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SPENT FUEL SURVEILLANCE AND MONITORING METHODS

PROCEEDINGS OF A TECHNICAL COMMITTEE MEETING
ORGANIZED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
AND HELD IN VIENNA, 27–30 OCTOBER 1987

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FOREWORD

Nuclear power now accounts for over 15% of world electricity production and, in some countries, that proportion is in the range of 50-70%.

At the end of 1986, there were 394 power reactors connected to electricity supply networks in 26 countries with 270 232 megawatts of electrical generating capacity (MWe). During the year, 21 reactors accounting for a total of 20 938 (MWe) were newly connected to the grid.

One of the most important issues of nuclear energy development was, is and will be the nuclear fuel cycle technology. Presently, spent fuel storage is the most actual one. At present about 45.000 metric tonnes of water reactor spent fuel had been discharged worldwide. Only a small fraction of this fuel (approximately 7 %) has been reprocessed.

The amount of spent fuel arisings will increase significantly in the next 15 years. Estimates indicate that up to the year 2000 about 200 000 MTHM [Metric Tonnes Heavy Metal] of spent fuel could be accumulated. Taking into account the large quantities of spent fuel discharged from nuclear power plants and future expected discharges, countries are involved in the construction of facilities for storage of spent fuel and development of effective methods for spent fuel surveillance and monitoring to ensure that reliable and safe operations of storage facilities are achievable over periods when the final disposal of spent fuel or high level wastes is feasible. For any chosen strategy for the back-end of the nuclear fuel cycle, long-term storage of spent fuel is assumed to be a highly important option for all countries with nuclear power programmes. Therefore, it is of actual interest to exchange information on an international level on spent fuel surveillance and monitoring methods during storage.

Spent fuel surveillance consists of all planned activities which are performed to ensure that the condition of fuel assemblies remain within the prescribed limits. Surveillance programmes usually include continuous or periodical monitoring of both spent fuel and the storage environment. Surveillance programmes are being prepared taking into account the examination techniques available.

The Technical Committee Meeting on "Spent Fuel Surveillance and Monitoring Methods" (27-30 October 1987) has been organized in accordance with recommendations of the International Standing Advisory Group on Spent Fuel Management during its second meeting in 1986. The meeting formed part of the IAEA programme on Spent Fuel Management as discussed in the Annex I of the IAEA-TECDOC-419, which contains the proceedings of an Advisory Group Meeting on Spent Fuel Management, March 1986.

The aim of the meeting was to discuss the above questions with emphasis on current design and operation criteria, safety principles and licensing requirements and procedures in order to prevent: inadvertent criticality, undue radiation exposure, unacceptable release of radioactivity as well as control for loss of storage pool water, crud impact, water chemistry, distribution and behaviour of particulates in cooling water, oxidation of intact and failed fuel rods as a function of temperature and burnup; distribution of radiation and temperature through dry cask wall, monitoring of leakages from pools and gas escapes from dry storage facilities, periodical integrity tests of the containment barriers, responsibilities of organizations for the required operation, structure, staff and subordination, etc.

The Agency wishes to thank all those who participated in the meeting. Special thanks are due to the Chairman Dr. C.W.E. Addison, and Co-chairmen Mr. E. Vitikainen and Mr. M.J. Sametband. The officer of the IAEA responsible for the preparation of this document is Mr. F. Sokolov, Division of Nuclear Fuel Cycle.

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SPENT FUEL SURVEILLANCE PROGRAMMES AND PRACTICE IN MEMBER STATES

(Session I)

Chairman

M.J. SAMETBAND
Argentina

THE WET SPENT FUEL SURVEILLANCE PROGRAM IN ARGENTINA

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Abstract

Two PHWR reactors are now being operated in Argentina : Atucha I, with a power of 345 MWe, started in 1974, and Embalse, with 600 MWe, which went into criticality in 1983. One more is being built: Atucha II, with 745 MWe, and 700 additional MWe are planned to be connected to the electric grid by the year 1999.

For spent fuel, wet storage is being used. At present, 4650 fuel bundles have been irradiated, with an average burn-up of 6500 MWd/MgU, totalling 805.845 Kg of UO_2. They are deposited in 6 pools at Atucha I,which have enough capacity to store the fuel irradiated during the whole life of the plant.

About 12.600 fuel bundles were discharged at Embalse since 1983, with 267.900 Kg of UO_2, and are deposited in the storage pool, which has capacity enough to hold the fuel discharged during 10 years operation.A separate pool is used for defected bundles.

The pools at Atucha I have an austenitic steel lining, whereas at Embalse they have an epoxi lining, with good results in both cases.

A description is made of the monitoring in the water pools, chemical controls and radioactivity measurements.

The systems for failed fuel detection are described, as well as the non-destructive methods used for the surveillance of the irradiated fuel at both plants. At Embalse, disassembling of the fuel bundles is also being performed underwater, in order to examine samples of the cladding.

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1. *INTRODUCTION*

The Argentine nuclear program foresees a power capacity of 2300 MWe before the end of the century, which will be generated by PHWR plants.

At present, two reactors are operating: Atucha-I, with 345 MWe, since 1974, and Embalse, 600 MWe, which went into criticality in 1983. A third one is under construction, Atucha-II, with 745 MWe, and about 700 MWe shall be connected to the electric grid by the year 1999.

Both operating reactors are using fuel with natural UO_2 pellets and Zircaloy-4 cladding. This will be also the fuel for the next reactor.

Until September 1987, 4650 fuel bundles have been irradiated at Atucha-I, with an average burn-up of 6500 MWd/MgU, totalling 805.845 Kg of irradiated UO_2. At Embalse, about 12600 fuel bundles were discharged since 1983, with 267.876 Kg of UO_2.

At both sites, wet storage is utilized for the spent fuel, and will be also used for Atucha-II.

2. *CHARACTERISTICS OF THE WET STORAGE AT THE POWER PLANTS*

2.1 *ATUCHA-I*

2.1.1 *GENERAL CHARACTERISTICS*

This plant has a pressurized vessel reactor designed and built by KWU.

It is moderated and refrigerated by heavy water, and the fuel bundles consist of 36 rods of Zry-4 filled with natural uranium dioxide pellets, and one structural rod, all fixed to an upper plate. The bundle is kept tight with 15 spacers, and the fuel rods are free standing, pressurized with helium.

The active length of a fuel bundle is 5300 mm, with 173.3 Kg of UO_2.

The average burnup is 6500 MWd/MgU; 4650 bundles have been irradiated, and a total of 2150 Kg of Pu has been produced since 1974.

2.1.2 *STORAGE*

The spent fuel is stored at the NPP site in pools with a total capacity of 10340 bundles, that is, for the whole lifetime of the reactor.

Initially, there was one fuel pool building, with a transfer pool and two storage pools. A second building was later constructed with one transfer pool and four storage pools, every one of them of concrete with an austenitic stainless steel lining.

All the pools have a depth of 16 m, and the storage pools have a length of 8 m and a width of 5.5 m.

The deionized water circulates continuously through a cooling and purification system, with a mixed bed ion-exchange filter.

Both spent fuel buildings are connected, and the filter serves all the pools.

The water has a controled temperature of 34°C, with an allowed maximum of 60°C, and a pH of about 7.

The chemical controls include the measurement of silica, chloride and sodium ions, pH and conductivity.

The chloride ion content is kept very low (≤0.2 ppm), since the presence of chlorides may accelerate localized corrosion in some of the materials used, e.g. stainless steel.

The radiochemical controls are carried out to keep the radioactivity levels within the allowed limits.
Those levels are the same as at every place in the controlled zone, that is, a maximum of 5 Rem/year.

The purification system with mixed resins and diatomea filters removes this type of contamination.

The radioactivity measurements at the storage buildings involve the evacuation of the place in case of exceeding the tritium levels (~ 5 derived limits) although this is improbable since there are no D_2O systems in these buildings.

The γ activity measured values vary between 10^{-4} and $10^{-3}\mu C/cm^3$, of which 85% correspond to Co^{60} and the other 15% to Ru^{106}, Cs^{137} and Cs^{134}.

There is monitoring of Sr^{90}, and in case of detection of noble gases, evacuation of the buildings is ordered.

There has been no evidence of biological growth, with the exception of one ocassion, where the biological contamination was brougth from other building, and was removed mechanically.

2.1.3 *FUEL EXAMINATION*

All the inspections of irradiated fuel are performed in the transfer pool were the fuel is maneuvred in and out of the reactor.

Three non-destructive systems have been installed:

- Visual, with a periscope coupled to a telescope or to a TV camera,

- Metrology, to measure the length variation of the rods. It uses the fuel transport bridge, which has a 14m long mast, where the bundles are coupled and displaced vertically and also rotated axially. Both extremes of the length to be measured are aligned by means of the periscope to which a Questar telescope provided with a reticle is attached.

- Gamma scanning, measuring the total gamma activity along the bundle.

2.1.4 *FAILED FUEL DETECTION*

The gamma activity of the heavy water contained in the primary circuit is measured several times daily.

When there is an increase of the activity which could be caused by a fuel failure, the NX system for detection and localisation of failed fuel is used.

This system has a manifold valve which receives selectively the water of all the 230 fuel channels, and measures the gamma peaks corresponding to fission gases.

Once the defected fuel is localised, it is extracted from the reactor and transported to the pool to be examined and canned if necessary.

The operating experience indicated the convenience of supplementing this NX system with another method, based on a gamma spectrometer monitoring the heavy water activity in the refuelling machine.

The refuelling machine changes a spent fuel for a new one, with an average frequency of 1.5 bundles per day. During the same operation, it reshuffles fuel, working at reactor pressure.
Once finished, it transfers the spent fuel to the pool, and for that the refuelling machine is depressurized, 40 minutes after reshuffling. At this moment, fission products released from rod defects, with lifetimes higher than 10 minutes, can be detected.

2.2 *EMBALSE*

2.2.1 *GENERAL CHARACTERISTICS*

The Embalse de Rio Tercero plant consists of a Candu reactor, 600 MWe, moderated by heavy water, and each one of the 380 horizontal pressure tubes contains 12 fuel bundles, 8 bundles being added or discharged during each refuelling operation.

The fuel consists of UO_2 pellets filling 37 tubes of Zry-4, with an active length of 479mm. Each bundle contains 21.26 Kg of uranium dioxide.

The average burnup is 6700 MWd/MgU; 12600 fuel bundles have been irradiated, with a total production of 799 Kg of Pu until September 1987.

2.2.2 *STORAGE*

The irradiated fuel is stored in the service building adyacent to the reactor, in a pool with a capacity of 44648 bundles, corresponding to ten years of plant operation with a load factor of 0.85.

All the four pools: for spent fuel discharge, reception, main storage and failed fuel storage, are built of concrete and have epoxi lining.

The main storage pool is 22.5 m long and 11.6 m wide, with a water depth of 7.6 m; the failed fuel storage pool has 2.4x5.5 m, with a water depth of 6 m, and a minimum of 4.1 m must remain between the top of the bundles and the water surface to ensure negligible radiation levels.

Through all the pools, demineralized water circulates through heat exchangers and mixed bed resins.

The controled temperature is 25°C, and the water is kept very clear. This is essential to make easier the handling and visual inspection of the fuel.

The radioactive contamination is also removed by the purification system; this material arises from defective fuel or from deposites formed on the surface of the fuel assemblies while they are inside the reactor core.

Chemical controls enable to keep the water very pure. A very low chloride ion content is specified ($\leq$ 0.2 ppm), and the average measured value is $\leq$ 0.1 ppm.

The specified pH is 6.5, and the conductivity should be below 2uS.

Under normal conditions, no enhanced corrosion should be expected in the storage pools.

The radioactivity level is controlled: the total activity is 2.10^{-3} $\mu Ci/cm^3$.
The average radiation exposure for the personnel working at the pools area shall not exceed 0.6-1.0 mrem/hr.

2.2.3 *FUEL EXAMINATION*

The inspection is performed in the transfer bay, using a Questar telescope and a periscope. Measurements of the dimensions are also performed.

It is possible to disassemble fuel rods underwater. The equipment is positioned on the observation frame, and the operation is viewed through the periscope.

A cleaning filter absorbs particles and debris which may be produced during the operation, by means of a pneumatic pump and a 10 micron filter similar to the one used in the primary circuit.

2.2.4 FAILED FUEL

The failed fuel is separated from the normal spent fuel in the discharge pool, and then is stored in a carousel type container for an initial decay period. After that, it is canned and

transferred to the failed fuel storage pool for long term storage.

The defective fuel is identified through two systems:

The Gaseous Fission Product System, with a gamma spectrometer of high resolution, which operates continuously and measures the gamma activity of some gaseous fission products and of I^{131} in two coolant sample lines, one for each Heat Transport System loop. It may determine when failed fuel appears, and in which loop.

In addition to this, coolant samples are taken periodically from a pump discharge line and from a steam generator inlet in each loop.

The Failed Fuel Location System by Delayed Neutrons, identifies the fuel channel containing the defective fuel and further indicates when the defective bundle have left the channel during the refuelling operation.

The system has sampling lines connected to each one of the 380 fuel channels.
Each line carries coolant to the sample coils immersed in a water filled moderator tank, where BF_3 thermal neutron detectors are lowered.
The system measures neutrons emitted by I^{131} and Br.

The location of the defective fuel bundle is then known, and so, on arrival in the spent fuel pool it is segregated manually, and transported to the failed fuel storage pool.

2.3.1 *ATUCHA-II*

This is a KWU pressurized vessel reactor, 745 MWe, being built in the neighborhood of Atucha-I.

The fuel is similar to Atucha-I, with natural uranium dioxide pellets in 37 rods, Zry-4 cladding, an active length of 5300 mm and a weight of 25 kg of UO_2.

The average burnup is 7500 MWd/MgU.

Five pools are being built: four for spent fuel storage and the fifth for reception and special operations.

Full storage capacity is of 3696 fuel bundles.

The future management of the pools will be similar to the one described for Atucha-I.

3. *CONCLUSIONS*

The experience in Argentina in the field of wet storage has been satisfactory, but taking into consideration that the fuel bundles may remain in the storage pools for long periods- more than 50 years- it is very important to study the possible mechanisms of degradation of the irradiated fuel materials, and of the components of the pools, like the linings and trays which hold the spent fuel.

More research on long term wet storage is needed, as well as on dry storage.

REFERENCES

(1) M. C. GELDSTEIN. Studies on the reliability of Materials used in the Back-End of the Nuclear Fuel Cycle in Argentina. Technical Committee Meeting, IAEA, Vienna, 2-5 September 1986.

(2) C. ARAOZ et al. Spent Fuel Management in Argentina. IAEA Advisory Group Meeting, Vienna, 11-13 March 1986.

(3) M. J. SAMETBAND et al. Fuel Examination for Atucha-I and Embalse Power Reactors. ED-WARF coordination Meeting, IAEA, Vienna, September 1985.

SPENT FUEL SURVEILLANCE IN SPAIN: SHORT REVIEW OF DIFFERENT METHODS USED IN NUCLEAR POWER STATIONS AND RESEARCH CENTRES

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Abstract

Eight power plants are operating in Spain, 5 of them are PWR, 2 are BWR and one GCR type.

All light water spent fuels are stored at pools in each NPP, being built of concrete, designed as seismic class I and stainless steel lined, and the saturation of their capacities is estimated to occur, on average, between 1993 and 2005. The GCR spent fuels are also stored in the NPP pool until its periodic sending to reprocessing.

Racks are designed to maintain the spent fuel with a $K_{eff} < 0.95$. The pool water is cooled, filtered and demineralized. Controls are carried out on water temperature, level in the pool, flow rate in the cooling circuit, radiation level and radioactive concentration in the air, water chemistry, etc. Irradiated fuel assemblies are vissually inspected by means of TV cameras and the cooling and purification systems are also visually inspected.

A strategy is being elaborated for intermediate storage of spent fuels, considering both wet and dry storage.

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1. INTRODUCTION

The Spanish nuclear power program has been revised in 1983. The National Energy Plan (PEN), approved by the Parliament, includes 10 nuclear power stations as shown in Table 1. All the NPPS are of LWR type, but Vandellós I (gas cooled,

natural uranium reactor). Spent fuel from this reactor (80-90 TmU/year) are being reprocessed by COGEMA, France.

TABLE 1. NUCLEAR POWER PLANTS IN SPAIN

NAME	GROSS POWER (MWe)	REACTOR TYPE	NSSS SUPPLIER	DATE OF COMMERCIAL OPERATION
José Cabrera	160	PWR	W	July 1968
Santa María de Garoña	460	BWR	GE	March 1971
Vandellós I	480	GCR	SFAC	May 1972
Almaraz I	930	PWR	W	July 1981
Ascó I	930	PWR	W	Dec. 1983
Almaraz II	930	PWR	W	Jan. 1984
Cofrentes	975	BWR	GE	Oct. 1984
Ascó II	930	PWR	W	Dec. 1985
Vandellós II	982	PWR	W	1988
Trillo I	1041	PWR	KWU	1988

W = Westinghouse.
GE = General Electric.
SFAC = Societé des Forges et Ateliers du Creusot.
KWU = Kernkraftwerk Union.

The total LWR spent fuel arisings are estimated at about 5500 tonnes U. Moreover, 114 tonnes, from José Cabrera (56 TmU) and Santa María de Garoña (88 TmU), were delivered to British Nuclear Fuel, Ltd., UK.

According to this program, short and medium term plans have been established to assure enough capacities of the spent fuel pools by installing compact racks. In Table 2 is shown the storage capacities of the nuclear power plants now in operation. At medium and long term, wet and dry storage of spent fuel are planned.

TABLE 2. SPENT FUEL CAPACITY OF LWR NUCLEAR POWER PLANTS.

NPP	POOL FUEL ELEMENT CAPACITY	CORE FUEL ELEMENT CAPACITY	EFECTIVE CAPACITY
José Cabrera	310	69	241
Santa María de Garoña	1727	400	1327
Almaraz I	612	157	455
Ascó I	588	157	431
Almaraz II	612	157	431
Cofrentes	3024	624	2400
Ascó II	588	157	431

2. GENERAL CONSIDERATIONS

Spanish LWR storage pools are built in concrete, stainless steel lined, and designed as seismic Category I.

Last five NPPS dispose of a fuel storage building designed to protect the assemblies against missiles, tornado winds, floods, seisms, etc. A ventilation system maintains the air in a confortable condition for people working in these buildings.

Spent fuels are vertically stored in racks, designed to keep a distance among them to maintain a subcritical situation ($K_{eff} < 0.95$) in a compact distribution.

One of the BWR (Santa María de Garoña) has its racks recovered by boral plates with stainless steel cladding. The five PWR type plants contain, as aditive, boric acid with Boron concentration between 2,000 and 2,500 ppm.

All plants pool water cooling and purification systems consit of heat exchanger, filters and demineralizer components.

The cooling systems are designed as seismic Category I and redundant, and its capacity is evaluated in order to maintain pool water temperature, in normal operation, below 52°C-57°C depending on the plant.

The purification systems are calculated to keep the water radioactive concentration below a level that allows a dose rate, in accessible areas, less than 2.5 mrem/hr. Normally this radioactive concentration is about 10^{-3} $\mu Ci/cm^3$.

Both systems allow a minimun water level in the pool between 3 m and 4,5 m, depending on the NPP, to obtain a water shielding to accomplish the radiological protection limit (2.5 mrem/hr). In normal operation water levels are above these figures and attain values between 6 m and 7.8 m.

Penetrations below the safe shielding level are designed to avoid a syphon effect.

Cranes and other transfer equipments are provided with interlocks to prevent from passing over stored fuels when fuel handling is not in progress.

All now in operation LWRS have capacities enough to contain its spent fuel generated in an operation period of more than 10 years. Taking into account the effective capacities shown in Table 2, and the total number of spent fuel stored in every pool, the saturation dates may be estimated between 1993 and 2005 for these plants.

The Vandellós I pool is able to store about 14.500 fuel elements, capacity enough for one year and a half production period (8000 FE/year). This pool has a water filtering system,

in continous, and a Cesium decontamination system for the event of a clad breaking during the storage period.

The research reactor (JEN-1) is swimming pool type, 3 Mw thermal power, using plate type fuel elements. Spent fuel are stored in the lower power zone of the pool. Includes water cooling and purification systems as well as air ventilation system and fuel handling equipment. This research reactor started its operation in 1959 and now is stopped.

3. SURVEILLANCE AND MONITORING

In all the Spanish nuclear power plants, the following parameters, regarding pool water, are controlled: Temperature, level, pH, conductivity, chemical composition, and radiochemical concentration.

Chemical and radiochemical analysis are periodic. The pH is controlled to avoid incompatibility between the water and the racks material.

The instrumentation of the other parameters gives local indications and alarm signals in the control hall when the measured values are up or below the levels stablished by design criteria or by normal operation values.

At the same time, the differential pressure between the upstream and dowstream sides of filters and demineralizers is measured as well as cooling system flow rate.

Periodicaly, water samples are taken in different zones of the pool for radiochemical analysis; for instance, in the purification system to verify the effectiveness of filters and demineralizers.

The fuel storage building air is continously monitored from the point of view of radiation level, and the air radioactivity concentration is measured periodicaly.

Spent fuel perfomed inspections are:

Visual by means of TV cameras, to detect, mainly, surface cracks and changes in the colours of welds that may be a consequence of corrosion products.

Sipping in some plants when the pool water isotopic analysis gives leak signal in a rod or assembly.

Active components of cooling and purification systems are visually inspected.

A leak detection system detects liner leakage and gives alarm signal in the control hall.

The research reactor spent fuel is inspected, in the pool, visual and radiologicaly. On each fuel element are perfomed two dose rate measurements, one using a Teledetector IF 103, to obtain the contact dose rate, and another one using a linear ratemeter, Babyline type, located 1 m over the horizontally placed fuel element.

All controlled parameters in the NPPS and the research reactor are registered and these registers are sent periodicaly to the Regulatory Body (CSN, Nuclear Safety Council).

By the other hand, post irradiation examinations have been performed in experimental fuel rods irradiated in the José Cabrera reactor, as part of a Tripartite Agreement among Westinghouse, Unión Eléctrica, S.A. and the Junta de Energía Nuclear (now CIEMAT).

These examinations consist on visual, fuel rod length measurements, gammascanning, released fission gas determinations and cruds analysis, all these measurements were carried out in the JEN metallurgical hot cells.

4. FUTURE PERSPECTIVES

The total amount of spent fuel to be generated in Spain is about 5,500 TmU.

In 1984 was created ENRESA, a national company with responsabilities to carry out all activities related with the management of radioactive wastes, including spent fuel. Some of the Spanish storage pools will reach their full capacity by 1993. ENRESA is elaborating the strategy for intermediate spent fuel storage, considering both wet and dry storage.

CONSTRUCTION AND OPERATING EXPERIENCE OF THE PFR IRRADIATED FUEL BUFFER STORE

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Abstract

Irradiated fuel and breeder subassemblies are removed from the core of the Prototype Fast Reactor (PFR) at Dounreay at the end of their useful life and are held under sodium until the fission product heating has decayed to below 2.5 kW. Each subassembly is then drained of sodium in the Irradiated Fuel Cave (IFC) and the top mixer breeder section and lower filter and spike sections are removed using a laser. The remaining section containing fuel pins is steam cleaned and water washed to remove the sodium, sealed in a spent fuel can and sent to an Irradiated Fuel Buffer Store Pond (IFBS) to await transfer to the Dounreay Fast Reactor Fuel Reprocessing Plant. The IFBS thus serves as a reservoir for spent fuel and eliminates the need to directly couple the preparation programme in the IFC to the operation programme of the Reprocessing Plant. It also serves as a long term storage facility.

The IFBS was commissioned in 1982. This paper discusses the construction and operating experience of the 247 channel storage pond which was built as an annexe onto the PFR Reactor Hall after the reactor had been operating for several years. Particular aspects covered are the fuel element storage cans used, the pond transporter and can draining facility, and the associated flask handling equipment.

Introduction

1. The Prototype Fast Reactor is a 250 MW(e) pool type sodium cooled fast reactor power station situated at Dounreay on the North Coast of Scotland. The reactor core comprises 78 fuel subassemblies, the remaining 132 channels containing breeder and reflector subassemblies.

2. Irradiated fuel is stored in three facilities within the reactor complex. Following discharge from the core with the off load refuelling machine, subassemblies are held under sodium in buckets in a 20 position storage rotor in the reactor's outer sodium pool. When fission product heating has decayed to below 15 kW a subassembly may be transferred in its bucket of sodium by a fuel transfer flask (FTF) from rotor storage to a primary storage tank (PST) in the Irradiated Fuel Cave (IFC) that is situated within the reactor building. The sodium filled section of the PST can hold 60 items under sodium. When the individual subassembly heat level has decayed to below 2.5 kW it is lifted and drained of sodium as the first step in its preparation for reprocessing. At this stage a laser is used to cut the top mixer breeder section and lower spike from the subassembly. The remaining 'fuel section' comprising 325 steel clad sealed fuel pins within a hexagonal wrapper section is then ready for steam cleaning and water washing to remove adhering residuesof sodium.

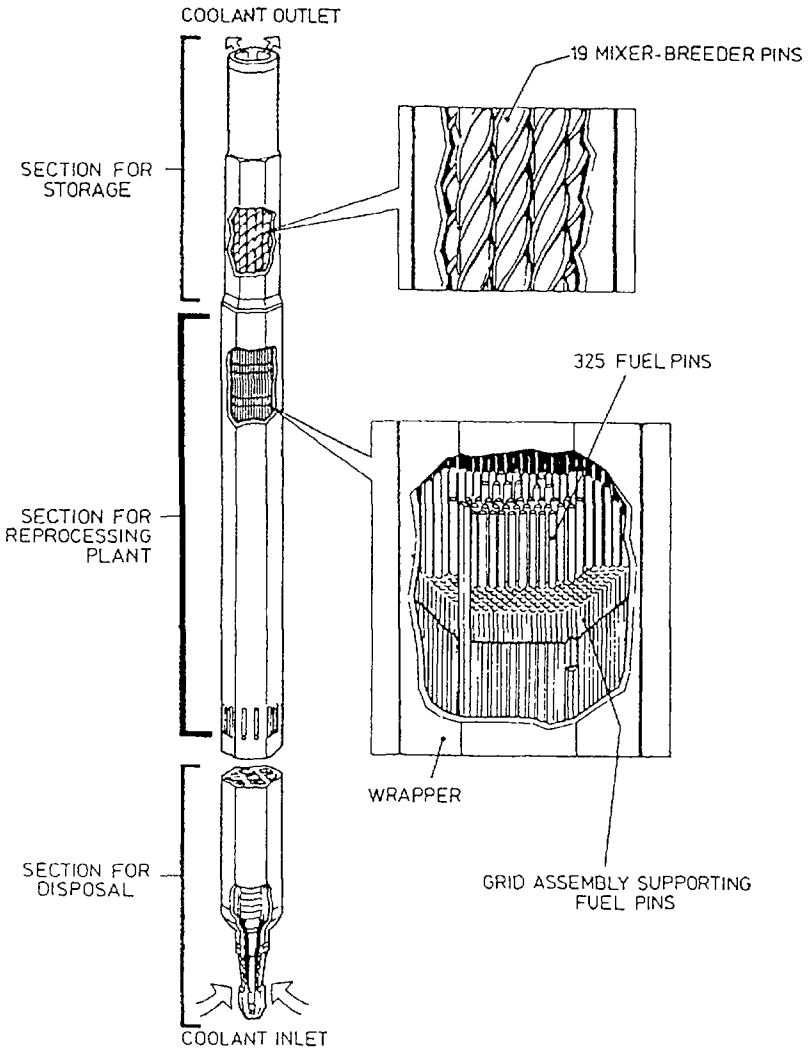


Fig. 1 PFR SUB ASSEMBLY SHOWING APPROXIMATE LASER CUTTING POSITIONS

Once 'topped and tailed' and cleaned the subassembly is placed in a sealed container, a spent fuel can, before it is moved out of the nitrogen blanketed IFC into the Irradiated Fuel Buffer Store. Figure 1 shows a subassembly and the position of the laser cuts: Figure 2 the physical arrangement of the spent fuel can.

