Practical Experience in the Application of Quality Control in Water-Reactor Fuel Fabrication

IAEA-Seminar held at Karlsruhe on 12-16 March 1984

Editor: D. Vollath
Institut für Material- und Festkörperforschung

Kernforschungszentrum Karlsruhe

PRACTICAL EXPERIENCE IN THE APPLICATION OF QUALITY

CONTROL IN WATER-REACTOR FUEL FABRICATION

IAEA-Seminar held at

Karlsruhe on 12-16 March 1984

Editor: D. Vollath

F O R E W O R D

Highly industrialized countries have gained vast experience in
manufacturing water reactor fuel. Manufacturing is followed by
a stringent system of quality assurance and quality control. The
Seminar on Practical Experience in the Application of Quality
Control in Water-Reactor Fuel Fabrication provided a forum for an
exchange of information on methods and systems of quality assurance
and quality control for reactor fuel. In addition, many developing
countries which have started or intend to set up a nuclear fuel in-
dustry are interested in the application of quality assurance and
quality control.

This meeting has been preceded by two different series of con-
ferences: the IAEA meetings 1976 in Oslo, 1978 in Prague and 1979
in Buenos Aires, and the Karlsruhe meetings on Characterization and
Quality Control of Nuclear Fuel held in 1978 and 1981.

Quality control and quality assurance has many different facets.
Unlike the purely technical aspects, covered by the Karlsruhe con-
ference series, the IAEA meetings always relate to a wider field of
topics. They include governmental regulations and codes for practi-
cal quality assurance.

This volume contains the papers presented at the seminar and a
record of the discussions.

I wish to thank the IAEA for having arranged this meeting and the
government of the Federal Repuclic of Germany for hosting it. The
generous financial support granted by the Kernforschungszentrum
Karlsruhe to the conference and to publication of the proceedings
is gratefully acknowledged.

<div align="center">D. Vollath</div>

CONTENTS

PHYSICAL METHODS FOR ANALYSES

Evaluation of the seminar

Supplement to Session I
  (Paper submitted after deadline)

Principles governing the organization of quality control
  in the fabrication of water reactor fuel elements
  (IAEA-SR-102/59)

OPENING SESSION

Chairman: D. Vollath
Kernforschungszentrum, Karlsruhe

ECONOMIC CONSEQUENCES OF QA AND QC IN FUEL AND FUEL ASSEMBLY PRODUCTION

A. A. Strasser

The S. M. Stoller Corporation
1250 Broadway
New York, N.Y. 10001

ABSTRACT

The planning of quality control and quality assurance programs for fuel fabrication must balance the cost of the programs, their effectiveness, and the economic consequences of failure to meet the product specifications. The cost of fuel failures can be very high in comparison to the cost of quality control, and this provides considerable economic justification for increasing the level of quality control if its effectiveness in reducing failure potential can be demonstrated. Typical costs and examples are discussed.

1. Consequences of Failures in the Quality Control System

The detailed planning of the quality control (QC) and quality assurance (QA) programs for an industrial product must balance the cost of the quality programs, their effectiveness, and the consequences of failure to meet the product specifications. The intent of the QC and QA programs for fuel and fuel assembly production is to provide a high level of assurance that these products will meet their specifications and in turn perform reliably at their design conditions. The intent is that this be achieved at an economically competitive cost, both in the fuels industry and the power generation industry.

Some design features have a minor effect on performance, and for this reason the extent of related quality control, i.e. cost, can be relatively low. Other design features, such as the pellet hydrogen content, are critical to reliable performance, and for these a higher cost of quality control can be justified.

The QC plan selected for each design parameter is based on a number of considerations:

- The relationship of fuel performance statistics to the design parameters
- Desired confidence level for meeting the specification
- Statistics of the fabrication process in meeting the specification
- Subjective engineering judgement

A failure of the production QC plan occurs when an off-specification design feature passes the plan and is accepted as part of the final product. A failure of the design QC plan occurs when the specification itself is inadequate. This paper discusses the economic consequences of QC failures in fabrication process control, process inspection, and final inspection. The economic consequences of design QC failures, however, are similar.

Three different limit levels are used in the design and fabrication of fuel. The process control limits, the tightest level, are held by the fabricator below the second level, the the specification limits. These in turn are below the failure threshold, the third level. A well known example of the relationship of these limits related to the moisture content of UO_2 is shown in Figure 1. The margin between these three limits is in recognition that variations and uncertainties exist in design procedures, materials properties, fabrication processes and predictions of performance requirements. If the failure threshold, or the relationship of the specification parameter to the failure threshold are uncertain, the margin between the two should be sufficiently large to take the uncertainty into account. Similarly, if the fabrication process produces a highly variable product, the margin between the process limit and the specification limit should be large to reduce the risk of accepting a deviate product. The costs associated with the maintenance of large margins is usually high, which provides an incentive to improve our knowledge of failure thresholds and to narrow the variability in fabrication processes.

The current state-of-the-art of fuel QC is sufficiently mature to shift the emphasis in today's QC programs toward controlling the extremes of the product quality distribution, and to reduce the small number of failures that do occur to an even lower level. The major concern, therefore, is to eliminate the extreme "tails" of the distribution, as shown in Figure 1. This type of graph should be viewed as applicable to all parameters critical to fuel performance, not just moisture content. The two approaches to eliminate the extremes are an increase in the quality control level to reject all components in this category, and/or to improve process control, as was done in this case by improving drying procedures. In most cases, increasing QC alone is not economical.

The consequences of QC failures can range from essentially nil, to the failure of the fuel assembly and release of radioactivity. If a product feature falls outside the specification limit but below the failure threshold, the consequences may be negligible. If several product features are off-specification but below the failure threshold, the evaluation of their potential effect on performance is more complex, because the interaction of effects may not be immediately obvious, and must be taken into account.

An off-specification product that exceeds the failure threshold is likely to result in economic losses. The failure of a fuel assembly to meet performance requirements will result in a variety of economic penalties and will be borne by a variety of organizations, and even individuals. Figure 2 presents a chart of how the financial losses are transmitted and eventually paid for.

The costs of fuel failures are compensated directly by the fuel vendor and the utility. The division of responsibilities for payment depends partly on the contractual relationship between the two organizations. In the majority of the cases the greater financial burden falls on the utility. The publicly-owned utility must recover these costs either from the customers it serves, in the form of increased rates for electricity consumption, or from its owners, the stockholders, in the form of decreased profit distributions. A state-owned utility recovers its increased costs through increased government taxation or also directly via increased rates charged to the consumers of the electricity.

The vendor has less options for cost recovery. The costs related to the fuel failures will increase his general overhead. The vendor then has the choice of recovering the costs by decreasing profits to stockholders or increasing the price of the fuel fabrication to future customers, an action that will make his fuel less competitive.

2. Fuel Failures and Their Cost Components

A definition of fuel "failure" is important to a discussion of its economic consequences. Generally, a failure is thought of as a clad breach that releases fission products to the coolant. While this is one failure mechanism, and certainly the most costly one, there are others that can have significant economic impact. In the many years that the S.M. Stoller Corp. has assisted utilities in negotiating fuels contracts, we have developed a number of definitions to suit specific situations. A typical definition is given in Table 1.

The costs of fuel failures are reflected primarily in the loss of electrical power as discussed later. Additional costs are represented by a large number of services required for the investigation and correction of the problem and the subsequent disposal of the fuel.

Fuel failures can be divided into three generic categories, each having a somewhat different set of cost components, as outlined below. The potentially highest cost components are indicated with an asterisk.

a. Repairable failure, no fission product release

Examples: Differential fuel rod growth

Spacer damage (moderate)

Failed hold down spring

Cost components: Pool examination

Hot cell examination (optional)

Engineering evaluation

Reports to regulatory agencies

Fuel repairs and reconstitution in the pool

Radioactive waste disposal

*Added outage time (lost power)

Development of engineering remedy by vendor

b. Non-repairable failure, no fission product release

Examples: Fuel rod bowing (potential power derate)

BWR channel corrosion, bulging, bowing

Power peaking due to improper quantity of U^{235} or burnable absorber

Cost components: Same as Item "a" plus

*Operation at derated power (lost power)

c. Non-repairable failure, with fission product release

Examples: Hydriding

Pellet-clad interactions

Fuel rod end plug weld failures

Clad oxidation, corrosion

Cost components: Same as Items "a," "b," plus

*Plant shutdown (lost power)

*Underburned U^{235} (lost energy)

Increased plant maintenance cost

Increased personnel exposure

Cost of new reload calculation

Non-optimal reloading

Decrease in plant maneuverability

Fuel storage, shipment, and reprocessing penalties

The fuel failure cost components are discussed below. Typical figures representative of their costs in the USA are given.

Pool Examination

Assemblies with leaking fuel rods are identified by sipping. The procedures developed are quite rapid; a complete PWR core can be sipped in three days.

Determination of the failure cause requires, at a minimum, visual examination of the fuel assembly. In most cases this is done in the spent fuel pool with a periscope or TV camera. Visual examinations identify significant dimensional changes, the clad surface condition, clad failures, fretting, cracking, corrosion and other observable anomalies.

Several types of non-destructive, quantitative measurements are made relatively routinely; these include fuel rod bowing, spacing, and growth. An oxide thickness measurement method has also been used successfully for Zircaloy cladding. Ultrasonic inspection to identify the rods with clad breach are becoming routine as well. Less frequent measurements made primarily for research and development objectives include clad profilometry, eddy current and in some instances fission gas release measurement.

The costs for poolside examinations vary depending on the number of assemblies and fuel rods examined, the number of shifts worked (size of the crew), the degree to which the examination tools are automated, and potential delays in the examination while the crew is on site due to other reactor operations.

The personnel doing the work are usually the vendor's or that of a subcontractor specializing in examinations. The utility may do some of the examinations, but at a minimum they must supply personnel to help move the fuel, set up the equipment in the pool, and provide a health physicist. Other utility engineering staff often participate, and these are also part of the fuel failure cost.

Typical pool examination costs are summarized in Table 2. The majority of the examinations are in the range of $200,000-$700,000.

Hot Cell Examination

The pool examination is insufficient in some instances to provide the information needed to determine the cause of failure. The destructive examination in

a hot cell permits detailed examinations typical of non-radioactive laboratories such as metallography, chemical analyses, x-ray diffraction, and mechanical and physical property testing. The important fission gas release measurements are generally made in a hot cell, although some have been made in a pool. The puncture in a pool requires mechanical resealing, not acceptable to every utility.

The visual examination can be made with considerably greater clarity and resolution in a hot cell than in the pool, and the flexibility for disassembly and maneuvering of parts is also greater. Some of the non-destructive examinations done in the pool are also repeated in the hot cell where they can be made with greater accuracy.

The cost range of typical hot cell examinations involving fuel is $300,000-$500,000. Their cost can be higher, but unlikely to be lower, unless shipping and examination are restricted to non-fueled components.

Engineering Evaluation

The cause of the failures is usually evaluated by the vendor and independently by the utility or its consultants. Sensitive commercial issues are in question: was the cause a failure in the vendor's fabrication or design QC system? or was it due to utility operating methods incompatible with the fuel performance limits?

The data from the pool and hot cell examinations must be reduced, evaluated, and interpreted. In many cases the rod or assembly power history must be calculated to determine the conditions the fuel was exposed to. The vendor will search his QC records looking for unusual occurrences that may have been recorded during fabrication. The typical cost of such an evaluation is in the range of $30,000-$100,000.

Reports to Regulatory Agencies

Fuel failures and proposed remedial actions are reported to the regulatory agencies. Depending on the severity of the case, additional meetings, analyses, and tests may be required. The cost of such interchanges can range between $10,000-$100,000.

Cost of New Reload Calculation
Non-Optimal Reload
Decrease in Plant Maneuverability

A new reload calculation and license are required for the cycle subsequent to the one the failures occurred in, if a significant number of fuel rods are replaced or several failed assemblies are discharged. The cost can range from $50,000-$200,000 depending on the complexity of the job.

The compatibility of the desired fuel cycles with the now, hybrid reload will probably not be optimal. In order to meet cycle energy and shutdown window requirements, a greater than optimum amount of U^{235} and number of assemblies will probably have to be used. The cost of a non-optimal reload will depend on the degree of mismatch, and can range from very small to large amounts. Unit costs in terms of fabrication and U^{235} losses are given in connection with the discussion of "Underburned U^{235}".

Previously burned, or fresh fuel assemblies will be used to replace failed ones. While they will be chosen so as to limit their impact on thermal margins, their presence may reduce plant operating flexibility in some cases. Again, the costs may range from small to large and will depend on the nature of the specific case.

Underburned U^{235}

The discharge of fuel prior to reaching its goal burnup represents a loss in energy (energy generated per unit weight of U_3O_8). This represents one of the largest cost components in the economics of fuel failures. The actual amount of the loss will depend on the burnup of the fuel at the time of its premature discharge. A fuel failure near the end of its life will cost significantly less than one in its first cycle. QC failures tend to occur early in life.

The loss in U^{235} value and fabrication cost due to failure of one PWR and one BWR assembly as a function of burnup are given in Figure 3. The U^{235} is paid for by the utility, and the fabrication cost, burnup pro-rated, by the vendor. Assuming an assembly average burnup of 10,000 MWD/MTU, the loss of an entire assembly is $150,000 (BWR) and $420,000 (PWR). Assuming only a fraction of the rods failed, and the assemblies can be reconstituted, the loss will be less. If 10% of the rods failed in 10 assemblies, the cost will be approximately the same as one assembly.

Fuel Repair and Reconstitution

The fuel assemblies that can be repaired, are disassembled and reconstituted with new or repaired components in the spent fuel pool. The process often requires the design, fabrication, and testing of specialized tools and inspection gages; other costs include those for the new components, the labor of the repair process itself at the site, and the re-inspection of the assembly. The repair process may necessitate an extension to the outage, which will result in a power generation revenue loss, discussed subsequently. Exclusive of the power loss, the repair cost is estimated to be in the $100,000-$300,000 range.

Radioactive Waste Processing and Disposal

Fuel failures will increase the concentration of radioactive isotopes in the coolant, and consequently contaminate areas that the coolant contacts. The most notable concentration of radioactive species will occur in the demineralizers which will produce resin and resin-related wastes contaminated with fission products. Additional radioactive wastes will be generated during storage in the spent fuel pool water during handling and during decontamination of the tools used for handling and inspection. The additional wastes will have to be processed, to convert the dilute wastes to a more concentrated form for disposal.

The additional waste processing and disposal cost will depend on the amount of radioactive species released from the fuel. A significant activity level can increase the costs by $100,000 or more.

Increased Plant Maintenance
    Increased Personnel Exposure

The increased level of radioactivity in various parts of the plant system will increase the cost of maintenance. Increased complexity of the work, longer time required for the same operation, and additional decontamination work are some of the causes for the increase. The potential increase of noble gas activity in the containment may delay or limit access, increasing the reactor down time.

The rate of personnel exposure will increase in high radiation areas; in extreme cases, additional personnel will be required to complete the work, as some workers reach their exposure limits. Since radioactivity from corrosion products are usually much higher than from fission products, the overall effect - except for extreme cases - is small. Nevertheless, the cost estimate is difficult to make, particularly as it relates to the monetary evaluation of added personnel exposure.

Fuel Storage and Shipment

Reprocessing

The storage and shipment of severely failed fuel may require a special container to minimize contamination of the pool, the containment building, and the shipping cask. In addition, the costs of decontaminating the pool and shipping cask have to be considered. The added storage cost for one failed assembly, including special containers, is $10,000-$20,000 and the shipping cost is $60,000-$70,000. If the reprocessor accepts failed fuel, a surcharge will probably be applied. The total cost of this category is probably around $100,000.

Plant Derate or Shutdown

As the plant's Technical Specification coolant activity or the plant's internal activity limits are approached, the reactor power must be reduced or derated to stay within specifications. The plant may be shut down if the coolant activities reach very high levels. Derates can also occur if the power peaking factors reach technical specification limits, or fuel rod bow exceeds regulatory limits.

All of these events, whether they are a derate, shutdown, or an extended outage, result in a loss of plant capacity factor, or power, potentially the most costly aspect of fuel failures. The power that would have been generated by the plant has to be replaced from another source.

Replacement cost of the lost nuclear power made up from an oil or coal fired plant will vary from one geographic area to another. A recent survey of replacement power costs of US utilities made by the S. M. Stoller Corporation, showed that the net increase in power cost, i.e. gross replacement power cost minus the value of unused nuclear fuel, is 20-40 mils/Kwh[*] for coal and 40-60 mils/Kwh for oil. The cost of replacing nuclear power will also depend on the size of the plant, and its normally achieved capacity factor. Figure 4 shows the daily cost of make up power if it is obtained from a coal fired plant, and Figure 5 if it is from an oil fired plant. In each case the costs are given as a function of plant size and net replacement cost at 70% capacity factor. The sensitivity to a capacity factor range of 60-80% is shown for the median coal and oil replacement costs. For a 1000 MWe plant at 70% capacity factor the cost can range from $250,000 to $1,000,000 per day. At these rates a few days of shutdown could equal and exceed all other fuel failure related costs. The replacement power costs are to the utility's account and are passed on as shown in Figure 2.

---

[*]1 mill/Kwh = 0.1¢/Kwh

3.   Relationship of Fabrication QC to Fuel Performance

Determination of the adequate QC level to assure good fuel performance is the key to a good QC plan. The majority of the fuel QC practices are standard and accepted by the industry. Three examples of controversial situations are discussed next which present more difficult decisions. These deal with cases that represent the "tail" of the distributions for Zircaloy irradiation growth, pellet chips in fuel rod gaps, and Zircaloy corrosion. How can infrequent deviations to these specifications be controlled? and what is the justifiable cost of control?

Irradiation Growth

During the past years several instances of unusual irradiation induced Zircaloy growth were noted in PWRs irradiated up to and beyond their design limits. Similarly in BWRs, there were indications that rods in the upper bound of the Zircaloy growth rate distribution reached the limit of the space allotted by the hold down spring between the fuel rod shoulder and the upper tie plate at high exposures. The hold down springs were observed to be "solid", a good indication for growth. These instances could have resulted in interference bowing. Photographs of the growth distribution are shown in Figure 6.

There is a need to determine the growth rate distribution of Zircaloy clad, and reject high growth rate lots. Alternately, a large margin must be allowed in the "shoulder gap" to take into account the variability in properties. An important variable that determines the growth rate is the crystallographic texture. Texture analyses are used for process qualification but not made routinely for quality control. Texture analyses cost $500-$1,000 each, and since there may be up to 200 lots in a reload, the cost of this QC step would be very expensive: $100,000-$200,000. One sample per lot may not be adequate, given the unique behavior of one tube shown in Figure 6.

Improved process quality control to reduce variability in properties may be an alternative, less costly way to resolve the problem. Since irradiation growth also depends on the microstructure produced in the clad tube by a combination of the final reduction and heat treating steps, the temperature gradients in the heat treating furnace could cause variability in subsequent growth rates. Tubes, or portions of tubes, in cooler than nominal temperature zones would grow at higher rates than those at nominal or higher temperatures. Process control could be improved by determining the temperature profile in the furnace, improving temperature homogeneity and monitoring it to assure that it is maintained. This approach is considerably less costly than the additional inspection.

Pellet Chips in Fuel Rods

Pellet chips can locate in the fuel-clad gap of the fuel rod during fabrication, shipping and handling. Excessive amounts will accelerate pellet-clad mechanical interactions (PCIs), potentially leading to clad failure. An extreme case of this type of PCI failure is shown in Figure 7, a stainless steel clad PWR fuel rod. The fuel-clad gap in UO_2-stainless steel rods is smaller than in Zircaloy rods, increasing the difficulty of loading pellets in long tubes. The loading operation lead to chipping, in this case, and the subsequent operation resulted in PCI failures. The chips were a major, though not the only, cause of the PCI. Similar occurrences have been suspected in Zircaloy clad rods, though not shown as clearly as in this figure.

Radiography of the fuel rods is a sensitive method by which chips can be detected in the gap, or missing pieces can be detected in pellets. Gamma scanning is a less sensitive, but less costly, detection method. A full length radiographic inspection of a group of 20 rods takes about three manhours and $250 in materials, a total cost of no more than $400. For a reload of about 10,000 rods the cost would be $200,000. This would include radiography of the welds as well. Is the cost of the complete radiography of fuel rods justified or should it be limited to the welded ends?

Alternative to additional QC is again improved process control, that is, the control of chipping during fabrication. Basic to such control is the manufacture of pellets with good mechanical integrity, good pellet grinding procedures, good pellet loading and fuel rod handling practice.

The relevance of inspection for chips has often been questioned. Since the pellets crack and form chips in the reactor during irradiation, in spite of the care taken during fabrication, is such inspection meaningful?

Nodular Corrosion

Nodular corrosion of Zircaloy in BWRs has contributed to the premature removal of Zircaloy 4 channels as well as heavy crud formation and subsequent failure of Zircaloy 2 cladding. The quality of the Zircaloy has a significant effect on the rate of nodular corrosion, although water chemistry also plays a significant role particularly in the case of the cladding.

The Zircaloy characteristics responsible for good or poor resistance to nodular corrosion are not clearly understood; for this reason, it is difficult to establish effective quality control to separate good lots of tubing from bad ones. The $400^\circ C$ ASTM autoclave test used for many years to identify poor corrosion resistance lots is too insensitive for nodular corrosion. In-reactor nodular corrosion occurs at normal clad surface temperatures of $360^\circ C$; however, ex-reactor autoclave tests must be made at $>400^\circ C$ to reproduce nodular corrosion. An accelerated autoclave test at $500^\circ C$ has been used as quality control to detect lots with unusually poor, as well as unusually good, corrosion resistance. The reproducibility of the test from laboratory to laboratory is in need of improvement. The correlation of ex-reactor test results with in-reactor performance is claimed to be reasonable, but knowledge in this area is limited.

An ASTM committee, of which SMSC is a member, is attempting to evaluate the high temperature autoclave test and improve its reproducibility. Variables being investigated in a round robin test of "standard" process samples are: temperature, pressure, time, sample area to autoclave volume ratio, refreshed vs. static environment, and the effect of oxygen. If this test were to give reliable results, its cost would be low. The equipment is about $50,000 and the capacity of the autoclave is sufficient for 10-20 lots. The inspection cost per reload of 100-200 lots would be about $40,000.

As in the other cases discussed, appropriate process control could reduce or eliminate the need for autoclave testing. Quenching the clad tube from the β or $\alpha+\beta$ phase during the final fabrication steps homogenizes the structure sufficiently to provide consistently good resistance to nodular corrosion. Such processes are under development and evaluation, and some are offered commercially.

In-reactor confirmation of the benefits of clad selection via autoclave testing and the improvement of clad corrosion resistance by process control are shown in Figure 8. The picture represents a BWR assembly that has been irradiated for 2 cycles in Ringhals 1. The effect of water chemistry is eliminated by testing the clads in the same assembly. The corner rods with nodular corrosion represent standard process cladding. Considerable improvement in nodular corrosion resistance is shown by the clads in the center of the assembly, which were made by special proprietary processes. The processes themselves were evaluated and screened successfully by the high temperature autoclave test.

4.   Comparison of Fuel Failure and QC Costs

The costs of fuel failures are compared to QC costs in Table 3.

The total QC costs for fabricated fuel are in the 20-30% range. Typical reloads for an 1100 MWe PWR and BWR have 24,000 and 36,000 Kg U respectively. At $200/Kg U fabrication cost, the total costs per reload are $4.8x10^6 and $7x10^6. An average 25% for QC corresponds to $1.2x10^6 and $1.75x10^6 for the total QC costs of a PWR and BWR reload. The examples of QC discussed previously are $100,000-$200,000 for texture measurements, $200,000 for radiography, and $40,000 for corrosion testing.

In comparison, the range of fuel failure costs that may accrue are about $2-6x10^6 dollars, a significantly higher range than the total QC costs and even higher than the special QC cases discussed. The leverage favoring the benefit of QC is considerable.

The cost of the QC examples can be easily justified economically, if their effectiveness in reducing failure potential can be demonstrated. The examples also indicate that process control may be used to decrease process variability and the fraction of marginal or poor product resulting from the process. Economic justification of this approach can often be made more readily than the increased QC level. Even though additional QC is justified by fuel failure costs, ineffective QC will raise fabrication costs to levels that are not competitive.

The development of a QC program that recognizes the economic consequences of failure and yet can successfully compete in the industrial environment, is a difficult task. An economically justified, well integrated, QC program must be based on an optimal combination of process control and inspection level, effectiveness of the inspection process in eliminating off specification product, recognition of the relationship between each product parameter and fuel performance, and evaluation of the cost of failure to meet performance requirements.

TABLE 1

TYPICAL DEFINITION FOR FUEL MECHANICAL FAILURE

One or more Fuel Assemblies shall be deemed to have Mechanically Failed when:

a. The rate of fission product activity release to the reactor coolant exceeds a mutually agreed upon value.

b. A condition exists which is generally accepted in the nuclear industry as evidence of Mechanical Failure, such as fission product leakage as determined by sipping.

c. A visible Fuel Rod cladding penetration is detected utilizing such instruments as periscopes, video cameras, boroscopes, ndt equipment, etc.; or

d. A Fuel Assembly (including all necessary fittings) exhibits any significant departure from design configurations or from normal operating conditions, including gross physical distortion of the Fuel Rods or physical distortion or failure of the Fuel Assembly control rod guide tubes, nozzles or spacer girds to an extent which renders the Fuel Assembly unsuitable for its intended purpose; or

e. Any conditions exist which by mutual agreement would be expected to result in a Mechanical Failure before the next scheduled refueling shutdown based on experience of other Seller-supplied Fuel Assemblies of a similar type operated under similar conditions.

TABLE 2

TYPICAL FUEL EXAMINATION COSTS IN THE SPENT FUEL POOL

(Exclusive of Equipment Costs)

Normal Examinations

Sipping (1/3 to whole core)	$100,000 -	$200,000
Visual examination (25%-40% of peripheral rods)	50,000 -	100,000
Ultrasonic for failed rods (15-60 assys.)	150,000 -	200,000
Rod bow, spacing, growth	100,000 -	150,000
Utility manpower	20,000 -	50,000
	$420,000 -	$700,000

Less Frequent Examinations (for 20-30 rods)

Profilometry, eddy current, length	$ 50,000 -	$150,000
Fission gas release	100,000 -	200,000
Gamma scanning	100,000 -	150,000
	$250,000 -	$500,000

Total range: $200,000 - $1,200,000

TABLE 3
COMPARISON OF FUEL FAILURE AND QUALITY CONTROL COSTS

Fuel Failure Costs

Pool examination	$200,000	-	$1,200,000
Hot cell examination	300,000	-	500,000
Engineering evaluation	30,000	-	100,000
Reports to regulatory agencies	10,000	-	100,000
New reload calculation	50,000	-	200,000
Non-optimal reload	?		?
Decreased plant flexibility	?		?
Unburned U^{235} - lost power	200,000	-	600,000
Fuel repair and reconstitution	100,000	-	300,000
Radioactive waste processing and disposal	100,000	-	?
Increased plant maintenance and increased personnel exposure	high		high
Fuel storage, shipment, reprocessing	100,000	-	?
Plant derate or shutdown - lost power for 3 days	750,000	-	$3,000,000
	$1,840,000	-	$6,100,000+

QC Costs

Zircaloy growth - texture analyses	$ 100,000	-	$200,000
Pellet chips in fuel rods - radiography	200,000		
Zircaloy nodular corrosion - high temperature autoclave test	40,000		
Total fuel QC costs for one 1100 MWe reload (25% of fab cost)	$1,250,000 (PWR) - $1,750,000 (BWR)		
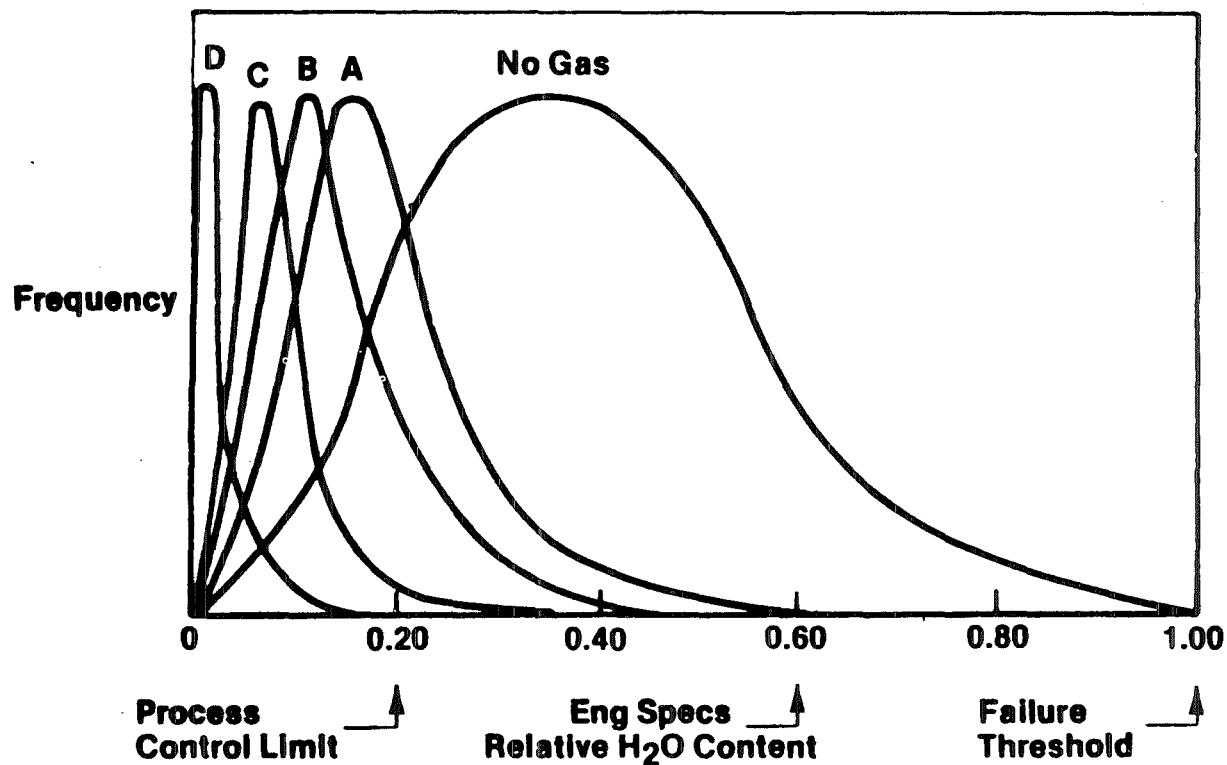
Fig. 1   UO₂ Batch Moisture Distributions

FIGURE 2   RECOVERY OF FUEL FAILURE COSTS

COST OF FUEL FAILURE

UTILITY   ← WARRANTY PAYMENTS ←   VENDOR

INCREASE ELECTRICAL
ENERGY PRODUCTION COST

INCREASE OVERHEAD
EXPENSES

INCREASE COST
OF POWER TO
USERS

PRIVATE UTILITY
DECREASE PROFIT
TO STOCKHOLDERS

STATE UTILITY
INCREASE TAXES

DECREASE PROFIT
TO STOCKHOLDERS

INCREASE FUEL FAB.
PRICE TO NEXT
CUSTOMER

ENERGY VALUE REMAINING IN ASSEMBLY AT DISCHARGE ($X10^3$ 1934 $)

DISCHAGE BURNUP (GWD/TU)

UNIT COSTS

ORE	= 25	$/LB U_3O_8
CONV.	= 6	$/KGU
SWU	= 135	$/SWU
FAB.	= 200	$/KGU

PWR

BWR

PWR FAB. COMPONENT
BWR

BWR BATCH AVERAGE DISCHAGE BURNUP

PWR BATCH AVERAGE DISCHAGE BURNUP

FIG. 3 ENERGY VALUE REMAINING IN PWR OR BWR ASSEMBLY AT
DISCHARGE (DUE TO FUEL FAILURE) VERSUS DISCHARGE BURNUP.
(REF.: W. FRANKS, S. M. STOLLER CORP.)
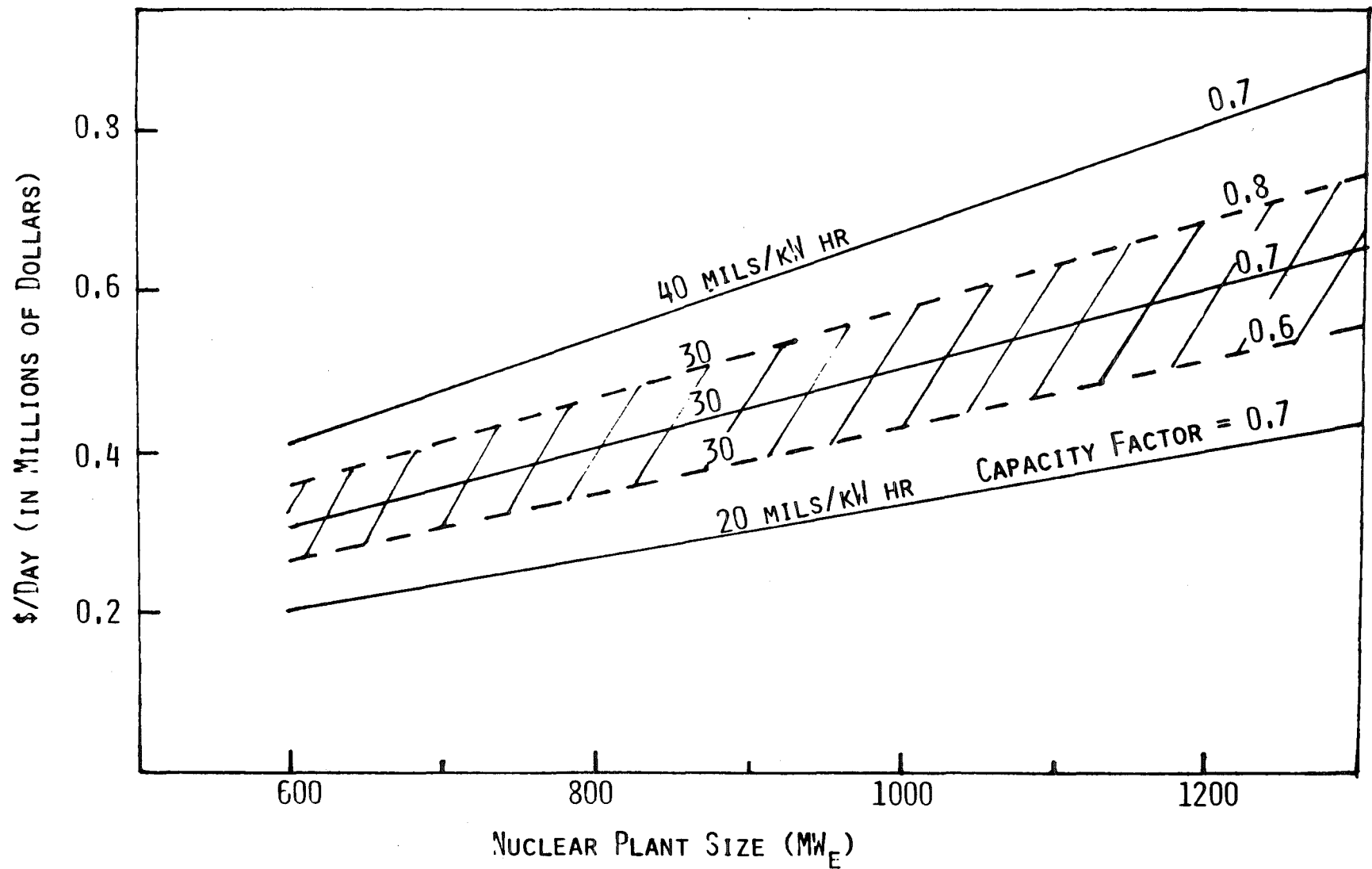
FIG. 4 REPLACEMENT POWER COSTS FROM A COAL FIRED PLANT.
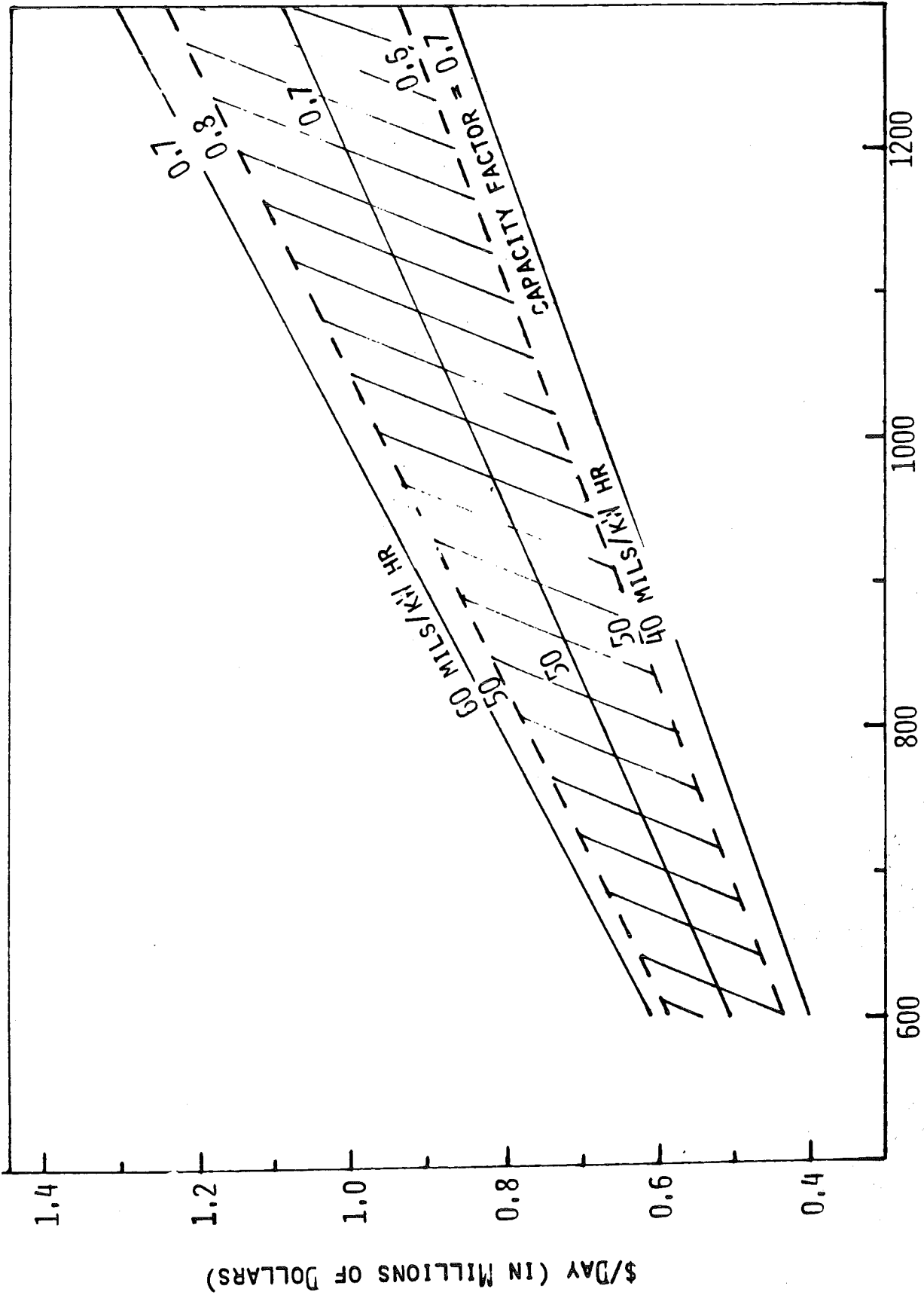
FIG. 5   REPLACEMENT POWER COSTS FROM AN OIL FIRED PLANT.
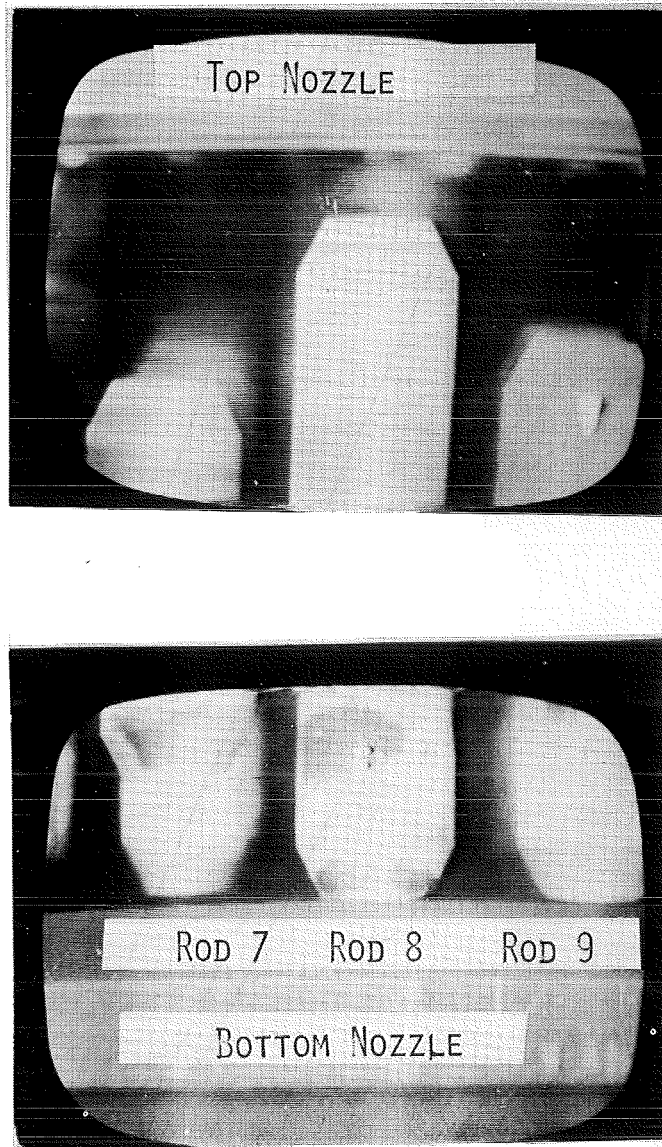
4 Cycle Exposure
46.5 GWD/MTU Assy. Ave. Burnup

FIGURE 6 Fuel Rod Growth in ZION PWR Assembly CE4 (Face 1)
(Ref.: R. Kaiser, J. Melehan, E. Roberts, H. Wilson.
"Primary Performance and Licensing Considerations for
Increasing PWR Fuel Discharge Burnup." IAEA Specialists
Meeting on High Burnup in Power Reactor Fuel, Mol, Belgium,
March 1981.)

Backup Ring   Cladding
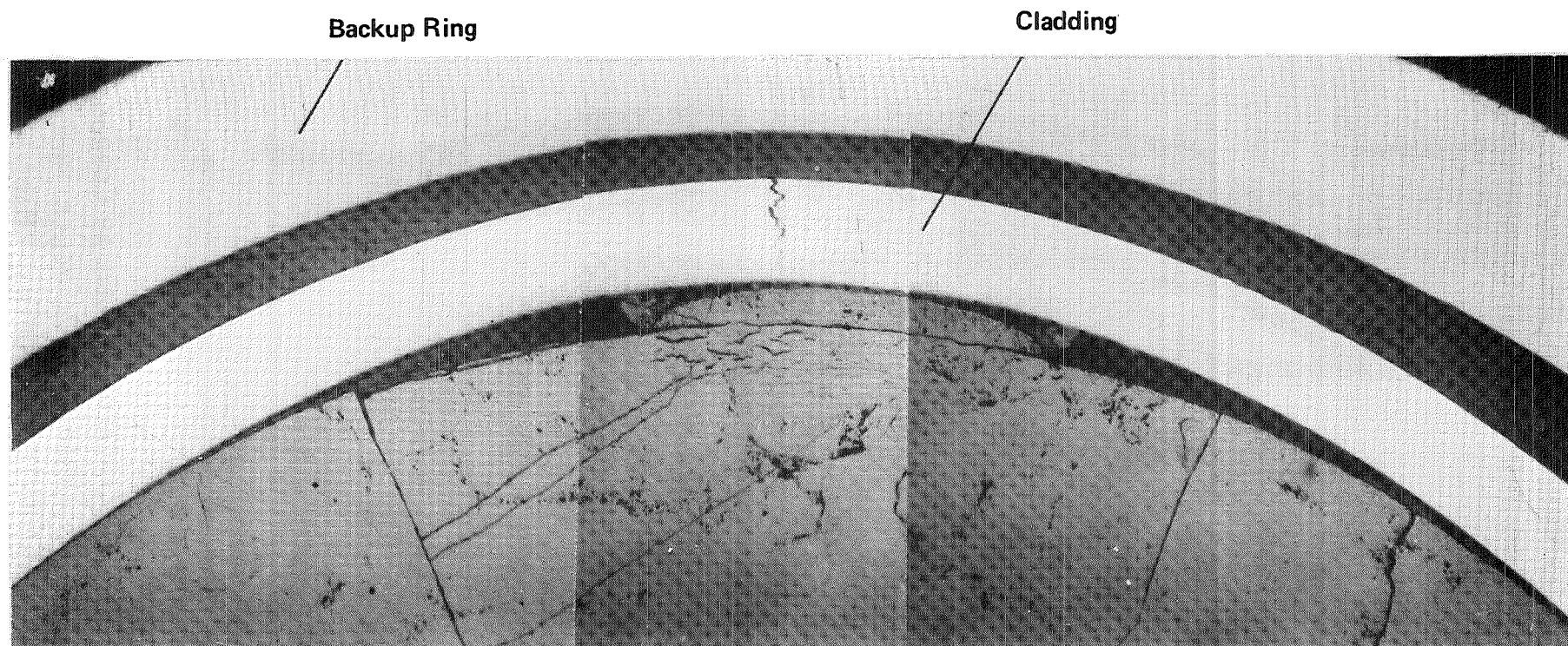
FIGURE 7   EFFECT OF PELLET CHIPS ON CLAD PERFORMANCE
            TYPE 304 STAINLESS CLAD UO_2 FUEL ROD IRRADIATED TO 38,400 MWD/MTU

                    REF.: V. PASUPATHI, R. KLINGENSMITH, "INVESTIGATION
                           OF STAINLESS STEEL CLAD FUEL ROD FAILURES AND
                           FUEL PERFORMANCE IN THE CONNECTICUT YANKEE REACTOR."
                           EPRI, NP-2119, NOVEMBER 1981.

PROCESS
EX-REACTOR
500°C, 24 HR.
AUTOCLAVE TEST }

SPECIAL

<100 MG/DM^2

STANDARD

100-500 MG/DM^2
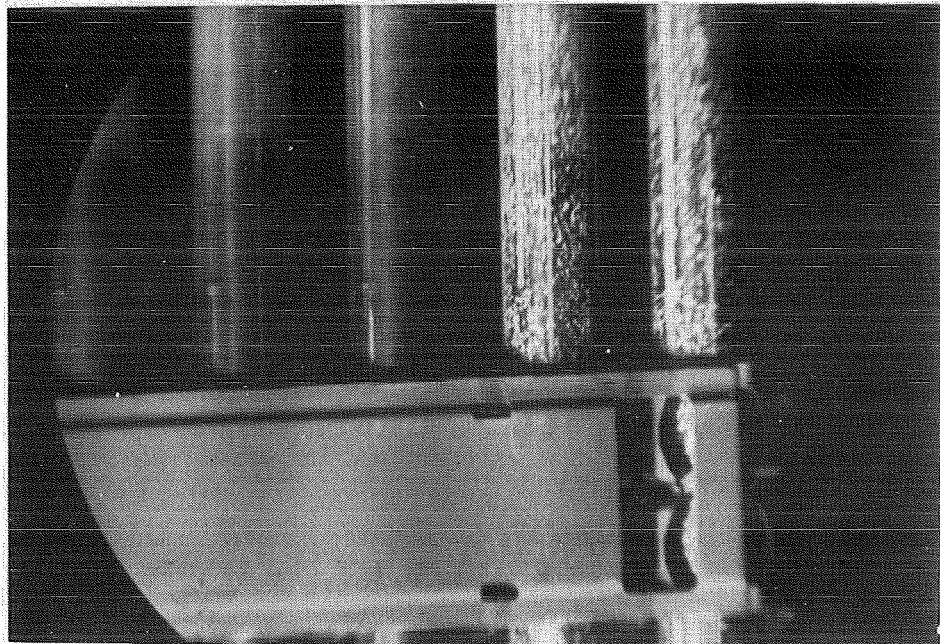
PROCESS
EX-REACTOR
500°C, 24 HR.
AUTOCLAVE TEST }

STANDARD

100-500 MG/DM^2

SPECIAL

<100 MG/DM^2
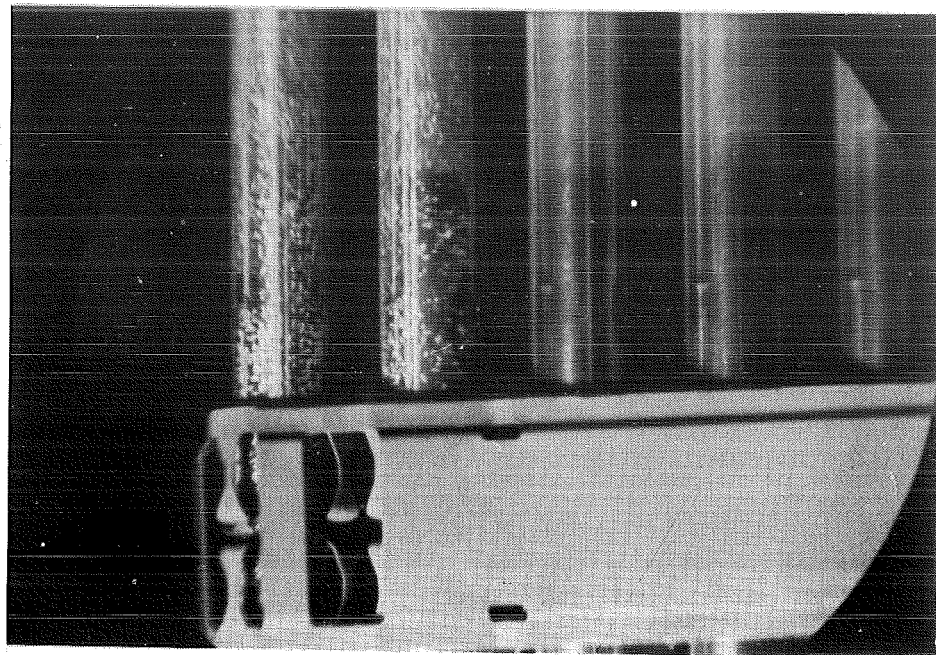
FIGURE 8   EFFECT OF PROCESSING ON ZIRCALOY-2 NODULAR CORROSION
ASEA-ATOM FUEL ASSEMBLY IRRADIATED 2 CYCLES IN RINGHALS 1 TO
10.6 MWD/KGU, WITH SANDVIK CLADDING MADE BY VARIOUS
PROCESSES

(PHOTO BY ASEA-ATOM)

DISCUSSION

R.S. RUSTAGI: a) How do you fix the fuel warranty cost? Is it decided at the time of placing the purchase order? How much is the cost? How is it decided that the failure occurred due to fabrication parameters going out of control?

b) In view of the current understanding that fuel failure costs are QC costs, can you please indicate what is the optimum fuel failure rate?

A. STRASSER: a) The warranties and the remedies are part of the fuel contracts and are negotiated at the same time as the remainder of the contract.

The cost remedy to the vendor, if the fuel fails, is usually the fraction of the fabrication cost equivalent of the unused, warranted burnup, as shown in Fig. 3.

The cause of the failure is decided by an Engineering Evaluation, as described on page 6. This is often a difficult task, and a clear cause cannot alsways be established, or the cause may be due to several factors. The remedy, if any, is then negotiated between the utility and the vendor.

b) The most economical failure rate is, of course zero. The economically acceptable number of failures must be determined on a plant by plant basis and will depend on the plant's internal activity limits, and regulatory activity limits: the status of the fuel supply, the fuel cycle design, the plant system design (such as cleanup capacity), the leak tightness of the plant systems, limitation to the plant storage system and the back end of the fuel cycle, and other factors not the least of which is the plant mangagement's operating philosophy.

K. BALARAMA MOORTHY: a) In order to minimize P.C.I., is there any thinking in the U.S.A. to use lubricated fuel tubing? If so, give details please.
b) Are there any trends in the U.S.A. to use any zirconium base alloys other than Zry2, to overcome nodular corrosion etc.

A. STRASSER: a) Zircaloy cladding coated on the ID with graphite has been developed by several vendors. It was originally thought the graphite would act as a lubricant between pellets and clad. Work

during the past years has shown that graphite does not act
as a lubricant, but can reduce stress corrsosion accelerated PCI
by acting as a fission product barrier or absorber.
b) Low level development efforts on other Zr alloys do exist, and
in fact Zr-Nb alloys with $\geq$ 2% Nb have shown better resistance to
nodular corrosion than zircaloy. However, then alloys have some
disadvantages such as greater difficulty in obtaining satisfactory
welds. The success of the new heat treating processes in reducing
Zircaloy 2 nodular corrosion has also reduced the incentive to change
alloys. The overwhelming reason for not introducing a new alloy for
large scale use is the high cost and difficulty to get it licensed.
For these reasons I would expect that in the U.S.A. we will continue
to use Zircaloy 2.

H. BAIRIOT: In evaluating the cost of failed fuel, the cost of
fuel assembly repair and of underburned U235 were added. Is this not
too pessimistic taking into account that most failed assemblies in-
corporate only one failed fuel rod and that the withdrawal of that rod
does usually not change the total energy produced by an assembly?

A. STRASSER: It is true that one failed rod alone will not
represent a significant loss, but rather "if a significant number of
fuel rods are replaced or several failed assemblies are discharged".
Our data base of identified LWR fuel failures indicates that most
failed fuel assemblies have more than one failed rod.

Seminar on Practical Experience in the Application of Quality Control
in Water-Reactor Fuel Fabrication, 12-16 March, Karlsruhe

TECHNOLOGY AND PERFORMANCE OF WATER REACTOR FUEL
(REVIEW OF IAEA ACTIVITIES)

I.L. Rybalchenko

International Atomic Energy Agency

P.O. Box 100

A-1400 Vienna

ABSTRACT

Nuclear power proved to be a reliable and economically advantageous
source of energy. Many countries are developing nuclear power
programmes. Development of nuclear fuel cycle technology is an important
part of each programme. The paper reviewes briefly some aspects of
nuclear fuel cycle developments in the world with emphasis on water
reactor fuel technology and performance and Quality Assurance and Quality
Control programmes. Strict QA and QC procedures have contributed
significantly to the high reliability of nuclear fuel. As more R & D
work is being done on improvements in fuel utilization and as more
manufacturers are entering the fuel suppliers group, additional efforts
are needed in standardization of fuel fabrication technology, fuel itself
and Quality Control programmes. All this could be achieved through
extensive cooperation on a multinational basis.

The IAEA plays an important role in international exchange of
information and cooperation. The paper gives a review of IAEA activities
and future programmes in this area.

1. Some aspects of nuclear fuel cycle developments in the world

Nuclear power cannot be developed without proper development of the whole nuclear fuel cycle technology and especially without reliable and effective technology of fuel fabrication. There is no common approach in the world to reactor type and nuclear fuel cycle strategy. In different countries different nuclear fuel cycle strategies were established because of various reasons but basically two types of nuclear power programmes were developed in a large commercial scale:

- Nuclear power programme based on natural uranium fuel with thermal neutron gas-cooled or water-cooled and graphite or heavy water moderated reactors;

- Nuclear power programme based on slightly enriched uranium fuel with thermal neutron water-water or water-graphite reactors.

Other programmes (such as fast breeder, high temperature etc.) are on a small demonstration scale and will not contribute significanly to nuclear power development at least during this century.

Both basic programmes and their slight modifications are well established now and many countries have a long, positive experience of their operation (more than 2666 reactor years). Some countries have nuclear share of electricity production up to 40%, (see Tables 1, 2). At the end of 1983, 313 power reactors were operational and 207 reactors were under construction. The total capacity of these plants is 380 GW(el) [1,2]. Most of these reactors are water-cooled reactors (PWR, BWR and LWGR).

The present IAEA forecasts for nuclear power development are 291 GWt by 1985, 376-420 GWt by 1990 and 580-850 GWt by the year 2000 which is much lower than previous forecasts by the IAEA and other organizations, (see Table 3).

For a relatively short period of development (about 40 years) nuclear industry has experienced both booms and depressions. During the past 20-30 years in a number of countries adequate nuclear fuel cycle technology was developed to meet requirements of the growing nuclear power programmes. This technology proved to be reliable, safe and economically feasible for many countries. In recent years, as the expectations of nuclear power growth have not materialized, projected demands for nuclear raw materials and fuel have decreased which led to their overproduction in some countries and a significant fall of the uranium market price, (see Fig. 1) [3].

In 1983 although the condition of the uranium industry continued to deteriorate in some countries, there has been some stabilization as the level of overproduction was reduced, requirements grew and uranium spot market price increased somewhat. Major cutbacks in exploration and production have occurred in several countries, perhaps below levels adequate to assure future supplies. A few projects related to new low cost deposits continued to be developed, especially in Canada and Australia. There is a continuing concern about long-term adequacy and economics of uranium, (see Fig. 2, 3) [4]. A number of developing countries continue to be interested in nuclear power development programmes which can be gauged by the large number of Member States' requests for technical assistance from the IAEA.

Uncertainties in the back-end of nuclear fuel cycle, have led to delays in reprocessing of spent fuel and, consequently, to an unexpectedly large accumulation of spent fuel assemblies and shortages of spent fuel capacities.

It is expected that steady growth of nuclear power in general will improve the situation of nuclear industry in many countries and the demand for nuclear materials and fuel cycle services will be stabilized.

The major part of the world's operating power reactors (about 90%) are water-cooled and have uranium dioxide as a fuel. Eleven countries were fabricating nuclear fuel for commercial use for quite a long time.

A number of additional states are developing manufacturing technology and conducting various R & D projects. The manufacturing processes employed are quite similar for most plants, although special requirements exist between the various types of fuel.

Practically all fuel assemblies of the existing power reactor types consist of UO_2 fuel rods in zirconium alloy cladding. The geometry of rods and fuel assemblies varies but physical and mechanical features are somewhat similar and different manufacturers may have a different organizational and technical approach to QA and QC procedures. As an example, a list of nuclear fabrication facilities is given in Table 4. Besides that, a number of developing countries are entering the suppliers group and are constructing their own nuclear fuel fabrication facilities (Brazil, Argentina, India) [5,6].

Experience with fuel fabrication shows very few fuel failures and continued improvement in fuel reliability. Problems of hydridization, UO_2 densification, fretting, and pellet-clad interaction have been studied in detail. Improvements have been made by changes to specifications, and improved Quality Control and fabrication techniques.

Strict QA and QC procedures have contributed significantly to the high reliability of today's fuel. Experience obtained over the past years, whereby, the present satisfactory stage of reliability of the fuel behaviour in operating conditions, has been reached. We can see how the efforts of utilities, industry and research to develop remedies against initial defects, have been largely successful. Average failure rates of the order of 0.01% failed rods per reactor year can be achieved for prevailing current mode of operation and current discharge burnups. R & D work is being done on improvements in fuel utilization (extended burnup), load following mode of operations, use of plutonium in thermal reactors, and the possibility of reconstruction of failed fuel. The trends to improve the fuel utilization in water-cooled reactors could be achieved by two parallel ways - change in the fuel management strategies and improvement of mechanical and metallurgical properties of the fuel. R & D efforts are being focused on such problems as - fission

gas release, water corrosion of Zirconium cladding, dimensional
stability, pellet cladding interaction, increased use of burnable
poisons. Most of these problems are not new, but prior to licensing, it
is necessary to check the behaviour of the fuel by experiments under
expected conditions. Computer models and codes in fuel behaviour
analysis proved to be useful for the development of the new designs.

The implementation of the improv ents do not present unsurmountable
difficulties but due to the nature of the work, i.e.: length of
irradiation time, necessity to check carefully the safety aspects, delays
to change the industrial procurements, the full benefit of improvements
in fuel utilization is not expected before the next decade.

Innovations in fuel technology would require additional or new QA
and QC measures which should also be proved to guarantee higher safety
standards for fuel operation.

A significant experience has also been gained in the back-end of
nuclear fuel cycle (storage, transport and reprocessing). At present, a
larger portion of irradiated nuclear fuel is being stored in water pools
at reactors. A smaller part of spent fuel is in storage pools at
reprocessing plants or in away-from-reactor storage sites. Only some of
the spent fuel has been reprocessed. Even with full operation of
reprocessing plants a large portion of spent fuel will not be reprocessed
for some time and will have to be stored. For example, through 1990 no
more than 6.500 MTU of oxide fuel can be reprocessed in Europe and an
inventory of some 10.800 MTU of unreprocessed fuel will accumulate by
that time. The, expansion of on-site storage and reprocessing
capacities, as well as, the provision of centralized storage facilities
should prevent spent fuel storage problems in the near future. But at
the same time, long-term storage of irradiated nuclear fuel requires
reliable behaviour of fuel (especially, fuel cladding integrity) even
after its discharge from the reactor. Although the environment at
storage is less hostile than in the reactor circuit, the residence time
in storage is significantly longer. It requires additional R & D work to

prove the reliability and safety of storage during extended period of time.

Steady growth of nuclear power in the world would much depend on reliable operation of nuclear fuels and continuation of international cooperation in this field would contribute to it significantly.

2. International cooperation in the area of nuclear fuel

Multinational cooperation in the nuclear field is being carried out successfully through the International Atomic Energy Agency. The activities in the field of nuclear fuel cycle are becoming more important as it is the only organization capable to collect and evaluate information and to provide support to Member States in the field of nuclear materials and fuel cycle technology on a large scale.

The main objectives and strategies of the Agency's activities in the area of Nuclear Fuel Cycle are to promote the exchange of information between Member States on technical, environmental and economic aspects of nuclear fuel cycle technology, to provide assistance to Member States in the planning, implementation and operation of nuclear fuel cycle facilities and to assist in the development of advanced nuclear fuel cycle technology. In particular, this activity is focused on such subjects as: uranium resources and geology, technology of ore processing and production of nuclear fuel and reactor materials; spent fuel management, operational data, status, prospects and economics of nuclear fuel cycle facilities throughout the world.

The described objectives are worked out by collecting, reviewing and disseminating information, organizing various meetings, coordination of research and development programmes in Member States; publication of technical reports, manuals, etc. and providing assistance to Member States in development of their nuclear programmes.

Many years ago the Agency had already shown great interest in fuel element technology. During the initial years, attention was paid more to the properties of materials for physicists' and designers' use, and only later, emphasis was shifted to the fabrication processes and performance of nuclear fuel at reactors. In 1960 the IAEA organized an International Symposium on "Fuel Element Fabrication with Special Emphasis on Cladding Materials" and in 1962 a monograph was published on "Fabrication of Fuel Elements". After an Expert Panel on "Quality Assurance and Control in the Manufacture of Metal Clad UO_2 Reactor Fuels", convened in Vienna, November 1974, the Agency has enhanced its interest in the field of fuel elements for water power reactors.

In 1976 the Agency established an "International Working Group on Water Reactor Fuel Performance and Technology". The main function of the IWGFPT is to provide advice and assistance to the IAEA regarding the Agency programme in the above-mentioned field. This programme includes the arrangement of Meetings (where representatives of producers and suppliers, utilities' operators and Governmental Agencies have open discussions) and the preparation of guidebooks and reviews. A list of most interesting IAEA publications in this area is given in Table 6.

At present the IAEA is coordinating research activities on such programmes as fuel performance computer modelling, interaction of water coolant with fuel cladding and methodologies of post-irradiation examination of reactor fuel.

In the framework of the Technical Cooperation the Agency provides assistance to countries which wish to set up nuclear fuel fabrication facilities. This assistance is in the form of contracts which are provided to developing countries, experts' advice on their programmes and help in the start-up of their facilities, fellowships for their engineers, and, purchase of scientific and industrial equipment.

In the area of nuclear fuel during recent years, assistance was provided to such countries as Indonesia, Egypt, Republic of Korea, Romania, Mexico, etc. Quality Assurance and Quality Control is one of the most important subjects in the technical assistance programmes.

At present, the IAEA is supporting the work of the Committee on Assurances of Supply of nuclear materials, fuels and fuel cycle services. The assured supply of reactor fuel is very important for many developing countries. Despite the fact that a number of countries are developing their own production capacities, many countries still would be dependent on suppliers. In order to assure the supply the countries may be interested in diversifying their sources of supply. In this situation standardization of fuel fabrication technology, fuel itself and Quality Control programmes are needed to a large extent. All this could be achieved through extensive cooperation and on a multinational basis.

3. Conclusions

Nuclear power has been proved to be a reliable and an economically advantageous source of energy in the world. Many countries are developing extensive nuclear power programmes. Development of nuclear fuel and its supply plays an important role in the effective operation of nuclear reactors.

Existing experience of water reactor fuel technology and performance is quite positive. Significant efforts are being done to make improvements in fuel manufacture, better uranium utilization at reactors and to higher safety standards of fuel performance. All this requires very effective programmes on Quality Assurance and Quality Control during fuel fabrication and reactor operation.

The IAEA plays an important role in international exchange of information and cooperation in this area. The IAEA will continue its

programmes in the nuclear fuel cycle area with emphasis on improvements of fuel technology and performance and higher safety standards. Among them is the subject of Quality Control of nuclear fuel. Importance of this subject will be increased because more countries are entering the suppliers group and more standardization is needed to assure the diversified supply of nuclear fuel.

References

1. Nuclear Power Reactors in the world. Ref. Data Series No.2
   IAEA September 1983 (page 7, Table 1; page 8, Table 2)

2. Status and trends of nuclear power worldwide
   IAEA report, September 1983

3. Monthly report on the Nuclear Fuel Market, No. 186
   NUEXCO, February 1984

4. Uranium, Resources, Production and Demand
   Summary Report OECD/NEA and IAEA December 1983, Paris

5. P. Knudsen et al. "Experience in fabrication technology and
   performance of water reactor fuel".
   Proceedings of the International Conference on Nuclear Power
   Experience, Vienna, September 1982

6. Nuclear Fuel Cycle Facilities in the World
   IAEA internal report, December 1983.

TABLE 1

Leading Countries in Nuclear Share of

Total Electricity Produced During 1982/83

Country	Nuclear Share, %	
	1982	1983
Finland	42.4	41.5
France	38.7	48.3
Sweden	38.6	36.9
Belgium	30.3	45.9
Bulgaria	29.2	27.4
Taiwan	28.7	36.8
Switzerland	27.6	29.3
Japan	19.5	18.1
Germany, Fed. Rep.	17.4	17.8
United Kingdom	15.2	17.0
German Dem. Rep.	12.5	11.8
United States	12.1	14.8
Canada	10.5	13.2

Source: IAEA Power Reactor Information System (PRIS)

TABLE 2

Nuclear Power Plants in Developing Countries
(as of 31 December 1982)

Country	Operating		Under Construction	
	No. of Units	Capacity MW(e)	No. of Units	Capacity MW(e)
Developing Countries Excluding CPE-Europe				
Argentina	1	335	2	1292
Brazil	1	626	2	2490
Cuba	–	–	1	408
India	4	809	4	880
Korea, Rep. of	2	1193	7	6227
Mexico	–	–	2	1308
Pakistan	1	125	–	–
Philippines	–	–	1	620
Taiwan	4	3110	2	1814
Yugoslavia	1	632	–	–
Total excluding CPE-Europe	14	6830	21	15039
Developing Countries in CPE-Europe				
Bulgaria	4	1632	2	1906
Czechoslovakia	2	762	6	2520
Hungary	1	408	3	1224
Romania	–	–	2	1320
Total in CPE-Europe	7	2802	13	6970
Total	21	9632	34	22009

Source: IAEA Power Reactor Information System (PRIS)

Scheduled Expansions of Nuclear Power Capacity   TABLE 3

up to 1990

Country Grouping	GW(e) in operation at End of Year		
	1982	1985	1990[1]
Industrialized Countries	163.5	269.3	337.6-381.3
Developing Countries			
a) In CPE-Europe[3]	2.8	7.1	10.3-17.1
b) Others[4]	6.7	15.0	18.7-22.1
c) Total of DC's	9.5	22.1	29.0-39.2
World Total	173.0	291.4	367-420

Source: IAEA Power Reactor Information System (PRIS).

TABLE 4

Commercial Fuel Manufacturing Plants

Country	Plant	Capacity t/yr	Fuel types fabricated
Belgium	FBFC	410	PWR & BWR
Canada	CGE	600	HWR
	W Canada	500	HWR
	C-E Canada	230	HWR
F.R. Germany	RBU	850	PWR, BWR & HWR
France	FBFC	370	PWR
Great Britain	BNFL	100	BWR & PWR
Italy	Agip Nucleare	200	BWR & PWR
India	DAE	135	HWR & BWR
Japan	JNF	480	BWR
	MNF	460	PWR
	NFI	50	BWR & PWR
Sweden	ASEA-ATOM	330	BWR & PWR
USSR	State Committee	700	PWR & BWR
USA	B & W	375	PWR
	C-E	265	PWR
	ENC	600	BWR & PWR
	GE	1 100	BWR
	W	800	PWR

Table 5

List of recent IAEA publications on reactor fuel

Title	Year of issue
Proceedings Series	
Fabrication of Water Reactor Fuel Elements (Prague)	1978
Nuclear Power Experience – Vol. 3 Nuclear Fuel Cycle (Vienna)	1982
Technical Report Series	
Guidebook on Quality Control of Water Reactor Fuel	1983
Technical Documents (unpriced)	
IAEA Specialists' Meeting on "Effect of Water Chemistry on Fuel Cladding Behaviour in Power Reactors" (Leningrad)	1983
IAEA Specialists' Meeting on "Water Reactor Fuel Safety and Fission Product Release in Off-Normal and Accident Conditions" (Joint IAEA/ OECD-NEA, Risoe)	1983
IAEA-CEC Joint Specialists' Meeting on "Power Ramping and Power Cycling Behaviour of Water Reactor Fuel" (Petten)	1983
IAEA Specialist' Meeting on "Water Reactor Fuel Element Performance Computer Modelling" (Preston)	1983
IAEA Specialists' Meeting on "Examination of Fuel Assembly for Water Cooled Power Reactors" (Tokyo)	1982
IAEA Specialists' Meeting on "Influence of Power Reactor Water Chemistry on Fuel Cladding Reliability"(San Miniato)	1982

Title	Year of issue
IAEA Specialists' Meeting on "High Burnup in Power Water Reactor Fuel" (Mol)	1982
IAEA Specialists' Meeting on "Pellet Cladding Interaction in Water Reactors" (Risoe)	1981
IAEA/OECD-NEA Joint Specialists' Meeting on "Safety Aspects of Fuel Behaviour in Off-Normal and Accident Conditions" (Helsinki)	1981

Figure 1

HISTORICAL EXCHANGE AND TRANSACTION VALUES — 29 February 1984
(DATA NUEXCO)

Figure 2

ANNUAL REACTOR URANIUM REQUIREMENTS AND
URANIUM PRODUCTION CAPABILITY - 1984-1995
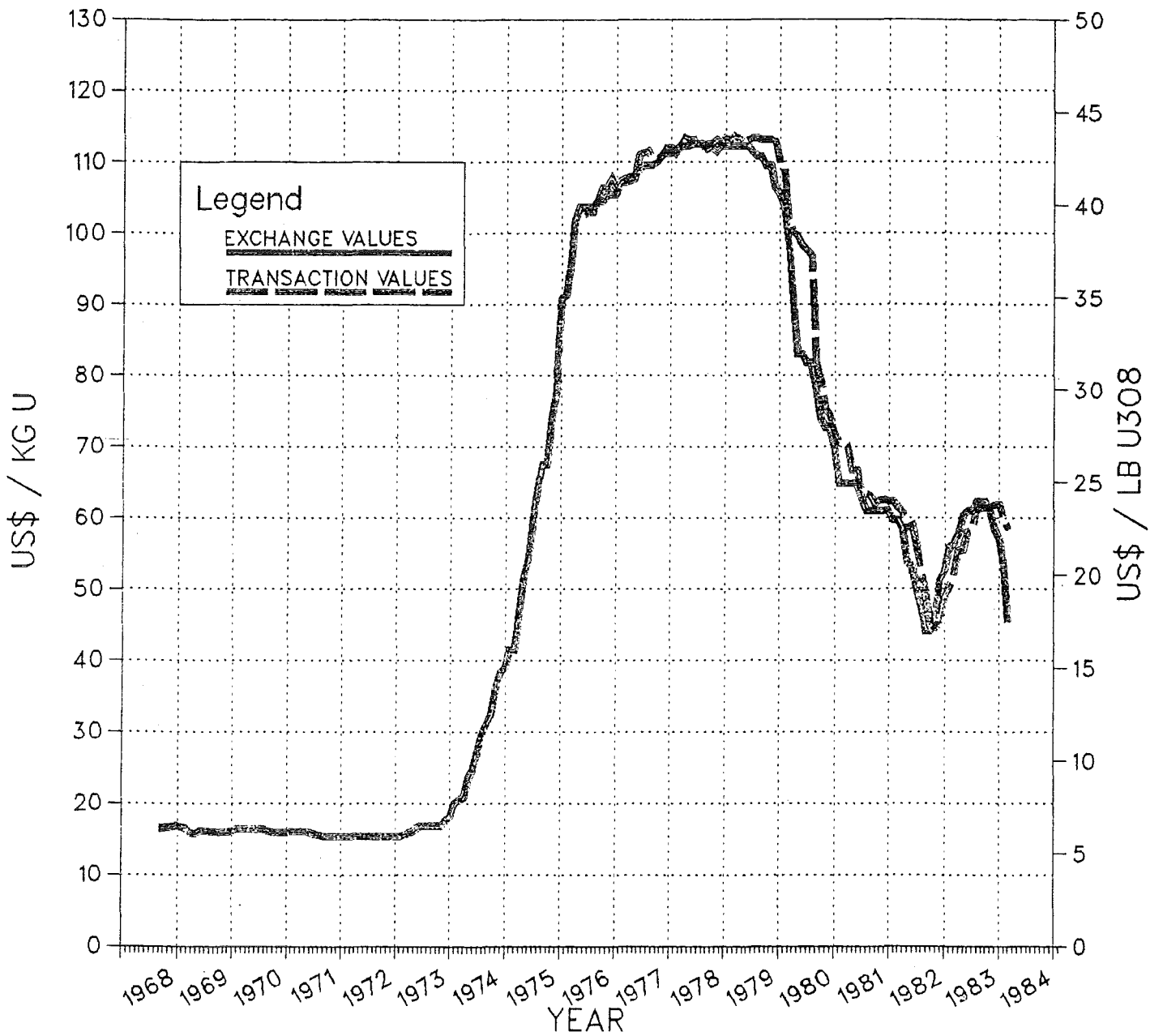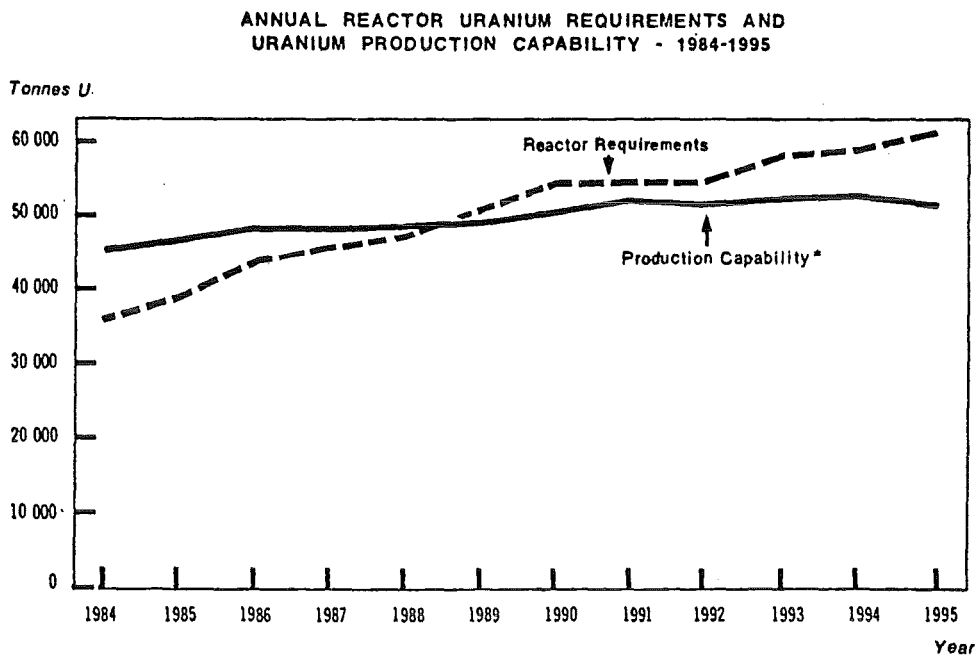
* Based on *existing* and *committed production centres* supported by *known resources* (RAR and EAR-I)
recoverable at cost of $ 130 / kg U or less.

Figure 3

ANNUAL REACTOR URANIUM REQUIREMENTS AND
URANIUM PRODUCTION CAPABILITY FROM KNOWN RESOURCES (< $ 130 / kg U)
1985 - 2025
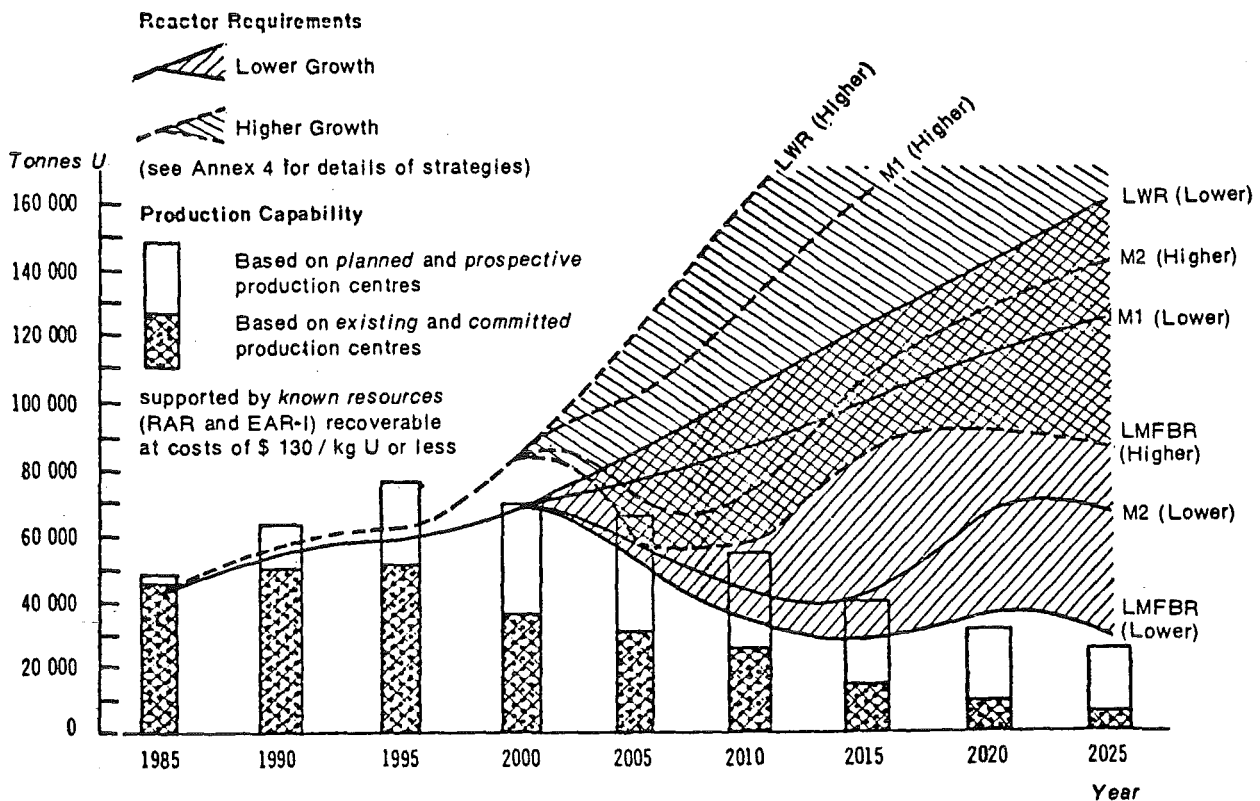
DISCUSSION

K.R. KUMMERER: You mentioned an average failure rate of 0.01% for the fuel pius. This figure is impressively low. One single failed pin, however, means that the whole subassembly has to be removed. Thus, the effective failure rate for the subassemblies is much higher (e.g. by a factor ~ 100).

I. RYBALCHENKO: It may be so, but there are no published statistics from utilities.

R. HOLZER: To my knowledge unscheduled plant shutdown has never occurred in the Fed.Rep. of Germany due to fuel failures. With the current very low fuel failure rates ($\approx 2 \times 10^{-5}$/rods.year) replacement of defective fuel assemblies by appropriate adjustment of the fuel management scheme is easy without prolonged outage time of the plants.

H. BAIRIOT: The statistics on plant outage do not show any more "fuel failure" as cause of the outage, because the occurrences of gross fuel failure necessitating plant shutdown result from a primary cause (e.g. baffle jetting, moving parts in the primary circuit). The statistics take indeed into account only the primary cause of plant outage.

SESSION I

OFFICIAL RULES IN PRACTICE FOR QUALITY
ASSURANCE IN DIFFERENT COUNTRIES

Chairman: D. Vollath
Kernforschungszentrum, Karlsruhe

THE ROLE OF THE GOVERNMENTAL AUTHORITY IN LWR FUEL FABRICATION QA ACTIVITIES IN BRAZIL

R. POLLIS

CNEN - Comissão Nacional de Energia Nuclear

Rua General Severiano, 90

Rio de Janeiro - RJ

BRAZIL

ABSTRACT

The paper deals with the position of CNEN, the Brazilian nuclear regulatory body, concerning QA in the manufacture of LWR fuel elements. As QA regulation, CNEN has adopted the IAEA's Code of Practice No.50-C-QA. A new regulation based on this code, but somewhat modified to fit Brazilian specific conditions has been developed. The activities conducted by CNEN on the fuel manufacturer are discussed. These activities include review and assessment of the QA Programme for compliance with regulatory requirements, and inspections/audits to verify the correct implementation of the measures described in the QA Programme.

1. INTRODUCTION

To meet the expected future energy demand and, on the other hand, considering a need to reduce as much as possible the dependence on oil imports, the Brazilian Government has decided to complement the domestic energy resources by means of nuclear origin power.

To achieve this aim, a Nuclear Power Programme was established, consisting in installing a set of nuclear power plants as well as in developing an associated nuclear industry. PWR-type nuclear power plants have been selected as the most adequate for the purposes of this programme. This decision was based on the experience of the world's existing nuclear power plants, taking into consideration aspects such as economics (capital costs and generating costs), safety, operating performance, and reliability.

The feasibility of the Nuclear Power Programme required the installation of a complete nuclear fuel cycle, including and enrichment plant. To save time, the Brazilian Government has decided to purchase technology from a highly experienced country in the nuclear field.

In 27 June 1975, an agreement was signed between Brazil and the Federal Republic of Germany, covering the following subjects:

- prospecting, extraction and processing of uranium ores and production of uranium compounds;

- design and construction of nuclear power plants and other nuclear facilities;

- uranium enrichment services;

- fuel element manufacturing and spent fuel reprocessing.

From the Brazilian side, a state-owned company (NUCLEBRÁS) was assigned to conduct the programme. An extensive effort has since then been carried out to promote the improvement of the domestic basic industry so that equipment and components meeting the high quality standards of nuclear industry could be manufactured.

NUCLEBRÁS has installed in Resende, State of Rio de Janeiro, an industrial complex (CIR) which will produce the fuel to be consumed by the nuclear power reactors. This complex comprises three units, namely, a UF_6 conversion plant, a uranium enrichment plant, and a fuel manufacturing plant.

2. LWR FUEL FABRICATION PLANT

NUCLEBRAS fuel fabrication plant (FEC) was planned to produce
fuel assemblies for light water reactors, and started its activities in
October 1982. At first, only PWR-type fuel assemblies will be produced.

The implantation of this plant has been planned to be performed
in three separate steps. The first step consists of manufacturing fuel rods
with purchased fuel pellets and zircaloy parts, manufacturing fuel element
structural components and finally assembling the fuel elements. The second
step consists in the production of fuel pellets from purchased UO_2 powder. The
third step consists in the conversion of enriched UF_6 into UO_2 powder.

FEC was designed to have a nominal capacity of 400 ton UO_2/year,
which is equivalent to a production of about 800 fuel assemblies/year. FEC's
present capacity is 100 ton UO_2/year.

In order to meet the plant licensing requirements, NUCLEBRAS has
developed and implemented a Quality Assurance System so that the fuel elements
to be produced will provide a safe and reliable performance in service. A
"Quality Assurance Programme for fabrication of nuclear fuel elements and
associated core components" was written by NUCLEBRAS to describe its QA system,
defining all actions and measures to be established. To manage this QA System,
NUCLEBRAS ought to develop an appropriate organizational structure.

A set of QA procedures, instructions, and guidelines applicable
to specific groups, departments and work areas, has been prepared by
NUCLEBRAS. These documents specify or detail how activities affecting product
quality are to be carried out in order to meet the commitments set out in the
Quality Assurance Programme.

3. REGULATORY POSITION

3.1 - General Philosophy

Safety is one of the aspects related to the Nuclear Power Programme which the Brazilian Government is most concerned with. In this respect, it is a governmental policy not to save any efforts to preserve the health and the safety of the public, to safeguard facilities and materials, and to maintain environmental conditions. The governmental organization responsible for establishing and enforcing nuclear safety requirements is the national nuclear energy commission (CNEN). Among the safety requirements applicable to items important to safety of nuclear facilities, are QA requirements.

Regarding specifically to QA, CNEN performs the following:

- Establishment of QA regulations for nuclear facilities;

- Review and assessment of submitted QA Programmes;

- Conduct of inspections and audits over approved QA Programmes to verify compliance with QA regulations.

3.2 - QA Regulatory Requirements

Owing to the lack of national regulations, in 1979, CNEN decided to adopt the IAEA Code of Practice No.50-C-QA "Quality Assurance for Safety in Nuclear Power Plants" as QA mandatory regulation. This document sets forth the basic QA principles and requirements which are to be adhered to in order to establish and implement an adequate QA system.

The principles and requirements of this Code are grouped in 13 sections covering different subject areas, usually referred to as the "13 criteria".

The adoption of the Agency's Code No. 50-C-QA as an interim regulation was very convenient to CNEN, since this document, based on the experience in QA methods and practices of many different countries, incorporates all internationally accepted QA basic principles and requirements.

However, considering that nowadays enough experience has been acquired in this subject, besides a need to prevent misinterpretations caused by translation problems, CNEN decided to develop and issue its own QA regulation, written in Portuguese, and applicable to all QA activities under its jurisdiction, i.e., nuclear field.

In May 1983, a working group was assigned to prepare this new regulation. In August 1983, a draft version was ready. This version is now under a process of review for final approval and issuing by CNEN's Department of Standards and Specifications. This document was based on the Code of Practice No.50-C-QA, and in its preparation consideration was taken of particular local conditions influencing QA practices, such as:

(a) The lack of previous industrial experience in manufacturing nuclear power plant components, in addition to the reluctance of some manufacturers in accepting purchaser enforcement of their QA measures, required more stringent QA requirements along with an adequate indoctrination of manufacturers' management and personnel on the purposes of the Code.

(b) The QA concept, as practised in Brazil, due to historical reasons, is a hybrid combination of the American (also IAEA) "system-related" approach, which places the emphasis on the achievement of quality through an adequate QA system based on prescribed QA requirements, and the German "product-related" approach, which places the emphasis on the achievement of quality through an increased control of the product quality, based on redundant verification activities performed respectively by the supplier, the purchaser, and an independent inspection organization. CNEN has been establishing measures to harmonize the individual features of both approaches, so that it can be prevented, in the areas where they overlap, a multiplicity of QA activities and documents, which could increase unnecessarily quality costs, without a corresponding significant improvement in the achieved quality.

The major modifications have been made in the first two chapters, namely "Introduction" and "Quality Assurance Programme". Some minor changes have also been made so that the regulation could be more specific and enforcing.

3.3 - Additional QA Guidance

The IAEA's Code of Practice states only the general principles and rules which are to be adhered to. Therefore, it gives rise to numberless ways to interpret and implement its requirements. To aid all organizations involved in a nuclear power plant project to perform their activities in a correct way, more detailed guidance is required.

In the lack of national standards, the IAEA's Safety Guides of the series 50-SG-QA (1 through 11) represent a precious help. They provide acceptable methods for establishing and implementing specific measures required by the Code. These guides are non-mandatory documents. Therefore, compliance whith them is not required. Nevertheless, CNEN fully recomends their use.

ABNT, a Brazilian technical standard association, has been making intensive efforts to fill the gap of lacking national standards in this area. Among the ABNT's 22 committees, one is assigned to nuclear energy subjects, (CB-20).

CB-20 is made-up of 7 subcommittees, two of which are:
SC-05 - Nuclear Fuel Cycle;
SC-06 - Quality Assurance.

The two above subcommittees have working groups engaged in developing standards on some selected topics. Subcommittee SC-06, in particular, has undertaken the task of preparing a series of standards covering the same subjects of the IAEA's Safety Guides 50-SG-QA 1 to 11, taking into consideration, as it has been discussed above for the development of CNEN's QA regulation, the specific local conditions influencing QA practices. Most of the standards have already been prepared and some of them have already been issued. This work is scheduled to be completed by June 1984.

4. QA PROGRAMME REVIEW AND ASSESSMENT

Since fuel elements are nuclear reactor safety-related items, the fuel supplier (NUCLEBRAS) was required to submit to CNEN for review and approval, a QA programme for nuclear fuel element fabrication.

The purpose of the review and assessment process is to ascertain whether:

(a) all applicable requirements of regulations have been considered in the QA programme;

(b) the commitments undertaken in the QA programme are consistent with these requirements;

(c) the duties and responsibilities of persons and organizations assigned to QA functions are described in sufficient detail to permit CNEN to determine if they have the required authority and organizational freedom to effectively perform their activities.

The bases for approval of a QA programme are the Agency's Code of Practice No.50-C-QA and the CNEN's Standard No.NE-1.09 - "Standard Model for Fuel Element Manufacturing Plant Safety Analysis Report".

As a result of the review and assessment process, some minor improvements in the NUCLEBRAS' QA Programme have been required. After these conditions have been satisfied, the QA Programme was approved by CNEN.

5. QA PROGRAMME INSPECTIONS AND AUDITS

Once the QA Programme has been approved, CNEN ought to verify whether the QA system has been implemented as described in the QA programme. Such an objective has been achieved through a process of inspections and audits of the manufacturer quality-related activities, which has started even before the License for Operation was granted, that is, in the phase of commissioning. These preliminary verifications were intended to ascertain that NUCLEBRAS was adequately prepared to start production of fuel elements meeting the prescribed quality requirements.

Due both to the large variety of a fuel manufacturer quality-related activities and documents and the limitation of resources, mainly manpower, available to CNEN, a thorough coverage of all elements of the QA programme during such inspections /audits is, obviously, impracticable. Therefore, CNEN has decided for a random selection of the points to be verified.

To ensure a uniform interpretation of the regulatory objectives, inspections and audits are conducted in accordance with CNEN procedure No.PF-01 - "Conduct of Inspections/Audits".

This procedure incorporates requirements and practices contained in the IAEA Safety Guides No.50-SG-G4 - "Inspection and Enforcement by the Regulatory Body for Nuclear Power Plants", and No.50-SG-QA-10 - "Quality Assurance Auditing for Nuclear Power Plants".

The technique used in the inspections/audits consists in verifying whether the inspected/audited organization has converted its own commitments (as stated in the QA programme) into actual measures and procedures and how these measures and procedures are being implemented. This is accomplished by searching both documentary and physical evidences of their implementation. Thus, a number of QA documents have been examined for completeness and conformance, and work areas have been inspected to verify if quality-related activities have been properly conducted and controlled. CNEN's inspectors ought to convince themselves that there are no conditions which could jeopardize product quality.

Local inspections have been conducted in the two laboratories which have been installed for the FEC's first implantation step, namely the material testing laboratory and the metrological laboratory (a chemical laboratory is foreseen for the next steps).

The purpose of these local inspections has been to verify aspects such as:

- status of measuring and testing equipment calibration;

- availability of procedures, standards, specifications and other technical documents.

- good housekeeping conditions, such as cleaning, lighting, space, safety, access control, temperature and moisture control, equipment conditions, available services (gas, compressed air, AC/DC electrical supply, etc.). Such conditions may affect the quality of work performed in the laboratories.

Special emphasis was placed into verifying whether reference standard equipment had been certified by INMETRO, the national metrological certification body.

Checklists are prepared in advance to serve as a help to the inspectors' memory, but they are not intended to limit their performance.

Results of all regulatory inspections/audits are documented to provide for further enforcement actions, if needed. Thus, after each inspection/audit, a report has been prepared.

CNEN's inspection/audit reports contain the following information:

(a) purpose and scope of the inspection/audit;

(b) list of participants;

(c) list of documents used as bases for the inspection/audit;

(d) summary of results;

(e) description of documents examined, work areas inspected, etc.;

(f) detailed description of findings;

(g) requirements for corrective actions;

(h) checklist used.

CNEN forwards a copy of the inspection/audit report to the management of the inspected/audited organization. In case any conditions adverse to quality have been reported, CNEN requires a written response describing and scheduling the intended corrective actions to identify causes, correct deficiencies and prevent repetition. Such a response is then assessed by CNEN for adequacy. Follow-up actions are taken during further inspections/audits, to verify implementation as scheduled of these corrective actions.

Most of the CNEN's requirements to NUCLEBRAS for corrective actions have been caused by minor administrative problems or lack of procedures. These kind of deviations are relatively easy to correct and whenever NUCLEBRAS has been urged, prompt corrective actions have been taken.

Besides CNEN's direct inspections/audits of its QA Programme, the fuel supplier QA activities are also controlled by CNEN through the fuel purchaser, i.e., the nuclear power plant owner. This other way of control consists in determining if the plant owner fulfils its responsibility of imposing upon its main contractors (one of which is the fuel vendor), the requirements contained in its own QA Programme which, as part of the plant licensing process, has previously been reviewed and approved by CNEN for consistency with regulatory requirements. The plant owner activities and documents concerning fuel procurement, in particular audits and surveillance of the fuel manufacturer, have been verified, including witness by CNEN of QA auditing activities.

Notwithstanding CNEN's inspections/audits, the plant owner, as the responsible organization for a nuclear power plant project, retains the responsibility to assure that the purchased fuel elements are in compliance with contractual requirements.

6. CONCLUSION

Nuclear fuel claddings are the first barrier to the release of the radioactive isotopes produced into the core of a nuclear reactor. Extended nuclear power station outages caused by fuel failures imply very high operating costs. To ensure a safe and effective operation of a nuclear power plant, one essential condition is that the QA principles and rules be successfully implemented in the fabrication of nuclear fuel elements.

CNEN, as a regulatory body, is responsible for verifying proper adherence to QA regulations. The QA system of the Brazilian fuel vendor (NUCLEBRAS) has been assessed through inspections and audits conducted in its fuel fabrication plant (FEC). Since this plant is quite recent, obviously not enough time has been ellapsed to get fuel performance information. However such a handicap can be suitably compensated by:

- use of a well proved technology;

- implantation of a well structured and managed QA system, based on qualified equipment, procedures and personnel;

- conduct of a close regulatory control.

The combination of the above actions correctly implemented, permited CNEN to conclude that NUCLEBRAS Fuel Element Fabrication Plant is capable to produce fuel which will perform satisfactorily in service.

/idc.

QUALITY ASSURANCE AND QUALITY CONTROL REQUIREMENTS
IN NUCLEAR FUEL PROCUREMENT IN FINLAND

M. OJANEN
Finnish Centre for Radiation
and Nuclear Safety
P.O. Box 268
SF-00101 Helsinki 10
Finland

The paper gives a description of the general procedures and requirements set by
the Finnish safety authorities for nuclear fuel procurement and it also discusses
the different organizations involved. The contents of separate guides for the super-
vision of fuel design and manufacture as well as nuclear fuel quality assurance
are discussed in more detail. Finally the operating experience of fuel in Finnish
nuclear power plants is briefly described.

1 Introduction

The share of nuclear power of the electricity produced in Finland has been about 40 % during 1983. The two operating nuclear power plants include four units, two BWR:s with the capacity of 660 MW(e) designed and constructed by Ab Asea-Atom and two PWR:s (type VVER) with the capacity of 440 MW(e) designed by the Soviet V/O Atomenergoexport (AEE). The first unit, a VVER-440, was loaded in 1976 and the fourth, a BWR-660, in 1979. The PWR-units (VVER:s) are owned and opereted by Imatran Voima Power Company (IVO) and the BWR-units by Industrial Power Company (TVO). The need for a fifth unit has been discussed quite intensively in connection with the future Finnish energy policy, but no firm decision has so far been taken.

The front end of the fuel cycle for both power plants, that is for both power companies, is covered totally by foreign services. For the VVER-440 reactors the Soviet AEE has made a principle agreement with the power company to supply fuel for the whole lifetime of the plant. There is also a principle agreement for returning the spent fuel back to the Soviet Union. For the two BWR-units the fuel is presently supplied by two separate fuel manufacturers on the basis of a reload agreement. The back end of the fuel cycle after long term storage is still open for the BWR:s.

2 Safety Control of Nuclear Facilities and Materials

The Finnish Centre for Radiation and Nuclear Safety (STUK), which is under the administrative control of the Ministry of Social Affairs and Health, is the national authority responsible for the supervision of radiation protection and nuclear safety in Finland. Partially its activities cover areas where the highest supervisory power lies within the Ministry of Trade and Industry. It was established in 1958 to organize the supervision of medical and industrial uses of radiation in accordance with the Radiation Protection Act. At the end of the 1960´s, the Centre´s field of activities was broadened to cover the supervision of siting, construction and operation of nuclear power plants in accordance with the Atomic Energy Act. The Centre was reorganized in 1975, and its name and administrative structure were changed in order to reflect more accurately its new scope of activities.

The permanent staff of the Centre numbers over 200, almost half of them being university graduates or specialists in their own fields. Over a third of the staff are engaged in research and development. The supervision of nuclear power, including both technical control of reactor safety and surveillance of the environment as well as the international safeguards control tasks, accounts for roughly half of the work of the Centre. The research carried out by the Centre is closely connected with its supervisory activities.

The Centre is divided into three main departments, an administrative unit and a separate medical research group. The major part of the supervisory and regulatory activities concerning nuclear energy is the responsibility of the Department of Nuclear Safety. The Department has a staff of about 50 nuclear experts.

The Department of Inspection inspects the working conditions as regards radiation and keeps files of persons exposed to radiation and of individual doses.

The Research Department carries out environmental monitoring around nuclear power plants. It also takes part in the offsite emergency planning, especially as regards the risk evaluations in the environment.

3 Regulations and Licensing

The legal framework for nuclear energy in Finland consists mainly of the Atomic Energy Act, passed in 1957 and the Radiation Protection Act of the same year. Both have been subsequently amended several times. Preparations for a new Nuclear Energy Act are in the final stage and for a new Radiation Protection Act under way. Several other areas of legislation, notably legislation on nuclear liability, pressure vessels, transportation of dangerous materials, building, regional planning and mining, contain provisions which apply to nuclear activities.

According to the Atomic Energy Act, a special permit is required for the production, possession, transport, use, import and export of materials suitable for the generation of atomic energy, as well as for trade and other transfer of such materials. The permit is also needed for the construction, possession and operation of a facility intended for the generation of atomic energy, and for the construction, possession and operation of a nuclear reactor. These permits are issued, according to the Atomic Energy Act, by the Ministry of Trade and Industry. The present practice, however, has been to take the major permits for the consideration of the Government.

According to the Radiation Protection Act, a special permit is required for the manufacture, use, transport, import and export, possession and trade of radioactive materials as well as for the use of equipment and installations producing radiation. These permits are issued, depending on the nature of the activity by the Ministry of Trade and Industry, the Finnish Centre for Radiation and Nuclear Safety or the National Board of Health.

When STUK is evaluating the license application for the import, transport, handling, storage, use and potential export of nuclear fuel, it will give special attention to the following aspects:

- Based on the information supplied by the license applicant and STUK's own inspections, STUK has gained assurance of the safety of the nuclear fuel and of its compatibility with the reactor.

- The license applicant has shown its competence and preparedness to supervise and to carry out safely the transport, storage, handing and use of the nuclear fuel.

- Based on the information supplied by the license applicant and STUK's own inspections, STUK has gained assurance of the sufficiency of the applicants competence and preparedness to arrange the safeguards control and security measures in an acceptable manner.

- The license applicant has presented an acceptable plan for:

    - the final and safe disposal of spent fuel or
    - the disposal of spent fuel outside Finnish territory or
    - the interim storage of spent fuel and for the final and safe disposal of wastes from reprocessed fuel to be returned to Finland.

- All the activities in question can be realized without violating the obligations of the international agreements signed by Finland.

The issued permits will normally contain conditions to ensure the achievement of the sufficent safety level and the fulfillment of the obligations of international agreements. After the permits have been granted STUK will carry out continuous safety control of the licensee's activities and special attention is also paid to the applied quality assurance systems.

The general information concerning the safety of nuclear fuel is presented in the safety analysis report of the plant. This report also includes the analyses dealing with disturbance and accident situations at the plant unit. Otherwise the fuel information is presented in the pre-inspection documents of each delivery batch. These documents are submitted to STUK for approval not later than one year before the commencement of manufacture. The pre-inspection documents of fuel include the following items:

- Design and manufacturing quality assurance
- Design bases
- Behaviour analyses and experimental studies
- Operating experience

- Material specifications
- Description of manufacture
- Quality control programme
- Drawings

STUK gives regulations, rules and measures regarding supervising the safety of the nuclear power plants and nuclear fuel. At present there are about 50 such safety-guides (YVL Guides) completed. Most of these quides have been translated into Eng-lish. The pre-inspection of fuel is discussed in more detail in the Guide YVL 6.3 "Supervision of Fuel Design and Manufacture". The general fuel licensing proce-dures are described in the Guide YVL 6.1 "Licensing of Nuclear Fuel and other Nuc-lear Materials".

4 Quality Assurance Requirements

4.1 General Requirements

The licensee has the main responsibility for the execution of the quality assurance measures concerning fuel. The general principles for the quality assurance activi-ties concerning nuclear fuel have to be defined in the quality assurance programme of the licensee. This quality assurance programme is submitted separetely to STUK for approval. The programme shall fulfil the requirements given in the Guide YVL 6.7 "Nuclear Fuel Quality Assurance". The principles of the programme have to be visible in the quality assurance programmes of other involved organizations.

Before the final order the licensee has to inspect and accept the quality assurance programmes of the fuel designer and manufacturer and also those of the most impor-tant subcontractors. By conducting inspections during manufacture the licensee must also ensure that these programmes are followed in practice.

The Finnish Centre for Radiation and Nuclear Safety will perform audits based on the licensee´s, designer´s and manufacturer´s quality assurance manuals concerning the implementation of quality assurance. If the designer or manufacturer is not known beforehand by the licensee, the first audit should be made before the commen-cement of manufacture, otherwise the audits are carried out on an annual basis. It is the licensee´s responsibility to take care of the timing of these audits in such a way that enough time is reserved for corrective actions.

4.2 Fuel Procurement

The nuclear fuel procurement documents (including uranium supplies, conversion, enrichment and transport) must be written so that the national licensing procedures

and the obligations of international agreements are taken into account. These documents must also include or give as reference the relevant technical requirements, standards and other guides. Also the requirements concerning article identification, recieving inspections, archive samples, packing, handling, transport and storage must be presented.

The acceptance conditions and procedures for products and services must also be defined in the procurement documents. The acceptance of an article is based on the inspection of the design documents as well as on the manufacturing control and recieving inspections. The treatment of deviations should also be defined in the procedures.

The representatives of the licensee and the safety authority must have access to the fuel factory and must be able to inspect documentation concerning the relevant fuel batch and its manufacturing methods. The manufacturer must reserve possibility to perform retests with archive sample materials.

Furthermore the procurement documents should define the manufacturing control results, that will be delivered to the licensee. The designer´s and manufacturer´s operating experiences with the corresponding fuel type should also be available.

4.3   Fuel Design

4.3.1   Design Requirements

The criteria for the design of the fuel assembly and fuel rods must be defined in the design documents. The design basis consideration should include normal operating conditions, off-normal disturbances and anticipated accident conditions. The designer must show with acceptable methods the fulfilment of these criteria.

The fuel design must consider the combatibility of fuel with the reactor and other relevant equipment at the plant. These considerations should include the structure of the reactor core, the thermal hydraulic and physical properties of the reactor, coolant chemistry of the primary circuit and storage pools, and fuel handling, transportation and storage systems.

The design evaluation should be based on calculational analysis, experimental investigations and operational experiences and the evaluation should include all design conditions. In addition to the normal design organization the evaluation should also be carried out by independent experts.

In case changes are introduced into the already accepted design, these changes must

be evaluated with the same procedure as the corresponding item in the original design.

The design documents should be stored for eventual inspections for the whole operational age of the fuel.

4.3.2 Design Evaluation by the Licensee

The licensee has the responsibility to evaluate the design information including drawings, design calculations, experiments and operating experiences. The evaluation must include additional independent calculational analyses and estimation of relevant operating experiences gathered by the licensee.

The introduction of major changes into the accepted fuel design gives reason to check the design information and it is possible that additional analyses have to be performed. Also the compatibility with the reactor may have to be checked.

When the fuel manufacturer and/or designer is not known beforehand by the licensee the design evaluation is carried out in its entirety. If, however, the fuel type is the same as that already used by the licensee, the evaluation may primarily concentrate on estimating the effects of manufacturing differences.

In general, if the fuel type is new or only limited operational experience is available, the design evaluation will include manufacturing and irradiation of lead test assemblies.

The evaluation of the fuel design will continue all the time during the operation of the fuel type, and the design requirements can be changed when new reloads are ordered if operating experiences give reason for that. Operating experiences are gathered according to an accepted fuel behaviour surveillance programme which includes the control of operating parameters and different kinds of pool side inspections.

4.4 Fuel Manufacturing

The fuel manufacturing control is aimed to ensure that the products meet the requirements set for them.

The licensee must evaluate the fuel manufacturing and quality control methods and specifications and follow the manufacture in a planned manner. The manufacturer must have documented instructions for all manufacturing and quality control steps. The instructions for quality control tests must give requirements for performing

the tests, for acceptable test methods and equipment, for the extent of the tests and for acceptable test results.

The manufacturing and quality control methods must be accepted for use according to written instructions. Those manufacturing and control methods which, because of their complexity, sensitivity or other reasons require periodical qualification or special skills from the personnel, must be separately defined. The validity of the qualifications for manufacturing and control methods and for personnel must be documented.

The fuel rod assemblies and channels and all their parts must be marked so that their identification is possible at all stages of the manufacture and that all components of the final product can be traced to the origin material.

The control of products and activities which do not fulfil the given requirements is of major importance and must be carried out according to special instructions. These instructions should include methods for identifying deviating products and activities, for separating the deviating or otherwise affected products, finding out reasons for the deviation and informing the involved organizations. Deviating products must be evaluated and rejected, accepted or repaired according to written instructions. All deviations accepted by the manufacturer must be documented and presented also to the licensee for acceptance.

The licensee will inspect the QC documentation during and after the manufacturing campaign especially to check that all planned QC measures have been taken and all deviating results reported.

The Finnish Centre for Radiation and Nuclear Safety will carry out its independent manufacturing inspections. In the case of the first fuel delivery from the manufacturer of the assemblies, the inspection of manufacturing and QC methods is performed before the actual manufacturing campaign starts. With subsequent deliveries these inspections are performed on an annual basis. After the manufacturing of each delivery lot, the Centre will also inspect the manufacturing QC documentation, including the deviation reports.

5   Practical Experiences

Up to the present day, the experience of fuel procurement in Finland is based on three fuel suppliers. The fuel to the Loviisa power stations is supplied by the main contractor of the plant, Atomenergoexport. The Olkiluoto plants, have up to now acquired fuel from Asea-Atom. However, also Kraftwerk Union AG from FRG is presently being qualified as a supplier. At the time when the first agreements

on fuel supplies for the Loviisa and Olkiluoto plants were made, the detailed procedures in fuel procurement were not established as regulatory guides.

5.1 QA/QC Arrangements

It has not so far been possible for AEE to arrange an opportunity for the Finnish organizations to visit the fuel factory in order to carry out the inspections described earlier. Therefore IVO has been obliged to take interim substituting measures. The USSR Chamber of Commerce and Industry (TPP) is being used as the licensee's agent in fuel manufacturing control. TPP has the right to inspect all the specified documentation in the factory and also to request extra QC-tests to a limited extent. TPP transmits to IVO the normal QC results as well as the results of any extra tests by TPP. TPP also performs quality assurance audits planned by IVO to the fuel factory and reports the results of these audits to IVO.

As a substituting measure IVO also takes one assembly from every fuel manufacturing lot for destructive QC-examinations, which are partly performed by IVO and partly by the Technical Research Centre of Finland (VTT). The QC results obtained by AEE, TPP, IVO and VTT are compared statistically.

The above described way of fuel procurement is, however, not considered totally satisfactory by the Finnish Centre for Radiation and Nuclear Safety and therefore also the fuel permits have been issued with a limited time of validity.

With both Asea-Atom and Kraftwerk Union as fuel suppliers to TVO's BWR reactors the quality assurance audits and manufacturing control by TVO and the Centre have been regular.

Also TVO has taken advantage of outside organizations in performing the practical work by using VTT as a consultant during control visits to the manufacturers and their subcontractors.

5.2 Fuel behaviour

Fuel behaviour in both Loviisa and Olkiluoto plants has been satisfactory. Up to the autumn 1983, altogether five failed fuel assemblies have been found in Loviisa and one in Olkiluoto.

It was already seen in the design evaluation of the Loviisa fuel that the space for axial growth between the fuel rod upper end plug shoulder and the upper tie plate is too small. This observation was also confirmed by later pool side inspections of spent fuel. The closure of the elongation space has not, however, accord-

ing to the present understanding caused any fuel rod failures. A change in the bundle design has now been accepted and the problem is expected to be removed from subsequent reloads.

The fuel assembly design of the Olkiluoto plants uses screws in fixing the channel and core cell leaf springs into the upper end plate and in fixing the channel to the transition piece. Originally these screws were fabricated of Incaloy 800 material. On the basis of experience gained from elsewhere, the fuel designer wanted to change the material into Inconel X-750, which has high tensile strenght. Experience showed, however, that Inconel X-750 was prone to stress corrosion (SCC) in BWR conditions. Different fabrication methods have been used in the Inconel X-750 screws to improve their SCC behaviour but in spite of that screw failures have been found in practically all fabrication types. Only four of these screw failures have been found in Olkiluoto but in other plants using Asea-Atom fuel the failures have been more numerous. Therefore the screw material has now been changed back to Incaloy 800 with some structural design modifications, because all the previous experience of Asea-Atom with that material was satisfactory.

References

1.   Guide YVL 1.0 "Safety Criteria for Design of Nuclear Power Plants"

2.   Guide YVL 1.1 "The Institute of Radiation Protection as the Supervising Authority of Nuclear Plants"

3.   Guide YVL 1.4 "Quality Assurance Program for Nuclear Power Plants"

4.   Guide YVL 6.1 "Licensing of Nuclear Fuel and Other Nuclear Material"

5.   Guide YVL 6.2 "Fuel Design Limits and General Design Criteria"

6.   Guide YVL 6.3 "Supervision of Fuel Design and Manufacture"

7.   Guide YVL 6.7 "Nuclear Fuel Quality Assurance"

DISCUSSION

S.V. RAGHAVAN: What is the frequency of the destructive testing carried out on a batch of fuel assemblies by the licensee, and what action is taken if found defective.

M. OJANEN: The average batch size which corresponds to one reload of both Loviisia units is about 250 assemblies. The destructive test frequency is one assembly per batch. If non-conformances are found, the licensee will contact the vendor and reach with him an agreement about future actions.
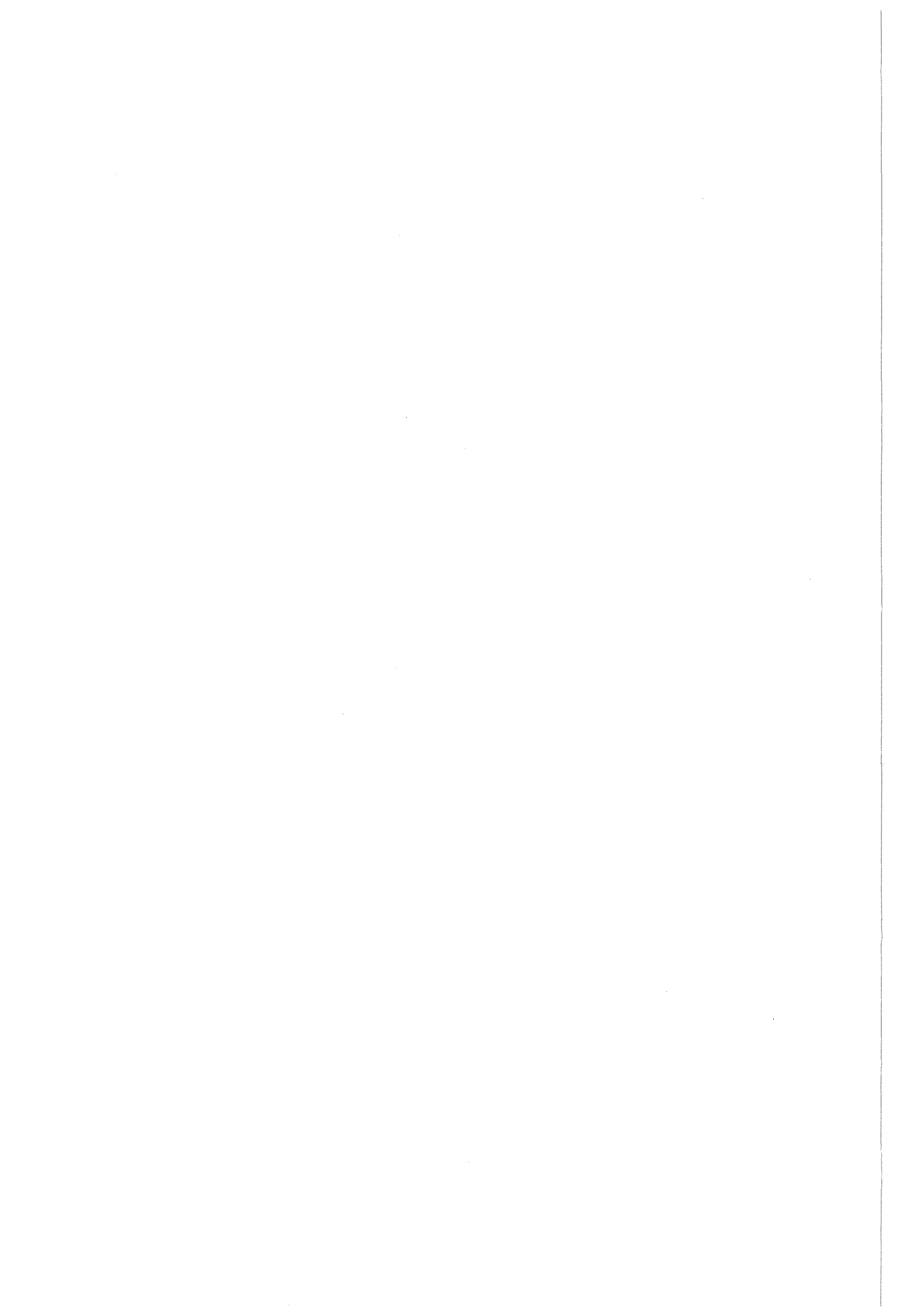
SESSION II

QUALITY ASSURANCE SYSTEM IMPLEMENTED IN
DIFFERENT PRODUCTION PLANTS   1.

Chairman: V. Gorskij
All Union Scientific and Research
Institute of Inorganic Materials, Moscow

SESSION III

QUALITY ASSURANCE SYSTEM IMPLEMENTED IN
DIFFERENT PRODUCTION PLANTS   2.

Chairman: R. Holzer
Kraftwerk Union AG, Erlangen

SESSION IV

QUALITY ASSURANCE SYSTEM IMPLEMENTED IN
DIFFERENT PRODUCTION PLANTS   3.

Chairman: K. Balaramamoorthy
Atomic Fuels Division
Bhabha Atomic Research Centre Trombay, Bombay

IAEA-SR-102/11
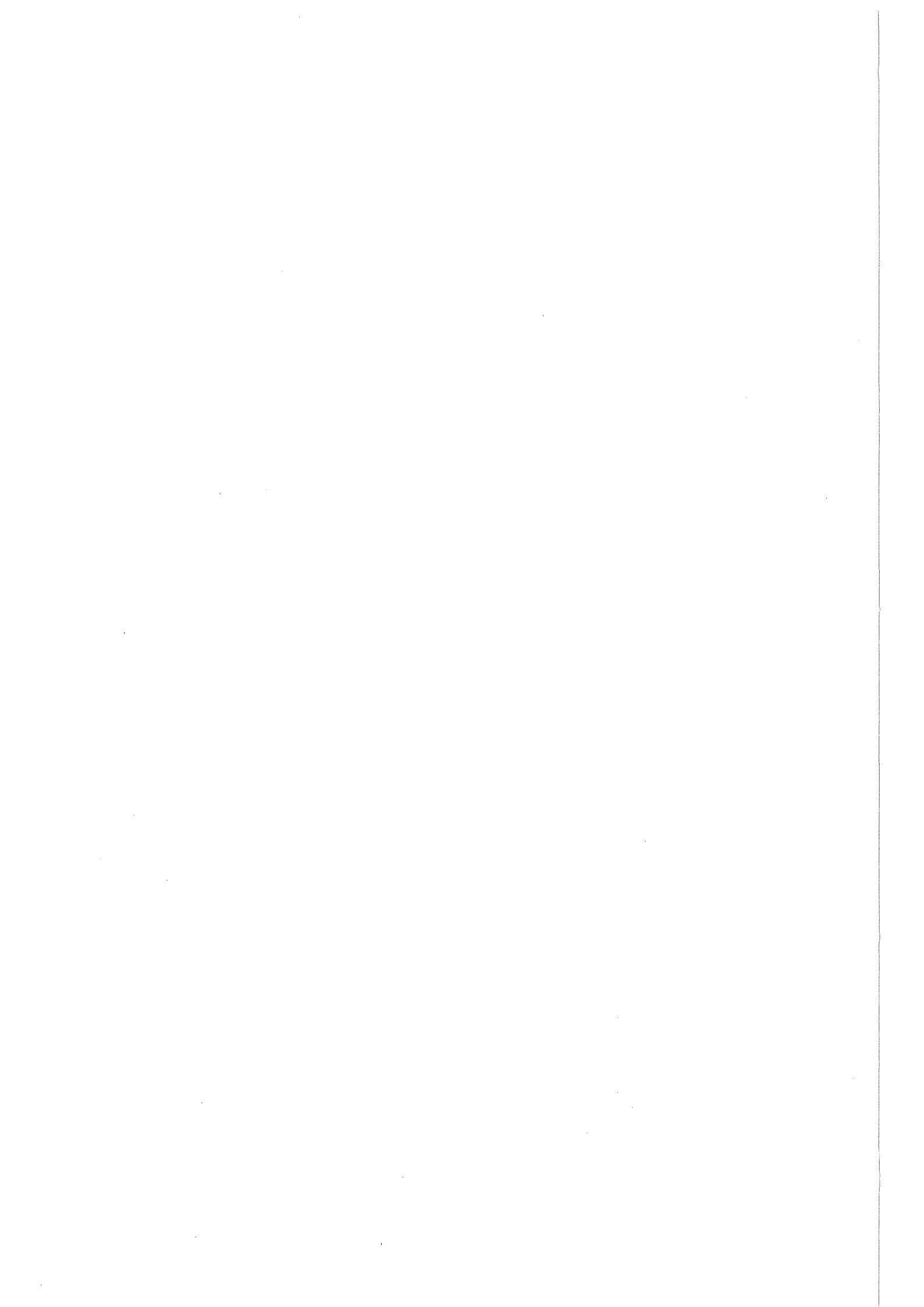
EXPERIENCE WITH QUALITY ASSURANCE
IN FUEL DESIGN AND MANUFACTURING

R. Holzer, F. Nilson
Kraftwerk Union Aktiengesellschaft
Erlangen, Germany

ABSTRACT

The Quality Assurance/Quality Control activities for nuclear fuel design and manufacturing described here are coordinated under a common "Quality Assurance System For Fuel Assemblies And Associated Core Components" which regulates the QA-functions of the development, design and manufacturing of fuel assemblies independent of the organizational assignment of the contributing technical groups. Some essential characteristics of the system are shown, using examples from design control, procurement, manufacturing and qualification of special processes. The experience is very good, it allowed a flexible and well controlled implementation of design and manufacturing innovations and contributed to the overall good fuel behavior.

