COMPUTER ASSISTED LEARNING FOR REACTOR PHYSICS EDUCATION AND TRAINING OF NUCLEAR PLANT DESIGNERS, OPERATORS AND MAINTAINERS

J. Brushwood and P.A. Beeley
Nuclear Department, HMS SULTAN
Military Road, Gosport, Hampshire, UK. PO12 3BY
physics_DNST@dial.pipex.com

R. Beadnell, J.L. Robertson, J.M. Warden
Nuclear Department, HMS SULTAN

ABSTRACT

This paper discusses the variety of teaching methods employed by the Nuclear Department in the education and training of nuclear practioners in support of the United Kingdom's Naval Nuclear Propulsion Programme (NNPP). Reactor physics is required to be taught at a variety of levels from introductory principles to postgraduate, a variety of teaching methods are employed to satisfy this requirement. This paper will introduce all of these teaching methods but will focus on the use of computer-assisted learning. The three main computational tools *Core XYZ*, *M32* and industry standards (*WIMS*, *MONK* and *MCNP*) will be described and examples of their application given. Some sample results and applications of industry standard codes from both the taught and dissertation phase of the Departments MSc. Nuclear Technology and Safety Management will be provided.

1. INTRODUCTION

The Nuclear Department at HMS SULTAN was formed in 1998 by the amalgamation of the Department of Nuclear Science and Technology, previously established at the Royal Naval College Greenwich in 1951, and the Nuclear Systems Group (HMS SULTAN). Approximately fifty courses are taught ranging from a one-week fundamentals course to a full-time one year Masters Degree in Nuclear Technology and Safety Management [1]. The University of Surrey provides external postgraduate validation for five of these courses. The courses are designed to provide suitably qualified mechanics, diagnosticians, plant engineers, maintenance personnel, health physicists and designers. Lecturing staff comprises Senior Lecturers from Flagship Training Ltd, Civil Service academics and Naval operators.

The Department aims to both educate and train personnel [2]. The accident at Three Mile Island and Chernobyl showed how important it is to achieve the correct balance between education and training. If education is neglected, then problems of understanding plant conditions could arise when, owing to plant failures, the plant deviates from the normal mode of operation [3]. Safe operation and control of the plant then depends to a great extent on how the operator has been educated in the theoretical background to his training. Owing to the wide variety of courses run by the Department, reactor physics must be taught at many different levels, building from a basic introduction to atomic physics up to transport theory and Monte Carlo methods. Prior to discussing the application of computer-assisted learning in reactor physics education and training, this paper will introduce all of the teachings methods and media encountered by students. Broadly, these support functions may be separated into three areas, namely; traditional lectures and workshops, laboratory based experiments, project work and simulator training.

2. EDUCATION AND TRAINING SUPPORT FOR REACTOR PHYSICS

2.1 LECTURE MATERIAL AND WORKSHOPS

The traditional lecture and workshop/tutorial format is used throughout the various courses. Reactor physics in introductory courses is based on an examination of the simplest formulation of the neutron life cycle, the one energy group three-factor formula. This is extended to the two energy group six-factor formula in later courses. Similarly, diffusion theory is introduced by a one-group model and then developed to two groups eventually examining multi-group models. Interwoven with these lectures is the use of the in-house computer assisted learning package *Core XYZ*, discussed later. The culmination of this development, at MSc. level, is the detailed discussion of transport theory.

The latter topic is covered in the “Advanced reactor physics” lecture series in which students are introduced to the various mathematical approximations and the analytical solutions used in transport calculations. Practising reactor physicists and engineers from Naval design authority and industry visit to lecture on the practical applications of reactor physics in design and support of the naval nuclear propulsion plant.

In all cases quite detailed, self contained, lecture notes are given to the students and extensive use of electronic information systems and multi-media is encouraged whenever possible. Lectures are consolidated and reinforced during tutorial sessions. As computer assisted learning becomes more available the old format of tutorial is being replaced by interactive workshops. A well-equipped library ensures the students have access to source academic and technical material, including the classical textbooks in the subject.

