# Neutronic Analysis of the PSBR Using a Burnup-Coupled MCNP Simulation with MURE

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> **Abstract** — The main goal of this study is to verify the accuracy of burnup-coupled neutronic calculations when employing the Monte Carlo Utility for Reactor Evolutions (MURE) and MCNP5 codes for modeling TRIGA-type reactors, in this case the Penn State Breazeale Reactor (PSBR) core. Research and educational requirements mainly direct the PSBR operating schedule. With such operating schedules, one particular area of concern, specifically relating to nuclear analytical applications, is time-dependent changes in the neutronic characteristics of the reactor, specifically within the irradiation positions. Particular concern exists among scientists performing neutron activation analysis measurements as to whether continuous variations in reactor operations would cause significant fluctuations in the neutronic characterization parameters of the irradiation positions. A secondary objective of this study is to analyze fluctuations in the neutronic characterization parameters and their dependence on various core conditions as examined by detailed burnup-coupled neutronic simulations. In this study, a burnup-coupled neutronic simulation model of the PSBR is developed using the MURE and MCNP5 codes. The simulation results are verified by a series of experiments including measurements of the core excess reactivity starting from the first core loading in 1965 to 2012, control rod worth, fission product buildup, temperaturedependent reactivity loss, integral control rod worth curves, individual fuel element worth, and neutron flux. Local neutronic calculations of the simulation are confirmed by measuring neutronic characterization parameters for one of the irradiation positions within the PSBR core, namely, dry irradiation tube 1. Analyzing time-dependent data predicted by the simulation, the neutron temperature and the measure of the nonideal epithermal neutron flux distribution are found to be reasonably static. Conversely, the thermal-to-epithermal neutron flux ratio and spectral index are found to be relatively responsive to alterations in the core.

**Keywords** — Burnup-coupled neutronic analysis, neutron activation analysis, neutron flux characterization.

**Note** — Some figures may be in color only in the electronic version.

## I. BACKGROUND/MOTIVATION

The Penn State Breazeale Reactor (PSBR) is a 1-MW TRIGA Mark III–type reactor. The first core, which was a Material Testing Reactor, went critical in 1955 and was replaced with a TRIGA Mark II core in 1965. PSBR has had a well-documented operational history since 1955. The current maximum operating power is 1 MW(thermal). The core power can pulse up to 2000 MW(thermal). PSBR serves as a teaching and research reactor at the Pennsylvania State University Radiation Science and Engineering Center (RSEC). In 1998, two additional irradiation positions were installed in the PSBR core for routine neutron activation analysis (NAA) research. These were air filled aluminum tubes, called dry irradiation tube 1 (DT-1) and dry irradiation tube 2 (DT-2).

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The primary goal of this study is to perform a detailed burnup-coupled neutronic analysis of the PSBR core starting with the core loading in 1965 to 2012 using a burnupcoupled neutronic analysis simulation developed with the Monte Carlo Utility for Reactor Evolutions (MURE) and MCNP5 (Version 1. 51) codes.<sup>1–5</sup> By performing the calculations, the accuracy and applicability of the MURE code for a TRIGA-type reactor, where the operation schedule is quite random, were investigated. As far as we know, this is the first attempt to use the MURE code for burnup-coupled neutronic analysis of a TRIGA reactor.

Neutron transport equations describe the neutron density distribution in the vicinity of an irradiation position. However, NAA methods, such as a single comparator or absolute methods, rely on simplified representations of the neutron density distribution, such as the Høgdahl and/or Westcott depictions.<sup>6–12</sup> A set of neutron flux characterization parameters, namely, the thermal-to-epithermal neutron flux ratio *f*, the measure of the nonideal epithermal neutron flux distribution  $\alpha$ , spectral index ( $r \sqrt{T_n/T_o}$ ), and neutron temperature  $T_n$ , are defined to reflect the individual neutronic characteristics of a specific location.

The operating schedule of the PSBR causes fluctuations in temperature distribution and fission product concentrations throughout the core. These fluctuations are known to cause unknown uncertainty in NAA measurements of the PSBR irradiation positions.<sup>13,14</sup> It is essential to evaluate whether changes in the core and reactor operations would affect these parameters. Specifically, when using single comparator NAA methods, changes may lead to inaccurate quantification of elemental masses. The simulation results were used to examine the effects of operation schedule and core changes on these neutron flux characterization parameters. Therefore, a secondary objective was to quantify and analyze changes in the neutron flux characterization parameters using the burnup-coupled neutronic analysis simulation.

The study in this paper is described in three sections followed by a summary. Section II describes the new cross-section data that were generated for these simulations and modeling details of the PSBR core and external structures. Section III reviews burnup-coupled neutronic analysis results and compares them with measurements. The final section verifies the local neutronic characterization capabilities of the simulation by analyzing an irradiation tube within the PSBR core.

In all figures and tables in this paper except for Sec. IV, error bars are  $2\sigma$  (95% confidence interval, considering calculational errors only) and 10% of the measured value for the calculated values and the measurement results, respectively. In Sec. IV, the measurement results are listed with  $1\sigma$  mea-

surement uncertainty, and the calculated values are given  $2\sigma$  (95% confidence interval) error bars.

