Annals of Nuclear Energy 96 (2016) 441–445

Contents lists available at ScienceDirect

Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

Advanced CANDU reactors axial xenon oscillation controllability validation

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ARTICLE INFO

Article history:
Received 15 March 2016
Received in revised form 25 June 2016
Accepted 29 June 2016
Available online 6 July 2016

Keywords:
MCNP
ORIGEN2
Fuel burnup
Controllability
Stability
Xenon oscillation

ABSTRACT

Advanced CANDU reactor fuel channel assembly MCNP model with top, bottom, and four lattice sides reflecting boundaries is used to generate a conservative flat axial neutron flux distribution for the ^{135}Xe oscillation controllability analyses. Burnup-dependent neutron Fission Tally and ^{135}Xe axial profiles are calculated using the Monte Carlo burnup script MCOS. The controllability of the ^{135}Xe oscillations is validated by the MCOS-calculated xenon reactivity worth swings with respect to the mean. The validated upper bound of the xenon reactivity swing band of ±3 mk is well within the zone controller reactivity bank of ±7 mk.

1. Introduction

The advanced CANDU reactor (ACR-1000) ACR-1000 Technical Summary has an effective fuel length of 495 cm. This is much longer than the diffusion length of the thermalized neutrons. In keeping the constant channel power, a ^{135}Xe disturbance started in top region can affect the loosely coupled bottom region. This can cause an out of phase xenon oscillation and neutron flux tally between top and bottom the fuel channel. A large oscillation can lead to a reactor trip or to dangerously high local fuel temperatures causing significant fuel meltdown damage. Furthermore, prolong xenon oscillations can burden the core materials with temperature cycling. This will lead to a premature materials failure. A regulating system, such as zone controller, can adequately control the bulk power and prevent large xenon oscillations in ACR-1000 reactor.

Monte-Carlo neutronics calculations are sufficiently fast using modern, massively parallel computers. One of the major sources of errors in power reactor fuel burnup calculations is the burnup-dependent resonance treatment of the nuclide cross-sections (XS). The Monte Carlo neutron and photon transport code MCNP (X-5 Monte Carlo Team, 2003) using the continuous-energy nuclear XS can be applied to complex reactor fuel assemblies. It also can handle the neutron spectrum transition between the actinides and the strong absorber fission products (SAFP) such as ^{135}Xe directly without the group XS preparation steps.

In this work, a typical ACR-1000 MCNP detailed axial fuel channels assembly (FCA) model is setup and discussed in Section 2. A brief description of the written Monte Caro fuel burnup BASH script MCOS used to link MCNP to ORIGEN2 for a commercial power reactor fuel cycle analysis is given in Section 3. Section 4 presents the results and discussion, beginning with the MCNP neutron fission flux tally (Fission Tally) and ^{135}Xe top and bottom layer out-phase oscillations versus depletion. This is followed by an evaluation of the burnup-dependent Fission Tally and the stability of the nuclides relative profiles. Next the space independent FCA model total averaged nuclides, such as, ^{235}U, ^{135}Xe, ^{239}Pu, and ^{149}Sm, atomic densities depletion and built-up versus depletion time are discussed. Finally, the ^{135}Xe reactivity worth oscillations are addressed and their controllability validated. Conclusions are presented in Section 5.

2. ACR-1000 fuel channel assembly axial model description

The advanced CANDU reactor (ACR-1000) fuel was investigated for xenon oscillation controllability using an unit lattice cell model

Abbreviations: ACR-1000, advanced CANDU reactors; AD, atomic density (atoms/b-cm); L/A, axial tally local to average ratio; FCA, fuel channels assembly; MCOS, MCNP Coupling with ORIGEN2-Script; Fission Tally, MCNP neutron fission flux tally; XS, nuclide cross-sections; 1σ, one standard deviation; mk, reactivity worth milli-K (10^{-3}); SAFP, strong absorber fission products; Xe-Δρ, xenon reactivity.

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http://dx.doi.org/10.1016/j.anucene.2016.06.039
0306-4549/

with 43 UO_2 fuel rods (ACR-1000 Technical Summary). ACR-1000 is the next-generation (Gen-III Plus) CANDU technology from Atomic Energy of Canada Ltd. which maintains proven elements of the existing CANDU design. The ACR-1000 fuel uses slightly enriched uranium (about 2.3%) to extend fuel life to three times so that the spent fuel waste volume is reduced by two-thirds (ACR-1000 Technical Summary). The chosen ACR-1000 FCA consists of a zirconium–niobium (Zr–2.5%Nb) pressure tube, centered in a Zr calandria tube. Fuel elements have an effective fuel length of 495 cm, a cladding thickness of 0.14 cm, and a lattice pitch of 24 cm. Other fuel element dimensions are presented in Table 1. The 43 fuel rods are arranged around a center rod (R1) and 3 Rings (R2 to R4) as shown in Fig. 1. The FCA fuel contains UO_2 with a 95% density and an average of 2.3 wt% of ^{235}U. All the thermalized neutrons were diffused from the heavy water moderator region into the fuel channels. We have higher linear heat generation rates in the outer ring R4 with an outer radius (OR) of 0.576 cm. To have a better flatten heat generation rates, the outer radius in fuel pins R1, R2 was increased from 0.576 to 0.675 cm.