3. The original intention was to reprocess PFR fuel at Windscale but the success in dealing with spent fuel at Dounreay from the Dounreay Fast Reactor (DFR) and other materials testing reactors during the 1960's and 1970's led to modification, improvement and expansion of the DFR Reprocessing Plant to accommodate the physically larger PFR fuel. A local buffer store was therefore necessary to ensure that the operational programmes of the reactor and reprocessing plant could be independent of each other.

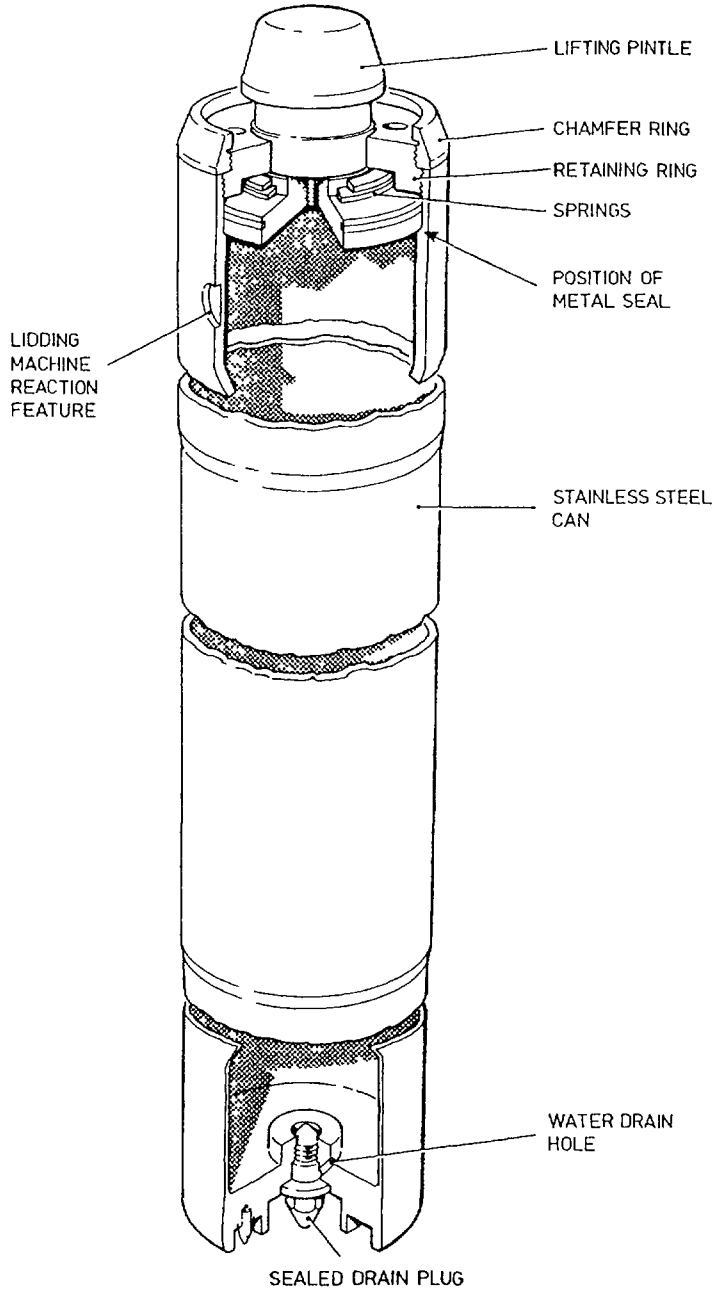


Figure 2 PFR SPENT FUEL STORAGE CAN MK III TYPE

The Irradiated Fuel Buffer Store (IFBS) or Buffer Store

4. The buffer store provides a short term 'buffer' between the Irradiated Fuel Cave and the Reprocessing Plant to avoid the direct coupling between these two facilities and a storage facility for mixer breeder units and breeder sub assemblies not likely to be reprocessed in the short term. The buffer store has sufficient storage capacity to accommodate all fissile reactor core components if it became necessary to discharge the total 210 reactor channels at the same time. With these considerations in mind the physical size and capacity of the under water storage pond was fixed as a 13 x 19 lattice giving 247 channels.

5. The buffer store is housed in a sealable containment building attached to the north west corner of the reactor building. It has its own ventilation system and is linked to its parent building by a personnel airlock and a sealing inlet transfer port valve.

The Pond

6. The storage pond construction consists of a high quality stainless steel tank having vertical stainless steel storage holes let into the base plate and projecting downwards. The storage tubes are set in porous concrete to allow any leakage of pond water to percolate to a drain point connected to a hold up tank for storage. This tank also serves as a radiation monitoring point for measuring the activity of any such leakage. The porous concrete is contained in a conventional type of pond with thick walls and base and having its own asphalt internal sealing membrane. The operating water level is 1.37 m above the top of the storage holes.

7. The outer walls and inner porous concrete mix are of sufficient thickness and density to attenuate radioactive emissions from the stored fuel. Figure 3 shows the basic design of the pond.

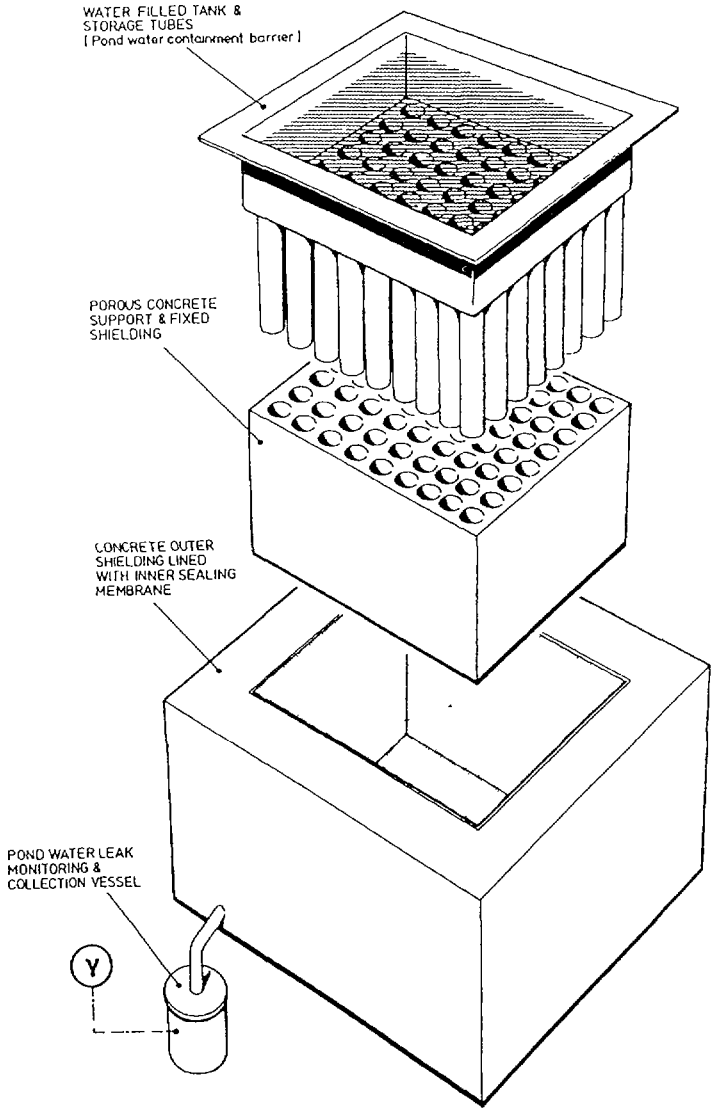
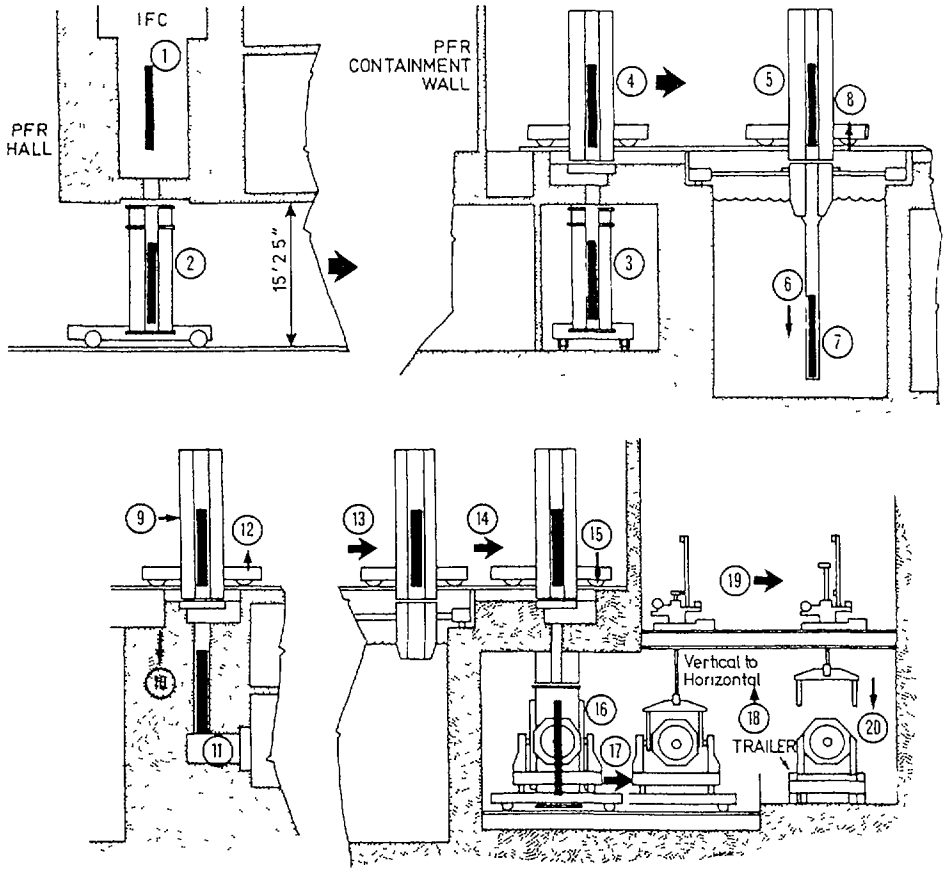


Figure 3 BASIC POND DESIGN

The Pond Transporter

8. The pond transporter is designed to move a spent fuel can to and from any pre-determined pond storage position and three other stations: the draining facility, the inlet port valve and the discharge port valve. Figure 4 shows diagramatically the flasks, transporter and can draining handling sequences. It is also possible to move a spent fuel can from one port to the other port without lowering into the pond water.



1. S/A CAN IN CELL
2. LOWER INTO TRANSIT FLASK
3. MOVE TO BUFFER STORE INLET PORT
4. RAISE S/A CAN INTO POND FLASK
5. MOVE POND FLASK OVER POND STORAGE HOLE
6. LOWER S/A CAN INTO STORAGE HOLE
7. LEAVE S/A CAN FOR STORAGE PERIOD
8. RAISE S/A CAN INTO POND FLASK
9. MOVE POND FLASK TO DRAINING FACILITY
10. LOWER S/A CAN INTO DRAINING FACILITY
11. DRAIN & RESEAL S/A CAN
12. RAISE S/A CAN INTO POND FLASK
13. MOVE POND FLASK OVER POND
14. MOVE POND FLASK OVER DISCHARGE PORT
15. LOWER S/A CAN INTO TRANSIT FLASK
16. TILT TRANSIT FLASK INTO HORIZONTAL POSITION
17. MOVE TRANSIT FLASK UNDER HANDLING CARRIAGE
18. LIFT TRANSIT FLASK
19. MOVE FLASK OVER TRAILER
20. LOWER FLASK ON TO TRAILER

Figure 4 POND, TRANSPORTER AND FLASK HANDLING SEQUENCES

9. Three main components make up the transporter. A travelling bridge; a crab carrying a winch and shielded container mounted on rails on the bridge and a shielded snout. The transporter is designed to be capable of future extension to handle full length canned irradiated subassemblies. The shielded container is constructed of high density concrete encased in 12.7 mm of carbon steel. Figure 5 shows the pond transporter.

10. When moving over the pond the transporter engages onto a shielded snout which is partially submerged in the pond water and which extends from the transporter carriage downwards to just above the storage tube support grid. The snout forms additional shielding to the water and is designed to minimise radiation at the interface between the transporter shield and the top of the snout. The purpose of the snout is to ensure adequate shielding to operators on the crab and on the pond walkways whilst an irradiated subassembly in its can is being lowered into or raised from its pond storage location.

11. The hoist mechanism is external to the shielded container for ease of access for maintenance and is shielded where its cable passes into the container to protect the building superstructure from radiation and to eliminate scatter to the working area. The hoist has two speeds: normal 4 metres per minute and creep of 0.2 metres per minute.

12. Electrical and instrument supplies to the transporter are carried by multicable catenaries to avoid the use of multipin connectors or slip rings which experience has shown to be unreliable.

13. The grab design was based on that used successfully at Dounreay in the Dounreay Fast Reactor (DFR) pond transporter. The grab jaws are powerfully spring loaded into the closed (engaged) position. Pneumatic pressure into an internal bellows unit overcomes the spring loading and permits the jaws to open (disengage). A 'slack rope' signal coincident with the correct grab depth permits the pneumatic pressure to be applied to open the jaws. Figure 6 shows the arrangement of the grab.

14. Adjacent to the pond and sealed by a shielded valve of the same design as those used at the buffer store inlet and discharge ports is the spent fuel can draining facility. A sealing plug in the base of the can is unscrewed to check that no water has leaked into the can during its under water storage. Hand operated tools working through shielding are used for this operation which is observed through a lead glass shielded viewing window.

15. Any water drained from a spent fuel can is passed from the draining facility into a storage vessel. The activity of this liquid is measured by a local gamma detector attached to the vessel. The design permits the drained water to be circulated through a 1 micron filter until its activity level is reduced to less than 200 micro Sv per hour when it is ready to be transferred to the PFR liquid effluent plant. The filter is changed remotely and is removed from the filter handling cell by a La Calhene sealed system into a shielded flask for disposal.

16. A grab delatch facility is incorporated into the top of the can draining facility. This allows the can to be supported at the correct height to permit a collar on the grab to be moved against the spring load of the grab jaws to release the grab in the event of the jaws failing to open when latched onto a can.

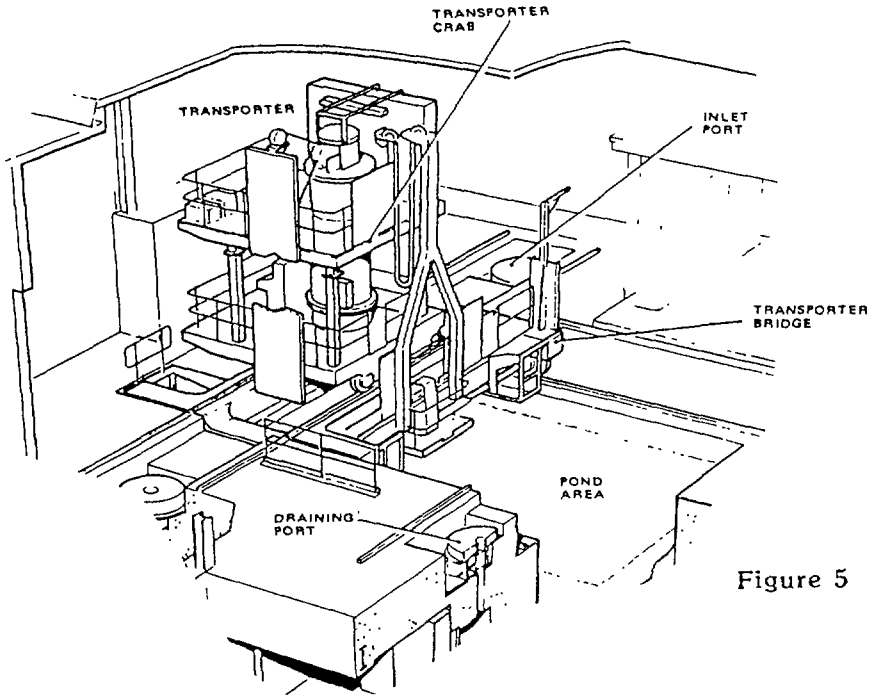
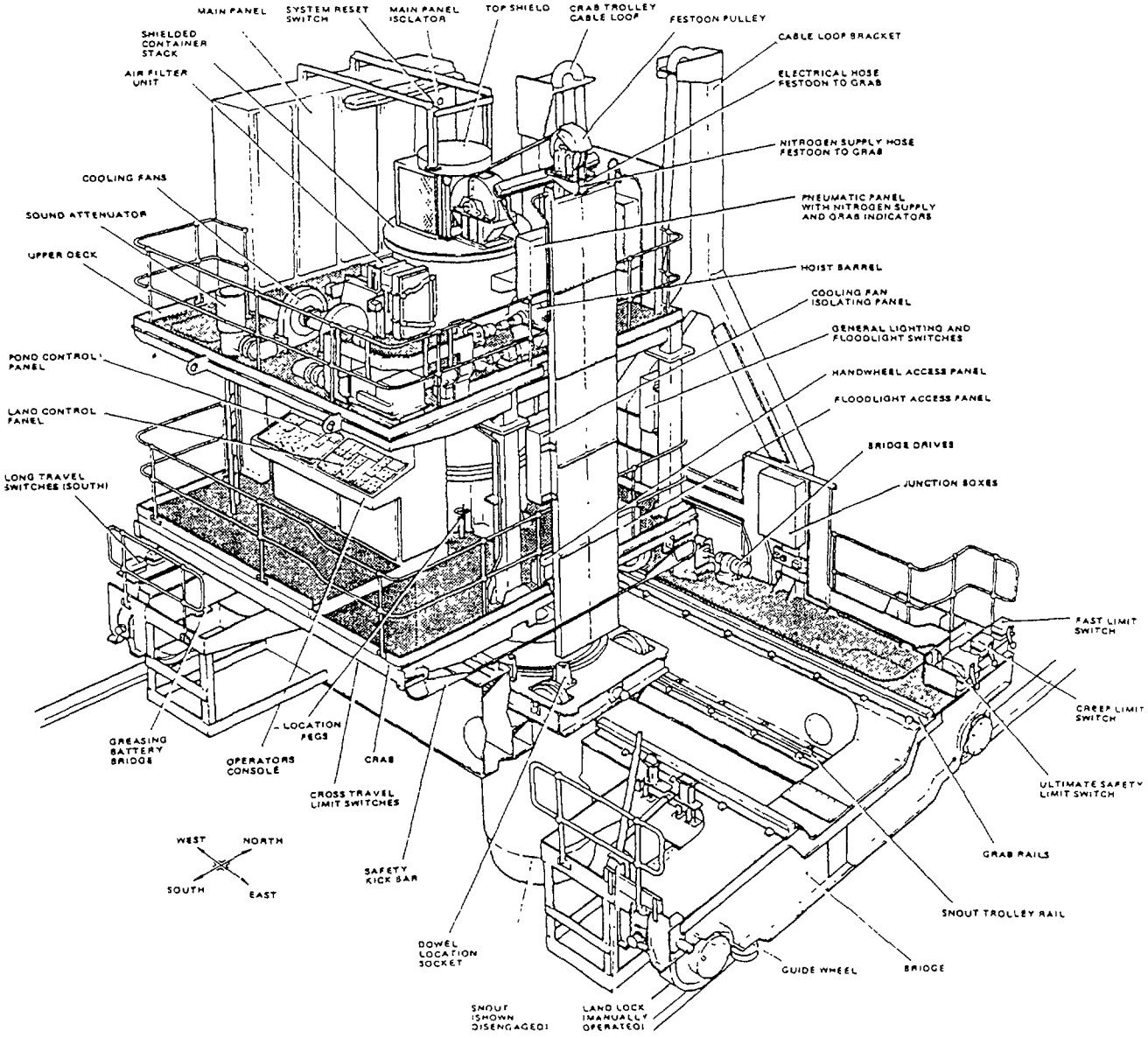


Figure 5 PFR POND TRANSPORTER



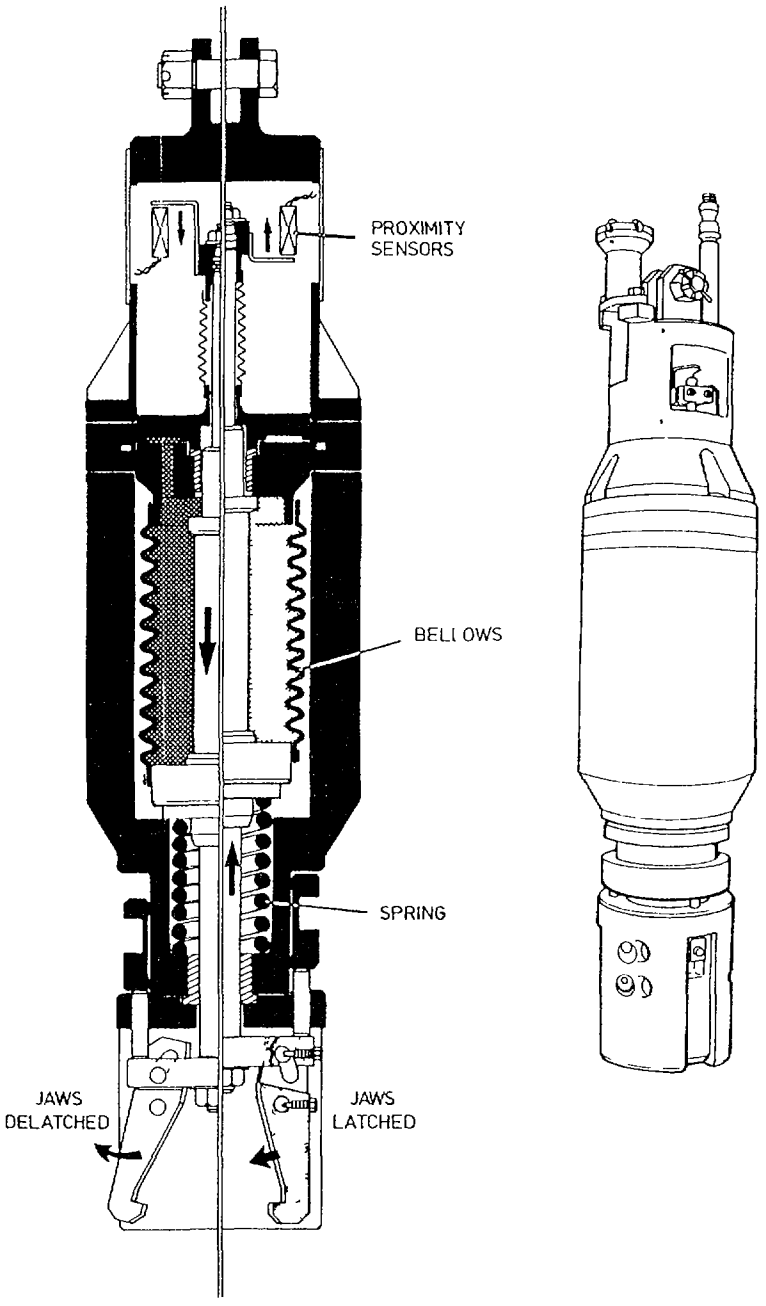


Figure 6 ARRANGEMENT OF BUFFER STORE PNEUMATICALLY OPERATED GRAB

17. Access to the inside of the can draining facility and the grab delatch position are via interlocked shielded doors that can only be opened when a subassembly is in its correct location in the facility. Figure 7 shows the arrangement of the draining facility and Figure 8 the drained liquid cleaning circuit.