#### **II. PSBR BURNUP ANALYSIS USING MURE**

The MURE advanced code library is written in C++for nuclear reactor neutronic analysis, including an integrated burnup calculation module and coupling mechanisms for MCNP and thermal-hydraulic calculation code libraries.<sup>5</sup> MURE consists of four operational modules. The first module deals with the creation of the MCNP input files for complex geometries. The next module then creates an abstract nuclear reaction/decay tree following possible transmutation reactions and radioactive decay schemes for elements within each cell of the MCNP geometry. The third module sets up the Bateman's differential equations for the abundance of each isotope in these cells and then solves them using an adaptive Runge-Kutta numerical integration. MURE uses the cell-averaged multigroup neutron flux (F4 tally) and cross-section data directly from MCNP5 output in these equations. The fourth MURE module is for coupling with thermalhydraulic codes, which can also couple to a simplified thermal-hydraulic code or the Coolant Boiling in Rod Arrays (COBRA-EN) subchannel code.

The change in atom density  $N_i$  within fuel elements is governed by decay, burnup, and production for isotope *i* as given by Bateman's equation (adapted from Ref. 4):

$$\begin{aligned} \frac{\partial N_i}{\partial t} &= -\lambda_i N_i - N_i \sum_i \int \sigma_i^r(E) < \phi(E, t) > dE \\ &+ \sum_{j \neq i} \lambda_j^{j \to i} N_j \gamma^{j \to i} + \sum_{j \neq i} N_j \int \sigma_j^{j \to i}(E) < \phi(E, t) > dE , \end{aligned}$$
(1)

where

 $\lambda = \text{decay constant}$ 

E = energy of neutrons

- i, j = index numbers for the target and product isotopes, respectively
- $\gamma^{j \to i}$  = proportion of decay branch of isotope *j* decaying to isotope *i*
- $\langle \phi(E, t) \rangle$  = multigroup cell-averaged neutron flux calculated by MCNP
  - $\sigma_i^{j \to i}(E) =$  multigroup cross section for a capture reaction transforming isotope j into i
    - $\sigma_i^r(E)$  = removal cross section at cell temperature.

# II.A. Temperature-Dependent Cross-Section Generation Using ENDFB-VII Data Files with NJOY

The PSBR core has a negative temperature feedback coefficient of reactivity. PSBR core fuel elements are deployed in two enrichment levels of uranium (uranium mass fractions of 8.5% and 12%) in ZrH. At higher temperatures, because of the negative Doppler feedback effect, hydrogen in the fuel negatively affects the neutron economy due to upscattering. Accurate modeling of this phenomenon is necessary to have a reasonable neutronic simulation for the PSBR core. The previously available 50 K interval cross-section data were not sufficient to accurately predict the excess reactivity loss, especially at power levels >500 kW (Ref. 15). New cross-section data were generated using NJOY for an extended list of isotopes, with 10 K intervals.

ENDF/B-VII incident neutron data files were used to produce A Compact ENDF (ACE) cross-section data files for 86 isotopes (as listed in Table I) at temperatures between 293.72 and 900 K with 10 K intervals. The NJOY dataprocessing code system commands (moder, reconr, broadr, heatr, gaspr, thermr, purr, and acer) were used to manipulate the data files.<sup>16</sup> The resulting ENDF files were then loaded for each isotope and converted into energy-dependent, pointwise PENDF data files. The PENDF data were then manipulated for a given temperature and then linearly interpolated for the resonance region. The resulting data were Doppler broadened for unresolved resonances, corrected for selfshielding, and saved as ACE formatted files for use in subsequent MCNP calculations.

#### TABLE I

List of Isotopes for Which Temperature-Dependent Cross-Section Data Were Generated Using NJOY

111	540	947	133.0	1511	2300
'H	<sup>34</sup> Cr	<sup>94</sup> Zr	155Cs	<sup>131</sup> Eu	<sup>239</sup> Pu
$^{2}H$	<sup>55</sup> Mn	<sup>96</sup> Zr	<sup>134</sup> Cs	<sup>152</sup> Eu	<sup>240</sup> Pu
$^{10}B$	<sup>54</sup> Fe	<sup>95</sup> Mo	<sup>135</sup> Cs	<sup>153</sup> Eu	<sup>241</sup> Pu
<sup>11</sup> B	<sup>56</sup> Fe	<sup>97</sup> Mo	<sup>139</sup> La	<sup>154</sup> Eu	<sup>242</sup> Pu
C-natural	<sup>57</sup> Fe	<sup>99</sup> Tc	$^{141}$ Pr	<sup>155</sup> Eu	<sup>241</sup> Am
$^{14}N$	<sup>58</sup> Fe	<sup>101</sup> Ru	<sup>143</sup> Nd	<sup>154</sup> Gd	<sup>242</sup> Am
<sup>24</sup> Mg	<sup>58</sup> Ni	<sup>102</sup> Ru	<sup>144</sup> Nd	<sup>155</sup> Gd	
<sup>25</sup> Mg	<sup>60</sup> Ni	<sup>103</sup> Rh	<sup>145</sup> Nd	<sup>156</sup> Gd	
<sup>26</sup> Mg	<sup>61</sup> Ni	<sup>105</sup> Rh	<sup>148</sup> Nd	<sup>157</sup> Gd	
<sup>27</sup> Al	<sup>62</sup> Ni	<sup>105</sup> Pd	<sup>147</sup> Pm	<sup>158</sup> Gd	
<sup>28</sup> Si	<sup>64</sup> Ni	<sup>108</sup> Pd	<sup>148</sup> Pm	<sup>234</sup> U	
<sup>29</sup> Si	<sup>83</sup> Kr	<sup>109</sup> Ag	<sup>147</sup> Sm	<sup>235</sup> U	
<sup>30</sup> Si	<sup>90</sup> Zr	113Cd	<sup>149</sup> Sm	<sup>236</sup> U	
<sup>50</sup> Cr	<sup>91</sup> Zr	<sup>129</sup> I	<sup>150</sup> Sm	<sup>238</sup> U	
<sup>52</sup> Cr	<sup>92</sup> Zr	<sup>131</sup> Xe	<sup>151</sup> Sm	<sup>237</sup> Np	
<sup>53</sup> Cr	<sup>93</sup> Zr	<sup>135</sup> Xe	<sup>152</sup> Sm	<sup>238</sup> Pu	

