Neutronic Analysis of the Penn State Breazeale Nuclear Reactor Using the MURE Computer Code

Dağıstan Şahin,[1,a] Kenan Ünlü,[1,2] Kostadin Ivanov[2]

Service Provided: Penn State Breazeale Reactor

Sponsors: The Penn State Radiation Science and Engineering Center

Introduction

The Pennsylvania State University (PSU) Radiation Science and Engineering Center (RSEC) hosts the Penn State Breazeale Nuclear Reactor (PSBR), a TRIGA Mark III-type 1 MW nuclear reactor. The PSBR has a well-documented and analyzed operation history starting from 1955. The initial reactor design was a Material Testing Reactor, but was replaced in 1965 with the TRIGA reactor core. The PSBR operates on a flexible schedule based on research, educational, and service needs. The specific power and timing requirements of this schedule causes a fluctuation in neutronic characteristics of the PSBR core [1]. Using the Monte Carlo Utility for Reactor Evolution (MURE) libraries, a burnup coupled neutronic analysis tool for the PSBR has been developed to simulate variant neutronic characteristics of the PSBR core; a simplified version of this simulation has been used previously to calculate neutron self-shielding factors for activated samples in the PSBR core [2]. For this work, the model has been extended to evolve a five-section fuel element. MURE offers an easy to understand object oriented environment which includes a powerful burnup calculation module [3], [4]. MURE uses cell averaged energy dependent neutron flux (F2-Tally) and cross section data from MCNP5 and builds Bateman's equations for each isotope in matrix form. Equations are then integrated using numerical algorithms. Results obtained using MURE for the PSBR neutronic simulations are compared both with measurements and an advanced neutronics simulation tool, called TRIGSIMS, results [5], [6].

Simulation Design

MURE is an advanced code library written in C++ for nuclear reactor evolution with integrated burnup calculations, MCNP, and thermal hydraulic coupling mechanisms [7], [8]. MCNP5 in MURE is used to perform neutronic calculations. MURE provides class libraries to create MCNP input and load results to effectively perform burnup calculations [9], [10]. MURE provides a cross-section generation tool and a user interface for data analysis.

Reactor Components

Technical drawings, documentation, and software manuals at the RSEC were analyzed for geometry and composition information for fuel elements, control rods, air tubes, graphite elements on the core periphery, the core lattice, the pool, and the heavy water (D_2O) tank [5], [11].

The PSBR bare core model in MCNP is shown in Figure 1. The PSBR core is modeled to be in a pool (width 2 m, thickness 2 m and height 3.25 m) that is filled with high-purity water at 300 K. A D_2O tank is also coupled to the PSBR core for beam port irradiation experiments. The D_2O tank is modeled as an aluminum cylinder container filled with D_2O, having a radius 29.52 cm and thickness 28.57 cm. There is a re-entry hole in the D_2O tank for the beam port neutron guides to conjugate.

Fuel Elements

The fuel meat of each fuel element is divided into five individual sections as shown in Figure 1. Material information and burnup calculations are performed separately for each of these sections. The temperature of each section is set individually using linear interpolation between temperature measurements performed for instrumented fuel elements and locally generated power within the section.

Control Rods

There are four control rods used to control and drive the PSBR nuclear reactor. Three of them are fuel-follower control rods, called the Safety Rod (SA), the Shim Rod (SH), and the Regulating Rod (RR). The last control rod is an air-follower control rod, called the Transient Rod (TR). The fuel meat of the three fuel-follower control rods is divided into five sections, as displayed in Figure 1.

Graphite Element

The graphite reflector elements in the aluminum cladding are used as a reflector around some parts of the PSBR core, since Core Loading (CL)-53G (2009).

[1] **Author 1 affiliation, e.g. Radiation Science and Engineering Center, The Pennsylvania State University, University Park, PA 16802**
[2] **Author 2 affiliation**
[a] **Author N affiliation**

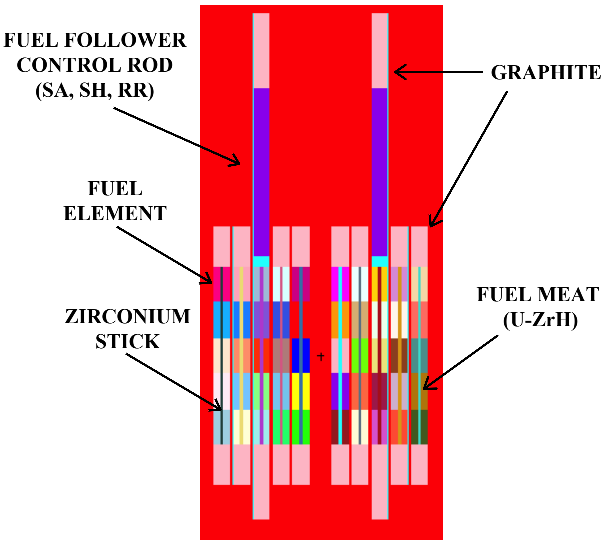

Figure 1: MCNP model of the PSBR core, side view.