The Spent Fuel Can

18. The spent fuel storage can consists of a 197 mm OD stainless steel tube with machined and welded on end fittings. The lower fitting has a central 20 mm OD draining hole that is sealed with a threaded plug and a gas filled metal ring. The top fitting has an internal thread and lip for carrying a metal 'O' ring seal and external features to engage a

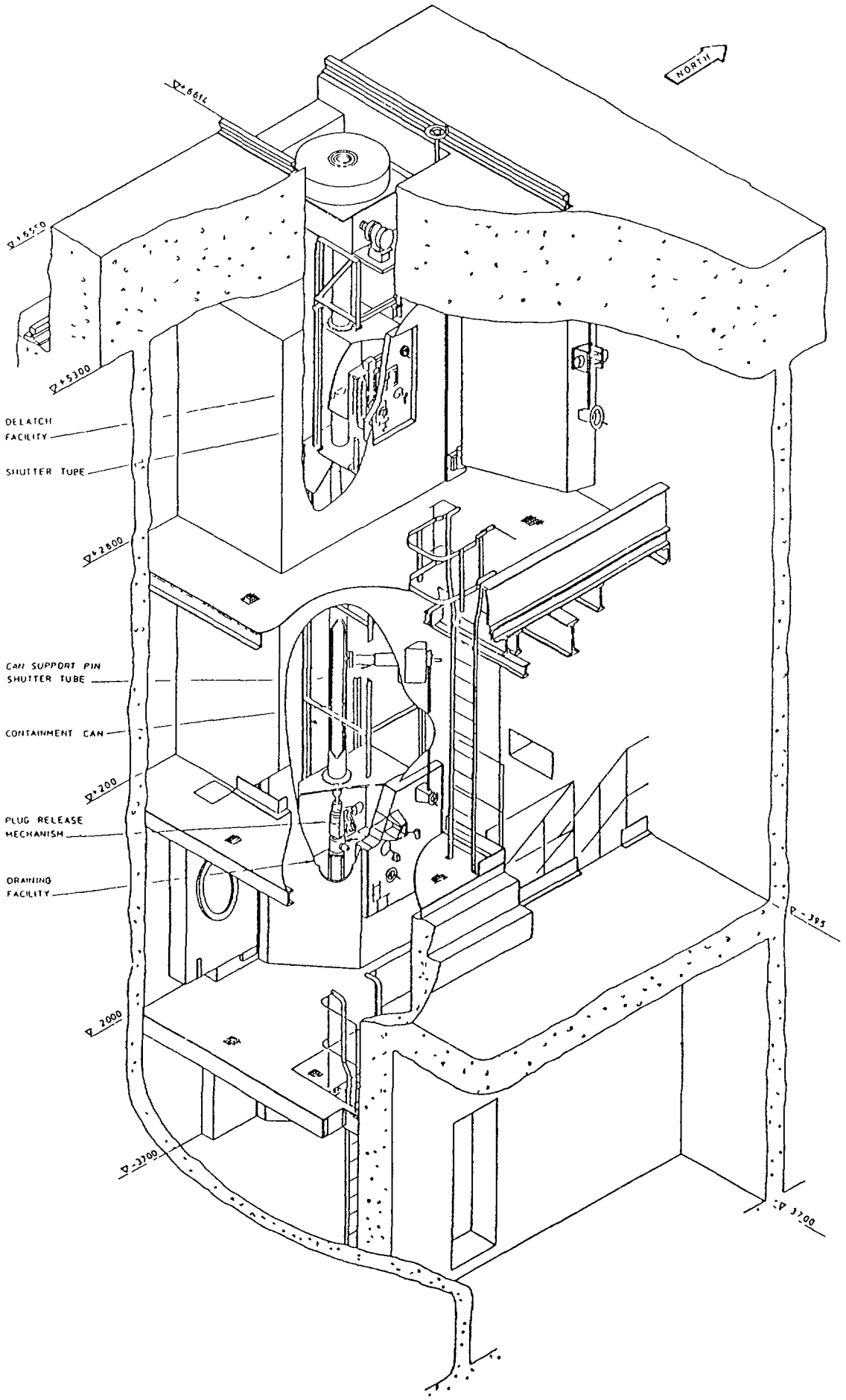


Figure 7 THE CAN DRAINING AND GRAB CHANGE FACILITY

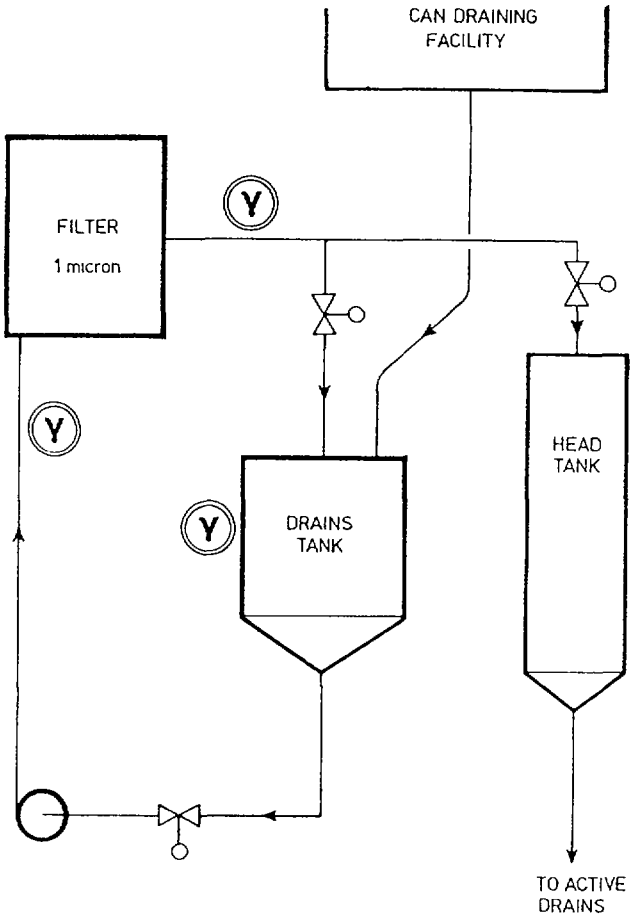


Figure 8 DRAINED WATER CLEANING CIRCUIT

lidding machine. A lid is located in the top of the can in the IFC and is sealed by a pneumatically operated lidding machine jack which holds the lid down against the 'O' ring seal. A retaining ring is screwed on to hold the lid down. The jacking load of 15 tonnes may then be removed allowing the compression load to be retained by the reaction of the screwed ring against the spring washers inside the lid. The spring washers were found to be necessary during the development stage of the lid and seal. Without the spring force to sustain the load, deformation of the can and screwed retaining ring caused during the load transfer from the lidding machine jack to the retaining ring allowed sufficient relaxation of the lid and 'O' ring to destroy the seal. On top of the lid is a handling feature in the form of a mushroom shaped pintel. Each can is identified with a unique alpha numeric code. A bevelled chamfer ring is fitted to each can after it has been lidded in the IFC to eliminate a sharp edged feature necessary for the lidding machine but which prevents the smooth horizontal handling of the can in the Dismantling Cave of the reprocessing plant.

Pond Cooling

19. The 148 m^3 of pond water is circulated via a pump through the pond cooling system designed to control the maximum water temperature between 30 and 35 ^{o}C. Water flow is from the pond above the baffle plate and the return is below the baffle plate being directed towards the storage holes via holes in the return header. Primary heat is given up to secondary water via a plate type water to water heat exchanger. The secondary water

is pumped through the heat exchanger and returned to twin evaporative water cooling towers having a heat dissipation capacity of 250 kW per tower.

20. The pond cooling system is physically arranged to satisfy two criteria. These are firstly to prevent accidental syphonic drainage and secondly to ensure that any primary water leaking or draining from pumps, valves and heat exchangers is channelled into a large catchment tray. All such water is carried into the PFR active effluent plant for processing. Figure 9 shows the schematic arrangement of the cooling system.

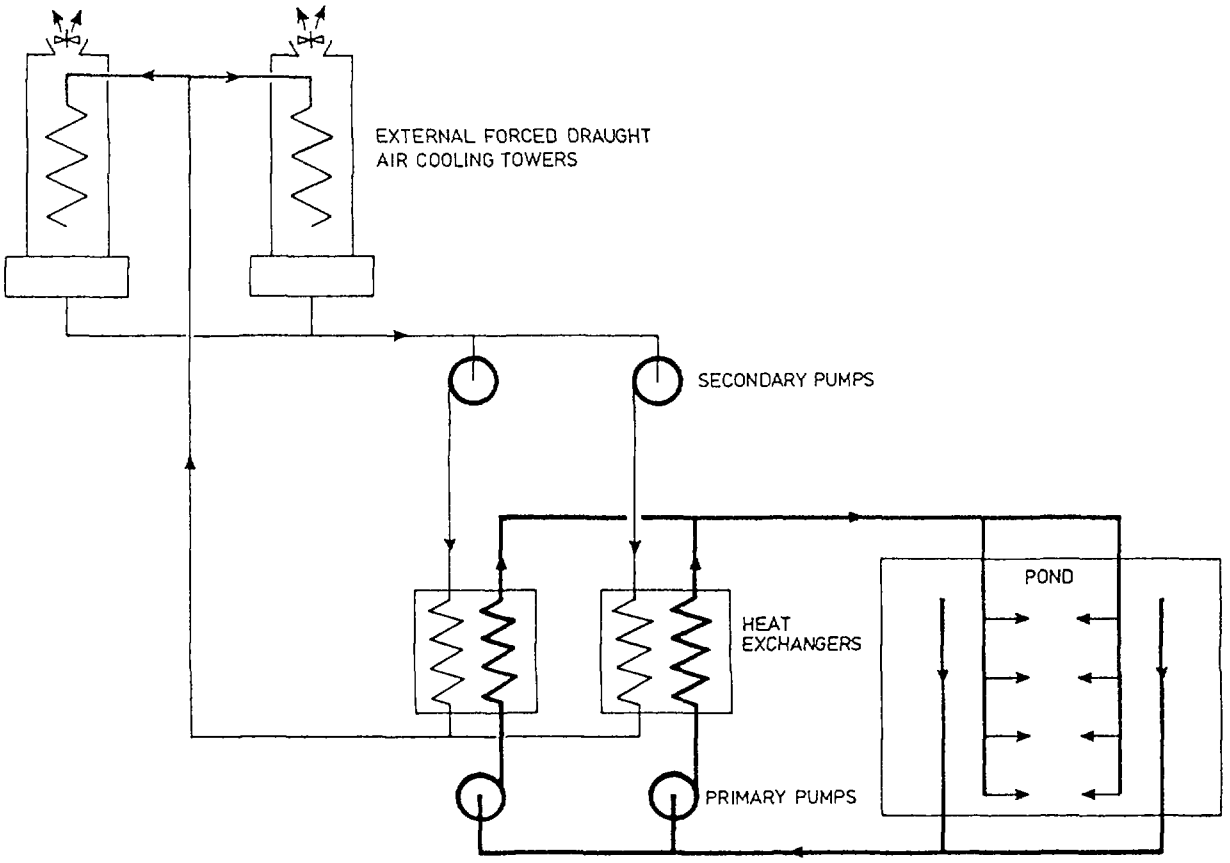


Figure 9 SCHEMATIC ARRANGEMENT OF THE POND COOLING SYSTEM

Building Ventilation

21. The design basis accident for the pond is postulated to be the rupture of two fuel pins in the building air space during transit between the flask and the pond. Such a release would not constitute a site or district hazard and for this reason High Efficiency Particulate Air (HEPA) filters are not installed. However the building can be sealed with conventional, not airlock, type doors. The exhaust ventilation system maintains a slight depression in the main pond hall, plant room and draining area to promote an air flow from non active areas into those of a higher hazard potential. Air is changed continuously at a rate of five times per hour and is exhausted from the building roof at a height of 16 metres above the ground. The building exhaust ducts are sealed by electro-pneumatic containment valves when radiation in the pond building reaches a pre-determined level of 200 micro Sieverts per hour.

22. The interior of the shielded grab change and draining facility is held at a depression of 35 mm WG by a dedicated two fan exhaust system running one on duty, the other on standby with automatic changeover controls. This system contains a replaceable HEPA filter to trap any airborne activity that may be released during can draining operations.

23. A separate extract system draws air from the transfer port valve areas to remove any active dust that may be present from the outer surface of the cans. If the building containment is sealed this fan automatically shuts down and its extract damper closes to complete the building seal.

24. The pond transporter shielded container has its own air extract system. When an item has been fully raised into the shielded container and the lower plate closed an extract fan is switched on to draw air out of the container and pass it through a HEPA filter. This design guards against possible activity release from contamination on the outside of the can or from a leaking can seal in a can containing 'ruptured' fuel.

25. The air exhaust rate from the building is almost matched to the inlet system air which is coarsely filtered. Inlet air can be heated from the station high pressure hot water system if operating conditions demand it.

26. The ventilation, cooling systems and containment sealing can be operated either locally or from the station central control room (CCR).

Activity Monitoring

27. There are three groups of activity surveillance equipment in the IFBS. One group comprises area gamma monitors working in the 10 micro Sieverts per hour to 10 Sieverts per hour range. The second group comprises gamma detectors which give localised monitoring in the range of 1 micro Sv per hour to 1 milliSv per hour and the third group of units are semi-mobile and can be plugged into several different locations to detect beta activity in air.

28. A signal from any one of the three area gamma detectors trips the fans and seals the buffer store containment. The trip levels are set at 200 microSv per hour.

Operating Experience

29. A total of 135 full cans have been successfully placed into the pond. Of these 81 containing fuel subassemblies have been successfully removed from the buffer store and sent to the reprocessing plant only six have leaked a maximum of a few ml. The only problems experienced during plant operation have been associated with the large sealing valves on the draining facility and the discharge ports and failure of the transporter grab to engage on a can on one occasion. The valve problem was common to both the discharge port and draining facility valves. Concrete dust residue from construction work had become dislodged by the dedicated ventilation extract system associated with the valves causing the heavily shielded valve gate to stick on its tracks. The valves are motor driven through a torque limiter and could not be closed. A hand drive facility is also available but this is not regulated by the torque limiter and during operation the large mechanical advantage from the hand drive caused the lead screw drive to be damaged. The hand drive has now been reinforced and the area thoroughly cleaned to remove the possibility of a recurrence of this problem. No further problems have been experienced.

30. After the completion of approximately 500 operations the pond transporter grab failed to open during a picking up operation. To effect a repair the grab was lowered through the open discharge port valve into a "controlled" area specially erected underneath the valve. The fault was traced to an electrical proximity sensor in the top of the grab that operates the gas valve to supply nitrogen to the grab bellows. Water had percolated through a seal and short circuited the winding preventing correct operation. The sensor and seals were renewed and tested for leak tightness before being reinstated.

31. Routine stocktaking and nuclear material inventory checks require the identity of each can held in the storage pond to be checked. The engraved lettering on the lid of each can can be viewed by using binoculars and a spotlight from the bridge of the transporter. However this is not totally satisfactory and a closed circuit television system is now used to identify the contents of each channel. Finely divided debris in the pond water is detrimental to clear visibility and it is sometimes necessary to wait for the dust to settle to prevent back scatter from the lighting. The dust also tends to fill in the engraved lettering and it is sometimes necessary to clean the can lids using a remote under water 'duster'.

32. The pond transporter is fitted with two other television cameras one to view coordinates marked on the pond wall and the other on the bridge to give approximate storage channel positions. A further camera is fitted to an intrascope on the snout to give accurate viewing of a marker adjacent to the targeted storage hole and used to enable accurate positioning ($\pm$ 5 mm) of the crab to take place. This system has worked very well and it has never been necessary to adjust the settings or repeat the approach to a storage position.

33. The spent fuel flask, which weighs 40 tonnes and is used to carry the canned sub assembly from the discharge port of the buffer store to the reprocessing plant, is moved between the under side of the discharge port valve to the flask road transporter by two pieces of equipment that have given trouble free operation. The first is the flask handling trolley which turns the flask from the horizontal plane to the vertical plane and jacks it up onto the discharge port valve to make a gas tight seal. The second is the flask handling carriage which transfers the flask between the flask handling trolley and the road transporter.

34. The total heat load from items stored in the pond has never yet elevated the water temperature above 30 ^{o}C and it has therefore not been necessary to operate the pond cooling system. With steady plant operation and more fuel and breeder subassemblies now being discharged through the fuel route the pond heat loading will increase and cooling will be demanded. There is a decay heat limit for each channel of 3 kW. The limit of 2.5 kW imposed for sodium draining in the IFC ensures that the higher value is never exceeded. No sodium bearing items are permitted in the buffer store pond.

Summary

35. Operation of the IFBS since it was commissioned in 1982 has been highly satisfactory. Although it has not been necessary to run the pond cooling system it is routinely tested. The building ventilation system has worked well and there have been no pond leaks. The transporter and flasks have all performed well as have the spent fuel containment cans of which 7% were found to have leaked a few ml's of water.

36. The well engineered buffer store allows a production line style of operation to be adopted for the PFR fuel route to the reprocessing plant.

SOME ASPECTS OF DESIGNING COMPACT SPENT WWER FUEL STORAGE

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Abstract

This paper discusses some possible ways of compact storage of spent fuel assemblies of WWER-440 and WWER-1000 reactors in cooling ponds.

Application of boron steel cans permit to increase the capacity of cooling ponds for WWER-440 and WWER-1000 spent fuel assemblies by a factor of 2 as compared to the design capacity.

The account of the actual isotope composition of spent fuel can permit a more compact arrangement of spent fuel assemblies in a pool and serve as an alternative or addition to special heterogenous absorbers to achieve the maximum usage of a pool capacity. The calculated coefficient of compactness of cooling ponds at WWER site is equal 1,27 for WWER-440 and 1,66 for WWER-1000 spent fuel assemblies.

At present on spent nuclear fuel (SNF) reprocessing it became necessary to have a compact arrangement of spent fuel assemblies (SFA) in existing cooling ponds at a NPP site and to construct a new long-term storage.

The cooling pond capacity can be increased only through a reduced pitch between SFA. The major problem in the design of a compact storage is to ensure nuclear safety. Heterogeneous neutron absorbers, e.g., boron steel sheets, are conventional and already used in some countries for a compact storage of SFA. The accomodation of WWER-440 and WWER-1000 SFA in hexagonal canning tubes made of boron steel with 1% boron content makes it possible to achieve a coefficient of compactibility of about 2.

Despite the feasibility of a reasonably high coefficient of compactibility the indicated method has several disadvantages. Boron steel has poor wedability which leads to a complicated design and is relatively expensive. Its corrosion resistance is to be

ensured up to 100°C; it is necessary to control the boron content of steel during manufacture and operation, boron distribution, the accuracy of determination of its content in the process of boron steel canning fabrication. The nuclear safety of a storage will depend on the fulfilment of the above requirements.

Therefore, attention should be paid to a way of a compact storage of SFA taking account of the spent fuel burn-up that does not require any additional expenses.

The tabulated results of calculations show that the account of the actual isotope composition of spent fuel can permit a more compact arrangement of SFA in a storage and serve as an alternative or addition to special heterogeneous absorbers to achieve the maximum usage of a storage capacity.

TABLE: SAFE PITCH OF FA ARRANGEMENT AND COEFFICIENT OF COMPACTIBILITY OF COOLING PONDS AT WWER SITE

FA type	Lattice pitch at K_{eff}=0.95		Coefficient of compactibility	
	Fresh fuel	Spent fuel	Theoretical	Practical
WWER-1000	380	310	1.5	1.66
WWER-440	220	200	1.21	1.27

The tabulated values of the pitch for spent fuels were calculated assuming the following burn-up: WWER-1000 - 40 MW.day/kg, WWER-440 - 30 MW day/kg. The practical coefficient of compactibility is determined from comparison of the area accounting for SFA with a safe pitch and a pitch adopted in the storage design; the theoretical coefficient of compactibility is determined in a similar way based on a safe pitch fresh and spent fuel.

One more possible way of increasing the capacity of SNF cooling ponds and storage is to use stainless steel tubes that are structural materials of cans and absorbers at the same time.

The design of a compact storage taking account of a spent fuel burn-up entails the following problems.

First, to ensure the nuclear safety of storage for SFA having a burn-up less than the design one. This problem can be solved by accomodating a portion of SFA with a pitch corresponding to the one permissible for fresh FA or by equipping some cells with boron steel cans.

Second, to rule out the possibility of achieving criticality at the low density of water. It is shown in/ 1/ that at the low density of water (0.05-0.4 g/cm^3) the maximum multiplication factor is observed for a large storage; its value can significantly exceed a unity at the pitches of FA arrangement at which K_{eff} =0.95 with a storage fully flooded with water. However, despite the theoretical possibility of $K_{eff} > 1$, in each particular instance it is necessary to consider the actual storage design and real situations that can take place. For spent fuel, e.g. for WWER-440 storage of a cooling pond type under normal pressure, the possible decrease of water density is within 1 to 0.958 g/cm^3 (0,01°C - 100 °C) or less than 0.01 g/cm^3 (saturated steam). For the above water densities K_{eff} remains less than 1 even without taking account of burn-up and structural materials. In a WWER-1000 spent fuel storage where water is under pressure it is theoretically possible to achieve critical steam conditions. In this case the density of water can vary from 0 to 0.286 g/cm^3 which is dangerous for rather a large storage. However, in this case too one should consider real situations to draw conclusions on possible boiling and reaching the above values of water density. The solution of the second problem is to rule out the possible decrease of water density. e.g., using a water make-up system. This method is mostly preferred for a long-term storage where heat release is not high.

The third problem is to control the burn-up of SNF for a spent fuel storage. This problem can be solved using a non-destructuve assay developed in the USSR. The non-destructive assay is based on recording the intrinsic neutron /2/ and gamma-radiation of SFA /3/.

Thus, on the basis of the results of the investigations the following conclusions can be drawn:

- as compared to the design capacity boron steel cans permit a factor of 2 increase of the capacity of cooling ponds for WWER-440 and WWER-1000 SFA;

- it is possible to design a long-term compact storage for spent WWER fuel taking account of the actual isotope composition, use of absorbers in the form of structural steel tubes and arrangement of control of SFA burn-up;

- the list of accident conditions is required that must be considered in cooling pond design;

- in the design of a water storage technical measures must be provided that rule out a possible decrease of water density to less than 0.95 g/cm^3 under normal and accident conditions.

REFERENCES

1. F.M.Alcorn. Criticality evaluation of low-density moderation in PWR fuel storage. TANS, v.27, 1977.
2. Fedotov P.I. et. al. Proceedings of 3 rd Annual Symposium on Safeguards and Material Management, Nuclear Material Management, - Karlsruhe, ESARDA, 1981, p.105.
3. Bibichev B.A. et al. Atomnaya Energiya, 1982, v.53, is.4, p.22.

EXPERIMENTAL METHODS DEVELOPED IN SUPPORT OF SPENT FUEL SURVEILLANCE PROGRAMMES

(Session II)

Chairman

E.A. VITIKAINEN
Finland

INTRODUCTION OF THE IAEA/BEFAST-II CO-ORDINATED RESEARCH PROGRAMME: RESEARCH SUBJECTS, PARTICIPATING ORGANIZATIONS, SCHEDULE

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Abstract

Extended spent fuel storage is an important activity for all countries with nuclear programmes because fuel after its discharge from the reactor is required to be stored before reprocessing or final disposal. The storage period is highly dependent upon the individual national strategies to complete the nuclear fuel cycle.

The results and experience gained in the co-ordinated research programme "BEFAST" (Behaviour of Spent Fuel Assemblies during Extended Storage), which was performed from 1981 to 1986 under the auspices of the IAEA, were so important for the participating countries that the IAEA already in September 1986 initiated this new follow-up programme "BEFAST-II" (Behaviour of Spent Fuel and Storage Facility Components during Long-Term Storage).

Together 16 organizations from 13 countries (Argentina, Canada, Finland, Germany, F.R., GDR, Hungary, Italy, Republic of Korea, Japan, Sweden, UK, USA and USSR) joined the programme.

The research activities going on in each participating country can be grouped under three major topics: (A) long-term behaviour, (B) surveillance, and (C) facilities & operation, which also form the basis of the BEFAST-II final reort "Research, Development and Practice for the Extended Storage of Spent Fuel".

The BEFAST-II CRP will be carried out during the years 1987-1991, so that the final report could be completed by the end of 1991.

The first Co-ordinated Research Meeting will be held in April 1988 in Budapest, Hungary. Two other CRP meetings are preliminary planned to be held in September 1989 and March 1991. The working procedures of the BEFAST-II CRP will be agreed upon during the Budapest meeting.

The general goal of the programme is to provide a forum for free information exchange on spent fuel matters between specialists in member countries.

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1 INTRODUCTION

Extended spent fuel storage is an important activity for all countries with nuclear power programmes because fuel after its discharge from the reactor is required to be

stored before reprocessing or final disposal. The storage period is highly dependent upon the individual national strategies to complete the nuclear fuel cycle.

This is why 11 organizations from 10 countries (Austria, Canada, CSSR, Finland, GDR, Hungary, Japan, Sweden, USA and USSR) joined a co-operative research programme "BEFAST" (Behaviour of Spent Fuel Assemblies During Extended Storage) under the auspieces of the IAEA for exhange of information on the spent fuel storage technology. The programme was performed from 1981 to 1986 and the information collected covered the following aspects:

- potential fuel degradation mechanisms during storage;
- spent fuel examination and surveillance programmes;
- impact on storage equipment.

The final report of the programme /1/ was published after the final Research Co-ordination meeting which was held in Leningrad, USSR, in May 1986 /2/.

The unanimous opinion of all BEFAST members was that the co-operation and the results obtained in the programme were important for each participating countries and the meeting recommended to continue this efford during the next five years (1986 -1991).

The new follow-up programme, "BEFAST-II, Behaviour of Spent Fuel and Storage Facility Components during Long-term Storage", was initiated by the IAEA in September 1986 and 16 organizations from 13 countries (Argentina, Canada, FRG, Finland, GDR, Hungary, Italy, Republic of Korea, Japan, Sweden, UK, USA and USSR) joined the co-operative research programme (CRP).

2 GOALS AND MAJOR TOPICS OF THE BEFAST-II PROGRAMME

Together 8 subjects were initially proposed to be covered by the BEFAST-II CRP: (i) Storage experience; wet/dry,

(ii) Spent fuel monitoring, (iii) Effects of decontamination and cleaning on materials, (iv) Handling and transport of spent fuel after storage, (v) Storage of defected fuel, (vi) Storage of fuel exceeding 100 a, extrapolation of present experience, (vii) Predictive models: Failure mechanisms, Material aspects, and (viii) Crud impact on spent fuel integrity.

In order to define how to improve the performance of the BEFAST-II CRP and accomodate it to needs and request of participating countries as well as to find its place among other IAEA activities in the nuclear fuel cycle, a consultants' meeting was held in October 1987 in Erlangen, FRG.

This consultants' meeting (consultants from FRG, USA, USSR and Finland as well as the IAEA's representative) discussed the activities which are going on in different countries under the BEFAST-II research agreements and decided that these could be grouped under three major topics: (A) Long-term behaviour, (B) Surveillance and (C) Facilities & Operation, which could also then form the basis of the BEFAST-II Final Report "Research, Development and Practice for the Extended Storage of Spent Fuel".

This final report thus is one of the goals of the CRP . The other goals are to provide a forum for free information exchange on spent fuel storage matters between specialists in the member countries.

Distribution of BEFAST-II research subjects within the proposed major topics is shown in Table 1 , and a proposal for the new research matrix is shown in Table 2.

TABLE 1. PROPOSED RESEARCH ACTIVITIES WITHIN THE THREE MAJOR TOPICS

1 MAJOR TOPIC	2 A LONG-TERM BEHAVIOUR	3 B SURVEILLANCE	4 C FACILITIES & OPERATION
SUBITEMS	*Materials aspect (claddings & components) *Degradation mechanisms and Models *Validation -experimental -experience	*Monitoring; Wet/Dry -environment -components -fuel assemblies -dose rate of workers *Fuel Conditions -operational -fabrication technology -defected fuel rods and assemblies *Different Reactor Types	*Dose Rate Reduction *System Performance *Changing Modes wet - dry *Capacity Enhancement -high density racks -reracking -double tiering -doped coolant -rod consolidation *Handling of Heavily Damaged Fuel

TABLE 2. PROPOSAL FOR THE NEW BEFAST-II RESEARCH MATRIX

1 SUBJECT / COUNTRY	2 A LONG TERM BEHAVIOUR: MATERIALS	3 A LONG TERM BEHAVIOUR: MECHANISMS & MODELS	4 A LONG TERM BEHAVIOUR: VALIDATION	5 B SURVEILLANCE: WET	6 B SURVEILLANCE: DRY	7 C FACILITIES & OPERATIONS	D OTHER
ARGENTINA				-wet storage experiments			
CANADA	====>	====>	====>	====>	====>	-wet storage of spent fuel 26a (1988) -dry: ERB, CEX-1, -2 (concrete canisters)	
FEDERAL REPUBLIC OF GERMANY	====>	-life-time prediction methods for dry-stored fuel -computation of spent fuel rod behaviour	-review of available dry storage data			<====	
FINLAND	====>		-wet storage experience	-periodical examinations			
GERMANY DEMOCRATIC REPUBLIC		====>				-verification of temperature calculation models in dry casks	
HUNGARY	-investication of local corrosion processes in Al and ZRYs			-corrosion monitoring methods		-compact storage in water pools (collection of basic data)	
ITALY				====>		-handling and transport of LATINA spent Magnox fuel	
REPUBLIC OF KOREA		-models for corrosion product concentration in water pools	-water pool storage experiments	-periodical examinations -controlled storage of defected fuel in water pools			
JAPAN	-oxidation of ZRY in dry storage -oxide properties			-pool component behaviour -periodical measurements of chemistry in the pools			
SWEDEN	====>			====>		-examination of damaged spent fuel -dissolution and leaching processes	
UNITED KINGDOM	-oxidation of UO_2 in dry storage	<====>	-wet storage experience	-monitoring of wet storage		<====	
USA	====>	====>	-wet storage experience	<====	-dry storage program	<====	
USSR	====>	====>	-wet storage experience	<====		<====	

3 SCHEDULE AND WORKING PROCEDURES

Together three co-ordinated research meetings will be held in the course of the programme: The first in April 11-15, 1988 in Budapest, Hungary; The second in September 1989 and the third in March 1991 (exact dates and places will be determined later). The final report of the programme is planned to be published by the end of 1991.