2.2 LABORATORY PRACTICAL AND PROJECT WORK

Experimental work is undertaken by virtually all courses and varies from, at the introductory level, simple atomic and nuclear physics practicals i.e. an examination of neutron interactions with matter using the Departments' Am/Be neutron sources to, at the MSc level, whole day assignments at the Imperial College CONSORT research reactor and the Department's 14 MeV d-T neutron generator facility. Reactor based experiments may include; (a) Control Rod Approach to Critical; (b) Determination of Integral Control Rod Reactivity Worth; (c) Excess Reactivity of the Core; (d) Measurement of the Moderator Temperature Coefficient of Reactivity.

The MSc. students are required to complete both a series of assignments and a dissertation. Students on the other postgraduate courses undertake a short project, approximately six weeks in length. The aim of the short project or assignment is to provide the students with the opportunity to apply their academic knowledge, computational and experimental skills in reactor physics to solving a reasonably challenging problem. A recent project offered was the determination of in-core power in low energy research reactors by measurement of ^{16}N and ^{18}F in the primary coolant [4]. In this project the deterministic code *LWR-WIMS* was used to estimate the thermal to fast flux ratios required in a set of ab-initio equations relating core power to ^{18}F activity in the primary coolant.

Postgraduate students are required to submit a written report and give an oral presentation on their projects, which are assessed by the staff and an external examiner.

2.3 SIMULATOR TRAINING

The initial training of submarine reactor plant operators, and their subsequent continuation training, is carried out after completion of courses at the Nuclear Department using full-scope simulators that replicate the reactor control room (reactor, electrical and propulsion control areas) of a submarine. These simulators use a combination of describing equations, look up tables and line fit extrapolations to display realistic indications on instrumentation to the operating team in real time. Students are trained to operate the plant in accordance with prescribed operating procedures and on completion of a very intensive course are ready to take up their appointment within the Fleet as trainee engineers.

The simulator training undertaken within the Nuclear Department [5] as part of the academic courses is of a more fundamental nature and is designed to convey basic physics, reactor engineering and plant operating principles in a manner that does not overwhelm the student with systems related information. Output from this Basic Principles simulator is displayed on an array of 90 cm monitors forming a display array system, or "telewall", shown in Figure 1a.

Physical laws were chosen as the basis of the reactor modeling as this would allow the reactor simulator to be taken outside the normal operating region to demonstrate various physical and failure processes. The reactor simulation was developed in house and the model uses a coupled point kinetic/thermo-hydraulic set of equations. The equations used in the modeling are covered in lectures: for example, reactor kinetics is modeled using a four node, six-delayed neutron group model.

A true simulation requires these equations to be solved at least once a second so that the results can be displayed in real time. In practice the equations must be solved more rapidly to make the eye sense continuous motion. This is problematic when solving differential equations with fast time constants. Mathematical techniques, such as the prompt jump approximation, are introduced with a consequent loss of accuracy to keep the simulation manageable [6]. It is possible to extract any of the modelled parameters and display time variations in a manner not available on the full scope simulators.