1.   INTRODUCTION

The importance of Quality Assurance/Quality Control activities in design and manufacturing of nuclear fuel as well as the necessity of a feedback of the quality related information obtained during the operation of the products respectively in connection with post-irradiation-inspection and performance evaluation is generally recognized.

In the performance of these tasks, which are part of a quality loop, normally various organizations and different groups within these organizations are involved, and their interaction must be carefully coordinated under an appropriate Quality Assurance System (QAS). To illustrate some essential characteristics of such a system, the following paper describes examples of the main functions and typical activities taken from the Quality Assurance System for Fuel Assemblies and Associated Core Components as applied in our organization, the Kraftwerk Union AG

(KWU) and in its affiliated Fuel Manufacturing Companies. The emphasis is put on
Quality Assurance (QA) rather than on Quality Control (QS). As explained in general
publications, (e.g. /1/) QA is the more comprehensive term and relates to the system.
Following a definition of the German standard, DIN 55350, QA means "all organiza-
tional and technical tasks directed towards the assurance of the quality of both
the design and its implementation". QA includes all planned and systematic activities
necessary to guarantee, with adequate confidence, that a system, component or
assembly will satisfactorily perform its intended function.

In that sense QA includes QC as "quality assurance actions which provide a
means to control and measure the characteristics of an item, process or facility in
accordance with established requirements." /1/ Examples for the QC-philosophy and
methods applied under our QA-System are described in other contributions to this
seminar. /2 to 5/.

2. REQUIREMENTS ON QUALITY ASSURANCE FOR DESIGN AND MANUFACTURING OF NUCLEAR FUEL

The basic Requirements of the Quality Assurance Rules and Regulations which
are applied for nuclear Power Plants and Nuclear Facilities have to be followed also
for the design and manufacturing of nuclear fuel assemblies. Some of these well-
known QA-Standards are shown in Fig. 1.

RBU   **KWU**

- Quality assurance for safety in nuclear power plants
  (50 - C - QA , IAEA)
- Quality assurance criteria for nuclear power plants
  (10 CFR 50 , App. B ; USA)
- Quality assurance program requirements for nuclear facilities
  (ANSI N 45.2 ; USA)
- Quality assurance program requirements for nuclear power plants
  (ANSI / ASME NQA - 1 - 1979 ; USA)
- General requirements for quality assurance
  (KTA 1401 ; FRG)

Basic Requirements on Quality Assurance for   **Fig.1**
nuclear power plants and nuclear facilities

The basis of every QA system, required by all rules and regulations, is a
written Quality Assurance Program. (QAP)

The Quality Assurance Program is a general document which defines the principles and objectives of the necessary QA-System, specific to the products to be designed and manufactured and specific to the organization. The program has to be authorized by the company management and declared to be binding.

3. EXAMPLES AND TYPICAL QA-FUNCTIONS AND ACTIVITIES UNDER OUR QA-SYSTEM FOR FUEL ASSEMBLIES AND CORE COMPONENTS

3.1 General

Under the overall responsibility of KWU, fuel assemblies for water reactors in a broad variety of types and configurations (PWR, BWR, HWR) have been developped, designed, manufactured and supplied to nuclear power plants. This was done in a broad but closely controlled cooperation between different organizational units, following the principles of our Quality Assurance System. The technical divisions for fuel technology as well as for fuel design and engineering belong to the KWU "Fuel Cycle Group." The first one is experimentally oriented, and its tasks include the development of manufacturing techniques, whereas the second prepares the manufacturing documents, (e.g. drawings, specifications etc.) partially with input from other engineering departments within the company. The direct fabrication work is done in our case by affiliated companies: nuclear fuel and fuel assemblies are made by Reaktor-Brennelement Union GmbH (RBU) respectively for mixed oxide fuels by ALKEM GmbH; the required Zircaloy tubes are produced by Nuklearrohr GmbH (NRG).

A joint Quality Assurance Program (QAP) has been developped and it is binding for all these organizations.It was originally established in 1975 on the basis of our own QA concept, on the German licensing practice and also under consideration of already existing rules. In its present form the QAP satisfies the basic requirements of national and international rules such as KTA 1401 (Germany),10 CFR 50 App.B (USA),IAEA Code of Practice 50-C-QA, ANSI-N 45.2 etc. It has been successfully used for the supply of fuel assemblies to many different countries, and the deliveries were not limited to reactors of our own design, they included also reload fuel for water reactors of foreign design, the so called third party reactors.

The philosophy and general form of our Quality Assurance System for Fuel Assemblies and Core Components could also be used in international cooperation, independent of organizational structures which are different from our own ones. A good example is our cooperation with NUCLEBRÁS in Brazil,where the know-how is transferred, local organizations are increasingly involved and the fuel manufacturing is already done in the Brazilian nuclear fuel fabrication plant.

RBU   **KWU**

Content:

1. QA-program, QA-system, QA-manual
2. Organization
3. Design control
4. Procurement control
5. Preparation and review of manufacturing, test and examination documents
6. Document control
7. Control of purchased parts and materials
8. Identification and control of materials, parts and components
9. Control of special processes
10. Tests examination and inspections
11. Procedure qualification tests
12. Control of manufacturing and test equipment
13. Handling, storage and shipping
14. Test and manufacturing status
15. Nonconforming materials, parts or components
16. Corrective actions
17. Documentation
18. Audits

18. points of the quality assurance program for fuel assemblies and associated core components   Fig. 2

Fig. 2 shows the structure of the QAP for fuel assemblies. It is not the purpose of this presentation to go through all of the 18 points of the QA-program, but rather to explain certain selected topics.

3.2 QA-procedures; QA-manual

The QA-program is more detailed than the QA-rules and regulations and must be specific to the organization and products to be manufactured, but it is not comprehensive enough to implement all the details of the QA-system.

Many additional supplementary documents are necessary to prescribe the adequate QA measures and procedures for design control, procurement control, inspection, control of special processes, control of non-conforming items and other main functions. Therefore the Quality Assurance Program for fuel assemblies and associated core components is supported by written procedures and instructions such as

- QA-procedures (system related)
- Procedures and instructions for engineering
- Procedures and instructions for manufacturing
- Procedures and instructions for inspection
- Manufacturing and inspection sequence plans
- etc.

These procedures, instructions and plans define and regulate in detail the individual activities which have to be followed by the persons which are directly in-

volved in the execution of quality assurance measures, independent of their organizational assignment to design, manufacturing or even development departments (e.g. to assure the validity of experimental design input data). The necessary control of interfaces between the individual companies and organizational units is a very important part in the QA-procedures. The procedures and documents for QA which are applied by our fuel design and manufacturing organizations are described in a QA-Manual together with a general summary of the overall QA-system.

The Quality Assurance Manual for UO_2 fuel assemblies and associated core components is a common handbook of KWU and RBU. Corresponding quality assurance manuals exist in relation to ALKEM for mixed oxide fuel and to NRG for fuel cladding tubes. Finally,separate quality assurance manuals exist also for subsuppliers of important semi-finished material or of components for fuel assemblies.

3.3 Design control

Important areas of design control are the control of design interfaces between the fuel design and engineering organization which gives input to,respectively receives feedback from many other units interacting with the fuel assembly organization,as well as the relations between design and manufacturing organizations (see fig. 3).

RBU   **KWU**

Interface areas in fuel assembly design   Fig.3
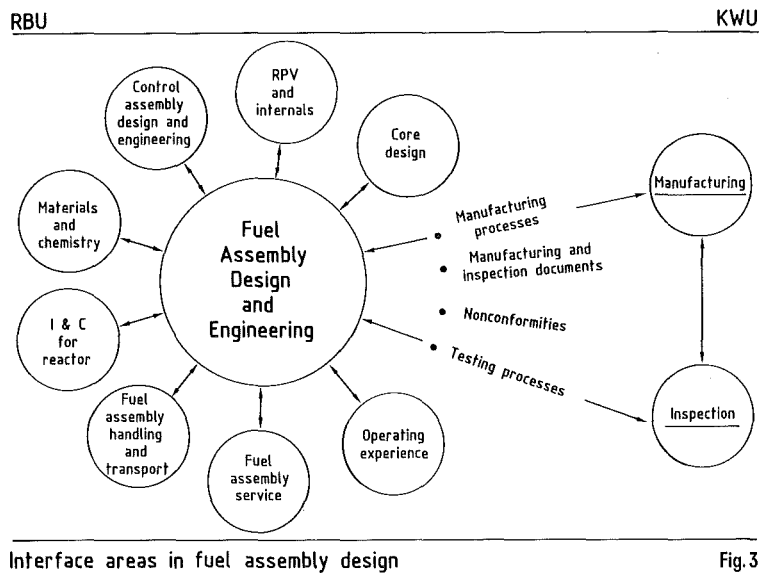
The interface control between the individual design units is very important to assure that verified input data and verified dimensions (compatibility) are used in fuel design and that the responsibilities for overlapping requirements between

the fuel design and other engineering departments are clearly established and assigned. Another important area is the appropriate consideration of the feedback resulting from experiments and operating experience.

Fig. 4 shows activities and interaction in design control with respect to design verification and design review.
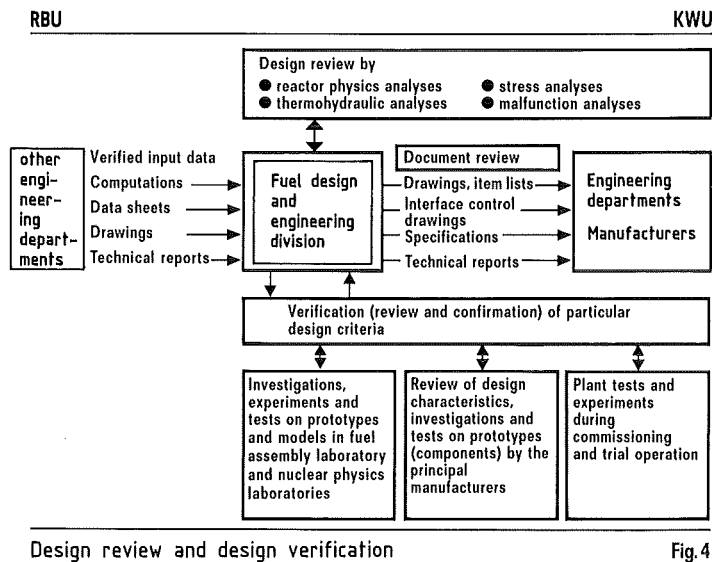
Design review and design verification   Fig. 4

3.4 Procurement control

The technical requirements for the product quality and for important manufacturing and inspection processes are defined by the design and engineering department in specifications and drawings.

These documents are the technical basis for the procurement of materials, parts and structural components from suppliers. Fig. 5 (next page) which is largely self explaining shows the supplier qualification by the design and manufacturing groups for products with special quality requirements (such as semi-finished products of Zry e.g. cladding tubes; structural components such as top and bottom pieces etc.) The fabrication is supervised by the purchasing organization. This can include inspections and audits at the suppliers as well as examinations of the products before delivery or upon receipt. The feedback of these quality related data is used for a continuous evaluation of the suppliers.

3.5 Inspection program

RBU   KWU

Selection of manufacturer (pre-selection)

⬤   General qualification

- Review of the QA-System (audit)
- Survey of manufacturing equipment
- Survey of test equipment

Selection and ordering

⬤   Product related qualification

- Review of the manufacturing and test documents

Trial manufacturing

- Supervision of trial manufacturing
- Analysis of the results obtained

Supplier evaluation

Release for manufacturing

Establishment of manufacturer's qualification   Fig. 5
for products with special quality requirements

The heart of the QA-program in this area is the manufacturing and examination sequence plan (FPP) which defines all the inspections required, the points in the process when inspections are to be made, the methods to be used, the scope or frequency of inspections and their documentation. Those FPPs exist for fuel assemblies respectively for major fuel assembly components such as fuel rods, spacers etc. With respect to the inspections they are supported by detailed instructions such as shop traveller, inspection instructions and examination procedures.

For the inspection of simple parts, for example end plugs, an Inspection Plan (PPL) is sufficient and used instead of the more comprehensive sequence plan. Both the FPPs and PPLs including the procedures for inspection and manufacturing have to be reviewed by the design and engineering organization units.

3.6 Qualification of special processes

Processes which may have an essential influence on the final performance of the product are defined as special processes. For example certain welding, heat treatment or cleaning procedures but also important tests can belong to this category. Before use of any special process in production, the process itself has to be qualified by a procedure qualification test.

Fig. 6 explains the course of a procedure qualification test. The qualification program has to be commonly established by the organizational units which are responsible for manufacturing, inspection, design and engineering. The procedure qualification has to be documented by a qualification report. The qualified process has to be released for using in production by the design and engineering organization units.

RBU	KWU
Planning (KWU/RBU)	Qualification program • Samples • Special tests and examinations • Process parameters to be considered
Execution (RBU)	Procedure qualification test • Production of samples • Measurements • Tests and examinations
Evaluation (RBU)	• Results of samples • Measuring and test results • Qualification report
Release (KWU/RBU)	• Assessment of the procedure qualification test • Definition of the procedure and parameters • Release of the procedure

Course of a procedure qualification test   Fig. 6

3.7 Control of non-conformities

The handling of products which deviate from design specifications respectively drawings or which have not been manufactured according to the required process specifications is specially addressed in the QA system. Any non-conforming items must be identified and segregated. A notification which contains all necessary information about the deviation has to be written by the QC-department of manufacturing. This notification must be reviewed by a "technical adviser" in the design department. Based on a thorough investigation of the foreseeable consequences of a deviation a decision has to be made about the eventual applicability of the product in the responsible design group in agreement with other engineering departments (e.g. material department) and the manufacturing organization. This can be

- reject
- use as is (if necessary with special provisions)
- rework or repair with reinspection

In cases, where it is not easily evident that the use of a deviating item does not adversly affect the product quality, independent experts or consultants acting for licensing authorities may be involved in the decision sequence.

3.8 Audits

Periodic as well as unscheduled (surprise) audits are performed by qualified QA personal (Auditors) in the manufacturing plants as well as in the design, engineering, or laboratory organizations. In case of any non compliance, corrective

actions are initiated to improve product quality as well as the effectiveness of the QA system.

4. SPECIFIC ASPECTS AND EXPERIENCE

By means of the binding application of a joint Quality Assurance Program in all organizational units contributing to the nuclear fuel assemblies and through all stages of the work - from the development, technology, design and engineering activities to the procurement of materials and manufacturing - a uniform basis of all QA actions is established. The interfaces are closely controlled, so that the influence of changes in one area on the subsequent steps become clearly evident.

The system enabels a quite flexible reaction on new design requirements. In this context the gradual introduction of design improvements such as low parasitic structure respectively Zircaloy spacers with optimized reload fuel into existing PWRs may be mentioned or the implementation of proven favorable design character- istics, for example reconstitutable assemblies, into third party plants based on the successful behavior in other reactors. Fortunately no compatibility problems were encountered in the supply of reload fuel due to effective design control in the fields of physics, thermohydraulic and mechanical design.

The quality assurance circuit for nuclear fuel technology and engineering is closed by formalizing also the feedback from operational fuel behavior and post irradiation examination into design. For this purpose we have a special product assurance group in engineering. Their work contributed to the successful remedy of some earlier fuel failure modes (e.g. hydriding or fretting). To avoid fretting, we measure 100 % of all our PWR spacer meshes on an automatic equipment which records also the contact forces between fuel rods and spacer springs, so that a means for immediate correction is provided if required.

In the manufacturing area a point of particular emphasis in our QA-system is the control of special processes. Quality must be planned and produced within the processes and should be only verified and documented by subsequent testing. Sort- ing out of non-conforming items is not the primary aim of QC-tests.

This can be illustrated by the welding processes between fuel rod clad and end plugs. The allowable range of geometrical and process parameters is determined during the procedure qualification tests and carefully controlled later on. Originally the welding was done by a TIG method and subsequently controlled by x-ray testing. Due to successful process control, resistance welding with alternate control methods was implemented. In both cases the weld quality is excellent and a subsequent He-leak test shows essentially zero indications.

In summary, the statement can be made that QA has contributed to the current very satisfactory performance of our LWR fuel which is characterized by an overall integrity of about 99.997% per cycle.

5. REFERENCES

/1/ IAEA, Vienna, 1983; Guidebook on Quality Control of Water Reactor Fuel, Technical Reports Series No.221, STI/DOC/10/221

/2/ H.G.Weidinger and K.-H.Kunz; Methods for QC of Zry Tubings, IAEA-Seminar 1983, Karlsruhe

/3/ H.Assmann and E.Steinberg; Statistical Methods in the Quality Control of UO_2 Pellets and Zircaloy Cladding Tubes for Water-Reactor Fuel Rods, IAEA-Seminar 1983, Karlsruhe

/4/ H.-J.Romeiser; Quality Control for PWR-Spacers by Means of a Computer Assisted Automated Test Equipment, IAEA-Seminar 1983, Karlsruhe

/5/ H.Engel; Use of a Software-System for the Quality Assurance of the Production of Fuel Elements for Power Reactors, IAEA-Seminar 1983, Karlsruhe

DISCUSSION

A. STRASSER: Have you developed a good non-destructive test for your Zircaloy end plug resistance welds?

R. HOLZER: Quality is controlled by process qualification and control of the manufacturing procedure.

R.S. RUSTAGI: In verification of design criteria, what are the specific tests done to simulate off-normal operating conditions or malfunctioning of fuel during reactor commissioning?

R. HOLZER: The tests are not specifically oriented towards experiencing off-normal conditions or malfunctioning. Their main aim is to verify the predictions of design models and methods in the allowable operating range of the plant and fuel.

K. BALARAMA MOORTHY: Are there any plans to adopt newer or novel welding methods such as Laser, or EBW for closure welds of fuel elements in your production plants?

R. HOLZER: Such methods are under consideration for development, but not used in current manufacturing.

FUEL MANUFACTURING FOLLOW AT FRAGEMA

G. PANCACCINI

FRAGEMA

149 Cours Lafayette

69006 LYON

FRAGEMA has set up a complete surveillance system totally independant of the one performed by manufacturers.

This paper exposes that system and the specific responsabilities of each party - subcontractors, manufacturers, FRAGEMA, customers.

The FRAGEMA involvment is presented throughout the fabrications starting with the subcontractors until the final release and shipment.

The data collection system which also allows feedbacks between design, manufacturing and reactor operation is briefly described.

I. POSITION PARTICULIERE DE FRAGEMA

FRAGEMA occupe, parmi les fournisseurs de combustible nucléaire une place particulière en ce qui concerne les fabrications.

Les sociétés FRAMATOME et COGEMA ont pour vocation la fourniture de matériels et de services parfaitement complémentaires : Chaudières nucléaires et cycle du combustible.

Parmi ces fournitures, la fabrication des éléments combustibles constitue le carrefour où le cycle du combustible interfère directement avec les performances des chaudières.

Reconnaissant ce fait, FRAMATOME et COGEMA ont convenu de s'associer pour unir leurs capacités et leurs compétences, afin de créer une filiale où ils sont associés à parts égales.

Cette filiale, FRAGEMA, a été mise en place le 1er Juillet 1981. Cette société, constituée autour de l'ancienne Division Combustible de FRAMATOME, conserve avec cette dernière, des liens tels, que le développement des éléments combustibles puisse être effectué avec toutes les garanties souhaitables quant à leur bonne intégration dans les chaudières. FRAGEMA continue également à s'appuyer sur les moyens expérimentaux considérables du CEA (Commissariat à l'Energie Atomique) qui ont déjà été largement mis à contribution pour le développement de l'Assemblage Combustible Avancé(AFA).

Dans ce contexte, FRAGEMA, Société d'ingéniérie, sous-traite la fabrication de combustible à deux contractants. Le premier, historiquement parlant, FBFC (Franco Belge de Fabrication de Combustible) avec ses deux usines de ROMANS (FRANCE) et de DESSEL (BELGIQUE) a une capacité totale de 1100 T par an d'uranium contenu. Le second, CFC (COGEMA-FRAMATOME) filiale, comme son nom l'indique de COGEMA et FRAMATOME, a été créée en même temps que FRAGEMA. Dimensionnée dans un premier temps pour 500 T par an d'uranium contenu, cette usine située sur le site de PIERRELATTE vient d'être opérationnelle. C'est pourquoi dans ce système relationnel, pour s'assurer de la conformité des produits livrés, FRAGEMA a été amenée à élaborer un système complet de surveillance des fabrications totalement indépendant de celui réalisé par ses contractants.

II. DOMAINES D'INTERVENTION DE FRAGEMA

2.1. Sous-traitants :

Sont ainsi appelés les fournisseurs des fabricants d'éléments combustibles.

2.1.1. Responsabilités

Directement responsable de ses fournisseurs, le contractant doit évaluer leur aptitude à satisfaire les différentes exigences applicables et notamment celles d'Assurance de la Qualité ; c'est-à-dire plus précisément:

- prononcer l'agrément du fournisseur,
- prononcer la qualification du processus de fabrication et/ou du produit commandé, lorsque celui-ci entre dans son domaine de responsabilité,
- passer des commandes aux fournisseurs dûment agréés et qualifiés,
- s'assurer du bon déroulement des opérations relatives aux commandes,
- s'assurer de la conformité des produits et matériels livrés par rapport au dossier technique référencé dans les commandes.

Dans ce schéma, FRAGEMA doit pour sa part élaborer les programmes de qualification et prononcer la qualification des produits et lignes de fabrication dans son domaine de responsabilité (selon les produits concernés).

Par ailleurs, les contractants adressent à FRAGEMA copie des commandes passées à leurs sous-traitants. FRAGEMA vérifie que celles-ci sont conformes au dossier technique d'application à la date de la passation.

2.1.2. Enquête Qualité

Pendant la réalisation de la commande et selon l'importance et l'étendue de la fourniture, le contractant effectue des enquêtes périodiques. Ces enquêtes ont pour but de s'assurer du respect des exigences de la commande et, le cas échéant, de faire mettre en place les actions correctives par le fournisseur.
FRAGEMA est associé à ces enquêtes en tant qu'observateur.

2.1.3. Réception

Après réalisation de la commande, le contractant réceptionne les produits et s'assure de leur conformité aux exigences du dossier technique.

Lors de ces recettes, les représentants de FRAGEMA, les représentants mandatés de ses clients sont formellement convoqués.

FRAGEMA assure lors de ces recettes une surveillance définie comme suit :

- Chaque sous-traitant sera visité au moins une fois par an,
- La fréquence d'intervention dépend de la catégorie(1) du composant et du niveau de surveillance tel que décrit dans le tableau ci-après :

CATEGORIE NIVEAU DE SURVEILLANCE	I	II
Normal	50 %	30 %
Réduit	30 %	15 %
Renforcé	100 %	50 %

Les premières interventions sont effectuées sur la base d'une surveillance normale ; les passages d'un niveau de surveillance à un autre étant alors précisés comme suit :

Normal ⟶ Renforcé : non conformité de même nature lors de 2 interventions consécutives

Renforcé ⟶ Normal : pas de non-conformité lors de 2 interventions consécutives

Normal ⟶ Réduit : pas de non-conformité lors de 3 interventions consécutives

Réduit ⟶ Normal : à la première non conformité lors d'une intervention

(1) Voir paragraphe III.2

2.2. Contractants

Au-delà de l'aspect conception pour lequel FRAGEMA est le seul maître-d'oeuvre, FRAGEMA a la responsabilité, vis-à-vis de ses clients, de l'Assurance de la Qualité de la fourniture d'éléments combustibles. A cet effet, FRAGEMA impose la mise en oeuvre, chez le contractant et ses sous-traitants, d'un Programme d'Assurance de Qualité et demande aux contractants de rédiger et transmettre pour approbation un Manuel d'Assurance de Qualité décrivant l'organisation mise en place.

Indépendamment de la surveillance des opérations de qualification, fabrication, contrôle, recette, qui seront présentés en détail ultérieurement, FRAGEMA se réserve le droit d'enquêter périodiquement chez les contractants afin de s'assurer que les dispositions prévues par le Programme d'Assurance Qualité soient strictement respectées.

2.3. Clients

Le système relationnel mis en place entre clients - FRAGEMA - fabricants - sous-traitants est très complet et les prérogatives de chaque partenaire sont clairement définies. Ainsi FRAGEMA est le seul interlocuteur habilité de ses clients, et toutes relations avec les autres partenaires doivent passer par cette plaque tournante qu'est FRAGEMA. Les relations contractuelles Client-FRAGEMA sont cependant susceptibles de varier de contrat à contrat.

III. SUIVI DE FRAGEMA EN USINE

Ce paragraphe présente successivement les éléments techniques du suivi des fabrications tels qu'exécutés par FRAGEMA à savoir :

- Qualifications,
- Points de surveillance et d'arrêt,
- Enquête d'atelier/enquête opération,
- Système de gestion des informations.

3.1. Qualifications

FRAGEMA a défini dans ses exigences les procédés qui réclament une attention particulière. Il s'agit des procédés qui influencent largement les propriétés ou caractéristiques d'un matériau ou composant, ou qui ne peuvent plus être vérifiés sur les produits finis. Les conditions d'exécution sont alors formellement qualifiées par FRAGEMA lorsqu'il s'agit d'un procédé global ou les contractants pour les procédés particuliers.

3.1.1. Qualification des modes opératoires de fabrication et de contrôle

On peut distinguer trois grandes catégories en ce qui concerne ces qualifications. Elles concernent : le contrôle non destructif, les procédés de liaison et les traitements thermiques.

Pour les deux premières catégories, FRAGEMA et ses contractants élaborent un dossier regroupant les exigences requises pour la qualification concernée, ressuage ou soudage par exemple et il appartient alors au contractant de gérer le personnel qualifié et de reprononcer les qualifications à échéance. FRAGEMA exerce une surveillance lors de ces requalifications qui peut être normale (50%), réduite (20%), ou renforcée (100%) suivant les résultats précédents.

En ce qui concerne les traitements thermiques, les fours doivent être instrumentés et les paramètres enregistrés à chaque traitement. La traçabilité doit également être sans ambiguité.

La durée de validité des qualifications est définie pour chaque cas par le dossier technique FRAGEMA et est généralement de l'ordre de deux ans.

3.1.2. Qualification des lignes de produits

Le système de surveillance mis en place par FRAGEMA est directement fondé sur la nécessité d'une qualification des moyens de fabrication des produits concernés. La qualification est prononcée par FRAGEMA, suivant les critères d'un programme de qualification, établi d'un commun accord avec les contractants.

3.1.3. Qualification pour les composants de catégorie I réalisés par

le fabricant de combustible

Avant que la réalisation de la présérie de qualification ne débute, FRAGEMA s'assure des moyens de fabrication, de contrôle et d'étalonnage, et que les gammes de fabrication et de contrôle sont discutées avec les fabricants. Lors de la réalisation de la présérie, FRAGEMA délègue un représentant pour suivre les principales étapes de la réalisation.

La qualification n'est accordée que si la présérie est conforme au dossier technique et aux exigences des programmes de qualification.

3.2. Points de surveillance et d'arrêt

Il est de la responsabilité de FRAGEMA de définir à ses fabricants les exigences qu'elle a concernant les contrôles des produits et ceux qui doivent être soumis à surveillance par FRAGEMA.

Dans ce schéma, le fabricant doit élaborer les gammes de contrôle et les faire approuver par FRAGEMA, certifier la conformité des produits et documenter les résultats.

3.2.1. Contrôles fabricants

Avant d'expliciter la surveillance FRAGEMA, il faut présenter les différents types des contrôles fabricants. Ces contrôles sont de deux sortes :

- Contrôle direct du produit
  Cette catégorie de contrôle doit être prévue à certaines étapes de la fabrication. La documentation doit expliciter la méthode (mesure, examen visuel, contrôle non destructif) et les critères d'acceptation.

- Contrôle des opérations
  Cette catégorie de contrôle consiste à vérifier, à intervalles préalablement définis, que les opérations de fabrication s'effectuent de façon satisfaisante.

Ces contrôles portent notamment sur la vérification :

- des paramètres de fabrication préalablement qualifiés,
- des conditions de fabrication.

et de façon plus générale sur les facteurs qui ne peuvent être vérifiés à postériori.

3.2.2. Surveillance de FRAGEMA

La surveillance exercée par FRAGEMA n'a en aucun cas pour effet de diminuer la responsabilité du fabricant qui doit mettre en oeuvre tous contrôles, épreuves, essais ou autres actions qui lui paraissent nécessaires pour obtenir l'assurance que le produit dont il a la charge est conforme au dossier technique, mais consiste en une vérification organisée à toutes les étapes essentielles de la fabrication des produits. Par ailleurs, la fabrication ne pourra se poursuivre qu'après accord formel de FRAGEMA en ce qui concerne les produits soumis à point d'arrêt. Ces produits -les pastilles UO2, les crayons combustibles, les assemblages combustibles et les grappes terminées- ne doivent être présentés qu'après contrôle final par le fabricant.

A chaque point d'arrêt ou de surveillance FRAGEMA exerce un contrôle des produits présentés en appliquant un plan de surveillance préalablement défini. A titre d'exemple, les schémas suivants ont été retenus :

. Pour les pastilles UO2 : contrôles volants journaliers suivant NFX 06-022 avec les NQA définis dans le dossier technique FRAGEMA.

. Pour les soudures de crayons combustibles, application de la NFX 06-022, NQA 0,1 à la relecture de clichés radiographiques.

. Pour le contrôle visuel des assemblages, application d'un plan CSP-1 de Dodge avec un AOQL de 0.4

De façon générale, lors d'une présentation d'un produit à un point de surveillance, le fabricant doit présenter un dossier justifiant la conformité des pièces présentées.

3.2.3. Non-conformités

FRAGEMA, associé à ses contractants, a élaboré un système précis définissant les exigences en matière de non-conformité des matériaux, pièces ou composants.

Ce schéma inclut les notions importantes suivantes :

- non-conformité : écart détecté entre la valeur réelle d'une caractéristique et la valeur prescrite par une exigence (plan, spécification...),
- incident : tout évènement susceptible d'affecter la qualité d'un produit y compris tous les écarts aux exigences techniques autres que les non-conformités,
- réparation : action documentée qui a pour objet de redresser une non-conformité,
- anomalie : non-conformité qui ne peut être éliminée par réparation, tri, etc... et qui est donc persistante.

Chacun de ces points fait l'objet d'investigations et d'un traitement par le contractant dans le cadre de ses compétences et par FRAGEMA pour les cas autres que les non-conformités telles que définies ci-dessus.

3.3. Enquête d'atelier - Enquête opération

L'objectif de ces enquêtes effectuées par FRAGEMA dans les usines de fabrication, est de s'assurer, que les exigences du dossier technique relatives aux conditions de fabrication et de contrôle sont respectées et ce, par une observation sur place, au cours d'opération de ces conditions de fabrication et de contrôle.

Ce type d'enquête est réalisé annuellement par site de fabrication pour chacun des domaines ci-après :

- pastillage,
- crayonnage,
- grilles,
- assemblages (squelette, tube-guide),
- grappes annexes,
- embouts supérieurs et inférieurs,
- manutention (stockage, mise en conteneur).

3.4. Système de gestion des informations

Au-delà de la fabrication, l'objectif d'un système de gestion des informations est de permettre une recherche rapide sur l'identification, la localisation et l'histoire de tout composant ou ensemble.

Le système de traçabilité défini par FRAGEMA, mis en place par les contractants, doit permettre de retrouver tout au long de la vie du produit l'origine de toute matière, pièce et composant.

L'ordinateur central basé à LYON (IBM 4331) permet de gérer les quelques 800 000 informations de traçabilité appelées par le dossier technique pour la fabrication d'un coeur de combustible. FRAGEMA utilise avec les usines, une ligne de transmission de données qui est une extension du système informatique.

Les différents fichiers contiennent les informations produites lors des contrôles de fabrication. Ces informations classées suivant des critères préalablement définis, subissent un traitement systématique permettant de s'assurer que le produit final, assemblages et grappes, est conforme.

Un "dossier d'identification" -par assemblage ou grappe- clé d'entrée de toutes les informations de traçabilité, est par ailleurs remis au client lors de la livraison des éléments. Ce dossier contient également la référence des anomalies acceptées pour les composants de cet assemblage ou grappe.

IV. NIVEAUX DE QUALITE

L'estimation de la qualité, effectuée par FRAGEMA, a pour but de situer le niveau de qualité du produit dans le cadre des spécifications et tolérances pour un certain nombre de caractéristiques. Cette estimation est effectuée pour les produits suivants : pastilles UO2, poudre d'UO2, crayons, squelettes, grilles, assemblages combustibles.

L'ensemble des données fournies, dans le cadre de la traçabilité et de l'estimation de la qualité, constitue la base du système d'analyse de la qualité de FRAGEMA.

Pour ce faire, FRAGEMA édite périodiquement des états de Qualité. Ces documents sont le résultat des traitements de l'information ; ils donnent une synthèse de la qualité en se dégageant de la multitude des informations pour ne garder que celles qui représentent les pivots de la qualité.

L'intérêt de l'analyse est de donner à l'information toute sa puissance et d'avoir ainsi une technique d'aide à la décision dans le cadre d'une connaissance du produit à tous les stades de son élaboration. La portée de l'analyse de la qualité est notamment à remarquer dans les domaines suivants :

- Suivi de fabrication

  Le produit est suivi aux divers stades de la fabrication et FRAGEMA peut ainsi la piloter en imposant des mesures préventives sur la base de l'exploitation statistique de résultats de contrôle (calcul de variance, tendance, etc...). L'Assurance Qualité joue alors efficacement son rôle en promouvant des mesures préventives plutôt qu'en imposant des actions correctives.

- Synthèse de la qualité

  La constitution des états de qualité où sont enregistrés les identificateurs et les résultats de mesures pour toutes les caractéristiques contrôlées, constitue la preuve de la conformité.

Pour conforter les différents résultats obtenus par les contractants, FRAGEMA procède régulièrement à des séries de contre-essais. Cette surveillance par contre-essais porte principalement sur les tests destructifs appelés par le dossier technique FRAGEMA et ce pour les produits de catégorie I.

V. RETOUR D'EXPERIENCE ENTRE EXPLOITATION CONCEPTION ET DE FABRICATION

Exploitation des résultats d'irradiation

A l'aide de l'analyse de corrélation, les résultats d'irradiation sont interprétés de façon à déceler les raisons des comportements particuliers du combustible et éventuellement déterminer l'origine d'un incident. La banque de données de fabrication peut permettre d'identifier les processus de fabrication, ou les caractéristiques des produits responsables de ce comportement, et conduire ainsi à une évolution du Dossier Technique dans le domaine de la conception ou de la fabrication.

VI. CONCLUSION

Les divers aspects du suivi des fabrications exposés dans ce document, soulignent l'importance de l'organisation mise en place par FRAGEMA. Elle est un élément fondamental de l'Assurance Qualité et l'automatisation de l'acquisition des données est un moyen efficace pour garantir la conformité des produits fabriqués.

Ce système est bien le centre des données, indispensable, sans lequel il serait impossible de tirer une conclusion de la multitude des informations disponibles. Il représente en outre un facteur d'économie par le niveau de sécurité qu'il apporte aux décisions.

Le suivi des fabrications de combustible, exercé de manière continue par un concepteur, permet d'intégrer les différentes contraintes de conception et de fabrication dans un produit pour lequel une fiabilité accrue est sans cesse recherchée.

ТРЕБОВАНИЯ К ТЕПЛОВЫДЕЛЯЮЩИМ ЭЛЕМЕНТАМ РЕАКТОРА
ВВЭР-1000 И ОБЪЕМ КОНТРОЛЯ ПРИ ИХ ПРОИЗВОДСТВЕ

Решетников Ф.Г., Головнин И.С., Бибилашвили Ю.К.,
Соколов Ф.Ф., Суханов Г.И.

Всесоюзный научно-исследовательский институт
    неорганических материалов

Москва, СССР, 1984.

АННОТАЦИЯ

В докладе описывается роль водо-водяных реакторов (ВВЭР) в программе развития атомной энергетики Советского Союза. Приведены основные условия эксплуатации и характеристики твэлов ВВЭР-1000.

Описаны требования, выдвигаемые к конструкции твэла в целом и к ее составным частям.

Приведены основные параметры, которые необходимо контролировать при производстве твэлов с точки зрения обеспечения работоспособности элементов в условиях эксплуатации АЭС с реакторами ВВЭР-1000.

В программе развития атомной энергетики Советского Союза водо-водяным реакторам (ВВЭР) отводится значительная роль.

На протяжении многих лет успешно эксплуатируются реакторы ВВЭР-440 на Нововоронежской, Кольской, Армянской и других АЭС.

Пущены и в настоящее время успешно эксплуатируются 5-й блок НВАЭС (1980 г.) и 1-й блок ЮУАЭС (1982 г.) с реакторами ВВЭР-1000. В ближайшие годы планируется ввести в строй еще ряд блоков с реакторами единичной мощностью, равной 1 млн.кВт. (табл.1) [1 , 2].

Стабильная, хорошо отработанная технология производства твэлов и ТВС, правильно выбранные конструктивные решения обеспечивают высокую надежность активных зон реакторов действующих АЭС. На первом этапе разработки твэлов для реакторов типа ВВЭР-1000 решалась главная задача - создание надежной конструкции твэла. При этом, естественно, отдельные, наиболее важные требования, предъявляемые к исходным параметрам твэла, обеспечивались введением более строгого контроля как технологического процесса, так и готовой продукции. Существенное возрастание масштабов производства потребовало на следующем этапе с одной стороны совершенно по иному подойти к организации серийного поточного производства твэлов и тепловыделяющих сборок, с другой - провести большой объем работ в направлении дальнейшего совершенствования комплексной системы управления качеством продукции.

Условия эксплуатации и характеристики твэлов ВВЭР-1000 представлены в табл.2,3.

Требования, выдвигаемые к конструкции твэла в целом и к ее составным частям, разрабатываются на основе анализа условий эксплуатации и изучения процессов, определяющих работоспособность твэлов с учетом экономических аспектов. Часть требований устанавливается исходя из нейтронно-физических и гидродинамических характеристик активной зоны (табл. 4 и 5).

При определении номинальных значений тех или иных исходных параметров твэла и допустимых отклонений от этих значений рассматриваются предельные эксплуатационные режимы. При этом выбирается достаточный запас между предельным эксплуатационным уровнем и уровнем отказа. Выбор параметров, контролируемых при производстве твэлов, и установление планов контроля осуществляются на основании анализа влияния самих параметров и величины их отклонения от номинальных значений на те факторы, которые определяют необходимость выдерживать

параметры в требуемых пределах.

При установлении планов контроля и уровня дефектности готовой продукции исходные параметры твэлов, из условий надежности конструкции, принято подразделять на три группы.

I группа - наиболее важные параметры твэлов или критические, т.е. те, которые решающим образом сказываются на работоспособности твэлов. Несоблюдение требований по этим параметрам могут быть причиной преждевременного выхода твэлов из строя, а при аварийных ситуациях вызывать такие повреждения, которые выходят за критерии, по которым оценивается безопасность активной зоны: максимальная температура на оболочке не более 1200 $^\circ$C, отсутствие плавления топлива, величина окисления оболочки не выше 18 % первоначальной толщины стенки , при заливке холодной водой оболочки не должны разрушаться[3].

К I группе параметров твэлов относятся:
- содержание водорода, влаги, фтора, хлора, углерода, азота в топливных таблетках;
- стехиометрия топлива;
- плотность топлива;
- диаметр топливных таблеток и геометрия оболочек твэла;
- дефекты в оболочке и готовом твэле;
- герметичность твэлов;
- качество сварных швов;
- технологические газы в топливе;
- ориентация гидридов в оболочках;
- микроструктура топлива;
- давление гелия в готовом твэле;
- загрязненность фтором внутренней поверхности оболочек;
- сплошность топливного столба.

Для параметров I-й группы необходим либо сплошной контроль, либо статистический с минимальным уровнем дефектности.

Для автоматически регулируемых технологических режимов величины отдельных параметров этой группы (микроструктура, плотность топлива, стехиометрия и др.) можно обеспечить в требуемых пределах технологией производства.

Ко 2-й группе исходных параметров твэлов относятся такие параметры, отклонения от которых могут привести к некоторому ухудшению технико-экономических показателей АЭС или отклонения этих характеристик от требований нормативно-технической документации можно

достаточно уверенно оценить расчетными исследованиями.

Ко 2-й группе относятся:

- свойства и структура материала оболочек, включая анизотропию свойств, коррозионную стойкость;
- доспекаемость топлива;
- овальность оболочек;
- обогащение топлива;
- суммарный борный эквивалент;
- высота таблеток;
- внешний вид таблеток;
- масса топлива в твэле;
- загрязненность наружной поверхности твэлов фтором.

Для 2-й группы применяются статистические виды контроля.

Для автоматически регулируемых технологических режимов многие параметры этой группы можно обеспечить в требуемых пределах технологией производства.

К 3-й группе относятся все остальные параметры.

Параметры этой группы либо обеспечиваются технологией, либо статистическими планами контроля.

Необходимость тщательного контроля параметров твэла, относящихся к 1-й группе можно проиллюстрировать несколькими примерами.

Одной из причин выхода твэлов из строя может быть локальное гидрирование внутренней поверхности оболочек с быстрым развитием сквозных трещин. Это происходит в результате воздействия остаточной и сорбируемой влаги и водорода, в том числе в виде углеводородов, находящихся в топливе и во внутреннем объеме твэла. Поэтому контроль таких примесей как влага, водород, является необходимым .

Применение топливных таблеток, плотностью не менее $10,4$ г/см^3 и контроль этого параметра, позволили снизить общее содержание влаги в топливном сердечнике до $0,0003$-$0,0004$ мас %, что резко уменьшило вероятность разрушения твэлов из-за локального гидрирования.

Известно, что повышенное содержание фтора и хлора под оболочкой твэла, может быть причиной повреждения защитной окисной пленки на внутренней поверхности оболочки. При потере защитных свойств окисной пленки резко возрастает вероятность и локального гидрирования и коррозии под напряжением оболочек твэл.

Допуск на диаметр таблеток вместе с допуском на внутренний диаметр оболочки гарантирует на начальной стадии эксплуатации (све-

жее топливо) заданную величину диаметрального зазора между топливом и оболочкой и, соответственно, его теплопроводность.

При увеличении этого зазора ухудшается теплосъем с топливного сердечника, что приводит к повышению температуры в центре топливного столба, увеличивается количество тепла, аккумулированного топливом. Это обстоятельство является особенно принципиальным для оценки аварии типа $LOCA$. Влияние этого типа аварии на поведение твэлов в консервативных предположениях проводится для "свежего топлива".

Контроль геометрических размеров перечисленных компонентов позволяет выдерживать оптимальную величину зазора, обеспечивающую необходимую теплопроводность и технологичность при сборке.

Наличие разрывов топливного столба более чем 3-5 мм ведет к заметному всплеску энерговыделения в твэлах, что влечет за собой увеличение средней температуры оболочки и топлива. Кроме того в переходном режиме в местах разрыва при заклинивании таблеток уровень напряжений выше.

Заполнение твэлов гелием с исходным противодавлением до 20-25 кг/см^2 позволяет не только предотвратить смятие оболочки, но и улучшить термомеханические характеристики твэла, особенно в переходных режимах эксплуатации[1].

Контроль сплошности топливного столба и наличие исходного давления во многом влияет на работоспособность твэлов

Аналогичные рассуждения можно было бы сделать и для всех остальных параметров твэлов.

Безусловно и очевидно, что обеспечение всех требований к твэлу должно прежде всего гарантироваться технологией изготовления, стабильностью технологического процесса, оптимальностью разработанных технологических режимов. Хорошо отработанная и стабильная технология с высокой степенью автоматизации и механизации процессов позволяет обеспечивать промышленный выпуск продукции высокого качества, а внедрение в производство автоматически регулируемых технологических режимов дает возможность снизить объем контрольных операций, что, конечно, приведет к уменьшению себестоимости изготовления твэлов, поскольку контрольные операции составляют существенную долю в общей себестоимости производства твэлов.

Таблица I. Действующие, строящиеся и планируемые до 1990 г.
АЭС с реакторами ВВЭР

№№ пп	Наименование	Число блоков	Мощность МВт	Примечание
I.	Нововоронежская 2,455 Гвт	I	210	Опытно промышл.
		2	365	
		3	440	Серийный
		4	440	
		5	1000	Серийный
2.	Кольская 1,7600 Гвт	х4	440	Заполярье холод- ные северные условия
3.	Октемберянская 0,810 Гвт	х2	405	Высота 1100 м Зона сейсмичес- ких условий
4.	Ровенская 3,88 Гвт	х2 х3	440 1000	
5.	Южно-Украинская 4,0 Гвт	х4	1000	
6.	Калининская 4,0 Гвт	х4	1000	
7.	Запорожская	х4	1000	
8.	Хмельницкая	х4	1000	
9.	Ростовская	х4	1000	
10.	Балаковская	х4	1000	
11.	Крымская	х2	1000	
12.	Куйбышевская	х3	1000	
13.	Башкирская	х3	1000	
	Итого		40,9 Гвт	

Таблица 2. Условия эксплуатации и характеристики
твэлов реактора ВВЭР-1000 (5-й блок НВАЭС)

Наименование параметра	Размерность	Тип твэла B-1000
1. Максимальные удельные нагрузки на твэл	Вт/см	525
2. Давление теплоносителя в активной зоне	кг/см^2	160
3. Максимальная скорость теплоносителя	м/сек	6
4. Температура теплоносителя на входе/выходе	°С	290/322
5. Максимальная температура оболочки твэла (наружней поверхности)	°С	350
6. Давление гелия под оболочкой твэла	кг/см^2	20-25

Таблица 3. Наиболее важные исходные характеристики
твэла ВВЭР-1000

Наименование параметра	Размерность	Значение параметра	
		по ТУ	средне-статисти-ческое
Плотность топлива	г/см^3	10,4-10,8	10,5-10,6
Содержание влаги	мас.%	0,0007	0,0003--0,0004
Отношение высоты топливной таблетки к диаметру	-	1,2-1,5	1,2-1,4
Состав внутритвэльной среды:	объем.%		
- содержание гелия		95	97-99
- содержание примесей		5	1-3
Давление гелия под оболочкой	кг/см^2	20-25	20-25
Величина диаметрального зазора между топливом и оболочкой	мм	0,19-0,32	0,25

Таблица 4. Общие требования к контролю

Факторы, определяющие требования к контролю	Контролируемые параметры
Физика активной зоны	Обогащение топлива Примеси в топливе, суммарный борный эквивалент Вес топлива в твэле Длина топливного столба Суммарный и единичные зазоры между таблетками
Безопасность (аварийные режимы)	Диаметр топливной таблетки Внутренний диаметр оболочки Суммарный и единичные зазоры между таблетками Плотность топливной таблетки Доспекаемость топлива
Конструкция ТВС и теплогидравлика	Наружный диаметр твэла Длина твэла Вес топлива в твэле Разрывы в топливном столбе
Возможность эксплуатации дефектного твэла до плановой остановки реактора	Содержание примесей в топливе (углерод, азот и др.) Стехиометрия топлива Плотность топлива

Таблица 5. Работоспособность твэла и требования к контролю

№№ пп	Контролируемый параметр	Характеристики, определяющие работоспособность твэла
I.	Общее содержание водорода в топливе	- Локальное гидрирование
2.	Содержание влаги в топливе и под оболочкой	- ВТО
3.	Содержание фтора и хлора в топливе	-
4.	Количество и состав технологических газов в топливе	- Коррозия под напряжением
5.	Загрязненность внутренней поверхности оболочек фтором	
6.	Стехиометрия топлива	
7.	Дефекты в оболочках	- ВТО
8.	Содержание азота и углерода в топливе	
9.	Механические свойства и анизотропия механических характеристик оболочек	
I0.	Содержание водорода, ориентация гидридов и текстура оболочек	
II.	Микроструктура и внешний вид топлива	- Коррозия под напряжением
I2.	Шероховатость поверхности таблетки и оболочки	- Выход осколочных продуктов деления (ГПД,ЛЛП)
I3.	Геометрия топливных таблеток и оболочек	- Устойчивость оболочек
I4.	Сплошность топливного столба	
I5.	Давление гелия в готовом твэле	- Устойчивость оболочек
I6.	Доспекаемость топлива	- Коррозия со стороны теплоносителя
I7.	Внешний вид и геометрия готового твэла	
I8.	Примеси в оболочке	- "Нодулярная"коррозия
I9.	Однородность структуры оболочек	- Коррозия со стороны теплоносителя
20.	Внешний вид твэла	- Коррозия под напряжением
2I.	Металлургическое состояние оболочек	
22.	Загрязненность оболочек фтором	
23.	Привес при автоклавных испытаниях	
24.	Контроль на гелиевых течеискателях	- Герметичность твэла;
25.	УЗД и радиографический контроль сварных швов	- Качество сварных швов

ЛИТЕРАТУРА

1. Решетников Ф.Г. и др. "Состояние и перспективы развития производства и использования ядерного топлива в реакторах типа ВВЭР. Nuclear Power Experience (Proc. Intern. conf., Vienna, 13-17 Sept. 1982), v. 3, IAEA, 1983, p. 621.

2. Суханов Г.И. "Состояние и развитие работ в области поведения и рабочих характеристик твэлов энергетических водяных реакторов в СССР. Доклад на заседании Международной Рабочей Группы, МАГАТЭ, Вена, 1982 г.

3. Общие положения обеспечения безопасности атомных станций при проектировании, сооружении и эксплуатации (ОПБ-82). "Атомная Энергия", т.54, вып.2, февраль 1983, стр. 151-160.

CONSIDERATIONS SUR LE CONTROLE DE LA QUALITE
DANS LA FABRICATION D'ASSEMBLAGES COMBUSTIBLES
POUR DES REACTEURS A UO$_2$ NATUREL

C.HAVRIS
Institut de Réacteurs Nucléaires
Enérgetiques - Pitesti - Roumanie

ABSTRACT

The paper points out the experience in the fabrication
and quality control of the fuel assembly. The following problems
are described :

- the conception phylosophy for "Inspection and Test Plan"
- the adequate plans choice for the fuel assembly quality
control
- the combined utilization modality of the standards for
the inspection by attributes and the inspection by
variables for percent defectives.
- the problem of the acceptable quality level and its
determination,
- the effects of the AQL choice for the fuel quality
evaluation.

The results and the statistical evaluation of some data
obtained in the fuel assembly quality control are presented at
the end of the paper.

1. INTRODUCTION

L'élément fondamental dans le contrôle de la qualité des
lots de grappes combustibles est le procédé d'obtenir l'informa-
tion sur la qualité du lot et la décision d'acceptation de ce
procédé conformément a un critère basé sur la théorie de la sta-
tistique mathématique. On connaît généralement ce procédé sous le
nom de plan de vérification de la qualité. Le choix d'un certain
type de plan de vérification des lots de produits est subordonné
au specifique de la situation pratique et au but suivi par l'ap-
plication du plan.

Dans les conditions de la production industrielle la
conception de la vérification de la qualité 100% pour chaque

caractéristiques de la grappe de combustible ne peut constituer quand même une solution, étant considérée non-economique, inefficace, et dans la majorité des cas impossible de realiser pratiquement.

De plus, cette manière de vérification de la qualité se trouve en contradiction avec la caractéristique principale de l'organisation de la production industrielle : la réalisation des colectivités de grappes combustibles liées par des proprietés statistiques.

Le caractère non économique de la verification de la qualité 100% est mis en évidence par les coûts exagerés, aussi bien que par le fait que par cela on ne pouvait pas réaliser un remède de fond de la qualité, mais seulement un remède de forme, par le triaje des grappes conformes de ceux non-conformes du point de vue de leur qualité.

De même, on peut affirmer que la vérification 100% n'est pas efficace 100%. La cause est retrouvée dans la fatigue des ouvriers qui s'occupent du contrôle, traduite par la diminution progressive de l'attention, sourtout lorsque les grappes combustibles sont nombreux dans le lot. Paradoxalement, la charge du renvoi des grappes avec des défauts devient plus dificile lorsque leur nombre est petit. Le phenomène est de nature psychologique, avec un petit nombre de grappes non-conformes du point de vue de la qualité impliquant une frequence réduite du stimulus pour maintenir vife l'atention des contrôlleurs.

On peut mentionner que lorsque les essays de vérification de la qualité sont destructives (la vérification de la résistance à la corrosion, la vérification de la résistance à la torsion des soudures d'assemblage de la plaque d'extremité, etc) la vérification de la qualité 100% est absolument inapliquable.

La vérification de la qualité 100% (essai nondistructive) est imposée seulement pour la caractéristique critique dans l'utilisation des grappes de combustible. Par exemple : la longueure de la grappe; la cote H (fig. 3.2.1) jusqu'a la mise au point du processus de sudage.

Mais, cette vérification ne représente pas une triage.

2. CONCEPTION ET MISE AU POINT D'UN PLAN DE CONTROLE
   DE LA QUALITE ET ESSAIS POUR LE CONTROLE FINAL
   DES GRAPPES.

2.1. Processus de fabrication et testes représentatives.

La diagramme du flux presenté dans la figure 2.1.1 peut être considerée comme une variante de schéma de présentation du processus final de fabrication des grappes de combustible.
Dans le flux sont representées en ordre, les opérations technologiques de fabrication et les opérations de côntrole de la qualité avec les procedures techniques afférentes.

2.2. Conception d'un plan de côntrôle.

Le plan de contrôle de la qualité et essais représente un document important du Programme de l'assurance de la qualité. La conception d'un pareil document est conditionnée par le project d'exécution et aussi par les spécifications de contrôle, par les exigences prévues dans le contrat de livraison du combustible, par les traits spécifiques de la tâchnologie de fabrication adoptée, etc. Il doit furnir aux moins les dates suivantes /2/ :

- le flux de fabrication et de contrôlè avec l'emplacement des points de contrôle et des opérations de vérification nécessaires;
- les spécifications de qualité et les valeurs imposées;
- l'équipement et les procedures de vérification;
- les caractéristiques du plan de contrôle;
- le mode d'échantillonement;
- les documents de contrôle de la qualité;

Dans le tableau 2.2.1 est présenté un model de "liste de vérifications" spécifique pour le contrôle final d'une grappe de combustible.

A remarquer que les operations de contrôle de la qualité: le contrôle de l'etancheité et le contrôle de la contamination ont été appliqués aussi aux éléments combustibles.
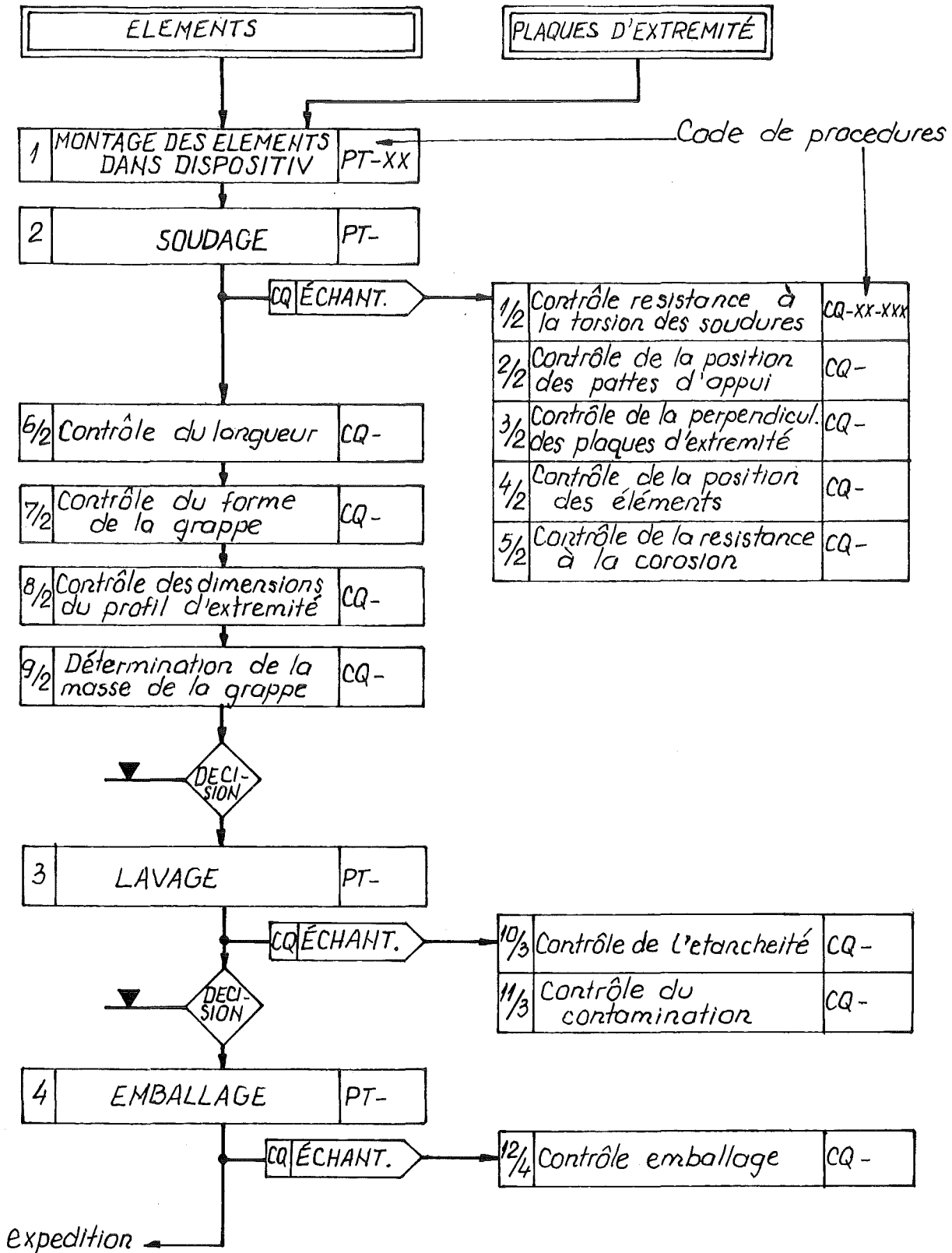
Figure 2.1.1.

Tableau 2.2.1.

LISTE DE VERIFICATIONS

No. Caractéristique de contrôle	Valeurs imposés	Code procédure	Plan de contrôle			Observations
			Type standard	N_c	AQL	
1. Résistance à la torsion des soudures			mesure	IV	0,4	Autorisation processus spécial; sur des échantillon-témoins
2. Position des pattes d'appui			attributs	II	2,5	
3. Perpendiculaire des plaques d'extremité			Mesure	IV	1	
4. Position des éléments			mesure	IV	0,25	
5. Résistance à la corrosion			0,2%	-	-	
6. Longueur	495±1 mm	CQ-XX-XXX	100 %	-	-	
7. Forme de la grappe			100 %			
8. Dimensions du profil d'extremité			100 %	-	-	
9. Masse de la grappe			100 %	-	-	
10. Etancheité			attributs	II	0,25	Control 100 % pour des éléments
11. Contamination			attributs	II	0,65	Contrôle par attributs pour des éléments
12. Emballage			attributs	II	1	

2.3. Choix des plan adéquats de vérification de
la qualité.

La problème du choix entre les diverses catégories des
plans de vérification de la qualité des grappes semble assez
simple. Les choses deviennent plus compliquées lorsqu'il est
nécessaire de décider entre le type de protection, le type
d'échantionnement, etc.

Les éléments fondamentaux qui conduisent à un plan de
vérification ont été établis selon les necessités et possibilites,
donc selon les conditions restrictives du problème (les valeurs
AQL, la modalité d'échantillonement, etc). Le controleur doit
recevoir une technologie claire des grappes du point de vue
technique (caractéristiques de la qualité, dispositifs de véri-
fication utilisée, etc) aussi bien que du point de vue méthodolo-
gique (le volume de l'échantillon nécessaire a prélever et véri-
fier, le calcul de la statistique de décision, le criterium de
décision sur le lot).

2.4. Utilisation combinée des standards de vérification
de la qualité du lot des grappes par attributs
et par mesure.

L'application exclusive dans l'activité de contrôle de
la qualité du standard de contrôle par attributs ou du standard
de contrôle par mesure seulement, n'offrent pas un solution
idéale. Une certaine catégorie de caracteristiques de qualité
conduit dans ces cas à des difficultés qui compliquent l'appli-
cation des standard mentionnés. Il s'agit du choix des plans de
vérifications pour les caractéristiques spéciales de qualité.
Ces caractéristiques peuvent être définies comme étant les ca-
ractéristiques dont la vérification de qualité peut être réalisée
seulement par l'utilisation d'un échantillon de petit volume a
cause de coût de la vérification (ex. la vérification de la ré-
sistance à la torsion des soudures d'assemblages).

La solution est celle d'utiliser les standards combinés
pour la vérifications de la qualité des lots des grappes par
attributs et par mesure, bénéficier ainsi des avantages suivants
de l'application des standards de vérification de la qualité
par mesure dans le cas des caracteristiques spéciales mesurables:

a) on augmente le domaine des valeurs AQL utilisables
pour les caractéristiques spéciales;

b) le volume de l'échantillon a vérifier est petit.

Tableau 2.4.1.

Volume de l'échantillon	AQL (%)	
(n)	Contrôle par attributs	Contrôle par mesure
3	4,0	4,0; 2,5
4	-	4,0; 2,5; 1,5; 1
5	2,5	2,5;1,5; 1; 0,65
7	-	2,5; 1,5; 1; 0,65; 0,4; 0,25
8	1,5	-
10	-	1,5; 1; 0,65; 0,4; 0,25; 0,15

Dans le tableau 2.4.1 on peut examiner les dates compara-
tives AQL qui peuvent être assureés avec des volumes des échan-
tillons ≼ 10.

2.5. Effect du choix du niveau de qualité acceptable.

La plupart des plans de vérification de la qualité des
lots des grappes de combustible à supposé le choix apriorique
d'une valeur des caractéristiques de qualité du lot, pour lequel
la probabilité d'acceptation soit grande. Dans le cas des stan-
dards /7/, /8/, cette caractéristique de qualité est AQL pour
lequel la probabilité d'acceptation des lots est environ 95 %.
Le succès de l'application efficace du contrôle statistique de
la qualité des lots de grappes dépend surtout de la corectitude
du choix des valeurs AQL pour chaque caractéristique.

La tendance erronée du choix arbitraire de certaines va-
leurs AQL, trop petites en comparaison avec les possibilités
courrantes à un moment donné du fabricant du combustible et même
en comparaison aves les nécessités objectives de l'acheteur a
exigé un contrôle à 100% et a crée donc un effet indirect, parti-
culièrement nocive, de ne pas stimuler le fabricant à rendre
meilleure la qualité du combustible fabriqué dans la periode
respective.

Pour mieux comprendre cette idée, considerons la representation graphique dans la figure 2.5.1. ou sont décrites les courbes CO des plans de vérification de la qualité pour AQL = 0,15% et AQL = 2,5 % pour le volume du lot N = 501 ÷ 1200, niveau normal, le niveau de vérification de la qualité N_c= II, lettre code I /7/.
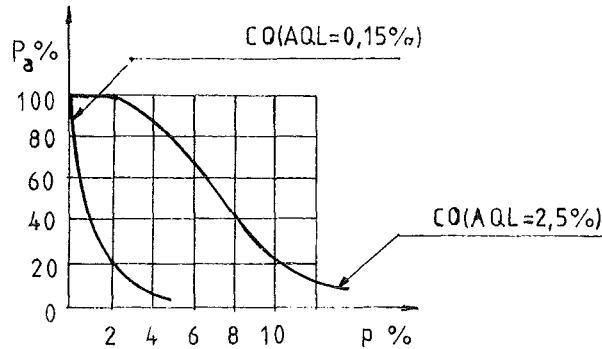
Fig.2.5.1.

Supposons que le fabricant du combustible par l'interemèdiaire du plan de contrôle de qualité appliqué peut assurer pour les caractéristiques de qualité majeures de la grappe de combustible une fraction défective moyenne p = 2 % et que ignorant cette chose, l'acheteur fixerait arbitrairement AQL = 0,15% avec le désir excessif d'accepter seulement des lots d'une très bonne qualité. Le résultat seraient dans ce cas loin de satisfaire l'acheteur parce que la majorité des lots seraient repoussés (Pa = 20%).

Si, par exemple on fixait par contrat une valeur AQL plus grande (AQL = 2,5 %) tous les lots serait acceptés (Pa= 100%) mais fait serait inconvenable pour l'acheteur et d'une certaine manière pour le fabricant de combustible aussi, qui ne pourrait ainsi être stimulé pour rendre meilleure la qualité de la production.

L'effet negatif est materialisé par la relaxation pas toujours justifiée du contrôle du fabricant sur le processus de production, fait qui conduirait à une détérioration progressive de la qualité du processus.

3. RESULTATS

3.1. Dates de contrôle obtenues pour la caracterisation des soudures d'assemblage

La caractérisation des soudures d'assemblage de la plaque d'extremité a compris deux etapes distinctes :

- par la métalographie des soudures
- par les testes de la résistance à la torsion des soudures

L'examination métalographique ne peut pas donner une
indication complète sur la qualité de la soudure. Dans les figures
3.1.1. et 3.1.2 sont presentées des sections transversales des
soudures d'assemblage provenues des lots de soudures executés sous
des régimes différents de soudure. Bien que les images indiquent
apparemment des soudures aproximativement identiques, les testes
de verification de la résistance à la torsion des soudures provenues
des memes lots ont mis en évidence des valeurs du moment de torsion
M_t dans le raport 2,4 /1.

Le teste de la resistance à la torsion des soudures a exigé
l'usage des échantillons-témoins (specifiques) pour la caracterisation
du processus de soudure. A mentionner que, pour l'adoptation de
cette méthode de contrôle ont été prises les suivantes mesures
concernant :

- l'autorisation de l'opération de soudure d'assemblage comme
processus spécial;

- la vérification de la concordance des résultats obtenus
sur des échantillons - témoins avec les valeurs de la resistance
à la torsion des soudures d'assemblage sur les grappes de combustible
prélevées de lots différents de fabrication.
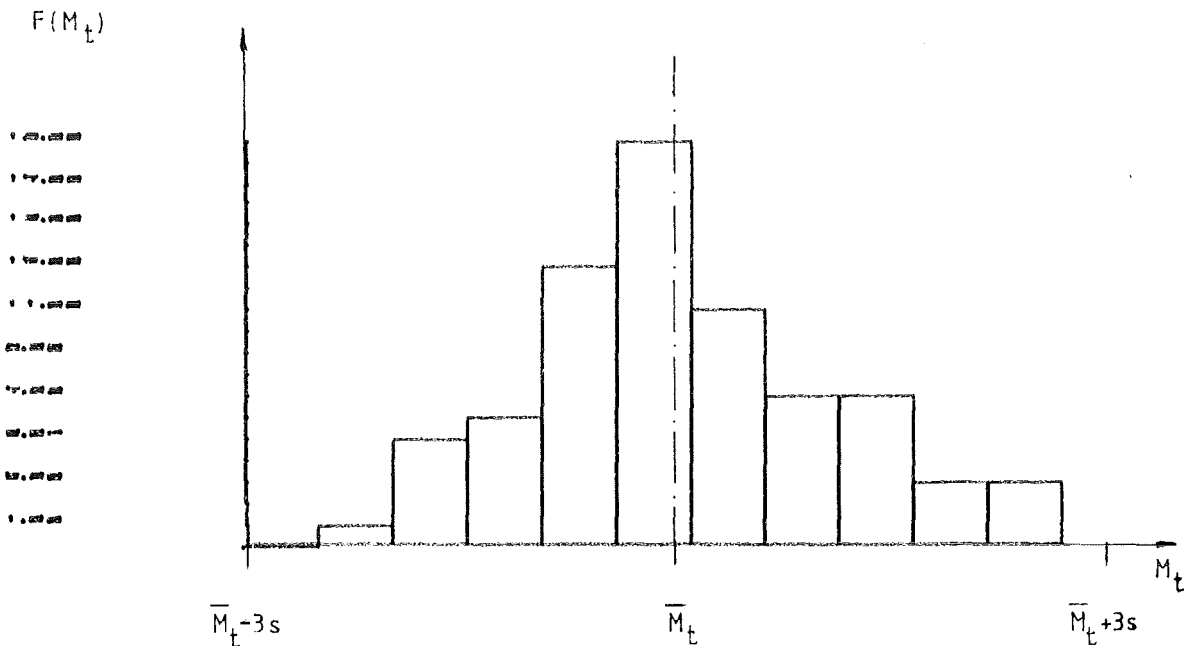
Fig 3.1.3
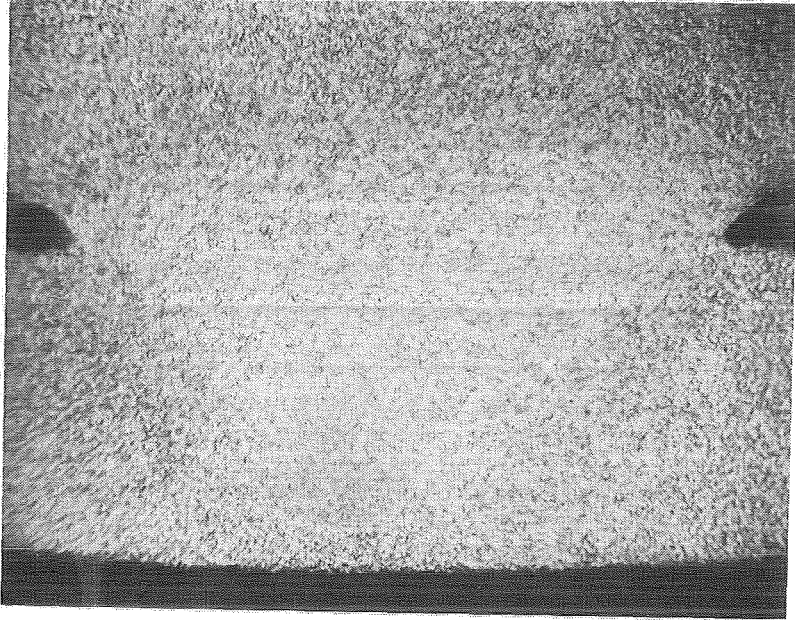
Figure 3.1.1.
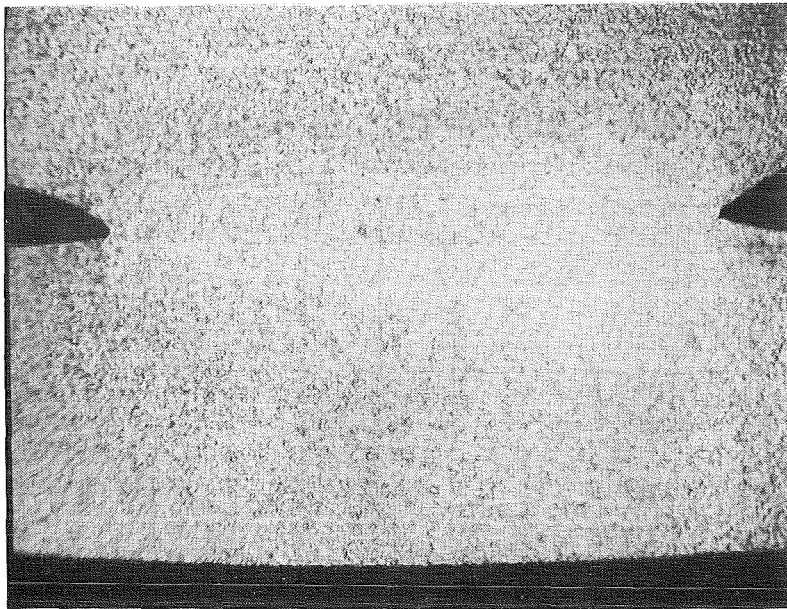
Figure 3.1.2.

Dans la figure 3.1.3. est représentée l'hystogramme des valeurs du moment de torsion M_t des soudures d'un échantillon.

3.2. Dates de contrôle obtenues pour la caractérisation de la cote H

La cote H (fig 3.2.1) représente une caractéristique de contrôle qui a crée au commencement des difficultés mais qui n'a plus représenté une cause d'apparition, des grappes non-conformes, après avoir determiné les paramètres optimals et assuré la stabilité du processus de soudure d'assemblages.
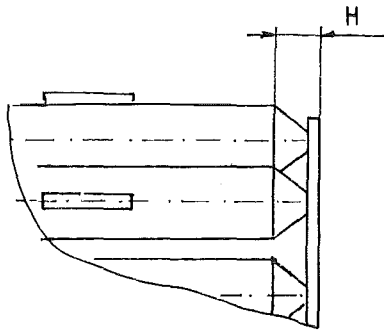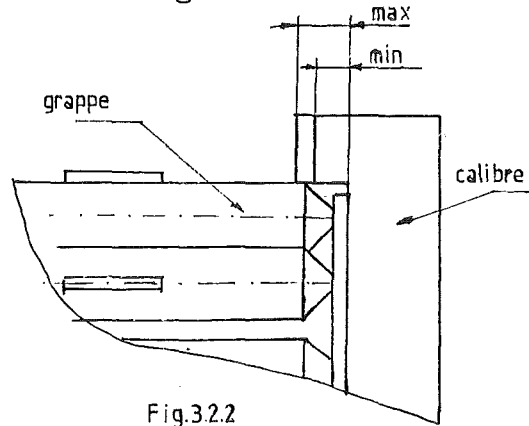
Fig.3.2.1.   Fig.3.2.2

Une calibre qui peut être utilisé pour le contrôle de la cote H est présenté dans la figure 3.2.2.
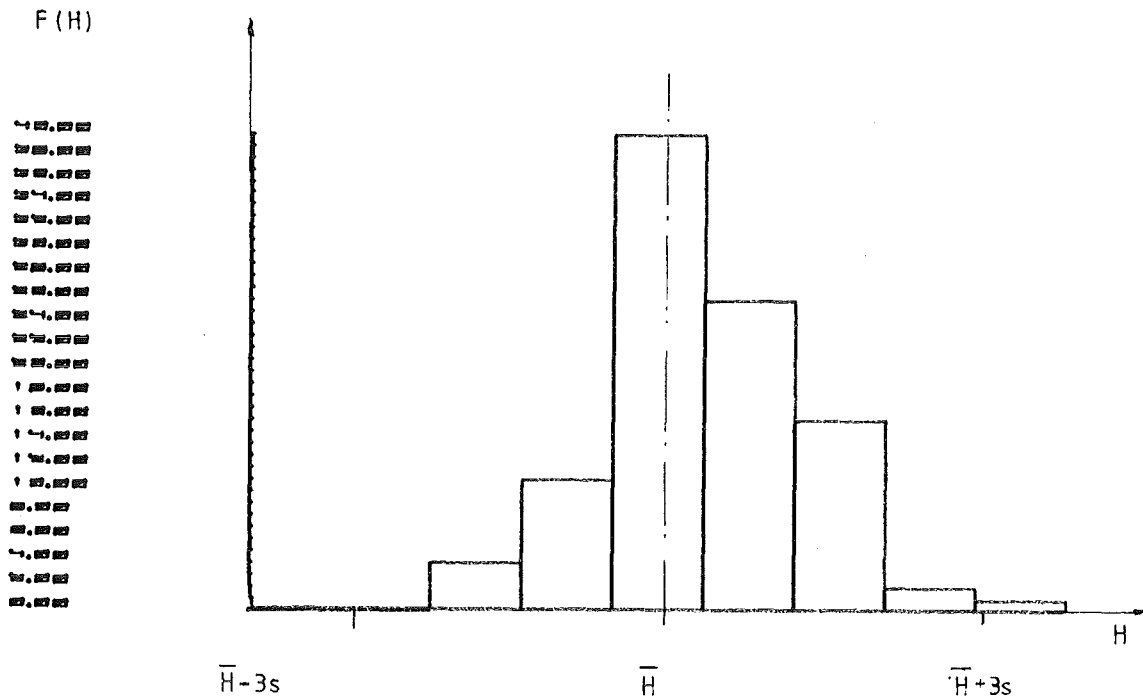
Fig.3.2.3

Dans la figure 3.2.3. est representée l'hystogramme des valeurs obtenues à la mesure de la cote H sur un échantillon d'un lot de grappes combustible.

4. CONCLUSIONS

Le contrôle de la qualité represente un levier important dans la technologie du combustible nucleaire.

Les dates qu'il met a la disposition du fabricant sont necessaires pour maintenir une qualité superieure imposée par l'acheteur.

Les valeurs de AQL comprises dans les plans de vérification de la qualité du combustible nucleaire ne doivent être envisagées comme un but fixe par l'intermediaire d'un standard de produit mais comme un élément à un dinamique propre qui tend vers une valeur optimale du point de vue economique dans certaines conditions (existentes au fournisseur et au acheteur) à la mesure du deroulement du contrat entre les deux parties.

REFERENCES BIBLIOGRAPHIQUES

/1/ Quidebook on Quality Control of Water Reactor Fuel Technical Reports Series No. 221 IAEA,Vienna 1983

/2/ CSA - Standard Z. 299-1978 - Quality Asurance Requirements

/3/ Juran, J.M. - Quality Control Handbook, McGraw Hill Book Compani, Third Edition, New York 1974.

/4/ Larrieu, J - Gestion de la qualité. Aspects economiques et techniques, Dunod, Paris, 1970

/5/ Duncan, A,J - Quality Control and Industrial Statistics, 3rd. edition, Honewood Ilinois R.D. Irwin Inc. 1965

/6/ Grant, E.L. - Leaveworth R.E. - Statistical Quality Control, 4-th edition New York, McGraw Hill Book Co, 1972

/7/ MIL STD 105 D - Sampling Procedures and Tables for Inspection by Atribute Washigton DC US Government Printing Office,1963.

/8/ MIL STD 414 - Sampling Procedures and tables for Inspection by Variables for Percent Defectives Washigton DC Government Printing Office, 1957.

/9/ Iliescu D.V. - IAQ EOQC Conference, Vienne, 1975.Some Thoughts on statistical Q.C.

/10/Iliescu D.V. - Voda Gh.V. - Statistique et tolerances, Ed.technique, Bucharest 1977.

EXPERIENCES IN THE APPLICATION OF QUALITY CONTROL AND
QUALITY ASSURANCE PROGRAMMES IN WATER REACTOR FUEL FABRICATION

K Balaramamoorthy, R Vijayaraghavan, PG Kulkarni
S V Raghavan and AK Bandyopadhyay

Atomic Fuels Division
Bhabha Atomic Research Centre
Trombay, Bombay – 400 085
INDIA

And

N Kondal Rao, R Rajendran, TS Krishnan, JB Patro
GVSRK Somayajulu, AR Javdekar and M Surya Prakash

Nuclear Fuel Complex
Hyderabad – 500762
INDIA

ABSTRACT

Nuclear fuel for Research Reactors and Pressurised Heavy Water Reactors (PHWRs) is being fabricated in India for a period of over two decades. The fuel is produced to conform to stringent quality control specifications. Generally, the performance of the fuel has been very good in the reactors. This is not only due to the high quality workmanship in the various stages of production but also to the meticulous care exercised in the planning and application of quality control and quality assurance procedures.

For the nuclear fuel used in Water Reactors, extensive material specifications have been compiled and they are periodically reviewed and revised. The specifications cover various aspects such as metallurgical and mechanical properties, non-destructive testing, dimensional and visual standard requirements. Similarly, detailed manufacturing engineering instructions (MEIs) and quality control instructions (QCIs) have been drawn.

For any deviations from the specified requirements, design concession committee considers all deviations and acceptance or rejection criteria are evolved. In this task, the design concession committee is supported by experimentation in various laboratories of the Department of Atomic Energy.

The Quality Assurance procedures have been evolved over a long period of time. They generally conform to the latest code and recommended guides of IAEA regarding Quality Assurance in the manufacture of fuel.

The paper details out results of various Quality Control and Quality Assurance procedures and analyses the results obtained.

1. INTRODUCTION

Quality Assurance is defined as 'planned and systematic actions to provide adequate confidence that an item or facility will perform satisfactorily in service'. This definition is applicable to the manufacture of fuel for nuclear reactors as well. Quality Assurance encompasses the whole realm of activities from design to performance evaluation and should generate feed-back to close the circuit. Thus, it covers 'womb to tomb' route, from conceptual ideas to manufacture and operations.

Nuclear fuel can be considered as the heart of a reactor. The supervision and control of nuclear fuel manufacture decisively determine the quality of the product on which its subsequent performance in the reactor depends. Fuel element failure can not only reduce the availability factor of a nuclear power station and the achievable burnup, but also increase the exposure of operation and maintainence personnel to radiation, thus becoming a point of considerable importance from safety aspects.

2. ORGANISATION

In India, all activities associated with nuclear fuel are carried out by the Department of Atomic Energy (DAE) of the Indian Government. The Nuclear Fuel Complex, Hyderabad is a constituent unit of DAE with the responsibility to manufacture fuel for the BWRs and the PHWRs. It is an integrated facility with plants to produce uranium oxide pellets from yellow cake, zircaloy finished components from zircon and plants for assembly of fuel bundles.

Nuclear Fuel Complex has essentially the following constituent units — Zirconium Oxide Plant (ZOP), Zirconium Sponge Plant (ZSP), Zircaloy Fabrication Plant (ZFP), Natural Uranium Oxide Plant (UOP), Ceramic Fuel Fabrication Plant (CFFP), Enriched Uranium Oxide Plant (EUOP), Enriched Fuel Fabrication Plant (EFFP) and Quality Control Laboratory for meeting quality control requirements of all plants.

The Power Projects Engineering Division (PPED), another constituent unit of DAE, has the responsibility for design (including fuel design) and construction and operation of nuclear power plants.

The Bhabha Atomic Research Centre (BARC) provides R & D support. The various tasks of the Fuel Development Committee are carried out mainly in the different Divisions of BARC. The Atomic Fuels Division (AFD) of BARC, as an

external auditing organisation carries out quality surveillance of the fuel manufactured by NFC. This is very effective in checking the usual defects like use of outdated drawings/specifications, mistakes committed due to non-incorporation of modifications and developments etc. This is also very useful in bringing the latest developments to be incorporated, research and development, including inspection methods and techniques, proper documentation of data and proper evaluation and in deciding the disposition of material which is slightly out-of-specifications.

The inter-relationship of the various units is depicted in Fig.1.

At NFC, in order to carry out the responsibility effectively, the Quality Control and Inspection Group is constituted independent of the production units and is responsible directly to the highest level of management viz. the Chief Executive. Also in conformance with the requirements of quality assurance, all the manufacturers and suppliers of raw materials and components outside NFC, either within DAE family or outside, are all taken as part of the manufacturing group and subjected to the same QC, QA programme.

3. QUALITY ASSURANCE FUNCTIONS

There are three primary documents on the basis of which the Quality Assurance function of nuclear fuel is executed.

a) Detailed specifications and drawings for parts and assemblies are provided by the Design Group of PPED.

b) Unambiguous and exhaustive MEIs and QCIs have been generated by the respective Groups of NFC.

c) Quality Surveillance plan and shipping release format is provided by AFD/BARC and reports of periodic QS visits are compiled and distributed.

The second tier of documents consists of a variety of forms and reports to ensure that all checks are made and properly recorded.

A set of documents consisting of the material evaluation reports and the final inspection reports along with special instructions, if any, accompanies each shipment of fuel.

Any deviations in the quality requirements of fuel are brought to the Design Concession Committee which consists of representatives of design, manufacturing, surveillance and operating station groups.

Any major modifications either in the design or in manufacturing route are discussed by the Safety Review Committees.

4. EXPERIENCES IN THE QUALITY FUNCTION

At IFC, in the last decade, about 1000 Nos. of enriched uranium fuel assemblies for two BWRs of Tarapur Atomic Power Station (TAPS) and 30,000 Nos. of natural uranium fuel assemblies for PHWRs have been fabricated conforming to the stringent quality requirements and supplied. The PHWRs are located at Kota (Rajasthan Atomic Power Station) and at Kalpakkam (Madras Atomic Power Station). Also a large number of zircaloy coolant tubes and calandria tubes have been manufactured by NFC.

The nuclear fuel, zircaloy fuel sheaths are one of the most important components. The flow-sheet with details of parameters controlled and QC points etc. in the fuel clad tube manufacture is shown in Fig.2. Some of the important specifications that are followed in Zircaloy Fabrication Plant for cladding tube manufacture are shown in Fig.3.

4.1 Quality Aspects of PHWR Fuel

4.1.1 Uranium Oxide Pellets

Uranium oxide powder is compacted into pellets which are sintered and ground and checked at different stages to meet the final specified requirements like density, chemical composition, metallographic structure, grain size and dimensions. More than 1000 lots have been processed so far. Majority of the lots met all the specified requirements and have been cleared for use and dis-charge burnups have also been very satisfactory. However, during fabrication stages some minor deviations have been noticed and in order to study the effect of these, if any, on irradiation behaviour, specified bundles have been fabri-cated with proper documentation. Some observations on deviations in (i) grain size, (ii) chemical composition and (iii) dimensions, are given below alongwith burnup level achieved.

Grain size (specified range - 5 to 25 microns)

A few uranium oxide lots were processed with large grain size and the maximum individual grain size accepted was 600 microns. Three of the bundles assembled with large grain size uranium oxide were documented. The maximum burnup achieved on irradiation was of the order of 9400 MWD/Te U. The fuel showed no abnormal behaviour and the bundles have been discharged after stipu-lated burnup levels.

Chemical Composition

It was observed, on chemical analysis in a few UO_2 lots that they had high gadolinium content (maximum 0.48 ppm as against the specified maximum of 0.10 ppm) and some bundles were assembled using these lots. There is no adverse report on irradiation of these bundles.

Dimensions

Dish depth of the uranium oxide pellets was found to be marginally more than the specified value in few lots and they were treated as normal pellets though a formal concession is raised. Few bundles loaded with such pellets have not shown any abnormal behaviour during the stipulated irradiation time.

4.1.2 Zircaloy Components

Some minor deviations in chemical composition of zircaloy ingots were observed for impurities like aluminium, nitrogen, oxygen and sometimes for alloying elements iron and tin. It was observed that ingots produced during a particular period were showing higher aluminium, nitrogen contents either individually or together. Maximum aluminium content observed in one ingot was as high as 227 ppm (130 ppm in product) as against the specified maximum of 75 ppm. Maximum observed nitrogen content in another ingot was 246 ppm as against the specified maximum of 65 ppm. Similarly in one case 1595 ppm oxygen was found against the specified maximum of 1400 ppm. However, based on extensive long term corrosion studies carried out, it was seen that even 175 ppm aluminium and 100 ppm nitrogen gave acceptable corrosion rate. Hence the requirements have been revised to 95 ppm of aluminium and 90 ppm of nitrogen.

4.1.3 Assembled Bundles

Dimensional deviation in Bundles

Flash pickling of zircaloy tubes was done, where low YS was encountered. Thus overall diameter of some bundles increased and further filing of bearing wires and wire wraps resulted in reduction in wire thickness. Most of these bundles were accepted on the basis of the tests carried out to ascertain that a minimum wire weld shear strength was maintained. However, now the problem is solved by maintaining minimum specified wire thickness coupled with increase in overall bundle diameter by 0.05 mm.

Helium Content in the Cover Gas

A few bundles were assembled with elements containing low helium in the cover gas. It was observed on analysis that the helium content was ranging from 43 to 97.5% in the cover gas. Heat transfer calculations were done and possible increase in O/U ratio due to the presence of excess oxygen was evaluated. The possibility of nitrogen pick-up in the end cap weld was examined by microhardness testing, gas analysis and metallography on representative samples and no abnormality was noticed. Based on these evaluations, these bundles were released for reactor use. No abnormal behaviour of these bundles was reported from the reactor site. Presence of low helium in the cover gas was traced back to malfunction of vacuum pump during welding. This was later rectified and corrective measures were taken to ensure the presence of helium to the specified extent by analysing samples more frequently.

Few bundles were intentionally assembled with elements containing argon and helium as cover gas in the ratio of 80% and 20%. Nothing abnormal was reported in irradiation of these bundles. On the basis of this result, we may use argon + helium mixture as cover gas in future, depending on the supply position of helium gas and other factors.

Autoclaving of Bundles

Autoclaving of bundles on 100% basis was done as a measure of check for any contamination and other metallurgical defects like segregation of impurities in the zircaloy fuel sheath. Occasional occurrence of greyish to whitish patches on elements, end caps, end plates and wires caused either rejection or acceptance under concession according to the degree of severity of these patches on the bundle.

Fluoride contamination or trace impurities in water used for autoclaving caused white patches. Contaminations like dust also caused faint brownish to greyish patches. A high degree of cleanliness is insisted upon to avoid such recurrence and NFC is now autoclaving bundles on 10% basis and occurrence of these patches is practically non-existent.

Bundles with graphite coated tubings

Twenty four bundles assembled with graphite coated tubes have been loaded in the reactor and are reported to be behaving well. This practice has been implemented to minimise PCI and for easy loading of fuel pellets into the fuel sheath. Plans are now on hand to adopt this practice in regular production.

End Plate/End Cap Welds

Some fine indications were observed in squeezed out bead portion of the welds in between end plate and end caps. Experiments conducted on weld strength on such samples showed acceptable weld strength and as such these bundles were accepted for reactor use. Performance of these bundles were reported to be normal.

Surface Roughness on Zircaloy Sheath

Elements from certain tube lots showed surface roughness of the order of 4 microns on sheath surface after pickling the elements. Roughness marks are attributable to pilgering/straightening marks. As roughness did not impair ultra-sonic evaluation of the sheath, elements having surface finish upto 2 microns (normal tubes have surface finish 0.8 microns or better) were accepted for further processing. Few bundles fabricated with these elements showed nothing abnormal in performance.

The bundles assembled with zircaloy material having slight deviation in chemical composition are treated as normal bundles, without any loading restriction in the reactor. Bundles assembled with tubes having TCE values less than 14.5% (as against the specified minimum of 15%) are restricted to low power zone in the reactor. Bundles made with fully annealed tubes ($\bar{x} - 3\sigma = 310$ MPa) have performed well in the reactor.

4.1.4 Development of Split-Spacer Bundles

For use in PHWRs, manufacture of split-spacer type assemblies in place of wire-wrap assemblies was taken up. The process adopted for these was resistance welding of the appendages (spacers and bearing pads) to the cladding tubes as per the specifications developed by PPED and development work carried out at AFD. A large number of random samples were collected from the production stream and they were destructively tested for shear strength of the welds. It was seen that the minimum strength requirement was met with 95% confidence. Weld monitors are under development in collaboration with Reactor Control Division of BARC for on-line monitoring the integrity of resistant welds and the tests conducted so far yielded very encouraging results.

4.2 Quality Aspects of BWR Fuel

Most of the UO_2 lots and zircaloy material lots for the BWR fuel met the requirements with regard to the specifications, with minor deviations in chemical composition. High degree of control over the entire process at various stages is exercised in achieving the specified quality level towards the procurement of components such as tie plates, spacers, springs, plugs, etc. Overall performance of fuel clusters in the reactors is quite satisfactory. However, after critical review of the existing specification, few changes have been incorporated as under

i) Prepressurisation with helium from 1 atmosphere abs. to 3.5 atmospheres abs., is done to increase thermal conductivity of pellet clad gap and hence decrease the fuel temperature and consequent fission gas release. These factors minimize PCI.

ii) Short chamfered flat pellets are used for better recovery and uniform microstructure and density. Chamfering is provided to minimise the chipping during processing and it eliminates ridging of elements during reactor operation. From the operational experience gained, it is felt that dishing of pellets is not necessary, as the space provided within the element is adequate to compensate the irradiation swelling even at maximum burnup level.

iii) Statistical quality control for mechanical properties of fuel sheath is adopted to have an overall control over the entire lot with a certain confidence level. Earlier zircaloy sheaths with 0.8 mm wall in stress relieved condition were used in fuel clusters. It was felt that the scatter in the mechanical properties within the lot was due to the critical range of stress relief temperature. Hence, tubes in annealed condition have been produced with higher wall (0.9 mm) to optimise the strength. The scatter in the mechanical properties has been considerably reduced giving a higher confidence level.

4.2.1 Deviations in Enriched Uranium Oxide Pellets

Evaluation of uranium oxide lots processed over a long period indicates that the majority of the lots satisfy the specified requirements of density, dimensions and chemical composition including moisture content. However, few

lots showed high density upto 97.5% of the theoretical density and were used with loading restriction. Also, minor deviations in chemical composition were observed in some lots for which the total boron equivalent content was well within the limits. Only noticeable deviation in chemistry for boron to an extent of 4 ppm maximum was observed in some powder lots. These lots were used as poison rods, with restrictions on pellet diameter and positioning.

4.2.2 A large number of set-up and process control samples of end plug welds were metallographed and the results show that the minimum weld thickness is achieved with 99% confidence.

4.2.3 Other components

It has become possible to manufacture all the fuel components such as stainless steel tie plates, zircaloy spacers, stainless steel springs, lock tab washers etc. indigenously. Constant QA efforts were made to bring out acceptable castings, spacers etc. by evaluating chemical analysis, heat treatment records, mechanical properties and microstructure. Liquid penetrant test, radiography and dimensions were also checked.

5.   CONCLUSION

The fuel manufactured in India for both PHWR and BWR reactors has given exceedingly good performance in service. This is due to the fine workmanship of the production personnel as well as due to the meticulous care exercised in the QC and QA functions by all concerned. The deviations highlighted in the paper only illustrate and stress that all points are carefully considered before disposition, and nothing is left to chance. The indigenous manufacture of nuclear fuel has given considerable experience and confidence, both in production and quality assurance methods and has helped to develop and grow trained man-power for the country's future nuclear power programme.

References

1.   IAEA Safety Code on QA

2.   IAEA Safety Guide SG-QA-11

3.   Balaramamoorthy, K. "QA & QC Procedures for Nuclear Fuel Manu-
      facture in India", IAEA Tech. Series No.173, IAEA, Vienna (1976).

4.   Balaramamoorthy, K. "QA in Manufacture of Nuclear Components –
      Indian Experience", NPP-QA Seminar, KAERI, Seoul, Nov. 1980.

5.   Balaramamoorthy, K., Rao, V.S.G., Kulkarni, P.G., "QA in Developing
      Countries with particular reference to Indian Experience",
      IAEA-SM-253, Paris, May 1982.

6.   Anantharaman, K., Babu, S., Anand, A.K., Mehta, S.K., "Design
      Criteria for Confidence in Manufacture of BWR Fuel Rods",
      IAEA-SM-233.

7.   Shah, B.K., and Balaramamoorthy, K., "Internal Report on Long Term
      Zircaloy Corrosion Testing".

FIG. 1

DEPARTMENT OF ATOMIC ENERGY

P.P.E.D.
Design Group
Fuel development
committee

Design
Concession
Committee

Safety Review
Committee etc.

NUCLEAR FUEL COMPLEX

MANUFACTURING ORGANISATION

A.F.D.
Quality
Surveillance

Quality Audit
External to N.F.C.

UTILITY

RAPS
TAPS
MAPS
FBTR
etc.
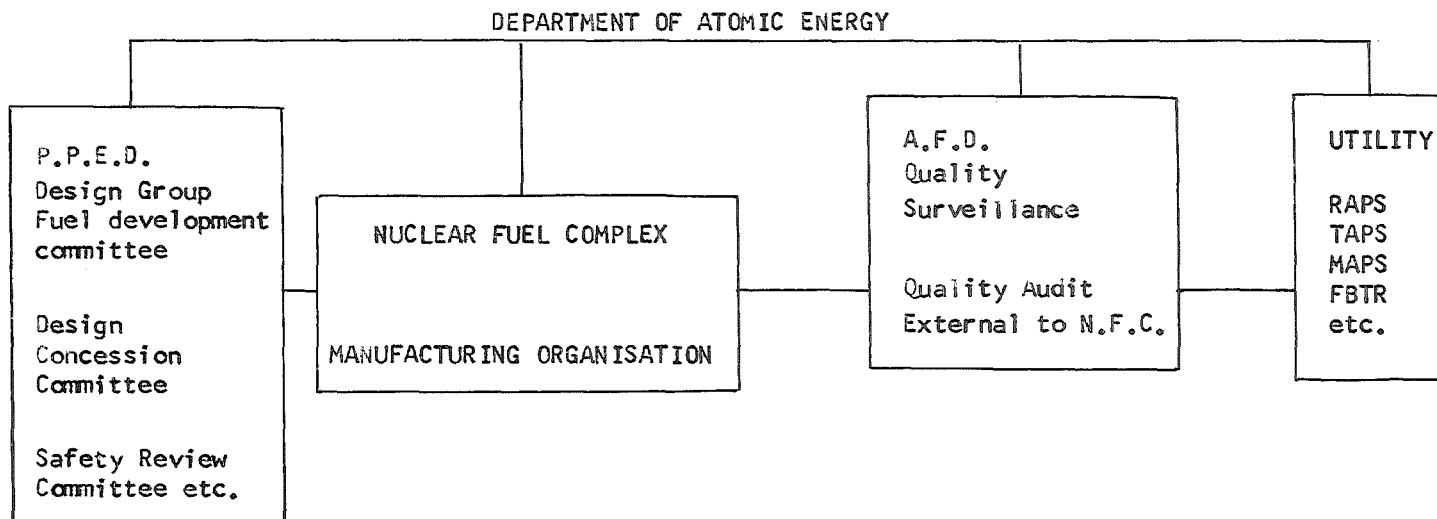
FIG. 2

QUALITY CONTROL IN ZIRCALOY CLADDING TUBE MANUFACTURE

	OPERATION		CONTROL PARAMETER	CONTROL TECHNIQUE
1.	RECEIVING ZR. SPONGE		CHEMICAL COMPOSITION, HARDNESS	100% LOT SAMPLING
2.	ALLOYING ADDITIONS AND ELECTRODE FORMING		CHEMICAL COMPOSITION, DENSITY OF COMPACTS, SURFACE CONTAMINATION	100% LOT SAMPLING
3.	DOUBLE ARC MELTING AND MACHINE		MELTING PARAMETERS	PARAMETERS CONTROL
4.	INGOT INSPECTION		CHEMICAL COMPOSITION, SURFACE QUALITY, INTERNAL DEFECTS—UT HARDNESS	100% SAMPLING 100% INSPECTION 100% SAMPLING
5.	DOUBLE EXTRUSION AND BETA QUENCHING		EXTRUSION PARAMETERS QUENCHING PARAMETERS	100% INSPECTION OF TOOLING, PARAMETER CONTROL, ADMINISTRATIVE CONTROL FOR PERIODIC QUALIFICATIONS OF FURNACE
6.	BILLET MACHINING AND INSPECTION		DIMENSIONS OD & ID, SURFACE FINISH, CHEMICAL COMPOSITION	100% INSPECTION 100% SAMPLING
7.	COPPER JACKETING & EXTRUSION		EXTRUSION PARAMETERS, SURFACE QUALITY	100% INSPECTION OF TOOLING PARAMETER CONTROL 100% INSPECTION
8.	DEJACKETING, ANNEALING, GRINDING, PICKLING ETC.		SURFACE QUALITY, STRAIGHTNESS	100% INSPECTION

9.	TUBE SHELL INSPECTION	▣	DIMENSIONS, SURFACE QUALITY VISUAL INTERNAL DEFECTS, UT, BOW METALLOGRAPHY, HARDNESS	100% INSPECTION 100% SAMPLING
10.	RECEIVING TUBE SHELL AFTER INSPECTION	◬		ADMINISTRATIVE CONTROL IN THE FORM OF TRAVEL CARDS WITH PROPER IDENTIFICATION INSTRUCTIONS FOR FURTHER PROCESS, SIGNATURES OF AUTHORISED PERSONNEL ETC.
11.	INTERMEDIATE PILGERING, CLEANING, PICKLING ETC.	⊙	PILGERING PARAMETERS DIMENSIONS OD & WALL VISUAL	PARAMETERS CONTROL 100% TOOLING INSPECTION SET UP SAMPLING
12.	INTERMEDIATE ANNEAL	⊙	ANNEALING PARAMETERS SURFACE APPEARANCE	PARAMETER CONTROL, ADMINISTRATIVE CONTROL 100% VISUAL INSPECTION
13.	FINAL PILGERING & CLEANING	▣⊙	PILGERING PARAMETERS DIMENSIONS OD & WALL, SURFACE DEFECTS, UT, VISUAL	SAME AS IN 11
14.	STRESS RELIEF	▣⊙	ANNEALING PARAMETERS SURFACE APPEARANCE MECHANICAL PROPERTIES	STRICT PARAMETER CONTROL QUALIFICATION & CALIBRATION OF FURNACE, 100% VISUAL LOT SAMPLING
15.	STRAIGHTENING, ID BLASTING, OD GRINDING, CLEANING	◯	BOW ID & OD FINISH DIMENSION OD	100% INSPECTION
16.	FINAL INSPECTION	▣	DIMENSION ID/OD, WALL & LENGTH, FLAWS UT VISUAL BOW	100% INSPECTION

17. FINAL INSPECTION ▣   STRENGTH AT ROOM & ELEVATED   LOT SAMPLING
                            TEMPERATURE, TCE, GRAIN SIZE,
                            HYDRIDE ORIE..TATION, HYDROGEN &
                            NITROGEN ANALYSIS, CORROSION IN
                            STEAM

18. PICKLING, CLEANING,   ◉   AUTOCLAVE PARAMETERS   PARAMETER CONTROL 100%
    AUTOCLAVING (WHERE   WATER QUALITY
    NECESSARY)   VISUAL FOR AUTOCLAVE FINISH

LEGEND : FINAL INSPECTION AND RELEASE ☐   OPERATION & QUALITY CONTROL ◯   STORAGE △

PERMANENT RECORD MADE •

FIG. 3

MAIN PARAMETERS AND SPECIFICATIONS APPLIED
TO FINISHED ZIRCALOY CLADDING TUBES

Parameters :-

SPECIFICATIONS

1. Chemical analysis

    a) Control of alloying elements

    ASTM - 146 as guide line

    b) Control of impurities like
       Cl, H, N etc.

    Lot sampling

    c) Control of impurities contributing
       to high neutron absorption cross
       section

2. Mechanical
    properties

    a) Tensile strength

    ASTM - E 8 & ASTM - E 21

    b) 0.2% offset Y.S.

    Lot sampling basis

    c) Percentage elongation in 50 mm

3. Burst Test

    a) Hoop stress

    1) Closed end burst test for RAPS
       tubes at room temp.
       Hoop stress : 620 MPa
       TCE : 15% Min.

    b) T.C.E.

    2) Open end burst test for TAPS
       tubes at room temp.
       Min Burst : 53 MPa
       TCE : 16%
       Lot sampling basis

4. Metallography

    a) Grain size

    ASTM - A - 112
    Lot Sampling
    Longitudinal : ASTM G.S. No.7
    Transverse   : ASTM G.S. No.8

    b) Texture

		c)	Hydride orientation & Fn fraction	
5.	Corrosion test	a)	Autoclave test in steam	Water PH : 7 ± 1 Resistivity : 500,000 ohms. Temp. : $400 \pm 5°C$ Pressure of Steam : 10.3 ± 0.7 MPa Black lustrous oxide film on all surfaces. Wt. gain : 3 days — 22 mg/dm^2 14 days — 38 mg/dm^2
6.	Dimension	a)	I.D. & O.D. by air gauging	100% inspn.
		b)	Wall thickness by ultrasonic pulse echo technique	100% inspn.
7.	Integrity	a)	Flaw detection by ultrasonic pulse echo technique Standard : 60°V notch 3.2 mm x 0.04 mm both Longitudinal and Transverse.	100% inspn.
8.	Straightness		1 in 1000.	100% inspection
9.	Surface roughness		1.6 microns	100% inspn. by comparison with visual standards.
10.	Visual	a)	As per drg. requirements	100% inspection by comparison with standards

———————

DISCUSSION

B.P. JENKINS: One of your slides indicated that you carry out a 100% dimensional check on pellets. What are the features you check for and could you explain the philosophy behind this?

K. BALARAMA MOORTHY: We carry out one hundred percent inspection on UO_2 pellets for usual quality and for outer diameter of pellets. We have stringent tolerances on diametral clearance (between pellet outer diameter and inside diameter of Zircaloy sheath) because of the collapsable sheath design. So we give importance to one hundred percent inspection.

V. GORSKY: What methods do you use to verify your weld joints in CANDU fuel rods?

K. BALARAMA MOORTHY: Set-up and process sample welds are examined metallographically. All production welds are visually examined and helium leak tested. Ultrasonic inspection techniques are also planned.
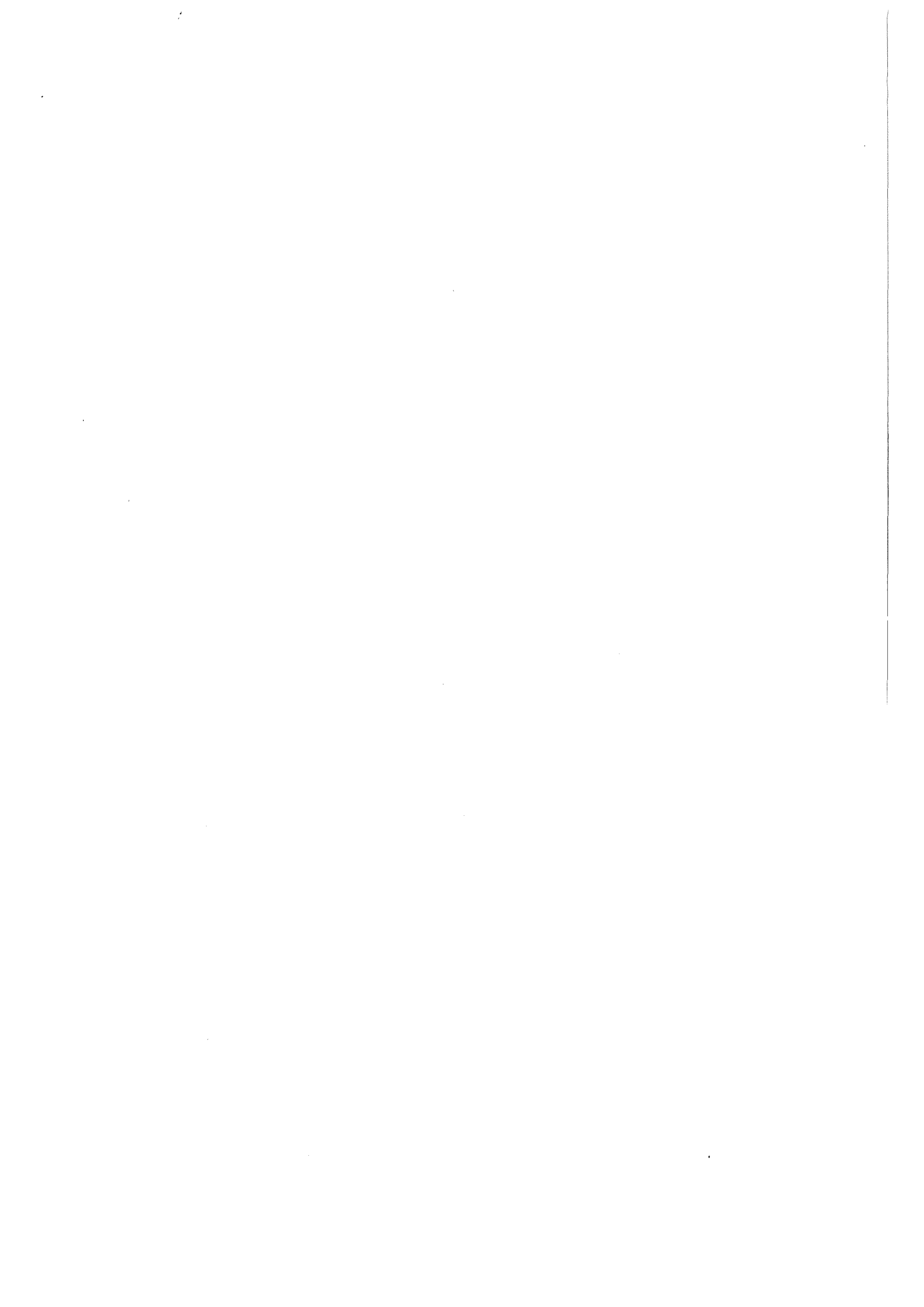
STATISTICAL EVALUATION OF FUEL AND FUEL ASSEMBLY PRODUCTION

M. ERNOTTE

FRAGEMA

149 Cours Lafayette
F - 69006 LYON

ABSTRACT

Statistical evaluation of product characteristics and manufacturing conditions is a valuable tool to improve both the designer's and manufacturer's knowledge of fuel technology. This knowledge may in turn be used to further improve the products and processes.

The types and conditions of the evaluations FRAGEMA is commonly running are described and examples commented. Some considerations on irradiation feedback and on the quality management information system are also developed.

STATISTICAL EVALUATION OF FUEL AND FUEL ASSEMBLY PRODUCTION

I - THE NEED FOR STATISTICAL EVALUATION

Statistical evaluation should aid the designer and the manufacturer in fully knowing their product and their manufacturing processes.

For the fuel designer, standard statistical quality control practices are rather frustrating : he often wants to know how many parts are outside tolerances, where exactely and how far out they are. He also likes to work with means and standard deviations and most of the time when he asks for such data, the QC engineer will tell him something like "There is a minimum probability of .95 that at least .98 of the lot is within tolerances but looking at the data, I don't think the distribution is quite normal ...".

In the field of nuclear fuel where increased reliability is a major goal in the entire industry and for the kind of objectives we are all shooting at, simple statistical quality control leading only to Accept/Reject decisions is just not adequate for some important characteristics. In a fuel core of a 900 MWe PWR, there are about 10,000,000 pellets and in a pellet production meeting the lowest AQL of MIL-STD-105 D which is .010, there are still about 1,000 defective pellets per core. Let's hope that these are not randomly scattered critical defects, otherwise one might be facing around 1,000 fuel rod failures per core or a disastrous failure rate exceeding 10^{-2} !

Another aspect which is sometimes overlooked and which is also of great concern to the designer is that rejected parts are also to be considered even if they have been scrapped, since they are clues for product quality which are as meaningful as the accepted parts.

If a process is under control, which means stable but if it is not correctly centered in respect to the design criteria, statistical sampling will inevitably reject some of the lots but these inspection operations will not modify the mean and the standard deviation of the process and thus of the accepted lots and the inspection operations are therefore totally useless if they are not associated to a statistical evaluation of the process output.

In this illustration not only is inspection uneconomical but the designer might get an optimitistic view of the actual product characteristics.

Conversely one last consideration about fuel design concerns the possibility of having a product in fact better than thought ; we might have been producing/supplied with products which far exceeded the requirements. In our type of product where operational experience is a source of justification, if the operation of such "good" products has been successful in the past, what validation does this experience represent, that of production or that of design requirements ?

That is why it is important to know not only whether a product meets requirements but also how closely these are met.

For the fuel manufacturer, the statistical evaluation of his production is an important tool which will help him in many respects as statistical sampling in itself will only allow quality to be measured without necessarily producing a quality product.

Any manufacturer who acts to discover and remove causes of poor quality will inevitably see improvements in the 3 traditional areas : quality, costs and planning. An unadapted process will certainly lead to the production of defective parts and most probably to reinspection, screening and even conflicts with customers ; all these result in cost overruns and scheduling problems.

It is however possible to combine both the designer's and manufacturer's interests in one simple approach : the best assurance for a smooth operation is to have an adequate process under control.

Is this idealized objective easily met in actual industrial situations ? Unfortunely not and reasons for this are discussed below.

DESIGN DIFFICULTIES

It is difficult for the designer to state in all cases exactly what is needed and this is evidenced by the subtle differences in expressions like requirements, recommendations, preferences, objectives and the like which are sometimes used in specifications.

The first problem is to define what is to be considered as acceptable or more generally what makes the quality of a product. A classical definition of quality is the capacity for an item to perform satisfactorily in service or "Fitness for use" as Juran called it.

In the case of a complex product like nuclear fuel this requires that the quality requirements for the final product be broken down into quality requirements applicable at each step of the fabrication process.

Some of these requirements may be temporary. The length of a fuel rod will be the total length of two end plugs and one fuel tube and the fuel rod designer may be more interested in the rod length than in the component lengths although if the end plugs are TIG welded, the end plug length will probably determine the weld location.

Some characteristics have an importance through their mean value, others through their individual values. In a grid, for example, the average mixing vane angle has an effect on the fuel assembly pressure drop but one single vane bent too far could cause a rod failure.

The same characteristic could have an entirely different effect depending on the location at which it is applied. Cleanliness is a good example. It is obvious that internal contamination of the fuel tube present when the pellets are loaded is much more of a concern that the same type of contamination on the outside of the same tube away from the weld areas.

Finally quality requirements are often specified on an attribute basis and a maximum percentage of defectives is quoted but the situation on FIG. 1 (a) is in most cases, potentially less harmful than the situation on FIG.1(b).

This is very important because these outliers in otherwise acceptable processes are probably the major cause of subsequent failures.
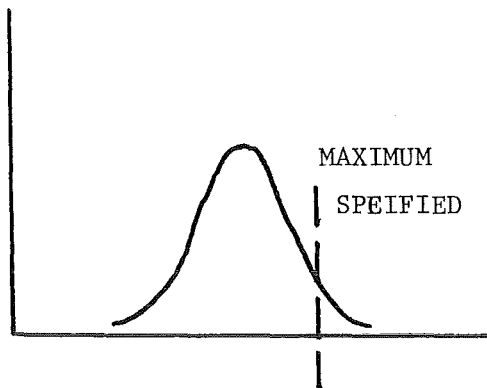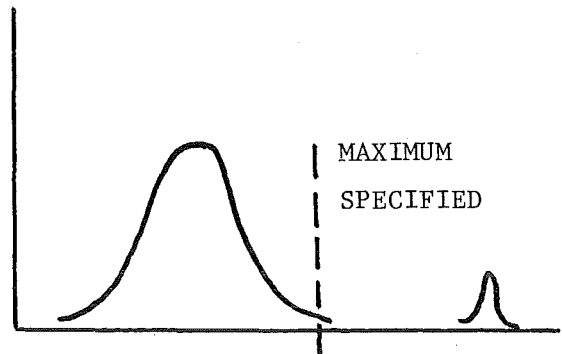
FIGURE 1 (a)   FIGURE 1 (b)

MANUFACTURING DIFFICULTIES

The designer will always think in terms of actual values or characteristics and will not be concerned by things like bias, difficulties in selecting truly random samples, inspection equipment capabilities, influence of operator errors, etc. which are often left to production or inspection organizations.

With 100 % inspection which was thought adequate in the past when a high quality product was required, such factors are the sole source of error and in some cases they are far from negligible.

One of the problems commonly met in manufacturing is the high number of variables which could intervene in a process and therefore considerable time and effort is needed to conduct experiments under actual industrial conditions.

High output production lines suppose automated processes which should be fairly reproducible but imply also a high input rate of components leading to rapid exhaustion of part lots and thus frequent line feed changes : for example, a tube lot is processed through a fuel rod line in one single workshift.

II - EVALUATION CONDITIONS AT FRAGEMA

After these preliminary considerations, the systems FRAGEMA has set up to monitor fuel product quality and fuel product fabrication will be described.

FRAGEMA is essentially a fuel design organization and contracts for fuel fabrication are placed either to FBFC (Franco-Belge de Fabrication de Combustible) or to CFC (COGEMA - FRAMATOME).

The chain of organizations involved in fuel supply is thus follows : subsuppliers - contractors (FBFC or CFC) - FRAGEMA - utility.

Within FRAGEMA, two departments (Technology and Quality) are responsible for defining quality requirements in the form of specifications. The Quality Department is also in charge of qualifications and surveillance of fuel fabrication and of quality and reliability analyses connected to manufacturing activities.

QUALIFICATIONS

Product surveillance starts with qualification of production lines for products whose quality cannot be assured only through post fabrication inspections. These include all assemblies (grids, fuel rods, nozzles, skeletons and final assemblies) and some special products like Inconel and Zircaloy strip, grid straps, guide thimbles, Zircaloy bar stock, fuel rod end plugs and fuel tubes.

Qualification is achieved through formal qualification programs established jointly by cognizant design and fabrication units which aim at defining permissible variations for selected significant process parameters.

For example, blending conditions for mixing UO_2 virgin powder, recycled U_3O_8 and other additives are determined during these programs.

The conditions include rotating speeds, blending cycles, amounts of materials and are selected after statistical evaluation of the homogeneity of the blends to assure a constant feed to the pelleting process and to validate the chemical analysis of subsequent pellet lots based on results obtained on a very small number of pellets.

Another example of evaluation taken from the pellet qualification program covers the homogeneity of green pellet density obtained on multi-die presses and the homogeneity of sintered density achieved within furnace boats. Both are evaluated through variance analysis with the objective already mentioned, i. e., process stability and justification of sampling practices on sintering boats as in this case, a truly random sample is not practical due to the stacking of pellets in the molybdenum boats.

Statistical sampling plans are also discussed with the manufacturer and tried out during the qualification programs, which end in the fabrication of a complete lot under industrial conditions to demonstrate that variables selected produce adequate process averages and standard deviation when compared to specification limits. Extensive additional measurements are required to make sure that this objective is met on the qualification lots.

The size of such lots is rather difficult to set. They must not be too small in order to be representative of an actual production but also they must not be too large for economic considerations whenever the optimum conditions have not been found and a new qualification run seems desirable.

For FRAGEMA products, a lot size equivalent to five fuel assemblies has been selected. This corresponds to 40 grids, 1,300 fuel rods, 2,500 kg of pellets, etc.

We thus ensure that proposed manufacturing and inspection processes are reproducible and well adapted to our requirements.

A production line is qualified by FRAGEMA for a period of two years and requalification for another two years is only granted after a thorough evaluation of the output of that line.

Such evaluations are based on all available information accumulated since the last qualification. It includes all non conformances and manufacturing incidents, lot acceptance sampling results for items procured from outside by the contractors and surveillance inspection results for items subjected to FRAGEMA hold points.

