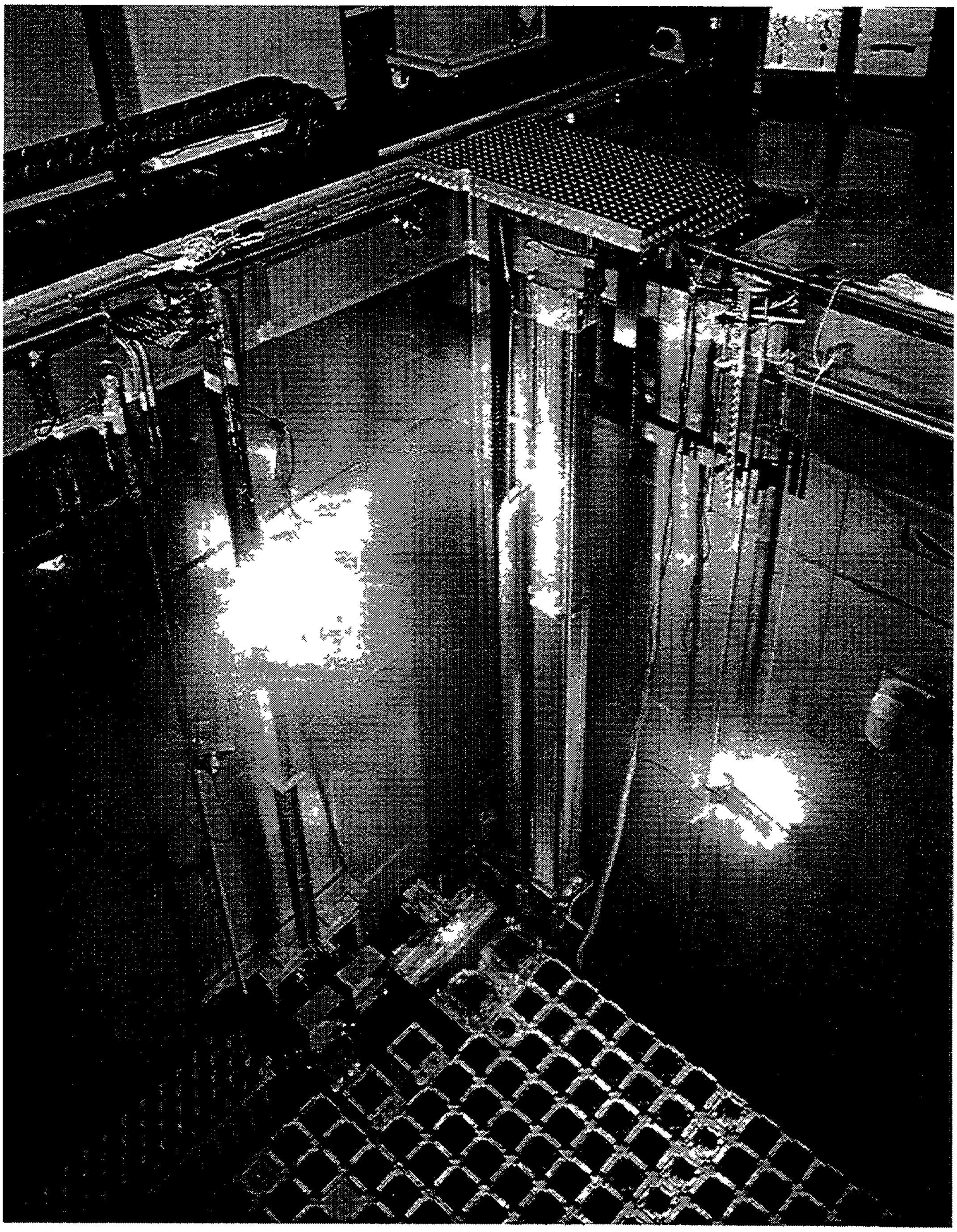


FIG 5 Multiinspection at Gosgen

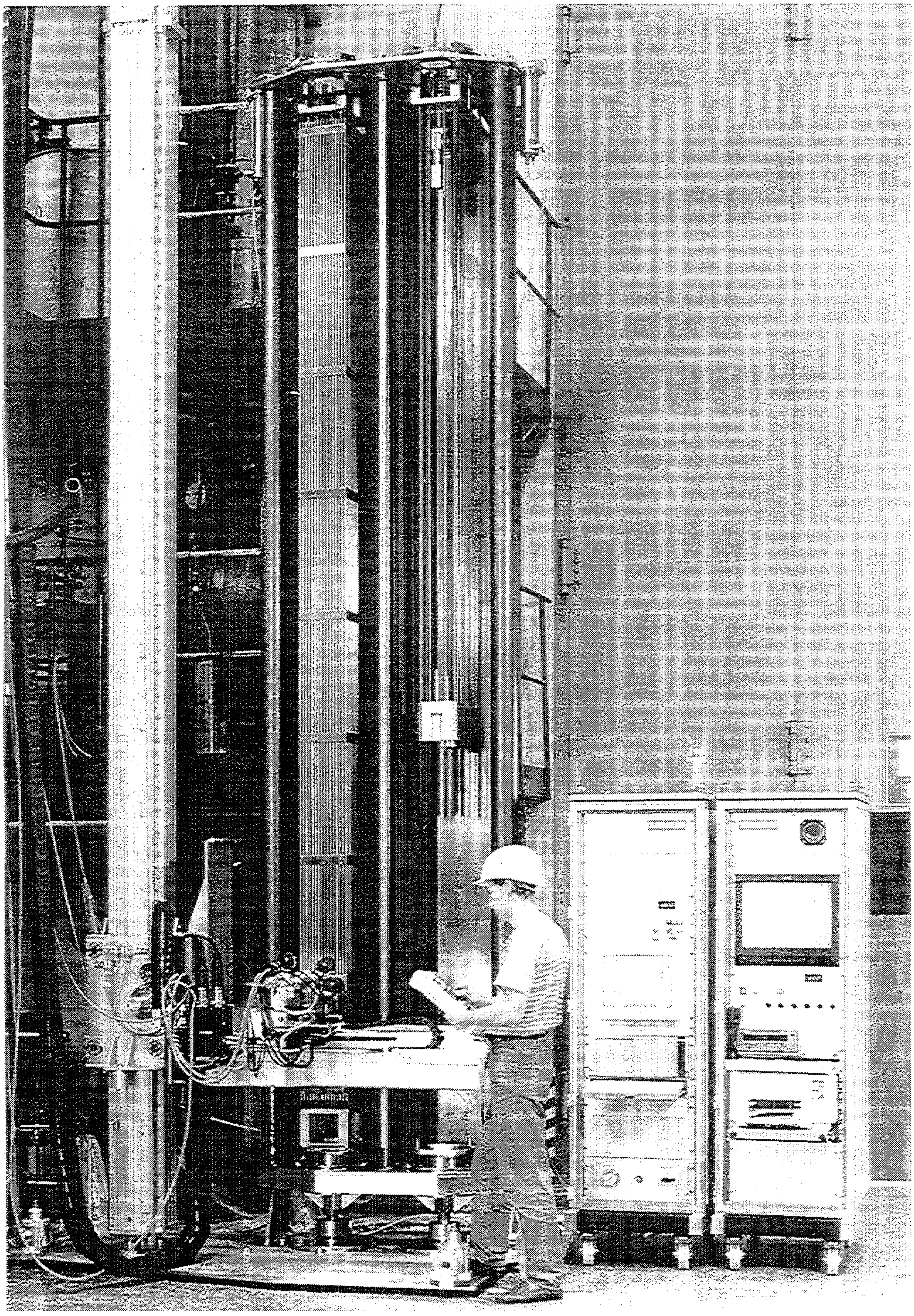


FIG. 6. *Multiinspection with mast*

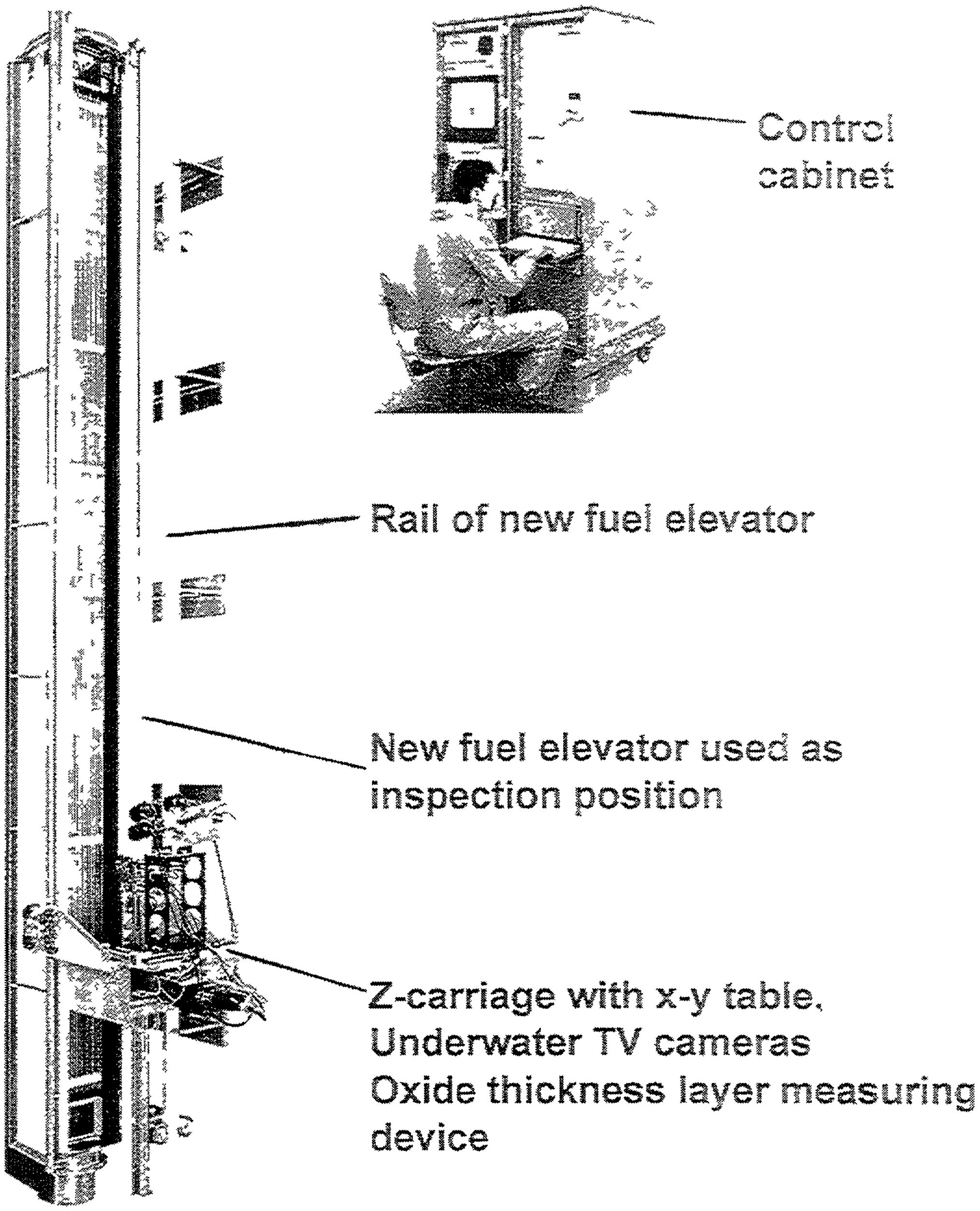


FIG 7 Multiinspection in combination with new fuel elevator

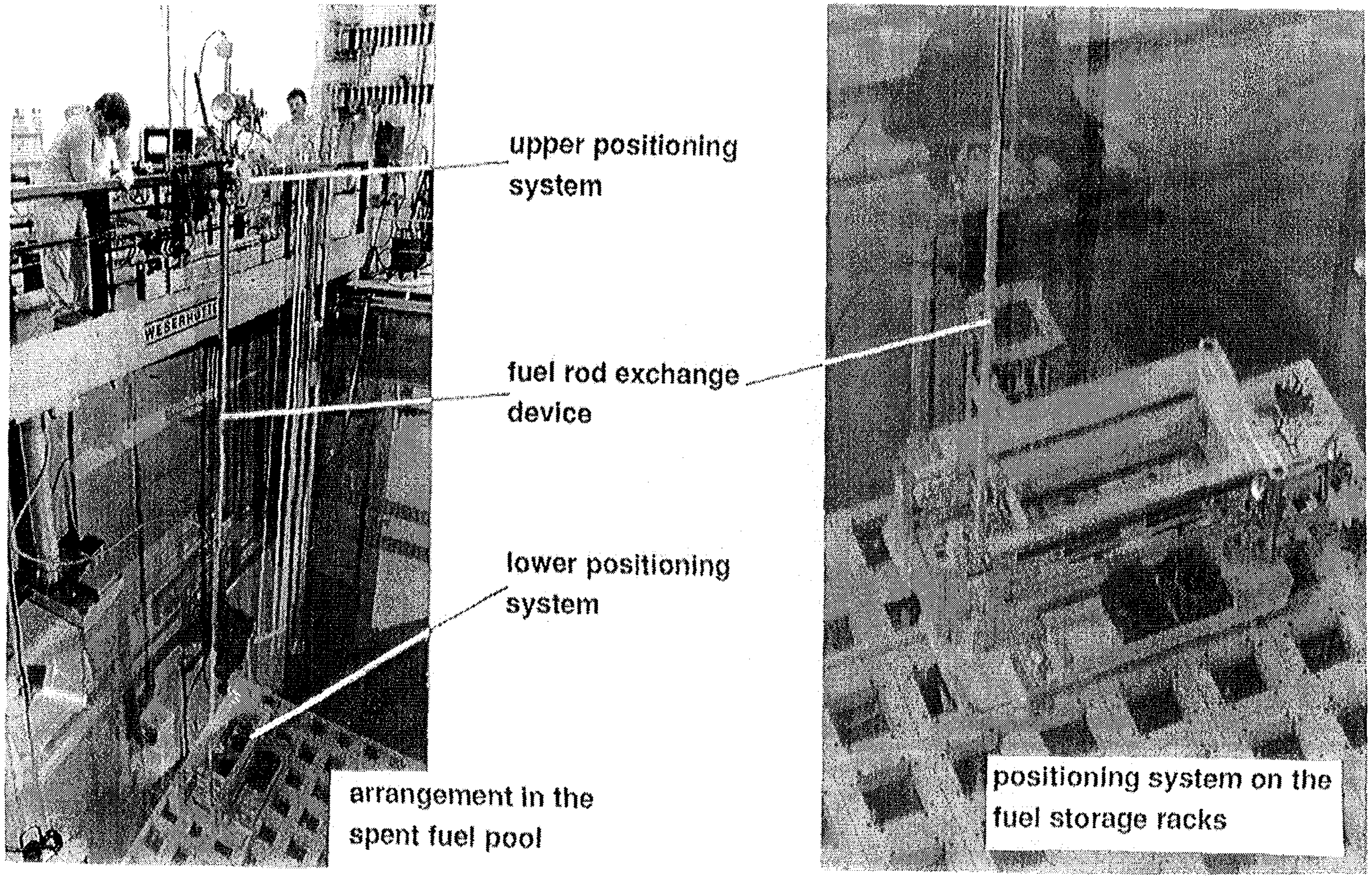


FIG. 8. PWR - fuel assembly repair, transfer of fuel rods into a new skeleton

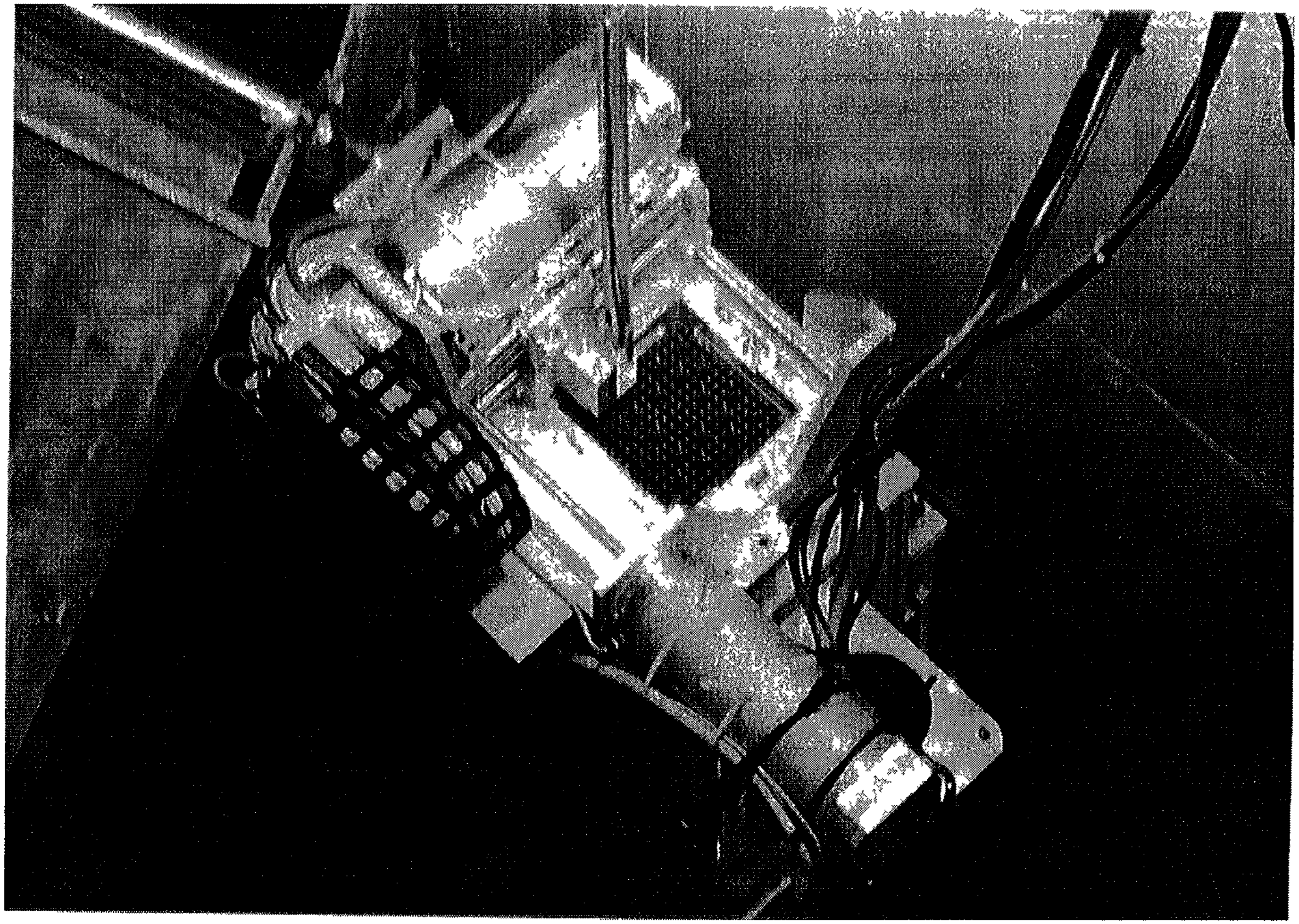


FIG. 9. Coordinate-controlled carriage for fuel rod exchange device guidance

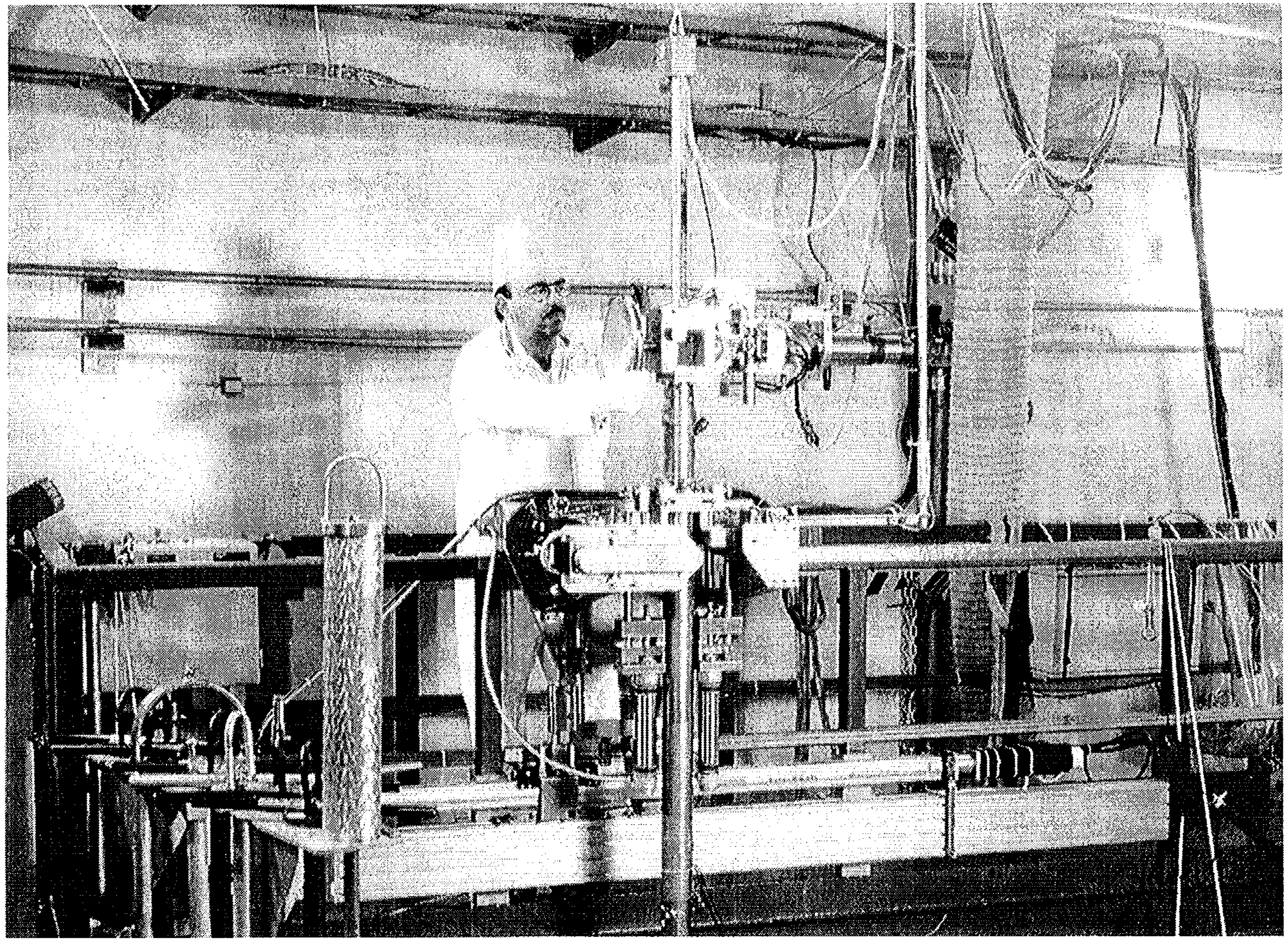


FIG. 10. Fully automated fuel rod exchange device for fuel rod testing and movement

3.1. Main new features of the reconstitution equipment for skeleton change

Our fuel rod exchange device with its pneumatically actuated fuel rod gripper provides fully automatic fuel rod handling. The computer-controlled drive system which raises and lowers the fuel rod follows a velocity/travel program. A sensor unit for monitoring the forces during withdrawal and insertion of fuel rods has been directly integrated in the draw bar of the fuel rod exchange device. The device halts the insertion process and automatically starts to retract the fuel rod if the thrust increases above a predetermined value which is dependent on the fuel assembly design.

The coordinate-controlled carriage moves the fuel rod exchange device to the position of the desired fuel rod location and is a particularly useful tool for fast and fully automated transfer of all fuel rods from a damaged assembly into an intact skeleton. A considerable amount of time is saved in comparison with earlier techniques. For example, it now takes only about 21 to 24 hours to complete fuel rod transfer between two PWR fuel assemblies depending on the type involved (15x15 or 16x16). The computer which controls and monitors the transfer operation also records process stages and important measurements such as friction force during rod insertion and eddy current testing at the same time.

3.2. Coordinate-controlled carriage for individual fuel assembly reconstitution

Siemens also uses an automatic coordinate-controlled carriage to test rod by rod without transferring fuel rods to new fuel skeletons. This new device can be integrated into the existing reconstitution equipment of a power plant. It allows fully automated testing of the integrity of the cladding tubes of all fuel rods in a fuel assembly in the relatively short space of time of less than 10 hours. This is equivalent to three service personnel shifts per fuel assembly less than needed with earlier inspection methods.

Fig. 9 shows the coordinate-controlled carriage placed onto a fuel assembly, while Fig. 10 shows the upper part of the fuel rod exchange device with an x-y carriage placed on the rail of a refueling bridge.

3.3. Process control of fuel rod movements

The fuel rod movements of all coordinate-controlled carriages are controlled by four units: The first unit controls the positioning system which is used to place the fuel rod exchange device vertically on an arbitrary fuel rod position in one of the three assembly positions.

The second unit controls the fuel rod exchange device, i.e. the automatic rod gripping process, rod withdrawal with maximum speed, rod re-insertion with limited force and varying speed, and release of the re-inserted fuel rod.

Unit three comprising a multi-channel A/D converter is used to digitize eddy-current and spring force measuring signals supplied by instruments of the fuel rod exchange device and triggered by the fuel rod upward or downward movement.

The host computer as the fourth unit coordinates the tasks performed by the above mentioned units.

4. ADVANTAGES OF NEW INSPECTION TECHNIQUES

The **mast sipping technology** to identify leaking fuel assemblies allows detection during removal of the fuel assemblies from the reactor. This provides earlier information on the condition of fuel assemblies which are scheduled for further operation. The time advantages of 60 hours can be used for changing the core design or carrying out preparations to reconstitute a defective fuel assembly.

Inspections carried out with MULTIINSPECTION allows parallel work with the fuel handling system while the inspections are being conducted. Inspections can be started during unloading of the fuel assemblies from the reactor and during change-outs of control rod assemblies and flow restrictors. This provides time savings up to four days during the shutdown period of a reactor.

The **automatic reconstitution equipment** based on coordinate-controlled carriages combined with our fuel rod exchange device allows rapid fuel rod transfer between a damaged fuel assembly to a new skeleton or a fast test rod by rod of a defective fuel assembly.

The following table provides an overview.

Work conducted	Time required	Time required with earlier methods
Skeleton change 15x15	21 h	50 h
Skeleton change 16x16	24 h	60 h
EC test on all fuel rods (type 16x16)	<10 h	40 h

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FUEL ASSEMBLY CONFIGURATION IMAGE ANALYZER

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XA9848394

Abstract

Neutron irradiation inside an operating nuclear reactor changes the dimensions of the reactor fuel assembly and its components. For example, irradiation can lengthen the fuel assembly and fuel rods, and change the gap between fuel rods.

Mitsubishi Heavy Industries, Ltd. and Mitsubishi Nuclear Fuel Co., Ltd. have jointly developed a new, computer-assisted system to measure such changes. Using this system, a fuel assembly can be videotaped with underwater cameras and its dimensions precisely analyzed through efficient processing and automatic measurement of the video images.

1. INTRODUCTION

Neutron irradiation inside an operating nuclear reactor changes the dimensions of the reactor fuel assembly and its components. It is important to measure those changes to verify fuel-assembly dimensions. The resulting data are then stored and used to evaluate the effects of irradiation on the fuel assembly.

In Japan, the integrity of all irradiated fuel that will be reused must be confirmed by visual inspection during the refueling outage. Irradiated fuel is inspected and videotaped with underwater cameras. The fuel assembly configuration image analyzer, a new system developed jointly by MHI and MNF, can accurately measure and analyze the videotaped images. This paper provides a general introduction to the new system.

2. GENERAL DESCRIPTION OF THE SYSTEM

2.1. Features

The system uses an image processor-analyzer with high-powered resolution and high-speed operating functions. It establishes measuring positions automatically by analyzing the shading and concentration of video images, providing quicker and more accurate measurements.

Lighting conditions and the characteristics of underwater video cameras vary slightly from one plant to another, affecting the quality of fuel images obtained through the usual method of inspection. The new image-processing system, however, is applicable to every plant.

Specifications of the image processor-analyzer are shown in Table 1. An analyzed image and measurements are shown in Figure 1.

2.2. System Configuration

Underwater TV cameras, ascending and descending on both sides of the fuel assembly, separately videotape it. The new system consists largely of image- and data-processing equipment, which automatically measures the right and left sides of the fuel assembly.

The images are processed and analyzed according to measurement instructions, after a correction (by TBC, Time Base Corrector) of distortions due to rotational variations in the videotape.

Image processing automatically provides coordinates for each measuring position. Data processing sequentially converts distances between image coordinates into measured dimensions.

The system configuration is shown in Figure 2, and the measurement processing flow is shown in Figure 3.

TABLE I. MAIN SPECIFICATIONS OF IMAGE PROCESSER-ANALYZER

Item		Specifications
Image shading		256 levels of gray
Image resolution	Horizontal	1,024 pixels (approximately 0.15mm per pixel)
	Vertical	512 pixels (approximately 0.30mm per pixel)
Main processing functions		-Smoothing process -Edge emphasis process -Binarization -Frequency analysis by Fast Fourier Transformer
Processing time		Approximately 3-6 sends per image

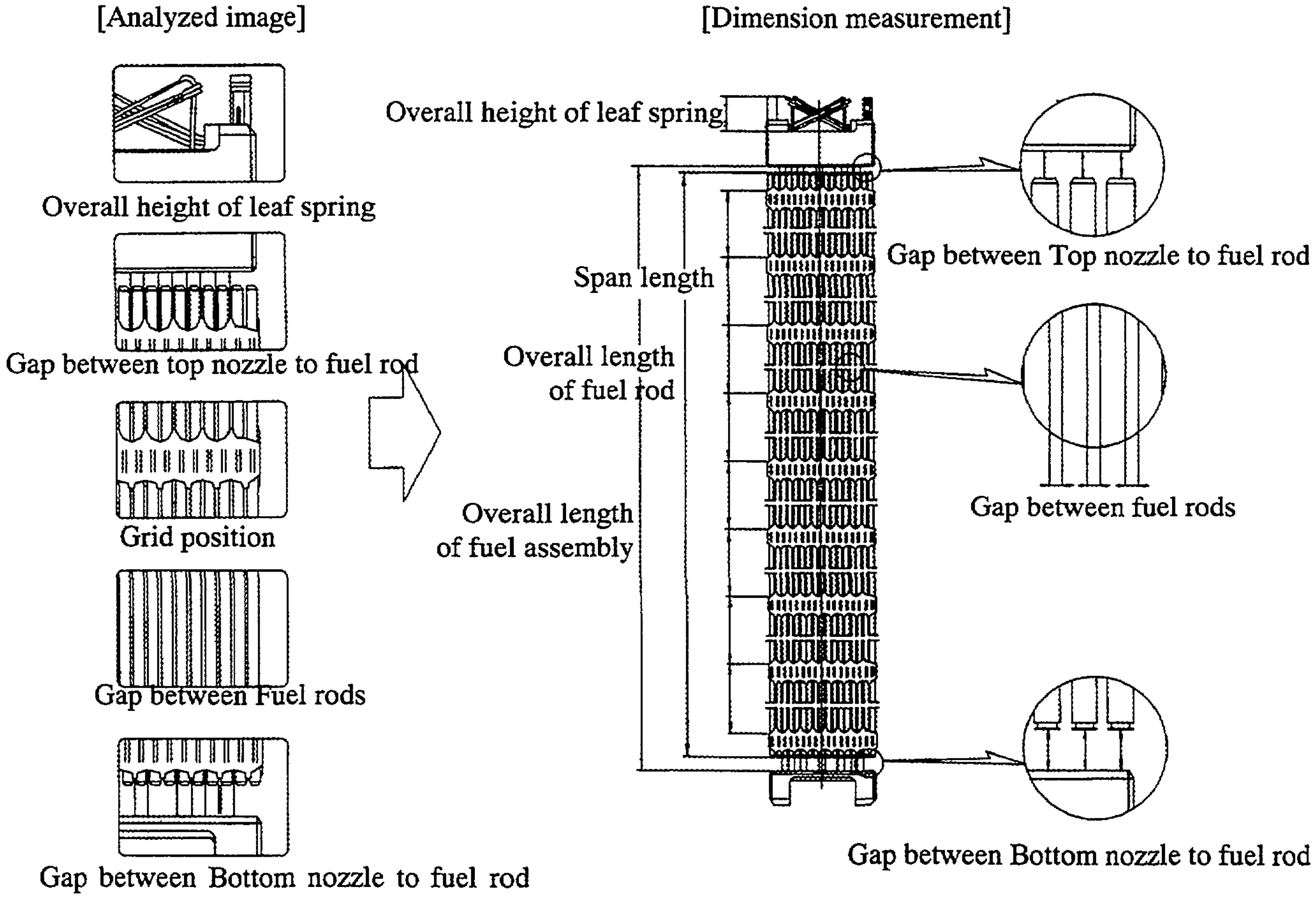

FIG.1. Analyzed image and measurement item

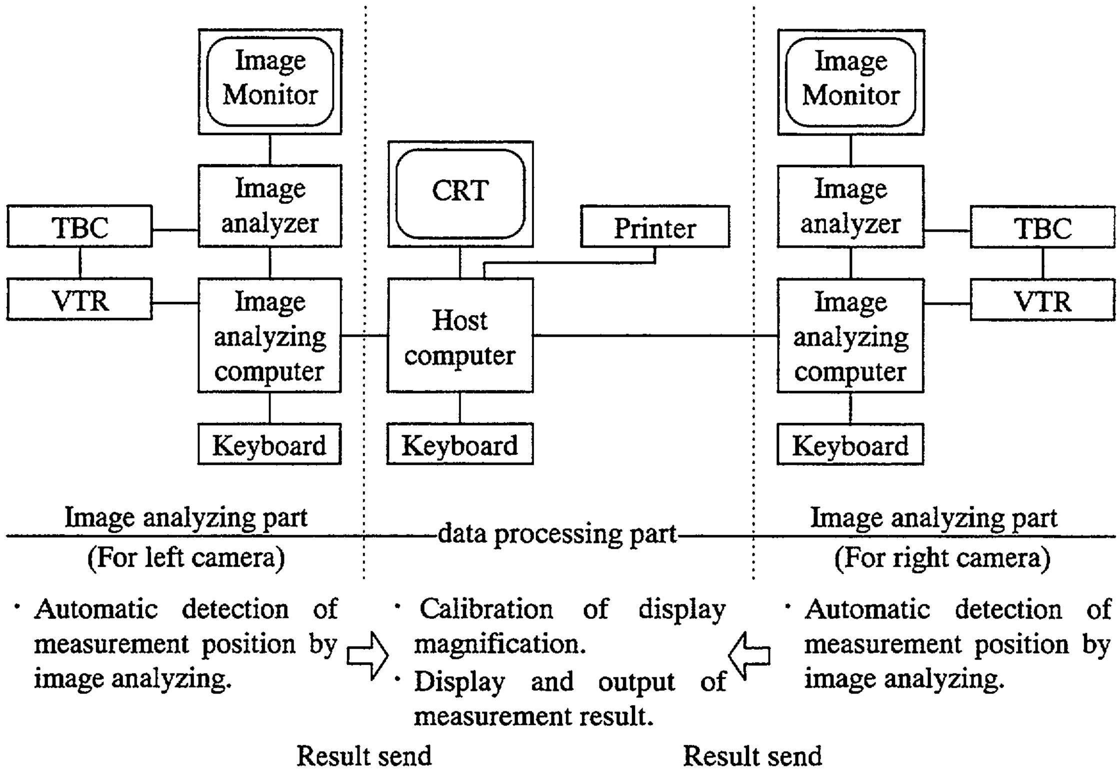

FIG. 2. System of fuel assembly configuration image analyzer

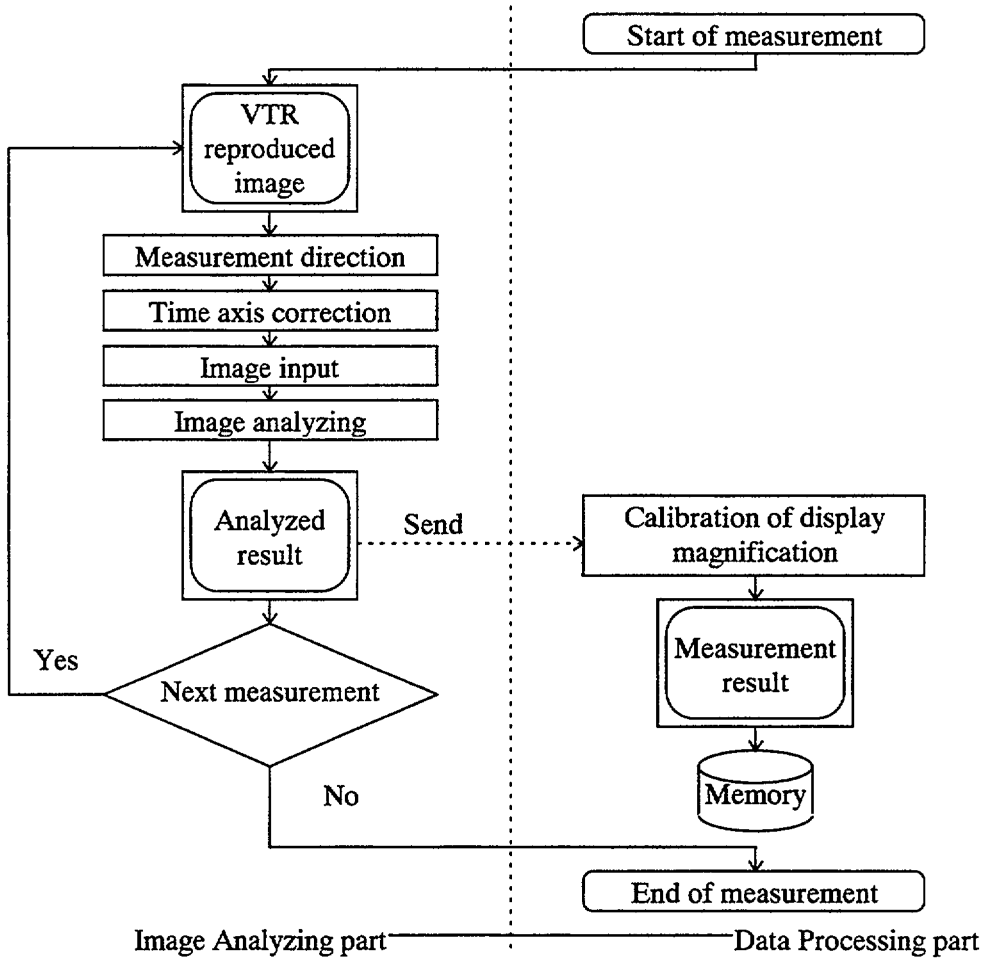

FIG.3. Flow Chart of Measurement

3. IMAGE ANALYZING METHOD

Automatic detection of measuring-position coordinates makes it possible to analyze five types of images. An outline of the system follows.