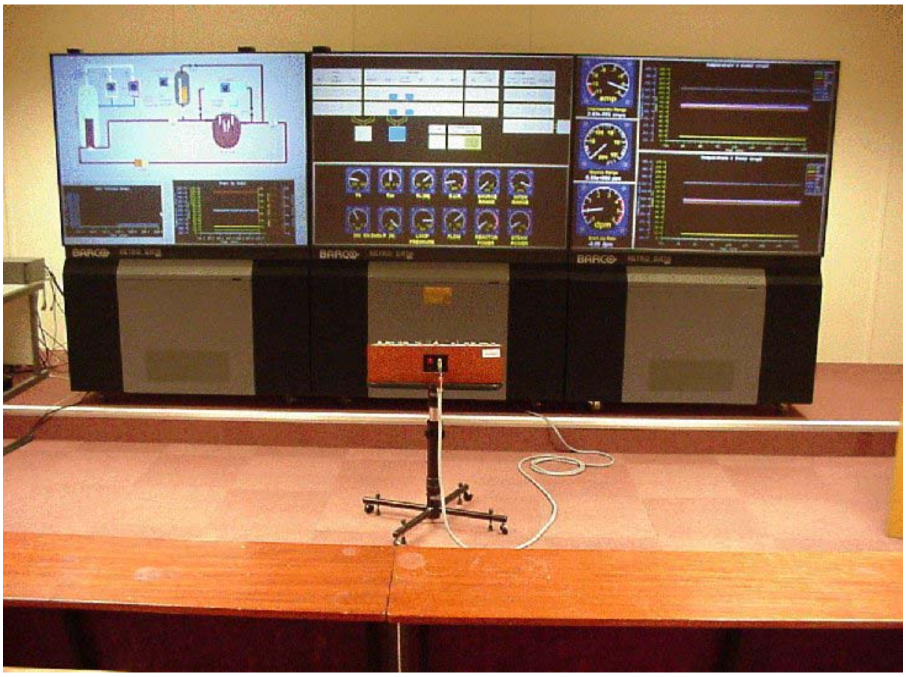

Figure 1a. Telewall Basic Principles Simulator

A typical use of the simulator is to take the "reactor" from a fully shut down state to critical, warm up primary coolant to the normal operating zone, then demonstrate the plants load-following or self – regulating behaviour. A typical output screen showing the variation of various parameters following a steam demand (yellow line) on a hypothetical civil plant is shown in Figure 1b. The age of the reactor can also varied in a simulator exercise to demonstrate longer-term effects on the plant, for example how changes in the temperature coefficient of reactivity affects power and temperature transients.

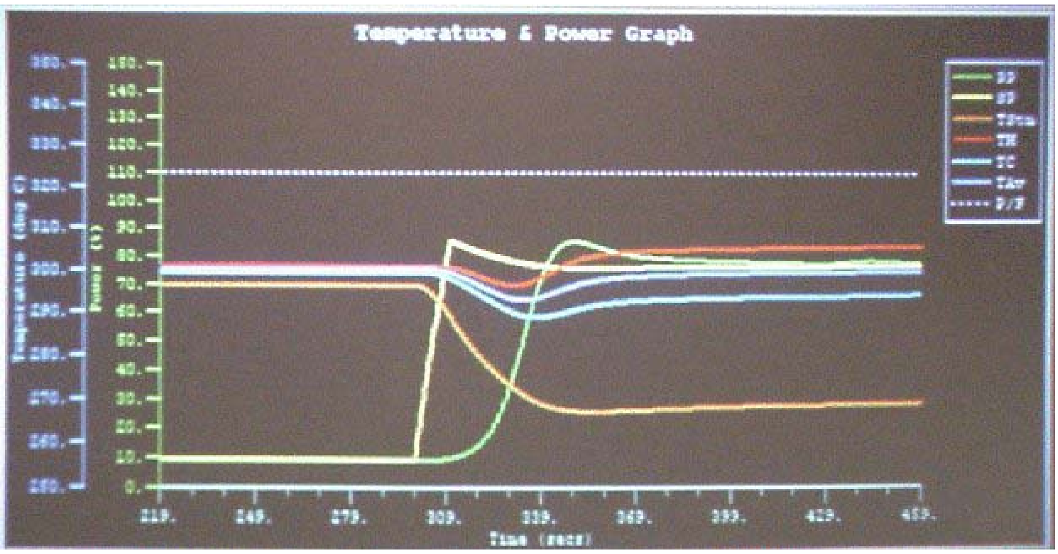

Figure 1b. Telewall screen showing response to increasing power demand

3. COMPUTER ASSISTED LEARNING

The use of computer assisted learning in education is now well established and full use of this technique is made in order to enhance the teaching of reactor physics. Three distinct approaches are adopted: the in-house two group point kinetic code, *Core XYZ*; a linked thermodynamic and two group neutronics code *M32*; and ultimately computer modeling using industry standard deterministic and probabilistic codes.