# II.B. Implementation of the PSBR Core Simulation Model

The PSBR core consists of about 100 fuel elements, including 2 instrumented fuel elements, 4 control rods, 10 graphite elements, and a detachable  $D_2O$  tank. The elevation and plan views of the MCNP model for the PSBR core are shown in Figs. 1 and 2, respectively. Simplifying assumptions were employed to limit the computational time. The fuel elements and the control rod bottom and top latching structures were ignored, due to relatively low importance within neutronic calculations. Each fuel element was divided horizontally into five homogeneous fuel meat sections. The fuel element radius remains constant (no swelling occurs)



Fig. 1. Elevation view of the MCNP model of the PSBR core.



Fig. 2. Plan view of the MCNP model of the PSBR core coupled to the  $D_2O$  tank.

even at higher temperatures. Burnup and decay calculations were also simplified. Only isotopes with a halflife >3 h and concentration at >10<sup>-10</sup> mol/m<sup>3</sup> were recorded for evolutions to the following step. Only significant neutron interaction probabilities were considered; i.e., the natural log energy-averaged reaction cross section  $\varepsilon$  for any neutron-induced reaction is >10<sup>-2</sup> b (1 b = 10<sup>-24</sup> cm<sup>2</sup>) (Ref. 4), where the natural log energy-averaged cross section is given as

$$\varepsilon = \frac{\int \sigma(E) \log(E) dE}{\int \log(E) dE} .$$
 (2)

A total of 17900 energy groups were logarithmically distributed within the thermal reactor neutron energy range (0 to 20 MeV) and used for the MCNP neutron flux tally (F-4) as listed in Table II.

## II.B.1. External Components

The PSBR core resides in a high-purity water pool kept at  $\sim 300$  K. A D<sub>2</sub>O tank is also coupled to the PSBR core for beam port irradiation experiments as shown in Fig. 2. The D<sub>2</sub>O tank model was approximated as an aluminum cylinder container. There is a penetration hole in the D<sub>2</sub>O tank, for the beam port neutron guides to conjugate.

## II.B.2. Fuel Elements

The fuel meat of each fuel element is divided into five sections. Material information and burnup calculations are performed separately for each of these sections. A schematic drawing of the fuel element elevation view, as implemented in the simulation, is shown in Fig. 3.

The temperature of each section of the fuel element model was set independently by a linear interpolation using temperature measurements from the instrumented fuel elements. Let us define the power generated in a fuel

TABLE	Π

Multigroup Energy Intervals Used in Neutronic Simulation

Energy Range	Energy Interval (eV)	Number of Groups
1	0 to 1	400
2	1 to 10	500
3	10 to $10^4$	15000
4	$10^4$ to $10^5$	1000
5	$10^5$ to $2 \times 10^7$	1000



Fig. 3. Fuel-follower control rod (left) and fuel element (right) models for the PSBR core, elevation view (dimensions in units of centimeters).

element section as  $P_{cell}$  and the local power of an instrumented fuel element center section as  $P_{inst}$ . An average temperature in any fuel element section  $T_{cell}$  can be calculated by linear interpolation using measured temperature values  $T_{inst}$  and  $P_{inst}$  using

$$T_{cell} = \frac{P_{inst,j+1} - P_{cell}}{P_{inst,j+1} - P_{inst,j}} (T_{inst,j+1} - T_{inst,j}) + T_{inst,j}, (3)$$

where index j and j + 1 mark the sections of instrumented fuel element below and above the cell temperature, respectively.

## II.B.3. Control Rod

There are four control rods used to monitor and drive the PSBR nuclear reactor. Three of them are fuel-follower control rods, where a neutron absorber  $(B_4C)$  was followed by a fuel element. Fuel-follower control rods are called the safety rod (SA), shim rod (SH), and regulating rod (RR). The last control rod is an air-follower control rod, called the transient rod (TR). The fuel meat of the first three fuel-follower control rods was divided into five sections as shown in Fig. 3. The TR is a B<sub>4</sub>C rod in an air-filled aluminum rod. All fuel-follower control rods were replaced once with new rods, in 1994 (Ref. 17).

## II.B.4. Graphite Element

Graphite elements with aluminum cladding were used at the perimeter of the core starting in 2009 as a reflector for the PSBR core. The graphite elements were later removed in the recent core loading in 2012.

# II.B.5. Dry Irradiation Tubes and Pneumatic Transmission System

Dry irradiation tubes and pneumatic transmission system (PTS) tubes are modeled as air-filled, infinite aluminum tubes in the simulation as displayed in Fig. 4. The PTS tube is not stationary; therefore, it was replaced with water in the model during burnup calculations.



Fig. 4. Dry irradiation tube and fuel elements, elevation view (dimensions in units of centimeters).