Process characteristics are computed from these data and when necessary action plans for the next two years are established. They may cover revised sampling plans to be applied by either the Contractors or by FRAGEMA. Special programs may also be initiated to investigate any abnormal situations. The need for such programs however is usually felt earlier through the other opportunities provided by our comprehensive evaluation system.

An interesting feature incorporated in the evaluations performed for requalification consists in the use of unbiased data obtained independantly from the manufacturer : those corresponding to the surveillance by Contractors on their subsuppliers and those from FRAGEMA surveillance on Contractors' fabrications.

The data are unbiased as they relate to all incoming lots presented by the manufacturer to the buyer. They are thus appropriate measurements of the overall effectiveness of the manufacturer's production and internal inspection systems.

Whenever possible, we compare our results with those known by the manufacturers and any discrepancy between the two sets of results is investigated in cooperation with the manufacturers. In our experience, the origin of most of such differences lies in the following areas : difficulty of truly random sampling, differences in comprehension and application of requirements between operators and differences due to product evolution between sampling stages.

PRODUCT OVERINSPECTIONS

FRAGEMA has identified to the manufacturers mandatory hold points for the following fabrication steps : UO_2 pellets, all subassemblies and final assemblies.

The objective of such inspection is normally not to verify that the product is acceptable but rather to collect independant information on the effectiveness of the manufacturer's production system as stated earlier.

The sampling plans used are tailored to each product and to each type of production. Some details about these sampling plans will be presented in another FRAGEMA paper in this Seminar. (1).

MANUFACTURING INCIDENTS

With all sampling plans and even with 100 % inspection there is always a finite possibility that some unacceptable products be in fact accepted.

A theoritical solution to that problem is to prevent the production of defective items and it requires that all abnormal situations detrimental to the quality of a product be thoroughly evaluated in order to identify preventive actions which could prevent recurrence of the conditions leading to a lack of quality.

Control charts are extensively used in fuel fabrication for FRAGEMA and they provide a convenient means of monitoring a production process when the requirements to be met are known and fixed.

Some situations are however more subtle and complex and require a different approach for resolution. We have set up with our Contractors a powerful tool to deal with some of these situations.

It consists of a formal system for reporting and investigating manufacturing incidents which are not simple non conformances and could have an effect on the overall quality of the fuel products.

For example, we consider we face an incident when low density pellets present in fuel rods are detected at the linear enrichment check, when fuel rods are found defective at the helium leak test...

The frequency of these events is fortunately very low but nevertheless full investigations are launched not only to reconsider other products already accepted which might also be affected but mainly to identify the cause of the problems and find possible durable remedies. All incidents are fully documented in formal reports and made available to our customers.

QUALITY ANALYSIS

The last evaluation tool we have consists of our Quality Analysis System.

For specific products and characteristics, banks of manufacturing data are constituted and used to monitor product quality and to compare it to design limits. These data are either logged in at FRAGEMA like chemical composition of Zircaloy tube lots or specially recorded by the manufacturers and transmitted to FRAGEMA, percentage of surface defects on UO_2 pellets for example.

Statistical characteristics are computed, chronologic charts and histograms are regularly produced and estimates of actual quality levels based on FRAGEMA surveillance data as mentioned before allow trends analysis, comparisons between several production lines or between several sources of supplies.

As stated before, such information may be used to trigger special investigation programs aimed at improving our knowledge of factors contributing to the production of defects. Table I presents the product characteristics which are covered by our Quality Analysis System.

III - PRACTICAL EXPERIENCE

Throughout the years, we have so been able to evaluate many different situations and some of the most representative will be commented.

PELLET DIAMETER

The case goes back several years ago and was revealed during the FRAGEMA surveillance inspection of pellet lots.

Occasionally pellets with out of tolerance diameter were sampled from lots accepted by the manufacturer. Even if the percentage of such pellets was acceptable based on a classic 95 X 5 confidence level, we thought that it was abnormal for a process like centerless grinding, to produce out of tolerance parts.

TABLE I - FRAGEMA QUALITY ANALYSIS

QUALITY ANALYSIS RESULTS	UO_2 PELLETS	CLADDING TUBES	FUEL RODS
Quality analysis reports (statistics, chronology, histogram)	- Geometric density - Thermal stability - Diameter - Length - Porosity spectrum - Grain size - Visual defects - Powder recycle - U 235	- Tensile test - CSR - Gas analysis - Hydride orientation - Ovality - Corrosion test - Chemical analysis - Transverse dimensions	- Weld defect rates (girth and seal welds)
Quality level estimation (based on surveillance results)	- Visual defects - Geometric density - Diameter - Length - Dishing dimensions	- Transverse dimensions - Internal soudness	- X-ray interpretation efficiency

The analysis we conducted provided the following indications : the equipment capability was adequate (low standard deviation, slow drift) but the centering of the diameter was sometimes incorrect because the operators lacked useful information on the output of the grinders as the inspection was done by QC later on completed pellet trays.

The problem was cured after modification of the ground rules for work : the process was monitored by the operator with a control chart based on a sample of consecutive pellets sampled at prescribed regular intervals and detailed instructions were given to operators for resample and for readjustment of the grinder and finally QC conducted regular surveillances of operator adherence to procedures and also took samples with a lower frequency to check the validity of the operator's measurements.

That has later been demonstrated to be highly effective since the rejection rate after FRAGEMA overinspections dropped to zero.

PELLET VISUAL INSPECTION

That inspection was traditionnally done on a 100 % basis and although every one was aware of the fact that the inspection efficiency was not 100 %, the inspection specification did not take it into account and inspectors from all parties (manufacturers, FRAGEMA and even customers) were more often than reasonably expected trapped in lengthy discussions and cumbersome rescreening and reinspections.

We started with an evaluation of the efficiency of pellet visual inspection and found figures around 60 % for end defects and 80 % for side defects and thus concluded that with such efficiencies several actions had to be taken : the designer had to state the product quality he wanted and requirements like "100 % visual inspection" had to be dropped, the manufacturer was also required to reduce the number of defective items produced and conduct his own surveillance of all 100 % inspections.

The first action lead to the definition of several acceptable quality levels depending on the postulated severity of defects (nail cracks, chips, pits ...) combined with the actual quality levels representative of the production at that time.

QC surveillance of manufacturing screening operators was conducted using sampling plans taken from MIL-STD-105 D on reasonably small sublots in order to permit rescreening, should the sublots be rejected by QC.

Finally, the manufacturer paid more attention to the various causes of surface defects on pellets and was successful in improving the quality of as-produced pellets.

The end results have again been the drastic reduction of lot rejections after FRAGEMA or customer inspections and an improvement of the overall quality of pellets as we have now process averages lower than the required AQL'S by factors 3 to 10 depending of the type of defect considered.

FUEL ROD WELD X RAY INTERPRETATION

X-Ray inspection is conducted on a 100 % basis by qualified inspectors and any true indication seen on the films is considered a welding defect and the weld has to be rejected.

During our surveillance inspection of fuel rods, we used to reread a limited sample of X-Ray films to check the QC inspector interpretations and any discrepency between the two inspections was attributed to the unavoidable human factor.

We later decided with the manufacturer to formalize both our and his internal surveillance procedures and to use sampling plans based on an AQL of 0.1. Manufacturer surveillance would be done on a daily basis and FRAGEMA surveillance would be done on rod lot basis roughly corresponding to 3 day production runs.

It took us several months to obtain consistent results but ultimately we knew exactly the reading efficiency of X-Ray QC inspectors i.e. if a defect is present on a film what is the probability that the defect be missed.

We were also surprised to observe that the top girth welds were more difficult to read which we think is due to the less favorable top end plug geometry caused by the presence of the seal weld hole.

OUTGOING FUEL ROD WELD QUALITY

A full paper has recently been presented on the subject so the logic will only be briefly discussed as it is a very good illustration of what a thorough evaluation of a process can lead to. (2)

Everybody specifies a sensitivity level for X-Ray in terms of ASTM levels 2-2 T or 2-1 T (or through similar standards) but the relationship between the probability of detection and the defect (porosity) size is not clearly known. As outlined in FIG. 2, the modelization of the distributions of the largest porosity present in the welds and of the detection probability curve was possible after extensive metallographic work.

Combining the two curves, the outgoing weld quality versus the observed reject rate could be determined and we fixed a threshold reject rate above which a 200 % X-Ray inspection would be necessary to maintian an acceptable outgoing weld quality.

This type of evaluation is summarized in FIG. 3. It applies to many processes and illustrates the different factors contributing to the quality of a product. A detailed and quantified knowledge of these factors definitely open new paths towards improvements in quality.

FREQUENCY DISTRIBUTION OF THE
MAXIMUM POROSITY DIAMETER PRESENT
IN WELDS

PROBABILITY OF DETECTION BY
RADIOGRAPHY OF WELD POROSITIES

POROSITIES
DETECTED

POROSITIES
LARGER THAN
ϕc NOT
DETECTED

OUTGOING WELD QUALITY VERSUS
WELD REJECT RATE

FIGURE 2

UNDETECTABLE DEFECTS
(method - intrinsic)

DEFECTS PRODUCED

DETECTABLE
DEFECTS

UNDETECTED DEFECTS
(accidental)

DETECTED
DEFECTS

NON CRITICAL DEFECTS

CRITICAL DEFECTS

RISK OF FAILURE

NO RISK OF FAILURE

FIGURE 3

GRID PRODUCTION

Grid assemblies are products which normally require many adjustments,
verifications and rechecks and although the manufacturing operators were
already heavily involved in an interactive process with QC, neither the
manufacturer nor FRAGEMA were quite satisfied with the grid situation as
most of the criteria were to be 100 % inspected without a clear indication
of the minimum efficiency needed or achieved.

The manufacturer proposed to fully integrate manufacturing and
verification operations and let QC conduct surveillance operations. This
was accepted by FRAGEMA.

NUMBER OF CHARACTERISTICS
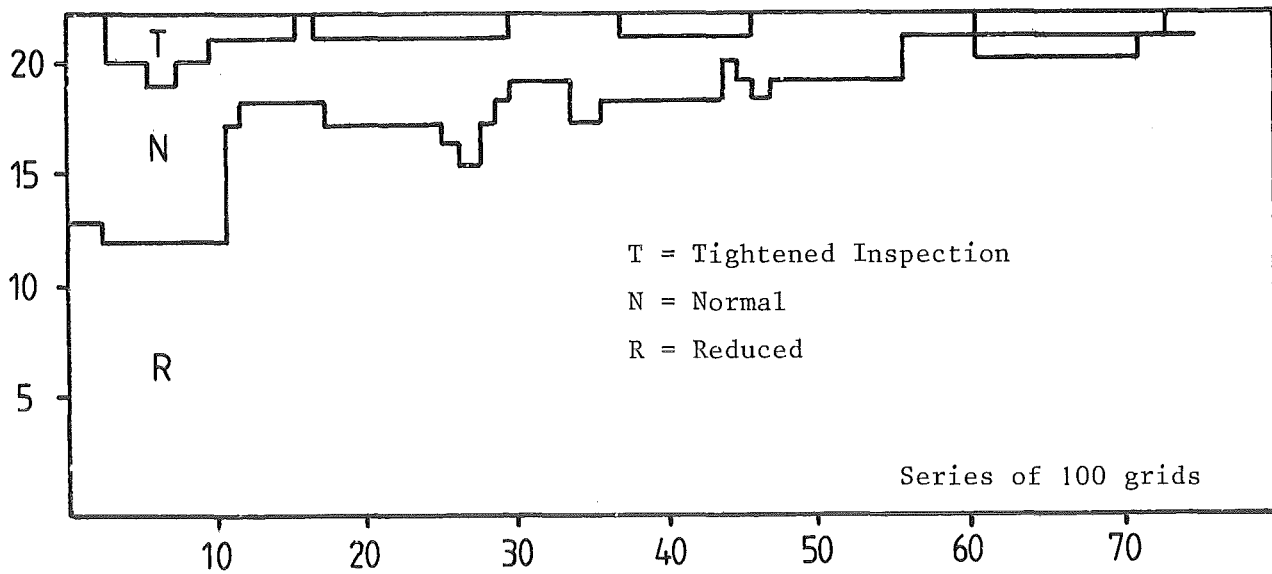
FIGURE 4 – EVOLUTION OF PRODUCTION UNDER MANUFACTURING
INTEGRATED INSPECTION

Twenty two different grids characteristics are covered by design criteria
and we started by defining that sampling plans corresponding to an AQL of
0.4 could be satisfactorily used to monitor an operation which was
specified to be 100 % performed.

An initial period devoted to the measurement of the quality levels
achieved for the 22 characteristics combined with the dispositions of
MIL-STD-105 D concerning the applicability of reduced inspection, allowed
a starting classification of each criterion to be established for normal
or reduced inspection.

Grids were grouped in 100 piece lots and submitted to QC surveillance :
one non conforming grid causing the entire lot to be returned to
manufacturing for 100 % screening for the defective characteristics. Each
time such a situation occured, a full investigation of the case was
conducted by the manufacturer and involved the designer if there was any
need to clarify the requirements or even to adapt them but always without
any compromise on the required minimum quality.

Such a process was rather painful but it worked out successfully in the long run as now all characteristics are consistenly under reduced inspection by QC which is the indication of the fact that it is now highly exceptional to find a grid accepted by manufacturing inspectors which would be rejected by QC.

Some other material on the possibilities of quality improvement offered by the integration of manufacturing and inspection activies will be presented in this Seminar. (3)

IV - IRRADIATION FEEDBACK

Feedback of irradiation data to design and manufacturing activities is a key element in improving fuel reliability. Here again, statistical evaluation techniques are used to ascertain whether there are correlations between fuel failures and as-built fuel characteristics or manufacturing conditions.

These are the only tools available experimentally to improve our technical knowledge about factors contributing to rod behaviour given the fact the we are already in an overall reliability range of 99.99 % to 99.999 % per year and considering that we are still looking for improvements !

With such low failure rates, these correlations could only indicate areas where potential gains could be expected and which should further be investigated pooling the resources from both the design and manufacturing organizations.

V - MEANS REQUIRED

They are two requisites for meaningful comprehensive fuel production evaluations : a good traceability system and a computerized data bank.

The FRAGEMA traceability system is documented in a manual which applies to all 3 manufacturing plants that FRAGEMA is involved in and has been fairly comprehensive since the early days. Data recording and transmittal had been worked out in detail long ago with the manufacturer so that the presently available data bank amounts to approximately 200 million bytes of information.

Data transmitted by the plants to FRAGEMA are handled by facilities composed of a 2048 K byte IBM 4331 processing unit, 193.5 M byte disk drives, tape units, floppy disk units, consoles and printers. The transmission and processing facilities are dedicated exclusively to traceability and manufacturing data, and used by personnel belonging to a Management System unit from the Quality Department.

The system software consists of a library of 160 programs containing a total of 70,000 instructions.

VI - CONCLUSION

The above considerations and examples have shown how statistical analyses of fuel production can increase both the designer's and the manufactuer's knowledge of fuel manufacturing technology. This additional knowledge may in turn be used to further improve fuel reliability through the definition of more stable and better adapted processes.

REFERENCES

1 - G. PANCACCINI, Le suivi des fabrications de combustible par FRAGEMA.
2 - G. KOPFF, C. MORIN, D. PACHTEM, C. RENAULT, P. VAN CRAENENDONCK,
   Détermination de l'efficacité du contrôle radiographique des soudures des crayons combustibles.
   6ème conférence internationale sur les essais non destructifs dans l'industrie nucléaire, Zurich december 1983.
3 - JC. NURY, JP. ROBIN, M. FOURE, C. MORIN, J. VAN BENEDEN, Système de contrôle en continu des fabrications.

DISCUSSION

V. GORSKY: In France a weld ultrasonic test facility was developed. Are you using it in the manufacture of fuel elements?

M. ERNOTTE: After comparison of the detection capabilities offered by ultrasonic tests and X-rays, we have concluded that, as the defect responsible for the majority of our weld rejects was porosity, we could not ensure the same outgoing weld quality after ultrasonic test as after X-ray. As a consequence, we are now looking towards automatic X-ray techniques.

R. HOLZER: Should complete traceability be kept in the future in view of the current very low defect levels, which make it very difficult to correlate the cause of isolated failures to the recorded QC - data?

M. ERNOTTE: This can be discussed; fuel rod data should be traceable; for various parts and materials different traceability requirements could be defined.

K. BALARAMA MOORTHY: What is the present practice in France - adoption of steam autoclaving the Zircaloy tubes before pellet loading or electropolishing the fuel rods?

M. ERNOTTE: We do neither autoclave fuel tubes before pellet loading nor electropolish the fuel rods. We corrosion test tube samples in 400° C steam as part of the QC operations at the tube manufacturer's and weld samples in 360° C water as part of the QC operations for rod manufacturing.

QUALITY CONTROL AND PERFORMANCE OF PHWR FUEL IN INDIA

R.S.Rustagi, M.Das, O.P.Arora
Department of Atomic Energy
Power Projects Engineering Division
Homi Bhabha Road, Colaba,
Bombay-400 005.

ABSTRACT

Indian Nuclear Power Programme is currently based on installation of a series of pressurised heavy water reactors in 235 MWe size, followed by 500 MWe size reactors in the nineties. Indigenous facilities for nuclear fuel design, fabrication and quality auditing have been developed. Statistical quality control and 100% inspection techniques are applied for ensuring the quality of fuel. Good operating experience on fuel over the last 10 years has validated the design and quality characteristics in fuel specifications. The paper describes the different designs of PHWR fuel bundles now fabricated in India and the quality control aspects evolved. Furthermore, some typical results are brought out.

1. INTRODUCTION

The Indian Nuclear Power Programme is primarily based on installation of natural uranium fuelled, pressurised heavy water reactors (PHWR) with the exception of Tarapur reactors of Boiling Light Water type. To date, three PHW reactors in the 220/235 MWe range have been completed (See Table-1) with a high degree of indigenously produced components and presently, five more units are in various stages of construction or commissioning. With the successful infrastructure thus set up and considering the need of more nuclear power plants in the country, it is proposed to set up a series of PHW reactors of standard design in the 235 MWe and 500 MWe sizes in the coming years, so as to have a total installed capacity of 10,000 MWe by the turn of the century.

The choice of PHWR's in the first phase of the 3 stage nuclear programme was influenced by two major factors, namely the ability to optimally use the limited reserves of natural uranium (indicated reserves: 73,000 tonnes of U) in the country and capability of indigenisation so as to achieve self-reliance in this vital field.

In the field of fuel, India has developed complete technological know-how right from the mining of ore to final reprocessing and reuse of spent fuel. Fig.1 shows the various operations in the nuclear fuel cycle now being carried out within the country.

With regard to fabrication, fuel for the first phase of the power programme has been under regular manufacture at the Nuclear Fuel Complex, Hyderabad since 1973, before which a pilot scale operation at the Bhabha Atomic Research Centre had produced in 1971, half of the initial fuel charge for the Unit 1 of the Rajasthan Atomic Power Station. At present, the NFC is capable of producing upto 200 t of finished fuel in a year; and it is estimated that a five fold increase in the fuel production capacity will be required to meet the demand in the coming decades. [1]

In this paper, the PHWR fuel designs and improvements, quality control characteristics with particular emphasis on statistical quality control, quality auditing and traceability and fuel performance in the Rajasthan reactors are discussed.

2. FUEL DESIGNS

As of date, over 22,000 fuel bundles have been produced indigenously and loaded in the reactors. The design characteristics of the bundles produced are given below.

2.1 19 ELEMENT DESIGN

The reference fuel design of the 19 element type for the Rajasthan and Madras reactors is shown in Fig.2. Table-2 shows the salient design data of the fuel assembly. Due to use of natural uranium, the strong emphasis on neutron economy has resulted in the use of thin wall collapsible cladding (radius to thickness ratio of 18) and high density of UO_2 pellets (10.6 Mg/m^3, 97% T.D.) which maximises the ratio of contained uranium to parasitic neutron absorbers. The clad being collapsible, the axial and diametrical clearances have to be kept highly optimised to avoid excessive sheath strain from collapse and creep down, at the same time providing adequate plenum

FIG.-1
PHWR NUCLEAR FUEL CYCLE IN INDIA

FABRICATION
NUCLEAR FUEL COMPLEX
HYDERABAD

DESIGN

• POWER PROJECTS
ENGINEERING DIVISION,BOMBAY

RESEARCH
AND
DEVELOPMENT

• BHABHA ATOMIC
RESEARCH CENTRE,BOMBAY

POWER
PRODUCTION

• RAPP – KOTA
• MAPP – KALPAKKAM

REPROCESSING

• POWER REACTOR FUEL
REPROCESSING PLANT,
TARAPUR

RAW
MATERIALS

DEPLETED
URANIUM

PLUTONIUM
RECYCLE

• NATURAL URANIUM :
URANIUM CORPORATION
OF INDIA, JADUGUDA

• ZIRCON ORE :
INDIAN RARE EARTHS,
ALWAYE

• FAST BREEDER TEST
REACTOR, KALPAKKAM

81.69 mm (MAX.)

Ø15.22

WIRE PAD

PRESSURE TUBE

ELEMENT

19 ELEMENT

81.74 mm (MAX.)

Ø15.22

Ø13.04

INTER ELEMENT SPACER

PRESSURE TUBE

ELEMENT

BEARING PAD

22 ELEMENT

FIG.-2
19 ELEMENT & 22 ELEMENT FUEL BUNDLES

space for fission gases and UO_2 thermal expansion/swelling. Majority of over 22,000 bundles irradiated are of the 19 element wire wrap type.

2.2   22 ELEMENT DESIGN

The third PHWR Atomic Power Station at Narora (NAPP) is India's first opportunity to apply significant operating experience to the design of a new standardised nuclear power plant, incorporating several major design modifications to improve safety, reliability and economy; and fuel is no exception. The 22 element fuel sub-division was adopted to obtain higher (15%) thermal power output, compared to the 19 element design, which is required for the 235 MWe rating. [2]

After considering the various alternatives of pin diameters, number of pins and their arrangements, a graded design of 22 elements, with two different element diameters was considered most optimum. This bundle is also shown in Fig.2. The salient design parameters are compared in Table-2. Several prototype bundles have been fabricated and successfully tested to establish the design and fabrication parameters.

2.3   OTHER DESIGN IMPROVEMENTS

Other notable design improvements incorporated in the fuel assembly as a result of operational feed back are:

(i)   inter-element spacing by split spacer appendages replacing the original wire wrap design. This development has emerged from two consideration, namely (a) the wire wrap design is not adoptable to the 22 element configuration and (b) concern of sheath fretting is significantly reduced in case of split spacer disign at higher flow velocities. Split spacer design is also found to be an economical bundle, contains less zircaloy and reduces the concern of pellet chipping during fabricabion and assembly which applies to the 19 as well as 22 element designs.

(ii)   graphite lubrication, in which the inside surface of the cladding tube is coated by a 5-9 μm   thick graphite layer. The layer acts in a number of ways to reduce the incidence of fission product stress corrosion cracking of the cladding on a power ramp.

3.   QUALITY CONTROL CHARACTERISTICS

Specifications for zircaloy and uranium oxide materials are prepared by the designers taking into consideration the reactor operational requirements, process capability and costs. At the Nuclear Fuel Complex, the manufacturing activities commence with conversion of concentrate to UO_2 pellets on the uranium side and zircon ore sands to finished tubing and components on the zircaloy side. Total quality assurance of nuclear fuel at the various constituent units of the Department of Atomic Energy (DAE) is achieved by disciplined actions in the processes of design, manufacturing and operation. No quality assurance programme can succeed without first assessing the capability of processes and equipment. Quality control during fabrication is thus exercised not only at the final product stage, but at various intermediate stages during material processing-covering the physical and chemical characteristics of the fuel. [3]

The various quality characteristics in the fuel specification are further classified according to their relative importance to the end use - i.e. the reactor

performance of the fuel bundles. This forms the most important basis of arriving at their SQC parameters, wherever such techniques may be applicable.

Following classification levels are applied:

Critical : where lack of control will cause increased fuel failure probability and may also result in violation of safety regulations.

Major : where lack of control puts economic penalty in reactor operation and may cause small increase in fuel failures.

Medium : where lack of control may result in economic costs and/or schedule delays.

Minor : all other characteristics.

Table-3 outlines some examples of the characteristics, their classification, how it affects the design and performance and the QC plan currently adopted for zircaloy cladding tubes and uranium oxide pellets.

3.1 ZIRCALOY TUBES

As seen in Table-3.1, ductility and strength characteristics of cladding tubes, being critical parameters, are determined to the stringent acceptance sampling plans to give an acceptable quality level (AQL) of 0.15%.

With a view to improve neutron economy and also to avoid longitudinal ridge, it was initially considered that strength rather than the ductility was the principal quality characteristic of cladding tubes and as such the cold worked and stress relieved condition was specified. Subsequently, from the reactor operational experience, it was recognised that about 2% post-irradiation ductility avoids low ductility clad failures. Accordingly, the stress relief annealing cycle was modified and a statistical method of acceptance sampling was introduced. For evolving the statistical acceptance criteria, a detailed statistical analysis of about 100 tube lots was undertaken. Out of these, about 10 tube lots were 100% destructively tested for verification of the criteria. Typical results obtained are given below:

(i) Contribution of personal error in TCE measurement was analysed from measurement data obtained from 3 different persons. It was found that the personal error was less than 0.5% in TCE values.

(ii) Analysis of data from tubes produced by modified heat treatment annealing cycle indicated that when yield strength showed a tendency to reduce, TCE as expected showed a tendency to increase. However, while the scatter in the yield strength values of sample tubes could be controlled, the scatter in TCE, as revealed by range, R values increased.

(iii) In the statistical experiment, the number of random samples was increased from 4 to 12. From this study, optimum number arrived at was seven (7), as against four (4) in the previous practice. It was seen that beyond 7 random samples, the statistical influence did not improve significantly for TCE as well as Yield Strength.

(iv) Further design analysis indicated that with the revised specifications of cladding tubes with respect to annealing cycle and sample size, the contribution of cladding tubes to the expected fuel failure rate will be less than 0.1%. Actual experience has validated this result.

However, in large commercial batches due to inherent nature of the manufacturing process, particularly in the 773-788 K range of annealing temperature, occasional low values of yield strength and ductility would occur. While R&D work to reduce the variability in the process is continuing, in the interim, suitable changes in the related design and operating parameters have been made to suit such lots. Experience has shown that this dynamic specification approach has been borne out by good fuel behaviour in the reactor.

The dimensional requirements such as I.D., O.D., wall thickness and defects, dictated from assembly and burst property considerations, are inspected ultrasonically, 100% basis. Short term and long term corrosion tests on zircaloy material were carried out to form the basis of specified limits and tolerance levels on chemical composition for alloying elements and impurities. This not only benefitted the manufacturer from many repeat analysis but also ensured quality of the material at the ingot and the final product stage.

3.2 UO$_2$ PELLETS

With regard to the UO$_2$ pellets, manufactured from both ADU and AUC routes, emphasis is laid on high density, chippability/crackability characteristics and visual standards. From irradiation tests under power reactor conditions, good data has been gathered on the performance of high grain size material, surface microcracks, pits, end-capping etc. This has enabled to evolve the basis for quality characteristics of the UO$_2$ pellets. Chippability/crackability tests at the pellet stage and decanning tests on random element samples are routinely performed to ensure the specified quality requirements, so as to simulate the fuel handling and movement during various stages of fabrication and even transportation to the reactor sites.

3.3 OTHER COMPONENTS

Besides the above, SQC techniques are extensively used in various other stages of fuel manufacture. Examples are : element machining, spacer/bearing pad weld strength, zircaloy structural components, dimensional requirements and metallography of UO$_2$ pellets. Considering the increased fuel production rate in years to come, statistical process control rather than product control techniques have been also introduced recently.

4. FUEL PERFORMANCE

Table-4 presents data on cumulative bundles loaded and their performance in RAPS-1, RAPS-2 and MAPS-1 which are the 3 operational PHW reactor units in India. Since, RAPS-1 was the first PHW reactor commissioned, this has given us bulk of the operating experience. Following are the salient observations:

i) The three (3) suppliers viz. NFC, BARC and Westinghouse Canada (WC) have respectively produced 80.7%, 11.1% and 8.2% of the total 22011 bundles loaded as on Oct. end, 1983. BARC and WC together supplied fuel only for the initial charge of RAPS-1 in

the early 70's. NFC is thus the prime fabrication plant for power reactor fuel and zircaloy products for current and future reactors in India.

ii) The bundles in RAPS reactors have been irradiated to the design burnup levels (average : 7130 MW.d/teU). Some of the bundles have seen even 14500 MW.d/teU, without failures.

iii) Out of the total 20 defect bundles, none was of the split spacer type, although 640 such bundles have been irradiated, which brings out the superior features of this inter-element spacing technique.

iv) The overall defect rate for the cumulative 22011 bundles irradiated in the three reactors comes to 0.09% thus meeting the specified target of < 0.1%.

v) For the fuel manufactured to earlier specifications, defect rate was as high as 0.3%, compared to 0.06% for currently manufactured fuel. This shows that introduction of statistical quality control has benefitted producer and reactor user alike.

vi) There has never been occasion in RAPS or MAPS when the fuel defects had restrained power operation of the reactor. So much so, the initial fuel charge of RAPS-1 had experienced as many as 180 power cycles due to the large number of reactor trips and power set backs in the initial phase of operation. Still, the fuel did not present any problem.

vii) Although, there has been no extensive in-pile experience obtained yet on the 22 element bundle design, it is expected to reduce the fuel defect rate further, or, the restrictions of preconditioning nature now placed during reactor start-up could be relaxed, as the split spacers and graphite lubricated tubes have been introduced right from the beginning in this design. This is also due to a marginal reduction in the maximum heat rating parameter in this design.

5. AUDITING AND TRACEABILITY

Quality Auditing and Traceability are important managerial functions of an effective QA system. For QA in India, the expert members from fuel design, research and development, fabrication, licensing and operations groups are utilised in performing this function in a systematic manner. At the pre-production stage, at design specification stage and at production manufacturing stage, independent technical reviews and checks are carried out by authorised representatives routinely.

Material evaluation reports, bundle and element record cards, and records of process and set up control parameters are maintained for all the production lots. Salient manufacturing data for all the bundles are further computerised and kept in magnetic tapes for effective traceability, as and when required, during the entire fuel cycle life. In addition, suitable archive samples of each material lot used in the manufacture are preserved for future reference and evaluation. The data on irradiation history and movement of the fuel in the reactor is maintained by a fuel accounting and management computer code.

The existing QA systems are now under review to see if the QA operating costs could be expressed in terms of pay-back or failure reduction costs in keeping with the ever increasing demand on product quality. The modified system is expected to

provide a logical and statistically defensible method for measuring the performance of a quality program.

6. CONCLUDING REMARKS

Two types of fuel bundle design viz. 19 elements and 22 elements have been adopted for the Indian Pressurised Heavy Water reactors in the 220/235 MWe sizes.

Quality control characteristics for fuel are classified as critical, major, medium and minor according to their relative importance to performance of fuel bundles in the reactor. The classification forms the basis for evolving the quality control parameters.

Statistical quality control is adopted for many areas in fabrication of pellets, cladding tubes and structural components. Specifications and sampling plans for cladding tubes were optimised from analysis of 100 production lots vis-a-vis the fuel failure probability.

Fuel behaviour over the last 10 years has been excellent and this has given us valuable feed back on the role of quality characteristics of uranium oxide and zircaloy materials.

Systems for quality auditing and traceability of fuel to include manufacturing data and irradiation history in the reactor exist, which are being reviewed to suit growth in fuel production.

7. ACKNOWLEDGEMENTS

Thanks are due to Mr. S.L.Kati, Director Engineering and Mr. S.K.Chatterjee, Chief Engineer (Nuclear Design) for review of the manuscript and many useful suggestions. Authors are grateful to Dr. M.R.Srinivasan, Director, Power Projects Engineering Division for kind permission to publish the work.

8. REFERENCES

(1)   Rao, N.Kondal,
       Indian Experience on Production and Quality Control of Nuclear Fuels,
       Journal of Nuclear Materials, 106 (1982) 273-278

(2)   Das, M. et.al.,
       Evaluation of Indian PHWR Fuel Designs with particular reference to 22
       element Fuel Assembly for the Narora Atomic Power Station.

(3)   QA + QC. IAEA Technical Reports Series No. 173, 1976.

TABLE — 1

INDIAN PHWR STATIONS

Station	Location	Gross Capacity, MWe	Criticality Date
RAPS	Kota, Rajasthan	2 x 220	1972 1980
MAPS	Kalpakkam, Tamil Nadu	2 x 235	1983 1984
NAPS	Narora, U.P.	2 x 235	1986 1987
KAPS	Kakrapar, Gujarat	2 x 235	1990 1991

- -

TABLE — 2

DESIGN DATA FOR 19—ELEMENT AND 22—ELEMENT FUEL BUNDLE

	19—Element	22—Element
1. BUNDLE		
No. of Elements	19	22
Outer Ring of Elements	12	14
Intermediate Ring of Elements	6	7
Central Element	1	1
Maximum overall bundle diameter	81.69 mm	81.74 mm
Length of Bundle	495.3 mm	495.3 mm
Weight of Bundle	16.72 kg	16.21 kg
Weight of UO_2	15.2 kg	14.82 kg
Maximum bundle power (Nom.)	420 kW	487 kW
2. SHEATH		
Sheath material	Zircaloy—2	Zircaloy—4
Thickness	0.38 mm	0.38 mm
3. PELLET		
Diameter (Nominal)	14.4 mm	14.4 mm & 12.2 mm
Density of UO_2	10.6 Mg/m^3	10.6 Mg/m^3
Length to diameter ratio	1.4	1.2

- -

TABLE - 3

CLASSIFICATION OF QUALITY CHARACTERISTICS

3.1 ZIRCALOY CLADDING

CHARACTERISTIC	CLASSIFICATION	AFFECTS	QC PLAN
Chemical composition	Critical	Corrosion, strength	Ingot and product analysis for each lot.
Ductility (Burst Properties)	Critical	End of life TCE and PCI	Acceptance sampling AQL = 0.15%
Strength	Critical	Ridge formation	Acceptance sampling, AQL = 0.15%
Texture	Critical	Ductility and Fracture strength	Process control
Dimensional requirements (ID, OD, wall thickness)	Major	Bundle assembly	100% testing
Defects	Critical	Crack initiation	100% ultrasonic testing.

3.2 URANIUM OXIDE PELLETS

CHARACTERISTIC	CLASSIFICATION	AFFECTS	QC PLAN
Chemical composition (H, Ag, Cl, Ca, F)	Critical	Cladding integrity	Lot basis
Equivalent Boron Content (B, Cd, Dy, Gd)	Major	Fuel Burnup	Lot basis
O/U Ratio	Major	Thermal conductivity and fission gas release	Acceptance sampling, AQL = 1.0%
Density	Critical	Thermal conductivity & fission gas release	Acceptance sampling AQL = 2.5%
Microstructure	Major	Fission gas release, densification	Acceptance sampling, AQL = 2.5%
Chippability/crackability	Major	Local clad straining and PCI	Lot basis and element Decanning Test.
Visual	Medium	Pellet integrity and moisture pick up	Acceptance sampling AQL = 2.5%

TABLE – 4

4.1 FUEL OPERATIONAL DATA

(TO END OF OCT., 1983)

	RAPS – 1	RAPS – 2	MAPS – 1
BUNDLES IRRADIATED	12363	5976	3672
BUNDLES RECYCLED	8735	2304	–
DEFECT BUNDLES	19	1	–
CUMULATIVE OPERATION, EFPD	1333.04	471.95	10.87
COMMENCEMENT OF FUELLING., EFPD	178	174	–
	(OCT.'74)	(NOV.'81)	
POWER CYCLES:	255	60	8
INITIAL FUEL	180		
REPLACEMENT FUEL	60		
MAXIMUM BUNDLE POWER, kW	460	460	
FUEL BURNUP MW.d/teU			
MAXIMUM	14500		
AVERABE	7130	7000	

- -

4.2 FUEL FAILURES IN RAPS –1&2

(TO END OF OCT, 1983)

NO. OF BUNDLES	FOREIGN SUPPLY	BARC	NFC	TOTAL
TOTAL IRRADIATED	1792	2442	17777	22011
DEFECTS	5	4	11	20
DISCHARGED PREMATURELY	40	32	104	176
DEFECT RATE,%	0.28	0.16	0.062	0.091

- -

(1) RAPS : Rajasthan Atomic Power Station   2 x 220 MWe capacity

(2) MAPS : Madras Atomic Power Station   2 x 235 MWe capacity

(3) EFPD : Effective Full Power Days

DISCUSSION

E. STEINBERG: What is the reason of using Zry-2 for one and Zry-4 for the other fuel element?

R.S. RUSTAGI: We started with using Zircaloy-2 for cladding tubes in the 19-element bundle in 1971 for the initial fuel charge of the Rajasthan reactor I. Subsequently, for the 22-element bundle and also for the 19-element bundle, Zircaloy-4 was used.

M. SALIM: What is the weight of Zircaloy in your 22 element bundle?

R.S. RUSTAGI: In the 19-element bundle the weight of Zircaloy-2 is 1.5 kg, whereas in the 22-element bundle it is 1.4 kg Zircaloy-4. the total weight of the bundles is the same, 16.2 kg each.

M. ERNOTTE: What kind of microstructural details do you follow in your microstructure examination of pellets.

R.S. RUSTAGI: The longitudinal section of the pellet is examined to look for grain size, distribution of porosity, presence/absence of patches and granules.
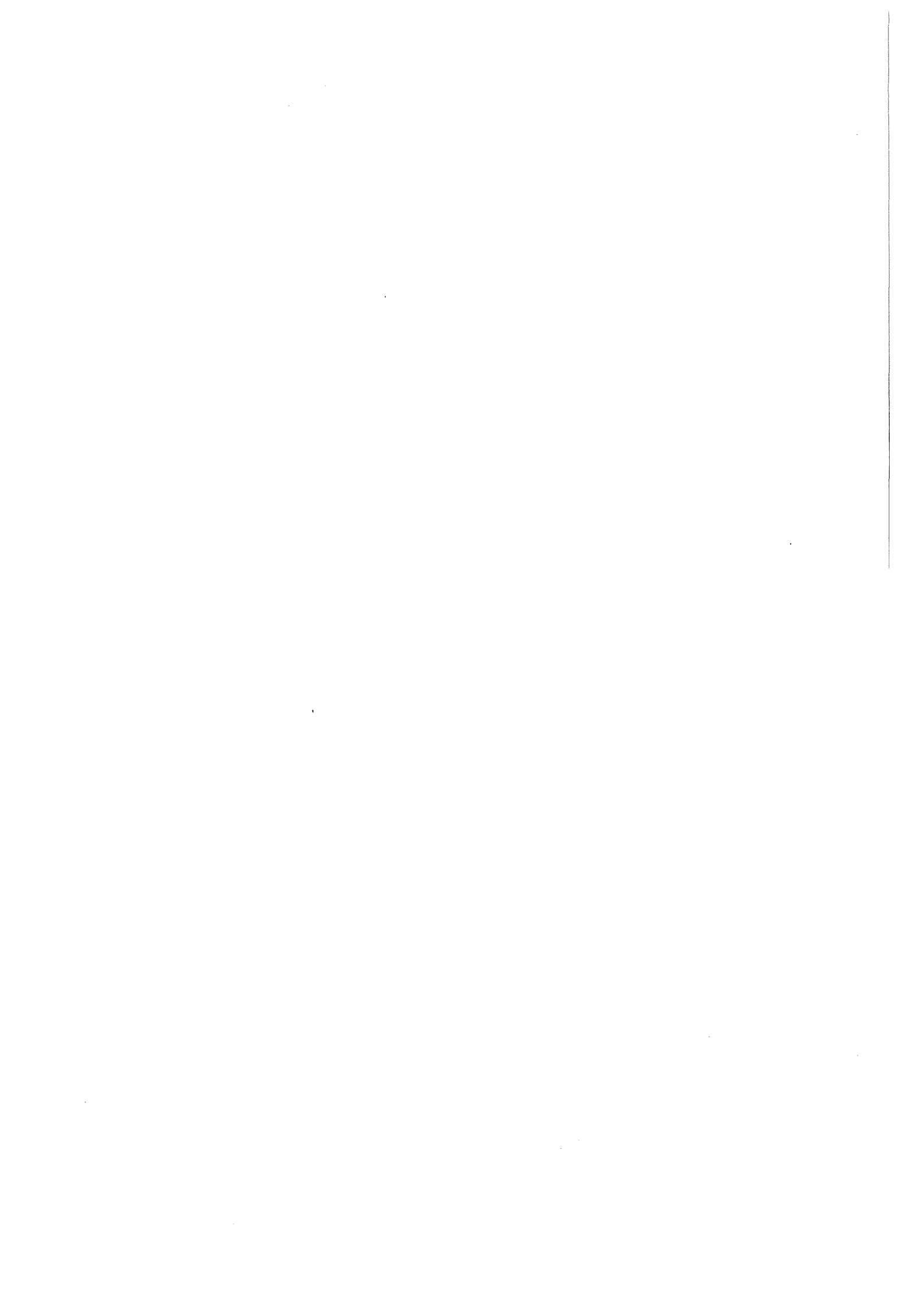
STATISTICAL METHODS IN THE QUALITY CONTROL OF UO_2 PELLETS AND ZIRCALOY CLADDING TUBES FOR WATER-REACTOR FUEL RODS

H. Assmann, E. Steinberg

Kraftwerk Union Aktiengesellschaft

D-8520 Erlangen, Germany

Abstract:

Statistical methods, based on random sampling, provide effective tools for the quality control in water-reactor fuel fabrication. This paper first presents a survey of major procedures for evaluating quality control data by distribution functions, their parameters and confidence regions as well as the testing of hypotheses on a given significance level. The practical application of those statistical methods is demonstrated by examples for diameter and moisture content of UO_2 pellets and for oxygen content, hydrogen content and elevated temperature yield stress of Zry cladding tubes. These examples show that a high and stable quality level can be secured by systematic product control, process control and control of testing techniques on a statistical basis.

1. INTRODUCTION

"Quality Control" (QC) has been subject to an extensive evolution during the past decades. Initially QC was limited to the determination of product properties in order to eliminate unsuitable products. Today, especially for products with high costs of manufacturing and the requirement of high reliability, QC is placed under the umbrella of "Quality Assurance" (QA).
Quality assurance is realized within a complete quality loop which is integrated into development, design, manufacturing and performance evaluation /1/. UO_2 pellets and Zircaloy cladding tubes for water-reactor fuel rods are such products of which the QA is established at Kraftwerk Union AG and her group of companies according to the "Quality Assurance System for Fuel Assemblies and Associated Core Components", described in another contribution to this seminar /2/. The following paper addresses QC under the aspects of statistical methods. Examples will be given for the application of statistical methods in product control, process control and control of testing techniques.

2. BASIC STATISTICAL METHODS FOR EVALUATING QC-DATA

Quality control applied to products as UO_2 pellets and Zircaloy cladding tubes requires sampling techniques. Therefore questions of the following type have to be answered: Which conclusions can be drawn from measured values of a sample in order to get information about the quality of the respective parent population? What is the reliability of such conclusions?
Can the sample size be economically adapted to the required quality level?

These questions can not be answered by statistical methods in a concise way before the analyst knows about or makes assumption on the distribution function of the measured values.

2.1 Distribution of Data and their Statistics

One has to distinguish between discrete probability distributions and continuous distributions. The most important distributions of both types are mentioned in Table 1. In practice, the normal distribution and the logarithmic normal distribution are used in many cases for QC-data.

The normal distribution is completely determined by its mean value μ and its standard deviation σ .

Often the logarithmic normal distribution has to be used when the value with the highest frequency is near zero and negative values do not occur.

Type of Distribution	Distribution Function	Example of Application
Discrete Distributions	Hypergeometric Distributions	Probability to find Defective Products in a Sample from a Population with known Number of Defective Products
	Binominal Distribution	Probability to find Defective Products in a Sample when the Probability for a Defective Product is known
	Poisson Distribution	Same as Binominal if Population Size is very Large and the Probability of the Occurance (e.g. Defects) is small
Continuous Distributions	Normal Distribution	Description of Q-data Distributions Decisions with Respect . . . to Acceptance and Rejection of Production Lots
	Logarithmic Normal Distribution	Same as Normal, if the Logarithmics of Q-data follow Normal Distribution
	Exponential Distribution	Failure Time of Complex Equipment
	Weibull Distribution	Lifetime of Products

Table 1 Survey of some Important Probability
Distributions and Examples of Application

Generally the evaluation of QC-data measured on a sample starts
by sorting the data in a tally chart (see Fig. 1) and plotting
the frequency histogram. In the next step the cumulative fre-
quencies are calculated and represented in a cumulative frequency
diagram (Fig. 1) by using a Gaussian scale. This diagram reads:
40 % of all values are not larger than 50 units, etc.
An ideal normal probability distribution yields a straight line
which intersects the 50 % level for the mean value μ and the
slope of which is determined by the standard deviation σ . The
smaller σ is, the steaper is the slope.

Class	Tally Chart	n	Σn_i	%
0–10				
11–20	I	1	1	1.7
21–30	II	2	3	5.0
31–40	‖‖ III	8	11	18.3
41–50	‖‖ ‖‖ II	12	23	38.3
51–60	‖‖ ‖‖ ‖‖ II	17	40	66.7
61–70	‖‖ ‖‖ IIII	9	49	81.7
71–80	‖‖ II	7	56	93.3
81–90	III	3	59	98.3
91–100				

a) $\qquad\qquad$ $\Sigma n_i + 1 = 60$
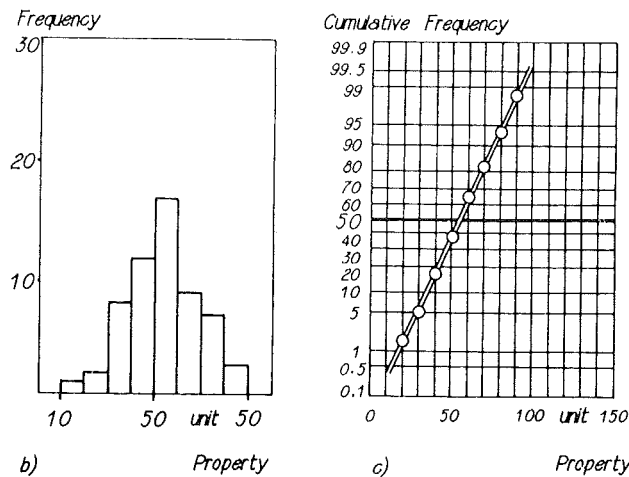
Fig 1 Representation of QC–Data
a) Table with Tally Chart b) Frequency Histogram
c) Cumulative Frequency Diagram

The following is valid:

68 % of all values are between μ + σ and μ – σ

95,5 % of all values are between μ + 2σ and μ – 2σ

99,7 % of all values are between μ + 3σ and μ – 3σ

In practice the QC-data obtained from a sample may approximately result in a straight line and we assume that the parent population can be described by a normal distribution.

2.2 Estimation of Parameters and Confidence Regions

The first task of our statistical evaluation is to estimate the parameters of the distribution function using estimates calculated from the sample distribution. Mean value $\bar{x}$ and standard deviation s of the sample, as defined according to Table 2 are estimates of μ and σ respectively.

By this method we get an approximation of the parameters which determine the distribution of the parent population. But we also need information about the quality of this approximation.

Parameters of the Sample	Parameters of the Parent Population	Probability Density Function
Mean Value $$\bar{X} = \frac{1}{n} \sum_{i=1}^{n} x_i$$	Estimate of μ	$$f(x) = \frac{1}{\sigma\sqrt{2\pi}}\, e^{-\frac{1}{2}\left(\frac{x-\mu}{\sigma}\right)^2}$$
Standard Deviation $$s = \sqrt{\frac{1}{n-1} \sum_{i=1}^{n} (\bar{x} - x_i)^2}$$	σ	

Table 2   Estimation of Parameters of the Normal Distribution

The second task of the statistical evaluation is to estimate intervals around the values of $\bar{x}$ and s within which the true unknown values of μ and σ are located with a distinct probability (e.g. 95 %, 99 %). This probability is called confidence level. Confidence intervals are calculated according to methods

described in the literature, see for example /3/. Table 3 gives a survey of the definitions of confidence intervals for the parameters μ and σ of a normal distribution and some rough information how to calculate them.

Confidence Interval for	Confidence Interval	Parameter a_i	Constant c_i from
μ (σ is known; large Sample Size)	$CONF \left\{ \bar{x}-a_1 \leqq \mu \leqq \bar{x}+a_1 \right\}$	$a_1 = C_1 \sigma/\sqrt{n}$	Normal Distribution
μ (σ is not known; small Sample Size)	$CONF \left\{ \bar{x}-a_2 \leqq \mu \leqq \bar{x}+a_2 \right\}$	$a_2 = C_2 s/\sqrt{n}$	Student's t–Distribution
σ	$CONF \left\{ a_3 S \leqq \sigma \leqq a_4 S \right\}$	$a_3 = \sqrt{\dfrac{n-1}{C_3}}$ $a_4 = \sqrt{\dfrac{n-1}{C_4}}$	χ^2 Distribution

Table 3   Estimation of Confidence Intervals for the Parameters of the Normal Distribution

For the calculation of confidence intervals we need constants c_i which are taken from the normal distribution function and the functions of two other important distributions: Student's t-distribution and Chi-square (χ^2) distribution. The values c_i for a particular confidence level γ (e.g. 95 %, 99 %) and a given sample size n can be taken from tables in the literature (e.g. /3/).

The confidence regions for μ and σ of the normal distribution can be represented in the diagram of cumulative frequencies in the form of a confidence belt (Fig. 2). The confidence belt is defined by the confidence limits for the mean value and by the confidence intervals for the standard deviation. The width of a confidence interval generally increases with higher values of the confidence level γ and decreases with larger sample size n. On the other hand, for a given confidence level and a given confidence interval, the correlated sample size can be minimized. Finally it can be concluded that with a probability of γ the true unknown distribution of the parent population is whithin the limits of the confidence belt.
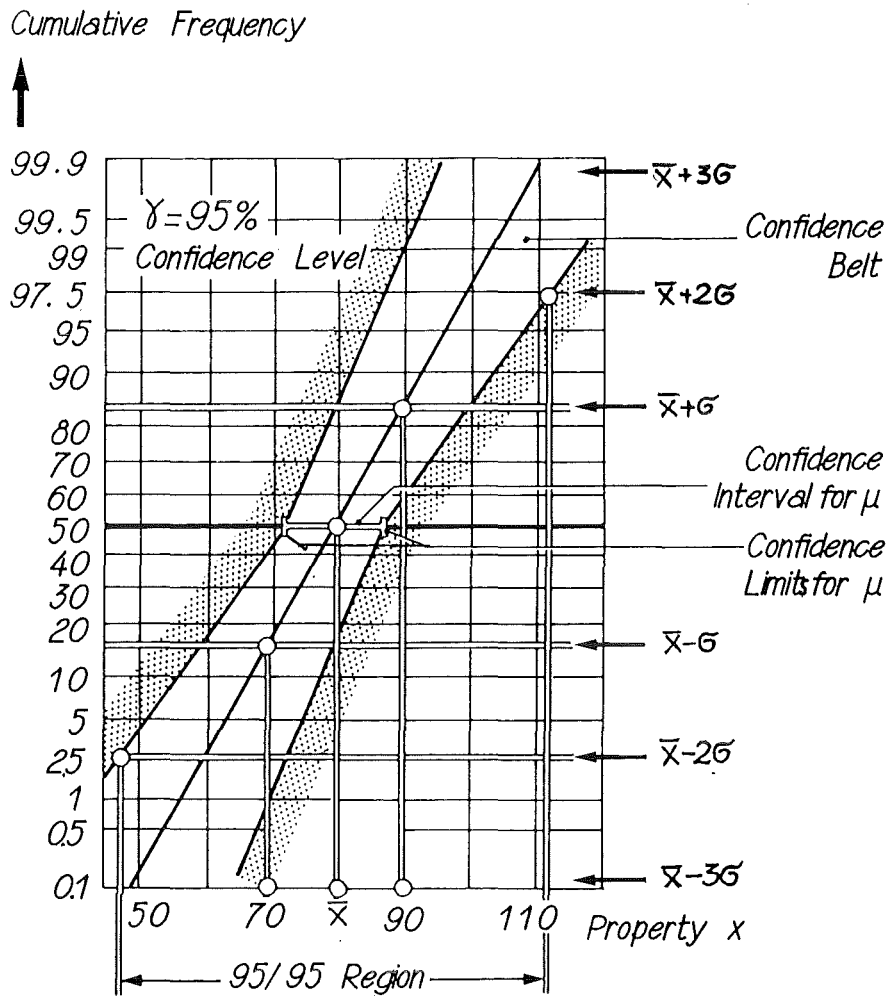
Fig2 Definition of Parameters and Intervals
of a Normal Distribution (schematically)

The meaning of the abbreviated expression "95/95 region" is: 95 %
of all values of the parent population are within this region
with a probability of 95 %.

2.3 Further Information about Parameters and Entire
Distributions

Another application of statistical methods is the testing of
hypotheses. A hypothesis is an assumption about the distribution
of a stochastic variable. A decision can be made whether a hy-
pothesis has to be rejected or not. This can be a basis for
acceptance and rejection actions in practical QC/QA in the pro-
duction. The first group of hypotheses considers the parameters
of the distributions.

Parameters of a sample distribution e.g. $\bar{x}$ and s, are compared with specified nominal values, with tolerances and their limits. What is the probability that the actual values of the product go beyond the specification limits? Are deviations stochastical or significant?

For problems of this kind we choose a desired significance level α (e.g. 5 %, 1 %). The meaning of the significance level is, the probability to reject a hypothesis which is right is α = 5 % or 1 %. In 95 % or 99 % of all cases we have met the right decision.

Table 4 shows some important examples. In any case a hypothesis H_0 is tested against an alternative hypothesis H_1. Of high practical interest are the questions whether the mean value $\bar{x}$ of a sample is equal to the mean value μ of the parent population and whether the standard deviation is not larger than e.g. one third of the specified tolerance. Other examples are: Comparison of the mean values and of the standard deviations of two normal distributions.

Task	Hypothesis	Method
Assessment of Estimation of Mean Value	$H_0 : \mu = \bar{X}$ $H_1 : \mu \neq \bar{X}$	Normal Distribution
Assessment of Estimation of Standard Deviation	$H_0 : \sigma \leq s$ $H_1 : \sigma > s$	χ^2 Distribution
Comparison of two Normal Distributions with Respect to Mean Value	$H_0 : \mu_1 = \mu_2$ $H_2 : \mu_1 \neq \mu_2$	t-Test
Comparison of two Normal Distributions with Respect to Standard Deviation	$H_0 : \sigma = \sigma_2$ $H_1 : \sigma \neq \sigma_2$	F-Test

Table 4 Testing of Hypotheses about Single Parameters of Distributions

Similar to the mathematical procedure for parameter and interval estimation for testing of hypotheses we need special test distribution functions. Those distributions are the normal probability distribution itself, the χ^2-distribution and Student's t-distribution, already mentioned and the F-distribution (or variance-ratio distribution) which can be found in the literature (see e.g. /3/).

The second group of hypotheses considers entire distributions. The problems may be (Table 5): Test that two populations, e.g. products manufactured in different periods of time or produced on different machines have the same type of distribution. Test that a parent population can be described by a normal distribution, the parameters of which are determined on a sample with small or large sample size. These questions can be answered by the use of the Kolmogorov-Smirnov two-sample test /4/ and the χ^2-Test /3/. All statistical methods mentioned in this section are in use at Kraftwerk Union AG and her affiliates especially in order to compare single populations, e.g. production lots of UO_2 pellets and Zircaloy cladding tubes with larger populations, e.g. the total production manufactured under the same conditions.

Task	Hypothesis	Method
Comparison of Distributions of two Populations P_1 ad P_2	$H_0 : D(P_1) = D(P_2)$ $H_1 : D(P_1) \neq D(P_2)$	Kolmogorov–Smirnov Two–Sample Test
Test of Normal Distribution of a Population	$H_0 : P = N(\bar{x}, S)$ $H_1 : P \neq N(\bar{x}, S)$	Small Sample: Kolmogorov–Smirnov One–Sample Test Large Sample: χ^2–Test

Table 5 Testing of Hypotheses about entire Distributions

3. APPLICATION OF STATISTICAL METHODS

3.1 General

Product control, process control and control of testing methods are equally performed in the production of UO_2 pellets as well as of Zircaloy cladding tubes. In process control the QC-data of precursor products and final products are followed up in the sequence of manufacturing steps and are correlated to particular process parameters. Thus, those parameters can be adjusted in order to obtain the desired properties of the product within the specified tolerance limits. Finally, statistical methods are suitable to control testing techniques. It is well known that the accuracy and variation of measured values can be assessed by statistical evaluation.

3.2 Examples of Quality Control on UO_2 Pellets

UO_2 pellets as the components of reactor fuel rods in which energy is generated by nuclear fission, have to fulfill many requirements with respect to their neutron-physics, thermal and mechanical behaviour. The properties of the UO_2 pellets which are submitted to a sophisticated QC program, partly by statistical methods, can be classified into the following major categories /5/:

1. Nuclear properties: e.g. enrichment
2. Chemical properties: O/U-ratio, impurities
3. Density and microstructure: density, pore-structure, grain size, etc.
4. Dimensions and surface conditions: e.g. diameter, length

In the following, two examples of statistical methods for UO_2 quality control will be given.

The **diameter of UO_2 pellets**, controlled by centreless grinding is specified within narrow limits ($\pm$ 0.010 mm) for optimization of the gap between pellets and cladding tube. The gap has to be

small in order to get a good heat transfer. On the other hand, a certain gap is required for filling in of the pellets. Therefore the knowledge of the statistical distribution of the pellet diameter is necessary. This distribution shall be described by a normal probability distribution function. So we can apply the statistical methods as summarized in the previous section.
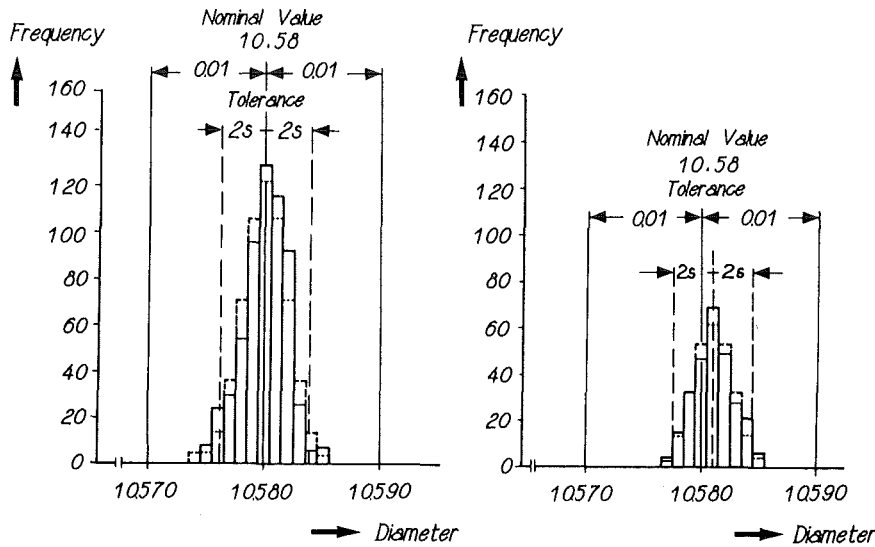
Fig 3   Frequency Histograms of UO$_2$ Pellet Diameter.
         Samples of Population P$_1$ (left) and P$_2$ (right)
         Observed (Bulk Lines) and Ideal Distribution (Dashed Lines)

We consider three populations of pellets. Two of them, P$_1$ and P$_2$ have been manufactured during different periods of time. The third population P$_{tot}$ is the total population which consists of 15 such single populations. Fig. 3 shows frequency histograms for the samples of P$_1$ and P$_2$ as compared to the ideal normal distributions. The tolerance limits are indicated in the figure. The χ^2-test for normality reveals that the hypothesis "The sample of P$_1$ is normally distributed" has to be rejected, whereas it does not have to for the samples of P$_2$ (and P$_{tot}$). This analytical result can neither be derived from the diagrams of Fig. 3 nor from the diagrams with the cumulative frequency lines of the three samples in Fig. 4. Disregarding this result we tested the hypotheses that the mean values and the standard deviations of the three samples are equal to eachother. Application of the t-test showed that all mean values are different on a significance level of 5 %. The F-Test showed that only the standard deviations of P$_1$ and P$_2$ can be considered to be equal on that significance level.
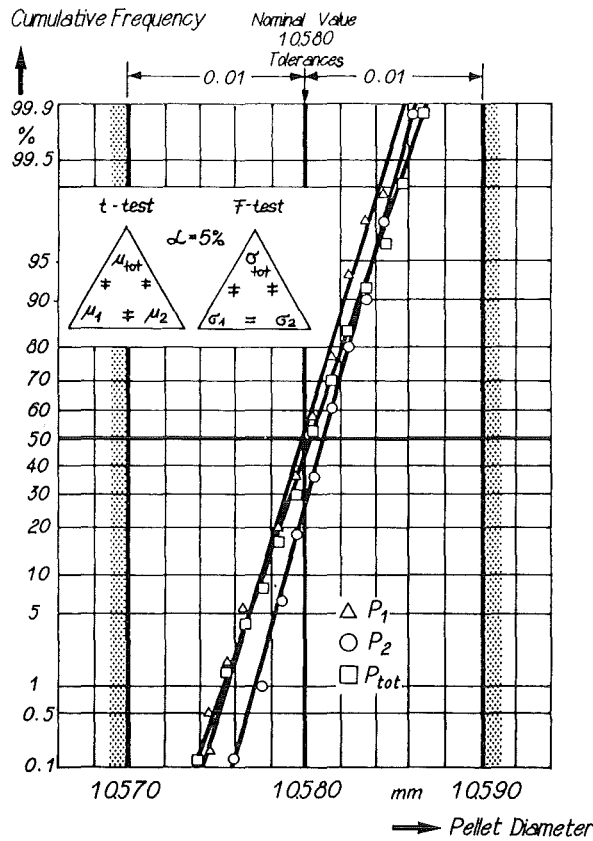
Fig. 4 Cumulative Frequency Diagram of UO_2
Pellet Diameter
Samples of Populations P_1, P_2 and P_{tot}

A special case of a normal probability distribution is the logarithmic normal distribution where the measured values have a natural limit. Such a limit we have in the case of the **moisture content of pellets** which only can be positive but are close to zero. In the calculation of parameters and confidence intervals (Table 2 and 3) the variable x_i has to be replaced by $\log x_i$. Therefore we obtain $\log \tilde{x}$ instead of the arithmetic mean value $\bar{x}$, where $\tilde{x}$ is the geometric mean value. Also we get $\log \mathcal{E}$ instead of σ (see the meaning of $\mathcal{E}$ in Fig. 5).

The moisture content of a sample of UO_2 pellets was measured. Fig. 5 shows the result of an estimation of the distribution parameters and confidence regions for normal and log-normal distribution together with a respective plot of the experimental cumulative frequencies. Although the log-normal distribution shows a better fit of data, the curve of cumulative frequencies also goes outside the confidence region. The S-shaped curve gives some indication that the sample is from two different parent populations. By a graphical method, described in /6/, the mixed

Cumulative Frequency
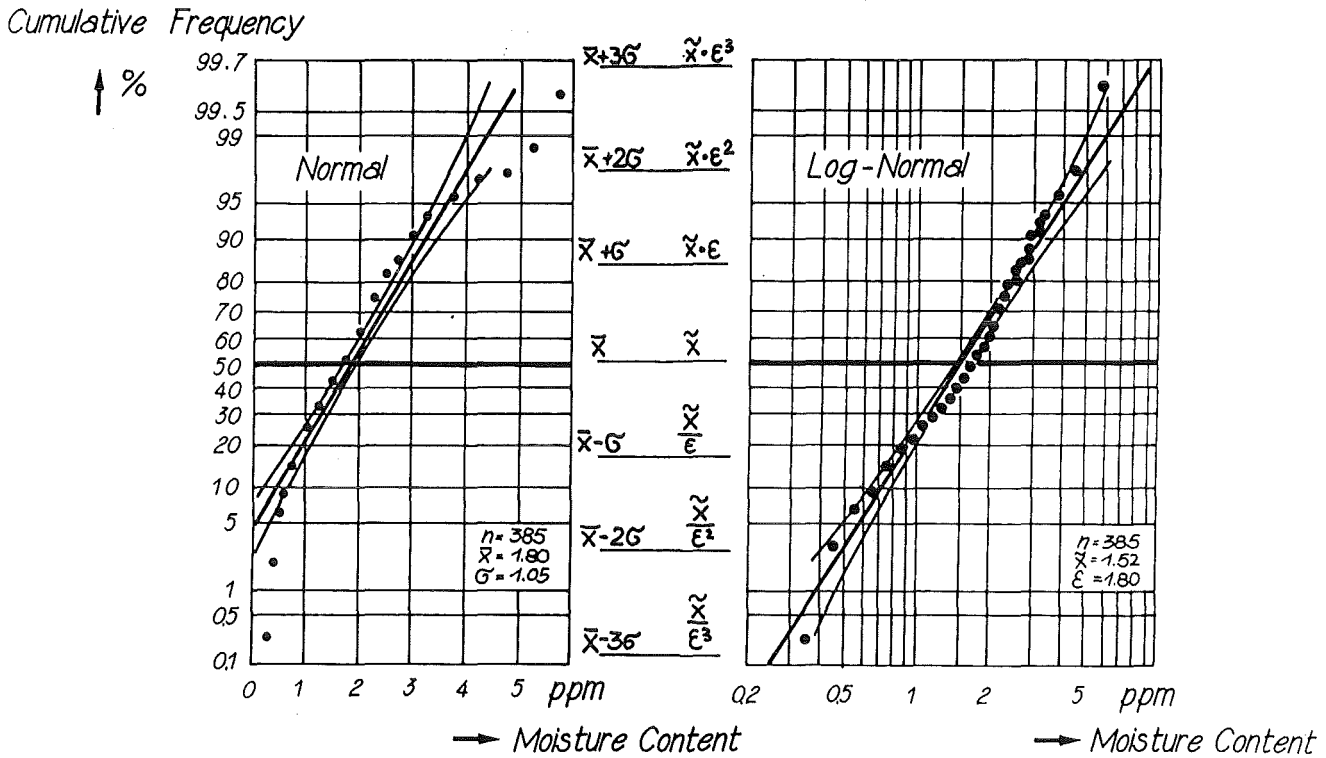
Fig. 5   Cumulative Frequency Diagram of UO_2 Moisture Content.
          Comparison of Normal and Log-Normal Models

distribution has been separated into two single collectives as
shown in Fig. 6. The calculated frequency distributions are
compared with the measured frequency histogram in Fig. 7. It
turned out that the two collectives are only different in the
geometric mean value $\tilde{x}$, whereas the logarithmic standard de-
viations are not significantly different. The average value of
the moisture content in the **fuel rod** is specified to be less than
10 wt-ppm.

From the calculated confidence intervals it can be concluded that
the predicted mean values for

collective 1   (n = 104): $0.68 < \tilde{x} < 0{,}79$ and
collective 2   (n = 281): $2.00 < \tilde{x} < 2.17$

are well below the tolerance limit.

The same mathematical method applied to the evaluation of the
open porosity of the pellets results in an equivalent separation
of two collectives. The complete analysis shows that there is a
relationship between moisture content and open porosity of UO_2
pellets.
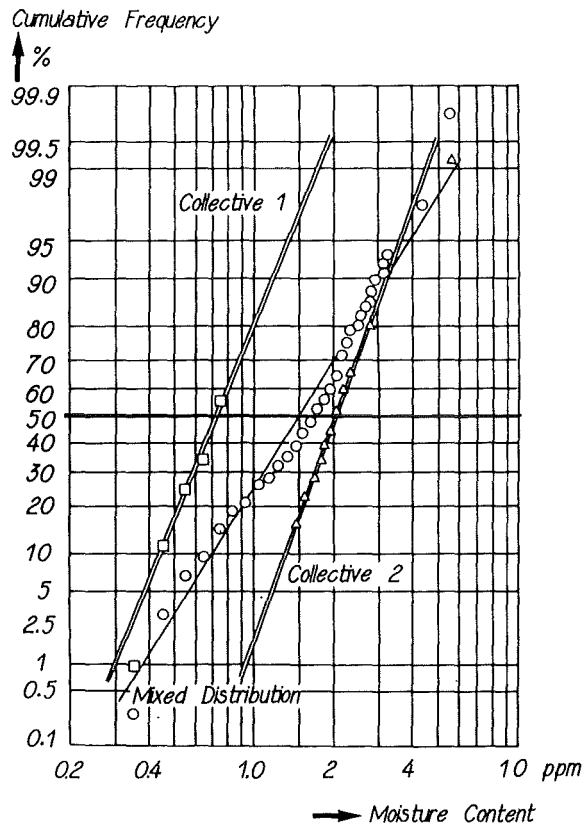
Fig 6 Moisture Content of UO₂ Pellets
Separation of a Mixted Distribution into two
Log-Normal Distributions
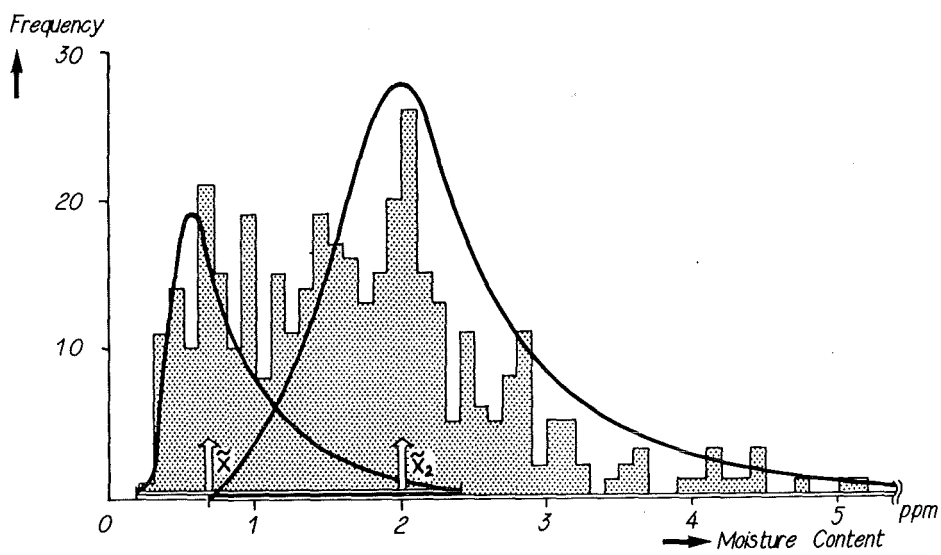
Fig 7 Moisture Content of UO₂ Pellets.
Frequency Histogram and Ideal Log-Normal
Distributions of two Populations

3.3 Examples of Quality Control on Zircaloy Cladding Tubes

Zircaloy cladding tubes are the most important barriers to prevent the escape of fission products from water-reactor fuel rods. High reliability during reactor operation is based upon a high level of product and process control during manufacturing of the tube /7/. The properties of the Zircaloy cladding tubes can be summarized by the following major items:

1. Chemical properties: composition, impurities
2. Dimensions, e.g. diameter, wall thickness, ovality
3. Mechanical properties and correlated microstructural properties: yield strength, fracture strength, etc.
4. Corrosion resistance
5. Integrity and surface conditions (e.g. ultrasonic test result)

The mechanical properties of Zry cladding tubes are mainly controlled by the chemical composition and by the thermal and mechanical treatment during the production of the tubes. One important example is **the dependency of the tensile strength on the oxygen content** of the material. Fig. 8 represents results from the measurements of tensile strength and oxygen content in the form of diagrams of cumulative frequencies.
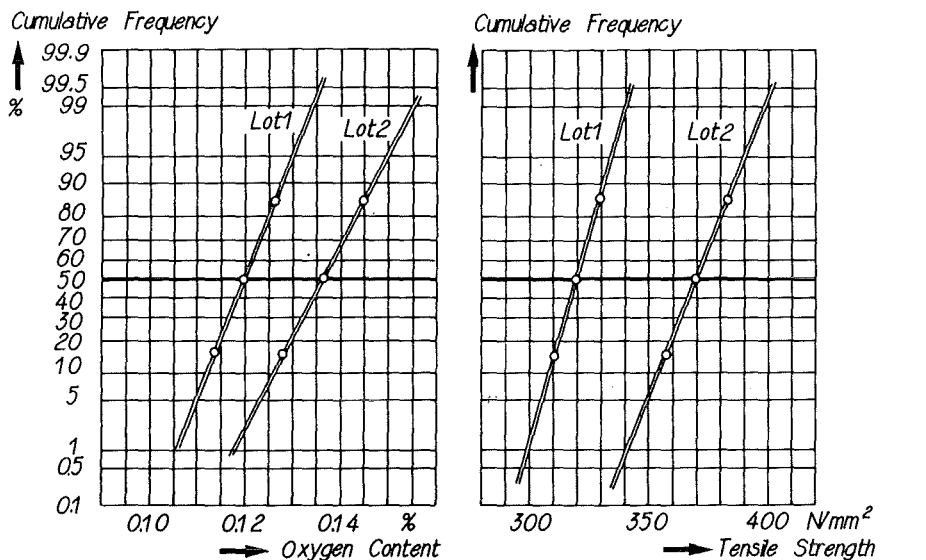
Fig 8 Influence of Oxygen Content on Tensile Strength

This example shows the application of statistical methods for two major tasks: comparison of the same product property on different lots of production, and comparison of different product proper-ties on the same lot of production. The confidence intervals of the mean values, not being shown in the figure, are not over-lapping. The single values of the oxygen content of the two populations are overlapping over a wide range. Thus, it can be seen that a separation of the two populations could not be done without statistical treatment of data. This is also true for the statistical distributions compared in Fig. 9.
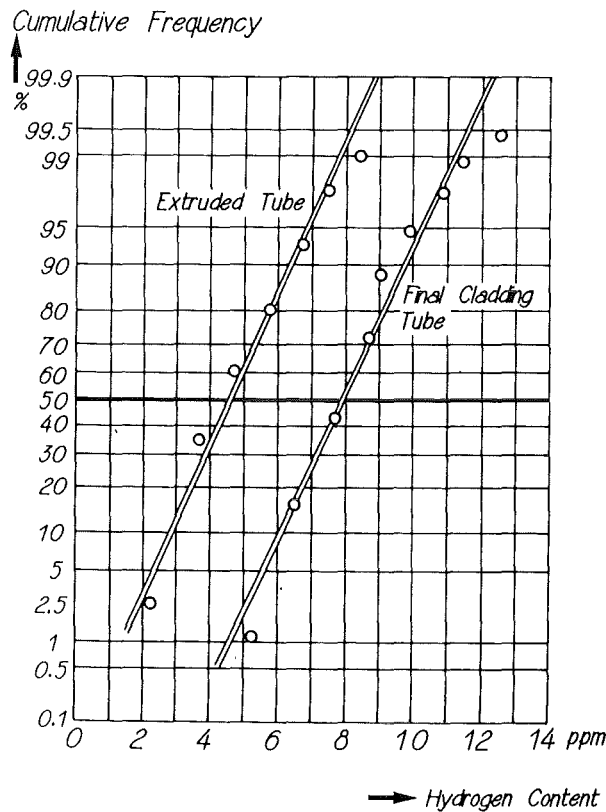
Fig.9 Hydrogen Content of Zry Cladding Tubes

The hydrogen content of the Zry cladding tube material has been measured in two stages of production (Fig. 9). Higher concen-trations of hydrogen are undesirable because the ductility of the material is being decreased by this. The hydrogen content is already determined by the manufacturing of the prematerial. The reported statistical evaluation reveals that the hydrogen content increases during the sequences of production which contain some intermediate annealing steps. In average, this rise is very small

and could not have been detected from single measurements because
of the scattering of values which - to a certain amount - are due
to the measurement technique.
The true value of hydrogen take-up cannot be derived before the
statistical evaluation has been done.

Statistical methods can not only be applied for product control
and assessment of process steps but also for the optimization
process parameter in relation to required product properties. **The
yield strength of Zry cladding tubes** shall be controlled by a
distinct manufacturing step. The specified range for the yield
strength at 400 °C is between 200 and 300 N/mm².

The question is: Is it possible to fulfill this requirement using
an available annealing furnace? In the furnace the material is
exposed to deviation from the nominal temperature as well as to
deviations from the nominal annealing time. The tubes at the rim
of the furnace reach the nominal temperature earlier than those
in the centre. Heat transfer and heat capacity play a certain
role.

A pilot run of the annealing process is performed with estimated
annealing conditions: e.g. ϑ = 525 °C, t = 5 h. During this total
annealing cycle the tubes are at the nominal temperature only
3.5 hrs. and the rest is the time for heating up and cooling
down. The results of tensile tests of samples from the pilot run
are statistically evaluated. From fig. 10 can be derived that
95 % of all tubes for a confidence level of 95 % are between
220 N/mm² and 307 N/mm².

Figure 11 shows the yield strength as a function of an annealing
parameter which contains the annealing time as well as the
annealing temperature - and is defined by
A = t exp(-Q/RT). Entering the measured range of yield strength
into the diagram of Fig. 11 the following correlations are found:

$$R_{p_{0,2}} = 307 \text{ N/mm}^2 \longrightarrow A = 3.35 \cdot 10^{-22} \text{ h},$$

$$R_{p_{0,2}} = 220 \text{ N/mm}^2 \longrightarrow A = 9.60 \cdot 10^{-22} \text{ h}.$$

Under the assumption that the annealing time for all positions in the furnace is the same, the variation of the yield strength can be explained by a temperature interval between 516 °C and 533 °C. The diagram of Fig. 11 also shows that the aim to control the yield strength within the region between 200 and 300 N/mm² has hardly been missed.
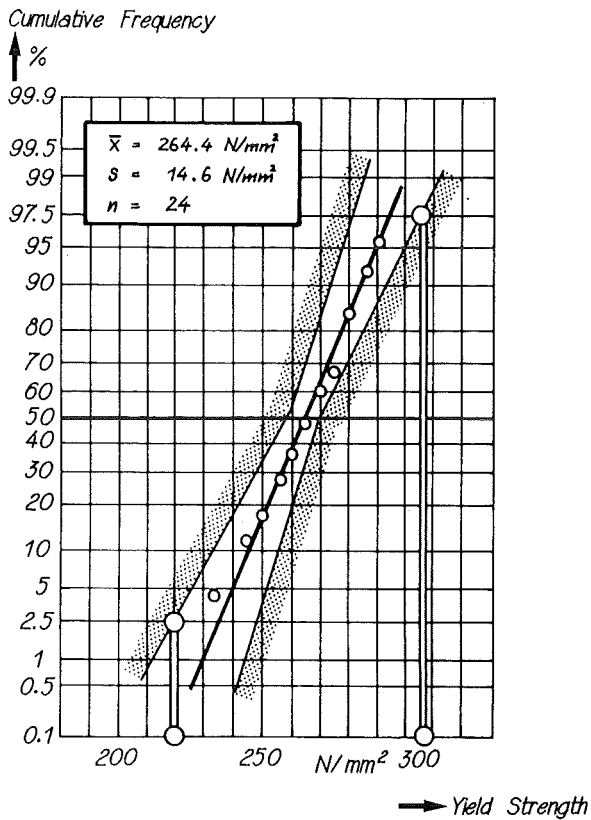
Fig 10 Yield Strength of Zry Cladding Tubes
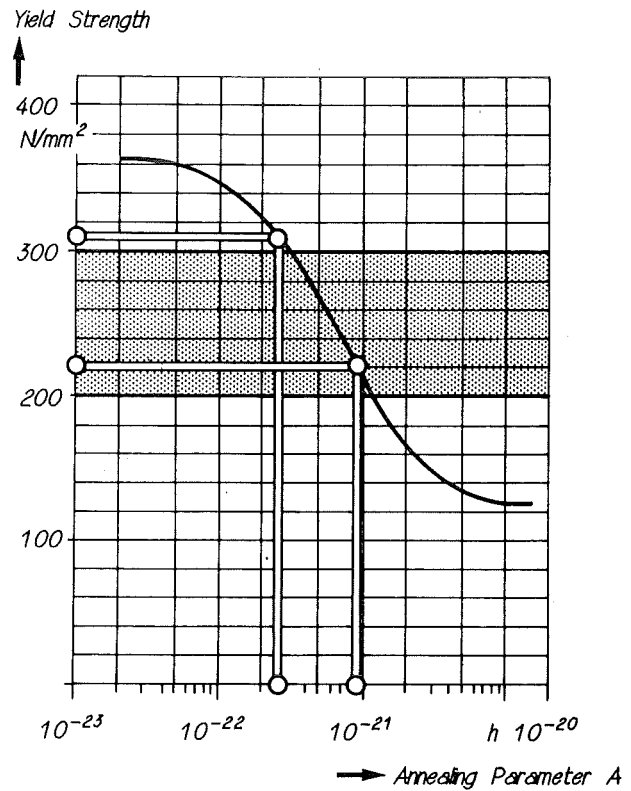
Fig.11 Yield Strength of Zry Cladding Tubes as a Function of the Annealing Parameter $A = t \cdot exp(-Q/RT)$

Using the parameter A it can be calculated that either the nominal temperature has to be increased by 2.5 °C or the annealing time has to be increased from 3.4 hrs to 4 hrs with unchanged temperature. Both measures lead to an average value of the yield strength of $R_{p_{0,2}} = 250$ N/mm² which is exactly in the middle of the desired range.

3.4 Control of Testing Methods (Example) /8/

A so called "round robbin test" provided informations about the accuracy of the **measurement of the yield strength** by the tensile test. Six laboratories of different companies have been involved in this investigation and have received samples from the same Zry tube for tensile tests at elevated temperature. The results have statistically been evaluated by an independent institution.
Fig. 12 shows the result. Taking into account the confidence region for a confidence level of 95 % it can be stated that 95 % of all measured values are within ± 5 % of the mean value.

It cannot be seen from the statistical data what the reasons for these deviations are. They can be caused by:

- inevitable variations of material properties within a single tube
- deviations of the testing temperature
- incorrect measurement due to the testing machine, e.g. deviations in the strain rate
- missing exactness in the evaluation of testing diagrams.
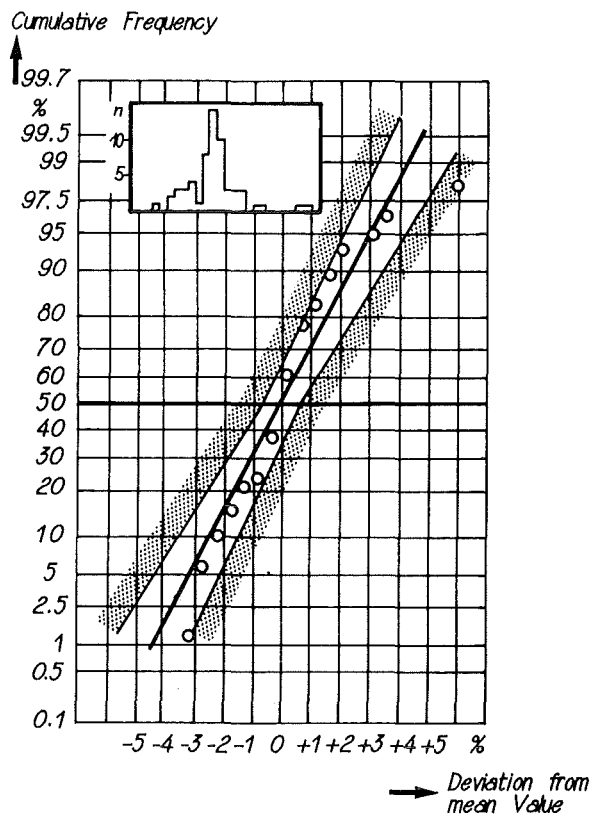
Fig. 12 Deviations in Tensile Test Results of Different Laboratories on the same Test Material

A thorough analysis of data revealed that two third of the uncertainty was due to the evaluation of the testing diagrams.

4. CONCLUSIONS

The statistical methods presented in this paper and their practical application in the quality control of UO_2 pellets and Zircaloy cladding tubes show that we have simple but strong tools for direct product control, for the quality assurance by means of process control and control of testing techniques.

These methods are being used at Kraftwerk Union AG and her group of companies during the standard production, for the qualification of advanced products and processes, in design and development. Data acquisition, documentation and evaluation are more and more being done by computer aided methods /9/. Another trend is the use of computerized on-line data processing /10/.

5. REFERENCES

/1/   IAEA; Guidebook on Quality Control of Water Reactor
        Fuel, Technical Reports Series No. 221, STI/DOC/10/221,
        Vienna, 1983

/2/   R. Holzer and F. Nilson, Experience with Quality Assu-
        rance in Fuel Design and Manufacturing, IAEA-Seminar
        1984, Karlsruhe

/3/   E. Kreyszig, Statistische Methoden und ihre Anwen-
        dungen, Vandenhoeck & Ruprecht, Göttingen, 6. Aufl.,
        1977

/4/   IMSL Library Reference Manual, IMSL LIB-0008, revised,
        June 1980

/5/   H. Assmann and H. Stehle, Oxide Fuel for Light Water
        Reactors - a Standardized Industrial Product, Energy
        and Ceramics, Elsevier Scient. Publ. Comp., Amsterdam-
        Oxford-New York, 1980, p. 893-912

/6/   H. Sauer H. and F. Wix, Graphische Häufigkeitsanalyse
        für Zeitstudien, REFA e.V., Darmstadt, 1960, 34-42

/7/   H. G. Weidinger and K. H. Kunz, Methods of Quality
        Control for Zircaloy Tubing, IAEA Seminar 1984,
        Karlsruhe

/8/   E. Steinberg and H. G. Weidinger, Statistische Aus-
        wertung von Daten aus der Qualitätskontrolle, am Bei-
        spiel dünnwandiger Rohre aus Zircaloy für die Reaktor-
        technik, Conference "Werkstoffprüfung 1983", Bad Nau-
        heim, Germany Dec. 8-9, 1983

/9/   H. Engel, Use of Software-System for the Quality Assu-
        rance of the Production of Fuel Elements for Power
        Reactors, IAEA-Seminar 1984, Karlsruhe

/10/   M. Streb, J. Steven and H. J. Romeiser, Quality Control
        for PWR-Spacers by Means of a Computer Assisted Auto-
        mated Test Equipment, IAEA-Seminar 1984, Karlsruhe

DISCUSSION

R.S. RUSTAGI: Would the statistical evaluation techniques described in your paper apply to the situation when the process may get out of control? How to test the data for identifying loss of control and how to treat the data for obtaining meaningful results?

H. ASSMANN: The statistical evaluation techniques explained in the paper show very easily when there is a tendency of process parameters to leave the desired range, and they help to keep the process under control. Identification of loss of control would be also possible by statistical evaluation procedures as can be derived from the example for the control of the yield strength.

A. STRASSER: Do you find that the type of distribution changes with time for any parameter in your process?

H. ASSMANN: We do not find a change in the type of distribution function for the particular items when the process is unchanged. But it is possible that the interpretation may change from an approximation to a more exact description, e.g. a log-normal distribution with small variation can be considered to be a normal distribution. Examples in this paper are the oxygen and the hydrogen contents of Zircaloy.

SYSTEME DE CONTROLE EN CONTINU DES FABRICATIONS

J.C. NURY and J.P. ROBIN
FRAGEMA, 149 Cours Lafayette
69006 LYON - F
M. FOURE, C. MORIN, J. VAN BENEDEN
FBFC, 26104 ROMANS SUR ISERE - FRANCE

The continuous inspection process is a method used in a Manufacturing plant in order to obtain a fast feed-back between inspection results and manufacturing process. It often requires a specific approach in design and a wide cooperation between Enginee-ring and Manufacturing people. This method leads to additional advantages such as high motivation, and better quality knowledge.

<u>1ère Partie</u> : Introduction - Les différents schémas de contrôle.

D'une manière générale, deux conceptions du contrôle de la qualité des produits fabriqués peuvent être envisagées : le contrôle séquentiel et le contrôle en continu.

Dans le cas du CONTROLE SEQUENTIEL (ou contrôle final par lot), même si les opérateurs de fabrication exécutent des vérifications de leur travail, le constat de la conformité repose, pour l'essentiel, sur les mesures et examens effectués par les inspecteurs d'un Service Contrôle Qualité indépendant. Bien que largement utilisée, cette façon de faire, présente comme nous le verrons plus loin, de sérieux inconvénients au plan, d'une part, de la motivation des opérateurs, d'autre part, de la connaissance et de la maîtrise de la qualité fabriquée et de la qualité sortante.

Dans le cas du CONTROLE EN CONTINU (ou contrôle intégré à la fabrication), les opérateurs de fabrication mettent eux-mêmes en oeuvre, le plus tôt possible dans le process, les moyens de vérification de la qualité de leur travail, dans le cadre de PLANS DE CONTROLE strictement définis. Les inspecteurs du Service Contrôle Qualité, qui conservent la responsabilité d'attestation de la conformité, appliquent alors des PLANS DE SURVEILLANCE visant à assurer que les PLANS DE CONTROLE ont été respectés et que la qualité sortante répond en permanence aux objectifs fixés. Un tel schéma rend les unités de fabrication responsables de la qualité qu'elles réalisent et leur permet de réagir promptement en cas de dérives ou d'incidents ; il respecte néanmoins le principe de jugement indépendant de la conformité finale des produits.

2ème Partie : Critères de choix d'une organisation du contrôle.

Sur le plan des principes, l'organisation des contrôles mise en place par l'entreprise doit être adaptée :

1. A la politique de qualité de l'entreprise :
(attribution des responsabilités, indépendance des services concernés, motivation du personnel).

2. Aux conditions de réalisation pratiques des contrôles et aux conséquences qu'il est possible d'en tirer sur le procédé de fabrication.

3. Au contexte historique de l'entreprise.

En termes pratiques, cela signifie tout d'abord, que le choix de l'organisation du contrôle résulte de la politique générale de la qualité de l'entreprise et de l'examen des avantages et inconvénients des deux organisations possibles définies dans la première partie (voir figure 1).

Les conditions techniques d'exécution de chaque contrôle peuvent également influer sur le choix de l'organisation.

Le contrôle séquentiel convient mieux aux contrôles très spécialisés ayant un temps d'exécution important ou à des pièces réalisées unitairement.

Le contrôle en continu est adapté à des fabrications de pièces de série ne nécessitant que des contrôles simples à exécuter, pouvant être réalisés au fur et à mesure de la production des pièces.

Enfin, il est important de noter que le contrôle en continu ne peut être implanté correctement que dans une entreprise ayant préalablement mis en place un système d'Assurance de Qualité. En effet, ce système nécessite la confiance du Service Contrôle dans les méthodes mises en oeuvre par la fabrication et nécessite donc, l'accord réciproque sur les procédures utilisées. D'autre part, l'application de ce système entraine une diminution des effectifs du Service Contrôle, un renforcement des effectifs de la fabrication, et un changement important des habitudes et des comportements. Ceci suppose donc, pour des raisons psychologiques, que la Direction de l'Entreprise ait placé l'objectif de qualité au premier rang, et que tout le personnel en ait clairement conscience.

FIGURE 1

AVANTAGES ET INCONVENIENTS
DE CHAQUE ORGANISATION

ORGANISATION 1 : CONTRÔLE CLASSIQUE (SÉQUENTIEL)

AVANTAGES	INCONVÉNIENTS
- ORGANISATION SIMPLE - NE NÉCESSITE PAS DE FORMA-TION PARTICULIÈRE DU PERSON-NEL DE FABRICATION AUX TECHNI-QUES DE CONTRÔLE. - LES MÉTHODES DE TRAVAIL DU SERVICE CONTRÔLE SONT SIMPLES À DÉFINIR CAR ELLES SONT INDÉ-PENDANTES DES VÉRIFICATIONS EXÉCUTÉES PAR LA FABRICATION.	- FAIBLE MOTIVATION DU PERSONNEL DE FABRICATION : IL NE SE SENT PAS RESPONSABLE DE LA QUALITÉ. - FAIBLE MOTIVATION DU PERSONNEL DE CONTRÔLE : IL NE MAITRISE PAS SA CHARGE DE TRAVAIL. - CONFLITS ENTRE FABRICATION ET CONTRÔLE. - QUALITÉ FLUCTUANTE. TRI ET RE-CONTROLE FRÉQUENTS. - RÉACTION LENTE ENTRE CONTRÔLE ET FABRICATION.

ORGANISATION 2 : CONTRÔLE EN CONTINU

AVANTAGES	INCONVÉNIENTS
- MOTIVATION ÉLEVÉE DU PER-SONNEL DE FABRICATION. - TRAVAIL D'ÉQUIPE. - QUALITÉ CONSTANTE : RÉACTION RAPIDE SUR LE PROCESS.	- UNE FORMATION PARTICULIÈRE DU PERSONNEL DE FABRICATION EST NÉ-CESSAIRE. - LES MÉTHODES DE TRAVAIL DU CON-TRÔLE DOIVENT ÊTRE ADAPTÉES POUR TENIR COMPTE DU TRAVAIL DE VÉRIFI-CATION RÉALISÉ PAR LA FABRICATION.

3ème Partie : Définition des exigences du concepteur

Adaptation au système de contrôle en continu

Pour un composant fabriqué en série et pour une caractéristique donnée, le concepteur se trouve devant le choix suivant pour définir son exigence de qualité :

1. Indiquer une tolérance et associer une exigence de contrôle statistique à cette tolérance.

2. Indiquer une tolérance et associer une exigence de contrôle à 100 % à cette tolérance.

Le choix qu'il fait dépend, bien sûr, de l'importance qu'il attache à la caractéristique en question, et des possibilités pratiques du contrôle industriel.

Dans le 1er cas le contrôle qui sera réalisé pour la caractéristique aura une efficacité qui dépendra de la formulation statistique utilisée, et cette efficacité, ou probabilité de détection des éléments défectueux, sera toujours inférieure à 100 %.
Il s'y ajoutera l'efficacité des opérateurs ou des moyens de contrôle, la précision des appareils de mesure, mais ces éléments sont en général d'un ordre de grandeur inférieur au facteur précédent et il est implicitement considéré que l'efficacité de la méthode de contrôle n'est due qu'à la formulation statistique utilisée. Cette approximation n'est pas gênante pour le concepteur lorsqu'elle n'intéresse que des caractéristiques d'importance mineure pour lesquelles un taux de défectueux de l'ordre du pourcent est admis.

Dans le cas du contrôle à 100 % les facteurs cités deviennent du 1er ordre pour la détermination de l'efficacité de la méthode de contrôle et ce serait une erreur de la part du concepteur que de croire, qu'avoir exigé un contrôle à 100 % le conduit à une efficacité de 100 %. Ce fait est d'autant plus important que cette exigence en général, est réservée à des caractéristiques majeures.

Dans le cas de caractéristiques majeures pour lesquelles un contrôle à 100 % est impossible, la formulation de l'exigence du Concepteur devra être telle qu'elle permette de tenir compte des facteurs autres que ceux liés à la formulation statistique, au cas où ces facteurs deviendraient du 1er ordre.

Nous voyons ainsi que dans chacun des cas ci-dessus, l'efficacité du contrôle sera inférieure à 100 %. Or pour la mise en application du contrôle en continu, la surveillance exercée par le Contrôle Qualité devra tenir compte de l'efficacité du contrôle réalisé par la fabrication. Ceci devra être possible, en particulier pour les caractéristiques contrôlées à 100 % et pour les caractéristiques pour lesquelles

plusieurs facteurs interviennent sur l'efficacité. Le Concepteur devra donc for-
muler son exigence de qualité de manière à ce que le fabricant puisse analyser
les moyens mis en oeuvre vis à vis d'un objectif synthétique, par exemple un
objectif de qualité sortante qui servira de référence commune pour les contrôles
de fabrication et les contrôles de surveillance, cet objectif pouvant être asso-
cié à une exigence de contrôle à 100 %.

Compte-tenu du caractère de la surveillance effectuée par le Contrôle Qualité,
à savoir :
- prise en compte de l'hypothèse d'un produit correspondant à l'exigence de
qualité car déjà contrôlé,
- produit fabriqué en série continue,

une pratique adaptée est, par exemple, d'utiliser pour le contrôle de surveillan-
ce, les plans d'échantillonnage définis par la norme MIL St 105 D ou NFX-06-022
en prenant pour valeur du NQA l'objectif de qualité sortante.

Nous voyons ainsi que l'implantation du contrôle continu conduit à une réflexion
sur l'efficacité des méthodes de contrôle utilisées, en particulier pour les carac-
téristiques contrôlées à 100 % et entraine une définition supplémentaire de l'ob-
jectif de qualité par le concepteur. Ceci exige une coopération étroite entre le
fabricant et le concepteur pour l'implantation de ce système de contrôle et cor-
respond à la démarche qui a été adoptée par FRAGEMA et FBFC.
L'implantation de ce système dans une unité en activité impose au concepteur de
s'interroger sur la validité de ses exigences, et impose au fabricant de recher-
cher les causes des défauts pour limiter les contrôles à mettre en place, compte
tenu de l'efficacité de ces derniers.

A titre d'exemple, nous pouvons citer :
- le contrôle de diamètre des pastilles, sujet développé dans la 4ème partie,
- le contrôle radiographique des soudures circulaires de bouchon sur gaine de
crayon combustible : sujet présenté lors de la 6ème conférence internationale
sur les essais non destructifs dans l'industrie nucléaire à Zürich en Décembre
1983 par FBFC et FRAGEMA.

4ème Partie : Implantation du contrôle en continu

Considérant une ligne de fabrication dans laquelle le contrôle séquentiel, tel que défini dans la première partie, est pratiqué souvent depuis plusieurs années, le passage au contrôle en continu ne peut se faire brutalement et nécessite, au préalable, deux étapes fondamentales :

L'évaluation technique de la situation et la formation du personnel.

1. EVALUATION TECHNIQUE

L'évaluation technique consiste à rassembler pour les caractéristiques spécifiées d'un produit donné, toutes les informations chiffrées permettant d'obtenir, sur une période de production suffisante, la connaissance objective de la qualité fabriquée ; il peut arriver qu'à l'occasion de cette phase d'analyse, des retouches des règlages du process soient introduites afin de parfaire la stabilité de la qualité. En bref, l'évaluation qui dans certains cas pourra s'étendre sur plusieurs mois et nécessitera un renforcement, voire un doublement des contrôles, consistera à déterminer la stabilité des machines et leurs variabilités, l'efficacité des procédés de contrôle et celle des opérateurs eux-mêmes, notamment pour les contrôles à 100 % (visuel et contrôles divers par attributs).

C'est dans cette phase d'approfondissement technique que le dialogue entre concepteur et fabricant prend toute sa valeur. Il permet en effet de fixer en toute connaissance de cause les niveaux de qualité acceptable et éventuellement de retoucher certains critères ou tolérances des spécifications. Il sera ensuite possible de bâtir, dans un langage commun les PLANS DE CONTROLE et de SURVEILLANCE qui seront appliqués par le fabricant et par les inspecteurs du client. Enfin, les services de fabrication et contrôle qualité pourront rédiger ensemble les procédures détaillées d'application et s'assurer que leurs opérateurs, procédés et instruments de contrôle sont parfaitement étalonnés entre eux.

2. FORMATION DU PERSONNEL

Le changement profond des habitudes que nécessite le contrôle en continu implique l'effort correspondant de formation, tant au plan technique que psychologique, de l'ensemble du personnel (techniciens, agents de maitrise et ouvriers). Les sessions de formation soigneusement organisées, portent essentiellement sur la définition du produit à obtenir, la description détaillée de l'ensemble du procès de fabrication contrôle, les bases statistiques nécessaires, l'utilisation des cartes de contrôle. Parallèlement, le contrôle en continu est replacé dans l'ensemble de l'organisation de la qualité et de ses objectifs et les responsabilités respectives de la fabrication et du contrôle qualité sont reprécisées.

Le message à faire passer peut être résumé de la façon suivante :
Les unités de fabrication disposant de tous les moyens pour maîtriser et vérifier
la qualité des produits qu'elles réalisent et pour réagir en temps réel en cas de
dérive, ne doivent présenter aux inspecteurs du Contrôle Qualité que des produits
correspondant au niveau de qualité exigé. Inversement, les agents de Contrôle Qua-
lité, moins nombreux mais plus polyvalents, doivent comprendre que leur tâche
n'est plus de trier des défectueux mais de s'assurer que le process demeure bien
sous contrôle et de détecter d'éventuelles dérives lentes.

3. EXEMPLE D'APPLICATION

Le contrôle en continu a été mis en place, à l'heure actuelle, sur la majorité
des lignes de produits dans les usines de F.B.F.C. Il serait trop long dans le
cadre de cette communication d'en présenter l'ensemble.
Nous choisirons de décrire succinctement le cas de la fabrication des pastilles
d'oxyde d'uranium.
En simplifiant et en excluant les contrôles destructifs (analyses chimiques et
isotopiques) on peut considérer que trois contrôles essentiels sont exécutés sur
les pastilles d'UO2 frittées et rectifiées : le diamètre, la densité et l'aspect.
Pour les deux premières caractéristiques, un contrôle à 100 % étant exclu, la vé-
rification par mesure par la fabrication porte sur un échantillon statistique
fréquent et donne lieu à la tenue de cartes de contrôle. Pour le diamètre, la
surveillance du Contrôle Qualité est exécutée au minimum une fois par poste de
travail et après chaque réglage de la rectifieuse ; alors que pour la densité,
le Contrôle Qualité recontrôle les mêmes échantillons que la fabrication afin de
s'assurer qu'il n'y a pas de dérive de la méthode et de l'équipement de contrôle.
L'aspect des pastilles donne lieu à un "contrôle à 100 %" (tri) par la fabrication
dont l'efficacité est ensuite reconnue par le Contrôle Qualité, par prélèvement
et schéma décisionnel basé sur des valeurs de NQA fonction de la nature des dé-
fauts (éclats, fissures, etc ...)

La figure 2 présente le schéma opérationnel pour le contrôle du diamètre.

Dans une telle organisation des contrôles, les pastilles d'UO2 sont libérées en
continu par train de quelques plateaux, alors que le contrôle séquentiel por-
tait autrefois sur des lots de 1500 à 2000 kg. On imagine aisément les problèmes
posés lorsqu'un lot était refusé par le Contrôle Qualité du fabricant ou par
l'inspecteur du client.

FIGURE 2

CONTROLE EN CONTINU
DIAMETRE PASTILLES
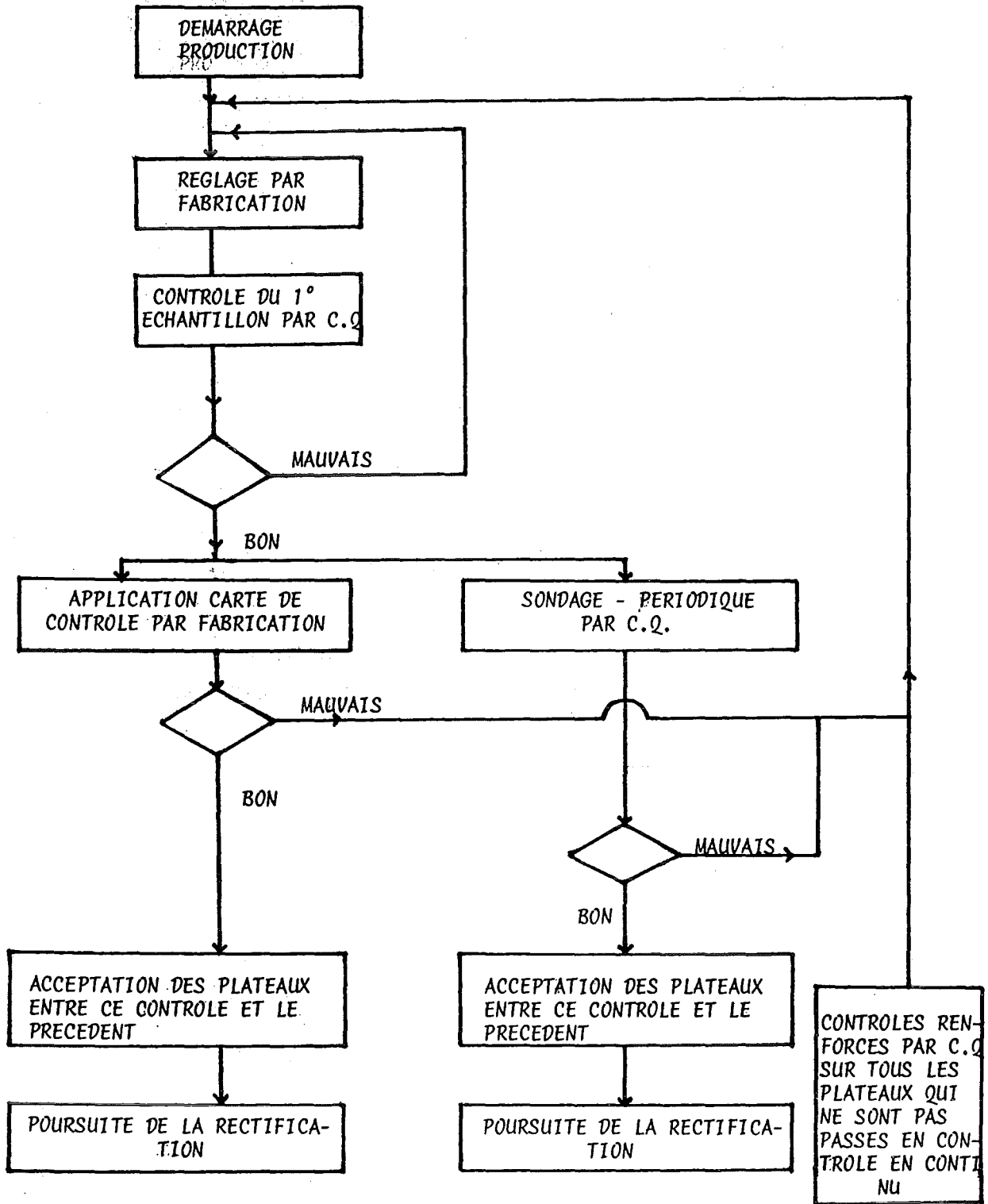
5ème Partie : Résultats

Il y a plusieurs manières d'apprécier les résultats de l'implantation du contrô-
le en continu dans une ligne de fabrication, selon que l'on considère des effets
immédiats ou l'amélioration à plus long terme de la productivité de l'atelier,
sachant en outre que la fabrication peut être de bonne qualité même si on prati-
que le contrôle séquentiel. Il est bon, par conséquent, d'avoir toujours à l'es-
prit que l'objectif fondamental du contrôle en continu est la mise sous contrôle
des équipements de fabrication ; lorsque celle-ci sera atteinte, non seulement
les rendements seront améliorés, mais les coûts de contrôle seront réduits au
strict nécessaire par suppression des tris, recontrôles et retours et par l'al-
lègement de la surveillance (passage du contrôle normal au contrôle réduit par
exemple).

Dans certains cas, des améliorations spectaculaires peuvent cependant apparaitre
lorsqu'il est possible de mettre rapidement sous contrôle un équipement défail-
lant ou plus généralement lorsqu'une corrélation forte est établie entre l'appa-
rition d'un défaut et un réglage particulier d'une machine. A titre d'exemple,
on peut citer le diamètre des pastilles d'oxyde d'uranium rectifiées pour lequel
depuis près de quatre ans le taux de refus par la surveillance du Contrôle Qua-
lité et par celle des inspecteurs des clients est quasiment nul.

Mais, en général, c'est à moyen terme que le bénéfice du contrôle en continu, vé-
ritable investissement technique et humain, sera obtenu. Outil de choix pour pi-
loter la qualité et régulariser les flux de production, le contrôle en continu
conduit les responsables et les opérateurs des ateliers de fabrication à adopter
une attitude dynamique vis-à-vis des problèmes de qualité. On note également une
importante amélioration des relations entre fabrication et Contrôle Qualité,
animés par la même motivation et en état de dialogue permanent.

6ème Partie : Perspectives d'évolution

FBFC et FRAGEMA, devant l'intérêt des résultats obtenus, ont décidé de poursuivre dans cette voie par l'implantation du contrôle en continu chez les principaux sous-traitants de FBFC, chez lesquels le contrôle séquentiel de réception est fort lourd et coûteux.

La connaissance précise acquise sur l'efficacité des méthodes de contrôle mises en oeuvre et la masse de données recueillies fournissent une connaissance pratiquement exhaustive de la qualité réelle des produits. Cet ensemble de données constitue une base indispensable pour le développement de nouveaux procédés de fabrication et de contrôle, en particulier les procédés automatisés. Les taux de défectueux détectés pour certaines caractéristiques du combustible se situent actuellement à des niveaux limites pour le contrôle humain.
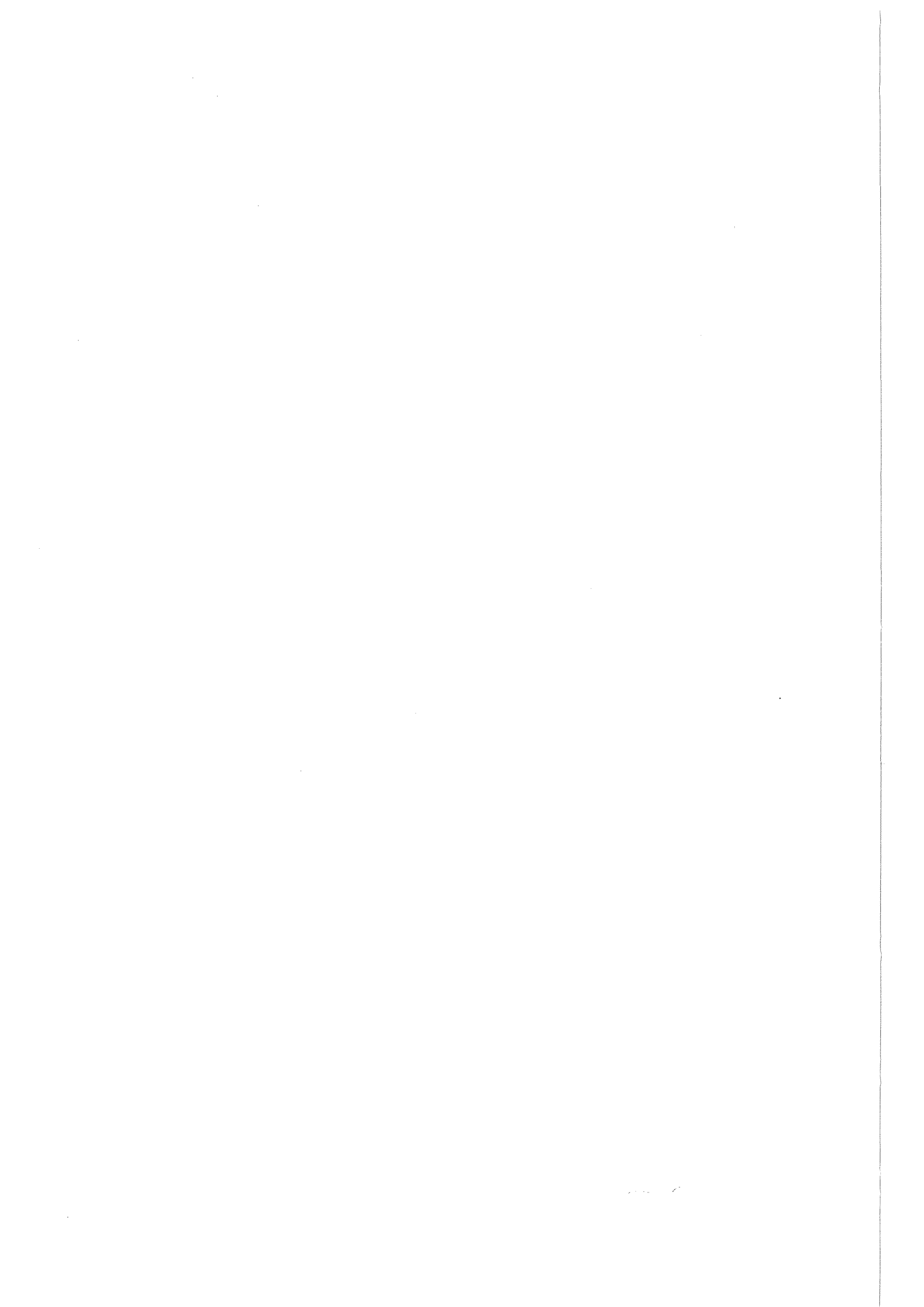
QUALITY TECHNIQUES IMPLEMENTED DURING THE FABRICATION PHASE OF ADDITIVATES AND BARRIER FUEL PELLETS AND BUNDLES

C. Coppola - P. Valle

Ansaldo Impianti S.P.A.
Via Gabriele D'Annunzio 113
16121 Genova

Abstract:

New methods were developed for QC activities related to the project mainly UT test and metallographic test during the characterization of ZR tubes with and without barrier.
A characterization process has been developed for additivated fuel pellets, fuel rods and segmented fuel rods, and for the whole bundles before loading into reactor.

INTRODUCTION: PROJECT DESCRIPTION

New fuel fabrication quality techniques were developed in

an experimental fuel production project.

The purpose of the project was to produce additivated and

barrier fuel and bundles to be loaded in the first reload

of Caorso Nuclear Power Plant.

The special assemblies were four ALTA (Additive Lead Test

Assembly) with additivated fuel and one SRB (Segmented

Rod Bundle) with additivitated fuel, special geometry

pellets and some Zirconium barrier rods and segments.

1.   STANDARD QUALITY TECHNIQUES FOR

        FUEL ASSEMBLIES FABRICATION - A SURVEY

One of the products that are to be industrially produced with the best achievable quality in spite of cost and schedule problems is nuclear fuel. This requires that quality is not a separate item to be considered but is unavoidable part of fuel fabrication. Quality assurance organization, procedures and techniques can't be a second-thought decision upon existing plant and machinery, but has to be originally planned in fuel fabrication plant construction.

The implementation of quality techniques is therefore a consequence of the plant layout.

Following the sequence of operation from powder reception to bundle on-site shipping the main quality-related aspects which require appropriate techniques are shown in exhibit 1.

In addition to these on-line techniques some data collection reduction and analysis activities are to be performed in a controlled way. In particular a characterization process takes place in order to collect selected data of characteristics of the bundle in order to keep a record of the manufacturing process results. These data may later be correlated to the characteristics of spent fuel. For data collection from production various standard

control charts are used.

Data reduction techniques are then used for some kinds of data: for example neutron scanning data are processed by a software package which identifies selected isotopes concentration.

Statistical analisis of critical data is then performed in various production phases. For single batches it may indicate that the related process is out of limit of a fixed degree of confidence even through the individual data are not out of limits.

2.   QUALIFICATION   AND   PRE-CHARACTERIZATION
        ISSUES

The characteristics of the special project ALTA-SRB which differ from the standard project are of three types.

The first is the introduction of special geometry pellets, "dished" and part cored pellets were inserted in SRB bundle. The impacted standard processes are pressing and sintering as the standard density and densification parameters should be maintained.

Fabrication of this kind of fuel is even more challenging as pre-characterization is required in order to get the correct shrinkage from the sintering process starting from a different geometry pressing head.

The second difference from a standard project is the Zr barrier tubes. These tubes are purchased by the manufacturer with appropriate requirements, but metallographic tests are required to control the grain size and the surface properties. UT test are in addition required for defect detectation both longitudinal and circonferential. The characterization of the tubes requires different process parameters for standard and Zr barrier tubes.

The third difference is the segmented rod configuration. In this rod the single segments shown in exhibit 2 are assembled toghether up to a full standard rod lenght. As you can see at both segment ends there is an absorber pellet. The pellets may be standard or special geometry ones, but maximum gap requirements are the same. Each segment is identified with a number metal stamped on the first plug. After inspection the retainer is inserted and the second plug is welded. Requirements are set also for helium filling of the free volume and for helium leak test. The rod assembling process is also controlled and plug parallelism and rod linearity must be within specified parameters. Machining applied to the end plug threads in addition to graphite coating is possible up to specified limits. The torque applied for assembling is also specified. The whole rod is then identified by appropriate markup. The segments may be then interchanged during plant refuelings.

3.   QC DATA STATISTICAL ANALYSIS, METHODS AND TECHNIQUES

The analysis of various statistical data leaded to modification to fabrication and control processes. The certified data of performance sintering test of the UO_2 powders were used for qualification of the ceramic process parameters.

In some cases has been necessary to scrap the production of some pressing heads as the statistical density achieved was not in accordance with specified limits. In other cases ceramographic tests did show that the granulometry of the additive was different from previous sinterization results. This could be traced to a different water concentration. A pre-pressing phase was introduced and qualified. The new sintering process qualification was conducted on a reduced basis but an acceptable degree of confidence was obtained by statistical analysis. It is interesting to note that the pre-pressing and granulation phase did contribute to improve the pellet surface roughness characteristics which was the most critical item for previous production.

With the highest enrichment and additive concentration batches, the roughness was within the previously established limits.

Another test which was influenced by the presence of the additive was the pellet densification test (a test of thermal simulation). Other experimental research were required. It is necessary to say that new specification limit were calculated for additivated fuel; lacking however proven data or adequate reference work the densification test should be considered only a reference test. In fact the test conducted with standard time and temperature parameters did show abnormal behaviours such as an irregular additive blistering on the pellet external surface, expecially the lateral one. The shrinking measured in this way was zero or negative and often it was impossible to get measures after the test. An engineering analisis of the densification test results determined that after the evaporation of the additive a deposition process took place upon cooling of the pellets surfaces. The time and temperature parameters were then changed with acceptable results.

A particular attention was required for "dished" and "part cored" pellets density measurements.

The dimensional aspect with very strict tolerances imposes the search of methods to execute accurate and reliable controls. For part cored pellets destructive controls were needed: ceramographies were taken and the results were evaluated on a statistical basis. Of particular interest is the evaluation of the volume of the cored parts. For its determination a water immersion test has

been used with accuracy of the order of 10%. The statistical evaluation has permitted to release the batch--by-batch production characteristics.

4. QC TECHNIQUES DEVELOPMENT-METALLIC COMPONENTS

In this area the new products developed are the end segment plugs. Four types of plugs were machined by a semiautomatic lathe from a Zircaloy-2 bar. Two of the types were the segment-to-segment plugs with male and female threads. The other two types were "extensions" of the segmented rod of appropriate lenght designed to clean out plugs from spacer positions. The upper plug machining has been particularly difficult due to the lenght of the bar (about 35 cm.).

The UT control after segment cutting has been repeated, but the limits have been restricted from an acceptability of a signal of 80% of a test defect signal, to 50%. Reject occurred for signal exceeding this limit.

Six rods were selected for operation with a special cladding. This cladding contains in the inner part a thin layer of Zr sponge co-extruded with Zircaloy-2 during the operation to thin-out the "shell".

Current welding techniques tube-plug were unsuitable for gap production in the Zr sponge.

The whole welding procedure had to be revised (different speed and current) and even the electrode position had to be changed. The operation was qualified with only the requirement of an accurate cleaning with usual methods not affecting the welded materials.

The welding operation qualification had to make reference also to a geometric layout in order to achieve the requirement of tube-plug parallelism.

The assembling of the SRB bundle required also new techniques development. In this bundle the coupling plates-rods should be better than that of a standard assembly. In fact the dismantling and re-assembling capability requires an enhanced dimensional evaluation of the 64 holes of the lattice. A study has been made to minimize the number of couplings with tolerance "not much lower" than the specified accepted tolerances. The feasibility in such condition of one or more dismantling-reassembling cycles in the fuel pool of a nuclear plant is still to be proven, but a compromise had to be done to allow standard plates to be used.