Results

PSBR core characteristics are measured annually and for each new core loading pattern. The developed simulation tool has been used to perform burnup coupled neutronic calculations of the PSBR for core loadings from 1966 to the present. Core excess reactivity and control rod worth values are calculated and compared to measurements. The maximum time step for any burnup calculation is limited to 10 days, in which excess reactivity loss is ~25 cents at an average operating power of 0.7 MW. Due to this low change in composition between two steps, it was not necessary to use the predictor-corrector algorithm for burnup calculations. Only major core loadings are represented in the calculations.

Core excess reactivity at the beginning of each core loading is calculated and compared with measured values and TRIGSIMS results as presented in Figure 2.

Rod worth for the four control rods (SA, SH, RR and TR) is calculated at the beginning of each core loading throughout burnup calculation in 1967-Sep 2011. It is then compared to the measured values. As an example, measured and calculated rod worths for SA are plotted in Figure 3.

Four fuel elements were removed one at a time out of the PSBR core and their worth was measured on April 26, 2004. The D_2O tank worth was measured on January 9, 2011. Experimental conditions are simulated and the predicted fuel element and D_2O tank worth values were compared with measurements, taking into account measurement uncertainties, in Table 1.

Discussion of Results

There is a large deviation between excess reactivity measurements and calculated values from 1973 to 1985, as shown in Figure 2. The PSBR core was controlled manually until 1991, when the first automatic control system was installed. Before 1991, hand plotted calibration data was used for excess reactivity and rod worth measurements. Furthermore, from years 1973 to 1985, various fuel elements were reported to be axially bent. This was due to the high radial flux and radial power gradient caused by frequent reactor pulsing experiments. This bending might have caused an unknown effect on reactivity

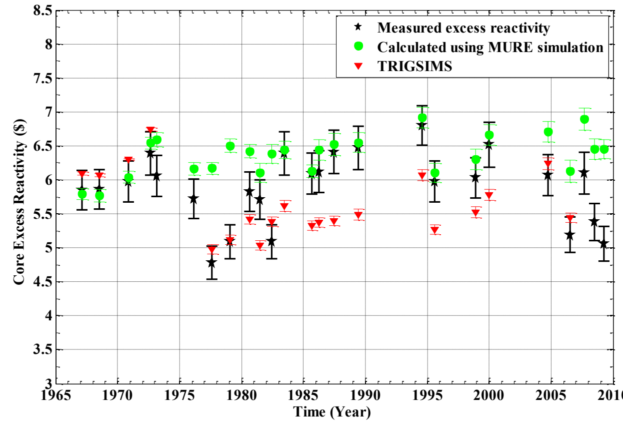

Figure 2: Measured and calculated PSBR core excess reactivity compared with TRIGSIMS results [5]

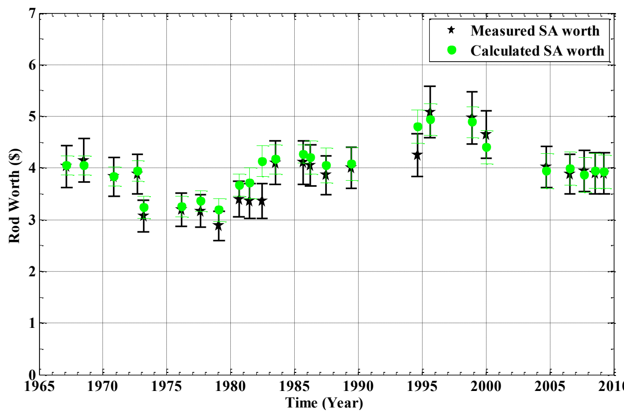

Figure 3: Measured and calculated Safety Rod (SA) worth

Table 1: Measured and calculated worth for D2O tank and fuel elements

Name	Worth (cent)	
	Measured	Calculated
Fuel Element #34	11±1	15±5
Fuel Element #203	23±2	26±5
Fuel Element #121	30±3	32±5
Fuel Element #126	45±5	42±5
D_2O Tank	66±7	69±7

worth measurements for the PSBR core. Later, fuel elements started to be rotated periodically to prevent such bending.