3.1. Measuring Grid Position

The grid is clearly reflected, but the intensity of its image changes abruptly near the outer edge. The image of the grid is accentuated at its outer edge, but only where the gradient of concentration changes significantly. Grid-position coordinates on the image are established through binarization, to select a concentration gradient above a certain threshold. (Grid-edge accentuation and an example of binarization are shown in Figure 4.) An analysis of the result provides a way to calculate span length and magnification-calibration values for the display image and the real image, by comparing them with the grid-design dimensions.

3.2. Measuring Fuel Rod to Rod Gap

Smoothing and integrating density distribution eliminates inconsistencies in the shading and illumination of fuel-rod-surface images. Coordinates for the right and left edges of a fuel rod are established by comparing the known outer diameter of the fuel rod to the density distribution of its video image.

3.3. Measuring the Gap between Fuel Rod and Top or Bottom Nozzle

It is difficult, through edge accentuation alone, to distinguish the top or bottom edge of a right fuel rod from that of other fuel rods. A density value around the top or bottom edge of the right fuel rod should be determined in relation to the other rods. Then determine the top or bottom edge of the right fuel rod and establish its coordinates. From this result, fuel-assembly length and overall fuel-rod length can be calculated.

3.4. Measuring Overall Leaf-Spring Height

It is difficult, using only the edge-accentuation process, to distinguish the top of a right leaf spring from other fuel-assembly components. All coordinates for the top edge of the correct leaf spring should be traced in their proper order. Determine the top edge of the leaf spring from the trace result that matches its shape, and then establish its coordinates.

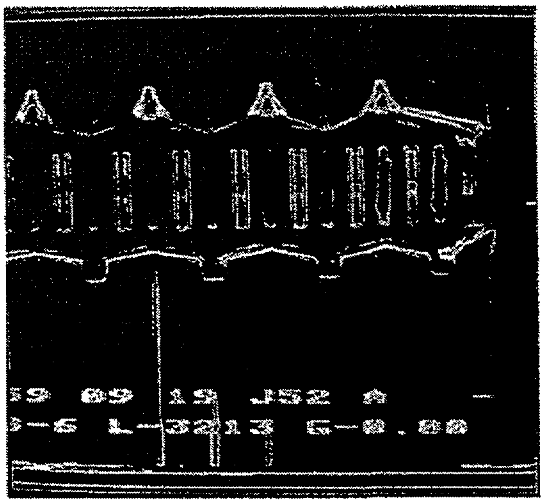

FIG.4. Grid-edge accentuation and an example of binarization

4. OUTPUT

During the usual visual inspection, a fuel-handling tool is used to hang the fuel assembly on an inspection stand. As a result, the fuel assembly can sway like a pendulum, causing a vibration in its video image. Underwater video cameras may shake when they move, which also causes image vibration.

The new system eliminates vibration, producing a clear image of the compressed fuel assembly. This system allows an accurate assessment of how irradiation affects fuel-assembly behavior In Figure 5, an output sample for a compressed fuel assembly clearly shows fuel-rod bowing.

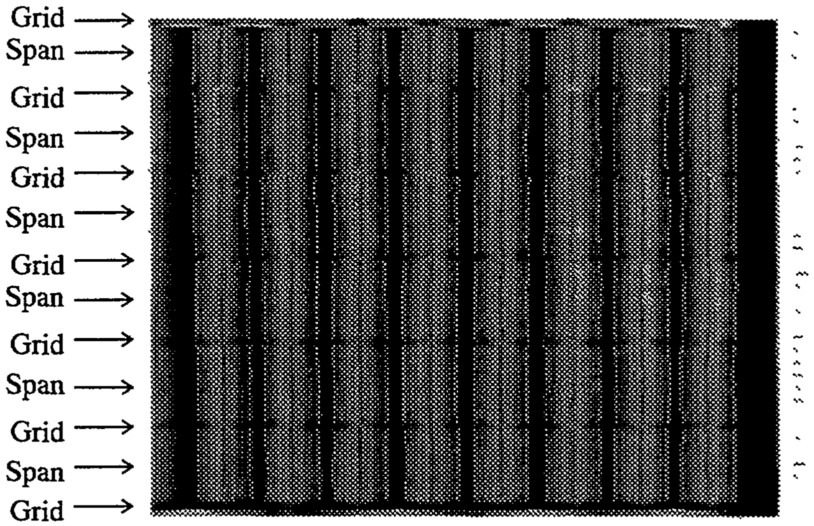

FIG.5 Image of Compressed Fuel Assembly.

5. PERFORMANCE EVALUATION

Dimensions were measured, using video tapes of the on-site system.
Measurement accuracy and automation rates were examined.

5.1. Automation Rate

Automation of up to 80 percent can be accomplished, considerably shortening the required measurement time. (Measurement time decreases from 8 hours to about 40 minutes for each fuel assembly.) The automation rate should increase with improvements in the image-processing unit.

5.2. Measurement Accuracy

Figure 6 shows the dispersion of measurement errors. It assumes errors of no more than 2 pixels, equivalent to acceptable errors due to image resolution. It also shows that the system can accurately measure variations in fuel assembly dimensions.

6. CONCLUSION

This system was developed to accurately measure grid position, leaf-spring height, top and bottom nozzle position, the gap between fuel rods, and the gap between fuel rod and nozzle by

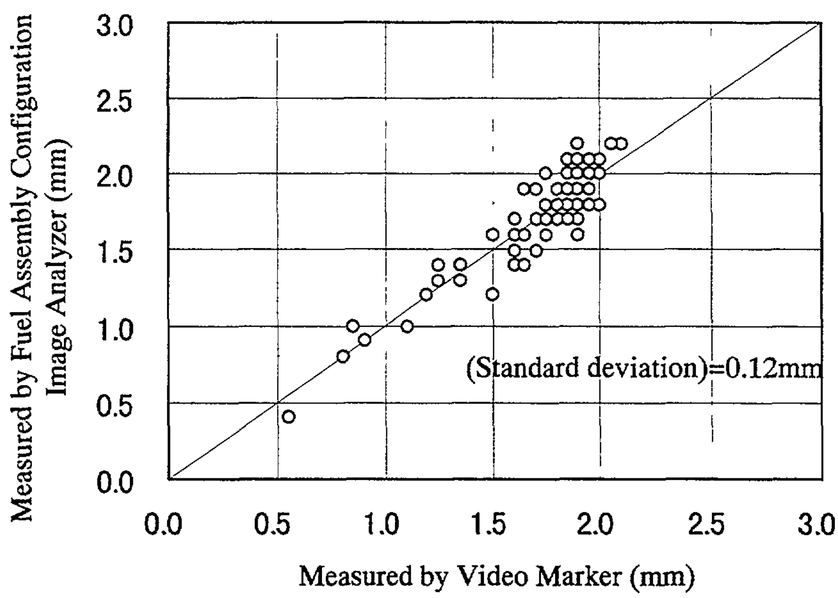

FIG. 6. Measurement accuracy

analyzing images from the video inspection of a fuel assembly. Fuel assembly length, grid-to-grid length and rod length can be determined from the measurement data. A clear image of the compressed fuel assembly can be produced by eliminating vibration in the underwater cameras and fuel assembly during video inspection.

On-site video tapes were used to determine measurement accuracy. Using this system, measurements took about 40 minutes for each fuel assembly.

XA9848395

DEVELOPMENT OF RCC GUIDE THIMBLE DISTORTION INSPECTION EQUIPMENT

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Abstract

An on-site inspection equipment for thimble distortion of fuel assembly, which has been focused concerning to incomplete RCC insertion, has been developed. The simple mechanism employing elastic deformation of flexible pipe successfully composed a reliable and easy-fabricated configuration of equipment. By the mock-up test its measurement accuracy was evaluated and its usefulness was confirmed.

1. INTRODUCTION

Irradiation deformation of the fuel assembly and consequent incomplete RCC insertion are considered to be the potential issues on high burnup of PWR fuel. For the help to study the cause and to find a solution, it is desired to develop an equipment to be used at poolside to measure the distortion of RCC guide thimbles which form the structural skelton of fuel assembly.

Mitsubishi has developed a handy inspection equipment to be able to access from the upper part of fuel without dismantling assembly. This paper presents its concept and the performance comfirmed by the mock-up test.

2. THIMBLE DISTORTION INSPECTION EQUIPMENT

2.1. Requirement for inspection equipment

In order to investigate the distortion of RCC guide thimbles during inservice inspection, measurement without dismantling the fuel assembly is indispensable although the thimbles prevent our visual access. Then the manner to insert sensors into the thimble is most practical approach. In this approach the size of equipment is required small enough to pass through the inner space of thimble tube which center line is eccentric due to the assembly bow up to several mm or over along the full length about 4 m. The existence of dashpot structure in thimble makes it difficult to inspect from inner space of thimble.

In order to satisfy above requirements we developed a flexible type of equipment suitable to insert into thimbles. This “flexible” is also the key concept concern with the principle of measurement for our developed system.

2.2. Configuration of equipment

The thimble distortion inspection equipment (TDIE) is the simple application of strain measurement. The main part of it locates in the lower portion and consists of a flexible pipe and stabilizers as shown in Fig.1. Three strain gauges are plastered on the pipe at the same elevation in 120 degrees circumferential intervals as shown in Fig.2. The material and dimension of the pipe are

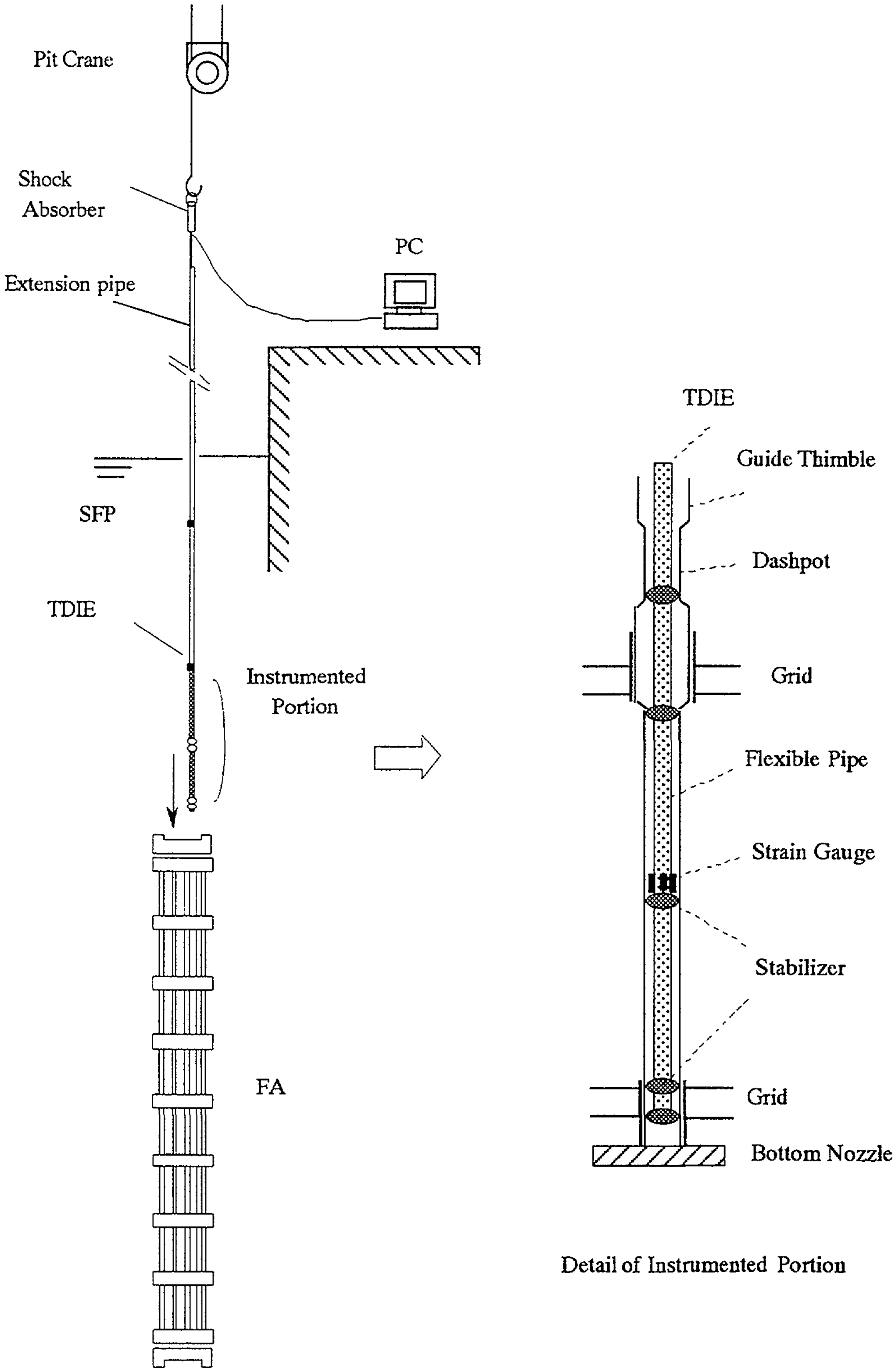


FIG. 1. Congiguration of TDIE

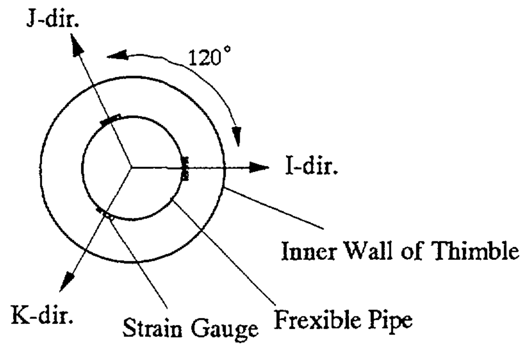


FIG. 2. Arrangement of Strain Gauges

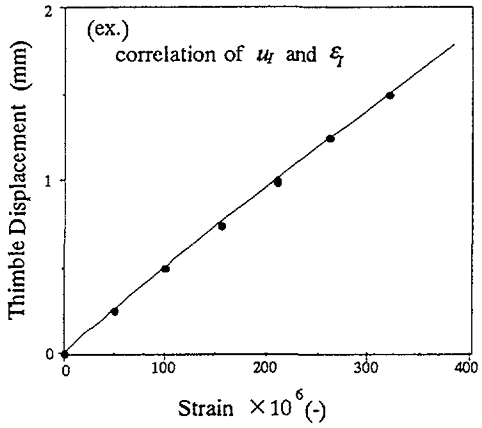


FIG. 4. Example of Calibration Data

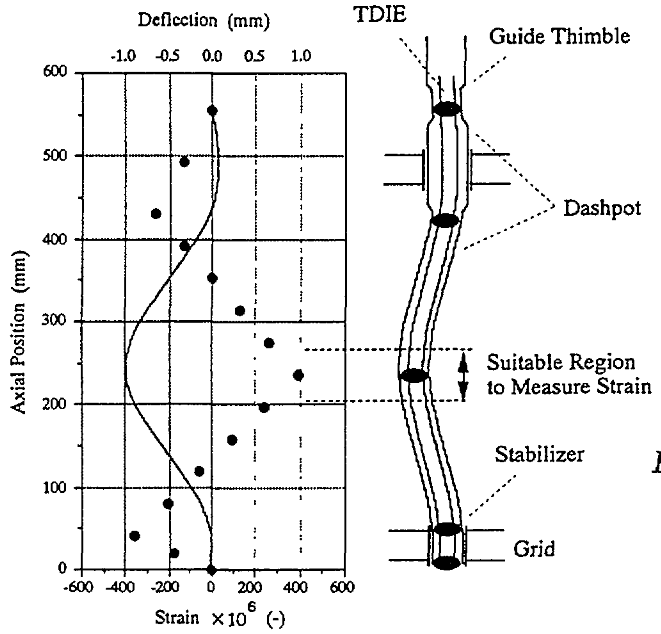


FIG. 3. Layout of TDIE for Typical Thimble Distortion

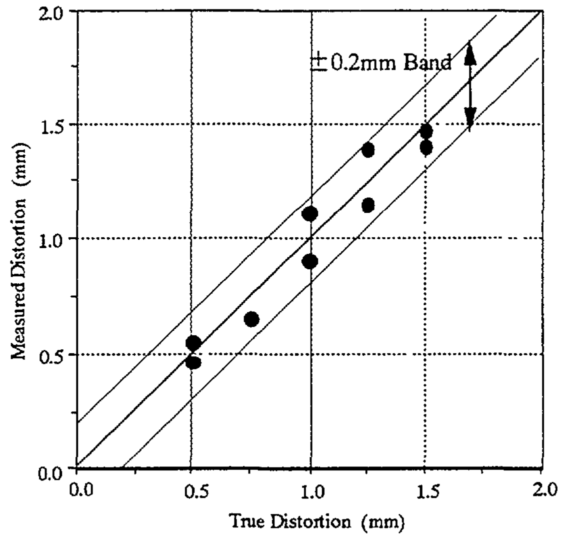


FIG. 5. Comparison between Converted Distortion and Actual one

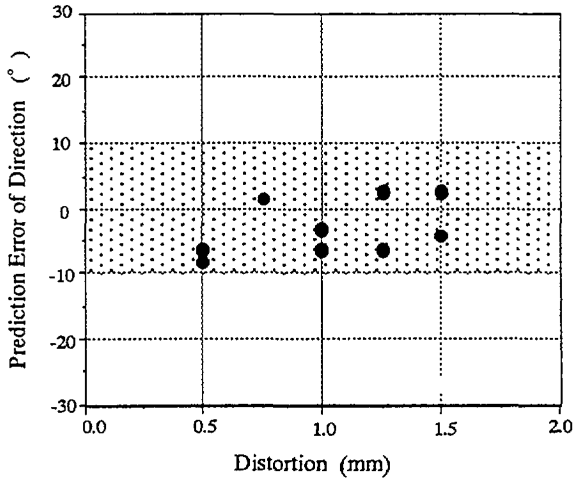


FIG. 6. Prediction Error of Direction

designed in the viewpoint of easy insertability with elastic deformation along the shape of distorted thimble. In the equipment thin stainless pipe of 6mm diameter with 0.3mm thikness is adopted.

The stabilizers control the performance of the equipment. The smaller clearance between stabilizer and inner wall of thimble the higher accuracy is obtained although too tight clearance leads to stick of equipment in the thimble. The material of it is also important bacause the low friction to inner wall of thimble is nessesary. Nylon is used for our equipment. The location of stabilizers determines the sensitivity of measurement. In the equipment two pairs of stabilizers are arranged in accordance with the both ends of 1-span thimble length and another one is used to fit with the expected distortion shape of thimble. This arrangement makes the pipe deform in the similar situation as both ends fixed type where it results in higher bending strain. The signals of a set of three strain gauges are analysed to determine the magnitude of thimble distortion and its direction.

Fig.3 shows the layout which is suitable to detect the typical shape of thimble distortion. For this type of distortion shape, the mid-span position is the best place to measure bending strain with high accuracy. Beside plots are calculated bending strain of the pipe when it deforms in the condition of hard contact between pipe and thimble at stabilizer's positions. In addition the equipment of axially multi-instrumented with suitable arrangement of stabilizers is more useful. It covers almost all the situations which will occur on the guide thimbles in fuel assembly.

At first above layout equipment was designed for exclusive use to bottom span considering narrow tube region of dashpot. On the other hand for the middle and upper span thimble inspection, another one was prepared to be fit the wider diameter of thimble with large size stabilizers. This type covers the measurement for all span except bottom.

2.3. Calibration

The cross section of flexible pipe at the elevation where strain gauges are plastered is shown in Fig.2. Let the three directions of gauges to I, J and K axes respectively.

The strain output of each gauge on the pipe is calibrated as follows.
- insert equipment into thimble at prescribed span position
- let thimble deflect to the I, J and K direction respectively
- detect and edit each strain output with magnitude of thimble deflection for each direction respectively

It is necessary to obtain the correlation between bending strain on flexible pipe and distortion of thimble (not instrumented pipe) because the unknown factor such as the stabilizer-thimble clearance effect exists. A sample correlation is shown in Fig.4. This calibration is performed at laboratory before poolside inspection.

2.4. Measurement

In the poolside inspection the handling tool is hung on the crane of spent fuel pit, and the equipment mounted on the extension pipe is inserted into the target thimble manually (Fig.1). When the equipment reaches the target span position, a set of strain gauge singals are recorded.

2.5. Data Processing

The thimble distortion is derived by the following procedure:

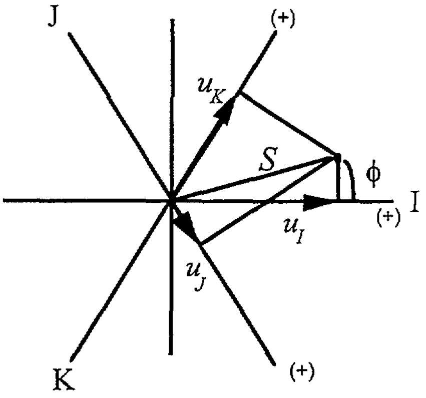
convert each measured strainto the displacement u for each direction using above calibration data

$$u_I = \quad sign\,(\varepsilon_I)\,|\,u_I\,|$$

$$u_J = -sign\,(\varepsilon_J)\,|\,u_J\,|$$

$$u_K = -\,sign\,(\varepsilon_K)\,|\,u_K\,|$$

calculate a magnitude of distortion $\boldsymbol{S}$ and its direction from 3 components of $\boldsymbol{u}$



$$S = \sqrt{u_I^2 + \frac{(u_K - u_J)^2}{3}}$$

$$\phi = \tan^{-1}\left(\frac{1}{\sqrt{3}}\frac{u_K - u_J}{u_I}\right)$$

3. MOCK UP TEST

In order to confirm the performance of the developed equipment, mock-up test was conducted at our laboratory. A skelton assembly (17x17 fuel type) was used because various pattern of thimble distortion could be easily produced. Test results on aforementioned equipment applied to bottom span of thimble are described here. Handling test was performed separately in the similar situation to SFP such as the fuel assembly in deep water.

Fig.5 shows the example plot of the converted distortion from measured strains and actual one. Fig.6 shows the prediction error of direction of distortion. The accuracy of measurment for the magunitude and direction are within 0.2mm and 10 degrees respectively.

In the poolside inspection at plant site the accuracy of measurement would slightly reduce due to the difficulties on handling. However the accuracy of 0.3mm on magnitude and 30 degrees on direction would be ensured.

4. CONCLUSIONS

A compact and reliable inspection equipment for thimble distortion has been developed. The measuring accuracy was confirmed within the useful range. This equipment makes the on-site inspection of thimble distortion perform easily and results in the useful information for the recent anxious issue of RCC incomplete insertion.

THERMOPHYSICAL INSTRUMENTS FOR NON-DESTRUCTIVE EXAMINATION OF TIGHTNESS AND INTERNAL GAS PRESSURE OF IRRADIATED POWER REACTOR FUEL RODS

XA9848396

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Abstract

The developed thermophysical method and technical instruments for non-destructive leak-tightness and gas pressure inspection inside irradiated power reactor fuel rods and FAs under poolside and hot cell conditions are described. The method of gas pressure measuring based on the examination of parameters of thermal convection that aroused in gas volume of rod plenum by special technical instruments. The developed method and technique allows accurate value determination of not only one of the main critical rod parameters, namely total internal gas pressure, that forms rod mean life in the reactor core, but also the partial pressure of every main constituent of gaseous mixture inside irradiated fuel rod, that provides the feasibility of authentic and reliable leak-tightness detection. The described techniques were experimentally checked during the examination of all types power reactor fuel rods existing in Russia (WWER, BN, RBMK) and could form the basis for new technique development for non-destructive examination of PWR (and other) type rods and FAs having gas plenum filled with spring or another elements of design.

1. INTRODUCTION

Nowadays power reactor fuel assemblies (FAs) have reached a high level of manufacturing quality and operational reliability. On the whole, reactor cycles are completed without fuel rod failures [1]. However, especially in the first period there are some failures of fuel rods (up to about 0.01% of rod sum total [2]). This results in fission product releases into coolant system.

Fission gas products (FGP) partially escape through a cladding defect and water entering a fuel rod starts reacting with fission products. As a result volatile chemical compounds are formed that may escape like FGP or dissolve and leave a fuel rod as dissolved species.

The resulting high activity concentration in the primary coolant circuit is largely responsible for increased irradiation exposure of plant personnel. The larger portion of FGP from failed fuel rods reach the atmosphere and poison it. Radioactivity releases to the atmosphere cause irreparable harm to people and nature.

Since nuclear plants have licensing limits to volatile fission product releases to the environment the careful monitoring of clad failure evolution becomes indispensable. Limitation on the activity concentration level by the licensing limit results in a loss of reactor operational flexibility.

On account of this the nuclear fuel reliability receives the highest priority in the nuclear industry today. Fuel manufacturers have programs aimed at achieving defect-free fuel operation [2,3]. These programs cover as a rule fuel quality inspection at the fuel manufacture plant and detailed on-site examination of spent FAs.

In order to decrease radioactivity releases due to fuel failures, the reactor is shut down and failed FAs are transferred to a cooling pool. Thus a large amount of almost fresh fuel is removed from the power cycle which entails high economical losses.

The poolside inspection of irradiated FAs to identify failed rods (leakers) - is the main part of FA reconstitution technology. Another reason of leaker identifying necessity (but already inside spent FAs and including the subsequent certain removal of the detected leakers) - is to ensure the FA transmission feasibility for dry storage and for reprocessing.

Nowadays the main economical, technical and technological aspects of poolside fuel inspection and repair (reconstitution) of FAs have been resolved in countries having high level of economics and atomic energy [1,4,5,6]. The effectiveness of inspection and reconstitution efforts evidently depends primarily on leak-tightness test probe quality (sensitivity, accuracy and reliability).

Moreover the internal rod gas pressure - is one of the main critical parameters that forms rod mean life in the reactor core, therefore this parameter is always in field of sight of the researchers. According to the stated reasons the aim of our efforts during last dozen years became the elaboration of method and technical instruments for non-destructive leak-tightness detection and gas pressure measuring inside irradiated fuel rods (both individual and within FA) under poolside and hot cell conditions.

2. THERMOPHYSICAL METHOD

Nowadays different information sources describe great number of methods and devices used for leak-tightness and pressure tests of rods and FAs, see, for example [7-11] and others. However, almost all of them are far from being both completed and utilized in practice and are interesting for us only as an illustration of the free flight of a technical idea.

Among well-known technical decisions two methods find practical application, one of them is based on ultrasonic sensing of rod plenum for the presence of water in it [See 4, or French modification named «Echo-330»], and the other one - on analyzing the intensity of Kr-85 gamma spectrum in the plenum area [12].

Each of them has some advantages, but at the same time many disadvantages are inherent in them that limit their application. For example, the above gamma-spectrometry method of gas pressure measurement inside irradiated fuel rods is comparatively intricate to realize under poolside conditions, it has numerous disadvantages of different methods based on radioactivity parameter analysis, such as:

- Burn up dependence (low sensitivity after low burn up),
- Influence of external radioactive background (need for careful collimating),
- Influence of irradiated fuel storage time,
- Influence of uncertainty in initial helium filler pressure in fuel rods, etc.

This method cannot be used for identification of defective rods within irradiated FA because of Kr-85 gamma-activity influence exerted by adjacent fuel rods.

The above-mentioned method of ultrasonic sensing (or a similar one) is free from the enumerated disadvantages and proved well for failed rod detection in in-pool irradiated FA inspection. But some limitations are inherent in this method, e.g.:

- Sensitivity - not less than 0.5 gram of water inside plenum,
- Only bottom plenum siting; insensitivity to other types of rod defects (for example, low gas leaker),
- Insensitivity to internal rod gas pressure, etc.

This method is not applicable for rod inspection on its short-term storage because of water evaporation as a result of afterheat.

Meanwhile the obtained results and the experience that have been gained in the field of our experimental work allow to be firmly convinced that the thermophysical non-destructive test instrument application in this area of fuel inspection will give not only huge ecological and economical benefits but also unbiased important and interesting results.

The developed principle of gas pressure measurement is based on the well known thermophysical method of the parameter measuring of an object under study, described in [13,14], where to measure the liquid flow rate and to detect defects and cracks the use is made of the perturbation effect of a heat flux going from a heater and changes are recorded in the temperature difference as dependent on the object position in relation to the heater as a result of the effect of the searched for parameter of the object.

The applied thermophysical method resulting parameter to be measured is gas pressure inside a fuel rod. In essence the thermophysical principle which forms the basis of pressure measuring method consists in the fact that in a local area of a gas plenum of a fuel rod a fixed thermal perturbation of cladding takes place leading to natural convection and accompanying it thermodynamical processes in plenum gas volume. Heat exchange between gas, warm cladding and environment depends upon the kind of a gas, its physical properties and pressure. By recording the cladding temperature field in space and time one gets information on a parameter under measurement. The information is interpreted with the help of graduated characteristics taken from standard specimens - rod imitators, filled with the same gas, as the rod under study, but having parameters known at high accuracy in the range needed.

It is obvious that in the measurement process the temperature field contains information not only on the parameter being measured but also to a significant extent on such affecting factors as:

- Manufacturing scatter of cladding geometrical sizes,
- Scatter in thermophysical properties of cladding material,
- Unstable environmental parameters,
- Multicomponent gas composition,
- Oxide film and crud at cladding surface,
- Fuel and solid fission products available in a fuel rod, etc.

Our design of the probe, method of measurement and special mathematical method of data processing allow us to single out the information on a gas pressure at adequate accuracy.

The technical problems are typical and have been resolved using the known approaches. This comprises choice of materials and elements, automation of measurements, fabrication of standard specimens of pressure and gas composition, introduction of the developed measuring device into the process line (either research or technological), metrological certification.

During the technique development two options of irradiated rod gas pressure non-destructive measurement system have been realized, based on the above-mentioned method, that is caused by a natural course of the method development and distinction in particular conditions of its application. Meanwhile, our thermophysical non-destructive rod internal gas pressure measuring method is universal from the point of view of rod inspection realization place and applicable both for poolside and hot cell conditions. In other words, the method, used for hot cell conditions of rod inspection will be efficient in poolside rod inspection conditions and on the contrary, the method, used for poolside rod inspection conditions will be efficient in hot cell conditions.