The flatness of the neutron flux profile shape and the higher power density can enhance the neutron flux tilts through ^{135}Xe oscillations (Randal and St. John, 1958). In this work, the FCA model has 48 axial uniformly divided fuel cells for each fuel rod. It has a total 43 × 48 = 2064 divided fuel cells. The FCA model also has six reflecting surface boundaries (top, bottom, and four lattice sides), which prevent neutron leakage from the lattice model. This model with its reflecting boundaries should have a well behaved flatted fission flux profile over the axial 48 divided cells. For the perfect calculated burnup parameters, such as Fission Tally, and depletion and build-up nuclides ^{235}U, ^{239}Pu, ^{135}Xe, and ^{149}Sm, their distribution profiles should be flat. So, all the axial local to average ratio (L/A) relative profiles should have a mean value of one. For each fuel rod channel, any deviations of burnup parameters L/A relative profiles from the mean value of one have all been calculated to one standard deviation (1σ) of accuracy. The perfect MCNP-calculated 43 fuel rod's axial 48 divided cells tally relative flat profiles should have L/A (1σ) equal to zero.

Table 1
ACR-1000 unit lattice cell parameters.

Lattice cell parameters	Radius (cm)	Inner radius (cm)	Outer radius (cm)
Fuel pin			
R1 radius (center rod)			0.675
R2 radius (7 rods)	1.75		0.675
R3 radius (14 rods)	3.14		0.675
R4 radius (21 rods)	4.50		0.575
Tube			
Pressure tube		5.30	5.950
Gap CO_2		5.95	8.250
Calandria tube		8.25	8.700

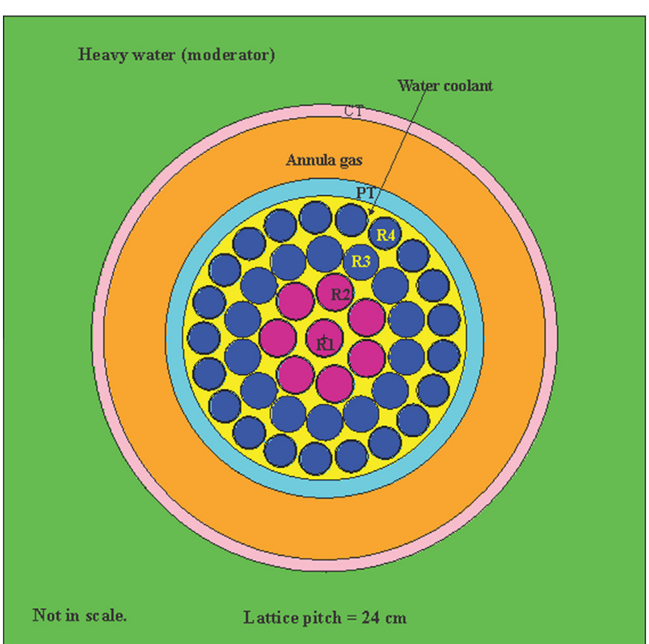

Fig. 1. ACR-1000 unit fuel channel lattice.

3. Monte Carlo fuel burnup analysis coupling bash script

The Monte Carlo burnup analysis methodology in this work consists of MCNP Coupling with ORIGEN2 BASH-Script (MCOS) Chang and Ryskamp, 2000; Chang, 2005. MCOS was written to link the MCNP with the depletion and buildup code ORIGEN2 (Croff, 1983; Ludwig and Croft, 2002; Bell, 1973). The Monte Carlo transport code MCNP is used to provide the one-group cross-section and neutron flux tally values for the calculation of radioactive decay and burnup calculation in the ORIGEN2 code calculation. The MCNP-generated reaction rates are integrated over the continuous-energy nuclear data and space within the region without using the group XS as required in the diffusion burnup code calculation. ORIGEN2 performs burnup calculations for MCOS using the matrix exponential method to calculate time-dependent formation, destruction, and decay for each divided fuel cell. These depletion calculations require MCOS to provide the following input to ORIGEN2: (1) the initial compositions and amounts of material, (2) one-group microscopic XS for each isotope, (3) material feed and removal rates (if desired), (4) the length of the irradiation period(s), and (5) the flux or power. In this work, we only calculated and updated the burnup dependent one-group XS of nuclides whose reactions are important to criticality, such as the major depleted actinides $^{234-238}$U and $^{239-242}$Pu, and SAFPs such as ^{135}Xe and ^{149}Sm. After depletion and buildup calculations are performed by ORIGEN2, MCOS passes isotopic compositions of materials back to MCNP to begin another burnup calculation time-step. For each time step, a MCNP KCODE option calculation with 5.0×10^4 neutrons is used in each of 115 calculation cycles. This required about 30 min of CPU/wall-time on a workstation with two dual-core 2.86 GHz XEON processors. The Fission Tally calculation for each fuel cell can achieve a tally relative error of 0.03 or less.

In summary, MCOS is a BASH shell script that couples the MCNP and ORIGEN2 computer codes. It can update the nuclides neutron spectrum weighted XS at the beginning of each time step, automatically from the fuel cycle beginning of life to the end of life without the need for any manual interface. The FCA model with MCOS burnup analysis can provide accurate neutronics burnup characteristics of the fuel axially divided cells vs. depletion time.

4. Results and discussion

MCOS was used in this work to compute the fuel depletion parameters relative profiles of the FCA model vs. depletion time. MCNP calculates all the FCA divided 2064 fuel cells burnup required neutron reactions and fission flux tallies. The Fission Tally is proportional to the product of the fission atomic density, fission cross section, and the neutron flux tally. The ORIGEN2 code uses the burnup calculation option, IRP, in the MCOS calculation. This allows automatic adjust neutron flux level to preserve the constant fission power during each burnup calculation time interval.