3.1 THE *CORE XYZ* PACKAGE

The *Core XYZ* model has been developed in Visual Basic as a computer assisted learning package that can provide a simple three-dimensional mathematical representation of a compact light water reactor. It takes the students through a series of exercises (Figure 2) for a homogeneous parallelepiped core in order to calculate reactor parameters and fluxes, with the reactor reflected and unreflected, and compares the one and two group results. Kinetic calculations are also undertaken, but these are limited to the one energy group model with one group of delayed neutrons. The students complete the exercises in *Core XYZ* in stages that correspond to the level of reactor physics being taught in the classroom. They thus gain confidence in their ability and consolidate their knowledge of the derivation of various reactor parameters, and their approximate magnitude.

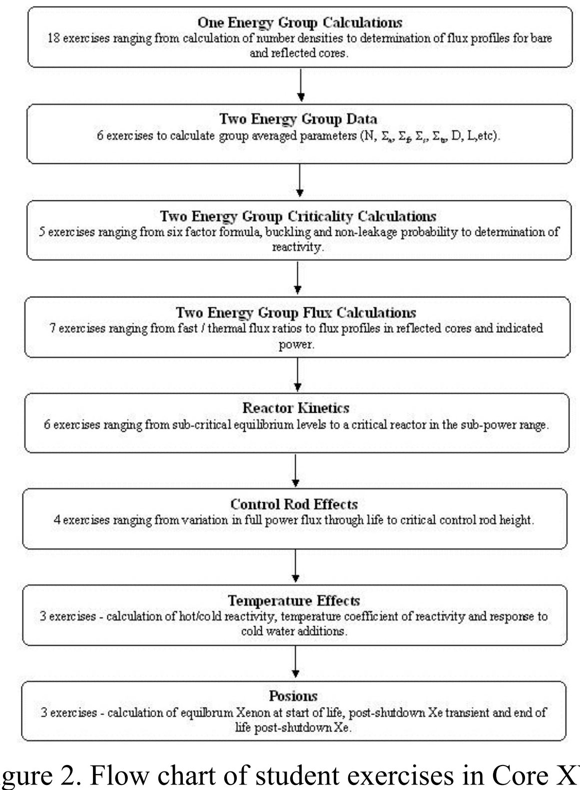

Figure 2. Flow chart of student exercises in Core XYZ

3.2 THE *M32* PACKAGE

The *M32* code has been developed by Moorby [7] in C+ as a simple reactor physics design tool for scopeing a more detailed design using industry standard codes. It is used to introduce to students the coupling that exists between the physics calculations, for example the determination of power distributions or reactivity, and thermal calculations such as the determination of coolant module inlet and outlet temperatures. *M32* allows the students to explore, through an interactive simulation, the three dimensional two energy group time dependent response of a compact pressurized water reactor to disturbances arising from control rod movements or changes to coolant temperatures or flow rates.

To achieve a code that can solve these coupled neutronics/thermo-hydraulic calculations yet still be implemented on in a “modest” PC platform requires the introduction of a number of approximations. The approach taken by *M32*, and introduced to the students in lectures, is to separate the neutronic and thermal calculations by assuming that in the quasi-steady state conditions encountered during routine operations the nuclear data that is driving the physics calculations will be only be weakly affected by the thermal variations. Under this assumption an initial power distribution using appropriately thermally averaged nuclear data based on an estimated initial thermal condition (e.g. whole core set to average coolant temperature) is determined. Subsequently this initial thermal distribution is refined but without re-calculating the power distribution.

Having determined a steady state solution the student can examine the effect of control rod movements or coolant properties. To allow for feedback from the varying thermal conditions simple curve fits are used to determine the coolant properties. A six group delayed neutron scheme is implemented for reactor kinetics calculations.

M32 assumes an xyz coordinate system with the z-axis aligned vertically, the core is assumed to be symmetric about the x = 0 and y = 0 planes. The core region is divided into a number of cells with dimensions Δx Δy Δz, a vertical stack of cells, i.e. constant z, is referred to as a stick. The user is required to input the required data such as dimensions and nuclear data for each of these cells. A finite difference scheme based on the conservation of fast and thermal neutrons (see Equations 1 and 2 respectively) is used for neutronic calculations.

