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Impact of Soluble Boron Modeling for PWR Burnup Credit Criticality Safety Analyses

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INTRODUCTION

The utilization of credit for fuel burnup in an out-of-reactor criticality safety evaluation necessitates consideration of fuel operating conditions, including an understanding of the sensitivity of fuel reactivity to variations in the operating conditions, to establish justifiable assumptions. Consequently, numerous studies [e.g., see Refs. 1,2] have been performed to determine the effects of depletion modeling assumptions on calculated spent nuclear fuel (SNF) isotopic compositions and reactivity.

Trends in reactivity have been established as a function of each relevant parameter to determine the conservative direction and the magnitude of the effect over a realistic operating range. The effects of variations in the operating conditions relevant to pressurized-water-reactor (PWR) burnup credit are fairly well understood. In general, changes in operating conditions that result in spectral hardening, especially late in burnup, tend to increase the reactivity of SNF. Thus, the reactivity of SNF increases with increased moderator temperature, soluble boron concentration, fuel temperature, and presence of fixed poisons (e.g., control rods and burnable poison rods). The effect of variations in specific power is a little more complicated and is discussed elsewhere.[1]

The operating conditions are generally assumed to be constant during depletion, which is certainly not the case for soluble boron. It is commonly recognized that the use of a constant (cycle-averaged) soluble boron concentration during depletion, as opposed to detailed modeling of the actual variation in boron concentration, provides a conservative modeling approximation for burnup-credit analyses. However, no studies to support this position are readily found in the literature. Therefore, the impact of soluble boron modeling was studied, and the results are described in this paper.

DESCRIPTION OF THE ACTUAL WORK

In PWRs, excess reactivity is suppressed by soluble boron. As the excess reactivity decreases with fuel burnup, the boron concentration is reduced accordingly, referred to as *boron letdown*. After an initial drop due to xenon buildup, boron letdown is fairly linear with burnup for typical reload cores without heavy use of burnable absorbers.

The objective of this study is to demonstrate the impact of assuming a constant (cycle-averaged) boron concentration, as compared to detailed modeling of boron letdown, on SNF isotopic compositions and reactivity. For the cases modeling boron letdown, the boron concentration was varied linearly from a maximum value at beginning-of-cycle (BOC) to zero at end-of-cycle (EOC). The depletion calculations were performed with SAS2H [3], assuming 3 cycles of 15 GWd/tonne U per cycle, time steps of 1 GWd/tonne U, conservative operational parameters for fuel temperature (1100 K), moderator temperature (610 K), and specific power (continuous operation at 60 MW/tonne U), and Westinghouse 17×17 fuel with 4.0 wt % ^{235}U enrichment. The boron letdown representation is shown graphically in Fig. 1 for a cycle-averaged boron concentration of 600 parts-per-million (ppm).

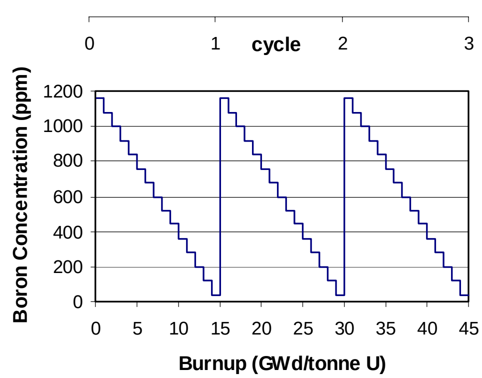

Fig. 1. As-modeled boron letdown for cycle-averaged concentration of 600 ppm.

Using the isotopic compositions from the SAS2H depletion cases, CSAS1X [3] one-dimensional calculations were performed to determine the effect on reactivity as a function of burnup for out-of-reactor conditions at burnup steps of 1 GWd/tonne U. The criticality calculations are based on an infinite array of spent-fuel pin cells using isotopics from the various depletion cases, and thus the effect of the soluble boron modeling is determined based on its effect on the depletion isotopics alone (i.e., no soluble boron is present in the criticality models).

RESULTS

Results from the sensitivity cases are shown in Table I for cycle-averaged soluble boron concentrations of 400, 600, 800, and 1000 ppm. The results are ratios of atom densities, atom density based on constant boron over atom density based on variable boron for the principal actinides and fission products important to reactivity. Relatively small differences are observed, indicating that the use of cycle-averaged boron is a fairly good approximation. Minor differences exist in the plutonium nuclides due to the increased spectral hardening late in burnup with the use of constant boron, as compared to variable boron in which very little boron is present late in burnup.

