M&C 2017 - International Conference on Mathematics & Computational Methods Applied to Nuclear Science & Engineering, Jeju, Korea, April 16-20, 2017, on USB (2017)

Verification and Validation of STREAM Source Term Calculation Capability

Bamidele Ebiwonjumi[1], Sooyoung Choi[1], Deokjung Lee[1, *], Ho Cheol Shin[2]

1. Department of Nuclear Engineering, Ulsan National Institute of Science and Technology, UNIST-gil 50, Ulsan, Korea
2. Core and Fuel Analysis Group, Korea Hydro & Nuclear Power Co. Central Research Institute (KHNP-CRI), 70 Yuseong-daero, Yuseong-gu, Daejeon, Korea
[*]*Corresponding author*: deokjung@unist.ac.kr

Abstract - *This paper presents the verification and validation of the source term calculation capability implemented in the reactor analysis code STREAM for the purpose of spent fuel analysis. Activity, decay heat, neutron and gamma source spectra of spent PWR fuel assemblies discharged are calculated. The verification is performed by comparison with ORIGEN-ARP calculations. Validation is done by comparing with experimental decay heat measurements from Swedish central storage facility for spent nuclear fuel, CLAB. The verification and validation results show good agreement with ORIGEN-ARP calculations and experimentally measured decay heat respectively.*

I. INTRODUCTION

The calculation of spent LWR fuel radiation source terms is important to support the back-end cycle operations: storage, transportation, reprocessing and final disposal. Determination of these radiological properties is necessary in the design and safety of nuclear installations. The loading of spent fuel into onsite pools, casks for interim storage puts a limit on the decay heat, gamma and neutron sources. Acceptance of spent fuel into disposal facility requires the activity concentration to be lower than specified regulatory limit. Spent fuel pools onsite are being filled up. Since it is impractically possible to obtain experimental measurement of spent fuel characteristics for all the assemblies, there is need to predict these source terms accurately at cooling times after discharge from the reactor so that the integrity of the storage facilities is maintained and excessive cooling time required for the radiation to reduce to acceptable levels is avoided.

In order to extend the capabilities of the STREAM (Steady state and Transient Reactor Analysis code with Method of Characteristics) code developed at UNIST, a source term calculation capability has been implemented to calculate radioactivity, decay heat, gamma source intensity (and spectra) and neutron source intensity (and spectra) from irradiated fuel. STREAM is a reactor analysis code that performs LWR whole core calculation to generate isotopic number densities for the radionuclide inventory during irradiation and decay, which of course is required for source term calculation. The goal of this work is to present the verification of this capability against ORIGEN-ARP [1] calculations and validation against experimental data.

II. METHODS OF CALCULATION

The source term calculation capability includes neutron sources due to spontaneous fission decay and (α, n) reactions. With the knowledge of the concentration of the isotopes present in the irradiated fuel, the source terms can be calculated for spent nuclear fuel assemblies or fuel assemblies in the core during downtime (end of cycle, due to shut down or refueling or cooling). The activation products, fission products and actinides present in spent fuel are generated from the nuclear reactions (capture, fission, absorption) and decay processes taking place in the fuel.

STREAM uses Chebyshev Rational Approximation Method (CRAM) [2] to solve the depletion equation for time-dependent nuclide concentrations. STREAM is a neutron transport code that uses the Method of Characteristics (MOC) to calculate the neutron flux. The one group reaction cross section used in the depletion calculation is generated through energy condensation using transport solution in STREAM. The flow of work in this method is shown in Fig. 1.

STREAM uses 72 energy group cross section generated from NJOY that uses ENDF/B-VII.0, then applies resonance treatment to generate effective multi-group cross section [3]. This operation is performed for each fuel assembly by using a two-dimensional (2D) arrangement to obtain problem-specific cross section which is used in the transport calculation. The flux obtained from the neutron transport solution is used to collapse the multi-group cross section into one-group cross section for the depletion calculation. The flux and cross section are then updated at every burnup step using the computed time-dependent nuclide concentrations.

The end product of this simulation is the radionuclide isotopic inventory of the irradiated fuel assembly. The nuclide concentrations are then used with decay data, photon data and neuron data to calculate the associated activity, decay heat and radiation source terms.