3. ROD TOTAL GAS PRESSURE MEASURING DEVICE

The described thermoconvective principle of gas pressure measurement inside irradiated fuel rods was realized for the first time by device, structural scheme of which is shown in Fig. 1.

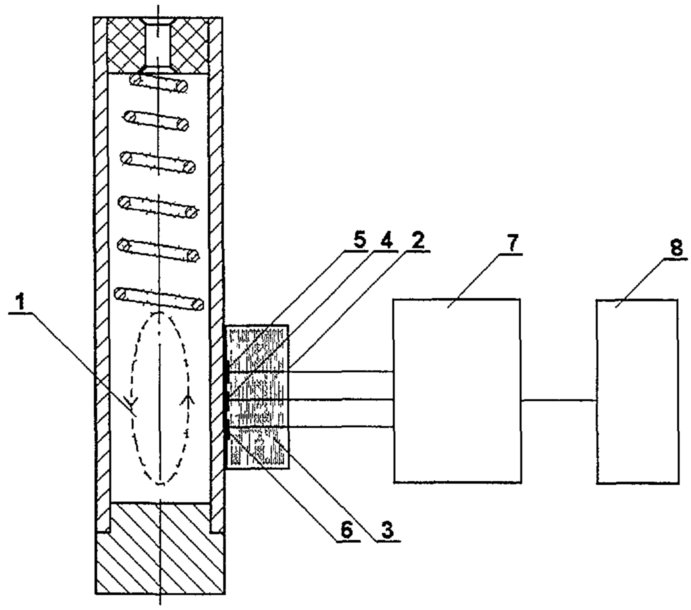

FIG. 1. Structural scheme of rod gas pressure measuring device.
1 - gas plenum; 2 - pressure probe; 3 - rubber bed, 4 - heater; 5,6 - thermoresistors;
7 - power and measurement block; 8 - personal computer.

Pressure probe 2 consists of a rubber bed 3 accommodating surface heater 4 and thermoresistors 5,6. Thermoresistors are connected to measuring bridge of block 7, that has controllable power source for measuring bridge, measuring amplifier, controllable power source for heater 4 and time interval setting mechanism. The automated control of the device blocks, the recording of measuring bridge unbalance voltage, that is proportional to a temperature drop in fuel rod cladding under study, as well as the computation of the searched for value of gas pressure are carried out with the help of personal computer 8.

The pressure probe is pressed on to a plenum of rod under study and the measurement process is carried out, recording the above-mentioned unbalance voltage during and after heat perturbation. The extreme value of the recorded curve has been taken as an informational parameter in the first experiments with irradiated rods. In the posterior experiments the special mathematical data processing and analyzing were applied to the recorded curve.

The experimental tests with variations in environmental temperature, cladding wall thickness and probe pressing conditions showed that using of this scheme the errors effected by those influencing parameters can be almost completely eliminated.

Thus, by thermoconvective gas flow excitation inside a leak-tight object through a short-term thermal perturbation of rod cladding and recording the cladding temperature drop in close proximity to the heated area it is feasible to measure gas pressure inside the object. It should be noted that this approach assumes that gas composition and content of the main gas mixture constituents are well known.

Feasibility of the developed method of total internal gas pressure measuring has been checked experimentally under BN-350 fast breeder reactor cooling pool conditions on a series of individual irradiated fuel rods. The so-called measuring manipulator (pressure probe with pressing mechanism) has been installed on the inspection stand. The inspection stand structural scheme and picture of applied measuring manipulator with hand remote control is shown in Figs 2 and 3.

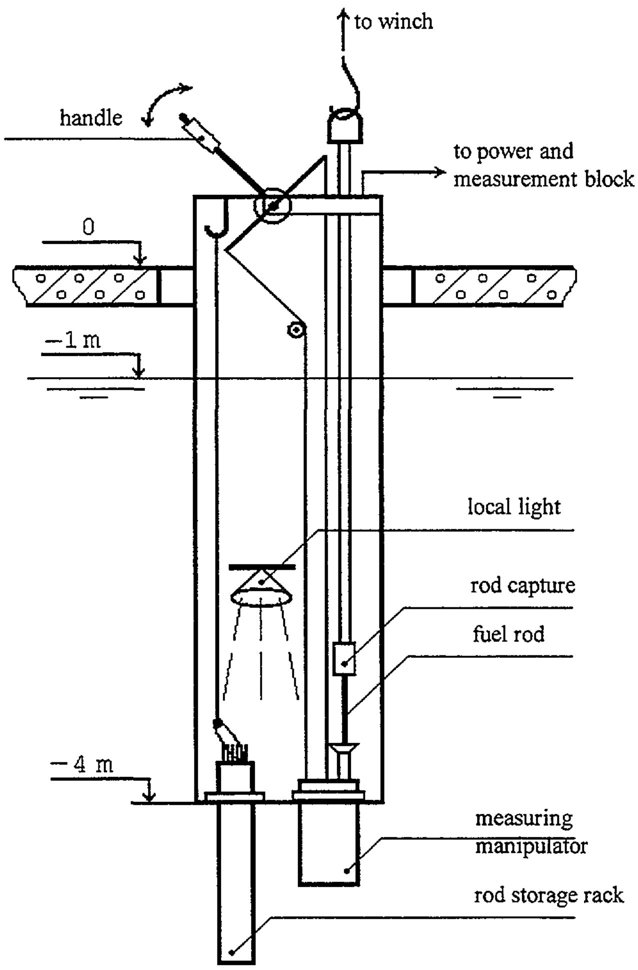

FIG. 2. Inspection stand structural scheme.

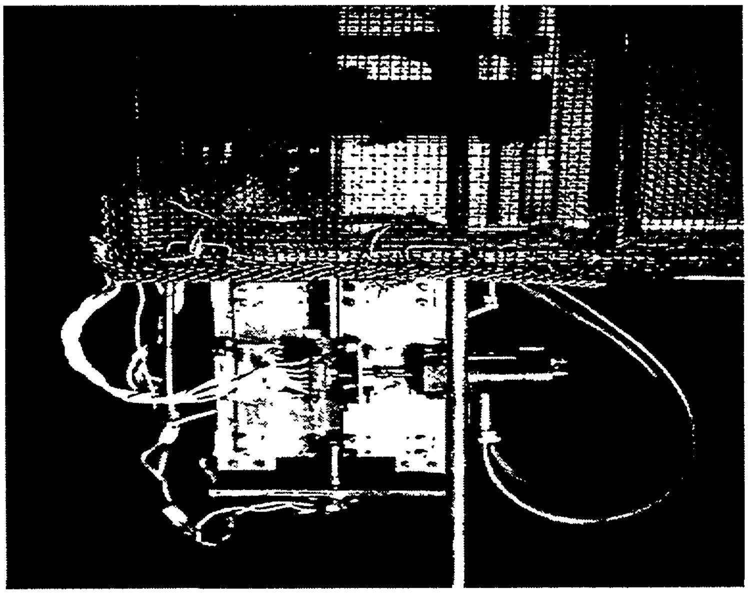

FIG. 3. In-pool rod gas pressure measuring manipulator.

A fuel rod with the bottom gas plenum was placed into the guiding funnels of the measuring manipulator that had two proximity sensors for fuel rod and its spacer wire orientation and positioning relative to the pressure probe. The first sensor is placed into lower guiding funnel and the other one - opposite to the pressure probe center. After the fuel rod and its spacer wire positioning an operator pressed the pressure probe on to the middle part of the rod gas plenum and the internal gas pressure measurement process started.

Fuel rod pressure measurements were accomplished using standard specimens filled with helium-xenon-krypton mixture with 7% of helium. The measured average results are in good agreement (at the accuracy up to 0.1 MPa) with the values calculated by the computer-calculating program for FGP releases from ceramic fuels as irradiated in BN-350 reactor. To experimental direct check the results of gas pressure was randomly measured by puncturing of several fuel rods placed under hot cell conditions. The comparison of the data received by destructive and developed non-destructive methods showed that there is full agreement of the results within the measurement error.

4. FUEL ROD INTERNAL GAS PARTIAL PRESSURE MEASURING SYSTEM

The presently used method and instruments have been improved, their technical potentialities have been widened, and the advantages of modern computer means of control and calculation of the experimental results have been applied. This assured the possibility of quantitative determination of the internal rod gas composition inside irradiated fuel rod and of improving the accuracy, efficiency and validity of the internal gas pressure measuring method. As a result, the unified system of non-destructive internal gas parameter inspection has been developed, it is designed for a leak-tightness test and measurement of total pressure and main gaseous constituent partial pressure inside irradiated power reactor fuel rods of any type existing in Russia.

The internal gas composition inside the cladding of the irradiated fuel rod is multicomponent, but we are talking about the inspection of a quasibinary gaseous mixture, that is quite enough for minimization of unstable gas composition influence on the method accuracy. We consider a model of a binary mixture consisting of two stable constituents: filler gas - helium and a gas that is formed under nuclear fuel operation - FGP. The parameters to be measured are partial pressure of helium and partial pressure of FGP.

It is apparent, that the internal gas composition inside irradiated power reactor fuel rods is not constant (i.e. relative gas constituent quantitative content in gaseous mixture of the particular rod is unequal to the relative content of the other particular rod) due to several reasons. For example, first, technological scatter in parameters of initial gas filling (sometimes up to 20%). Second, scatter in fuel properties and its operational conditions. And third, possibility of leakage (primarily helium) due to a cladding defect. In this connection the elaboration of a device to measure the gas pressure inside irradiated fuel rods that does not need the stable gas composition of rod under study was undoubtedly an urgent problem.

The physical principle and the main measuring scheme that are the basis of the developed system have been described above (See Fig.1). As distinct from them, the system is calibrated using standard specimens of pressure and composition, filled with mixture of helium, xenon and krypton with accurately known filling parameters (composition and pressure) in the needed range. As a result of the calibration the graduated relationships were derived in the form of calibration surfaces, which were used to compute the searched for internal partial gas pressures.

Nowadays the data processing procedure have been improved and carried out by special mathematical method with using of the preliminary classification of observations. In essence the data processing procedure consists in deriving of the graduated relationships of searched for the rod under study parameters as a function of some its generalized features. The generalized features are obtained from the a priori set of all rod measuring features by the special kind functional nonlinear optimization procedure application. The graduated relationships and generalized features are calculated with the help of training set application. The so-called training set is the feature vector observations of the standard specimens.

The developed system is a personal Intel-compatible computer base apparatus-program complex, hardware in the form of electronic blocks and devices, and a pneumatic measuring manipulator with pressure probe. All operations relevant to measurement and data processing are accomplished in automated mode. The measuring manipulator can be installed in a fuel inspection rig for poolside inspection or independently in a hot cell. The main technical characteristics of the measuring system are given in Table I.

The system is capable of operation under two modes. The first one is designed for express grading fuel rods into three groups: leakers, those having pressure close to design one and those having pressure significantly higher than the design one (time of measurement is 1 minute). The second mode is intended for quantitative measurement of helium and FGP partial pressures (time of measurement is 15 minutes). The availability of two modes makes the inspection process more flexible and efficient in terms of time used.

The system has been checked experimentally under hot cell conditions at Research Institute of Atomic Reactors in Dimitrovgrad [15] on a series of individual irradiated fuel rods. The results obtained and the experience gained point to the reliable and highly accurate operation of the system. The first results of non-destructive examination of actual fuel rods have been received in 1992 for standard spent WWER-1000 fuel rods and are given below in Tables 2 and 3. The fuel characteristics and operational conditions were as follows: 3-year cycle; 3.6% enrichment; burn-up of 34.7 MWd/kgU.

TABLE I. MAIN TECHNICAL CHARACTERISTICS OF THE PRESSURE MEASURING SYSTEM

Technical characteristics	Value
Range of pressure measurement, MPa	0.10-5.0
Range of helium pressure measurement, Mpa	0.10-3.0
Error of total and helium pressure measurement, Mpa	(0.15
Range of FGP pressure measurement, MPa	0.10-2.0
Error of FGP pressure measurement, Mpa	(0.10
Total time of single fuel rod measurement, min.	15
Time per single fuel rod in sorting, min.	1
Environment	water, air
Temperature of environment, (C	15 – 60

TABLE II. RESULTS OF WWER-1000 FUEL ROD INSPECTION

NN	Fuel rod No.	Date	Time	He pressure, MPa	FGP pressure, MPa
1	224	07.08.92	09:55:22	2.19	0.09
2	224	07.08.92	10:14:58	2.14	0.06
3	291	07.08.92	15:24:57	2.17	0.02
4	291	07.08.92	15:48:54	2.11	0.03
5	291	07.08.92	16:06:14	2.11	0.03
6	60	10.08.92	12:47:32	2.17	0.01
7	60	10.08.92	13:03:52	2.18	0.02
8	60	10.08.92	13:21:51	2.25	0.01
9	161	10.08.92	14:01:27	2.30	0.00
10	161	10.08.92	14:22:37	2.31	0.02
11	159	10.08.92	14:46:40	2.45	0.07
12	159	10.08.92	15:12:53	2.43	0.05
13	104	10.08.92	15:59:19	2.38	0.10
14	138	10.08.92	16:21:28	2.15	0.06
15	138	10.08.92	17:21:26	2.20	0.06
16	192	11.08.92	10:46:14	2.20	0.02
17	192	11.08.92	11:03:17	2.10	0.03
18	157	11.08.92	11:25:24	2.41	0.01
19	157	11.08.92	11:44:09	2.42	0.02
20	157	11.08.92	12:05:59	2.50	0.01
21	34	11.08.92	12:38:18	2.45	0.08
22	224	12.08.92	09:21:41	2.26	0.04
23	224	12.08.92	09:43:08	2.18	0.05
24	224	12.08.92	10:02:12	2.19	0.05
25	224	12.08.92	10:30:26	2.19	0.04
26	224	12.08.92	10:47:12	2.24	0.04
27	224	12.08.92	11:37:22	2.22	0.04
28	224	12.08.92	12:52:21	2.29	0.04
29	224	12.08.92	13:07:40	2.24	0.11
30	104	12.08.92	13:32:04	2.24	0.06
31	104	12.08.92	13:48:17	2.31	0.07
32	104	12.08.92	14:10:06	2.31	0.04
33	104	12.08.92	14:28:16	2.29	0.08
34	104	12.08.92	14:43:44	2.29	0.04

TABLE III. RESULTS OF ROD RANDOM PUNCTURE TEST

Fuel rod No.	224	104
Free volume, cm^3	28.08	27.73
Pressure, MPa	2.396(2.27)[a]	2.400(2.37)[a]
He, %	97.81	97.88
N_2, %	0.11	0.09
O_2, %	0.02	0.02
Kr, %	0.22	0.22
Xe, %	1.84	1.78

[a] Middle value of NDT test

The comparison analysis of the above results of the random rod pressure measurement by puncture and developed non-destructive method with the help of the pressure measuring system shows their absolute agreement within the measurement error.

Further improvement of the thermophysical method and technical instruments has resulted in the measuring system variant development for leak-tightness test and internal gas partial pressure measuring of peripheral rods in RBMK type FA. The picture of the given system measuring manipulator is shown in Fig. 4.

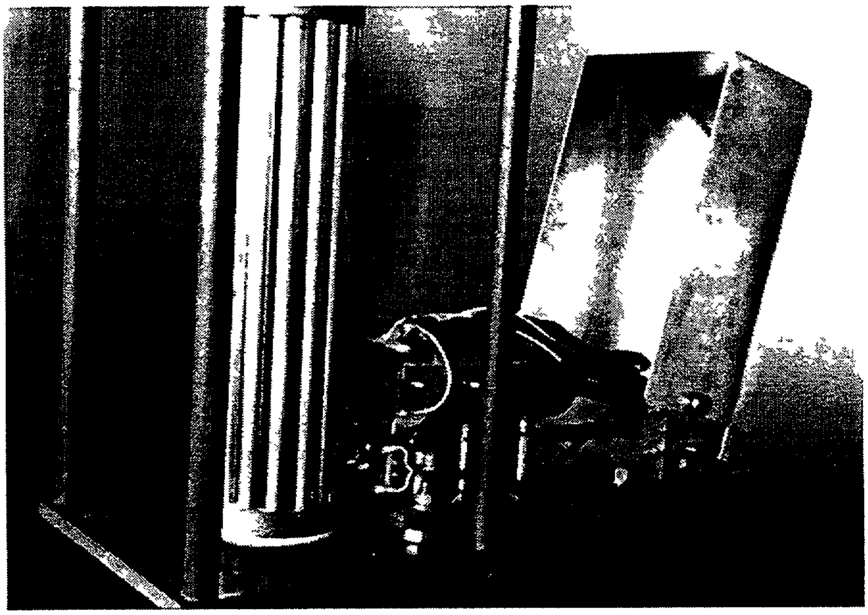

FIG. 4. Measuring manipulator for FA inspection.

5. CONCLUSION

A positive experience in development and operation of thermophysical instruments for NDT of internal gas pressure of Russian NPP rods in poolside and hot cell conditions [15] as well as in fuel production manufacturing conditions [16] have been gained to the present time. Many problems relevant to reliability, maintainability and convenience in operation for the equipment have been resolved. As a result the following advantages of the thermophysical method are revealed:

Extended range of its practical application, since neither rod afterheat level, nor rod plenum siting (top or bottom) influence on the quality of test results and the inspection of both single rods and FAs (peripheral rod line) is feasible

Universality in terms of rod inspection realization place (applicability both for poolside and hot cell conditions)

Extended information volume, since the information quality is comparable to the obtained only by rod puncture test (partial pressures of main gas constituents and gas mixture total pressure, rod leakage and tightness)

Reliable and authentic detection feasibility of every defective rod (rod with any probable defect type) because of realization of one (or more) of the following criteria under investigation:

- Low helium partial pressure (or its complete absence) in rod internal gas mixture,
- Abnormal low (or high) rod total internal gas pressure,
- Presence of water or steam inside rod.

Today the work is under way in the following directions:

- Improvement of features of the equipment now in action,
- Improvement of the method and technical instruments to inspect other fuel rod types (PWR type and others),
- Defective rod detection system development to inspect FA internal rod lines (without FA dismantling).

As to the third direction realization level, which among listed undoubtedly has the greatest potential practical importance and potentially the most beneficial, it should be noted, that we have already proved experimentally in laboratory poolside conditions the feasibility of the given development for Russian NPPs with the help of 1.5 mm thickness pressure probe variant and Russian type FA mock-up. By other words, the practical realizability of the given development does not cause any doubts for us.

The analysis of state of the art and the presented practical results of our work in the given area of science and engineering gives good prospects and the right to approve the expediency of wide application and further improvement of thermophysical method and technical instruments for inspection of standard, experimental as well as refabricated fuel rods and FAs in both poolside and hot cell conditions with a view to reconstruct defective FAs or investigate the individual rods during the realization of the program relevant to fuel perfection, increase of rod mean life and reliability of operation inside reactor core.

The described thermophysical method and technique have been experimentally checked during the examination of all types power reactor fuel rods existing in Russia (WWER, RBMK, BN) and undoubtedly could form the basis for new technique development for non-destructive examination of PWR (and other) type rods and FAs having gas plenum filled with spring or another elements of design.

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TELESCOPE SIPPING
THE OPTIMUM FUEL LEAK DETECTION SYSTEM

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Abstract

The TELESCOPE Sipping technology is an evolutionary development from previous ABB fuel leak systems used in LWR reactors. The system utilizes the existing dynamics that cause numerous fission products to leak from a failed fuel rod when the fuel assembly is raised from a reactor core during core fuel alterations. The system can also be used by repair work in pool side inspection in order to detect leaking rods or to verify reconstituted assemblies as non leakers.

1. SYSTEM PROPERTIES

- The assemblies are tested during the time they are lifted from a shutdown core when the leakage of fission products from a rod with damaged cladding is at maximum.
- By a total off-load of the fuel, the sipping is carried out simultaneously with the off-loading, without loss of time.
- On-line gas detection provides an immediate result as an integrated activity signal relative to the bundel test time. No later laboratory water or gas analyses are needed, but they may be run for additional information.
- Such factors as low contamination risk and high stability in the on-line background detection signal create an ideal situation for detection a leaking assembly with a high degree of sensitivity.
- By using a sipping procedure at pool side repair work of leaking fuel a more significant leak or non leak information is given as by using i e EC detection.

2. SYSTEM FUNCTION

Elevating the fuel assemblies forces fission products out of a damaged fuel rod. These accumulate in the water inside the fuel assembly channel (BWR) or in the refueling machine mast (PWR).

Water samples are drawn on-line from inside the mast and are degassed in the process system. The gas is detected in a gas detection system connected to the degassing unit.

Owing to a patented system for the apparatus and to the function of the gas detection circuit, whereby a small gas volume inside the loop is isolated from the ambient air, no dilution of the fission gases can occur. The air that is placed in the circuit at the start of sipping is forced out and is replaced by inactive gases and fission gases which are separated from the test water. The advantage here is that the content of fission gases in comparison to the inactive gases in the circuit is constant when non-leaking fuel assemblies are tested. Because of the constant background in the on-line gas detection loop, leaking assemblies are easily detected with a high degree of sensitivity.

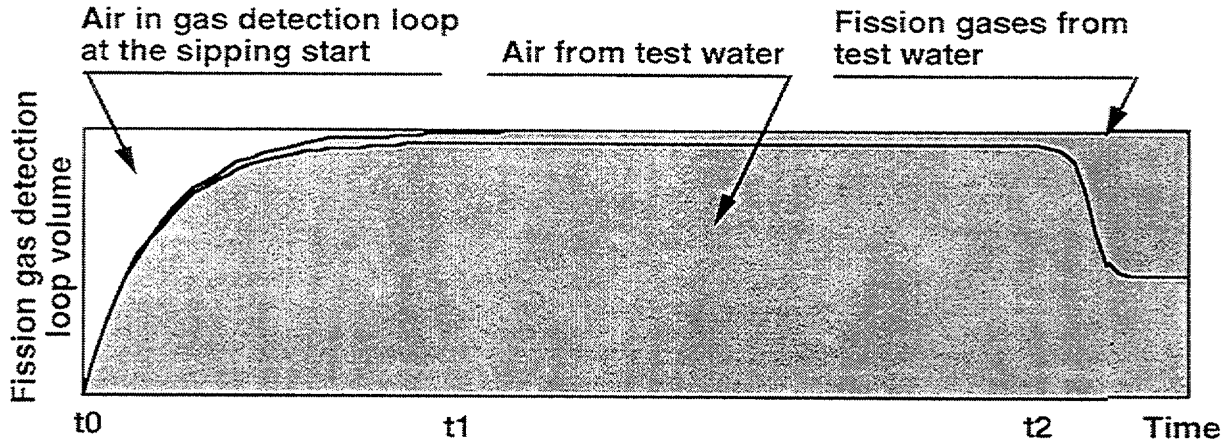

FIG. 1: Proportion of fission gases and non-active gases in the gas detection loop

The gas process in the gas detection loop is shown in figure 1. Non-leaking background assemblies are tested between t_1 and t_2. Here, a leaking assembly is detected at t_2. After cleaning the water and gas loop, the test can be restarted at t_0.

Detector signal registration and analysis are shown in figure 2. The registered signal can be analysed as a function of the testing time with respect to Δ cps, area, and regression by storing the signal during the test cycle.

3. COMPLEMENTARY USE OF THE TELESCOPE SIPPING SYSTEM

The TELESCOPE Sipping equipment can preferably be used in combination with pool side work of leaking fuel. The equipment can be used for single rod leak detection by a repair work, and for verification of an assembly as being a non leaker after a finished repair.

4. OTHER SIPPING SYSTEMS

The advantages and drawbacks of conventional sipping systems in comparison to the TELESCOPE Sipping system are stated below.

4.1. IN-CORE sipping (BWR)

General IN-CORE Sipping can be performed by using a hood to collect water samples from fuel assemblies. Hoods have been developed to test from 1 to 16 fuel assemblies at a time. The hood is placed over the upper end of the fuel assemblies. A compressed air cushion in the hood is usually applied over the upper part of the fuel channels to stop the flow of cooling water.

Fission products in a damaged fuel rod are forced out due to the rise in temperature.

Water samples are drawn from each fuel assembly. The radio activity of the water samples are spectral analysed in the plant laboratory.

The ABB Atom IN-CORE Sipping system uses not only the effect of increased temperature to force out fission products from a leaking fuel rod, it also uses the effect achieved by elevating each fuel assembly to a height of 0,5 meters. Elevation brings about a fission product outflow effect equivalent to 10-15° C.

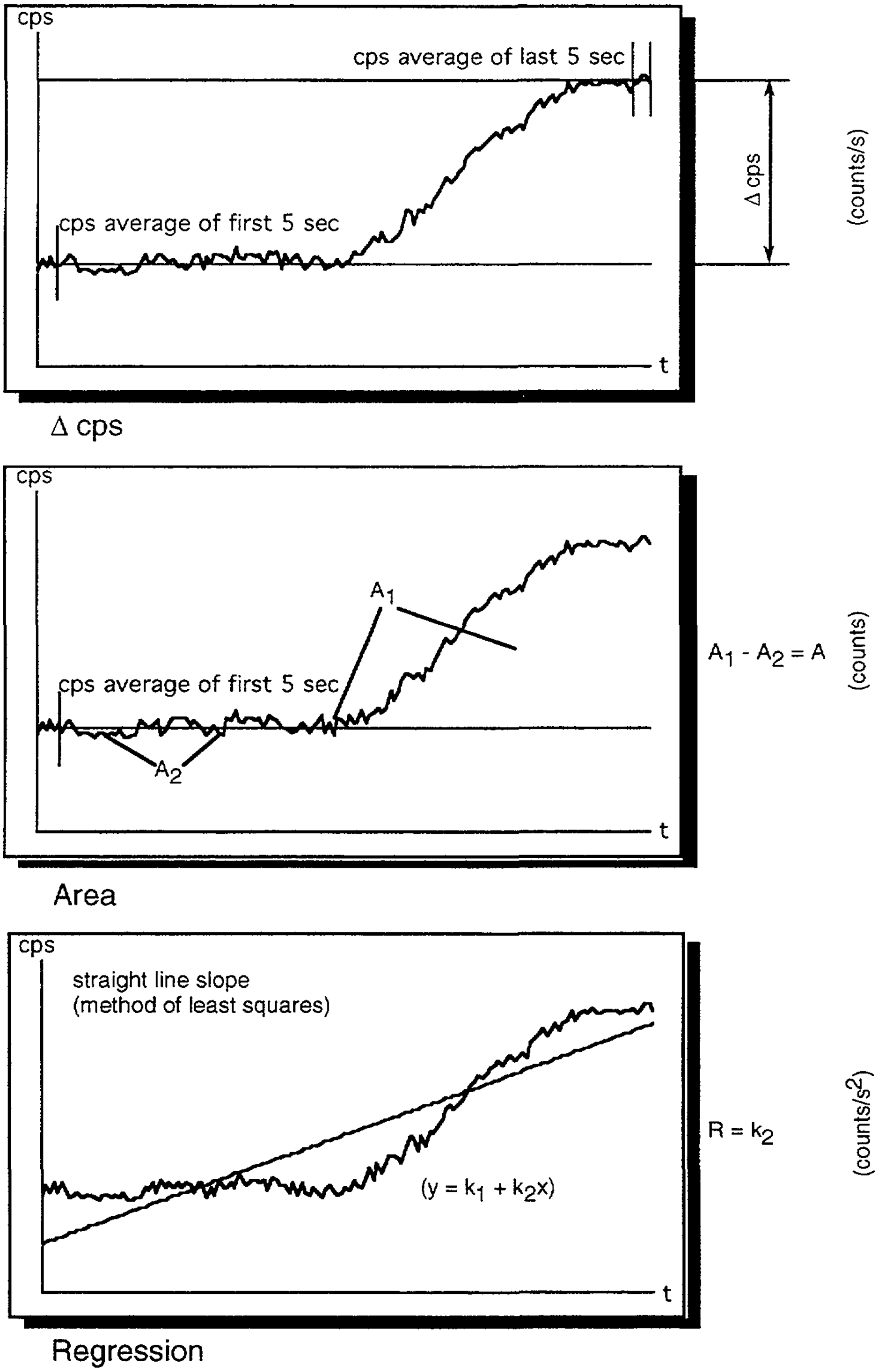

FIG. 2. Detector signal evaluation

The system also has a unique patented mammoth water pumping system that on-line degasses the water. Water and gas samples are taken simultaneously for subsequent spectral analysis in the laboratory.

4.2. ELEVATE sipping (BWR)

The system is an ABB Atom Sipping system used in a fuel service rig in a spent fuel pool. Placing a hood over a single assembly and raising the assembly to a height of four meters, results in a fission product outflow effect equal to about 150° C.

The system is used in combination with "Flux Tiltings" and ELEVATE Sipping of only the suspected assembly and some of the assemblies adjacent to the suspected one in the core.

The detection sensitivity is more than 10 times higher than the IN-CORE Sipping system. Drawbacks of the system are the costs in connection with the "Flux Tilting" and the risk of creating additional fuel failures by tilting.

4.3. CANISTER sipping (BWR, PWR)

In BWRs as well as PWRs, fuel leakage detection is performed in canisters in the spent fuel pools.

The fission products are forced out of the damaged fuel rods by raising the temperature and/or reducing the pressure in the canisters. A temperature increase is created by residual fuel power and sometimes by extra heating arrangements. A pressure reduction is created by reducing the pressure in an air cushion in the canister top, mostly down to a pressure of about 0,4 atm.

In some systems water samples are taken from the canister and are analysed in the plant laboratory. In plants using the "vacuum" system, on-line detection of the gases is performed. Fission products are significantly diluted in the canister water and in the gas detection circuit.

4.4. IN-MAST sipping (PWR)

When a PWR fuel assembly is transported in the reactor or the spent fuel pool, it is always lifted into a tube-like mast. That makes it possible to detect fission gases entering the upper air section of the mast. In order to increase the fission gas concentration in that air, a gas pumping circuit is established where the air is bubbling through the water surrounding the fuel in the mast. Sometimes compressed air is used to separate the fission gases from the mast water. A fission gas detection loop is connected to the circuit and on-line gas detection is carried out.

The fission gases are highly diluted in the process air and a high variation in background values is inevitable.

Water samples cannot be taken and analysed by the system.

5. SIPPING SYSTEM COMPARISON

In figure 3, BWR, and figure 4, PWR, the above-mentioned advantages and drawbacks are quantified as sipping efficiency factors. In the BWR comparison, IN-CORE Sipping is set as a base reference and, in the PWR, the CANISTER General System is chosen as the base reference. Figures 5 and 6 present the efficiency factors in the form of a staple diagram.

	Influence factor	Laboratory analysing factor for	On-line detection factor for	Sipping efficiency factor for		Sipping time/ assembly
		Water	Gas	Water	Gas	Min
INCORE General System	1	1	1	1	1	4
INCORE ABB Atom System	2 [1]	1	2 [2]	2	4	4
ELEVATE	[3] [4] 0,5 x 15	1	[2] 2	7,5	15	10-15
CANISTER Vacuum	[3] [5] 0,5 x 22,5	[6] 0,5	[7] 0,5	5,6	5,6	10-15
TELESCOPE Incore	11,25 [8]	1	2 [9]	11,25	22	3
TELESCOPE Off load	[10] [11] 0,5 x 45	1	[9] 2	22,5	45	0

FIG. 3. BWR sipping efficiency factors

1) Due to elevation 0.5 m
2) Due to effectiveness and low dilution in on-line gas separation with mammoth pumping
3) Due to loss of fission products during transportation to spent fuel pool
4) Due to elevation 4 m
5) Due to an underpressure of 0.4 atm
6) Due to dilution in the canister water
7) Due to dilution in the gas separation tank
8) Due to elevation 3 m
9) Due to detection sensitivity and stable background
10) Due to loss of fission products by water exchange elevation 12 m

	Influence factor	Laboratory analysing factor for	On-line detecting factor for	Sipping efficiency factor for		Sipping time/ assembly
		Water	Gas	Water	Gas	Min
CANISTER General System	1 [1]	1	1	1	1 [2]	10-15
INMAST Off load	2 [3]	-	0,5 [4]	-	1	10-15
TELESCOPE Off load	[5] [6] 0,5 x 45	[7] 0,5	[8] 2	11,25	45	10-15

FIG. 4. PWR sipping efficiency factors

1) Canister with an underpressure of 0.4 atm
2) Canister sipping with degassing tank
3) Due to first lift from core
4) Due to dilution in the gas volume of the mast
5) Due to loss of fission products by water exchange
6) Due to elevation 12 m
7) Due to large water volume in the telescope compared to the canister
8) Due to detection sensitivity

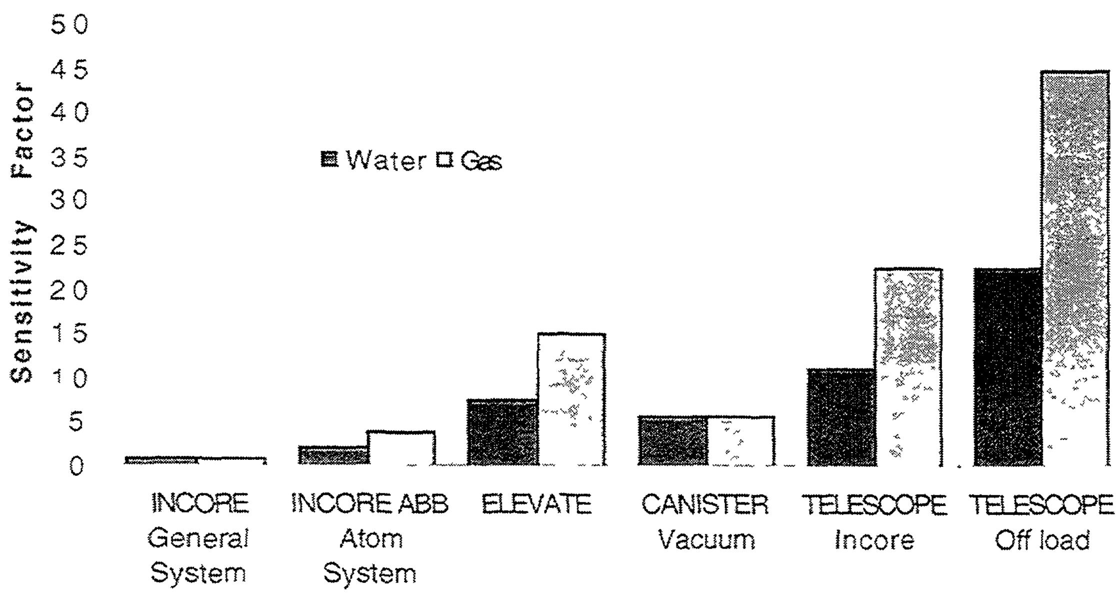

FIG. 5. BWR fuel sipping system comparison

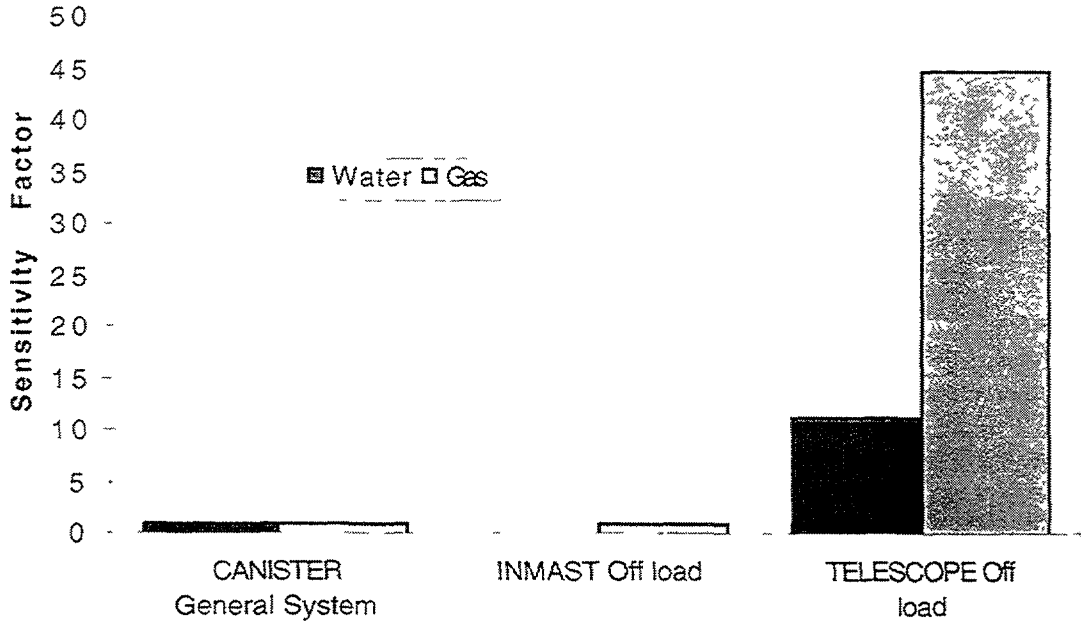

FIG. 6. PWR fuel sipping system comparison

The factors are set in rough values in accordance with the theoretical differences between the respective systems and in accordance with empirical l ~t.~

It has to be said that wide variations in results always exist in conjunction with practical sipping performances due to various facts, such as type of rod failure and the time between the shutdown and the performed test.

