Analysis of Decay Heat Measurements for BWR Fuel Assemblies

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INTRODUCTION

Due to advances in fuel and core design and fuel cycle optimization, the burnup range of assemblies discharged from commercial reactors has been steadily extended to higher values. In order to determine the accuracy of decay heat predictions for high-burnup spent fuel, current analysis methodologies and computer codes need to be validated to cover a larger enrichment and burnup domain. The scarcity of available experimental measurements required for this validation makes the endeavor rather difficult.

An experimental program recently initiated at the Central Interim Storage Facility (CLAB) in Sweden [1] to perform calorimeter measurements on full-length assemblies provides valuable thermal decay heat data for both BWR and PWR assembly designs over a large burnup range. Oak Ridge National Laboratory (ORNL), a collaborator in this program through support from the U.S. Nuclear Regulatory Commission (NRC), is evaluating the measurements with the SCALE 5 computer code system [2].

The analysis of BWR decay heat measurements discussed here is only a part of an extensive validation effort at ORNL intended to support the extension of the NRC Regulatory Guide 3.54 to include high-burnup spent nuclear fuel. The experimental data used to support the development of this guide included measurements on BWR assemblies with burnup up to 28 GWd/MTU; the analysis presented in this paper includes measurements that extend this burnup range up to 47 GWd/MTU.

ASSEMBLY DESCRIPTION

The decay heat measurements performed at CLAB that are discussed here include 34 BWR assemblies from eight reactor units. These assemblies can be arranged by the type of assembly design into four groups: 8x8, 9x9, SVEA-64 and SVEA-100. Each of these groups can be further subdivided in subgroups based on the details of the assembly configuration such as fuel pellet or rod diameter, assembly pitch, number of water rods, number of burnable absorber (BA) rods, concentration of the absorber material in the BA rods, etc.

The assemblies studied cover not only a large variety of assembly configurations, but also a large range of burnup (14 to 47 GWd/MTU) values and cooling times (11 to 26 years). A summary of the measured assemblies is presented in Table I.

Table I. Summary of assemblies measured.

Design	No.	Burnup (GWd/MTU)	Enrichment (wt % ^{235}U)	Decay time (years)
8x8	25	14.5 – 41.1	2.09 – 2.97	11.4 – 26.7
9x9	3	35.1 – 37.9	2.94	12.5 – 13.4
SVEA-64	4	32.4 – 46.7	2.85 - 2.92	12.4 – 15.4
SVEA-100	4	31.3 – 40.4	2.71 – 2.77	12.5 – 13.4

A 2-D layout for selected 8x8, 9x9 and SVEA-64 assemblies is shown in Fig. 1. The UO_2 fuel rods are shown in black (60, 70 and 59 for the 8x8, 9x9 and SVEA-64 assemblies, respectively), the UO_2/Gd_2O_3 fuel rods are shown in light grey (3, 6 and 4 for the 8x8, 9x9 and SVEA-64 assemblies, respectively), whereas the water rods are pictured in dark grey.

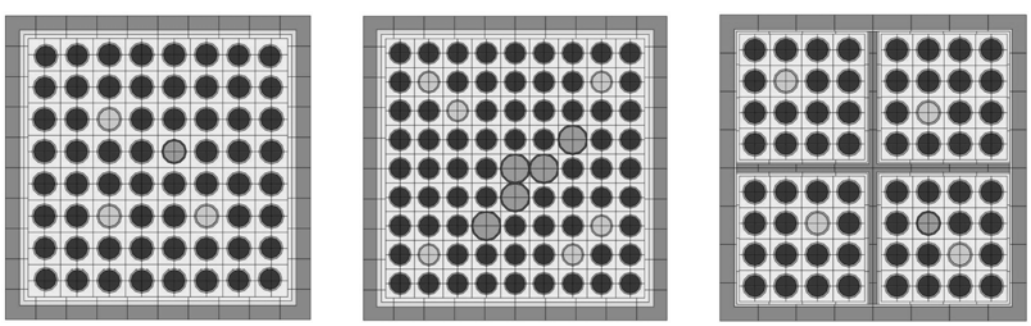

Fig. 1. Selected 8x8, 9x9 and SVEA-64 assembly configurations

There is one water rod in each of the 8x8 and SVEA-64 assemblies and five water rods assumed to be clustered toward the center in the 9x9 assembly. A unique feature of the SVEA design is the use of a large water cross moderating region that separates the fuel rods effectively into four subassemblies. Characteristic of most 8x8 assemblies considered is the presence of fuel rods at the corners of the assembly that have a pellet and rod diameter slightly smaller that the corresponding value for a regular fuel rod.

ANALYSIS METHODOLOGY

The analysis methodology is based on the 2-D depletion sequence TRITON and the point depletion module ORIGEN-ARP of the SCALE system. The analysis involves two steps: (1) performing TRITON simulations for each group of fuel assemblies with similar configurations to generate burnup-dependent cross section libraries for use with ORIGEN-ARP, and (2) performing stand-alone ORIGEN-ARP runs to simulate the depletion of each individual assembly using the library corresponding to the particular assembly design.

Burnup-dependent cross section libraries are generated with TRITON for different enrichment and coolant density values. In order to obtain the cross sections for ORIGEN-ARP simulations, the ARP module in SCALE is used. ARP interpolates the set of cross section libraries generated with TRITON by employing a low-order Legendre polynomial fit. The parameters of the interpolation are fuel enrichment, coolant density, and burnup.