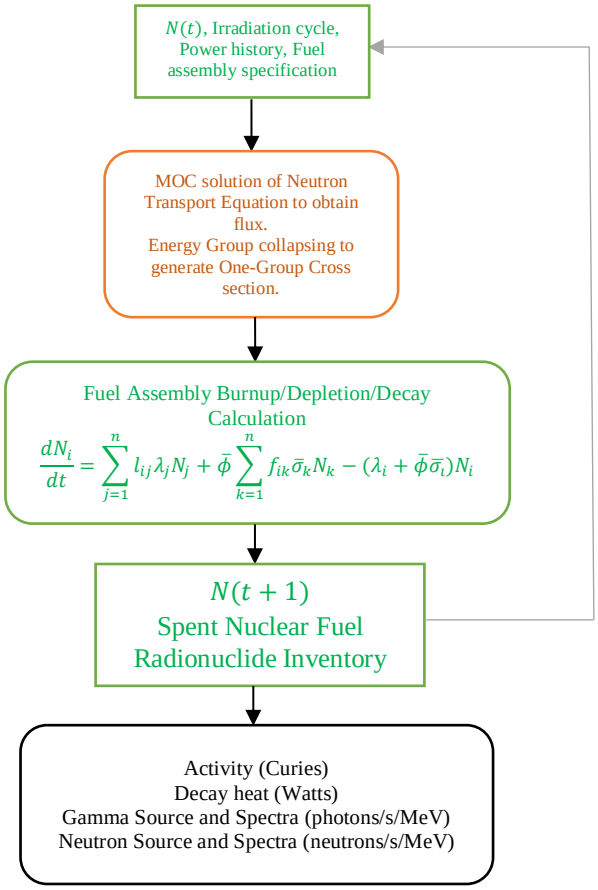

Fig. 1. Workflow for depletion calculation with STREAM.

1. Radioactivity and Decay Heat

During reactor operation, decay heat constitutes about 7% of the total energy recoverable from fission. Irradiated fuel assemblies contain radioactive isotopes whose radioactivity lead to decay heat as the emitted particles lose their kinetic energy in the spent fuel. This process goes on for long even at the end of irradiation when the heat generation is due solely to decay heat. Heat generation rate in spent fuel assemblies determines the peak clad temperature. The decay heat from spent fuel can lead to high temperatures where the fuel cladding and other shielding materials are degraded. This information is required to ensure that design and licensing specifications are not exceeded.

The results of STREAM's depletion calculation are used with decay data and photon data from ENDF/B-VII.0 libraries and ENSDF respectively, to calculate the source terms. The product of the decay half-lives and recoverable energy released per decay with the nuclide number density gives the radioactivity and decay heat in units of curies and watts respectively.

2. Gamma Sources and Spectra

Source spectra is important in the design of shields for radiation protection. This is because the probability of radiation interaction with shielding material depend, to a large extent, on the energy of the emitted photons and neutrons. Also, in shielding analysis, radiation sources are required in a given energy-group structure that is similar to existing cross section data. Thus it is important that gamma and neutron sources be described according to their energies. The photon library contains photon yield per disintegration which when multiplied with the activity can be converted to photon source intensity and spectra. The photon data accounts for x rays, gamma rays, decay gamma, spontaneous fission gamma, (α, n) reaction gamma and bremsstrahlung. Gamma sources in spent fuel are dominated by fission products.

3. Neutron Sources and Spectra

Neutron sources in spent fuel are majorly due to actinides which decay by α-emissions and spontaneous fission. Neutron sources due to spontaneous fission and (α, n) reactions are the two major contributor to the total neutron source in spent fuel. The spontaneous fission branching fraction and the neutrons yield per spontaneous fission are required to calculate the spontaneous fission neutron source. Neutron spectra due to spontaneous fission is calculated from the Watt distribution

$$N(E) = Ce^{-E/a}\sinh\sqrt{bE} \qquad (1)$$

where a and b are Watts fission spectrum parameters, E and C represent normalization constant and the neutron energy respectively.