$$\begin{aligned}\frac{1}{v_1}\frac{d\phi_1}{dt} = &\frac{d}{dx}\left(D_{x1}\frac{d\phi_1}{dx}\right)+\frac{d}{dy}\left(D_{y1}\frac{d\phi_1}{dy}\right)+\frac{d}{dx}\left(D_{z1}\frac{d\phi_1}{dz}\right)\\ &-\phi_1\Sigma_{a1}-\phi_1\Sigma_{s1}+\phi_2\Sigma_{s2}+\frac{v}{k}(1-\beta)(\Sigma_{f1}\phi_1+\Sigma_{f2}\phi_2)+\sum_{i=1}^{6}\lambda_i C_i+S_1\end{aligned} \tag{1}$$

$$\begin{aligned}\frac{1}{v_2}\frac{d\phi_2}{dt} = &\frac{d}{dx}\left(D_{x2}\frac{d\phi_2}{dx}\right)+\frac{d}{dy}\left(D_{y2}\frac{d\phi_2}{dy}\right)+\frac{d}{dx}\left(D_{z2}\frac{d\phi_2}{dz}\right)\\ &-\phi_2\Sigma_{a2}-\phi_2\Sigma_{s2}+\phi_1\Sigma_{s1}+S_2\end{aligned} \tag{2}$$

Heat flow in the fuel elements is assumed to occur only in the direction normal to the heat transfer surface (i.e. cladding) and is modeled with an equation of the form indicated in Equation 3.

$$(\rho c_p)\frac{dT}{dt} = \frac{d}{dn}\left[K\frac{dT}{dn}\right] + q(n) \qquad (3)$$

where n is the direction normal to the heat transfer surface

K is the thermal conductivity

q is the heat generation rate per unit volume of fuel

3.3 INDUSTRY STANDARD CODES

With regard to the use of industry standard codes the MSc. students principally use these in the dissertation phase of their course where a design study is undertaken. The objective in using these codes is to allow the students to develop the skills and rationale required to design and or model real reactor cores.

Available within the Department are a wide variety of reactor physics codes. These are traditionally separated into deterministic and Monte Carlo. Use is made of both commercial packages such as the Serco Assurance ANSWERS suite of codes *WIMS 8A* [8] and *MONK 8B* [9] and NEA databank supplied codes such as *MCNP 4c2* [10] and *WIMS-ANL* [11].

MONK 8B and *MCNP 4c2* are neutronic calculation codes operating on the basis of the Monte-Carlo method. It is primarily aimed at the calculation of the multiplication factor (k) of some system, however quantities such as neutron flux and hence reaction rates can also be determined through the use of so-called tallies. To calculate k the code simulates the birth, migration and disappearance of a finite number of neutrons within the system, so called histories.

The *WIMS 8A* is the latest commercial version of the well known deterministic reactor physics code *WIMS-D* [12]. The code has an open structure that comprises a set of methods (or solvers), which the user can link together to form a calculational scheme to solve a wide variety of problems encountered in thermal reactor physics. The user controlled open structure of *WIMS* allows problems from simple homogeneous cells to complex whole core calculations to be performed. The *WIMS-ANL* code was developed by the Argonne National Laboratory from the original *WIMS-D* code. Three important capabilities have been added; the generation of broad-group burnup dependent cross-sections, an ENDF/B-V based nuclear data library and a supercell option.

Primarily, the NEA codes are used in the final taught phase of the MSc. where three short optional modules are completed. A recent example of such a study undertaken in the "Computational methods in radiation transport" option was the evaluation of a proposed low enriched TRIGA type uranium-zirconium hydride pin fuel, shown Figures 3 and 4 are the initial pin cell model and the 5 by 5 fuel assembly studied.