The working procedures of the BEFAST-II CRP will be agreed upon during the Budapest meeting.

The consultants' group has recommended that the participants will be devided in three working groups responsible for correlating the material to be presented at the meetings for each major topic and preparing the corresponding sections of the final report. These working groups will meet in connection with the CRP meetings and, if needed and possible, also once between the CRP meetings. Other information between the meetings will be exchanged by mail through the chairman of the CRP.

REFERENCES

/1/ Behaviour of Spent Fuel Assemblies during Extended Storage, Final Report of the Co-ordinated Research Programme BEFAST, Phase 1 (1981 - 1986), Sponsored by the IAEA, IAEA-TECDOC-414, Vienna, 1987.

/2/ Long Term Wet Spent Nuclear Fuel Storage , Proceedings of a Technical Committee Meeting on Behaviour of Used Fuel Assemblies and Storage Equipment at Long-term Wet Storage Conditions, Leningrad, USSR, 26-30 May, 1986, IAEA-TECDOC-418, Vienna, 1987.

SPENT FUEL SURVEILLANCE AT BNFL, SELLAFIELD NUCLEAR REPROCESSING PLANT

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Abstract

A wide range of irradiated nuclear fuels are stored in ponds in BNFL, Sellafield, UK, prior to reprocessing. Magnox fuel from the first UK power generating reactors has been stored and reprocessed since the mid 1950's, and stocks of irradiated AGR (Advanced Gas-cooled Reactor) and Water Reactor fuels are increasing prior to the start of reprocessing oxide fuel in the THORP (Thermal Oxide Reprocessing Plant) facility in the early 1990's.

The methods of surveillance for each type of fuel are described. Underwater visual examination in ponds and more detailed examination in shielded caves are carried out. Limited surveillance is necessary for Magnox fuel as storage periods are short. For AGR fuels with longer storage periods the retrieval and element breakdown are described, followed by detailed metallurgical examination of pins and element components.

The effect of water chemistry on corrosion performance of irradiated AGR fuel during pond storage is illustrated. The methods of confirming satisfactory water storage of water reactor fuel are outlined. The major results of surveillance of each type of fuel are summarised.

The organisation of the work programmes is given. The paper concludes that, mainly through monitoring and surveillance, backed by experimental programmes optimum storage regimes have been developed for the various types of fuel, consistent with planned storage periods prior to reprocessing.

Introduction

1 Irradiated nuclear fuel has been stored, prior to reprocessing, at the Sellafield, UK, site since the mid 1950's (see Figure 1). The first UK power generating reactors were gas cooled and used natural uranium fuel rods sheathed in Magnox cans. Over 1000 te of Magnox fuel are reprocessed annually, and fuel is stored for limited periods in ponds treated chemically to limit corrosive attack of the cladding.

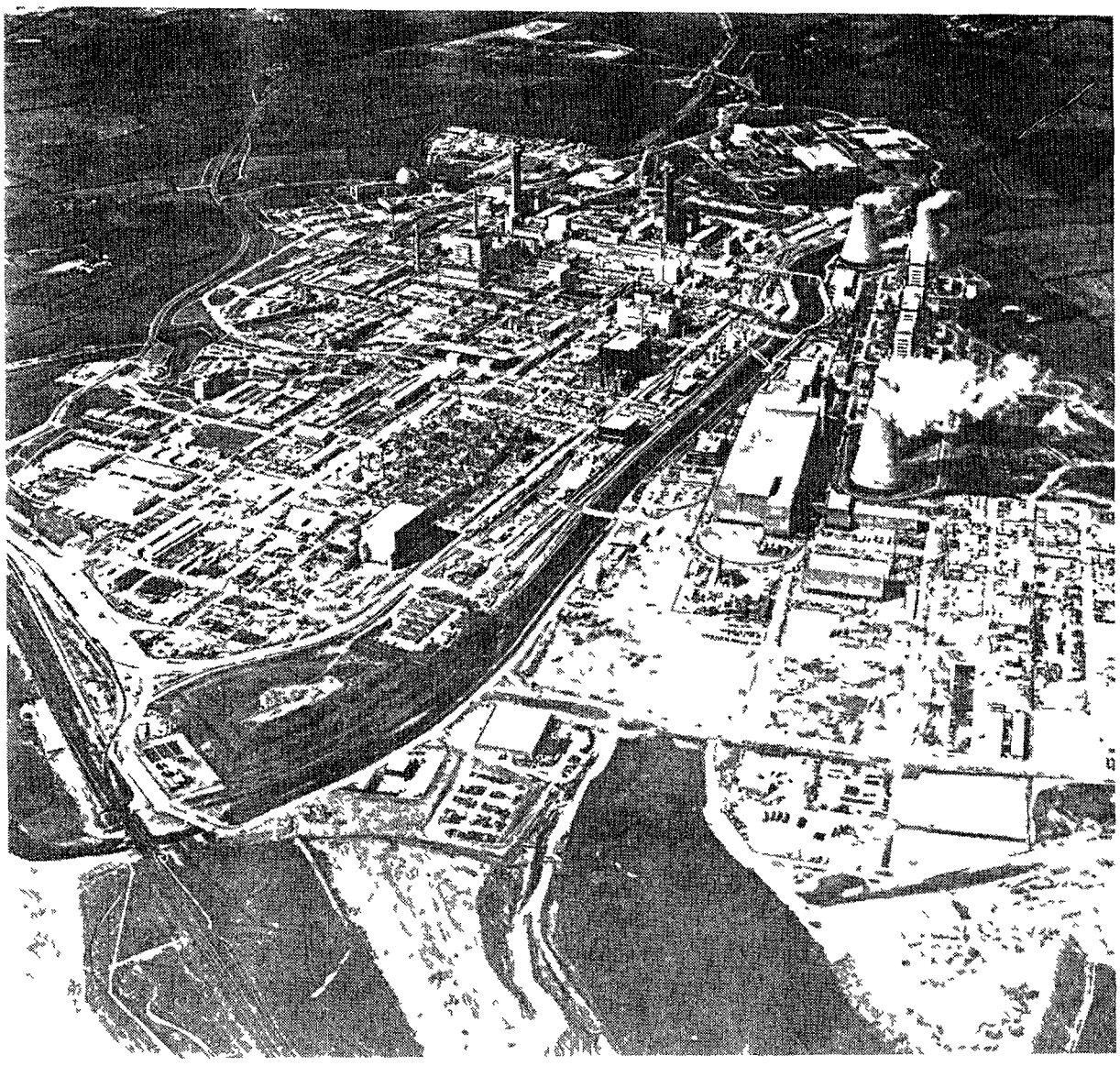


FIG 1 BNFL, Sellafield.

2 The next UK power generating reactors are the AGR (Advanced Gas Cooled Reactor) of which five stations are operating to date. In order to improve thermal efficiency, operating temperatures are higher than in Magnox reactors, therefore fuel consists of enriched uranium dioxide fuel pellets in stainless steel cans. Fuel from AGR power stations has been received at Sellafield since 1978 for storage in ponds prior to reprocessing in the new THORP (Thermal Oxide Reprocessing Plant) facility due to be operational in 1992.

3 Much of the planned throughput for the first 10 year campaign of THORP is water reactor fuel, mostly from outside the UK. This fuel includes PWR, BWR and SGHWR types. Water reactor fuel has been received at Sellafield since 1968 for storage underwater prior to reprocessing.

4 As can be seen from above, a wide range of irradiated fuel has been and is currently being stored at Sellafield [1]. Comprehensive programmes have been instituted to monitor the condition of fuel to endorse continued suitability for storage until reprocessing, including any interim handling or rod consolidation.

5 This paper describes the surveillance methods used, and the results obtained for each particular type of fuel. In addition the methods of general monitoring of ponds are outlined, together with the organisation and management of the programmes.

Magnox Fuel Surveillance

Background

6 "Magnox" fuel is named after the canning alloy used, which consists of Mg 99.2%, Al 0.80%; this has a low neutron capture cross section, as required for use with natural uranium fuel. Figure 2 shows various designs of Magnox fuel element. Magnesium is a strongly reactive metal

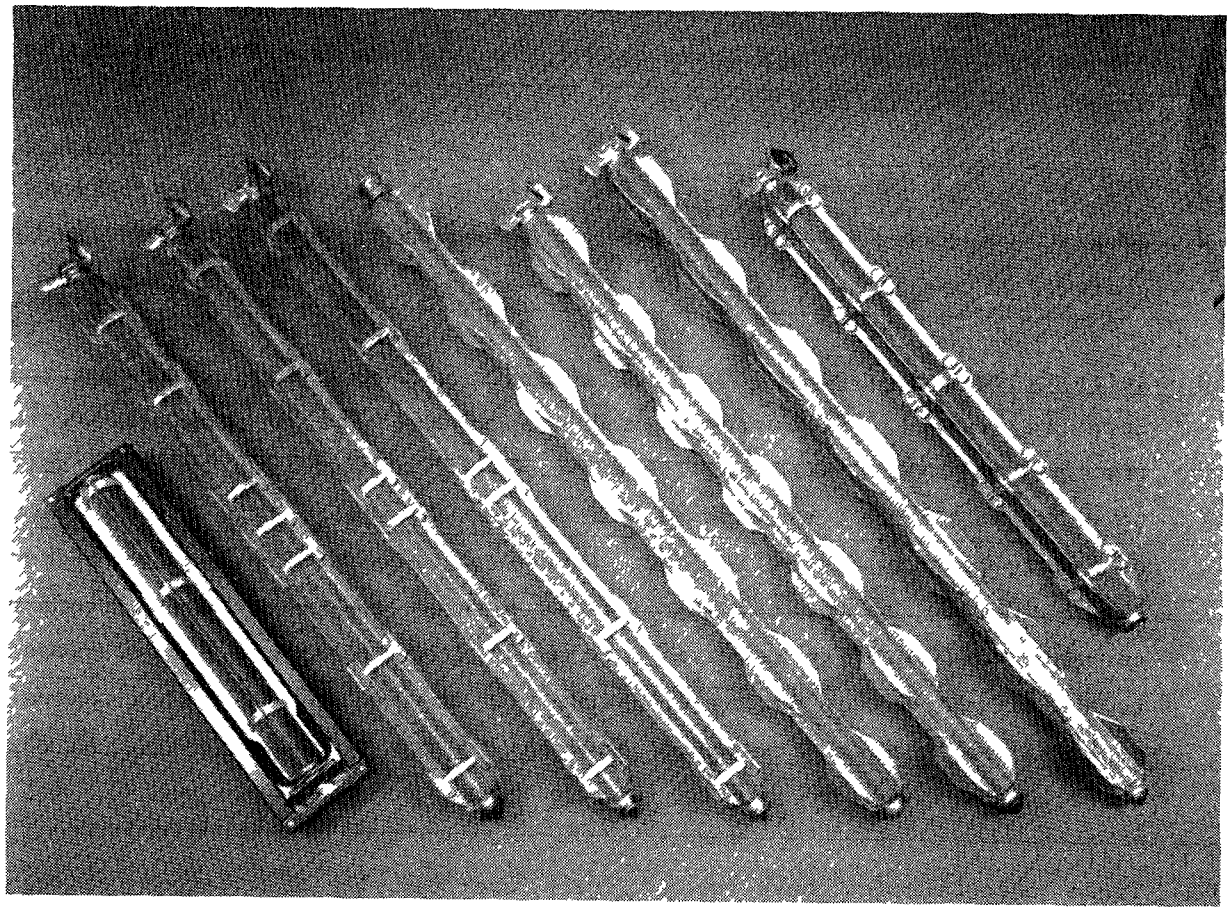


FIG.2. Representative designs of Magnox fuel element.

and is susceptible to corrosion attack in neutral water. This problem can be alleviated by increasing the pond water pH by NaOH dosing and minimising the aggressive ion concentrations, especially chloride and sulphate.

7 Practical considerations, arising from carbon dioxide absorption, tend to limit the pH of bulk pond water exposed to the atmosphere to a pH of approximately 11.5. Recently, to allow for more extended storage (for example in the event of outage of reprocessing plant), BNFL has adopted a more inhibitive chemistry of pH13, Cl^- + SO_4^{2-} less than 0.5ppm, with skips of fuel stored inside closed containers isolated from the bulk pond water. This method of storage is now used in the FHP (Fuel Handling Plant) at Sellafield.

Surveillance Methods

8 Monitoring of Magnox fuel stored in open skips in at-reactor ponds at pH 11.5 has been carried out by the Power Station utilities when required using their own cave facilities or those at Sellafield. Elements are extracted from skips in the Reactor Station ponds and examined visually, radiographically and then destructively in shielded PIE (Post Irradiation Examination) facilities.

9 Fuel stored at Sellafield has not been subject to routine surveillance, however inspection of the fuel element is possible during the decanning process, Figure 3.

10 Activity release from Magnox fuel elements normally occurs only as the metallic uranium fuel corrodes, and the release rate is proportional to the uranium corrosion rate. Therefore, bulk pond water analysis is able to show the presence of exposed uranium in the pond. This method, used on early ponds at Sellafield, does not necessarily indicate can penetration by Magnox corrosion since uranium exposure, and therefore corrosion and activity release, may result from mechanical damage caused to the element either during discharge from the reactor or during removal of splitters or lugs (components external to the fuel can itself) in the reactor pond to improve packing density during transport.

11 These uncertainties have been partly overcome by the latest storage method at Sellafield using containers isolated from the bulk pond water.

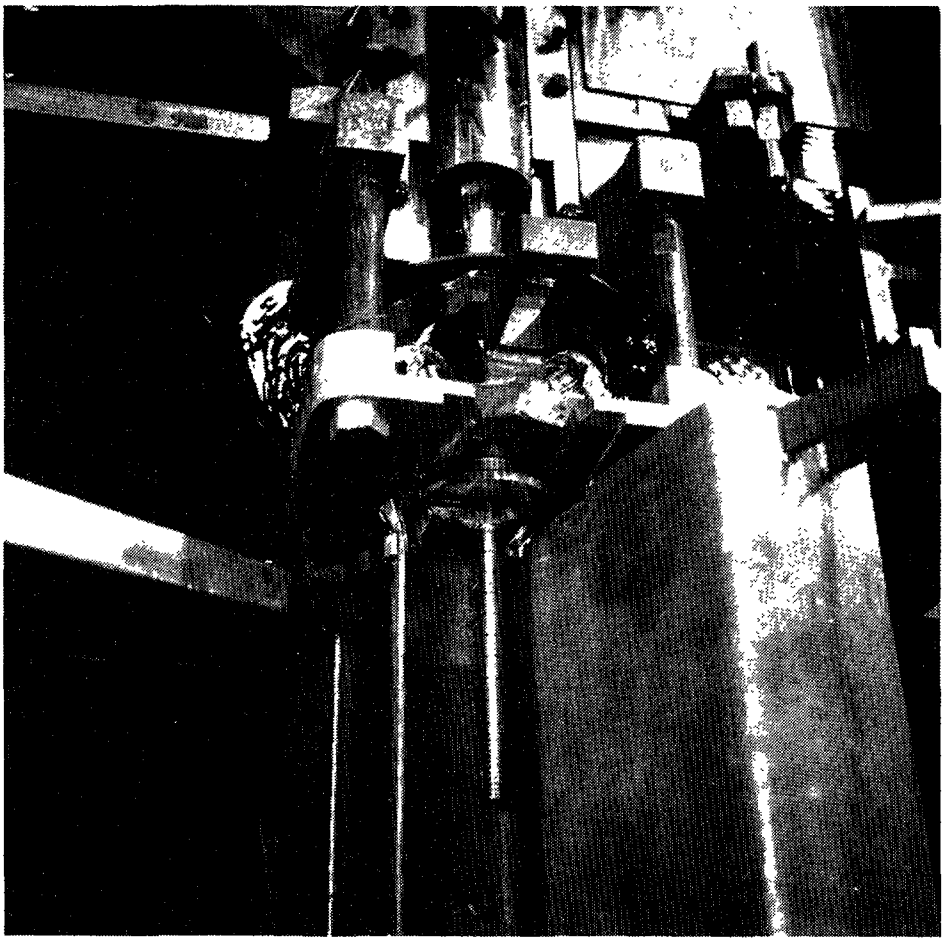


FIG.3. Magnox decanning facility, fuel handling plant.

Now it is possible to measure caesium releases directly for individual skips of fuel, compare these with fuel from different stations, and relate any activity release to handling damage and other factors. The new decanning caves in the FHP also provide a superior viewing capability to inspect elements during decanning.

Results of surveillance

12 Inspection of the fuel, either in post irradiation examination caves or during decanning, following pond storage for periods of up to approximately 2 years usually show little corrosion, Figure 4. Corrosion tends to be increased by the presence of Magnox corrosion product sludge $Mg(OH)_2$ settling on the finned element surface. The rates of general corrosion are low enough (0.1 mdd ≡ 2um/yr) to withstand penetration through the 2mm thick can wall for several hundred years, but much more rapidly penetrating localised attack tends to initiate at fin roots, Figure 5. Although the attack may spread around the can, the rates of

FIG.4. Moderate corrosion of a Magnox can after storage at pH 11.5.

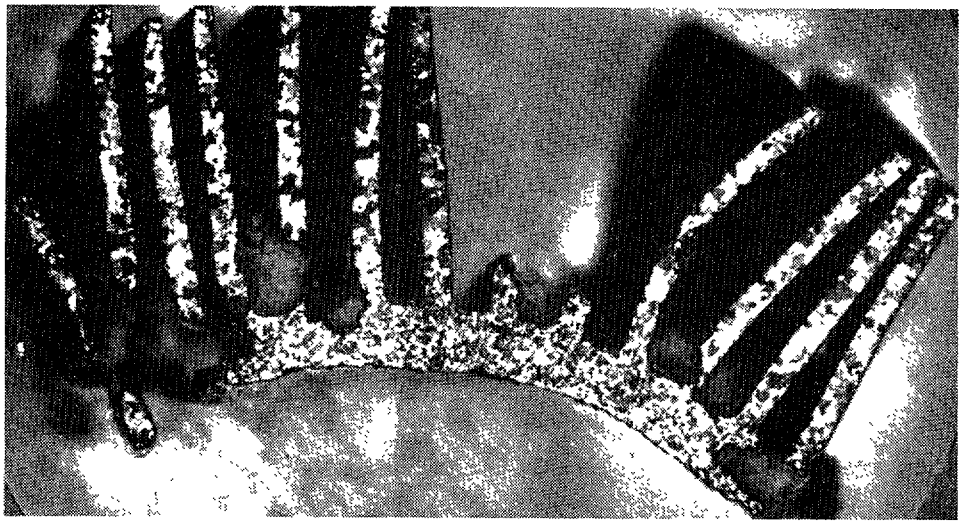


FIG.5. Section through a Magnox can showing localized attack at fin roots.

localised attack are usually low enough to avoid can penetration within two years. Nevertheless, there are failures and for this reason Magnox fuel is not routinely stored for periods longer than 1 - 2 years.

13 Storage, for more than 1 year, has occasionally allowed can penetrations to occur exposing uranium, whose corrosion was detected in the early ponds monitoring programmes. This also showed that small deviations from the specified pond water composition also may cause more rapid localised corrosion.

14 Fuel from the more inhibitive regime in the FHP, inspected in the decanning facility, has been in excellent condition. It is observed that any corrosion which initiated in the pH 11.5 regime (at-reactor pond) has been stopped in the pH 13 low Cl^- + SO_4^{2-} conditions. Monitoring of hydrogen evolution from inside FHP containers also confirm that the corrosion rate of Magnox is low (0.1 mdd or less); work is continuing to correlate the caesium release with mechanical damage.

Summary of Magnox

15 Surveillance of the condition of Magnox fuel has allowed the benefits of more protective pond storage regimes to be realised. From monitoring and laboratory experiments, it has been possible to develop an aqueous storage environment where sound Magnox fuel elements could be stored for 5 - 10 years, without significant degradation, if the need arises.

AGR

Background

16 The original predictions and surveillance of pond storage behaviour of irradiated AGR fuel showed that the 20 Cr/25 Ni, Nb stainless steel cladding could be prone to localised attack in neutral water. Extensive research investigations have found that some cladding becomes sensitised by a radiation induced segregation (R.I.S) mechanism in addition to any thermal sensitisation. These phenomena can leave the grain boundaries depleted of chromium to such a level that the material is susceptible to intergranular corrosion. The range of irradiation temperatures causing the sensitisation is 360 - 520^{o}C. In the reactor channel of 8 elements, conditions are such that the mid-pin position of element 1 (the lowest element, nearest to the coolant gas inlet) and pin ends of elements 2 and 3 may be sensitised. In addition the pin support braces in elements 3, 4 and 5 could be similarly affected. Figure 6 shows an element with support components.

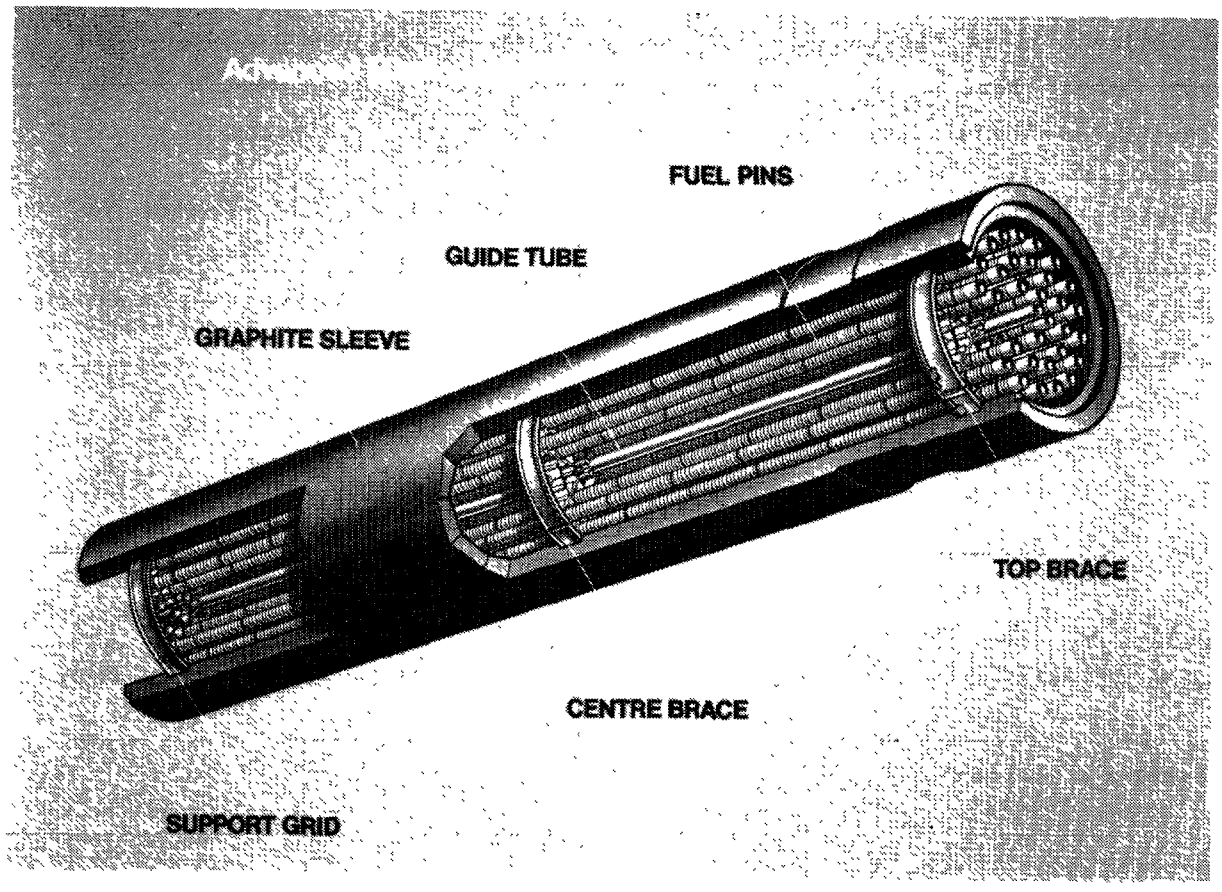


FIG.6. AGR element.

17 Since the 1977 public inquiry into the proposed construction of THORP and especially since the susceptibility to in-pond corrosion was recognised, a comprehensive fuel surveillance programme has been carried out on fuel stored in the original AGR ponds and more recently in the FHP alongside Magnox fuel at Sellafield. The Central Electricity Generating Board (CEGB) and the South Scotland Electricity Board (SSEB) separately monitor fuel behaviour in at-reactor ponds. The original AGR ponds at Sellafield are operated at pH7 with a demineralised water purge and chloride levels up to lppm because they are open to the atmosphere. AGR fuel storage in the FHP differs from Magnox in that storage is in flooded containers ie. allowing some mixing between container and bulk pond waters. Thus storage of AGR fuel in FHP is at pH 11.4 with low chloride (0.05ppm).

Monitoring Methods

18 The major methods of monitoring AGR fuel have been by visual examination in pond, and detailed examination including dismantling and metallography in shielded facilities.

19 Like Magnox fuel, AGR fuel is stored in at-reactor ponds before transport to Sellafield. Periodic underwater surveys have been carried out by the CEGB and the SSEB in the ponds.

20 At Sellafield more extensive examinations of elements are performed. Elements are selected from detailed records of reactor and storage histories, and transferred by shielded flask to the PIE facility. The elements are given a very thorough visual examination, which records a standard set of features including the condition of all the element components, Figure 7. If the element support components are the main area of interest, the element is dismantled by pushing the pins out of the element supports and breaking off the graphite sleeves. The grid and braces are then revealed for closer examination and are mechanically tested. Anmalysis of the response of the component to the mechanical test enables the behaviour of the component in subsequent dismantling to be predicted, see para 23.

SUMMARY OF PIE OPERATIONS ON POND STORED CAGR FUEL

	OPERATION	REASON FOR OPERATION
1	Visual examination of fuel assembly prior to dismantling	To establish extent of corrosion and distribution of corrosion products on fuel pins, braces and guide tubes (Pin support components)
2	Dismantling Operation	Fuel pins impacted clear of bottom grid and examined on removal from assembly. Pin support components examined for signs of degradation during dismantling
3	Detailed visual examination	To establish extent of attack on pins and pin support components.
4	Mechanical testing of grid and braces	To assess strength of components after irradiation and storage.
5	Fuel pin leak testing	Pins leak tested (nitrous oxide technique) to confirm pins intact.
6	Metallographic Examination	Optical Metallography carried out on sections of fuel element assembly to assess type and extent of attack.
7		Note: Some corrosion testing (Strauss) carried out on pin sections to confirm susceptibility to corrosion (sensitisation).

FIG.7. Post-storage examination procedure for pond stored AGR fuel.

21 Alternatively, if the condition of the fuel pins themselves is the main concern, the graphite sleeve is cracked and removed from the element, exposing the outer ring of 18 pins for closer examination. Details such as pin surface condition, degree of surface deposition and spalling are recorded. After complete dismantling of the element, a selection of pins is made for further mechanical testing, leak testing and metallography. Mechanical testing may involve tensile tests and/or drop tests to simulate operations in the production dismantler. Leak tests are carried out on a purpose-built rig using a nitrous oxide sniffing technique. Metallography includes Strauss tests to assess the degree of sensitisation across the can wall and progressive metallography at regions of interest, for example any leak sites or regions showing corrosion deposit.