The conclusion to be drawn from the comparison is that TELESCOPE Sipping is an outstanding fuel leak detection system.

EXPERIENCE AND NEEDS
(Session 2)

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FRAMATOME EXPERIENCE IN FUEL ASSEMBLY REPAIR AND RECONSTITUTION

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Abstract

Since 1985, FRAMATOME has built up extensive experience in the poolside replacement of fuel rods (For repair or R&D purposes and the reconstitution of fuel assemblies (i.e. replacement of a damaged structure to enable reuse of the fuel rod bundle).

This experience feedback enables FRAMATOME to improve in steps the technical process and the equipment used for the above operations in order to enhance their performance in terms of setup, flexibility, operating time and safety. In parallel, the fuel assembly and fuel rod designs have been modified to meet the same goals.

The paper will describe:

- the overall experience of FRAMATOME with UO_2 fuel as well as MOX fuel; the usual technical process used for fuel replacement and the corresponding equipment set;
- the usual technical process for fuel assembly reconstitution and the corresponding equipment set. This process is rather unique since it takes profit of the specific FRAMATOME fuel assembly design with removable top and bottom nozzles, so that fuel rods insertion by pulling through in the new structure is similar to what is done in the manufacturing plant.
- the usual inspections done on the fuel rods and/or the fuel assembly,
- the design of the new reconstitution equipment (STAR) compared with the previous one as well as their comparative performance.

The final section will be a description of the alternative reconstitution process and equipment used by FRAMATOME in reactors in which the process cannot be used for several reasons such as compatibility or administrative authorization. This process involves the pushing of fuel rods into the new structure, requiring further precautions.

1. INTRODUCTION.

As of September 1997, the total number of irradiated FRAMATOME fuel assemblies is more than 40,000, mainly 17x17 fuel assemblies (97%), in nearly 80 different plants reactors designed by FRAMATOME or other N.S.S.S supplier.

Some of these assemblies have been damaged during their irradiation lifetime. This damage takes the following forms:

- Leaking fuel rods, mainly due to debris in reactor coolant;
- Skeleton damage (grids), mainly due to handling incidents.

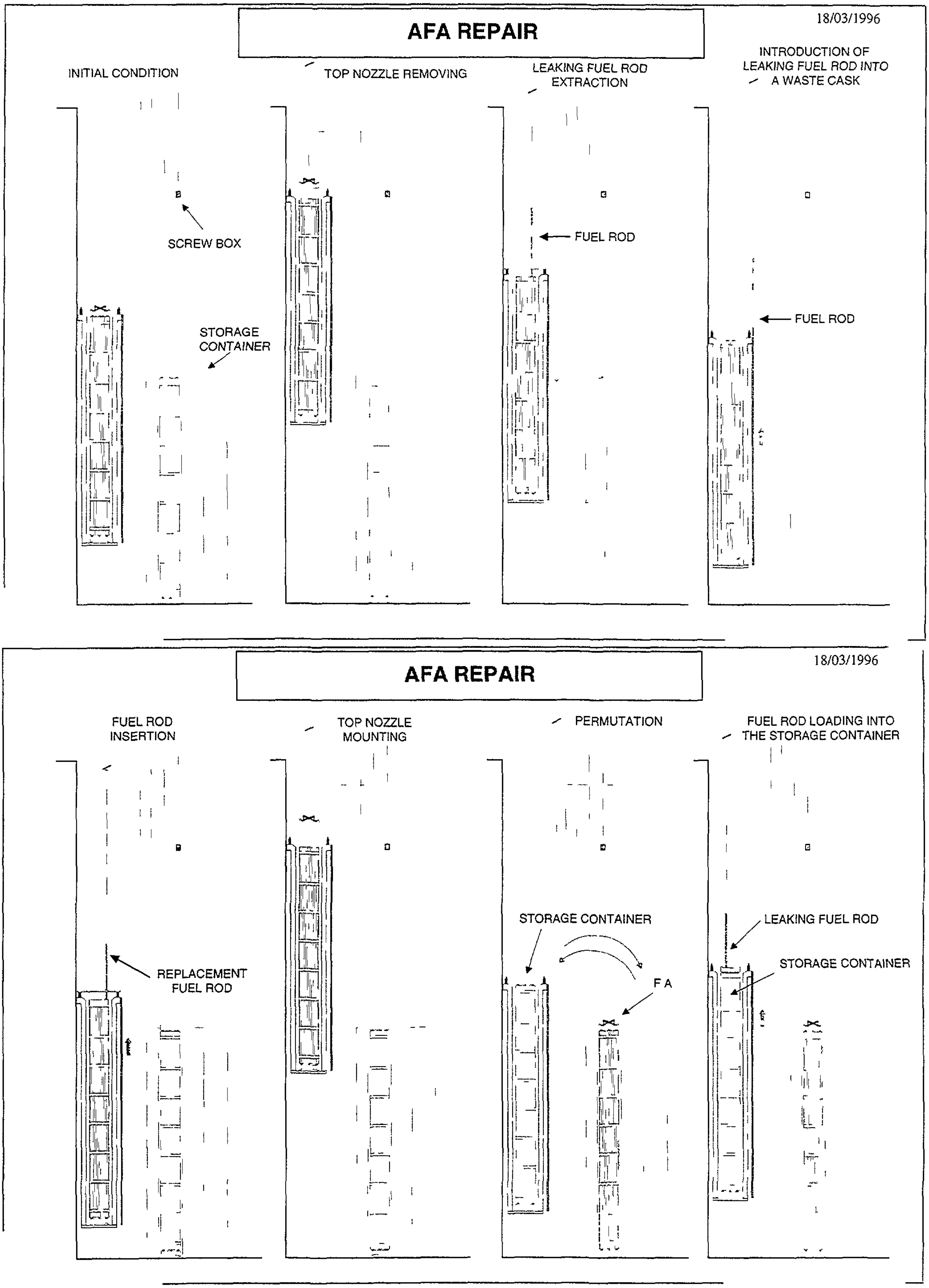

FIG. 1. AFA repair process - principle

In order to allow the reuse of some of the damaged fuel assemblies which have not reached the end of their service life, FRAMATOME has developed since 1985 several sets of equipment for fuel rod and skeleton replacement.

These sets of equipment have been developed and are operated on the basis of the AFA design, which allows the dismantling of the top as well as the bottom nozzle, and fuel rod handling from both the top and bottom of the fuel rods.

These two AFA features allow different kinds of repair, depending on the severity of the damage.

2. AFA[1] REPAIR TOOLS

This is a simple, easy-to-use set of tools, which has been designed for the replacement of fuel rods in fuel assemblies.

These tools can replace one or several fuel rods after removal of the top nozzle. The main feature of this repair process is the use of the fresh fuel elevator to support the fuel assembly during the repair operations.

The principle of the repair operations is described in figure 1 below. The fuel rod gripper remains in fixed position, while the elevator moves the fuel assembly up and down during rod extraction and insertion. During this operation, the whole fuel rod is surrounded by a protecting sleeve in order to prevent any external damage.

To increase the experience feedback on leakage causes, the equipment includes a set of tools for on site examinations by the visual (underwater cameras) and eddy current techniques.

Photograph 1 below gives an overview of the complete set of tools installed in a fuel pool. The main components of this tooling are:

- The fuel rod gripper and its handling device.
- A set of tools for fuel assembly nozzle setup and removal: screwdriver, expansion tool, nozzle handling tool, guide plate for proper positioning of screwdriver and/or expansion tool.

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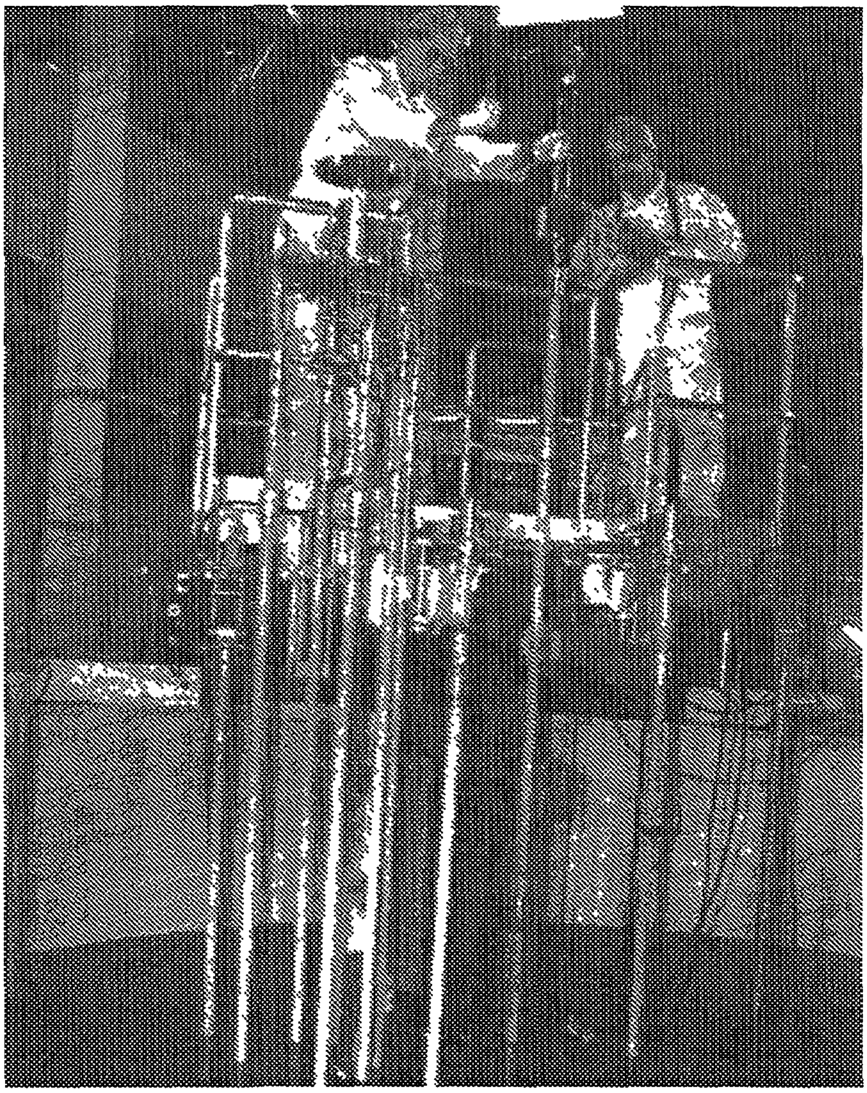
[1] AFA= Advanced Fuel Assembly (With removable bottom and top nozzles).



PHOTOGRAPH 1. AFA repair tools

Due to its very simple design, the AFA repair equipment can be operated in a very short time: it takes less than 60 hours to perform the repair of one fuel assembly, including setup and removal of the equipment. This performance allows fuel assembly repair during outages. As of September 1997, FRAMATOME has replaced about 400 fuel rods in 130 fuel assemblies.

3. FUEL ASSEMBLY RECONSTITUTION: STAR STATION

For skeleton replacement on fuel assemblies, FRAMATOME has developed a process in which the fuel rods are pulled from the damaged skeleton and transferred directly to a new one. As shown in figure 2 below, the two skeletons are stacked in such a way that the corresponding rod location in each skeleton is perfectly aligned.

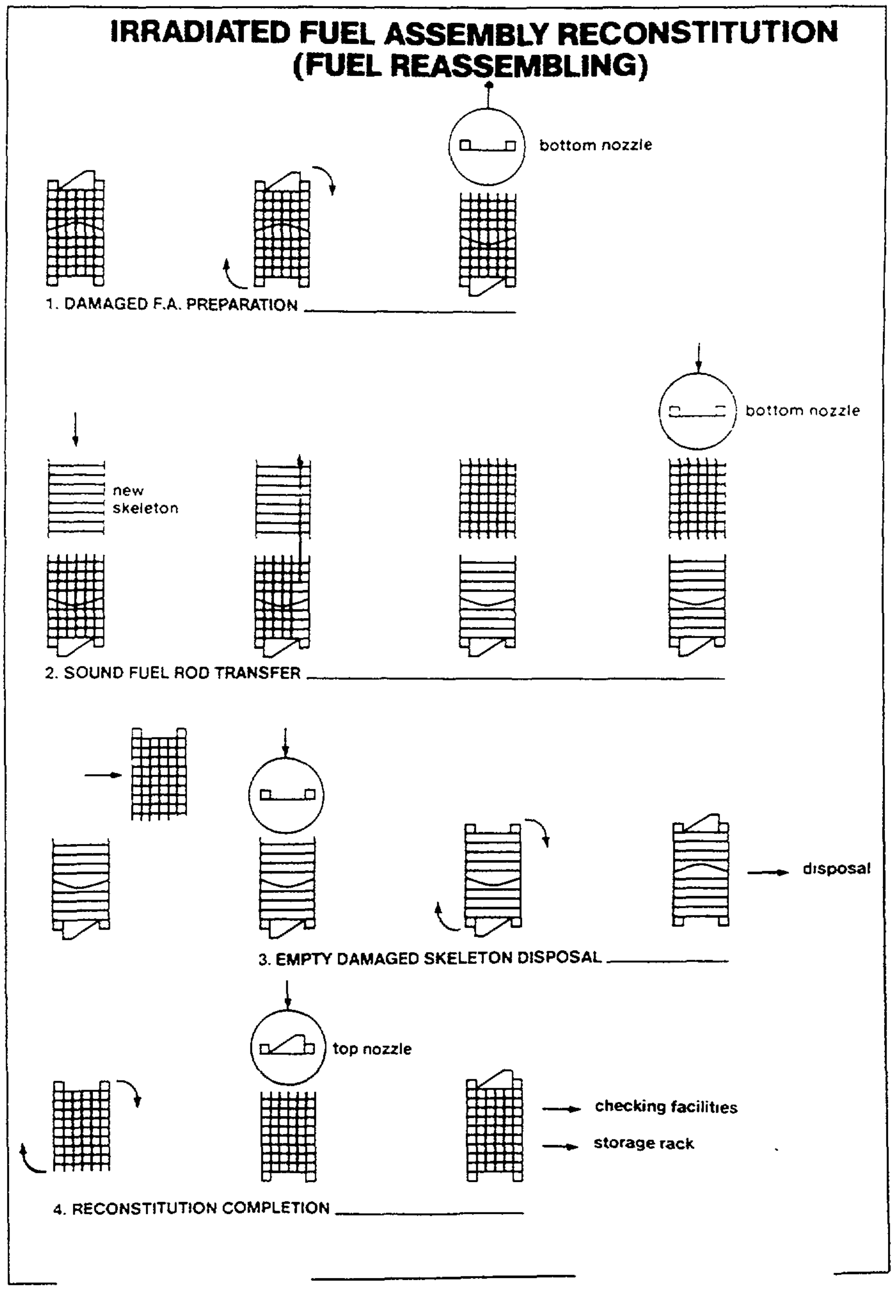


FIG. 2. Reconstitution process

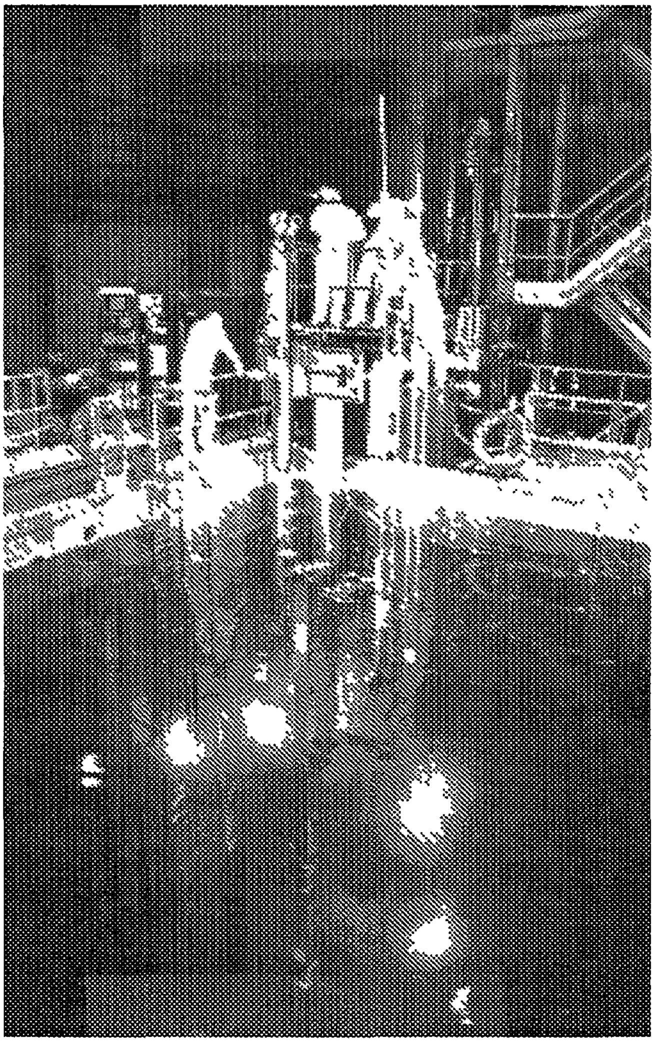
The main advantages of this method are as follows:

- The perfect alignment of the two skeletons prevents any possibility of location error during the transfer of the fuel rod. This feature is particularly important in case of heterogeneous rod bundles such as MOX or fuel assemblies with burnable poisons rods.

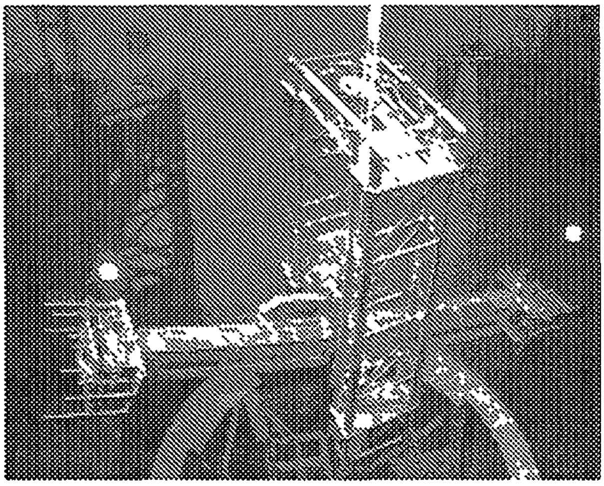
- Insertion of the fuel rod by pulling reduces significantly the risk of interference between the rod and the new skeleton. Furthermore, the rod never undergoes compressive loads, which prevents any risk of rod damage by buckling.

- During the transfer, the rod is always held in a minimum number of grids of both skeletons, and no horizontal movement of the gripper is needed, so that the risk of accidental drop of the fuel rod during its handling is ruled out.

Photographs 2 and 3 show the equipment used for fuel assembly reconstitution. This equipment, called STAR, consists of:



PHOTOGRAPH 2. Star station



PHOTOGRAPH 3. Basket inverter

- two baskets for accommodating the damaged fuel assembly and the new skeleton, and providing their proper alignment during the operations,

- a fuel rod transfer tool including a motor-driven gripper for rod pulling. This gripper is positioned on the fuel rod by the mean of a XY table for the upper part of the gripper, and a centering plate for the lower part. The XY table is manually operated.

- a basket inverter, A two level bracket used to support the baskets and the rod transfer tool. A set of tools for fuel assembly nozzle setup and removal similar to those used for AFA repair (See para 2).

A complete set of safety features have been implemented. In particular, the transfer sequence of the fuel rod is completely automatic and computer controlled, and such parameters as insertion forces, insertion speeds, altitudes and gripper status are constantly monitored. Part of this equipment can also be used for repair, as an alternative method to the AFA repair, for example in case of plants where a fuel elevator is not available.

As of September 1997, FRAMATOME has successfully reconstituted 60 fuel assemblies, corresponding to the transfer of nearly 16,000 fuel rods.

The STAR station, which is the last generation of reconstitution tools, has been in operation since the beginning of 1997. This equipment is the result of significant improvements made on the previous generation of tools (OSIC station), providing major performance enhancements, especially in terms of flexibility and setup speed, as shown in table 1 below:

TABLE I. FUEL ASSEMBLY RECONSTITUTION PERFORMANCE

	OSIC	STAR
Weight	30 t	9 t
Volume	200 m^3	50 m^3
Setup time	200 h	80 h
Reconstitution time (1 FA)	64 h	64 h
Removal time	160 h	60 h

Additional improvements are planned on STAR to reduce the reconstitution time per fuel assembly down to 48 H in the near future.

4. FUEL ASSEMBLY RECONSTITUTION: ALTERNATIVE METHOD.

In spite of the flexibility of the STAR equipment, there are some plants where this equipment cannot be easily operated. This is the case for the plants with (16x16 and 18x18 fuel assemblies), where a different reconstitution process (fuel rod insertion by pushing) and the related tools have already been qualified by safety authorities in the post.

For this reason, FRAMATOME has developed the tools for this alternative reconstitution process, consisting in inserting the fuel rod in the new skeleton by pushing.

Figure 3 below gives the functional description of this process and the associated tools. The damaged fuel assembly and the new skeleton are installed side by side directly in the storage racks. A fuel rod gripper tool, similar to the STAR gripper, is successively positioned on each fuel rod location by means of two XY tables. the movements of which are electrically synchronized.

Like STAR, the fuel rod gripper is fully automated, and the main parameters (extraction/insertion forces, speeds, altitudes and gripper status) are monitored. In addition, due to the larger work distance (10 m underwater) required by this process, the XY tables are also computer controlled, so that no manual operation is needed during the rod transfer. In this way, maximum efficiency and safety can be preserved.

Presently, this new tool is undergoing qualification tests, and its first use on a German plant (UNTERWESER) is scheduled for the beginning of 1998. Photograph 4 below gives an overview of this new transfer tool.

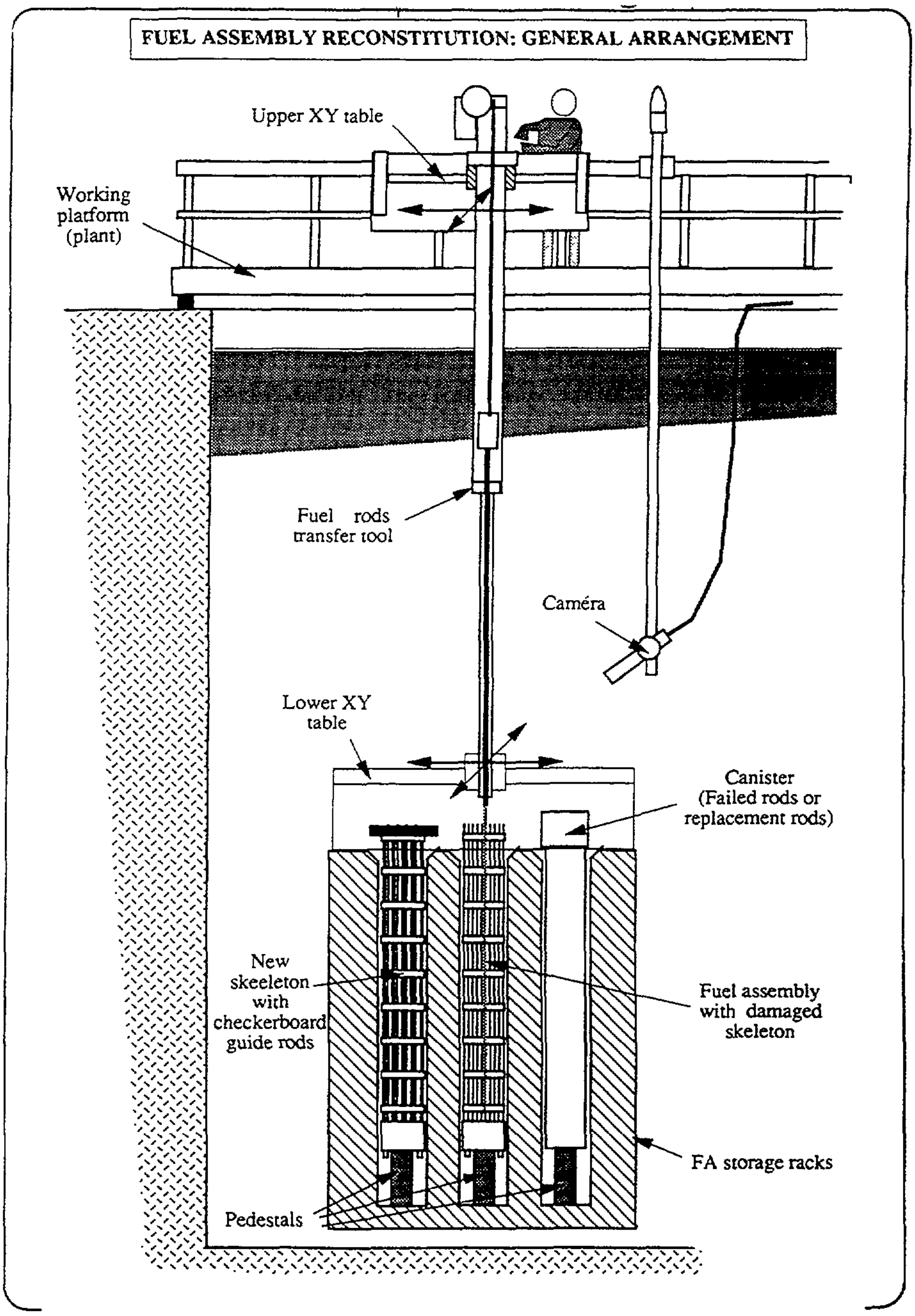
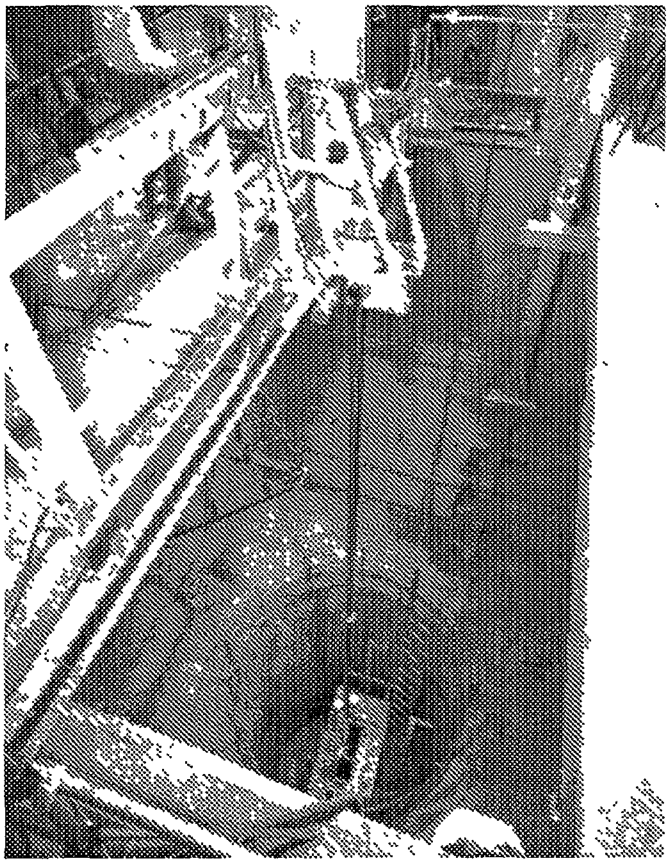


FIG. 3. 16x16 and 18x18 fuel assembly reconstitution



PHOTOGRAPH 4. 16x16 and 18x18 fuel rod transfer tool

5. CONCLUSION

FRAMATOME has built up extensive experience in the replacement of fuel rods and skeletons on irradiated fuel assemblies.

Since 1985, about 110 campaigns have been run, involving the repair of fuel assemblies with more than 400 fuel rod replacements and 60 skeleton replacements.

Moreover, to increase experience feedback, FRAMATOME takes advantage of repair equipment to pull specific demo fuel rods for on site or in hot cell examination (60 R & D campaigns).

Thanks to these sets of equipment, FRAGEMA has the means to both provide repair services to its customers and to accumulate experience feedback experience on its fuel assembly designs.

POOLSIDE INSPECTION FACILITY FOR PWR FUEL ASSEMBLIES

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Abstract

Pool side inspection programme for LWRs started in India with the inspection of BWR fuel assemblies at Tarapur and this involved sipping, visual inspection, UT and Eddy current testing. In view of the possibility of having VVER type of reactors in our country, a R&D program has been initiated for study of behavior of these type of fuel. The program would involve irradiation, pool side inspection and hot cell examination of specially designed fuel assemblies. Well characterized fuel assemblies irradiated in research reactor are transferred to the fuel pool with the help of fuel transfer system. The fuel assemblies are taken out of the transfer system, sipping test performed and de channeled using under water handling and cutting tools. The fuel pins are then taken out of assembly and loaded on to the stand for underwater UT and Eddy current testing. The details of the handling and inspection facilities provided in pool for inspection of the hexagonal fuel assemblies has been discussed in the text. Dismantling and inspection procedure used for control assembly pins have also been discussed.

1. INTRODUCTION

The pool side inspection program would involve irradiation, pool side inspection and hot cell examination of specially designed fuel assemblies. The fuel assemblies irradiated in research reactors are transferred to the fuel storage pool using a flask. The flask is dismantled using under water handling tools. Using a fuel grappler the fuel bundles are handled remotely and taken for inspection. A schematic is shown in Fig 1.

The pool side inspection program for PWR fuel in India will have the following facilities.

1. Sipping facility to detect failed fuel assemblies.
2. Visual/dimensional inspection test rig.
3. Fuel assembly de channeling/cutting facility.
4. Eddy current testing facility.
5. Ultrasonic testing facility.

Each of these facilities is explained in the following paragraphs.

2. SIPPING FACILITY

Gross fuel failures can be detected by wet sipping and the water samples are analyzed for Iodine and Caesium levels using a Gamma ray scanning chamber. The fuel assembly is separated from bulk of the pool water by transferring it in to a canister using the fuel grappler.

The canister (Fig. 2) is a 400 mm dia double walled SS vessel about 3.75 Mts. long mounted vertically on a stand. The top and bottom portions of the vessel are connected by a pipe line, pneumatic valves and a hermetically sealed motor pump set for mixing the water inside the canister. It has also got a heating coil at the bottom for heating the water in it.

After required parameters are reached (for ex. temp of water inside the canister, flow rate etc.) water samples are cooled and passed through a Gamma ray scanning chamber. The gamma ray scanning chamber consists of a lead shielding for water samples, a detector unit, a counting device and a channel analyser. The gamma activity of the sample is measured using high performance Germanium detector. The results are compared with the back ground levels. Depending up on the Energy peaks obtained on the CRT monitor the isotopes are identified.