α-emissions from actinide decay interact with the oxides in reactor fuel resulting in neutron emissions as the alpha particles slow down and are stopped. The neutron yield from this reaction is required to calculate the (α, n) neutron source. The neutron yield in oxide is calculated using the (α, n) reaction cross section, stopping cross section of α particle and the alpha particle energy.

$$\mathrm{Y}_i(E_\alpha) = \frac{N_i}{N}\int_0^{E_\alpha}\frac{\sigma_i(E)}{\varepsilon(E)}dE \qquad (2)$$

where $\sigma(E)$ is the (α, n) reaction cross section, $\varepsilon(E)$ is the stopping power expressed in terms of stopping cross section, E_α is the energy of emitted alpha particle N_i is the number density of the target nuclide which in this case is oxide material and N is the total number density of all nuclides present.

III. VERIFICATION

Source terms calculations for a given problem, a 17 x 17 PWR fuel assembly, with 3.1% enrichment, irradiated to a burnup of 60 MWd/kg with power density 40 W/g was verified by comparing the result of STREAM with ORIGEN-ARP. The comparisons for activity, decay heat and the total neutron source for up to 1000 years of cooling times is shown in Fig. 2, Fig. 3 and Fig. 4.

The agreement with ORIGEN-ARP shows that the source term calculation capability in STREAM can be used to accurately predict activity, decay heat and radiation source terms due to neutron and gamma emissions from spent fuel.

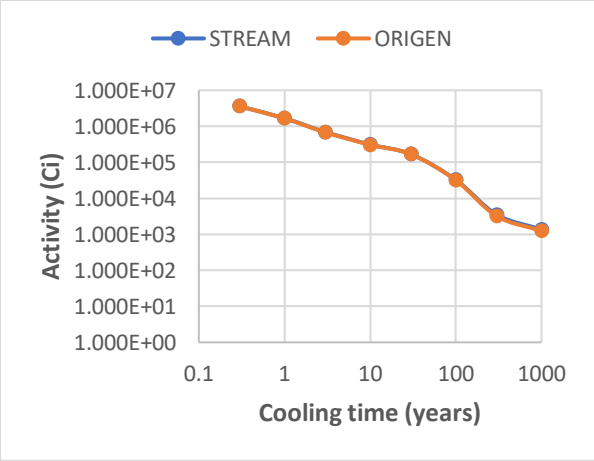

Fig. 2. STREAM activity comparison with ORIGEN-ARP.

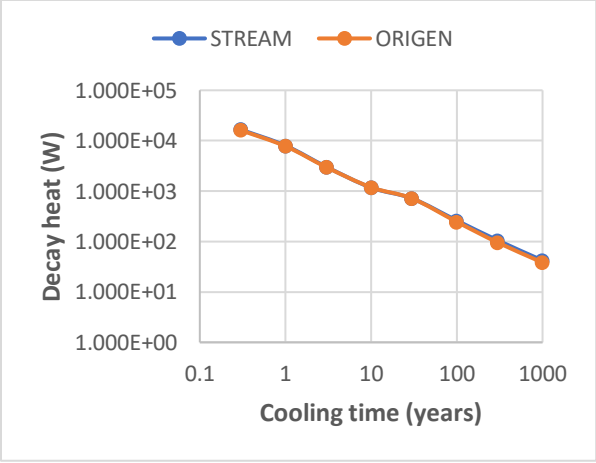

Fig. 3. STREAM decay heat comparison with ORIGEN-ARP.

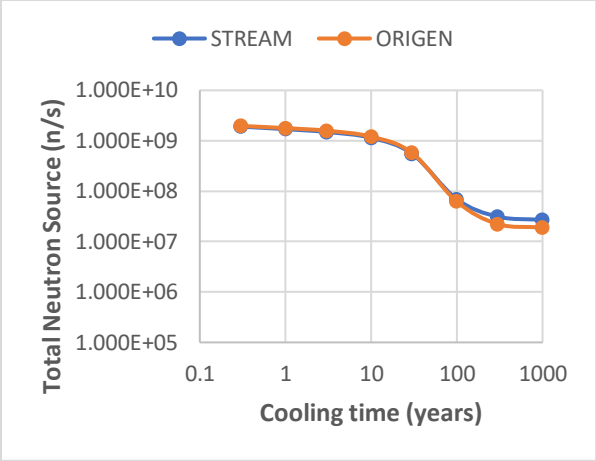

Fig. 4. STREAM neutron source comparison with ORIGEN-ARP.

