Progress in Nuclear Energy 81 (2015) 184–195

Contents lists available at ScienceDirect

Progress in Nuclear Energy

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

Variability in spent fuel inventory data

Kristina Yancey*, Pavel V. Tsvetkov

Texas A&M University, 3133 TAMU, College Station, TX 77843-3133, USA

ARTICLE INFO

Article history:
Received 31 March 2014
Received in revised form
14 November 2014
Accepted 2 December 2014
Available online 2 March 2015

Keywords:
Applicability range
Nuclear power
Spent fuel
Nuclide
Actinides
Fission products

ABSTRACT

A new methodology has been developed to assess how representative a collection of nuclide mass estimates is of real-world spent fuel. The analysis is approached as an "applicability range" evaluation, which quantifies the fraction of the historical population that is represented by one mass estimate. The new methodology is applied to a database containing spent fuel inventory estimates for operating U.S. reactors and uses historical assembly designs as a reference. The evaluation consists of two major steps: the implementation of a sampling and randomization scheme and the calculation of the applicability range scores. The mass estimates are assigned scores for each of the assembly designs they should represent, and the scores are averaged together to find overall applicability range scores for the database estimates. The results showed that the estimates for newer, PWR assembly classes had much higher scores than estimates for older or BWR assembly classes. The applicability range methodology can be extended beyond the database analyzed in the present work to any study that needs to use one value to represent a population with broad variations and limited knowledge of the underlying distributions.

1. Introduction

This paper presents a new methodology to quantify how representative an estimate is of a variable based on the known historical variation of its input factors. The method is applied to the analysis of the Spent Fuel Database (SFD), which was created for advanced fuel cycle and waste management studies (Yancey and Tsvetkov, 2014). The database was built using publicly available information to model individual spent fuel assemblies in ORIGEN-ARP (Gauld et al., 2009) and contains estimates of the nuclide inventories in those assemblies. The new method was developed as a way to understand some of the uncertainty of the mass estimates in the SFD.

1.1. Spent fuel database

The models used to create the SFD are generalized versions of their real-life counterparts. The database covers the 103 United States (U.S.) nuclear reactors that were operating in January 2012. The model for each reactor used available information about power levels, capacity factors, and more to generate mass estimates for 98 nuclides, including actinides and 20 fission products, on a per assembly basis. The models included certain assumptions that, together with the variability in the input data, could distort the estimates in the database. A previous paper discusses the broader implications of the assumptions that were made (Yancey and Tsvetkov, 2014). This paper explores two specific assumptions that could affect how many real-world assemblies the SFD estimates would be able to represent: enrichment and initial uranium content.

In 2012, a paper was released by Oak Ridge National Laboratory (ORNL) reporting the results of a technical review conducted by the U.S. Department of Energy's Office of Fuel Cycle Technologies (Wagner et al., 2012). This report greatly augmented publicly available information about spent fuel. For the models used to create the SFD, the ^{235}U enrichment was assumed to be 4.0 wt%, and the initial uranium content was chosen for each reactor based on context clues in (EIA, 1995). The ORNL report included tabulated information about these two input variables that facilitated the development of the new method presented here. While it did not give the exact enrichment and initial uranium content used by each reactor over the course of its lifetime, the report did present average values along with maximum and, in the case of enrichment, minimum values for historical assembly designs used by industry up until 2004. Fig. 1 was compiled based on the average enrichment values given in the ORNL report. The histogram shows the variation in enrichment from 1.5 to 5.0 wt% enrichment in 0.05%

* Corresponding author. Tel.: +1 979 492 4082.
E-mail addresses: kristina.yancey@gmail.com (K. Yancey), tsvetkov@tamu.edu (P.V. Tsvetkov).

http://dx.doi.org/10.1016/j.pnucene.2014.12.002

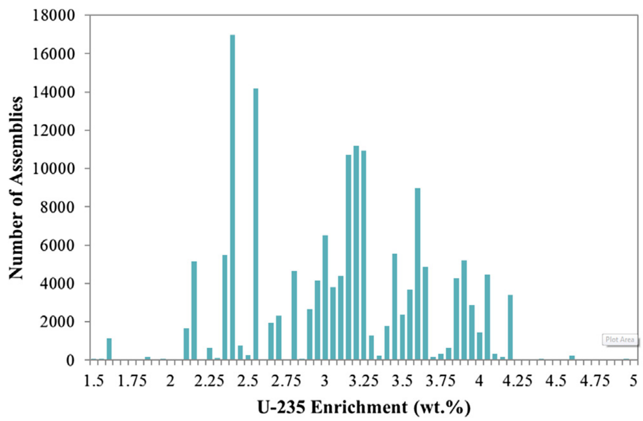

Fig. 1. The variation of ^{235}U enrichment in fresh fuel assemblies used in U.S. reactors up until 2004.

increments. As shown in Fig. 1, the spread around the assumed value of 4.0 wt% is quite large, with 92.8% of the assemblies falling below that value.

Expanding on the data presented in Fig. 1, Fig. 2 shows that even if the historical average enrichment level was chosen instead of 4.0 wt%, one standard enrichment level poorly represents the population. Fig. 2 was created using JMP[1] (2012) and shows that a normal distribution does not fit the data. Using the KSL "goodness of fit" test, the normal distribution was ruled out as a sufficient model at a significance level of 95%, with a *p*-value of 0.001. These figures suggest that variation in enrichment must have some impact on the database. No single enrichment is a good representation of the levels used in the past.