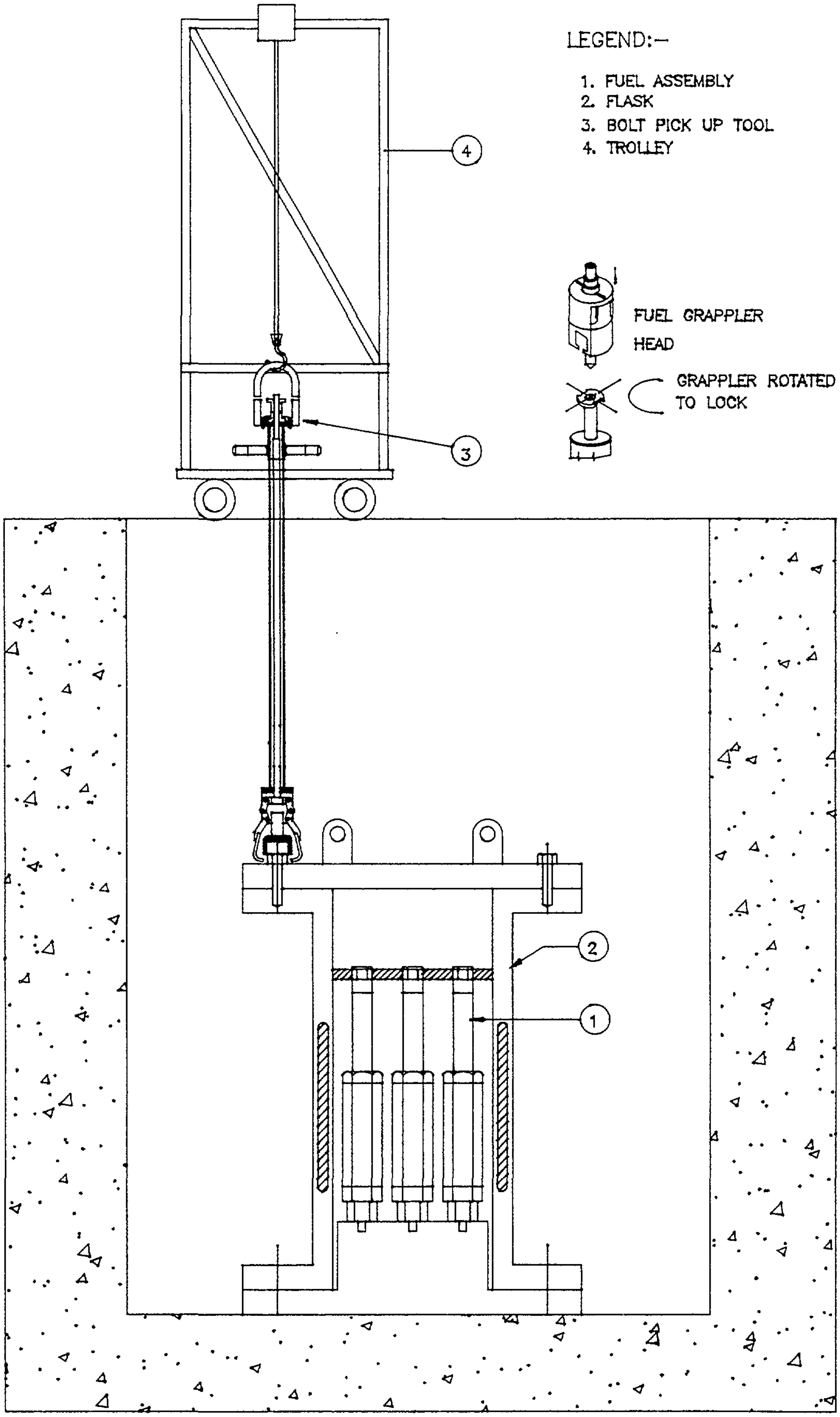


FIG. 1. Fuel transfer flask inside the pool

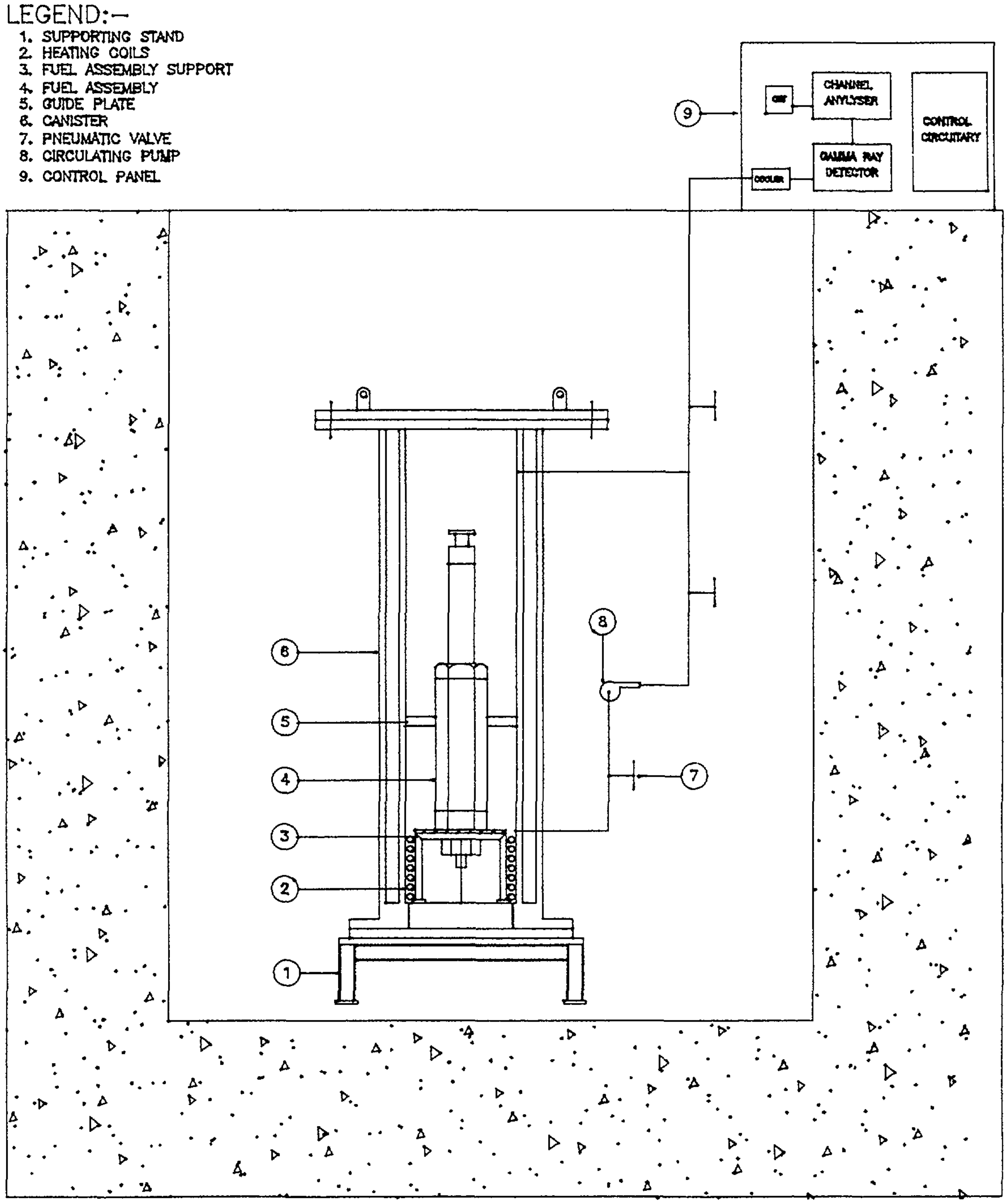


FIG. 2. Under water sipping facility

3. VISUAL/DIMENSIONAL INSPECTION TEST RIG

In this facility overall condition of the fuel assembly like shape, length, surface condition of the fuel assembly/pins can be inspected. The fuel assembly is inspected before and after de channelling using this facility. This facility consists of a specially designed fixture on which fuel assembly can be guided and rested on a rotary supporting platform. Visual inspection of the fuel assembly is carried out using a flexible video image scope arrangement as shown in the Fig[3].

The video image scope consists of a light guide's fiber bundle for illumination, an objective lens and a CCD (Charge Coupled device) at the distal end. The image of the object is converted in to electronic signals by the CCD based on the intensity of reflected light from the object. This signal is fed to a Camera Control Unit that processes and outputs them as video signal and displayed on a TV monitor. The depth of field and angle of vision can be changed using proper optical adopters at the inspection end of the scope. Additional illumination can be obtained by under water lights inside the pool if required. The images can be stored permanently for future references if required.

The length of the fuel assembly can be measured by rotating a screwed shaft that moves the pointer on a calibrated scale between the reference points on the fuel assembly. The pointer moves up and down on accurately machined guides by the screw shaft whose rotational motion is calibrated using an electronic encoder. The pointer can be moved across the length of fuel assembly by a separate motor. The pointer motion is controlled from the control panel thus giving the measurement accurately. The video image scope viewing head is used to move the pointer to the required position.

4. FUEL ASSEMBLY DE CHANNELING/CUTTING FACILITY

The fuel assembly is de channelled (removal of outer hexagonal channel) for surface examination of fuel pins. The control rods along with its spider are dismantled by removing the lock nut and the total control cluster is kept on a storage rack, before proceeding to the cutting of outer hexagonal channel. In some of the variations of fuel assembly it is possible to de channel the assembly without cutting operation. In this case the channel locking fasteners are spark eroded and the channel is lifted up using a channel handling tool. But in the present design, the fuel assembly is de channelled by cutting the hexagonal shroud by slitting saw. Water hydraulic actuating systems are used as all the operations are to be carried out under water.

The outer hexagonal channel of the fuel assembly is removed by slitting the channel at pre determined locations (based on fuel assembly design). This facility consists of fuel assembly gripping table, hydraulically operated slitting saw enclosed in a chamber. The cutting operation can be seen from the top of the chamber Water is circulated through the chamber to collect cutting chips and debris present if any using a filter that can be replaced after decontamination.

The fuel assembly is held horizontally by hydraulic gripping heads of the gripping table. An air motor drives the slitting saw and can be moved on guide ways along the length of the fuel assembly. Each face of the hexagonal channel is cut by successive indexing of gripping heads. Top and bottom portions are removed from the chamber for reuse after decontamination and visual inspection.

The active fuel portion of the fuel assembly is separated from the hexagonal channel and taken for visual and dimensional inspection using under water handling tools. After visual inspection the active fuel portion is transferred to reconstitution facility for removing individual fuel pins to carry out pin level examination. A schematic diagram is shown in Fig [4].

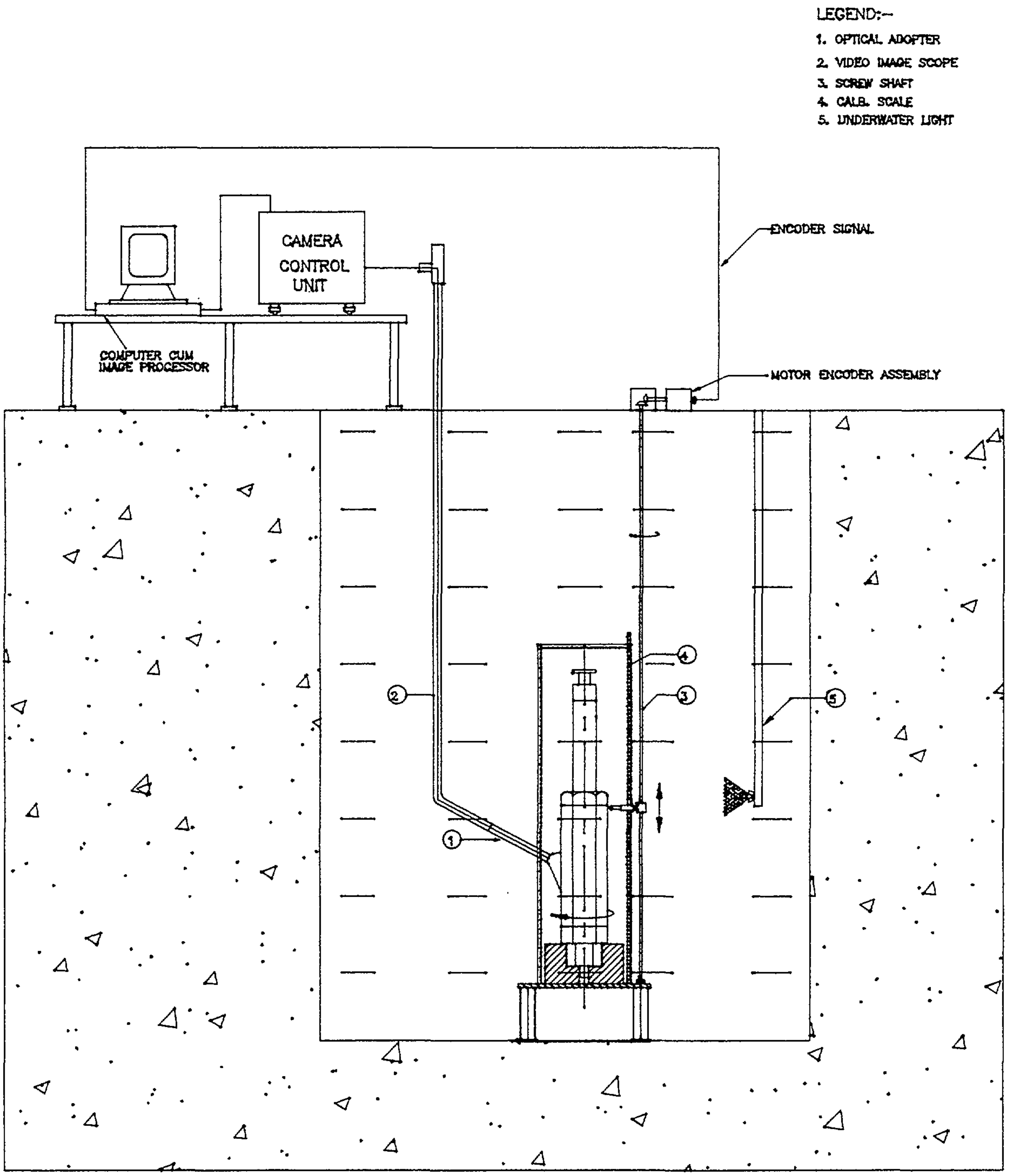
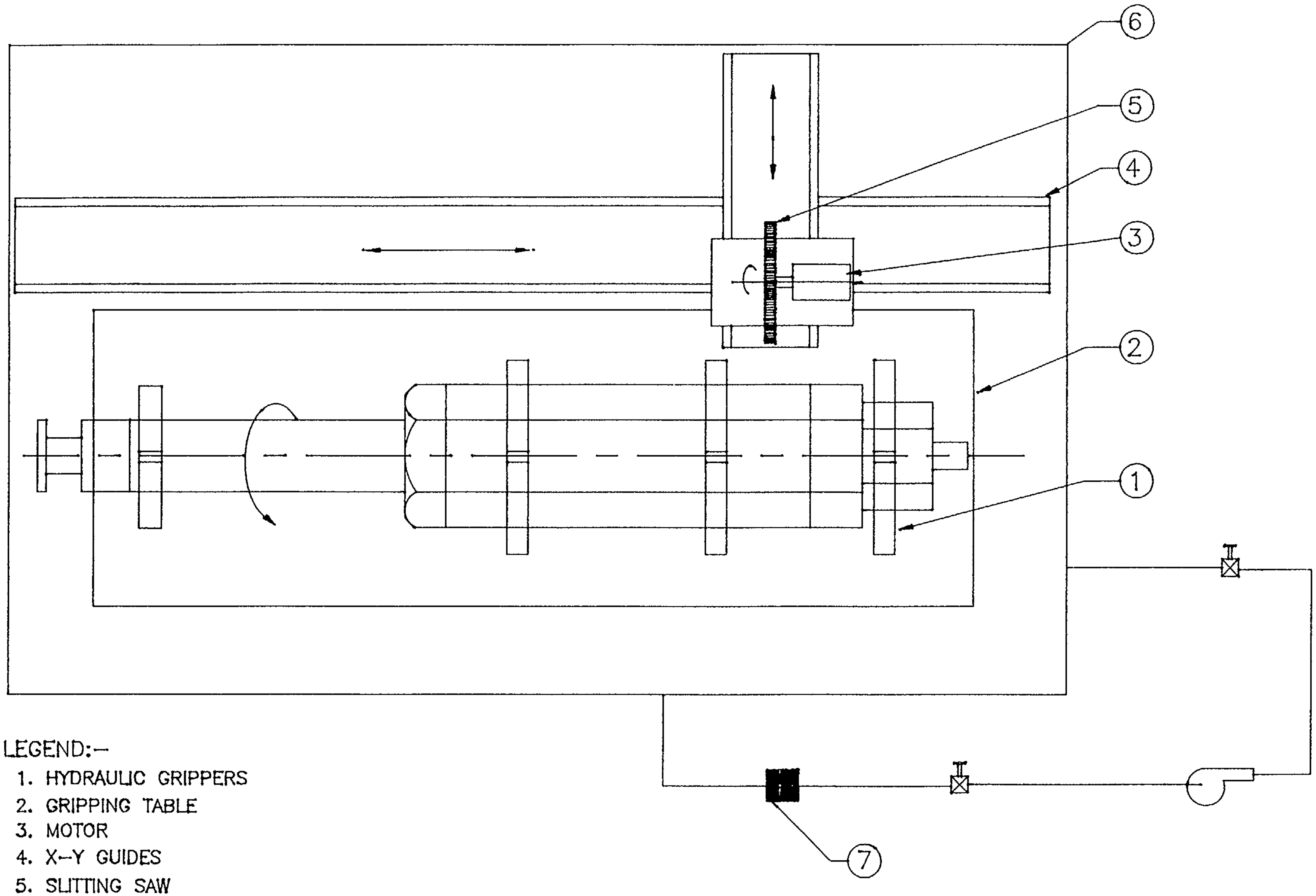


FIG. 3. Visual and dimensional test rig



LEGEND:—
1. HYDRAULIC GRIPPERS
2. GRIPPING TABLE
3. MOTOR
4. X—Y GUIDES
5. SLITTING SAW
6. CUTTING CHAMBER
7. FILTER

FIG. 4. Fuel assembly dechanneling/cutting facility

5. EDDY CURRENT TESTING FACILITY

In this facility fuel pins and control rods are tested for discontinuities like cracks, wear of clad tube, hydride formation, hardness of clad tube etc.

This facility consists of a stand to support fuel pin/control rod handling tools in vertical condition, cylindrical measuring coils, reference coils, an AC oscillator unit ,Eddy Current channel analyser, a personal computer connected with a plotter. After visual examination individual fuel pins are transferred to this facility using fuel pin grappler. The fuel pin is inserted concentrically in to the measuring coils held by the testing fixture. Two nos. of encircling type measuring coils are used, one is used while lowering fuel pin and other while rising it. Separate measuring and reference coils are used for control rod testing. The fuel pin is moved up and down concentric to the coils and the outputs from the measuring coil and a reference coil is compared by EC channel analyser and the cross sectional variations can be plotted along the length of the fuel pin. In a similar manner the control rods of the fuel assemblies can also be tested to know the control rod wear characteristics. The arrangement is shown in Fig [5].

6. ULTRASONIC INSPECTION FACILITY

Ultrasonic inspection is done to characterize the discontinuities precisely those grossly indicated by the eddy current testing. In ultrasonic inspection the fuel pins are inserted in to the test stand and gripped in position by the spring loaded grippers. The measuring head is moved up and down along the length of the fuel pin. The measuring head consists of two ultrasonic transducers placed diametrically opposite and can be moved back and forth with reference to the fuel pin center line. The probes are also rotated inside the measuring head around the fuel pin. A wedge of ultrasonic energy is given by line focusing probe at approximately 26 ° angle to the surface to be tested which gets converted into a 45 ° shear wave after refraction that passes all around the circumference of the tube. Another probe is a spot focusing probe which is used for determining the cross sectional variations of the clad. Separate UT channels are used for each of the probes. The results are stored and plots can be taken for records.

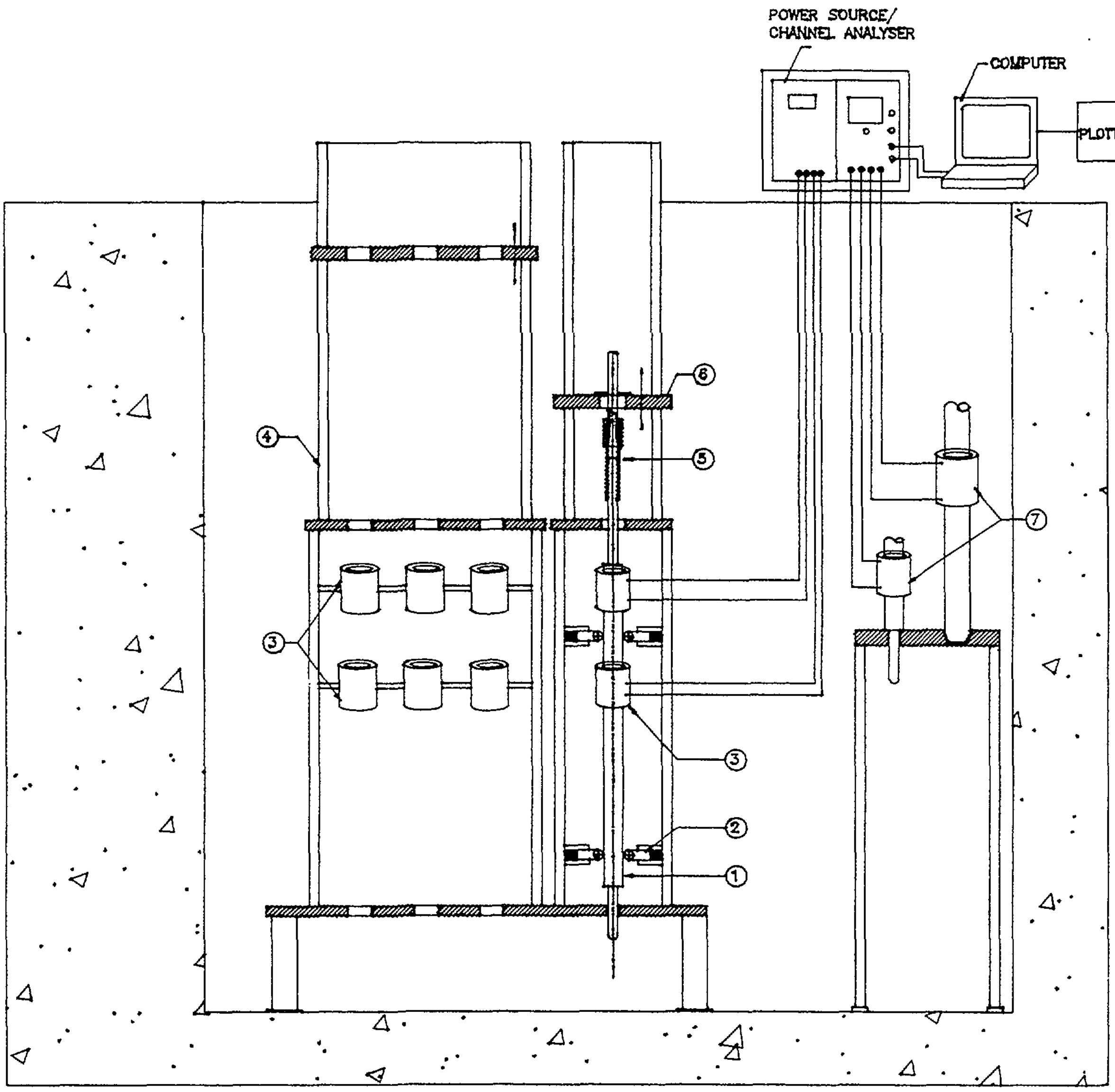
Using this facility it is possible to detect the fuel clad failure by sensing the presence of entrapped coolant between clad and the fuel and also the cross sectional variations of the fuel/control rod clad can be measured accurately to know the wear characteristics of the clad. The details are shown in Fig [6].

7. CONTROL ROD INSPECTION

The control cluster along with its spider is dismantled by removing the lock nut and kept on a storage rack. Visual inspection will be carried out on each of the rodolets for fretting corrosion, cracks and clad wear. The individual control rods can be disassembled and can be taken for Eddy Current, Ultrasonic testing. The facility design is such that the cluster also can be inspected without disassembly.

In Eddy current testing, the control cluster with spider is rested on a movable table. The table is moved up and down in such a way that each control rod moves concentric to the measuring coils. The control rod inspection will be completed with one up and down motion of the table. Individual Eddy current signals are compared with a single reference coil signal. By selecting the appropriate channel on EC channel analyser the results can be recorded.

In ultrasonic testing, a control cluster with spider is made to rest on the top guide table of the testing fixture. Individual control rod is passed through the measuring head and



LEGEND:—

1. FUEL PIN
2. GUIDING JAWS
3. MEASURING COILS
4. CONTROL ROD TEST SET-UP
5. PIN HOLDING COLLET
6. MOVING TABLE
7. REF. COILS

FIG. 5. Eddy current testing facility

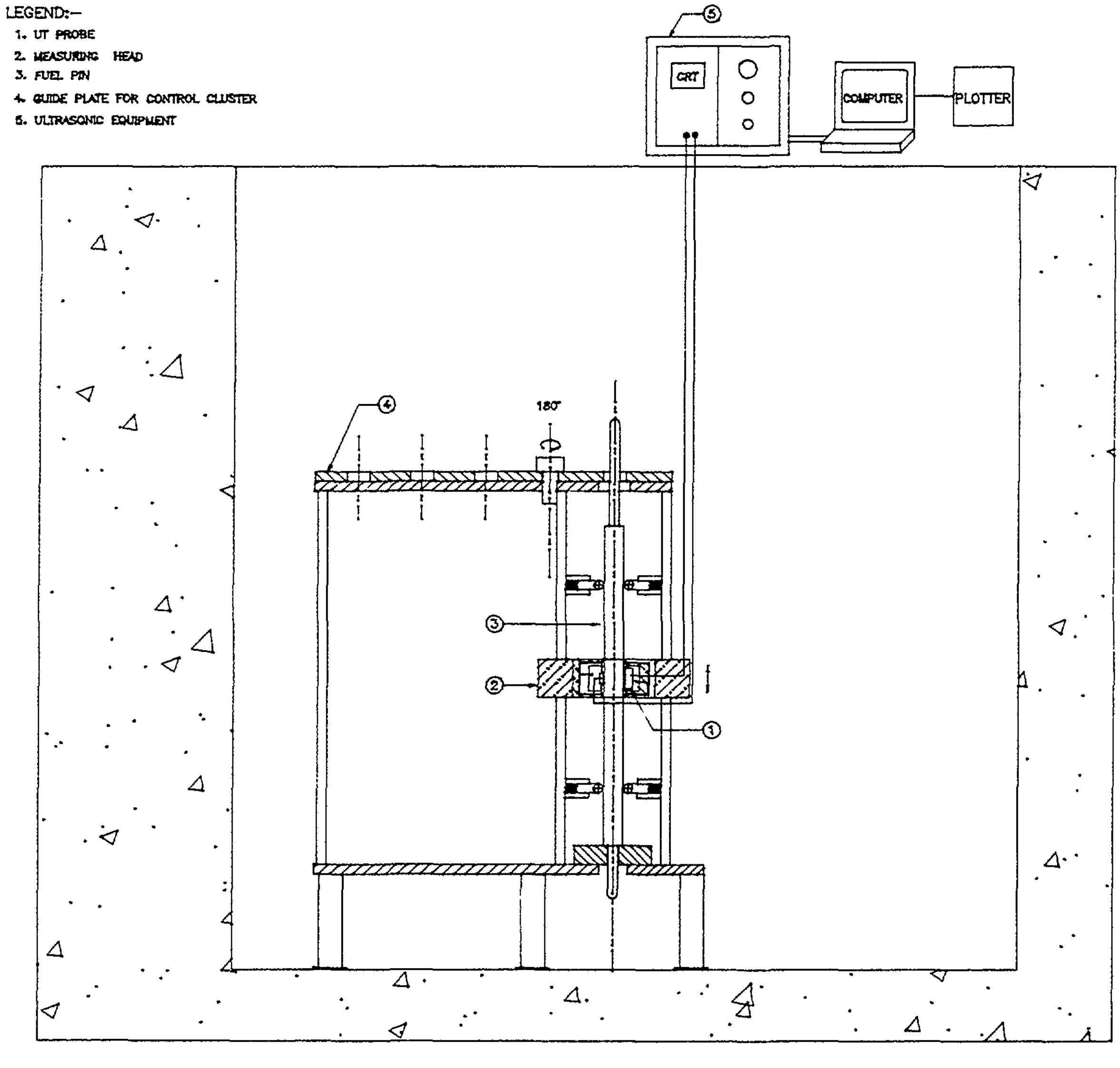


FIG. 6. Ultrasonic testing facility

inspection is carried out. In a similar manner, other rodolets are brought to the centre line of the measuring head by lifting and rotating the cluster before resting on the table.

Normally three types of rod failures are expected during the operation.

i) Fretting corrosion between rod and the outer guide tube.

ii) Wear of the rodlet bottom tip due to non parallel motion between spider shaft and the rodlet.

iii) Cracking of the rod clad tube due to radiation swelling.

8. CONCLUSIONS

1. Pool side inspection facility being established is intended to be used basically for identification of failed assembly and supplement to the inspection in hot cells.

2. The facility when commissioned will be used for all the LWR type irradiated fuel assemblies.

3. The facility design can be modified to suit variations in the fuel designs.

4. Reconstitution facility can also be added to the pool side inspection program for new fuel assembly designs.

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POOLSIDE FUEL ASSEMBLY INSPECTION CAMPAIGNS PERFORMED AT KERNKRAFTWERK LEIBSTADT DURING SUMMER 1997

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Abstract

In order to minimise fuel cycle costs, fuel assembly discharge burnup and average U-235 enrichment were increasing over past years in the Kernkraftwerk Leibstadt (KKL) plant. In parallel, high burnup verification programs were defined in collaboration with fuel suppliers. The aim of these programs is to demonstrate safe and reliable fuel performance up to the designed burnup limit and to identify any problems in due time. This is not only achieved by detailed poolside inspections of lead test assemblies, but also by hotcell post-irradiation examination of selected rods.

In the frame of a hotcell examination campaign, enhanced localised corrosion in the vicinity of spacers on SVEA-96 fuel rods was identified in May 1997 as a potential problem. The average rod burnup of the investigated rods was around 50 MWd/kgU after 5 one year cycles of operation. As fuel operation up to six cycles is foreseen in KKLs fuel management plans, the risk of fuel failures caused by enhanced localised corrosion could not be excluded. An action plan was therefore developed in order to identify the root cause. Part of the action plan were two poolside inspection campaigns:

1. Visual inspection of 38 assemblies unloaded during refuelling outage 1996 after 5 cycles in operation. This campaign was performed in June 1997. It gave a broader data base to develop a concept for fuel management for the upcoming refuelling outage scheduled in August 1997.

2. Visual inspection, oxide layer thickness measurements, crud sampling and rod diameter measurements on 29 assemblies with different operation histories. This campaign was performed during the outage. A large portion of the inspected bundles was re-inserted for continued operation. The collected data confirmed that assumptions made for reload licensing and safety analyses were conservative.

The inspection campaigns performed at KKL during summer 1997 by ABB Atom demonstrated that it is possible to address unexpected problems in a short time. Planning and data evaluation done in close co-operation between fuel supplier and plant, and a well trained, highly motivated inspection team, supported by experienced operators of the fuel handling equipment, applying well proven equipment, are the prerequisites for a successful inspection campaign.

1. INTRODUCTION

Under Swiss conditions, more than 50% of fuel cycle costs are backend costs. In a first approximation, they are proportional to the amount of spent fuel. By increasing the U-235 enrichment of fresh bundles, the average burnup of assemblies discharged from the Leibstadt reactor was raised from around 20 to well above 40 MWd/kgU (Fig. 1). In parallel, several fuel performance programs have been established in close collaboration with the fuel suppliers. The aim of these programs is to demonstrate safe and reliable fuel performance up to the designed burnup limit and to identify potential problems in due time. This is not only achieved by detailed poolside inspections of lead test assemblies, but also by hotcell post-irradiation examination of selected rods.

In the frame of a hotcell examination campaign, enhanced localised corrosion in the vicinity of spacers on SVEA-96 fuel rods was identified in May 1997 as a potential problem. The average rod burnup of the investigated rods was around 50 MWd/kgU after 5 one year cycles of operation. As fuel operation up to six cycles is foreseen in KKLs fuel management plans, the risk of fuel failures caused by enhanced localised corrosion could not be excluded. An action plan was therefore developed in order to identify the root cause and ameliorating actions. Part of this plan were two poolside inspection campaigns, the first one in June 1997, during normal reactor operation, where 38 discharged bundles were visually inspected, the second one during the refuelling outage in August 1997. The aim of this paper is not to discuss results obtained during these campaigns, but to show that it is possible to address unexpected problems in a short time, when work planning and data evaluation are performed in close co-operation between supplier and plant.

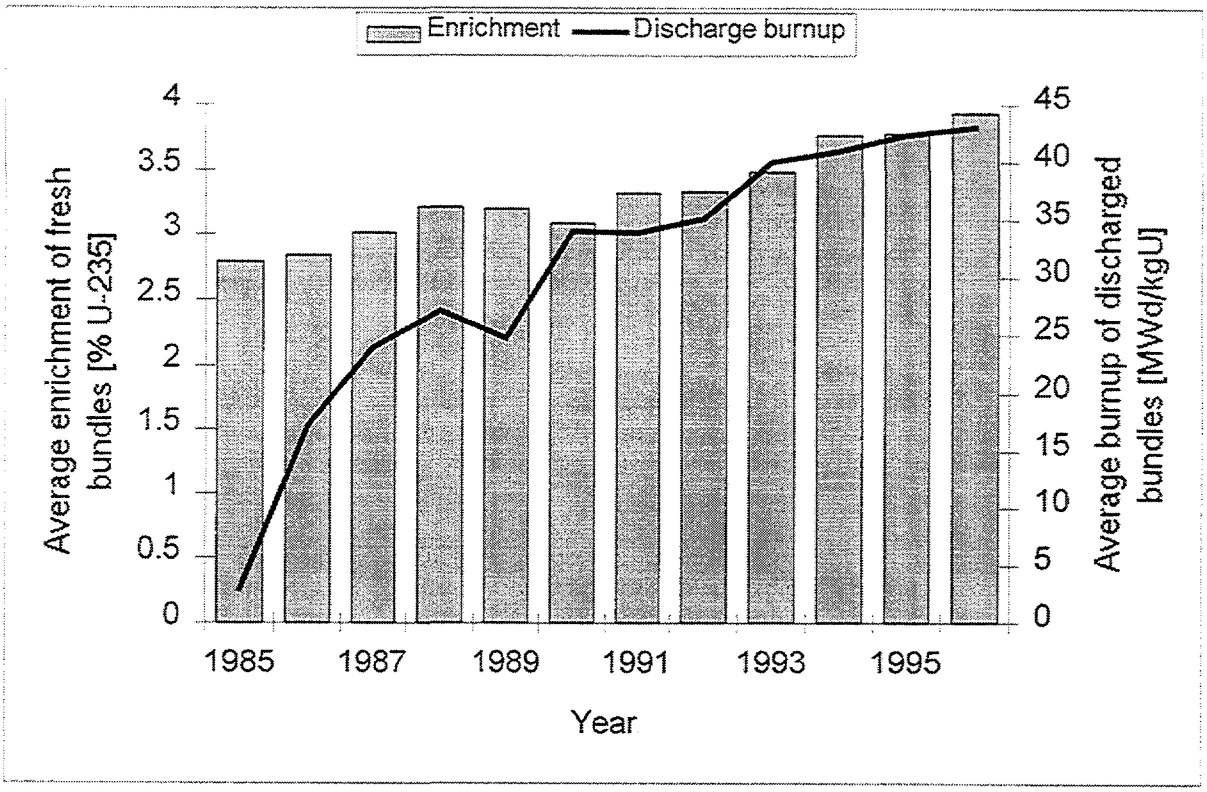

FIG. 1. Development of fresh fuel enrichment and discharge assembly burnup at KKL

2. FUEL IN THE LEIBSTADT REACTOR

The Leibstadt reactor, a General Electric BWR/6 Mark 3 boiling water reactor, started commercial operation in 1984. The core contains 648 fuel assemblies and is operated at a thermal power of 3138 MW. The initial core contained GE6 assemblies with fuel rods arranged in an 8×8 array. In cycle 3, the first fuel assemblies with a zirconium liner as a PCI remedy were loaded. With the aim to qualify alternative fuel vendors, ABB Atom SVEA-64 and Siemens-KWU 9-9Q lead use assemblies were introduced in cycle 3 and 4 respectively. In cycle 8, the first reload of ABB Atom's SVEA-96 10×10 assembly was introduced. The SVEA-96 bundle is composed of four 24 rod subassemblies placed in a fuel channel with a central diamond shaped water channel and water wings between the subassemblies. The spacers consist of Inconel-750. During cycle 10, a mid-cycle outage was performed in order to unload some failed fuel assemblies. The 13th one year cycle was completed on July 28, 1997. An overview of fuel assembly types in the KKL core up to cycle 13 is shown in Table I.

