FUEL BURNUP CHARACTERISTICS FOR THE NRU RESEARCH REACTOR

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ABSTRACT

The driver fuel of the NRU research reactor at AECL, Chalk River is a low enriched uranium (LEU) fuel alloy of Al-61 wt% U_3Si, consisting of particles of U_3Si dispersed in a continuous aluminum matrix, with 19.8% U235 in uranium. This paper describes the burnup characteristics for this type of fuel in NRU, including the determination of fuel depletion using the neutronic simulation code TRIAD, comparisons between simulated and measured burnup values, and the regulatory licensing operational average fuel burnup limit.

1. Introduction

The National Research Universal (NRU) reactor at Chalk River began operation in 1957. It is used to carry out research in basic science and in support of the CANDU power reactor programs, such as the fuel bundle and material development programs. It is also a major supplier of medical radioisotopes in Canada and the world. The NRU reactor is heavy water cooled and moderated, with on-line refueling capability. It is licensed to operate at a maximum thermal power of 135 MW, and has a peak thermal flux of approximately 4.0×10^{18} $n.m^{-2}.s^{-1}$. The hexagonal lattice pitch is 19.685 cm.

The NRU reactor consists of many different types of rods, such as driver fuel rods, Mo-99 production rods, absorber rods and control rods. The number of driver fuel rods in the core varies from 85 to 95, depending on the nature of experimental programs being undertaken and the demand for radioisotope production. The NRU driver fuel is a low enriched uranium (LEU) fuel alloy of Al-61 wt% U_3Si, consisting of particles of U_3Si dispersed in a continuous aluminum matrix, with 19.8% U235 in uranium.

This paper describes the fuel burnup characteristics for the NRU reactor, including the determination of fuel depletion using the neutronic simulation code TRIAD, comparisons between simulated and measured burnup values, and the regulatory licensing operational average fuel burnup limit.

2. Fuel Burnup Simulation

The fuel burnup simulation of the NRU reactor is performed by the TRIAD [1] code, which is a three-dimensional neutron diffusion code in two energy groups. The diffusion equations are solved numerically using a finite difference method. The TRIAD code has two main modules. The first is the neutronic simulation module which calculates steady-state neutron flux and power distributions in the reactor. The present TRIAD version incorporates 301 hexagonal sites (227 rod sites plus 74 D_2O sites around the outside), each comprising six triangular prisms, with 18 axial segments of variable heights. There are a total of 6×301×18 triangular prisms (or meshes) for the whole reactor. The 18 axial cells representing a rod in NRU can be of different cell types, each with uniform neutronic properties. The detailed flux shapes and neutron spectra through each type of cell are determined using the WIMS-AECL neutron transport code [2]. The homogenized cell parameters, in two energy groups, are then calculated by flux- and volume-weighting the region material properties. Examples of cell parameters are the diffusion coefficient, and various types of absorption and scattering cross sections.

The second module of TRIAD is the core-following module which tracks reactor assemblies by name, reactor position and fuel rod burnup as they move into and out of the reactor. Reactor snapshots are stored as core loadings, which contain the necessary data for all axial segments of each type of rod in the reactor, such as the rod type in each site location, and the linear burnup (in MWd/cm) in each axial segment for fuel rods. To begin the core following process, it is necessary to specify an initial core loading and define a run period. The run period can be chosen between two TPD (Total Power Days) values, in MWd, which refers to the total thermal energy produced in the reactor from all fuelled sites in a specific time interval. The reactor power level is measured by the thermal power based on flow and temperature measurements of the river water in the secondary cooling circuit. The initial TPD is that for the initial core loading, and will correspond to the end of the previous run.

The run period between two snapshots may be further divided into smaller rod-change intervals as in the case of refuelling, and the fuel depletion (burnup accumulation) for each interval is calculated using a fixed channel power distribution. Both measured and simulated channel power distributions for the reactor core are used. For each change interval, the accumulated burnup for a rod in the reactor is calculated by distributing the reactor total power days (TPD) increment among the power generating rods in the core. The axial burnup increments over the run period for a particular rod are calculated using the ratio of axial section power to total rod power, from simulation (there is no measured distribution).

After a depletion step is complete, the rod change that occurs at the end of the interval is carried out. The data changes at this TPD are made using the rod power and burnup information for the rod taken out or put back into the reactor. The total reactor power for the next depletion interval is also adjusted. In this way, the accumulating burnup for each rod is calculated only for times during the run period when it is in the reactor.

3. Comparisons between Simulated and Measured Burnup Values

In 2010, axial fuel segment burnup measurements were performed on fuel samples collected from two NRU driver fuel rods, FL-1528 and Fl-1540, that had reached their end-of-life burnup of 342.1 and 310.2 MWd, respectively. Uranium isotopic ratios of U235/U238 and U236/U235 for these samples were measured using thermal ionization mass spectrometry. The measured exit burnup isotopic ratios were analysed using

WIMS-AECL to obtain the corresponding rod segment burnups in MWd/cm, which were compared to the TRIAD burnups at the same locations. There was reasonable agreement between measurements and TRIAD burnup values, with the measured-to-simulated burnup ratios varying from 0.95 to 1.07. More measurements are planned in the future to determine the measured-to-simulated burnup ratios more accurately.

4. Regulatory Licensing Operational Limit for the Average Fuel Burnup

Driver fuel rods in NRU may be shuffled a few times to different sites during their life cycle in the core, which is about 1 year. At different phrases of fuel burnup, site changes are necessary in order to take advantage of the different neutron flux levels in various region of the reactor. Usually, fresh fuel rods are placed in the outer part of the core, and the rods are then gradually shuffled into the centre. As the rods burnup further, they are moved back towards the outer part of the core. These higher burnup rods will remain there until they reach an exit burnup of ~340 MWd, and are finally removed from the core permanently.

For a typical NRU core loading, there are fresh and old fuel rods with burnups ranging from 0 to 340 MWd. The average driver fuel burnup value is usually maintained at ~190 MWd. The regulatory licensing operational limit for the average fuel burnup is 200 MWd per rod, regardless how many driver fuel rods in the core. This licensing limit was set to limit the buildup of excess fission products in the core, which is undesirable because it would cause a greater dose to the public in the event of an accident of fuel leakage.

5. Summary

This presentation reviews the fuel burnup characteristics and calculation methods applied to the NRU research reactor. Although the technique of tracking fuel burnup may be unique for this reactor, some of this experience for fuel burnup management may be useful to designers and operational staff working with other types of small reactor.

6. References

[1] T.C. Leung and M.D. Atfield, "Validation of the TRIAD Code Used for the Neutronic Simulation of the NRU Reactor", Proceedings of the 30th Annual Conference of the Canadian Nuclear Society, Calgary, Alberta, Canada, 2009 May 31 - June 3.

[2] J.D. Irish and S.R. Douglas, "Validation of WIMS-IST", Proceedings of the 23rd Annual Conference of the Canadian Nuclear Society, Toronto, Canada, 2002 June