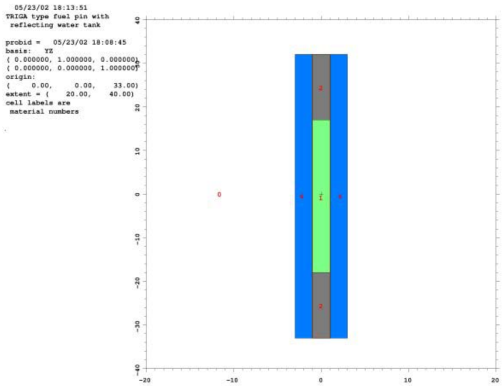

Figure 3. Pin cell model of TRIGA element

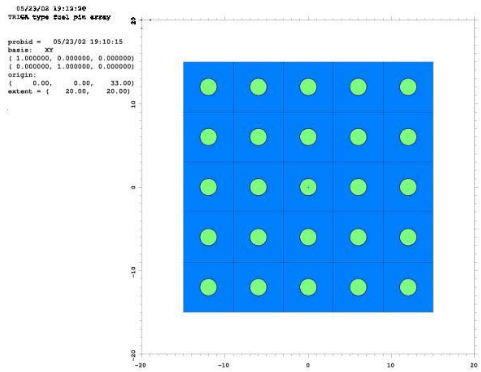

Figure 4. 5 by 5 fuel assembly

3.4 DESIGN STUDY

The culmination of the MSc. is the submission of a dissertation; this is traditionally conducted along the lines of a "design study". The design study is considered important in providing the students with the opportunity to work as part of an engineering team. Assessment, again, is by staff and an external examiner. Each student works on a specific design aim such that the combined efforts of all the students' results in a complete nuclear plant design. For example; one student may chose to concentrate on the core design calculations; a second focusing on the thermo-hydraulics/kinetics study and a third examining radiation protection and shielding. Although primarily responsible for their own study areas, each student is required to maintain a good understanding of the other study areas to appreciate the impact that choices he or she may make upon the overall plant design. In addition to the individual reports the students are required to produce a joint project management report that is an integral part of the whole study. A recent MSc. design study examined the feasibility of a high-temperature reactor for marine applications [13].

The starting point for the design study is the production of a user requirement document (URD) by staff members in the Department. The URD specifies the requirements such as thermal power and any constraints on the design for example the maximum allowable uranium enrichment. By way of example the URD for the recent design study of a high temperature reactor for marine applications specified that the plant must have a thermal output of 180 MW with a maximum power density of 3 MW m^{-3}, use a basic fuel element of "similar" design to those of the DRAGON [14] reactor and fit within a hypothetical 10 m cubic reactor compartment. The final design study core configuration is shown in Figure 5 and required 169 hexagonal graphite blocks each containing 19 embedded fuel elements with each fuel element containing 771300 fuel microspheres.

Extensive use was made of both *MONK* and *WIMS* to study amongst many other issues the initial and through life reactivity for un-poisoned and gadolinium poisoned cases, shown in Figure 6. The overall result of this study and optimization was to achieve a near constant k_{eff} of close to 1.1 up to a total burnup of 30000 MWd/T.