TABLE I. FUEL ASSEMBLY TYPES LOADED IN KKL CORE UP TO CYCLE 13

Assembly type	KKL Cycle													
	1	2	3	4	5	6	7	8	9	10A	10B	11	12	13
GE6	648	648	540	384	268	109	88	56	4					
GE7B			104	104	104	103	8				5			
GE8B				152	268	428	436	348	290	170	168	64	8	
GE10							104	144	144	144	142	142	70	
GE11-LUA							4	4	2	4	4	4	2	2
KWU-9-9Q				4	4	4	4	4						
SVEA-64			4	4	4	4	4	4						
SVEA-96								88	208	330	329	430	448	414
SVEA-96/L													64	64
SVEA-96+/L												8	56	168

3. INSPECTION CAMPAIGN PERFORMED IN JUNE 1997

The first 38 SVEA-96 fuel assemblies had been unloaded from the KKL core after 5 cycles of operation during the 1996 refuelling outage. They had reached average assembly burnups between 46 and 49 MWd/kgU. Within the frame of a high burnup verification program, 6 similar assemblies had been inspected in the pool during the outage. Four of them were re-inserted for another cycle. From the remaining two assemblies, one rod was selected from each one for detailed hotcell post-irradiation examinations. One of the rods showed extensive crud and oxide spalling at the lower spacer positions. The second rod was considered to be a "typical" five cycle SVEA-96 rod of that generation. The results of destructive hotcell examinations were available in May 1997. They showed that localised enhanced corrosion in the vicinity of the lower spacers had led to a wall thickness reduction of about 50% in the "typical" 5 cycle rod. The rod showing extensive spalling had as little as 25% remaining wall thickness at some spots. Moreover, some local surface hydride concentrations were found. Later on, the examinations were extended to a 3 cycle and a 4 cycle rod already in the hot cell. The latter locally showed a wall thickness reduction of about 25%, whereas the 3 cycle rod revealed a normal behaviour with an oxide layer thickness below 50 μm even within spacer positions.

As a basis for the development of a fuel management concept for the next cycle scheduled to start at the beginning of September 1997, a broader data base was needed. It had to show whether the observations made in the hotcells were representative for all rods or for some single cases only. Within a few days, a campaign was initiated and planned for the visual inspection of all 38 discharged SVEA-96 fuel assemblies. All peripheral rods in each subassembly were inspected (60 of 96 rods). A qualitative assessment of the nature of the oxide near and under the spacer grids was made for each rod in all six spacer positions. A scale from 1 to 6 was applied (1 = extensive spalling, 3 = onset of spalling, 6 = thin oxide). The whole inspection was documented by video tape. The work was performed from June 9 to June 19, 1997. Fuel assembly transfer from storage pool positions into the fuel handling equipment and back to the storage rack was performed by operators belonging to the regular KKL staff. Subassembly handling and inspection work was done by two ABB Atom inspectors. Two shift groups worked from 6 a.m. until 10 p.m. The data were evaluated in close co-operation between KKL and ABB Atom. They are plotted in Fig. 2. The results showed that about 4% of all inspected rods were at least once rated with 1. The affected rods were more or less evenly distributed across all inspected bundles with no clear preference for certain positions, but heavy spalling was not seen in the uppermost spacer positions.

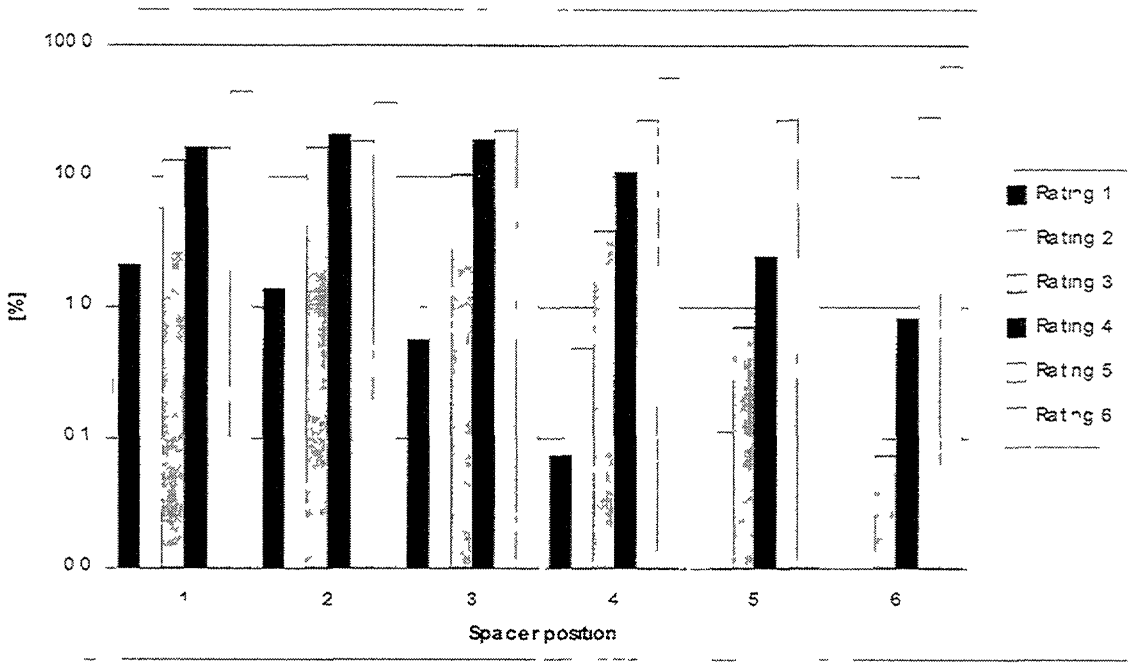

FIG. 2 Visual inspection of 5 cycle SVEA-96 fuel assemblies and qualitative assessment of nature of oxide near and under spacer grids (rating 1 = extensive spalling, 3 = onset of spalling, 6 = thin oxide). Observe logarithmic scale!

4. INSPECTION CAMPAIGN DURING THE REFUELLING OUTAGE 1997

4.1. Scope of inspections

The discharged bundle inspection performed in June 1997 showed a picture of the enhanced localised corrosion as it appeared at "end-of-life" for a group of assemblies with very similar irradiation histories. In order to learn more about the development of the phenomenon and possible influencing factors like rod power history, residence time, core location, time of fabrication and of introduction into the core, data from assemblies with differing irradiation histories and residence times are needed. Moreover, assumptions made on the basis of the hotcell examination results in order to establish a revised thermal mechanical operating limit (TMOL), taking into account enhanced localised corrosion, had to be confirmed by measurements performed on a significant amount of assemblies. As it was decided in an early stage of core design work for cycle 14 to unload all four and five year old SVEA-96 bundles in order to save time for a detailed analysis of the consequences of enhanced localised corrosion, the inspection program focused on three and four year old bundles. Bundles with as different irradiation histories as possible were selected, including bundles from core symmetry positions. Fuel rods with slightly different Zircaloy-2 cladding fabrication processing variables were included. An overview of selected bundles to assess enhanced localised corrosion is given in Table II.

In addition, two GE11 assemblies and a prototype SVEA-96 debris filter were visually inspected. Finally, in three defective assemblies the failed rod had to be localised and the primary failure cause identified.

TABLE II. FUEL BUNDLES SELECTED FOR DETAILED POOLSIDE INSPECTIONS FOR ENHANCED LOCALISED CORROSION ASSESSMENT

Residence time [cycles]	Number of bundles	Burnup range [MWd/kgU]	Remarks
1	4	11.2-11.8	1 assembly LK2, 1 assembly LK2+
2	2	23.5-23.8	1 assembly LK2, 1 assembly LK2+
3	10	25.2-34.8	2 assemblies with LK2, LK2+ and LK3 cladding 3 symmetry pairs
4	12	23.1-32.6	1 quad. symmetry, 3 symmetry pairs
5	3	28.6-41.6	1 symmetry pair

4.2. Inspection techniques

On two subassemblies per bundle, spacer 3 was shifted downwards. On a selection of subassemblies, spacers 1, 2, 5 and 6 were also shifted.

An underwater TV camera and video recording was used for visual inspection. Special attention was paid to the occurrence and lateral extent of spalling. At the same time, the general appearance and geometry of subassemblies were checked.

For oxide thickness measurements, a standard eddy-current (EC) lift-off (Fischerscope) equipment was used. Outside spacer regions, pointwise measurements were taken. In the spacer regions, continuous measurements of 100-200 mm length were performed. Special attention was paid to the problem of erroneous EC measurements caused by the magnetic properties of tenacious crud formed on part of the cladding, because zinc injection is applied at KKL [1, 2]. Therefore, soft and at least part of the tenacious crud layer was removed on some selected rods by mechanical grinding over a length of 20 mm. Zirconium oxide is not significantly affected, as it is much harder than crud. Comparison of EC signals within and outside the grind marks revealed whether the measuring signal was disturbed or not. If it was disturbed, the true oxide and crud layer thickness was always overestimated. Profilometry over the edge of the grind marks resulted in an estimation of the crud layer thickness, allowing for a (conservative) calculation of the net oxide layer thickness from the EC signal.

Mechanical metrology was applied for fuel rod diameter measurements. The profilometry results were used to verify EC oxide layer thickness measurements, to determine the crud layer thickness and to calculate the remaining metal wall thickness.

Four inspection teams consisting of 2-3 ABB Atom inspectors and two KKL operators were formed. During refuelling, when fuel assembly inspections were on the critical path of the outage activities, work at the spent fuel pool went on during 24 hours a day. Afterwards, the schedule was reduced to two shifts of 8-9 hours. The fact that the spent fuel pool and refuelling floor are in different buildings at KKL, was a big advantage, allowing for more or less interference free activities at both sites. The whole campaign took place between August 5 and August 28, 1997.

4.3. Evaluation procedures

Based on oxide layer thickness measurements in non-spalled regions with uniform oxide, the cladding inner diameter was determined as follows:

$$t_w = t_{orig} - \frac{t_{ox}}{PB}$$

$$ID = OD_{ox} - 2 \times \left(t_{ox} + t_w\right)$$

where

t_w is cladding metal wall thickness,

t_{orig} is original cladding wall thickness (0.63 mm),

t_{ox} is oxide layer thickness,

PB is the Pilling-Bedworth ratio for the conversion of oxide thickness into corroded metal layer thickness,

ID is cladding inner diameter,

OD_{ox} is outer cladding diameter including oxide layer, measured by profilometry.

The minimum remaining metal wall thickness in the spacer region, assuming axisymmetrical corrosion and spalling, was calculated as follows:

$$t_{w(\min)} = \left|\frac{OD_{ox} - ID}{2} - \left(t_{ox} + t_{crud}\right)\right|_{\min}$$

where

$t_{w(\min)}$ is minimum remaining metal wall thickness,

t_{crud} is crud layer thickness.

As it was difficult with the applied measuring techniques to exactly match the oxide layer thickness and diameter measurements, a conservative approach was chosen by using a maximum value for t_{ox} and a minimum local OD_{ox} value within each spacer region. The calculation is applicable for spalled and non-spalled positions.

4.4. Some results

Data evaluation is still going on and far from being completed. Moreover, this paper does not aim at presenting and discussing inspection results. Nevertheless, a selection of results is presented in order to show that the campaign was not only a heavy work load for all involved people, but also led to important conclusions. The following assumptions were made, when a revised TMOL for cycle 14 operation was established:

- No influence of enhanced localised corrosion on the TMOL up to a local burnup of 36 MWd/kgU (typical for 3 cycle fuel), provided that the average enhanced oxide thickness within the spacer region does not exceed 100 μm.

- Minimum remaining wall thickness of 50% at a local burnup of 50 MWd/kgU (typical for 4 cycle fuel).

- Minimum remaining wall thickness of 25% at a local burnup of 60 MWd/kgU (typical for 5 cycle fuel).

As can be seen in Fig. 3, maximum oxide layer thickness values, corrected for the crud layer and averaged over the enhanced corrosion spacer region, are smaller than 100 μm in all measured spacer grid positions for all measured 3 cycle rods. In Fig. 4, the minimum remaining wall thickness values in all measured spacer grid positions for all measured 4 cycle rods are plotted. The remaining metal wall thickness is well above 50% of the as-fabricated wall thickness. Therefore, the assumptions for establishing the revised TMOL and one of the prerequisites for cycle 14 operation were fulfilled and the authority, the Swiss Federal Nuclear Safety Inspectorate HSK, was able to give their cycle operation release.

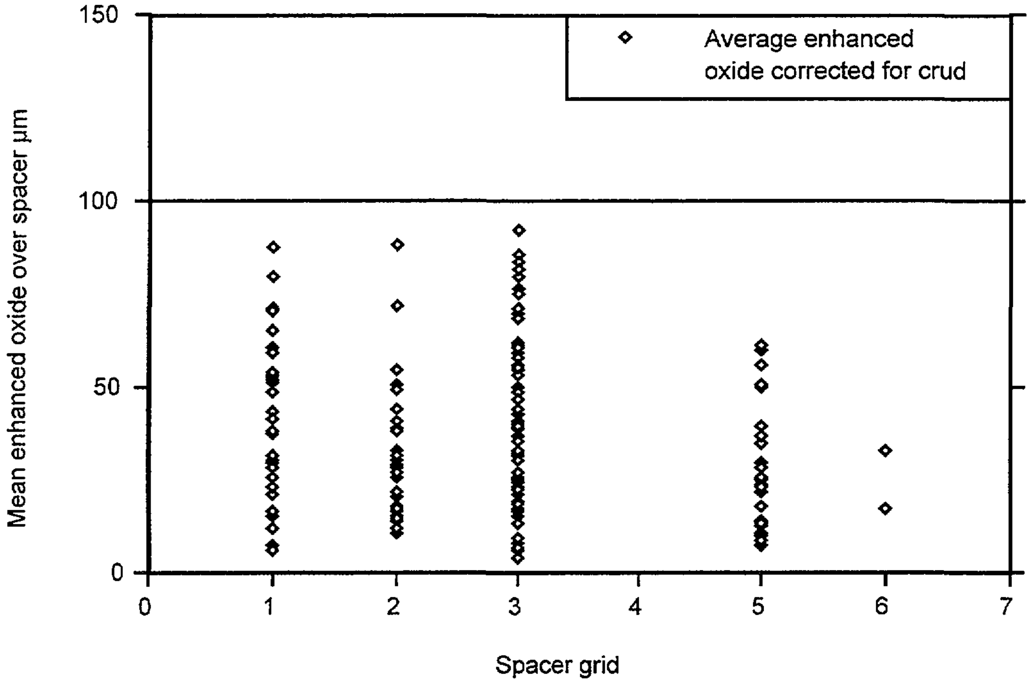

FIG.3. Mean enhanced oxide layer thickness averaged over spacer region in all measured grid positions for all measured 3 cycle rods.

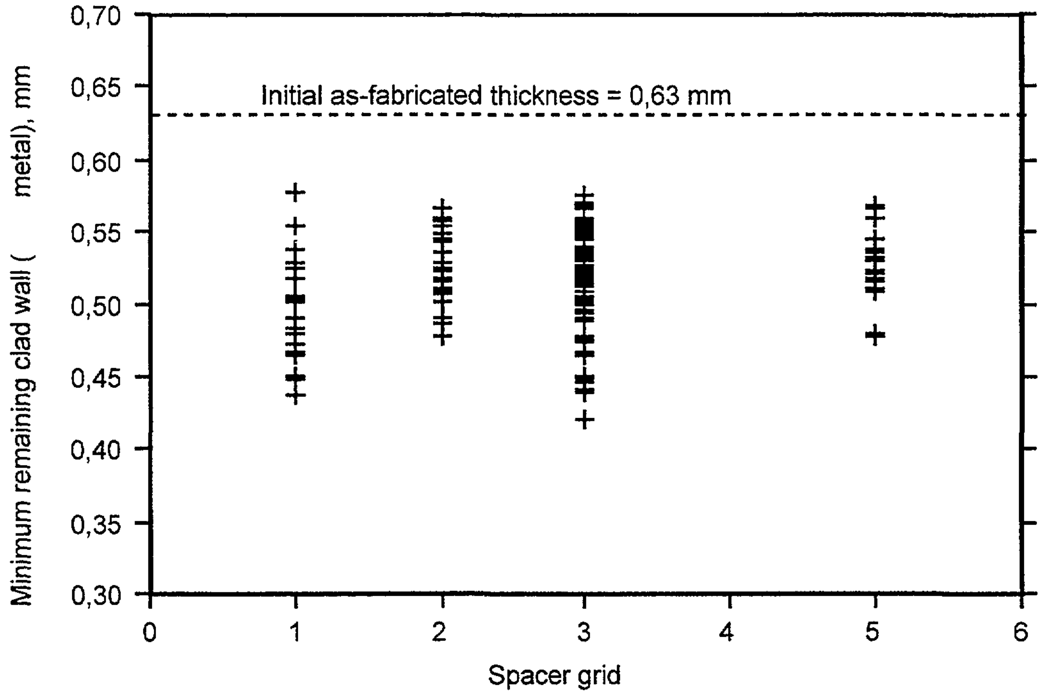

FIG. 4. Minimum remaining metal wall thickness in all spacer grid positions for all measured 4 cycle rods.

5. CONCLUSIONS

The inspection campaigns performed by ABB Atom during summer 1997 at KKL demonstrated, that it is possible to address unexpected problems in a short time,

- when planning and data evaluation are performed in close co-operation between fuel supplier and plant representatives,
- when a well trained, highly motivated inspection team, supported by experienced operators of the fuel handling equipment, is applying well proven equipment,
- when technically competent representatives of the authority are willing to closely follow the measurements, data evaluation and analysis.

ACKNOWLEDGEMENTS

Poolside inspection campaigns as they were performed during summer 1997 in KKL are only possible thanks to a large, versatile team working all out. Major contributions were made by

- A. Flühler and H.R. Gerber, KKL, who planned the campaigns, co-ordinated the efforts and organised support whenever needed,
- R. Lundmark, KKL, who assisted in setting up the inspection programs and prepared all necessary assembly handling lists together with E. Obst,
- J.O. Erling, who performed a great job as ABB Atom's inspection team leader,
- all the ABB Atom inspectors and KKL operators, working long shifts at the spent fuel pool under difficult and demanding conditions, and
- B. Anderssön, ABB Atom, who co-ordinated data handling and evaluation.

The willingness of H. Wand, HSK, to closely follow-up work progress by frequent visits on site is highly appreciated.

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EXPERIENCE OF DEVELOPMENT OF THE METHODS AND EQUIPMENT AND THE PROSPECTS FOR CREATION OF WWER FUEL EXAMINATION STANDS

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Abstract

The report presents the basic methods and equipment developed for inspection of the fuel elements and fuel assemblies in the spent fuel pools. It considers their characteristics and results of the tests under laboratory and experimental fuel examination stand conditions. In particular, the following techniques are presented: visual inspection, measurement of the geometrical dimensions, definition of the form change in fuel assemblies and fuel elements, detection of the failed fuel elements, etc. The experience of the experimental fuel examination stand operation is generalized. The concept of the creation of the WWER-440 and WWER-1000 FA and FE inspection stands is presented. The concept is based on the modular principle which runs as follows. A set of the basic functional blocks is being developed based on which it is possible to make such a stand configuration which is necessary to fulfil the specific program of the examination at the particular nuclear power plant.

1. INTRODUCTION

Within the frame of the program for development of the stands for WWER and RBPC (Reactor Big Power Channel) spent fuel inspection the experimental inspection stands were developed and manufactured. The basic methods for non-destructive examination of fuel elements and FAs were improved, the detectors and executive mechanisms were examined, the design techniques were checked. The report presents the experience of the methods and equipment creation for irradiated fuel inspection in the spent fuel pools and the concept for creation of stands for WWER-440 and WWER- 1000 FA examination.

2. EXPERIMENTAL INSPECTION STANDS

To test the equipment and to work out the inspection methods using the fuel element and FA models in the laboratory a special stand was developed (Fig. 1) [1]. The stand represents a frame construction of 9 m in height. Along two vertical guides the carriage with a mobile table moves. There are detectors and devices for FA examination on it. The FA model is established in a jack. The FA is rotated and the carriage is moved by the engines located on the control board. The mobile table on the carriage is moved by the submersible step by step engine.

The examination of the following methods were carried out at the stand:

- visual inspection by TV camera;
- eddy current testing of the fuel elements of the FA peripheral line by pancake coil;
- measurement of the oxide film thickness;
- measurement of the peripheral fuel element diameter and gaps between them;
- measurement of FA geometrical sizes (cross sizes, bowing size and twisting corner, length);
- detection of leaky fuel elements in FA.

Various detector blocks and facilities were also tested at the stand.

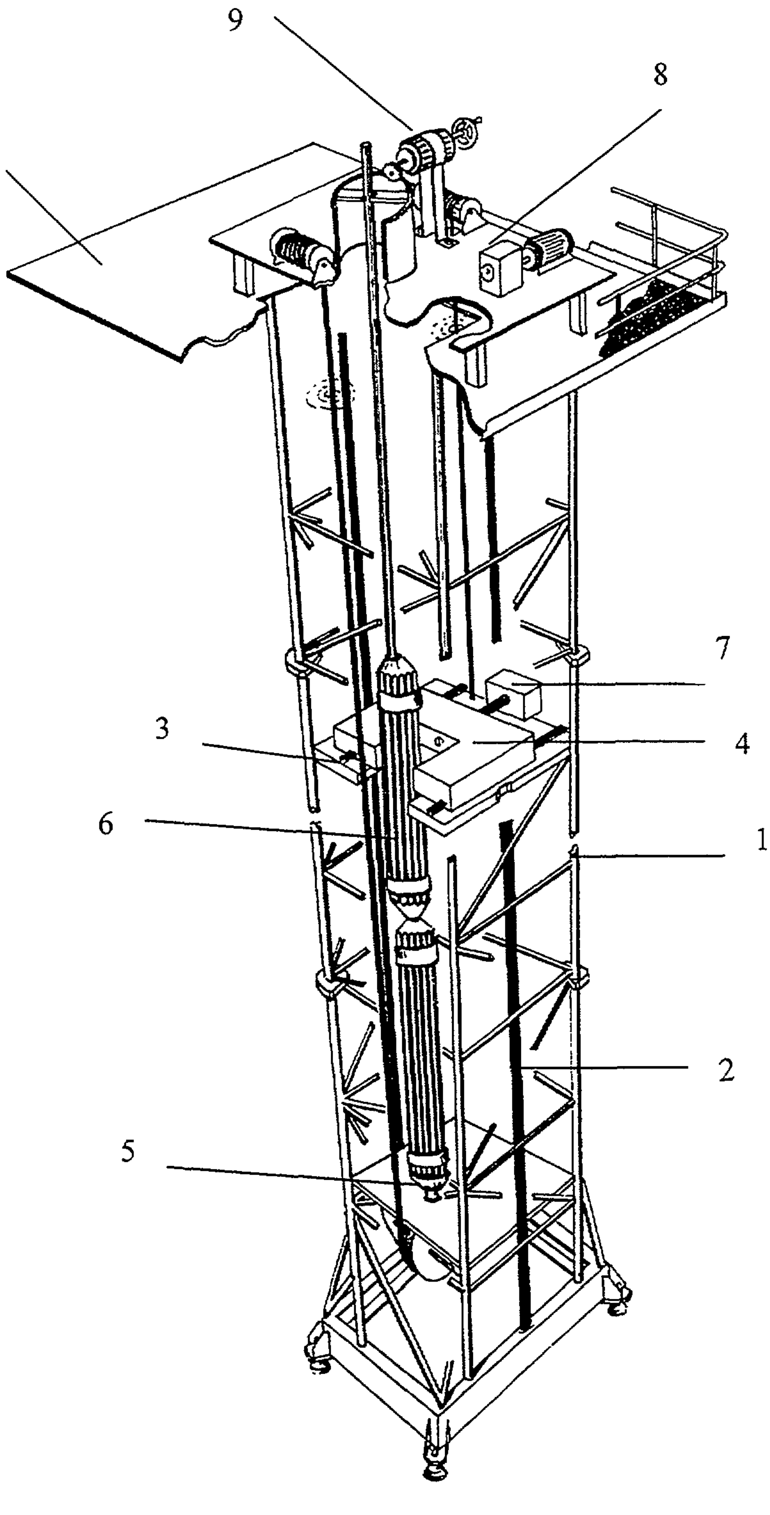


FIG. 1. Laboratory inspection stand.
1 - frame construction; 2 - guide; 3 - carriage; 4 - mobile table with detectors; 5 - FA support; 6 - FA; 7 - table driver; 8 - driver of carriage and FA support; 9 - FA rotating driver; 10 - servicing platform.

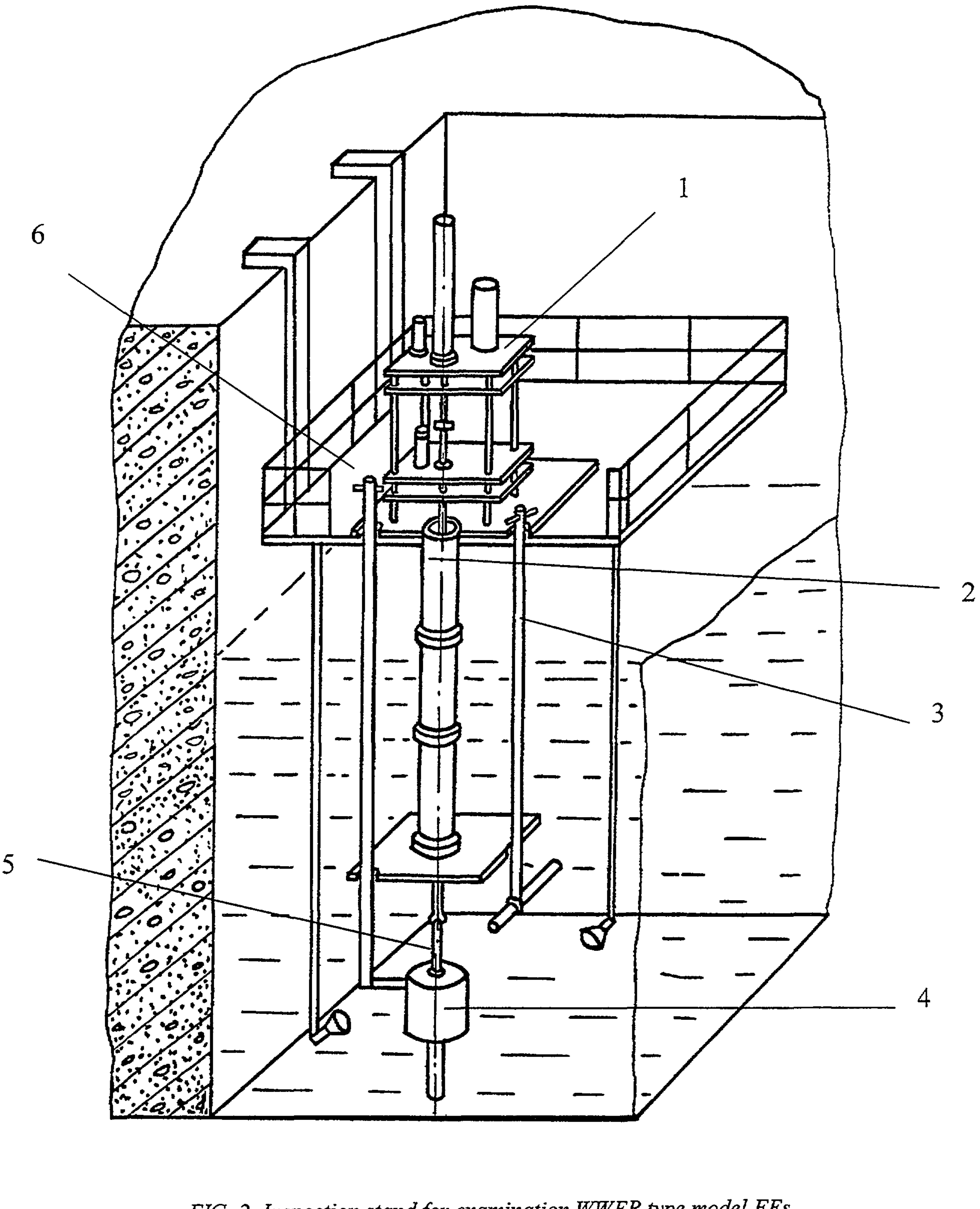


FIG. 2. Inspection stand for examination WWER type model FEs.
1 - fuel element transport mechanism; 2 - guide; 3 - TV camera mounting boom; 4 - probe unit; 5 - fuel element; 6 - servicing platform.

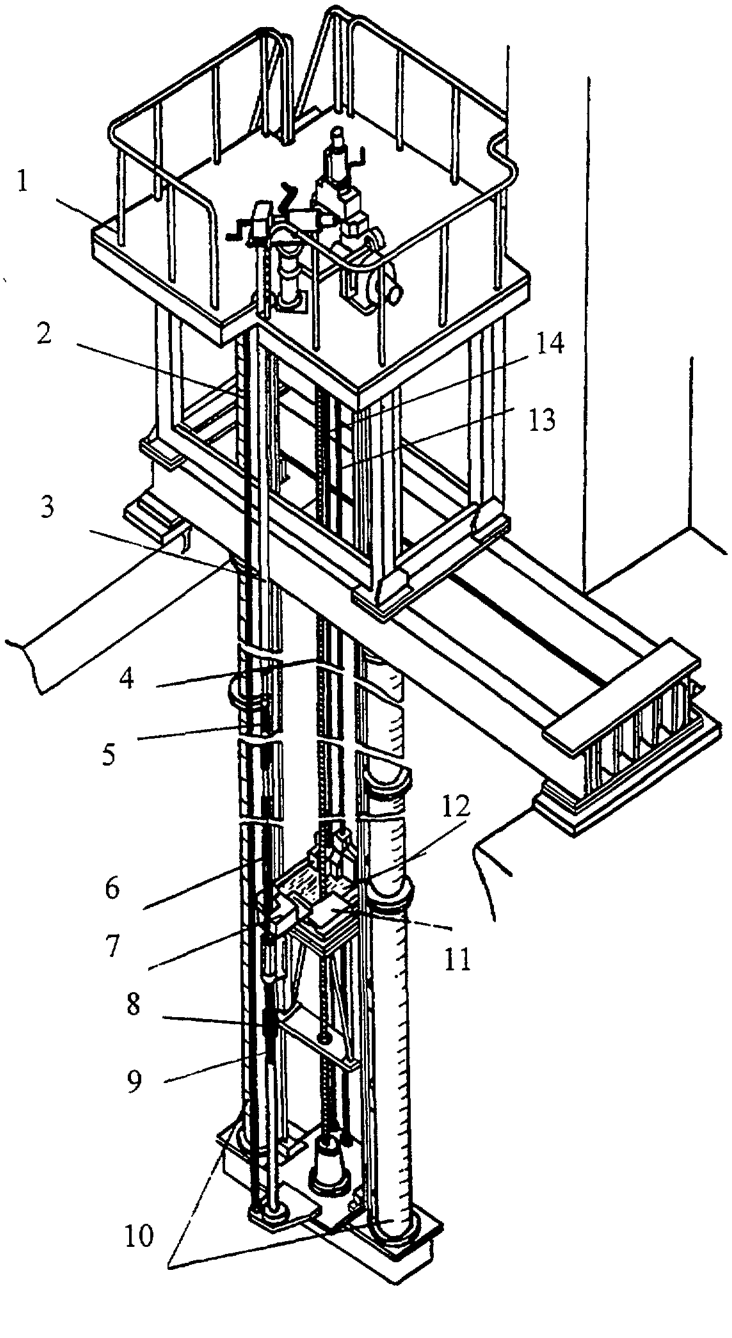


FIG. 3. FA inspection stand.
1 - servicing platform; 2 - ruler; 3 - FA holder; 4 - screw; 5,6 - FA;
7 - block with detectors; 8 - glass; 9 - test sample; 10 - column with guides;
11 - table; 12 - carriage; 13,14 - bars.

To improve methods for single fuel element examination the inspection stand for WWER type model fuel elements irradiated in the research MIR reactor was developed [2,3]. The stand (Fig. 2) represents the assembly construction consisting of the mechanism of fuel element moving and rotation, cylindrical guide, bars with TV camera and bars with measuring detector block. The stand design allows to carry out its quick installation and dismantling. Installation and the set-up of the measuring equipment takes from four till six hours.

The following NTD methods are applied at the stand:

- fuel element visual inspection;
- dimension measurement;
- eddy current testing;
- leak testing of fuel rods.

By the present time more than 50 WWER model fuel elements have been inspected at the stand.

Experimental inspection stand for RBPC FAs (Fig. 3) represents a suspension construction the basic elements of which are two columns of 300 mm in diameter [4]. The guides are attached to them. The carriage moves along them by means of a pair of screw and nut. On the carriage there is a table of two-co-ordinate moving. The measuring block with detectors are set on the table. The table is moved due to rotation of two bars transmitting a twisting moment from the drivers to the executive mechanisms.

All electric drivers are in the top part of the stand on the control board. The examined FA is brought into the stand sideways and fastened to the hanger. The FA bottom part is fixed in the special glass to avoid side moving. The bottom part of the glass simulates a small site of the FA and it is a calibration assembly for setting-up and checking of the detectors for dimensions measurement, devices for defectoscopy and visual inspection. The total stand height is 18 m, the cross dimensions are 1.5 x 1.9 m. The examination techniques of the stand is similar to those of the laboratory stand.