5.   COMPARISON WITH THE STATE-OF-THE-ART

As we have already pointed out the organization of a nuclear fuel manufacturing plant is a very rigid one. While virtually every manufacting and control process has to be considered a special process, a slight modification to the expected results often requires not only a new process qualification, but also hardware modification. The experience gained with the ALTA-SRB project, however, demonstrates that relevant changes to current production and control method are possible. Conditions for this result are the leadership of a fuel design organization with respect to manufacturing problem solving and the capability of the project quality organization to develop new methods and techniques in real time according to the manufacturing schedule. Software and statistical control techniques development is one of the instruments needed for overcoming hardware constraints.

These techniques together with process modifications are not substantially different from standard techniques and processes, but they are not proven and moreover there is no workmanship experience both for software and hardware applications.

The key for succesful production is then the quality

awareness which can both be affected by budget and schedule problems. Unlike hardware flexibility capabilities, quality oriented workmanship is very important in a fuel fabrication plant and a new project as the ALTA-SRB one, which requires it in a structural way, has good chances of being succesful.

EXHIBIT 1

QUALITY TECHNIQUES FOR FUEL ASSEMBLY FABRICATION

- Powder water content analysis for safe geometry
  storage
- Enrichment on-line verification
- Pressing heads qualification
- Pressing process qualification
- Density measurements
- Chemical analysis
- Sintering process qualification
- Density measurements
- Densification measurement
- Blistering analysis
- Ceramographic analysis
- Grinding process qualification
- Plug welding processes qualification
- UT test on rods
- Administrative control on rod loading
- Neutron scanning analysis
- Dimensional control
- Characterization tests (rod & bundles)
- Assembling process qualification
- Handling and storage procedures
- Shipping procedures & test
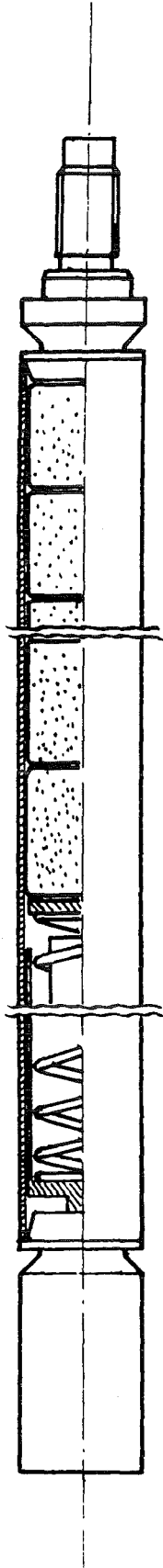
EXHIBIT 2

ADDIVATED FUEL SEGMENT
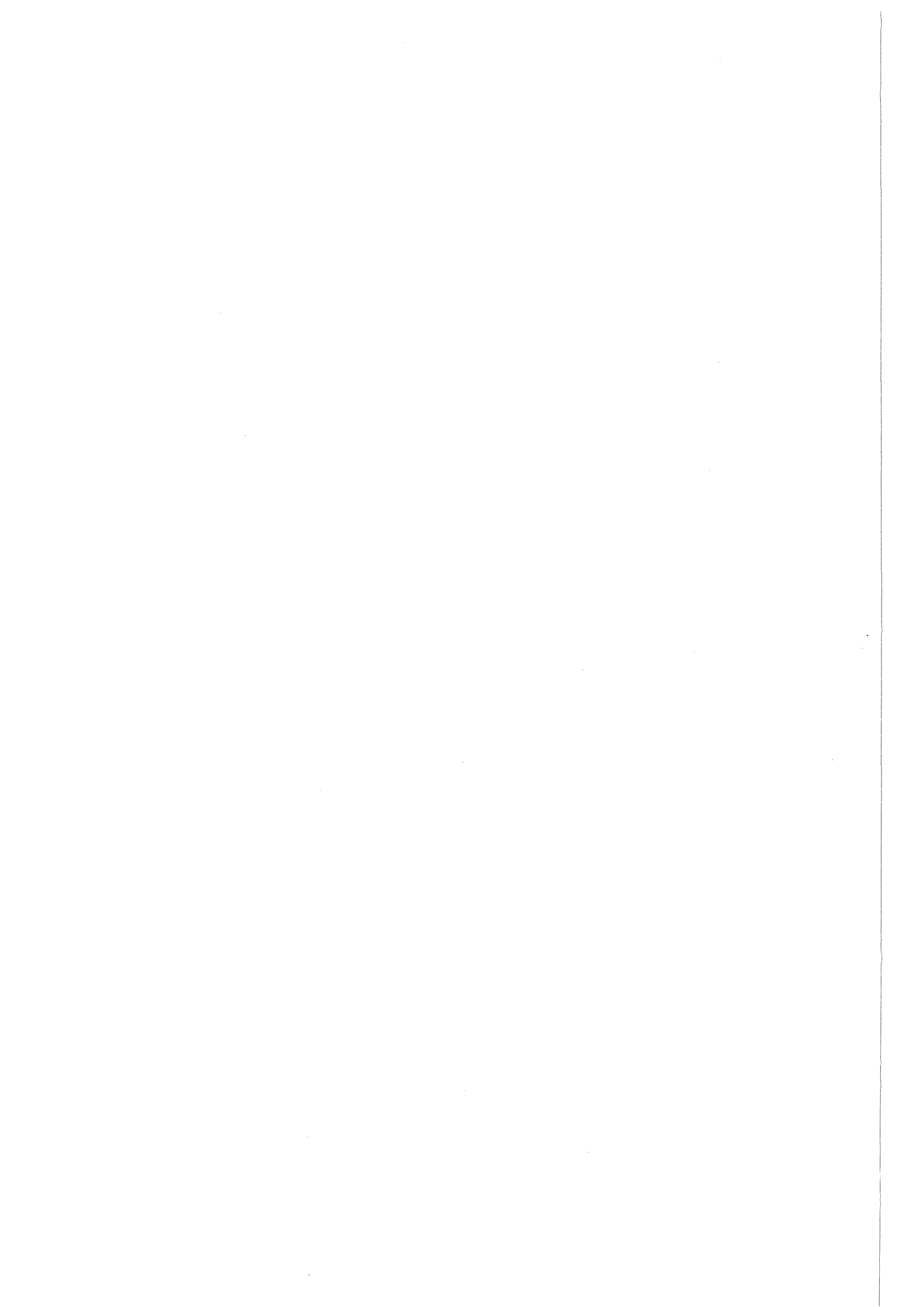
SURVEILLANCE PAR EDF DES FABRICATIONS D'ASSEMBLAGES PWR

G.P, Hochard

EDF/Service Contrôle des Fabrications

BP 143 71104 Chalon/Saone

N.M. Vial

EDF/Service des Combustibles 23 bis avenue

de Messine 75384 Paris

M.M.Galimberti

EDF/SEPTEN Tour EDF-GDF 92080 Paris La Defense

Abstract:

This paper describes the EDF's practice for quality assurance requirements, in nuclear fuel manufacturing, applicable to suppliers of water reactor fuel assemblies.

It covers methods and programmes used by "ELECTRICITE DE FRANCE" (E.D.F) to verify how the quality will be obtained in the procurement, design and manufacture of fuel assemblies by contractors and their sub-contractors.

In the course of the year 1984 this programme will be scheduled with informatic assistance, by the "EDF-MAGESTIC system", increasing efficiency for corrective actions.

SOMMAIRE

1 - INTRODUCTION

La présente communication fait état des dipositions prises par "ELECTRICITE DE FRANCE" (E.D.F.), relatives à la surveillance des fabrications d'assemblages de combustible des centrales électronucléaires de la filière à eau légère, du type réacteur à eau sous pression (R.E.P.).

Elle traite notamment des méthodes mises en oeuvre depuis l'origine et des évolutions en cours, notamment de la mise en place d'un système de gestion informatisé (MAGESTIC) dans le courant de l'année 1984.

Depuis la livraison en 1976 des 157 assemblages de la centrale de "FESSEN-HEIM 1" environ 8000 assemblages de combustible ont été fabriqués.

Cette fourniture comprend la livraison des assemblages des premiers coeurs, réalisés au titre des contrats passés au constructeur de la chaudière nucléaire, et les recharges de combustible actuellement renouvelées sur la base de campagnes annuelles portant sur des tiers de coeur (52/64 assemblages).

Tenant compte d'un parc nucléaire de 52 tranches en service (54000 MWe) à l'horizon 1990, il convient d'envisager la fourniture globale (1ers coeurs et recharges) d'environ 2000 à 2500 assemblages par an (fig. 1), sur la base du programme actuellement engagé.

Les prévisions au-delà de 1990 devraient être révisées en légère baisse, ceci résultant :

- des études en cours portant sur le développement d'assemblages à taux de combustion massique élevée (45000 MWJ/T) autorisant la mise en oeuvre de campagnes allongées à 18 mois.

- de l'effet de lissage du programme nucléaire.
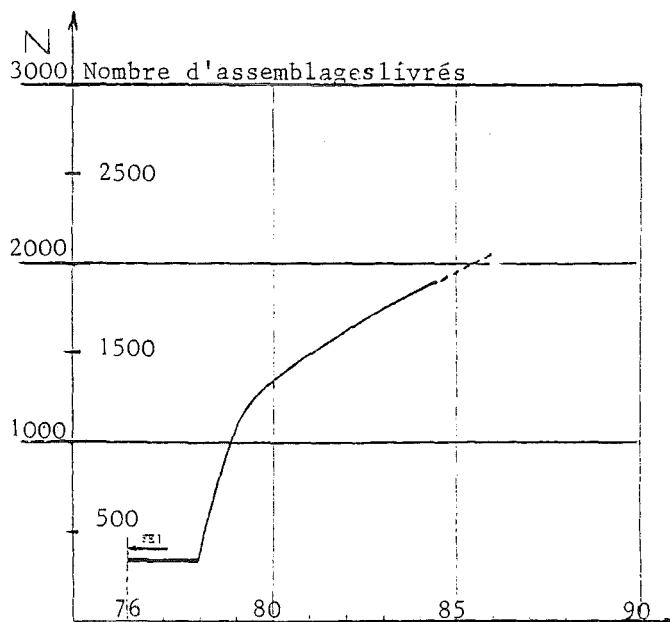
FIG.1 : PARC ELECTRONUCLÉAIRE EDF
EVOLUTION DES BESOINS ANNUELS EN
ASSEMBLAGES DE COMBUSTIBLE

Les assemblages du type 17x17 se présentent (fig. 2) sous la forme d'un squelette réalisé à partir de 2 pièces d'extrêmités en acier inoxydable austénitique reliés par des tubes de structure en alliage de zirconium auxquels sont fixées des grilles d'espacement.

Les crayons de combustible - au nombre de 264 par assemblage - forment un réseau de pas carré. Ils sont constitués par des tubes en alliage de zirconium, dans lesquels sont empilées des pastilles de bioxyde d'uranium. Les tubes sont obturés à leur extrêmité par des bouchons soudés.

Les tubes de structure servent de tubes-guides aux crayons absorbants des grappes de règlage, et permettent aussi l'insertion des crayons de poisons, crayons sources et grappes de bouchons.

La pièce d'extrêmité supérieure autorise la manutention verticale de l'assemblage, elle comporte un système de ressorts permettant le maintien de l'assemblage dans le coeur du réacteur.

Les assemblages des paliers 900 MWe et 1300 MWe ne diffèrent géométriquement que par leur longueur (12 ft/14 ft) et le nombre de grilles d'espacement.

Les assemblages courants actuellement chargés ne sont pas démontables et sont équipés de grilles d'espacement en alliage de nickel.

Les développements en cours portent sur l'emploi d'assemblages démontables, équipés de grilles d'espacement en alliage de zirconium.

Par ailleurs, l'emploi de crayons comportant des pastilles mixtes de bioxyde d'uranium et d'oxyde de gadolinium est en cours de développement dans le cadre de la politique d'allongement des campagnes : les premiers assemblages comportant des crayons gadoliniés ont été livrés dans le courant de l'année 1983.

2 - PRINCIPES GENERAUX RELATIFS A LA SURVEILLANCE DE LA FABRICATION

Les méthodes de surveillance mises en application par E.D.F. sont basées sur l'application des principes portés dans le code de bonne pratique de l'AIEA (n° 50 CQA) et les guides de sûreté, dont le n° 50-SG-QA11 publié en décembre 1983.

La surveillance exercée fait intervenir différents services d'E.D.F. : les services gestionnaires des contrats responsables vis-à-vis des constructeurs des actions contractuelles, le Service Etudes et Projets Thermiques et Nucléaires (SEPTEN) chargé des études de conception, le Service Contrôle des Fabrications (SCF) plus spécialement chargé de la surveillance en usine.

Celle-ci est appliquée à trois niveaux : celui des constructeurs, des fabricants d'assemblages et de grappes et de leurs sous-traitants. Elle se concrétise par :

- la vérification des systèmes d'Assurance de la Qualité des intervenants,

- un contrôle des études,

- un contrôle des approvisionnements et de la fabrication des assemblages de combustible,

- une inspection des assemblages à la réception sur le site.

Par ailleurs, des prélèvements de produits et composants effectués sur les chaines de fabrication à l'initiative d'E.D.F., sont transmis pour analyse à des laboratoires différents de ceux des constructeurs.

Les exigences correspondantes font l'objet de prescriptions portées dans les contrats, dont le Cahier des Clauses Administratives Générales (CCAG) et des cahiers de spécifications techniques.

Ces documents font appel à des règles techniques, par exemple au recueil RCCC (Règles de Conception et de Construction applicables aux assemblages de combustible des centrales nucléaires PWR) publié en juillet 1981 sous l'égide de l'AFCEN. (*).

Les règles correspondantes élaborées en 1980 viennent progressivement en remplacement de celles portées dans les Cahiers de Prescriptions de Fabrication et Contrôle (CPFC), utilisées antérieurement.

Une édition du RCCC programmée dans le courant de l'année 1984 définira les exigences d'E.D.F. relatives à la conception, à la fabrication et au contrôle, applicables à tout fournisseur de combustible pour les centrales à eau légère comportant un réacteur à eau préssurisée.

3 - CONTROLE DES ETUDES

Le constructeur est chargé d'effectuer des études de conception et de comportement des assemblages.

Néanmoins, conformément au guide de sûreté de l'AIEA n° 50-SG-QA.6, E.D.F. assume la responsabilité de l'efficacité générale du programme d'assurance de la qualité pour la conception du produit : à cette fin, E.D.F. s'est doté de moyens d'étude sans pour autant se substituer au concepteur.

Les vérifications effectuées portent notamment sur l'examen des dossiers de conception et des dossiers de compatibilité. Elles s'étendent aussi à l'examen du dossier technique de l'assemblage en vue de vérifier la cohérence avec les règles de conception.

Le dossier technique comporte les plans et spécifications techniques définissant les exigences détaillées requises pour la fabrication des assemblages, il fait l'objet d'un examen en concertation par les différents services d'E.D.F. concernés.

Une approche identique est effectuée pour l'instruction des anomalies de fabrication rencontrées (§.6).

(*) AFCEN : association française pour les règles de conception et de construction des matériels des chaudières électronucléaires, constituée le 19 octobre 1980 par E.D.F., FRAMATOME, NOVATOME.

Afin de satisfaire à ces différents objectifs, E.D.F. utilise les moyens d'études qui suivent :

- des codes généraux de calculs mécaniques, hydrauliques et neutroniques et radioprotection applicables aux configurations de calcul souhaitées
  par exemple : des codes de calcul aux éléments finis (INCA, TEDEL...).

- des codes spécifiques à la modélisation du comportement du crayon, conçus par E.D.F. (Ex. : CYRANO).

Par ailleurs, des calculs théoriques de vérification du dimensionnement des composants sont effectués.

Suivant les principes d'assurance de la qualité, l'action d'E.D.F. est conduite, soit en vérifiant la cohérence d'ensemble des documents présentés par les constructeurs, soit en réalisant des contre-études sur des éléments, sous-ensembles, et l'assemblage de combustible.

Au vu des conclusions tirées, il est demandé aux constructeurs des justifications sur les hypothèses retenues ou des compléments d'étude, éventuellement des essais supplémentaires, afin d'étayer le dossier de conception pour des hypothèses ou des résultats de calcul controversées.

4 - CONTROLE DES APPROVISIONNEMENTS ET DE LA FABRICATION

Le Service Contrôle des Fabrications d'E.D.F. est chargé de la surveillance de la construction en usine des assemblages de combustible.

Il effectue à ce titre :

- une vérification des spécifications techniques proposées par les constructeurs, leurs fabricants et sous-traitants.

- des enquêtes pour la vérification de l'organisation de la qualité mise en place par les constructeurs, les fabricants et leurs sous-traitants.

- des enquêtes procédés relatives à la mise en oeuvre des procédés de fabrication et contrôle dans les ateliers.

  Ces enquêtes font l'objet d'une programmation et d'un suivi des actions correctives à mettre en oeuvre à tous les niveaux de la chaine de fabrication.

- des inspections techniques dans les usines de fabrication d'assemblages et chez les fournisseurs de produits.

- des prélèvements sur les chaines de fabrication.

Il s'assure par ailleurs du traitement satisfaisant des non-conformités relevées et de la traçabilité des procès-verbaux portés dans les rapports de fin de fabrication.

4-1 - Contrôle des approvisionnements

Les vérifications effectuées par E.D.F. portent sur :

- la qualification des produits utilisés pour la fabrication des
  assemblages combustible et le respect des paramètres importants
  permettant d'en garantir la qualité (fig. 3).

- la participation par sondage aux opérations de recette des produits
  et composants, et à l'instruction des non-conformités éventuellement
  relevées.

- la réalisation de prélèvements (cf. § 4-3).

Les interventions sont effectuées dans certains cas au niveau des
matériaux de base et dans la plupart des cas au niveau de l'élaboration
des demi-produits et produits (cf. fig. 4).

Les postes relatifs à la conversion de l'UF6 en bioxyde d'uranium de même
que la fabrication des lingots et trex en alliage de zirconium font
l'objet d'enquêtes spécifiques, la ligne de conduite actuellement retenue
étant de s'assurer de la qualification des procédés de fabrication
correspondants.

4-2 - Contrôle des fabrications

Les interventions effectuées par E.D.F. concernent la fabrication de
différents composants mis en oeuvre (pastilles, grilles, pièces d'extrê-
mités, squelettes...), la réalisation du crayonnage et la constitution
des assemblages et grappes associées.

Elles viennent en complément des contrôles effectués par les services
d'assurance de la qualité des constructeurs, et des services de contrôle
de la qualité des fabricants et fournisseurs.

La recette finale des assemblages et des grappes fait l'objet de la levée
d'un point d'arrêt par E.D.F., faisant suite à une vérification de la
conformité et de la traçabilité de la documentation fournie par le
constructeur pour constituer les rapports de fin de fabrication, qui sont
mis à disposition de l'exploitant.

Depuis 1983, la fabrication des assemblages de combustibles fait l'objet
d'une surveillance programmée.

Une transposition dans le cadre d'une application du plan informatique
E.D.F. (système MAGESTIC) est actuellement en cours.

Le système MAGESTIC est conçu sur la base d'un ordinateur (CII-HB-DPS
7/65) installé au siège du service à EVRY et relié à des consoles
implantées sur les lieux de travail des agents par l'intermédiaire des
réseaux TRANSPAC, RETINA et EURONET pour l'étranger.

Les premières applications ont été développées à la fin de l'année 1983.

Celles-ci sont basées (cf. fig.5), pour ce qui concerne la surveillance programmée sur l'emploi de quatre types de documents :

a) des plans types de surveillance qui définissent pour un matériel donné (ex. : assemblage de combustible, cuve...) les actions de surveillance à exercer par des interventions sous forme d'enquêtes procédés et des inspections par sondage ou individuelle.

b) des procédures qui font état des vérifications à effectuer au cours d'une intervention programmée.
Il s'agit de l'équivalent d'une check-list portant sur le contrôle de la documentation technique et des opérations de fabrication associées.

c) des plans d'action, établis pour une usine ou un site donné, qui font état des fréquences d'intervention retenues pour un exercice calendaire, et des procédures à utiliser.

d) des compte-rendus d'action informatisés, autorisant un suivi des actions correctives demandées.

Le diagramme porté sur la figure 6 fait état de l'articulation de ces pièces et des relations entre le poste de travail implanté sur les lieux ou s'exerce l'inspection - ou à proximité - et l'ordinateur central.

FIG.5 : SCHEMA GENERAL DE MAGESTIC
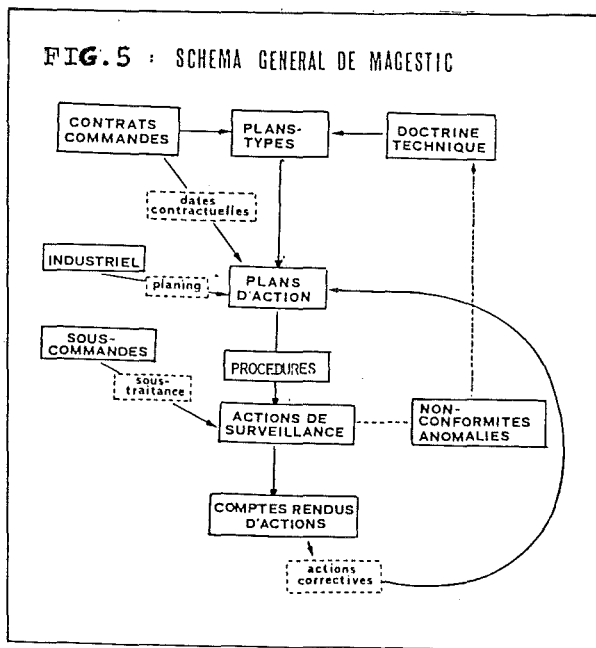
4-3 - Prélèvements de produits et composants

Ceux-ci sont effectués sur la base de plans de prélèvements annuels établis en fonction de la charge de l'usine et du caractère "prototype" de certaines fabrications.

La réalisation d'expertises complémentaires par des laboratoires différents de ceux des constructeurs, a pour but d'améliorer la connaissance des produits et de mettre en évidence éventuellement des phénomènes particuliers non décelés par les contrôles de fabrication.

Une partie des échantillons prélevée fait l'objet d'essais, le solde des prélèvements est archivé dans un laboratoire E.D.F..

Les enseignements tirés font l'objet d'une analyse immédiate si nécessaire, et d'un examen annuel.

5 - INSPECTION DES ASSEMBLAGES A LA RECEPTION SUR LE SITE

Celle-ci porte sur l'examen des assemblages à la sortie des conteneurs, elle est assurée par des agents du site de production nucléaire qui ont fait l'objet d'une formation appropriée.

6 - TRAITEMENT DES NON-CONFORMITES

Conformément aux exigences du Code de bonne pratique de l'AIEA (§ 2-2), toutes les activités qui influent sur la qualité doivent être exercées en conformité avec des procédures écrites, des instructions ou des plans appropriés, comportant des critères d'acceptation quantitatifs ou qualitatifs.

Les écarts relevés par rapport à ces documents font l'objet d'une instruction pour le règlement des non-conformités conformément au § 10-2 du code AIEA n° 50 CQA.

Les modalités pratiques en fonction des systèmes de gestion développés par les constructeurs sont précisées dans les contrats.

La position prise par E.D.F. est la suivante :

En règle générale, tout matériel non conforme, pour lequel une remise en conformité n'a pas été prévue dans les documents techniques, est rebuté.

Pour des cas exceptionnels, E.D.F. accepte toutefois d'examiner les propositions de constructeurs relatives à :

- la mise en conformité par d'autres moyens que ceux prévus dans les documents techniques.

- l'acceptation en l'état du matériel (avec ou sans action corrective).

Préalablement à toute intervention sur le produit, le constructeur doit transmettre un dossier justificatif indiquant la nature et les modalités du traitement préconisé.

Les justifications présentées à cet égard font l'objet d'une fiche d'anomalie soumise à l'avis d'E.D.F.

Par ailleurs, dès qu'apparaît un incident qui pourrait nuire à la qualité du produit, le fournisseur est tenu d'en informer E.D.F. dans les délais les plus courts.

Conformément aux exigences portées dans les règles fondamentales de sûreté (RFS V.2-e) de l'administration française; les fiches d'anomalies et rapports d'incidents pouvant mettre en cause la sûreté font l'objet d'une déclaration.

7 - UTILISATION DU RETOUR D'EXPERIENCE

L'analyse du comportement en réacteur du combustible est effectué, tant par les constructeurs que par E.D.F., qui établit des statistiques sur l'origine des ruptures de gaines.

Les moyens de ressuage utilisés sur les sites permettent de déceler les assemblages défectueux et l'engagement de recherches sur les causes probables de la déficience avec un retour sur les conditions de fabrication de ceux-ci.

D'autres moyens sont utilisés dont ceux de laboratoires chauds appartenant ou non à E.D.F., et une cellule d'examen qui est en cours de montage au laboratoire E.D.F. de CHINON.

8 - CONCLUSION

Les moyens mis en oeuvre par E.D.F., dans le cadre de la surveillance de la fabrication des assemblages de combustible pour réacteurs à eau préssurisée, viennent en complément de ceux développés par les constructeurs et leurs fournisseurs.

E.D.F. ne participe pas directement aux opérations de contrôle et recette des produits, composants et assemblages, mais s'assure par un programme de surveillance approprié que le constructeur et ses sous-traitants ont pris toutes dispositions utiles pour l'obtention de la qualité requise.

Les vérifications effectuées à ce titre, en application des principes définis par l'AIEA comportent essentiellement des enquêtes sur l'organisation de la qualité des constructeurs et de leurs sous-traitants aux différents niveaux de la chaine de fabrication, des enquêtes techniques sur les procédés de fabrication et contrôle mis en oeuvre, des inspections techniques programmées sur les lieux de fabrication.

L'emploi d'un système informatique de gestion (MAGESTIC) à partir de 1984 contribue à améliorer l'efficacité de cette surveillance, notamment pour ce qui concerne le suivi des actions correctives demandées.

0 0

0

ASSEMBLAGE COMBUSTIBLE / PWR

MODELE 17 X 17

Caractéristiques communes

Dimensions : 214 x 214 mm

Nb crayons   :   264

Nb tubes guides   :   24

Nb de grilles   :   8 ou 10

PASTILLES   :   UO2

Caractéristiques particulières

PALIER	900	1300
Nb assemblage/coeur	157	193
Nb assemblage par recharge	52	64
Hauteur	3,85 m	4,49 m
Poids total de l'assemblage (Kg)	665	745

MATERIAUX

Gaines et tubes guides : ZIRCALOY 4

Grilles et ressorts   : Inconel 718 *

Manchons   :   Acier austnénitique

Embouts   :   Acier austénitique

* Emploi envisagé de Zircaloy 4
  pour les grilles

Araignée
SPIDER-Assembly

Crayon de contrôle
Absorber Rod

Ressort de maintien
Top nozzle spring set

Embout supérieur
Top nozzle assembly

Grille supérieure
Top grid

Tube guide
Thimble tube

Grille de mélange
Vanned grid
Crayon combustible
Fuel rod

FUEL ASSEMBLY

Grille inférieure
Bottom grid

Embout inférieur
Bottom nozzle

FIG. 2

QUALIFICATION DE PRODUITS

SPECIFICATION

"N" PARAMETRES

DOSSIER

TECHNIQUE

PROGRAMME TECHNIQUE
DE FABRICATION
"p" VARIABLES
DEFINIES ET TOLERANCEES

VERIFICATION PORTANT
SUR L'OBJECTIF : CONTROLE
DE LA CONFORMITE
DU PRODUIT

PROGRAMME D'ESSAI

QUALIFICATION DU
PRODUIT (*)

FABRICATION DU
PRODUIT

RAPPORT DE QUALIFICA-
TION DU PRODUIT

VERIFICATION PORTANT
SUR LES MOYENS UTILISES
POUR ATTEINDRE L'OBJEC-
TIF : CONTROLE DES
GAMMES ET PROCEDURES
ASSOCIEES

DOMAINE DE VALIDITE
DE LA QUALIFICATION

"q" VARIABLES
ESSENTIELLES TOLERANCEES
$x \pm k$ AVEC $q \leqslant p$

(*) PIÈCE TYPE CU PRÉSÉRIE.

FIG : 3

FIG. 4

FABRICATION D'ASSEMBLAGES DE COMBUSTIBLE POUR LES CENTRALES A EAU LEGERE DE L'E.D.F.

DOMAINE D'INTERVENTION EN MATIERE DE SURVEILLANCE DES FABRICATIONS

☐ Points d'intervention   ⌐ ⌐ Surveillance des procédés de fabrication

MISE EN OEUVRE	TYPES DE PRODUITS			
	Composés d'uranium	Alliage de zirconium	Aciers inoxydables et alliages de nickel	Produits divers
MATERIAUX DE BASE	Minerai uranium	EPONGES / LINGOTS	FERRO-ALLIAGES / LINGOTS / FERRAILLES	B4C – AIC Pyrex ...
FOURNISSEURS DE DEMI-PRODUITS	UF6 / Enrichissement	EBAUCHES TREX / FEUILLARDS / BARRES	PRODUITS PLATS ET PRODUITS LONGS – FILS / PIECES MOULEES	
FOURNISSEURS DE PRODUITS ET COMPOSANTS		TUBES DE GAINAGE / TUBES GUIDES / BOUCHONS	GRILLES / RESSORTS / EMBOUTS / TUBES INOX	PASTILLES / BARRES
FABRICANTS D'ASSEMBLAGES ET DE GRAPPES ASSOCIES	CONVERSION DE L'UF6 EN POUDRE UO2 / PASTILLES UO2	CRAYONS / ASSEMBLAGES	SQUELETTES	GRAPPE DE REGLAGE ET CONTROLE

PLAN D'ACTION

PLAN TYPE DE SURVEILLANCE

PLAN TYPE

ORDINATEUR   CENTRAL

CII - HB - DPS 7/65

CONSOLE   IMPLANTEE
SUR LES LIEUX DE
TRAVAIL OU A PROXIMITE

COMPTE-
RENDU
D'ACTION

ENQUETE
OU
INSPECTION

PROCEDURES

SYSTEME   DE   GESTION

MAGESTIC

APPLICATION "SURVEILLANCE
PROGRAMMEE"

SUIVI DES ACTIONS CORRECTIVES

FIG.6

TECHNICAL AND ADMINISTRATIVE IMPLEMENTATION OF QA AND QC DURING

FABRICATION OF ADDITIVATED AND BARRIER FUEL PELLETS

C. Coppola - P. Valle

Ansaldo Impianti S.P.A.

Via Gabriele D'Annunzio 113

16121 Genova

Abstract:

The importance of the project requires the responsibility to
implement QA and QC activities in order to achieve the maximum
data and avoid any possible mix-up. The great number of fuel
composition, the small size of fuel batches, the necessity to
utilize "Qualification during fabrication" procedures, were the
main aspects to take in account. Major conditions adverse to
quality or manufacturing losses, were identified, and corrective
action taken. The experimental characteristics of certain pro-
cess to undetermined status of the quality and special kinds of
control were requested in order to determine if the process
parameters could be safely applicated.

1.   PROJECT DESCRIPTION

As a part of a PCI remedy research two specific projects were developed by A.I. Company for fabrication of special fuel bundles to be inserted in the first reload of Caorso Nuclear reactor.

The special assemblies were four A.L.T.A. (that is Additive Lead Test Assembly) with additivated fuel to be irradiated in a central zone of the core for four reactor cycles and one S.R.B. (that is Segmented Rod Bundle) carrying as PCI remedies additive fuel, special geometry pellets and some Zirconium barrier rods. The SRB is irradiated in a peripheral zone of the core and contains 29 segmented rods (4 segments each) interchangeable in the following reloads.
The most exposed segments will be available for further studies in experimental reactors mainly with power transients and post-burnup analysis.

The main results expected, in addition to demonstrate the good behaviour of the fuel, were to get licensing of the prototypes in a commercial plant, to develop new fabbrication processes of commercial value upon satisfactory completion of the experimental test, to develop a Fuel Surveillance Program which allows a complete analysis of fuel performance in various operating phases.

This kind of results are strictly correlated with a proof and a traceability of the industrial processes involved in fuel fabrication.

The fuel has consequently been fabricated under current industrial conditions without any laboratory shortcut even for very limited batches.

2.   PROJECT AND QA ORGANIZATION

The design organization has a fundamental role in fixing the manufacturing constraints through conservative calculations, which may or may not be maintained during the manufacturing process.

The interface control between the designers and with the fabrication organization (in a different company) is therefore one of the critical conditions not only to resolve nonconformities, but mainly to minimize frequency of occurrence and impact of manufacturing losses and rate of rejection re-analysing, if necessary, the design conditions.

An interdipartimental approch has been chosen for the project design organization. In exhibit 1 the organization chart is shown.As you can see the design organization is strictly tied to Caorso Reload 1 Project, but it is not in a

subordinate status.

This implies that the plant reload scheduling has to be maintained, but that problems which affects the experimental bundles do have an impact on the whole reload project.

You also can see that the quality organization is not shown for the reload project as the quality for proven bundles is currently assured through the company QA section while new procedures and techniques are needed both for the design and the manufacturing phase of the experimental project.

In exhibit 2 is shown the licencing interfaces between the various departments. The responsibilities involved are the ones shown previously. On the right the project department PRO Section PR2 Unit U11 on the left the engineering departments and various section/units.

3.   QA AND QC DURING FABRICATION

The main problems which a QA organization faces when new procedures/techniques are needed in working QA systems, is to comply to the maximum possible extent to the existing technical conditions such as qualification procedures, production by batches and by fuel compositions, standard QC controls. Additional controls (non

destructive and destructive) and preliminary process qualifications have to be procedured in a controlled way. On-line process qualification is then needed to determine through destructive controls undetermined quality status. Manufacturing losses have then to be taken into account.

The first issue in pre-production qualification is to verify the process feedback to the inclusion of additive. A depleted uranium powder with the same characteristics of enriched powder has been used with various additive concentrations. After blending the local concentration of additive within the powder has been measured and a homogeneous distribution has been reached. After pressing the green density is the main process parameter to be controlled. The acceptable range of green density has not been significatively modified with reference to a standard lot. Then the sintering parameters were consistently modified in order to achieve standard density and densification of the pellets.

The qualification of the sintering process was the most critical item in the shop. Result from analisys had to be reproducible for all the batches and all the process involved.

Statistical evaluations with a 3σ confidence limit had to be carried out both for qualification and production runs. Such engineering activities had to be performed by the manufacturer organization and result had to be discussed and approved during inter company meetings.

Empirical adjustments to the statistical data were made to take into account the strict confidence limits and the small size of the populations.

The sintering process, in addition, could not be entirely reproduced in on-line fabbrication due to lack of sufficient feed material in the press silos, condition determining an irregular feeding of the press cavities.

The grinding process qualification has evidenced the unability of the machinery to produce pellets within the acceptable surface finish limits. This has been demonstrated to be a consequence of the additive surface migration during the sintering process.

Comprehensive analysis were then carried out in engineering departments but the designers could not accept worse roughness standards.

A study of inside roughness coupling tube-pellet evidenced the possibility to consider acceptable roughness as an additive characteristic of both tube and pellets. Roughness limits for the tubes were then made more strict in order to make pellet roughness acceptable. Random-selected tubes from each lot were then cut in longitudinal direction and after inspected for roughness characteristics with very sofisticated instrumentation. The statistical interval with 3 σ confidence limits was then evaluated for each lot and the combined desired characteristics of rods and pellets were fixed as a manufacturing limit.

Due to the high number of fuel compositions a very strin-
gent adiministrative control was introduced and the tra-
ceability of the rod composition was demonstrated by
neutron scanning analysis.

This kind of control was then used for loading of the
segmented rods in SRB bundle. In this case neutron
scanning analysis wasn't possible due to the inclusion of
special neutron poison pellets.

4. NONCONFORMITIES AND QA GROUP ACTIVITIES

In the production phase the project QA group had to face
the quality aspects of several technical problems which
inexpectedly arose from the shop practice. Their main
concern was not to provide technical solutions on a
problem-oriented basis but to include in the activity
planning the new issues, assuring that the "problem
solving" activities were carried out in a traceable way
from the engineering changes to quality-related documents
to the procedures and instructions issued for the shop
activities.

The remaining non-conformities which were to be treated
in the standard way with MRB (Material Review Board)
were consequently very limited (only 10 cases).
In exhibit 3 are shown non conformities and MRB dispo-

sitions. An "analysis" of such has determined that the non conformities are well within the operating experience limits of the fuel fabrication plant both as total number of occurents and as statistical disposition records (no scraps).

Interesting is the third MRB corresponding to Inspection Report N. 413. The value of density 97.23% obtained for a lot of pellets is usually accepted as standard project with the condition to classify such pellets as "type B" subject to use only in a particular zone of the rod. The ALTA-SRB project did however require that only superior grade material should be used (that is only "type A" pellets). In any case it has to be noted that only one production lot has been affected by the described deviation.

Unfortunately however this impressive quality record does not imply that the rate of rejection was low or zero, but shows that the scraps came from the experimental characteristics of the project and that the workmanship was adequate or good thus indicating that once determined the optimum process parameters, the state-of-the-art of fuel fabrication techniques is ready to develop an industrial production of additivated fuel with the required characteristics.

A further step in quality demonstration for future production is the ability to determine if the process parameters could be safely applied to a current production. On-line extensive QC has determined that the qualification

did not cover the production of pellets with different enrichment and additive concentration. Qualification during fabrication procedures were then used and destructive test were performed such as vacuum impregnation test for density measurement and isotopical analysis with enrichment analyzer.

Ceramographic tests were extensively performed to determine acceptable standards for additive concetration distribution within the pellet.

From such tests the manufacturer determined a step by step analisys of machining capability to reproduce the qualification results. The critical operation was determined to be pressing and some pressing heads were re-qualified for this production.

The QA aspects of the process required a prompt customer QA/QC examination of the quality level during fabrication in order to avoid major conditions adverse to quality or manufacturing losses over the available powder quantity. A resident inspector was present during shop activities at the manufacturer's site in order to control both the acceptability of QC test and data and the manufacturer QA/QC organization performance.

An extensive audit/design review was performed then by an audit team on the manufacturer. The lead auditor was chosen by the company QA organization outside the project QA group to assure indipendency of judgement, but engineering and QA representatives of the project group

were included in the team.

The audit plan was aimed to the control of certification, data collection, and the verification of the process engineering through the analysis of QC data.

The traceability of these data was verified and the reproducibility of the processes assured. The limits of validation were traced out.

As a temporary conclusion of the research, waiting for post-burnup data, a considerable fabrication and control know-how was achieved for additivated fuel commercial production.

EXHIBIT   1

Fig. A02.30.01 - Struttura operativa ed organizzativa

EXHIBIT 2

Programma "PCI-REMEDY"
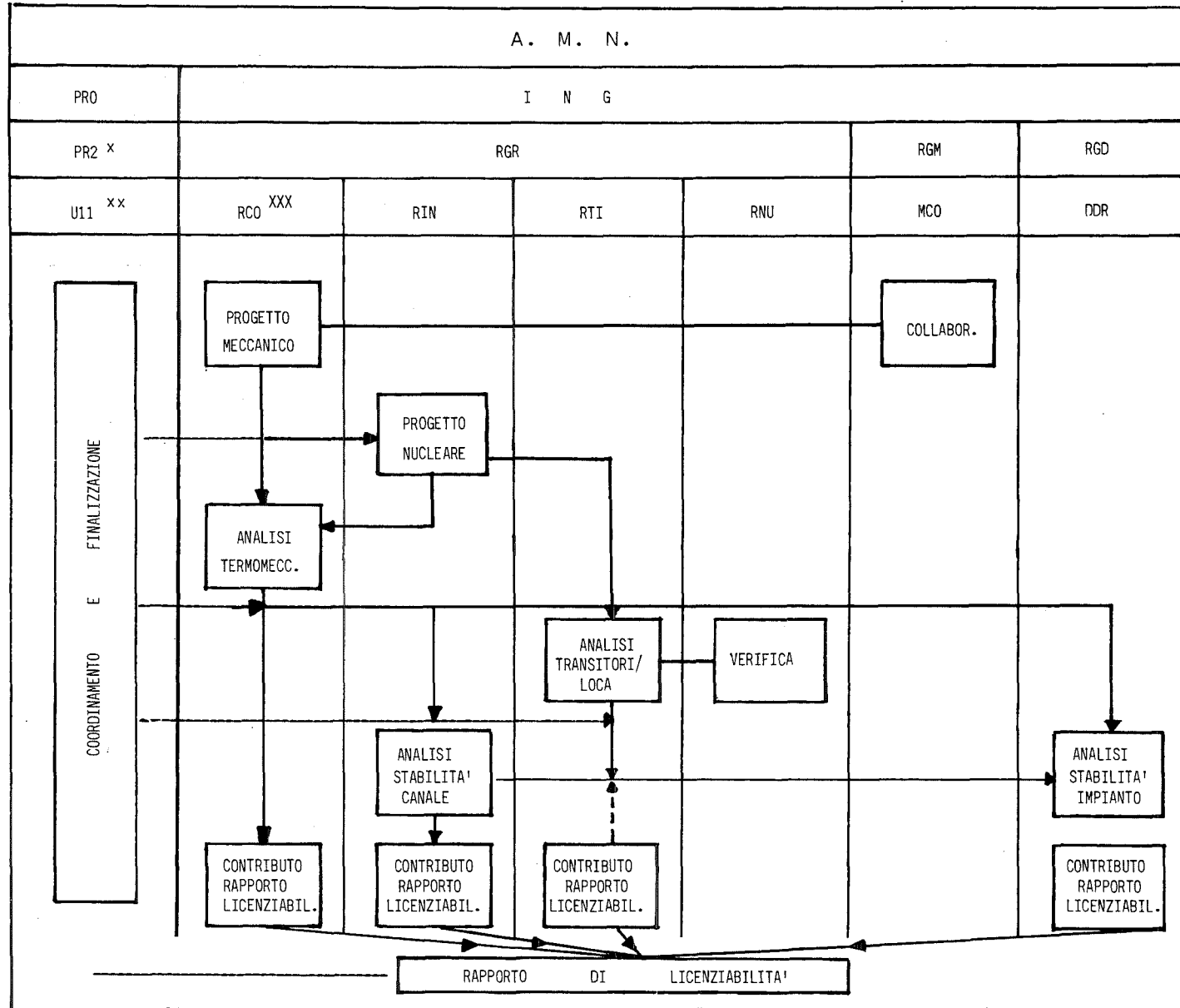
EXHIBIT 3

TABLE 10 NONCONFORMITIES – MRB DISPOSITION
——

I.R. N°	DESCRIPTION	DISPOSITION
–	Plug dimension out of limits (1st case)	Use as recommended
–	Plug dimension out of limits (2nd case)	" "
413	Density UO$_2$ + Additive > 97%	" "
415	Densification > 0.004	Use as is
423	Fissile column lenght > 158.7 mm	" "
439	"Part Cored": pellets dimensions out of limits	" "
440	"Dished": pellets dimension out of limits	" "
429/433	Segments lenght less than minimum	Use as recommended
435	Fissile column lenght less than minimum	Use as is
441	Dimension "corner" spacer/cornes rod less than minimum	" "

DISCUSSION

H. ASSMANN: a) Which additives do you use and which is the percentage of additives?

b) What is the purpose of the additives?

C.E. COPPOLA: a) Fuel fabrication using additives is covered by TDA (Technical Development Agreement) between various companies and I am not authorized to speak about it. I am sorry - if you need additional information, please send your request directly to me.

b) The purpose of additives is to minimize the pellet cladding interaction as a possible remedy.

QA/QC FOR NUCLEAR FUEL
FABRICATION AT ASEA-ATOM, SWEDEN

Aart van Santen
AB ASEA-ATOM
FUEL DEPARTMENT
QUALITY CONTROL
Box 53, S-721 04 VÄSTERÅS, SWEDEN

ABSTRACT

This paper describes the ASEA-ATOM system for quality assurance and quality control for manufacturing of nuclear fuel. The system is basically organized in accordance with the requirements stipulated in US NRC 10 CFR 50, Appendix B. Some examples are given to illustrate how these basic requirements are applied in practice. The QA/QC organization, some specific inspection methods and the machines used are described briefly. A general trend in inspection methods is increased automation and computerization, which minimizes inspection error and improves productivity.

DISCUSSION

H. ASSMANN: a) Which additives do you use and which is the percentage of additives?
b) What is the purpose of the additives?

C.E. COPPOLA: a) Fuel fabrication using additives is covered by TDA (Technical Development Agreement) between various companies and I am not authorized to speak about it. I am sorry – if you need additional information, please send your request directly to me.
b) The purpose of additives is to minimize the pellet cladding interaction as a possible remedy.

LIST OF CONTENTS

3. INTRODUCTION

The ASEA-ATOM fuel factory has a capacity of about 400 tons UO_2 fuel per year. Both BWR and PWR fuel is manufactured. The manufacturing of fuel assemblies may be divided into four main steps:

- Conversion of UF_6 to UO_2 powder
- Manufacturing of UO_2 pellets
- Manufacturing of fuel rods
- Assembly of fuel rods and components into fuel assemblies.

The factory also manufactures fuel assembly components such as spacer grids, tie plates, etc., as well as other core components, a.o. BWR control rods and fuel channels.

To be able to compete in the light water reactor fuel market, a vendor has to comply with very stringent quality assurance and quality control requirements. This paper gives a short description of a successful QA/QC system. Fuel fabrication and the associated quality control at ASEA-ATOM are under continuous development; the paper describes the present situation, a probable future development is further automation and computerization.

4. QUALITY ASSURANCE

4.1 Codes and standards, quality assurance program

There are several international codes and standards that deal with quality assurance for nuclear power plants. One of the earliest, and perhaps the most well known, is the US NRC 10 CFR 50 App. B. There are also the ANSI/ASME N 45.2 and the ISO-6251. More recently, there is also a safety guide which IAEA (50-SG-QA 11) issued. Most of these standards are very similar and generally it can be said that if a company has a QA-system that fulfils the requirements of one of these standards, then it is highly probable that the requirements of the other standards are also met. For several of these standards the requirements are expressed in a systematic way in the well known "18 Criteria" from the US NRC 10 CFR 50 App. B.

At this point it might be appropriate to define quality assurance. Different people and organizations may phrase the definition differently, but most agree on the content, so it is sufficient to cite just one, for instance the ISO version: "All those planned and systematic actions necessary to provide adequate confidence that a structure, system or component will perform satisfactorily in service." The words planned and systematic should be stressed when talking about a QA-program.

The ASEA-ATOM quality assurance program is basically organized according to the requirements stipulated in US NRC 10 CFR 50 App. B. It defines the structure, responsibility, levels of authority and the arrangement of the organizational units and personnel involved. Written instructions, procedures and drawings are used in all activities affecting quality. It also assures these activities are performed by adequately trained and supervised personnel and documented according to written instructions. The program also provides for regular review by management to assure suitable corrective actions can be taken.

The QA-system is presented in a program description and organized according to the well known "18 Criteria". Under each heading is explained in general terms how the requirements for that particular criterion are fulfilled. Behind these general terms there is of course a multitude of detailed instructions and

procedures. Most of these are organized in a "Quality Assurance Manual for Nuclear Fuel". It would take too much space to deal with all of the requirements in detail, but a few examples may serve to illustrate how the criteria are met.

4.2 Organization

One of the more important criteria for a QA-system concerns organization. The main objects are to ensure sufficient independence for the QA and QC organizations as well as clearly defined responsibilities, duties and authority for all organizational units. Also the lines of communication must be clearly defined.

Fig. 1. Organization of ASEA-ATOM's Fuel Department

Fig. 1 shows the organization of the ASEA-ATOM Nuclear Fuel Department. Quality assurance for the whole of ASEA-ATOM is under the direct control and supervision of the technical director, who is a vice president and member of the board of directors. Within QA there are several subsections, a.o. one for QA systems and audits.

Within Production there are a Fuel Factory, a Component Factory and an independent section for Quality Control. Detailed instructions define the various interfaces and responsibilities. Authority and responsibilities are also documented in job descriptions.

The QC-section's main interfaces outside the Production Department but still within the company are with Fuel Engineering and the Quality Assurance Department, as Fig. 2 illustrates.

```
        ┌──────────┐    — QA - Manual
        │    QA    │    — Audits
        │          │    — Approval of
        └──────────┘       deviation reports
                        — Approval of
┌──────────┐               certificates
│  Fuel    │
│  eng.    │
└──────────┘
                        ┌──────────┐
— Specifications, drawings,  │ Quality  │
   inspection plan           │ control  │
— Development of new         └──────────┘
   inspection techniques
— Education and training
   of NDT-personnel
```
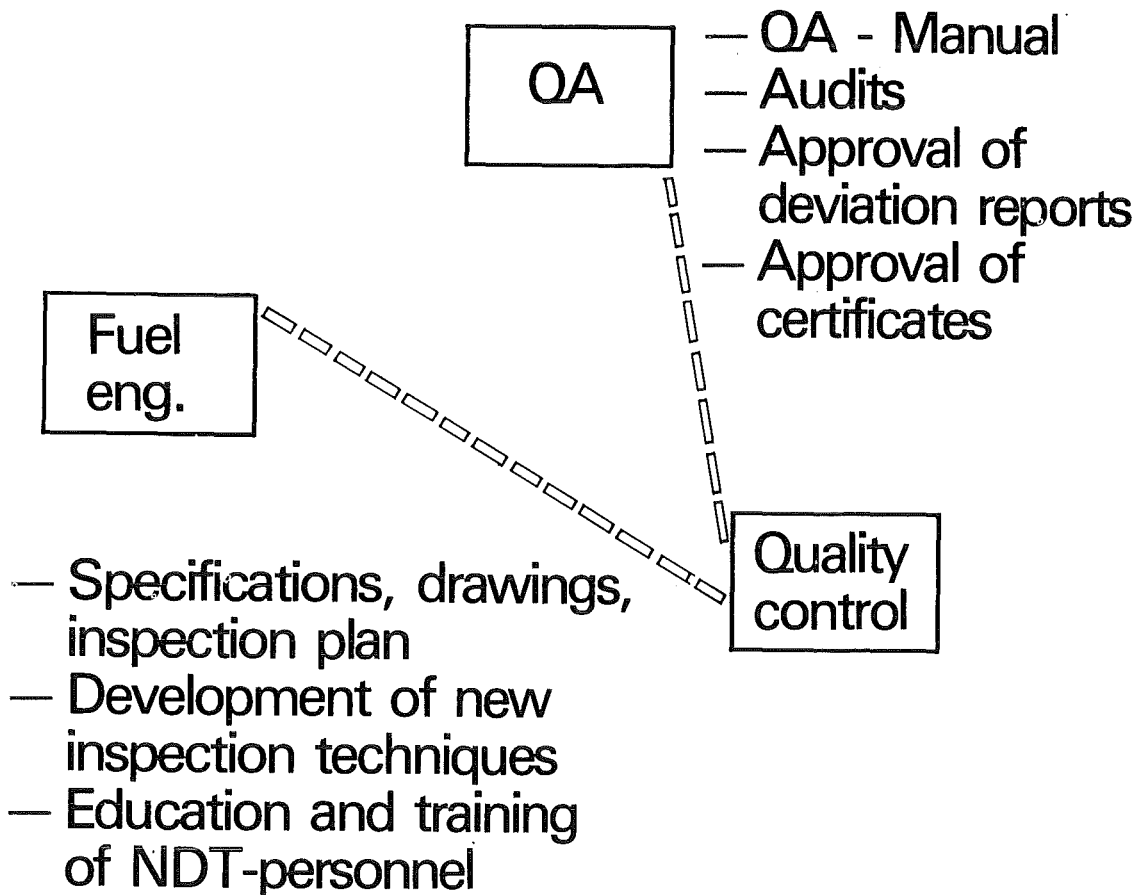
Fig. 2. QC main interfaces outside production department

Fuel Engineering not only provides the necessary drawings, specifications and inspection plans but is also responsible for development of new inspection techniques and processes as well as education and training of NDT-personnel. All level III NDT personnel are employed by Fuel Engineering.

The Quality Assurance Department issues the QA-manuals for nuclear fuel and component manufacturing and performs internal and external (supplier) audits. It is also involved in the review and approval of deviation reports and inspection certificates. Internal audits cover not only the production department, but also design and procurement.

Fig. 3 shows a more detailed division of responsibilities within the QC-section. It is divided into four subsections. Three of these are each responsible for quality control and inspection of well defined areas of the production department. Inspection is carried out and supervised by QC-personnel who have not performed the work being inspected. The fourth QC-section is responsible for records and documents, which includes filing and micro-filming of QC-records as well as quality control planning.

Fig. 3. The quality control organization
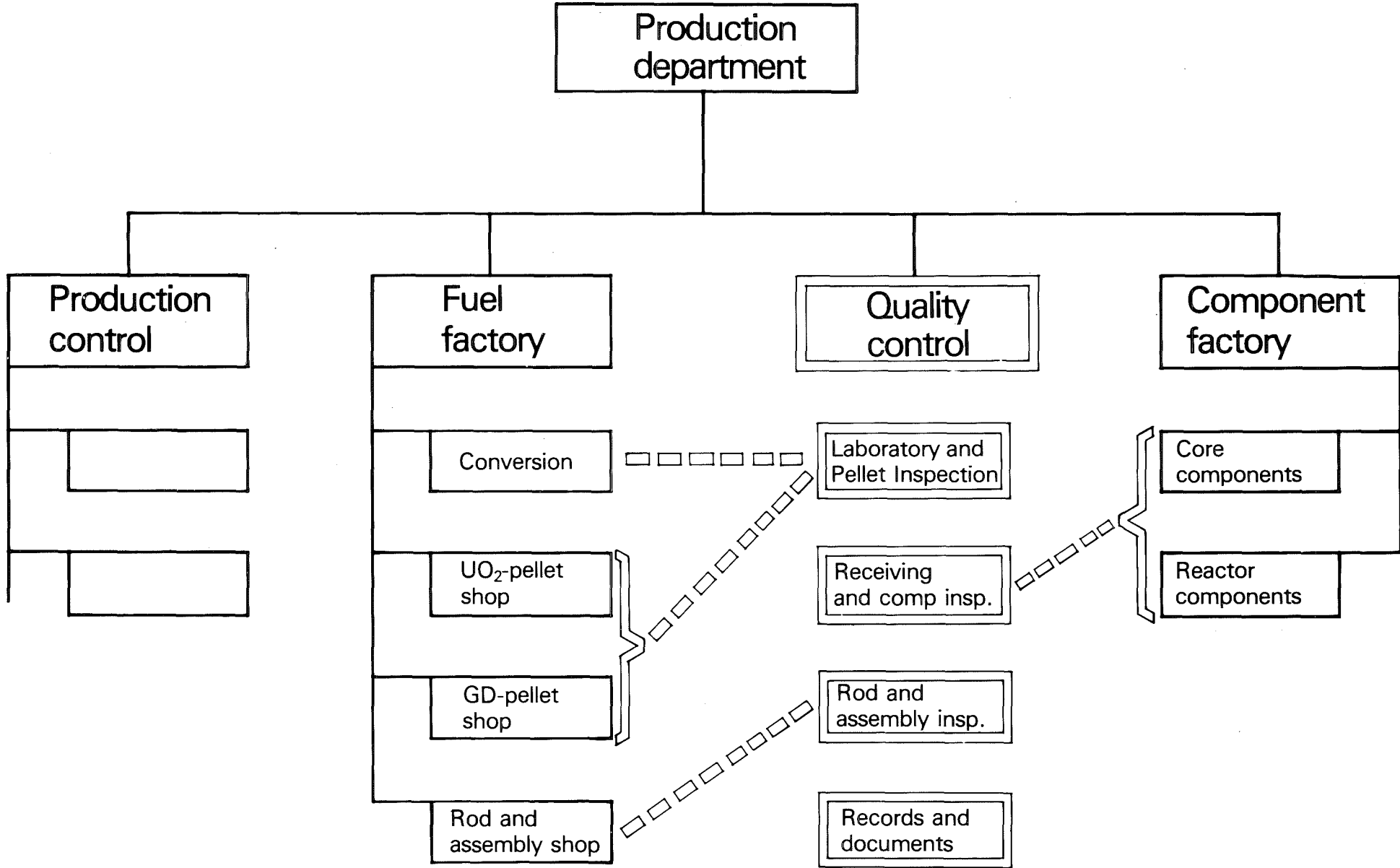
4.3 Practical applications

To make it easier to understand how the general criteria can be applied in practice, three criteria have been selected for a more detailed description to explain their implementation in the ASEA-ATOM QA-system.

- "Instructions, Procedures and Drawings"

The system requirement for this criterion states that activities affecting quality shall be prescribed in and performed according to documented instructions, procedures or drawings of a type appropriate to the circumstances. Where applicable, these documents shall include quantitive criteria such as tolerances, operating limits, comparative samples, etc.

Fig. 4 illustrates how this requirement is met. The Fuel Engineering section prepares a set of drawings, inspection plan, specifications and a total product specification, which is approved by QA. These engineering drawings and specifications contain all necessary tolerances and other quantitive requirements, and the inspection plan defines the required inspection. These documents are "translated" into a process flow outline, with operating procedures and inspection procedures, by manufacturing planning in cooperation with the QC-subsection for records and documents. The process flow outline is in principle a list that identifies every single step, manufacturing as well as inspection. For each step there is an operating procedure or an inspection procedure. Before process flow outline, operating procedures and inspection procedures are issued they are checked by the Fuel Engineering department. Operating procedures and inspection procedures describe in detail how operations and inspections, including the documentation thereof, shall be performed. They also include acceptance criteria such as operating limits, tolerances, etc. Fig. 4 also shows how inspection and process records are generated and filed so that an inspection certificate can be issued in conjunction with the delivery of the product. When filing the records, the inspection plan from engineering is used as an index. The final inspection certificate is checked and approved by the QA department.

CUSTOMER

ORDER

INSP. CERT

PRO-DUCT

SALES

FUEL ENG. | QA DEPT.

QA APPROVAL

INSP. CERT

PRO-DUCT

DRAWINGS   SPEC

INSPECTION PLAN

CHECK

MANUFACTURING PLANNING

QC RECORDS AND DOCUMENTS

QC FILE

PROCESS OUTLINE

OPERATING PROCEDURES

INSPECTION PROCEDURES

MANUFAC-TURING

PRO-DUCT

PROCESS REC

INSPEC-TION RECORDS

INSPECTION

PRO-DUCT

PRO-DUCT
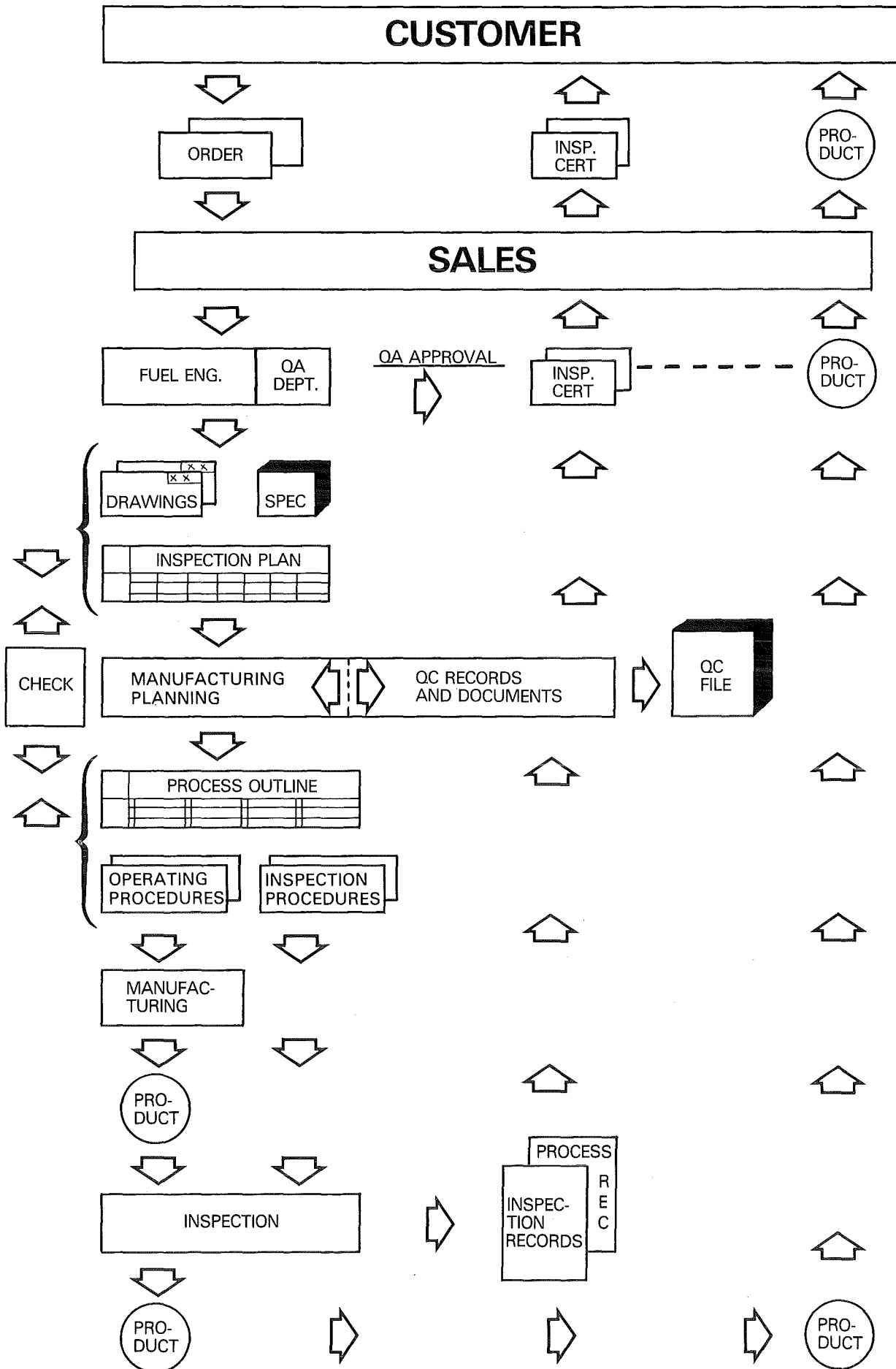
Fig. 4. "Instructions, Procedures and Drawings"

Responsible section MNGR / Document	QA	QA - Subsection	Fuel engineering	Fuel eng. subsect.	Manufact. planning	Manufacturing	QC	QC - Subsection	QC - Supervisor	⇨ Etc
Product spec.	A		I	P						
Part drawing			A I	P C		S		S		
Techn. spec.			A I	P C		S		S		
Inspection plan			A I	P C		S		S		
Procurement doc.	A	C		C	P	I				
Inspect. procedure				C			A	C I P		
Deviation report		A		C					I	P
Final insp. certificate	A	C						I	P	

⇩ Etc.

P = Prepared by
C = Checked by
A = Approved by
I = Issued by
S = Consultation

Fig. 5. "Document Control"

- "Document Control"   The basic requirement for document control is that individuals or organizational units responsible for preparation, review, approval and issue of documents, including changes thereof, are clearly identified. It also calls for a controlled system for release and distribution, and timely information on revisions. At ASEA-ATOM these responsibilities are clearly identified in a company instruction. A table gives an overview of who is responsible for what. Fig. 5 shows an example of such a table. It can be seen, for instance, that the QC-subsections are responsible for preparing, checking and issuing inspection procedures. However, these three items are not the responsibility of one and the same QC-subsection. There are four different QC-subsections, three for actual inspection, and one for records and documents. The general rule is that the subsection for records and documents does the preparing and issuing, and the procedure is checked by the QC-subsection responsible for the inspection activity in question. The procedure is also always checked by a Fuel Engineering subsection and approved by the QC-manager before it is issued for use. Take Deviation Reports as another example; these are prepared by a QC-supervisor and then issued by the QC-subsection manager. After that, they go to a Fuel Engineering subsection for a technical decision and are finally approved by QA. This leads us to the requirement concerning non-conforming materials.

- "Non-conforming Materials, Parts and Components"

This requirement calls for control of non-conforming items, basically a proper identification and marking of items as well as established procedures for review and disposition of non-conformity. There are of course very detailed company instructions for handling non-conformity, but the general principle can be seen from Fig. 6.

The left column of rectangles represents the Production department, the right column the Fuel Engineering and QA-departments. In the middle is the QC-department. The circles represent the non-conforming item and the small squares the paper flow between QC, Engineering and QA. If a non-conforming item is not an obvious reject and cannot be reworked by a qualified procedure, there still may be a possibility that the item can be repaired or used as is. In this case the item is clearly

marked and separated from approved items. A deviation report is written by QC, and Fuel Engineering makes a technical decision. This decision must be approved by QA. In the case of repair, QA must also approve the re-inspection report.
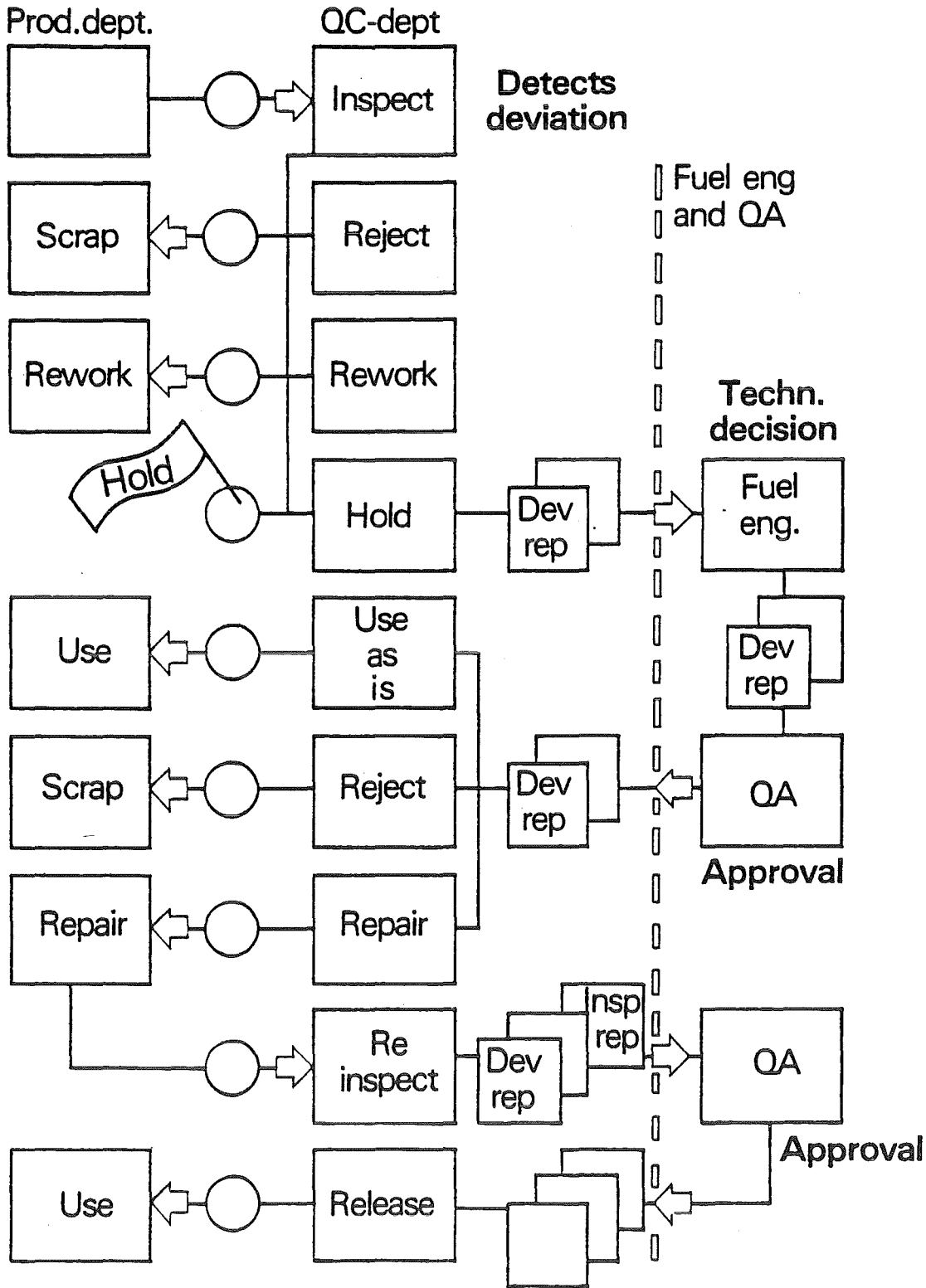
Fig. 6. Non-conforming materials, parts or components

Education

	Course A	Course B	Course C							
⊠ Has followed course ⧄ Should follow course within ⊗ months										
Inspector X		⊠	⧄6							
Inspector Y	⧄3		⊠							
Inspector Z	⊠	⊠	⊠							

Table 1

4.4 Education

Proper education and training of all personnel performing activities that affect quality is essential in all departments and in the Quality Control department in particular.

Personnel who carry out non destructive testing are educated, trained and qualified according to detailed company instructions. These are basically in accordance with the ASNT recommended practices SNT-TC-1A. Records stating the qualifications of all NDT-personnel are kept.

Not only NDT-personnel but also all other QC-inspectors are educated and trained according to company instructions. It is the responsibility of each QC-subsection manager to keep and periodically review an "education status" record of his personnel. This status is summarized in a table similar to Table 1. The status is reviewed once every quarter.

Since it is necessary to maintain a certain flexibility in each QC-subsection it is the responsibility of each QC-supervisor to keep and periodically review an "ability status" record of his personnel. This assures inspection is carried out only by properly trained personnel. The ability status is recorded in a form similar to Table 1, the main difference being that it lists "jobs" instead of "courses" and thus indicates which jobs each inspector is capable of performing or should be trained on. Most of the on the job training of inspectors is performed by the QC-supervisor.

Dimensional measuring	Conventional	All products and components
		Spacers and other components
	Coordinate measuring machine	Fuel cladding
	Automated and/or computerised	Fuel rods
		Fuel channels
		Pellet density
Destructive testing	Metallography and Mechanical testing	Welds
		Receiving inspection of materials
	Ceramography	UO₂-pellets
		GD-pellets
	Puncturing	Rod He-pressure
		Gas content
		Moisture
Chemical analysis	Conventional	UO₂ powder and pellets
		Boron carbide
		Incoming materials
		Welds contamination
	Computerized	UO₂ powder and pellets (impurities)
Nondestructive testing	X-Ray	Rod fill hole weld
		Fuel channels
	He-leak testing	Fuel rods
		Control rods
	Rod ɣ-scanning	Fuel rods
	Ultrasonic testing	Fuel cladding
		Rod circumferential weld
	Eddy current testing	GD-pellets
		Fuel channels
	Dye penetrant testing	Tie plates
		Control rod drive comp
		Other components
Administrative systems	Computerized	Fuel assembly (right rod in right place)
		Component traceability
	Manual	Records filing and certification

Table 2

5. QUALITY CONTROL METHODS

5.1 General

Many different quality control methods are used in the fuel factory. These methods may be subdivided into five main groups as shown in Table 2.

This overview of QC-methods used in the QC-section shows several computerized and automated items. Under the heading NDT, several processes are also highly automated and/or computerized even though the table does not show this. Also the coordinate measuring machine has a computer and is fully automated. This has been a general trend in inspection development at ASEA-ATOM over the last several years and is still continuing. The reasons for this automation are partly economic (improved productivity) and partly a quality goal to decrease operator (inspector) errors.

Most inspection steps are built in as part of the production flow, as can be seen from Fig. 7 which gives a general view of the process flow. This enhances fast feedback of inspection results, which promotes timely corrective actions.

The general overview of the process flow is divided up in several blocks, conversion, UO_2-pelletizing, Gd_2O_3-UO_2-pelletizing, rod shop and the assembly area. All the circles in the production flow scheme indicate some form of QC inspection.

In the following, the process flow and associated inspection points for pellets and rods are described in some detail.

Before the powder is taken into the pelletizing area, a performance test is made on samples from every powder lot. This determines the amount of U_3O_8 to blend into the powder as well as the pressing and sintering parameters. Both UO_2 and U_3O_8 powder lots are also analyzed for impurities after homogenisation and before blending in 900 kg containers. After blending, each container in the lot is analyzed for enrichment by gamma-counting on a sample, and released for pressing. During pressing, process control is performed by checking height and green density. During sintering, at least one pellet per sintering boat is checked for density. These values are treated statistically for each lot for a decision on acceptance.
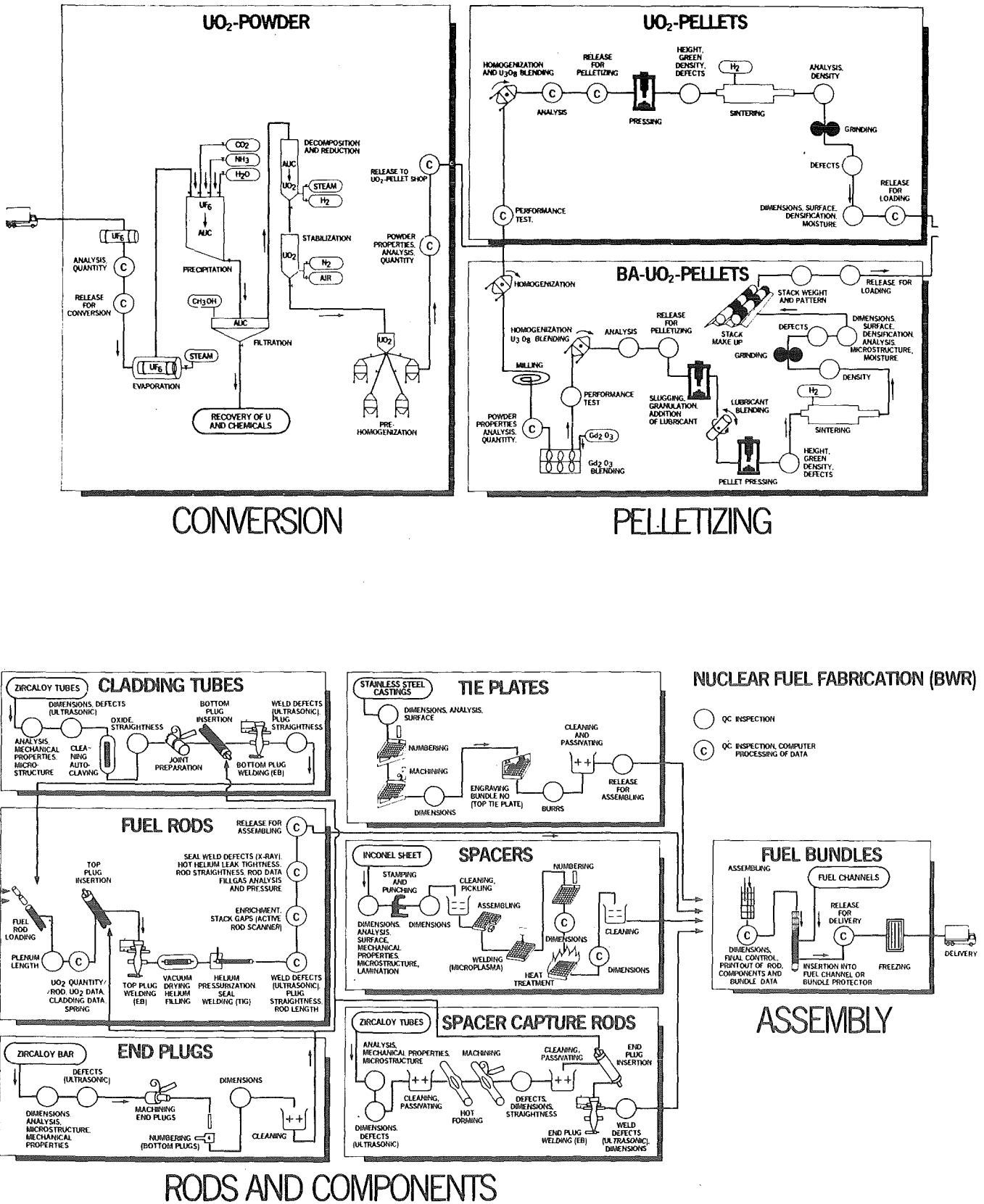
Fig. 7. The production flow

All the test pellets from the density check are kept on a special "test tray" until the complete pellet lot has been sintered. From the test tray, representative samples are taken (after grinding) for impurity analysis, moisture check, densification and enrichment. At this point the enrichment check is done by masspectrometer. After grinding, the pellets are also checked for diameter and surface defects. Before releasing for loading, all inspection results and all documentation for each lot are checked for correctness and completeness and the shop travellers on the pellet containers are QC-stamped for release for loading.

After fuel rod loading, the plenum length is checked and all relevant data, including UO_2 quantity and enrichment, cladding, plug and spring-data for each individual fuel rod number are entered into a computer file. After plug welding, vacuum drying, helium back filling, pressurization and seal welding the circumferential weld is checked by ultrasonics in a machine with preset acceptance limits. Also the rod length and plug straightness (BWR) are checked at this point. The next inspection step is an active rod gamma-scanner for checking rod enrichment, single pellet enrichment deviation, pellet stack length and gaps between pellets. This machine is fully automated, with preset acceptance limits and a sorting system for separation of accepted and rejected rods. Next, the seal weld is inspected by X-ray and all rods are helium-leak-tested at elevated temperature. The last inspection station is an automated machine for straightness checking. After a final visual examination all accepted rods from a lot are released for assembly. Rods are handled in lots of about 225 of the same type and enrichment. Rejected rods from all inspection stations are reported into the computer file so they are blocked from acceptance in the assembly.

General	— Identified and marked containers — Shop travellers — Release points — Computerized rod- and assembly data tracing

UF6-receiving	— Evaluation of independant lab certificate
	  (masspectrometer)
	— 1 δ -sample per "master" cylinder
	  (safety reason, check < 4 %)

Conversion	— 1 δ -sample from first batch after
	  enrichment change
	— 1 δ -sample for each lot (composite sample
	  from all powder containers)

Powder preparation	— 1 δ -sample from each u_3O_8-lot — 1 δ -sample from each powder container after UO_2-U_3O_8 blending
Pellets	— 1 masspectrometer analysis per lot (composite sample from 4 pellets)
Rods	— 100 % δ -scan for rod average enr. — 100 % δ -scan for single pellet deviation
Rod bundle	— Computerized rod data tracing

Table 3

5.2 <u>Enrichment Control</u>   When manufacturing nuclear fuel for BWRs, many different enrichments are processed in the shop at the same time. Enrichment control therefore becomes a primary concern. The basis for a proper enrichment control is a very strict system of identification and marking. This is combined with a controlled system of release points with enrichment measurements at each point. In the rod and assembly area, a computerized system for rod data tracing is an essential part of enrichment control.

An overview of the enrichment control system is given in Table 3.

5.3 <u>Traceability</u>   The traceability system for fuel rods and assembly components is largely computerized. The rod data tracing system starts at the pellet loading station, see Fig. 8.
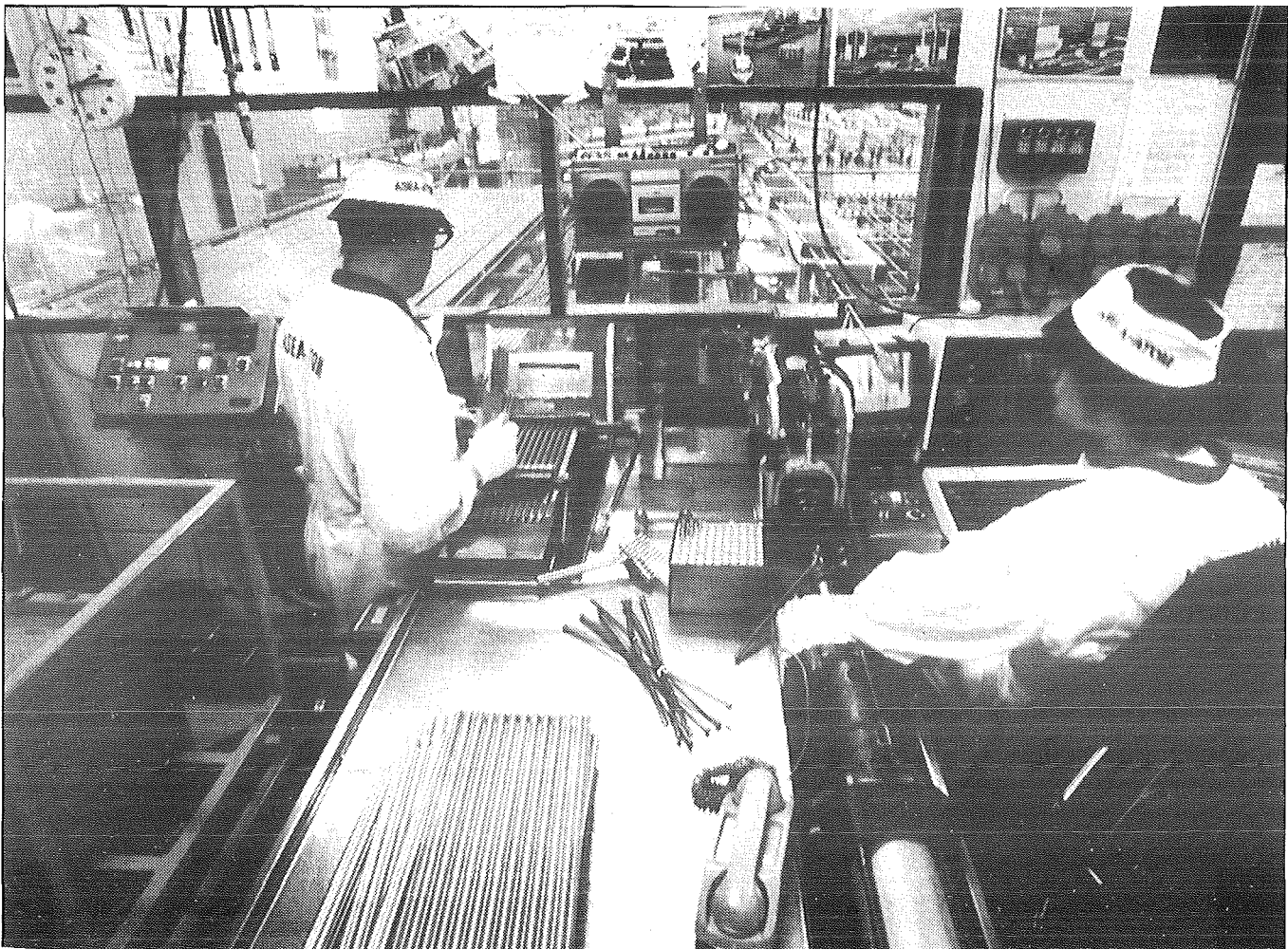
Fig. 8. The pellet loading station

At this point there is a computer terminal, and the rod data file is initiated. Each rod has an individual number on the bottom plug. By weighing first empty and then loaded tubes, the UO_2 weight is automatically entered into the rod file. Other data, such as enrichment, plug and spring lot numbers, which are the same for the rod lot, are entered into the computer terminal once for each lot. The tube material lot number is already linked to the individual number on the bottom plug at an earlier station (bottom plug welding). The rods are weighed individually but loaded ten at a time on an inclining surface.

One QC-purpose of the computerized rod data file is to make it possible to perform a computer check at the assembly area to verify that rods of different type and enrichment are assembled at their correct position in the rod assembly. Other purposes are to assure that no rods rejected at any QC-station are assembled at all, data traceability and uranium accounting.

Since a BWR-assembly may have four or five different enrichments, different rod diameters in the corners, some Gd-containing rods, four or more tie-rods and one spacer capture rod, one assembly may contain around 10 different rod types. It is of course very important that no mistakes are made in putting the right rod type in the right place. Assembling of BWR assemblies is done in the vertical position, see Fig. 9 which shows the top of the assembly. Before starting assembly, the correct numbers of each rod type for one assembly are collected in a special fixture, which is lifted to the top of the assembly station. During assembly the top guide of the assembly fixture is fitted with templates with different hole patterns for each new rod type to be assembled to assure correct positioning. At the same time, the operator will enter the rod number at the indicated position on the computer terminal screen. The assembly sequence is preprogrammed in the computer.

Fig. 9. Top view of the BWR assembling station

Using this information (rod number and assembly position) the computer checks the rod data file to verify that the rod has been loaded with pellets of the correct enrichment for this position, that it has not been reported as rejected by QC at any station or reported as assembled before and that it has been released for assembly at the end of the rod inspection line. After completed assembly and QC acceptance, a print-out of all relevant assembly data is obtained. While waiting for the computer acceptance, the top tie plate is assembled and final dimensional inspections are made.

5.4 Some examples of inspection equipment

- The active rod gamma-scanner, see Fig. 10, has two parallel channels and is fully automated and computerized. Acceptance limits are set by evaluating a large number of runs with a "test rod" containing pellets of well defined deviating enrichments. The machine not only checks for single pellet enrichment deviation but also for rod average enrichment, pellet stack length and gaps between pellets.
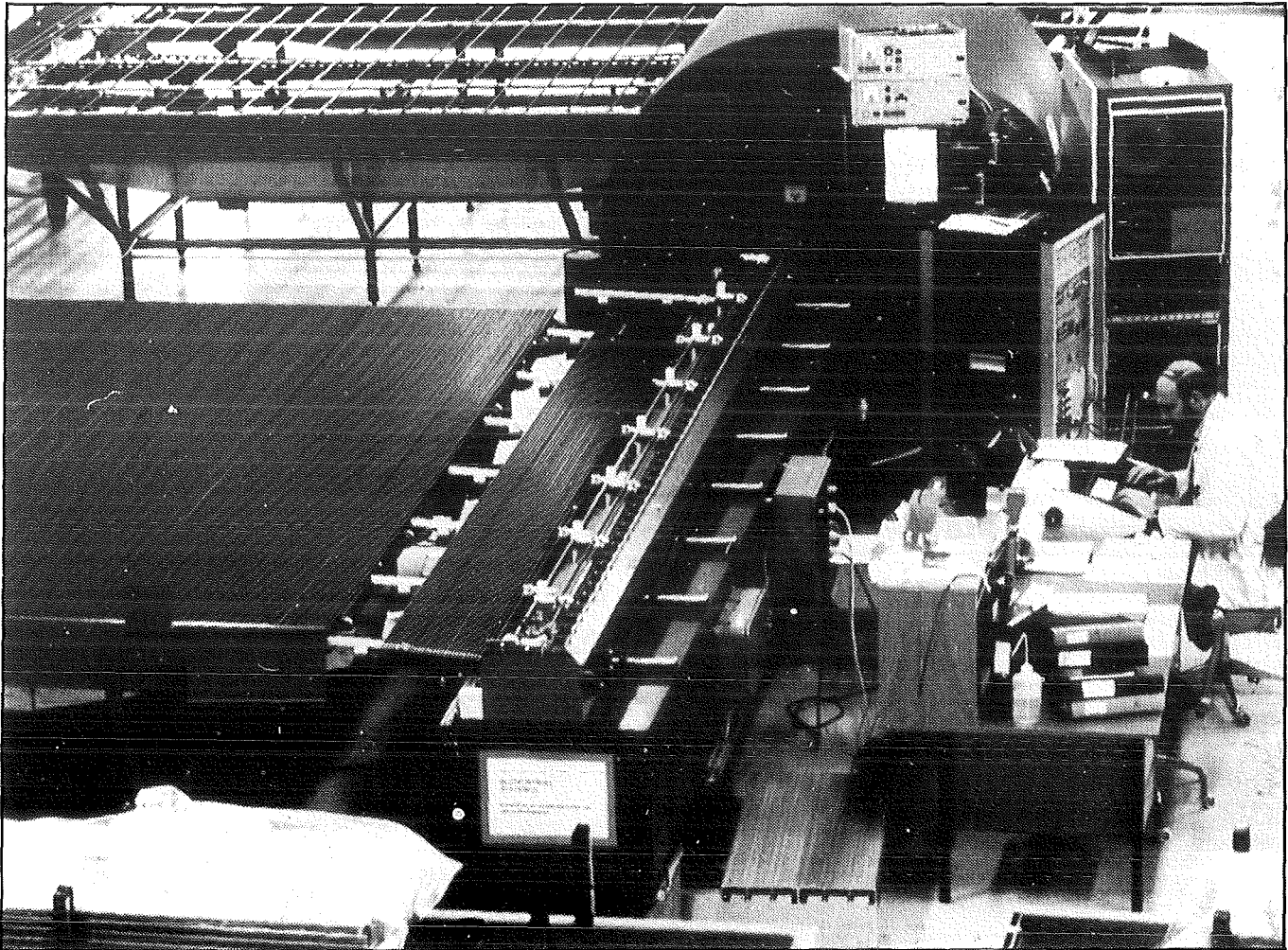
Fig. 10. The active rod gamma-scanner

Rejected rods are sorted out and separated from accepted rods automatically. The machine has a large input store and two output stores (one for accepted and one for rejected rods) so it can operate completely unattended during the night once the input store is filled at the end of the day.

- Receiving inspection of cladding tubes is another example of automated inspection equipment, see Fig. 11. The tubes are fed through an ultrasonic test chamber with seven channels. Two (opposing directions) are for longitudinal defects, two for transverse defects, two for "pits" on the inside diameter and one for dimensions. In the channel for dimensions the OD and cladding thickness are measured and the ID, ovality and excentricity calculated automatically. All dimension and defect signals are calibrated with standards and an acceptance limit for each channel is preset in the machine.
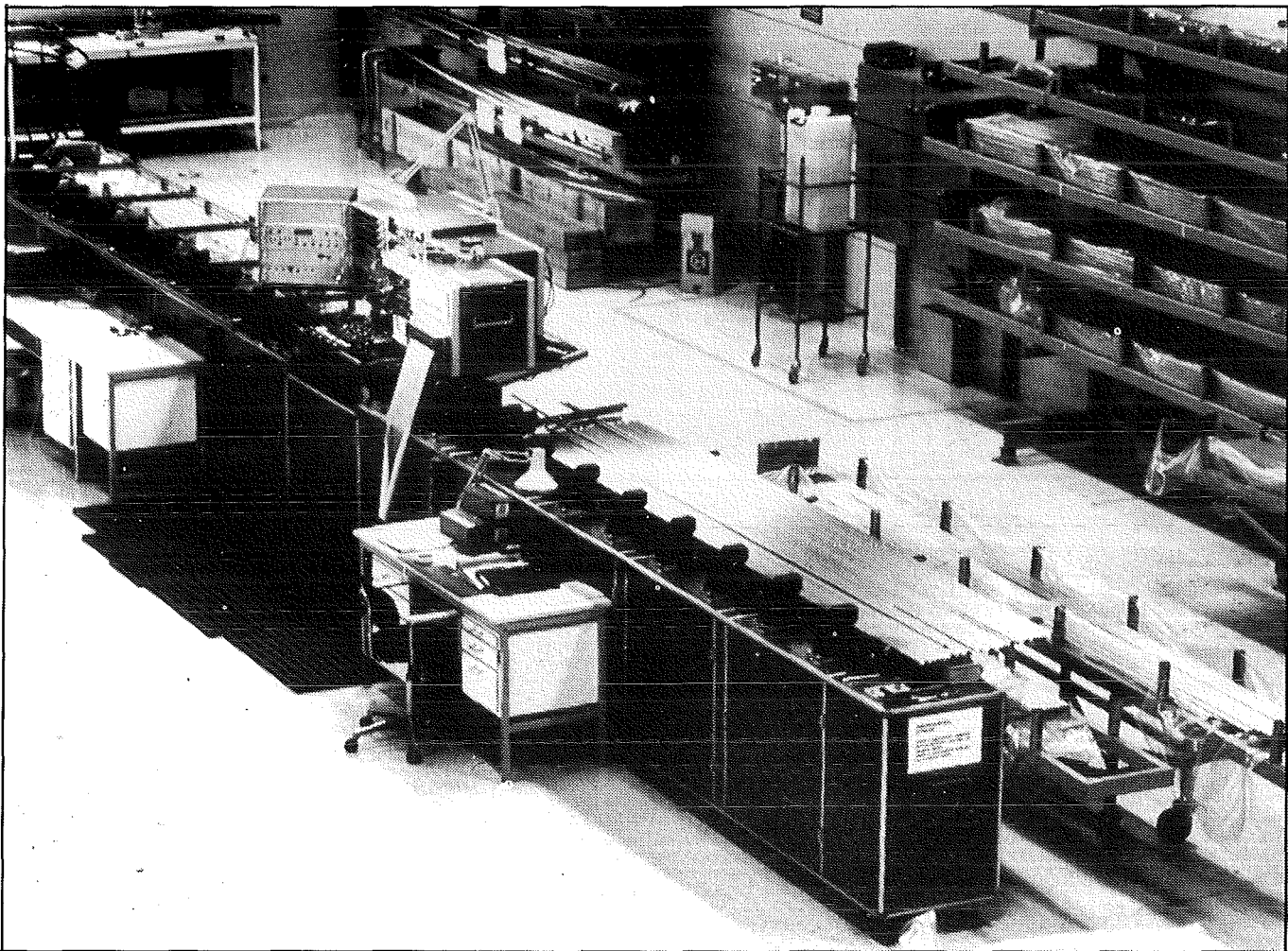
Fig. 1. Receiving inspection of cladding tubes

Rejected tubes are automatically fed out about 0.5 m further from the machine than accepted ones for easy separation. As in many other inspection machines, the inspector (operator) can not make any judgements on acceptability; all limits are set in the machines to avoid inspector error. As a point of interest it may be mentioned that the tubing suppliers are required to do 100 % inspection of tubes. Then the tubes are inspected 100 % again as receiving inspection at ASEA-ATOM.

Also in the pellet area some semi-automatic inspection equipment is used. One example of this is the pellet density check, see Fig. 12. Pellets are first weighed in air and then in water. From these two weights the density can be calculated. The inspector does not have to perform any calculations, only move the pellets to the different positions. Data are automatically fed into a microprocessor which calculates pellet density. Lot acceptance is based on a statistical evaluation of a large number of measurements.
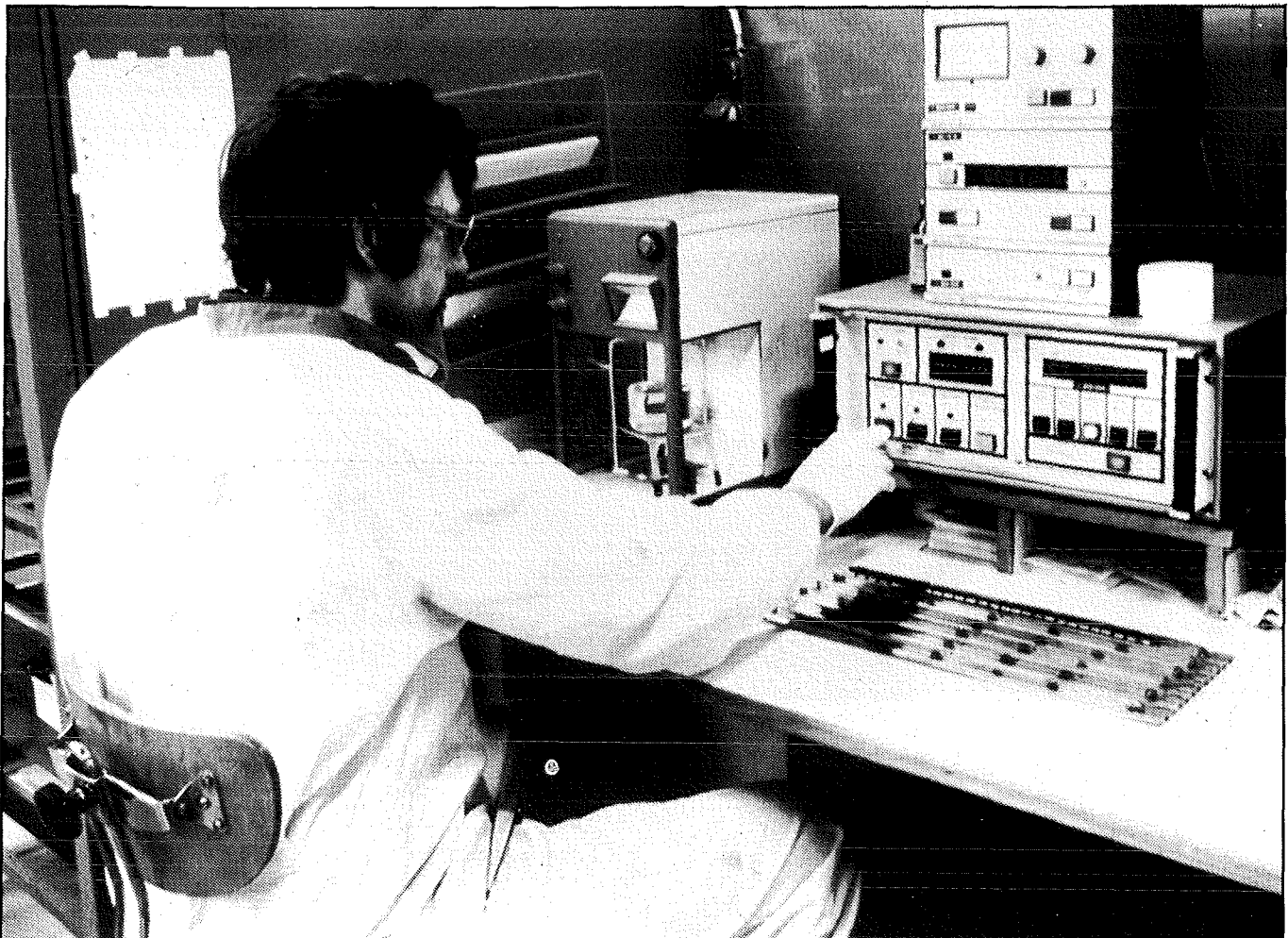
Fig. 12. Density checking of pellets

- Another example of how less inspector error can be combined with increased productivity is the use of a computerized coordinate measuring machine, see Fig. 13. The machine can be used for inspection of a large variety of components, a.o. spacer grids. Quality control of spacers includes a large number of difficult and precise measurements. Through automation and computerization, inspector error can be minimized, while at the same time making it possible to load the machine with many spacers and let the machine perform the measurements without supervision, for instance during the night.

Fig. 13. Inspection of spacer grids in the coordinate measuring machine

- Inspection of PWR fuel assemblies requires a large number of measurements at different locations in the assembly. Envelope, straightness, tilt and twist have to be checked as well as the distance between fuel rods at many locations.

A special computerized machine has been developed for this purpose, see Fig. 14. This machine will perform all the necessary measurements automatically according to a preprogrammed sequence and provides a print-out of the results, including indications of any deviation.

Fig. 14. PWR assembly inspection machine

DISCUSSION

C. FRANS: After end plug welding, you have got on the fabrication line a ultrasonic test device to control the quality of the weld. Could you, if possible, briefly explain why you have chosen this way instead of the "traditional" X-ray?

A. VAN SANTEN: In the mid-sixties we had some bad experience with TIG welding. After we changed to EB welding we also qualified ultrasonic test inspection of welds. This technique is specially suited for EB welds with a small heat affected zone and a smooth outer surface. In this application ultrasonic test is superior to X-ray and it also lends itself to automation much easier than X-ray.

M. ERNOTTE: ASEA requires 200% ultrasonic test on fuel tubes, 100% at the supplier plus 100% during receiving inspection. What is the rate of rejection in the receiving inspection and what type of defects do you discover?

A. VAN SANTEN: Less than one percent, mainly small dimensional deviations, maybe border cases.

R.S. RUSTAGI: What may be the inspection cost as a fraction of overall fuel fabrication cost and what is generally the basis of costing decisions related to inspection and quality control?

A. VAN SANTEN: No exact figure can be given, but the estimates as given in Mr. Strassers paper seem reasonable, e.g. about 20 - 25% of fabrication costs.
Apart from profitability, QC - costs are motivated by the competition on the market.

K. BALARAMA MOORTHY: Is the eddy current test you mentioned for UO_2 - Gd_2O_3 pellets intended for determining the percentage of the Gd_2O_3 content or for checking homogeneity?

A. VAN SANTEN: Not for homogeneity , but for: Gd_2O_3 content, the pattern of Gd containing pellets interjected with non Gd containing pellets (which is used to vary the Gd content for different parts of the rod).

GARANTIA DE CALIDAD PARA LA FABRICACION DE COMBUSTIBLES
NUCLEARES CON URANIO NATURAL

José Helmut Koll
COMISION NACIONAL DE ENERGIA ATOMICA
Unidad Sectorial Combustibles Nucleares
Depto. Garantía de Calidad
Avda. del Libertador 8250
1429 Buenos Aires
Rep. Argentina

ABSTRACT

The experience obtained through seven years of development
and fabrication of nuclear fuel, has shown that it is possible
to implemented a quality assurance system stepwise. The most
important factor is adecuately trained personell, at all levels,
which has had the opportunity to work at the actual industrial
equipment during the development phase.

Experienced laboratories for testing the quality of materials,
products and for the qualification of the process, is another
vital factor .

1. INTRODUCCION

La implementación de un sistema de garantía de calidad para el suministro de combustibles nucleares es una tarea que requiere de una gran experiencia si se involucra desde el inicio todas las etapas que van desde el diseño, adquisición de materiales y equipos, capacitación de personal, procesos de fabricación y control de calidad y hasta la inspección y entrega final de los elementos combustibles a la central nuclear.

Alternativamente puede implementarse un sistema de Garantía de Calidad en forma progresiva, si se planifica de tal forma desde el inicio y si se presentan condiciones favorables, tal como en los reactores a uranio natural, en los que un combustible puede ser introducido o retirado sin interrumpir el normal funcionamiento del reactor.

Esta implementación progresiva es aplicable especialmente cuando se trata de combustibles nucleares ya conocidos o con sólo ligeras modificaciones, siendo facilitada si además se disponen de materiales aprobados, equipos apropiados, procesos calificados, personal capacitado, así como de la documentación necesaria.

Queda por resolver, cómo se obtienen estos elementos arriba citados, es decir los materiales, equipos, procesos, personal y documentación.

1. Diseño de Elementos Combustibles

    Cálculos - Ensayos para diseño - Planos y Especificaciones.

2. Desarrollos de tecnología

    Procesos de fabricación
    Técnicas de control, ensayos y análisis

3. Equipamiento industrial

    Para Fabricación, para Control de Calidad

4. Materiales: Documentación de Compra - Recepción

5. Laboratorios de Ensayos

6. Personal - Capacitación

7. Ingeniería: Documentación de Fabricación y Control -
    Procedimientos

8. Calificación de Procesos

9. Instalaciones industriales

10. Capacidad industrial operativa

    Organización - Calificación

Abreviaturas usadas:

CNEA   : Comisión Nacional de Energía Atómica
FAE (Zry) : Fábrica Aleaciones Especiales
CFC (UO$_2$) : Complejo Fabril Córdoba
EC :· Elemento Combustible

2. DESCRIPCION TECNICA

Para poder describir adecuadamente las condiciones dentro de las que se ha desarrollado el presente trabajo, debe mencionarse lo siguiente:

Los participantes principales a considerar para un sistema de Garantía de Calidad para combustibles nucleares son:

a) El organismo licenciante.
b) La Central Nuclear.
c) El suministrador de EC
d) Diseño y evaluación de EC.
e) Garantía de Calidad Combustibles Nucleares.
f) Fabricante de EC.
g) Proveedor de polvo UO$_2$.
h) Fabricante de semiterminados de Zry.
i) Subcontratistas de partes y piezas.
j) Proveedores de insumos directos e indirectos.

La filosofía general adoptada es disponer de personal adecuadamente capacitado para realizar las tareas que corresponden a cada uno de los participantes, con suficiente nivel de tal forma que sea normal trabajar respetando básicamente requerimientos de garantía de calidad. De esta manera, cuando se impongan el cumplimiento estricto de garantía de calidad, no se vea alterada esta forma de trabajo.

Los tipos de combustibles considerados son:

Atucha I, CANDU y Atucha II

Las características de estos elementos combustibles ya son ampliamente conocidos. En particular el EC Atucha I ha sido descripto en varias publicaciones, así como el EC CANDU 600 Mw usado en la Central Nuclear Embalse. El EC para la Central Nuclear Atucha II es una versión modificada del EC Atucha I.

La diferencia más importante reside en la forma en que son tratados los desarrollos y la fabricación de estos EC por CNEA, como se describirá a continuación y que permitieron cumplimentar la mayoría de los requerimientos de garantía de calidad.

La técnica aplicada ha sido entonces la de desarrollar primero el conocimiento, para luego y en base a este conocimiento, integrar las distintas actividades hasta conformar finalmente el sistema de garantía de calidad.

2.1 Línea Elementos Combustibles Atucha I (1976-1982)

1. Diseño   Conocido-proveniente de KWU

Probado en uso en la CNA-I desde 1974
Planos y Especificaciones existentes

2. Desarrollos de tecnología

Desarrollos realizados con equipos de laboratorio para
conocer los procesos.
Desarrollo de técnicas de control, especialmente ensayos
menos comunes (corrosión en autoclave, hidruración, explo-
sión, etc.) y análisis químicos.

3. Equipamiento industrial

La etapa de desarrollo puede comenzarse con equipos de la-
boratorio pero luego se exige la puesta a punto de las téc-
nicas de fabricación con los equipos industriales.
Equipos especiales: adquiridos a través del fabricante ori-
    ginal del combustible.
Equipos convencionales: adquiridos en el mercado comercial.

4. Materiales y partes componentes

Adquisición de polvo UO_2 y semiterminados de Zircaloy a fa-
bricantes reconocidos que ya operan bajo un sistema de ga-
rantía de calidad, en el mercado internacional. De igual
modo se compran otros materiales directos tales como Helio
para soldadura y aleaciones para brazing. Debe implementar
se la recepción de los materiales, así como sus controles.

En caso de existir partes componentes complejas, debe consi
derarse también su adquisición a fabricantes conocidos.

En una segunda etapa de integración se incorporarán estos
materiales de polvo UO_2 y zircaloy de producción local.

5. Laboratorios de ensayos

Es necesario desde el inicio disponer de los servicios de
laboratorios que tengan experiencia de primer nivel. Sólo
así puede asegurarse que los resultados de ensayos sean los
correctos. En general es posible obtener estos laboratorios
para análisis químicos y ensayos mecánicos a nivel nacio-
nal. Más difícil resulta disponer de servicios de metalogra
fía, análisis de gases residuales, de corrosión y de ensa-
yos no destructivos especializados en tubos de reducidos
espesores.

Si bien resulta posible sin dificultades realizar el con-
trol dimensional adecuado, no siempre se dispone del servi
cio de calibración requerido, debiendo por ello tomarse las
precauciones desde el inicio.

6. Personal

Personal capacitado es el componente clave en todo sistema
de garantía de calidad. No es suficiente con "enseñar", es
absolutamente necesario dejar que el personal se ejercite
en sus tareas, a fin de adquirir la experiencia necesaria.
Esta experiencia debe incluir no sólo los casos cuando
"todo anda bien" sino también es fundamental que tenga que
resolverse casos cuando "algo falla".

El hecho de poder trabajar desde fecha temprana con los

equipos industriales permite aumentar notablemente la validez de la experiencia obtenida.

7. Ingeniería de Fabricación. Documentación.

Una vez desarrollados los puntos 1 a 6 arriba descriptos y con la experiencia obtenida, se podrá ahora completar la documentación requerida para la fabricación y el control.

La documentación preexistente en forma preliminar, se irá adaptando a la experiencia que se vaya obteniendo, pudiendo aumentar o disminuir los requerimientos. En general, al inicio se tomarán precauciones adicionales, que luego pueden ser innecesarias.

Documentos de fundamental importancia son:

El plan de inspección y ensayos
Las planillas de control de calidad
Los procedimientos de fabricación existentes en cada
   lugar de trabajo (no sólo en los archivos)
Los procedimientos de ensayo, análisis y control
La calibración de instrumental de medición y ensayos

8. Calificación de Procesos

Consiste en calificar el Proceso, el equipo, el personal de fabricación y el de control, el producto y si cabe el término, también la documentación.

La calificación es una de las etapas de más responsabilidad en todo el sistema de garantía de calidad, justamente porque implica decir y escribir que todo el conjunto de tareas para una dada actividad se está realizando correctamente.

Una vez calificado un puesto de trabajo, debe verificarse periódicamente si se mantienen las condiciones de la calificación.

9. Instalaciones industriales

Todas las tareas hasta ahora descriptas en los puntos 1 a 8 pueden ser realizadas en Plantas Piloto. Paralelamente se debe construir la Planta Industrial. Sin embargo, cuanto antes se disponga de ésta, tanto menor serán las incertidumbres en el momento de la transferencia de equipos, su reinstalación y puesta a punto.

10. Capacidad Industrial Operativa

La última etapa en la integración de las actividades que permiten implementar la fabricación de EC a escala industrial bajo un sistema de Garantía de Calidad es obtener la Organización Operativa. Llegado a este nivel se completarán la totalidad de los requerimientos del Programa de Garantía de Calidad. Ello se verificará durante la Calificación del Fabricante.

2.2 Línea Elementos Combustibles Embalse (CANDU) (1978-1984)

La filosofía general es la misma en cuanto se refiere al desarrollo progresivo del conocimiento y su aplicación. Respecto de

los detalles existen, sin embargo, diferencias muy importantes en relación a la Línea Atucha I.

Estas diferencias son:

1. Diseño: sólo existía un diseño conceptual. No existían planos y especificaciones.

2. Tecnología: se realizó en forma similar, comenzando con los procesos a nivel de laboratorio, pero pasando rápidamente a escala industrial.

3. Equipamiento industrial: hubo necesidad de desarrollo propio de equipos especiales y, con elló, la calificación de los mismos.

4. Materiales especiales: se procede igual a la línea Atucha-I.

5. Laboratorio de ensayos: se disponía parcialmente de la experiencia desarrollada para la línea Atucha-I, completándola.

6. Personal: fue necesario formar un nuevo grupo de personal capacitado, de manera similar a la línea Atucha-I.

7. Ingeniería de Fabricación÷Documentación: se desarrolló en forma similar toda la Documentación de Fabricación y Control, pero con planos y especificaciones propias.

8. Calificación de Procesos: se procede en forma similar a la línea Atucha-I.

9. Instalaciones Industriales: dado que, a la fecha de comienzo del desarrollo de esta línea de Combustible Embalse, ya se disponía de instalaciones industriales, se pudo hacer uso de las mismas, prácticamente, desde el inicio.

10. Capacidad Industrial Operativa: se desarrollará como una transferencia,en un mismo lugar físico,de la responsablidad de un "Proyecto" a una empresa industrial.

2.3 Línea Elementos Combustibles Atucha-II (1981-1986)

En este caso, aún en etapa de su inicio, se procederá de modo diverso, dado que ya se dispone de capacidad operativa a nivel industrial.

Sin embargo, se agrega a la lista de tareas, la de diseño incluyendo ensayos para diseño, en colaboración con expertos a nivel internacional.

Ya desde el comienzo, se utilizará polvo UO_2 y Zircaloy de origen nacional.

3. RESULTADOS OBTENIDOS

La experiencia obtenida después de 7 años de trabajo, ha demostrado que ha sido posible desarrollar en forma progresiva,e ir cumplimentando los requerimientos de un sistema de garantía de calidad, hasta llegar a integrar las actividades desde el diseño,

incluyendo la ingeniería y la fabricación de elementos combustibles.

Durante las etapas de desarrollo se han fabricado 240 EC-Atucha-I, y durante los dos primeros años de fabricación industrial, 240 EC por año que equivalen, en total, a casi 3 núcleos del reactor. Para estos EC se ha utilizado polvo UO_2 y Zircaloy de origen importado.

Durante las etapas iniciales de fabricación, la calidad queda asegurada mediante un Manual de Garantía de Calidad, con la documentación necesaria que permita el registro de los datos requeridos.

Con la experiencia obtenida durante la etapa de desarrollo se redactaron los procedimientos e instrucciones, pero su implementación y efectivo uso quedó en manos del fabricante industrial.

Mediante auditorías de garantía de calidad y un servicio de evaluación de desviaciones podía asegurarse que se había logrado trabajar bajo los principios de la "Garantía de Calidad". A pesar de ello, quedan aún muchos detalles por ajustar o mejorar.

En la actualidad, se está comenzando a aplicar este sistema para incorporar a la línea de fabricación polvo UO_2 y tubos de Zircaloy de fabricación local.

4. CONCLUSIONES

La implementación de un Sistema de Garantía de Calidad para la fabricación de combustible nuclear a escala industrial puede realizarse comenzando por la elaboración de toda la documentación requerida, entregada a una organización preestablecida y capacitando personal en forma masiva.

Una alternativa de mayor contenido nacional es la de realizar los desarrollos de procesos y equipamiento y entrenamiento de personal en forma incremental, hasta llegar a la etapa industrial, generando los procedimientos y controles a medida que se vaya ganando experiencia.

En la fase inicial, la utilización de materia prima de calidad asegurada, facilita las actividades de garantía de calidad del producto final.
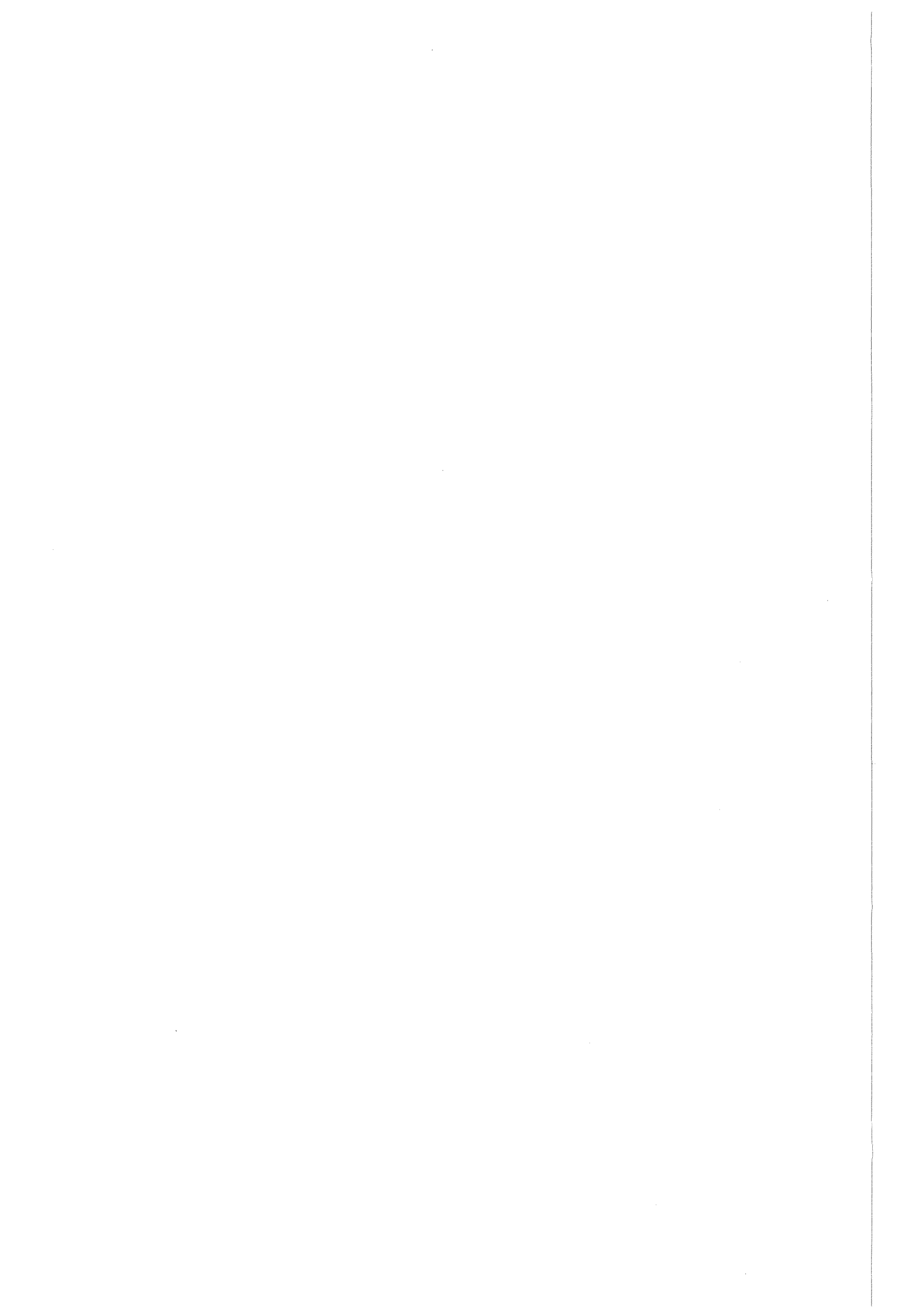
USE of the D O M SOFTWARE SYSTEM IN QUALITY ASSURANCE WITH THE MANUFACTURE OF FUEL ELEMENTS FOR NUCLEAR REACTORS.

Authors: Herbert Engel,   Reaktor Brennelemente Union, Hanau

Jürgen Peuker,   Reaktor Brennelemente Union, Hanau

Eberhard Herrmann,   Siemens AG, Zentralbereich Technik, München

Helmut Hackl,   Siemens AG, Zentralbereich Technik. München

D O M is a program system for the PR 300-16 process computer with the aid of which defined objects (workpieces) are followed trough the individual production and testing operations and the production and test data acquired in this way can be stored and evaluated in relation to the object.

The D O M program system is a real time system, i.e. all program functionsdescribed above are available to the user at any time.
This is indeed of great significance for the application of data selection and evaluations functions.

Use of the DOM software system in quality
assurance with the manufacture of fuel elements
for nuclear reactors

1.0 General objectives

1.1 The RBU and its manufacturing range

The manufacturing program of the RBU comprises the
production of fuel elements for power and ex-
perimental reactors on the basis of weakly enriched
uranium dioxide (UO_2) according to the pressurized
and boiling water concept.

This includes the manufacture of the following
individual products:
- Sinterable uranium oxide powder (from uranium
  "UF_6" or uranyl nitrate "UNH" according to
  AUC process.

- Sintered UO_2 pellets

- Sintered UO_2 pellets with neutron absorbers (Gd_2O_3)

- Complete fuel rods and control rods

- Fuel element structural parts such as spacers,
  control rod guide tube, skeleton, etc.

The individual parts mentioned (Fig. 1: Schematic
display of the manufacture of a pressurized water
fuel element) are assembled by the RBU either to
fuel elements or control and absorber elements
or they are delivered as piece parts to the
customers.

