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STATUS OF UKAEA LOW POWER RESEARCH REACTORS

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ABSTRACT

The NESTOR, DIMPLE and ZEBRA low power reactors, located at the AEA Winfrith Technology Centre in the South of England, offer a comprehensive research capability. These versatile reactors are used to investigate performance and safety issues relevant to the entire fuel cycle, covering fuel manufacture, power production, spent fuel storage and reprocessing. Recent reviews, in preparation for licensing by the UK Health and Safety Executive, have confirmed the safety of the reactors for future operation and identified no factors which will limit their useful life.

INTRODUCTION

The UK Atomic Energy Authority (AEA) own and manage three versatile low power research reactors at its Winfrith site in the South of England. The reactors, NESTOR, DIMPLE and ZEBRA, have complementary functions and are run by the same physics and operations team in a coordinated programme. This paper provides a brief description of the reactors and their capabilities, together with their current status. In addition, the paper gives an overview of the regulatory framework governing the operation of these research reactors and the procedures involved in executing experiments within defined operating limits.

THE LOW POWER RESEARCH REACTORS

NESTOR

NESTOR is a light water cooled, graphite and light water moderated reactor, which operates at a power of up to 30kW and is used as a source of neutrons for a wide range of applications. A general view of the reactor, and its associated facilities, is given in Figure 1.

The core of the reactor is contained within an annulus formed by two concentric aluminium vessels through which demineralised water circulates. The fuel elements are made up of an assembly of 80% enriched uranium-aluminium alloy plates. The alloy fuel region is 0.5mm thick and clad in aluminium. Typically, sixteen fuel plates, 1.5mm thick, 70mm wide and 640mm long, are brazed together at the top and bottom to form a 70mm square fuel element. Channels exist for two rows of fuel elements, the current arrangement being shown in Figure 2. Vacant positions are filled with aluminium canned graphite blocks, with the spaces between adjacent channels occupied by similarly canned graphite wedges. The inner vessel is filled with graphite to form an inner reflector, in the centre of which is a californium start-up source. The outer tank is surrounded by an external graphite reflector, which contains six control plate slots adjacent to the vessel wall. In the present loading, four of the positions are occupied by two safety plates, one coarse control plate and one fine control plate. Each of these plates consists of a cadmium absorber, clad in stainless steel.

An important feature of NESTOR is the flexible design of the biological shielding around the reactor, which makes it possible to incorporate a wide range of experimental facilities. This is accomplished through the use of shielded caves on each face of the reactor, identified as Cave A to Cave D, and an extra shielded facility on top of the reactor. Boral shutters between the core and the caves allow experimental arrangements to be isolated from the reactor.

NESTOR first went critical in 1961 and commenced an intensive programme of thermal and fast reactor experiments which continued into the early seventies. This was followed by a shift in emphasis away from core measurements to the study of benchmarks and practical shield configurations. Cave C has been specially adapted to accommodate a shielding facility known as ASPIS (Activation and SPectroscopy In Shields) to provide experimental data on the penetration of neutron and gamma radiation through the shields of reactors, fuel reprocessing plants and other installations.