The stand was mounted in 1988 on the II block of Ignalina NPP and it is currently used to monitor the FAs unloaded from the reactor, i.e. the FA visual inspection is carried out by means of TV camera. According to it the primary information on the fuel elements and their state is obtained.

The stand to improve the method using a full-scale WWER-1000 FA model was also developed.

3. INSPECTION METHODS

The conventional methods of non-destructive examination of fuel elements and fuel assemblies in hot cells such as visual inspection, dimension measurement, eddy current testing were adapted for under water work in the spent fuel pools. The new methods being never used before in hot cells were developed. They are ultrasonic methods of leaky fuel element detection and methods for determination of change in form of the rod cluster control assemblies (RCCA).

3.1. Method of reconstruction of cross-section form WWER-1000 FA RCCA

The necessity to monitor change in form of RCCA is caused by the attempt to prolong their operation on the one hand and by providing the required level of safe reactor operation on the other hand.

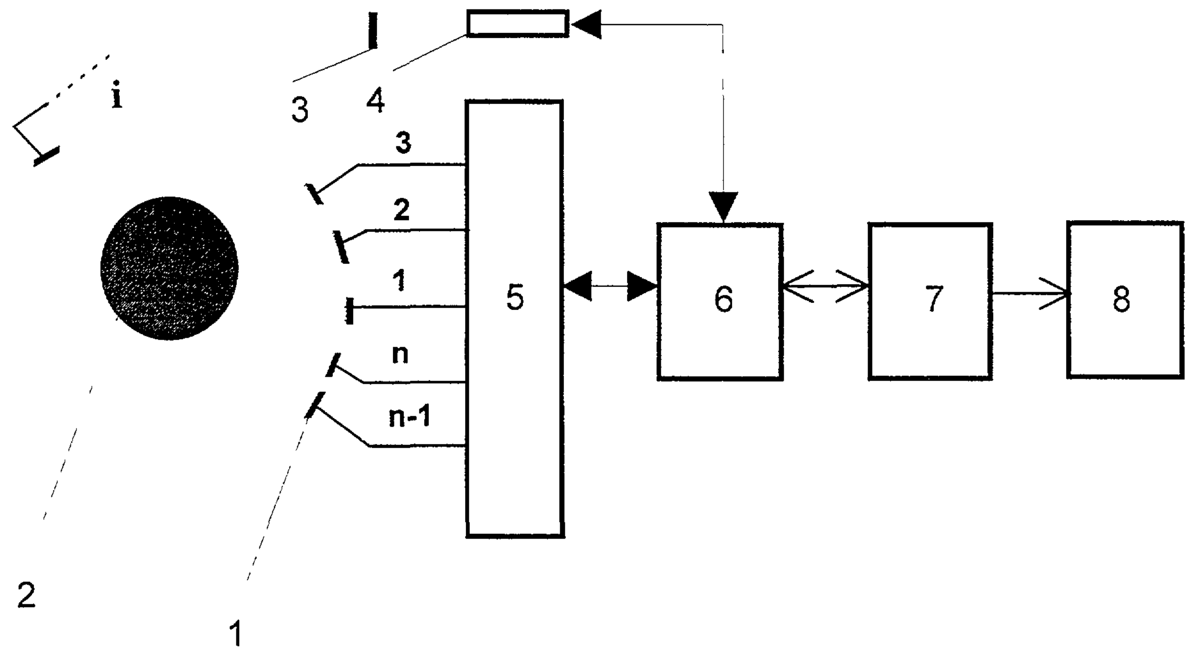


FIG. 4. Ultrasonic testing (UT) inspection principle.
1 - UT transducer; 2 - tested rod; 3,4 - device for correction sound velocity;
5 - switching device; 6 - UT channel; 7 - computer; 8 - printer.

During operation of RCCA two principal defects can appear [5]:

1. RCCA cladding tube external wear due to:

   - friction of FA guide tubes and RCCA at their vertical moving;
   - vibration of RCCA under the influence of coolant flow.

2. Cladding cracking of RCCA in the side of the bottom end due to swelling of the absorbing material under the influence of the irradiation and vibration.

The method was developed to identify the type of the cladding wear, to determine the maximal wear depth and to measure the rod cross size due to the swelling of the absorbing material.

The measuring device represents a ring, the inner diameter of which is 23 mm and outer diameter is 45 mm, inside of which RCCA under examination is located (Fig. 4). The outer diameter of the device is limited by the distance between the absorbing rods in the cluster.

Along the perimeter of the inner ring 104 ultrasonic detectors are located, which are connected to the measuring and computing system through the switching device.

The detectors operate in echo-pulsing mode, i.e. radiate ultrasonic waves in the direction of the RCCA surface and register them after reflection. Based on the obtained time of signal arrival and the sound velocity in water the distance from every detector to the RCCA surface is determined, and by means of specially developed algorithms the form of the RCCA cross-section is restored.

To correct sound velocity in water the method of the reference channel is used. From time to time it determines the sound velocity based on time of wave distribution from the detector of the reference channel to the reflector located at a particular distance from the detector.

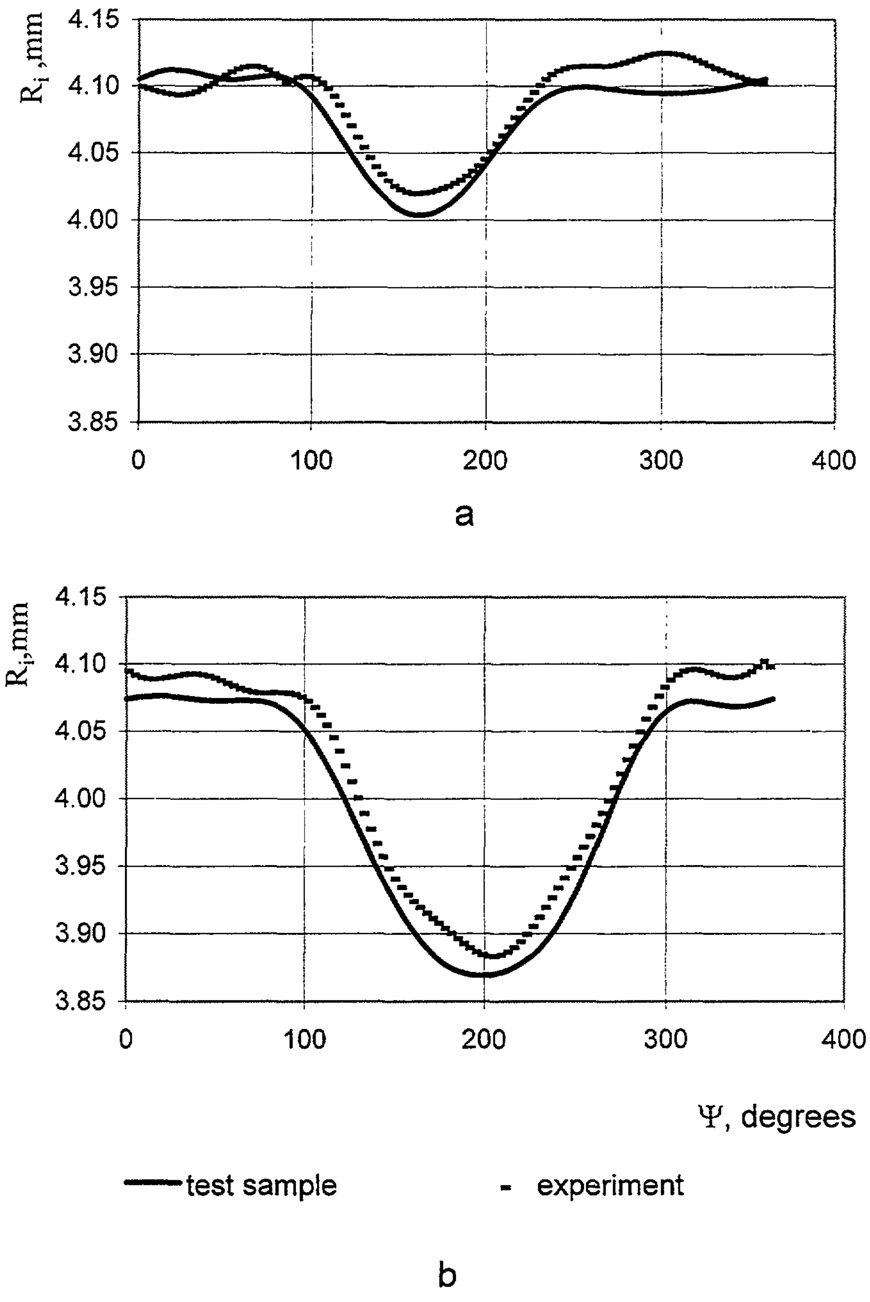
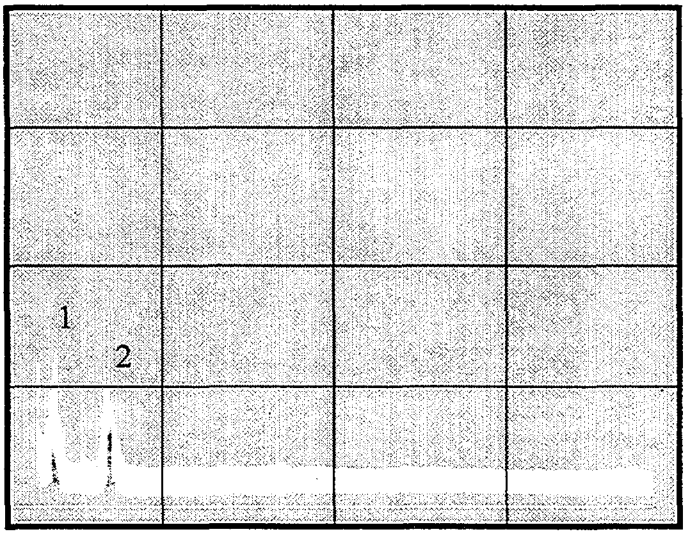


FIG. 5. Dependence of radius on the angle for the test samples.

The method was examined on the cylindrical rod test samples and test samples with imitation of worn RCCA cladding. Figures 5 a and b present dependences of radius of two test samples with maximal wear of 97 μm and 202 μm, respectively on the angle. It gives good conformity of the restored figure with the initial one. The accuracy of determination of maximal wear depth of the test samples is about 0.03 mm. The accuracy of determination of the material loss is about 2%. The given method can be applied for determination of the change in form of WWER fuel element cross-section.

a



b

FIG. 6. Examples of signals for sound (a) and leaky (b) WWER claddings at wave generation on the lateral surface:
1 - transmitter pulse; 2 - "useful" signal; 3 - multiple signals.

3.2. Methods for detection of leaky fuel elements in WWER-440 and WWER-1000 FA

Detection of the leaky fuel elements in FA is based on detection of water under the leaky fuel element cladding. The water under the cladding is detected by ultrasonic methods [6,7]. In the cladding the waves of certain type are generated. At wave propagation in the cladding their decay occurs due to energy radiation into environment. If there is water under the cladding, the additional easing of waves occurs and reduction of the wave amplitude testifies that fuel element is leaky.

According to the way of wave generation in the cladding the ultrasonic methods can be divided into two classes. The first one is generation and detection of waves on the lateral surface of the cladding. At that thin probes with emitter and receiver of ultrasonic waves are inserted into the space between the fuel elements in FA. The method of firm BBR and method EXO-330 belong to them [6].

The second method is generation and detection of waves on the fuel element top plug applied by firm Fragema [7]. The ultrasonic detector operated in a mode of radiation-reception is set on the plug. The wave is propagated along the fuel element cladding and detected by the same detector after reflection from the bottom plug.

The design of the WWER-1000 FA does not allow to use methods where the thin probes with detectors are inserted into the space between the fuel elements in FA, since the guide tubes of the RCCA block the gaps between the fuel element rows in FA. Therefore to detect the leaky fuel element in the WWER-1000 FA the method is applied, at which the monitoring is carried out from the side of the fuel element top plugs.

For WWER-440 FAs the monitoring can be carried out by all three methods after FA shroud tube removal.

Calculating and experimental examination of methods are carried out, the conditions of wave generation in WWER fuel element Zr claddings are determined. The basic parameters and specifications of methods are determined.

The efficiency of leaky fuel element detection was checked using the laboratory test samples and irradiated WWER fuel elements.

Figure 6 presents the characteristic signals for a sound and leaky cladding at wave generation on the lateral surface. Signal 2 corresponds to the wave going from emitter to receiver. The wave amplitude for the sound cladding is several times more than for the leaky one.

For the sound cladding a series of signals decreasing in signal amplitude (signals 3 in Fig. 6a) is observed which go one after another in equal intervals. These signals correspond to waves, which go around the cladding once, twice, etc.. For the leaky cladding these signals are not detected due to strong wave decay.

Thus, the detection of the leaky fuel elements can be carried out based not only on the amplitude of the signal of the first accepted wave (signal 2), but also on the presence of the sequence of signals 3 from waves, having gone round the cladding several times that raises reliability of correct identification of leaky fuel elements.

Figure 7 presents the example of distribution of signals for leaky and sound WWER-440 and WWER-1000 fuel elements at generation and detection of waves on the side of the top plug. The amplitude of signals for sound WWER-4400 fuel elements changes from 3.0 to 10.0 V, for leaky fuel elements the signal was lower than the noise level. For sound WWER-1000 fuel elements the amplitude changes from 1.0 up to 2.5 V, and for leaky fuel elements the signal is also lower than the noise level.

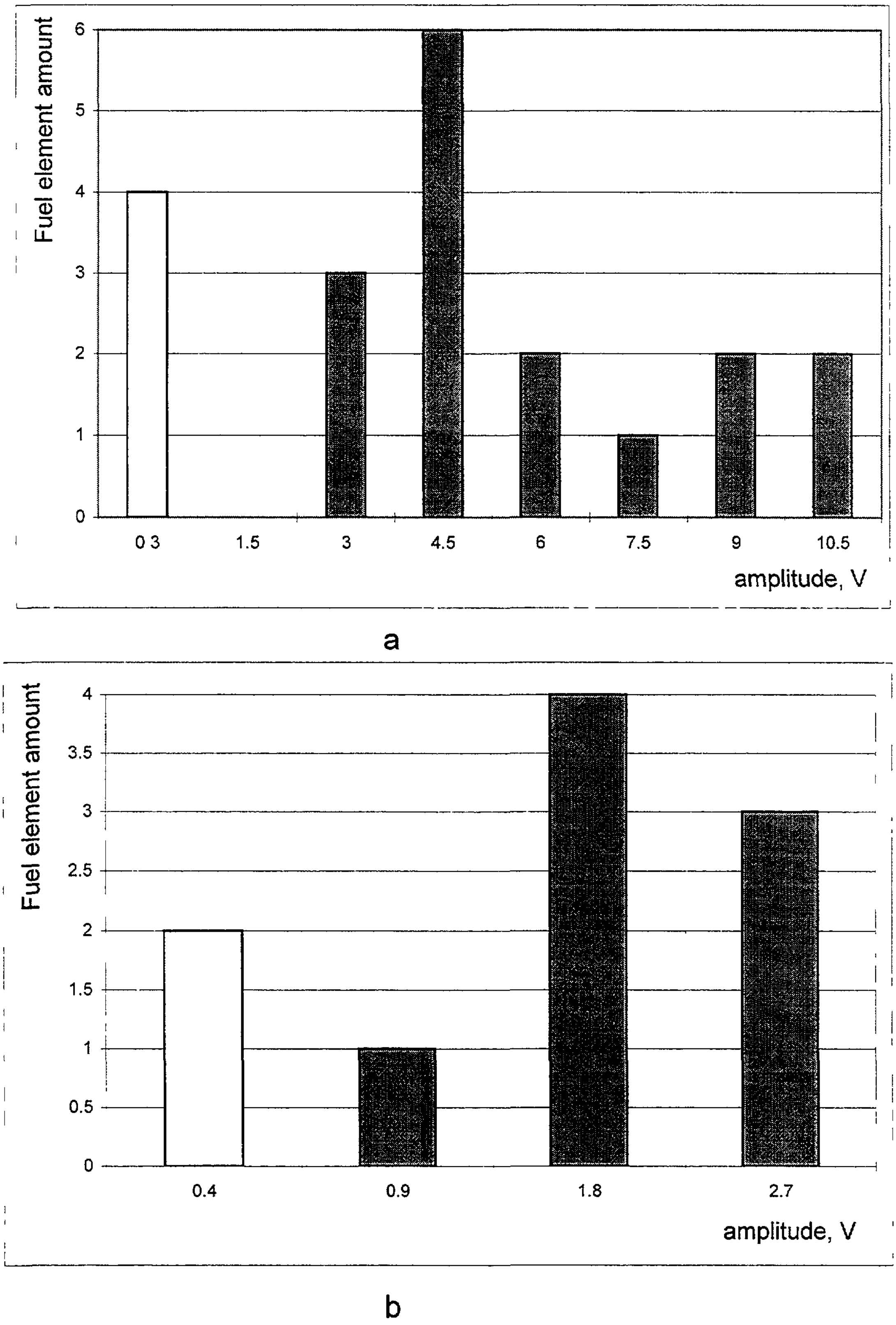


FIG. 7. Distribution of the signal amplitude for leaky and sound fuel elements for WWER-440 (a) and WWER-1000 (b) FAs.

- leaky fuel elements, - sound fuel elements.

4. CONCEPT FOR CREATION OF THE EQUIPMENT FOR POOLSIDE INSPECTION AND REPAIR OF DISMOUNTABLE WWER-440 AND WWER-1000 FAs.

Taking into account the world experience for creation of stands for inspection, repair and reconstruction, and experience for creation and operation of the experimental stands the concept of stand creation is developed to inspect dismountable WWER-440 and WWER-1000 FAs.

The analysis of possible sets of stands and their methodical providing and hardware has shown that despite the distinction in designs of WWER-440 and WWER-1000 FAs in technological operations during inspection, in set of the methods, etc., which will no doubt affect the design of the stand for every listed type of FA, a significant unification of these stands is possible. Therefore, a modular principle is the basis of the concept. A set of the basic functional blocks is developed. Their combination allows to make a stand for particular program examination.

The set of the functional blocks consists of:

- FA inspection stand being a rod of the whole equipment complex. Its basis is the mechanism which allows to move measuring devices and equipment for technological operations.

- Inspection stand of the single fuel element which is installed nearby the FA inspection stand. Before installation of the stand into the spent fuel pool the measuring devices being necessary for the particular program examination are established.

- Measuring devices for examination of the fuel assembly and separate fuel elements. These devices are made as an independent ones which are easily installed on the inspection stands of FA and single fuel element.

- Technological equipment for dismounting and mounting of FA. It includes tools for taking off and installing of the FA shroud tube and head, tools for removal of fuel elements from the fuel assembly.

The stands should be carried out in such a way as to realize their transportation from block to block within the NPP as well as between NPPs.

According to the concept the sequence and nomenclature of the experimental and design work should be the following.

1. The FA inspection stand for methods of visual inspection, dimension measurement and detection of leaky fuel elements in the FA as the most frequently used facility at FA inspection in the spent fuel pools is necessary to create.

2. Then providing the stand with other FA inspection methods, take an opportunity to carry out non-destructive examination of fuel assembly (without its dismounting), similar to the examinations in hot cells.

3. Development and creation of equipment for mounting and dismounting of the FA and inspection stand of the single fuel element. It allows to carry out examination of the single fuel elements removed from the fuel assembly, and further to repair the leaky FAs.

The realization of the given concept allows to obtain a variety of stands for inspection, repair and reconstruction by means of a small set of basic functional blocks.

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PWR FUEL INSPECTION AND REPAIR TECHNOLOGY DEVELOPMENT IN THE REPUBLIC OF KOREA

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Abstract

As of September 1997, 10 PWRs and 2 PHWRs generate 10,320MW electricity in Korea. And another 8 PWRs and 2 PHWRs will be constructed by 2006. These will need about 400 MTU of PWR fuels and 400 MTU of PHWR fuels. To improve average burnup, thermal power, fuel usability and plant safety, better poolside fuel service technologies are strongly recommended as well as the fuel design and fabrication technology improvements.

During the last twenty years of nuclear power plant operation in Korea, more than 4,000 fuel assemblies has been used. At the site, continuous coolant activity measurement, pool-side visual inspection and ultrasonic tests have been performed. Some of the fuels are damaged or failed for various reasons. Some of the defected fuels were examined in hot cell to investigate the cause of failure. Even though 30 PWR fuel assemblies were repaired by foreign engineers, fuel inspection and repair technologies are not established yet. Various kind of design for the fuel make the inspection, repair and reconstitution equipments more complex. As a result, recently, a plan to obtain overall technology for poolside fuel inspection, failed fuel repair and reconstitution through R&D activities are set forth.

1. INTRODUCTION

1.1. Status of Nuclear Energy in Korea

Since 1978, nuclear energy has been one of the major sources of energy along with hydraulic, coal and oil, considering availability, long-term supply, economics and technology (Table 1). 10 PWRs and 2 PHWRs are now operating with total capacity of 10,316 MWe and supply 43.2£¥ of the national electricity requirement. Additionally, 14 PWRs and 2 PHWRs with total capacity of 16,600MWe are planned to be built and one PWR is planned to be decommissioned by 2010. By the time, total capacity of nuclear power plants will be 26,330MWe.

TABLE I. NATIONAL ENERGY PROSPECTS
[UNIT: MWH (%)]

Year	Nuclear	L N G	Oil	Coal	Hydraulic	Total
1993	7,616 (28.0)	6,198 (22,9)	5,574 (20.5)	5,260 (19.4)	2,498 (10.4)	27,153
1995	8,620 (26.8)	6,740 (20.9)	5,920 (18.4)	7,820 (24.3)	3,090 (9.6)	32,180
2000	13,720 (26.0)	14,200 (26.9)	5,140 (9.8)	15,830 (30.0)	3,880 (7.3)	52,760
2005	18,720 (27.5)	16,210 (23.9)	5,500 (8.1)	22,030 (32.4)	5,480 (8.1)	67,930
2010	26,330 (33.1)	22,010 (27.7)	3,530 (4.4)	21,700 (27.3)	5,980 (7.5)	79,550

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Various types of nuclear reactors are generating electricity in Korea (Table 2). Westinghouse designed Ko-Ri units 1&2 of two loop reactors, Ko-Ri units 3&4 and Young- Gwang units 1&2 of three-loop reactors. Ul-Jin units 1&2 of three-loop reactors are designed by Framatome which are slightly different from those of Westinghouse. Young-Gwang units 3,4,5&6 and Ul-Jin unit 3,4,5&6 are designed based on ABB C-E System 80 design concept. Furthermore, two CANDU reactors are being operated and another two are under construction. Considering the scale of nuclear power production capacity in Korea, these variety of plant design makes it complicate to conduct construction, licensing, maintenance, improvement and repairs of the plants as well as fuel design, fabrication and service.

1.2. Status of Nuclear Fuel Cycle in Korea

Figure 1 shows a block diagram of fuel cycle in Korea. Conversion, reconversion, fabrication process of CANDU and PWR are established. Nuclear fuel design technology, materials and components development, increase of fissile resource usability, fuel services such as fuel inspection, examination and repair and waste disposal are tasks on a waiting list.

1.2.1. Status of PWR Fuel

From 1989, home-made nuclear fuels have been supplied to all the power plants. A new fabrication plant with capacity of 350MTU/y of PWR fuels and 700MTU/y of CANDU fuels is ready to supply fuels to the existing nuclear power plants as well as new ones to be constructed. These nuclear power plants are using several type of foreign or home-made fuels(Table 2). 14X14, 16X16 and 17X17 rod arrays of 'Standard Fuel Assembly(STD)', 'Optimized Fuel Assembly(OFA)', 'Advanced Fuel Assembly(AFA)', 'KWU Optimized Fuel Assembly(KOFA)', 'Korea Advanced Fuel Assembly(KAFA)' and 'Vantage 5H' for Westinghouse type PWRs, ABB C-E type 16X16 fuel (Figure 2) for ABB C-E type PWRs and 37 fuel rod cluster type fuel (Figure 3) for CANDU type PHWRs has been used. Such a variety of fuel type lead Korea Nuclear Co. Ltd.(KNFC) to be rich in technical experiences as well as problems. Based on their experience and international cooperation, KNFC is now trying to have his own design, fabrication and related technologies.

1.2.2. PWR Fuel Irradiation Performance

Since Ko-Ri PWR Nuclear Power Plant unit 1 starts its operation in 1978, numerous fuel assemblies have been burned (Table 3), but some of them are damaged due to flow induced vibration, debris, handling mistakes and unknown reasons(Table 4).

1.2.3. CANDU Fuel

Since 1983, numerous CANDU fuels, 100 MTU(¡-5,000 fuel clusters)/year, were supplied by KAERI and Canadian GE. The new fabrication plant of KNFC will supply fuels to the four CANDU reactors. No elaborate efforts are paid to examine the fuel integrity because any safety problem due to the fuel has not been issued yet and price of the fuel is relatively cheap(less than $1,500/fuel cluster).

2. FUEL INSPECTION, TEST AND EXAMINATION

2.1. Current Status of Pool-Side Inspection for Refueling

Considering that so many power plants are being operated, constructed and scheduled to be constructed in Korea, it seems to be urgent to establish technologies to supply more reliable engineering services; i.e., pool side fuel inspection, repair and integrity examination, in time.

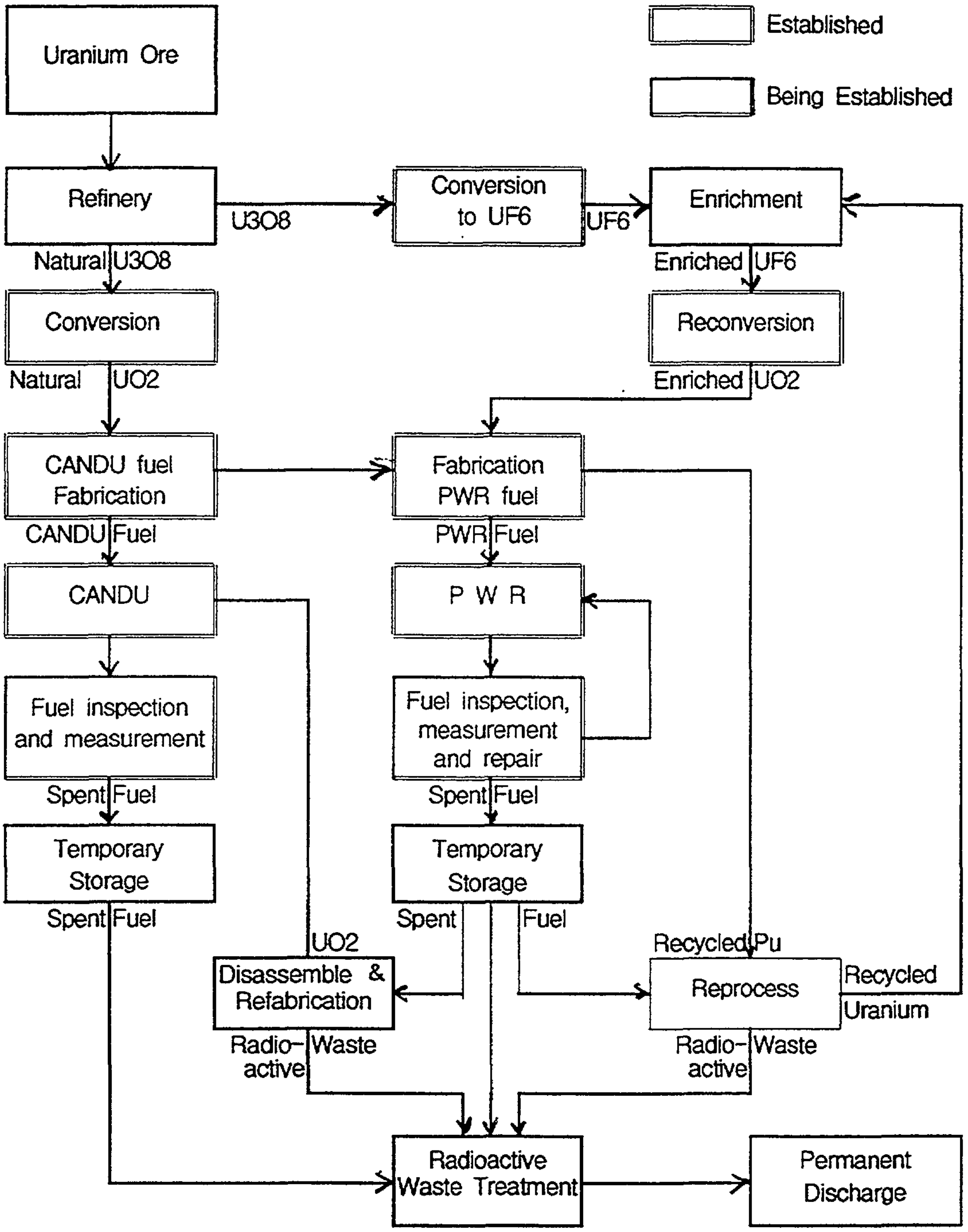


FIG. 1. Fuel cycle in Korea

TABLE II. STATUS OF NUCLEAR POWER PLANTS AND FUELS IN KOREA

Nuclear Plant unit #		Fuel Type	Fas in Core	Output (MWe)	Commerc. Start-up	Cycle (Month)	Enrich-ment(%)	Batch Burnup (MWD/MTU)
Ko-Ri	1	14x14, W	121	587	'78. 4	15	3.8	35,000
	2	16x16, W	121	650	'83. 7	15	3.8	33,000
	3	17x17, W	157	950	'85. 9	18	4.2	40,000
	4	17x17, W	157	950	'86. 9	18	4.2	40,000
Young-Gwang	1	17x17, W	157	950	'86. 8	18	4.2	40,000
	2	17x17, W	157	950	'87. 6	18	4.2	40,000
	3	16x16, CE	177	1000	'95. 3	12	3.1/3.6/4.1	43,000
	4	16x16, CE	177	1000	'96. 1	12	3.1/3.6/4.1	43,000
	5	16x16, CE	177	1000	'01. 6	12	3.1/3.6/4.1	43,000
	6	16x16, CE	177	1000	'02. 6	12	3.1/3.6/4.1	43,000
Ul- Jin	1	17x17, W	157	950	'88. 9	18	4.2	40,000
	2	17x17, W	157	950	'89. 9	18	4.2	40,000
	3	16x16, CE	177	1000	'98. 6	12	3.1/3.6/4.1	43,000
	4	16x16, CE	177	1000	'99. 6	12	3.1/3.6/4.1	43,000
	5	16x16, CE	177	1000	'03. 6	12	3.1/3.6/4.1	43,000
	6	16x16, CE	177	1000	'04. 6	12	3.1/3.6/4.1	43,000
Wol-Sung	1	37rod cluster	4560	679	'84. 4	12	natural	8,000
	2	37rod cluster	4560	700	'97. 6	12	natural	8,000
	3	37rod cluster	4560	700	'98. 6	12	natural	8,000
	4	37rod cluster	4560	700	'99. 6	12	natural	8,000

- additionally, eight PWRs will be constructed until 2010.

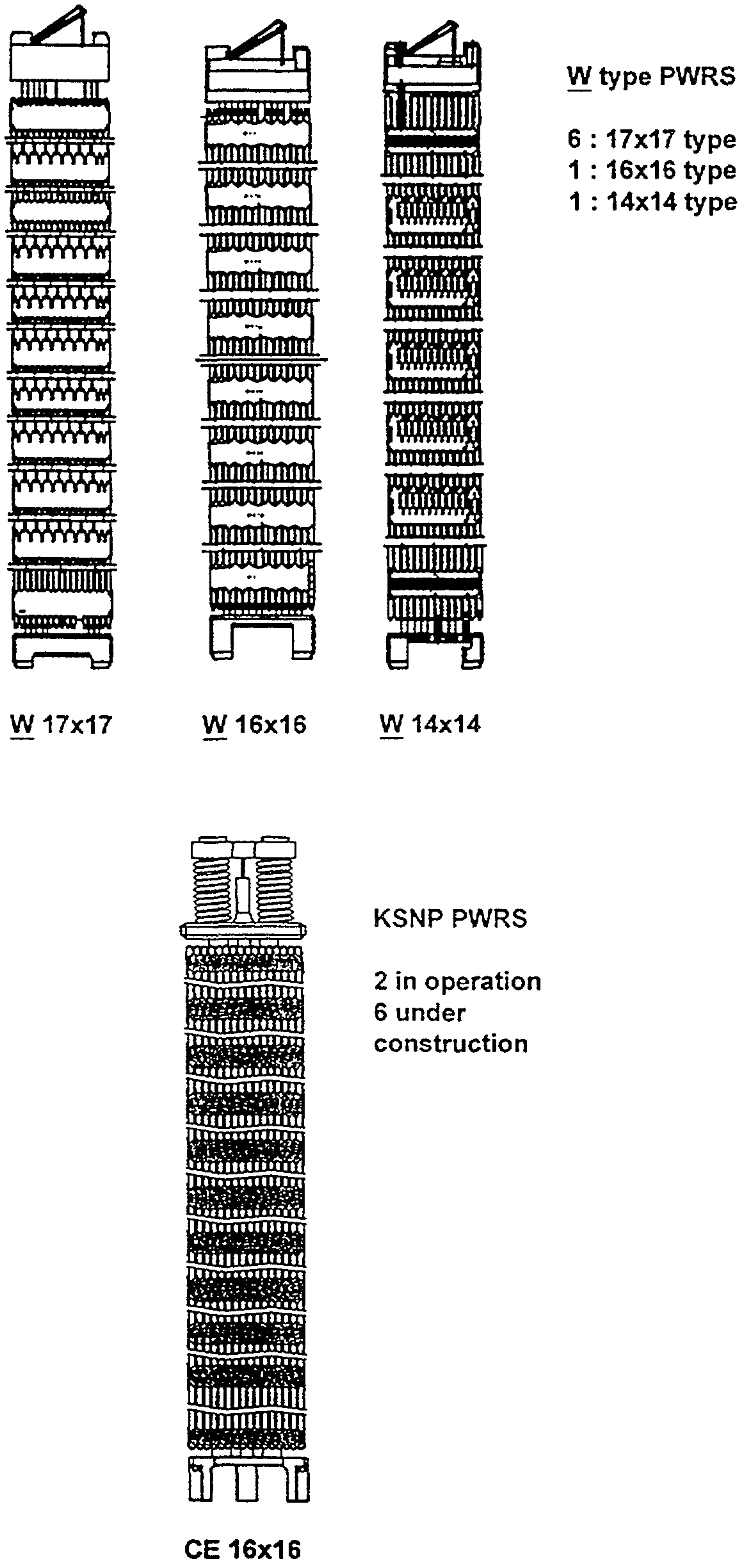
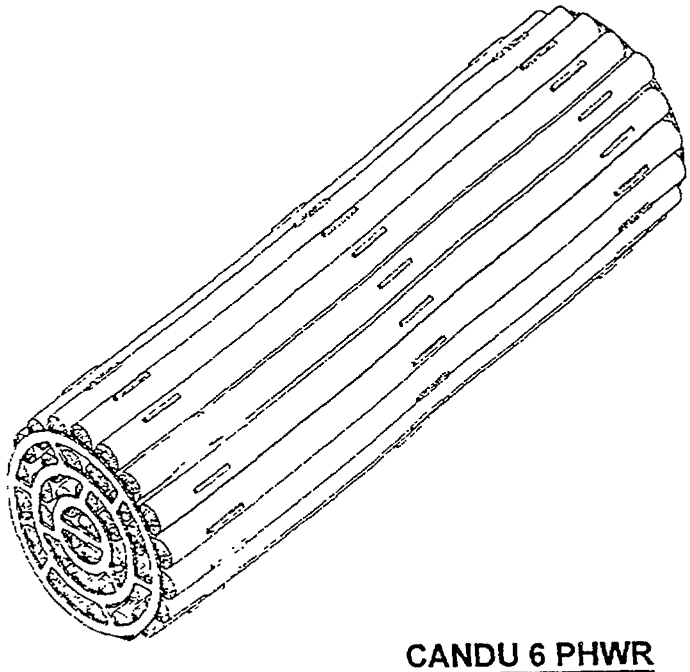


FIG. 2. *PWR fuels used in Korea*



CANDU 6 PHWR
2 in operation
2 under construction

❑ Component
- 37 Fuel Pin
- 2 Zry End Plate
- Spacer & Bearing Pad

FIG. 3. CANDU fuel bundle

TABLE III. DISCHARGED FUEL ASSEMBLIES AND FAILED FUEL ASSEMBLIES

Year	'78-'81	'82-'84	'85-'87	'88-'90	'91-'93	'94-'96	total
Discharged FA	80	161	420	1,000	1,209	1,137	4,007
Failed FA	5	12	6	5	31	3	62

TABLE IV. ROOT CAUSE OF FUEL FAILURES

Root Fault	Events	Failed Fuels	Comment
Baffle Jet Injection	5	17	Coolant Flow Path Design
Assembly Vibration	2	29	Fuel Design
Debris	6	11	
Manufacturing	2	2	
Fuel Handling	1	1	
Unknown	2	2	
Total	18	62	

2.1.1. Visual Inspection

The first step to examine integrity of irradiated PWR fuels is visual inspection using an under water TV camera. Even though this technology is well used, it is needed to be improved to have image analysis.