# III. MEASUREMENTS AND VERIFICATION OF THE NEUTRONIC MODEL

The accuracy of the neutronic simulation using MCNP depends on careful geometrical and compositional modeling, as well as the number of particle histories simulated, and the effective and ineffective numbers of cycles used in the computation. One measure of accurate MCNP simulation is to perform a convergence analysis.

# III.A. Convergence Analysis for MCNP Simulation of the PSBR Core

Convergence analysis of an MCNP simulation is as essential as complete and accurate modeling of the system. Figure 5 shows the relative entropy for the effective multiplication factor  $k_{eff}$  calculated by MCNP within a burnup step, at 900 kW for the first 150 inactive simulation cycles. The numbers of particles per cycle (histories), inactive cycles, and active cycles was  $5 \times 10^4$ , 300, and 1000, respectively. Relative entropy was calculated (assuming each cycle has a discrete distribution) using the Kullback-Leibler formulation, given as<sup>18</sup>

$$R_{e,i} = k_{eff,i+1} \times \log_2\left(\frac{k_{eff,i+1}}{k_{eff,i}}\right), \qquad (4)$$

where  $R_{e,i}$  is the relative entropy for the *i*'th cycle and  $k_{eff}$  is the calculated effective criticality. Although  $k_{eff}$  values converge quickly as seen in Fig. 6, this does not necessarily mean that the fission source distribution has converged. The Shannon entropy source distribution is a critical parameter that is available in the MCNP output, and it is used to validate MCNP simulation source convergence.<sup>19</sup> It is possible to have a converged  $k_{eff}$  without having a converged fission source distribution for the same MCNP simulation is shown in Fig. 7. In Fig. 7, the Shannon source entropy value for each cycle is plotted against the subsequent cycle value to show the convergence of source entropy.

As shown in Figs. 6 and 7, it is clear that the source convergence occurs after the first five MCNP cycles.

## III.B. Burnup Calculations for the PSBR Spanning Years 1965–2012

To perform burnup calculations, core loadings since 1965 (after the core conversion to TRIGA Mark II) were identified by investigating the PSBR operation logbooks. Burnup and decay times, excess reactivity, mean operating power, average critical control rod positions (given in



Fig. 5. Relative entropy in effective multiplication factor calculated by MCNP.



Fig. 6. Effective criticality constant for  $5 \times 10^4$  histories/cycle.



Fig. 7. Shannon fission source iterative convergence at each cycle in sequence.

percent of inches extracted from the core), and control rod worth calibration values were tabulated from 1965 to 2012. Ten days was the maximum time step used in the burnup calculations. Ten-day continuous operation at 0.7 MW results in a core excess reactivity loss of  $\sim 25 \text{ }\text{e}$ .

The measured excess reactivity values were obtained by extrapolating the measured control rod worths at each reactor cycle. This process is performed at the beginning of each core loading, and results were recorded in PSBR logs. Since 1965, core excess reactivity at the beginning of each core loading is calculated and compared with measured values as presented in Fig. 8. Core excess reactivity  $\rho_{ex}$  is calculated using the MCNP-calculated  $k_{eff}$  at each core loading, by having all control rods fully taken out of the core. The excess reactivity is given by

$$\rho_{ex} = \frac{k_{eff} - 1}{k_{eff} \times \beta} , \qquad (5)$$

where  $\beta$  is the total delayed neutron fraction taken as 0.007 for the PSBR (Ref. 17).

Significant inconsistencies between excess reactivity measurements and calculated values were noted between the years 1973 and 1985, as shown in Fig. 8. For instance, the absolute error in the calculated excess reactivity for the year 1977 was 29.1%. The PSBR core was controlled manually until 1991 when the first automatic control system was installed. Before 1991, hand-plotted calibration data were used for excess reactivity and rod worth measurements. Furthermore, from the years 1973 to 1985, various fuel elements were reported to be axially bent.<sup>20</sup> The bending was due to the high radial flux and radial power gradient caused by repeated reactor pulsing experiments. This bending might be the cause of discrepancies.

Later, fuel elements started to be rotated periodically to prevent such bending.

In 1994, the measured and calculated excess reactivity values also had considerable discrepancies, as much as 18.5%. After further research through the RSEC documentation for the PSBR, it has been found that in 1994, core excess reactivity had been measured inaccurately.<sup>20</sup> Operators using the newly automated control system did not wait long enough between reactivity insertions for the reactor to become stable. Based on the PSBR Safeguards Committee meeting report, excess reactivity of the PSBR core in 1994 should have been  $\sim 6.8$  \$, which compared well with the calculated value of 6.92 \$ (Ref. 20). Total control rod worth should have been 12.25 \$ instead of 11.74 \$. Among the control rods, 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 were remeasured as 6.64 \$ and 5.05 \$, respectively, in 1995 (Ref. 20).

## III.B.1. Control Rod Worth Analysis

The rod worths for the four control rods (SA, SH, RR, and TR) at the beginning of each core loading were calculated and compared with the measurements. As an example, the calculated TR rod worth values are compared to the measured values as shown in Fig. 9. It is clear that the simulated values are in agreement with the measurements. Each control rod worth is calculated by having all the other rods at a critical position. While the other control rods were held at the critical position, the  $k_{eff}$  values were calculated by MCNP when the control rod is fully inserted and fully taken out. The difference is the



Fig. 8. Measured and calculated PSBR core excess reactivity values from 1965 to 2012.



Fig. 9. Calculated and measured TR worth from 1965 to 2012.

calculated control rod worth. Measured control rod worths are taken from PSBR logbooks.