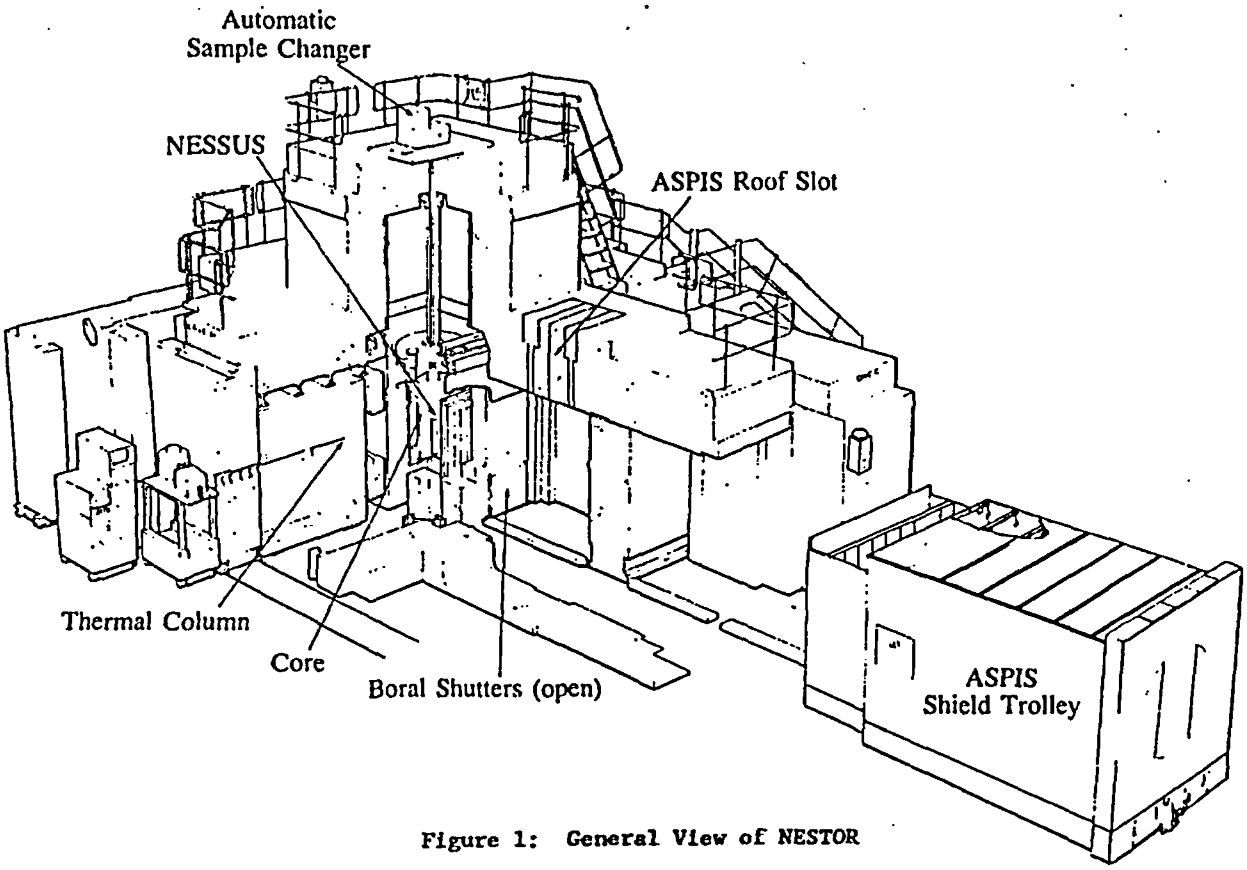

Figure 1: General View of NESTOR

The ASPIS facility is shown in Figure 3. Shield components, mainly comprising slabs or tanks, are mounted vertically in a mobile tank with an internal cross-sectional area of 1.8m x 1.9m and a length of 3.7m. A fission plate, of which there are several designs, is located within the experimental shield arrays. With the mobile tank positioned inside Cave C, thermal neutrons from the outer graphite reflector of the NESTOR core impinge on the fission plate and produce a fast neutron source of about 3×10^8 $n.cm^{-2}s^{-1}$. The absolute source strength is accurately determined by fission product counting and the spatial distribution via detailed flux mapping with activation detectors. Spectral shaping of the ^{235}U fission source for a given simulation is achieved by incorporating combinations of a range of materials between the fission plate and test shield. A slot in the roof of Cave C, above the test shielding configurations (see Figure 3), can be created for the study of cavity streaming problems.

An example of the application of the ASPIS facility is provided by a comprehensive study of neutron penetration through light-water reactor radial shields, including a simulated nozzle region, for the validation of calculation methods and data.[1] ASPIS is also the European fast reactor shielding facility, with a current experimental programme of neutron penetration studies through sodium and boron carbide.[2]

Cave C has also been used to irradiate a tank of sodium acetate solution, which was then pumped to a separate shielded compound to provide a pure high energy gamma-ray source. Adjacent to the compound is a dedicated gamma-ray facility known as ARCAS (Attenuation of Radiation using Cobalt Activated Sources).[3] It comprises a large shielded cave containing twenty-seven ^{60}Co sources with a total strength of 2×10^{12} Bq. Complex radiation fields are simulated by the computer controlled movement of the individual sources. ARCAS is used to calibrate dosemeters, perform energy deposition measurements with involved source distributions and provide experimental data for the assessment of gamma-ray code packages.