2.1.2. Sipping Test

To identify failed fuel assembly(ies) with leaking fuel rod(s), sipping test was performed, but, is no more applied now because failed fuel rod could not be identified with it.

2.1.3. Failed Fuel Rod Detection System (FFRDS)

FFRDS using ultrasonic test technology is used to identify leaking fuel rods. Even its performance is satisfactory, more reliable and convenient inspection system are anticipated.

2.2. Post irradiation examination at hot cell

2.2.1. PIE facility for fuel irradiation performance

Korea Atomic Energy Research Institute(KAERI) has a PIE facility which is composed of a pool, hot cells and radiochemical laboratory. Major jobs are as follow;

- Pool Side Examination : visual inspection, dimensional measurements, eddy current test and sampling of crud on the fuel rod cladding;
- NDT of Fuel Rod : visual inspection, dimensional measurement, axial gamma-scanning, X-ray radiography and eddy current test;
- DT of Fuel Rods : fission gas sampling, rod ovality and length variation, residual gap measurement, micro/macro ¥ã\och-scanning, burnup measurement and metalography, density measurement and chemical analysis of the pellet fragment;
- Radiochemical Laboratories : radiochemical analysis of the radioactive materials, such as crud, fission fragments, cladding materials etc.

2.2.2. R£¦D in the hot cell

2.2.2.1. PWR Fuel Post Irradiation Examination

Since 1986, seven irradiated PWR fuel assemblies and two fuel rods, which are burned for 1,2,3 or 4 cycles, have been examined in hot-cell to evaluate their irradiation performance.

2.2.2.2. Control rod examination

Since 1980, Hafnium rods have been used along with Ag-In-Cd rods as control rods in Korea. Post irradiation examination was performed with several damaged Hafnium rods.

2.3. New hot-cell at KMRR

In 1994, Korea Material Research Reactor(KMRR) with seven concrete hot cells and seventeen lead hot-cells was constructed to be used to produce radio isotopes and examine experimental specimens irradiated in the KMRR.

2.4. Pool-side fuel integrity test and dimensional measurement system development

KAERI developed an inspection system comprising with visual inspection equipment and dimensions, oxidation layer and holddown spring force measuring systems to investigate irradiation performance and root cause of fuel failure. With KAERI and international technical cooperation, an improved pool-side inspection facility is planed to be jointly developed from 1998 to 2002(Table 5). The scope is as follows;

2.4.1. Joint design and fabrication of the equipment

- fuel inspection stand
- fuel assembly inspection system;
    - periscope, underwater TV camera system
    - dimensional measuring devices(FA twist and bowing)
    - oxide thickness measurement device and crud sampling device
    - top nozzle spring force measurement device, etc.
- rod inspection system
    - periscope, underwater TV system
    - dimensional measuring device(ovality, diameter, rod length, profilometry)
    - eddy current test

- oxide film thickness, etc.
- skeleton inspection system
    - grid spring force measurement
    - borescope
    - guide tube dimensional change, twist and bowing
    - equipment for transportation
    - trailer and boxes

TABLE V. MILESTONE OF POOLSIDE INSPECTION EQUIPMENT DEVELOPMENT

year	1998	1999	2000	2001
Hardware	● Conceptional design ● Detailed design	● Parts make or purchase ● Fabrication	● Test ● Inspection ● Modification	● Practical use
Software	● Database buildup ● Training	● Inspection manuals ● Training	● Training	

Equipment will be fabricated in Korea or overseas depends on its convenience.

2.4.2. Training

- staffs need more practical experience through international cooperation

3. SPENT FUEL REPAIR

3.1. Fuel repair in Korea

In Korea, numerous Westinghouse type PWR fuels have been burned as they are scheduled, however, some of the fuels are failed(Table 3). Failed fuels are substituted with fully burned fuels and several fuels located in symmetric positions and are collected and repaired(Table 6) if the fuels can be burned for one or two cycles but slightly damaged. Furthermore, even when the Westinghouse type fuels is failed, the fuel is better to be repaired for use as scheduled in the view of fuel cycle management as well as plant management.

In 1995, two fuels are failed during startup test of the first cycle of Young-Gwang unit 4. When fuels burned in ABB C-E System 80 reactor are happen to be failed, they are better to be repaired to avoid extension of overhaul period as two or three months are needed to design emergency core. 12 ABB-CE engineers came with repair equipment(Figure 4), trained KNFC personnel and repaired the failed fuels. It cost quite a lot because the repair job has to be done in very short time and new fuel repair equipment have to be purchased.

3.2. Plan for fuel repair technology development

Fuel repair equipment is scheduled to be developed from 1998 to 2002 with a fund of about \$5,500,000 (Table 7). International and internal cooperation are expected to design and fabricate the equipment, train personnel and participate fuel repair job as follows;

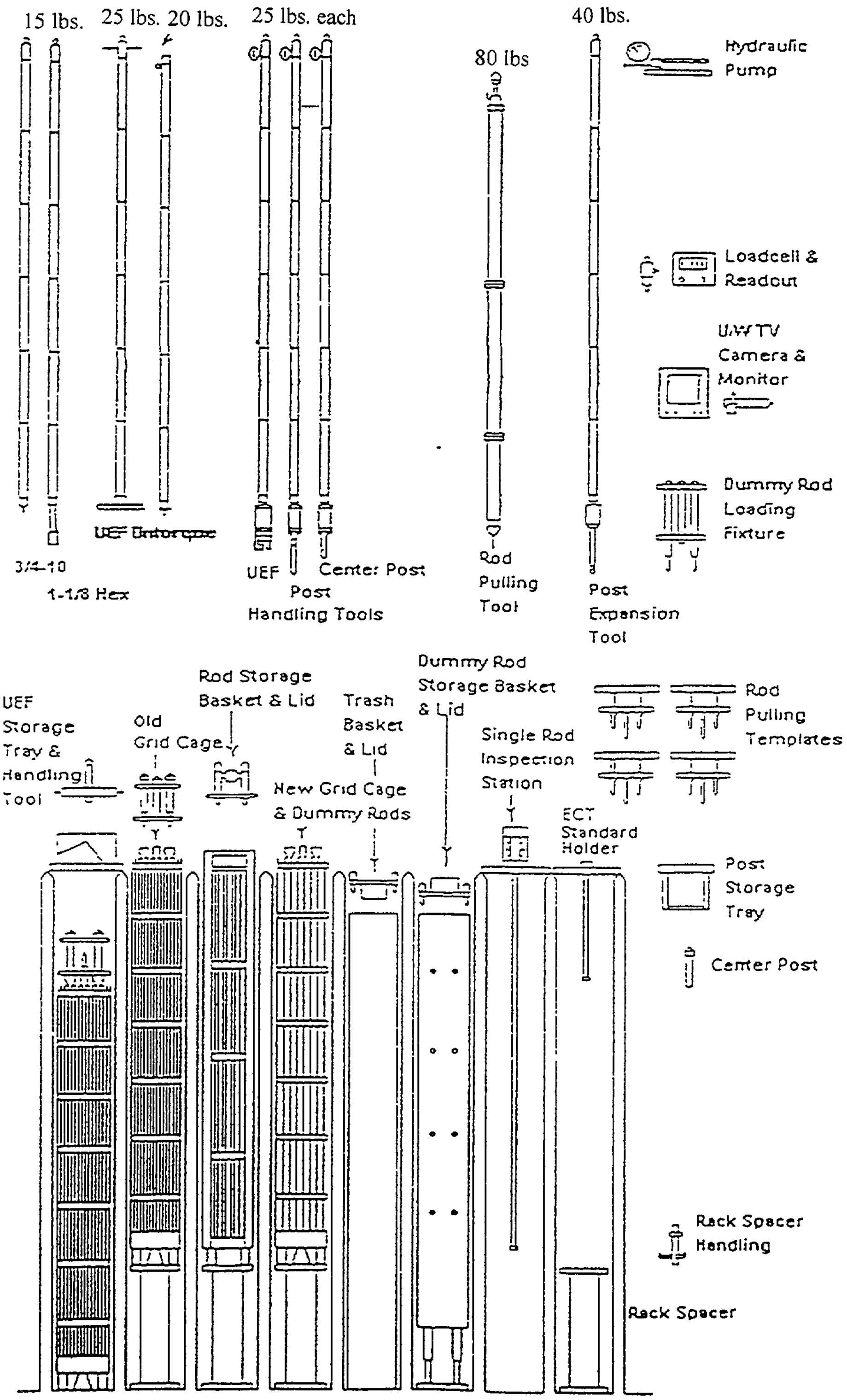


FIG. 4. ABB-CE fuel reconstitution tools

TABLE VI. REPAIRED FUEL ASSEMBLIES

Repaired Year	Design origin	Fuel Type	Repaired FAs	Repaired by
1985	W	14x14	17	Westinghouse
1991	W	14x14, 17x17	11	Westinghouse
1995	ABB-CE	16x16	2	ABB-CE/KNFC
Total	30			

TABLE VII. MILESTONE OF FUEL REPAIR EQUIPMENT DEVELOPMENT

year	1998	1999	2000	2001	2002
Hardware	●Concept design	●Detailed design	●Purchase or make parts ●Fabrication	●Test ●Inspection ●Modification	●Modification ●Repair failed fuel
Software	● Cooperation contract	● Tainting	●Database buildup Training	●Licensing Training	●Repair manuals

- equipment will be fabricated in Korea or overseas country depend on its convenience.

3.2.1. *Joint design and fabrication of fuel repair equipment*

- work station
- top nozzle disjoint/fixing tools for 14x14, 16x16 and 17x17 of STD, OFA, KOFA, KAFA, V5H and future fuels
- rod removal/insertion equipment for the fuel rods for 14x14, 16x16 and 17x17 of STD, OFA, KOFA, KAFA, V5H and future fuels
- fuel assembly turning(up side down and up) equipment for 14x14, 16x16 and 17x17 fuels
- equipment for transportation ; trailer and metal and wooden boxes
- one set of ABB-CE type fuel repair tools
- baskets for bolt/nut etc.
- fuel inspection equipment; underwater TV camera system, periscope

3.2.2. *Training*

Staff need more practical experience through international cooperation by performing joint repair work in Korea as well as overseas countries.

4. FUEL DESIGN IMPROVEMENT

4.1. **Development of advanced nuclear fuel**

An advanced fuel having higher thermal margin, burnup and zero-defect is planed to be developed(Table 8). Target goals of the fuel are 5,5000MWD/MTU of batch burnup, 15% of thermal margin, 0.3g of seismic strength and longer cycle length. Hardware development is scheduled from 1998 to 2000 and lead assemblies are scheduled to be irradiated from 2001 to 2004(Table 8). Development of new cladding and grid materials for the fuels is also an important part of the development to get better fuel integrity.

4.2. Fuel design modification for debris resistance

4.2.1. Rod removal/insertion process improvement to prevent scratches

Pushing fuel rods into or pulling out from a skeleton may induce scratches on the fuel rod surface and also damage grid spring because of friction between the rod and grid spring. To avoid this, a method of rod turning and simultaneously pushing into or pulling out from a skeleton has been developed. The fuel rods might have helical shape scratches, but not deep, on the cladding surface. This helical shaped scratches are considered to be less harmful than the longitudinal one.

4.2.2. Debris resistant bottom nozzle development

One of the major causes of fuel failure is debris induced fretting wear. We are developing a three dimensional helmet shaped tip with a number of smaller size holes to be attached under the bottom nozzle hole. It is assumed that debris which are big enough to induce fretting wear could not pass through the flow holes. Total flow area of the holes on a tip is designed to be bigger than the

TABLE VIII. MILESTONE FOR ADVANCED FA DEVELOPMENT

Phase 1	Phase 2	Phase 3
Development of improved KSNP FA		
• Jointly with a foreign vendor • Use reference design	Verification of In-Reactor Performance	
		Commercial Supply of Improved KSNP Fas
	Conceptual Design of FA Components	
	• Exclusive ownership for all PWRs in Korea	Development of Demo Fas for PWRs

flow area of the original flow hole of the bottom nozzle so that flow disturbance and pressure drop be minimum.

4.2.3. Fuel rod surface hardening by ion implantation

Most of the debris induced fuel failure occurred on surface of a part of fuel rod located just under the lowest grid. One of method to avoid the fuel rod failure is to make the surface of the fuel rod harder and stronger. High energy ion implantation technology is studied to make the fuel rod surface hard. The facility for the ion implantation may be complex but small in size and easy to control.

5. SUMMARY

In Korea, nearly half of national electricity requirement has been fulfilled with nuclear energy. To support these power plants more reliable fuel service technology are needed.

Fuel integrity examination technology is required to inspect spent fuel integrity to use them next cycle, examine fuel performance and investigate fuel failure mechanism. Even efforts had been devoted to build a pool-side spent fuel integrity examination equipment, more improved one is required to be developed to have more accurate and various information about spent fuels.

The fuels burned in Westinghouse type PWRs as well as System 80 reactor had better to be repaired to be used as scheduled. So, multi-purpose fuel repair equipment are required to be developed.

Plans to develop technologies for spent fuel pool-side inspection and repair are going to be set up which are very important part of a long-term national plan for fuel design improvement

Annex

PAPERS PREPARED FOR THE MEETING BUT NOT PRESENTED

XA9848403

THE RESEARCH AND DEVELOPMENT OF THE IN-MAST SIPPING TEST DEVICE

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Abstract

Sipping test device is used to identify the tightness of the irradiated fuel assembly during refueling campaign. The gas is selected as the medium and the Xenon 133 is selected as the indication nuclide. The device consists of the gas system, γ activity detection and measurement system, the power supply and signal system, the mechanical components and parts. There are satisfactory functions in the device e.g. easy operation, indication in instrumentation, chart record and acoustic alarm which can meet the operation demand of the nuclear reactor.

1. INTRODUCTION

Sipping technique is used to identify the tightness of the fuel. By isolating the fuel assembly to be tested and increasing the pressure inside the fuel (by heating) or decreasing the pressure outside the fuel, the fission products release from the defective fuel will be accelerated. The tightness of the fuel can be identified by detecting the fission products.

The Sipping technique has been developed since 1960's up to date there are the sipping taking the water as the medium, Iodine and Cesium isotopes as the indication nuclide; and the sipping taking the gas as the medium, Krypton and Xenon as the indication nuclide; the poolside sipping using the sipping cell as the isolator; and the in mast sipping using the mast of the refueling manipulator as the isolator.

The objective of this subject is to set up a detection device to identify the tightness of the irradiated assembly for Qin Shan phase two NPP. The device should be more sensitive and easy to operate.

2. THE OPTION SELECTION

It was learnt from the literature that the escape factor of fission gas krypton and xenon from the defective fuel into the coolant are about five times the factor of soluble fission products iodine and cesium, the activity of Kr85 and Xel33 in the coolant are higher than that of I131 and the activity of Xe133 in the coolant is about 150 times that of Kr85, he Xe133 has the more portion of γ activity with high peak which is easier to detect, Xe133 with a shorter half life can be detected during refueling operation. For the more sensitive device the Xe133 is the best indication nuclide.

The in mast sipping used the mast as the isolator is more simple than the sipping with the special sipping cell, and it is easy to operate only a bit more operation time is necessary without special operator.

The in-mast sipping with gas medium and Xe133 indication nuclide was selected.

3. IN-MAST SIPPING TEST DEVICE

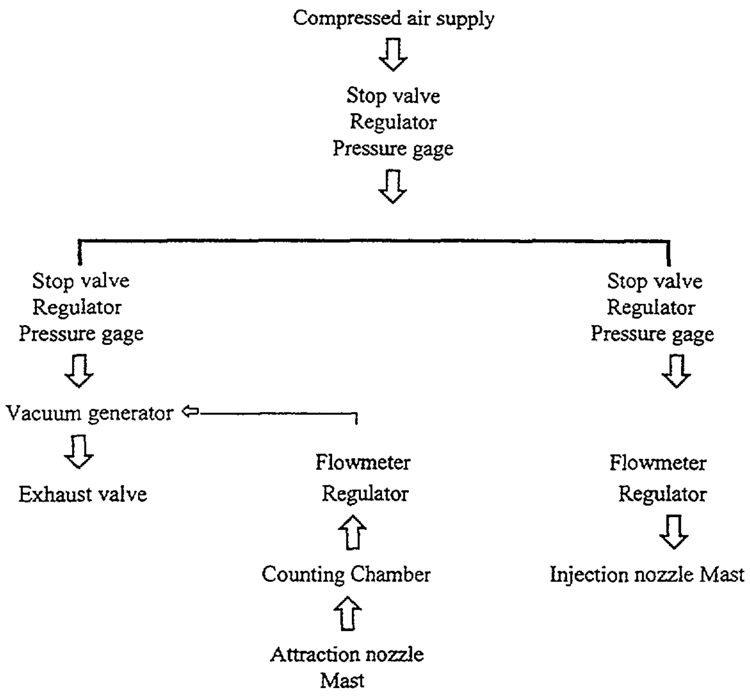
As the fuel assembly griped by the gripper raises inside the mast of the refueling manipulator from the reactor core to the top gripper position for about 9m, the Xe133 release through the defective clad of the fuel will be accelerated. The compressed air injected into the mast at the bottom raising around the fuel rods will carry the Xe 133. The air will be attracted from the mast at the top to the counting chamber where the Y activity of the air will be detected and measured by the γ activity detection and measurement system to identify the tightness of the fuel assembly.

The sipping device consists of the gas system , the γ activity detection and measurement system the power supply and signal system and the mechanical components and parts.

3.1. Gas system

The gas system is used for the compressed air supply, the air injection and the air attraction to the Counting chamber.

The compressed air flow is divided into two ways, one of them supply the injection air in certain pressure and flowrate to the bottom of the mast, the another way supply the air in certain pressure to a vacuum generator to attract the air from the mast through the counting chamber in certain flowrate. The system is as the scheme.

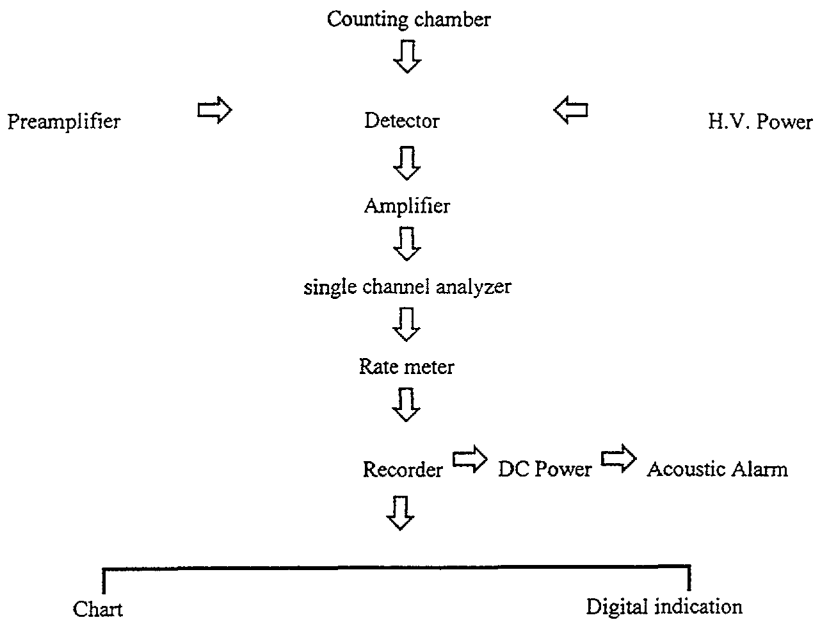


The Calibration of the gas system is done in two steps : the tightness test of the whole circuit section by section; the adjustment of pressure and flowrate to the value.

3.2. The γ activity detection and measurement system

The γ system is used for detection, measurement and record of the γ activity of the gas inside the counting chamber.

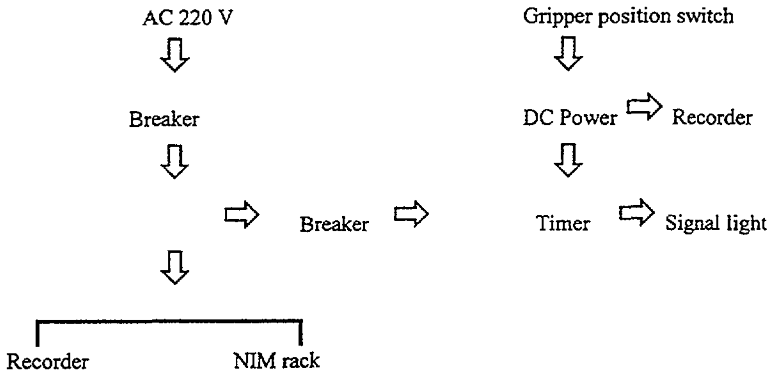
The system consists of a series of Nuclear Instrument Module (NIM), the detector and the recorder. The system is as the scheme.



The calibration of this system is done in three more steps: the adjustment of the NIM; the drawing of a curve in cps - threshold with standard solid γ source; the setting of the threshold corresponding to the cps peak of the indication nuclide.

3. 3. The Power Supply and Signal System

The power supply and signal system is used for supply the power to the NIM rack, the recorder and the signal light. The system consists of the breaker, the timer, DC power source and signal light. The system is as the scheme.



3. 4. The Mechanical components and parts

The mechanical components and parts are: the counting chamber, the lead cask, the source holder and the cabinet. All of these matter are out of vendor catalog. The design, manufactory and quality assurance should be done specially.

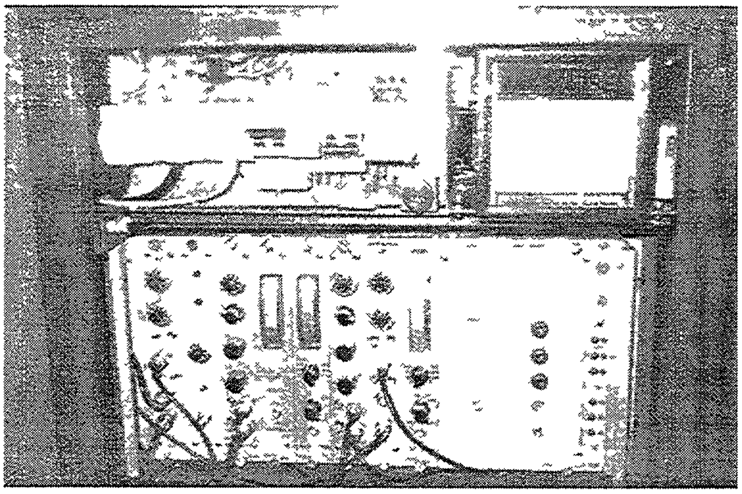
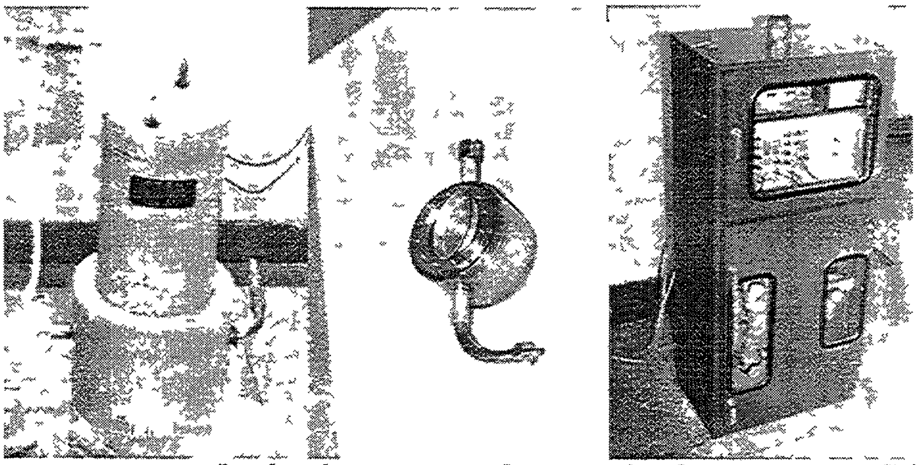
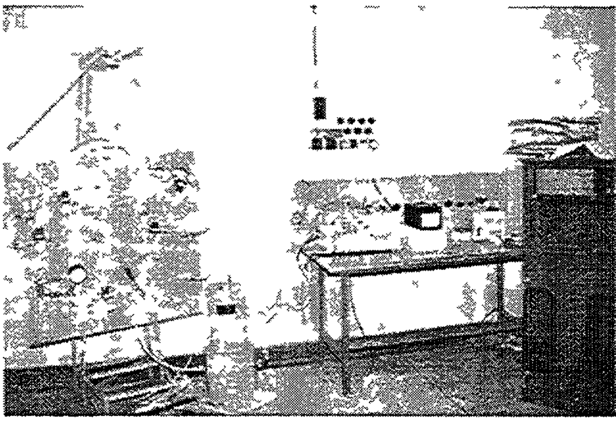


FIG 1 Some standard parts



Lead cask *Counting chamber* *Cabinet*

FIG 2 Some special parts



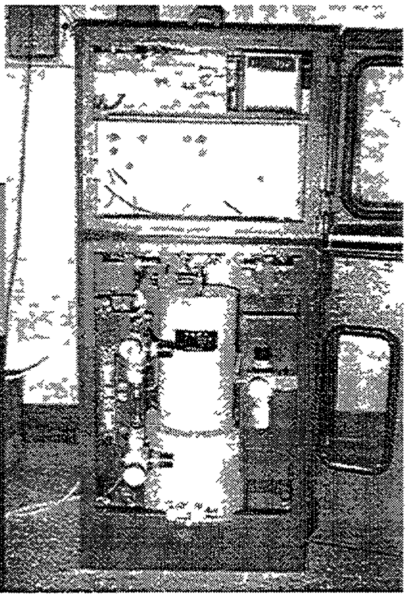


FIG 3 Assemble and calibration outside cabinet *FIG 4 Assemble and calibration*

3. 5. The key behaviour of the device

The key behaviour of this device are:

- The sensitivity of the γ system;
- The tightness, and attraction ability of the gas system;
- The calibration parameters of the systems in the device;
- The indication and record functions of the device.

3. 6. The approach to the goal

The subject was successfully completed step by step as following:

- The investigation done by consulting the information about existing sipping device worldwide from the experts and literature;
- The option selection based on the investigation,
- The engineering work;
- The procurement of the standard parts within the vendor catalog (Fig 1);
- The manufactory of the special parts (Fig 2);
- The Assembling and calibration of individual system outside the cabinet separately (Fig 3);
- The assembling and calibration of whole device inside the cabinet (Fig 4);
- The acceptance of the key behaviour.

3.7. The Performance of the device

There are satisfactory functions in the device e.g. easy operation, indication in the instrumentation, chart record and digital indication alarm etc. The calibration parameters are as they should be.

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XA9848404

THE STATUS, NECESSITY AND FEASIBILITY OF POOLSIDE INSPECTION, REPAIR AND RECONSTITUTION OF WR FUEL ELEMENTS IN CHINA

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Abstract

This paper mainly describes the status, necessity and feasibility of poolside inspection, repair and reconstitution of WR fuel elements in China. According to the status, some proposals about conducting poolside inspection, repair and reconstitution program are given in this paper.

1. THE STATUS OF NUCLEAR POWER PLANT AND ITS DEVELOPING PLAN

There are two nuclear power plants being operated now in China. They are Da Ya Bay Nuclear Power Plant (2x900MW) and Qin Shan Nuclear Power Plant (1x300MW). During the ninth five year project, four other nuclear power plants (eight reactors) will be built around the sea area in China. It is estimated that they will have been built and put into operation from 2000 to 2005. These are Qin Shan 2-phase (2x600MW), Da Ya Bay 2-phase (2x100Mw), Qin Shan 3-phase (2x700MW) and Lian Yun Harbor (2x100MW). Besides those mentioned above, many provinces (such as Zhejiang, Guangdong, Jiangsu, Shandong, Liaoning, Jiangxi, Fujian, Anhui, Gansu, Hunan, etc.) are planning to build new nuclear power plants at the beginning of the 21th century. Thus, China will become one of the fastest countries in developing nuclear power plants.

2. THE STATUS OF POOLSIDE INSPECTION, REPAIR AND RECONSTITUTION OF FUEL ELEMENTS

The sipping test device of fuel elements was installed in Da Ya Bay Nuclear Power Plant and in Qin Shan Nuclear Power Plant when these two plants were built. Seven failed fuel elements were detected in Da Ya Bay Nuclear Power Plant from its operation by using the sipping test device. Up to now, there are no failed fuel assemblies detected in Qin Shan Nuclear Power Plant. But we have got some experience in sipping tests. Concerning the research and development of the sipping test device, several institutes such as the Beijing Institute of Nuclear Engineering and the Shanghai Institute of Nuclear Engineering are being engaged in researching and developing the sipping test device. At present, the Beijing Institute of Nuclear Engineering has the ability to develop this device. An ultrasonic inspection device for detecting failed fuel rod in assembly is not equipped in the above mentioned nuclear power plant now. So, we are not engaged in the ultrasonic inspection of a failed or damaged fuel assembly yet. With regard to the repair and reconstitution of a failed or damaged fuel assembly, as the number of failed assemblies detected is few and we do not have the repair and reconstitution device, we are not engaged in the repair and reconstitution of failed assemblies yet.

3. THE NECESSITY OF POOLSIDE INSPECTION, REPAIR AND RECONSTITUTION OF THE FUEL ASSEMBLY

According to our country's develop plan of nuclear power, four nuclear power plants (total 8 nuclear reactors) will be built during the ninth five year project. Six nuclear power plants will be put into operation at the end of 2003. Estimated by the present failed probability of the fuel assemblies, that is 1% to 1.6%, damaged fuel assemblies will be increased as time goes on.

It is well known that radioactive gases fission products of fuel rod will escape from the cladding and get into the primary after the fuel assemblies have failed or are damaged, thus increasing the area of the power plant and the individual dose of the operating person. So, from the point of safety, when the radioactivity induced by failed fuel assembly reaches a given level, the power plant should reduce its power. In the case of seriousness the power plant should be shut down to replace the failed assemblies.

The economy of inspection, repair and reconstitution of a failed fuel assembly is considerable, especially to these operated only one cycle in the reactor. The value of a fresh fuel assembly is about 50,000US$. But that of inspection, repair and reconstitution of a failed fuel assembly is about 4,000 to 6,000US$. In addition , a failed fuel assembly stored in the spent fuel pool will increase the radioactive level of the surrounding area and the individual dose of the operating person. The direct processing if failed fuel assembly will cause waste of nuclear fuel resource. So, economy and safety of nuclear fuel resources are the direct reasons for inspection and repair of failed fuel assemblies.

4. THE FEASIBILITY OF POOLSIDE INSPECTION, REPAIR AND RECONSTITUTION OF FUEL ASSEMBLY

As mentioned above, the Beijing Institute of Nuclear Power Engineering has the ability to develop a sipping test device. The failed fuel rod detection and repair equipment has been purchased by China from France. We are planning to build a simulated repair and reconstitution station. So that the elementary hardware for inspection, repair and reconstitution of a fuel assembly could be reached.

There are three nondestructive testing centers under Nuclear Industry Corporation of China. Many specialists who have rich experience on nuclear fuel elements inspection are working there. Besides, there are several nuclear fuel elements factories. They also have a great number of specialists and scientists on manufacture and assemblage for various kinds of fuel elements. These specialists could be considered as solid foundation for fuel elements poolside inspection, repair and reconstitution.

5. SUGGESTION

According to the present condition and the 9th five year project in China, equipment for poolside inspection, repair and reconstitution is mainly imported from foreign countries. But at the same time, this kind of inspection and repair equipment is also under development with the cooperation of several big institutes in China.

As in foreign countries, special training for inspectors should be performed when an organization for fuel elements poolside inspection, repairing and reconstitution is established. These inspectors could only work when they have had the necessary certification and they will take charge of poolside inspection, repair and reconstitution for all the nuclear power plants.

6. CONCLUSION

According to the plan for nuclear power development in our country and the present nuclear power developing standard, it can be seen that fuel elements inspection, repair and reconstitution is very necessary and also possible in China.

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