## III.B.1.a. Control Rod Model Adjustment

After several attempts analyzing control rod worth curves and comparing with measurements, it was found that the density of the B<sub>4</sub>C absorber material in the control rods had to be adjusted. The absolute errors between the measured rod worth values and the calculation results were as much as 21%, specifically toward the upper parts of the rods (after 25 cm). A possible effect is the fact that the impact of the fuel composition and neutron spectra was neglected in the analysis. The control rods were kept at the critical positions of each fuel cycle during long-term burnup calculations, limiting accurate capture in a compositional change in the fuel portion as well as absorber portions. Furthermore, the control rods were kept in operation from 1965 to 1994 without replacement. The lack in accurate composition due to burnup could be easily simulated by iteratively adjusting the absorber density until the calculations are in agreement with the measurements.<sup>17,21</sup> Elemental compositions 1 and 2 as listed in Table III employ the manufacturer-specified elemental compositions and adjusted density values for the fuel-follower and air-follower control rod absorber materials, respectively. The final densities were found by iterative calculations and comparisons to measurements.

A control rod calibration was performed as a periodic measurement on August 29, 2011. The corresponding

**⊗ANS** 

TABLE III

Elemental Compositions and Densities of Control Rod Absorber Material Used in This Work

Number	<sup>10</sup> B Mass Fraction (%)	<sup>11</sup> B Mass Fraction (%)	Carbon Mass Fraction (%)	Density (g/cm <sup>3</sup> )
1	3.18	12.82	84	2.5
2	3.18	12.82	84	1.13

measurement results and calculated values before and after the density adjustment were compared as shown in Table IV. After the adjustment, the maximum absolute error between the measured rod worth and calculated rod worth values was 4%. In Table IV, the agreement between the calculated values with the measured values was quantified based on the *z*-factor. The *z*-factor is given as<sup>22</sup>

$$z-\text{factor} = \frac{|V_{measured} - V_{calculated}|}{\sqrt{\sigma_{measured}^2 + \sigma_{calculated}^2}}, \quad (6)$$

where

$$V_{measured}$$
,  $V_{calculated}$  = measured and calculated values,  
respectively

 $\sigma_{measured}, \sigma_{calculated} =$  uncertainties of the measured and calculated values, respectively.

TABLE IV
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Measured and Calculated Control Rod Worth Values Using Before and After Adjustment of Densities

Name	Measured	Calculated Worth Using Original	Calculated Worth Using Adopted
	Worth (\$)	Densities (z-Factor) (\$) <sup>a</sup>	Densities (z-Factor) (\$) <sup>a</sup>
Transient rod Safety rod Shim rod Regulating rod	$\begin{array}{c} 2.78 \pm 0.28 \\ 3.64 \pm 0.36 \\ 2.73 \pm 0.27 \\ 2.81 \pm 0.28 \end{array}$	$\begin{array}{c} 3.15 \pm 0.06 \ (1.3) \\ 4.42 \pm 0.06 \ (2.2) \\ 3.20 \pm 0.06 \ (1.7) \\ 3.27 \pm 0.06 \ (1.6) \end{array}$	$\begin{array}{l} 2.79 \pm 0.06 \; (0.04) \\ 3.68 \pm 0.06 \; (0.1) \\ 2.83 \pm 0.06 \; (0.4) \\ 2.84 \pm 0.06 \; (0.1) \end{array}$

<sup>a</sup>While keeping other rods at the critical position, each control rod is fully inserted and removed to calculate total rod worth. The z-factor values are given in parentheses.

The closer the z-factor is to zero, the better is the agreement between the values. If the z-factor is >3, the result is not acceptable.

For further verification of the adopted densities, the adopted control rod elemental compositions were verified against an integral control rod worth measurement on July 11, 1999. Integral control rod worth curves were calculated and compared with the measurements. As an example, calculated and measured integral rod worth curves of the SA rod were in agreement as shown in Fig. 10.

## III.B.1.b. Measured and Calculated Temperature-Dependent Reactivity Loss

A temperature-dependent excess reactivity loss measurement was completed for core loading 52 on November 15, 2004. Calculated reactivity loss values were compared with the measured values as presented in Fig. 11. Figure 11 clearly demonstrates the ability to model the temperature reactivity feedback for the PSBR core.

## III.B.1.c. Fission Product Buildup

For power reactor applications, changes in neutron characteristics related to startup and shutdown are minor compared to long-term operation. However, for a TRIGA-type, low-power research reactor, such as the PSBR, these changes are significant. An experiment was performed to evaluate the impact of fission product buildup/decay in the PSBR core. During the experiment, the PSBR (after a long cooldown, when most of the effective fission products such as <sup>135</sup>Xe have decayed sufficiently) was operated at a constant power of 700 kW for 45 h. Control rod positions were used to record core excess reactivity, every 5 h. The measured and calculated values were in agreement as shown in Fig. 12. The measured excess reactivity values are based on extrapolation of measured control rod



Fig. 10. Measured and calculated integral control rod worth for SA.



Fig. 11. Measured and calculated core excess reactivity loss at operating powers 0.05 to 1000 kW.



Fig. 12. Measured and calculated excess reactivity loss due to fission product buildup in 45 h at 700 kW.

worth values. The calculated excess reactivity values were obtained by taking all control rods out at each step to calculate  $k_{eff}$ . The difference between the  $k_{eff}$  values at each step with the control rods at the critical position and control rods fully out is converted to equivalent reactivity using Eq. (5).