TABLE I. Ratios of Calculated Quantities for Constant versus Variable Soluble Boron Modeling at 45 GWd/tonne U and 5-year cooling

Cycle-Averaged Boron (ppm)	400	600	800	1000
	Ratio (Constant Boron / Variable Boron)			
^{234}U	1.000	1.000	0.999	0.999
^{235}U	0.999	0.998	0.997	0.997
^{236}U	0.999	0.999	0.999	0.998
^{238}U	1.000	1.000	1.000	1.000
^{238}Pu	1.002	1.003	1.005	1.006
^{239}Pu	1.007	1.010	1.012	1.015
^{240}Pu	0.996	0.994	0.993	0.991
^{241}Pu	1.006	1.009	1.011	1.014
^{242}Pu	0.997	0.996	0.994	0.993
^{241}Am	1.005	1.008	1.010	1.012
^{237}Np	1.002	1.003	1.004	1.006
^{95}Mo	1.000	1.000	1.000	1.000
^{99}Tc	1.000	0.999	0.999	0.999
^{101}Ru	1.000	1.000	1.000	0.999
^{103}Rh	0.999	0.998	0.998	0.997
^{109}Ag	0.998	0.997	0.997	0.996
^{133}Cs	0.999	0.999	0.999	0.999
^{143}Nd	1.000	1.000	1.000	1.000
^{145}Nd	1.000	0.999	0.999	0.999
^{147}Sm	0.998	0.996	0.995	0.994
^{149}Sm	1.020	1.029	1.039	1.048
^{150}Sm	1.000	1.000	1.001	1.001
^{151}Sm	1.014	1.021	1.027	1.033
^{151}Eu	1.014	1.021	1.027	1.033
^{152}Sm	0.996	0.994	0.992	0.990
^{153}Eu	1.001	1.001	1.002	1.002
^{155}Gd	0.992	0.989	0.986	0.983
k_{inf} (ao)	1.001	1.002	1.002	1.002
k_{inf} (a+fp)	1.001	1.001	1.001	1.002

The impact of the isotopic differences on reactivity is illustrated in Fig. 2. The behavior is consistent with expectations: (1) where each case includes a greater concentration of boron, and subsequently harder neutron spectrum, there is a tendency toward more reactive isotopic compositions and (2) the impact of a hardened spectrum is greater later in burnup. Because the constant boron modeling has significantly more boron present late in burnup, it results in slightly more reactive isotopic compositions for discharged SNF. Therefore, it is a conservative modeling approximation to employ in burnup-credit evaluations. Finally, note that three-dimensional KENO calculations with a 32-assembly cask model produced results consistent with those shown in Fig. 2.

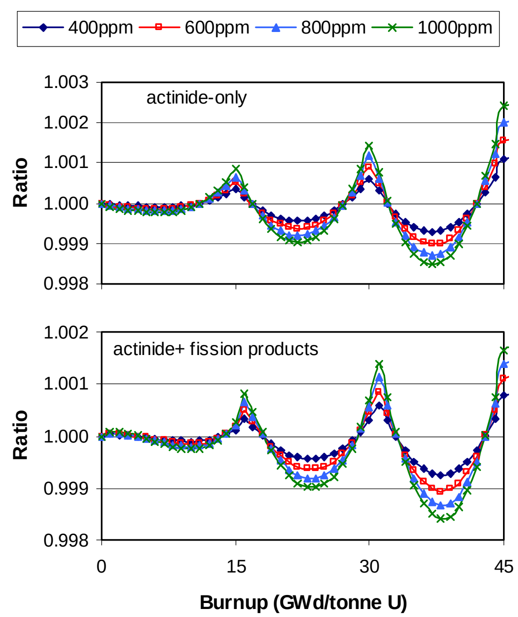

Fig. 2. Ratio of k_{inf} values (constant boron/variable boron) as a function of burnup.

REFERENCES

1. C. V. PARKS, M. D. DEHART, J. C. WAGNER, *Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel*, NUREG/CR-6665 (ORNL/TM-1999/303), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, February 2000.
2. J. C. WAGNER, C. E. SANDERS, *Assessment of Reactivity Margins and Loading Curves for PWR Burnup-Credit Cask Designs*, NUREG/CR-6800 (ORNL/TM-2002/6), U.S. Nuclear

Regulatory Commission, Oak Ridge National Laboratory, March 2003.

3. *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation,* NUREG/CR-0200, Rev. 6 (ORNL/NUREG/CSD-2/R6), Vols. I, II, and III, May 2000. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-545.