TRITON is a module that couples the 2-D arbitrary polygonal mesh transport code NEWT with ORIGEN-S in order to perform a depletion simulation. At each depletion step, the transport flux solution from NEWT is used to generate the cross sections for the ORIGEN-S calculation; the isotopic composition data resulting from ORIGEN-S is employed in the subsequent transport calculation to obtain cross sections for the next depletion step, and so forth.

Another possible analysis approach is to individually simulate with TRITON each of the 34 assemblies considered; the nuclide concentration file generated by TRITON can be postprocessed with the OPUS utility code from SCALE in order to extract the decay heat data. The drawback of this approach is the lack of time efficiency, as separate TRITON simulations are required for each individual assembly, including those with a similar design. In addition, if multiple measurements (i.e., different decay times) are available for a particular assembly, a separate simulation is necessary for each value of the decay time.

One of the advantages of the ORIGEN-ARP methodology is its computational efficiency; only one TRITON simulation is performed to generate the library for each group of assemblies with similar design. Once the required cross section library is available, the running time for an ORIGEN-ARP simulation of an assembly is much shorter than a TRITON simulation of that assembly (typically seconds rather than hours), depending on the complexity of the modeled assembly.

For a unique geometry configuration, TRITON simulations are carried out for each combination of values from a set of fuel enrichment and coolant density points that covers all the assemblies having that configuration. A burnup-dependent cross section library is therefore generated for each of these combinations. A set of two enrichment points (2.0 and 3.0 wt % ^{235}U) and five coolant density points (0.1, 0.3, 0.5, 0.7, and 0.9 g/cm^3) are considered in this work, as these values are sufficient to simulate the measured assemblies. The libraries are generated for 24 equidistant burnup points to cover the 0 to 72 GWd/MTU burnup range, with a burnup step of 3 GWd/MTU.

RESULTS

The ORIGEN-ARP simulations performed for each of the BWR assemblies studied account for the contribution to the decay heat of the light elements in those structure materials (e.g., spacers) that are not present in the TRITON model. The contribution of these elements to the total thermal decay heat is not negligible, as it can add up to a few percent. For each assembly, the density of the coolant and the burnup are assembly average values obtained by axially averaging the void and burnup distributions obtained from utility data. The effect of modeling the axial variation in burnup and coolant density on the calculated decay heat for the assembly is not addressed in this work.

A total of 45 BWR measurements were available, as multiple measurements were performed for some of the assemblies. The average of the calculated-to-experimental decay heat ratio (C/E) was 1.005, with a standard deviation of 2.4%. The comparison of the measured and calculated decay heat values, as well as of the C/E ratio as a function of burnup is presented in Fig. 2.

The distribution of the C/E values around the mean is quasi-normal, as shown in Fig. 3. A least squares fit with a Gaussian shape gives the square of the correlation coefficient as $R^2 = 0.956$, which is close to unity.

The results obtained demonstrate the capability to accurately model decay heat measurements for BWR assemblies using SCALE 5, for a large burnup range, long cooling times, and a variety of assembly configurations. Work is in progress at ORNL to estimate the code bias and uncertainty for decay heat predictions corresponding to a large set of measurements on both PWR and BWR assemblies that were performed at different experimental facilities. This validation work will extend to higher burnups the decay heat predictions data previously used to support the development of the NRC Regulatory Guide 3.54: from 39 GWd/MTU and 28 GWd/MTU for PWR and BWR, respectively, to 50 GWd/MTU and 47 GWd/MTU.

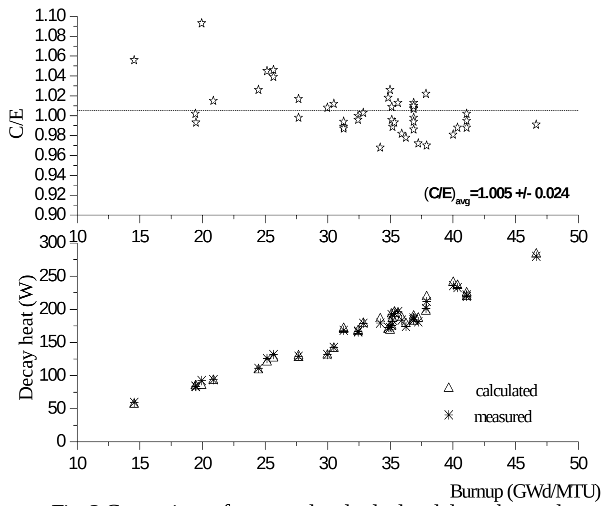

Fig. 2 Comparison of measured and calculated decay heat values.

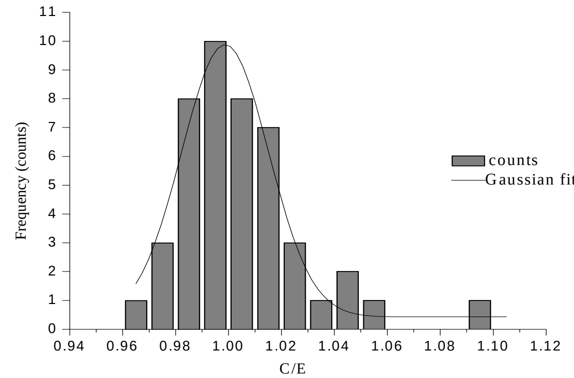

Figure 3. Distribution of (C/E) values

ACKNOWLEDGMENT

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