#### III.B.1.d. Fuel Element and D<sub>2</sub>O Tank Worth

Further validation of burnup calculations was completed for individual fuel elements. Four fuel elements were removed one at a time out of the PSBR core. The reactivity worths for these fuel elements were measured on April 26, 2004. Additionally, the  $D_2O$  tank worth was measured on January 9, 2011. The measured and calculated fuel element and  $D_2O$  tank worth values are given in Table V.

# III.B.1.e. Neutron Flux Measurements at the PSBR Irradiation Positions

Local neutronic calculations were verified against neutron activation measurements using gold-aluminum wires (AuAl, 0.112% Au). The positioning and dimensions of

#### TABLE V

Neutronic Simulation Calculated and Measured Worth for D<sub>2</sub>O Tank and Fuel Elements

Name			Worth (¢)	
	Ivanic	Measured	Calculated	z-Factor
	Fuel element 34 Fuel element 203 Fuel element 121 Fuel element 126 D <sub>2</sub> O tank	$ \begin{array}{r} 11 \pm 1 \\ 23 \pm 2 \\ 30 \pm 3 \\ 45 \pm 5 \\ 66 \pm 7 \end{array} $	$     \begin{array}{r}       15 \pm 5 \\       26 \pm 5 \\       32 \pm 5 \\       42 \pm 5 \\       69 \pm 7     \end{array} $	0.8 0.6 0.3 0.4 0.3

the gold aluminum wires in the DT-1 and DT-2 irradiation positions were as shown in Fig. 4. After irradiation, the wires were cut into 2.54-cm pieces, and gamma spectroscopy was performed using a Canberra<sup>TM</sup> Model GC1518 high-purity germanium detector. Self-shielding, decay, interference, background, and efficiency corrected saturation activities were analyzed and compared with simulation results. Calculations of saturation activities were performed using the burnup-coupled MCNP simulation with the calculated compositions at the measurement time for the gold-aluminum wires.

The AuAl wires were irradiated for 2 min at 500 kW for the first measurement performed on September 16, 2011. The measured and calculated saturation activities are shown in Fig. 13. Reproducibility of these results was verified with subsequent measurements in March 2012 and on June 15, 2012, at 800 kW.

The final saturation activity measurement was performed after the installation of a commercial sample. The commercial sample was aligned to the fuel element center at a position close to the DT-1 position, one fuel element apart. The sample was 15.2 cm in length and contained an approximate mass fraction of 22% natural boron. The neutron flux depression observed in DT-1 due to this commercial sample in measured and calculated saturation activities is presented in Fig. 14.

# IV. ANALYSIS OF THE NEUTRONIC CHARACTERIZATION PARAMETERS FOR THE DRY IRRADIATION TUBES

Once the neutronic simulation was verified by measurements, time-dependent analysis of the neutron flux characterization parameters was performed for the PSBR dry irradiation tubes. The neutronic model was first used to calculate the reaction rate probabilities in the PSBR irradiation position, DT-1, for a set of isotopes. The resulting reaction rate probabilities were iteratively fitted to modified Westcott functions to calculate the neutron flux characterization parameters.<sup>6-12</sup> The calculated Westcott thermal, resonance, and fission spectrum averaged (fast) neutron flux in DT-1 are given in Table VI. Calculated thermal-to-epithermal neutron flux ratio f, measure of the nonideal epithermal neutron flux distribution  $\alpha$ , spectral index  $(r\sqrt{T_n/T_o})$ , and neutron temperature  $T_n$  are given in Table VII. Normalized values of the measure of nonideal epithermal neutron flux distribution  $\alpha$  and thermal-toepithermal neutron flux ratio f are plotted together in Fig. 15 for a 7-month period to visualize the changes in these



Fig. 13. Measured and calculated saturation activity of gold wire segments along DT-1 (irradiated for 2 min at 500 kW on September 16, 2011).



Fig. 14. Measured and calculated saturation activity of gold wire segments along DT-1 (irradiated for 2 min at 800 kW on July 5, 2012).

#### TABLE VI

Westcott Thermal, Resonance, and Fast Neutron Flux in DT-1 During Irradiations Performed at 800 kW (PSBR Core Coupled to the D<sub>2</sub>O Tank)

Date		Westcott Neutron Flux $\left(\frac{n}{cm^2 \cdot s}\right)$	
	Thermal (10 <sup>12</sup> )	Resonance (10 <sup>11</sup> )	Fast (10 <sup>12</sup> )
November 16, 2011 November 21, 2011 November 30, 2011 January 31, 2012 February 6, 2012 February 9, 2012 February 15, 2012 February 23, 2012 February 28, 2012 March 22, 2012 June 12, 2012 June 20, 2012	$7.99 \pm 0.42$ $7.84 \pm 0.41$ $7.87 \pm 0.39$ $7.94 \pm 0.43$ $8.04 \pm 0.42$ $7.82 \pm 0.35$ $7.81 \pm 0.46$ $7.94 \pm 0.47$ $7.92 \pm 0.47$ $7.93 \pm 0.46$ $7.93 \pm 0.45$ $7.47 \pm 0.41$ $7.40 \pm 0.43$	$\begin{array}{c} 3.11 \pm 0.18 \\ 3.41 \pm 0.19 \\ 3.39 \pm 0.18 \\ 3.30 \pm 0.2 \\ 3.25 \pm 0.19 \\ 3.50 \pm 0.16 \\ 3.31 \pm 0.2 \\ 3.28 \pm 0.21 \\ 3.16 \pm 0.21 \\ 3.24 \pm 0.21 \\ 3.20 \pm 0.2 \\ 3.70 \pm 0.18 \\ 3.62 \pm 0.19 \end{array}$	$\begin{array}{c} 3.72 \pm 0.87 \\ 3.81 \pm 0.85 \\ 3.61 \pm 0.75 \\ 3.20 \pm 0.92 \\ 3.30 \pm 0.78 \\ 3.76 \pm 0.82 \\ 3.72 \pm 0.15 \\ 3.33 \pm 0.91 \\ 3.53 \pm 0.76 \\ 3.23 \pm 0.85 \\ 3.24 \pm 0.87 \\ 4.47 \pm 0.97 \\ 4.26 \pm 0.9 \end{array}$
July 12, 2012	$6.70 \pm 0.36$	$3.39 \pm 0.16$	$3.85 \pm 0.8$