Fig 1: Schematic display of the manufacture
of a pressurized water fuel element
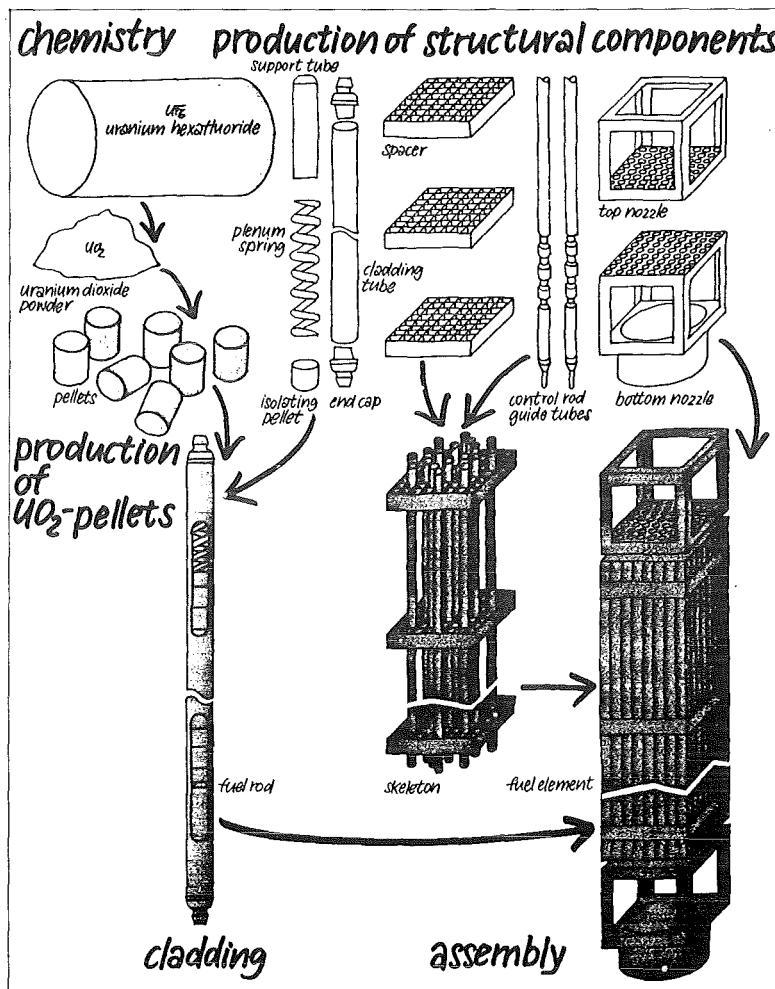
Furthermore, the RBU is concerned with the recovery
of non-irradiated uranium containing production
scrap as well as with the planning and construction
of fuel element manufacturing plants or sections
of them.

1. Objectives of the EDP system

The objectives were monitoring for manufacture
and testing according to specification of fuel
elements and their components for light water
reactors accompanying the manufacturing
process.
In addition, complete and faultless documentation
of the quality data for the individual objects
should be ensured (Fig.2: Schematic display of
the objectives).
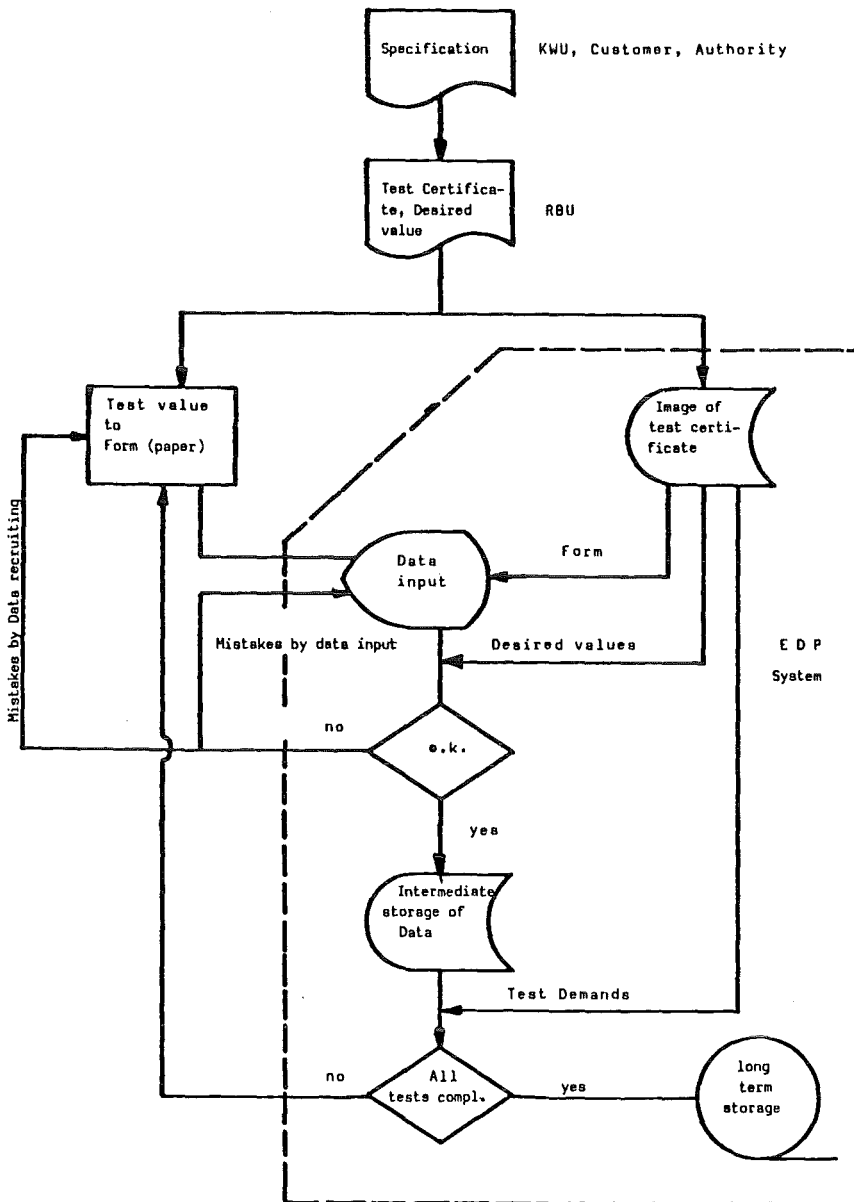
Fig. 2:

Schematic display of the objectives

Within the framework of these objectives, large quantities of data must be acquired, assigned selectively to individual objects, processed specifically with regard to order, object and acquisition variables, documented and archived.

- Data acquisition specific to the order and object is a decisive feature of the EDP system. A material or immaterial unit, to which short term, medium term or long term information can be assigned is understood under the term "object". With RBU, the information consists of the quality and production data of the objects.

- The data acquired must be checked quickly and reliably to ensure that they conform with the <u>specified characteristics</u>.

- Apart from this checking of the individual data, the <u>completeness</u> of all data of a testing and production operation as well as carrying out the required complementary tests (e.g by TüV (Technical Inspection Board) or customer) are ensured.

- A further test which must be carried out by the system concerns observation of the prescribed test operation sequence and the completeness of the testing sequence for a certain object.

- In the case of permitted <u>rework loops,</u> the above-mentioned tests must also be carried out taking account of the different types of rework.

- Releasing of individual objects must be monitored by the system.

<u>An object is released:</u>

1. when all production and testing operations required according to order specifications have been performed properly and the result was in accordance with the specification;

2. when any deviation reports have been released or rework has be properly terminated;

3. when the object possesses clear identification within the order.

- It must be guaranteed that only released objects,
  i.e. only those according to specification reach
  <u>further processing.</u> In this way it is achieved
  that objects can only be incorporated in an object
  of higher order if all tests have been carried
  out properly. This also applies for piece parts
  of an object in batches; e.g. a sleeve tube
  delivery batch must be released before a sleeve
  tube from this batch can be used for an individual
  fuel rod.

- The individual objects making up each object are
  recorded by <u>cross references</u>. Thus it is possible
  to have recourse to the manufacturing history
  of the piece parts of a complex product and to
  countercheck once more the release of the parts
  making it up. These cross references can cover
  several hierarchy levels.

- All test results, cross references and manufacturing
  data required by the specification (e.g.
  operating data such as plant, operator or important
  process parameters) must be <u>documented</u> related
  to the object <u>long term</u>. The same applies for
  released deviation reports and rework.

- All test specifications used during an order
  (e.g. ideal value of a certain quality feature)
  must also be documented together with all <u>modific-</u>
  <u>ations</u> made during the course of an order. The
  date of the modification is recorded.

2.0   Technical description of the DOM software system

DOM[1] is a program system for the PR 300-16 process
computer with the aid of which defined objects
(workpieces) are followed through the individual
production and testing operations and the production
and test data acquired in this way can be stored
and evaluated in relation to the object.

DOM system components for data acquisition are:
The dialog system for form processing DIF for the
data input and output by data monitor and a standard
DOM program package for taking over data from user
programs which realize   for instance, the coupling
between decentral unit and test station computers.
Programs for immediate processing of data on input
as well as for the preparation of outputs are con-
nected in series to that section. They undertake
for instance testing for tolerance limits or they
determine with a data output the mean value of several
input individual values.
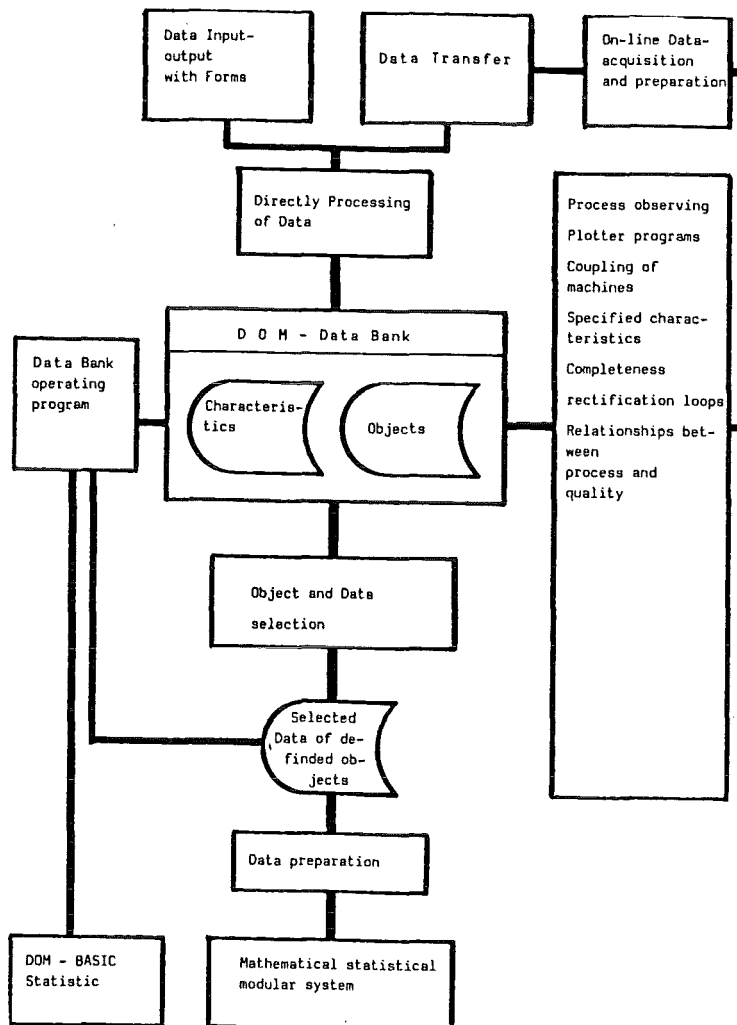
Fig. 3:

Basic structure of

the DOM software

system

The heart of the DOM software system is the DOM data
base. It is based on the Siemens standard software
product DATORG 300, a program system for data organ-
isation and administration. This system also provides
the basic elements for the DOM data base modules
(standardized subroutines) with which both DOM programs
and also arbitrary user programs specific to a project
can have access to the data base.
A data bank operating program (DBBEDI) makes it
possible to perform conveniently (operation using
self-explanatory operating masks) at the system level
modifications, extensions and corrections to the
data base or to initiate outputs (e.g surveys) in
different degrees of detail.

By means of a selection system, problem-oriented
data sets can be constructed from the initially in-
homogeneous data quantity of ideal and actual values
of different objects and these can be made accessible
to statistical evaluations. In this way such things
as exact and quick statements on the actual status
of the manufacturing process and the quality of the
finished products (position and spread as well as
the dependence on time of the corresponding process
and quality parameters; fault statistics,   etc.)
can be obtained.

The data base operating program affords the facility
of selecting data of defined object groups based
on simple selection criteria - at the level of the
object classification. The selected data are brought
into a form compatible for BASIC programs. A DOM
BASIC statistics package is available for the eval-
uation of these data series; these are programmed
procedures of descriptive and assessing statistics,
e.g. frequency analyses, scatter plots and re-
gression analyses, contingency table test, variance
analyses, time series display with trend analyses.

A defined data selection is possible using the DOM selection system. In this any stored data can be referred to as criterion for data selection, in addition also still the variables implicitly recorded by the DOM system, such as the time of data acquisition. For problems posed, such as the investigation of the spread of a quality parameter of workpieces which run in a certain period over a certain production process and are manufactured there under certain process conditions, the data material to be investigated can be prepared with the aid of the DOM selection system.

The object-related acquisition of data on the manufacturing history and the quality achieved permits statistical parameter investigations which should lead to greater understanding of the relationships between process and quality and should thus in the final analysis lead to quality and process improvements.

The statistical program package MASTABA[1] has been conceived primarily for these parameter investigations and important parts of it have been realised.

In particular comprises by multivariante procedures such as multiple stepwise linear regression analyses and multiple variance analyses.
The input values can be transformed by a built-in MASTABA[1] module before evaluation by these statistical modules. Thus it is possible even to process expected non-linear relationships between parameters and target variables with these procedures.
(Fig.4 ; Fig.5)

1) Please refer to explanation of terms on page 11

Static process characteristic (def. period)	Dynamic process characteristic	Internal process structures and relationships

process   variable   quality   variable

quality variable   date   t   process variable
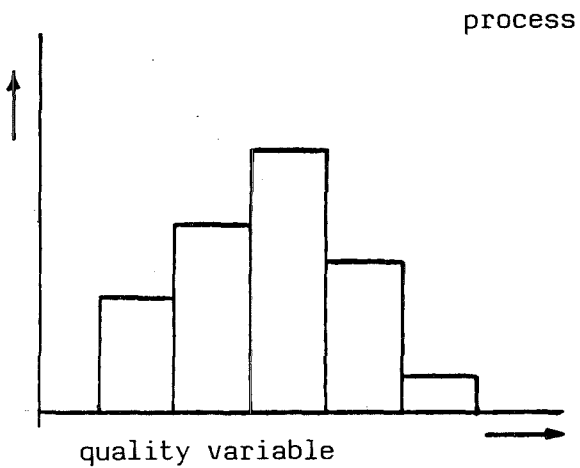
Procedure:

. frequencly analysis
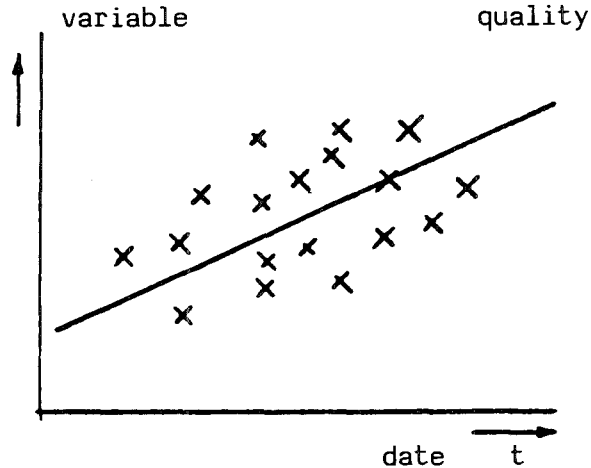
Procedure:

. time series display and analysis

. regression any variance analysis with parameter date
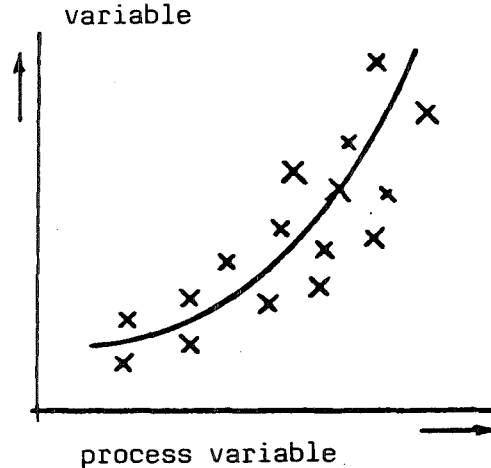
Procedure:

. regression analysis
. variance analysis
. contingency table test
. comparision of distribution
. discrimination analysis
. covariance analysis

Fig.: 4   Statistical methods for process analysis

PROCESS

SETTING
INSTRUCTIONS

MARGINAL
CHECKING

COMPUTED VALUES
X_{MAX}, X_{MIN}, INT.
DIFFERENCE, $\overline{X}$, etc.

$\underline{M O D E L}$
CURRENT-PROCESS-SIM.
IDEAL/ACTUAL COMPAR.
CALCULATE OF CONTROL
VARIABLES

LONG TERM
OBJECT-RELATED
STORAGE

STATISTICAL EVALUAT.
(GRAPHIC + CALCUL.)
E. G. CALCULATION

CUSTOMER

DOCUMENTS
RBU-QA
RBU-MAN.

DEVELOPMENT OF A
PROCESS-MODEL
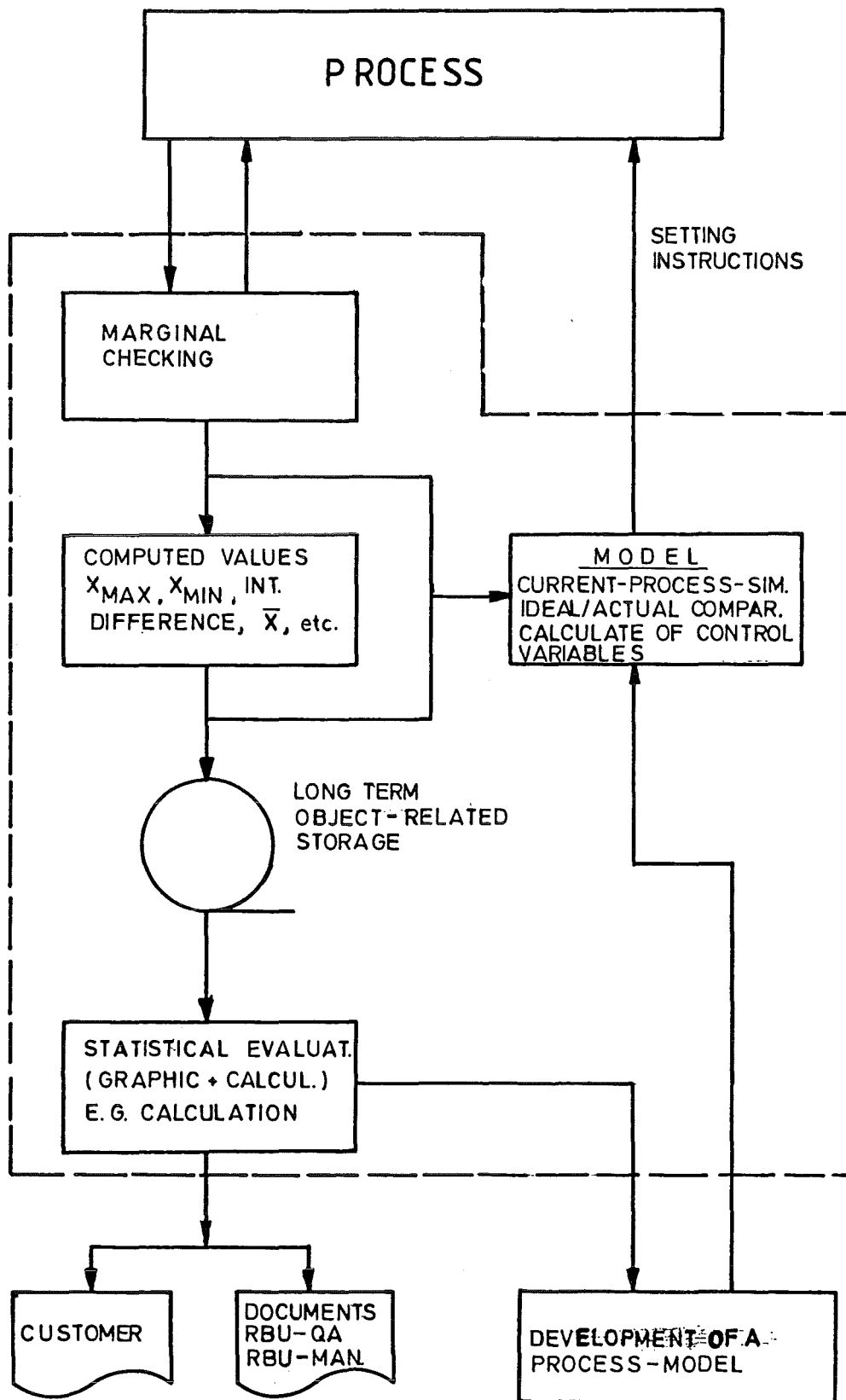
Fig. 5: Schematic display of the development of a process model

3.0   Results of use so far

The introduction of the system was realised step by step from May 1982 onwards according to the manufacturing and test procedure of the fuel elements.

According to the use so far of the DOM system, it can be stated that the expectations were fulfilled.

Expensive subsequent reworking was avoided or the accumulation of unnecessary costs was prevented by the early indication of omissions and irregularities in the manufacturing and testing procedure.

This concerns the following points in particular:

- the staff-intensive concluding documentation checking and possible supplementing can be dispensed with;

- missing inputs and tests can follow on immediately in the correct sequence;

- errors in reworking required are avoided and necessary sortings out are ensured;

- parts which do not conform with the prescribed ideal procedure are not released and thus cannot be further processed.

There has already been a noticeable "educational effect" by errors and omissions being detected immediately after their occurrence, by them being indicated to the operator and by their immediate rectification being enforced.

The system showed itself to be very robust (after
the usual teething problems) in dealing with faulty
inputs by the operators. Errors in the manufacturing
or testing procedure occurring or faults in the product
are indicated to the operator on the spot in such
detail that he can take the necessary measures without
calling on a system specialist in order to make further
working possible. These system properties permit the
system to be used in the 2nd and 3rd shifts as well
where supervision is low.

It is just as important that DOM was shown to be safe
against system failure caused by software and that
no data losses caused by software occurred. Since
the process computer hardware used also proved to
be reliable, the system availability was high.

The good acceptance by the users at introduction of
the system is encouraging, in particular at the lowest
user level. Apart from the detailed reporting back
already mentioned and the robustness towards faulty
inputs, this is to be attributed in particular to
the practical dialog guidance which was achieved as
a result of the input forms individually set up by
the user.

High demands were placed at RBU on the running time
characteristics since many input dialogs run in
parallel and complex input data processing is performed
in real time mode.
This led even in the introductory phase at RBU to
a number of system improvements being incorporated
to accelerate the dialog processing.

The running time characteristic is thus currently
satisfactory but it requires further measures in view
of technical modifications aimed at in the manufact-
uring and testing process.

Effects on the quality assurance system

A manual process with the same performance criteria
is not possible for time and staff reasons. However,
it is just the processing accompanying the process
which has a favourable influence on quality-conscious
work; a learning effect is achieved by the immediate
reports back.
Use of the system creates better clarity so that
points of weakness in the manufacturing and testing
process can be detected faster and more easily.

The additionally available facilities for data eval-
uation also enable the static and dynamic behaviour
of the processes to be identified and interfering
effects to be discovered.

In this way at all operator levels the relevant know-
ledge about the processes and consequently the
technological and organisational procedure are
improved.

Because of the versatility of the system, possibilities
of use arose for objectives where the intention
originally was to use special program packages, e.g.

- supervising the availability for use of the measuring
  equipment stock (approx. 11000 measuring devices)
  of the RBU, as well as their administration within
  the task framework of the B test station.

- supervision of the admissibility of manufacturing
  and testing equipment for order processing.

- assistance with the administration of the specific-
  ations and drawings permissible for manufacturing
  and testing.

- administration of documents for the approval pro-
  cedure according to the Nuclear Law, Paragraph 7.

4.0   Conclusion

EDP has made considerable inroads into the technical
area in recent years.
The areas of application are divided from one another
by terms such as CAD (Computer Aided Design)

CAM (Computer Aided Manufacturing)

CAE (Computer Aided Engineering)

CAQ (Computer Aided Quality Control)

etc.   (Fig. 6)

Fig. 6 :
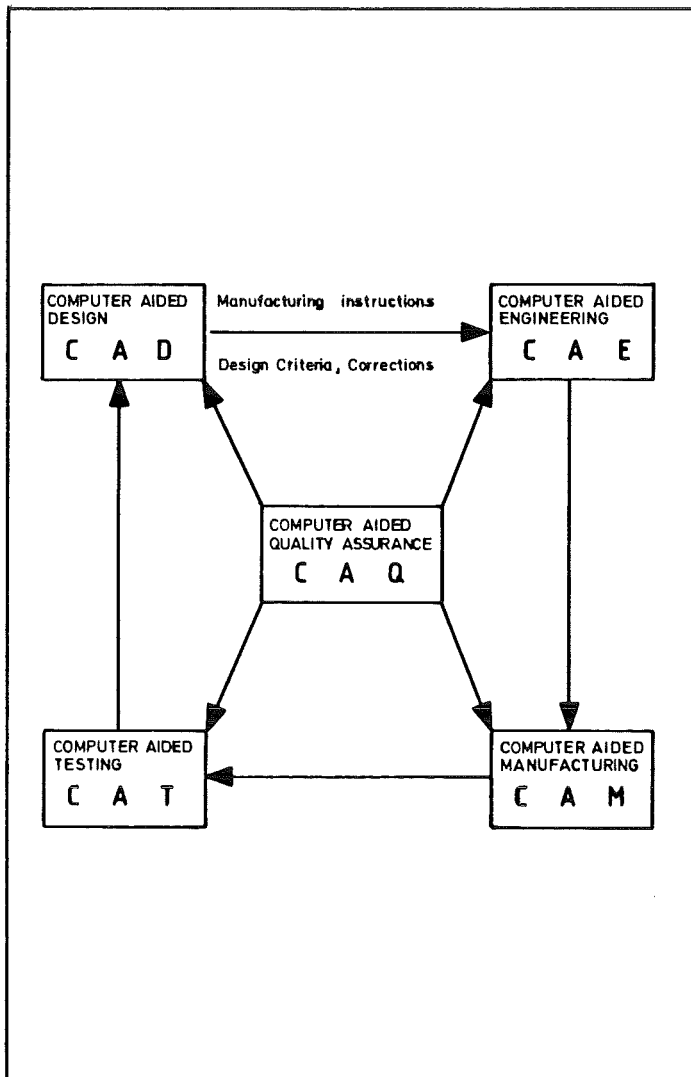
The DOM software system covers partial areas of CAM
and CAQ, the focus being on CAQ.

It is planned, extending beyond the original objectives,
also to employ DOM increasingly for tasks in the field
of process analysis and process optimization.
It this way the DOM system will become increasingly
a management system which can bring data from widely
differing process operations together.

In the further extension, computer-assisted manu-
facturing and testing stations will be installed
at the work station level, the results of which will
be transmitted in compressed form over direct trans-
mission paths to DOM.
The decentralisation of the computer assistance into
autonomous partial systems connected with this can
be administered unproblematically by DOM in its
present stage of development.

Thus this system will also fulfil in the near future
the requirements on a software system for quality
assurance at RBU.

The DOM program system is a real time system, i.e.
all program functions described above are available
to the user at any time. This is indeed of great sig-
nificance for the application of data selection and
evaluation functions.

The conception and structure of the DOM system makes
possible at different levels easy adaptation to the
circumstances specific to a project: Thus the "objects"
for which the data should be acquired, processed,
stored and evaluated can be freely defined initially.
This also applies for the number and type of the
variables to be acquired for these objects. The DOM
system programs also offer the possibility at a further
level of incorporating user programs specific to the
project and thus of adapting to the project problems.
The use of standardized DOM program modules for the
data base access in user programs also simplifies
extensions specific to programs. The possibility of
being able to initiate DOM system programs from user
programs with tasks specific to a project also serves
this purpose. Not least, an information exchange
between user programs and the DOM system can take
place over simple data interfaces.

Meaning of the program or program package abbreviations

DOM	: Documentation of the manufacturing and quality history of objects in discrete processes with modelling by statistical methods
DIF	: Dialog system for form processing
DATORG 300	: Data file organisation and administration
DBBEDI	: DOM data base operating program
MASTABA	: Mathematical-statistical modular system
AENEAS	: Construction of input-interfaces for parameter analyses and other statistical evaluations

SESSION V

PHYSICAL METHODS FOR ANALYSES 1.

Chairman: A.A. Strasser
S.M. Stoller Corporation, New York

SESSION VI

PHYSICAL METHODS FOR ANALYSES 2.

Chairman: G. Dressler
EXXON Nuclear GmbH, Lingen
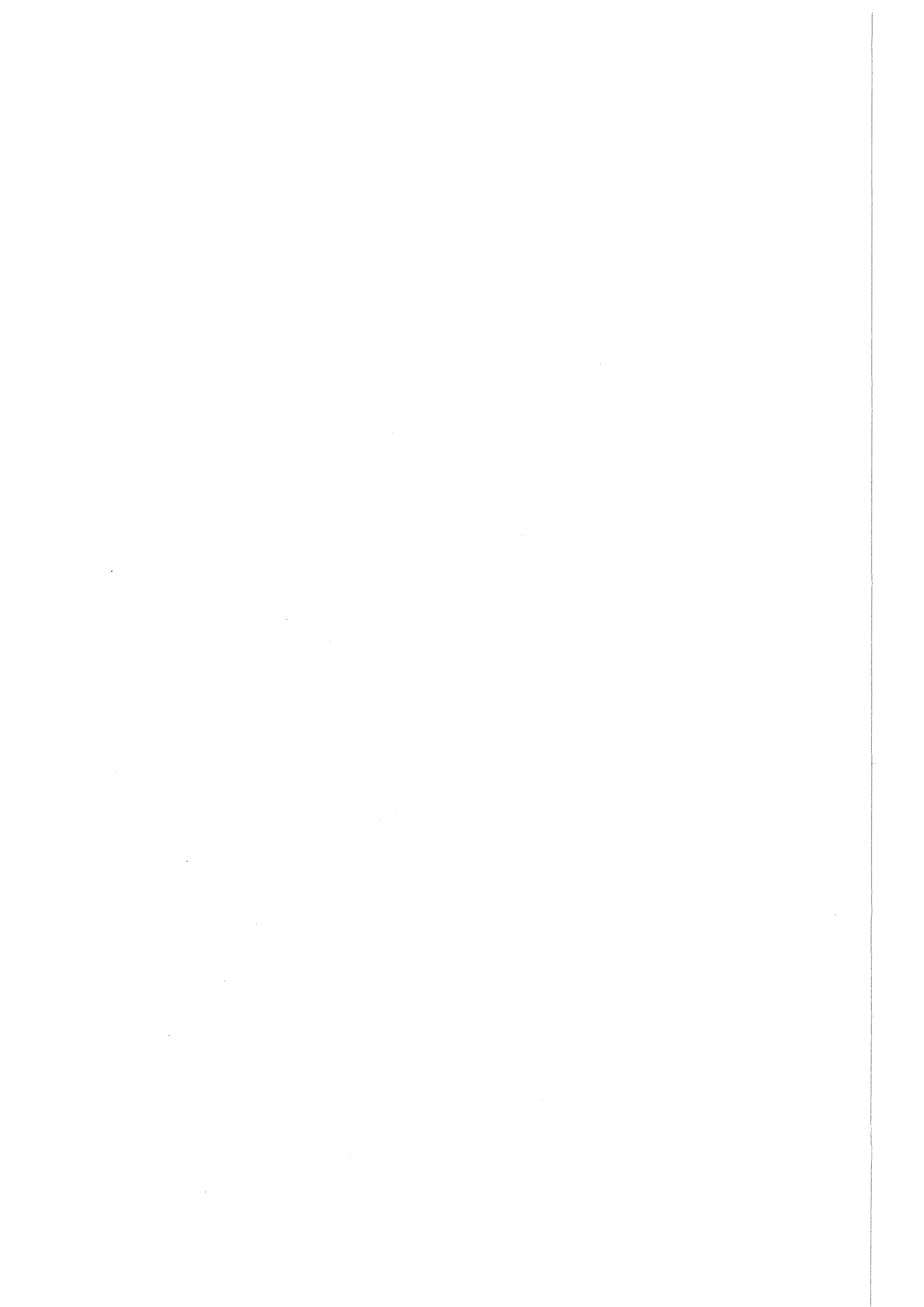
PHYSICAL METHODS OF QUALITY CONTROL OF FUEL PELLETS

D. Vollath

Institut für Material- und Festkörperforschung III

Kernforschungszentrum Karlsruhe Postfach 3640,

D - 7500 Karlsruhe, Federal Republic of Germany

H.-J. von Wachtendonk

Reaktor Brennelement Union GmbH, 6450 Hanau 11

Federal Republic of Germany

Abstract:

To guarantee a uniform quality of the final product it is indispensable that,
beginning with the powder, all intermediate products are subject to control.
In this paper control methods are presented both for methods of quality control
and methods of characterization. Since a complete description of all possible
methods goes beyond the framework of this contribution only some methods are
presented by a way of example.

1. Introduction

The quality control of uranium dioxide or uranium/plutonium dioxide pellets must not start as late as on the fabricated pellets. To guarantee a uniform quality of the final product it is indispensable that, beginning with the powder, all intermediate products are subject to control. In the controls preceding final control of the pellets a basic distinction must be made between two types:

- Controls directly related to specified variables, e.g., enrichment.

- Controls the results of which serve as a controlling parameter for subsequent fabrication steps, e.g., properties of the powder.

In this paper control methods will be presented for both types of control. The control methods can be further broken down into:

- methods of characterization, and
- methods of quality control.

Methods of characterization are applied in qualifying a fabrication method and a new product, respectively, or if one simply wishes to know more about a product. The methods of characterization are not suited in routine quality control; they are much too expensive and time consuming. Also a great number of the methods of quality control are applied only at the beginning of a fabrication lot if it has been ensured by qualifying the fabrication method that the variables will not be subject to variations.

Since a complete description of all possible methods would go beyond the framework of this contribution, only some methods will be presented now by way of example. In special those methods are explained, having a potential for future application, even if they are not introduced to industrial production until now.

2. Fabrication of UO_2 Pellets and Accompanying Quality Controls

Fabrication step	Control
Powder fabrication	Metal content (% U)
	Enrichment (% U-235)
	O/U ratio
	Fluorine content
	Metallic impurities
	BET surface
	Grain size
	Free flowing property
	Bulk density
Pressing	Tap density
	Green density
Sintering	
	Fluorine content
	Density
Grinding of curved cylinder surface	Metal content (% U)
	Enrichment (% U-235)
	O/U ratio
	Chemical assay for F, Cl. C, N_2, H_2
	Metallic impurities
	Residual gas
	Geometry
	Density
	Microstructure
Rod filling	Surface quality

Figure 1 is a greatly simplified flowsheet of pellet fabrication and of
the control methods to be applied following the individual fabrication
steps.

If one looks at this figure, one sees that the source powders are com-
pletely analyzed for their chemical impurities and the properties which
are significant with a view to fabrication. The whole chemical assay
is repeated on the fabricated pellets, also in cases where no bonding or
lubricating agents have been added to the powder. The reason is that the
impurity content may be reduced during the process of sintering through
evaporation (e.g., fluorine) or the portion of metallic impurities may
increase, e.g., by mechanical abrasion.

3. Control of the Source Powders

3.1 Basic Controls

The controls of the source powders must cover the following items:

- chemical and physical compositions,
- properties in terms of conveyance and compaction behavior,
- sintering properties.

Of the controls relating to the composition of the powder only the gamma
spectroscopic determination of enrichment will be explained here.

At the receiving control stage a powder sample is collected, homo-
genized and a small fraction of it dissolved. The solution is given
into a sample receptacle placed in a borehole-NaI-detector where the
characteristic gamma radiation of U-235 and the radiation emitted by the
uranium decay products are measured. Since besides the characteristic
radiation of U-235 (187 keV) also the radiation of the daughter products
is measured, the measured value of the characteristic uranium radiation
must be corrected accordingly.

For this purpose, the gamma spectrum to be measured is split into different energies in a multichannel analyzer (resolving power about 7% relative to the corresponding gamma energy) and the counting rates are summed up in the range of characteristic U-235 radiation. In the energy ranges adjacent to the characteristic uranium radiation two windows are placed in which the background radiation of the uranium daughter products is measured. Then the uranium counting rate is corrected accordingly so that the net counting rate is proportinal to the U-235 content. The system is calibrated by means of NBS or EC standards. Acquisition and correction of measured values are performed on a minicomputer /1/.

Table 1: Accuracy of measurement of U-235 assay in solution

U-235 concentration in powder (wt.%)	Measurement error (2 σ) (wt.%)
0.2 ÷ 3.0	0.015
3.0 ÷ 4.0	0.020
4.0 ÷ 5.0	0.045

Immediately before compaction the enrichment of the powder is controlled a second time. For this purpose, the filler is preceded by a chamber to which a. NaI detector has been flanged which is shielded by lead against background radiation. The chamber is so designed that it is alsways filled with UO_2 powder at the same level.

The detector signal is amplified and split in a multichannel analyzer according to the different energies. The resolving power of the NaI detector is again approx. 7%. The spectrum is evaluated on a microcomputer / 1 /. For automated energy calibration a weak Am-241 source has been installed in the NaI detector which emits a reference radiation of 59.9 keV. The software seeks this peak and calibrates with it the energies of the

various channels so that the characteristic radiation of U-235 at
187 keV is always exactly found. The system is calibrated in quanti-
tative terms by means of a second powder chamber having the same geo-
metry and filled with UO_2 powder of known enrichment. The accuracy
(2 σ) for the range from 0.2 to 5.0 wt.% U-235 is ± 0.05 wt.% U-235
for a measuring time of 500 seconds per measured value.

The behavior of the powder when conveyed through pipes or poured is
described by the "flowing behavior." Coefficients describing the fill
of the dies are the bulk density and the tap density. No standardized
control methods exist as yet for these controls so that the individual
firms have developed control methods which satisfy their requirements
to the best possible extent. However, it is to be expected that within
the foreseeable future uniform rules of measurement will be developed
to standards by DIN or ISO /2/.

For powders fabricated under the same method the BET surface consti-
tutes a good measure of the sintering capability. When the surface is
determined according to Brunauer, Emmet and Teller a gas, normally nitro-
en, is deposited on the surface of the sample at the temperature of li-
quid nitrogen. Under this method, the surface can be estimated in a
good first approximation from the amount of gas adsorbed, assuming a
monomolecular layer, or, alternatively, several adsorption isotherms
are recorded which provide information about the structure of surface
density. With this additional information it is then possible to calcu-
late more precisely the surface of a sample /3/.

In case of variable powder sources knowledge of the specific surface
is not sufficient to make statements about the behavior of the powder
during the process of fabrication. Supplementing assays must be perfor-
med. Mostly the particle sizes are subject to analysis.

3.2 Sieving Behavior

The powder particles visible under the optical microscope are mostly

agglomerations made up of much smaller particles, the primary particles. The primary particles can be recognized only under the electron microscope, above all in case of UO_2 ex ADU. Consequently, a "particle size analysis" is normally an analysis of size of the secondary particles formed by agglomeration. This type of analysis is performed in most cases by sieving. Besides, the powders are generally sieved before they are processed. This sieving ist mostly a vibration sieving. If powders must be processed which are highly susceptible of agglomeration sieving by means of vibration sieves poses problems because in the course of sieving new agglomerates are continuously formed which quickly become greater than the meshes of the sieve. Sieving stops altogether as soon as this point is reached.

The remedy consists in using an air jet sieve. The air jet sieve relies on the principle that the woven wire screen is first flushed by an air stream directed upwards which originates in a slit type nozzle rotating below the screen. The feed is maintained in a free flying movement. Then, the direction of the air stream is reversed, which means that it passes the meshes from top to bottom and entrains that material which is finer than the mesh. The coarse material is left on the screen and weighed at the end of the separation process. The fact that the screen is repeatedly flushed from bottom to top implies that the feared blinding of meshes is avoided / 4 /. Moving of the feed in an air stream also prevents the particles from agglomerating.

Figure 2 / 5,6 / shows plots of the sieving behavior of the air jet sieve compared to a vibration sieve. The feed was UO_2 ex AUC. It can be seen that practically no more sieving effect is attained if a 20 µm screen bottom is used in the vibration sieve w ilst the same powder can still be sieved well on the same sieve bottom exposed to an air jet. This becomes equally clear when the 40 µm sieve bottom is used. Whilst in the vibration sieve a residue of about 12% is left after approx. 100 minutes which can no longer be sieved, the powder passes the sieve within about 5 minutes under the same conditions leaving a residue of about two to three percent. This simple comparison makes evident the
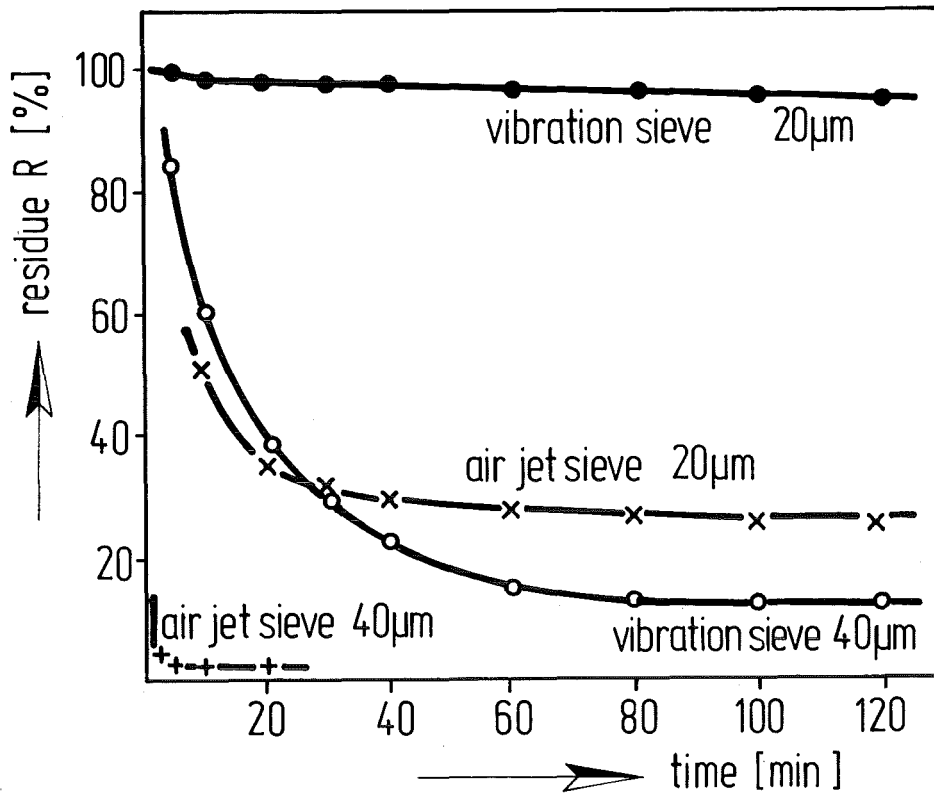
Fig. 2: Comparison vibration sieve vs. air jet sieve for UO_2 ex AUC powder

advantages offered by the air jet sieve. But also with the air jet sieve
the measured values obtained for the grain size distributions cannot al-
ways be regarded as absolute measures of size distribution of the secon-
dary grains since the permeability plots greatly depend on the surface
loading of the sieve bottom. Figure 3 /5,6/ shows this phenomenon by a
series of measurement in which a 20 μm sieve bottom was loaded with
25, 50, 100 and 200g UO_2 ex AUC. It can be recognized that the residue left
after 120 minutes of sieving varies between 20% (25 g) and 30% (200 g).
Under the same conditions the amount left in a vibration sieve varies
between 95% (25 g) and 99% (200 g). It is evident from these results of
measurement that an exact sieving analysis is neither feasible with the
help of an air jet sieve if powders with a high tendency to agglomeration
are to be sieved. However sieving of the powder prior to further processing
is at least ten times faster in the air jet sieve and implies much smaller
amounts of residues.
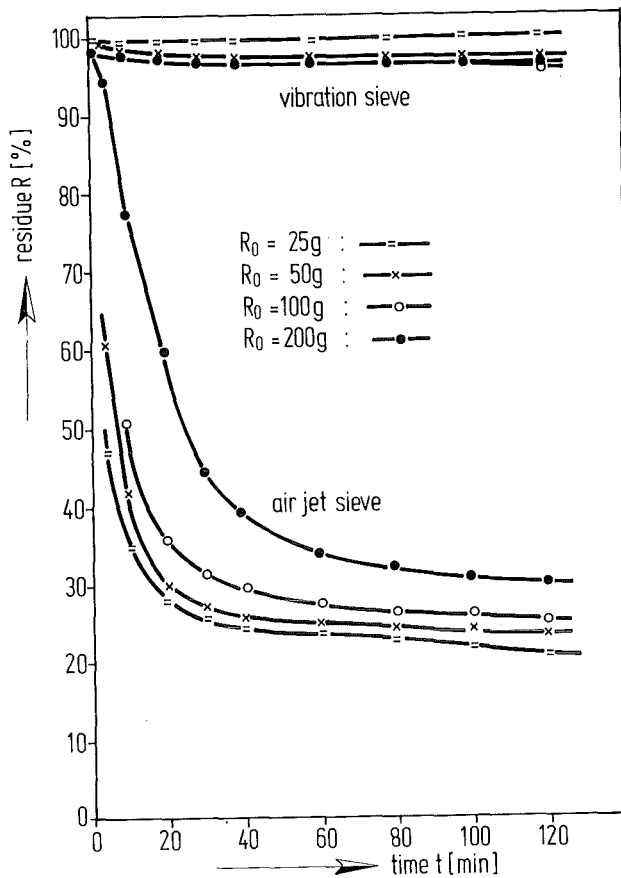
Fig. 3: Comparison vibration sieve vs. air jet sieve for UO_2 ex AUC powder. Influence of sieve loading R_0

But the situation in characterizing powders still waits for a satisfactory solution; even if an air jet sieve is used. We have not decided for determining new absolute data of powder characterization because of the high expenditure to be expected. We have rather tried to make available reference parameters to the manufacturer who certainly possesses a wealth of experience concerning the behavior of individual powder qualities. The said parameters allow to state which batch, for which experience has already been collected, is similar to a new one. The comparison is made by means of image analysis on the basis of the PACOS image processing system / 7 /.

3.3 Powder Characterization by Optical Means

The structural particularities of the powder particles which determine the sintering and flow behavior cannot be ovserved normally under the optical microscope because these structures can be detected exclusively

by an electron microscope. Therefore, the secondary grains of the powder were selected for determination in the optical analysis. The following aspects were in favor ot this selection:

- The secondary grains are usually so large sized that they can be observed and measured without problems using an optical microscope.

- The secondary grains in terms of shape and size reflect the agglomeration behavior of the powder. Since the agglomeration behavior greatly depends on the morphology of the primary grains, it provides, together with the specific surface, an indication of the sintering capability.

Consequently, an appropriate measuring method must be capable of measuring the different powders by grain size and grain shape and compare the measured values obtained with the values applicable to other powders. If the reference parameters are conveniently selected, an assignment is possible.

Powder classification by shape and size uses a technique which had been developed for the PACOS image analysis system / 7 /. The powder particles are compared with templets (structuring elements) having different shapes and sizes. If now the image transformation "erosion" / 7 / or "opening" / 7 / is performed in the images of the preparations for these templets and the area is subsequently measured, one obtains a vector of the measured value for each of the templets. A method of assignment was developed / 8 / according to which an unknown powder is assigned to one of several specimen powders known. However, the specimen and the sample must be measured under the same conditions. Obviously, the specimen powder must be measured only once because the measured values are stored in a computer as a "specimen library."

For testing the efficacy of the templets and the significance of assignment special procedures were developed. The theoretical background of this classification procedure is described in /8/.
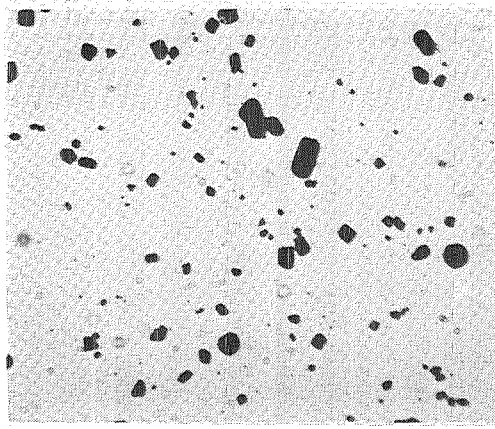
UO₂ ex AUC
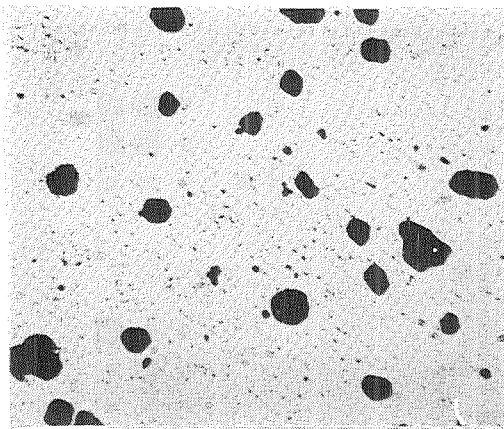
UO₂ ex ADU
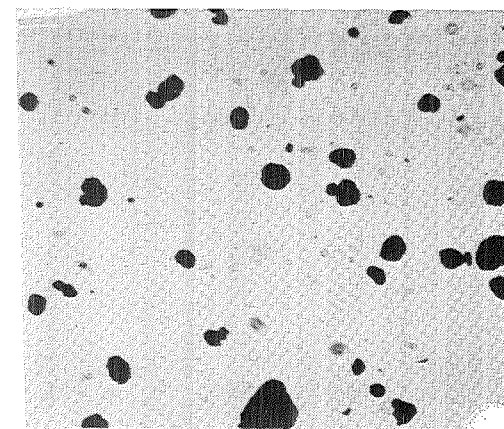
UO₂ ex ADU 30'
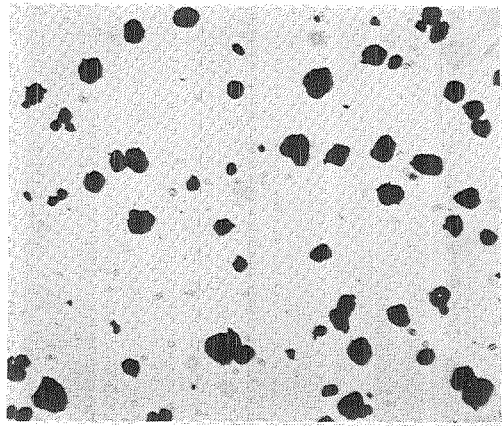
UO₂ ex ADU 60'
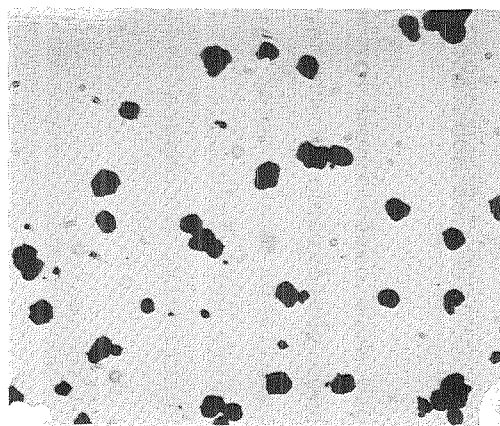
UO₂ ex ADU 150'

Fig. 4: Set of test powders used for powder characterization

In the following table a typical example of practical application will be given which was used to test the separating power of the method. The following powders were used:

Specimen no.	Powder type		
1	UO_2 ex AUC		
2	UO_2 ex ADU		
3	UO_2 ex ADU	30 minutes at 900 $^\circ$C, annealed in CO_2	
4	UO_2 ex ADU	60 minutes under above conditions	
5	UO_2 ex ADU	150 minutes under above conditions	

These powders were conditioned for microscopic analysis.

Figure 4 shows photographs of these preparations as used for measurement. To obtain results of measurement which are reproducible in an unambiguous way, the powders were focused automatically / 9 /. Of each specimen and sample, respectively, a comparatively large number of image fields were measured.
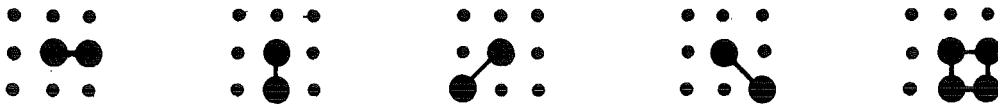
Fig. 5: Set of templets used for powder characterization

Figure 5 shows the set of structuring elements used for evaluation. For this preparation the vectors of measured values were determined and stored in a file in conformity with the selected set of structuring elements. Subsequently, a second series of preparations were produced. In the assignment experiments both specimen series were measured.

Table 3: Assignment of unknown test powders by use of d_k values. The unknown specimen is assigned to that test powder, for which the discriminator d_k is maximum

UNKNOWN SPECIMEN NO.	STANDARD SPECIMEN NO.				
	1	2	3	4	5
1	3.2×10^{-1}	4.4×10^{-9}	4.9×10^{-8}	3.5×10^{-10}	0.0
2	3.1×10^{-5}	3.4×10^{-1}	1.7×10^{-2}	3.4×10^{-2}	0.0
3	0.0	0.0	2.1×10^{-1}	1.1×10^{-2}	1.4×10^{-1}
4	0.0	0.0	2.2×10^{-1}	5.2×10^{-1}	7.0×10^{-3}
5	1.3×10^{-5}	0.0	7.6×10^{-3}	1.0×10^{-3}	2.3×10^{-1}

Table 3 shows the values of the dicriminator d_k for assignment of these five test samples to the individual classes. It can be seen that the assignment is correct. Even if the assignment is obviously correct, the significance of assignment must be examined in a detailed evaluation / 8 /.

4. Testing of Pellets

4.1 Test on Whole Pellets

After sintering and grinding of the pellets the geometry and density must be examined. Since these are very frequent measurements it is necessary to use a fully automated device. In this context, the following parameters have to be measured: diameter, length, dishing, volume, perpendicularity, chamfer width and height. The first four of these attributes can be measured by an automated device. Additionally, the "geometrical" density of the pellet is calculated from these four values. As an example for such an automated device, an instrument used at the RBU workshop at Hanau will be described:

The device consists of three measuring gauges and a small robot arm moving the samples from gauge to gauge. The gauges are placed on a circle with the robot arm in its center. The robot takes a pellet from a magazine and moves along the whole circle perimeter. First, the pellet is placed into the diameter gauge where the diameter is measured by three sensors at three different spots of the pellet surface. In the next step the pellet is placed into a combined meter which measures the pellet length, the dishing depth and the perpendicularity. The length and dishing depth are measured by normal path sensors, the perpendicularity is measured by a cardanically suspended plate with four induction coils on its back side. The last gauge is a small balance.

These gauges are connected to a microprocessor which controls all gauges and the robot arm, takes over the measurement values, and monitors the results. The computer calculates the diameter from the three values furnished by the sensors and compared it to the tolerances specified. From the dishing depth it calculates the dishing volume.

Figure 6 depicts a scetch of the entire measuring facility. Figure 7a shows the station for diameter measurement and Figure 7b the station for measurement of pellet length, dishing depth and perpendicularity.

In addition, the computer needs the value of the chamfer width which is measured independently by a mircoscope and has to be entered manually into the system. This measurement is carried out at the beginning of each shift and a new measurement is necessary only if the punch is changed during the shift.

From these geometrical data and the pellet weight the computer can calculate the "geometrical" density. All values obtained are compared automatically with the valid values in the respective specifications and drawings. In case of deviations the system demands either repetition of measurement (second sample) or scrapping of the material involved. All data accepted are transferred to a connected central computer for data storage and final documentation.
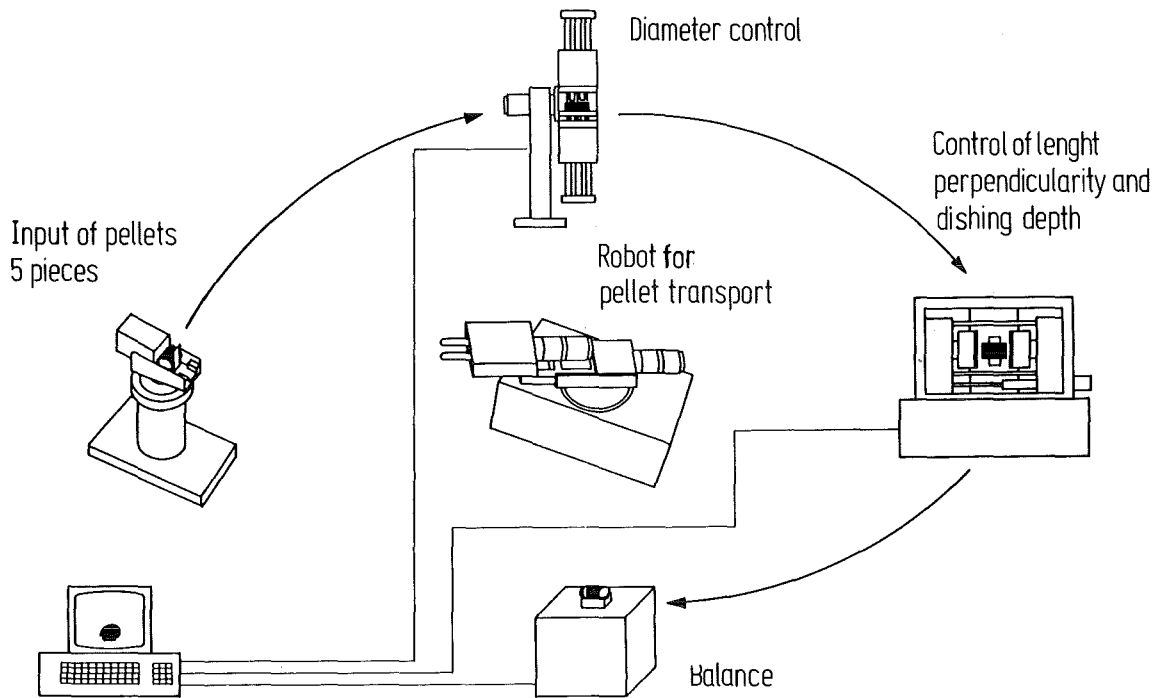
Fig. 6: Automatic control of pellet geometry and density at RBU Hanau
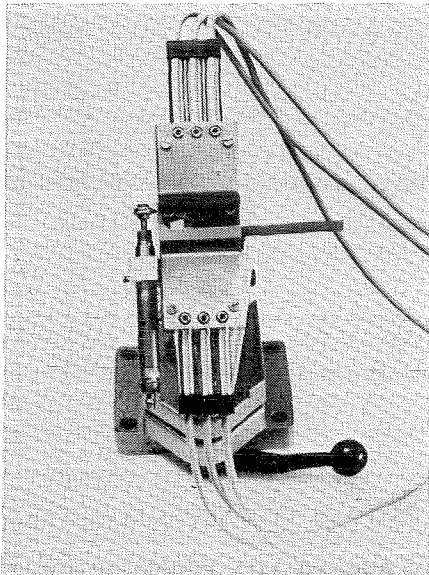
Control of diameter
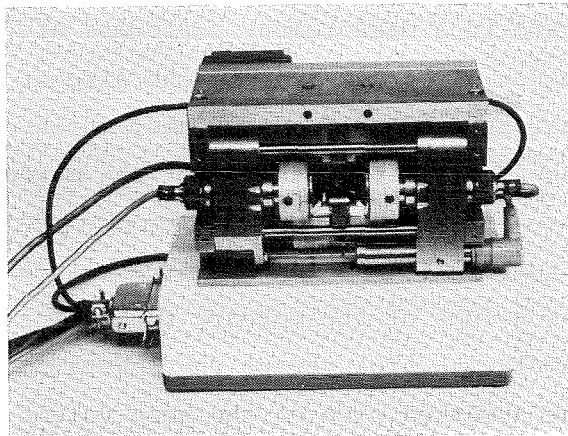
Control of lenght
perpendicularity and
dishing depth

Fig. 7a and 7b: Automatic device for determination of pellet geometry

The pellet density is defined either as the "geometric density" or density of lift. Since the geometry of the pellets can often be measured but inadequately on account of chamfer and dishing, the density of lift is determined in nearly all cases. Generally, water is used as the buoyancy medium. The accuracy of density determination is 0.2% (2σ) in case the buoyancy method is used; the error in the determination of the geometric density is approximately twice as high /10/.

Another possibility of determination of the density is by assessing the volume using a mercury pycnometer or in connection with mercury porosimetry, a method by which also the pellet volume is determined.

Mercury porosimetry, in addition, provides information about the fractions of open and closed porosities.

Under this method mercury exposed to high pressure is forced into a specimen evacuated in advance. A pore size distribution can be determined from the pressure applied and the volume of mercury forced into the specimen. The mercury forced into the pores at given pressure does not completely fill the pores, but approximates the pore shapes to a surface characterized by the minimum radius of the mercury meniscus. This radius corresponds to the smallest channel into which mercury can penetrate in a measurement performed at the pressure. This means ecactly spoken, we are measuring the size of the channels inbetween the pores and that part of the pore volume, which can be accessed by channels corresponding to the pressure of the mercury.

Figure 8 /11/ reflects the results of such a measurement. In this diagram the volume of the mercury forced into the open porosity was plotted in a cumulative plot versus the logarithm of the pore size. In this example we see a marked rise of the pore volume accessible from the outside in the range of meniscus radii between 0.2 and 0.1 µm and in the range below 0.01 µm. If the mercury pressure is reduced after the maximum pressure has been attained, the mercury, exposed to the pressure applied by its surface tension, is again forced out of the pores. If one carries out an
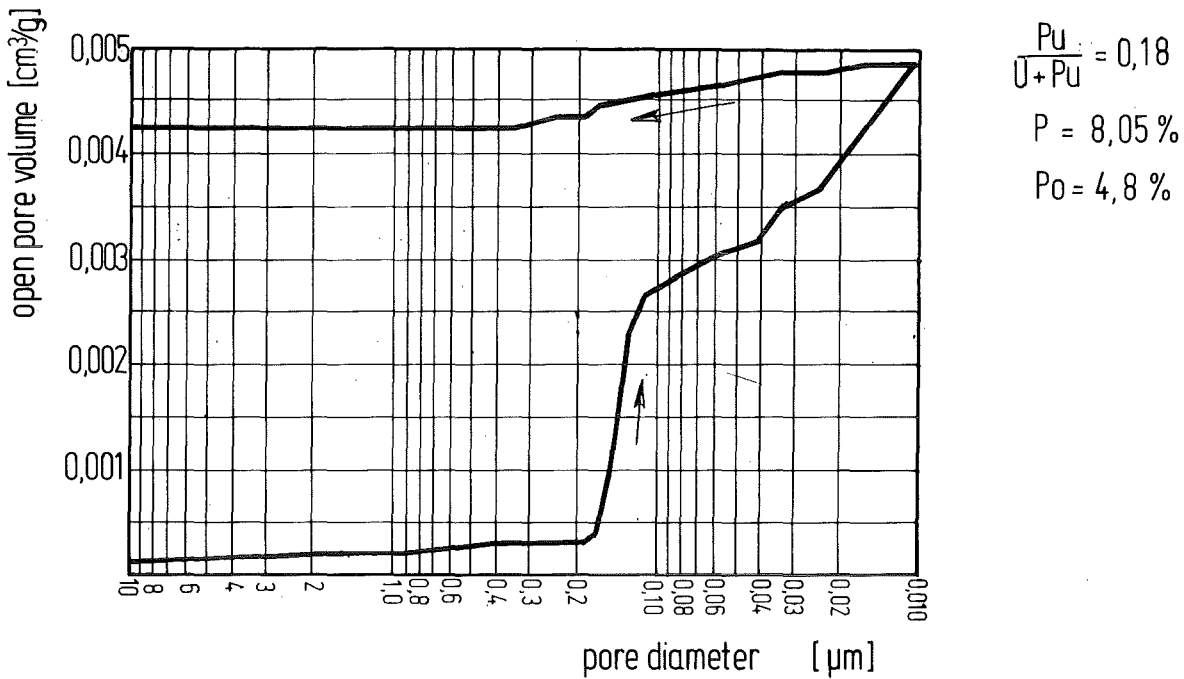
Fig. 8: Typical result of mercury intrusion porosimetry measurement

experiment of this kind, one recognizes pronounced hysteresis between
the pore volume when the pressure is increased and the pore volume asso-
ciated with pressure reduction. This allows conclusions to be drawn with
respect to the pore structure.

A significant estimate which can be made from this hysteresis consists
in the discrimination between the volume of pores accessible from outside
and the volume of the channels connecting these pores. Mercury porosimetry
can also be used to detect cracks and other surface flaws. These volumes
can be discriminated quite easily from those of the pores because they
are voids with dimensions normally not encountered in pellets / 11 /.

From the travel time of elastic waves which can be measured conveniently
and quickly the velocity of elastic waves can be calculated using the
known linear sample dimensions. This velocity in turn depends on the elas-
tic properties of the workpiece. In case of longitudinal elastic waves the
following relation is established between the Youngs's modulus E and the
rate of propagation (" sound velocity").

$$v_L = \sqrt{\frac{E}{p}}$$

where p is the density of the sample. If the measurement is performed
with transverse waves the Young's modulus in the relation above must be
replaced by the shear modulus. On account of the close relationships
existing between the velocity of elastic waves and the structure this
measurement is particularly suited to characterize porous samples because
these properties greatly depend on the porosity and the shape of the pore
/ 12 /. By such measurements both green compacts and sintered bodies can
be studied.
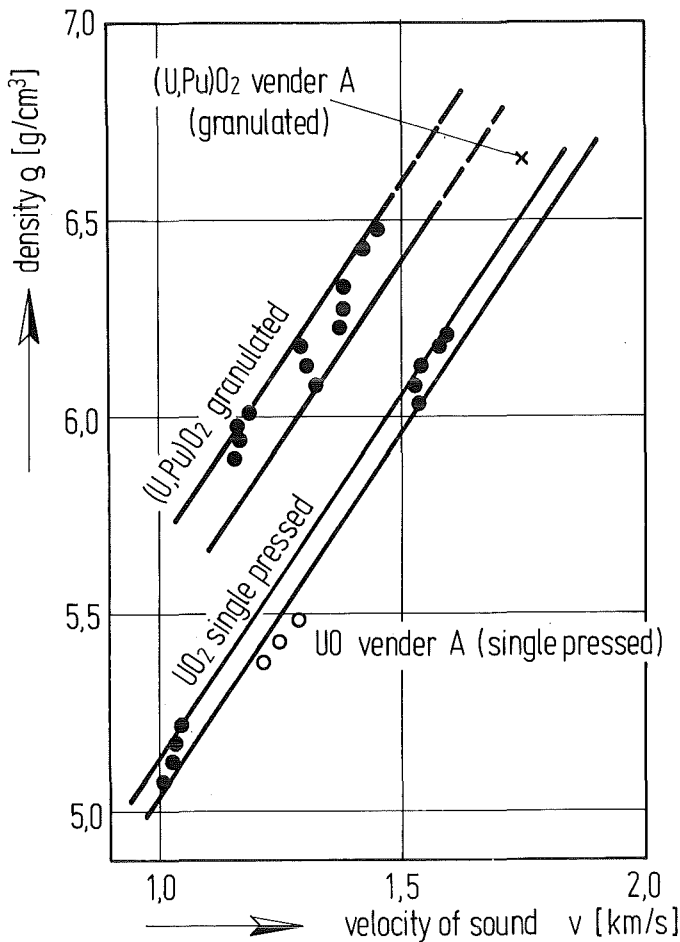
Fig. 9: Velocity of sound vs. density for green pellets made from different powder preparation

Figure 9 / 5,6 / shows the dependence of the velocity of sound on the
density of the sample. In this diagram the measured values have been
combined for two types of green compacts, namely green compacts fabrica-
ted from granulated powder ((U, Pu)O$_2$ granulated) and from non-granulated

powder (UO$_2$ single pressed). It can be noticed that the measured values for the two types of green compacts can be combined into two distinctly separated scatter bands. The difference between these two types of sample is attributable primarily to the different structures and not to the differences in plutonium content. The difference in sound velocity of these two sample types can be explained by the fact that samples having rather spherical pores have a higher sound velocity than the samples with rather flat pores. Experience shows that pellets fabricated from granulated powder possess a greater fraction of flat or tetrahedral pores. Although some samples from a different manufacturer do not lie within thes scatter bands, they, basically, follow the same laws. It can be noticed that the modifications of any parameters during the manufacturing process which lead to a change in structure can be detected with great sensitivity by measurement of the sound traveling time.

Similar statements can be made for sintered sample material as well. Figure 10 /5,6/ shows the same as Fig. 7a sound velocity-density diagram for sintered samples. It is apparent that similar to the situation encountered for green compacts, a scatter band exists in which samples occur which have been fabricated according to one single fabrication method. If samples of other manufacturers are considered, one sees that some of them lie very distant from this scatter band. For evaluating the homogeneity and reproducibility of a fabrication technique it is interesting to find out that the sound velocity scatters within a wide band for samples A, B, C, D, E fabricated according to the same method. Whilst the samples D and E lie within the traced scattering band, the sound velocity of the samples A, B, C is clearly lower. A reduced sound velocity indicates rather oblong pores. In the case considered here the reduction of the sound velocity is an indicator of a more or less high fraction of microcracks in the sample. The sample F also supplied by the manufacturer A but fabricated under a different method, likewise occurs within the scattering band of granulated U, PuO$_2$ pellets.

These simple examples show the high sensitivity with which a qualitative structural analysis can be performed by measurement of the sound velocity.
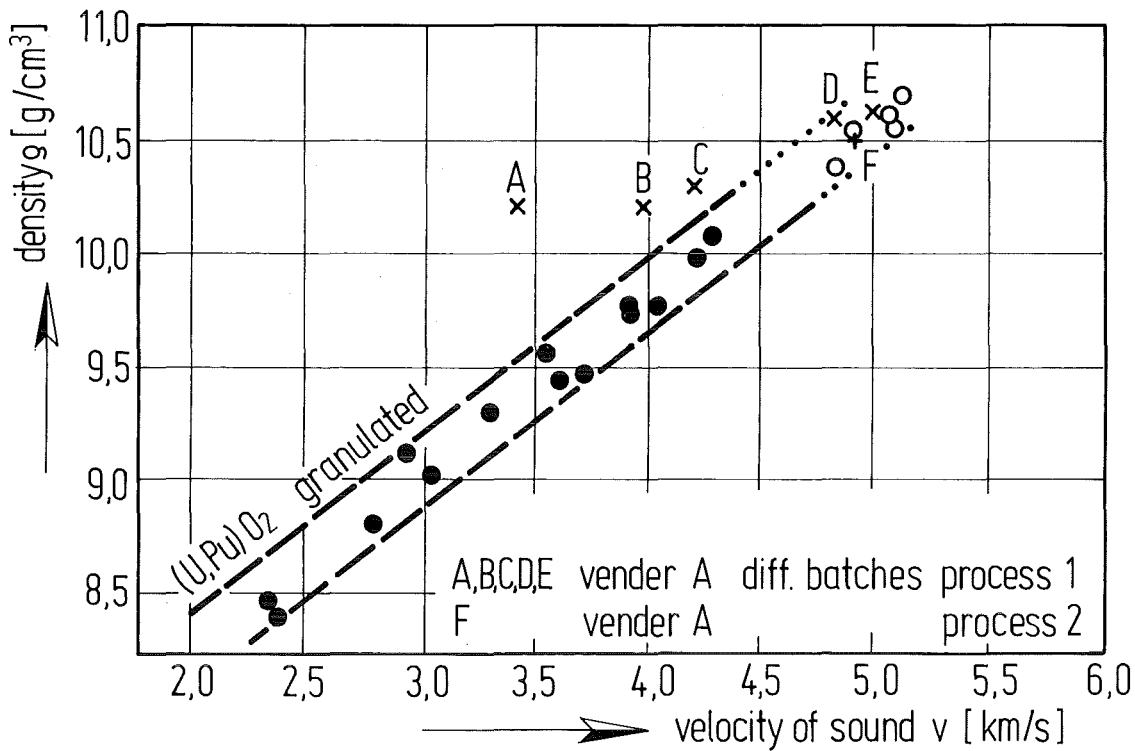
Fig. 10: Velocity of sound vs. density for pellets made of granulated powder

If one considers in addition that only few seconds are required to make such a measurement, the potential of this measuring method for quality control can be assessed.

4.2 Controls of Sections of Pellets

For ceramographic investigations of the pellet structure the sample is cut into two halves. To obtain a reliable statement about structure, both longitudinal and transverse sections of the samples must be investigated. By ceramographic methods and subsequent image analysis the following properties of the pellets can be determined:

- homogeneity of the structure,
- grain sizes,
- pore sizes,
- presence of flaws (cracks, inclusions).

Whilst it is helpful to use unetched samples for the determination of the pore sizes, grain sizes can be measured only on etched samples. The grain sizes are described by either chemical or by thermal etching. If the grain boundaries in the samples prepared ceramographically have been developed in a reliable manner, the problem of size determination of pores and grains are rather identical. The basic problem is that a size per se does not exist for these two-dimensional bodies. Every size description is directly linked to a shape. The shapes used for description are capable of approximating but coarsely the shapes occuring in nature. A proven and theoretically well founded description of the shape of such pores or grains consists in the determination of the largest inscribable gauge of a defined pattern /13, 8/. If this is made with different patterns the size distribution of these patterns in the sample can be determined. If counting of particles by different classes of sizes and patterns is adequate for the respective purpose, classification is peformed by different steps of the image transformation "erosion". If areas are to be measured enabling to determine volumetric fractions, one relies on the transformation opening or templet matching. If, as a matter of fact, surfaces are to be estimated, only templet matching is left as a method of image transformation. In the following paragraphs the practical application of the analysis of the pore structure of samples will be explained which had been fabricated by two different techniques. In this analysis a distinction is to be made between the nearly tetrahedral pores present between large powder particles (granulate grains) and the rather spherical pores. This disdrimination allows to influence specifically the fabrication process because the pore shapes are typical of different steps in pellet fabrication.

To make a distinction between rather spherical and rather tetrahedron shaped pores, this analysis can be performed once with a triangular and once with a square templet /8,14 /. For triangular patterns it must be consideres in addition that in a discrete Cartesian coordinate system four congruent equilateral rectangular triangles exist which can be converted into each other by rotation. These four possible orientations must be taken into account in the measurement. After the measurements have been made with triangles and squares, the individual results of measurements have to be assigned to these figures.

Two assumptions can be made for this separation. Assuming that the shapes of the pores can be described rather well by triangles or squares, one supposes subsequently that in each figure into which a square can be inscribed also a triangle of equal size can be inscribed (assumption A). This means that these triangles cannot be counted. However, also a square of half the lateral length can be inscribed in each triangle. Since in this case the square is the smaller figure, it is eliminated from counting. If the figures cannot be so well assigned, the assumption B is made. On this assumption the figure is evaluated as being a square in case one can inscribe either a triangle of equal edge length or if the area of an inscribable square is greater than that of the inscribed triangle. Consequently, figures having the same area constitute the limit case.

Figure 11 shows measured results obtained from a sample made of granulated powder. In the left part of the figure the rough measured data are entered with the results of the four orientations of the triangles added up. In the right part the classification by shape and size has been entered. Since approximated triangular pores have to be treated here, the assumption A could be used for evaluation.

We see that in the sample under consideration the greatest pores are rather triangular than square. If we compare these results with measurements obtained from single-pressed material, we recognize major differences. The results of measurement shown in Figure 12 could be evaluated only by making the assumption B, which means that the given shapes do not agree well with that of the pores. In this figure the dominance of the rather round pores is clearly apparent. The fact that the small pores are rather triangular pores might not be a very significant result, since in this range of sizes the falsification of the pore shape by the given point screen might already be considerable.

The measurements described before enable the manufacturer to adjust the hardness of the granulate and the compacting pressure in such a way that for granulated powder a specified pore structure is obtained.
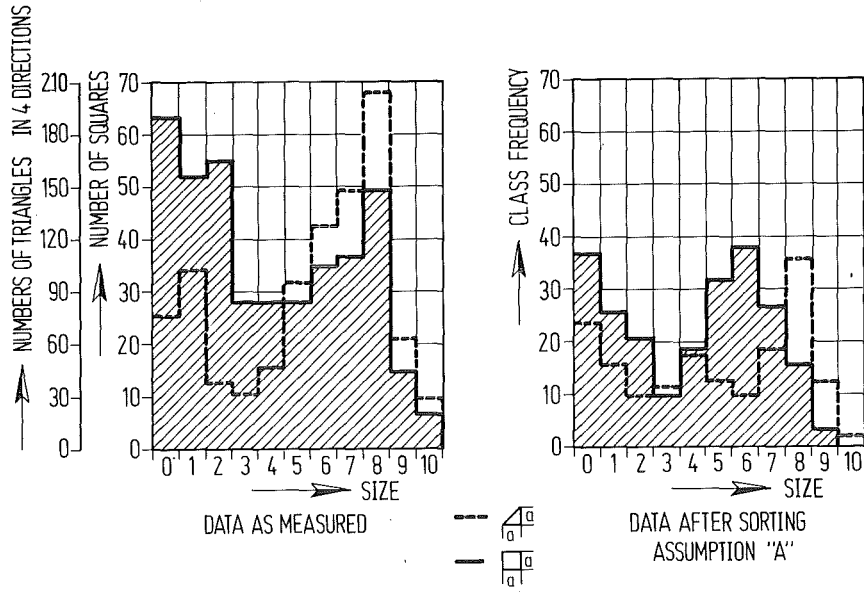
Fig. 11: Classification of pores of a pellet by size and shape (granulated powder)
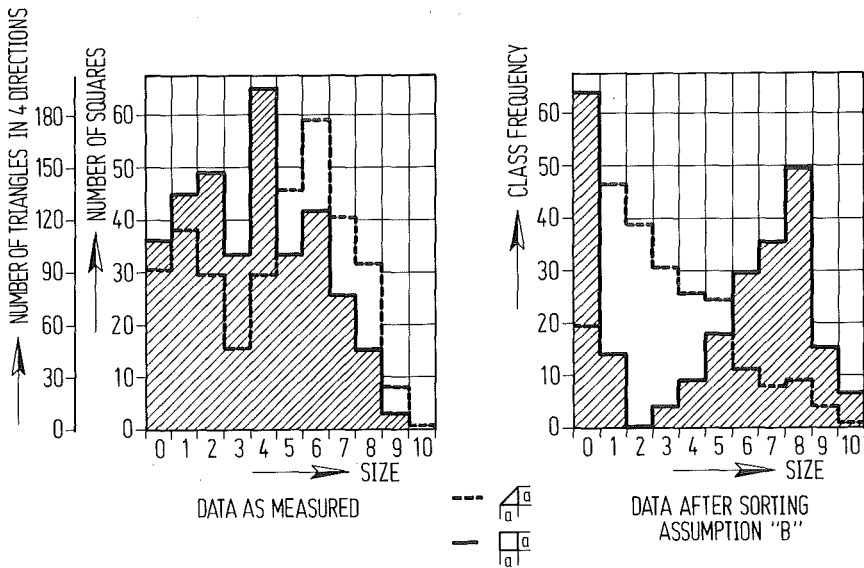
Fig. 12: Classification of pores of a pellet by size and shape (single pressed powder)

By comparison of the pore size distribution determined by means of a nearly circular gauge with the results of mercury porosimetry interesting relations can be established between the pore sizes and the channels connecting the pores / 11 /.

Problems encountered in LWR fuel with plutonium as the fissile material concern the distribution of the plutonium in the pellets. Since these problems are particularly important in fast breeder fuel, they are preferably discussed in publications dealing with these subjects. The problems of solid solution formation are of major importance in this context; they are studied by means of X-ray diffraction techniques /15, 16/. Problems relating to the size of zones rich in plutonium are investigated by evaluation of alpha autoradiographies by means of image analytical evaluation / 11 /.

5. Open Questions

The number of problems which cannot be handled by suitable measuring and testing methods is very low.

One of these problems is the control of pellets for freedom from flaws. At present, possible surface flaws such as spalling and chipoff are compared visually with the help of surface standards. This method is not satisfactory because the person applying it gets tired and the results of this control are normally not uniform. Automated recognition by image processing techniques would be desirable. Such a control system could read and control in addition the label indicating the percentage of enrichment.

References

1   H. Eberle, P. Matussek, I. Michel-Piper, H. Ottmar
Elemental and Isotopic Concentration Analysis on
Nuclear Fuels Using Nondestructive Assay Techniques
J. Nucl. Mat. 81 (1979) 203 – 214)

2   H. Assmann   private Communication (1983)

3   P.H. Emmett   Catalysis Vol I; Fundamental Principles
Book Division Reinhold Publ. Corp. (1956)

4   O. Sauer   Feinheitsmessungen an technischen Stäuben
Alpine A.G. Augsburg (1963)

5   H. Wedemeyer   private Communication (1982)

6   D. Vollath   Charakterisierung von Mischoxidbrennstoff-
tabletten und deren Vorprodukten   in
Schnellbrüter Brennelemententwicklung KfK 3505 (1983) 57 – 69

7   D. Vollath   The Image Analysing System PACOS
Prakt. Metallographie 19 (1982) 7 – 23, 94 – 103

8   D. Vollath   Classification of Stochastic Specimen by a Set
of Standard Specimen
Acta Stereologica 2 Suppl. I (1983) 117 – 120

9   D. Vollath   Automatic Focusing of Every Image Point of an
Image
United States Patent 4, 350, 884 (1982)
Europäische Patentschrift 0017726 (1983)

10   W. Dörr, H. Assmann, G. Maier, J. Steven
Bestimmung der Dichte, offener Porosität, Porengrößenverteilung
und spezifischer Oberfläche von UO_2 Tabletten
J. Nucl. Mat. 51 (1979) 135 – 141

11 D. Vollath Methods of Characterizing Fuel Structure
    J. Nucl. Mat. 81 (1979) 115 - 136

12 J. Boocock, A.S. Furzer, J.R. Mathews
    The Effect of Porosity on the Elastic Moduli of UO_2
    as Measured by an Ultrasonic Technique
    AERE - M' 2565 (1972)

13 P. Delfiner A Generalization of the Concept of Size
    J. Microscopy 95 (1972) 203 - 216

14 D. Vollath Optical Methods for Application in Nuclear
    Fuel Characterization
    J. Nucl. Mat. 106 (1982) 71 - 80

15 H. Riella-Gracher, D. Vollath Röntgenographische Be-
    stimmung der Mischkristallverteilung in UO_2 - PuO_2 Tabletten
    J. Nucl. Mat. 106 (1982) 93 - 100

16 H. Riella-Gracher Ermittlung der Plutoniumkonzentrations-
    verteilung in $(U, Pu)O_2$ mit Hilfe der röntgenographischen
    Analyse
    KfK 3619 (1983)

DISCUSSION

V. GORSKY: Why do you measure the velocity of elastic waves in the pellets? From the slide one can see that the error is much too high in the density range of 10.2 - 10.6 8/cm^3.

D. VOLLATH: The scattering band indicated on the slide was due to a small deviation in pellet structure and not due to a measuring error. This slide indicated – as I pointed out – the advantage of the methods described for monitoring changes in pellet structure.

A. STRASSER: To what extent can the powder shape and size characterization be used, for product sinterability?

D. VOLLATH: Using such a powder characterization technique, the band in which sintering esperiments have to be performed in the case of a new unknown powder is smaller.

K. BALARAMA MOORTHY: Considering the importance, are there any plans to develop appropriate specifications and test methods for determining or assessing chippability of uranium oxide pellets.

D. VOLLATH: Until now we have not tried to develop methods for testing the chippability.

R. POLLIS: How long has the automatic equipment for pellet dimension control been used in RBU?

H.J. von WACHTENDONK: For one year.

DETERMINATION OF THE U-235-CONTENT
IN UF$_6$ WITHIN THE SHIPMENT CONTAINER

H.-J. von Wachtendonk, R. Baumann
Reaktor-Brennelement Union GmbH,
Rodenbacher Chaussee 6, 6450 Hanau 11
West Germany

Abstract:
RBU controlles all received UF$_6$-cylinders for the U-235-
content by means of a direct measurement, through the
steelwall of the shipment container. This measurement
is now performed for about two years in routine practice.
More than 400 UF$_6$-containers have been measured up to now.
The results of these measurements and the obtained precision
are presented in this paper.

DETERMINATION OF THE U-235-CONTENT
IN UF$_6$ WITHIN THE SHIPMENT CONTAINER

H.-J. von Wachtendonk, R. Baumann

Reaktor-Brennelement Union GmbH,

Rodenbacher Chaussee 6, 6450 Hanau 11

West-Germany

RBU produces UO$_2$-powder and pellets in a large scale. Normally, the uranium is fed in form of UF$_6$ to the process. This UF$_6$ is received from various enrichment plants. Each delivery is accompanied by a certificate in which all necessary date concerning the UF$_6$ are maintained: gross weight, net weight, U-content, U-235-enrichment, content of impurities etc. From these date the weight and the U-235-enrichment are controlled.

The U-235-determination at the receiving of the UF$_6$-delivery shall not be a precise measurement, but only a mix up control. The precision we wanted to achieve with this, new method should be of the same order with which we had determined the U-235 content with our previous applied method.

Description of the previous applied method

After receiving of the UF$_6$-container a gaseous sample had to be taken. The valve was connected to some cooling traps cooled by liquid nitrogen. The cooling traps were evacuated and purged with dry nitrogen gas to avoid decomposition of the UF$_6$ to non volatile UO$_2$ F$_2$ which may clog the gas pipes.

Subsequently, the valve of the UF$_6$-container was opened and a gaseous sample was taken. Especially in winter time when the environmental temperature was low this procedure took a large time because of the low vapour pressure of the UF$_6$.

Additionally, because the cooling traps consisted of glass the sampling apparatus could easily break. This could endanger the operator by liberatiing hazardous gas (hydrogen fluoride) and by contamination through uranium.

After sampling the sample had to be prepared for the measurement: hydrolysis and conversion to AUC (ammonium uranyl carbonate), calcination to U_3O_8 and/or dissolution to uranyl nitrate. The U_3O_8 powder or the uranylnitrate solution were taken for analysis. These steps cause, of course, the danger of nitroducing analytical mistakes or systematic errors. These disadventages are avoided by the new method.

Description of the new method and the apparatus used for it

After receiving of UF_6 container it is weighed immidiately. During the weighing procedure the enrichment check is performed. For this purpose the radiation of U-235 (187 keV) is measured by an intrinsic germanium detector. First, the container surface has to be prepared. Varnish and rust have to be removed by grinding completely. This is necessary to result a good ultra sonic measurement of the wall thickness.

The wall thickness has to be measured to correct the weaking of the U-235 radiation by the steel wall of the container. Normally, the wall thickness varies between 12.5 mm and 13.5 mm, but there are some containers, too, which have a wall thickness of about 11 mm.

The wall thickness is determined with an ultrasonic gauge which measures the thickness with an precision of $\pm$ 0.01 mm (1 s). This precision can't be improved further as inhomogeneities within the steel wall limit the precision. All types of cavities and cracks within the steel wall give wrong results in thickness measurement. In this case another spot at the cylinder surface has to be prepared.

The prepared measurement area has a diameter of about 10 cm. At five spots of this area the wall thickness is determined and the mean value of these five measurements is taken for correcting the radiation weakening by the steel wall.

In the next step the detector is positioned to the prepared area. A collimator is mounted to the detector and both are installed in a mavable container so that a fixed geometry is obtained. This is very important because small deviations in the geometry result in a remarkable measurement error. When all these steps are finished the measurement can be started.

The enrichment analyzer is placed in the chemical laboratory of RBU, the detector unit at the weigh scale, about, 180 m away from the analyzer. The detector is connected to it by a double shielded wire and doesn't need any intermediate amplifier between the detector and the steering unit.

As enrichment analyzer the system TN 4000 from Tracor Northern is used. It contains a fast amplifier, a multichannel analyzer and an electronic data processor which operates all parts automatically.

Calibration

To calibrate the system 60 UF_6 containers of different and well known U-235-content were used ranging from 0.71 % up to 3.80 % U-235. The enrichment within those cylinders which were used for calibration was check three times each: certified value supplied by the enrichment plant, γ-spectrometric determination by the chemical laboratory of RBU (gaseous sampling) and mass spectrometric analysis by Alkem Company. The certified value was taken as the true value. The both analytical determinations should confirm it.

Check of drifts

A special container containing UO_2-pellets in a plastic matrix can be connected to the detector so that the same geometry at each check is guaranted. This special device should result the same countrate at each measurement. If the result deviates from the expected value the system has to be recalibrated. For this purpose one or more UF_6-containers of known U-235 content are taken depending on the calibration range which has to be recalibrated.

Additionally, time by time the system is checked with an UF_6-cylinder of known U-235 content (every one or two month).

Results

Up to know more than 400 UF_6-containers have been measured by this method with sufficient results. It is not the aim to measure the true enrichment, but only to avoid an enrichment mix up and to detect UF_6-containers of deviating U-235 content. The deviation limit we can tolerate is + or - 0.05 % U-235 content (absolute value). This detection limit has to be achieved with 2 sigma confidence. Table 1 shows that normally this goal is achieved. Some standard deviations are a little bit high, but this depends on the small number of evaluated data.

There is a good linear correlation between the measured and the certified values. The ideal equation would be

$$y = x$$

We have obtained following equation:

$$U\text{-}235_{certified} = 0.009 + 0.998 \; U\text{-}235_{measured}$$

with a correlation coefficient of

$$r = 0.9952$$

which indicates a quite good correlation,

There is a small positive bias whose origin we don't know. We expect that the radiation correction function which considers the shielding by the steel wall is not valid over the full range of wall thickness we have measured. (10.5 mm up to 13.8 mm). We cannot improve the ultra sonic wall thickness determination. The value of 0.01 mm (1 s) at a wall thickness of about 13 mm is quite excellent. So we will try to improve the radiation correction function. We have found a weak correlation between the wall thickness and the U-235 content. At smaller wall thickness there is a tendency to higher enrichment values and, vice versa, at thicker walls there is a tendency to lower enrichment values. We reduce this result to a non ideal radiation correction function.

Our aim that the measured value deviates less than 0.05 % of U-235 content from the certified value over the whole U-235-enrichment range (from 0.2 % up to 5 % U-235 content) with 2 sigma confidence is achieved. The measured absolute deviation is nearly independent from the certified enrichment, that means, that the relative error increases with lower U-235-enrichment. It exists a logarithmic relation between the relative error and the U-235 enrichment.

$$\log \frac{U\text{-}235}{U\text{-}235_{cert.}} \cdot 100 = \log 0.36 - 0.16 \cdot U\text{-}235_{cert.}$$

with a correlation coefficient of

$$r = 0.960$$

This means, that the relative error decreases from 2.3 % at relative to depleted uranium to 0.48 % relative to 4.3 % enriched uranium.

Table 1:

certified value [% U-235]	mean value [% U-235]	standard deviation
0.2005	0.2070	0.0162
0.7110	0.7229	0.0134
1.6046	1.6086	0.0224
1.9186	1.9293	0.0206
2.1087	2.1039	0.0226
2.4006	2.4104	0.0205
2.9502	2.9508	0.0236
3.0009	3.0117	0.0331
3.1984	3.2214	0.0129
3.2431	3.2282	0.0213
3.3023	3.2772	0.0180
3.3527	3.3534	0.0272
3.4372	3.4207	0.0236
3.5078	3.5162	0.0288
3.8027	3.8232	0.0214
3.9646	3.9646	0.0329
4.0625	4.0621	0.0256
4.2583	4.2622	0.0230

mean value of standard
deviation: 0.0226

linear correlation between measured and certified value:

$$U\text{-}235_{cert.} = + 0.009 + 0.998 \; U\text{-}235_{meas}$$

correlation coefficient:

$$r = 0.9952$$

1. R.B. Walton
   The Feasibility of Nondestructive Assay Measurements in
   Uranium Enrichment Plants
   LASL Rep. 87545 Apr. 1978

2. R.B. Walton, T.D. Reilly, J.L. Parker, J.H. Menzel
   Measurements of UF_6-Cylinders with Portable Instruments
   LASL, Nuclear Technology Val. 21 Febr. 1974

3. J.W. Rowson, S.A. Honzeas
   Uranium Isotopic Ratio Determination by Gamma-Ray
   Spectrometry
   Nuclear Instruments and Methods 154 (1978) 541-48

4. E. Dermendjiev, N. Beyer
   Non-destructive Gamma-Ray Enrichment Measurement
   of large UF_6-Cylinders
   IAEA - STR - 67

5. E. Dermendjiev, N. Beyer, D.E. Rundquist
   Non-destructive Measurements of UF_6 Stored in 5" and
   30" Cylinders
   ESARDA 1st Symposium
   Apr. 1979, Brussels

6. H.J. von Wachtendonk
   Determination of the U-235-Content within the UF_6-
   Shipment Container
   Procedings of the 5th ESARDA Symposium
   Apr. 1983 Versailles

CHARACTERIZATION OF NUCLEAR FUEL BY NEUTRON - INDUCED
AUTORADIOGRAPHY BASED ON RELIEF AND DYED IMAGE FORMA-
TION IN POLYMERS

R. Ilić, M. Najžer, D. Sušnik, M. Humar, A. Loose

J. Stefan Institute, E. Kardelj University
of Ljubljana, Jamova 39,
61 000 Ljubljana, Yugoslavia

A B S T R A C T

A new neutron induced autoradiographic technique for the determina-
tion of uranium and gadolinium in nuclear fuel is developed. This
technique is based on enhanced dissolution or dyeing abilities of
polymers exposed to high fluence of charged particles. Upon irradi-
ation by neutrons of fuel samples which are in close contact with
the polymer detector a latent image is formed in the detector. This
image consists of radiation damage tracks of fission fragments and/
or internal conversion electrons emitted by ^{235}U (n,f) and ^{157}Gd (n,
γ) reaction respectively. A visible relief or dyed image is formed
by the soaking the irradiated polymer foil in pure water or water
containing an organic dye. Investigations showed that an image of
superior quality is obtained with respect to the track-etch techni-
que. The method was applied to the determination of the inhomogene-
ity of (U, Gd) O_2 fuel. The required thermal neutron fluence using
pure gelatine as an image detection material, was found to be bet-
ween 10^{15} and 10^{16} cm^{-2}. The spatial resolution of the method was
found to be about 5 μm.

INTRODUCTION

The most important element for use as a burnable poison is gadolinium (1-4). Mixing of Gd_2O_3 in UO_2 fuel pellets to form a solid solution of (U, Gd) O_2 has been used for many years to provide a burnable poisons in BWR-s and is currently getting an increased attention for application in PWR-s. Fuel element designer and modellers assume that a perfect homogeneous mixture of Gd in UO_2 is formed. In practice this however may not represents the real situation. During fuel pellets manufacturing is therefore very important to control Gd inhomogeneity which should meet exceed acceptable limits. In addition during the irradiation of a fuel pin, depending on dose rate, fuel temperature, oxigen-to-metal ratio gadolinium may be redistributed both radially and axially. The information about the redistribution of Gd and its burn-up is very important to modellers studying the behaviour of fuel during the irradiation as well as to fuel reprocessors. While pure Gd_2O_3 particles can be detected by the variety of the methods, Gd content variations in the solid solution matrix is less easy or rather more costly to characterize (1-2).

Recently new Neutron Induced Autoradiographic (NIA) techniques based on relief and/or dyed image formation in polymers were developed (5-7). These techniques are based on enhanced dissolution or dyeing abilities of polymers exposed to a high fluence of charged particles emitted during the irradiation by neutrons. It was shown that isotopes which have high cross section for (n, p), (n, α) and (n, f) reactions as well as for (n, γ) reactions followed by internal conversion or β-decay can be mapped (6). In the present work the applicability of the techniques for the determination of inhomogeneity in (U, Gd) O_2 fuel using a pure gelatine as an image detection material was investigated. The response of the gelatine to fission fragments and internal conversion electrons via ^{235}U (n, f) and ^{157}Gd (n, γ) reaction was measured and the optimal conditions to produce relief and/or dyed image were determined. Comparison of these techniques with NIA and α-autoradiography utilizing solid state nuclear track detectors is given and illustrated by a few autoradiograms of (U,Gd) O_2 fuel.

P R I N C I P L E S

In this technique the fuel pellet to be examined is polished by a standard ceramographic procedure and a foil of a suitable polymer is placed against this surface. The pellet plus polymer foil is then irradiated to a high neutron fluence, Fig. 1a. This can be achieved by irradiation in the nuclear reactor core. After irradiation the latent image is transformed to relief image by soaking the irradiated foil in water, Fig. 1b.
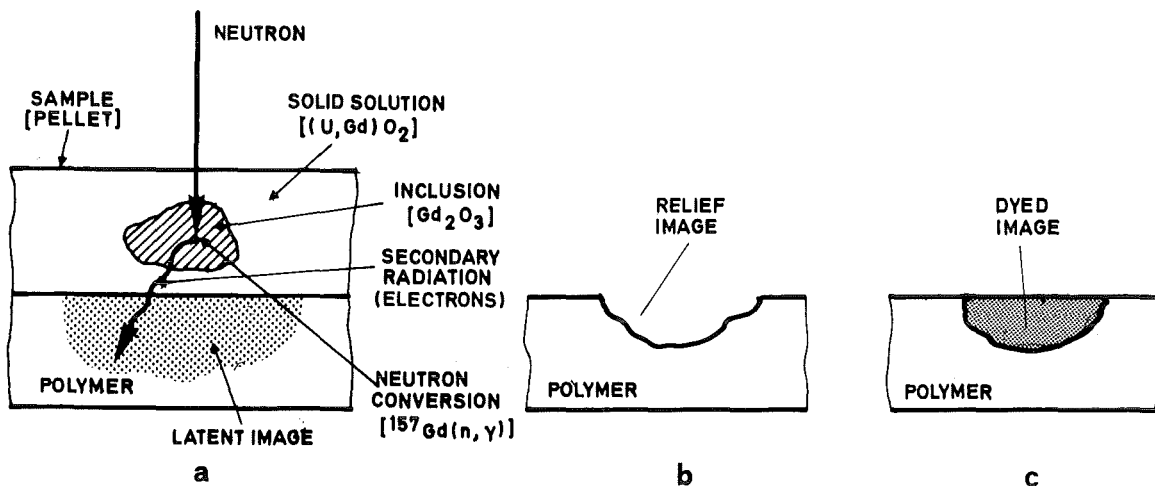
Fig. 1: Principle of the method:
a) irradiation
b) relief image after soaking in water,
c) dyed image after soaking in an organic dye

If the fluence of the charged particles is approximately one order of magnitude less than that required for a relief image a dyed image is obtained by soaking the irradiated polymer foil in an organic dye, Fig. 1c. The mechanism of a dyed and relief image formation is described elsewhere (6-8). In our previous work (6-7) it was shown that the response of the given detector depends on the type and energy of particles. Since the required fluence of electrons to produce a relief or dyed image is approximately two orders of magnitude greater than that of heavier ions (7), gadolinium concentration variation in nuclear fuel pellets at the proper exposure can be observed as a variation of residual foil thickness or optical density (Fig.1).

Naturally ocuring gadolinium has two isotopes ^{155}Gd and ^{157}Gd, having very high neutron capture cross section of 61000 and 250000 b respectively (12). Reaction products emit gamma rays and conversion electrons. These internal conversion electrons are initially mo-

noenergetic.However those actually emitted by the specimen of fini-
te thickness will have a continious energy distribution. Conversion
electrons of 11 different energies have been identified following
neutron absorption in gadolinium (9). Over half of these electrons
have an initial energy in the range 70 to 80 keV (9). The range of
these electrons is about 6 µm in the fuel and 40 µm in a polymer
foil. Thus using a thin polymer foil the spatial resolution of about
5 µm can be achieved by the imaging of gadolinium. The range of a fi-
ssion fragments emitted by ^{235}U (n, f) reaction is smaller indicating
that an image of better spatial resolution can be obtained by the
imaging of uranium.

E X P E R I M E N T A L

Materials

Gelatine 40 µm thick, backed on a 100 µm thick polyester base/produ-
ced by Fotokemika Zagreb/, was used as the image detection material.
For comparison autoradiographs were also made by some solid state nu-
clear track detectors: LR-115, CR-39 an Makrofol KG.

The response of the gelatine to neutron-induced charged particles was
measured by the irradiation of thin samples of gadolinium and uranium.
For this purpose 25 µm thick Gd-foil and 93 % enriched uranium fission
foil were used. To illustrate the applicability of the new techniqu-
es for the characterization of nuclear fuel, UO_2 - Gd_2O_3 pellets with
different Gd_2O_3 contents (from 0-10 wt %) were prepared. In order to
get different Gd_2O_3 distribution in the samples two different process-
es were used. In the first one, Gd_2O_3 was presintered at 1500°C for
2 h and crushed. Then UO_2 powder was added and green cylindrical pe-
llets (∅ = h = 8 mm) were obtained by mixing with alcohol, drying
at 100°C and pressing at 300 MPa. Pellets were than sintered in hydro-
gene atmosphere at 1700°C for 4 or 24 h. Using this process inhomoge-
neous mixtures of UO_2 and Gd_2O_3 were obtained. The second process was
similar to the first one. Only Gd_2O_3 was not presintered and sinte-
ring was made at 1800°C for 3 h. This led to homogeneous mixture of
(U, Gd) O_2 .

Irradiation

All the irradiations in NIA experiments using gelatine as an image detection material were performed in the core of the TRIGA Mark II reactor at the J.Stefan Institute, using aluminium pressure type cassettes. The thermal neutron flux was 4×10^{12} cm^{-2}s^{-1} and the gold Cd ratio about 4. The neutron irradiation in NIA experiments with solid state nuclear track detectors was performed at the exposure room of our reactor. The thermal neutron flux was 8.9×10^{7} cm^{-2}s^{-1} and Cd ratio about 40.

Image Processing

A relief image was obtained by soaking the irradiated gelatine in H_2O at 20°C for several minutes. For a dyed image, the water contained 0.1 wt % of methylene blue. Latent image obtained on solid state nuclear track detectors was transformed to a visible image by chemical etching. LR-115 detector was etched in 2,5 N NaOH at 60°C for 70 minutes. The etching conditions for CR-39 and Makrofol KG were 6.25N/NaOH/70°C/30 min and 5.5 N/ KOH/60°C/25 min respectively.

The optical density of the dyed image was measured by a microphotometer using 2 x 0.02 mm scanning beam. The depth difference between the irradiated and unirradiated region was measured by a TESATRONIC electronic lenght measuring device.

RESULTS

The response of the detector for a dyed image is expressed as the optical density, D, defined as:

$$D = \log \frac{Io}{I} \qquad (1)$$

where Io is the intensity of incident light and I is the intensity of that transmitted by the detector.

The response of the detector for a relief image is expressed as the normalized residual thickness, h, defined as:

$$h = 1 - \frac{h_i}{h_o} \quad ; \quad R \geq h_o$$

$$h = 1 - \frac{\Delta h_i}{R} \quad ; \quad R \leq h_o \tag{2}$$

where h_o is the thickness of the detector foil before irradiation, h_i is the residual foil thickness, R is the range of the particles and $\Delta h_i = h_o - h_i$ is the thickness of the dissolved material.

The optical density and the normalized residual foil thickness vs neutron fluence Φ_n, for gelatine - Gd foil combination is shown in Fig.2.
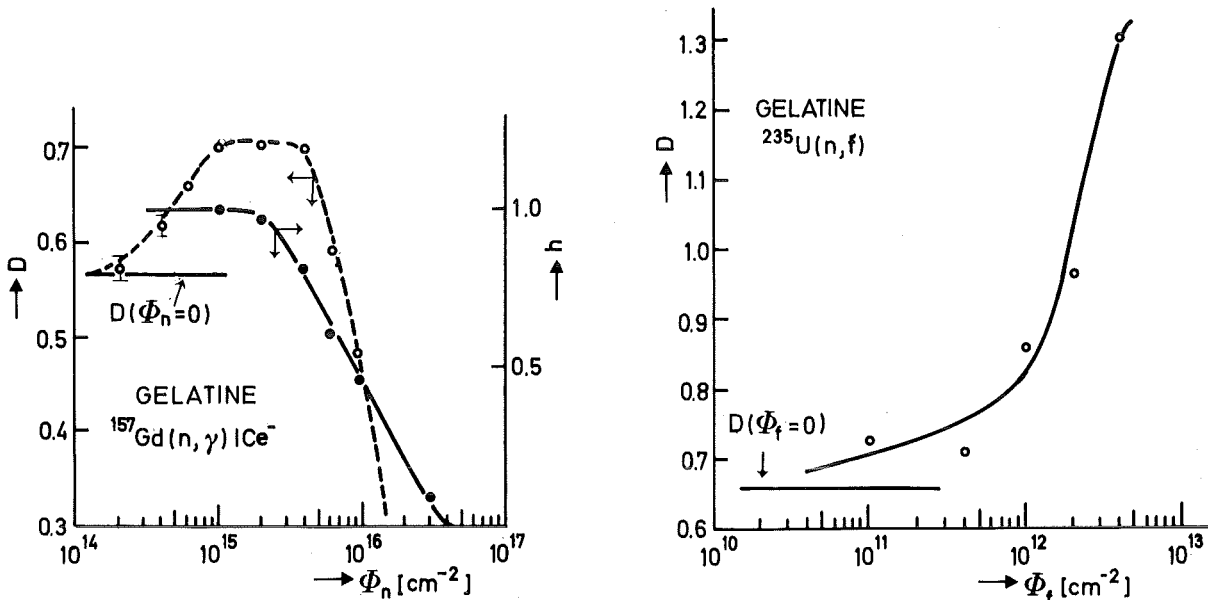
Fig.2: Optical density, D, and residual foil thickness, h, vs neutron fluence, Φ_n, for gelatine - Gd (25 µm) combination

Fig.3: Optical density, D, vs fluence of fission fragments, Φ_f

From this figure it can be seen that first a dyed image is formed at a neutron fluence of about 10^{14} cm^{-2}. The optical density increases with increasing fluence up to $\sim 2 \times 10^{15}$ cm^{-2}, which is a threshold value for the formation of a relief image. Due to the dissolution of the gelatine the optical density descreases at higher fluences. From the Fig.2 one can see the following characteristics of a relief image i) at low neutron fluences ($< 2 \times 10^{15}$ cm^{-2}) there is no observable dissolution effect, ii) in the range of neutron fluence from about 2×10^{15} to 2×10^{16} cm^{-2} there is a very steep variation in the dissolution rate of irradiated area, iii) finally the effect reaches satu-

ration at the neutron fluence of about 5×10^{16} cm^{-2} when the removed layer thickness is equal to the range of the conversion electrons.

Optical density vs fluence of fission fragments Φ_f is given in Fig.3. From this results it can be seen that a high contrast of dyed image can be achieved by fission fragments. Our experiments showed that the minimum fluence of fission fragments to produce a relief image is about 5×10^{12} cm^{-2}.

The overall internal conversion coefficient (number of electrons emitted per neutron absorbed) for natural gadolinium was found to be 0.6 (11). Since the number of fission fragments emitted per neutron absorbed is two, the question arrise what is an optimal neutron fluence to produce an image from which Gd_2O_3 and UO_2 particles can be observed in (U, Gd) O_2 fuel. The neutron fluence required to produce relief or dyed image on gelatine for different fuel phases is summarized in table 1. This results shows that Gd inhomogeneity in (U,Gd) O_2 fuel can be measured from a relief autoradiogram obtained at neutron fluence from 10^{15} to 10^{16} cm^{-2}, which is too small to produce an uranium image.

Table 1: Neutron fluence, Φ_n, required to produce relief and dyed image on gelatine by the conversion electrons and fission fragments

$$\Phi_n{}^{1)} \ (cm^{-2})$$

Material	Reaction	Dyed image	Relief image
Gd	$^{157}Gd \ (n,\gamma)ICe^-$	$10^{14} - 10^{15}$	$>10^{15}$
Gd_2O_3		$5 \times 10^{14} - 5 \times 10^{15}$	$>5 \times 10^{15}$
(U, Gd) O_2 10at % Gd		$4 \times 10^{15} - 4 \times 10^{16}$	$>4 \times 10^{16}$
U	$^{235}U \ (n,f)$	$3 \times 10^{16} - 3 \times 10^{17}$	$>3 \times 10^{17}$
UO_2		$10^{17} - 10^{18}$	$>10^{18}$

1) assuming that material thickness is greather than the range of internal conversion electrons and/or fission fragments

Feasibility of the new techniques for the determination of inhomogeneity in (U, Gd) O_2 fuel and its superiority over track etch detectors is illustrated in Fig.4,where microstructure of UO_2-Gd_2O_3 nuclear fuel is presented. Optical micrograph (Fig.4a) shows that inhomogeneous mixture of UO_2 and Gd_2O_3 is formed. α-autoradiogram (Fig.4b) and neutron induced fissiogram (Fig.4c and 4d) showes that the bright areas in an optical micrograph corresponds to crystal grains of Gd_2O_3.
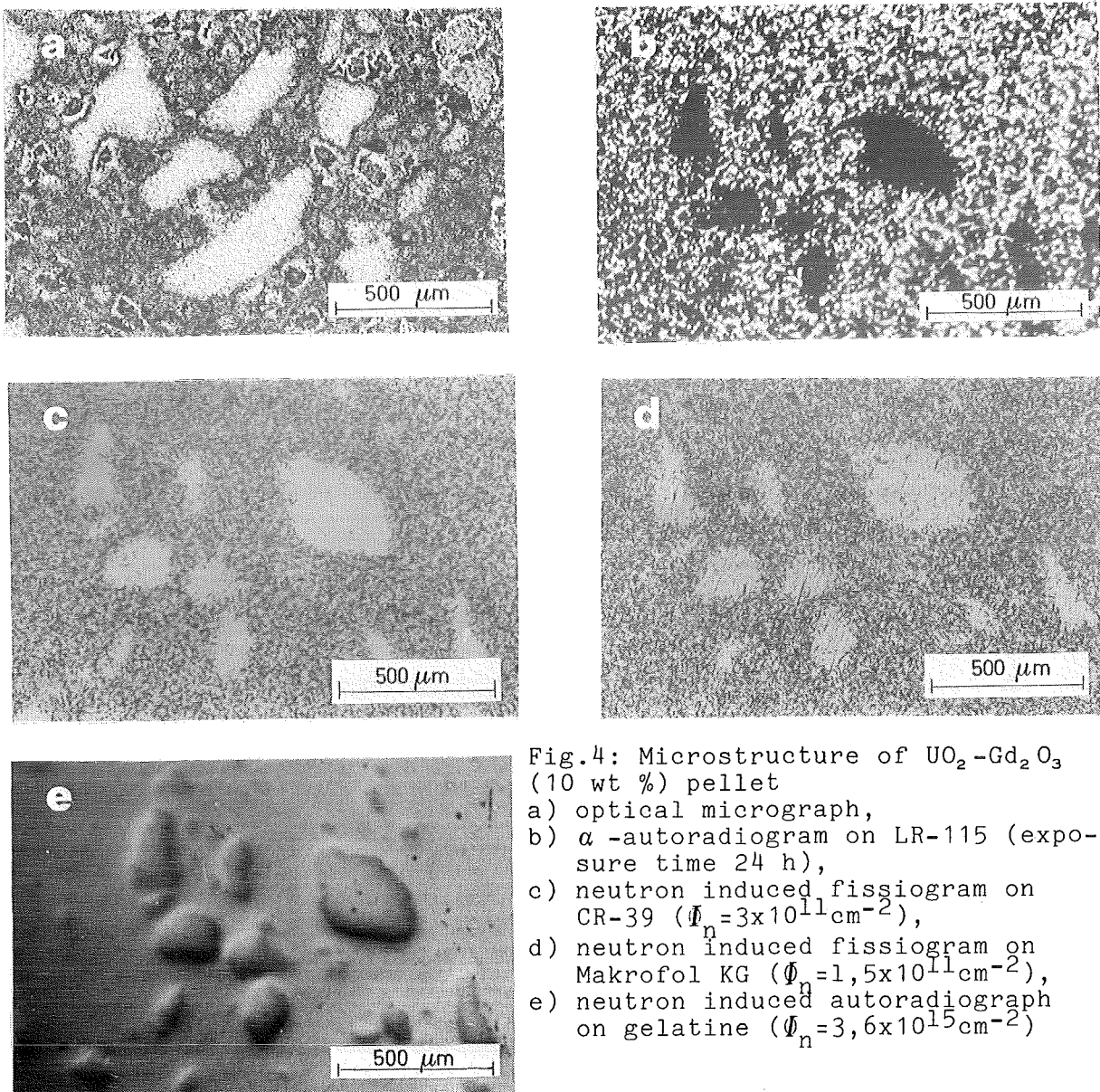
Fig.4: Microstructure of UO_2-Gd_2O_3 (10 wt %) pellet
a) optical micrograph,
b) α -autoradiogram on LR-115 (exposure time 24 h),
c) neutron induced fissiogram on CR-39 ($\Phi_n=3 \times 10^{11} cm^{-2}$),
d) neutron induced fissiogram on Makrofol KG ($\Phi_n=1,5 \times 10^{11} cm^{-2}$),
e) neutron induced autoradiograph on gelatine ($\Phi_n=3,6 \times 10^{15} cm^{-2}$)

On the relief image (Fig.4e) very small inclusions of Gd rich pha-
ses can be clearly observed due to the much better spatial resolution
and detail discernment of the new technique.

CONCLUSIONS

NIA based on relief and dyed image formation in polymers is a method
complementary to other autoradiographic techniques.Our preliminary
work showes that this method has advantage in comparison with other
autoradiographic techniques based on application of solid state nucle-
ar track detectors and X-ray films, because of the following:

i) it is more universal. Burnable poisons which have high cross sec-
   tion for (n, α) as well as (n, γ) reaction followed by internal
   conversion or β -decay and fissionable materials can be mapped,

ii) it is selective. Response depends on the type and the energy of
    particles and the detector material

iii) it is practically insensitive to γ -rays, and

iv) it has much better spatial resolution and detail discernment.

The present paper reports the concept of the method and a first
attempt for application in the characterization of nuclear fuel. Fur-
ther studies, both theoretical and experimental, are in progress.

REFERENCES

1. D. Vollath, K.Kummerer, Eds., Characterization and Quality Control of Nuclear Fuels, J.Nucl.Mater., 81 (1979)

2. Ibid, 106 (1982)

3. R.A.Murgatroyd, B.T.Kelly, Technology and Assesment of Neutron Absorbing Materials, Atomic Energy Review, 15, 1 (1979) 3

4. B.Frost, Nuclear Fuel Elements, Pergamon Press, Oxford, 1982

5. M.Najžer, M.Humar, R.Ilić, Microautoradiography with Gelatine, Nucl.Tracks Suppl. 3 (1982), 77

6. A.Loose, I.Hunyadi, G.Somogyi, Zs.Varga, M.Najžer, R.Ilić, Neutron – Induced Autoradiography Based on Relief and Dyed Image Formation in Polymers, Nucl.Tracks and Rad.Meas. 8 (1984), in press

7. G.Somogyi, I.Hunyadi, R.Ilić, A.Loose, Zs.Varga, Formation of Ion-Beam Images in Gelatine and Various Polymers, Ibid.

8. A.Loose, Neutron Induced Microautoradiography Based on Relief and Dyed Image in Polymers, Diploma Work, E.Kardelj University of Ljubljana, Ljubljana, 1984

9. M.R.Hawkesworth, Neutron Radiography: Equipment and Methods, Atomic Energy Review 15, 2 (1977) 169

10. A.A.Harms, G.R.Norman, The Role of Internal Conversion Electrons in Gadolinium – Exposure Neutron Imaging, J.Appl.Phys. 43, 7 (1972) 3209

11. A.A.Harms, G. McCormack, Isotopis Conversion in Gadolinium – Exposure Neutron Imaging Nucl.Instr.Methods 118 (1974), 583

12. D.Van der Hardt, H.Röttger, Neutron Radiography Handbook, D.Reidel, Dordrecht, 1981

DISCUSSION

K. BALARAMA MOORTHY: I am rather surprised to see a slide projected by you showing free gadolina in UO_2 - 8% Gd_2O_3. Is this made specially for your experiments? Are there any test results on other sintered pellets?

R. ILIC: This specimen was specially prepared for the illustration of the capability of the method to study the inhomogeneity of $(U, Gd)O_2$ fuel. We got also some results on fully sintered pellets. If we have a homogeneous mixture, i.e. a perfect solid solution of $(U, Gd)O_2$ relief and dyed image cannot be obtained.

A. STRASSER: Do you believe that neutron sources, like Pu-Be or californium, can be used for the irradiation?

R. ILIC: The required thermal neutron fluence to obtain relief image is between 10^{14} and 10^{16} cm^{-2}. In the case of a dyed image this value ranges from 10^{13} to 10^{15} cm^{-2}. Using Pu - Be or Cf 252 neutron source it would be rather difficult to get the required neutron fluence in a reasonable irradiation time.
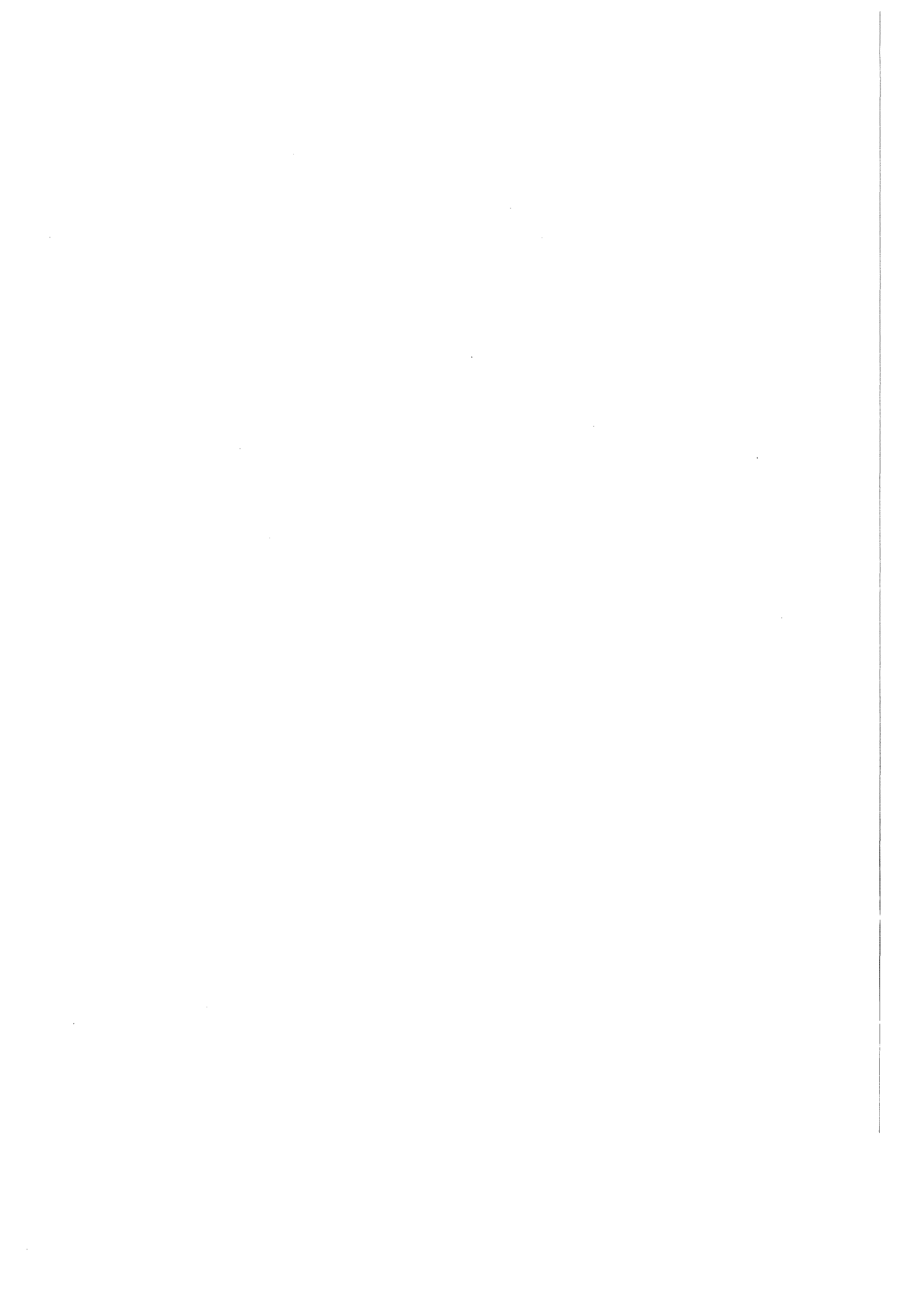
Fabrication and quality control of mixed oxide LWR fuel with
regard to homogeneity of fissile content

K. Gruber, G. Latzel, H. Roepenack

ALKEM GmbH, D-6450 Hanau 11, FRG

A process for the fabrication of a soluble mixed oxide fuel is
described.