In 1994, measured and calculated excess reactivity values again had a large difference. After further research through the RSEC reports, it has been found that in 1994, core excess reactivity had been measured incorrectly [12]. Operators using the new automated control system did not wait long enough between reactivity insertions for the reactor to become stable. Based on the report, excess reactivity of the PSBR core in 1994 should be approximately $6.8, which compares well with the calculated value of $6.92. Total control rod worth should be $12.25 instead of $11.74. Among the control rod worths, the worth of SA had the biggest deviation. The calculated worth in 1994 was $4.8 and the reported value in 1994 was $4.25. For the same core loading, the excess reactivity and SA rod worth have been re-measured as $6.64 and $5.05, respectively, in 1995 [12].

Statistical fluctuations in calculated core excess reactivity and control rod worth occur due to several other aspects, such as:

- Simplified thermal hydraulics assumptions based on measurements
- MCNP statistical uncertainties
- Approximations in geometry and material compositions
- Discarding core loadings with short lifetimes

Burnup calculations are always performed based on reactor log data, except one extra step. Simulation has been over predicting excess reactivity in 2010 (CL-53H). An additional 42 MWD burnup has been applied to match measured excess reactivity in January 2011.

Conclusions

An accurate and computationally feasible burnup coupled neutronic simulation application for the PSBR core has been developed using MURE libraries. Burnup calculations have been performed for PSBR cores starting in 1967 and for all major core loadings. Measured and calculated core excess reactivity and control rod worth values showed agreement within experimental uncertainty.

References

D. K. Hauck, "Dendrochemistry: Seeing the forest through the trees," PhD Thesis, Pennsylvania State University, Pennsylvania State University, 2008.

D. Şahin and K. Ünlü, "Determination of self shielding factors and gamma attenuation effects for tree ring samples," *J Radioanal Nucl Chem*, vol. 291, no. 2, pp. 549–553, Jun. 2011.

O. MÉPLAN, "MURE Web Site." [Online]. Available: http://lpsc.in2p3.fr/gpr/MURE/html/MURE/MURE.html . [Accessed: 17-Jan-2012].

[O. MÉPLAN, A. Nuttin, O. Laulan, S. David, F. Michel-Sendis, and J. Wilson, "MURE: MCNP utility for Reactor Evolution - Description of the methods, first applications and results," *ENS-ANS-SFEN*, 2005.

C. Tippayakul, "DEVELOPMENT OF A PRACTICAL FUEL MANAGEMENT SYSTEM FOR PSBR BASED ON ADVANCED THREE-DIMENSIONAL MONTE CARLO COUPLED DEPLETION METHODOLOGY," Pennsylvania State University, University Park, PA, USA, 2006.

C. Tippayakul, K. Ivanov, and C. Frederick Sears, "Development of a practical Monte Carlo based fuel management system for the Penn State University Breazeale Research Reactor (PSBR)," *Annals of Nuclear Energy*, vol. 35, no. 3, pp. 539–551, Mar. 2008.

O. MÉPLAN, "UserGuide," *MURE, MCNP Utility for Reactor Evolution*. [Online]. Available: http://lpsc.in2p3.fr/gpr/MURE/html/UserGuide/UserGuide.html. [Accessed: 10-Feb-2011].

N. Capellan, J. Wilson, S. David, O. Méplan, J. Brizi, A. Bidaud, A. Nuttin, and P. Guillemin, "3D coupling of Monte Carlo neutronics and thermal-hydraulics calculations as a simulation tool for innovative reactor concepts," in *Proceedings of Global 2009*, Paris, France, 2009, pp. 1358–1367.

A. Nuttin, P. Guillemin, A. Bidaud, N. Capellan, R. Chambon, S. David, O. Méplan, and J. N. Wilson, "Comparative analysis of high conversion achievable in thorium-fueled slightly modified CANDU and PWR reactors," *Annals of Nuclear Energy*, vol. 40, no. 1, pp. 171–189, Feb. 2012.

J. Brizi, O. Méplan, S. David, A. Bidaud, N. Capellan, P. Guillemin, A. Nuttin, and J. Wilson, "Sodium-cooled fast reactors: void coefficient and waste minimization. Neutronic studies using MURE," in *Proceedings of Global 2009*, Paris, France, 2009, pp. 1957–1966.

Penn State Breazeale Reactor, *PSBR Safety Analysis Report*. RSEC Penn State University, 2005.

"RSEC Safeguards Committee Meeting," Pennsylvania State University, Radiation Science and Engineering Center, Agenda, Jul. 1995.