Cave D is filled with a 1.8m cube graphite stack, providing a "thermal column" facility (see Figure 1). The well-thermalised neutron flux ranges from 10^6 to 10^9 $n.cm^{-2}s^{-1}$ when the reactor is operating at full power. Its main function is for calibrating dosimetry packages and detectors, and the testing of components. In addition, by removing demountable sections of the graphite column, a horizontal beam hole of 95mm diameter is created. A set of thin borated polythene annuli of 60mm internal diameter are mounted in the shield door to provide a collimated neutron beam of $1.5 \times 10^5 n.cm^{-2}s^{-1}$. The beam is used to irradiate samples for the prompt gamma analysis determination of low levels of trace elements.[4]

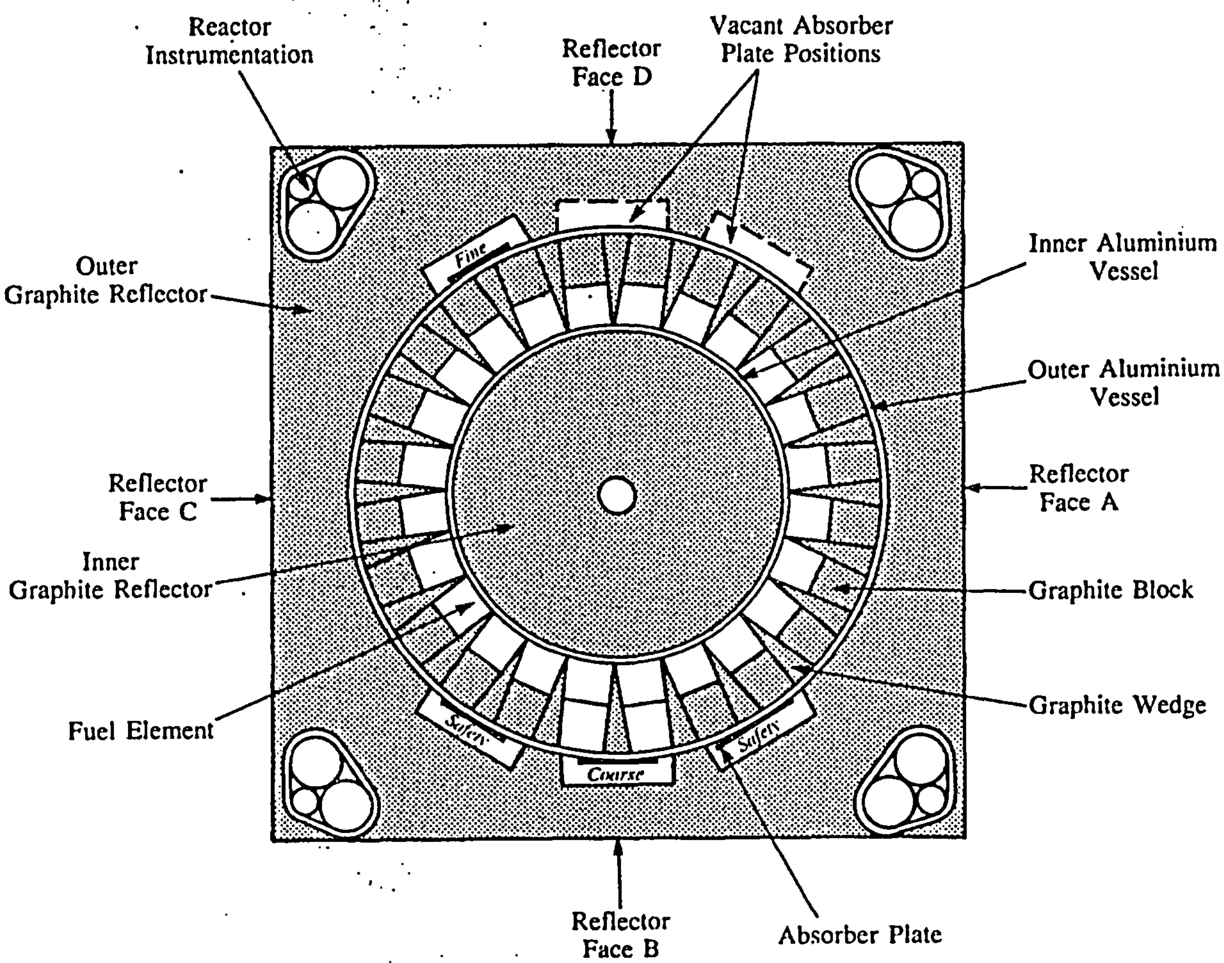

Figure 2: Plan View of the NESTOR Core Arrangement

A standard reference field has been established at the centre of the inner graphite reflector, with a peak thermal neutron flux of 3×10^{11} $n.cm^{-2}s^{-1}$ and fast flux (>0.1 Mev) of 5×10^{10} $n.cm^{-2}s^{-1}$. The neutron and gamma-ray spectra are well-characterised and provide a stable reference field for precision measurements with a wide range of integral detectors. In addition, the reference field is used to perform non-destructive analysis for the nuclear and petrochemical industries, employing both neutron activation and reactivity measurements.[4] These reactivity measurements are used for composition verification against reference standards and deriving neutron absorption cross-sections. The NESSUS facility[5] (Neutron Energy Spectrum Standard Uranium Source) provides a simple method of locating samples in the reference field. An automatic sample changer (see Figure 1) reproducibly inserts up to twelve individual samples in any desired sequence. The NESSUS facility is also used for the production of radio-isotopes.