Burnup effects were neglected in this study even though the ORNL report also listed average and maximum values for this factor. It is recognized that burnup will significantly affect the accuracy of the database mass estimates, but the methodology used to generate the ORIGEN-ARP models did not accommodate a simple perturbation in this respect. It was assumed that each assembly went through three 1.5 year cycles in the reactor and that the cycles were separated by refueling outages at 0% power. The length of an outage was determined by the capacity factor for an individual reactor in a given year. With this setup, changing the burnup would affect more than one parameter. For each sample run, it would need to be decided if the burnup should only perturb the length of each cycle, or if it would be more accurate to reduce the number of cycles to two. While the choice may not significantly affect the dominant fissile nuclides, it would affect the higher actinides, such as the concentration of ^{242}Cm. To evaluate the SFD required over 54,000 new runs, so incorporating these additional complications into the analysis was beyond the scope of this study. In a formal uncertainty quantification, burnup must be considered.

While the ORNL report did not connect exact values to individual reactor units, the assembly design information can be connected to specific assembly classes. These classes are listed in Table 1, which is reprinted here for clarity (Yancey and Tsvetkov, 2014). Table 1 presents average power histories for each assembly class, calculated using the collected information from all of the units belonging to a certain class (EIA, 1995; NRC, 2011a, c). The assembly classes are abbreviated company names followed by a multiplication factor, denoting the size of the assembly. Here, "B&W" stands for Babcock & Wilcox, "CE" stands for Combustion

[1] JMP was created by John Sall and originally stood for "John's Macintosh Program" (Shipp and Lafler, 2012). Today, it is pronounced, "jump."

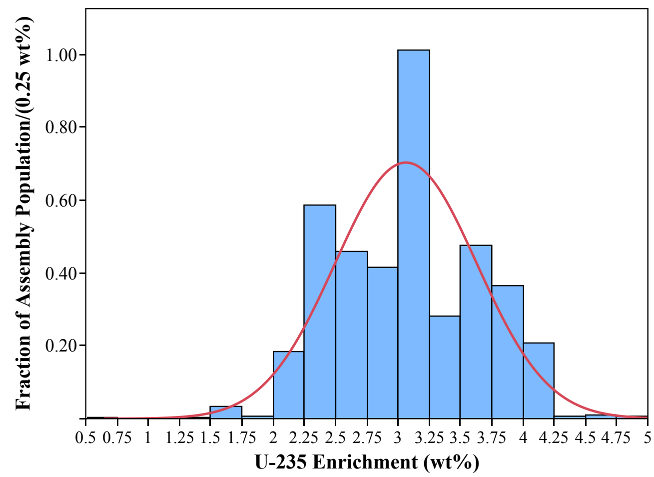

Fig. 2. ^{235}U enrichment levels used in U.S. reactors up until 2004 with a Normal (3.06584,0.56754) continuous fit overlay.

Engineering, "GE" stands for General Electric, and "W" stands for Westinghouse. The assembly designs listed in the ORNL report were connected to each class through cross-referencing (EIA, 1995). To emphasize the distinction between assembly design and assembly class, "assembly class" refers to the structure of a reactor, while "assembly design" refers specifically to the type of assembly going into the reactor. A reactor's assembly class cannot change, while the assembly design often changes to accommodate higher burnups and other technological improvements.

1.2. Applicability range

A traditional sensitivity analysis could not be performed on the mass estimates because the sample size within the database was too small, and the choices of enrichment and of initial uranium content by industry were not random. The levels for both were decided based on what was best commercially for each reactor according to specific utility practices. Therefore, the distributions of the enrichment levels and the initial uranium content used over the past forty-some years are not normal. Within the field of statistics, certain analysis methods are available to understand non-normal distributions, but they are beyond the scope of this work. Instead, a simple methodology was developed to assess what will be called the "applicability range" of the database.

For the purpose of this research, the ***applicability range*** (AR) is defined as the fraction of assemblies used over a reactor's lifetime

Table 1
Reactor classification groups and averaged power histories for units belonging to the assembly design class.

Design class	Number of units in class	First month of operation	Initial power (MWth)	Month of uprate	Final power (MWth)	Number of assemblies
B&W 15 × 15	6	May 1975	2609	–	2610	177
CE 14 × 14	4	May 1976	2560	Apr. 1989	2719	217
CE 16 × 16	4	June 1983	3246	July 2002	3405	217
CE 16 × 16 Sys 80	4	Apr. 1985	3554	Dec. 2000	3749	241
GE BWR 2,3	8	Aug. 1971	2191	Aug. 2001	2426.4	724
GE BWR 4,6*	27	Mar. 1981	2980	Dec. 2003	3263	764
W 14 × 14	6	Dec. 1972	1488	Feb. 2007	1664	121
W 15 × 15	8	Nov. 1973	2671	Aug. 2000	2799	157
W 17 × 17	32	Jan. 1985	3227	Nov. 2000	3355	193

*This class also has an additional uprate between start-up and the final uprate. This average middle uprate occurred in June 1992, increasing power to 3146 MWth.