22 Because AGR fuel is stored in containers in the FHP, individual skips (1 container holds 1 skip equivalent to 0.86 te U) can be monitored for activity release. Although the container is flooded, the exchange rate between container and bulk pond water is small.

23 In order to reduce storage volume, the Sellafield production dismantler has recently been commissioned, so it is now possible to monitor the condition of elements during the dismantling process in which pins are removed individually from the element. Records are kept of any abnormal features, such as excessive pin removal forces. This is important as the individual pins will be stored for a further period underwater, prior to reprocessing in the THORP plant. There is the intention to monitor the condition of dismantled fuel in the future.

Results Of Surveillance for AGR Fuel

24 As extensive monitoring and surveillance of AGR fuel has taken place, only the major results are described below.

25 Metallography of fuel element braces have revealed corrosive attack in the crevice regions of braces that have become sensitised in reactor, Fig 8. Considerable variability in the extent of attack has been found and this variability is thought to be materials related, possibly caused by small differences in composition. The above are observations made on fuel stored in the original storage ponds at Sellafield. Because of improved water chemistry in the FHP significant reductions in the extent of brace corrosion are expected.

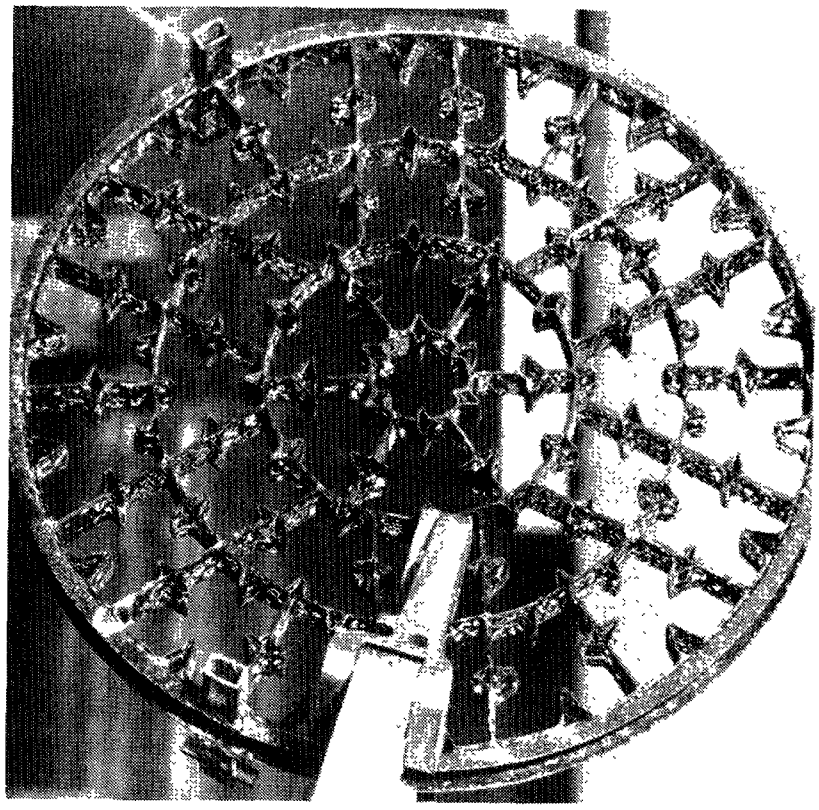
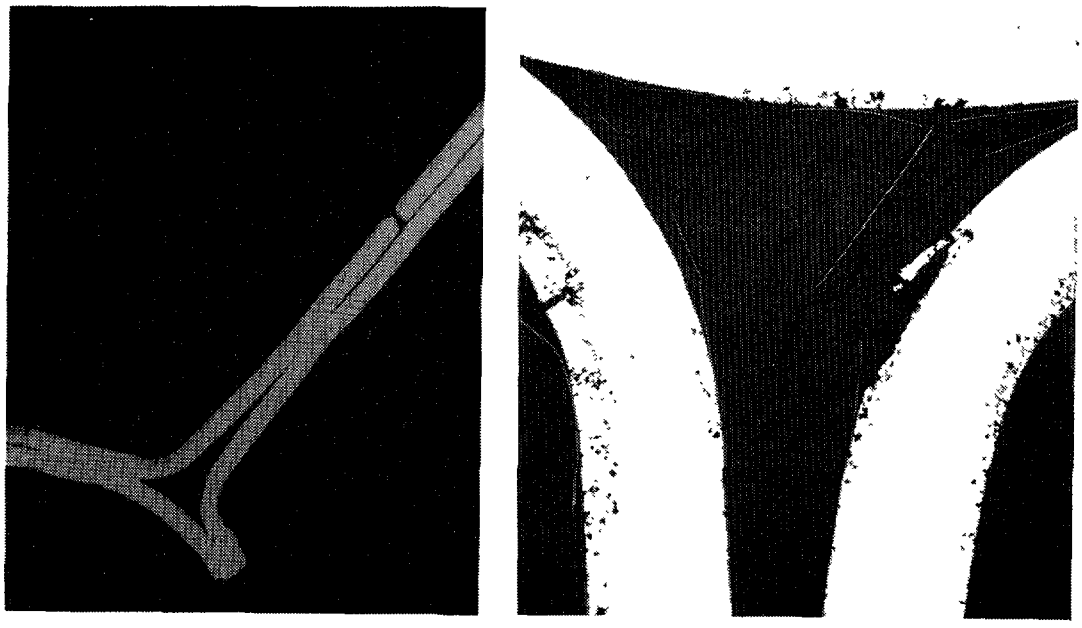


FIG.8a. View of centre brace after dismantling operations (note corrosion product on edges and faces of cell structure).



FIGs 8b and 8c. Metallographic examination of strip section showing evidence of intergranular corrosion and grain loss.

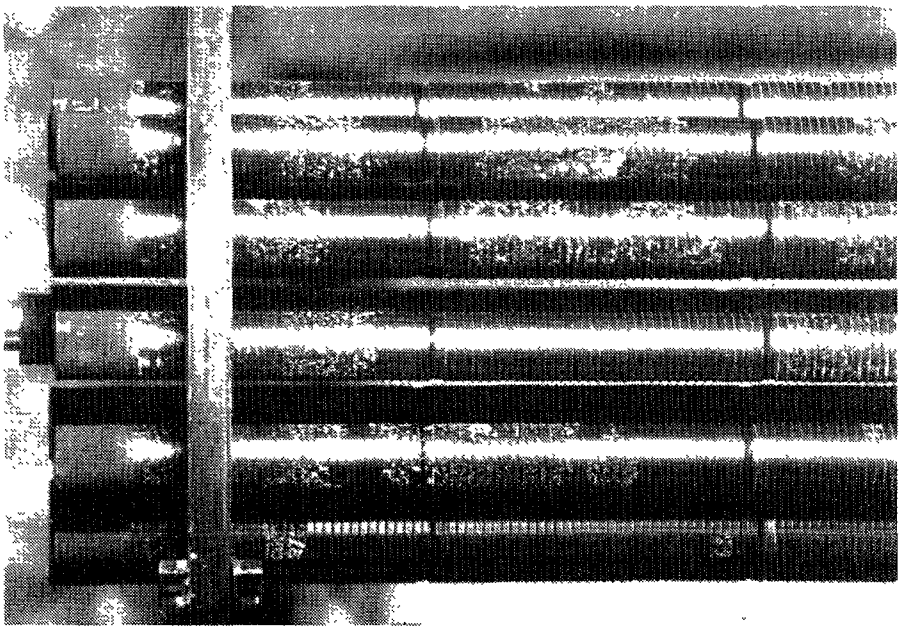
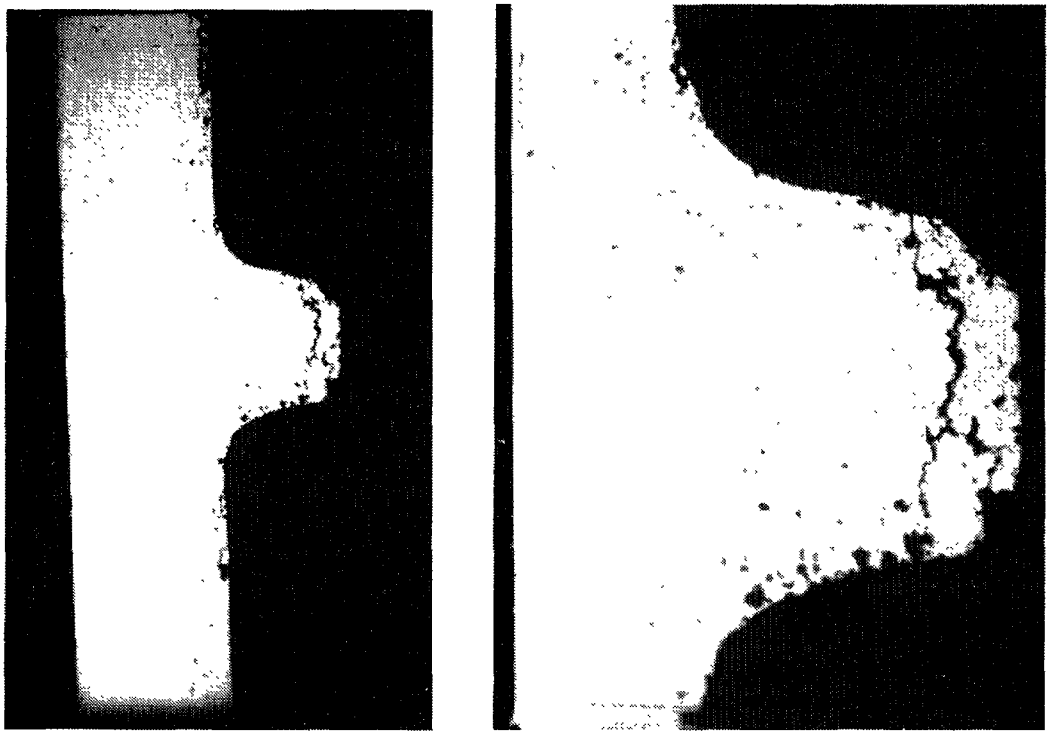


FIG.9a. View of top end of element 1 pins in assembly showing corrosion products on pin surfaces.



FIGs 9b and 9c. Metallographic examination of longitudinal section showing intergranular corrosion and grain loss on rib profile.

26 Strauss tests at Sellafield have shown that standard AGR fuel cladding may become sensitised both by carburisation at the outer surface layer and by R.I.S (Radiation Induced Segregation) through the cladding wall. There is considerable variability in the response of nominally similar materials to pond storage environments, and this variability is thought to be materials related. PSE (Post Storage Examination) has shown that fuel with low irradiation (low levels of R.I.S) has only superficial corrosion after storage in BNFL's original, open, ponds, where the Cl^- level was maintained below 1ppm. As irradiation levels have increased more extensive intergranular attack has been found at higher chloride levels. These effects are shown in Fig 9.

27 Container monitoring of fuel stored for up to 1.5 years in the improved water chemistry of the FHP (pH 11.4, 0.05ppm Cl^-) has shown that this environment is suitable for fuel with current irradiation levels. PSE of this fuel is planned in 1987.

28 Observations of fuel being dismantled in the production plant show the satisfactory condition of fuel stored in FHP at Sellafield, Figure 10.

Summary on AGR fuel monitoring

29 Extensive surveillance on irradiated pond-stored AGR fuel has shown that the original storage conditions of pH7 and low Cl^- are adequate for the earlier lower burn-up fuel. The conditions in the FHP (pH 11.4, Cl^- 0.05ppm) are superior for the more recent, higher burn-up fuel.

30 PSE has shown variability in the extent of corrosion in pond conditions, and this area is currently being investigated.

Water Reactor Fuel

Background

31 Water reactor fuel has been received and stored on site since the 1960s. Because only a very small quantity was reprocessed in the 1960s, the remainder, and current receipts, must be stored until reprocessing begins in the THORP complex in 1992/93.

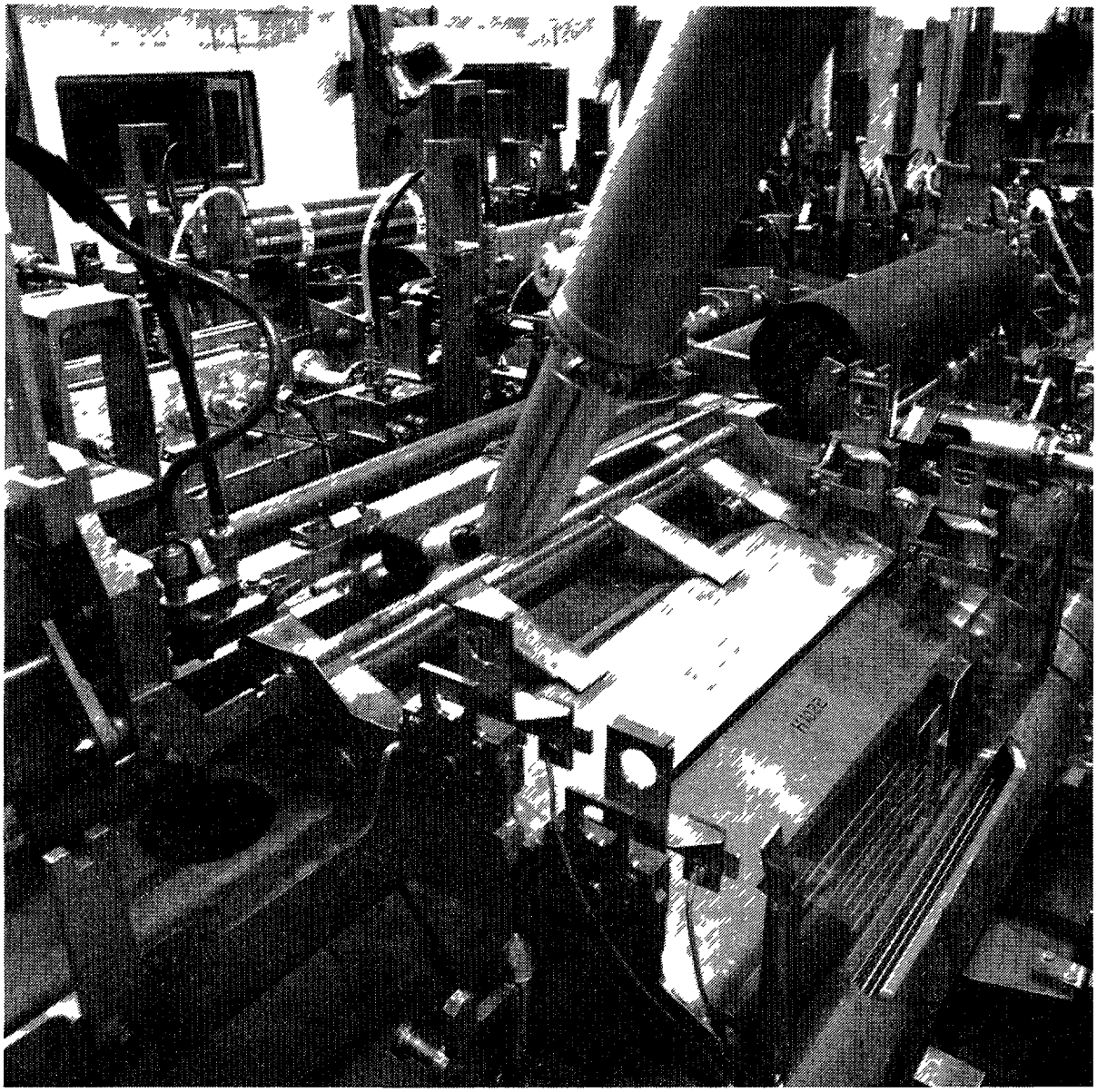


FIG.10. AGR dismantling cave in fuel handling plant.

32 Since the 1977 THORP Inquiry, a programme of PSE has been carried out to maintain confidence in the condition of water reactor fuel.

33 The method of storage at Sellafield has progressively changed from open skips to individual bottles to multi-element bottles (MEB), Fig 11, which originally contained reactor pond water. MEB's contain a neutron absorber material, BORAL, which can also reduce oxygen in the gas space (ullage). Thus MEB's containing sheathed BORAL, ie. not exposed to the internal ullage atmosphere, are flooded to vent out the ullage atmosphere, but MEB's containing exposed BORAL are sealed, so the fuel elements in this case are exposed to reactor pond water only.

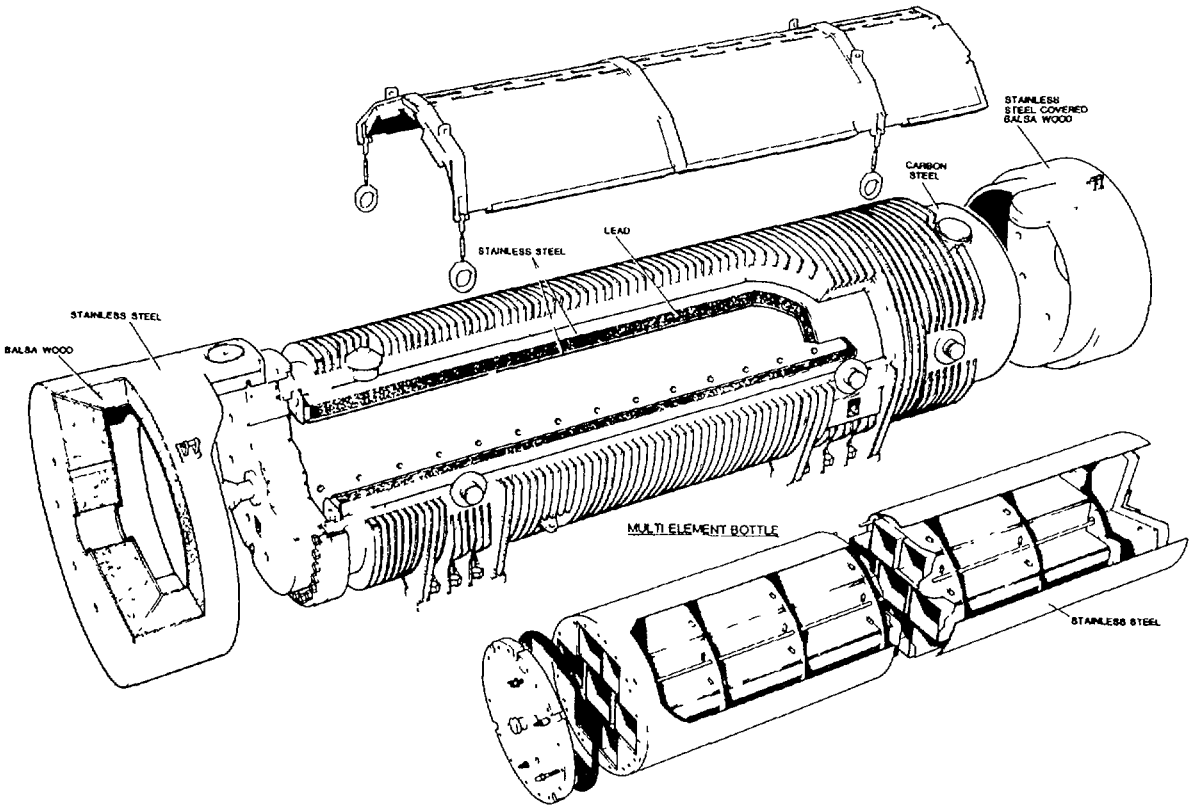


FIG.11. Arrangement of an MEB (multi-element bottle) for storage of water reactor fuel.

Methods of Surveillance

34 Methods used have been underwater visual examination followed where appropriate by more detailed in-cave examination and possibly metallography, and MEB monitoring.

35 For underwater visual examination the lid is removed from the MEB and the fuel element removed. High resolution colour closed circuit television video recordings are taken on each face of the element by passing the element vertically past the camera, turning the element through 90^{o} and repeating. If an interesting feature is found on the element, the camera can examine the particular area at higher magnification.

36 The equipment consists of a specially developed pan and tilt head supporting a high resolution colour video camera, housed in a waterproof casing. The video recordings can be displayed on a monitor while the examination is taking place. For subsequent viewing, the equipment is being modified to allow digital frame storage of selected views. These will be recorded in digital form at the camera resolution, ie. about

twice as good as video tapes. These views can then be replayed for conventional still photography using a special monitor system. Figure 12 shows a schematic arrangement of the current system.

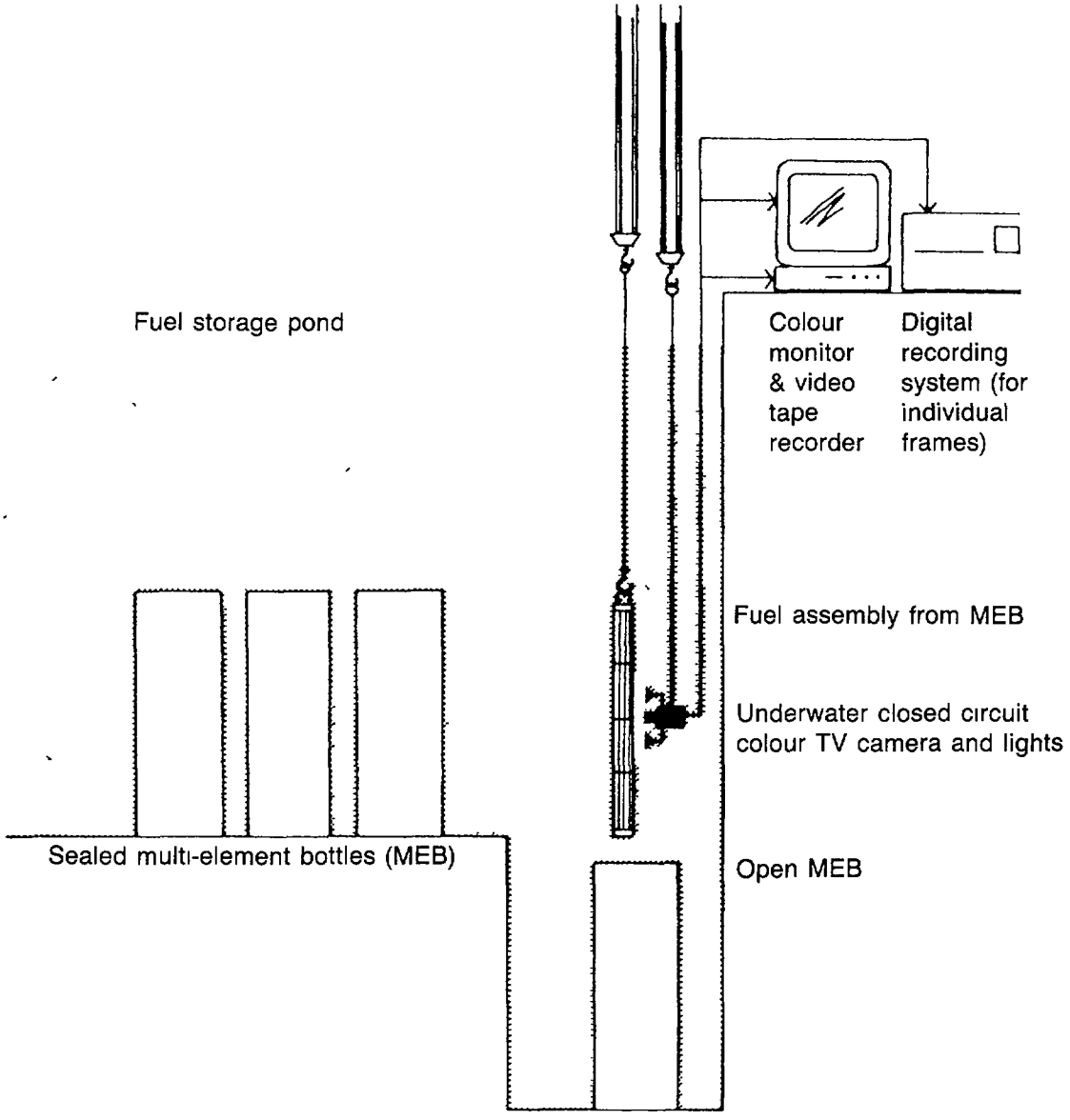


FIG.12. Water reactor fuel inspection method.

37 Although the equipment has some radiation resistance, improvements are being instituted by incorporating a radiation-hardened periscope system. This is being combined with developmental fuel assay programmes where fuel moves past a monitor. Thus advantage will be taken of the assay work to enable a greater number of elements to be examined than would otherwise be possible.

38 Certain features, observed during the visual examination, may be examined further within shielded facilities. For example cladding and pellet metallography, and crud characterisation may be carried out.

39 In addition non-fuel components eg. neutron adsorbers are examined. Samples of BORAL have been exposed to both the bulk pond water and to internal MEB conditions, and periodic visual and destructive examination carried out.

40 To supplement the PSE of individual elements, single pins of specific interest (eg. high burn-up, heavily crudded, broken, or damaged) have been identified, and selected for storage in a retrievable storage basket. It is intended to examine these 'worst case' pins periodically to check for any potential degradation.

41 In addition to PSE, MEB monitoring for release of radioisotopes is carried out on a regular basis. MEB washing trials have also been undertaken to check for crud loosening over extended storage periods.

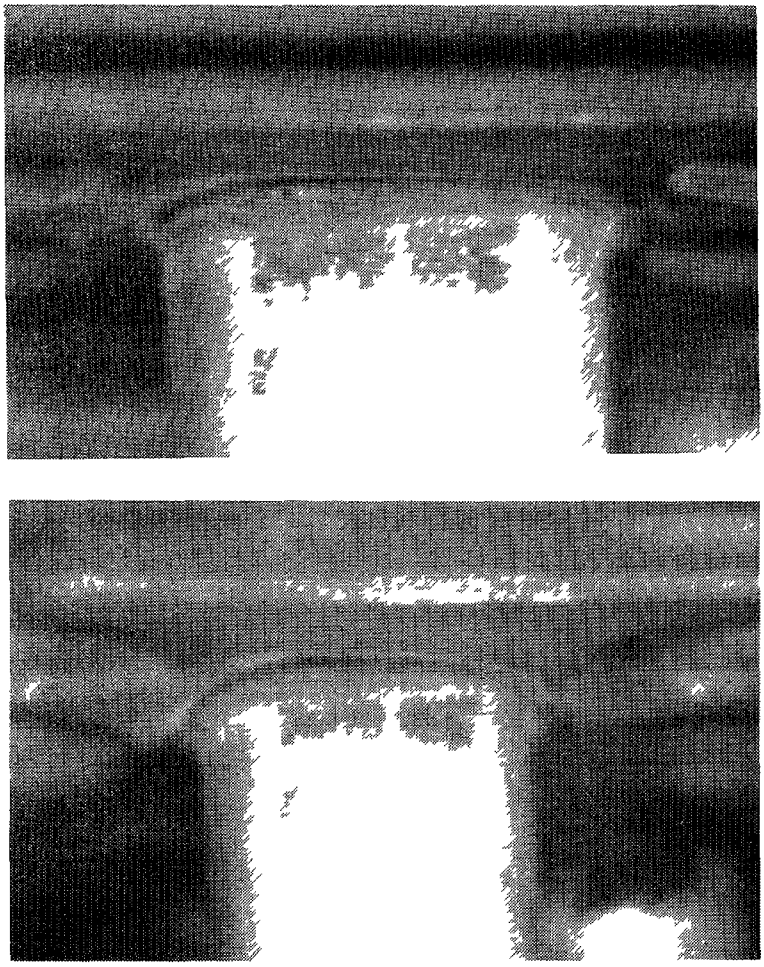
Results of Surveillance of Water Reactor fuel

42 The major conclusion of the PSE programme is that no significant degradation appears to be occurring during extended underwater storage of water reactor fuel. Some unexpected features on fuel pins have been examined in greater detail and found to be only surface markings. This agrees with general world wide experience.