parameters. Even the changes for  $\alpha$  mostly remain within the uncertainty of the calculations; significant changes in *f* values (specifically after the core loading change in May 2012) are visible as shown in Fig. 15.

The computed neutron flux characterization parameters were verified against two subsequent measurements. Thermal-to-epithermal neutron flux ratio f was measured using the bi-isotropic method.<sup>23</sup> The measure of nonideal epithermal neutron flux distribution  $\alpha$  was measured by using the bare triple method.<sup>23</sup> Spectral index  $r \sqrt{T_n/T_o}$  was measured using the bare dual monitor method.<sup>24</sup> Neutron temperature  $T_n$  was measured using lutetium wires.<sup>24</sup> Uncertainties in the measured neutron flux characterization parameters f,  $\alpha$ ,  $r \sqrt{T_n/T_o}$ , and  $T_n$  were calculated as the sum of the fixed uncertainty from the nuclear data and the experimental uncertainty due to the measured activities, as explained in Ref. 23.

#### TABLE VII

		_		
Date	f	α	$r\sqrt{T_n/T_o}$	$T_n$ (K)
November 16, 2011	37.18 ± 5.02	$-0.080 \pm 0.023$	$0.039 \pm 0.003$	324
November 21, 2011	$33.24 \pm 4.39$	$-0.068 \pm 0.029$	$0.044 \pm 0.003$	327
November 30, 2011	$33.54 \pm 4.37$	$-0.068 \pm 0.018$	$0.043 \pm 0.003$	328
January 31, 2012	$34.70 \pm 4.67$	$-0.071 \pm 0.025$	$0.042 \pm 0.003$	326
February 6, 2012	$35.76 \pm 4.83$	$-0.067 \pm 0.022$	$0.040 \pm 0.003$	324
February 9, 2012	$32.30 \pm 4.10$	$-0.056 \pm 0.024$	$0.045 \pm 0.003$	326
February 15, 2012	$34.12 \pm 4.74$	$-0.074 \pm 0.021$	$0.043 \pm 0.004$	329
February 23, 2012	$35.02 \pm 4.90$	$-0.075 \pm 0.021$	$0.041 \pm 0.004$	327
February 28, 2012	$36.22 \pm 5.05$	$-0.076 \pm 0.031$	$0.040 \pm 0.003$	324
March 22, 2012	$35.46 \pm 4.93$	$-0.072 \pm 0.034$	$0.041 \pm 0.004$	324
April 23, 2012	$35.72 \pm 4.91$	$-0.076 \pm 0.027$	$0.040 \pm 0.003$	326
June 12, 2012	$23.84 \pm 3.19$	$-0.071 \pm 0.023$	$0.050 \pm 0.004$	330
June 20, 2012	$24.18 \pm 3.28$	$-0.075 \pm 0.023$	$0.049 \pm 0.004$	331
July 12, 2012	$23.32 \pm 3.09$	$-0.064 \pm 0.033$	$0.051 \pm 0.004$	328
	1			1

Thermal-to-Epithermal Neutron Flux Ratio, the Measure of the Nonideal Epithermal Neutron Flux Distribution, Spectral Index, and Neutron Temperature in DT-1



Fig. 15. Average values of the measure of the nonideal epithermal neutron flux distribution and spectral index in DT-1. [The core loading was changed on May 22, 2012, and a strong absorber sample was installed in a nearby (R1) position on July 5, 2012.]

Nuclear data and self-shielding factors for the foils and wires used in the measurements are given in Table VIII. The temperature-dependent Westcott *g*-factors for <sup>176</sup>Lu were taken from Ref. 14. The gold and lutetium wires were diluted (0.112% gold in aluminum and 0.1% lutetium in aluminum, respectively), so neutron selfshielding was negligible.<sup>25</sup> The zirconium foils were pure and thick enough to cause a significant resonance neutron shielding. Resonance self-shielding factors for <sup>94</sup>Zr and <sup>96</sup>Zr were calculated using the following empirical formulations<sup>26</sup>:

$$G_{res}({}^{94}\text{Zr}) = 1 - 1.543 \times 10^{-4} \times t_{\text{Zr}} + 1.143 \times 10^{-7} \times t_{\text{Zr}}^2$$
(7)

and

$$G_{res}({}^{96}\text{Zr}) = 1 - 2.327 \times 10^{-4} \times t_{\text{Zr}} + 1.487 \times 10^{-7} \times t_{\text{Zr}}^2, \qquad (8)$$

where  $t_{Zr}$  is the thickness of zirconium foil in micrometers.