Finally, the NESTOR reactor plays an important role in operator training and, as with the other two reactors, supports university projects.

DIMPLE

DIMPLE is a versatile, low power (<200W), water moderated reactor used to investigate performance, safety and safeguards issues relevant to the entire nuclear fuel cycle. A general view of the reactor is given in Figure 4.

DIMPLE's initial programme at Winfrith in the early sixties was largely aimed at validating the calculation methods and data used in reactor design and operational analysis. In September 1983, following refurbishment of the plant and the incorporation of modern nucleonics, criticality studies were included in the programme. Whilst the reactors varied experimental programme has been closely coupled to developments in the nuclear industry, many of the studies are of a generic nature and have provided a unique opportunity for technique development.

DIMPLE can accommodate a wide range of experimental configurations. Conventional assemblies consist of fuel pins supported, and precisely located, between upper and lower lattice plates inside a large aluminium tank

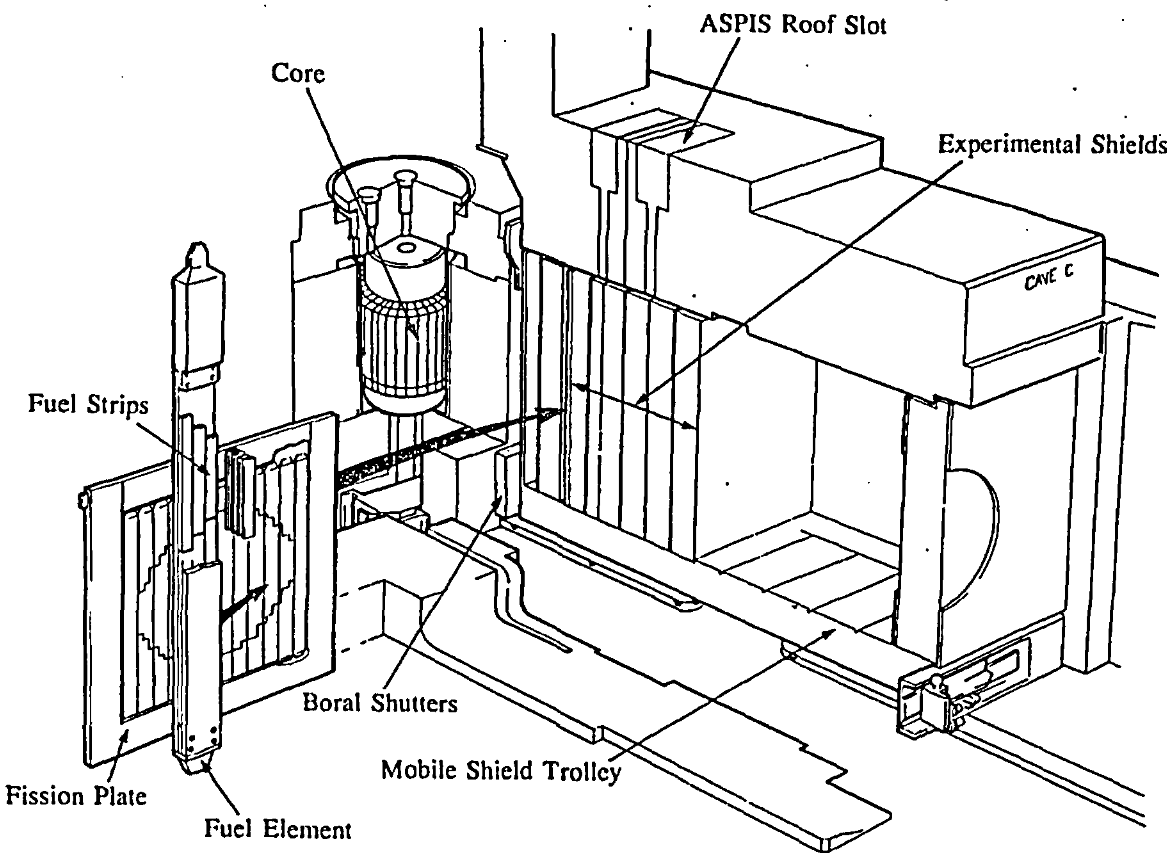

Figure 3: The ASPIS Shielding Facility

(2.6m diameter and 4m high). Standard DIMPLE fuel pins comprise sintered uranium dioxide pellets, of 2% to 7% enrichment, stacked within 10mm diameter cans. A range of mixed-oxide pins are also available. Both simple geometry fuel pin benchmarks and more complex configurations, representative of operational or accident conditions, can be built. Flexibility is accomplished by varying the lattice plate design, fuel type and the inclusion of non-fuel components such as structural or absorber materials. Designs have been investigated for other fuel geometries (eg plate fuel and solutions) and systems with neutron spectra ranging from fast to well thermalised. The ability to control the reactor by means of moderator level alone