43 Detailed examination has shown that significant degradation has not occurred on BORAL neutron adsorbers, Figure 13, or at crevice locations on PWR top nozzles, see Figure 14.

44 Activity monitoring in MEB's has shown that pin failure has not occurred progressively with ongoing storage ie. caesium values have remained fairly constant. Some crud loosening has been found during element washing trials, but insufficient to make handling operations more difficult in the Head End pond in the THORP complex.

FIG 13 Boral plate, inspected after 2009 days of storage



(a)

(b)

FIG 14 Views of underside of PWR top nozzle showing corrosion products on sleeve of thimble tube after (a) 2412 days and (b) 2882 days storage

Summary Of Water Reactor Fuel Surveillance

45 Ten years of monitoring the condition of water reactor fuel pins stored at Sellafield, for periods of up to 20 years, has revealed no significant deterioration. Non-fuel components have shown limited effects. Overall, Sellafield experience reinforces world data on the confidence in extended storage of water reactor fuels.

46 It is intended that monitoring and surveillance will continue until the fuel is reprocessed in the THORP facility from 1992 onwards.

General Pond Monitoring

47 As well as detailed examination of individual fuel elements, bottled elements or MEB's, comprehensive bulk water pond monitoring is maintained at Sellafield.

48 Storage ponds at Sellafield use a purged water flow system. Recently the new SIXEP (Site Ion Exchange Plant) has been commissioned. Pond purge rates, water temperature, aggressive ion concentration (Cl^- and SO_4^{2-}), pH and radiochemical species are all measured on a routine basis.

49 This approach enables the storage parameters affecting fuel element corrosion or activity release to be adjusted to give fuel protection. For example, Magnox fuel in containers in FHP is kept at pH 13; Cl^- less than 0.05 ppm and the pond water is cooled, typically to below 15^oC, and AGR fuel is stored at pH 11.4, Cl^- less than 0.05ppm. Fuels stored in these conditions have been shown by monitoring to be in good condition after the usual storage periods.

50 An example of the usefulness of bulk pond monitoring was in the older Magnox pond at Sellafield, with open skip storage, when analysis of the ^{137}Cs to ^{134}Cs concentration ratio in the bulk pond water was used to identify leaking fuel. The caesium ratio varies with fuel burn-up and storage time from greater than 13 to 4. For AGR fuel, a caesium ratio of less than 1 signifies graphite as the source, whereas a value of greater than 2 means leaking fuel pins.

Organisation of Monitoring Programmes

51 Long experience of Magnox fuel from manufacture to reprocessing, coupled with the optimisation of the storage chemistry in the FHP at Sellafield and the limited storage period before reprocessing means that only limited surveillance needs to be maintained. There is Utility/BNFL/United Kingdom Atomic Energy Authority (UKAEA) co-operation in monitoring the effects of irradiation on the fuel element, and if necessary this can be extended to post storage examination in-cave.

52 For AGR fuel, at present both the utilities (CEGB/SSEB) and BNFL carry out in-pond fuel surveillance and monitoring, the results are discussed and programmes co-ordinated by a committee composed of representatives of those concerned. The container monitoring is undertaken within BNFL, and the PSE is done under contract by UKAEA. A hierarchy of committees ensures that fuel, as designed and fabricated, is suitable for both reactor endurance and pond storage. These committees also ensure the two-way flow of information necessary for the assessment and experimental validation work required.

53 Water reactor fuel surveillance is organised by BNFL and some of the underwater examination work and metallography are carried out in BNFL facilities, operated by UKAEA staff.

Summary And Conclusions

54 This paper outlines the considerable resources allocated to the surveillance and monitoring programmes designed to ensure that fuel stored at Sellafield remains compatible with any handling or dismantling requirements and reprocessing.

55 Mainly through the results of monitoring fuel, an optimum storage chemistry for Magnox fuel has been developed, and important results on AGR fuel's susceptibility to corrosion have been found. PSE on AGR fuel has also validated the efficacy of a raised pH regime for inhibition, if the latter is required in the event of future increases in burn-up or fuel variants.

56 Monitoring has maintained confidence of minimal degradation of water reactor fuel over extended periods in good quality water.

Acknowledgement

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REFERENCES

(1) IAEA-TECDOC-418
Long Term Wet Spent Nuclear Fuel Storage, Leningrad, May 1986.
"Irradiated Nuclear Fuel Storage at the Sellafield UK Reprocessing Plant", C W E Addison, R D Shaw, BNFL.

NEUTRON AND GAMMA SCANNING OF WWER-440 FUEL ASSEMBLIES IN THE COOLING POND OF KOZLODUY NUCLEAR POWER PLANT

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Abstract

The results of neutron and gamma scanning of WWER-440 fuel assemblies in the cooling pond of NPP Kozlodui are presented. The purpose of the measurement was to develop the experimental technique and the data treatment method. The measurements were performed by the UNIT-6 device, which consists of two fission chambers, Be target flask, ionization chamber and the recording apparatus. 13 spent fuel assemblies from UNIT-1 of the Kozlodui NPP were measured, with initial enrichment of the fuel 1.6, 2.4 and 3.6%, a mean calculated burnup from 18 to 37 MWd/tU and a cooling time from 500 to 1600 d. The burnup distributions along the axis of the fuel assemblies were obtained from the photoneutron count rate and the ionization chamber current measurements. The correlation between the burnup value and the neutron count rate measured by the UNIT-6 device was investigated. It was shown that the ionization chamber current measurements can be used for the spent fuel assemblies cooling time determination.

Introduction

Neutron and gamma radiation measurements of spent fuel assemblies in the NPP cooling pond provide useful information for surveillance purposes. It is possible to obtain fuel burnup and cooling time data from these measurements /1 - 3/. This paper presents the results of the measurements of spent WWER-440 fuel assemblies at the Kozloduj NPP in 1984 - 85. The purpose of the measurements was to develop the method of carrying out neutron and gamma scanning measurements of fuel assemblies in the NPP cooling pond.

Measuring technique

In scanning of spent fuel assemblies the following values are measured: the counting rate for neutron selfradiation from the fuel assembly (N_S); the counting rate for photoneutrons arising from the reaction (γ,n) in a Be(N_{Be}) target, and the ionization chamber current (I_γ). The measurements were performed by the UNIT-6 device, which consists of a detector head and a recording apparatus.

The detector head (See fig. 1) consists of a detecting unit (1), target flask of Be (2) and a mechanism for moving them (3, 5).

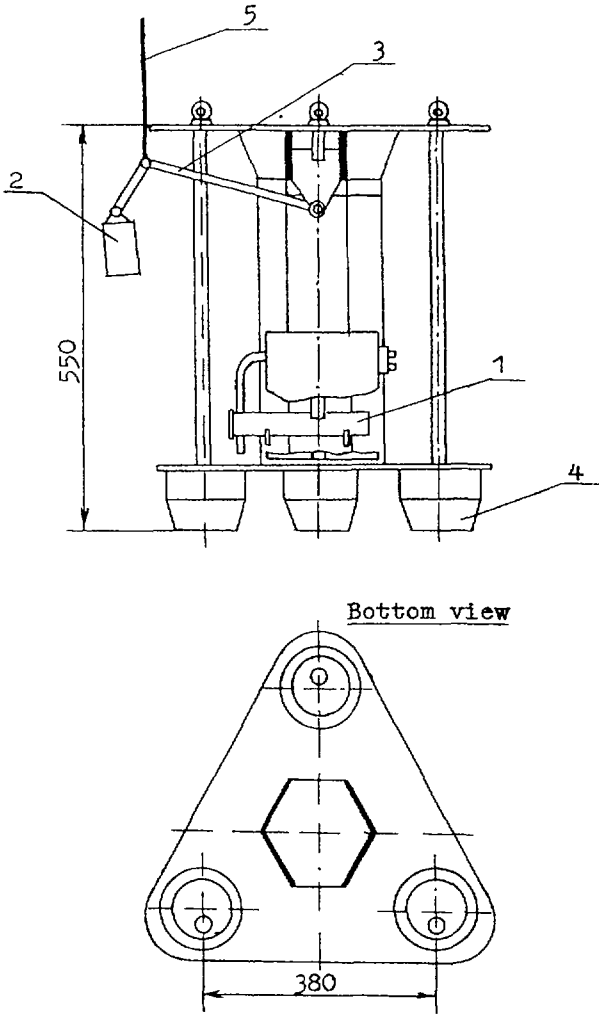


FIG.1. Detector head of the UNIT-6 device.
(1) detecting unit; (2) target flask; (3) target flask moving mechanism; (4) pegs; (5) wire rope.

The detector unit is an assembly of two KNT-31 fission chambers with ^{235}U and an ionization chamber. The fission chambers are arranged on opposite sides of the working channel in the form of a fork, with the ionization chamber placed along side one of the fission chambers. The detecting unit is hermetically sealed. The target flask containing Be is attached to the arm (3) wich is moved into the working position by means of a wire rope (5). The detector head is fastened to the grid of the transport case in a rigor ously fixed position by means of pegs (4).

The recording apparatus consists of a preamplifier for the fission chambers and the ionization chamber, an amplifier - discriminator, a scaling unit, a digital voltmeter and a printer.

Before the measurements were taken, the detector head was placed on the upper grid of the transport case and lowered into the cooling pond together with it. The fuel assembly selected for the measurements was picked up by the refuelling machine, transferred under water to the detector head and lowered into its working channel. Each assembly was measured at 10 positions along its axis with the coordinates given in fig. 2. Also shown in the figure is the coordinates for which fuel burnup calculations were made. The measuring time per position was 100-200s. Altogether, 13 spent fuel assemblies from unit 1 of the Kozloduj plant were measured, with initial ^{235}U enrichment of the fuel Po = 1.6, 2.4 and 3.6%, a mean calculated burnup W_{calc} = = 18 - 37 Kg/tU and a cooling time t_{cool} = 500 - 1600 d. The principal characteristics of the WWER-440 assemblies studied are given in Table. Fig. 3 shows as an example the relative distributions of the measured values, and also the calculated fuel burnup (W_{calc}) along the axis of assembly P 3,6 4-860. The W_{calc} values were obtained using BIPR code /4/. Fig. 3 indicates that the most inhomogeneous distribution along the axis of the assembly is that of the quantity N_S. The distributions of N_{Be} and I_γ are similar to the distribution W_{calc}.

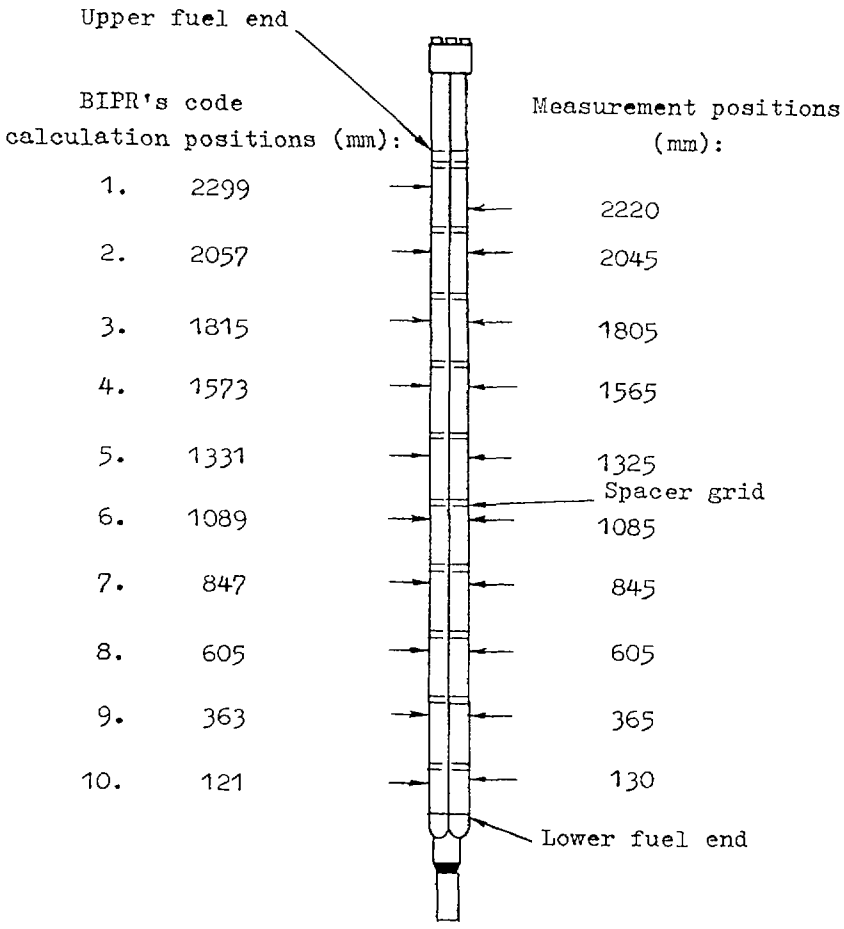


FIG.2. Co-ordinates of neutron measurements.

TABLE: CHARACTERISTICS OF THE FUEL ASSEMBLIES

Item No.	Fuel assembly identification number	Initial enrichment in ^{235}U, %	Irradiation time, days	Cooling time, days	Mean calculated burnup, kg/tU
1	P 3.6 4-308	3.6	1004	881	34.91
2	P 3.6 4-353	3.6	1004	881	34.90
3	P 3.6 4-426	3.6	1004	881	28.60
4	P 3.6 4-860	3.6	999	516,881	33.58
5	P 3.6 4-799	3.6	999	881	34.58
6	P 3.6 01681	3.6	971	510	34.34
7	P 3.6 01703	3.6	971	510	26.90
8	P 2.4 4-808	2.4	999	516,881	26.12
9	P 2.4 33	2.4	999	881	28.61
10	P 2.4 02691	2.4	971	510	30.66
11	P 1.6 302	1.6	640	510	18.02
12	K 2.4 4-2	2.4	1138	1611	30.85
13	K 2.4 4-6	2.4	1138	1611	26.63

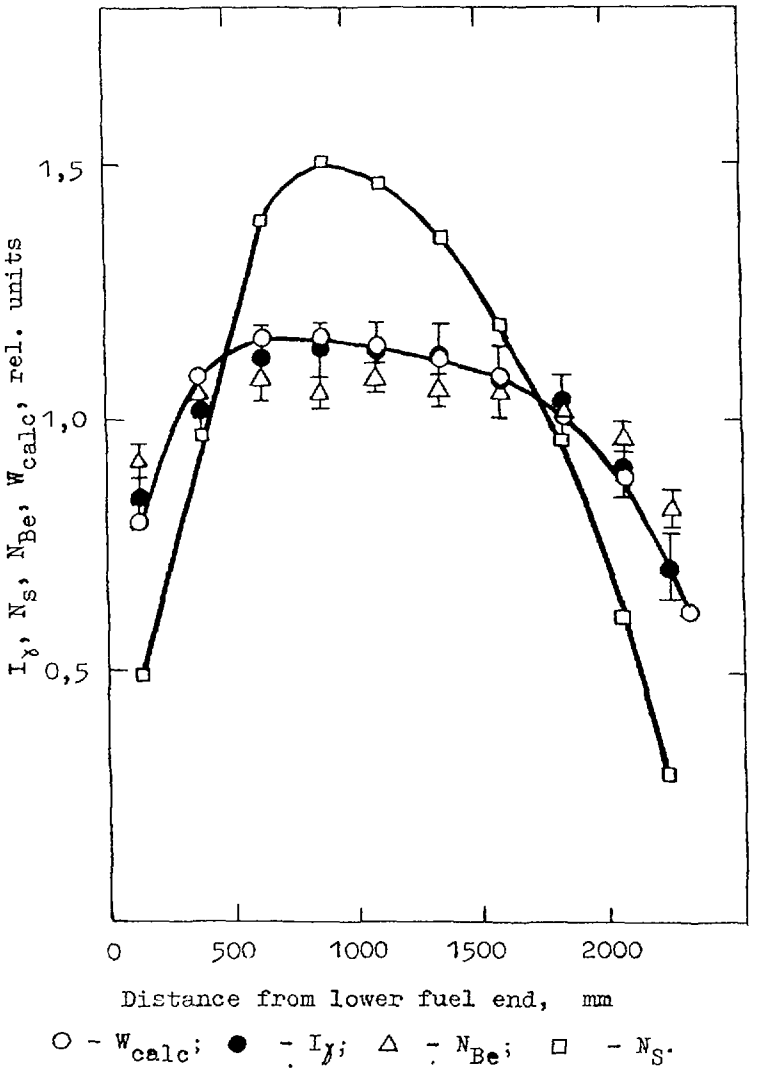


FIG.3. Relative distributions of measured values along the axis of fuel assembly R 3.6 4-860.

Data treatment

The burnup (W) distribution along the axis of the fuel assembly can be estimated from the N_{Be} and I_γ measurements. However, because of strong dependence N_{Be} and I_γ on irradiation history and t_{cool} of the fuel assembly, it is more difficult to determine the values of W from the N_{Be} and I_γ measurements. Therefore, the correlation between W and N_S was investigated. Fig. 4 shows the dependences between W_{calc} and N_S obtained for 10 points along the axis of measured fuel assemblies. The dependence W_{calc} on N_S was approximated by a function:

$$W = b(p_o)N_S^a. \qquad (1)$$

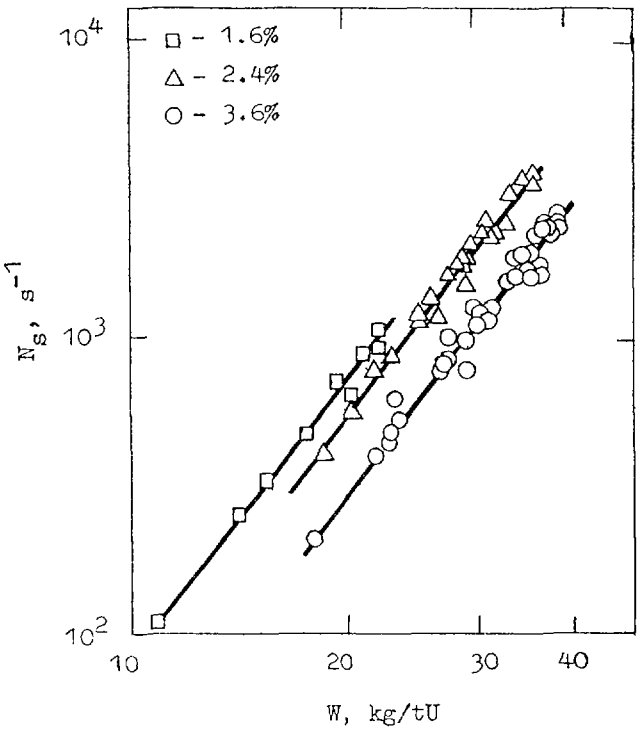


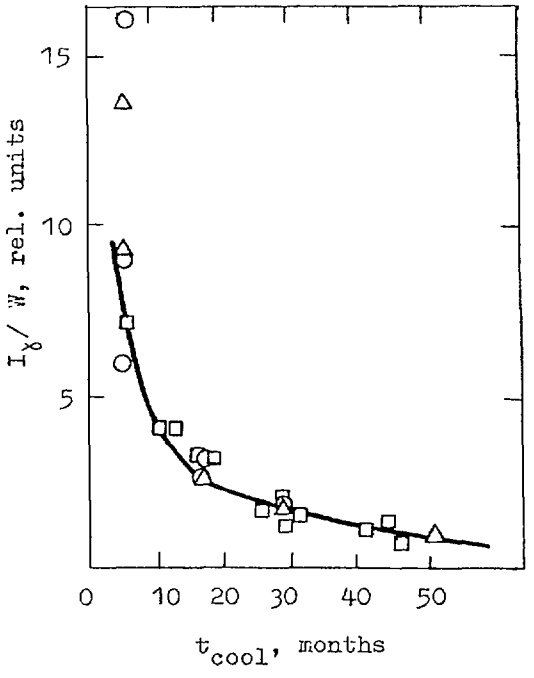
FIG.4. Neutron counting rate for spent WWER-440 fuel assemblies as a function of calculated burnup.

Next values of parameters a and b were obtained by means of the least square method: a = 0.294 ±.008; b(1.6%) = 2.78 ±.10; b(2.4%) = 3.12 ±.08; b(3.6%) = 3.78 ±.08. Note that the values of a obtained do not depend on p_o. These values of a and b can be used for neutron scan data treatment at the UNIT or ION /2/ units type. For this purpose it is necessary to introduce a calibration coefficient φ in formula (1):

$$W = \varphi\, b(p_o) N_S^a.$$

In order to determine the value of φ it is sufficient to measure N_S at one of the positions at which the value of W is known. The method of W determination presented in this paper is applicable at 1.5 years $< t_{cool} <$ 6 years.

The dependence of I_γ on t_{cool} can be used for t_{cool} determination /5/. Fig. 5 shows the dependence of the mean values of I_γ/W_{calc} on t_{cool} for measured fuel assemblies and the data from Ref. /3/. It is seen from fig. 5 that there is a good agreement between our data and the data from Refs /3, 5/. As the I_γ/W ratio is measured in relative units, so it is necessary to carry out calibration measurement of I_γ/W for fuel assembly with a known t_{cool}.



— - Ref. /5/; □ - Ref. /3/;
△ - our data, 1984; ○ - our data, 1985.

FIG.5. Dependence of Iγ/W on the fuel assembly cooling time.

In conclusion, the described in this paper the method of neutron and gamma radiation measurements of WWER-440 fuel assemblies in the cooling pond and the data processing method can be used for the control of fuel assemblies other types of LWR.

REFERENCES

/1/ KOHAR, S., PETENYI, V., LIPTAK, R., KRAJEI, L., The fuel burnup determination by combined passive neutron and gamma-spectrometric ND-measurements, Progress report to IAEA N 2639/Rb, July 1981.

/2/ TARVAINEN, M., RIIHONEN, M., Spent fuel measurements at Loviisa Nuclear Power Station May 1982. First interim report to technical contract N 2950/TC between IAEA and STUK.

/3/ DERMENDJEV, E., FILATKIN, A., KAIEDA, K., UTYATNIKOV, V., KASCHIEV, G., Passive neutron and gross gamma-ray safeguards verification measurement of WWER-440 spent fuel. IAEA-SM-260/42, Vienna, Austria, 8-12 November 1982.

/4/ PETRUNIN, D. M., BELYAEVA, E. D., KIREEVA, I. M., BIPR-5, a program for calculating three-dimensional energy release and fuel burnup fields in single-group diffusion approximation for WWER-type reactors /in Russian/, I. V. Kurchatov Institute of Atomic Energy, Preprint IAEA-2518, Moscow (1975).

/5/ LEE, D. M., et al, A new approach to the estimation of LWR irradiated fuel assemblies using simple gas chamber techniques. LA-7655-MS (Ispo-48), March 1979.

VISUAL INSPECTION OF SPENT FUEL INTEGRITY BY UNDERWATER TELESCOPE

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Abstract

In addition to leakage detection it could be useful to examine the cladding surface for corrosion and mechanical damage in order to locate the source of leakage or to anticipate loss of integrity. The check of non-dismountable assemblies could also be informative. Highly radioactive and densely packed spent fuel is hardly accessible, only remote viewing is possible either in wet or dry storages. Television cameras and periscopes or telescopes are in general utilized for such purposes, in most cases designed for underwater use.

We developed an Underwater Telescope for such purposes and have broad experience in the inspection of LWR fuel assemblies in storage ponds and of other reactor components and structures. Experiments with HWR spent fuel assemblies are also in course.

Visual examination of surfaces can be carried out down to about 10 m water depth with very good resolution and image stability, therefore, in addition to direct viewing or video recording photographs can be taken for documentation and comparison. The device has no radiation sensitive element at the front end, so both remanent damage and temporary malfunctioning is completely avoided.

The mechanical design allows to set up arrangements meeting the actual requirements: various lengths, diameters, vertical or horizontal viewing, special lamps and other options are available.

The design and possible applications are briefly illustrated and the experience obtained in the examination of corrosion and other defects is reported.

In addition to other surveillance methods it is advisable to examine visually the accessible surfaces of fuel cladding and further containment barriers of spent fuel storage facilities. Timely detection and precise location of corrosion or mechanical damage can be an aid to periodic integrity tests. The highly radioactive and densely packed spent fuel allows only remote viewing techniques. Features required are high resolution, radiation resistance, easy handling and image recording. Television cameras and periscopes or telescopes are in general utilized for such purposes, in most cases designed for underwater use.

An Underwater Telescope was developed in the Institute of Isotopes, Budapest for the examination of various reactor components and structures and for safeguards purposes. It consists of an optical unit (telescope) and a tube system (Fig.1).

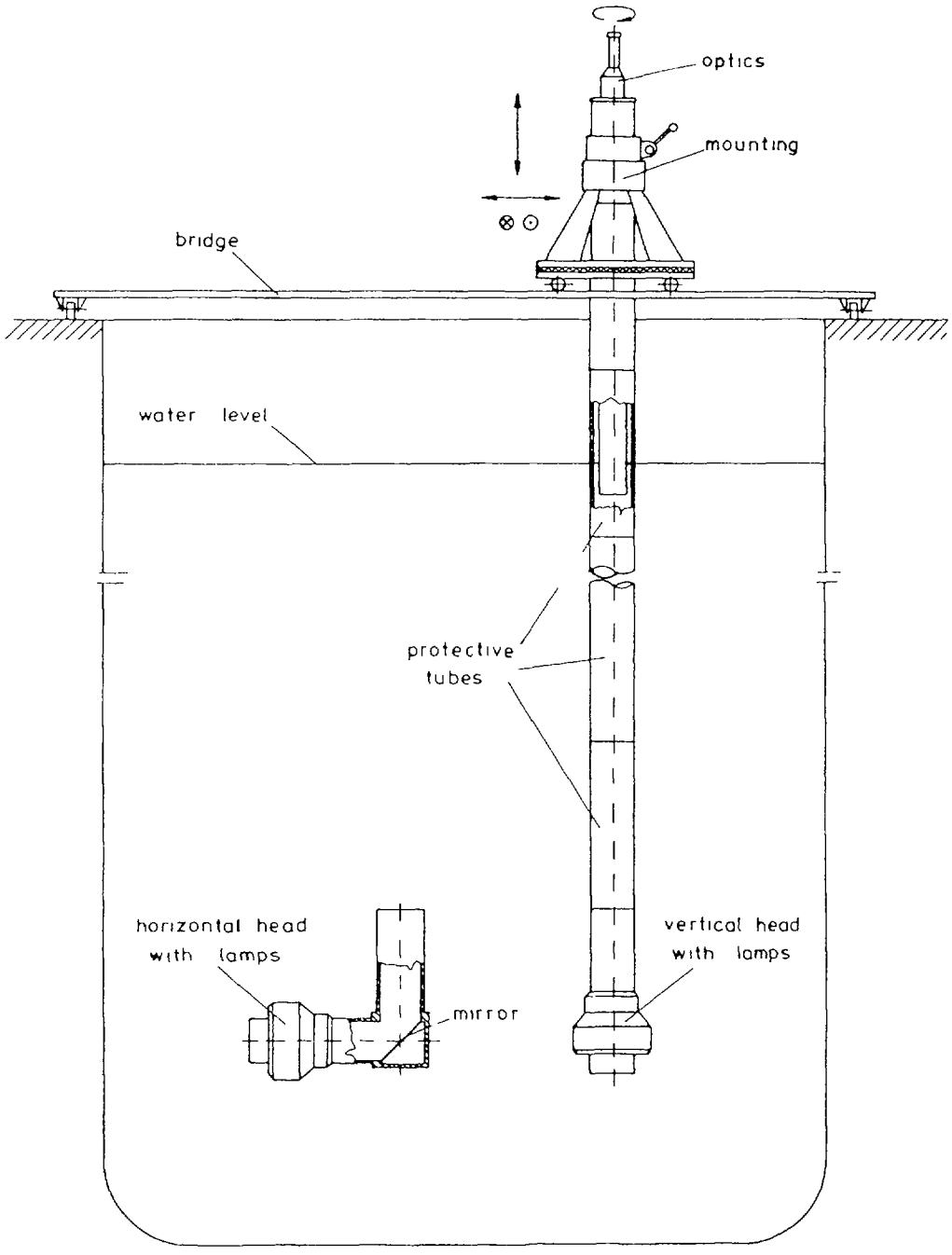

Fig.1. Layout of the Underwater Telescope

The tube is filled up with water providing complete shielding to the optics and the viewer. No radiation sensitive element is at the front end, so both remanent damage and temporary malfunctioning is completely avoided. The optical unit is

provided with zoom and variable magnification allowing easy target positioning and close viewing. The maximum shooting distance in transparent water is about 20 m, high resolution (i.e. 0.1 mm) can be achieved within 10 m. A modified version can be used even in non-transparent water, then the protective tube is filled up with clean water. Owing to the stabilized image photographs or videorecords can be taken for documentation. Horizontal viewing is also possible.