Introducing this new technique some quality characteristics were
found which were unknown at that time. Improvements of the test
methods and analysis of test results lead to modifications within
the production process which is now well under control.

Introduction

For the production of U-Pu-Mixed Oxide Fuels the distribution of
plutonium is an essential feature of quality.

Generally this item is double specified, i.e. in terms of the
macroscopic and microscopic distribution.

The macroscopic distribution is described in most specifications
as Pu-content per length whereas the microscopic distribution is
characterized by the mean and maximum particle diameter where the
Pu-content is also taken into account.

Under these aspects a maximum particle diameter of approximately
300 µm is allowed for the Pu-bearing component containing 30 w/o
Pu-fiss.

Another recently imposed demand refers to the solubility of U-Pu-oxide in HNO_3. It has been proved by experiments that U-Pu-mixed crystals with a Pu-content of about 35 % - according to latest trends - are soluble in HNO_3, contrary to pure Pu-oxide.

Such soluble mixed crystals can either be produced chemically by precipitation of U-Pu-mixed crystals or physically by mixing and intense milling and successive sintering.

ALKEM has developed special manufacturing processes for both possibilities, the OCOM- and AUPuC-process. Here the experiences will be reported with the mixing/milling process, the so-called OCOM-process (= optimized co-milling process).

Up to now approximately 5000 fuel rods for light-water reactors with more than 10 t mixed oxide-pellets have been produced according to this procedure. The in-pile behaviour has shown to be excellent during the past three years [ref.].

Description of the OCOM-Manufacturing Process

The production and quality control process of U-Pu-mixed oxide can be shown by the following figure 1.

The OCOM-process starts with the production step "Co-milling", where a so-called 'master-mix' is produced out of Pu- and U-oxide. This mixture of U- and Pu-oxide containing approximately 30 w/o Pu-fiss is intensively milled and then diluted to its appropriate value in a second mixer. In case of light-water reactor-fuels the plutonium content is between 2 and 4 w/o.

In order to examine the macroscopic Pu-distribution, the Pu-contents of the powder or the pellets are assayed wet-analytically by potentiometry, and lately also by photometry, or nondestructively by γ-spectrometry.
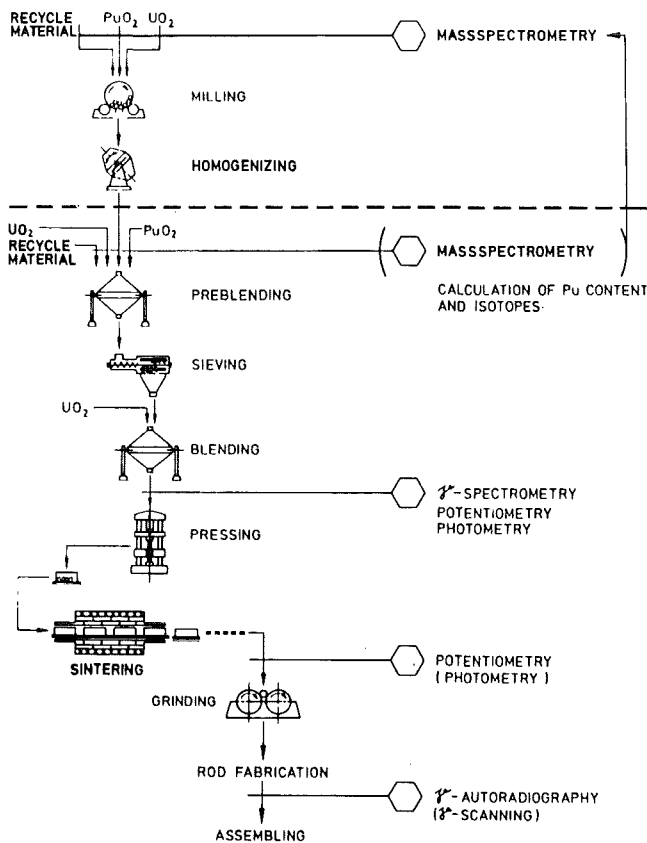
Fig. 1:

OCOM-process: production and quality control scheme

Since a mix up of pellets from different process charges has to be excluded, it is also possible to classify the γ-autoradiography on fuel rods as a macroscopic test.

The fuel rods were put on x-ray films. The γ-radiation of the fuel pellets produce a blackening on the film. This test method was utilized up to the year 1981.

Since 1982 this procedure has been replaced by γ-scanning. Here the radiation of the fuel rod is examined with a NaJ-detector.

With figure 2, I would like to give you a survey of the test methods which are in use to prove the plutonium contents in the basic materials, the fuel pellets and the fuel rods. On this chart the relation between test accuracy and test expenses, which is most important for the test planing is shown.
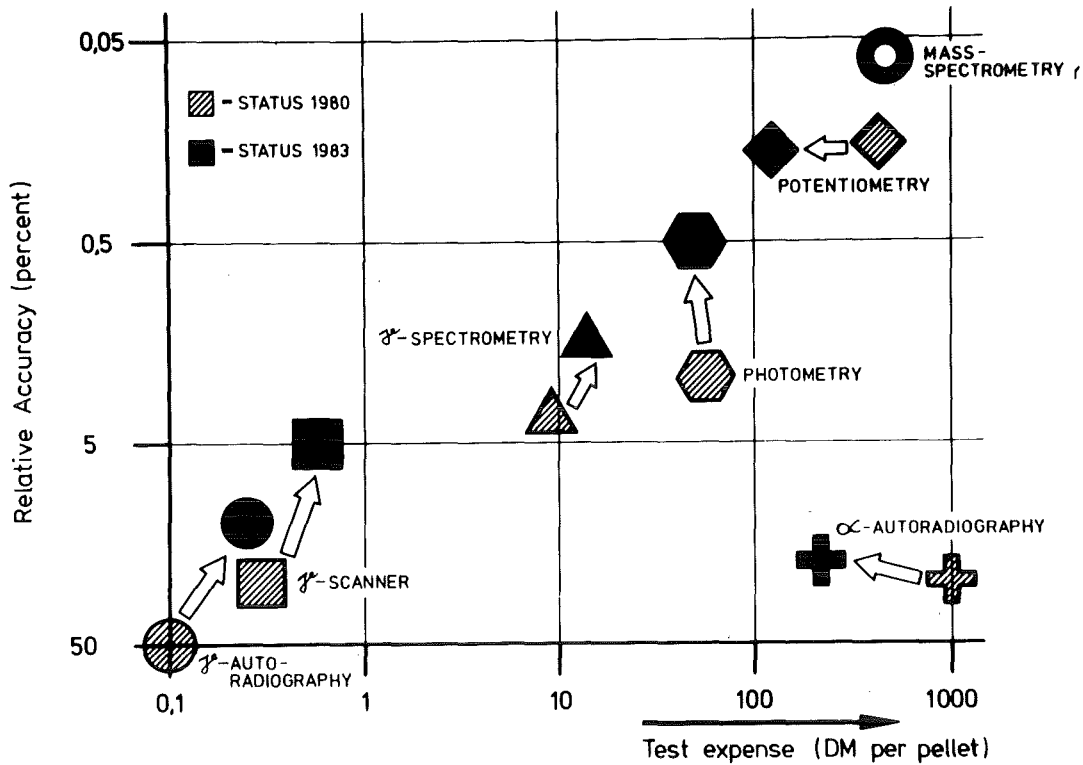
Fig. 2:

Comparison of accuracy and expense of the test methods used for the determination of Pu-contents.

Moreover the improvements which have been achieved through our own developments are made visible by the cross-hatched marks as well as by the arrows attached.

It is worth mentioning that for the first time we were successful in applying the photometry for an exact Pu-analysis. You may realize by this picture that the photometric method closes the gap between the fast but inaccurate γ-spectrometry and the more accurate but also more expensive potentiometry.

Presentation of production experience

Reviewing former fabrication projects, the mathematical-statistical analysis of the distribution of the Pu-values found potentiometrically always showed a slightly spread shape which nevertheless was within the limitation of the specification requirements (figure 3).
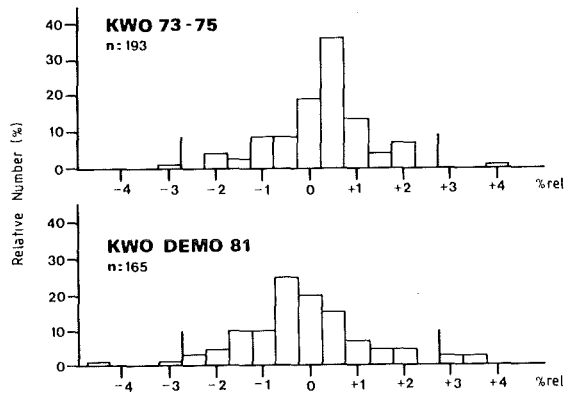
Fig. 3:

Statistical Pu-fiss-
distributions of
LWR-fabrication projects

In this picture you see two distributions of the results of plutonium-
analysis which were found with the potentiometric analysis method.

The comparison of the distributions of former fabrication projects
with the distribution of values from the 'early days' of the OCOM-pro-
ducts shows that the OCOM-process at the beginning in 1981 had a strong
tendency to form outliers towards the direction of a higher enrichment.

Obviously related to these deformed distributions the γ-autoradio-
graphy showed some new phenomena.

These phenomena could be classified in three groups:

- dark, sharp-cornered spots
- dark, cloud-like structures
- light and dark pellets

Pellets with the above mentioned effects were taken from the rods and
examined by ceramography.

The pellets with the first mentioned effect "dark, sharp-cornered spots"
showed the particle distributions represented in figure 4.

(a)　　　　　　　　　　　　(b)

1000µm
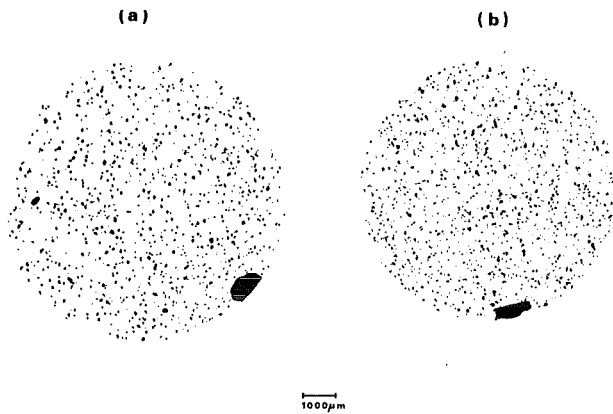
Fig. 4:

α-autoradiography of pellets containing large plutonium containing
particles of different origin

In figure 4 you can see at the pellet edges larger plutonium-containing
particles which are slightly different from one another as far as their
shape is concerned. The particle in picture 4a has an oval, the one of
picture 4b has an edged, splintery shape.

Later on it could be proved that these U-Pu-agglomerates are of
different origin. The oval agglomerate was built up within the powder
while it is moved. The splintery agglomerate came from sediments at
powder processing equipments.

The second effect "dark, cloud-like structures" was also examined with
ceramographic micrographs. For this purpose a pellet was ground on
different levels. The α-autoradiography is shown in figure 5a and 5b.

(a) Top　　　　　　　　(b) Center

(c) Top
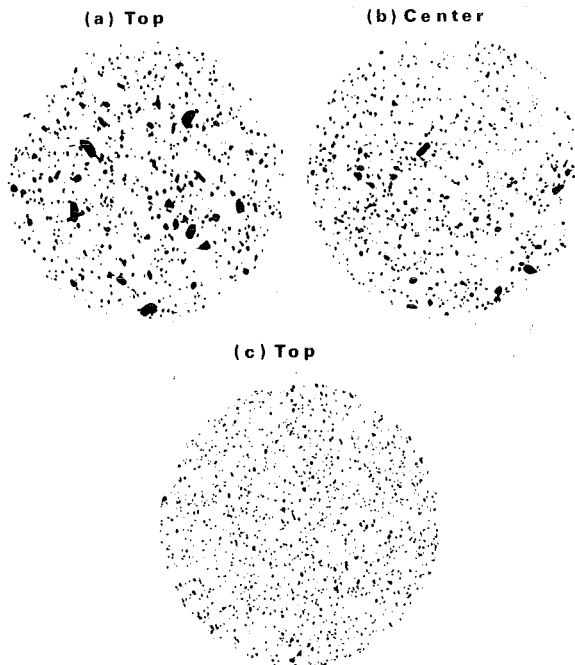
Fig. 5:

α-autoradiographies

(a), (b): top and center, the
　　　　　　pellet contains
　　　　　　agglomerates of
　　　　　　approx. 500 µm diameter

(c)　　: normal pellet

The figures distinctly show that, here too, the agglomerates, and especially their inhomogeneous arrangement is responsible for the irregular (asymetrical) blackening (cloud formation) of the γ - autoradiography film. The figure 5c shows a homogeneous distribution of the mixed crystal particles.

Consequently, the third effect "differently strong-blackened pellets" is also explainable. Here the smaller agglomerates are in homogeneous distribution. Their quantity, however, is appropriately enlarged.

Parallel to that further findings for these observations were revealed by the measurement results of the γ-scanner. This scanner is an ALKEM-development which replaced the γ-autoradiography-method for most of the ranges of application.

Here we use the 60 keV Am-241- and the 208 keV-Pu-241 radiation out of the total γ-spectrum.

Its mathematical ratio is an important material indicator. As Am builds up relatively fast in Pu, and as it owns a distinct natural radiation component it can also be used as tracer material. Thus
- during the production of the basic mixtures Pu-materials are used which differ in terms of Am-content, and which therefore can identify mixtures later on.A computer program provides this identification.

The OCOM fuel rods of the 1982 production which were tested by the scanner showed the following characteristics (see figures 6 and 7):

- broad peaks in the scanning charts from pellets which the computer could not identify, which therefore could not come from previous productions lots (figure 6).

- Narrow peaks for which an identification with a mixture was also
  impossible. These narrow peaks could be made invisible by turning
  the rod under the detector (figure 7).
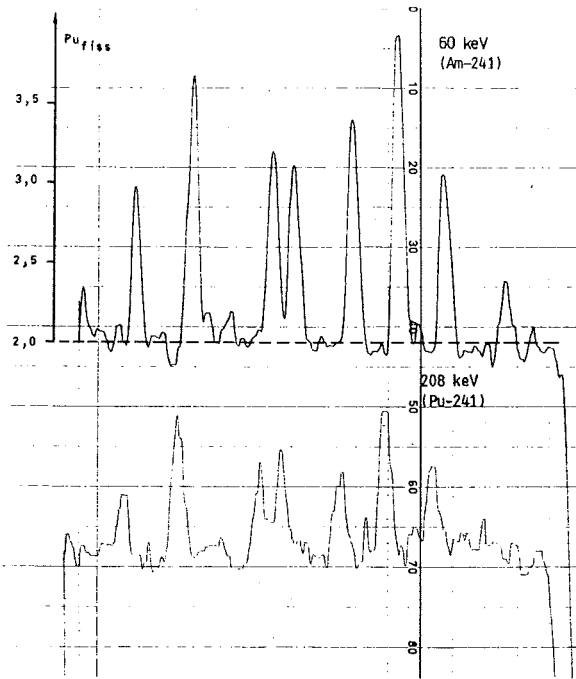
Fig. 6:

Scanning chart of a test rod
containing selected off Spec.-
pellets

ORIGINAL SCANNER CHART   RESPONSE OF THE PARTICLE AT DIFFERENT ANGLES
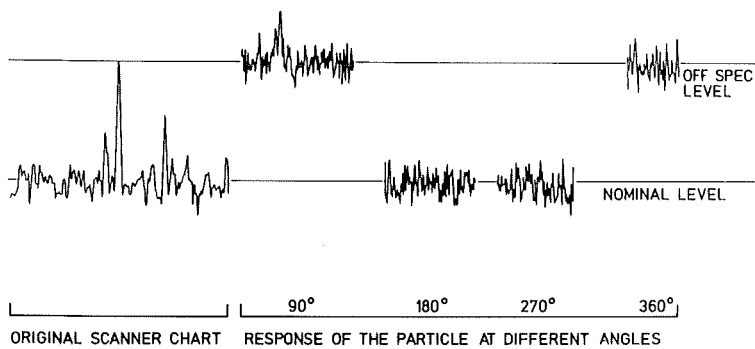
Fig. 7:

Scanning chart showing a narrow peak.
Note: the response disappeares at certain azimuthal angles.

Further investigations showed that the "broad peaks" were identical
with the second and third effect of the γ-autoradiography as
explained before namely cloud-like and homogeneous blackening of
the film.

The "narrow peaks" were to be explained by the first effect the
agglomerates at the edge of the pellets as shown in picture 4.

In order to assure the quality and exclude any out of specification
fuel rods all the fuel rods out of the mixture concerned were
examined by the scanner. Cross-checks were made on some of the
rods by means of the γ-spectrometry. The examinations proved
a large conformity of both methods on one hand, on the other hand
the Pu-distribution with a tail towards high Pu-values could be
reproduced (figure 8).

- ESTIMATED TOTAL DISTRIBUTION
  (Qualitative)

- SCANNER REJECT LEVEL

2,0   2,5   3,0   3,5   Pu$_{fiss}$

| 99,5 % | 0,5 % |
| ACCEPTED GOOD PELLETS | REJECTED BAD PELLETS |
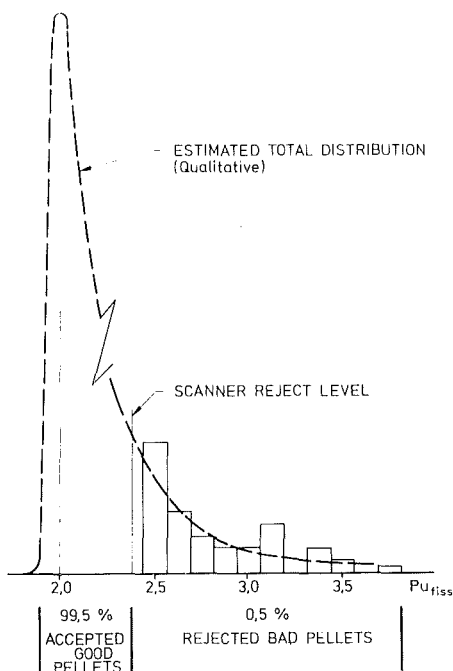
Fig. 8:

Reconstruction of the upper
tail of the Pu-distribution
by scanner assay.
Note: the number of deviant
pellets detected is approx.
0,5 % of the total number of
assayed pellets.

Because of the conformity of the γ-scanner results with the other
testing methods, the γ-scanner could be used as a 100 %-measuring-
and sorting device.

It was possible to persue the entire segregation process by means of
the fast measuring system of the γ-scanner together with all the
other further developed measuring systems like the γ-spectrometry and
especially the photometry. The critical elements within the production
process could thus be traced, and the efficiency of process improvements
was affirmed.

These improvements of the production technique on one hand consist
of the optimization of the mixing process by finding the appropriate
choice of powder quantities to be mixed, and on the other hand
segregation processes within the powder were avoided by careful
handling of the powder.

In addition the fabrication process and the equipment were improved
in such a way that bigger Pu-containing agglomerates were eliminated.

By these changes the macrohomogeneity as well as the microhomogeneity
could be improved.
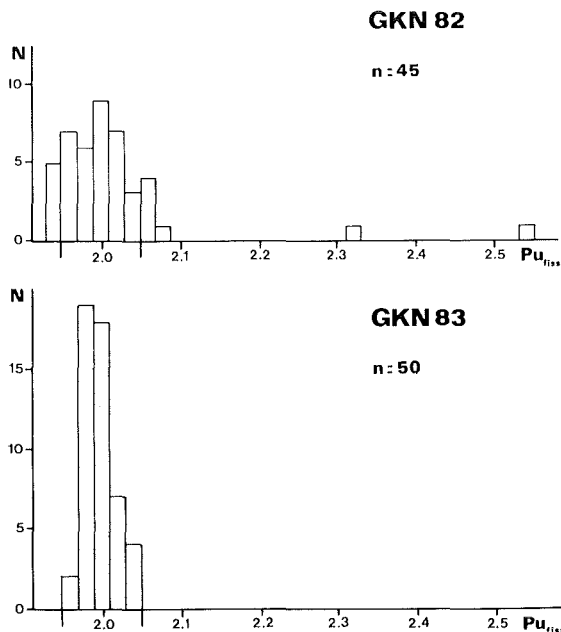
Fig. 9:
Result of the improvements in
the production process.
Upper part: original OCOM-process
Lower part: improved OCOM-process

Final considerations

The process which has been shown here is typical for interactions which can happen between the production and the quality assurance, if new production methods are put to test, or if existing production methods are changed.

These changes are certainly submitted to preliminary tests within so-called qualification charges for a successive production process; nevertheless possible disturbances and difficulties cannot be completely avoided, even if the qualification charges provide unmistakeably good results.

This can be explained by the fact that the qualification consists of smaller production quantities, and therefore problems depending on the quantities cannot be realized, and that perhaps other machines have to be used.

Furthermore it has to be taken into account that in the course of the normal production defects are found rather delayed in particular if the characteristics are microscopic or if the sampling too small.

This includes questions about the test duration, test expenses, and whether the test can be carried out nondestructively or destructively.

In the case that has been depicted here an decision was initiated by statistic collection, evaluation, and judgement of test results. Furthermore it was possible to verify this statistically based statement by a well-graded test concept, so that consecutively the non-homogeneities could be discovered in time, and their further appearance could be avoided.

Reference

D. Hanus, J. Krellmann, R. Löb   Jahrestagung Kerntechnik 1982
                                   Mannheim, Mai 1982

DISCUSSION

A. STRASSER: What is the homogeneity you would like to attain?

K. GRUBER: This depends on the actual specification, e.g. for
LWR approximately ± 2% rel. There are two approaches:
1. Mean and std.-deaviations of the statistical distribution of
potentiometric values taken from a number of samples.
2. 100% inspection by γ-scanning, but note that the sensitivity
is not good enough to meet the specification requirement mentioned
above.

B.J. BUESCHER: Would it not be preferable to inspect the fuel
rod prior to final assembly for inhomogeneities?

K. GRUBER: It would be preferable but there are currently
technical difficulties with the γ-detectors in the fabrication
equipment.

METHODS OF QUALITY CONTROL FOR ZIRCALOY TUBING

H. G. WEIDINGER
Kraftwerk Union AG
Erlangen, Federal Republic of Germany
K. H. KUNZ
Nuklearrohr-Gesellschaft mbH
Duisburg, Federal Republic of Germany

ABSTRACT

Quality control of cladding tube production has to take care of three aspects:

- quality of the incoming material
- process control of all steps between incoming and outcoming material
- control of the final product

This paper reports - besides a consideration of the design relevant requirements - the present status of all those three aspects.
It focuses specificly on the importance of and the experience with process control. Besides a systematic discussion, four examples out of the practice of cladding tube production are described: process control with respect to surface roughness, uptake of nitrogen, trex and cladding tube eccentricity and influence of process control on dimensional deviations.

METHODS OF QC FOR ZIRCALOY TUBING

1. ## Introduction

Cladding tubes have a key function: they are the most important component for the operation of nuclear fuel assemblies as a barrier to retain fission products and actinide elements generated in the fissile material. Therefore specific attention always was paid to keep the quality of this product under particularly close control.

In the meanwhile a high level of experience could be obtained due to 20 to 30 years of commercial production in many countries all over the world. During this time the failure rate of nuclear fuel operated with Zircaloy cladding was in most cases extremely low /1, 2, 3/.

There was also other material used, as for instance "Magnox" for the gas cooled reactor and also iron or nickel base alloys, for instance for some early pressurized water reactors as well as for advanced reactor types and for breeder reactors. But since the commercial interest was concentrated on water cooled reactors, the most important cladding materials became Zircaloy 2 and Zircaloy 4 /4/.

Therefore this paper only reports aspects of quality control of cladding tubes made of these materials. Nevertheless, many of the described methods and facts also hold for any other nuclear fuel cladding material.

Cladding tube is not the only tubular component made of Zircaloy in nuclear reactor technology. But other Zircaloy-tubings as guide tubes are being fabricated much the same way as cladding tubes. The QC requirements are similar or less stringent. So it makes sense to concentrate on Zircaloy cladding tubes.

It was always clear that the properties of the final cladding tube are not only influenced by the steps of tubes manufacturing but also by all Zircaloy fabrication procedures, at least back to the melting of the alloys /5/. To stay within a reasonable scope this paper will concentrate its considerations on the cladding production that starts with an extruded (and sometimes already once rocked) tube hollow which is called trex.

But some considerations will also be given to the QC-aspects
of the prior production steps back to the melting of the ma-
terial.
During the mentioned period of 20 to 30 years of commercial
production of Zircaloy cladding tubes there was a development
of the philosophy how to control the quality of this product
as well as a development of the technology which is respon-
sible for the todays high standard of quality /6, 7/.
The development of this philosophy and technology to the pre-
sent status shall be considered shortly in the following
chapter.

2. **General Aspects of Quality Control of Zircaloy Cladding Tubes**

Basicly speaking, the quality of any product has to be achieved
by reasonable measures during fabrication and by checking the
result of this fabrication with respect to the required proper-
ties and their limitations. But the better the influences of
the fabrication procedures and the process parameters of the
used tools and auxiliar materials are understood, the better a
systematic approach is feasible to "make" quality by measures
during fabrication and not by measurements after fabrication.
As in the meanwhile mentioned several times (compare f.i./8, 9/)
an optimized combination of those measures and measurements
is also important for an economically optimized product.
Consequently quality control of cladding tubes production has
to consider three aspects:

1. quality of the incoming material and its control

2. process control of all steps between incoming and out-
   coming material

3. control of the final product

Of course, by all three aspects important contributions are
possible to an improved quality of the final cladding tubes
and, as a matter of fact, they were obtained during the

history of this technology. But the first and the third aspect are mainly of interest for medium and long term development.

Because, once the incoming materials are in the plant - and dispositions have to be made in the order of one year in advance - the only really effective tool to improve or to lose the standard of quality is to work with aspect No. 2, the process control.

This process control has to be considered twofold as

- control of the process parameters

- control of the properties of the intermediate product

for each production step (comp. Fig. 1). In practice process control may be considered to be devided into 5 actions (comp. Fig. 2):

1. Identify   the specific targets of these fabrication steps
                - process parameters
                - intermediate product properties

2. Perform   this fabrication step starting with given
                - operating process parameters
                - properties of the incoming product

3. Check   during this fabrication step the actual
                - process parameters
                - outcoming product properties

4. Compare the result of 3.) with 1.) and feed back this information for: 5.)

5. Corrective actions at 2.) (and change target at 1.) if failure in target identified.

The sequence of the actions 1.) to 5.) has to be a closed loop of informations and consequent actions. The more one can break down the whole production into production steps, each

controlled by such a closed loop sequence, the more flexible response is possible to qualify deviations. The more contineous information is available from action 3.) and the shorter the loop of feed back of information and correction action can be kept, the more effective any quality control during production will be.

As a matter of fact in the early days of Zircaloy-cladding tube production it was difficult to define other intermediate targets than intermediate dimensions OD and ID and even those dimensions were not always optimized. Thus there was a high risk that some of the final requirements could not be met and the final inspection was the first opportunity to realize how much of the product had to be rejected, very often without a chance of rework.

Therefore much emphasis was given to very consistent and stringent final checks with methods of high accuracy and reliability. Thus for instance the highly efficient non destructive ultrasonic test procedures to control dimensions and material soundnes were developed /10, 11/, which are today in common use in all Zircaloy-cladding fabrication plants. Several destructive tests have been developed and steadily improved /8, 12, 13, 14, 15, 16, 17, 18/ as summerized in chapter 6.

Much effort was also applied to improve the reliability of the incoming material properties. In the next chapter this aspect shall shortly be considered.

3. Quality of Incoming Zircaloy Tube Hollows

For the cladding tube producer three types of requirements to the incoming material are essential:

1. To get the dimensions of the tube hollows with sufficient close tolerances;

2. To gain <u>reliability</u> with respect to <u>workability</u>;

3. To assure sufficient <u>uniformity</u> in those <u>metallurgical</u> <u>material properties</u> that influence the <u>final cladding</u> <u>properties</u>

To 1.:

Of course O.D. and I.D. are basic requirements for every cladding tube manufacturer. But the point of most concern during the years of developing a stable quality level at the prematerial vendor was the reliability to stay within the limits of the requirements for the eccentricity over the whole length of the tube hollow. The critical production step for this dimensional requirement is the extrusion and the steps of prior preparation. It took years of development to reach the todays' standard, where every qualified production will stay safely away from about 5 % of the wallthickness.

To 2.:

Workability mainly means safety against splitting or cracking during rocking. Today a high degree of reliability has been reached, but still on a mainly empirical basis of knowledge. This means that it is still very difficult to quantify the influences of the various production steps between melting and extrusion with regard to this requirement.

To 3.:

There are several final cladding tube properties that depend on the technology of the prematerial vendor. As example two of the most important cases are:

- Strength and creeprate depend on the chemical composition even within the specification limits. Well known influences are those of the Sn, O and C contents /5/.

- Corrosion depends on a proper β-quenching /19, 20/ prior to all subsequent deformation and annealing steps which have strictly to stay in the α-range (i.e. below about 800 °C).

In the meanwhile these influences are under safe control by the prematerial manufacturers. Some other possible effects are still the objective of experimental investigations /21/. There are for instance possible influences of chemical inhomogeneities on special corrosion effects as nodular corrosion, or the impact of impurities on the recrystallization behaviour of Zircaloy. As a result of many years of fruitful cooperation between cladding tube manufacturers and Zircaloy producers a set of checks turned out to be useful for both sides to be applied on the Zircaloy material before cladding tube fabrication starts. Table I and II illustrates how an example of this set of requirements and tests may look like.

4. Design Criterias Relevant to Zry-Cladding Tube Quality

Two major objectives have to be observed /22, 23/ to assure the operational targets of Zircaloy cladding tubes:

- mechanical integrity and

- dimensional stability.

In table III (a and b) groups of design criteria are shown which result from these general objectives and which set of material properties has to be considered as a consequence of those design criteria.
Numerical values for strength and ductility and consequently for residual coldwork and degree of recrystallization vary for different designs between values for just stress relieved up to fully annealed material.
Maximum allowed corrosion rate today is almost everywhere specified according to ASTM G2.
The maximum depth of accepted flaws varies, but is mostly in the range of 5 - 10 % of wallthickness.

Surface roughness figures also vary depending on design cal-
culations. Hydride orientation is no longer generally required;
but some specifications still prefer a tendency to circumferen-
tial orientation. Some specifications ask for texture orien-
tation factors, preferring a tendency to radial orientation of
the basal poles.
In most cases cladding tube specifications do not ask for quan-
titative requirements with regard to microstructure. But never-
theless those properties will be followed qualitatively by
the designers of fuel rods. Major deviations from experience
would not be accepted.

Dimensions of course follow the needs of specific designs. But
the tolerances are in most cases very close.

It is now the task of the cladding tube manufacturers not only
to meet those sets of design relevant properties but also to
make sure that there is a sufficient level of confidence that
all figures of a value remain in a close scatterband, at least
for those tubings that belong to one lot of delivery, better
for the whole fabrication campaign.

5. Quality Control in Zircaloy Tube Fabrication

Figure 3 gives an overview on all typical steps of fabrication
and quality inspections of the intermediate and the final pro-
duct during the production of Zircaloy cladding tubes from the
incoming tube hollow or trex. The number of cold rolling and
intermediate annealing steps may vary /24/, depending on

- size of the incoming material

- amount of coldwork applied during one step

- final size of tubes

Typically there are 3 steps when the starting size is
63,5 x 10,92 mm (OD x Wth.). As an example /24/ the change
of dimensions may be as follows:

1. step	63,5 x 10,92	30,0 x 4,5	80 % CW
2. step	30,0 x 4,5	16,7 x 2,2	73 % CW
3. step	16,7 x 2,2	10,75 x 0,72	76 % CW

In order to get a homogeneous structure and texture, through
the wall, over the circumference and along with the whole
length of the tube and to avoid local surface imperfections
on the one side a very thorough shaping, sizing and surface
finish of the rolling tools /25/ is essential; on the other side
the degree of deformation and specificly the ratio of wall-
thickness to diameter deformation and the annealing paramters
of the final sequence of rocking and annealing has to be ad-
justed /26, 27, 28, 29, 20, 31, 32/.
Today there are rather sophisticated machines in use for
grinding the tools to shape and size as well as measuring de-
vices to check the calibration of the tools. Electronic con-
trol of the grinding process and electronic data acquisition
after the measuring procedures are becoming state of tech-
nology.

After each cold rolling step the Zircaloy tube has to be cleaned
very thoroughly in order to avoid that impurities may diffuse
into the surface during the subsequent annealing. This might
lead to local hardening and thus be the origin of surface
cracks produced by the next cold rolling step. During the
intermediate annealing step the material has to be completely
recrystallized, but grain growth has to be avoided. This
annealing has to be performed under vacuum ($< 10^{-3}$ mbar) in
order to prevent uptake of gases (O, N, H) by the Zircaloy.
Experience has shown that only watching the vacuum when the
pumps are working is not sufficient. The tightness of the sys-
tem has to be checked by a static leakage test.

Straightening may be necessary after annealing to assure straightness before the next cold rolling or the finishing procedures. Also a slight pickling often is in use after intermediate annealing before entering the next cold rolling mill to assure a metallic bright surface.

Most relevant to product quality during these intermediate steps are the procedures

- cold rolling

- cleaning

- annealing

Therefore a systematic and comprehensive process control in the sense as discussed in detail in chapter 2 has to be applied to each of these procedures in each production sequence. This contains in each case the control of

- process parameters and

- intermediate product properties.

In table IV the technical measures and the necessary equipment is characterized.

During the final sequence of cold rolling - cleaning - annealing not only the final size is given to the product. Very important design relevant properties of the Zircaloy tube as strength/ductility, creep, growth and also the final microstructure and texture will be fixed. Most of those properties are being adjusted by the proper combination of coldwork and annealing temperature and time /33, 34, 35, 36, 37, 38/. Today there are analytical methods available /39/ to precalculate the necessary combinations of fabrication parameters, if some basic material constants are known from experiments.

The combination of temperature and time for the annealing
process in a production furnace may be critical, if a speci-
fic degree of stress relief or recrystallization in the
cladding tube is required by the fuel rod design. In those
cases very close control on time and temperature for all po-
sitions in an annealing muffle has to be assured by QC.
When starting with material from a new vendor or after changes
in the prematerial production a pilot annealing is highly re-
commendable.

The required surface condition of the cladding tube will be
obtained by the finishing processes. Different procedures are
in use. In some cases a final pickling on O.D. and/or I.D. sur-
face is applied, sometimes controlled by ultrasonic techniques
/40/; in other cases no pickling after the last annealing is
permitted. In those cases grinding of the O.D. surface and
sand blasting of the I.D. surface is commmonly in use. Recent
experience has shown that the I.D. sand blasting may be dropped
if a highly efficient hot cleaning was applied before the
final anneal/41/.
As a matter of course all those finish procedures again have to
undergo a QC surveillance, where again process parameters
and intermediate product properties will be controlled (com-
pare table V).

6.   Quality Control for Final Acceptance

The quality of the produced cladding tubes has been estab-
lished after the last finish step. The basic task of the
final inspection now should be to verify this quality and to
quantify and document the result. As a matter of fact it
still is also a final barrier, where nonconforming material
can be rejected, if necessary. But in modern Zircaloy cladd-
ing tube production this case will be an exception. In table
VI and VII a survey is given on a typical set of final in-
spection tests, test techniques and applied frequency for Zir-
caloy cladding tubes. The test frequency may vary depending
on specifications.

More details of test techniques, sensitivities and resolutions, accuracies and other practical experiences were recently compiled and published /5/.

Since the primary target of the final inspection test is to verify and quantify the obtained quality level, the work of QC in this stage is not finished when it can be stated and documented that the produced tubing fullfilled the specified requirement.

The quantitative results of the final inspection should now be evaluated with the methods of statistics that are developed and available today /42, 43, 44, 45, 46/.
In all cases of statistical evaluation it is essential to know the type of the distribution function to be applied to the available set of data. Otherwise, wrong conclusions might be drawn from wrong statistical figures. Not all properties follow a "normal" Gauß-distribution (like f. i. strength data). Some follow a "log normal" distribution (like f. i. the uniform burst elongation). Figures 4 and 5 give some examples of typical distributions of both types.

It is very informative and useful to compare those plottings, or at least the average values and standard deviations between different fabrication lots or campaignes. Thus trends or unexpected deviations may be detected and counter-actions may be taken in time.

7. **Examples of Practical Experience and Consequences from Zry-Tube Fabrication with Integrated Process Control**

At the end of this systematic description of QC methods applied to Zircaloy cladding fabrication it might be of interest to discuss some examples about the cladding production practice. They shall show which quantitative results may be obtained for the quality level of Zircaloy claddings and also how these results influence the economy of this type of production.

7.1   First Example: Process Control on Cold Rolling with Respect to Cladding Surface Roughness

In Fig. 6 is shown how the method of closed loop of information and action (as generally explained in Figure 2) works in the case of process control on cold rolling in order to keep the roughness requirements within narrow limits of specification. One can see that it is the interaction of control of process parameters, i. e. mainly tool characteristics and rolling parameters, and of the product property roughness, that leads to the required stability of quality level. In Figure 7 the practical measures, their documentation and the final statistical evaluation of the obtained quality level is illustrated. The final result is a stable and quantitatively well characterized level of quality for one specific quality requirement, in this case the surface roughness. Basicly the same procedure will be successful for many other requirements, as for instance dimensions, ultrasonic rejection ratios etc.

7.2   Second Example: Uptake of Nitrogen during Cladding Fabrication

Zircaloys are known to have a high affinity to hydrogen, oxygen and nitrogen. The nitrogen content of Zircaloy has to be kept under specified limits because too high values will reduce the corrosion resistance of Zircaloy in reactor water. On the other hand each annealing process has to be very thoroughly controlled to avoid enrichment of oxygen and nitrogen in Zircaloy from air contaminations or leckage in the furnace muffle.

It would be out of the scope of this paper to discuss all possible sources of air uptake during annealing and how this can be controlled. But only systematic and reliable control of all sources garantuees that the acquired limits can be met. The result of good process control can typically be shown in Fig.8. There is compared the nitrogen content in the delivered tube hollows with the nitrogen content in the final cladding tubes made out of this material.

By approved mathematical methods it can be shown that the
small differences between both distributions are "statistically
not relevant", i.e. the nitrogen content is the same in both
cases.

7.3   Third Example: Influence of Trex Eccentricity on Cladding
                   Tube Eccentricity

It is an old experience of rocking technology that eccentricity
appearing in the prematerial (tube hollow or trex) can not be
improved by the later rocking steps. Only the opposit is po-
ssible, i.e. some eccentricity may be generated during the
rocking steps between tube hollow or trex and final cladding
tube. Fig. 9 shows both effects in the statistical evaluation
of two projects, where project A was later in the development
of Zircaloy cladding technology then project B. In this evalu-
ation the maximum eccentricity of the single tube (trex and
cladding tube) was used and therefore the absolute figures for
trexes and cladding tubes can not directly be compared. But
nevertheless it can be very clearly seen in these figures that
a systematic improvement of the trex eccentricity also leads
to better values of the cladding tubes.
As a second information one can also conclude from this eva-
luation that the rocking process control could be improved with
regard to eccentricity for project A as compared with project B.

7.4   Fourth Example: Influence of Process Control on Dimensional
                   Deviations

In Fig. 10 the history of the rejection level for dimensional
deviations (O.D., I.D., Wallthickness) is shown for a high volume
of Zircaloy cladding production. This figure demonstrates that by
introducing systematic process control in addition to incoming
material inspection and acceptance tests the rejection ratio can
significantly be reduced to about one third of the values that
had been observed before.

This example also demonstrates that by a consistent interaction
of all types of QC-measures not only a good and stable quality
level can be obtained but also the overall economy of the
production will be improved.
Low rejection ratios mean less material losses. This is speci-
ficly important when the ratio of material costs is as high as
it is in all Zircaloy productions.

In this case it has also to be taken into account that a signi-
ficantly increased and stable quality level, assured by process
control, allows to reduce the costs for the final inspection.
These improvements of quality and economy go together.

8. Conclusions and Outlook

This survey should show how the joint efforts of process control
and product control lead to a high and stable level of quality
for a nuclear component which is as important as cladding tubes.
But in order to really reach these targets, two additional con-
ditions have to be fulfilled:
The one is a close cooperation between the producer of Zircaloy,
the manufacturer of the cladding tubes and the party that is in
charge of the overall Zircaloy technology within the scope of the
nuclear fuel technology.
The second essential is detailed and reliable technical knowledge
of material properties and process mechanisms within all three
mentioned parties.
There is a large amount and a high level of such knowledge
everywhere in the world where there is Zircaloy technology
practiced today. But there are not many published data available
in open literature as far as fabrication and control knowledge is
concerned.
The authors therefore have to thank for the openess of the
NRG* management to release data of practical experience from
a modern Zircaloy cladding fabrication.

---

*) NRG: Nuklear-Rohr GmbH, Duisburg, West-Germany

Thus not only theoretical considerations but also realistic demonstrations could be given to describe the methods and the status of modern Zircaloy cladding quality control.
On the basis of this status the future development will now concentrate on increasing mechanizing and automizing of the fabrication and control steps. This may increase again reliability and stabilize the economy of Zircaloy cladding tube production. But it will not substitute ongoing efforts to still increasing scientific and technological understanding of the material Zircaloy and the processes to be applied to this material.

An interesting new development in Zircaloy is the addition of a liner of pure Zirconium to the inner surface of a Zircaloy cladding tube /47/ in order to improve the resistance against iodine stress corrosion attack.
To achieve a good bonding between die liner and the base material both components are being coextruded in the same stage of production where also regular tube hollows are being fabricated.

There are no new requirements to be observed during cladding tube fabrication and no problems arise if there is a good control on the rocking processes.
There are also no new types of QC methods necessary to control liner thickness and bonding. But it may turn out that non-destructive test methods with higher resolution for flaw detection and dimensional determination will be useful to improve the effectiveness of the non-destructive final testing of such tubes.

As a matter of fact the evolution of Zircaloy cladding tube fabrication is still under way, even after more then 25 years of success - a good sign for the potential of and the vitality in this area of technology.

LIST OF FIGURES

LIST OF TABLES

LIST OF REFERENCES

/1/   "Light Water Reactor Fuel Performance" ANS
Topical Meeting 1979, Portland, USA

/2/   "Reactor Safety Aspects of Fuel Behaviour" ANS/ENS
Topical Meeting 1981, Sun Valley, USA

/3/   "LWR Extended Burn Up - Fuel Performance and Utili-
zation" ANS
Topical Meeting 1982, Williamsburg, USA

/4/   D. L. Douglas, "The Metallurgy of Zirconium"
IAEA (1971), Vienna

/5/   "Guidebook on Quality Control of Water Reactor Fuel"
IAEA, Techn. Report Series No. 221
Vienna (1983) 111

/6/   B. Lustman, WAPD-T-2715 (1978) and
"Zirconium in the Nuclear Industry"
ASTM-STP 681 (1978) 5-18

/7/   K. H. Matucha, P. Wincierz
Journal of Nuclear Materials 100 (1981) 119-127

/8/   G. Dressler, H. G. Weidinger
"Nuclear Fuel Quality Assurance"
IAEA, Oslo (1976) 227

/9/   "Guidebook on Quality Control of Water Reactor Fuel"
IAEA, Techn. Report Series No. 221
Vienna (1983) 5, 8, 27

/10/   B. T. Cross, D. J. Abbot
"Zirconium in Nuclear Applications"
ASTM-STP 551 (1973) 31

/11/   H. Rauscher, 4ième Colloque International sur les
Méthodes du Contrôle Non Déstructive
Grenoble, 11.-14. Sept. 1979

/12/   P. E. C. Bryant, "Applications-Related Phenomena for
Zirconium and its Alloys"
ASTM-STP 458 (1969) 360

/13/   P. D. Kaufmann, P. Danielson, E. F. Barock
"Zirconium in the Nuclear Applications"
ASTM-STP 551 (1974) 52

/14/   R. Pfau, G. Dressler
Prakt. Metallogr., 6 (1969) 614

/15/   G. Kraft, A. Kahler
Z. Anal. Chem., 281 (1976) 285

/16/   H. Stehle, E. Steinberg, E. Tenckhoff
        "Zirconium in the Nuclear Industry"
        ASTM-STP 633 (1977) 486-507

/17/   Th. Andersson, A. Wilson
        ASTM-STP 681 (1979)

/18/   J. E. Lewis, G. Schoenberger, R. B. Adamson
        "Zirconium in the Nuclear Industry"
        ASTM-STP 754 (1980) 35

/19/   A. B. Johnson Jr., "Application-Related Phenomena
        for Zirconium and its Alloys"
        ASTM-STP 458 (1969) 271

/20/   Th. Andersson, G. Vesterlund
        "Zirconium in the Nuclear Industry"
        ASTM-STP 754 (1982) 75

/21/   E. Tolksdorf, "Nuclear Fuel Quality Assurance"
        IAEA, Oslo (1976) 247

/22/   H. G. Weidinger, H. Aßmann, R. Holzer, K. Raab
        "Fabrication of Water Reactor Fuel Elements"
        IAEA, Prag (1979) 151

/23/   "Guidebook on Quality Control of Water Reactor Fuel"
        IAEA, Techn. Report Series No. 221
        Vienna, (1983) 25, 88

/24/   L. Moulin, S. Reschke, E. Tenckhoff
        "Zirconium in the Nuclear Industry"
        6th ASTM Conference, Vancouver 1982, to be published

/25/   J. H. Schemel, R. W. Mc Kenzie
        "Zirconium in Nuclear Applications"
        ASTM-STP 551 (1973) 169

/26/   D. O. Hobson, "Application-Related Phenomena for
        Zirconium and its Alloys"
        ASTM-STP 458 (1969) 37-49

/27/   D. O. Hobson, Trans. ANS 13 (1970) 572

/28/   B. A. Cheadle, "Zirconium in the Nuclear Industry"
        ASTM-STP 633 (1977) 457-485

/29/   E. Tenckhoff, "Zirconium in the Nuclear Industry"
        ASTM-STP 754 (1980) 5

/30/   N. Nagai, T. Kakuma, K. Fujita
        "Zirconium in the Nuclear Industry"
        ASTM-STP 754 (1980) 26

/31/   N. Nagai, T. Kakuma, T. Matsushita, M. Kimpara
        "Zirconium in the Nuclear Industry"
        ASTM-STP 754 (1980) 63

/32/   E. Tenckhoff, Materialkundlich-technische Reihe Nr. 5
        Gebr. Bornträger/Berlin, Stuttgart (1980) 58

/33/   E. D. Hindle, G. F. Slattery
        J. Institute of Metals (1966) 245-249

/34/   H. Richter, E. Schelzke
        Z. f. Metallkunde 61 (1970) 415-420

/35/   H. Yamamoto, Chem. Economy a. Eng. Rev. 3 (1971) 19-28

/36/   K. Kjellström, Scand. J. Metall. 4 (1975) 65

/37/   R. J. Beauregard, G. S. Clevinger, K. L. Murty
        BR-1093, RDTPL 76-11 and Proceedings of the
        4th internat. Conference on Structural Mechanics in
        Reactor Technology, San Francisco (1977)

/38/   G. Dressler, H. Richter
        Symp. "Wärmebehandlung" der Deutschen Gesellschaft
        für Metallkunde, Bad Nauheim (1977)

/39/   E. Steinberg, A. Schaa, H. G. Weidinger
        "Zirconium in the Nuclear Industry"
        6th internat. ASTM Conference, Vancouver 1982, to be
        published

/40/   Y. Tanaka, Y, Komatsu, N. Fujii, H. Kubota
        "Zirconium in the Nuclear Industry"
        6th internat. ASTM Conference, Vancouver 1982, to be
        published

/41/   Personal communication by "Nuklear Rohr GmbH"
        (NRG), West-Germany

/42/   W. Masing, "Handbuch der Qualitätssicherung"
        K. Hansa Verlag (1980) München, Wien

/43/   L. Sachs "Statistische Methoden"
        Springer Verlag (1979) Berlin, Heidelberg

/44/   H. J. Kleibeler, H. Richter
        Metallkd. 65 (1974) 341

/45/   E. Steinberg, H. G. Weidinger
        "Werkstoffprüfung 1983", Deutsche Gesellschaft für
        Metallkunde, Oberursel, to be published

/46/   H. Aßmann, E. Steinberg
        "Practical Experience in the Application of Quality
        Control in Water Reactor Fuel Fabrication"
        IAEA Seminar 1984, Karlsruhe

/47/   J. S. Armijo, L. F. Coffin Jr.
        Patentauslegeschrift No. 2550029
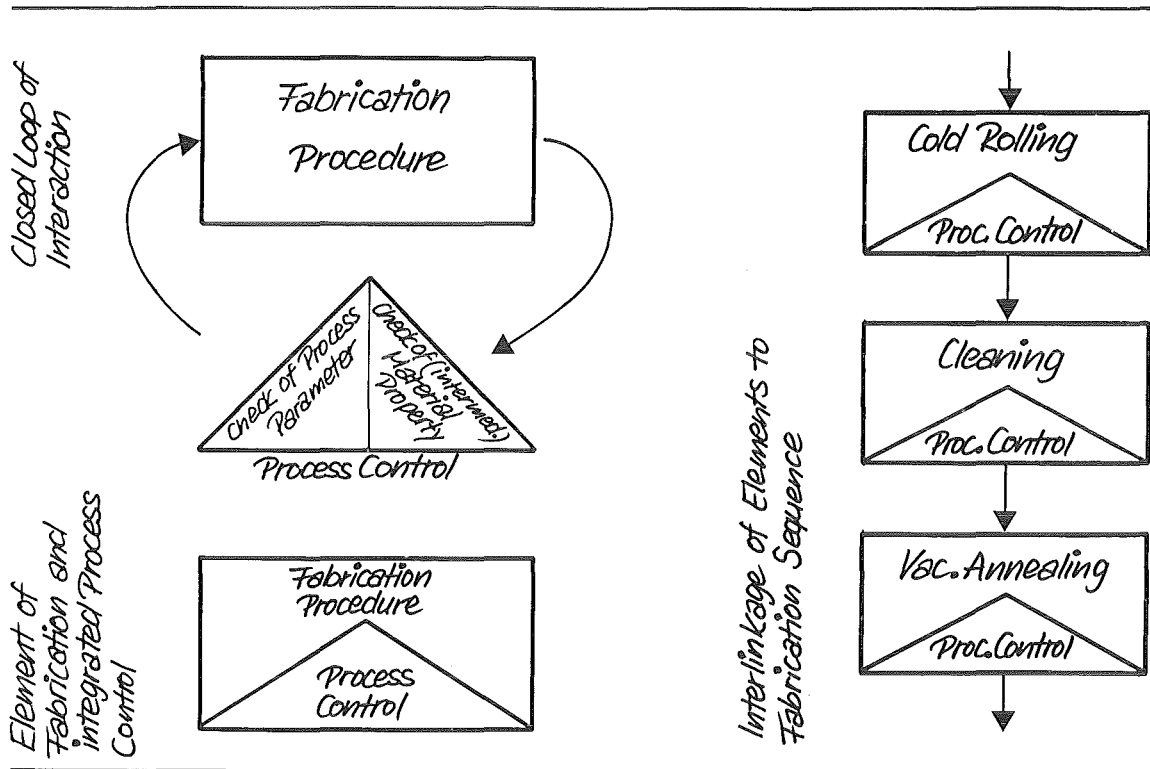
Figure 1   Interaction of Fabrication and Integrated Process
Control and Example for Fabrication Sequence with
Interlinked Fabrication Elements
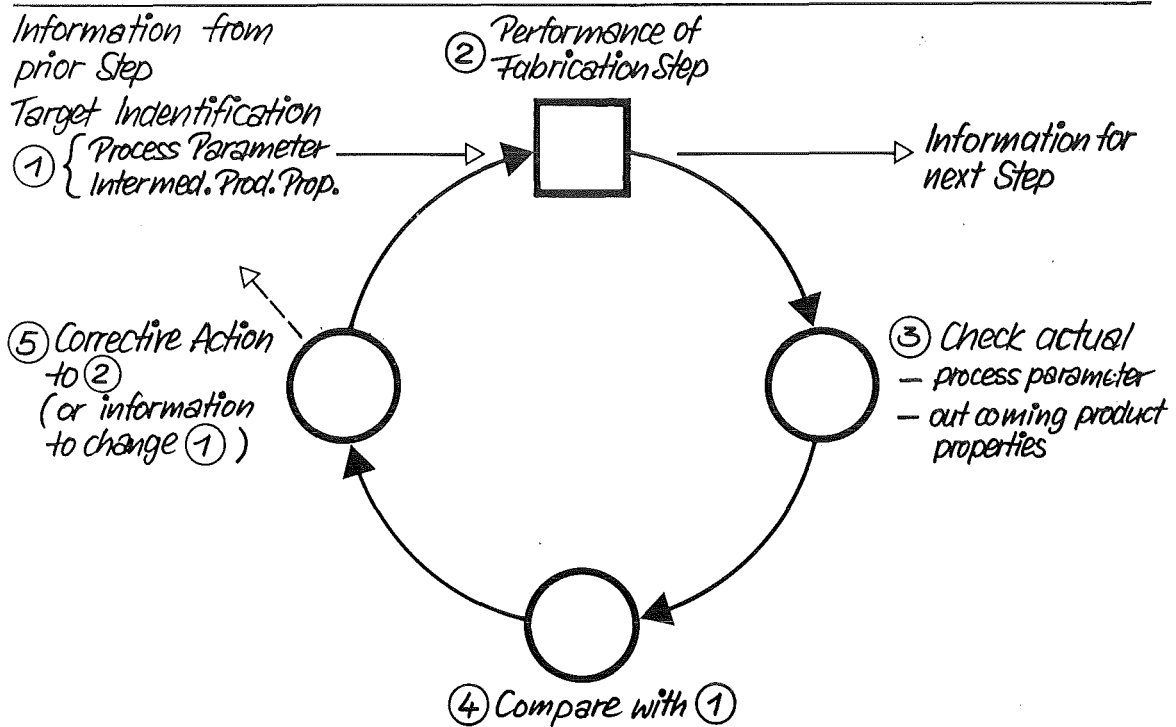
Figure 2   Flow of Informations and Actions for Integrated
Process Control

Incoming   Intermed.Proc.Final Proc.   Finish   Final Acceptance
Material   2-3x   Packing
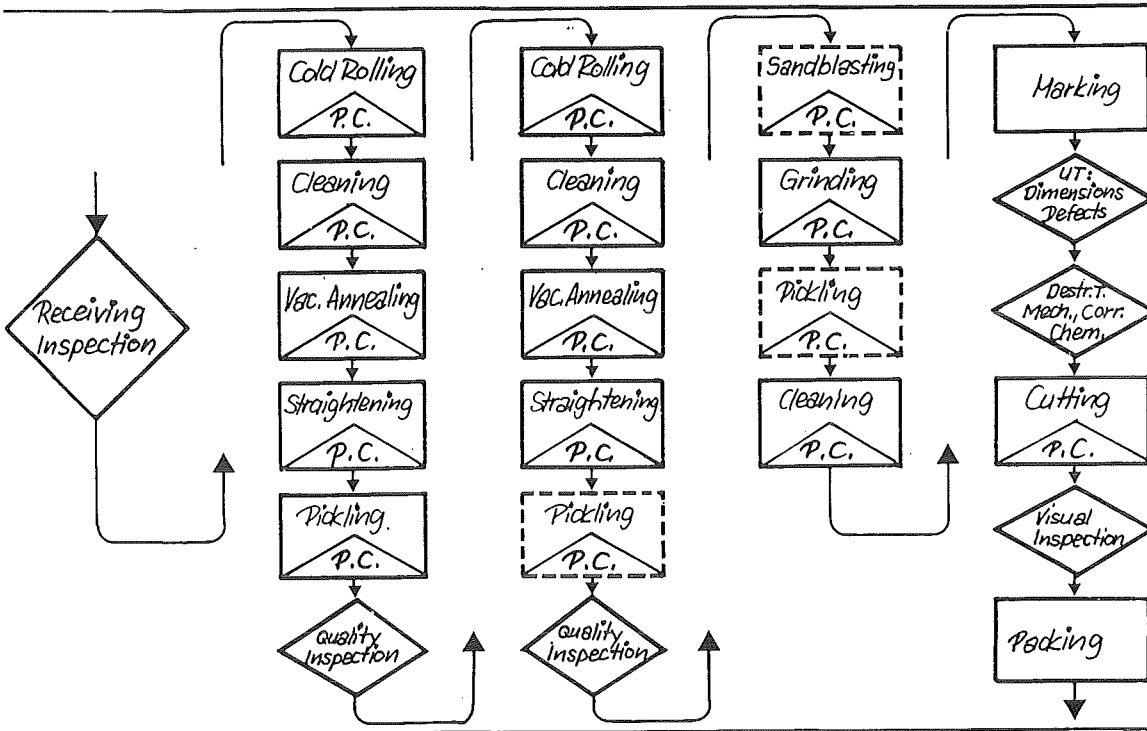
Figure 3   Schematic Flow Diagramm of Zircaloy Cladding Tube
                Fabrication and Quality Control
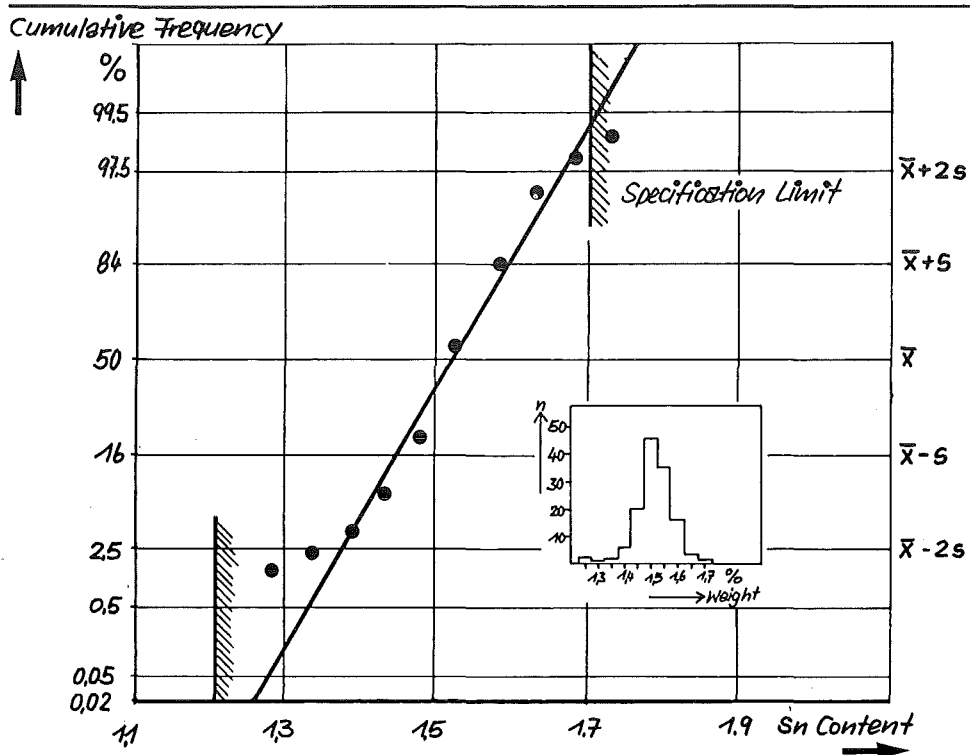
Figure 4   "Normal" Distribution of Sn Content Zircaloy 4
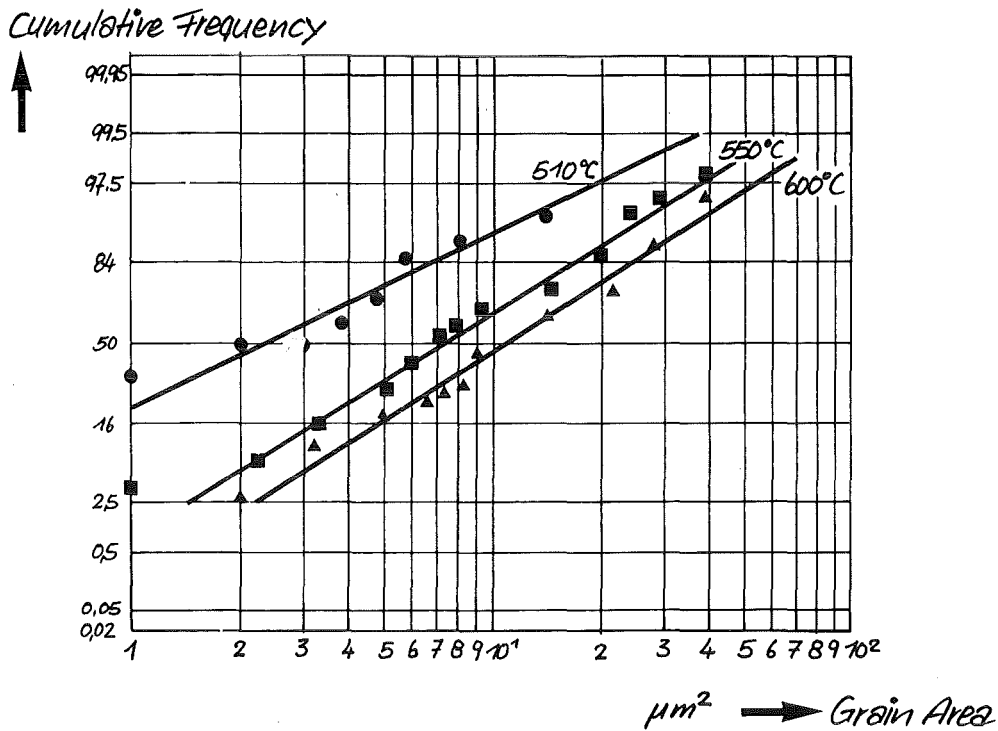
Figure 5   "Log-normal" Distribution of Grain Area of Zircaloy 4 Depending on Temperature of Final Annealing
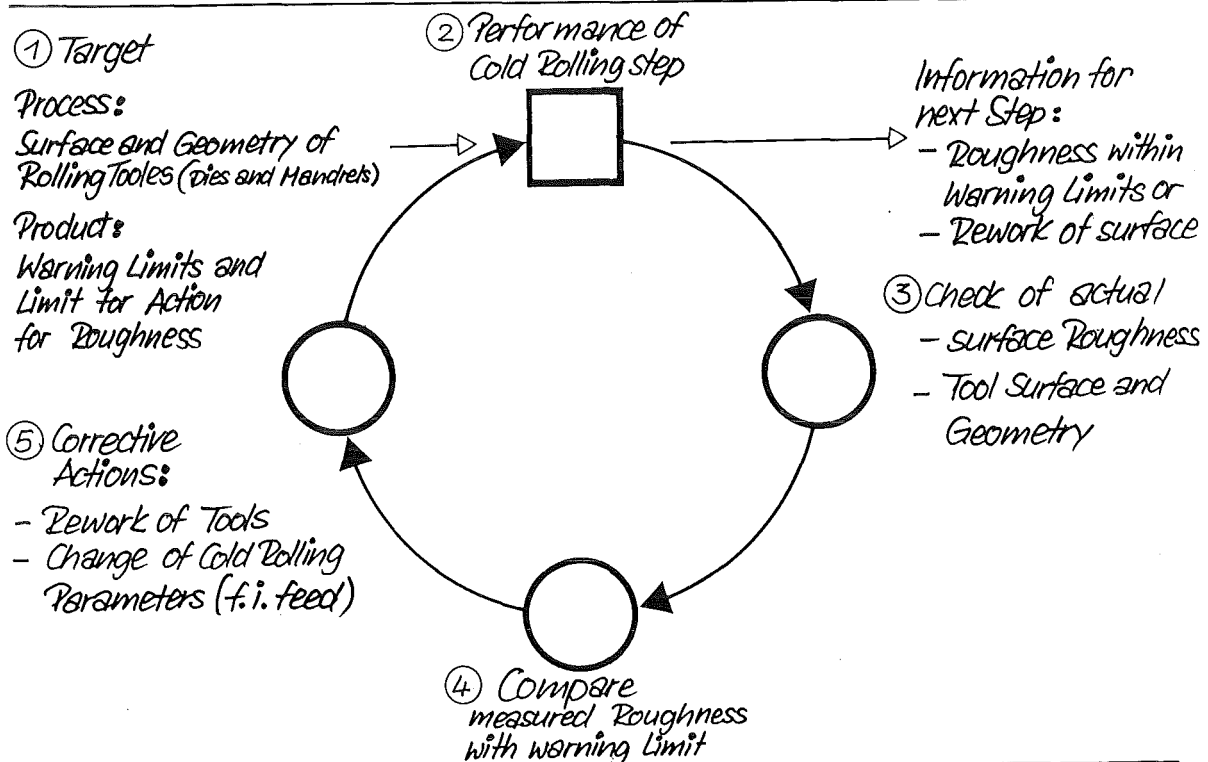
Figure 6   Process Control during Cold Rolling on Surface Roughness
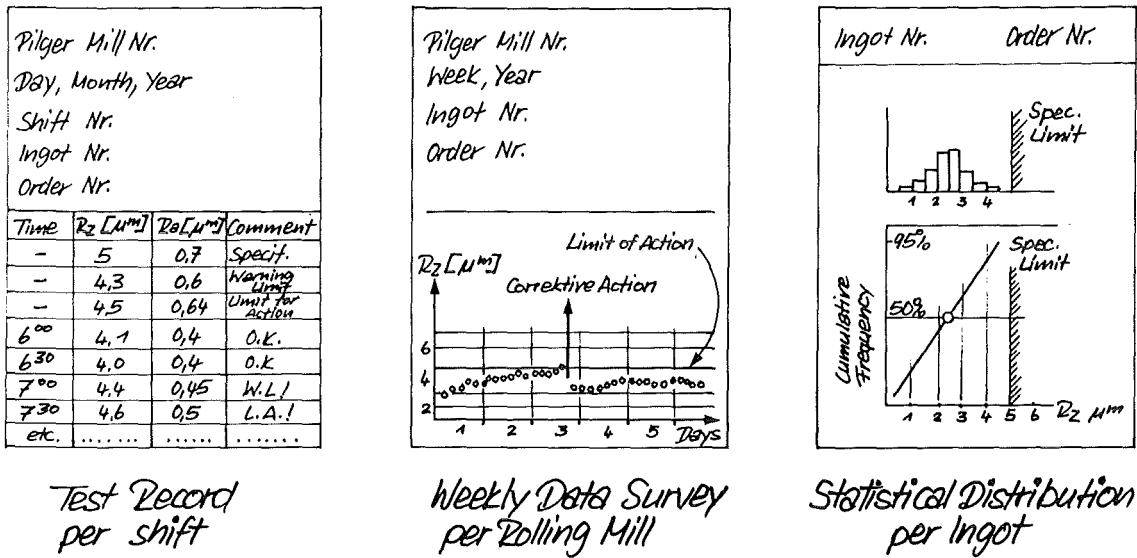
Figure 7   Evaluation and Documentation of Surface Roughness
Values from Process Control during Cold Rolling
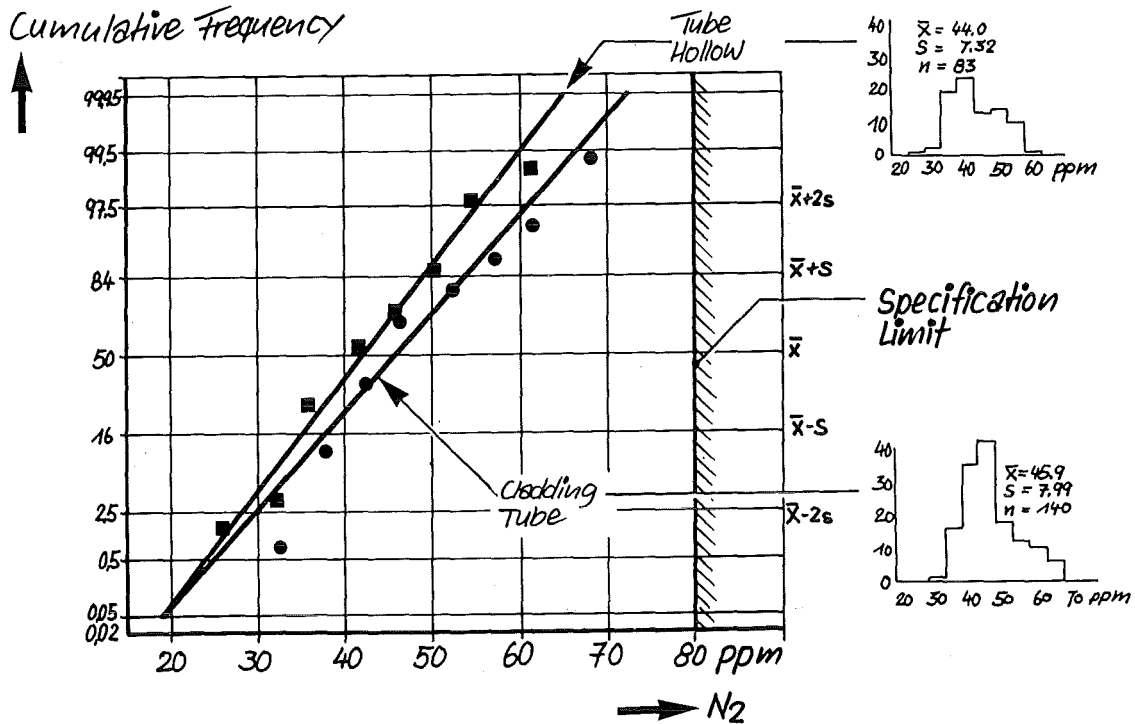
Figure 8   Comparison of Nitrogen Content in Tube Shell and
Cladding Tube
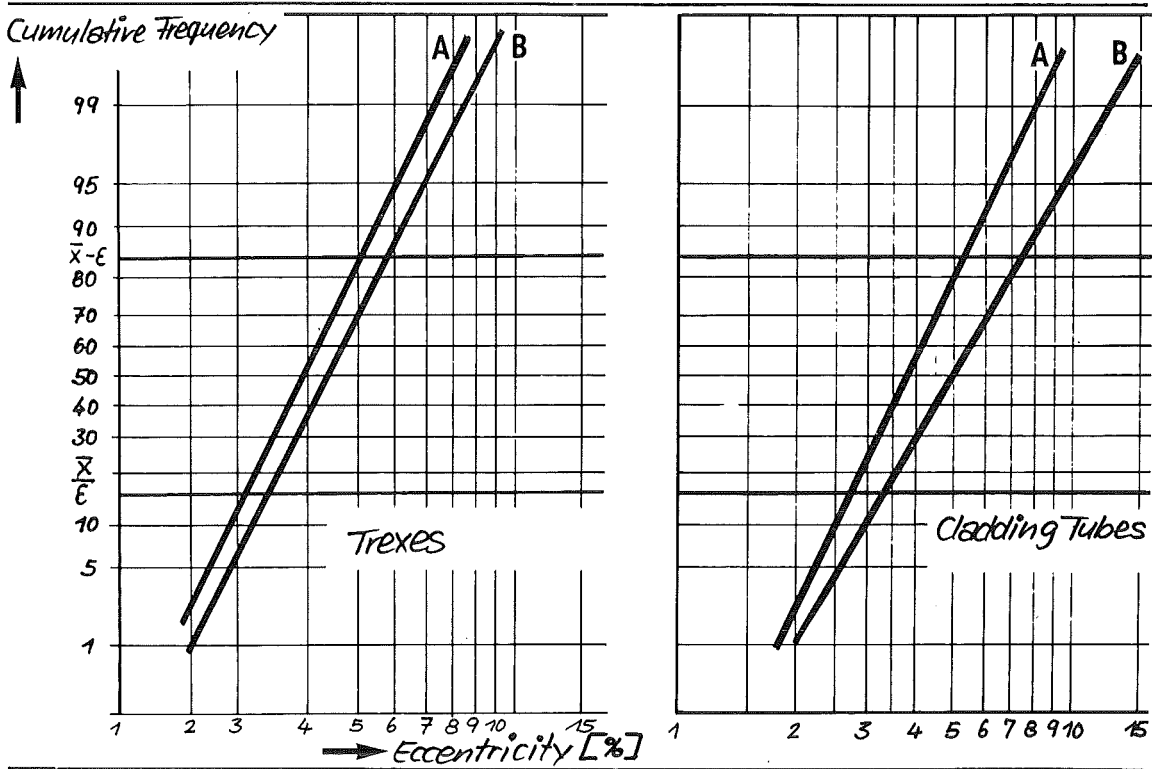
Figure 9   Influence of Trex Eccentricity on Eccentricity of Cladding Tubes, Comparing Statistical Evaluations for two Projects: A and B
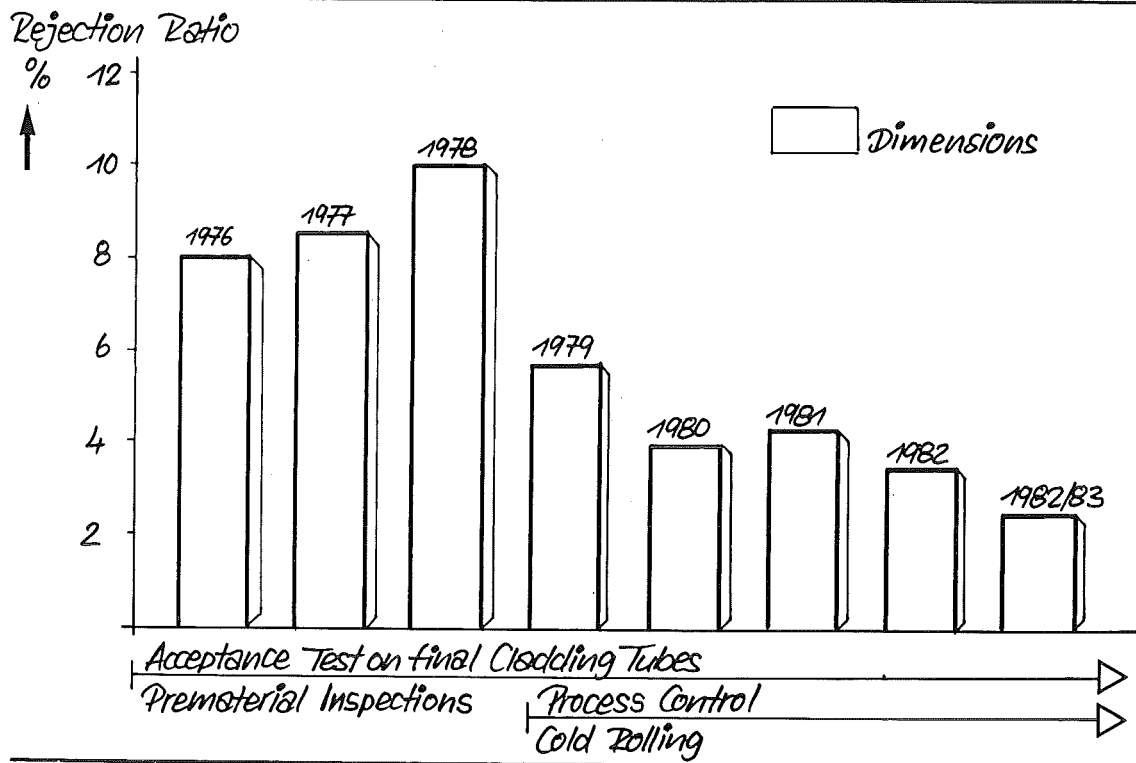
Figure 10   Rejection Ratio for Dimensional Deviations Influence of introduced Process Control on Cold Rolling

Test Target		Inspection of		Test Technique
General	Specific	Certificates	Product	
Chemical Composition	Alloy Elements (Cr, Fe, Ni, etc.)	3-5% per Ingot	4 per lot	Spectrographic Chemical Analysis Atomic Absorption
	Impurity Elements (ASTM)	3-5% per Ingot	—	Spectrographic Chemical Analysis Atomic Absorption
	Sensitive Elements (O_2, N_2, H_2, C)	3 per lot	4 per lot	Hot Extraction
Mechanical Properties	Grain Size	2 per lot	2 per lot	Metallographic Exam.
	Grain Structure	2 per lot	2 per lot	Metallographic Exam.
	Workability			Tentative Fabrication or Compression Test
	Hardness	2 per lot	2 per lot	Hardness-Test (Rockwell HB)

Table I   Quality Control at Incoming Zircaloy Tube Hollows
or Trexes
Part 1: Chemical Composition and Mechanical Properties

Test Target		Inspection of		Test Technique
General	Specific	Certificates	Product	
Dimensions	Outer diameter	100%	Random Sample	Ultrasonic transit time method
	Wall thickness	100%	Random Sample	Ultrasonic transit time method
	Inside diameter	100%	Random Sample	By calculation, O.D. − 2×WTh*
	Eccentricity	100%	Random Sample	Ultrasonic transit time method
	Length	100%	Random Sample	Ruler measurement
	Straightness	100%	Random Sample	With Thickness Gauges on a Reference Surface
Surface	Imperfections Contamination	100%	100%	Optical, Microscope, Visual
	Identity	100%	100%	Visual Inspections
	Roughness	Random Sample	Random Sample	'Pick-up transformer
Defects	Cracks, Inclusions Imperfections	100%	Random Sample	Ultrasonic Pulse Echo Technique
Nuclear Properties	Neutron Cross Section Only on special request in Purchase order	1 per Ingot		Nuclear Test

Table II Quality Control at Incoming Zircaloy Tube Hollows
or Trexes
Part 2: Dimensions, Surface, Defects, Nuclear Properties

General Criteria	Specific Criteria	Cladding Tube Properties
● Mechanical Integrity	Strength/Ductility (Mechanical Pellet, Cladding Interaction)	Min (and max) axial Yield / Ultimate Strength and Plastic Deformation (unif./rupture) at elevated Temperatures (E.T.) Min. circumferential unif./Burst Elongation at E.T.
	Corrosion / H-Pick up (From Coolant)	Corrosion rate in pressurized Water Steam at E.T.
	Stress Corrosion Cracking (From Fission Products)	Max Surface Roughness [SCC-Susceptibility in Standard Test]
	Material Soundness	Max. Size of Flaws (stand def. ≤ 5% WTh.)

Table IIIa   General and Specific Design Criteria for Zircaloy-Cladding Tubes and Resulting Set of Mechanical Properties
/   /: not Objective of Acceptance Test

General Criteria	Specific Criteria	Cladding Tube Properties
● Dimensional Stability	Irridiation induced Growth	Dimensions : O.D., I.D., min WTh. Eccentricity, Ovality (Max. circumferential Creep Rate at E.T.)
	Creep	[Cold Work or Degree of Recrystallization] [Microstructure and Texture] (Hydride Orientation)

Table IIIb   General and Specific Design Criteria for Zircaloy-Cladding Tubes and Resulting Set of Mechanical Properties
/   /: not Objective of Acceptance Test

Procedure	Check of Process		Check of intermed. Product	
	Process Parameters	Test Method	Product Property	Test Method
Cold Rolling	Tool Geometries Feed Speed of Lift Lubrication	Spec. Measuring Equipment } Inspection	Dimension Surface Inspection	Micrometer Inspection Pulse Echo U.T.
Cleaning (Hot detergent, high pressure application)	Bath Composition Temperature Pressure	Chem. Analysis } Termocouples a. } spec. Gadgets + Recording	Cleanliness	Wiper Test Visual Inspection
Vacuum Annealing	Temperature Time Vacuum	} Termocouples } a. spec. Gadgets } + Recording	Surface Mechanical Properties (Final Anneal only)	Inspection Tensile Test Burst Test Structure and Texture Investigation

Table IV   Process Control at Cold Rolling, Cleaning and Vacuum Annealing for Zircaloy-Cladding Tube Fabrication

Procedure	Check of Process		Check of intermed. Product	
	Process Parameter	Test Method	Product Property	Test Method
Straightening	Straighten Triangle	Machine Adjustment	Straightness	Gauges check space to standards
Pickling	Bath Composition PH Time, Temperature	Chem. Analysis Ph-Meter Watch, Th. Couples	Weight Surface Dimensions	Scale Inspection Gauges
Sandblasting	Granulation Pressure	Sieve Technique Pressure Gauge	Roughness Cleanliness	'Pick up' transformer Inspection
Grinding	Granulation Speed	Visual Inspection Speed Gauge	Roughness Surface	Material Removal 'Pick-up'transformer Inspection
Cutting to length	—	—	Length	Ruler measurement (Gauge calibration)

Table V   Process Control at Diverse Fabrication Procedures for Zircaloy-Cladding Tubes

Test Target		Test Technique	Test Frequency
General	Specific		
Dimensions	Outside Diameter Wall thickness Wall variation Inside Diameters	} Ultrasonic transit Time Method Comput. Calcul. O.D. -2×WTh.	100% 100% 100%
	Length	Ruler measurement	100%
	Straightness	With thickness Gauges to standards	100%
Defects	Cracks, Imperfections −outside parallel −outside transverse −inside parallel −Inside transverse	Ultrasonic Pulse Echo Method	100%
	Flat defects	Indicated by Wall Thickness Measurement	100%
Surface	Imperfections Contaminations	Optical and visual	100%
	Roughness	'Pick up' transformer	2 per lot

Table VI   Quality Control for Final Acceptance of Zircaloy-Cladding Tubes I: Dimensions, Defects, Surface

Test Target		Test Technique	Test Frequency
General	Specific		
Chemical Composition	Gases (H_2, O_2, N_2)	Hot extraction	2 per lot
Mechanical Properties	Tensile Test (RT)	Tensile Test	3 per lot
	Tensile Test (E.T.)	Tensile Test	3 per lot
	Creep elongation (E.T.)	Creep Test	3 per lot
	Burst elongation (E.T.)	Burst Test	3 per lot
Structural Properties	Corrosion at sample	Auto clave, steam (ASTM-G-2)	2 per lot
	Grain Size	Metallography	1 per lot
	Hydride orientation	Hydrogenation, Metallography	only for qualification
	Texture	X-Ray Goniometer	only for qualification
	Micro-Hardness	Vickers Hardness	only for qualification
	Micro-Structure	Metallography, TEM, SEM	only for qualification
	Jodine-SCC	Standard-Test SCC	only for qualification

Table VII   Quality Control for Final Acceptance of Zircaloy-Cladding Tubes II: Chemical Composition, Mechanical Properties, Structural Properties

DISCUSSION

K. BALARAMA MOORTHY: a) Your opinion regarding the necessity of carrying out β-quenching the tube shells just prior to the last cold rolling (Pilgering) step.
b) Use of barrier fuel tubing and quality control methods adopted for assessing tube-quality thickness and bonding of inner layer,etc.

H. WEIDINGER: a) The later in the sequence of cold rolling and annealing β-quenching is applied, the more certain you may be with respect to corrosion. But you have to be aware of possible work-ability problems after β-quench.
b) The Zr-barrier tube can be produced and controlled by the same technology as developed and practiced for standard tubes, except for additional QC-checks which are specifically necessary to control the interesting properties of the liner (i.e. thickness etc).

D. SLABU: In 1983 the rate of the rejected tubes was given as 3%. Are these tubes finally rejected or will they be repaired?

H. WEIDINGER: If the type of deviation are (dimensions) they will be rejected. But other deviations (i.e. roughness, etc.) may be repaired.

V. GORSKY: a) Do you use ultrasonic test in your tubes at an intermediate stage of working, e.g. after the first rolling?
b) What is your working frequency in ultrasonic testing of the finished tubes?

H. WEIDINGER: a) No we do not. b) Standard test frequency is 5 Mhz.

COMPUTERIZED ULTRASONIC TUBE TESTING SYSTEMS WITH SPECIAL
REFERENCE TO THE INSTALLATION IN TUBE MILLS FOR NUCLEAR
TUBING OF FUEL FABRICATION FACILITIES

R. Rauscher,   G. Buken

NUKEM GmbH

GB Nutronik

Rodenbacher Chaussee 6

D-6450 Hanau 11

During the last years testing systems for high quality tubes had to offer more
and more possibilities except flaw detection and dimensional measurement. These
additional facilites, which are required by the tube manufacturers on one hand
and the tube customers on the other hand, can be summarized in the slogan
"statistics". The possibilities of a modern tube testing system in connection
with digital "on-line" data processing are shown by two typical examples.

During the last years testing systems for high quality tubes had to offer more and more possibilities except flaw detection and dimensional measurement. These additonal facilities, which are required by the tube manufacturers on one hand and the tube customers on the other hand, can be summarized in the slogan "statistics". I would like to show you the possibilities of a modern tube testing system in connection with digital "on-line" data processing in using two typical examples which are standard configurations of NUKEM rotating head systems:

1. System qualification and monitoring by using the socalled "control card technique".

2. Production monitoring by statistical methods.

To 1.: Control Card Technique

For several years customers for nuclear tubes, especially for cladding material for fuel rods, ask for a demonstration of the accuracy and reproducibility of the testing system which is used by the tube manufacturer as well as for periodic checks of these data. For this reason the production test starts with a series of test runs of a calibrated standard tube. The reproducibility is checked by comparing the results of the test runs and printout of a control card. Then, during normal tube production, the standard tube is run through the system more or less periodically and the results are compared with the initial control card.

First of all let's have a look to a typical standard tube with its standard notches in longitudinal and transverse direction as well as the dimensional standards for outer diameter, wall thickness etc.

The following slide shows the distances and lengths of the windows which are given into the computer for every channel and every piece of the standard tube. We assume that after several test runs the ultrasonic or eddy current signals of one channel will result in a mean value $\bar{x}$ and a deviation . During production test tolerances are introduced which are responsible whether a tube is accepted or rejected. $\bar{x}$ is defined as the sum of all individually measured values devided by the quantitity of values. The deviation   is defined as

$$\sigma = \sqrt{\frac{(x_i - \bar{x})^2}{n - 1}}$$

For practical tube testing during production the main question is: how far is it allowed to compare results coming from a standard tube with practical results coming from natural defects. In other words: which will be the factor of unsafety for production check in case of deviations in standard tube test.

First of all, I would like to give you some rough information on the testing system itself:

The ROTA 25 RD is a high speed ultrasonic rotating head machine for testing high quality tubes in the diameter range of 1/4" to 1". The speed of rotations is continuously variable up to 8000 rpm. Flaw detection and dimensional check is done in one single pass of the tube, whereby the max. pulse rep. rate for the flaw detection is 20 kHz for every channel and for the dimensional check is 5 kHz. These parameters result in a pulse density on the tube surface of 1/100 of an inch for the flaw detection an 4/100 of an inch for the dimensional measurement.

The transducers (up to 8) can be adjusted in the rotary head during standstill without any additional system. Combined tube drive and centering units care for tube transport and perfect guidance, even for delicate, thinwalled tubes. The tube drive speed is continuously variable. A typical throughput speed for a 100 % check according to nuclear specifications is 16 m/minute, which is roughly 50 ft/minute.

For production check using statistical methods the comparison between tests made at random and tube lot results is done as follows: in case single values are picked out of lot values it is possible to use the same definitions for mean values and standard deviations for these samples out of a lot: The mean values of the samples $\bar{x}$ as well as the standard deviation of the mean values which will in the following be called $\tilde{\sigma}$. It is self evident that the sample values of a lot: mean value and deviation always have got a socalled normal distribution.

For the valuation of production processes the socalled control card with know $\bar{\bar{x}}$ and known standard deviation is used on a broad basis. If $\bar{\bar{x}}$ and   are known for a certain production process it is quite well possible to indicate the probability a certain value will exceed the required standard when using the natural deviation of the mean value $\bar{x}$ of samples.

Slide: 95,9 % of $\bar{x}$; inbetween $\bar{\bar{x}} \pm$ 2

99,7 % of $\bar{x}$; inbetween $\bar{\bar{x}} \pm$ 3

If $\bar{\bar{x}}$ and   are unknown, these values must first be verified by taking samples out of a lot. Transferred to tube testing this means the frequent run of the standard tube.

For some time such an on-line data processing system and a programmable statistic valuation is available for system qualification in ultrasonic tube testing. Depending on customer's demands first of all a preset run of the standard tube is carried out in order to print out the "control card". A typical control card is shown in the next slide.

During production test frequent runs of the standard tube are carried out again more or less periodically, let's say once a shift or once a week.

The referring printout is going to be compared with the initial control card. By these means it is possible to verify whether the system is operating correctly or whether there is a "systematic defect", for instance diminuation of the gain of one channel, deterioration of the mechanical guidance etc.

Using the control card and statistical methods it is also possible to preset monitoring or reject levels. We think it should even be possible to shift the reject levels during tube testing in favour of the tube manufacturer. But this is point which is exclusively part of the philosophy of the tube manufacturer.

The complete configuration consists of two microcomputer systems type DEC LSI 11/23, which are connected to each other. The necessary peripheral equipment as floppy disc and input/output terminal are part of whole system.

To 2.: Production monitoring

During normal production test the configuration takes over the following tasks:

2.1 Printout of a certificate for every tested tube, consisting of:
- tube number according to the automatically counted and displayed number
- test result with the informations:
  accepted or
  flaw defects with indication: channel and number of flaws per channel
  dimensional defects with indication: kind of defect and quantity of
  "over high level" and "under low level"
  (the ovality values have got two valuation levels).

2.2 Storage of data per tube. The following data are stored for every tube until conclusion of the test of a whole tube lot:

- quantity of flaw for each channel
- minimum, maximum and medium OD
- minimum, maximum and medium WT
- minimum, maximum and medium ID
- maximum ovality
- maximum wall variation.

2.3 Printout of a certificate per tube lot. Tube lot end is indicated to the computer by the operator. Consequently the printout of the certificate for the tested tube lot an interrogation of the parameters for the new lot are carried out. The output certificate includes the following data:

- free space for comment
- date, time
- operator
- lot-no.
- quantity of tested tubes (without standard tube runs)
- quantity of tubes per sorting criteria, which means:
  . acceptec
  . outer diameter under low level
  . wall thickness under low level
  . ovality exceeding
  . inner diameter under low level
  . flaw defect
  . outer diameter over high level
  . wall thickness over high level

- for od, wt and id each:
  . absolute minimum value
  . absolute maximum value
  . mean value of the mean values $\bar{x}$
  . standard deviation of the mean values ($\bar{x}$)

- for ovality, wall-variation each:
  . absolute maximum value
- interrogation, whether a histogram should be printed
- if yes: interrogation from which value, maximum, minimum or mean value of id, wt or od, and max value of wv and ovality. The classification of the histogram is calculated using statistical formulas. The display is graphic with the resolution given by the print.

When starting a new tube lot the data of the old one are automatically erased.

Of course it is possible to connect the whole system to a centralized big computer in a tube mill in order to either store the data for future checks or to get a more and more complete survey on the whole production with a possibility to control the different steps.

Parameters of testing system, positions of test tube

	Distance	Window
T1 (ext):	1628	10
T1 (int):	1772	10
L1 (ext):	1705	10
L1 (int):	1560	10
L2 (ext):	1704	10
L2 (int):	1561	10
T2 (ext):	1629	10
T2 (int):	1766	10
WT + :	1700	20
WT - :	1800	20
OD + :	1900	20
OD - :	2000	20
ID + :	2100	20
ID - :	2200	20
EC x :	2300	20
EC y :	2400	20

Control-card print-out "US-flaws"

Control card critere	Number of X /	Test-Tube S	X/ Lcl	Lch	Lcl	S Lch	00-APR-83 13:30:50 T
2 T1 (ext):	15.756	2.1293	11.634	19.879	0.209	5.383	9.155
3 T1 (int):	25.355	0.4065	24.568	26.142	0.040	1.028	24.095
4 L1 (ext):	25.476	0.8041	23.919	27.033	0.079	2.033	22.983
5 L1 (int):	16.473	1.5906	13.393	19.552	0.156	4.021	11.542
6 L2 (ext):	27.378	2.2851	22.953	31.802	0.224	5.777	20.294
7 L2 (int):	18.954	2.5239	14.067	23.840	0.247	6.380	11.130
8 T2 (ext):	20.209	2.1126	16.119	24.299	0.207	5.341	13.660
9 T2 (int):	24.800	1.8669	21.185	28.414	0.183	4.720	19.012

Print-out after control run, after produced control-card

Control-runs   Number of test tube   00-APR-83   13:07:36

2 T1 (ext): X/ = 15.303 s = 0.9522 Xi = 14.630   16.650   14.630
         : *

3 T1 (int): X/ = 25.442 s = 0.0157 Xi = 25.453   25.453   25.420
         : *

4 L1 (ext): X/ = 24.956 s = 0.6875 Xi = 24.100   25.783   24.983
         : *

5 L1 (int): X/ = 18.092 s = 0.7695 Xi = 18.310   17.060   18.907
         : *

6 L2 (ext): X/ = 28.630 s = 0.1605 Xi = 28.733   28.403   28.753
         : *

7 L2 (int): X/ = 15.420 s = 0.3537 Xi = 15.067   15.903   15.290
         : *

8 T2 (ext): X/ = 19.682 s = 1.5439 Xi = 20.523   21.007   17.517
         : *

9 T2 (int): X/ = 25.896 s = 0.0042 Xi = 25.897   25.890   25.900
         : *

Statistic definitions

$$\bar{x} = \text{mean value} \qquad\qquad = \text{dispersion}$$

$$= \frac{\sum Xi}{i} \qquad\qquad = \sqrt{\frac{(Xi-\bar{x})^2}{n-1}}$$

$\overline{\overline{X}} + 3\acute{o}$

$\overline{\overline{X}} + 2\acute{o}$

$\overline{\overline{X}}$

$\overline{\overline{X}} - 2\acute{o}$

$\overline{\overline{X}} - 3\acute{o}$
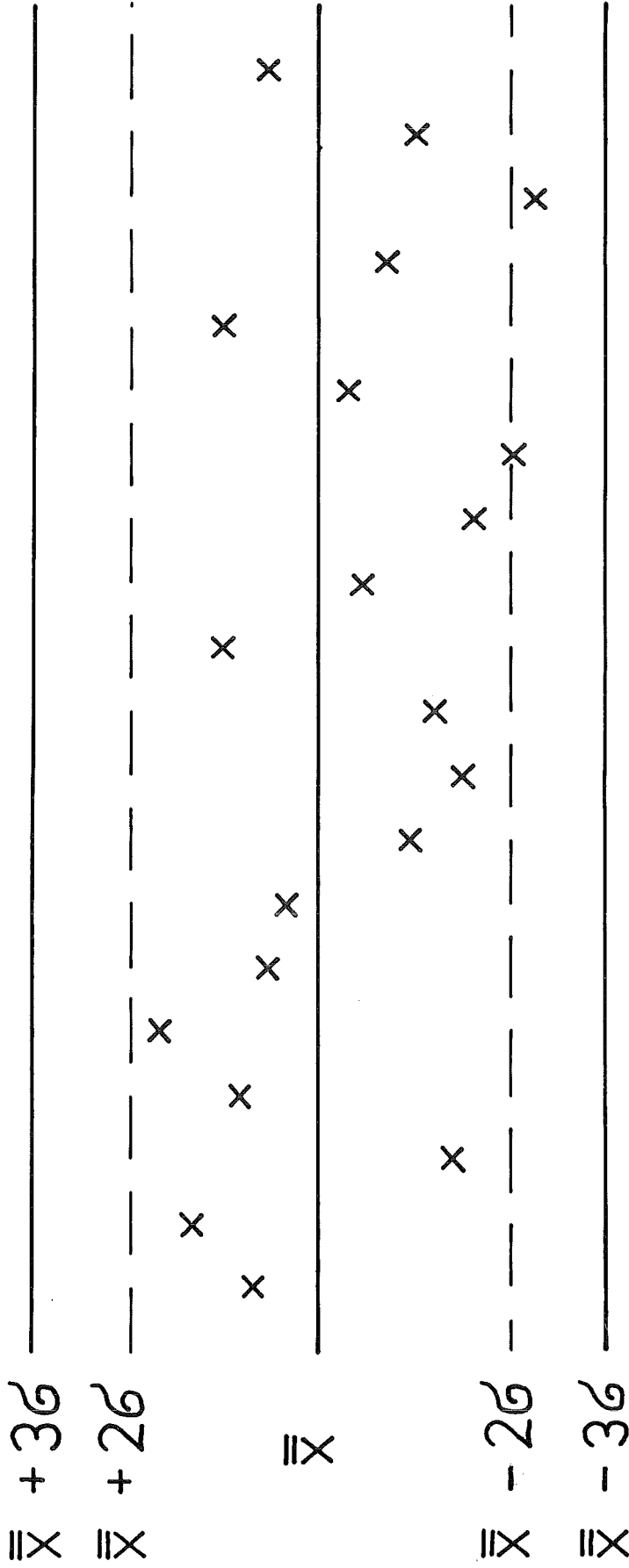
REGELGRENZE, RÜCKWEISGRENZE BEI MEHRFACHPRÜFUNG
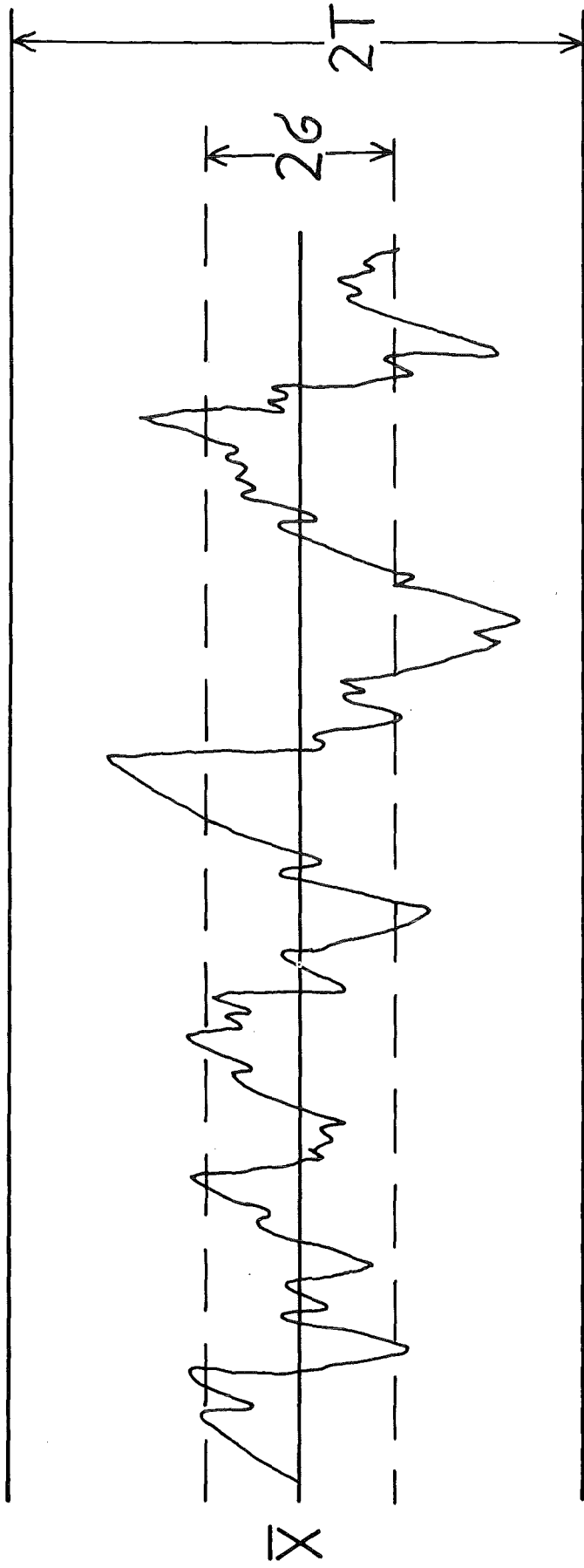
ANZEIGEN SCHWANKUNG BEI MEHRFACHPRÜFUNG
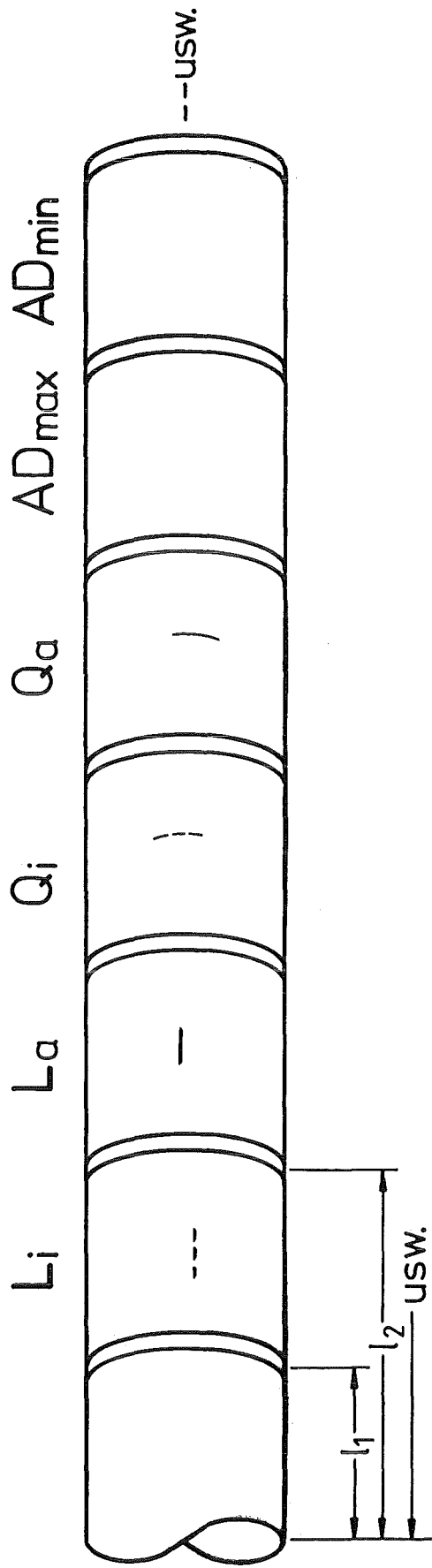
TESTROHRAUFBAU

Methods for QC of Fuel Assemblies

G. Dressler and F. Hetzler

Exxon Nuclear GmbH, Lingen, Federal Republic of Germany
Consultant, Tacoma, Washington, USA

The QC of a fuel assembly may consist of more than 100,000 individual attribute measurements to produce the necessary number of approved fuel pins to make up a completed assembly. A "skeleton," made up of spacers, tie plates and tie rods or control rod guide tubes is required to provide the critical geometric positioning of the fuel rods. The "skeleton" is critical to fix each rod in a precise predetermined 3-dimensional position within a reactor core and maintain that position during a service life where the rod will be subjected to severe conditions of thermal and hydraulic forces.

The main inspection attributes for PWR and BWR fuel assemblies are as follows:

Main Inspection Attributes

PWR and BWR

1. Cleanliness and workmanship (visual).
2. Orientation and identification of assembly and components (spacers and tie plates).
3. Length.
4. Envelope.
5. Perpendicularity.
6. Rod/rod (rod/guide tube) separation.

PWR

7. Functional tests with reactor interface gauge (tie plates only).
8. Insertion test with plugging device and rod control cluster (withdrawal force).

BWR

9. Rod type and position (by enrichment marks and templates).
10. Straightness of end cap shanks following assembly.

In general, PWR fuel assemblies require a much greater effort to control specified rod-to-rod and rod-to-guide tube clearances. The data generated are submitted for computer analysis, bow and twist are calculated at the same time defining the total fuel assembly envelope. A BWR fuel assembly envelope is controlled with a short "channel" gauge. The inner envelope of the "channel" gauge is dimensioned to control the maximum envelope including bow and twist of the assembly.

BWR fuel assemblies, however, require much more attention with regard to enrichment to assure that differently enriched rods are placed according to specification. Typical control methods include administrative measures, special notches on the end cap shanks, serial numbering to indicate enrichment and rod type and template control of finished fuel assemblies.

Two examples of QC practices exercised by Exxon Nuclear at its USA (Richland, Washington) and European (Lingen, Federal Republic of Germany) fabrication facilities address the above. Functional gauging and fuel assembly characterization both share practical application in the Exxon Nuclear fuel fabrication environment. Both practices also relate to verification of fuel assembly design specifications and/or in-core performance.

Functional Gauging

Functional gauging is probably most familiar in the application of "Go/No-Go" rings, pins or bars to control simple geometric attributes. Examples include maximum diametral dimension of end plugs and pellets and pin-to-hole clearances. In addition to checking the critical dimensions on individual fuel components, Exxon Nuclear also applies functional gauging to check reactor interface features of fuel assemblies such as dowell hole locations, fuel assembly envelopes, fuel handling surfaces and guide tube locations. For PWR fuel assemblies, functional gauges are used to check the compatibility of upper and lower tie plates with respective reactor core components.

A specific example of functional gauging practiced by Exxon Nuclear is that accorded to tie plates. Tie plates are stainless steel castings or weldments machined to final dimension by conventional and numerically controlled milling machines. The tie plates are attached to each end of a fuel assembly with interconnecting mechanical fasteners (BWR tie rods or PWR guide tubes) to secure the individual fuel assembly components into place.

Figure 1 shows a typical BWR lower tie plate and the corresponding functional gauge. The functional gauge shown on the left is used to inspect the true position of the hole pattern in the tie plate. It is not practical
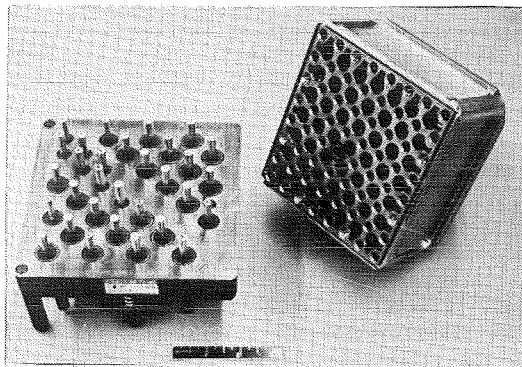
Figure 1: Functional Gauging of a BWR Tie Plate

to measure all holes at once because of the close pitch and the total number of holes. Instead, one-half of the holes are checked at one time by having 50% of the required "Go/No-Go" pins installed in the gauge. The tie plate is set onto the gauge, then removed and rotated a half turn and set back on the gauge to inspect all holes. The "Go/No-Go" pins can be removed from the gauge, so that if a tie plate does not fit onto the gauge, individual pins can be removed and replaced until the dimensional problem is located.

Exxon Nuclear also applies the same relationship gauging principle to PWR end plates to check the true relationship of dowell holes to each other, the dowell holes to the nozzle, and between top and bottom tie plates. In addition, the tie plate envelope is measured with dial indicators attached to the gauge. A typical PWR bottom tie plate mounted in a functional gauge is shown in Figure 2.
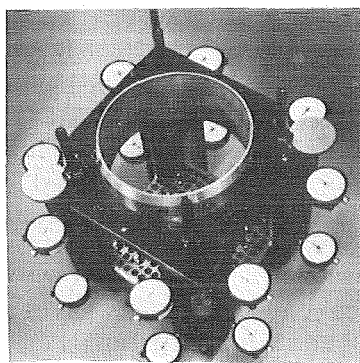
Figure 2: Functional Gauging of a PWR Tie Plate

The QC benefits gained from the use of tie plate (and similar) functional gauges include:

a) Elimination of potential errors inherent in subjective optical projector or comparator measurements through application of objective measuring techniques.

b) Specially trained personnel are not required to accurately interpret measurement results.

c) Measurement is easily repeatable for recheck, for example, to satisfy customer or third party authority quality audits.

d) Elimination of long discussions with vendors regarding interpretation of measurements and subsequent acceptance or rejection of purchased parts (Exxon Nuclear policy is to provide the pertinent function gauges or the gauge design to vendors).

e) Gauge accuracy certification is very compatible to overall QA administration.

f) Translation of fuel assembly design requirements directly into the gauge.

Additional benefits are also realized to fuel fabrication operations:

g)   Exxon Nuclear observed a 50% reduction in tie plate QC manpower requirements in the first year of functional gauge application.

h)   Convenient, easy-to-use equipment compatible with fabrication process control, particularly processes which apply functional machining fixtures.

Fuel Assemby Characterization

Fuel assembly characterization is typically applied to initial fabrication of new designs. Exhaustive measurements are recorded from a very limited number of fuel assemblies during and after fabrication but prior to irradiation. The purpose is to provide a very well documented dimensional reference base history for new "as-built" fuel assemblies which can later be used to compare post irradiation observations and measurements.

The same characterization also serves as an excellent tool to check fuel assembly fabrication reproducibility. A PWR fuel assembly, for example, may contain about one-half million individual QC attribute checks. These are combined with design characteristics such as spacer spring to fuel rod contact forces to produce a rather sophisticated set of inter-acting attributes for an individual fuel assembly. Exxon Nuclear, typically characterizes fuel-rod-to-fuel-rod spacings to:

a)   Determine the actual rod-to-rod spacings (average and standard de-viations) in comparison with specified minimums. (This check is also important to avoid hot spots or accidental burn-out when the fuel assembly is operating at power.)

b)   Determine any localized design or fabriction deficiencies along the length of the fuel assembly.

c)   Initiate any necessary corrections in the design of specific com-ponents within the fuel assembly to assure reproducibility in subsequent fuel assembly series fabrication.

The following example shows the characterization of the fuel-rod-to-fuel-rod spacing in two BWR 9x9 array fuel assemblies. The specified minimum rod-to-rod spacing is 2.90-mm. Variables measurement has been substituted for the attributive measurement normally used in series production and the results are tabulated in Table 1, which shows a range of dimensions in millimeters (x) and the frequency of measurements (n_1 and n_2) found within those ranges. Figure 3 shows the measurements in graphical form. The total population (Σn_1 and Σn_2) for each fuel assembly is 1120 which is equivalent to 140 spacer-to-spacer span mea-surements multiplied by 8 spans. The average rod-to-rod spacings ($\bar{x}_1$ and $\bar{x}_2$) are shown in the normal distribution curve together with the respective standard deviations (s_1 and s_2).

Table 2 tabulates the average rod-to-rod measurement (x) in each of the 8 spacer-to-spacer and spacer-to-end-plate spans together with the respective standard deviations (s).

The results from Figure 3 and Table 2 show that:

a)   The average values ($\bar{x}_1$ and $\bar{x}_2$) and the standard deviations (s_1 and s_2) of both fuel assemblies were nearly identical. This indicates a good and consistent fabrication history.

b)   All individual measurements were well above the specified minimum.

c)   There was no significant difference between the values measured for each of the spacer spans.

d)   The measurement data fit well into normal distribution curves which should apply to the type of measurement exercise performed.

Conclusions

Function gauging has been proven to provide a very economical approach to accurate and easily reproducible QC measurements. It is an objective technique and thereby easily interpreted by non-specialists.   The method is very compatible for fabrication process control, especially operations which incorporate functional tooling.

Fuel assembly characterization, such as rod-to-rod spacing measurement, provides an important reference point from which to compare post irradiation observations and measurements.   Characterization also produces measurements valuable to correlate fuel assembly fabrication reproducibility.

TABLE 1. ROD-TO-ROD SPACING FREQUENCIES

RANGE, X, mm	X, mm[1]	N_1[2]	N_1 CALCULATED[3]	N_2[2]	N_2 CALCULATED[3]
3.10 - 3.15	3.125	-	-	1	1
> 3.15 - 3.20	3.175	2	3	6	2
> 3.20 - 3.25	3.225	5	9	3	6
> 3.25 - 3.30	3.275	29	25	10	16
> 3.30 - 3.35	3.325	58	57	51	38
> 3.35 - 3.40	3.375	87	106	62	71
> 3.40 - 3.45	3.425	167	161	123	118
> 3.45 - 3.50	3.475	238	198	175	160
> 3.50 - 3.55	3.525	180	198	196	185
> 3.55 - 3.60	3.575	147	161	180	178
> 3.60 - 3.65	3.625	91	106	112	145
> 3.65 - 3.70	3.675	65	57	78	99
> 3.70 - 3.75	3.725	35	25	71	57
> 3.75 - 3.80	3.775	13	9	39	27
> 3.80 - 3.85	3.825	2	3	7	11
> 3.85 - 3.90	3.875	1	1	4	4
> 3.90 - 3.95	3.925	-	-	2	2

1) MID POINT

2) FREQUENCY PER RANGE

3) CALCULATED FROM AVERAGE AND STANDARD DEVIATION, FIGURE 3

TABLE 2. AXIAL ROD-TO-ROD SPACING DISTRIBUTION

(140 DATA POINTS PER SPACER SPAN)

LOCATION OF MEASUREMENT	X, mm	S, mm
UPPER TIE PLATE/1ST SPACER	3.53	0.13
1ST/2ND SPACER	3.51	0.12
2ND/3RD SPACER	3.49	0.09
3RD/4TH SPACER	3.48	0.09
4TH/5TH SPACER	3.48	0.11
5TH/6TH SPACER	3.50	0.10
6TH/7TH SPACER	3.49	0.11
7TH SPACER/LOWER TIE PLATE	3.53	0.12
FUEL ASSEMBLY 1	3.50	0.11
FUEL ASSEMBLY 2	3.54	0.12

FIGURE 3

ROD-TO-ROD SPACINGS OF TWO BWR 9x9 FUEL ASSEMBLIES

$\bar{x}_1 = 3.50$ mm
$s_1 = 0.11$ mm
$\Sigma n_1 = 1120$

$\bar{x}_2 = 3.54$ mm ⎫ DOTTED
$s_2 = 0.12$ mm ⎰ DISTRIBUTION
$\Sigma n_2 = 1120$

CALCULATED DISTRIBUTION CURVES
(FROM NORMAL DISTRIBUTIONS WITH
$\bar{x}_1$, s_1, AND $\bar{x}_2$, s_2)

SPECIFIED LOWER LIMIT

FREQUENCY

ROD/ROD SPACINGS, mm

—414—

2/84
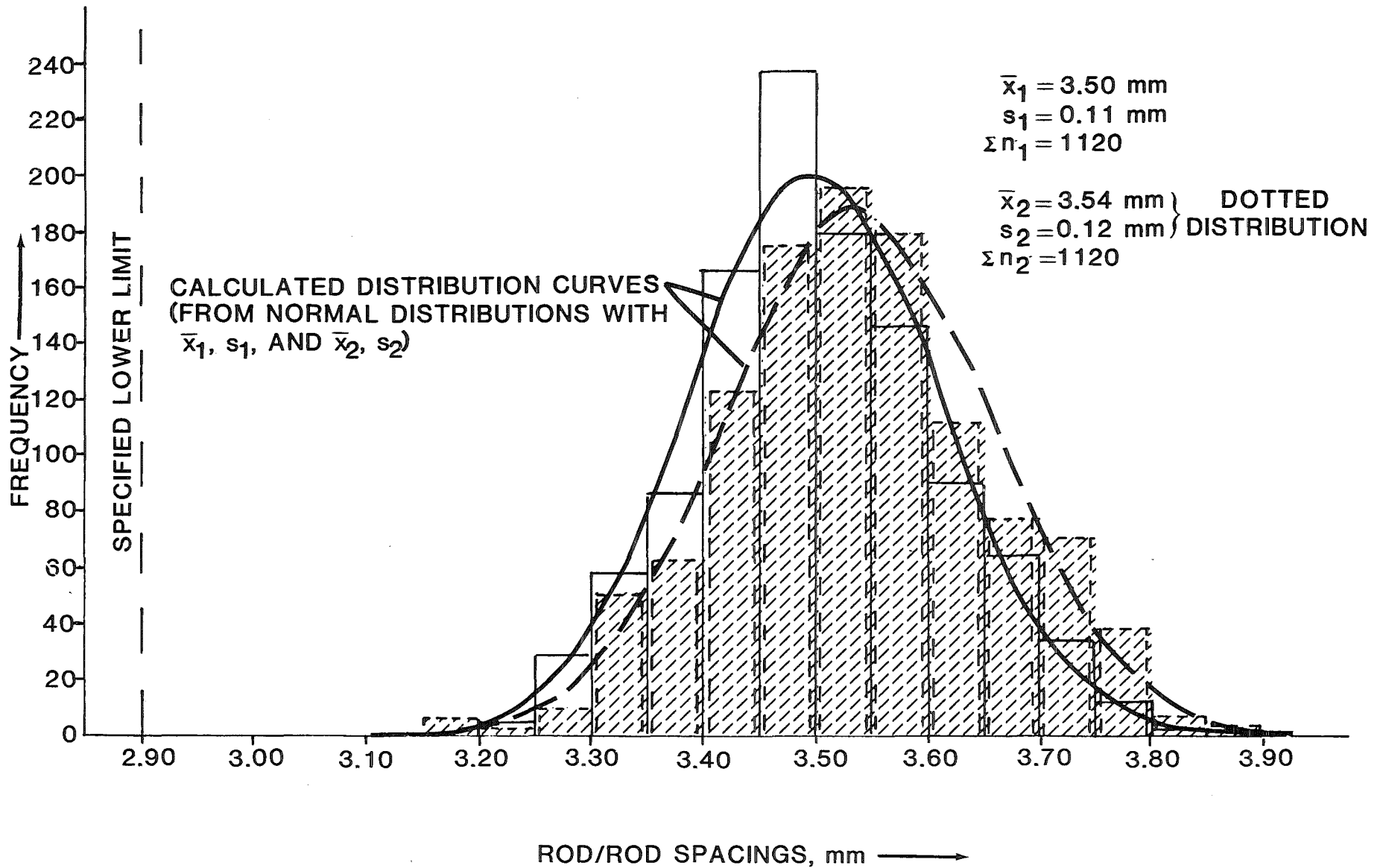
QUALITY CONTROL OF PWR SPACERS BY MEANS OF
A COMPUTER ASSISTED AUTOMATED TEST DEVICE

M. Streb
Reaktor-Brennelement Union, Hanau
J. Steven
Kraftwerk Union, Erlangen
H.-J. Romeiser
Kraftwerk Union, Erlangen

An automated measuring device has been developed for quality control of PWR-spacer grids. Spring forces, pitch and position of knobs are measured using a special transducer which is equiped with a piezo force gauge and two displacement gauges. The spacer is automatically positioned by means of a numerically controlled x/y-coordinate table to each spacer cell to be measured. Test results are stored, evaluated, arranged and recorded by a process computer. Unsatisfactory results are printed in correlation to cell position. After reworking a retest program is generated by the computer which initiates retest of only the reworked cell positions.