permits sub-critical and critical assemblies to be studied without the complicating perturbation of control rods. A bell jar mounted beneath the reactor core (see Figure 4) provides a fast internal-dump system. During reactor operation, the bell jar traps a bubble of air which, when released via the fast dump valves, results in the moderator level dropping by about 300mm in one second.

Lattice studies in DIMPLE are used to demonstrate the attainment of power reactor operational target accuracies for parameters such as pin power, absorber worths and void coefficient. Recent examples of such studies are provided by a series of benchmarks which form part of a step-by-step, diagnostic approach to the assessment of pin power predictions.[6] To extend studies in water reflected cylindrical systems to power reactor geometries, a cruciform array of 3% enriched uranium dioxide fuel pins was assembled. This simulated the rectangular corner configuration of a PWR and effectively represented twelve fuel elements. Extensive reaction-rate distribution measurements throughout each assembly have provided a wealth of data for the assessment of pin power predictions. The measurements, employing a wide range of fission and other activation foils, were performed by the internationally renowned Winfrith Counting Laboratory. In addition, absolute threshold reactions were measured at various locations spanning the equivalent PWR barrel position outside the cruciform core. The objective of this study was to assess the uncertainties in the radial fast neutron fluxes important to ex-core surveillance procedures.[7] The measurements extended the ASPIS studies outlined above by including the complex source geometry of the reactor core.

Similar lattice-type studies are used to investigate the impact of design changes on fuel manufacturing, transport, storage and reprocessing situations. Simulations of actual configurations provide a realistic test for the