The mechanical design allows to set up arrangements meeting actual requirements: various lengths, diameters, lamps and other options are available.

The Underwater Telescope can be used both in wet and dry storages. Positioning and scanning needs special support and mounting systems designed for actual conditions. [1,2]

The applicability of the device was tested in a number of various LWR facilities and now it is extensively used in periodic safety inspections. Core and pool liners, vessels, racks, tools, containers, heat exchangers, fuel assemblies and other structures and components are examined. [3,4] Based

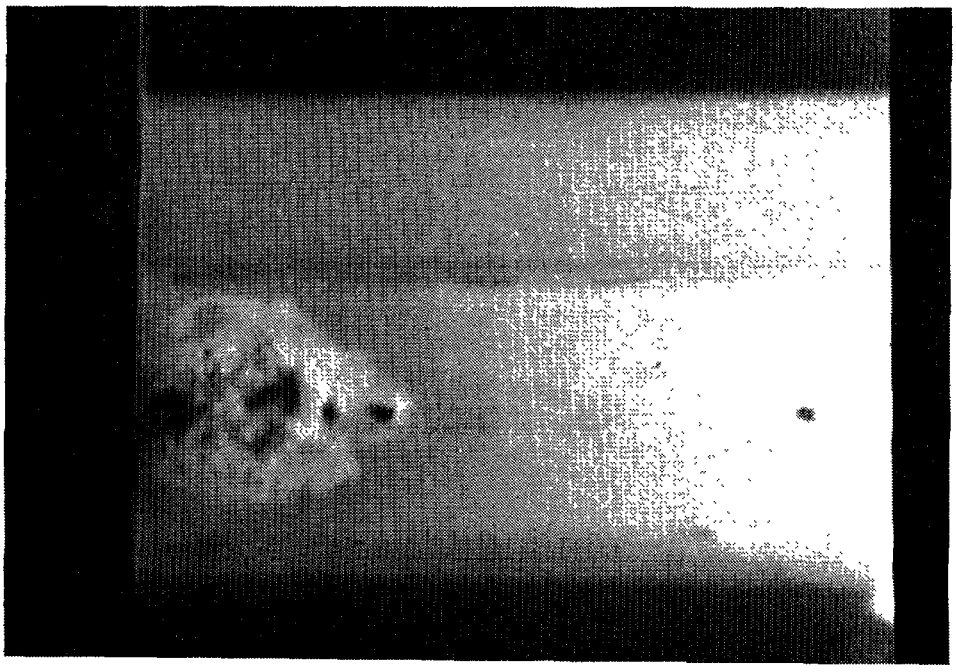

Fig.2. Corrosion on WWR-SM spent fuel

on this experience systematic investigations of spent fuel integrity has been carried out since last year at the Hungarian research and power reactors. Both longitudinal and azimuthal scanning of more than 100 WWR-SM fuel cladding has been completed. (Figs 2-3) Damaged assemblies have been separated and encapsulated. Several other fuel types were also examined showing that corrosion and other defects are not so unusual as expected. Pool liners, both of stainless steel and aluminium exhibit significant corrosion as well, in particular at the weldings. (Fig. 4) Systematic inspection of WWER type power

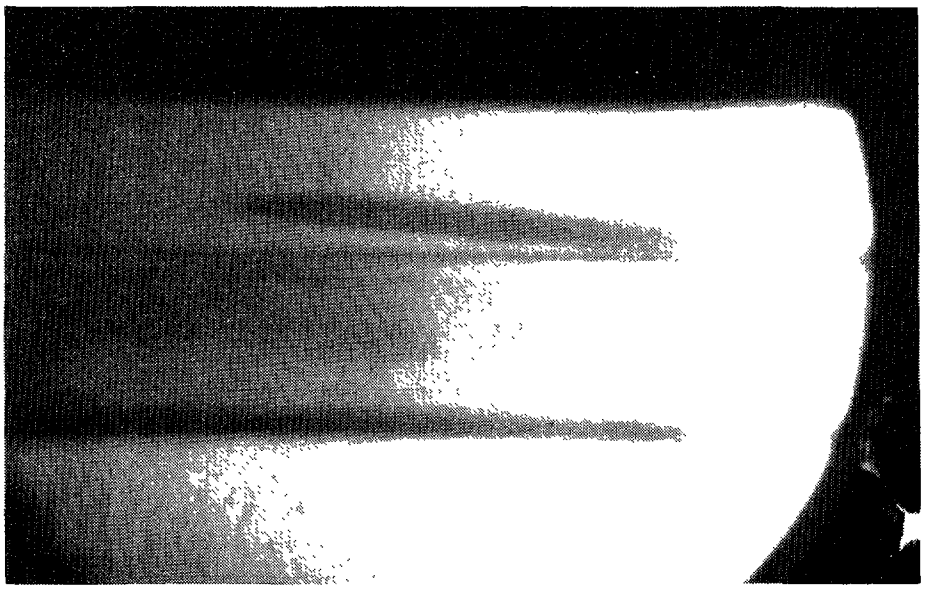

Fig.3. Blow-out on WWR-SM spent fuel

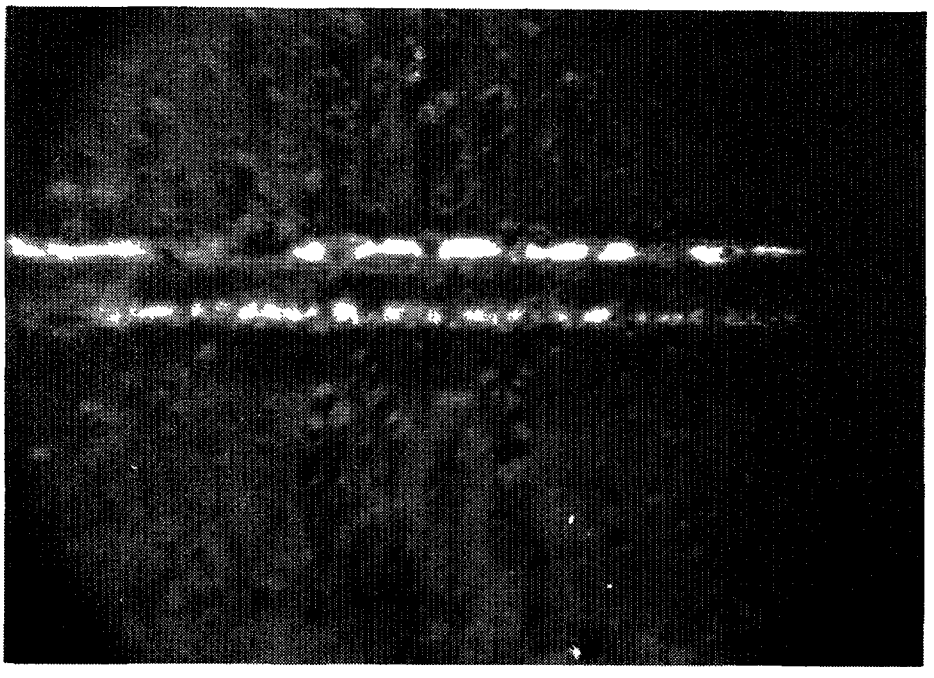

Fig.4. Corrosion around a weld on spent fuel pool liner

reactor fuel assemblies was also commenced. Assembly heads were examined to check the legibility of identification numbers affected by corrosion, deposition and mechanical damage. This type of fuel is non-dismountable, so only the external assembly cladding could be examined. Defects could not only be detected but also evaluated with important consequences on their future fate (Figs 5, 6).

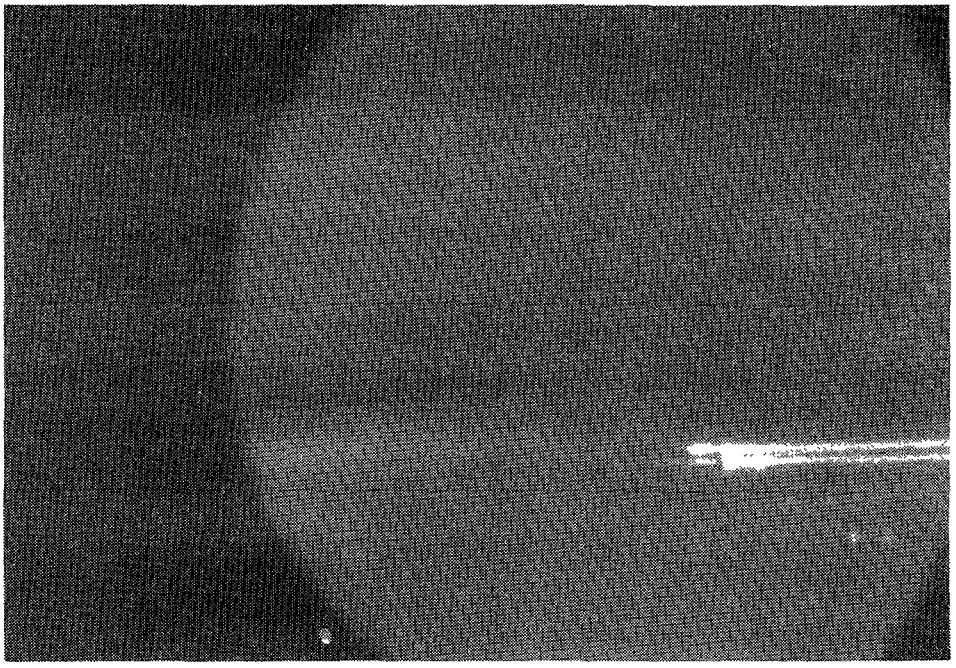

Fig.5. Scratches on external cladding of WWER spent fuel assembly

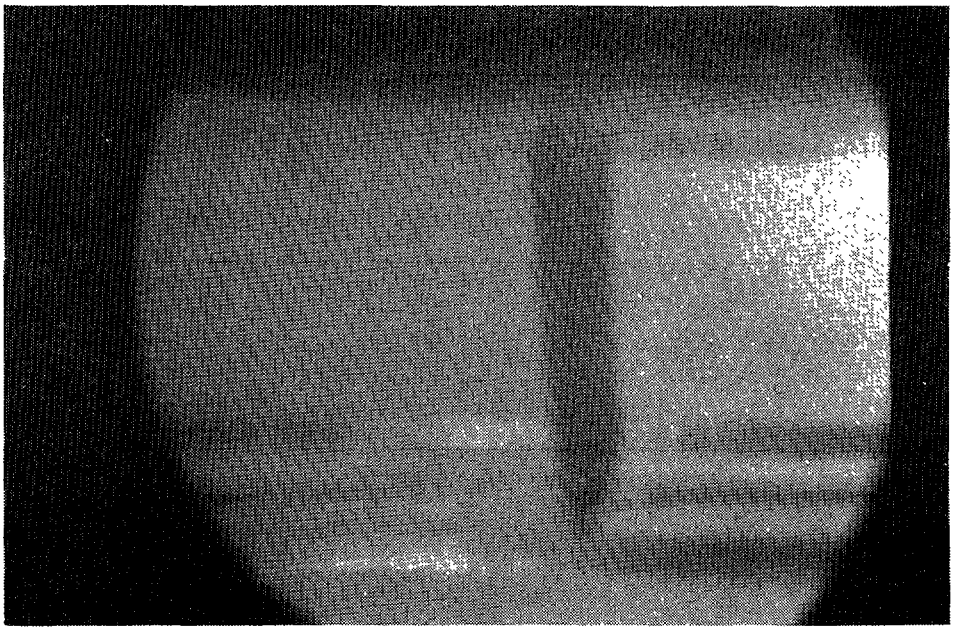

Fig.6. Dents in external cladding of WWER spent fuel assembly

REFERENCES

1. BIRÓ, T., LAKOSI, L., LENDVAI, O., VARGA, K.: Some studies on finger-printing nuclear fuel assemblies by underwater photography, Nuclear Safeguards Technology 1982 (Proc. Symp. Vienna, 1982), Vol. 1, IAEA, Vienna (1983) 423-24.
2. VARGA, K., BIRO, T., VIZDOS, G.: An underwater telescope designed for verifying nuclear fuel identity, ibid., pp. 424-25.
3. BIRO, T. et al.: Verification and integrity studies on spent fuel assemblies by underwater viewing and photography, Proc. 5th ESARDA Annual Symp. on Safeguards and Nuclear Material Management, Versailles, 1983, Joint Res. Centre, CEC, Ispra (1983) 147-51.
4. VARGA, K., BIRO, T.: New developments and experience with an underwater telescope in nuclear fuel identification, Proc. Int. Symp. Nuclear Material Safeguards Technology 1986, IAEA, Vienna, V.1, pp. 515-521 (1987)

STUDIES OF THE CONTINUOUS MONITORING OF FISSILE SYSTEMS USING NEUTRON MULTIPLICATION METHODOLOGY

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Abstract

The development of chemical processes for fissile material processing and spent fuel handling need new methods of surveillance and criticality control enabling an improvement of plant throughput.For this purpose the methodology based on the measurement of epithermal neutron multiplication presumably has practical emphasis. When applied with heuristical approach for date processing, it gives a new continuous criticality control policy.

The presented contribution examines the state-of-the-art in the neutron multiplication methodology, mainly applied to the in-line procedure of negative reactivity surveillance. The results indicate that it is possible to infer nuclear reactivity of fissile systems with accuracy of about 10-20% for very deep subcriticality and 10% for typical range of fissile system reactivities occurring in practice.

The experimental programme for the final verification of neutron multiplication methodology using subcriticality benchmarks is postulated. As a reference the ^{252}Cf driven neutron noise measurements are convenient.

INTRODUCTION

Arrangement of spent fuel assemblies in existing and compaced interim fuel storage facilities need within the range of control and surveillance methods - the procedure to detect nucear reactivity enhancement. Such situation can occur due to perturbations in the physico-chemical state of structural and neutron moderator/absorption materials existing in the facility. The criticality safety of dry storage pools is especially complicated due to the optimum moderation conditious which could appear at rather low water density. The criticality control policy shall be evaluated

during fire-fighting works. The proper evaluation of fuel burn-up credit is also important from this point of view. The problem concerns from one side eigenvalue and source-mode calculations to predict reactivity variations and system response for the external neutron source overriding inherent fuel neutron radiation. Secondly the method applicable in practice to detect nuclear reactivity variation is also desirable. Moreover, it is necessary to validate existing and developed neutron transport computer codes (mainly based on the Monte Carlo method). For this purpose the subcriticality benchmarks should be established as a reference for computer codes validation.

The aim of this researches was to invent continuous criticalitycontrol system to be applied for the nuclear fuel storage assemblies. In the opinion of authors the method based on the epithermal neutron multiplication seems to be applicable in realistic cases.

METHODOLOGY

The Effective Neutron Multiplication Factor, i.e. the ratio between the number of neutrons in the n-th generation and n+1-th generation, is not a directly measurable parameter. That means that it is necessary to consider other physical quantities that, permit to inter k_{eff} using theoretical models and supplementary calculations.

The oldest variable used in the reactor physics for k_{eff} inferring is the external source multiplication, i.e. the ratio of the number of neutrons produced by any neutron introduced to the fissile system (plus number of source neutrons) to the number of source neutrons.

For finite system the neutron multiplication factor can be inferred taking account a function:

$$k_{eff} = f\ (M,\ r_s,\ FMS,\ RC) \qquad (1)$$

where: M is the external source multiplication,
r_s is the source position,
FMS describes fissile material state, and
RC reflection conditions.

Taking into account the neutron leakage fluxes in a fissile material arrangement and an external neutron source, let us define:

$$M_{epi} = \frac{\phi_A^I}{\phi_A^o} \qquad (2)$$

where: M_{epi} is the epithermal multiplication,

ϕ_A^I is the epithermal neutron leakage flux in the point A considering the presence of fissile material in the system, and

ϕ_A^o is the epithermal neutron leakage flux in the poin A when the fisile material was excluded, and the same external source is being used.

Schulze and Wurz[1] established the following formula:

$$k_\infty = 1.747\ (1-M_{epi}^{-0.4368}), \qquad (3)$$

which correlates the calculations of k_∞ with that of the two-dimensional epithermal multiplication, considering an infinite vessel filled with different solutions, an infinite line source touching its external surface and a detector placed at the opposite side of the cylindrical vessel.

Considering the leakage pobablity PL and k_∞ it is possible to calculate:

$$k_{eff} = k_\infty(1-PL). \qquad (4)$$

EXPERIMENTAL TESTS

The first set of experiments[2] had the objective of reproducing the conditions of the heterogenous fissile material distribution Tests were performed using the model of fissile system consisting of ten plexiglass vessels, each in the form of a cylindrical annulus having a total volume of 1092 cm^3. The stack of the 10 annuli was located on the bottom of the measuring assembly.

In the fuel storage facilities the existence of variable fissile material concentration gradients is possible. The variable continuous concentration gradient was simulated by ten discrete stages in such a way that the appropriate set of fissile sections was filled with aqueous solution of uranyl nitrate, uranium enriched 80% in ^{235}U, with different uranium concentrations.
The rest of the sections were flooded with water. Measuring assembly is surrounded by 20 cm plexiglass reflector and the outer side 10 cm thick shielding ring filled with boric-acid/parafine mixture. In the central hole of the annuli stack a teflon pipe was

put. An external ^{252}Cf neutron source was moved up and down through this pipe and placed in a precise position.

The measuring assembly consisted of 16 proportional BF_3 counters located around the system.

In these experiments the measures physical variable was neutron multiplication.

A second set of experimental trials concerned the filling of a tank at different levels with solution of uranyl-nitrate 45 gU/l, uranium was enriched to 80% in ^{235}U. In each experiment a source was placed at height of about the half of the solution level, within the teflon pipe in the axis of the setup. The reflection, measurements condition were the same as in the first set of experiments.

In this set of experiments the measured physical variable was the neutron epithermal multiplication.

CALCULATIONAL TEST

Another series of homogeneous configurations was calculated to cover a range of k_{eff} near criticality. In this computations the same solution levels, solution composition and reflection conditions of the previous experiments were considered, increasing only systems' dimensions

In the last computations the calculational models were used in order to represent the homogeneous experiments in which only the neutron source position changes from the axis of the setup to the position 0.5 cm above the external surface of the system, keeping the source in a vertical arrangement with respect to the system bottom. The detectors were positioned in the opposite side with respect to the source, covering a cylindrical surface comprised in an angle of 120 degrees with respect to the system axis. This type of configurations was named "asymmetrical source", as opposed to the configurations described before, where the source was placed on the axis of the system ("centrally located source position").

The k_{eff} calculations were performated with KENO IV[3] Monte Carlo code. For the first set of calculations the Hansen-Roach 16 energy groups cross section library was used, while for the second set of calculations 27 group cross section library produced with NITAWL[4] from SCALE[5] master cross section library has been applied.

The criticality calculations were also executed with MORSE C.G.[6] Monte Carlo code using SUPERTOG 4[7] 100 group cross section library produced *ad hoc*, with one thermal energy group. The MORSE C G. criticality calculations were repeated, modifying the ^{235}U atomic densities, up to the point in which the resulting k_{eff} values were comprised in a range of 2σ with respect to the KENO IV values. In this way, despite the 100 group cross section library, the MORSE C.G. flux calculations correspond to the same KENO IV k_{eff}.

The flux computations were performed with MORSE C.G. code using the subroutine BDRYX called each time when particle changes physical medium.

The code calculated:

$$FLUX\ (I+1) = FLUX\ (I) + WATE/ABS(W) \qquad (5)$$

where: WATE is the statistical weight of the particle and
W is a cosine of the angle between the particle and the normal to the boundary.

The subroutine BDRYX was programmed to apply eq.(5) only when neutrons pass cylindrical surface placed at the detectors position.

COMPUTATIONAL RESULTS

To analyse the first set of experiments it was necessary to calculate the epithermal multiplication. It was supposed that excellent agreement between multiplication measured and calculated values (see Fig 1) was extended to the epithermal multiplication values

Fig 2 shows the good agreement existing between experimental and calculated epithermal multiplication values for the homogeneous configurations.

It is clear from Fig.3 that the agreement between k_{eff} values obtained from the measured epithermal multiplication and calculated data and KENO IV calculation results improved with the increasing of k_{eff} values. The relative differences become comprised in the calculation error range (about 10%) for k_{eff} values greater than 0.65.

Fig.4 demonstrated that for asymmetrical source configurations the agreement between k_{eff} calculated values from epithermal multiplication and KENO IV data, in the calculation error range (relative differences lower than 10%) begins for k_{eff} values greater

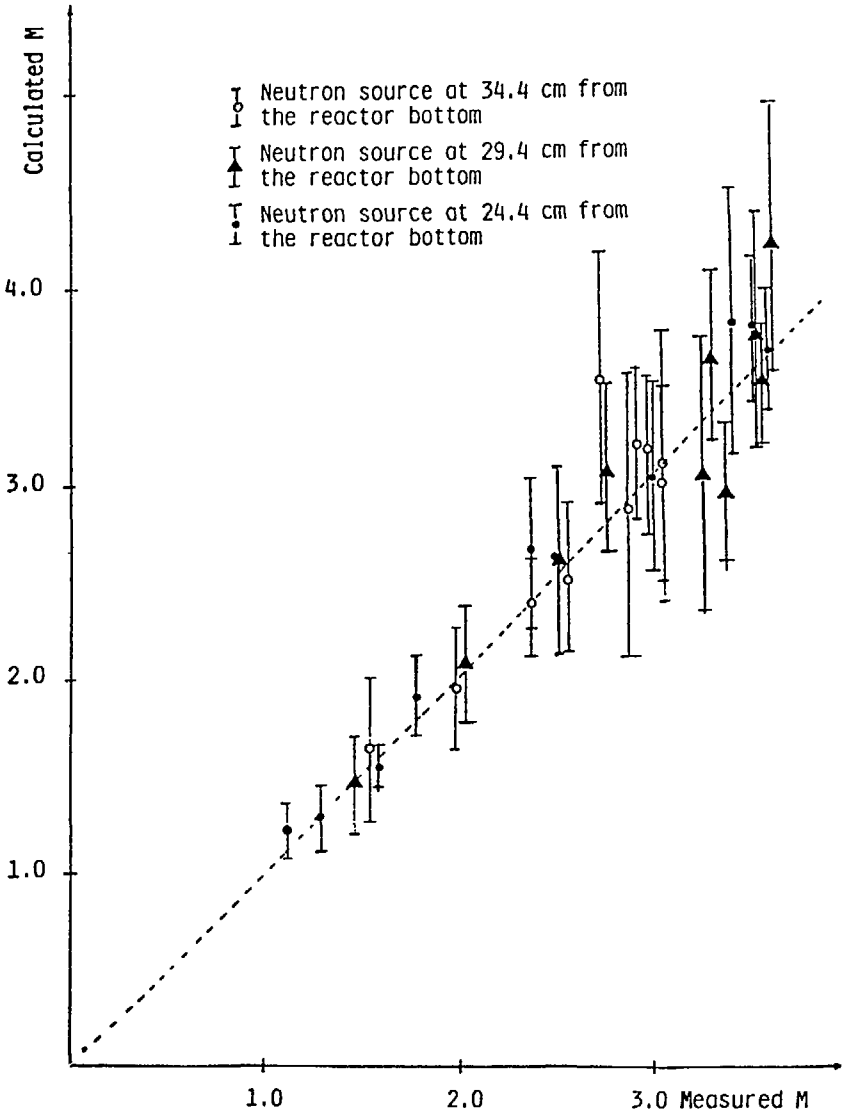

Fig. 1. Comparison between measured and calculated multiplication values, heterogeneous tests.

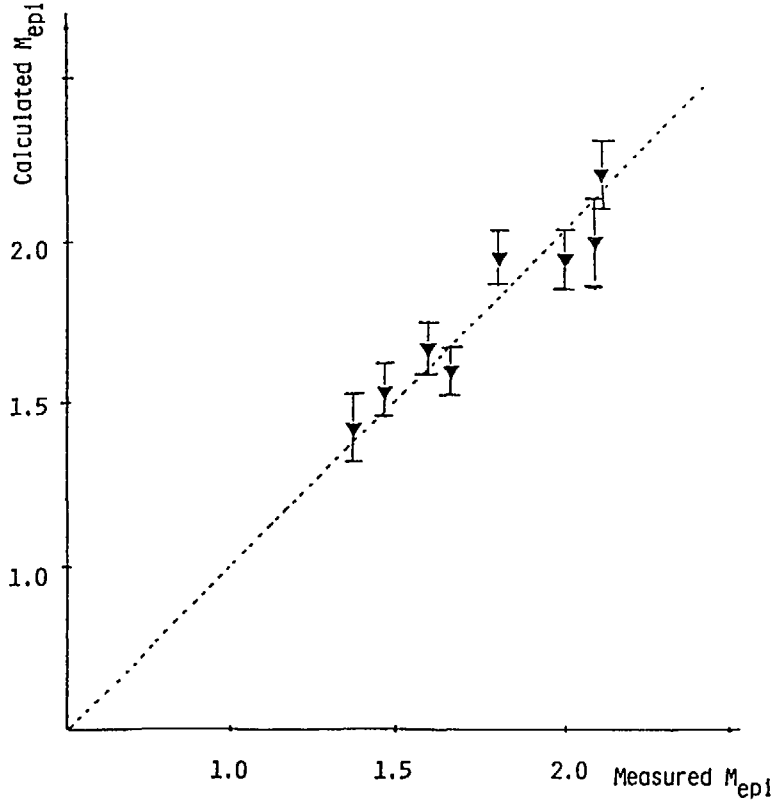

Fig. 2. Comparison between measured and calculated epithermal multiplication values, homogeneous tests.