1 Introduction

Spacers are important structural components of nuclear fuel assemblies. Their function is to fix the fuel rods in an exact radial position and to enable their thermal and irradiation-induced axial elongation in order to avoid high axial loading of fuel rods which could cause fuel rod bending. The spacers consist of strips that are assembled to a grid. The nodes of the grid are welded or brazed (Fig. 1). Each cell that is destined for taking up a fuel rod contains one or two springs, which press the fuel rod against a number of fixed contact areas on the opposite side of the cell (Fig. 2). The fixed contact areas are knobs which are stamped into the strips. The position of the knobs largely determines the position of the fuel rod, while the spring establishes the fixture force. The minimum fixture force is limited by the requirement to assure defined positioning of the fuel rod without relative tangential movement between rod and spacer. The maximum force is limited by the friction forces which arise during axial elongation of the fuel rods.

The strict requirements which result from this in respect to attaining the close tolerances required for the grid pitch, the position of the knobs, and the spring forces call for large scale quality assurance measures. Owing to the large number of tests involved in the envisaged random checking of grids pitch and knob position and the 100 % checking of the force of each spring, a large volume of data is produced which can hardly be handled by means of conventional methods such as manual measurement and listing of data in test protocols.

Therefore the objective was to develop a computer-assisted test device for an automatic measuring procedure that stores, evaluates, arranges and documents the test data. Moreover, the test device was to be integrated into a data logging system which documents all the quality data during the whole fabrication history of a spacer (Fig. 3). The data from the further tests, i. e. material document checks, material acceptance test results, fabrication checks, and final inspection of overall dimensions, have to be input into the central processing computer manually via terminals for final quality documentation.

2 Description of Concept

The values of the knob pitch T, knob position a and spring force F which are indicated in Fig. 4 have to be determined on the finished spacer. The testing of knob pitch and knob position hitherto been done by means of a manually positioned coordinate measuring device whereas spring forces were measured by a manually handled force transducer. Because the testing of these three characteristics always required the same positioning of different measuring devices into the spacer, it seemed reasonable to combine the seperate tests into one test.

2.1 Measuring Transducer

The fundamental module for the requisite measuring system was the spring force transducer which has already been verified in manual operation. This kind of transducer is shown in Fig. 5. For measurement of the spring force the transducer is pushed into the cell to be tested. The spring presses the transducer, which has a diameter corresponding to that of the fuel rod, against the knobs. A piezo quartz measuring disc is located in the region of the contact area between spring and transducer, which transfers the spring force into a proportional voltage signal.

Because the inserted transducer indicates the real postion in the spacer cell, there is a relationship between the position of the transducer and the position of knobs and the pitch. Use of two displacement transducers located at a fixed distance along the axis of the force transducer finger makes it possible to determine its vertical position. The position of the knobs (angularity) can be determined from the difference between the signals from the two displacement transducers. Moreover, the mean absolute deviation from the ideal position can be used to determinate the pitch once the actual position of the neighbouring cell has already been determined.

2.2 Structure of the Measuring Device

The measuring transducer will be positioned in each cell of the spacer by means of a suitable handling system. The handling system consists of a x-y-coordinate table in combination with a rotating table onto which the spacer is clamped in a special fixture and is positioned at intervals corresponding to the pitch.

The spacer can be turned through an angle of 90° by means of the rotating table for testing of the second spring in the cell.

Because the testing direction remains the same, only one axis needs to have a linear displacement scale for the exact determination of the real position. Because of the symmetry of the spacer grid, it can be divided into four quadrants, each with its own measuring transducer. A pneumatic lifting mechanism is provided for vertical movement of the measuring transducers. When the measuring transducer is pushed into the spacer cell, the transducer will be unblocked to self-center between knobs and spring.

Fig. 6 shows the hardware of the measuring device. Besides the mechanical equipment, it includes a process control computer, a data logging system for all measurement signals, and a numerical positioning control unit. The measurement signals are transmitted online by means of a multiplexer to the computer. The multiplexer contains fourteen inputs. Twelve analog inputs are provided for the eight displacement transducers and the four force transducers. Two digital inputs are provided for the linear scale incremental counter.

The process computer coordinates the complete testing procedure and evaluates the test results. Dialog with the test equipment is by means of a data monitor.

The initiation of the several subfunctions such as input of identity data, automatic test run or positioning of transducer are effected via a portable input terminal. The test results from each spacer cell position or the deviations thereof can be checked at the monitor or printed by the printer. Data exchange with the positioning control unit is established via a serial I/O-interface at which the control unit gives a checkback to the computer when it has reached the desired position. A magnetic disc recording system is used for storage of the operating system, the application programs and the specific data of the various spacer types. The operating system manages and coordinates the application programs and the input/output operations. The application programs contain the subprograms, for instance for calibration, input of identity data (spacer type), the automatic test procedure with positioning coordinates, and several auxiliary and test programs. The process computer, which is located at the measuring device, can be connected to the central computer by a modem operating system.

Before commencement and after a certain operating period, the measuring transducers have to be calibrated (Fig. 7). The calibration device is then clamped into the spacer fixture. This device is used to calibrate the transducers for spring force and displacement.

3 Performance and Results

The test procedure for a spacer involves the following steps:

Input of identity data: Spacer type
                                           Drawing No.
                                           Specification No.
                                           Pitch of knobs
                                           Position of knobs
                                           Force of springs

Generation of data for
numerical positioning control

Calibration of transducers
before start and after a
certain period of operation

Measurement of a spacer in
automatic test run

Comparison of test results
with specified values;
storage of evaluated
test results;
Recording of test results: Order No.
                                           Component No.
                                           Object No.
                                           Spacer type
                                           Specification No.
                                           Drawing No.

Result: e. g.: all requirements met
otherwise a protocol for
repair is issued

The test results can be issued in form of lists in which the results are corre-
lated with the cell positions (Fig. 8). In the case of unsatisfactory results,
a protocol for repair can be issued containing all objectionable results correlated
to the cell position.

Experience from the testing of spacers up to now shows that the objections relate
mostly to the spring forces. The unsatisfactory springs have to be recalibrated
and to be tested again. For this retesting run a special program for positioning
is generated by the process computer to avoid an overall retest. The results of
the retest are likewise issued for the reworked positions only.

All data for each characteristic are summarized for final documentation. The spe-
cified value, the number of tests, and the result of comparison between specified
value and test result are issued for each characteristic i. e. spring force, knob
pitch and knob position. The archived results can be used both for process control
and for the analysis of product quality and the evaluation of defects.

A statistical evaluation of test results is shown in Fig. 9. The histogram of
the spring forces reveals the measure of dispersion in correlation to the specified
tolerance limits.

4 Alternative Measuring Systems

Taking into account that experience reveals a relatively high probability of un-
satisfactory spring force and almost no objections as regards position and pitch
of knob, an alternative measuring system would be of interest. It would be possible
with a modified measuring transducer to calibrate the spring in question during
the measurement procedure itself such a procedure would enable the level of au-
tomation to be increased, because manual recalibration und retesting could be eli-
minated.

Such a modified force transducer is shown in fig. 10. The measuring finger is con-
nected to a piezo-force transducer outside the spacer cell. After insertion of
the finger into the cell it is displaced toward the knobs. At the moment of contact
between the knobs and the movable fulcrum of the finger, positioning is stopped.
This contact position is the reference point for displacement of the finger in
the opposite direction toward the spring by an amount corresponding to the fuel
rod diameter. In this position the effective spring force is transmitted to the
gauge which measures exclusively the shear force. If the measured force exceeds
the specified value, a special subprogram of the positioning control unit is ini-
tiated which conducts the calibration of the spring. This procedure is demonstrated
in Fig. 11 by a spring force vs. displacement diagram.

The finger is displaced in steps of increasing amplitude until the spring force
reaches the specified value. After each step, the finger returns to the measuring
position for retesting of the spring force.

5 Conclusions

The introduction of an automated measuring device for testing of spacers for nu-
clear fuel assemblies has made a contribution to quality control which has several
important advantages over the manual procedure used previously. The reliability
of measurement is increased and documentation is more precise and easy to survey.
The fast evaluation of results makes the level of quality immediately available.

Besides the rationalization effect frequently emphasized, the use of a computer-
assisted measuring device presents a high level of flexibility. It is possible
to use the same measuring device even if the product changes simply by modifying
the programs, which have to be adapted to the new test object. Last but not least
the ergonomic aspect should be mentioned.

Fig. 1:   PWR Spacer Grid
Fig. 2:   Detail of Fuel Rod Clamping
Fig. 3:   Data Flow Sheet of Spacer Quality Control
Fig. 4:   Characteristics of Spacer to be Controlled
Fig. 5:   Former Testing Device for Spacer Spring Forces
Fig. 6:   Flow Sheet of a Computer-Assisted Test Device
Fig. 7:   Calibration Device for Force Transducer
Fig. 8:   List of Test Results
Fig. 9:   Accumulation Distribution of Spacer Spring Forces
Fig. 10:   Modified Force Transducer
Fig. 11:   Adjusting of Spring Forces
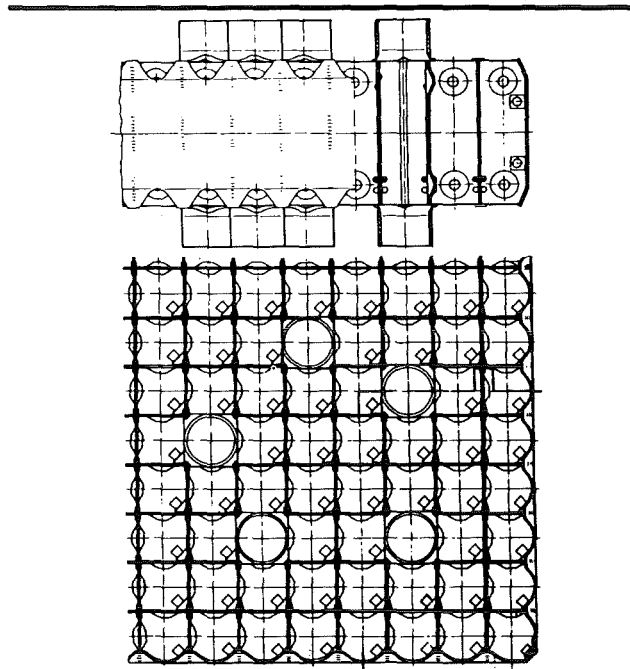
Zircaloy - Spacer for PWR -Fuel Assembly

Fig. 1

Kraftwerk Union

Inconel   Zircaloy

Detail of Fuel Rod Clamping

Fig. 2

Control of Documents

Acceptance Test Certifi-
cates of Subsuppliers
Sheets, Strips, Springs, Sleeves, Bra-
zing Material u o

Data Input
Terminal

Central Process-
Computer   Plant I

Acceptance Testing

Dimensions
Mechanical Properties
Other Requirements

Data Input
Terminal

Manufacturing Control

Assembling
Complete Brazing / Wel-
ding, Quality of Joints
Other Requirem.acc. to Spec

Data Input
Terminal

Materials Testing

Mechanical Strength
Metallographic Investigation
Other Requirem. acc. to Spec.

Final Inspection

Dimensions
Flatness
Other Requirem. acc to Spec.

Data Input
Terminal

Testing Automaton
Spring Force
Position of Knobs
Pitch of Knobs

Data Input
for Identification

Measuring Device -
Computer   Plant II

Data Flow Sheet of Spacer Quality Control

Fig. 3

—

Force F   Pitch of Knobs T   Position of Knobs a

Charateristics of Spacer to be Measured

Fig. 4
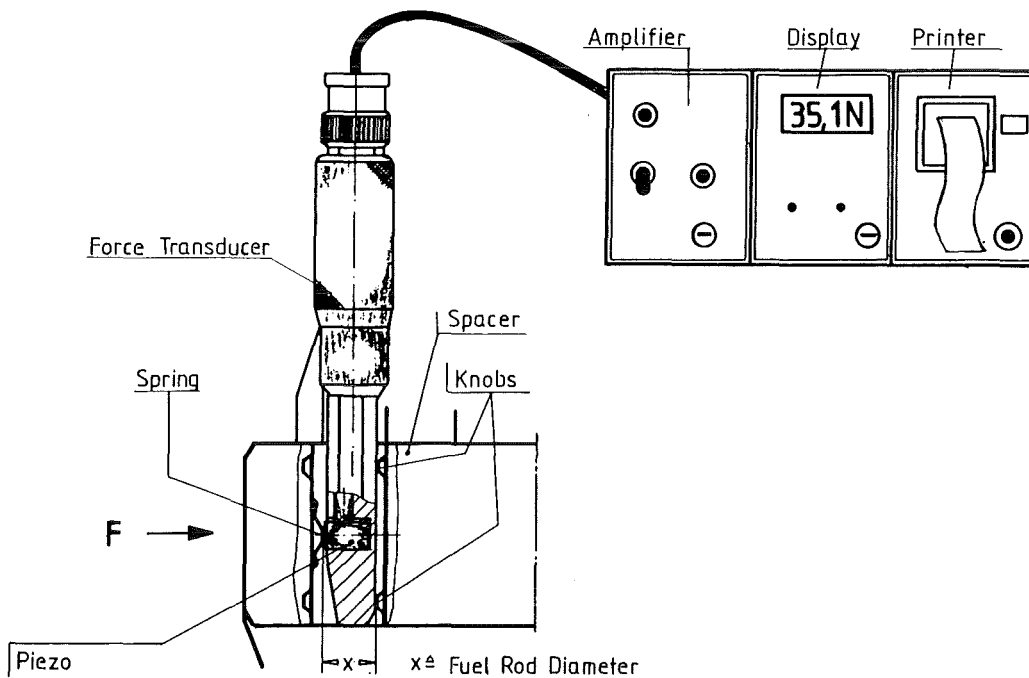
Kraftwerk Union

Former Testing Device for Spacer Spring Forces

Fig. 5

Position of Knobs a

Multiplexer

Force F

Pitch of
Knobs T

Magnetic Disc

Printer

Measuring
Device
Computer

Positioning Control

Data Screen

Operator Desk Terminal
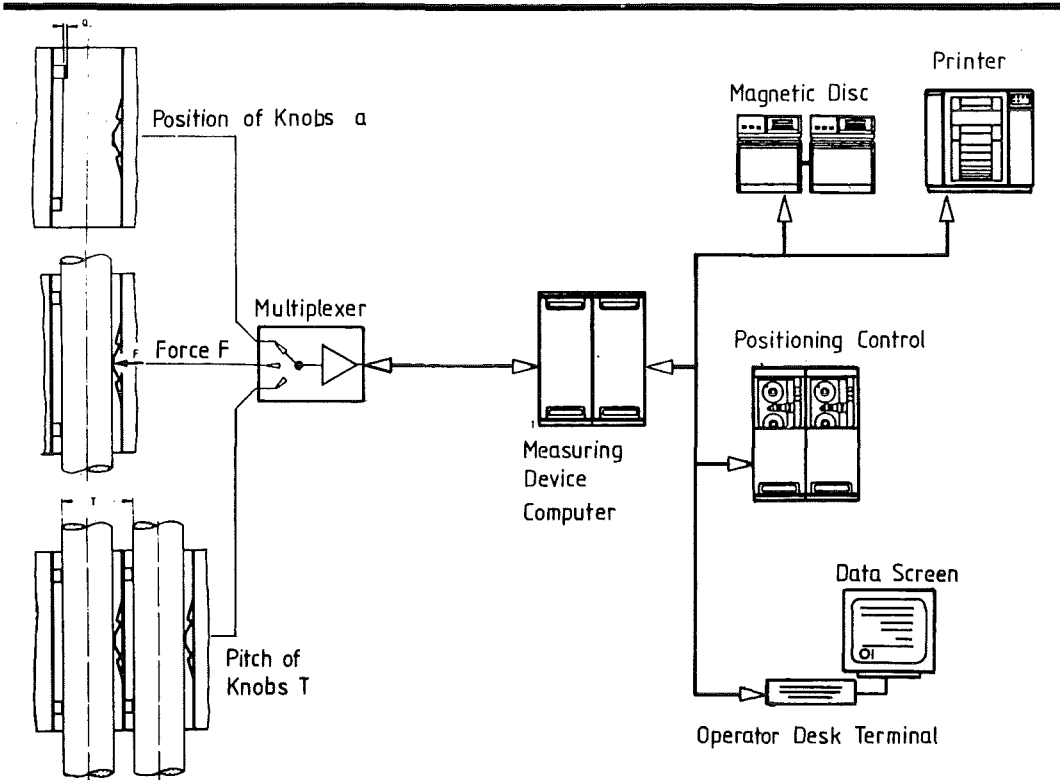
Flow Sheet of a Computer Assisted Test Device (Spacer Test Automaton)

B221.12.83

Fig. 6

Kraftwerk Union

Displacement
Transducer

x ≙ Fuel Rod Diameter

Calibration Device

Loading Roll

Piezo-Force Tranducer

Tensile Spring
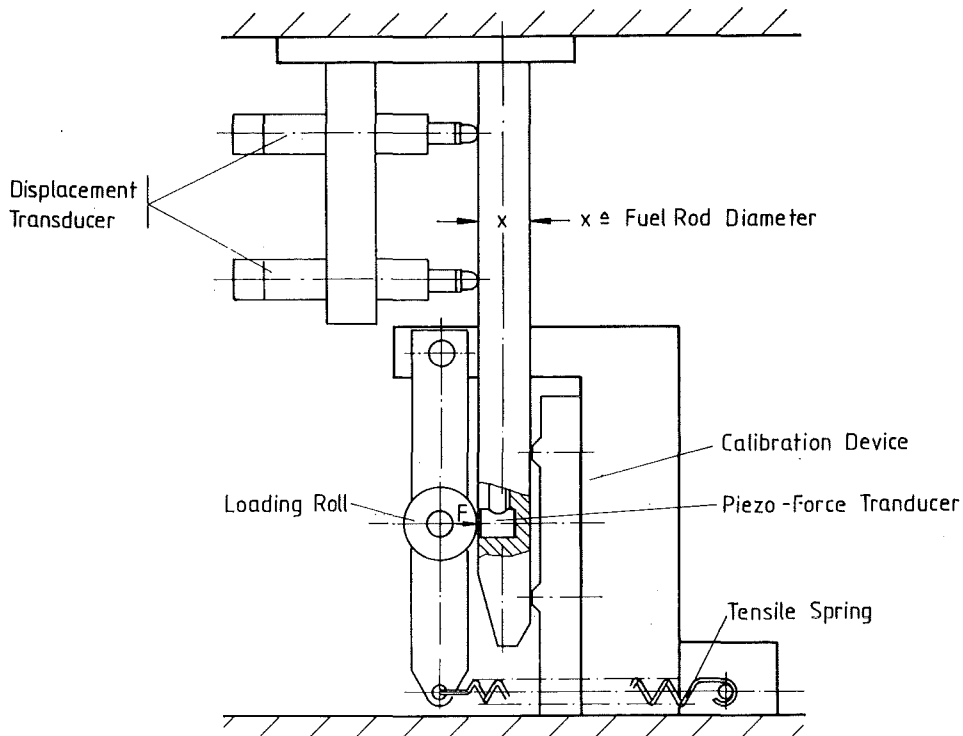
Calibration Device for Displacement und Force Transducers

Fig. 7

	A	B	C	D	E	F	G	H	K	L	M	N	P	R	S
1	417	483	410	448	409	451	445	383	439	418	450	418	482	418	489
2	365	412	337	407	484	360	385	378	448	462	495	444	474	437	444
3	393	378		410	367		460	486	474		412	469		423	409
4	421	403	423	378	388	378	343	520	482	488	439	444	417	426	409
5	384	362	469	375		418	475		477	469		448	423	411	454
6	369	452		434	392	392	437	450	472	477	434	482		468	447
7	368	352	379	363	323	386	435	510	476	415	479	459	467	427	472
8	455	422	480	449		488	413	468	471	389		466	440	435	359
9	434	467	481	483	450	471	467	457	431	455	406	420	362	405	429
10	425	510		447	451	484	463	449	425	418	446	391		422	391
11	397	452	454	497		495	470		495	424		408	445	466	435
12	436	420	495	493	458	459	453	421	429	462	429	419	428	460	412
13	399	479		464	506		465	440	433		412	373		426	398
14	381	462	480	503	463	462	494	450	407	419	354	438	448	429	412
15	494	461	484	392	440	407	497	402	432	453	470	438	444	451	418

Spacer Type   AH 55
Zircaloy-4

Max. Spring Force 500 N
Min. Spring Force 300 N

Too high	Too low
B 10	
D 14	
E 13	
H 4	
H 7	

Spring Force in $N \cdot 10^{-1}$

Spacer No. 8-48

Protocol of Test Results of Spring Forces of a Spacer

Fig. 8

B221.12.83

Kraftwerk Union
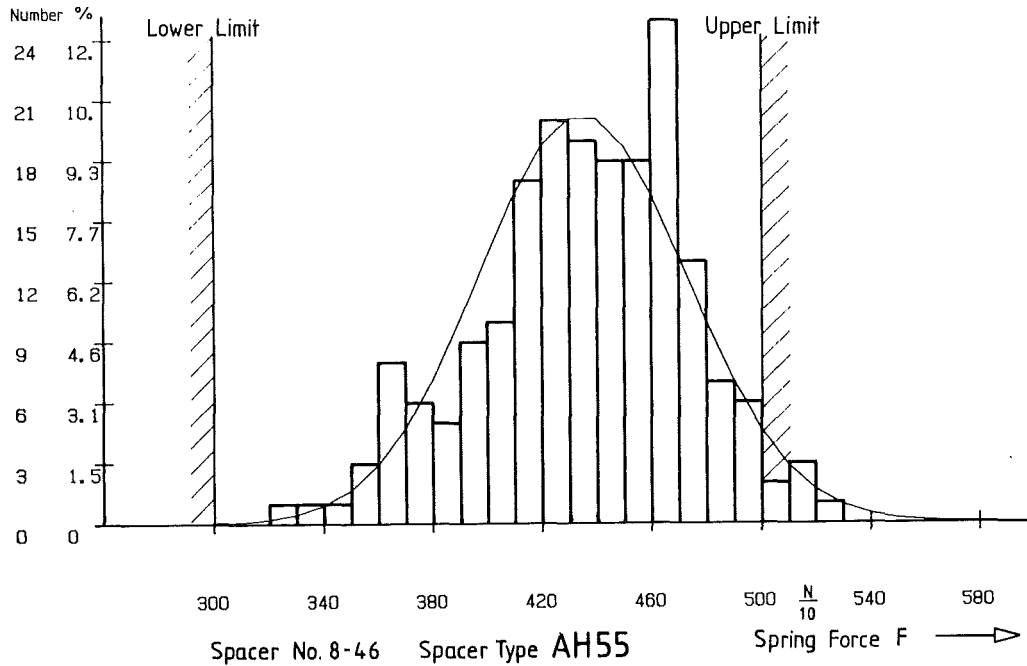
Accumulation Distribution of Spring Forces of a Spacer

Spacer No. 8-46   Spacer Type AH55

Spring Force F

Fig. 9

Measuring Finger

Piezo -Force
Transducer

Spacer Grid

Knobs

Numerical Controlled
Coordinate Table

x
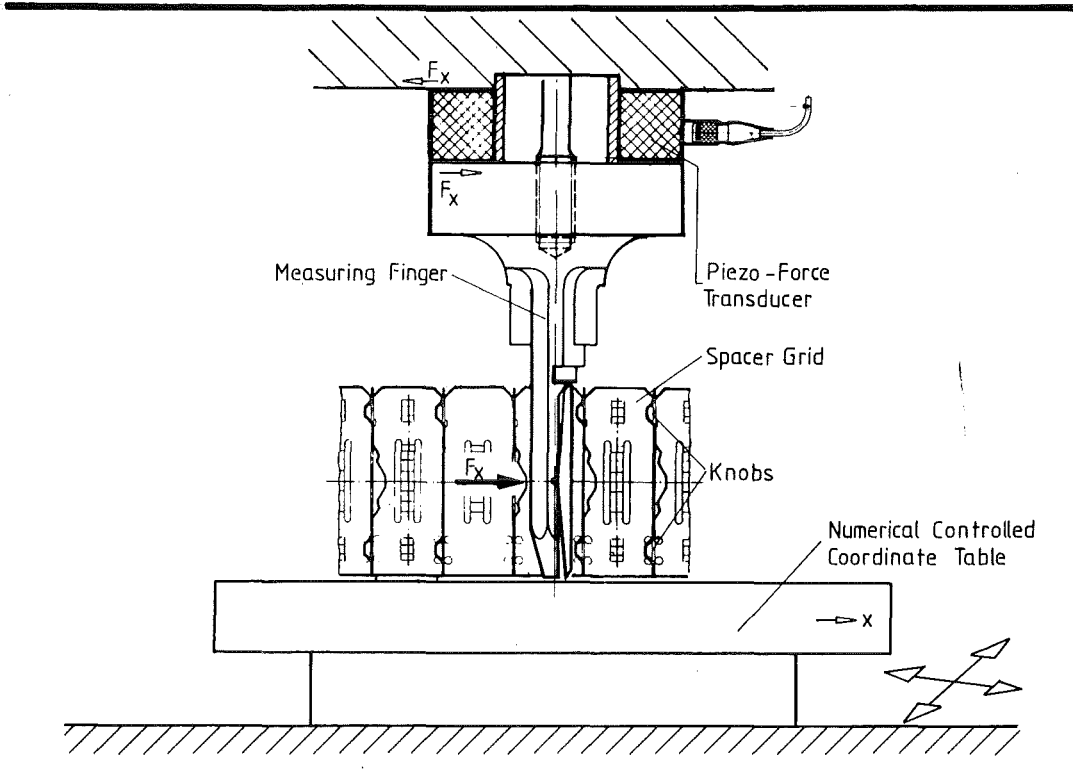
Measuring Device for Modified Force Transducer

Fig. 10

8221.12.83

Kraftwerk Union

Spring Force F (N)

Δf — Calibration Step

Spring Force before
Calibration

Upper Limit

Spring Force after Calibration

Lower Limit

Test Position ≙
Fuel Rod Diameter

Displacement of Spring f (mm)
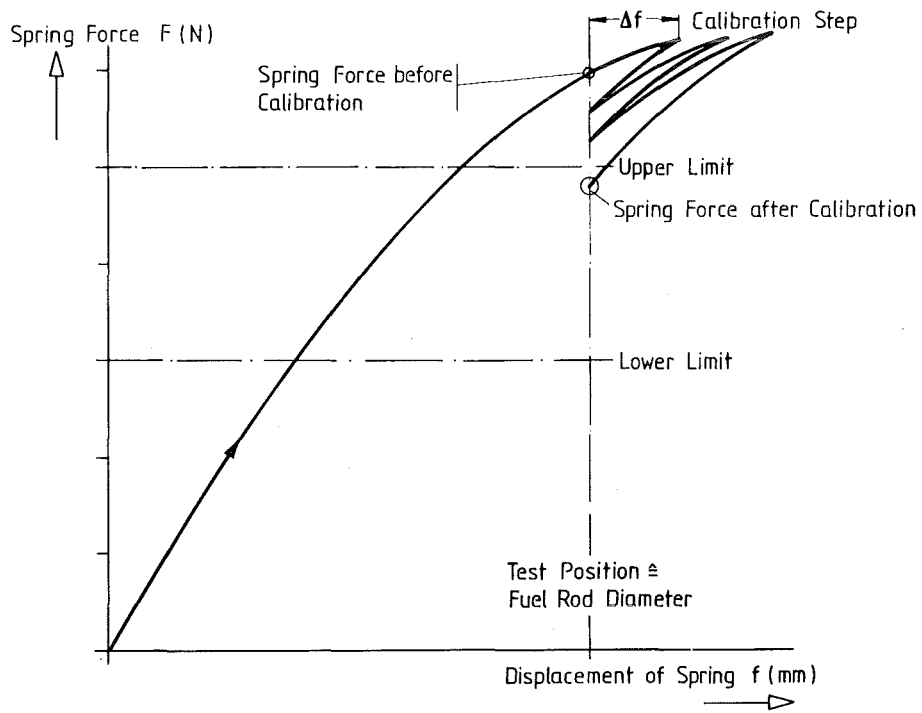
Adjusting of Spring Forces by Loading Into the Region of
Plastic Deformation.

Fig. 11

DISCUSSION

M. ERNOTTE: Is it possible with the system described for guide spring measurement to adjust the springs in both directions?

H.J. ROMEISER: No, it is only possible to calibrate springs in the direction to lower spring forces; but it is possible to manufacture springs near the upper tolerance limit to have the possibility of automatic calibrating.
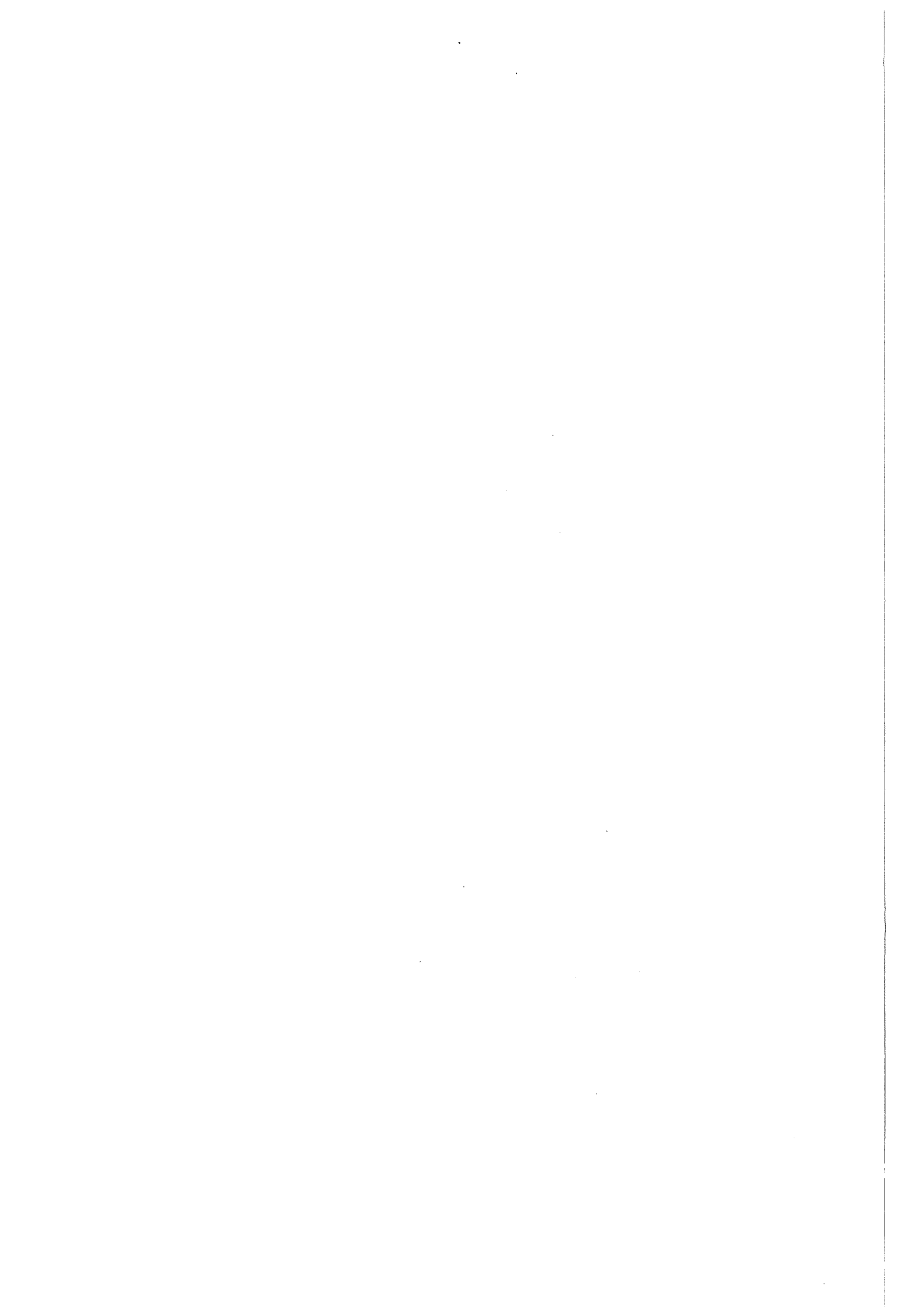
SESSION VII

METHODS OF CHEMICAL ANALYSES
TRAINING OF PERSONNEL

Chairman: M. Ernotte
Fragema, Lyon
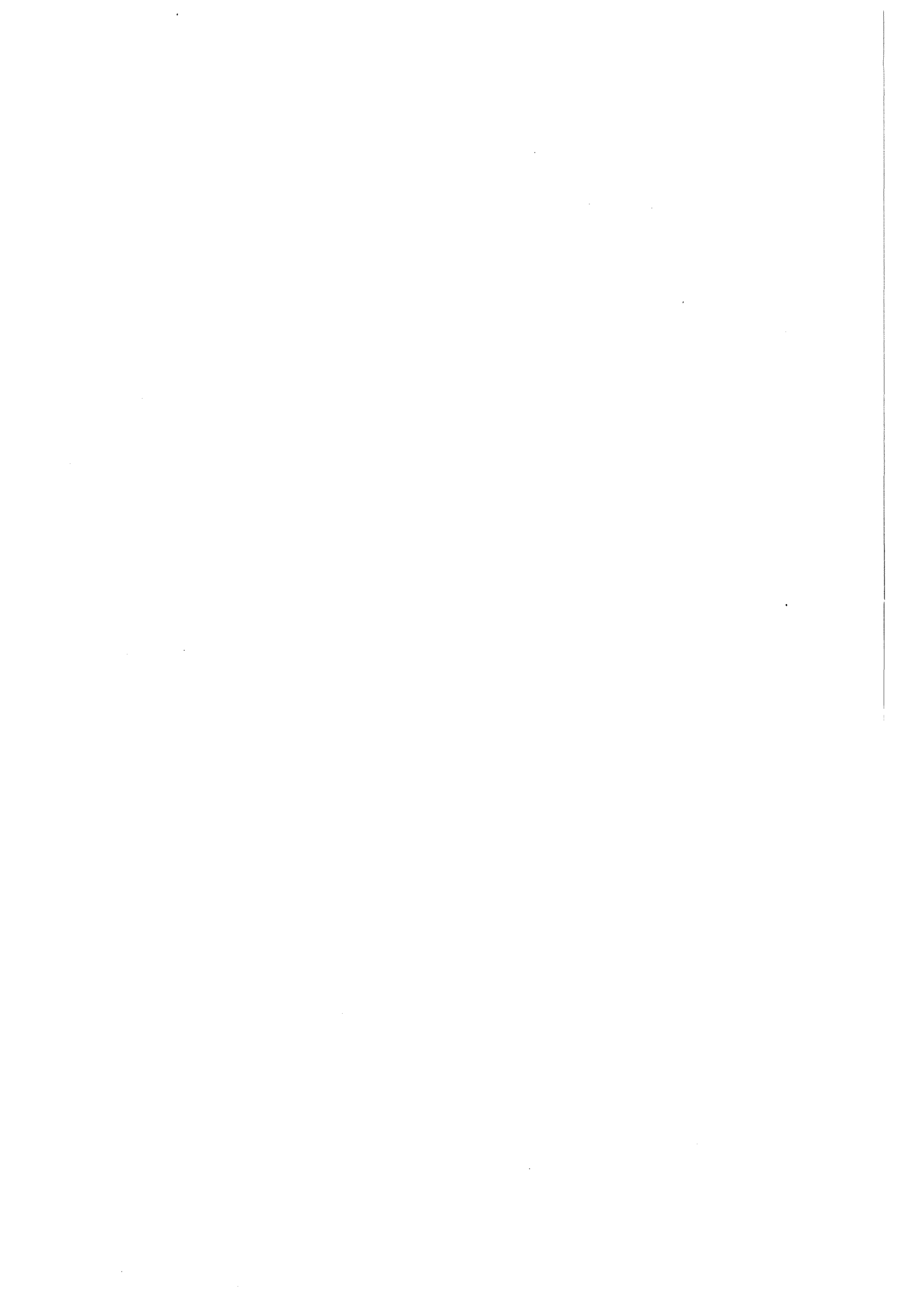
CHEMICAL ASSAY AS AN ELEMENT OF QUALITY CONTROL
OF LIGHT WATER FUELS

E. Mainka

Kernforschungszentrum Karlsruhe

Institut für Radiochemie

Postfach 3640, 7500 Karlsruhe

Federal Republic of Germany

Abstract:

A review will be presented of the various analytical techniques applied to the quality control of light water reactor fuels. The status of development of these methods has well advanced. Novel analytical methods now being introduced must be compared with existing performance data. The minimization of costs and wastes should play a role by selecting one of the available method.

A-1 <u>Introduction</u>

The European Organization for Quality Control (EOQC) defines quality control as "a system for programming and coordinating the efforts of various groups in an organization to check or maintain quality or to improve quality at an economical level"[1].

Today oxide nuclear fuel used in water cooled reactors is an industry manufactured product with a high degree of standardized reliability. In the USA and in Europe alone, more than three million fuel rods have already been successfully used in light water reactors [2]. The properties of oxide nuclear fuel itself have contributed substantially to this success. The most important characteristic features in this context are:

> high melting point,
> adequate thermal conductivity,
> low thermal dilatation,
> good behavior under irradiation conditions (dimensional
> stability, fission product retention),
> compatibility with water (in case of a fuel rod defect).

Another important factor contributing to this success consists in continous quality assurance during fuel manufacturing. Quality assurance is guaranteed by the specification and quality control.

A-2 <u>Specified Requirements</u>

The product properties of the fuel pellets can be broken down into four groups [2]

(1) fissile material content,

(2) chemical purity and stoichiometry,

(3) density, structure and resintering behavior,

(4) tolerance compliance and surface.

Chemical assay as an element of quality control deals exclusively with the first two requirements specified. Only these two points will be discussed in this report.

The process and product control of oxide powder and pellets to be used in LWR fuels should be discussed.
The fabrication of UO_2 pellets from UF_6 or uranyl nitrate hydrate (UNH) solutions as the source material takes place in a two-step process.

(1) Production of UO_2 powder by a chemical process.

(2) Fabrication of UO_2 pellets from this powder using metallurgical techniques.

Since the UO_2 pellet characteristic is determined also by the properties of the powder, not only the final product, i.e. the pellet, will be controlled but also the powder as the intermediate product.

Quality control plans for both UO_2 powder and pellets are given in [3].
The quality controls of the products to be performed on a routine basis depend not least on the state of the art in technology and on experience gathered by the respective industrial branch, i.e., with growing experience product control may, e.g., be reduced to relate only to control elements. In the final phase those controls will be carried out which have been specified by the reactor operator. Regarding the chemical composition the following assays are normally required:

The assays requested for UO_2 powder concern

(a) the uranium content and uranium-plutonium content, respectively,

(b) the isotopic ratio,

(c) the O/M ratio,

(d) the H_2O, F' and Cl'-contents,

(e) some metallic impurities.

The assays requested for the pellet concern

(a-c) the same items as for the powder,

(d) the C-content specified in addition to the H_2O, F' and Cl'-contents,

(e) metallic impurities as specified by the reactor operator,

(f) the residual gas content.

B   Chemical Assay as an Element of Quality Control of Nuclear Fuels

General Remarks

The status of development of wet chemical analytical techniques used in the quality control of nuclear fuels has well advanced. Relevant summarizing descriptions can be found in the literature [3, 4, 5]. Novel analytical methods now being introduced must be compared with existing performance data and the advantages and disadvantages must be discussed.

B.1 Analytical Methods for the Determination of Uranium and Plutonium Concentrations

Table 1 is a compilation of the respective analytical techniques which can be used in the quality control of individual products. The variation coefficients indicated (s and e) for the individual techniques have been elaborated by the

Table 1: Compilation of Analytical Methods for the Determination of Uranium and
Plutonium Concentrations

Analytical method Product of assay	Element	Redox/ titration		Coulo- metry		Gravi- metry		X-ray fluorescence spectroscopy		K-edges absorptiometry	
		s	e	s	e	s	e	s	e	s	e
U-solutions (very pure)	U	0.15	0.15	0.15	0.15	0.1	0.15	0.5	0.5	0.15	0.2
Uranium oxide powder, nuclear grade	U	0.15	0.15	0.15	0.15	0.1	0.1	–	–		
Uranium oxide pellet, nuclear grade	U	0.15	0.1	0.15	0.15	0.05	0.1	–	–		
U-Pu nitrate	U	0.2	0.2	0.2	0.2	–	–	0.5	0.5	0.15	0.2
Solution, very pure	Pu	0.2	0.2	0.2	0.2	–	–	0.5	0.5	–	–
$(U,Pu)O_2$-Mox LWR	U	0.3	0.2	0.3	0.2	–	–				
	Pu	1.0	0.5	0.2	0.2	–	–				

s = standard deviation of a single assay

b = standard deviation of the systematic error (e.g. calibration error)

responsible ESARDA (European Safeguards Research and Development Association) Group to the extent the said techniques have been available there.

Before the attempt will be made to compare the methods with each other, the performance data of the individual working techniques will be briefly described.

B-1a Redox/Titration of Uranium [7, 8, 9] and Plutonium [10, 11]

Measuring principle for uranium: Uranium is first reduced to U IV. The U IV→U VI oxidation step is measured potentiometrically.

Measuring principle for plutonium: Plutonium is reduced to Pu III using CuCl.
The Pu III→Pu IV oxidation step is measured potentiometrically.

Performance Data

The technique has much advanced and been tested in routine operation.
If ∿ 100 mg samples are used a high accuracy (0.15% for uranium and 0.2% for plutonium) can be achieved.
The technical complexity of the analytical technique calls for trained staff to do the analysis.
The cost of the equipment at present amounts to TDM 30-50.

B-1b Coulometric Method [13, 14]

Principle of measurement: Coulometry with a controlled potential relies on the Faraday law. The substance to be analyzed is oxidized and reduced, respectively, on the work electrode.
The current consumed in this process is measured by means of the coulometer.

Performance Data

The working technique according to [13] has been generally
introduced and tested in routine operation. It is a draw-
back that U IV reduction takes place on a mercury bath
electrode and is irreversible. Therefore, mention should
be made of the further improvement of this technique [14].

The amount of sample needed is very small ($\sim$ 20 mg).

The high accuracy of measurement of 0.15% adds to the
importance of this technique, the more so, since it is an
absolute method.

The technique can be automated at a high degree.

Trained staff should be available for processing.

The cost of the equipment amounts to approx. TDM 100.

B-1c Gravimetric Method [15-19]

Principle of measurement: The sample material is converted
by heating into U_3O_8 [15-17] and PuO_2 [18, 19], respectively.
The uranium and plutonium concentrations are determined from
gravimetric data. Trace amounts not capable of evaporation
are analyzed by emission spectroscopy and included in the
computation as corrections.

Performance Data

Gravimetry is one of the simplest and most accurate analytical
technique. However, 5-10 g of sample material must be used
in order to obtain an accuracy of 0.1%. But the material can
be returned into the process because no foreign materials
have been added.

The cost of the equipment amounts to approx. TDM 0.5.

The assay can be performed conveniently.

B-1d <u>X-ray Fluorescence Spectroscopy [20-24]</u>

<u>Principle of measurement</u>: The assay relies on the measurement of the X-ray fluorescence of fissile material spectral lines. The L-series of X-ray absorption edges is preferred.

Excitation: X-ray tube (e.g., Ag-anode).

Analyzer: LiF crystal {110}.

Detector: Scintillation counter.

<u>Performance Data</u>

The method is a proven one. When working with an internal standard (e.g. Th) accuracies of 0.5% are attained. Sample conditioning is simple; 30 mg samples are used.

The cost of the equipment amounts to approx. TDM 300.

B-1e <u>X-ray Absorptiometry [25, 26]</u>

<u>Principle of measurement</u>: The assay relies on a differential measurement of photon transmission directly below and above the L and K X-ray absorption edges, respectively, of the element to be assayed.

Excitation: X-ray tube with W-anode.

Detector: 200 mm^2 Ge-detector.
Resolution: 550 eV at 122 keV.

<u>Performance Data</u>

The method is being tested. Accuracies of measurement of 0.2% have been attained when comparing the methods [26].

The quantity of sample should be $\sim$ 500 mg.

Since the samples do not undergo changes during the measurement, the material to be analyzed can be returned into the process.

No sample conditioning is required so that the technique can be conveniently applied.

The cost of the equipment amounts to approx. TDM 250.

B-1a-e <u>Comparison of Methods</u> (Table 2)

All analytical methods described for the determination of uranium and plutonium concentrations conform to the specified requirements. If one compares the quantities of analytical samples required, the complexities of the working techniques, the equipment expenditures, and the waste arisings, considerable differences appear so that the choice of the method will depend on the possibilities offered.

B-2 <u>Determination of the Isotopic Ratio [27, 4]</u>

The isotopic ratio is usually determined by gamma spectroscopy directly on the powder and pellet, respectively. If the material is present as a solution, mass spectroscopic assay would be another possible means. Both techniques are proven and do not call for further discussion.

B-3 <u>The O/M Ratio [3, 28, 29]</u>

The following reasons are essential for specifying the O/M ratio:

(a) The thermal conductivity of the fuel is maximum at stoichiometric composition.

(b) Oxygen in excess leads to cladding tube oxidation.

(c) The O/M ratio exerts an influence on the chemical and physical behavior of the iodine and cesium fission products [2].

An overview of the methods is given in [3, p. 168]. The chemical methods eligible are Redox-titration, coulometric method and also gravimetric method, which have been already described in B1a-c. Determination by polarography should be mentioned as well.

Regarding UO_2 pellets and powders, we participated in an interlab test [30]. The results have been summarized in the following table (Table 3).

Table 2: Comparison of Methods

Analytical technique	Quantity of sample required [mg]	Cost of equipment [TDM]	Complexity of technique	Waste arising
A Redox/titration	100	30 - 50	Relatively complex	100 mg/analysis
B Coulometry, old version	20	100	Simple	Mercury as waste
C Coulometry, new version	20	100	Relatively complex	20 mg/analysis
D Gravimetry	5 000 10 000	0.5	Simple	Material assayed may be recycled into process
E XRFA	30	300	Simple	30 mg/analysis
F X-ray absorptiometry	500	250	Simple	No waste similar to D

Table 3: Interlab Test for the Determination of the O/M Ratio

Laboratory	A	B	B	C	D	E
Method	Titration	Titration	Photometry	Polarography	Titration	Polarography
	2.061	2.055	2.064	2.058	2.063	2.0564
	2.060	2.058	2.058	2.057	2.060	2.0564
	2.060	2.055	2.063	2.060	2.061	2.0563
	2.060	2.057	2.063	2.060	2.060	2.0564
	2.063	2.060	2.059	2.062	2.061	2.0568
	2.064	2.061	2.064	2.057	2.062	2.0562
	2.058	2.062	2.061			2.0568
	2.058	2.060	2.064			2.0569
		2.059	2.069			2.0569
Mean value (M) of single measurement	2.061	2.059	2.063	2.059	2.061	2.0566
RSD in % of single measurement	±0.10	±0.12	±0.16	±0.10	±0.06	±0.01

$\bar{M}$, the mean value derived from the lab mean values, is 2.060 ± 0.002 (0.1% RSD).

It is visible from the table that the methods used yield comparable results. When selecting the technique one will rely on the experience accumulated at the respective laboratory.

B-4 Determination of the Total Hydrogen Content [31, 32]

On account of its potential harmful effect hydrogen which causes hydride defects in the Zircaloy clad tube must not attain 1 ppm in the fuel according to the specification.

For determination carrier gas extraction together with final determination by coulometry are frequently used. By placing a CuO cell between the extraction furnace and the vessel used for final assay hydrogen is oxidized into water so that in this way the total hydrogen content is recorded. Also this method has proved its worth in routine operation.

B-5 Determination of Fluoride and Chloride [33-36]

The halogens fluorine and chlorine may play a part in the generation of hydride induced defects. Therefore, specifications have been fixed for both elements and they must be controlled. For the assay traces of fluoride and chloride are separated by pyrohydrolysis at high temperature with oxygen saturated in water vapor. The detection is preferably performed by ion specific electrodes.

Instead of the ultimate detection by ion specific electrodes ion chromatography has become a proven technique in recent time. The final assay in this case is performed by means of a thermal conductivity measurement.

Compared with the technique using ion specific electrodes, this technique is more sensitive and needs less sample quantities (µl).

B-6 Carbon Assay

The sample material is heated in a furnace in the oxygen stream. The carbon produced is converted into CO_2 which, subsequently, is titrated by coulometry. Also this working technique has proven its worth in routine operation.

B-7 Trace Assay by Emission Spectroscopy [37-40]

The fuel specification in all cases means a limitation on trace impurities by metals. In the individual case the number of the specified elements and the maximum permissible overall concentration of the trace elements undergo variations as well. Specifications have been rather frequently established for up to 27 metallic trace elements and therefore they must be controlled. In this context, emission spectroscopy is preferably used as the analytical technique. This technique does not only allow to attain the required detection limits; it also offers the possibility of making a multi-element assay. This is, e.g., a factor not to be neglected as compared to atomic absorption spectroscopy.

The direct current arc is used mainly as the source of excitation in emission spectroscopy. The oldest working technique is the Carrier distillation technique [37]. But also pre-evaporation must be mentioned in this context. Both analytical techniques are directly applied to the sample powder. The more volatile traces are separated from the matrix by specific heating which is an absolute necessity for analysis. However, in this way it is not possible to analyze, e.g., the rare earth elements and thorium for which specifications have also been fixed. With respect to these elements, a separation step precedes the spectral analytical assay proper after the sample material has been dissolved. This means that two techniques must be used in order to make all determinations required of trace elements.

At this point it is worthwhile trying to find out whether
by means of an improved analytical technique all specified
trace elements can be assayed in one working step.

At the IRCH of KfK a technique of extraction has been tested
under which all requested trace elements can be separated
from the matrix. The extracting agent used is a TBP/kerosene
mixture, ratio 1:4. The elements are extracted from 8 M HNO_3.
500 mg of sample are dissolved in 10 ml solvent acid.

The trace elements contained in the aqueous phase are then
analyzed by emission spectroscopy following excitation by ICP
(Inductively Coupled Plasma).

Since this is a relatively new technique of excitation, the
performance of an ICP torch will be described by reference
to the following figure.

Fig. 1   Schematic representation of a typical ICP used in
          analytical spectroscopy
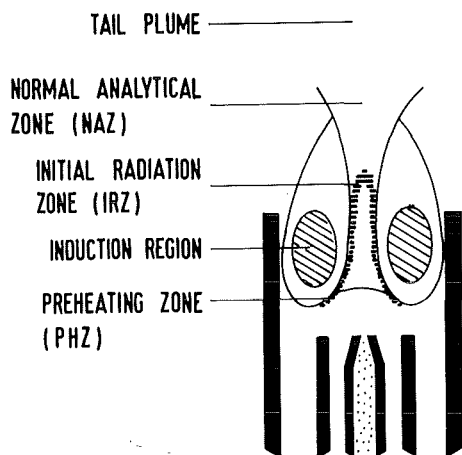
The figure shows how a typical plasma torch is fuctioning.
The induction coil surrounds the open end of a quartz tube
system. By the radio frequency current energy is supplied
to the plasma. It is apparent from the figure that the
argon flow carries and stabilizes the plasma. The bottom
side of the plasma is toroidal. The sample is introduced
as an aerosol into the aperture of the toroid with the help
of a carrier gas. Thanks to this shape the sample aerosol
can be introduced into the plasma very effectively. The
temperatures of the plasma are so high in certain zones that
the sample material becomes ionized. Therefore, the inter-
element effects playing a role in arc excitation can be
neglected.

To be able to perform a multi-element analysis also with this
excitation technique, we installed the ICP torch into an alpha
box. The configuration was so selected that besides the
sequential spectrometer also the already existing 3.5 m
grating spectrograph can be used for analysis.

Figure 2   shows the layout of the system.
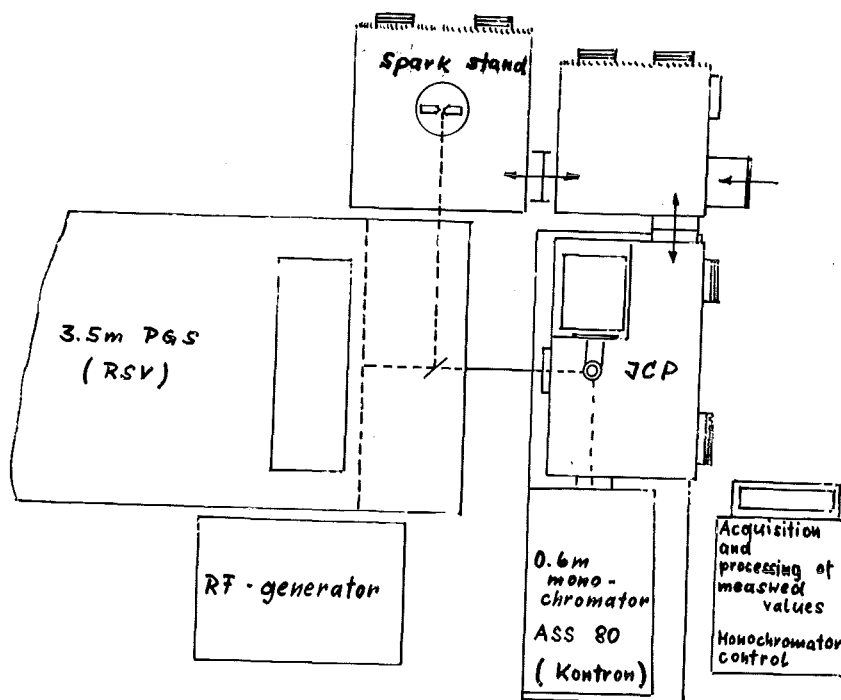
The detector for the grating spectrograph are photographic plates which are evaluated automatically. This means that at this point the technique is comparable to arc excitation.

A comparison of the methods with the previously used spectroscopic working techniques shows that the quantity of sample consumed can be reduced by at least the factor 2 since all elements can be assayed in one working step.

It is neither difficult to backextract the fuel from the TBP bearing organic phase so that the material can be returned into the process and no fuel-waste be produced at this point.

Summary

First of all it is important in quality control that the specifications can be satisfied with the requested accuracies.

It can be stated that the analytical techniques described conform to the requirements.

If a report entitled "Quality Control of Nuclear Fuels - Technical and Economic Aspects" [41] says that 20-40% of total powder and pellet fabrication costs are spent for quality assurance and control, it is worthwhile considering at which point cost savings might be possible without impairing quality.

The analytical techniques presented here - some of them novel ones - the majority of which are easier to apply, could certainly make a contribution to this effect. Also the minimization of wastes should play a role in this context.

Likewise one should think about the extent to which, e.g., the assay of 27 trace elements is meaningful if one can

evaluate the operating behavior of the fuel equally well with so-called control elements [42].

If, in the future, recycled material is to be used more frequently, it will certainly be necessary to give some thought to the relevant meaningful specifications.

Literature

1] Guidebook on Quality Control of Water Reactor Fuel
   IAEA Technical Reports Series No. 221 (1983) p 5

2] H. Assmann; H. Stehle
   Anforderungen an LWR-Brennstoff aus der Sicht des
   Betriebsverhaltens
   J. Nucl. Mat. 81, (1979) 19 - 30

3] H. Assmann and H. Bairiot: Process and Product Control
   of Oxide Powder and Pellets For Reactor Fuel Application
   IAEA Technical Reports Series No. 221 (1983) pp 149 - 180

4] Standard Methods for Chemical, Mass Spectrometric, and
   Spectrochemical Analysis of Nuclear-Grade Uranium Dioxide
   Powders and Pellets
   ASTM-C 696 - 80
   Annual Book of ASTM Standards Part 45 (1982) 180 - 225

5] E. Mainka; W. Coerdt; W. König and H. G. Müller
   Entwicklungsstand Chemisch-Analytischer Kernbrennstoff-
   untersuchungen
   J. Nucl. Mat. 81, (1979) 193 - 197

6] ESARDA private communication

7] W. Davies and W. Gray: A Rapid and Specific Titrimetric
   Method for the Precise Determination of Uranium Using
   Iron II Sulphate as Reductant. Talanta 11, (1964)
   pp 1203 - 1211

8] A. R. Eberle; M. W. Lerner; C. G. Goldbeck and C. J. Rodden
   Titrimetric Determination of Uranium in Product, Fuel and
   Scrap Materials after Ferrons ion Reduction in Phosphoric
   Acid: Part 1 Manual Titration
   NBL-Report 252 (1970) 1 - 25

9] J. Slanina; F. Bakker; A. J. P. Groen and W. A. Lingerak:
    Accurate and Precise Determination of 2 - 25 mg Amounts of
    Uranium by Means of a Special Automatic Potentiometric
    Titration; Z. Anal. Chem. 289, (1978) pp 102 - 105

10] W. Davies and M. Townsend
     Analytical Method for the Titrimetric Determination of
     Plutonium using Cuprous Chlorid as Reductant
     TRG - Report 2463 (1974) 1 - 45

11] C. Corpel and F. Regnaud: Dosage du Plutonium par Potentio-
     metrie a Intensite Constante; Anal. Chim. Acta 35, (1966)
     pp 508 - 513

12] W. D. Shults: Application of Controlled-Potential Coulometry
     to the Determination of Plutonium Talanta 10, (1963)
     833 - 849

13] Standard Test Method for Uranium by Controlled Potential
     Coulometry
     ASTM E 217 - 70 (Reapproved 1978)
     Annual Book of ASTM Standards Part 45 (1982) pp 889 - 897

14] G. Phillips and D. Crossly
     Recent developments in the application of controlled
     potential coulometry to the determination of plutonium and
     uranium.
     Nuclear Safeguards Technology (1978) IAEA
     STI / PuB 497 pp 665 - 675

15] Standard Methods for Chemical, Mass Spectrometric,
     Spectrochemical, Nuclear and Radiochemical Analysis of
     Nuclear-Grade Uranyl Nitrate Solutions
     C 799 - 78
     Annual Book of ASTM Standards 1982 Part 45 pp 563 - 592

16] J. R. Jewett: High-Accuracy Determination of Uranium in
     Concentrated Relatively Pure Solutions of Uranyl Nitrate
     RHO-ST-Report - 22 (1978) pp 1 - 5

17] O. A. Vita; C. R. Walker and M. E. Litteral: The Gravi-
    metric Determination of Uranium in Uranyl Nitrate.
    Anal. Chim. Acta 64, (1973) pp 249 - 257

18] L. Lindner and A. v. Baeckmann: Verfahren zur chemisch-
    analytischen Bestimmung von Plutonium und Uran in oxidi-
    schen Kernbrennstoffen
    KFK 701 (1967) 50 pages

19] J. R. McGowan; C. R. Johnson and K. A. Swinburn
    Oxygen/metal ratios in plutonium/uranium oxide fuels:
    A study of gravimetric methods
    Proceedings of a symposium on Analytical Methods in
    the Nuclear fuel cycle IAEA STI/PuB 291 (1972) 3 - 21

20] P. A. Pella and A. v. Baeckmann: The X-Ray Spectrometric
    Determination of Uranium and Plutonium in Solutions of
    Spent Nuclear Fuels.
    Anal. Chim. Acta 47, (1969) 431 - 438

21] D. Ertel: Simultaneous Determination of Uranium and
    Plutonium in inactive purex process solutions by X-Ray
    Fluorescence. J. Radioanal. Chem. 2, (1969) pp 205 - 209

22] A. v. Baeckmann: Determination of uranium and plutonium
    in solutions of spent nuclear fuels by X-Ray fluorescence
    analysis. IAEA STI/PuB - 260 II (1970) pp 45 - 51

23] A. v. Baeckmann; D. Ertel; J. Neuber: Determination of
    Actinide Elements in Nuclear Fuels by X-Ray Analysis
    Advan. X-Ray Anal. 18, (1975) pp 62 - 75

24] E. Mainka; J. Neuber; H. Wertenbach; R. Berg; B. Stoyanik
    and B. Fuger: Fully Automated Wavelength Dispersive X-Ray
    Fluorescence Analysis of Uranium and Plutonium in Repro-
    cessing Plant Solutions.
    Nuclear Safeguards Technology 1982 IAEA Vol. I STI/PuB 629
    pp 249 - 260

25] O. Cristallini; E. Gantner and A. v. Baeckmann:
Bestimmung von Uran in wässrigen Lösungen mittels
γ-Absorptiometrie. KFK 1703 (1972) pp 1 - 26

26] H. Ottmar; H. Eberle; P. Matussek and J. Michel-Piper:
Qualification of K-absorption edge densitometry for
applications in international safeguards.
Nuclear Safeguards Technology 1982 IAEA
STI/PuB 629 Vol II pp 147 - 163

27] H. Eberle; P. Matussek; J. Michel-Piper; H. Ottmar
and H. Allex: Elemental and isotopic concentration
analyses on nuclear fuels using nondestructive assay
techniques.
J. Nucl. Mat. 81, (1979) pp 203 - 214

28] E. Mainka; W. Coerdt; W. König and H. G. Müller
Entwicklungsstand Chemisch-Analytischer Kernbrennstoff-
untersuchungen. J. Nucl. Mat. 81, (1979) pp 193 - 197

29] G. Schumacher: Auswirkungen der Ausgangsstöchiometrie
auf das Betriebsverhalten von Oxidischen Brennstäben
Schneller Reaktoren. J. Nucl. Mat. 81, (1979) pp 53 - 62

30] T. Gorgeniy; E. Mainka; D. L. G. Smith
D. P. - Report 983 (1977) pp 4, 5

31] E. Brandau and L. Kohler: Wasser- und Wasserstoffgehalt
in Oxidischen LWR-Brennstoffen und in Borcarbid.
J. Nucl. Mat. 81, (1979) pp 257 - 262

32] W. Bartscher; H. Kutter: Beitrag zur Bestimmung des Ge-
samtwasserstoffs in Oxidischen Kernbrennstoffen.
J. Nucl. Mat. 81, (1979) pp 263 - 266

33] R. H. Powell and O. Merris: Separation of fluoride from
Inorganic compounds by Pyrolysis. Anal. Chem. 30, (1958)
pp 1546 - 1549

34] K. H. Neeb: Beiträge zur Bestimmung von Spurenverunrei-
nigungen in UO_2-Kernbrennstoff. Z. anal. Chem. 215,
(1966) pp 161 - 175

35] E. Mainka; W. Coerdt: Bestimmung von Fluorid-, Chlorid-
und Borspuren in plutoniumhaltigen oxidischen Kernbrenn-
stoffen. KFK 1320 (1971) pp 1 - 20

36] E. Mainka; W. Coerdt: Anionenbestimmungen in Kernbrenn-
stoffmaterial. In preparation

37] B. F. Scribner and H. R. Mullin: Carrier Distillation
Method for Spectrographic Analysis and its Application
to the Analysis of Uranium-Base Materials.
J. Res. NBS 37, (1946) pp 379 - 389

38] E. Mainka; H. G. Müller: Emissionsspektroskopische Spuren-
bestimmung in Kernbrennstoffen. KFK - 1261 (1970) pp 1 - 34

39] W. Coerdt; R. Crubellati; E. Mainka and H. G. Müller
Die Gadoliniumbestimmung in Kernbrennstoffen - ein Metho-
denvergleich - J. Nucl. Mat. 106, (1982) pp 109 - 114

40] E. Gantner; E. Mainka and H. J. Ache: Analytical Research
at the Institute of Radiochemistry, Nuclear Research Center
Karlsruhe, FRG.
Proceedings of Twenty-Sixth ORNL/DOE Conference on Analyti-
cal Chemistry in Energy Technology, Knoxville October 11 - 13
(1983) in printing

41] A. A. Strasser: Quality Control of Nuclear Fuels - Technical
and Economic Aspects. J. Nucl. Mat. 106, (1982) pp 3 - 12

42] O. Götzmann: Die Bedeutung der Brennstoff-Verunreinigungen
für das Brennstabverhalten. J. Nucl. Mat. 106, (1982)
pp 301 - 308

DISCUSSION

G. PAUWELS: What are the detection limits for the measurment of:
1.1 Cl-content with specific electrode after pyrohydrolysis,
1.2 rare earth content with ICP after TBP separation from U
(limit per rare earth)?

E. MAINKA: 1.1 Cl : 15 ppm/UO_2.

1.2 Per rare earth : 0.5 ppm/UO_2.
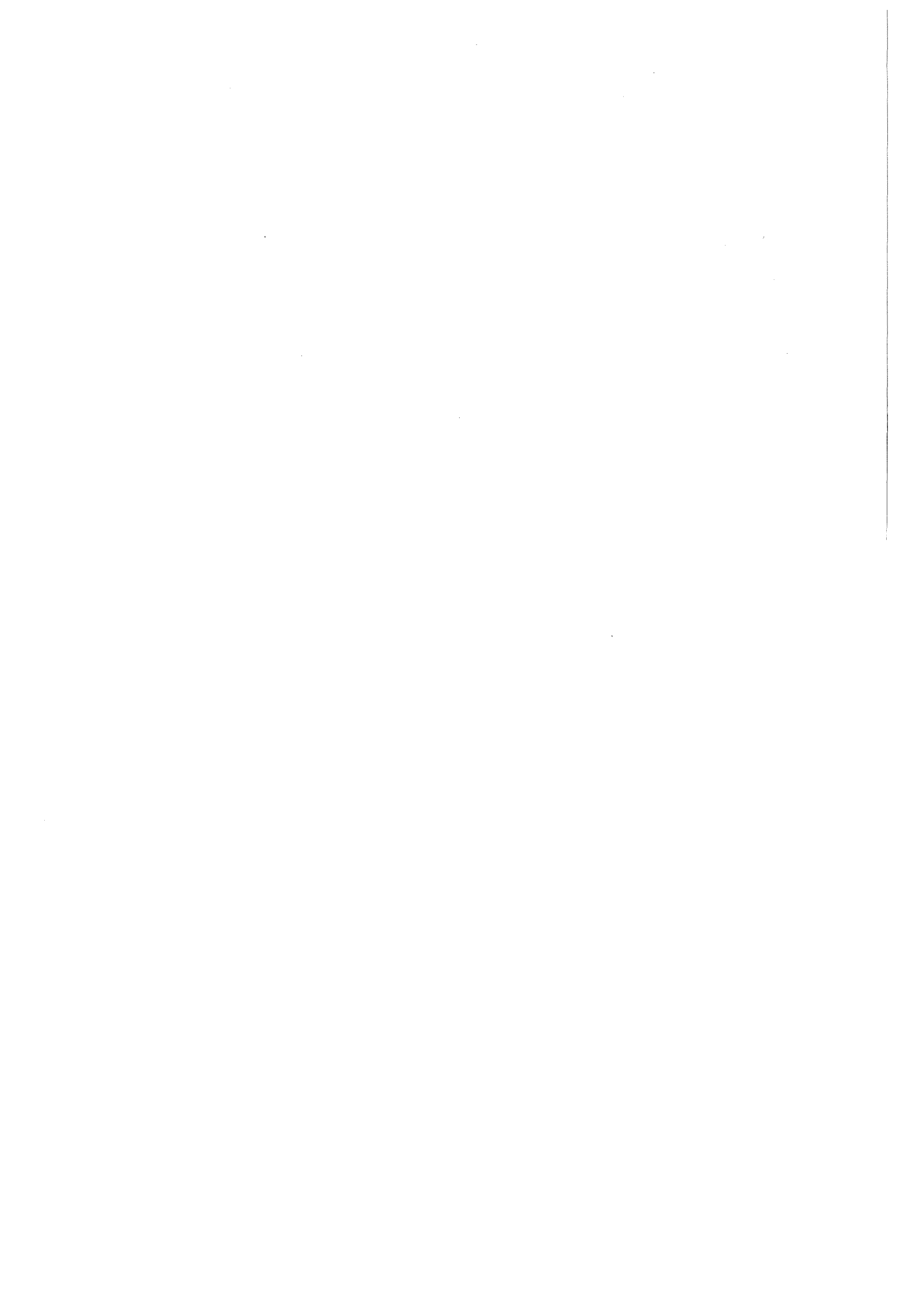
METHODS AND PROCEDURES OF CHEMICAL ANALYSES OF ZICALOY TUBING

Kaydell Bowles

Sandvik Special Metals corporation
P. O. Box 6027
Kennewick, Washington 99336 U.S.A.

Abstract:

To determine conformance to the chemical composition requirements, the zircaloy ingot is sampled either in accordance with procedures outlined in ASTM B350 or as approved by the tubing manufacturer.

The samples are analyzed for the alloying elements and impurities by methods and standards adopted by the manufacturer, but not necessarily those used worldwide. When possible, standards traceable to NBS are used for verification, but in most part the standars are made in-house by mixing highpurity zirconium oxides or solutions with oxides or solutions of the elements of interest.

Historical

The first zirconium available in large quantities was crystal bar. When this material was subjected to extended corrosion tests in water and steam, the material exhibited variable results. The reasons for the variations observed were due to impurity levels of nitrogen in the metal.(1)

The variability in the crystal bar gave momentum toward development of zirconium alloys and to make use of the less expensive Kroll process for sponge zirconium.(2)

The direction for alloy development of zirconium was furnished by the Naval Reactor Branch of the A.E.C. The early work was performed in the laboratories of the U. S. Bureau of Mines (Albany, Oregon), Argonne National Laboratory, Metallurgical Laboratory at MIT, Battelle Memorial Institute, Iowa State College, Oak Ridge National Laboratories and Bettis Atomic Power Laboratory.(3) Out of these early studies the alloys, Zircaloy-1, Zircaloy-2, Zircaloy-3 and Zircaloy-4 were born. The compositions were not optimized based on in-reactor corrosion rates, but on their performance and behavior observed under laboratory conditions.

The first zircaloy, called Zircaloy-1, was a 2-1/2% tin alloy. This alloy showed accelerated corrosion in high temperature steam (650°F) and was deemed unsuitable for reactor use. It did have good strength, but the poor corrosion resistance deterred its use.

Zircaloy-2 was developed and patented by several people at Bettis Atomic Power Laboratory; namely D. E. Thomas, R. B. Gordon and K. M. Goldman. The active corrosion program at Bettis Atomic Power Laboratory, which provided corrosion behavior data, allowed this alloy to be selected for reactor use in the Nautilis.(4)

Zircaloy-3, which was comprised of zirconium plus 1/2% tin and 1/2% iron had a short lifetime. It had good corrosion resistance, when properly heat treated. Unfortunately, this alloy was more sensitive to processing variables. Zircaloy-2 and Zircaloy-3 exhibit similar water corrosion properties; however, Zircaloy-2 was not as sensitive to processing variables. Therefore, the application of Zircaloy-3 was discontinued.

Zircaloy-4 was introduced when it was determined that the nickel in Zircaloy-2 enhanced the pick-up of hydrogen from the corrosion reaction. The Zr-4 alloy had the nickel removed and a similar amount of iron added on a mole fraction basis.

The impurity levels in the alloys of zirconium were initially set by a somewhat unscientific method of analyzing the element, constructing cumulative distribution curves, and selecting the specific limit at 90% of the cumulative distribution curve. These limits were satisfactory initially, but were refined over the years as more information became available on their effect on the corrosion resistance of zircaloy and neutron absorption.

The elements such as boron, cadmium and hafnium have always been maintained at low levels.

For a period of time, there was relatively little control of oxygen content, but then it was found that closer control on oxygen was beneficial for it imparted uniformity of mechanical properties and allowed a fairly consistent temperature range for the Alpha + Beta phase region. The latter permits consistency of heat treatments to achieve corrosion and mechanical properties.

Manufacture

Briefly, the zirconium metal is manufactured as follows:

The ore, zircon sand, is mixed with coke and undergoes chlorination. The tetrachloride is converted into a zirconium-bearing solution from which the hafnium is removed. The solution is converted to an oxide, which is rechlorinated and then reduced with magnesium or sodium in an inert atmosphere to what is commonly called sponge metal.

During the manufacture of zirconium, chemical analyses are performed on the feed stocks, liquors, oxides and ultimately the sponge. These analyses must meet internal requirements so that the resulting sponge may meet the criteria for release to melting.

From each sponge blend, a sample is split, and this sample is double arc melted under vacuum in a small laboratory furnace. The resulting ingot is then analyzed for impurities and hardness. Based on internal acceptance standards, the sponge blends are released for compacting into charges for subsequent melting into product ingots.

The compacted sponge, solid scrap and alloying elements, tin, chromium, iron and nickel are made into an electrode. This electrode is then at least double arc melted under vacuum. The final resulting ingot is about 24 inches in diameter and its length may vary from 6 to 12 feet. The ingot, after full chemical analysis, is released and fabricated into either bars, plates or billets.

Zirconium Alloys

The major alloys today are Zircaloy-2 used in cladding elements for the BWR's and Zircaloy-4 used in cladding elements of PWR's and PHWR's and channels for BWR's. The chemical composition for these two alloys is outlined in Table 1.

Sampling Methods for Chemical Analysis

The sampling method used to certify their product varies with the manufacturer. Two manufacturers remove samples from the sidewall of the ingot. The number of samples is in accordance with ASTM B350; i.e., top sample is within 5 inches (127 mm) of the top face and then one sample along the length at a distance not to exceed one ingot diameter. Another manufacturer removes the sample by taking a cross-section of the forged bar. The number of samples taken and the location corresponds to the number and place of the respective ingot.

TABLE 1

ASTM B350 Chemical Requirements for Zr-2 and Zr-4 Ingots

Alloying Elements Weight Percent	Zr-2 (R-60802)	Zr-4 (R-60804)
Tin	1.20-1.70	1.20-1.70
Iron	0.07-0.20	0.18-0.24
Chromium	0.05-0.15	0.07-0.13
Nickel	0.03-0.08	--
Iron & Chromium & Nickel	0.18-0.38	0.28-0.37
Oxygen	(A)	(A)

Impurities, ppm		
Aluminum	75	75
Boron	0.5	0.5
Cadmium	0.5	0.5
Carbon	270	270
Cobalt	20	20
Copper	50	50
Hafnium	100	100
Hydrogen	25	25
Magnesium	20	20
Manganese	50	50
Molybdenum	50	50
Nickel	--	70
Nitrogen	65	65
Silicon	120	120
Titanium	50	50
Tungsten	100	100
Uranium	3.5	3.5

TABLE 2

Method of Analyses for Zircaloys by the Manufacturers

Element	ASTM-E146	Manufacturer A	Manufacturer B	Manufacturer C
Sn	Iodate Titration (V)	(ICP)	(ICP)	AAS
Fe	Ortho-Phenanthroline (P)	" "	"	OES
Cr	Diphenylcarbazide (P)	" "	"	AAS
Ni	Dimethylgloxine (P)	" "	"	"
N$_2$	Nessler (P)	Nessler (S)	IGF-GC	IGF-GC
H$_2$	HE	HE (Air carrier)	IGF-GC	IGF-GC
C	CC	CC	CI	CI
O$_2$	Inert gas fusion	RM&I	IGF-GC	IGF-GC
Al	8-Hydroxyquinoline (F)	(ICP)	(ICP)	OES
B		MAS	"	"
Cd		(ICP)	"	"
Co		"	"	"
Cu	Neo-Cuproine (P)	"	"	AAS or OES
Hf		"	"	OES
Mg		"	"	--
Mn	Perioclate (P)	"	"	OES
Mo		"	"	--
Si	Molybdenum Blue (P)	"	"	OES or MB
Ti	5-Sulfosalicylic Acid (P)	"	"	OES
W	Thiocyanate (P)	"	"	"
U	Fluorometric Method	FA	"	FA
Cl		Potentiometric Titration	Ionalyzer	Silver Nitrate with X-ray Fluorescence

(V) = Volumetric
(P) = Photometric
(F) = Fluorometric
(ICP) = Inductive Coupled Plasma
CI = Combustion - Infrared
(S) = Spectrometry
HE = Hot Extraction
FA = Fluorometric Analysis

CC = Combustion Conductometric
RM&I = Reduction Melting & Infrared Absorption
MAS = Molecular Absorption Spectrophotometry
IGF-GC = Inert Gas Fusion with Gas Chromatographic separation of gases using thermal and/or infrared readout
AAS = Atomic Absorption Spectrophotometry
OES = Optical Emission Spectroscopy
MB = Molybdenum Blue (Spectrophotometric)

The Methods of Chemical Analysis

The method each vendor uses for analysis of the alloying element and impurity is as given in Table 2. Also given in this table are the methods of analyses outlined in ASTM E146 "Standard Methods for Chemical Analysis of Zirconium and Zirconium Alloys".

Please note the methods outlined in ASTM E146 differ from that of the manufacturers. This does not mean that the methods by the manufacturers are obsolete or less precise but that the methods in ASTM E146 have not kept pace with the state-of-the-art.

METHODS FOR ANALYSIS OF IMPURITIES AND ALLOYING ELEMENTS

Plasma Emission Spectroscopy (ICP & DC)

One primary method for analyzing the impurities and the alloying elements is plasma emission spectroscopy. Two manufacturers are currently using the inductive coupled plasma emission known as (ICP) while another vendor will soon be using the direct current plasma emission known as DCP.

The sample for analysis is placed in solution and small droplets of the solution are introduced into the concentrated flow of argon. Both types, DCP and ICP, cause the flowing argon and droplets of solution to become electrically conducting and heated to a high temperature, with a maximum between $7,000\text{-}15,000^{\circ}C$. The emission spectrum is then measured for that element. The unknown concentration is determined by comparing measurements on standards of known composition.

The plasma emission method has the capability to analyze all refractory elements such as boron, phosphorus, tungsten, niobium and uranium and to perform simultaneous multielement analyses up to as many as 60 elements.(5)

Atomic Absorption

In the atomic absorption methods of chemical analysis, a portion of the sample which is in solution is converted into an atomic vapor, and the absorbance of

light by this vapor is measured at a specific wavelength which is characteristic of the element to be determined. The unknown concentration is determined by comparison with absorbance measurements on standards of known composition.

Spectrographic Analysis

The spectrographic analysis uses metal samples where the metals are oxidized in a muffle furnace and the resultant oxide is ground to a fine powder. The oxide is then mixed with carriers such as graphite and excited by a high voltage in order to volatilize the sample carrier mixture. The spectrum of the volatilized element or elements is recorded photographically and the selected spectrum lines are measured photometrically. The line intensities are visually compared to standards or are converted to relative intensities and the concentrations are determined using appropriate standards of known composition.

Methods for Oxygen, Nitrogen and Hydrogen Determination

The three manufacturers use similar techniques for determination of oxygen, nitrogen and hydrogen. The method for obtaining the gases so they can be analyzed is placing the sample into a flux basket (nickel or platinum) held by a single-use graphite crucible. The sample and basket are fused under inert atmosphere (helium or argon) at high temperatures. The high temperature fusion releases the gases from the sample and the gases are swept through various scrubber columns and trapped. These gases are then quantified by either an infrared detector or a thermal conductive meter or a thermistor bridge detector.

Carbon Determination

The manufacturers use a spectrophotometer for determination of carbon. The sample is placed into a crucible containing a fluxing alloy. Upon combustion, carbon is liberated as CO_2, swept into an analyzing chamber, and then quantified using an infrared detector.

The other manufacturer uses a combustion or conductometric method for determining carbon in the sample. The sample is burned in a stream of oxygen, and the carbon dioxide in the evolved gases is absorbed by a barium hydroxide or sodium hydroxide solution. The amount of carbon present in the sample is determined by measuring the change in conductance of the absorbing solution and comparing the change with a calibration curve based on the carbon content of known standard samples.

Chlorine Determination

Chlorine analysis is by titration, fluorescence and by analyzing of ions.

The determination of chloride in zircaloys by x-ray fluorescence involves dissolving the sample and precipitating the chloride by additions of silver nitrate. The silver chloride is filtered and dried. The filter pad is then placed on a sample holder and the intensity of the Ag K-alpha radiation is measured and recorded. The Ag intensity is converted to ppm chlorine.

The determination of chloride by titration involves dissolution of the sample by hydrofluoric acid and the titration of chlorine with silver nitrate solution. The method utilizes an apparatus for the automatic plotting of the titration curve with detection of the equivalent point by potentiometric measurement at constant current.

Uranium Determination

The determination of uranium is either by fluorometry or plasma emission.

The determination of uranium by fluorometry involves the dissolution of the solid sample by nitric-hydrofluoric acid. The uranium is extracted from the solution by an organic complexing agent. The uranium solution is then evaporated, dried, and fused with a mixture of fluoride and carbonates into a small bead. The characteristic uranium fluorescence emitted by the bead under ultraviolet light is measured with a fluorometer.

Precision

Table 3 gives the interval of confidence at 95% for readings situated at the normal specified limits for impurities and within the specified composition ranges for alloying elements.

TABLE 3

Manufacturer's Reported Precision of Analyses of Zircaloys

ASTM B350		Manufacturer A[1]	Manufacturer B[2]	Manufacturer C (%)[3]
Sn	Wt %	1.40 ±0.030	1.78 ±0.04	4
Fe	ppm	0.200 ±0.0030	0.221 ±0.006	5
Cr	"	0.100 ±0.0030	0.126 ±0.004	5
Ni	"	0.050 ±0.0020		5
N_2	ppm	80 ±8		10
H_2	"	25 ±2		5
C	"	124 ±28		5
O_2	"	1182 ±30		5
Al	ppm	75 ±8	60 ±3.4	5
B	"	.5 ±.2		8
Cd	"	.5 ±.3		10
Co	"	20 ±3		7
Cu	"	50 ±5	59 ±3	7
Hf	"	100 ±10	53 ±8	5
Mg	"	20 ±2		--
Mn	"	50 ±6		10
Mo	"	50 ±6	20 ±1.6	--
Si	"	100 ±6	79 ±5.2	5
Ti	"	50 ±10		5
W		50 ±6		10
U		3.5 ±0.8		10

[1] This gives the interval of confidence for the ICP method at 95% for readings situated at the normal specified limits for impurities and in the middle of the composition range for alloying elements.

[2] This gives the interval of confidence at 95% for the ICP method when samples are taken from zircaloy and analyzed separately.

[3] This is the estimated percent relative precision.

Standards

The standards are not necessarily traceable to certified standards; i.e., NBS JAERI, nor are the elements to be determined certified in the NBS standards; e.g. copper and titanium.

The manufacturers use their own in-hand synthetic standards prepared by mixing high purity zirconium oxide or zirconium solutions with oxides or solutions of the elements of interest. These standards are the ones commonly (with exceptions of some elements and methods) used for calibration and standardization of their equipment.

At given frequencies, a metal standard or synthetic standard is analyzed by the equipment. The composition limits for the element or elements are checked against standard curves and the values for the standard must be within limits. This overcheck assures proper calibration and operation of the equipment for the samples being analyzed.

Acknowledgements

Much of the data presented in this paper is the result of the work of the following personnel:

Dr. J. Wille and Mr. J. H. Schlewitzt of Teledyne Wah Chang Albany; Mr. J. G. Goodwin and Mr. C. E. Taylor of Western Zirconium; and Mr. D. Mills of CEZUS. I thank them for their efforts and contributions.

References

(1) Stanley Kass, THE DEVELOPMENT OF THE ZIRCALOYS. ASTM Special Technical Publication No. 368 (1964) Page 6.

(2) Stanley Kass, THE DEVELOPMENT OF THE ZIRCALOYS. ASTM Special Technical Publication No. 368 (1964) Page 4.

(3) Same as (1) and (2).

(4) Admiral H. G. Rickover, L. D. Geiger and B. Sustman, HISTORY OF THE DEVELOPMENT OF ZIRCONIUM ALLOYS FOR USE IN NUCLEAR REACTORS. U. D. Report TID-26740 (1975).

(5) T. J. Hanson and R. O. Ediger, COMBINING AA AND ICP. American Laboratory, (March 1980) Pages 116-123.
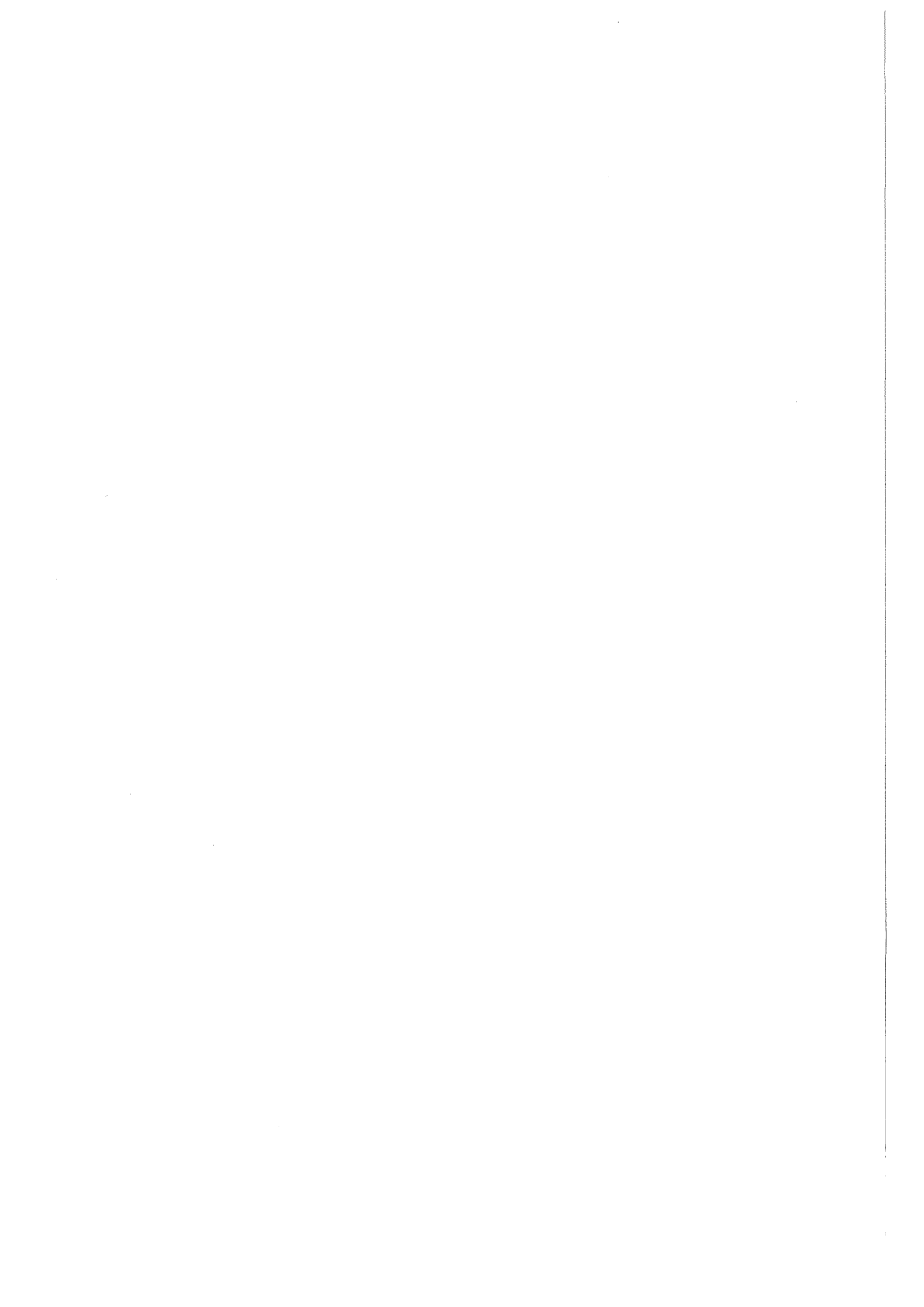
MEASUREMENT OF THE INNER PRESSURE
OF A FUEL ROD AND DETERMINATION
OF THE FILLING GAS PURITY

H.-J. von Wachtendonk, W. Ebbinghaus
Reaktor-Brennelement Union GmbH,
Rodenbacher Chaussee 6, 6450 Hanau 11
West Germany

Abstract:

Fuel rods have to be controlled for their inner pressure and
for the purity of the filled helium gas. This happens on the
one hand during production, but on the other hand it las to
be ensured that the specificd parameters are met at the final
fuel rod. This can only be actieved by destructive means - re -
opening of the welded fuel rod and measuring the inner pressure
and analysing the helium gas contained. In this paper a simple
method is presented which allows both measurements within a
short time.

MEASUREMENT OF THE INNER PRESSURE
OF A FUEL ROD AND DETERMINATION
OF THE FILLING GAS PURITY

H.-J. von Wachtendonk, W. Ebbinghaus
Reaktor-Brennelement Union GmbH,
Rodenbacher Chaussee 6, 6450 Hanau 11
West-Germany

Introduction

The fuel rods in bundles of nuclear reactors are filled with heliumgas which is the
heat conveyer from the UO_2-pellet where the heat is generated to the zircalloy shell.
Helium owns the best heat conductivity of all inert gases and doesn't react with or
corrode any part of the fuel rod. Therefore, the helium has to be of excellent puri-
ty. The helium gas suppliers normally deliver, highly pure products, but it has to
be ensured that the filling gas quality within the fuel rods equals the one of the
delivered gas.

Because all other gases deminish the heat conducitivity of He, the tolerable amount
of impurities has to be very low. The task of the quality department or the chemi-
cal laboratory is to ensure on the one side the excellent purity of the delivered
gas and on the other side that this highly pure helium gas is encapsulated within
the fuel rods.

The fuel rods have to be re-opened to allow a determination of the filling gas com-
position. This is a destructive analysis for the fuel rod because the end plug or
the tube has to be damaged to release the filling gas to be gas analyzer. This
causes that only few fuel rods can be analyzed. Therefore, the homogenity of the
manufacturing during production (oxigen and moisture content) at the welding machines.
Additionally, the fuel rods of pressurised water reactor fuel bundles have to con-
tain a high pressure to withstand the force of the pressurized water circuit within

the reactor. During production the fuel rod pressure is controlled by automatic welding machines which are locked if the minimum pressure isn't achieved. After finishing a lot of fuel rods some of them are taken for controll of the inner helium pressure. The sample size (some per mille) depends on the lot size. In small lots (less than 1000 fuel rods) at least one rod is taken for analysis.

Description of the development

The gas analyzer which was built and developed at RBU consists of two parts, the lock system, where the fuel rod is opened and the fuel rod inner pressure can be determined, and the gas analyzer, a commercial available gas chromatograph (Carlo Erba) which was modified for the special purposes.

Measurement of the fuel rod inner pressure

For the pressure measurement the plenum end of the fuel rod is gas tightly clamped into a lock of known volume. The tightness is controlled by evacuating it to low pressure (10^{-3} bar) and maintaining it for a short period (about 5 minutes). If no pressure increase or only a very small increase (some millibar) is observed the system is considered to be gas thight.

Now, the fuel rod is punctured by a mandrel bar and the gas is released to the evacuated lock. Before measuring the pressure after the equalization the mandrel bar has to be replaced to the identical position it had before the puncturing. This is necessary to avoid a volume fault in the pressure determination.

The volume of the lock alone and the combinded volumes of the lock and the fuel rod have to be determined. In both cases the procedure is the same. The system is evacuated to a very low pressure (better than 10^{-3} bar). It is connected to another cell with well known volume and pressure. The connectiong valve is opened and a pressure equalization is performed. The unknown volume is calculated according to the Boyle-Mariotte law.

$$p. V = constant$$

Under the environmental conditions helium behaves like an ideal gas. Consequently all necessary calculations are performed applying the ideal gas law.

Following steps have to be performed successively:

- Clamping the fuel rod
- evacuating the lock
- pressure equalization with a cell of known volume and pressure
- reading the equlibrium pressure
- evacuating the lock (without the other cell)
- punchuring the fuel rod
- reading the equlibrium pressure
- (taking a sample for gas analysis)
- evacuating the lock and the open fuel rod
- pressure equalization with the other cell
- reading the equalibrium pressure
- calculating the fuel rod pressure

The measurement accuracy of the fuel rod pressure is better than 1 % relative.

Analysis of the filling gas purity

The gas composition measurement – or better – the determination of impurities within the filling helium gas is performed during the pressure measurement. As I have explained above only a limited content of other gases within the helium can be tolerated. The helium content has to be greater than 96 % of the total volume. The delivered helium is highly pure: 99.998 % or better. This purity is checked with the same gas analyzer as the filling gas. But on the way from the storage tank to the welding machine contaminations by other gases, especially air, are imaginable. These contaminations may be originated from gases absorbed at the inner surface of the pipe system or from small leaks in the system where air may diffuse into. This is not very probable because the pipe system contains the helium with a higher pressure than the environment has. Another source of contaminations are repairs at the pipe system, through which air penetrates into it. When repair is finshed the system is carefully purged with helium, but it is possible, that gases absorbe at the inner pipe surface and desorbe slowly later on. We couldn't detect such a contamination source up to now, but we take care that it wan't happen.

The best indicator for air contamination of the helium is the product – the fuel rod – itself. The second end plug is welded to the zircaloy tube under 22 bar helium atmosphere. First, there are very sensitive oxygen detectors in the helium support tube near welding chamber which allow to detect very small oxygen contaminations and which blocks the electrical power automatically at a pre-selectable value of oxygen content. Second, the zirkaloy itself is a very sensitive indicator for oxgen contamination. But normally no annealing colors appear during the welding process.

Nevertheless, in the gas analysis of the fuel rod filling gas small amounts of impurities are found. From where are they originated? Most of them come from gases (air, hydrogen, argon and others) which are adsorbed or included into the UO_2-pellets and which diffuse slowly out of them. At RBU the UO_2-powder is produced via AUC (Ammonium uranly carbonate). From this process a small amount of carbon dioxide remains in the UO_2-powder. During the sintering process which is performed under hydrogen atmosphere containing some percent nitrogen most of the carbon dioxide is removed, but in the closed pores these gases (CO_2, H_2, N_2) are encapsulated. After sintering the pellets are ground and stored under air. The fuel column is laid and filled under air, too. Before welding and pressurising with helium the total fuel rod and the welding chamber are evacuated to low pressure. But gases are adsorbed at the pellet surface and included into the pores from which the pellet surface and included into the pores from which the gases are released with the time to the filling helium. In case of PWR-fuel rods which contain a helium pressure of about 22 bars the impurities range near the detection limits (some vpm for the highest), but in case of BWR-fuel rods which contain only 2 bars of helium pressure these trace gases can be detected easily. The sum of all impurities remains far away from the specification limit which allows an impuritiy level up to 4 percent. Normally, the impurities lie far beyond 1 percent.

Description of the gas analyzer

The composition of the filling gas is analyzed by a gas chromatograph. The carrier gas is helium itself. The principle of the analysis is as follows. The impurities are separated from helium and mutually by distribution between a stationary phase (solid or liquid phase within the separation column) and the moving phase (helium). Because of the slightly different physical behavior the impurities distribute different between the stationary and the mobile phase. Gases of low molecular weight and unpolar gases are weaker adsorbed by the stationary phase than heavy and polar gases. So a light gas like hydrogen moves faster through the column than a heavy gas like carbon dioxide.

The applied gas chromatograph is able to separate and to detect H_2, N_2, O_2, Ar, CH_4, CO and CO_2. It is not possible to separate all these gases with one column, so we use two different stationary phases - a molecular sieve column and a Porapak-Q-column to separate the gases. Ar and O_2 give a joint signal in both columns. For a separate determination of them it is necessary to remove the oxygen by reaction with zirconium sponge which can be switched into the gas flow before entering the separation columns. To measure Ar and O_2 two runs are necessary, first without the zirconium sponge solumn to give a combined signal, than with it to result the Ar value only. The Ar-value is subtracted from the combined signal. The resulting value represents the oxygen content.

Procedure

The helium filled lock is connected to two sample loops. Each of them is connected to one separation column containing the molecular sieve or porapak - Q, respectively. First the loops are purged with the helium to be analyzed. Then they are closed to give a sample of constant and well known volume. The loops are connected to the carriergas flow (He, N 6.0) of highly pure helium and transported to the separation columns.

A He ionisation detector with a weak tritium source is used for detecting the impurities. The detector signal is recorded on a plotter and evaluated automatically by an integrator.

The sensitivity of the system lies between 0.1 Vppm and 1 Vppm depending on the gas to be analyzed. The reproducibility is better than 1 % relative and the overall accuracy is better than 5 % relative.

References:

1. K. Voldum-Clausen,

   Effective Removal of Oxygen from Notrigen Carier Gas
   in Gas-Liquid Chromatografic Analysis
   Analyst 94, (1969) 800-502

2. R.T. Parkinson, R.E. Wilson
   Anomalous Response of a He-Ionisation-Detector
   J. Chromatog. 24 (1966) 412-414

DISCUSSION

K. BALARAMA MOORTHY: What methods do you adopt for assessing or determining the internal pressure in the scaled elements. Do you adopt or develop acoustic emission methods?

H.J. VON WACHTENDONK: We do not apply such methods in final fuel assembly.

M. ERNOTTE: a) What is the frequency of production tests for rod inner pressure?
b) Do you correlate the rod experimental void volume with the theoretical one calculated from design requirements?

H.J. VON WACHTENDONK: a) One rod per lot, at least, or one per 1000 if the lot size exceeds 1000 fuel rod. Additionally, there are some internal analyses as demanded by production people.
b) No, we don't do this.

G. PAUWELS: Do you also measure the moisture content during destructive testing of fuel rods, togehter with the gas content?

H.J. VON WACHTENDONK: No, we measure the moisture content of the UO_2 pellets prior to loading, and we continuously monitor the H_2O content of the filling gas during welding.

IAEA-SR-102/16
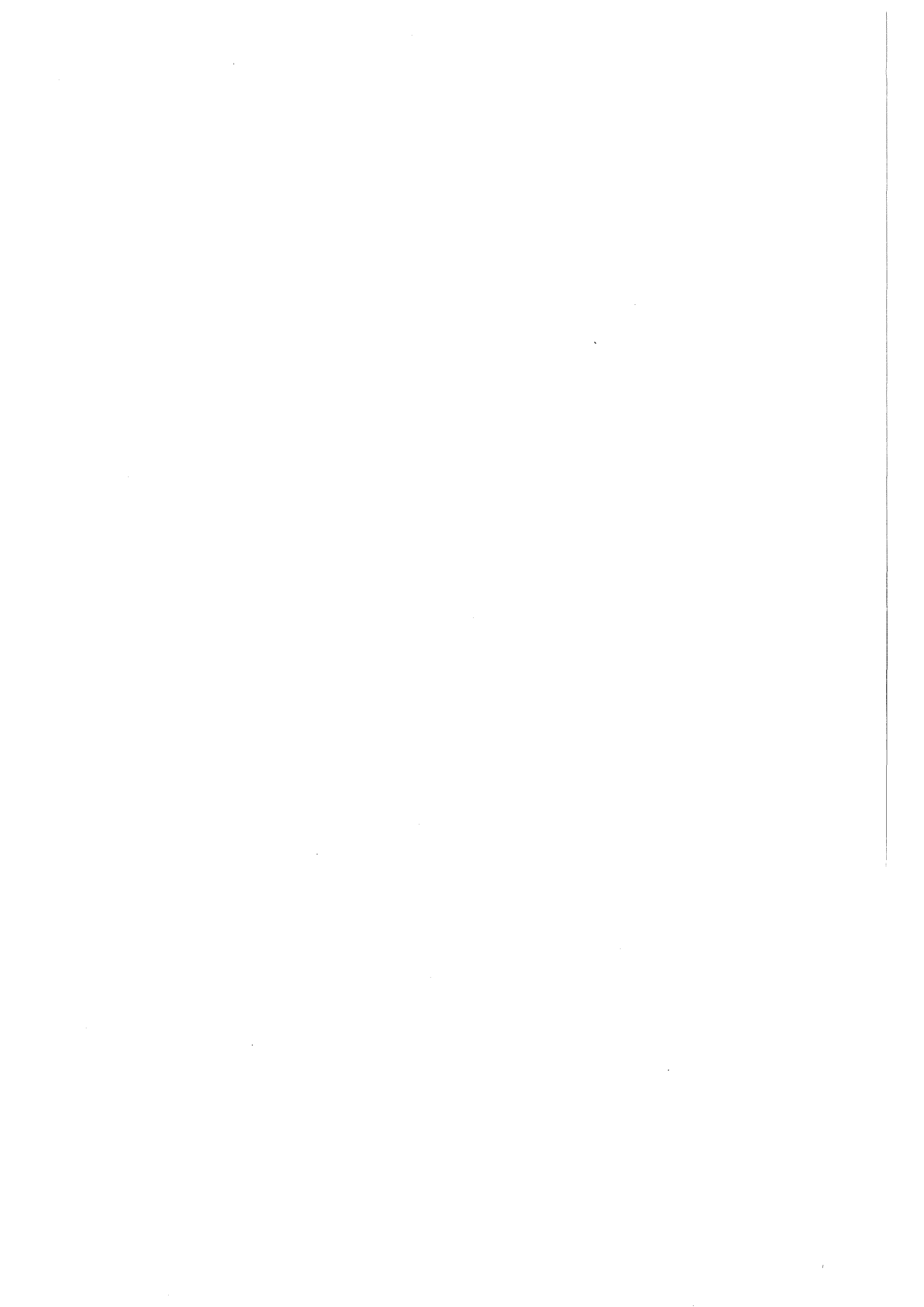
LR-O RESEARCH REACTOR EXPERIMENTAL FUEL PRODUCTION CONTROL

P. Veselý, J. Drexler, B. Fiala
Nuclear Fuel Institute
Prague - Zbraslav

J. Bardoš
Nuclear Research Institute
Prague - Rez

Abstract:

VZUP - Nuclear Fuel Institute, Prague-Zbraslav, in cooperation
with the Nuclear Research Institute, Rez by Prague, have designed
and manufactured VVER-440-LR 0 experimental fuel assemblies. During
the production the necessary control system was applied. This paper
gives basic information concerning the control system used as well
as the results obtained.

КОНТРОЛЬ ПРОИЗВОДСТВА ЭКСПЕРИМЕНТАЛЬНОГО ТОПЛИВА ДЛЯ

ИССЛЕДОВАТЕЛЬСКОГО РЕАКТОРА ЛР-0

П.Веселы, Я.Дрекслер, Б.Фиала
БРУП - Институт ядерных топлив
Прага
ЧССР

Й.Бардош
Институт ядерных исследований
Ржеж
ЧССР

Аннотация:

БРУП - Институт ядерных топлив (Прага - Збраслав) в сотрудничестве с Институтом ядерных исследований (Ржеж у Праги) сконструировал и изготовил экспериментальные тепловыделяющие сборки ВВЭР - 440 - ЛР 0. В производстве была использована соответствующая система контроля. В настоящем докладе приводятся основные информации об использованной системе контроля и о достигнутых результатах контроля.

1. Введение

Исследовательский реактор ЛР-0, находящийся в ИЯИ (Ржеж), предназначен для физических исследований активных зон типа ВВЭР. Для этого реактора БРУП - ИЯТ (Збраслав) и ИЯИ (Ржеж) совместно спроектировали и изготовили экспериментальные тепловыделяющие сборки ВВЭР - 440 - ЛР 0. Тепловыделяющие элементы производились на специально приспособленной экспериментальной лабораторной линии производительностью 300 кг UO_2/год. Каждое производство требует специальные контрольные операции. В настоящем докладе будут описаны операции контроля, примененные на всех этапах дабного производства.

2. Параметры экспериментального топлива

Параметры экспериментального топлива в основном соответствовали литературным данным для энергетических реакторов ВВЭР /1/. Однако, учитывая предназначение исследовательского реактора ЛР-0, некоторые из них (главным образом, структурные) не требовались, в результате чего можно было несколько упростить систему контроля.

Заказчик требовал определить и обеспечить следующие параметры:

Порошок UO_2 – изотопный состав – подтвердить аттестом поставщика и 10 контрольными измерениями из контейнера;

      – химический состав – подтвердить аттестом поставщика и 2 анализами на 10 выбранных элементов.

Таблетки – химический состав – содержание C – 200 ппм,

      содержание Fe – 500 ппм,

      содержание U – 87,7 %,

      отношение O/U = 2,00 ÷ 2,02;

– плотность $\rho_S \gtreqless 10,1$ г.см$^{-3}$;

– геометрические размеры – диаметр $\emptyset_{нар} = 7,56 \pm 0,04$мм,

      – осевое отверстие

      $\emptyset = 1,3 \begin{smallmatrix} +0,3 \\ -0,0 \end{smallmatrix}$ мм,

      – высота $h = 12 \begin{smallmatrix} +0,0 \\ -3,0 \end{smallmatrix}$ мм.

Тепловыделяющий элемент – геометрические размеры – см. рис. 1;

      – высота столба таблеток – 1250 ± 5 мм.

Тепловыделяющая сборка – геометрические размеры – см. рис. 2.

3. Основные операции контроля производства экспериментального топлива

3.1. Технология производства экспериментального топлива

Для принятого технологического процесса, изображенного схематически на рис. 3, характерны узел гранулирования с использованием вспомогательных органических веществ (поливиниловый спирт и стеариновая кислота) и операции перед спеканием /2/. Хотя этот процесс в настоящее время в промышленности

уже по⊄уществу не применяется, однако,ввиду его высокой универсальности, он целесообразен для экспериментального производства с низкой производительностью. Используя его, мы имели в виду еще одно его преимущество - меньшую зависимость от параметров порошка.

3.2. Контроль производства экспериментального топлива

В литературе (например, /3, 4, 5, 6/) приводится целый ряд процессов и организационных схем как осуществления собственно контроля, так и обработки полученных данных. При разработке проекта и реализации системы контроля производства экспериментального топлива мы придерживались обычного порядка, т.е. по этапам от проектирования и подготовки экспериментальной линии до собственно производства, как показано схематически в таб. 1.

Объем и включение принятых операций контроля полностью ясны из схемы на рис. 3.

3.3. Принятые аналитические методы

В ЧССР в ряде предприятий были разработаны аналитические методы для определения ряда примесей в металлическом уране и его соединениях ядерной чистоты. Одновременно был изготовлен и стандартный образец урана /7, 9/.

Для сравнительных определений содержания Fe, Cu, Mn, Ni, Cr, Cd, Mo, Si были использованы, главным образом, спектрофотометрические методы.

Для определения С был внедрен и использовался кулонометрический метод определения после сжигания в потоке кислорода ▬▬▬▬▬ при одновременном применении флюсов.

Отношение O/U определялось полярографическим методом добавок /8/ с двумя растворами при относительной погрешности определения 5 %.

Уран определялся титрованием бихроматом калия.

3.4. Остальные методы контроля

В качестве основного метода определения величины и формы частиц исходного порошка UO_2 нами был использован анализ фотографий, полученных на растровом электронном микроскопе /9/.

Кроме того, дополнительные данные были получены ситовым анализом.

Величина зерна и пористость спеченных таблеток определялась оптической металлографией при помощи обычной полировки алмазной пастой и проявления структуры химическим травлением серной кислотой и перекисью водорода.

Значительным дополнением были результаты, полученные технологическим испытанием спекания по ASTM C 753 - 73.

4. Процесс и результаты, полученные при контроле производства экспериментального топлива

4.1. Входной контроль (ВК)

Входному контролю подвергался по существу только порошок обогащенной UO_2, поставляемый из СССР. Результаты контрольного определения обогащения на 12 образцах приведены в гистограмме на рис. 4. Произведенные химические определения избранных элементов показали в основном соответствие с аттестом; полученные результаты даны в таб. II.

Наблюдения морфологии и величины частиц порошка при помощи растрового электронного микроскопа проявили очень широкий диапазон размеров частиц: от единиц до сотен мкм. Типичный вид порошка показан на рис. 5.

Проведенные технологические испытания спекания (1650°C, 4 часа) подтвердили хорошую активность спекания; при плотности сырых таблеток около 5,8 г.см$^{-3}$ была получена окончательная плотность таблеток выше 10,3 г.см$^{-3}$.

4.2. Производственный межоперационный контроль (МК)

Производственный межоперационный контроль на схеме показан между операциями 3 - 10. Он производится непосредственно обслуживающим персоналом технологического оборудования, в случае аналитики - в лаборатории института.

В узле подготовки гранулята (операции 3 - 6) производится контроль влажности смеси и параметров гранулята. Влажность колеблется в пределах 12 - 14 %. Параметры гранулята, такие, как текучесть, насыпной вес и их воспроизводимость, которые так важны для прессования, определялись в простом приспособлении, применяющемся обычно для испытаний металлических порошков.

Типичная скорость текучести, определяемая на 100 г гранулята, составляет 3,7 - 4,4 г/сек. По весу 25 см^3 слегка насыпанного материала рассчитывается насыпной вес, который достигает значений 2,2 - 2,4 г/см^3.

Химические анализы узла грануляции направлены на определение гомогенности содержания углерода и его полного содержания в шихте. Одновременно определяется повышение содержания железа вследствие истирания материала в ножевом смесителе. Среднее содержание углерода колеблется около 9800 ппм, а повышение содержания железа достигает значений 50 - 70 ппм.

При прессовании сырых таблеток производится регулярный контроль давления прессования и регулярный контроль приблизительно 1 % производимых таблеток. На регулярно отбираемых таблетках контролируется непосредственно их геометрия, геометрическая плотность, а также производится визуальный контроль при 25-кратном увеличении цилиндрической поверхности и торцов, особенно в области центрального отверстия.

После предварительного спекания контролируются, главным образом, визуально поверхность таблеток и аналитически содержание углерода, которое снижается под 100 ппм.

После спекания значение плотности спекания, определяемое методом двойного взвешивания, колебалось в пределах 10,3 - 10,65 г.см$^{-3}$. При измерениях возникают некоторые затруднения в результате задерживания пузырька воздуха в центральном отверстии. Всего контролируется 1,2 - 1,5 % спеченных таблеток.

После шлифования контролируется главным образом диаметр и остаточная влажность.

4.3. Выходной_контроль_(ВыК)

Выходной контроль был направлен на определение основных свойств таблеток, т.е. содержания урана, кислородного коэффициента, уточнение содержания некоторых примесей, плотности, геометрии и 100-процентный визуальный контроль таблеток.

Содержание урана в таблетках минимально 88,05 %, кислородный коэффициент до 2,01, содержание углерода до 100 ппм, содержание железа до 100 ппм. Плотность и геометрия соответствовали техническим условиям. Общая сумма данных дополнялась сведениями о структуре. Для иллюстрации на рис. 6 показана типичная структура таблеток.

Для каждого твэла для обеспечения гарантий определяется вес металла и изотопа, а кроме того, определяется и средняя плотность столба топлива перерасчетом на среднее значение толерантного поля.

4.4. Статистический анализ данных

Неотделимой составляющей контроля серийного производства является применение статистических методов/6, 9, 10, 11, 12/. При обработке данных использовались различные статистические методы - от наиболее простых, как определение уровня параметров выборки $\underline{X}$ и $\underline{S}$, определение доверительного интервала и построение гистограмм, до более сложных процессов анализа, как составление регрессионных моделей и использование факториальных испытаний.

Регрессионная модель разрабатывалась, например, для определения зависимости между плотностью прессования и давлением прессования. По этой модели были определены пределы для давления прессования, в которых давление пресса регулируется.

Недостатком перечисленных выше методов является то, что они не охватывают различные зависимости между отдельными параметрами. Поэтому была проверена возможность использования факториальных испытаний, а именно - двух- или трехфакториальных. В качестве примера можно привести исследование влияния "сырой" плотности, параметров предварительного спекания и размещения в печи спекания на конечную плотность спекания. В нашем случае, при использовании лабораторной шахтной печи с периодическим рабочим циклом, оказалось, что очень значительным фактором является размещение таблеток на этажах печи. Для этого случая рассчитанное значение $F \gg F_{крит}$.

4.5. Анализ эффективности принятой системы контроля

При анализе необходимо принять во внимание, что данное производство не превысило экспериментальный лабораторный уровень и что качество производимого топлива было изменено в соответствии с предназначением его к экспериментальным целям. Целью организации системы контроля было также получение доступного максимума информаций о технологических операциях и вывод соответствующих обратных связей. Большое влияние при этом оказывали, конечно, такие важные объективные факторы, как состояние оборудования лаборатории современной техникой и

приобретение необходимого производственного опыта.

В результате этого контроль отличается высокой долей рабочей деятельности при использовании обычной лабораторной техники. Поэтому абсолютно нельзя сравнивать данный случай с контрольной техникой в производстве топлива высокой производительности и с высокой степенью автоматизации.

Тем не менее мы, совместно с заказчиком, уверены, что данная система контроля была весьма эффективна и выполнила свое задание.

5. Заключение

Целью данного доклада было привести основные сведения о проведенных работах и достигнутых результатах в области контроля производства экспериментальных тепловыделяющих сборок ВВЭР - 440 - ЛР 0 для исследовательского реактора ЛР-0, установленного в Институте ядерных исследований в Ржежи у Праги.

Литература

/1/ Коллектив авторов: Опыт эксплоатации Нововоронежской атомной электростанции. ČSKAE ÚISJP, 1976, SNTL

/2/ Henke M.: Kernenergie 24, 1981, No 12, p. 450 - 455

/3/ Quality Assurance and Kontrol in the Manufacture of Metal - Clad UO_2 Reactor Fuels. Technical Reports series No 173, IAEA VIENA 1976

/4/ Bairiot H.: IAEA - SM 233/46 Prague 1978 ČSSR

/5/ Горский В.В.: Ат. техника за рубежом, 1981, № 4

/6/ Senevat J., Nury C.: IAEA - SM - 233/4 Prague 1978

/7/ Radil V., Jaroš J., Grammetbauer P.: Jaderná energie 24, 1978, 8, p. 297

/8/ Rubeš V.: Stanovení vodíkového indexu v keramickém palivu. VZUP - ÚJP, č. 321 (не публиковано)

/9/ Sokol L. a.kol.: Kontrola výroby jaderného paliva ze slinutého UO_2. VZUP - ÚJP, č. 300 (не публиковано)

/10/ Senevat J.: IAEA - SR - 7/6 OSLO 1976

/11/ Eckschlager K., Horsák I., Kodejš Z.: Vyhodnocování analytických výsledků a metod. SNTL/ALFA, 1980

/12/ Сакато Сиро: Практическое руководство по управлению качеством. Москва, Машиностроение, 1980.