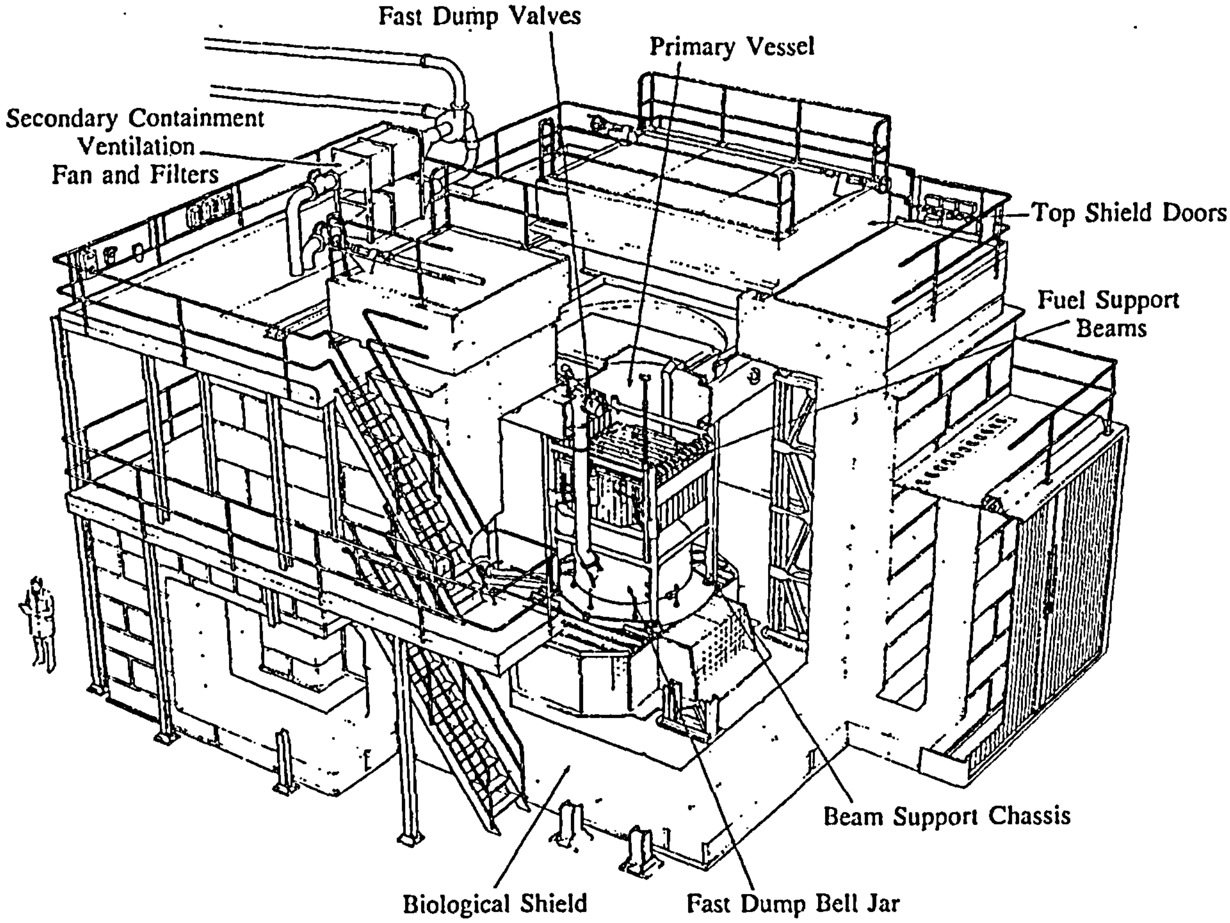

Figure 4: General View of DIMPLE

calculation methods and the assessment of margins for error. Criticality safety is vital to all fissile material operations and studies include both sub-critical and critical assemblies, allowing situations ranging from normal to extreme accidents to be simulated. An example of such an investigation is provided by a unique series of fuel pin arrays grouped within a prototypic boron-steel walled skip insert of the type used for the transport and storage of power reactor fuel.[8] An additional benefit of the monitoring techniques developed for the sub-critical investigations is their potential safety assurance and safeguards applications in a plant environment.[9]

Finally, measurements in DIMPLE with a range of irradiated fuel samples are providing experimental data on actinide and fission product reactivities, neutron source strengths and gamma spectra which, when coupled with chemical and isotopic analyses, is establishing a comprehensive, integrated data-base.[10] Such a database is fundamental to efficient and safe reactor fuel management, criticality assessment and providing the primary source data for shielding and accident analyses.

ZEBRA

ZEBRA was built for the purpose of performing zero power fast reactor physics studies and, in this role, had a continuous twenty year programme. The flexible design of the system (see Figure 5), and the use of simple plate and pin geometry components, allowed a wide range of different assemblies to be constructed. These ranged from small benchmark cores through to large complex assemblies simulating 1000MW conventional or heterogeneous power reactor designs.

For the past seven years ZEBRA has been on programmatic standby, with current core physics studies for the European Fast Reactor programme being performed at Cadarache. ZEBRA has therefore assumed the role of an experimental facility for multi-design studies, where it can be used for large "dry" core physics experiments inappropriate for DIMPLE. A recent example is the potential utilisation of ZEBRA for space reactor studies. Conventional type assemblies up to 4.0m x 3.4m x 2.7m high can readily be accommodated, with the $10^3 m^3$ containment providing the potential for housing even larger systems.