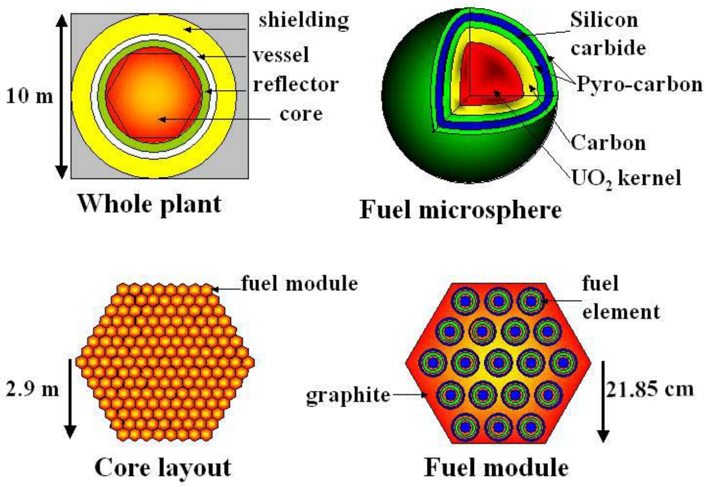

Figure 5. Marine HTGR core configuration

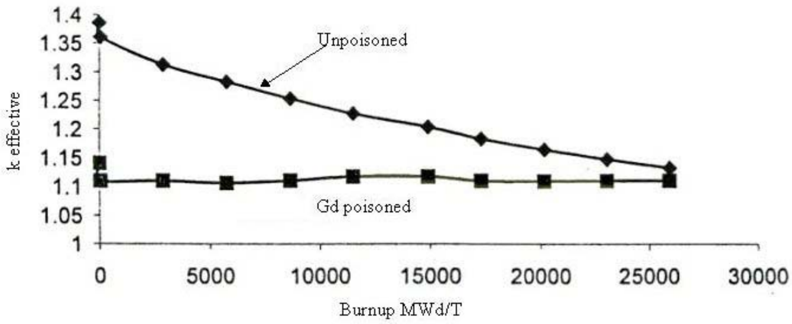

Figure 6. k_{eff} vs burnup for a Marine HTGR plant

4. CONCLUSIONS

The teaching of reactor physics, especially at the postgraduate level, is aimed at:

a. Providing a sound theoretical basis through lectures and workshops;
b. Consolidating and supplementing this knowledge with computational and experimental exercises;
c. Improving modeling and numerical skills using computer-assisted learning;
d. Training of plant operations within and outside normal operating regimes using digital computer simulators.

Over the last decade computer assisted learning has become a integral part of all reactor physics teaching and has greatly enhanced the students ability to assimilate and consolidate difficult concepts in this subject. It is believed that the balance between education and simulator training provides the nuclear engineer and associated personnel with the ability to support the design, build, operation and maintenance of the naval nuclear plant to the highest standards of safety.

5. DISCLAIMER

Any views expressed are those of the authors and do not necessarily represent those of the Nuclear Department or those of Her Majesty's Government. Nor should any mention of any commercial product be taken as an endorsement by the Nuclear Department or Her Majesty's Government.

6. REFERENCES

1. J.R.A. Lakey and D.J. Robb, "25th Anniversary of nuclear courses at the Royal Naval College", *Nuclear Engineering*, **25**, pp.148 (1984).
2. J.G. Kemey, *President's commission on the accident at Three Mile Island*, Pergamon Press, New York USA (1979).
3. P.G. Lisley and P.A. Beeley, "Reactor physics in support of the Naval nuclear propulsion programme", *Proceedings of the International Conference on Reactor Physics and Reactor Computations*, Tel-Aviv Israel, January 23-26, (1994).
4. P.A. Beeley, J.M. Brushwood, M.G. Henesy, M.W. Collins, C.A. Haywood, "Determination of in-core power in low energy research reactors by measurement of ^{16}N and ^{18}F in the primary coolant", *Journal of Radioanalytical and Nuclear Chemistry*, **215(1)**, pp. 135, (1997).
5. M.A. Gale and A.C.Thompson, "Simulating reactor dynamics", *Proceedings of the 4th National Conference of the UK Simulation Society*, St Catharine's College Cambridge UK, April 7-9, (1999).
6. A.G. Wills, "Pressurised water reactor simulation in the training environment", *Proceedings of the 2nd European Simulation Symposium*, Schliersee, Germany, October 22-24, (1990).
7. J.H.T. Moorby personal communication (2001)
8. WIMS User Manual, Serco Assurance (2000)
9. MONK User Manual, Serco Assurance (2000)
10. J.F. Briesmeister, *MCNP – A general Monte Carlo N-Particle transport code LA-12625-M*, Los Alamos National Laboratories (1993).
11. J.R. Deen, W.L. Woodruff, C.I. Costescu, L.S. Leopando, *WIMS-ANL user manual ANL/RERTR/TM-23,* Argonne National Laboratory (2001).
12. M.J. Halshall, *A summary of WIMS-D4M input options AEEW-M-1327,* Atomic Energy Establishment Winfrith (1980).
13. P. Lobet, R. Seigel, A.C. Thompson, R. Beadnell, P.A. Beeley, "A high temperature reactor for ship propulsion", *Proceedings of the 1st International Topical Meeting on High Temperature Reactor*, Petten, the Netherlands, April 22-24, (2002).
14. J.L.Head and A.N. Kinkead, "Graphite fuel element structures for high temperature gas cooled reactors", *Nuclear Engineering and Design*, **18**, pp. 115, (1972).