Measured neutron flux characterization parameters of the DT-1 position were compared with the calculated values

Nuclear Data for Reactions Used for Neutron Flux Characterization Measurements					
Reaction	k <sub>0,Au</sub> (Ref. 27)	<i>Q</i> <sup>0</sup> (Ref. 27)	<i>s</i> <sub>0</sub> (Refs. 24 and 27)	$\frac{\overline{E_r}}{(\text{Ref. 27})} (eV)$	Westcott g-Factor at 293 K (Ref. 28 and Ref. 29, p. 10)
$^{94}$ Zr $(n,\gamma)^{95}$ Zr $^{96}$ Zr $(n,\gamma)^{97}$ Zr $^{197}$ Au $(n,\gamma)^{198}$ Au	$2.0 \times 10^{-4} (1.3)^{a}$ 1.24 × 10 <sup>-5</sup> (0.3) <sup>b</sup> 1 1.25 × 10 <sup>-2</sup> 0	5.31 (3.3) 251.6 (1.) 15.7 (1.8)	5.21 279 17.2	6260 (4) 338 (2.1) 5.65 (7.1)	1.0007 1.0004 1.005

TABLE VIII

<sup>a</sup>Gamma rays of 724 and 756 keV were combined.

<sup>b</sup>Gamma ray of 743 keV.

<sup>c</sup>Gamma ray of 112.9 keV.

in May 2012 (during PSBR core loading 53) as shown in Table IX. A subsequent measurement was performed for verification. The measured and calculated neutron flux characterization parameters in August 2012 (during core loading 54) were as given in Table X. There are significant differences in the measured and calculated values of thermal-toepithermal neutron flux ratio f and nonideal epithermal neutron flux distribution  $\alpha$  following the new core loading in May 2012.

## V. DISCUSSION OF RESULTS AND CONCLUSIONS

The MURE and MCNP5 codes were used to develop a burnup-coupled neutronic simulation of the PSBR. The reactor fuel elements, control rods, irradiation positions,

#### TABLE IX

Measured and Calculated Neutron Flux Characterization Parameters in DT-1

Parameter	Measured Value	Calculated Value	z-Factor
$ \begin{cases} f \\ \alpha \\ r \sqrt{T_n/T_o} \\ T_n (\mathbf{K}) \end{cases} $	$31.4 \pm 2.4$ -0.12 ± 0.03 0.046 ± 0.004 Not measured	$\begin{array}{c} 35.5 \pm 5 \\ -0.081 \pm 0.03 \\ 0.041 \pm 0.004 \\ 326 \pm 20 \end{array}$	0.7 0.9 0.9 N/A

TABLE	x
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Measured and Calculated Neutron Flux Characterization Parameters in DT-1 (Repeated Measurement)

Parameter	Measured Value	Calculated Value	z-Factor
$ \begin{array}{c} f \\ \alpha \\ r \sqrt{T_n/T_o} \\ T_n (\mathbf{K}) \end{array} $	$\begin{array}{c} 23.1 \pm 1.8 \\ -0.067 \pm 0.02 \\ 0.051 \pm 0.004 \\ 308 \pm 19 \end{array}$	$\begin{array}{c} 23.3 \pm 3.1 \\ -0.064 \pm 0.03 \\ 0.051 \pm 0.004 \\ 328 \pm 20 \end{array}$	0.1 0.1 0 0.7

⊗ANS

and external structures were carefully modeled. Using the MURE libraries, burnup-coupled MCNP calculations were executed for the PSBR starting from the initial TRIGA reactor core loading in 1965 until 2012.

The new simulation of the PSBR core has been validated by core excess reactivity, fuel element worth, integral control rod worth, fission product buildup, and temperature-dependent excess reactivity loss measurements. It was revealed that the MURE libraries and MCNP5 can be successfully applied to predict the neutronic behavior of the PSBR core following a daily operational schedule.

Optimum control rod absorber elemental compositions were found and validated by integral rod worth measurements. The material density for the SA rod was found to be higher than the other fuel-follower control rods. Nevertheless, such variation is extremely unlikely because all fuel-follower control rods are set in symmetrical positions in the PSBR core and manufactured by the same company. A lower absorber density for the TR rod is expected, due to its extensive service since 1965.

Local neutron flux calculations in the PSBR neutron irradiation positions were compared with gold aluminum wire activation experiments. Specifically, in the region of DT-1, measured saturation activity values matched with calculations within the experimental and computational uncertainties. The measured neutron flux characterization parameters of the DT-1 position were found to be in agreement with the calculations.

It has been found that the neutron flux characterization parameters for the PSBR irradiation positions do not vary significantly due to daily operation of the reactor. The neutron temperature and the measure of the nonideal epithermal neutron flux distribution were found to be moderately static. As expected, changes in some neutronic characterization parameters were observed only following noteworthy modifications in the core, such as the deployment of a new core loading or the insertion of a strong absorber within close proximity of an irradiation position.

1972.

The thermal-to-epithermal neutron flux ratio and spectral index were found to be relatively more responsive to such changes in the core. The major changes in the thermalto-epithermal neutron flux ratio imply a requirement for repeated measurements of the neutronic characterization parameters after core modifications.

## Disclaimer

Certain commercial equipment, instruments, codes, or materials are identified in this study in order to specify the experimental procedure adequately. Such identification is not intended to imply recommendation or endorsement by the National Institute of Standards and Technology, nor is it intended to imply that the materials, codes, or equipment identified are necessarily the best available for the purpose.

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