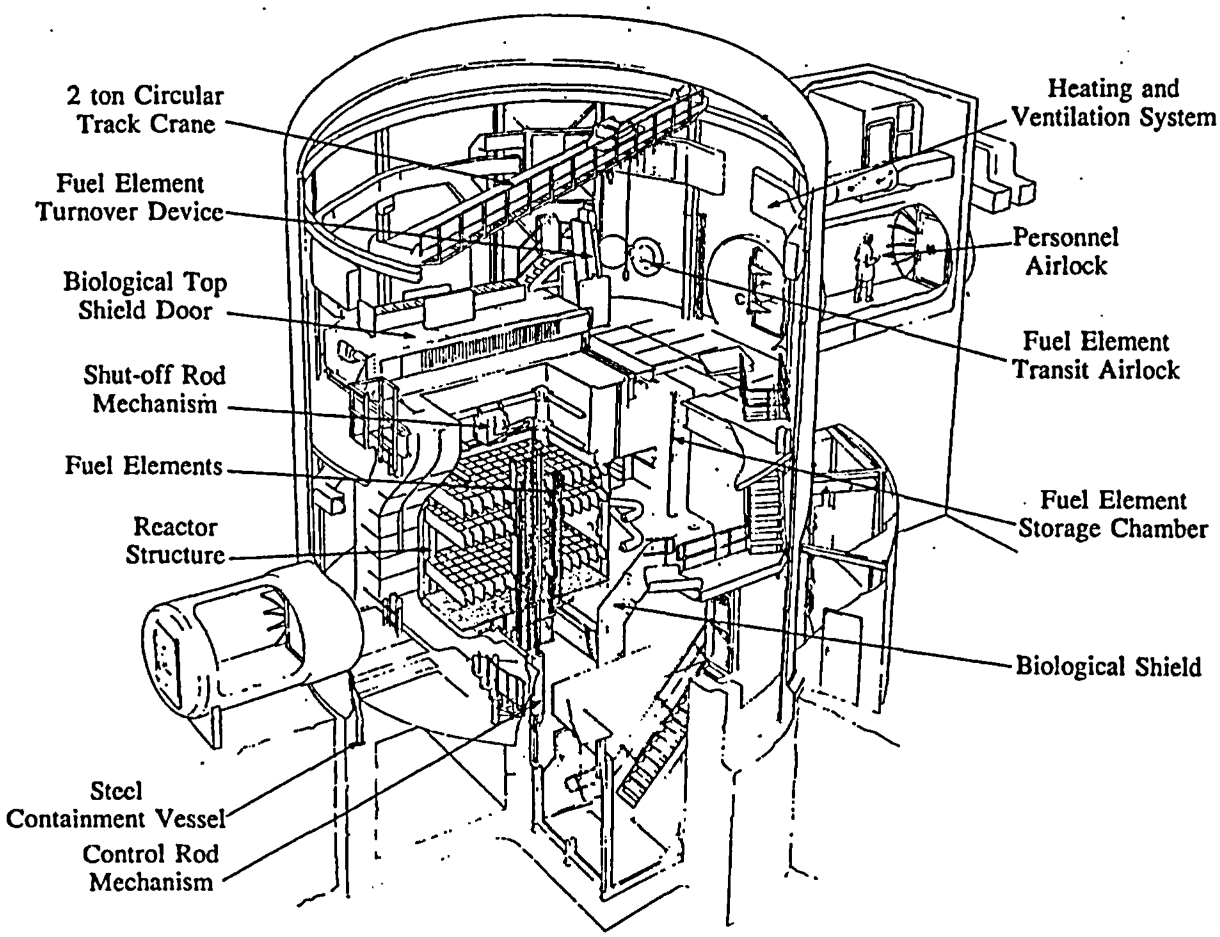

Figure 5: General View of ZEBRA

SAFETY OF THE LOW POWER RESEARCH REACTORS

Background

From October 1990, the Winfrith site will be licensed by the UK Health and Safety Executive (HSE) and regulated by the HSE's Nuclear Installations Inspectorate (NII). The HSE is the executive arm of the Health and Safety Commission (HSC), both bodies being established by Act of Parliament. Up until now, AEA sites have operated under a Government Directive to maintain standards equivalent to HSE licensed sites and therefore the new arrangements will not result in any major change in safety policy. NII inspectors already visit Winfrith under general powers which include the enforcement of the UK Ionising Radiations Regulations. The principal aim of the NII is to ensure a safe operating regime for civil nuclear installations and this is carried out by the performance of statutory functions under various Acts of Parliament. The objective is to achieve the compliance of operators with their duty to secure the health and safety of workers and of the public who might be affected by the hazards of ionising radiation from activities on the sites or from accidental releases.

All major plants on a licensed site require individual safety documentation. This is prepared by the plant operator for submission to the HSE. The safety cases for the Winfrith low power research reactors are peer-reviewed prior to submission and judged against corporate safety criteria and NII safety assessment principles.[11]

The Safety Document

The Safety Document lays down the safety principles adopted in the design and modification of the plant, and reviews the plant against modern standards. The safety management arrangements are also described, together with the requirements for supervision, manning and training of all staff concerned with operations. Finally, the operational experience of the plant is reviewed, in terms of both incident and dose statistics.

In the case of the Winfrith low power reactors, there has been no significant hazard, injury or radiation dose to personnel as a result of minor incidents or occurrences. By the careful planning of reactor operations, maintenance and experiments, the total radiological doses accumulated by workers are kept as low as reasonably practical, resulting in an average annual dose of less than 1.5mSv. At the time of writing, this compares with a statutory annual dose limit of 50mSv per worker. New lower dose limit recommendations are expected in the near future from the International Commission of Radiological Protection (ICRP) and the HSC. However, these are still likely to be at least an order of magnitude greater than the average values associated with low power reactor operations.