Таб. 1. Схема реализации контроля экспериментального топлива

Проект и оборудование экспериментальной линии					Технологические испытания				Производство		
План контрольных процессов					Проверка контрольных операций				Проведение контрольных опер.		
Спецификация топлива и технические условия	Включение контрольных узлов в технологию	Разработка проектов моделей	Обеспечение и подготовка лабораторного оборудования		Проверка лабораторных методов	Определение рабочих пределов	Определение параметров моделей	Определение количества изделий для контроля	Уточнение объема контрольных выборок	Уточнение моделей	Статистические методы контроля и анализа данных

Таб. 2. Результаты анализа избранных элементов в UO_2

Элем.	Аттест /ппм/	Анализ /ппм/	Элем.	Аттест /ппм/	Анализ /ппм/
Fe	35	21	Cd	0,3	0,05
Cu	1	0,6	Mo	10	0,6
Mn	1	0,4	Si	10	15
Ni	10	3	C	60	78
Cr	10	5			
Отношение O/U: аттест 2,045; анализ 2,098					
Уран /%/: аттест 87,71; анализ 87,38					
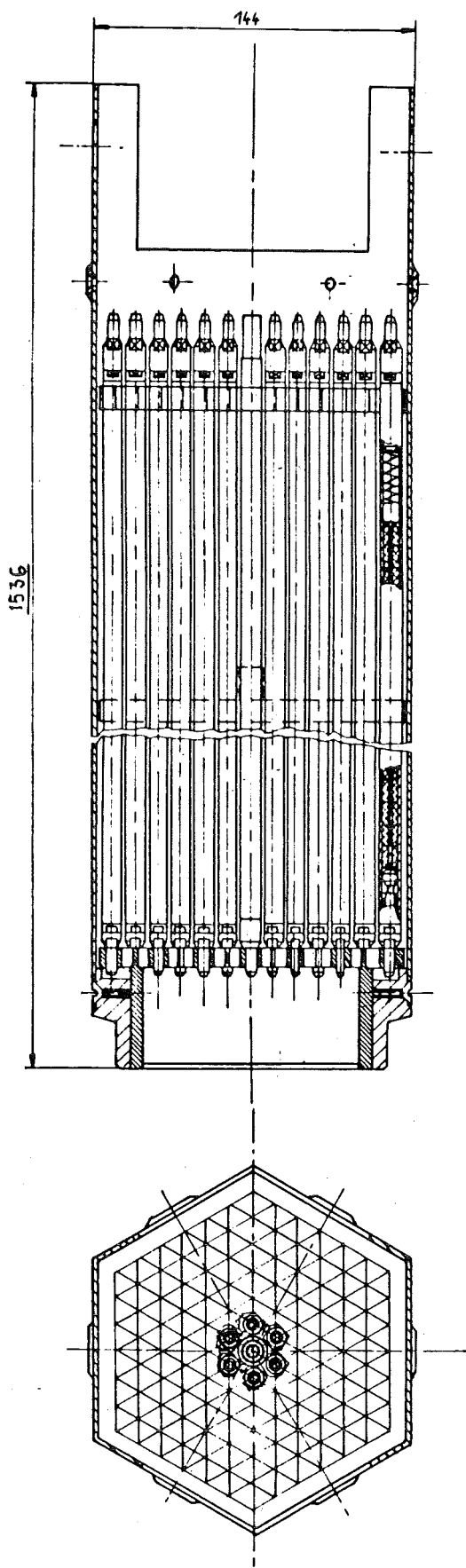
Рис. 2. Экспериментальная тепловыделяющая сборка
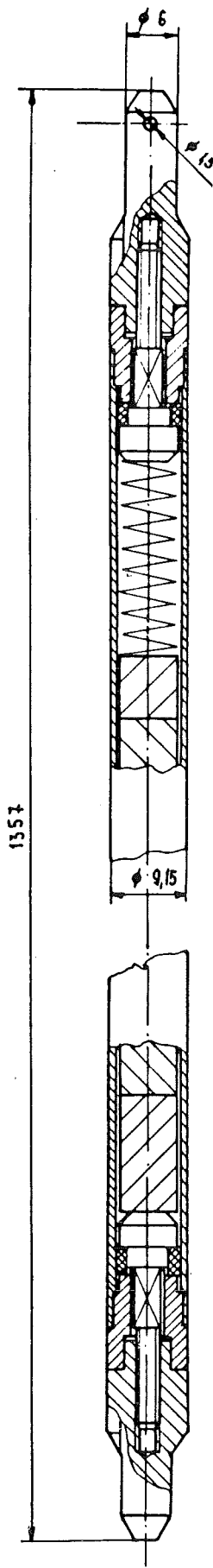
Рис. 1. Экспериментальный тепловыделяющий элемент

Рис. 3. Схема технологического процесса производства
топлива

контейнер UO_2	
отбор 50 кг	1
развес по 5 кг	2
смешивание	3
сушка	4
грануляция	5
классификация	6
прессование	7
предварительное спекание	8
спекание	9
шлифование	10
промывка	11
сушка	12
сборка твэла	13

H_2O C_2Cl_6 ПВС, СК

подготовка растворов связующего и смазки

ВК

МК

МК

МК

МК

МК

МК

МК

МК

ВыК

оболочки твэлов

завешивание твэла

упаковка твэлов в контейнеры
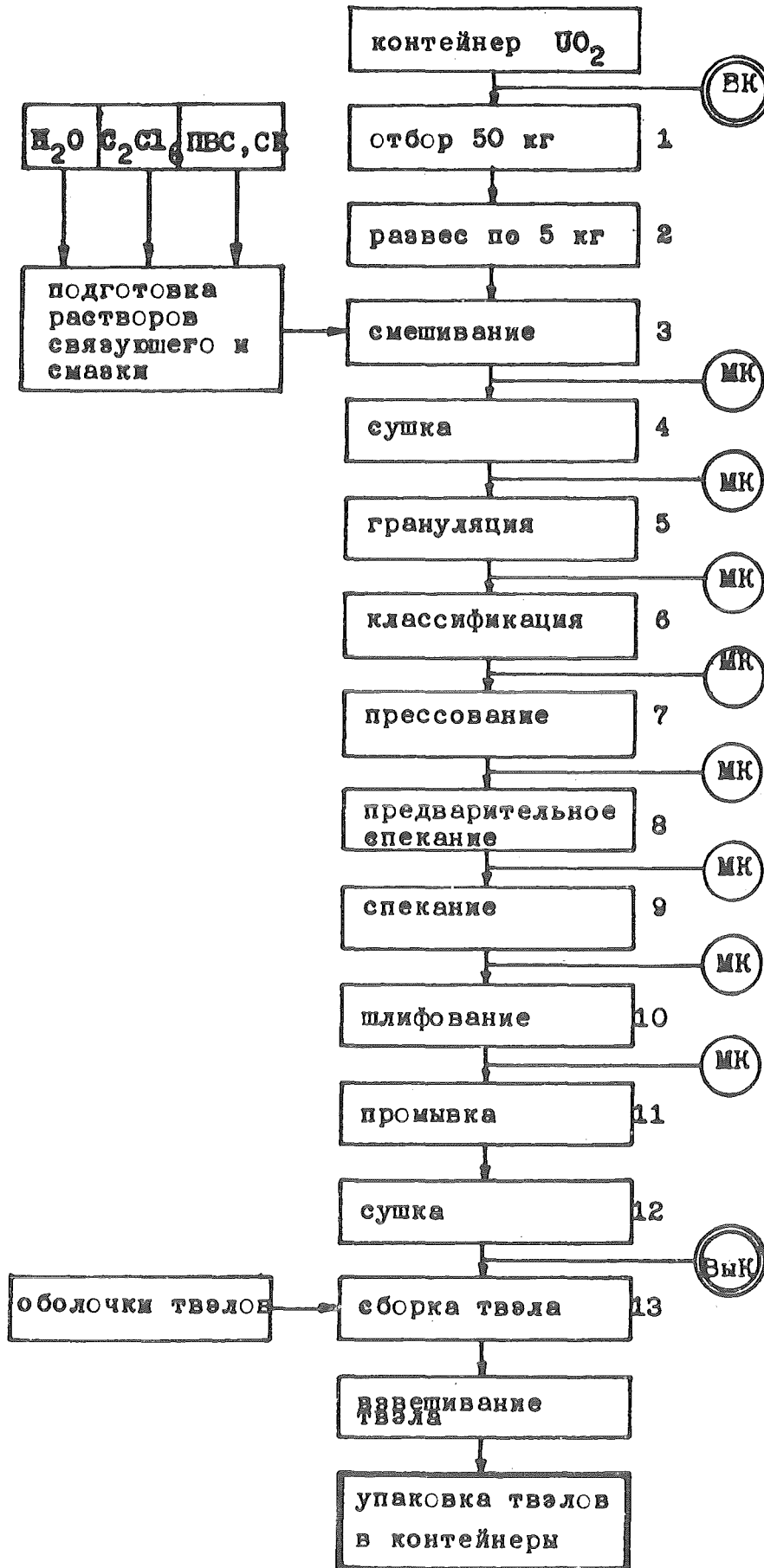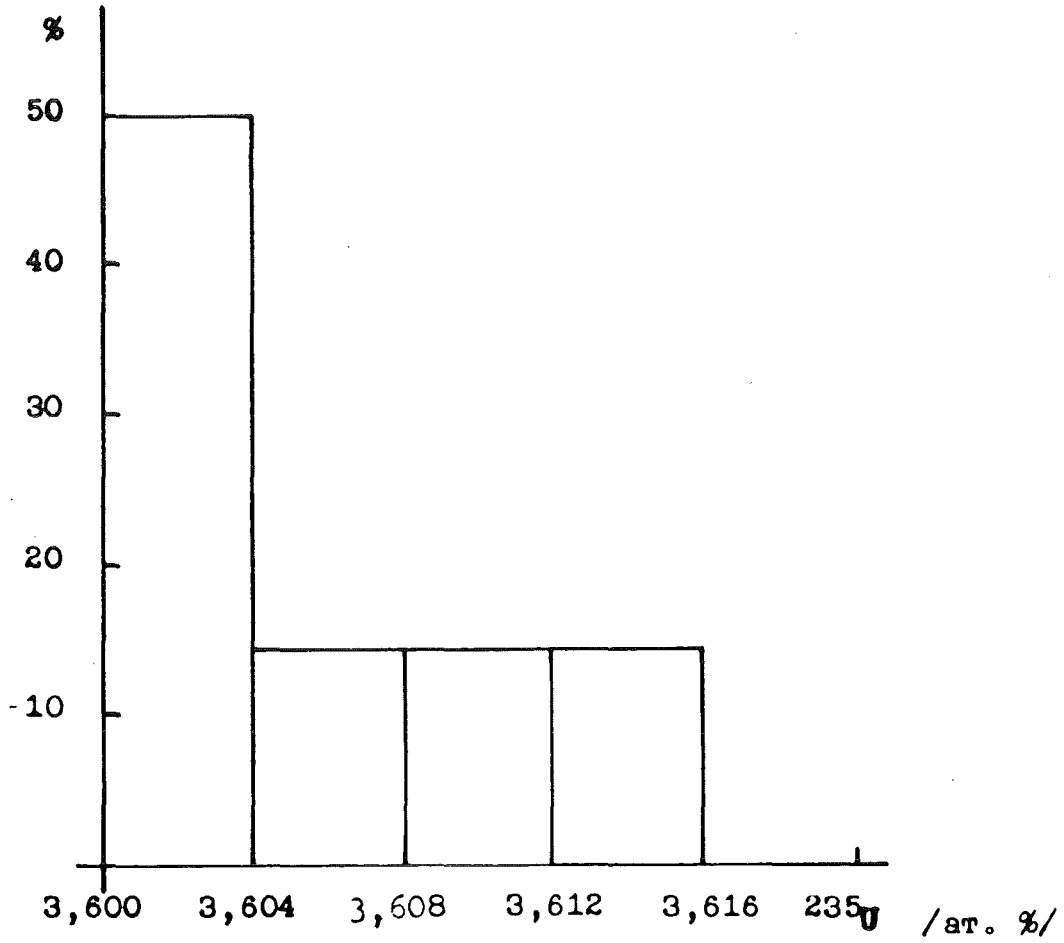
Рис. 4. Гистограмма обогащения топлива
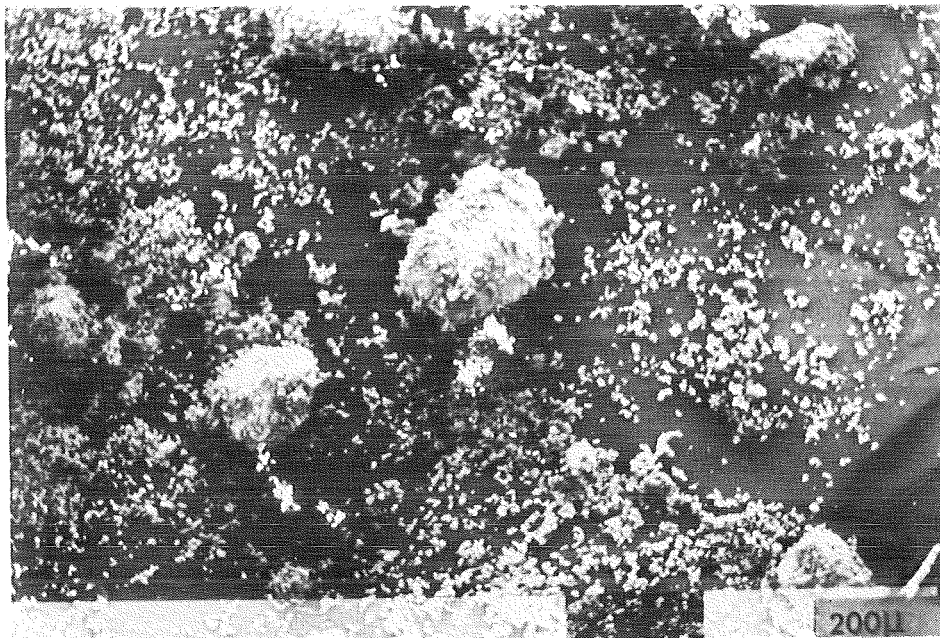
а)
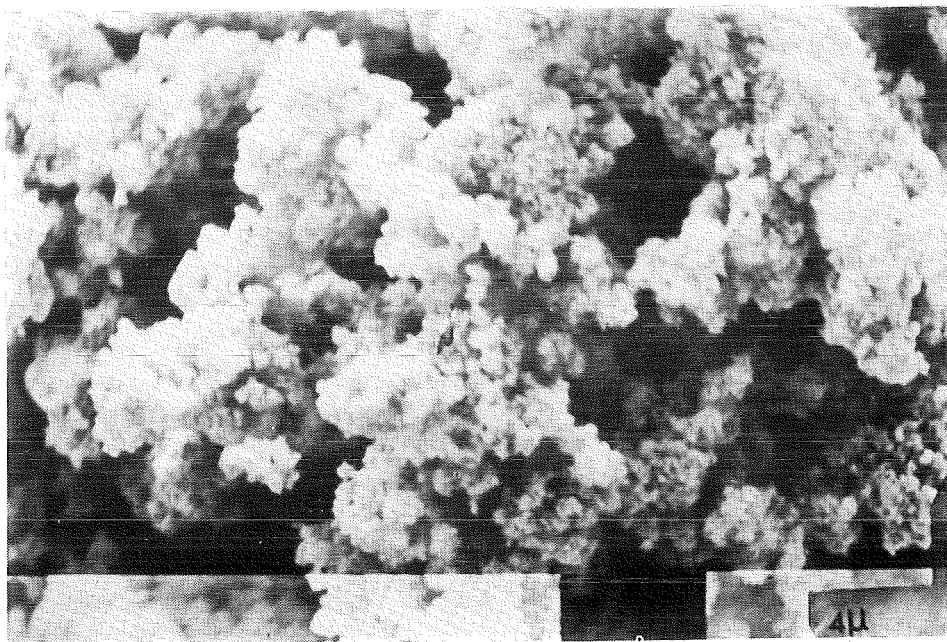
б)

Рис. 5. Порошок UO_2: а) ув. 90
б) ув. 5000
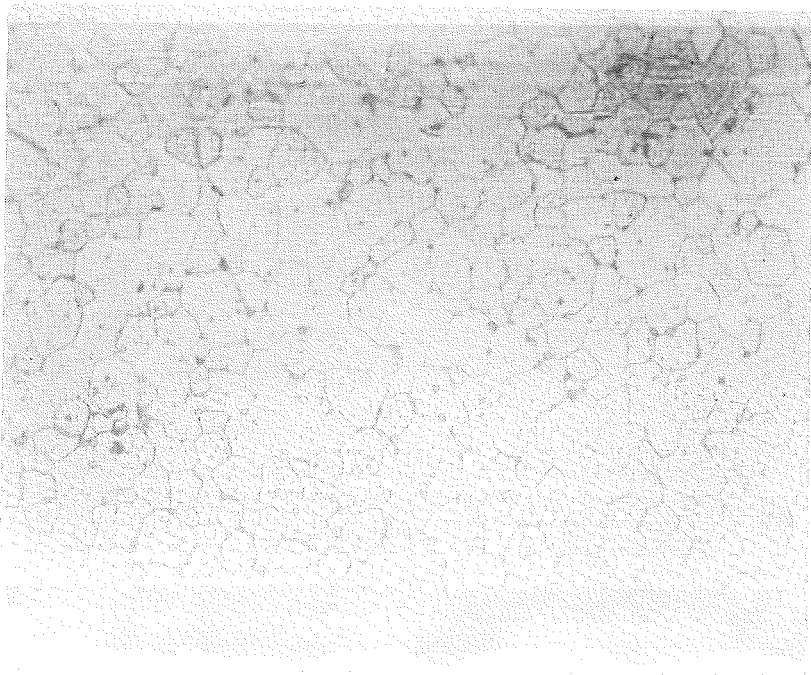
Рис. 6. Микроструктура спеченной таблетки

(ρ = 10,6 г.см$^{-3}$, ув. 500)

QUESTIONS/PROBLEMS OF QA/QC
PERSONNEL TRAINING AND QUALIFICATION
IN DEVELOPING COUNTRIES.

K. BALARAMAMOORTHY

BHABHA ATOMIC RESEARCH CENTRE

TROMBAY, BOMBAY-400 085, INDIA

A B S T R A C T

In India, nuclear power stations are designed, constructed, commissioned, operated and owned by the Department of Atomic Energy. The Indian nuclear energy programme is based on utilization of indigenous resources for the economic generation of power. The need for self-sufficiency in nuclear fuel fabrication is imperative for a sustained nuclear power programme. Since training of scientific and technical manpower is an important activity in a developing country, a manpower training programme was initiated in India several years before the introduction of nuclear power plants. It is essential to have a very broad based planning of manpower training in all its aspects for the successful implementation of nucler power plants. The paper deals in detail with the practices of establishing the manpower needs, training of requisite personnel, problems faced and how they were resolved. It also deals with the organizational philosophy and highlights the areas of concern for special training.

I INTRODUCTION

Indian Nuclear Energy programme is based on utilization of indigenous resources for the economic generation of power, developing its own know-how. The introduction and development of nuclear power coupled with the policy for self-sufficiency in any country will create unprecedented requirements. The technical complexities and unique safety and reliability requirements make it essential that highly qualified and properly motivated manpower be available from the beginning of and throughout the nuclear power programme. Hence a comprehensive manpower development programme should always be the integral part of nuclear power programme and should be consistent with the national practices and policies. This programme is essentially the integrated activity of planning and implementation of theoretical and practical multi-disciplinary education and training required to fulfill the manpower requirement for a country's nuclear power programme.

In India, for example, nuclear power stations are designed, constructed, operated and also owned by Department of Atomic Energy. Indian nuclear power programme has one of its main objectives - maximum utilization of indigenous manufacturing capacity for nuclear and conventional equipment (Table 1).

The need for self-sufficiency in nuclear fuel fabrication is imperative for a sustained nuclear power programme. Considering the meagre infrastructure within the country, it has been realised that nuclear fuel production in all aspects has to be arranged within the country.

After the experience and confidence gained with the manufacture of metallic uranium fuel for the research reactors and about 40 T of fuel (fuel bundles with natural UO_2 loaded in Zircaloy) for the intial loading of the first unit at Rajasthan, a decision was taken to set up fuel manufacturing facilities to meet the entire initial and reload fuel requirements both for BWR and PHWRs. At the Nuclear Fuel Complex set up at Hyderabad on the above objectives, production plants are in operation for the manufacture of reload fuel for the twin units at TAPS, natural uranium oxide fuel for PHWRs and other associated zircaloy components such as fuel sheaths, calandria tubes, pressure tubes, channels etc. All the plants are in operation and the fuel that is fabricated and supplied since mid Seventies is loaded in the reactors and very successfully irradiated. Plans are now on hand to expand the production facilities as the demand for initial and reload fuel is increasing.

II MANPOWER DEVELOPMENT - PLANNING

The manpower requirements for a nuclear power programme is substantial and varied and depends on many factors. It should be emphasized that it is necessary to have competitive quality engineers who have a higher degree of expertise in quality and reliability than is demanded by non-nuclear organizations. The demand of QA personnel has changed significantly in the recent years. While executing a nuclear power programme in a developing country, local industrial participation is not only desirable and feasible

but sometimes indispensible and essential. However, the extent to which participation takes place is a debatable point and will in fact depend, largely on local situation, facilities, resources etc.

Manpower planning is usually linked with the industrial base and socio-economic factors and the infrastructure available in the educational sector. In developing countries, a nuclear power programme very often symbolises introduction of modern science and technology both in its fundamental and applied aspects. As such, it is essential to have a very broad based planning of manpower training in all its aspects for a successful implementation of nuclear power plant.

Nuclear fuel technology can be considered as a closed loop process, starting with the design and development activities, followed by manufacturing and its specific quality control activities (in process as well as for product), and finally closed by the feed back of the operational experience, both for design and development efforts. The development of the fuel design commences with the initial conception of the reactor followed by activities such as proto-type fabrication, out-of-pile testing, inpile testing and post-irradiation examination. It is realised that the fuel design development requires extensive proto-type fuel testing before large scale production work is undertaken for reactor loading. All these activities are undertaken methodically in the Department of Atomic Energy.

During the manufacture of nuclear fuel, a comprehensive system of quality control is normally established since this is the method of applying all the actions necessary to control and measure the characteristics of an item or process. This practice is universal but satisfactory fuel is not necessarily the outcome of this management function alone. In addition overall QA is required that details and delegates authority for all those planned or systematic actions necessary to provide adequate confidence that fuel produced will perform satisfactorily in service. On the basis of the overall responsibilities, schedules envisaged and availability of the infrastructure, manpower planning and training assumes more importance for nuclear fuel needs.

III DETERMINATION OF MANPOWER REQUIREMENT

Manpower requirements, that is, adequate number of personnel with the necessary qualification and their availability at right time are influenced and dependent on the quality objectives. In the nuclear field, this objective has been developed with an emphasis on safety aspects and precision. The determination of the total QA manpower is rather difficult as it is not always easy to draw a line precisely between the QA work and other related engineering activities. The experience of a specific country cannot directly be applied to another country's problem but it could certainly provide some useful data if properly analysed and evaluated with respect to (a) scope and schedule of nuclear power programme (b) scope and schedule of national participation in the programme (c) constraints and limitations imposed by national manpower resources and infrastructure for development of manpower (d) level of industrial participation (e) phase of the project (design, construction or operation) and their contribution to manpower and resources.

In assessing manpower requirements, care should be taken to include, in addition to those people who actually will be needed to perform the task and functions, an adequate number of reserves/replacement personnel. In developing countries, as is experienced in India, there is usually a shortage of qualified manpower and consequently a high demand for qualified professionals, technicians and craftsmen.

IV TRAINING OF MANPOWER

Training of scientific and technical manpower is an important activity in a developing country. In India, for example manpower training in nuclear power programme started even several years before the introduction of nuclear power projects. All efforts were concentrated in developing manpower in basic sciences related to nuclear power, for example, Nuclear Physics, Chemistry, Metallurgy of Nuclear materials, Chemical engineering etc. The setting up of the research centre, Atomic Energy Establishment Trombay, now named as Bhabha Atomic Research Centre was an important milestone in this direction. Training School at Bhabha Atomic Research Centre was established in the year 1957 in which fresh graduates in Science and engineering from the various universities are recruited every year. The aim of the training programme is primarily to strengthen the knowledge in the basic subjects as well as to induct the trainees into nuclear science & technology in a broad way emphasizing the interplay of various disciplines because these subjects are not covered in the general curriculae of university graduates. Every year about 200 outstanding young graduates in Science & Engineering are recruited for admission to the training school in disciplines such as physics, chemistry, engineering (chemical, mechanical, electrical, electronics) and metallurgy. Table-2a shows BARC Training School statistics and Table-2b shows an example of distribution of persons in various disciplines who are presently undergoing training.

In order to meet the specific needs for the reactor operations and maintenance purpose, a separate training programme is organised since August '72 for training of technicians and craftsmen. It is realised that the availability of qualified technical manpower is a necessary condition for more viable nuclear power programme implementation and for the successful expansion of the technology. This is successfully accomplished in India by special training given to technicians at nuclear power stations.

Qualification requirements for inspection, examination and test personnel are general in nature and applicable to all phases of nuclear power plant, such as construction, manufacturing, fabrication, commissioning and operation.

V SPECIFIC ORIENTATION FOR QA/QC

Experience of many years of successsful fuel technology all over the world demonstrates that there are two basic aspects how this could be achieved and maintained. Quality has to be planned, produced and also verified. It should be realised that quality is just not a product of formalised actions and on the other hand it is the consequence of logical approaches on the basis of scientific and technical understanding of the products and processes (and also operational requirements. The whole area of fuel technology

can and has to be understood as a system of many feed-backs from sub-systems that are quality related and relevant.

Since past few years, Department is imparting orientation training by conducting familiarisation lectures on QA on the basis of IAEA Code and Guides periodically to the regular trainees. In addition to bringing general awareness, familiarisation lectures being given to middle and senior level scientific/engineering personnel.

It is generally assumed that recruiting a sufficiently highly qualified number of people resolves the problem. It could however, be said that for the daily routine work of QC in nuclear fuel manufacturing, it is not necessary and also not always possible to work mainly with very highly qualified people. It is absolutely essential that these people are always highly motivated and dedicated to their tasks. Needless to mention that these people must be experienced with regard to the various pertinent process steps in which they are connected and in application of the requisite control methods.

With the industry placing an increasing emphasis on Non-destructive Testing in the areas of quality control, liability, production efficiency and economics, it is more important than ever that NDT personnel have a thorough working knowledge of all types of NDT techniques. Participants are also required to know which type of testing should be done for numerous applications, why and how to apply NDT procedures and also how to interpret the results. Apart from some general training programme, specific training programmes both for technical and managerial staff, tailored to suit the particular job requirements of each and every person have been considered to be a must.

Generally the people are qualified in three different levels. Level I person should have capability for performing, documenting the inspection and test in accordance with the procedures, codes, standards and practices. Level II person in addition to Level I capability, should have capability to perform inspection and test planning, supervision and also evaluation of results. Level III person should possess the capabilities of Level II person and is expected to evaluate the adequacy of specific programmes of training and evaluation of the inspection and testing personnel and they should understand the responsibilities of the people they supervise.

Normally the recognition of the qualifications combined with the experience and categorisation with respect to the Levels is done in a developed country by professional societies. However in the absence of such societies in a developing country, the training and qualifying a person for a particular job or Level is the responsibility of the employer. In India, the DAE and some public sector undertakings who manufacture nuclear and other critical conventional equipment, a few recognized educational institutions and some technical professional societies organise frequently training programmes in fields such as Non-destructive Testing to meet the manpower demand at different "Levels". The curriculum for theoretical classes and practical training is carefully drawn and the course programme, qualification procedures both by written test and viva-voce tests are based generally on the pattern of ASNT requirements. Indian Standards Institution similar to BSI in U.K., is now formulating minimum requisites for certification of NDT personnel and these are comparable to International Standards with respect to the Levels

of proficiency. The basis used for certification include factors such as, education, experience and training, test results, capability demonstration etc.

Personnel for nondestructive examination are certified for one or more of NDT methods such as ultrasonics, radiography, magnetic particle, eddy current, dye penetrant tests etc. In order to qualify the 'professionals' for shouldering specific responsibility for executing QA/QC work, successful trainees are given specialised on-job training for a few years. Major topics for lectures in the training curriculae for quality assurance personnel include (a) Introduction to quality assurance (b) Quality assurance and Realiability (c) NDT techniques (d) Destructive examination techniques (e) Document, design and material control (f) Auditing (g) Metallurgy and properties of the metal and (h) Metrology.

VI ORGANIZATIONAL PHILOSOPHY

The principal philosophy being followed in establishing quality control organization is that the product must be of an acceptable quality. The quality control organization is made responsible for ensuring that all materials and components procured for use in the manufacture of fuel meet all requisite quality requirements and for certification prior to despatch that all products meet the customers' stated requirements. The quality of materials and components purchased from contractors is verified both at contractors' works by regular surveillance visits of QC group and by receipt inspection when the materials or components are delivered to the stores. During production, various stage and final inspections are carried out to confirm the product quality in accordance with the detailed quality plan using a wide variety of inspection equipment, from simple go/no-go gauges upto quite sophisticated NDT inspection facilities. Final certification that the product meets all requirements is not given until all the quality plan requirements have been fully satisfied. The typical organisational chart relevant to nuclear fuel is given in Table-3. Bearing the above in mind, the fuel production group and in-house quality control group are made to report to Chief Executive, NFC independently. The production group is responsible for production of required quantity to the required level of acceptance. QC Group, on the other hand, is responsible to assure that the products moving out of the plant are acceptable as per pertinent specifications.

Over and above in-house inspection, separate quality surveillance group exists for performing audit functions. This group is made responsible to evaluate and analyse primarily various OC methods and procedures followed at fuel fabrication and other plants' to fully assure conformity with contract requirements. The products leave the production plants only on issuance of a certificate from the QS group. All these activities detailed above need adequate number of scientific and trained manpower for effective functioning of the system.

VII SPECIAL TRAINING

The integrated approach is not considered to be complete till training is given to the personnel in both managerial and technical aspects.

The technicians are trained persons who are broadly knowledgeable and these are more highly trained in specific areas of technology. Therefore, they are expected to have better understanding in particular areas. The areas of concern which need specific attention in fuel fabrication and particularly for special training both from the view point of end product quality and to achieve higher yield are briefly detailed below:

(a) Production aspects:

O raw material control.

O handling of green pellets.

O sintering conditions, including rate of heating, soaking time and rate of cooling.

O grinding, including handling methods, dimensional control, surface finish requirement etc.

O drying conditions.

O cleanliness on fuel sheaths, other components and in general particularly in the assembly area.

O welding techniques employed for consistently achieveing requisite integrity of the weld.

O pressure and percentage of helium in the fill gas.

O chemistry of the weld.

O post-weld machining/cleaning operations on sealed elements.

O handling methods, sequence etc. for assembly of elements into a fuel bundle.

(b) Inspection aspects

O raw material clearance on scrutiny of test certificates and random inspection.

O appearance and density of green pellets.

O visual quality of sintered and ground pellets, and dimensions.

O moisture levels.

O weld integrity checks by both destructive and non-destructive tests.

O conformity checks for percentage of helium and for the fill gas pressure..

O dimensional control and internal soundness of the end closure welds.

O statistical quality control methods.

O fissile content and enrichment checks.

O Overall checks for the elements in the assembled bundle, including dimensional control, sequence of the elements etc.

O surface contamination checks.

Inspectors should be knowledgeable to understand thoroughly the process employed for fabrication, thereby enabling them to "look" for the presence/absence of defects and or deviations.

It is believed that it is necessary to place emphasis on the reliance mostly on In-service and on-the-job training programme.

When problems do arise during production of components which cannot be attended by QC crew of regular qualifications, these will no doubt have to be attended to and resolved by a highly qualified and experienced personnel. It will be essential for these

expert personnel to keep in constant touch to fuel technology laboratory where part of the time is used for research and development, and rest of the time for attending to special quality problems.

VIII CONCLUSION

Determination of the total QA manpower is dependent on various aspects such as type and range of activities handled, quality objectives, practices and conditions in the country, schedules etc. In most cases a high qualification level competent engineers/ personnel with good professional background and specialized training are necessary for performing the requisite functions. A comprehensive staffing and training programme for the professional/technicians/craftsmen is essential and has to be planned in advance. Manpower planning in advance, for identification of skills available in the country recruiting and training in the desired specific areas of high technology - such as production of green pellets, grinding of sintered pellets, enclosure welds for fuel element, critical NDT inspection and metrology steps, is found to be absolutely necessary.

It is also equally important to impart training in local environment in order to obtain the best results. In developing countries, manpower training is to be a continuous activity in order to minimize the effects of brain drain. Adequate renumeration to the employee commensurate with the importance in relation to the quality and final yield of the product, and intricacy of the task, is necessary. Like in any other organization, administrative restrictions sometime do arise to minimize 'over-employment' due to various reasons and in such instances, through careful planning, deployment of personnel from one specific activity to the other activity may have to be done to meet the demands of the work without in any way relaxing the 'quality' aspects either of the work or the employee.

The fuel plants set up by the Department of Atomic Energy designed and built in a time span dictated by the reactor programme have made the country self-reliant in meeting the fuel and component demands. Based on irradiation performance of indigenously fabricated fuel, research as well as power reactors, it is realized that QA & QC steps followed are adequate and in the proper direction in the nuclear fuel industry.

With proper attitude which recognizes the necessity of importance of QA, there will be commitment and with commitment quality will result.

***

DEGREE OF INDIGENISATION IN NUCLEAR POWER PROJECTS

KAKRAPARA-I & II

NARORA-I & II

MADRAS-I & II

RAJASTHAN II

RAJASTHAN I

TARAPUR

TABLE-1

— % INDIGENISATION (COST-WISE)

100
90
89
88
80
70
60
55
50
40
30
20
10
0
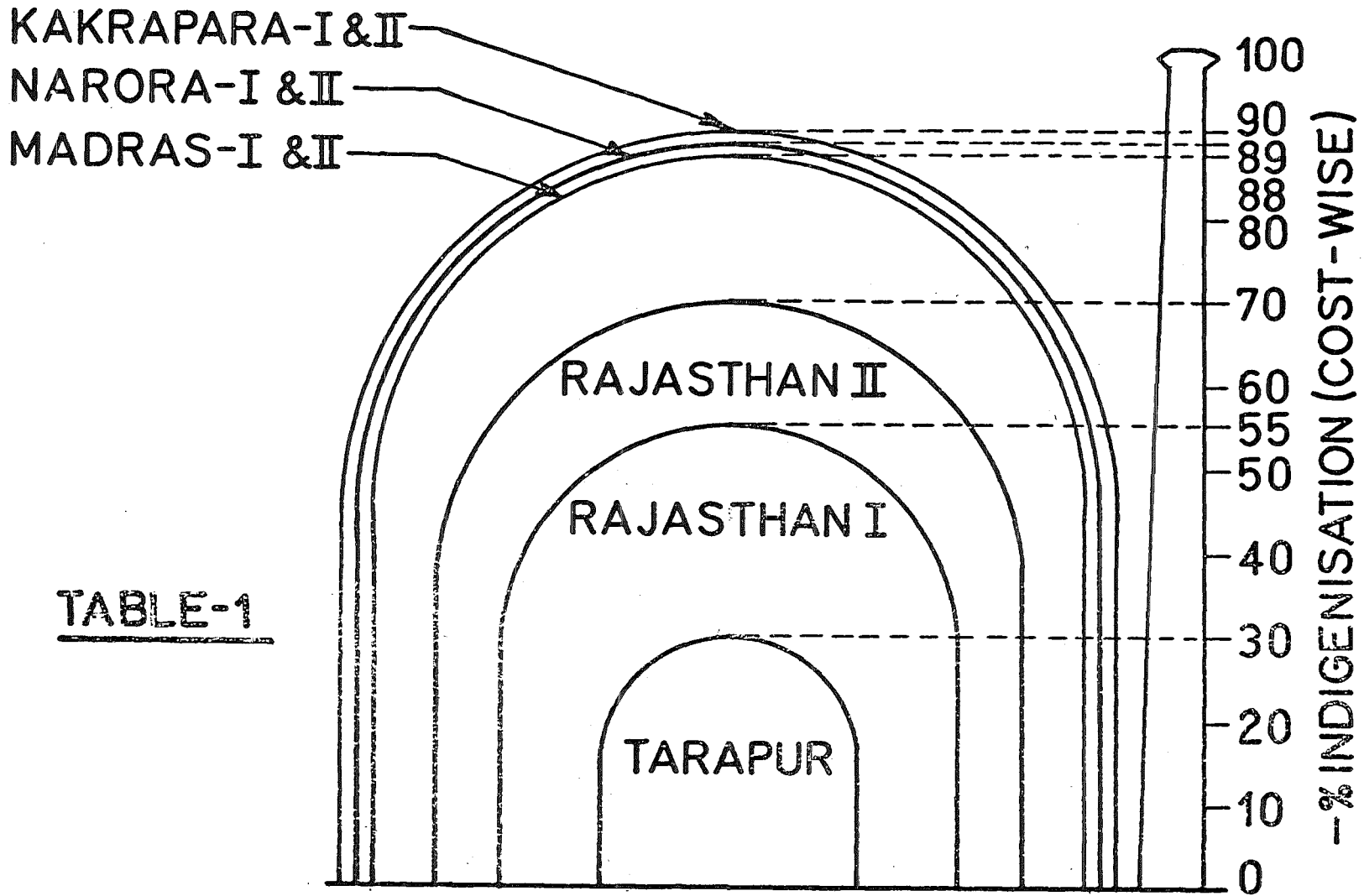
TABLE - 2a
BARC TRAINING SCHOOL STATISTICS
1957 to 1983

Course	Year	Applied for Admission	Called for Interview	Selected for Joining
1st	1957-58	6605	1897	255
5th	1961-62	3183	2310	260
10th	1966-67	4542	3165	242
15th	1971-72	8952	6738	245
20th	1976-77	8098	6805	266
24th	1980-81	3200	2554	369
25th	1981-82	6225	4579	381
26th	1982-83	4025	3055	272
27th	1983-84	3871	3062	279

TABLE - 2b
TYPICAL DISTRIBUTION OF PERSONS
WHO ARE UNDERGOING TRAINING 27th BATCH (1983-84).

	Regular Trainees	Inplant Trainees			
		NTC	HWP	DHRUVA	TOTAL
Physics	25	-	-	-	
Chemistry	23	-	-	-	
Mech. Engg.	22	9	4	3	16
Elec. Engg.	10	4	1	2	7
Instrumentation	7	-	-	-	
Chemical Engg.	8	3	6	-	9
Electronics Engg.	18	5	2	2	9
Metallurgical Engg.	6	-	-	-	
	119	21	13	7	41

DEPARTMENT OF ATOMIC ENERGY

PPED — BARC — NFC

PPED:
- NUCLEAR SYSTEMS
- POWER STATIONS
- FUEL DESIGN

BARC:
- MATERIALS GROUP
- REACTOR GROUP
  - REACTOR ENG. DIV.
- RADIO MET. DIV.
- ATOMIC FUELS DIVISION
- QA/QS

NFC:
- QUALITY CONTROL GROUP
- FUEL PRODUCTION GROUP
  - ZOP
  - ZSP
  - ZIRCONIUM FAB. PLANT
  - UOP
  - EUOP
  - FUEL FAB. PLANTS.

——— DIRECTION, RESPONSIBILITY AND ADMINISTRATION

———— DIRECT COMMUNICATION AND WORK CO-ORDINATION

TABLE-3 ORGANIZATION-RELEVANT TO NUCLEAR FUEL

References

/1/   K. Balaramamoorthy, Manpower requirements for QA programme.
       IAEA training course on QA for NPP - KAERI, Feb. 1983.

/2/   IAEA, QA + QC. IAEA Technical Report Series No. 173, 1976.

/3/   N. Kondal Rao et.al., Integrated planning for a fuel industry
       with emphasis on minimum size to fabricate own fuel.
       IAEA-CN-36/386(II.4)

/4/   K. Balaramamoorthy et.al., QA in developing countries with
       particular reference to Indian experience. Proceedings of
       IAEA symposium 'QA for NPP' Paris, May, 1981.

/5/   P.K. Iyengar et.al., Indian Experience in the Training of Manpower
       for Nuclear Power Programme.
       IAEA-CN-36/459 (VII. I) (b)

SESSION VIII

EVALUATION OF THE SEMINAR

Chairman: M.H. Bairiot
Belgonucléaire, Dessel

Chairman: M. Ernotte
Fragema, Lyon
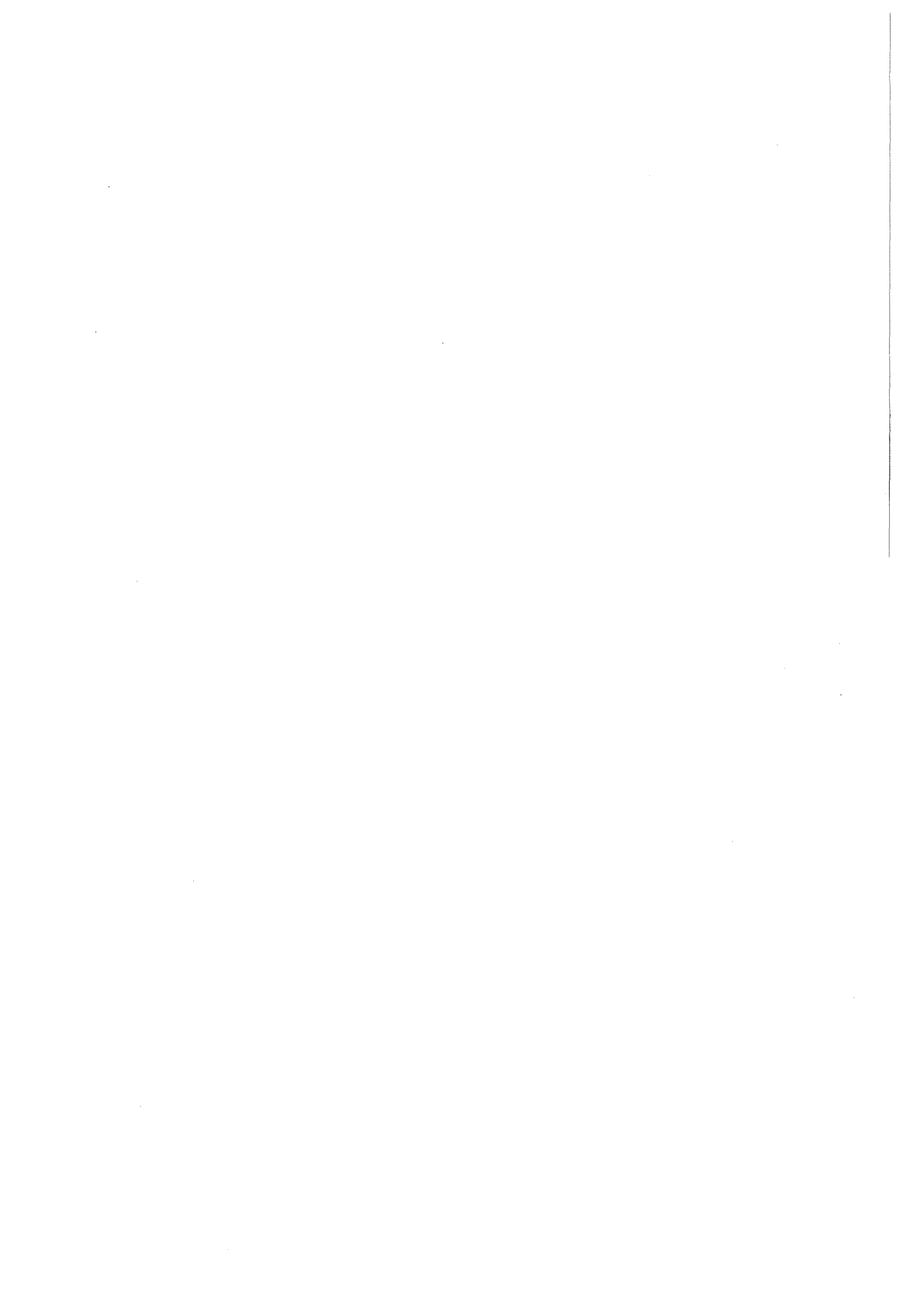
IAEA -SR-102/60

"PERFORMANCE-RELATED CHARACTERISTICS OF WATER REACTOR FUEL-
AN UPDATE"

P. Deramaix - H. Bairiot
BELGONUCLEAIRE - Brussels - Belgium

Various manufacturing techniques are utilized to fabricate fuel rods,
giving different characteristics to the fuel and a different response
to irradiation duties.   Additional differences arise from design
traditions specific to each fuel vendor.
On the ground of an appropriate data base complemented by proper
modelling, equivalent quality can be obtained from all these fuel
types and fuel rod designs.
The paper discusses the particular aspects linked to the fuel
structure as obtained by various conversion processes and the
utilization of various specifications of Zr cladding.

The fuel assemblies vary not so much by the manufacturing technique as
by the design itself.   The various aspects related to the quality of
the fuel assemblies are reviewed hereafter.

Finally, considerations are given on quality-related aspects of fuel
management at the site of the power plant, respectively from the point
of view of handling, in-core fuel management and spent fuel storage.

1. INTRODUCTION.

The presentation [1] at the previous Karlsruhe meeting in 1981 was limited to the fuel pellet attributes, since the topic of that conference was restricted to the fuel itself. In this paper, two objectives will be pursued :
- to discuss additional aspects relating to the data presented in the previous paper [1]
- to extend the presentation to the fabrication steps not considered in the previous paper.

Very little is to be added to the considerations outlined in the introduction and in the section on pellet manufacturing routes. Our viewpoints and beliefs had not to be altered from the experience accumulated over the three years elapsed since then.

The IAEA Guidebook on Quality Control of Water Reactor Fuel [2], to which we had contributed, is now published. It is an important document, perhaps imperfect and certainly incomplete, as any first attempt; the aim was to present as soon as possible an overview of the state of art, as resulted from the regional seminar organized by the IAEA in November 1979. Improvements are possible and certainly desirable. The material contributed to the present seminar is certainly a good opportunity to launch a revised edition of the guidebook.

The good behaviour of nuclear fuel is recognized [e.g.3], for the operating conditions experienced up to now; the failure probability of commercial fuel is currently lower than 10^{-4}. The trend to increase discharge burnup and in-core residence time is being backed by a data base [4] which indicates that present fuel is adequate. Improved power manoeuvering capability is being investigated to add flexibility to BWR plant operation and to expand the load following capacity of PWR plants; the results of the experimental programmes addressing those aspects are likely to cast additional light on the performance-related characteristics of nuclear fuel.

In this frame, one should not overlook that the learning cycle is very long in the field of nuclear fuel. Accelerated performance tests are often not representative and deceiving. Fig.1 presents the main feedback loops to be considered in accumulating enough experience for a commercial fuel or any improvement thereto. In this perspective, commercial fuel is likely to improve over the next decade and quality control will continue its trend to concentrate on the real performance related attributes.

This factual improvement of nuclear fuel provides a real challenge to QA and QC. In ANSI N-45.2, QA is defined as "all those planned or systematic actions necessary to provide adequate confidence that an item or facility will perform satisfactorily in service" and QC as "those QA actions which provide a means to control and measure the characteristics of an item process or facility in accordance with established requirements". If the requirement is to maintain a failure level lower than 1 rod out of 10 000 rods, does it still make sense to control to a 95/95 or even 99/99 confidence level, by satistical sampling of the product ? The only raisonable response is to put all the emphasis on a few 100 % controls of only key fabrication process parameters at product characteristics.

2. FUEL PELLETS.

The pellet manufacturing routes mentioned in the previous paper have not been modified over the last 4 years in commercial fabrication plants. The most noticeable alteration from previous practice is probably the adaptations in the MOX plants to assure a more uniform dispersion of the Pu in the fuel matrix, in order to improve the dissolubility in nitric acid [e.g. 5,6]. For U fuels, developments are being pursued to establish manufacturing routes for alternative fuel concepts such as duplex pellets or pellets coated with a burnable absorber.

Most of the quality control techniques are well established and continue to be improved to assure a good reproducibility, eliminate biases and reduce error margins. The large impact of quality control and assurance on the manufacturing costs has fostered a general trend to reassess the validity of the specified limits and even the necessity of some of the specified attributes. Whatever the economic incentive may be to broaden the specifications, the economic consequences of a deterioration of fuel quality are so overwhelming that the trend to simplify specifications can only progress at a reduced pace [12]. The evolution is consequently slow. Only few remarks will therefore be made on what has been presented at the previous meeting.

2.1. Plutonium.

The local distribution of the Pu has been improved with the adoption of manufacturing techniques assuring the dissolubility of MOX fuel. As a result, the historical emphasis on detecting to an acceptable confidence level the presence of high Pu-content grains or agglomerates has lost most of its significance. When confidence in process qualification in this respect will have built-up, the analytical control of Pu-rich grains should be simplified or even no more applied in routine fabrication [11].

2.2. O/M ratio

The performance effects of stoechiometry were discussed in the previous paper. When a certain burnup level is established, the stoechiometry of water reactor fuel reaches an equilibrium stage mainly influenced by the presence of the cladding, the fission products and the additives to the fuel (e.g. Gd or Pu). For the fluctuations of as-fabricated O/M ratios usually encountered in commercial manufacturing plants, the specified range should therefore depend on whether or not the fuel is likely to reach limiting conditions at low burnup (e.g. fuel for on-load fuelled reactors).

The O/M ratio was however found to play an indirect role in fuel characteristics such as H content, density, etc... The desire to maintain quite tight stoechiometry limits may therefore result in the future not from performance considerations but from manufacturing constraints.

2.3. Impurities.

The accumulated experience with prepressurized fuel rods confirms, as was predicted by our COMETHE modelling code [7], that thermal feedback effects occur less frequently but still play an important role in some power histories. The considerations presented earlier [1] on the impact of gas content are still to be taken into account but are usually not limitative for prepressurized rods.

The H content remains an important specification point. The emphasis should perhaps be placed not so much on a reconsideration of the specified limits, as on the coherence between the control process, the specification and the intrinsic response of each fuel to the extraction process used to analyse the H content.

2.4. Structure.

Fuel rods differing by their pellet structure (density, fabrication route,...) have shown to behave differently during irradiation, in particular during power changes (ramp tests). COMETHE sensitivity analyses have identified significant contributors : porosity, grain size and shape, tendency to form chips,... [e.g 9]. Those attributes should be maintained within qualified ranges; and departure from previous experience should thoroughly be evaluated.

3. FUEL CLADDING

3.1. Thermal and mechanical process steps are differing slightly from one commercial cladding tube manufacturer to the other.

T h e tube reducing process, and the annealing temperature and atmosphere history, in particular during the final process steps, are the most sensitive in controlling the mechanical properties and texture and the resulting behaviour of the cladding tubes under irradiation (irradiation growth, creep, maximum circumferential elongation).

In qualifying a tube manufacturer, the variability within a tube lot and from lot to lot should be scruttinized [13].

3.2. A discharge burnup increase from the actual standard values
(28,000 to 35,000 MWd/tM for BWR and PWR fuel assemblies respectively)
to target values such as 40,000 to 50,000 MWd/tM for BWR and PWR fuel
assemblies respectively is conditioned by a good corrosion resistance
of cladding and by a reduced loss of ductility of the cladding due to
hydrogen pickup. Corrosion resistance is primarily influenced by the
coolant water chemistry, but can be, to some extent, improved by a
fine dispersion of some metallic constituents, by limiting some
impurity contents (i.e. N) and by the final process steps; a better
understanding of the key parameters is likely to emerge from the
various experimental programs presently going on . Hydrogen content is
limited to less than 20 ppm by fabrication : combined to adequate
texture, the additional hydrogen pickup will not result in loss of
ductility margin.

3.3. Fuel performance under pellet-clad mechanical interaction and
operational or accidental transients is affected by inner diameter
tolerance (and the resulting pellet-clad gap variation), and by
ovality and wall thickness variation. These variations start at the
stage of machining hollows and are directly affected by the successive
tube reducing process steps. Specified tolerances and acceptance of
non-conforming material must be evaluated with regard to future rod
duty requirements.

3.4. Surface contamination (i.e. by F) has been the source of fuel rod
failure : special inner and outer surface treatments applied
commercially today have practically eliminated this cause of failure.

3.5. Surface defects and inclusions find their origin in the defects
of the starting raw material and in the internal tearing or surface
laps during the reducing process. Their effects depend on their
length, depth, location and pellet-clad mechanical interaction but are
limited by rejecting any defect exceeding standards proven acceptable
by irradiation experience.

3.6. Cladding tubes consist in large fabrication lots and the best
control is to insure that the process is run within the qualified
range of parameters.
   The cladding tubes are therefore controlled destructively only on
a very limited number of samples representative of the top, middle and
end of the fabrication lots.
   They are however 100 % controlled for outer diameter, wall
thickness and surface defects taking benefit of rather fast and
accurate ultrasonic and eddy current equipments.

4. FUEL RODS.

The main performance related requirements specified for the fuel rod are dealing with :

- the end plug welds quality ;

- the filling gas pressurization and composition ;

- the internal hydrogen content ;

- the potential occurrence of chips between the pellets or between the pellets and the cladding ;

- the fissile material contained in the fuel rod ;

- the overall length and plenum length.

4.1. End plug weld quality.

The TIG welding process has been used for the top and bottom end plugs and for the venting hole for prepressurization, for most fuel rods which have been manufactured for BN.

The welding process is first subjected to a qualification programme to show that all welds produced with the established parameter tolerances are within the specified requirements through non-destructive and destructive testing.

Afterwards, the welding operation relies on extensive weld process monitoring but periodic weld samples are produced during the course of fabrication to demonstrate the effectiveness of the process monitoring.

All the welds are submitted to visual examinations, X-rays and leak testing.

Among the other welding techniques, the electro magnetic resistance welding has also been applied for end plug welding. This technique is also submitted to an extensive process monitoring, 100 % visual examination and leak testing, but the very exhausting and questionable 100 % X-ray examination is replaced by simple go-no-go weld flash control. This efficient welding technique (even applied in case of cladding tube and end plug materials incompatible for TIG welding) has been used for test fuel rods, irradiated without problems.

4.2. Filling gas pressurization.

Zircaloy clad LWR fuel rods are routinely prepressurized to reduce cladding creepdown rate and to improve pellet-clad gap conductivity. Initial pressurization is in the range of 3 to 7 bars for BWR fuel rods, and in the range 20 to 30 bars for PWR fuel rods. Dry helium is generally used as the internal initial gas. Only a limited dilution of air is allowed, depending on the initial prepressurization : max. 5 % dilution by air is generally acceptable. BN irradiation experience as well as COMETHE sensitivity analyses has shown that a rather large tolerance is acceptable with regard to the initial pressure and the maximum allowed dilution of helium by air.

Filling gas prepressurization is controlled by monitoring the filling process; puncture tests are carried out on a very limited amount of fuel rods, namely those which are rejected for other defects (i.e. welds).

In the case of mixed oxide fuel rods, substoechiometric mixed oxide fuel pellets show a tendency to oxidize with the moisture contained in the pellets and in the internal gas, liberating hydrogen during the few days after sealing of the rod. That hydrogen is diluting the internal gas, but has not to be taken into account as a filling gas impurity.

4.3. Potential occurrence of chips between pellets or between pellets and clad.

Chips occurring at the outer surface of the pellets, either during loading the pellets in the fuel rods or as a result of fuel movements during handling of the assemblies may be the cause of fuel rod failure due to pellet-clad interaction. The probability of such occurrence is depending also on the fuel rod heat rating and power change rates.

X-ray control of the complete length of the fuel rod enables to detect the presence of chips between pellets or between pellets and cladding. Such a control is not feasible on all the fuel rods of a fabrication campaign, but only periodically i.e. on the fuel rods of the qualification lot. If the welds are 100 % X-ray controlled during routine fabrication, the control of the absence of chips in the plenum volume can provide an inexpensive verification of any departure from the qualified fabrication process.

The most effective way to avoid occurrence of chips is to rely upon high pellet quality standards and upon a well-designed rod loading device.

Furthermore, in case of mixed oxide plutonium fuel rods, this procedure has allowed BN to produce fuel rods with practically no outer alpha contamination.

4.4. Fissile material contained in fuel rods.

The fissile material amount in each fuel rod and in the finished fuel assemblies is important from following points of view :
- relative rod to rod power distribution in fuel assemblies ;
- reactivity lifetime of fuel assemblies ;
- safeguards.

Direct weight measurements are routinely carried out in the fabrication plants to guarantee the fissile material amount in each rod. Relative pellet to pellet fissile material content in each fuel rod is important to guarantee that the maximum authorized peak power is not exceeded. Administrative procedures and statistical control of the enrichment identification mark of each pellet in the dish should largely ensure that no pellet with an unspecified enrichment is loaded inadvertently in the rods. Additionally, an active or passive interrogation technique can be used not only for quality control of the fuel, but also as integral fissile material accountability of the plant (within the safeguard system); for performance considerations, it is only justified to measure the rods containing pellets with a lower enrichment than the enrichment of pellets simultaneously processed in the plant.

4.5. Overall and plenum lengths.

A 100 % control of the overall length is specified to guarantee t h e clearance provided for in the fuel rod and fuel assembly design, to account for Zircaloy cladding growth under irradiation.

The fuel rod thermomechanical design analyses (i.e. maximum internal pressure) are carried out assuming a minimum plenum volume, based on the worst combination of the tolerances, including possible cladding length reduction due to some end plug repairs. Fabrication procedure should ensure that the minimum plenum volume is met. A 100 % control of the minimum plenum length gives an added assurance, not only for the plenum length, but also for possible pellet to pellet gaps.

5. FUEL ASSEMBLY STRUCTURE.

The recent years have confirmed, mainly in modern PWR's, the trends towards the utilization of Zircaloy-4 as structure material for the spacer grids and guide thimbles and towards higher discharge burnup. The main performance-related characteristics involved in these progressive changes are discussed hereafter in terms of quality control methods and procedures (Fig. 2).

5.1. The spacer grids consist of a square network of Zircaloy-4 blades which constitutes the frame in which the fuel rods are supported by rigid buttons and springs. In the same spacer grids, the springs which apply forces to the fuel rods to fix them accurately in position against rigid buttons are integral. In some other cases, they are made of Inconel and separately attached.

Stiffness is given to the spacer grids by welding each crossing point and to the assembly structure by welding or mechanical deformation of spacer-guide thimble connections.

5.2. <u>Corrosion resistance</u> of spacer grids and guide thimble material is subject to the same corrosion problems as the fuel cladding. The quality assurance is based on specified corrosion tests on incoming materials and on process control monitoring to avoid contaminants.
Similar procedures and methods are applied to ensure corrosion resistance of the welds.
The quality is essentially dependent on a qualified manufacturing procedure. This qualification is established by metallographic examinal on corrosion tests and mechanical tests.

5.3. Irradiation-induced growth of Zircaloy.

The Zircaloy material structure and metallurgy state as well as its dimensions are selected during the design phase, such that spacer to spacer gap between fuel assemblies and total fuel assembly length remain within the limits taken into account in the design, in hot and cold reactor conditions.
As for cladding tubes, rolling processes with regard to the longitudinal direction of the Zircaloy blades and annealing temperature, mainly during the final process steps are the most important in controlling the growth of the Zircaloy spacers during irradiation.

5.4. Spring load.

The spring is designed such that the spring load be within acceptable design limits, taking into account worst combination of tolerances and irradiation induced relaxation effects.
Bimetallic spacer grid springs of BN design present a smooth load-deflection characteristic allowing for a large rod to rod distance tolerance, which guarantees the minimum spring load up to the highest anticipated fuel burnup. Careful process monitoring ensures that springs meet the design requirement.

5.5. Finished assemblies

The final control of the assembly consist in elaborate dimensional measurement performed on a control tower to check rod-to-rod spacing, straightness, bow, twist, enveloppe, etc. The quality of as-fabricated fuel should be maintained at an adequate level during the full assembly lifetime. The control of the quality at this stage is usually called "fuel surveillance" and has been discribed more extensively elsewhere e.g. 8, 10 . It will only be discussed here to check coherence with fabrication controls.

6. IN-REACTOR FUEL PERFORMANCE ASSESSMENT.

Fuel surveillance at the power plant (Fig.3) starts with the incoming inspection of the fuel at its arrival at the power plant, is pursued with in-core surveillance to detect the occurrence of the fuel defects and with pool-side examinations, in particular of lead assemblies, and includes quality control of reconstituted fuel assemblies after their repair, before their loading in the core for further irradiation.

6.1. Incoming inspection.

The incoming inspection is performed to verify that the fuel assemblies were not damaged during the handling and transportation operations. This incoming inspection includes therefore the followings steps (Fig.4) :
- Visual inspection of the shipping containers and the checking of their accelerometers to verify that the containers were not submitted to abnormal chock during their handling and transportation ;
- Identification of each assembly ;
- Visual inspection of the fuel assemblies to check overall conformance with the drawings and the absence of defects caused by handling or transportation ;
  assemblies ;
- Functional controls such as the introduction of a plug assembly or a full length rod control cluster, in case of RCC type fuel assemblies;
- Health physics measurements, i.e. alpha surface contamination in case of mixed oxide fuel assemblies.

6.2. Core surveillance.

Coolant water is continuously monitored from water chemistry and radioactivity points of view. Water chemistry characteristics control is important to avoid extensive corrosion of fuel assemblies and other components of the primary circuit and the accumulation of radioactive cruds. The subject is discussed in other specialists meetings.

Radioactivity increase indicates the occurrence of fuel failure. The main leakage indicators are I, Cs, Xe and Kr isotopes, from which it is possible to detect the number of failed fuel rods in a core, the burnup and the region of the failed fuel.

6.3. Pool-side examinations

The efforts to reduce uranium requirements and to allow for power plants operation in daily load-follow mode lead to the progressive increase of extended discharge burnup, and more advanced design fuel. On-site fuel characterization programmes are therefore set up to verify the capacity of the fuel to meet performance requirements, to identify and isolate failed fuel components, and to demonstrate the benefits of the design changes under consideration. These on-site fuel characterization programmes are conducted at fuel supplier's or plant operator's initiative. Sometimes they are imposed by the Regulatory Requirements.

Many fuel performance characteristics are directly observable from on-site non-destructive examination techniques. The relationship among the examination techniques and performance characteristics are tabulated in Table 1 and have been reviewed recently in details [8], [10].
Fuel examination facilities have been designed and installed in spent fuel storage pool of all the Belgian power plants. As a minimum these facilities allow visual examination and sipping of the fuel assemblies. In particular, spent fuel sipping is always required before shipping the fuel to the reprocessing plant.

6.4. Fuel repair.

6.4.1 There is an obvious economic incentive to repair fuel assemblies which have failed early in their life. The fuel examination facilities can generally be used to repair the defective assemblies.

6.4.2 In case the examinations show that the defective assembly structure is in good condition and can be kept, the repair consists of :
- removing the dismountable end nozzle ;
- locating the defective rod(s), e.g. by removing each rod almost completely out of the structure, through an eddy current probe mounted at the extremity of the fuel rod extraction tool and to reintroduce the sound rods immediately ;
- replacing the defective rods by either Zircaloy bars or water rods or by fuel rods (fresh with appropriate enrichment or irradiated at appropriate burnup) ;
- replacing the dismountable end nozzle and securing it.

This method has been proven practical : all the operations are carried out under visual controls (camera, endoscope...) and the extraction forces, never exceeding predetermined values, prevent the assembly structure from deformation. It remains however that this repaired fuel assembly goes back to the reactor without any elaborate dimensional control similar to those mentionned in para 5.5.

6.4.3 In case the defective assembly structure is damaged too, the most practical way to repair is to transfer all the sound fuel rods from the damaged structure to a new one and to replace defective fuel rods by Zircaloy bars or appropriate fuel rods (fresh or irradiated).
This is generally the case at present with low burnup fuel as the failure cause is not in the fuel itself (i.e. baffle water jetting problems).

The main quality control problems to be solved in such a reconstitution are :
- the correct location traceability of each identified sound rod. This can be guaranteed by an appropriate working procedure ;
- the securization of the dismountable end nozzle to the assembly structure after reconstitution. Qualified procedures and tools have proved to be able to guarantee reliable securisation.

7.CONCLUSION.

Although quite complex, the relative influence of individual attributes becomes now reasonably understood. It enables to progressively shift from a specification and Q.C. plan based on a "repeat what has been done earlier" to a more founded approach, which should enable to focuse Q.C. efforts on attributes directly related to fuel performance and to enlarge acceptance of non-conforming material.

In this respect, we are operating and continuously updating a consistant package of deterministic codes in the various aspects of the design of fuel assemblies (thermomechanical design of fuel rods, pressure drop of spacer grids, thermohydraulic design of fuel assemblies and core, mechanical design of spacer grids and assembly structure, nuclear design of fuel pellet, fuel rod, fuel assembly, core) [9]. The calibration and the sensitivity of these codes have shown their capability in a quick and inexpensive assessment of non-conforming products to justify their acceptance or their definite reject.

REFERENCES

[1] PERFORMANCE-RELATED CHARACTERISTICS OF WATER REACTOR FUEL,
H. Bairiot, Y. Vanderborck, G. Dumbruch, "Characterization and
Quality Control of Nuclear Fuels", Karlsruhe, June 2-5, 1981.

[2] GUIDEBOOK ON QUALITY CONTROL OF WATER REACTOR FUEL, IAEA,
Vienna, 1983.

[3] EXPERIENCE IN FABRICATION TECHNOLOGY AND PERFORMANCE OF WATER
REACTOR FUEL, P. Knudsen (Risø), H. Bairiot (BN), S. Sandklef
(SSPB), "IAEA International Conference on Nuclear Power
Experience", Vienna, Sept. 13-17, 1982.

[4] HIGH BURNUP : STATUS AND EXPECTATIONS, H. Bairiot, "IAEA
International Conference on Nuclear Power Experience, Vienna,
Sept. 13-17, 1982.

[5] FABRICATION IN BELGIUM OF MIXED OXIDE FUEL FOR PLUTONIUM THERMAL
RECYCLING, J.M. Leblanc.

[6] LAYING THE FOUNDATIONS FOR PLUTONIUM RECYCLE IN LWR's,
H. Bairiot, "Nuclear Engineering International", Jan. 1984.

[7] FUEL CHARACTERISTICS REQUIRED FOR LWR FUEL ROD CALCULATIONS,
E. De Meulemeester, "Characterization and Quality Control of
Nuclear Fuels, Karlsruhe, June 2-5, 1981.

[8] ON-SITE NON-DESTRUCTIVE EXAMINATION TECHNIQUES FOR IRRADIATED
WATER COOLED POWER REACTOR FUEL - A GUIDEBOOK, IAEA-TECDOC-245.

[9] OXIDE BEHAVIOUR MODELLING PROGRESS IN COMETHE, J. van Vliet,
N. Hoppe, "IAEA Specialists' Meeting on WR Fuel Element
Performance Computer Modelling, Preston, March 1982.

[10] CONTRIBUTION OF POOL-SIDE AND HOT CELL EXAMINATIONS TO LWR FUEL
SURVEILLANCE PROGRAMS, H. Bairiot et al, "IAEA Specialists'
Meeting on Examination of Fuel Assemblies for Water Cooled Power
Reactors, Tokyo, November 1981.

[11] PLUTONIUM ENRICHED THERMAL FUEL PRODUCTION EXPERIENCE IN
BELGIUM, J.M. Leblanc et al, "International ENS/ANS Conference -
New Directions in Nuclear Energy with Emphasis on Fuel Cycles,
Brussels-Belgium, April 26-30, 1982.

[12] ECONOMIC CONSEQUENCES OF QA AND QC IN FUEL AND FUEL ASSEMBLY
PRODUCTION, A.A. Strasser, Paper presented at this Meeting.

[13] STATISTICAL METHODS IN THE QUALITY CONTROL OF UO_2 PELLETS AND
ZIRCALOY CLADDING TUBES FOR WATER-REACTOR FUEL RODS, H. Assmann
et al, Paper presented at this Meeting.

EXAMINATION TECHNIQUES (CHAPTER) / **FUEL CHARACTERISTICS**

Columns 5–8 (VISUAL COMPARISON, REPLICATION, LVDT, EDDY CURRENT) fall under the heading **DIMENSIONAL TECHNIQUES**.

FUEL CHARACTERISTICS	FUEL MONITORING METHODS DURING OPERATION (5)	SIPPING (6.1)	VISUAL (6.2)	GAMMA SCANNING (6.3)	VISUAL COMPARISON (6.2)	REPLICATION (6.4)	LVDT (6.4)	EDDY CURRENT (6.5)	IDENTIFICATION OF FAILED RODS (6.5)	CRUD SAMPLING (6.6)	OXIDE THICKNESS MEASUREMENT (6.7)	LOAD-DISPLACEMENT TECHNIQUES (6.8)
ASSEMBLY BOWING			X		X							
ASSEMBLY HOLD-DOWN SPRING FORCE												X
ASSEMBLY IDENTIFICATION			X									
ASSEMBLY INTEGRITY (MECHANICAL)			X									
ASSEMBLY TWIST			X									
CLADDING CORROSION											X	
CLADDING DIMENSIONS (DIAMETER, OVALITY, RIDGE HEIGHT)							X					
CLADDING INTEGRITY (ESPECIALLY HERMETICITY)	X	X	X	X	X			X	X			
CRUD COMPOSITION										X		
CRUD PATTERN			X									
CRUD THICKNESS			X								X	
FLOW CHANNEL CORNER WEAR			X		X	X						
FLOW CHANNEL DIMENSIONS (BOW, FLATNESS, DISPLACEMENT, TWIST)							X					
FUEL COLUMN AXIAL DIMENSIONS (GAPS, LENGTH)				X								
FUEL COLUMN RELATIVE POWER				X								
GUIDE TUBE DIMENSIONS (INSIDE DIAMETER AND WALL THICKNESS)								X				
ROD IDENTIFICATION			X									
ROD LENGTH						X	X					
ROD-TO-GUIDE TUBE SPACE						X	X					
ROD-TO-ROD SPACE						X	X					·
ROD-TO-END FITTING SPACE						X	X					
ROD WITHDRAWAL FORCE												X
SPACER GRID INTEGRITY			X									
SPACER GRID POSITION						X	X					
SPACER GRID SPRING FORCE												X
SPACER GRID CELL DIAMETER						X	X					

TABLE 1. IRRADIATED FUEL CHARACTERISTICS OF INTEREST AND THE RELATED EXAMINATION TECHNIQUE

(FROM REFERENCE (8))

FIG. 1 — FUEL QUALIFICATION SCHEME

```
┌─────────────────┐  ┌─────────────┐  ┌──────────────┐  ┌──────────────┐  ┌──────────────────┐
│  ZR BLADES      │  │  IN SPRINGS │  │GUIDE THIMBLES│  │GUIDE THIMBLES│  │ TOP & BOTTOM     │
│                 │  │             │  │SMALL HARDWARE│  │   TUBES      │  │NOZZLES COMPON.   │
│ . VISUAL        │  │ . VISUAL    │  │ . VISUAL     │  │ . MATERIAL   │  │ . MATERIALS      │
│ . CUTTING DIR.  │  │ . CORROSION │  │ . CORROSION  │  │   STRUCTURE  │  │ . VISUAL         │
│ . CORROSION     │  │ . DIMENSIONS│  │ . DIMENSIONS │  │ . VISUAL     │  │ . CORROSION      │
│ . DIMENSIONS    │  │ . SPRING LOAD│ │              │  │ . CORROSION  │  │ . SPRING LOAD    │
│ . MIX'G VANES   │  └─────────────┘  └──────────────┘  │ . DIMENSIONS │  │ . DIMENSIONS     │
│   ORIENTATION   │                                     └──────────────┘  └──────────────────┘
└─────────────────┘
```

ZR BLADES
- VISUAL
- CUTTING DIR.
- CORROSION
- DIMENSIONS
- MIX'G VANES ORIENTATION

IN SPRINGS
- VISUAL
- CORROSION
- DIMENSIONS
- SPRING LOAD

GUIDE THIMBLES SMALL HARDWARE
- VISUAL
- CORROSION
- DIMENSIONS

GUIDE THIMBLES TUBES
- MATERIAL STRUCTURE
- VISUAL
- CORROSION
- DIMENSIONS

TOP & BOTTOM NOZZLES COMPON.
- MATERIALS
- VISUAL
- CORROSION
- SPRING LOAD
- DIMENSIONS

FUEL RODS
- WELD'G PROCESS
- PRESSURIZING PROCESS
- WELD X-RAY ...
- WELD DIAMETER
- ROD LENGTH
- PLENUM LENGTH
- FUEL WEIGHT
- FUEL ENRICH.
- LEAK TEST
- CONTAMINATION
- VISUAL
- WELD SAMPLE DESTR.

SPACER ASSEMBLING
- WELD VISUAL
- WELD DIMENSIONS
- RCC LATTICE
- ROD LATTICE
- OVERALL DIMENS.
- WELD SAMPLE DESTR.

GUIDE THIMBLE ASS'G
- WELD VISUAL
- WELD DIMENSIONS
- DIMENSIONS
- WELD SAMPLE DESTR.

TOP & BOTTOM NOZZLES ASS'G
- VISUAL
- DIMENSIONS
- WELD VISUAL
- WELD SAMPLE DESTR.

SKELETON
- COMPONENT ORIENTATION
- DIMENSIONS
- WELD VISUAL
- WELD SAMPLE DESTR.
- VISUAL

FUEL ASSEMBLY
- ROD VERIFICATION
- LOADING SEQUENCE
- DIMENSIONS
- VISUAL
- CONTAMINATION

FIG.2 – TYPICAL FUEL ASSEMBLY TEST AND INSPECTION SUMMARY

Fig.3 - ON-SITE FUEL HANDLING, STORAGE & SURVEILLANCE
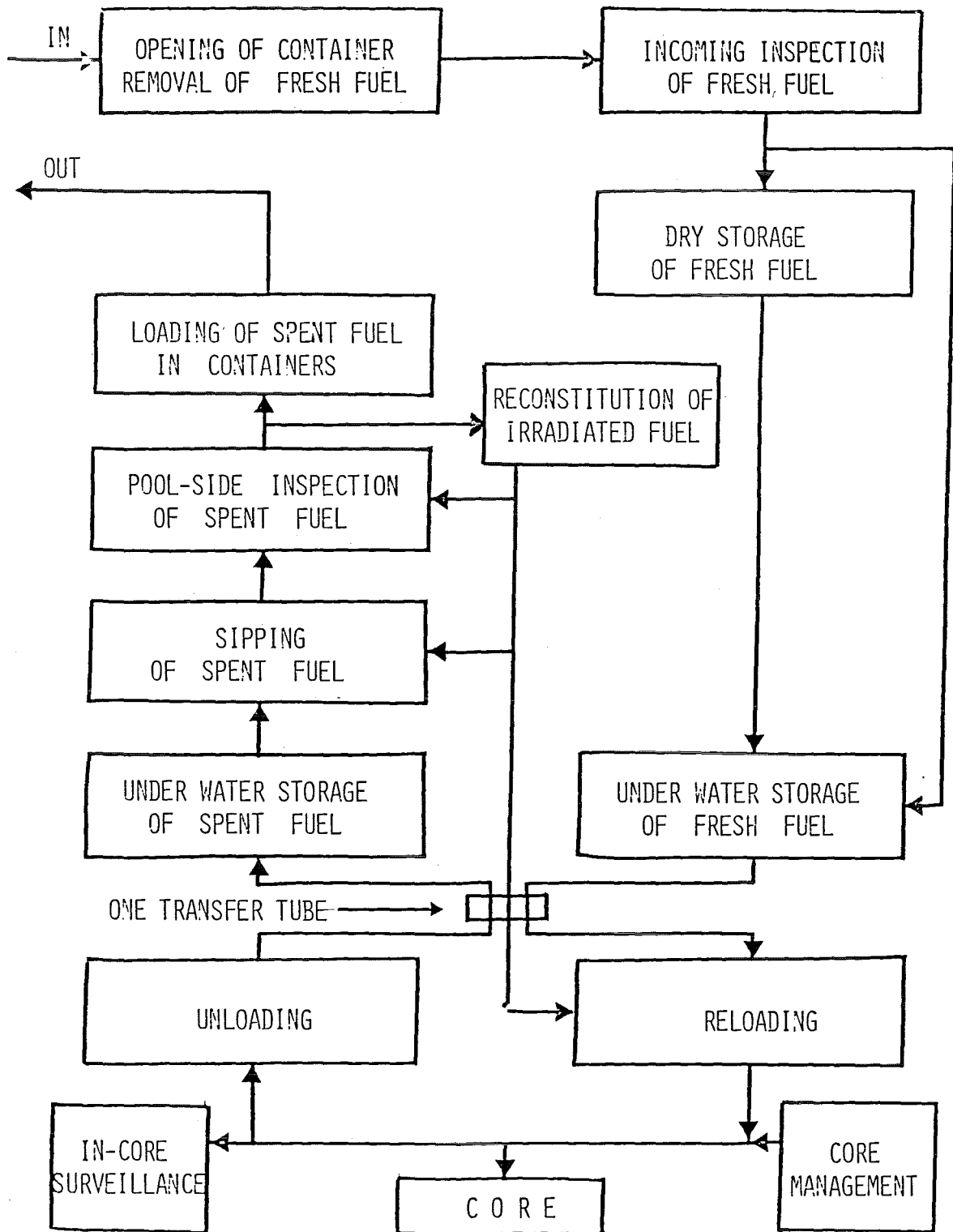
FIG.4 - INCOMING INSPECTION

. INCOMING RECEPTION OF SHIPPING CONTAINERS
  - HORIZONTAL
  - ACCELEROMETERS

. REMOVAL OF FRESH FUEL ASSEMBLIES :
  - VERTICAL POSITION USING STRONG BACK
  - VISUAL INSPECTION
  - HEALTH PHYSICS MEASUREMENTS : NO SURFACE
    CONTAMINATION.

DISCUSSION

Y. BARBIER: You presented a graph showing the evolution of
fuel central temperature with helium internal pressure. There is
a minimum temperature at about 1 MPa. Do you have a physical explana-
tion to this phenomenon?

H.F. BAIRIOT: The graph represents the peak conditions, which
do not occur in the same lifetime for the various prepressurization
levels. At low He pressures, the clad collapses against the fuel quite
early, before peak rating conditions are occurring (typically BOC 2
in a PWR); in this range, the peak fuel temperature decreases with
increasing He pressure as a result of reducing the effect of He dilution.
At high He pressures, the gap remains open; the higher the pressure is,
the longer the gap remains open and the more it becomes contaminated
with fission gases, resulting in an increased peak fuel temperature
during design basis operational overpower transients (most affected
during cycle 3).

ST. CARLSON: a) Do you use cold or hot He - leak tests and if
hot at what temperature?
b) What type of rod scanner do you use?
c) What's the detection limit on the individual pellet level in this
   machine?

H.F. BAIRIOT: a) Based on previous experience only cold He leak
testing is utilized.
b) Our rod scanner is able to work in the active and passive modes,
depending on the type of fuel and on the specifications. For LWR fuel,
only passive γ-scanning is utilized.
c) For routine application, the sensitivity is typically set to detect
a pellet varying by 10% from the nominal Pu content. If necessary, this
sensitivity can be increased by increasing the counting time.

R.S. RUSTAGI: On one of your slides you showed the prospect of
improvement in fuel burnups during the next 6 years by as much as 25%.
Could you please indicate the broad areas of design changes proposed
for fuel in the comming years?

H.F. BAIRIOT: Based on the results of ongoing demonstration programs, no major fuel design changes are required to obtain the contemplated burnups. Minor adoptions (such as an increase of the plenum length) will be implemented to meet the design criteria.

EVALUATION OF THE SEMINAR

H. Bairiot

BELGONUCLEAIRE

Europalaan 20, B-2480 Dessel

1. SUMMARY OF THE MEETING

1.1. Session 1 : Official rules in practice for QA in different countries.

Chairman : D. Vollath

The first technical session was related to the IAEA activities in the fields of QA & QC [50] and the underlyning principles of the application of QA and QC in USA [51], Brazil [2] and USSR [59]. From those papers and other contributions to the meeting [e.g. 13, 5, 60], apears that most regulations are based on the IAEA rules and recommendations for QA. It was realized that they coincide in many points with the regulations developed in the USA.

In most countries, the organization of QA and QC and its implementation are audited by gouvernmental organizations and/or by independent official organizations.

In all these cases QA is controlled by a regulatory body whos aim it is to
- establish regulations,
- license procedures appicable to the product and,
- audit and verify the implementation of the QA-system.

In one paper [59], examples of sophisticated equipment for QC were given, e.g. instruments for controlling canning, pellets, weld seam and for high temperature leak testing.

Refering to economic considerations [51] :
- nothing is more expensive than inadequate quality,
- competition in improvement of fuel quality is the most effective fuel vendor motivation for proper QA,
- to maintain competitive fabrication prices, unnecessary controls must be identified and cancelled.

NB : [*] refers to the paper in the official numbering system, i.e. IAEA-SR-102/*

1.2. Session 2 : - Official rules in practice for QA in different coun-
tries (cont'd)
- QA system implemented in different production plants

Chairman : V. Borsky

The question of controlling and assuring the quality in the nuclear
industry has been looked after for more than 10 years in many countries.
The history of the establishment of the whole system of Quality Control
shows a remarquable evolution from a rather general view of things to a
more specific definition of the required documentation in quite a number
of countries.

Although Finland [3] does not produce the fuel, a very responsive
system for controlling the fuel that they receive was developed, including
destructive control of a full assembly if auditing was not possible at the
manufacturing plant.

The documents which establish the interaction between the designers
and the fabricants of fuel in the Federal Republic of Germany [11] is
based on a documentation system developed for the group and on the 10 CFR
50 from the USA.

The achievements made in France [13] are examplified by the central-
ized data bank in which is stored the entire history of the production of
fuel elements and the connection or link-up of all the plants; it helps in
determining the causes of fuel failure at any power plant.

A USSR paper [17] presented an interesting approach to quality rank-
ing, i.e. to divide up in three groups those parameters that relate to the
fuel element attributres; in the first group no great deviation from spe-
cified values are tolerated; in the last group, more deviation is accepted
All this is in fact based on the experiments carried out in respect of the
various attributes (e.g. grain size of the fuel).

Success has been achieved by Indian scientists [5] in developing
their own national programme for the fabrication of fuel elements and or-
ganizing quality control.

1.3. Session 3 : Quality Assurance System implemented in different production plants

Chairman : R. Holzer

The contributions to Session 3 can be summarized by the following statements :

There is an increased use of analytical processing of the QC data by basic statistical methods, e.g. many papers in this seminar [57, 9, 30] presented histograms, cumulative frequency diagrams, confidence intervals, etc... Statistical evaluation is not only applied to the assessment of product quality, but it is recognized to be a tool for product control, process control, and control of testing methods itself [21].

Quality control is integrated into the sequence of manufacturing steps. Statistical evaluation can directly be used to control process parameters, e.g. the annealing temperatures during the manufacturing of Zircaloy tubes [6].

Quality Control implies a common language for designer and producer. It helps to understand the technology of fuel manufacturing and to improve product properties [4, 14]. A classification of quality characteristics according to their relative importance to the end use could help to improve appropriate sampling plants [4]. The classification may be based upon the levels : critical – major – medium – minor.

Quality Control should not only be applied for the standard production of fuel assemblies but also for the qualification of advanced products and processes, e.g. for the manufacturing of UO_2 pellets with additives and Zry-liner tubes [6].

Because of the large number of QC data, computerized evaluation and computerized data banks are necessary [57]. The computerized recording systems give the chance to trace back eventual fuel failures to respective batches of material and occurances during the production [57]. But during the discussion, cost objections have been raised with respect to a complete traceability. This aspect should further be discussed.

1.4. Session 4 : Quality Assurance system implemented in different production plants (cont'd)

Chairman : K. Balaramamoorthy

Papers were presented by France [12], Italy [22], Sweden [21], and Fed. Republic of Germany [20] detailing the QC practices and also QA Systems. All of them fulfill the well known 18 points criteria and/or the requirements and recommendations laid down in IAEA Code of Practice on QA and in Safety Guide QA-11. Procedures and Systems are tailored to meet the large scale production of high quality fuel assemblies. In this respect the tendency is to develop and to implement computer aided systems for systematically documenting the data and for eliminating possible human errors. The consequences of defects that might be missed in fuel rods (final stage of pellets, final sheath inspection, closure weld quality, moisture control, etc.) has justified implementation of improved methods and refinement of conventional production and inspection techniques.

1.5. Session 5 : Physical methods for analyses.

Chairman : A.A. Strasser

Four papers were presented on fuel powder and pellet inspection methods, and one paper on the development of an inspection and test plan.

A comprehensive paper [54] discussed various methods of characterizing fuel powder and pellets : currently used techniques, advanced methods of analyses for characteristics that have not been measured satisfactorily previously and work methods for current techniques. Most measurements are computerized and readily adaptable to a process plant.

Paper [18] described a new non-destructive method of measureing the U-235 enrichment directly in UF_6 containers with a Ge detector. While the method does not have the accuracy of the main spectrometric analysis of a sample, it does have the advantage of providing a nominal reading rapidly.

Paper [26] discussed a new test method and gave examples of the use of films sensitive to neutron irradiations. Measurements of the distribution of fissionable and absorber materials can be made by exposing the materials, covered by a polymer, to a neutron source.

Paper [23] presented improved methods for processing and inspecting for plutonium homogeneity in mixed oxide pellets. Inspection techniques used were Pu potentiometry, gamma-spectroscopy, alpha-autoradiography and rod gamma-scanning. Their relationship to economics was discussed.

Paper [15] discussed the proper method for the development of an inspection and test plan for fuel assemblies. The most important points related to the determination of the acceptable quality level and the choice of AQL level and its effect on fuel quality evaluation.

1.6. Session 6 : Physical methods for analyses (cont'd)

Chairman : G. Dressler

The majority of the papers dealt with the production and subsequent inspection of Zircaloy products, specifically cladding tubes [53, 7]. The main efforts undertaken during the past years focussed on increasing the process reliability by well balanced and tailored process controls (with the useful application of statistical methods), and the automatization of process, inspection and subsequent documentation . No new developments, for example in the field of improving or optimizing material properties were reported. This seems to be a general trend in the fuel manufacturing area, due to a lack of severe performance problems because of quality deficiencies. Under this aspect, it might be useful to not only think about rationalizing, improving and modernizing the existing QC equipment and techniques, but also about reducing the QC efforts on the basis of our learning curve which is described by a constant decrease of quality related fuel failures in all countries and for all suppliers.

It seems to be worth to carefully judge the applied QC techniques for complex parts like spacers [10] and discuss the questions of substituting expensive techniques (which locate variables data and many kilograms of paper) by simple attributive measurements with functional gaging as outlined in one of the papers [56].

1.7. Session 7 : Methods of Chemical Analyses, Training of Personnel and Update of fuel characteristics

Chairman : M. Ernotte

From an interesting review of the various analytical techniques applied in the quality control of UO_2 powder and pellets [58],a disturbing conclusion concerned the usefulness of the specification of the 27 trace elements.

Paper [24] presented procedures for chemical analysis of Zry and paper [19] described a device enabling to measure the inner pressure of rods and to sample and analyze (by gas chromatography) the filling gas in fuel rods.

Paper [52] exposed interesting considerations about QA/QC personnel training and qualification in India, linking the manpower requirements in these areas to the extent and the schedule of the intended nuclear power programme, with due consideration to the constraints imposed by the local possibilities. The training opportunities were addressing to personnel of all levels, ranging from post-university courses to familiarisation lectures and in-service or on-the job training programmes related to operation production and inspections aspects.

A Czechoslovakian paper [16] described the evaluations and verifications performed during and after the fabrication of UO_2 pellets for experimental fuel assemblies for test reactors.

A paper [60] summarized the Belgian situation which has the peculiarity to involve many of the world fuel manufacturers. Since the present trends in reactor performance concern the increased discharge burnup and fuel manoeuverability, nuclear fuel is a product which has a very long feed back time and this requires reliable engineering tools to evaluate quality related parameters. International research programmes are efficient in improving our knowledge of the basic phenomenons influencing in reactor fuel behaviour.
Some considerations were presented on the effects of MOX fabrication route on fuel dissolution, pellet hydrogen content, fuel densification, rod pressurization and the need for process controls on these important characteristics. Fuel surveillance after fabrication and during reactor operation was also stressed. The general conclusion presented was that simpler design and requirements could probably be achieved by a better knowledge of the relation between fuel assembly quality and irradiation behaviour.

1.8. Panel discussion

Chairman : H. Bairiot

A quality confidence level determined by quality control is usually limited to 99/99 (or even 95/95 for many attributes). It is insufficient by itself to assure a fuel quality characterized by the presently observed in service failure level of 10^{-5} rods per year. In this respect it is worth reconsidering the significance and the necessity of some tests like the helium leak check and gamma scanning of fuel rods, in the light of a consistent product quality with zero rejects over many years. Everybody still performs ASTM corrosion tests on Zircaloy samples and knows very well that the test is useless because of the fact that the critical part of the sample is removed by etching prior to testing; a panel member could not remember any rejected samples during its 15 years experience in this field.

Despite the increase of manufacturing experience and product reliability and quality, the associated audit activities show a drastic increase, mainly to satisfy requirements which were established many years ago under a quite different quality environment and maintained only for reasons of marketing strategies. Open discussion should be encouraged about the necessity of some established tests and inspection in the light of reducing the overall fabrication costs

The most sophisticated QC techniques have evolved from automated ("hands-off") to completely computerized ("brain-off") systems.

Many papers have referred to the IAEA guidelines and guidebooks on the quality of nuclear fuel. The Agency should be encouraged to update and complete these documents.

An inconclusive discussion took place on the difference of QC level observed at the fabrication plant site and the reactor site.

In most fabrication plants, the implementation of a stringent QA program has so much enhanced the quality of the final product that, at several inspection stages, the rejection rate has fallen to zero; QA procedures have evidently paid for themselves. The time might have come to review QC plans and replace those tests and analyses which never reveal negative results by a QA procedure ensuring that the work was performed in strict conformity with the qualified flowsheet.

2. CONCLUSIONS

2.1. The Seminar reviewed successively the three levels of QA/QC activities involved in the fuel fabrication :
- the establishement, development and implementation of a QA system and a QA program.
- the planning and application of QC for product and process control, including the statistical background and evaluation.
- the development, introduction and assessment of the testing procedures (both physical and chemical).

2.2. More than 80% of the presentations were useful.

2.3. General considerations on QC philosophy, regulations, principles and general production flow sheets with QC hold points have been covered adequately in this and past meetings, and need not to be repeated. It did not reveal new features.

2.4. Specific descriptions of methodologies for process control, inspection techniques and associated experience were and should continue to be presented. Some statistical evaluation techniques were discussed; this aspect deserves more emphasis.

2.5. The seminar was in line with the excellent meetings organized previously (1978 and 1981) by KfK on QC and characterization of fuel pellets; but the scope was extended to quality related features of as-fabricated assemblies, i.e. including cladding, fuel rods, assembly hardware, etc. No paper, but one, considered fuel quality thereafter (shipping, on-site handling, storage).

3. RECOMMENDATIONS

3.1. This Seminar was successful in presenting and confronting the practices adopted in the various countries. A future meeting (at a 3-year interval) would be mutually beneficial in updating the specifications and practices. It should be devoted exclusively to QC techniques, associated experience and feedback to the fabrication processes. To avoid duplication, it could again be organized as a cooperative undertaking between the periodic Karlsruhe meetings on Characterization of Nuclear Fuels and an IAEA Specialists' Meeting.

3.2. It should incorporate inspection of the fuel at the reactor site and discuss consistency with the fabrication plant practices, the transport operations and the storage methods.

3.3. A critical review on the relevance of some quality requirements should be encouraged. Cost/benefit considerations should be part of the criteria.

3.4. Developing countries indicated a need for further technical education and training in the field of QC methods.

3.5. The IAEA should pursue its activities in elaborating guidelines and guidebooks on QA and QC of nuclear fuel. A timely updating and revision of the Guidebook on QC of Water Reactor Fuel should be considered.

IAEA-SR-102/59

SEMINAR ON PRACTICAL EXPERIENCE IN THE APPLICATION
OF QUALITY CONTROL IN WATER-REACTOR FUEL FABRICATION

12-16 March 1984
Karlsruhe, Federal Republic of Germany

Principles Governing the Organization of Quality
Control in the Fabrication of Water Reactor Fuel

Elements

V. Gorsky

All-Union Scientific Research Institute of
Inorganic Materials, Moscow, USSR

Abstract

In the USSR nuclear power developments are based on water
cooled and moderated (VVER) and boiling graphite/water (RBMK)
reactors. The modern commercial fabrication of fuel elements and
assemblies is impossible without the complex system of the product-
ion quality management, which covers all problems of the quality
management in the production of structural and fissile materials,
semi-finished products, and as-fabricated fuel elements and
assemblies. The quality management of the fuel elements and
assemblies is carried out beginning with the branch of industry
and finishing with a shop, manufacturing structural materials.

The transition to more powerful reactors and the resultant
substantional growth of the fuel manufacture have required the

development of more elaborate and efficient means of control, suitable for the application in the automated mass line production. More rigid requirements on fuel elements and some of their parameters made it imperative to increase in some cases the measuring accuracy and to introduce a 100% control.

To control the quality of the structural components and fuel elements several new installations have been designed:

(1) complex tube inspection system for ultrasonic flaw detection and dimension determination

(2) ultrasonic bench for flaw detection in zirconium alloy rods

(3) high-efficiency radiometric devices for pellet density determination

(4) semi-automatic ultrasonic bench for the flaw detection in the welds of fuel elements

(5) high-efficiency radiometric device for inter-pellet gaps detection in a fuel stack

(6) leak testers with heated vacuum inspection chambers for the leak-testing of fuel elements.

------- 0 -- 0 -- 0 -------

In the USSR nuclear power developments are based on water cooled and moderated (VVER) and boiling graphite/water (RBMK) reactors. The successful operation of the reactors of the first generation has shown not only the high reliability of the fuel element and assembly designs but also the effectuality of the complex system of management of the production quality and applied methods and technical means of control and evaluation.

Started several years ago the transition to the construction of reactors with increased unit power (as, for example, unit 5 of the Novovoronezh nuclear power plant with the VVER-1000 reactor) and the resultant fundamental growth in the production of fuel elements and assemblies required the development of more sophisti-

cated and efficient means of control, suitable for automated mass line production.

Furthermore, requirements become more stringent for the measurement precision of some fuel element parameters [1]. The structure of the quality management in the fuel element and assembly production as well as some control means developed during the last decade are described below.

1. Complex Quality Management System

1.1. Organizational Structure

The complex quality management system (CQMS) has been functioning in the manufacture of fuels for nuclear power plants for many years. It covers all the problems of the quality management in the production of structural and fissile materials, semi-finished products and as-fabricated fuel elements and assemblies.

The branch quality management of the fuel element and assembly production is realized through the main administration, scientific-technical administration of industry, leading scientific research institutes, pilot design and project planning organizations.

The plant quality management of the fuel element and assembly production is implemented by chief engineers, technologists, designers, instrumentalists and their subordinated services.

The functions of the state control are fulfilled by the quality inspectors of the State Committee on Supervision for Security of Work Conducted at Nuclear Power Plants of the USSR.

1.2. CQMS at Plant

The organizational and technical activities and means of management of quality of fuel element and assembly production are fulfilled within the scope of the CQMS, which is a whole complex

of engineering, technical, technological, control, inspection and other processes. The CQMS establishes and regulates the functions of the plant main services and their management influence on the assurance of the specified quality. It realizes the following functions:

(1) prediction of the product technical level and quality

(2) planning of the product quality upgrading,

(3) working out and preparation of the product organization (a design assurance of the quality)

(4) technological preparation of the production (a technological assurance of the quality)

(5) material and technical assurance of the quality

(6) metrological assurance of the quality

(7) selection, arrangement, provision and training of the personnel (a qualificational assurance of the quality)

(8) provision of the stable high-quality production at each step of the manufacturing process

(9) technical quality control

(10) state supevision of the introduction and observance of the standards, technical specifications and the condition of measuring means at the plant.

The CQMS activity is regulated by the state, branch and plant standards.

1.3. Subsystem of CQMS in Plant Shops

The subsystem of the quality management of the fuel element and assembly structural components manufactured on automated and mechanized lines in shops incorporates the following management functions:

(1) control of incoming raw materianls, structural materials, components, and semi-products

(2) control of incoming measuring equipment and other technological devices

(3) adjustments and corrections of technological operations in accordance with the specified operational technical requirements and tolerances

(4) operator's self-control and an application of the active control instruments during the product manufacture process

(5) statistical adjustment of the technological process

(6) operation control

(7) internal inspection of the manufactured products

(8) control before delivery

(9) external inspection of manufactured products

(10) metrological provision of production lines and the forced calibration of measuring equipment

(11) periodic check-up of production equipment with respect to the technological precision

(12) periodic check-up of the technological process maintenance and culture of serving equipment

(13) systematic analysis of defect products (kind of a defect, reasons of its appearance and who is to blame and discussion of the problem together with the plant administration to take urgent measures to remove the causes of the reappearance of such defects).

1.4. Provision of Production with Measuring Equipment

Methods and technical means of the quality control of structural materials, semi-products, fuel elements, assemblies, and control instruments for technological processes and production equipment are developed by plant laboratories of the automation and nondestructive testing. The plant laboratory of the automation and mechanization conducts a whole cycle of operations from

the development or mastering the instruments and their testing under shop conditions to introduction of the measuring equipment and personnel. The measuring equipment in shops is attended by shop's services of the controlling and measuring instruments and automation.

1.5. Technical Control Departament

The technical control departament (TCD) at the plant is most important in CQMS. Legally the TCD is independent on a production administration. The chief controller is subordinated to the plant manager. Using effective methods the TCD controls the quality of incoming structoral materials between operations and before delivery of semi-finished products; withdraws from production the nonconforming materials, structural components, fuel elements and assemblies, carries out examinations and technical analyses of the rejects and defects as to the types, causes and guilty persons, systematically holds conferences with the administration to consider the nonconforming materials and imperfect products and work out managemental influences to remove the causes, leading to rejects and defects.

The TCD has the exceptional right of deciding on the fitness or unfitness of the manufactured items. The TCD draws up a document on accepted products and writes a notification about rejected items (or an act on revealed spoilage).

1.6. Quality Management Service

It coordinates the plant's subdivisions activities in the realization of their functions, carries out or organizes an engineering analysis of the accumulated information on a product quality and the causes of defect appearance, prepares for the plant manager the project management decisions aimed at improving the product quality and removing defects and spoilages in as-fabricated products, that cause claims and reclamations from consumers.

1.7. State Quality Inspection

Inspectors of the State Committee on Supervision for
Security of Work Conducted at Nuclear Power Plants of the USSR
control semi-finished products, structural components, as-fabri-
cated fuel elements and assemblies and conduct periodic evaluat-
ions of technological operations at plants as to their conformity
to the requirements of the standards and technical specifications.

Control operations are carried out with the measuring instru-
ments that are used by the TCD of the plant.

2. Technical Control Equipment

2.1. Controlled Parameters

The list of controlled parameters, methods and extent of
control are set by the technical specifications on fuel elements
and assemblies. As far as the fuel elements and assemblies for
VVER and RBMK nuclear reactors are concerned, the list of the
controlled parameters of the incoming structural and fissile ma-
terials, semi-finished products, as-fabricated fuel elements and
assemblies coinsides with the recommendations and requirements of
the IAEA [2,3]. Some distinctions are only in the control methods.

2.2. Metrological Provision

The CQMS of the plant incorporates a whole complex of control
operations conducted by the destructive and nondestructive evaluat-
ions and tests. The metrological provision of the quality control
is accomplished by the chief metrologist service. It takes part
in the development and introduction of control methods and means
and inspects the condition of the technical control means applied.

The analytical control and tests of materials and items are
carried out by plant laboratories in accordance with the state,

branch and plant standards or approved instructions.

The nondestructive evaluation of structural materials and finished items is conducted on equipment permitted for use by the state check-up service. The equipment sensitivity is adjusted by means of the branch and plant standard specimens or approved test-specimens or product control specimens.

2.3. Quality Control of Cladding Tubes

Zirconium alloy (Zr+1 wt.% Nb) tubes are controlled in lots. A tube lot consists of tubes of the same fabrication process and final heat treatment. The approval of a tube lot is realized by the TCD based on the results of control of all the parameters indicated in the specifications. The control and tests of tubes are carried out using generally adopted methods, as, for example, given in the standard ASTM B 353-77a.

Recently to increase the control speed much work has been done to develop a complex inspection system for the ultrasonic flaw inspection and geometric dimension measurements of thin-walled tubes. The design of such a system for an ultrasonic inspection of tubes 5-20 mm dia. and 0.3-1.0 mm of wall thickness is described in [4] . The tube to be controlled moves in the axial direction through two acoustic blocks with rotating ultrasonic transducers. The first one is intended for flaw inspection of tubes. The tube wall is sounded across and along its axis in opposite directions at a frequency of 5 MHz with two pairs of transducers in accordance with the "classic" scheme. The distinctive peculiarity of this block as compared, e.g. with the similar block in the ultrasonic inspection system ROTA 40/6000, produced by the firm NUKEM (FRG) is an application of plain bearings lubricated with water serving as an acoustic coupling medium in the flaw inspection block. The rotation speed is 6000 rev./min.

The second acoustic block serves to measure the wall thickness on two diametrically-opposite sides of a tube and outer diameter of a tube. The outer diameter of a tube is measured with two rotating ultrasonic transducers from acoustic pulses transmitted normally to the tube surface. It is proportional to the travelling time of the acoustic pulses from tranducers to both sides of the outer surface of the tube.

The wall-thickness is measured by two ultrasonic thickness-meters "Metal-6M" on the diametrically-opposite sides of a tube. The operation of this instrument is founded on the measurement of the resonsnce vibration cycle of the wall after an excitation by two acoustic pulses during the outer diameter measurement.

The inner diameter is calculated as the difference between the outer diameter and the two wall thickness. The accuracy of the measurement of the outer diameter and wall thickness is < 5 μ m, that of the inner diameter calculation is < 10 μ m. To decrease the influence of water temperature changes on the accuracy of the outer diameter measurement there is an additional transducer in this block generating the base signal.

The control results are recorded on a diagram paper. There are digital indicators of the tube geomentric sizes in the measuring instrument.

2.4. Quality Control of Rods

Zirconium alloy (Zr+1 wt.% Nb) rods are controlled in lots. A rod lot consists of rods of the same fabrication process and final heat treatment. The approval of a rod lot is realized by the TCD based on the results of control of all the parameters indicated in the specifications. The control and tests of the rods are carried out using universally adopted methods.as, for example, given in the standard ASTM B 351-79. The rod metal soundness is controlled

by an ultrasonic method. Every rod is subjected to an ultrasonic control on a bench having two transducers. The rod moves rotating through a local bath filled with water. The first transducer (combined, having separate transmitting and receiving crystals) serves to evaluate transverce defects and defects along the rod axis. The second transduser is intended to reveal longitudinal surface and subsurface defects (frequency of 5 MHz). The sensitivity is adjusted using the standard calibration rod with artificial defects - the longitudinal and transverse marks 0.2 mm in depth and 1.5 mm in length at the outer surface and holes 1.5 mm in diameter, drilled along the axis and radius of the rod and having a depth equal to the rod radius.

2.5. Quality Control of Pellets

UO_2 pellets of different enrichment are controlled in lots manufactured from similar initial powder. The approval of a pellet lot is realized by the TCD based on the results of control of all the parameters indicated in the specifications. The quality control of powder and pellets is carried out by the universally adopted methods as, for example, given in the standards ASTM C 696-80 and C 776-79. To reduce the water content of pellets to below 3-4 ppm the lower density limit of 10.4 g/cm^3 was set [1]. This required a more efficient equipment to be designed for the density control of green and sintered pellets. The well known γ-absorption method similar to that described in [6] is used in these controlling systems. The γ-ray source is ^{137}Cs. There is a compensation device to exclude the influence of sintered pellet diameter variations on the measuring accuracy during the density control by means of a collimated beam of γ-rays penetrating a pellet along its chord. By applying the multiposition principle of measurement (with the channels placed around a γ-ray source) and high-speed

electronics it was possible to increase the control speed to
5000 pellets per hour. The root-meansquare is ~ 0.6% (2σ). The
outer diameter of ground pellets is controlled on a rotating cross-
axial shaft device with a variable gap between shafts. This permits
the division of the controlled pellets into three groups: waste
pellets, normal pellets and pellets having an outer diameter above
the upper limit. To decrease rejects during pellet grinding on
centerless grinding tools the outer diameter of pellets is measured
on statistical samples. This permits the time regulation of the
grinding instrument position.

2.6. Quality Control of End-Plug Welds

In accordance with the CQMS requirements the quality of fuel
element end-plug welds is controlled by various methods at diffe-
rent stages of the fuel element production. The primary attention
is paid to a welding equipment because the stability of welds de-
pends on the welding equipment state. The electron-beam welding is
performed with   welding machines equipped
with automatic television systems for directing the electron beam
to joints being welded. Machines of this kind made it possible not
to weld "production test samples before manufacture" for destructi-
ve testing. At present the samples are welded only after the repla-
cement of a cathode in an electron- beam gun to control the stabi-
lity of welding parameters. The quality of the welded samples is
determined metallographically. The geometric sizes, outer surface
condition, corrosion resistance and leak tightness of fuel element
welds are subjected to control.

However, the most important method of the weld quality control
is radiography. The radiographic technique of fuel element weld
control is well known. The description of the equipment used and
an apparatus for developing films is given in [7]. The welds must

be free from any internal defects and pores 0.2 mm in diameter.

The disadvantages of radiographic examination are its high cost, low capacity and the impossibility of automating the process of radiographic film interpretaton. In this respect an ultrasonic method is very attractive. Semiautomatic equipment has been designed for an ultrasonic examination of electron- beam welds of VVER and RMBK fuel elements. Ultrasonic vibrations are introduced directly both into the weld seam and the fuel can side [8,9]. The operation experience has shown that compared to radiography this ultrasonic equipment permits a better detection of lack of fusion in the weld root more than 0.1 mm. But the reliability of pore detection is worse than in radiographic examinations. During ultrasonic examination less than 80% of pores 0.25-0.3 mm in dia. are detected. This is explained by the outer surface roughness of weld seams, shape and location of pores in welds, surface roughness of fit surfaces, irregular contacts between an inner surface of a can and an outer surface of an end-plug near the weld root due to the softening of canning metal during welding and so on. All this produces a high level of noises and false signals. The results of our ivestigation correlate with the results, described in [9]. From our point of view the modern condition of the ultrasonic technique does not permit the radiographic examination to be fully rejected as a method of evaluation of fuel element welds.

2.7. Quality Control of Fuel Column Continuity

An increase of the fuel element power density and a coolant pressure in the core of high power reactors (e.g. VVER-1000) required a more precise control of inter-pellet gaps in a fuel column, which must not be more than 3-5 mm [1]. An automatic γ -absorption facility has been designed to control inter-pellet gaps, a plenum length, the position of a fuel column holders, the sum of

inter-pellet gaps in the fuel column and to detect fuel crumbs in a plenum. The collimated (0.4x7.0 mm) beam of a γ-radiation source of ^{57}Co (20 m Ci) and a detector are intended to determine the beginning and end of a gap. The gap extent is measured from the number of pulses produced by the controlled frequency generator, that is operated by that γ-relay. In the course of its passage the inspected fuel element rotates a measuring roller of the pulse generator, that generates 1 pulse per each 0.1 mm of the fuel element passage. The facility measures the plenum lengh, the gap between the bottom end-plug and the first pellet, the maximum inter-pellet gap in the fuel column, the sum of all inter-pellet gaps in a fuel column, the presence of fuel crumbs in a plenum and the position of the fuel column holders. Small end chips of pellets are registered as an equivalent increase of an inter-pellet gap. The control results are processed by a special programme and typed. The test results of the VVER-1000 fuel elements showed that errors in measuring the inter-pellet gaps do not exceed ±0.3 mm (2σ), those for the plenum length are ±2.0 mm and those for the total inter-pellet gap in a fuel column are ±1.5 mm. The control speed is 400 m/hour.

Fuel elements condemned as defective are radiographed or investigated on the screen of an X-ray-television set to find out the reasons of the detected deviations from the tolerance in an as-fabricated fuel element.

2.7. Leak-Testing of Fuel Elements and Assemblies

Fuel elements and assemblies are leak-tested by helium mass spectrometry using a vacuum chamber method. The leak-testing technique, the means of the static and dynamic sensitivity determination of testing facilities, the rules of the preparation of fuel elements for leak-testing and so on do not differ from the gene-

rally adopted practice.

Fuel elements are leak-tested in lots for the presence of respectively "large" and "small" leaks. Final leak tests for the presence of small leaks are usually conducted by heating them in a vacuum chamber to 300-400°C in a leak-testing facility or directly after vacuum heating in an oven with their subsequent transfer to the vacuum chamber of the leak-testing facility.

Fuel elements to be evaluated for the availability of comparatively "large" leaks are subjected to helium pressurization. The leak-testing is performed at room temperature. In case of the evaluation of several fuel elements the rejection level is the very same as for a single fuel element.

As-fabricated assemblies are subjected to vacuum heating prior to leak-testing at room temperature.

References

1. Reshetnicov F. e.a. Status and Development Prospects for the Fabrication and Utilization of Nuclear Fuel for VVER-Type Reactors. Power Experience (Proc. Intern. Conf., Vienna, 13-17 Sept. 1982), v.3, IAEA, Vienna, 1983, p.621.

2. Guidebook on Quality Control of Water Reactor Fuel. Tech. Rep. Series N 221, IAEA, Vienna, 1983.

3. Quality Assurance in the Procurement, Design and Manufacture of Nuclear Fuel Assemblies. A Safety Guide. Safety Series N 50-SG-QA11, IAEA, Vienna, 1983.

4. Веремеенко С.В., Горский В.В. и др. Автоматизированная ультразвуковая установка Атлант-I для комплексного контроля труб малого диаметра. Дефектоскопия, 1984, № I, с.55.

5. NUKEM GmbH. Rohrprüfanlagen.

6. Ayer I., Schmitt D. A Gamma-Ray Absorptiometer for Nuclear Fuel. Nuclear Technology, 1975, v.27, N 3, p.442.

7. Akopov V.S. e.a. Fuel Element Radiographic Inspection. The Development of Technique and Apparatus Complex. Tenth World Conf. on NDT, 22-28 Aug. 1982, Moscow, USSR, Rep.6-23.

8. Mattson I.C. EB Welding and QC of Nuclear Reactor Fuel Rods at ASEA-atom. 3 rd Intern. Colloq. on Welding and Melting by Electrons and Laser Beams. Lion, France, 5-9 Sep., 1983, v.2, p.757.

9. Rep. WAPD-TM-1228, 1979.

10. Kokhov E.D. e.a. Radiometric Non-Destructive Evaluation of Nuclear Materials Distribution in Fuel Elements. Tenth World Conf. on NDT, 22-28 Aug. 1982, Moscow, USSR, Rep.4-15.

CHAIRMEN OF SESSIONS

Session 1	D. VOLLATH	Federal Republic of Germany
Session 2	V. GORSKIJ	USSR
Session 3	R. HOLZER	Federal Republic of Germany
Session 4	K. BALARAMA MOORTHY	India
Session 5	A.A. STRASSER	United States of America
Session 6	G. DRESSLER	Federal Republic of Germany
Session 7	M. ERNOTTE	France
Session 8	H. BAIRIOT	Belgium
	M. Ernotte	France

SECRETARIAT

Scientific Secretary:	A.F. MANIN	Division of Nuclear Fuel Cycle IAEA
Scientific Adviser:	D. VOLLATH	Kernforschungszentrum Karlsruhe
Administrativ Secretary:	E. PILLER	Division of External Relations, IAEA
Liaison Officer:	G. HERRMANN	Federal Republic of Germany

LIST   OF   PARTICIPANTS

BELGIUM

Bairiot, M.H.   Belgonucléaire
                                       Europalaan 20,
                                       B-2480 Dessel

Deramaix, P.   Belgonucléaire
                                       25, Rue du Champs de Mars
                                       1050 Brussels

Frans, C.D.G.   Electrobel SA
                                       Place du Trone, 1
                                       B-1000 Brussels

Pauwels, G.   Franco-Belge de Fabrication
                                       de Combustibles
                                       12, Europalaan, B-2480 Dessel

BRAZIL

Pollis, R.   CNEN
                                       Rua General Severiano, 90
                                       Botafogo, Rio de Janeiro

CHINA

Xu, Shijiang   Institute of Nuclear Energy
                                       Technology
                                       Tsinghua University
                                       Peking

CZECHOSLOVAKIA

Vesely, P.   Insitute of Nuclear Fuels
                                       Uranová 379
                                       255 01 Praha 5 Zbraslav

DENMARK

Neven, Z.V.   Elkraft Power Company Ltd.
                                       Lautruphoej 5, PO Box 230
                                       DK-2750 Balleup

Toft, P.   Risø National Laboratory
                                       Postbox 49
                                       DK-4000 Roskilde

FINLAND

Granroth, M.K.   Imatran Voima Oy
                                        Mechanical Department
                                        PO Box 138, SF-00101 Helsinki 10

Lunabba, R.H.   Technical Research Center of
                                        Finland, Metals Laboratory
                                        Metallimiehenkuja 2,
                                        SF-02150 Espoo 15

Moisio, J.M.   Imatran Voima Oy
                                        PO Box 138, SF-00101 Helsinki 10

Ojanen, M.U.   Finnish Centre for Radiation and
                                        Nuclear Safety
                                        PO Box 268

Pääkönen, J.K.   Teollisuuden Voima Oy
                                        Industrial Power Company Ltd.
                                        SF-27160 Olkiluoto

Patrakka, E.   Tellosuuden Voima Oy
                                        Industrial Power Company Ltd.
                                        Fredrikinkatu 51-53 B
                                        SF-00101 Helsinki 10

FRANCE

Barbier, Y.   CFC Cogema et Framatome
                                        BP No. 3
                                        F-26701 Pierrelatte Cedex

Chaussabel, L.   CEA/CENG/LCAC
                                        F-38041 Grenoble

Ernotte, M.   Fragema
                                        149, Cours Lafayette
                                        F-69006 Lyon

Godard, H.   Electricite de France
                                        23 bis, Av. de Messine
                                        F-75384 Paris Cedex 08

Hochard, G.P.   Electricite de France
                                        SCF, Route des Centrales
                                        BP 143,
                                        F- 71104 Chalon sur Saone

Morin, C.

Franco-Belge de Fabrication de
Combustibles
F-26104 Romans

Pancaccini, G.

Fragema
149 Cours Lafayette
F-69006 Lyon

Pasco, Y.J.P.

Ministère de l'Industrie et de
la Recherche, SCSIN
99, rue de Grenelle
F-75007 Paris

Robin, J.P.

Fragema
149, Cours Lafayette,
F-69006 Lyon

Vial, N.M.M.

Electricite de France
23 bis, Av. de Messine
F-75384 Paris Cedex 08

GERMANY, FED. REP. OF

Assmann, H.

Kraftwerk Union AG
Postfach 32 20
Hammerbacher Strasse 12-14
D-8520 Erlangen

Buken, G.

NUKEM GmbH
Notronik Division
Röntgenstrasse 2
D-8755 Alzenau

Dams, W.

ALKEM GmbH
Postfach 11 00 69
D-6450 Hanau 11

Dressler, G.A.

EXXON Nuclear GmbH
Postfach 14 65
Industriepark Süd
D-4450 Lingen 1

Elbel, H.

Kernforschungszentrum Karlsruhe
Postfach 3640
D-7500 Karlsruhe

Engel, H.   Reaktor Brennelement Union GmbH
                                   Postfach 11 00 60
                                   D-6450 Hanau 11

Gruber, K.   ALKEM GmbH
                                   Postfach 11 00 69
                                   D-6450 Hanau 11

Hackl, H.R.   Siemens AG
                                   Otto Hahn Ring 6
                                   Postfach 83 09 53
                                   D-8000 München

Herrmann, W.E.   Siemens AG
                                   Otto Hahn Ring 6
                                   Postfach 83 09 53
                                   D-8000 München

Hofmann, P.   Kernforschungszentrum Karlsruhe
                                   Institut für Material- und
                                   Festkörperforschung
                                   Postfach 3640
                                   D-7500 Karlsruhe

Holzer, R.W.   Kraftwerk Union AG
                                   Hammerbacherstrasse 12+14
                                   D-8520 Erlangen

Kral, W.A.   Wissen und Wirtschaft
                                   Eschenweg 1
                                   D-6837 St. Leon-Rot 2

Kruschel, M.   NIS GmbH
                                   Donaustrasse 23
                                   D-6450 Hanau 1

Kummerer, K.R.   Kernforschungszentrum Karlsruhe
                                   IMF II, Postfach 3640
                                   D-7500 Karlsruhe 1

Latzel, G.   ALKEM GmbH
                                   Postfach 11 00 69
                                   D-6450 Hanau 11

Märtens, D.   TUEV Hannover
                                   Am Tuev 1
                                   D-3000 Hannover

Mainka, E.   Kernforschungszentrum Karlsruhe
                                    Institut für Radiochemie
                                    Postfach 36 40

Peuker, J.   Reaktor Brennelement Union GmbH
                                    Postfach 11 00 60
                                    D-6450 Hanau 11

Robers, J.   Technischer Überwachungsverein
                                    Nordd. e.V.
                                    Grosse Bahnstrasse 31
                                    D-2000 Hamburg 54

Romeiser, H.J.   Kraftwerk Union AG
                                    PO Box 32 20
                                    Hammerbacher Strasse 12+14
                                    D-8520 Erlangen

Schaub, K.H.   Gemeinschaftskernkraftwerk
                                    Neckar GmbH
                                    Postfach
                                    D-7129 Neckarwestheim

Schettler, R.   Gemeinschaftskernkraftwerk
                                    Neckar GmbH
                                    Postfach
                                    D-7129 Neckarwestheim

Schlieper, T.   Sandvik GmbH
                                    Heerdter Landstrasse 229-233
                                    D-4000 Düsseldorf

Schmidt, F.   INTERATOM
                                    Abt. T 263, Postfach
                                    D-5060 Bergisch-Gladbach 1

Schuck, V.F.   Nuclear Ingenieur Service GmbH
                                    Donaustrasse 23
                                    D-6450 Hanau 1

Stehle, H.   Kraftwerk Union AG
                                    Hammerbacher Strasse 12+14
                                    D-8520 Erlangen

Steinberg, E.   Kraftwerk Union AG
                                    Hammerbacher Strasse 12+14
                                    D-8520 Erlangen

Stiller, P.   Kernforschungszentrum Karlsruhe
                                     IMF III, Postfach 3640
                                     D-7500 Karlsruhe 1

van Dornick, J.J.H.   TUEV Baden EV
                                     Dudenstrasse 28
                                     D-6800 Mannheim 1

Vollath, D.   Institut für Material und
                                     Festkörperforschung III
                                     Kernforschungszentrum Karlsruhe
                                     PO Box 3640
                                     D-7500 Karlsruhe 1

von Wachtendonk, H.J.   Reaktor Brennelement Union GmbH
                                     Rodenbacher Chaussee 6
                                     D-6450 Hanau am Main 11

Wedemeyer, H.   Kernforschungszentrum Karlsruhe
                                     Postfach 3640
                                     D-7500 Karlsruhe

Weidinger, H.   Kraftwerk Union AG
                                     Postfach 32 20
                                     Hammerbacher Strasse 12+14
                                     D-8520 Erlangen

HUNGARY

Szabo, J.   Nuclear Power Plant paks
                                     PO Box 71, H-7031 Paks

INDIA

Balarama Moorthy, K.   Atomic Fuels Division
                                     Bhabha Atomic Research Centre
                                     Trombay, Bombay-400085

Raghavan, S.V.   Atomic Fuels Division
                                     Bhabha Atomic Research Centre
                                     Trombay, Bombay-400085

Rustagi, R.S.   Power Projects Engineering Div.
                                     Department of Atomic Energy
                                     Homi Bhabha Road, Colaba
                                     Bombay 400005

INDONESIA

Muchlis, B.

Nuclear Metallurgy Division
Research Center for Nuclear
Techniques, PPTN Batan
jl. Taman Sari 71
Bandung

ITALY

Coppola, C.C.

ANSALDO IMPIANTI Spa
Via Gabriele D'Annunzio, 113
I-16121 Genova

Marucci, G.

ENEA Casaccia
Via Anguillarese,
I-00064 Roma

Monari, M.

ENEA Bologna
Via dell'Arcoveggio, 56/23
I-40129 Bologna

Valli, G.

ENEA Casaccia
Cirene Unit − Fuel Supply
Via Anguillarese,
I-00064 Roma

Zorini, D.

NIRA Spa
Via dei Pescatori, 35
I-16129 Genova

NETHERLANDS

Van Engen, W.R.

KEMA
Utrechtseweg 310
NL-6812 AR Arnhem

PAKISTAN

Salim, M.

Pakistan Atomic Energy Commission
PO Box 1114
Islamabad

ROMANIA

Havris, C.D.   Institut de Reacteurs Nucléaires
                                      Energetiques
                                      CP 78
                                      0300 Pitesti

SOUTH AFRICA

Lämmermeyr, H.U.   Electricity Supply Commission
                                      PO Box 1091 Johannesburg

Olckers, C.   Electricity Supply Commission
                                      PO Box 1091 Johannesburg

Tillwick, D.L.   Nuclear Development Corporation Ltd
                                      Private Bag X256, Pretoria 0001

Venter, S.   Nuclear Development Corporation Ltd
                                      Private Bag X256, Pretoria 0001

SPAIN

Calleja Maestre, J.M.   Central Nuclear de Valdecaballeros
                                      C/ Claudio Coello, 53, Madrid-1

Cuallado Gutierrez, G.   Central Nuclear de Valdecaballeros
                                      C/ Claudio Coello, 53, Madrid-1

Sala Candela, A.   Hidroelectrica Espanola, SA
                                      Calle Hermosilla 3, Madrid 1

Lobon Saez, J.   Empresa Nacional del Uranio, SA
                                      Santiaga Rusinol 12
                                      Madrid 3

SWEDEN

Carlson, S.   Swedish State Power Board
                                      Nuclear Fuel Department
                                      S-162 87 Vällingby

Harfors, C.B.   Barsebeck Nuclear Power Plant
                                       Box 524, S-240 21 Löddeköpinge

Persson, A.   AB Sandvik Steel
                                       S-811 81 Sanviken

van Santen, A.   AB ASEZ-Atom
                                       Box 53
                                       S-721 04 Vasteras

SWITZERLAND

Pagani, A.   Kernkraftwerk Gösgen-Däniken AG
                                       Postfach 55, CH-4658 Däniken

Slabu, D.V.   Nordostschweizerische Kraftwerke AG
                                       Parkstrasse 23, CH-5401 Baden

USSR

Gorskij, V.   All Union Scientific and Research
                                       Institute of Inorganic Materials
                                       Ulitza Rogova 5-A, Moscow

Sokolov, F.   All Union Scientific and Research
                                       Institute of Inorganic Materials
                                       Ulitza Rogova 5-A, Moscow

UNITED KINGDOM

Foster, K.   British Nuclear Fuels Ltd.
                                       QAD Bldg. 382
                                       Springfiel Works
                                       Salwick, Preston, Lancashire
                                       PR4 OXJ

Jenkins, B.P.   British Nuclear Fuels Ltd.
                                       Springfiel Works
                                       Salwick, Preston, Lancashire
                                       PR4 OXJ

Moore, G.   Rolls Royce Ltd. Nuclear
                                       PO Box 31, Raynesway,
                                       Derby, DE2 8BJ

Whetton, P.A.   Rolls Royce Ltd. Nuclear
                                         PO Box 31, Raynesway,
                                         Derby, DE2 8BJ

USA

Bowles, K.   Sandvik Special Metals
                                         PO Box 6027
                                         Kennewick, WA 99336

Hetzler, F.   1753, 57th Street N.E.
                                         Tacoma Washington 98422

Strasser, A.A.   S.M. Stoller Corporation
                                         1250 Broadway
                                         New York NY 10001

YUGOSLAVIA

Ilic, R.   Jozef Stefan Institute
                                         POB 53
                                         61111 Ljubljana

Kocic, A.   Boris Kidric Institute
                                         PO Box 522
                                         11070 Belgrade

Susnik, D.   Jozef Stefan Institute
                                         Jamova 39
                                         61000 Ljubljana

Trontelj, M.   Jozef Stefan Institute
                                         Jamova 39
                                         61000 Ljubljana

ORGANIZATION

Rybalchenko, I.   Division of Nuclear Fuel Cycle
                                         Wagramerstrasse 5
                                         PO Box 100
                                         A-1400 Vienna

O B S E R V E R S

Pranowo, T.   Jl. Muararajeune Kulon 48
                                    Bandung / Indonesia

Sunaryadi, D.   Jl. Rajamantri Tangah 12
                                    Bandung / Indonesia

Guerrieri, C.   Viale Regina Margaretha 137
                                    Roma / Italy

Buescher, B.J.   EG & G
                                    Idaho, Inc.
                                    PO Box 1625
                                    Idaho Falls, Idaho 83415
                                    USA