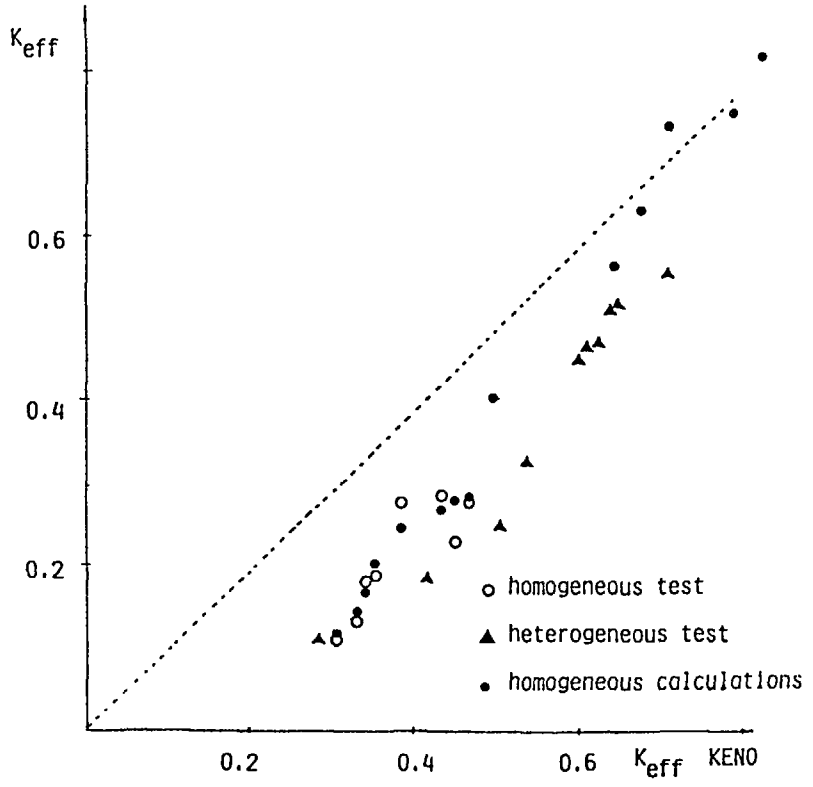

Fig. 3. Comparison between K_{eff} values obtained from epithermal multiplication measured and calculated data and KENO IV calculation results, centrally located neutron source configurations.

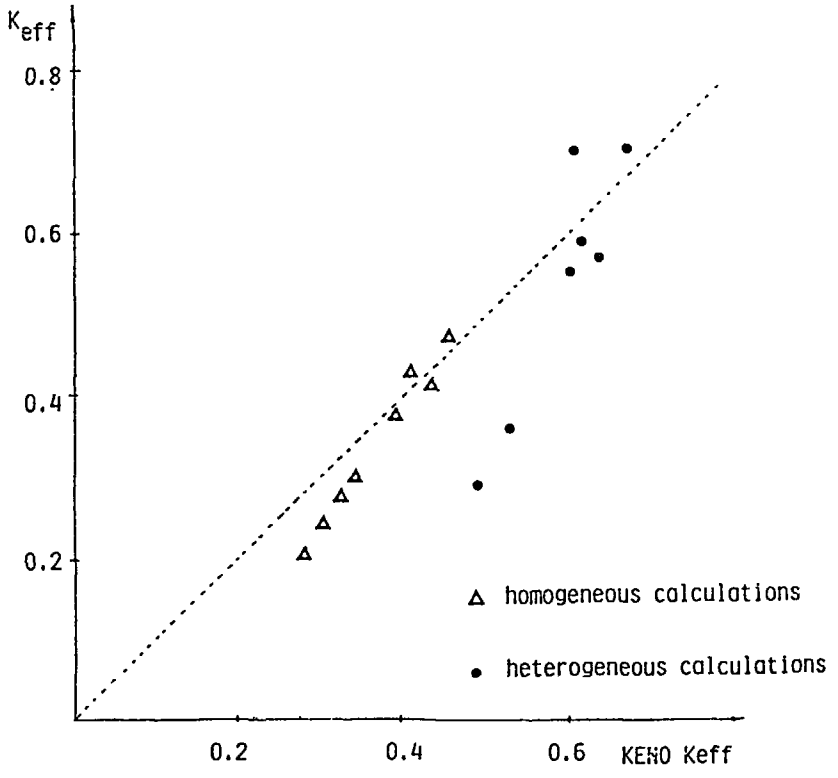

Fig. 4. Comparison between K_{eff} values obtained from epithermal multiplication calculated data and KENO IV results, in the case of asymmetrical source configurations.

TABLE 1. STATISTICAL EVALUATION OF THE HEURISTIC k_{eff} INFERRING METHOD

	Fissile solution level							
	1	2	3	4	5	6	7	8
Learning k_{eff}	0.0462	0.1649	0.2517	0.3431	0.3763	0.4355	0.4441	0.5023
σ_{keff}	0.0017	0.0020	0.0027	0.0029	0.0029	0.0029	0.0032	0.0034
N	18	19	18	19	17	19	19	19
Evaluation $\overline{k}_{eff}$	0.1125	0.1469	0.2370	0.3412	0.3780	0.4220	0.4483	0.4995
$\sigma_{k_{eff}}^{ev}$	0.0478	0.0576	0.0542	0.0359	0.0337	0.0335	0.0248	0.0051
Learning ϱ (\$)	-2949	-723	-424	-273	-236	-185	-178	-141
Evaluation $\overline{\varrho}$(\$)	-1127	-830	-460	-275	-235	-198	-176	-143
σ_{ϱ}^{ev} (%)	42.5	38.1	22.9	10.5	8.8	7.9	5.5	1.0

Note: N is the number of recognized states in effect of the LFRCA action.

than 0.37. That means that the correlation can also correctly describe deeply subcritical systems if the neutron source is located externally to the vessel at the opposite side of the detectors.

This observation agrees with results of simulation study performed by T. Żółtowski and H. Würz[9].

In the homogeneous experiments the agreement between k_{eff} values calculated from epithermal multiplication and KENO IV data is better than the agreement between the same parameters in the heterogeneous experiments. This is due to the influence at the physical state of the fissile material in the k_{eff} values.

Evidently, calculations considering the asynnetrical source correspond to the configurations more suitable ,, the formula (4) application. This circumstance explains the good results obtained.

In is important to mention that the use of artificial intelligence techniques (heuristic methodology) to the problem of inferring the negative reactivity of fissile system. T. Żółtowski[9] have reported final statistical elaboration of evaluation trals using heuristic method and hierarchical pattern clustering (with LFRCA decision rule) for pattern recognition. Table 1 compares the k_{eff} values obtained from the neutron transport calculations with the mean k_{eff} values inferred for expected particular evaluation fissile subcritical systems.

CONCLUSIONS

Neutrom multiplication and epithermal multiplicatioh are the physical parameters reproduced well by neutron transport calculations at deep subcriticality. This property is convenient from the point of view of criticality surveillance of fuel storage facilities and their compaction. The calculations and experimental tests performed in this work show that neutron multiplication methodology is sufficient for criticality surveillance. Moreover, when heuristic method[9] of k_{eff} inferring from M_{epi} values is additionally used - the enough precision of k_{eff} for system reactivities even above -300$ is obtained (see Table 1).

REFERENCES

1. G. Schulze, H. Wurz, "In-line Determination of Fissile Material in High Activity Solutions", Nucl. Technol. 35,663 (1977).
2. T.S. Żółtowski, "Dynamic Approach to the Continuous Criticality Control of Spent Fuel Reprocessing Facilities", Digital Computer Application to Process Control, Proc. 5th IFAC/IFIP Conf. North Holland, Amsterdam, pp.351 (1977).
3. L.M. Petrie and N.P. Cross, "KENO IV - An Improved Monte Carlo Criticality Program", ORNL - 4938, Oak Ridge National Laboratory (1975).
4. L.M. Petrie, N.M. Greene, J.L. Lucius and J.E. White, "NITAWL: AMPX Module for Resonance Self-Shielding and Working Libraries Production", ORNL-TM-3706, Oak Ridge National Laboratory (1976)
5. "SCALE, A Modular Code System for Performing Standarized Computer Analysis for Licensing Evaluation", CCC-450, Oak Ridge National Laboratory (1984).
6. "MORSE C.G., A General Purpose Monte Carlo Multigroup Neutron and Gamma-Ray Transport Code with Combinatorial Geometry", Oak Ridge National Laboratory, CCC-203 (1970).
7. R.Q. Wright, J.C. Lucius, N.M. Green and C.W. Craven Jr., "SUPERTOG, A Program to Generate Fine Group Constant and Pm Scattering Matrices from ENDF/B", ORNL-MM-2679, Oak Ridge National Lab. (1968).
8. T.S. Żółtowski, H. Wurz, "External Neutron Source Multiplication Studies Using the Monte Carlo Transport Code MCNP", Kernforschungzentrum Karlsruhe, Primarbericht, 04.OT.07P, (1986).
9. T.S. Żółtowski, "Studies of Neutron Methods for Process Control and Criticality Surveillance of Fissile Material Processing Facilities", INCT 2054/II/R/A/, Institute of Nuclear Chemistry and Technology (1988).

SUMMARY OF THE MEETING

Representatives from eleven countries were present: Argentina, Finland, Hungary, Italy, Japan, Poland, Spain, Sweden, UK, USA and USSR.

The meeting was chaired by Mr. C.W.E. Addison, Session I co-chaired by Mr. M. Sametband and Session II co-chaired by Mr. E. Vitikainen.

Session I: SPENT FUEL SURVEILLANCE PROGRAMMES AND PRACTICE IN MEMBER STATES

The following is an extract from the papers presented during the Session I:

Argentina:

The nuclear power program consists of HWR reactors: two in operation (Atucha-I, 345 MWe, Embalse, 600 MWe), one under construction (Atucha-II, 745 MWe), and another 700 MWe to be installed before the year 2000.

For spent fuel, wet storage is being used. At present, about 1,070 MTHM have been deposited in pools at the NPP sites.

Monitoring of spent fuel at the pools is done mainly with the objective of localising and segregating defected bundles. For this visual inspection methods are used, and at Embalse a device for dismantling fuel rods for further destructive tests has been installed.

Heat exchange, for the deionized water and mixed bed ion-exchange filters are used in all pools, and the water chemistry is controlled to maintain a low chloride content and activity concentration below 10^{-3} $\mu Ci/cm^3$.

The pools lining is made of austenitic stainless steel in Atucha-I, and of an epoxi coat in Embalse, in both cases with good results.

United Kingdom:

A paper was presented by the representative of UKAEA discussing the construction and operating experience of the PFR irradiated fuel buffer store pond at Dounreay.

Irradiated fuel and breeder subassemblies are removed from the core of the Prototype Fast Reactor (PFR) at Dounreay at the end of their useful life and are held under sodium until the fission product heating has decayed to below 2.5 kW. Each subassembly is then drained of sodium in the Irradiated Fuel Cave (IFC) and the top mixer breeder section and lower filter and spike sections are removed using a laser. The remaining section containing fuel pins is cleaned to remove the sodium, sealed in a spent fuel can and sent to an Irradiated Fuel Buffer Store Pond (IFBS) to await transfer to the Dounreay Fast Reactor Fuel Reprocessing Plant. The IFBS thus serves as a reservoir for spent fuel and eliminates the need to directly couple the preparation programme in the IFC to the operation programme of the Reprocessing Plant.

The pond has a storage capacity large enough to accommodate all fuel bearing reactor core components if it is necessary to discharge the total 210 reactor channels at the same time.

The cooling system and air activity are monitored. The operation of this buffer store has been highly satisfactory.

Spain:

Eight power plants are operating, five of them are PWR, two BWR and one GCR.

All light water spent fuels are stored at pools in each NPP, and the saturation of their capacities is estimated to occur, on average, between 1993 and 2005. The spent fuels of the only GCR are also stored in the NPP pool until its periodic sending to reprocessing.

LWR pools are made of concrete, designed as seismic class I, and lined with stainless steel.

Racks are designed to maintain the spent fuel with a K_{eff} <0.95. The pool water is cooled, filtered and demineralized. Controls are carried out on water temperature, level in the pool, flow rate in the cooling circuit, radiation level and radioactive concentration level in the air, water chemistry, etc. Irradiated fuels and assemblies are visually inspected, inside the pool, by means of TV cameras, and cooling and purification systems are also visually inspected.

A strategy is being elaborated for intermediate storage of spent fuel, considering both wet and dry storage.

U.S.S.R.:

A paper was presented on some possible ways for the compact storage of spent fuel assemblies of WWER-440 and WWER-1000 reactors, in existing ponds at the NPP sites. The increase of the storage capacity can be achieved by the use of boron steel cans, in which case the coefficient of compaction is nearly 2.

As an alternative way, the account of the actual isotope composition of the spent fuel may permit a more compact arrangement in the pool. This can also be used in addition to special heterogeneous absorbers in order to achieve the maximum usage of the pool capacity. It is shown that with this account of the spent fuel burn-up, the calculated lattice pitch at a K_{eff} = 0.95 determines a coefficient of compaction of 1.27 for WWER-440 storage pond, and of 1.66 for WWER-1000.

Session II: EXPERIMENTAL METHODS DEVELOPED IN SUPPORT OF SPENT FUEL SURVEILLANCE PROGRAMMES

All together five papers were presented in the session on the following subjects:

- international co-operation on long-term storage of spent fuel
- surveillance and monitoring of fuel assemblies and storage facilities
- monitoring of fissile material and fission products.

1. International cooperation on long-term storage of spent fuel.
The results and experience gained in the coordinated research programme "BEFAST" (Behaviour of Spent Fuel Assemblies during Extended Storage) which was performed from 1981 to 1986 under the auspices of the IAEA, were presented.

A new follow-up programme for the years 1987-1991 "BEFAST-II" (Behaviour of Spent Fuel and Storage Facility Components during Long-Term Storage) which was initiated by the IAEA in September 1986, was introduced:

The research activities going on in each participating country (by October 1987, 16 organizations from 13 countries were participating in

BEFAST-II) can be grouped under three major topics:

i) long-term storage, ii) surveillance and iii) facilities and operation, which also form the basis of the BEFAST-II final report: "Research, Development and Practice for the Extended Storage of Spent Fuel". The general goal of the programme is to provide a forum for free information exchange on spent fuel matters between specialists in the member countries.

2. Surveillance and monitoring of spent fuel assemblies and storage facilities.

The methods of surveillance for the various types of fuel in the UK were presented: Underwater visual examination in ponds and more detailed examination in shielded caves are carried out. Limited surveillance is necessary for Magnox fuel as storage periods are short (under 2 years). For Advanced Gas Cooled Reactors (AGR) fuels with longer storage periods the retrieval and element breakdown can be performed, followed by detailed metallurgical examination of pins and element components.

The effect of water chemistry on corrosion performance of irradiated AGR fuel was illustrated.

Mainly through monitoring and surveillance, backed by experimental programmes optimum storage regimes have been developed for the Magnox and AGR fuels, consistent with planned storage periods prior to reprocessing. Monitoring has also maintained confidence of minimal degradation of water reactor fuel during storage over extended periods in good quality water.

As an example of the useful tools needed in pool-side examinations, an underwater telescope was introduced.

The underwater telescope can be used for visual examination of surfaces down to about 10 m water depth with very good resolution and image stability. In addition to direct viewing or video recording, photographs can be taken for documentation and comparison. The device has no radiation sensitive element at the front end. The mechanical design allows to set up arrangements meeting actual requirements: various lengths, diameters, vertical or horizontal viewing, special lamps and other options are available.

The experience obtained in the examination of corrosion and other defects in both research and power reactor pools was reported.

3. Monitoring of fissile material and fission products

The results of neutron and gamma scanning of WWER-440 fuel assemblies in the cooling pond of NPP Kozlodui were presented.

The measurements were performed using a special device (Unit-6) which consists of two fission chambers, Be-target flask, ionization chamber and the recording apparatus. Thirteen spent fuel assemblies with different initial enrichments (1,6; 2,4 and 3,6%) were measured (burn-ups from 18 to 37 MWd/kgU and cooling times from 500 to 1600 days).

The burn-up distributions along the axis of the fuel assemblies were obtained from the photoneutron count rate and the ionization chamber current measurements.

The correlation between the burn-up value and the neutron count rate measured was investigated. It was shown that the ionization chamber current measurements can be used for the cooling time determination of the spent fuel assemblies.

Studies of the continuous monitoring of fissile material processing facilities using neutron multiplication methodology were presented.

The state of the art in the neutron multiplication methodology, mainly applied to the on-line procedure of negative reactivity surveillance, was described.

The results indicate that it is possible to infer nuclear reactivity of fissile systems with accuracy of about 10-20% for very deep subcriticality and 10% for typical range of fissile system reactivities occurring in practice.

The experimental programme for final verification of neutron multiplication methodology using subcriticality benchmarks is postulated. As a reference the Cf-252 driven neutron noise measurements are convenient.

SUMMARY OF THE PANEL DISCUSSION

Introduction

During the presentation of the papers, it was noted that there was a diversity in the methodology utilized for surveillance and monitoring of storage of spent fuel. Thus, the panel discussion was established to identify and categorise the current state of the art. Subjects discussed were present monitoring and surveillance practice, methods of surveillance, need for monitoring, dry storage issues, criticality factors, future trends and recommendations.

In general, it was noted there were four schools of thought: those involved in light water reactor fuel, heavy water reactor fuel, gas cooled fuel and for fast reactor fuel storage.

I. Light water reactor fuels

The spent fuel from light water reactors has been stored in water pools over the past 40 years. The fuel from commercial nuclear power facilities is typically stored in at-reactor pools or in separate Fuel Handling Buildings (FHB). Evaluations have shown negligible fuel degradation with utilization of proper storage water temperature and chemistry. The need for spent fuel pool surveillance and monitoring extends from the requirements to ensure the safety of facility personnel and general public. The spent fuel pools are monitored through use of radiation instrumentation in the FHB, and through effluent sampling of the water to be processed through the Spent Fuel Pool Cooling and Cleanup System and of the air in Heating and Ventilation/Air Conditioning System. The FHB environment and pool are monitored for the various types of radioactive materials which could be released as an inadvertent result of fuel failure or corrosion product release, including
1) fission products off gas, including Kr-85 and Xe-131 to the FHB environment,
2) fission products released to the pool coolant, including Cs-137, etc.
3) release of activation corrosion products to pool, including Co-60, Fe-59, etc. In the event of detection of abnormally high levels of radiation the defective fuel assembly is identified and sometimes placed in a container to prevent further release of radioactive material. In cases where the pool fuel

has been reracked (with neutron absorber material included in the storage racks) to accommodate additional spent fuel, periodic test specimens of the absorption material are removed and monitored and the welds and stainless steel examined. In addition, most light water reactor pools lined with stainless steel are monitored for leakage.

Surveillance of spent fuel is performed through periodic visual inspections of the pool, and also the instrumentation associated with the cooling and ventilation systems. All procedures at each facility are performed to ensure that as low as reasonably achievable radiation doses are maintained.

During the UK 1977 THORP inquiry a recommendation was made to monitor the water reactor fuel to confirm confidence in long term (~ 20 years) storage prior to reprocessing. Thus, a minimum programme is carried out on water reactor fuel to achieve this objective.

II. Heavy water reactor fuel

More emphasis is placed in monitoring the fuel from heavy water moderated reactors than the light water reactor fuel. For example, in Argentina in addition to pond water monitoring, fuel is examined in the transfer area where fuel is manoeuvred in and out of reactor. Three non-destructive systems are used:

i) Visual, with a periscope coupled to a telescope or to a TV camera;

ii) Metrology, to measure the length variation of the rods. The bundles are coupled and displaced vertically and rotated axially. The ends of the fuel pins are aligned and measured using a telescope.

iii) Gamma scanning, measuring the total gamma activity along the bundle.

At one station, Embalse, dismantling of fuel bundles is carried out to allow close visual examination of pins and to examine samples of cladding. The BEFAST-I programme also supports this position.

III. Gas cooled reactor fuel

Due to the nature and wide range of fuels stored in Britain, fuel surveillance is considered necessary. Magnox fuel was not designed for long term (>2 yr) storage and monitoring is available during decanning at Sellafield. AGR fuel can in some instances be prone to corrosion, and extensive monitoring and surveillance is carried out to ensure satisfactory storage prior to dismantling (rod consolidation) and eventual reprocessing. Water reactor fuel is also stored extensively at Sellafield. In addition to extensive visual examination in pond, a selection of elements is extracted from wet storage and transferred to shielded facilities for more detailed examination. This includes complete element breakdown, followed by mechanical and leak testing of pins and element support structures (grids and braces) if appropriate.

IV. Fast Reactor Fuel

In Britain fast reactor fuel is routinely reprocessed and therefore long-term storage is not envisaged. Fuel subassemblies are cleaned to remove adhering sodium coolant from the sealed fuel pins and assembly wrapper. The fuel is then sealed in stainless steel cans before it is transferred to the storage pond. There are therefore two sealed boundaries between the fuel material and the pond water.

Surveillance of the pond involves monitoring the cooling and ventilation systems for leaks of radioactivity. Leakage of water into the cans from the pond is checked prior to dispatching the fuel to the reprocessing plant.

V. General discussion

It is considered that fuel monitoring becomes more necessary as storage times increase. As extensive monitoring programmes - including fuel and non fuel components - are not required, international collaboration such as BEFAST-II is strongly supported.

In spite of the different approaches above, many common features do exist among the countries represented. A matrix shows for each participating nation the various types of fuel stored and the associated activities for monitoring and surveillance.

Most nations carry out examinations of non-fuel components such as neutron absorbers, pond liners and fuel racks or grids.

An interesting approach to surveillance was monitoring the K_{eff} in storage ponds.

It was agreed that much scope existed to refine the calculational methods and codes to achieve realistic K_{eff} values, which may at present be considered conservative. Subcriticality benchmarks are required.

An application of this was discussed for the Swedish facility of CLAB where it will be investigated if relaxation of calculated K_{eff} values may be achieved (e.g. by consideration of fuel burn-up). This may enable denser storage in the existing facilities.

There was also a consensus of opinion that the present examination and surveillance equipment gave adequate resolution for interpretation of data. Colour video equipment combined with still photography provides a comprehensive examination and recording system. Also video and photography via a telescope system is used.

It was felt that after the preliminary trials into dry storage, less monitoring and surveillance would be required compared with wet storage. An area for development involves humidity measurements inside dry storage containers.

To conclude, all participants welcomed the exchange of information on fuel surveillance and monitoring, and would appreciate an update in 3-5 years time on this topic, possibly to coincide with the completion of the BEFAST-II programme.

CONCLUSIONS AND RECOMMENDATIONS

1. It is considered that fuel monitoring becomes more necessary as storage times increase to ensure safe storage.
2. The monitoring and surveillance of spent fuel has been discussed between participating countries. The extent and scope of the programmes depend in part on the type of fuel stored.
3. Work to date on LWR and HWR with zircaloy cladding fuel supports world experience in minimal degradation in water storage while maintaining recommended water chemistry and conditions.
4. Work on fuel intended for short term storage and reprocessing (including some gas-cooled fuel) has involved more intense monitoring and surveillance to ensure compatibility with any handling before reprocessing.
5. The importance of the verification of the condition of non fuel components (for example pond liners and neutron absorbers) has been stressed.
6. Much work is continuing into refining the calculational codes for estimation of K_{eff} values. This could give benefit to enable higher density storage to be achieved. The experimental programme for neutron multiplication methodology for nuclear reactivity surveillance is postulated.
7. The transition to dry storage may reduce the extent of monitoring and surveillance required. Development of humidity instrumentation may be required.
8. It is recommended that the BEFAST-II CRP and other Member States would address the definition of factors which could limit storage periods.
9. It is recommended that further work is carried out into all aspects to enable safe and efficient long term storage of irradiated fuel.
10. It is recommended that further work into methods of non-destructive examination is carried out.
11. It is recommended that an update meeting be held in 3-5 years time to include factors involved in the transition to dry storage and the completion of the BEFAST-II programme.

Country	Storage Mode (Type)	Fuel Types	Planned Storage Time	CONTINUOUS (C) PERIODICAL (P) OR EXCEPTIONAL (E) SURVEILLANCE OF : Pool Water	Air in Strge. Rooms	Pool Liner & Components	Fuel Assemb.	Other What ?	MONITORING METHODS AVAILABLE (A) PLANNED (P) NOT PLANNED (NP) Visual (V)	γ- Scann. of FA	Sipping for leak detect.	Eddy Curr. Oxide	Air Monitors	Water Analys. (Samplg.)	Other What ?	Limitations or special requirements (by authorities)	Experience, Results, other remarks
Argentina	Wet	PHWR	15 yrs.	P	C	P	P		A	A	A	P	A	A	Facility for dismantling FA		Minimal degradation of fuel convenience of research on alternative materials for Pool equip. & lining
Finland	Wet	BWR WWER (PWR)	BWR 40 a	P	C	P	P		P	A	A	A	A	A	Facility for dismantling the WWER-		
Hungary	Wet	WWER (PWR) RR (A1)		C P	C -	P P	P P		A A	A A	A A	A A	A NP	A A			
Italy	Wet	MAGNOX MTR, BWR PWR(SS) CANDU	2 yrs. 25 yrs.	P	C	P(E)	P(E)	-Boraflex specimens surveillance -sludge(P) analysis (pool bottom)	A A A	NP A A	NP P A	NP P NP	A A A	A A A	Monitoring of storage room (C)		Few cases of high degradation of Magnox fuel
Japan	Dry Wet	RR (A1) (BWR) (PWR)						Continuous temp + FP monitoring of storage atmosphere -periodical destructive fuel examinations					A				
Poland	Wet	RR (A1)		P	C	-	-		P	P	P	NP	A	A	Continuous Criticality Control Planned		Dev. of epithermal multiplication methodology for continuous criticality control, heuristic approaches to Keff inferring
Spain	Wet (in NPPS) Planned Wet+Dry (Intermediate)	PWR BWR	on average 10 years up to 40 years	P	C	P	P	-	A	NP	A	NP	A	A		Max. Radition rate in accessible area 2.5 mR/h	
Sweden	Wet	BWR PWR	max. 60 yrs.	C	C	C	P		A	A	A	NP	A	A			Extensive reconstitution has been performed at some reactors. No such activities planned at CLAB
United Kingdom	Wet (Dry)	FBR AGR MAGNOX LWR (ZRY,SS)	10-12yrs.max 2 yrs. max 20 yrs. max	P P	C C	C P	- P	Boral skips P	- A	- A	- A	- A	A A	A A	Can draining Dismantling facility observation while decanning or dismantling	Not identified	7% of cans have leaked few CC's of water Water reactor fuel-min.degradation MAGNOX & AGR - condition compatible with handling & reprocessing
USA	Wet (Dry Planned)	PWR BWR	5 yrs wet (up to 40 years)	C	C	P	P	Poison rack	P	NP	A	NP	P	P		Not identified	Water storage safe & required for 5 years. Dry storage safe & acceptable after 5 years cooling.
USSR	Wet	WWER (PWR) RBMK	10 a 10	P	C	P	P		A	A	A	P	A	A	Neutron scanning & Y spectroscopy		5-8 years storage does not change noticeably the mechanical properties of the fuel element cladding.

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