The Safety Document contains a systematic and comprehensive assessment of potential hazards, covering the mis-handling of fuel, reactor transients and externally initiated events such as earthquakes, impact or fire. Those events with non-trivial consequences are analysed in-depth to provide an estimate of their probability of occurrence and the associated risk to workers and the public. The possibilities of accidents in the Winfrith research reactors having off-site consequences are few and have an extremely low probability of occurrence. This is demonstrated by the fact that for DIMPLE the overall risk is dominated by the external events, in particular direct aircraft crashes. However, even under these extreme circumstances, the off-site consequences are small.

Accident initiating events of major importance to power reactors are shown to be irrelevant for the low power research reactors. For example, complete loss of coolant from NESTOR, which is the only research reactor operating at any significant power level, poses no threat because the moderation function of the water would also be lost and result in reactor shutdown. The decay heat level is such that it would be removed by natural convection of the air and conduction to surrounding structures. No situation involving total or partial loss of electrical supplies has been postulated which would not result in shutdown of the reactor, and there are no subsequent requirements for power to maintain safety functions.

The risk assessments for the Winfrith research reactors have used grossly pessimistic assumptions, such as the ground level release of the entire fission product inventory from the fuel and no local condensation or plate-out. Despite such pessimisms, the upper limit risk of an early death to the most exposed member of the public is still well within the site target of 10^{-6} per annum and the share allocated to each of the low power reactors of 3×10^{-8} per annum.

Finally, although the reactors are expected to be available into the foreseeable future, the Safety Document details the philosophy of decommissioning. The long term objective is that the plant areas will be cleared completely of all reactor components and radiological hazards, making the sites available for alternative use. DIMPLE has already been decommissioned at the AEA's Harwell site prior to its move to Winfrith in the early sixties. In addition, its sister reactor, JUNO, has been successfully decommissioned to clean laboratory levels. In the case of NESTOR, no special techniques or equipment will be required beyond those already developed for use in the periodic shutdowns.

Performing Reactor Experiments

The safety case for the low power research reactors basically results in an enabling document specifying the operating rules, conditions and limits. Specific reactor experiments are designed to be carried out within this envelope, thereby minimising the impact of licensing activities on the programmes of the research reactors. Site safety committees, containing external members, are appointed to review all submissions before approval to operate is agreed.

Central to the safety procedures is the preparation of a Core Certificate, which must be fully signed before any major change. The Certificate has two main purposes.

(a) It ensures a cautious approach by requiring the principal features to be specified in advance. Formal signatures ensure decisions have been made at the proper level of responsibility as laid down in the Safety Document.

(b) It specifies those features, important to safety, which must be checked once the experimental change is completed.

In the case of DIMPLE, for example, the descriptive features of the Core Certificate include a lattice diagram showing the location of the fuel and detectors, a summary of the quantity and dimensions of the fuel used, the maximum fuel beam loading and the location of the neutron source. Details are also given of the quantity of moderator, the rates of moderator addition and the efficiency of the fast dump system, all of which are obtained from measurements in a dummy assembly prior to loading fuel. Before the lattice is assembled, calculations are performed to estimate its reactor physics properties and to demonstrate that, in conjunction with the measured parameters, the overall requirements of the Safety Document are met. The calculated data include the delayed neutron fraction, prompt neutron lifetime and temperature coefficients. Predicted values for the critical height, or k-value in the case of a sub-critical assembly, the water worth at a range of specified heights and the flux levels at which the reactor instrumentation will activate a trip, are subsequently verified by measurement.

Once the safety and reactor physics properties of a reactor core have been confirmed, experimental studies may be performed within additional well-defined operating limits specified in the Safety Document. These include a limit on the change in reactivity resulting from modifications carried out at shutdown (eg alterations to a DIMPLE lattice loading or ASPIS shield configuration) and a limit on the step reactivity change in the critical reactor. For example, the maximum changes permitted for sample reactivity measurements in NESTOR and DIMPLE are 60mNile and 120mNile, respectively.

SUMMARY

The low power reactors of the AEA provide a comprehensive research capability. Their versatility and continual enhancement have assured a successful, on-going experimental programme. Procedures for the safe execution of experiments have operated effectively at Winfrith for thirty years. Critical reviews of the plants, in preparation for licensing by the UK Health and Safety Executive, have confirmed the safety of the reactors for future operation and identified no factors which will limit their useful life. The ability of the AEA's low power research reactors to under-pin future advances in performance and safety throughout the fuel cycle is therefore available to the nuclear industry world-wide.

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