IV. VALIDATION

The source term calculation capability was validated by comparing decay heat calculations with decay heat calorimetric measurements performed at the Swedish central interim storage facility for spent nuclear fuel, CLAB [4]. The PWR fuel assemblies measured are of two design types: 15 x 15, 17 x 17 and they cover the enrichment range 3.095 – 3.404 (wt% ^{235}U), burnup range 20 – 51 GWd/MTU, and cooling time 12 – 27 years. The 15 x 15 fuel assemblies are 18 in number, while the 17 x 17 fuel assemblies are 16. Some assemblies are measured more than once. A total of 71 measurements is used in this validation. Comparison of calculated decay heat with measured decay heat is illustrated in Fig. 5. The ratio of calculated decay heat to experimental decay heat (C/E) and the difference between calculated and measured decay heat (C-E) is shown in Table 1. The average, overall C/E is 1.010±0.017. These results are based on using a constant power history in STREAM's depletion calculation.

Table I. Calculation-experiment comparison

Assembly design	No. of measurements	C/E		C-E (W)	
		Mean	Std. Dev	Mean	Std. Dev
15 x 15	33	1.011	0.017	4.95	7.23
17 x 17	38	1.009	0.016	5.11	8.98
Overall	71	1.010	0.017	5.04	8.22

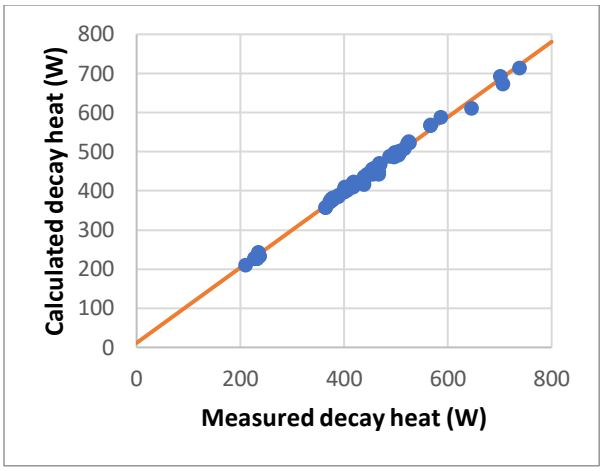

Fig. 5. Calculated vs. measured decay heat.

The comparison between calculated and measured decay heat shows good agreement as shown in Fig. 5 by the solid line. The uncertainty in measurement is reported [4] at 95% confidence level to be ±9.2 W (3.7%) at 250 W and ±18.2 W (2.1%) at 900 W. With this, the uncertainty in each measurement is obtained by linear interpolation. For the most of the 71 measurements, the calculated decay heat lies within the range of the measurement uncertainty. Only 9 calculated results lie outside the range of uncertainty in measurement. For some of these calculated results, the assemblies were rebuilt (fuel rods removed and reconfigured), while for the others, repeated measurements are performed. It may also be due to modeling data uncertainty. With the use of detailed power history information, it is expected that better agreements will be obtained in the calculated results.

V. CONCLUSION

Source term calculation capability implemented in STREAM has been used to examine its depletion capability. A set of 71 PWR fuel assembly decay heat measurements carried out at the Swedish central interim storage facility for spent nuclear fuel, CLAB, has been analyzed with STREAM. The PWR measurements involved 34 spent fuel assemblies with burnup ranging from 20 – 51 GWd/MTU and cooling times ranging from 12 – 27 years. The comparison between calculated and measured decay heat shows good agreement. The average calculated-to-experimental decay heat ratio is 1.010±0.017. The average bias in the calculation is 1%, with an uncertainty of 1.7%. Future work will include more validation studies with decay heat measurements performed for US PWR fuel assemblies at the Engine Maintenance and Disassembly (EMAD) facility of Hanford Engineering Laboratory (HEDL) and General Electric Morris Operation spent fuel storage facility.

REFERENCES

1. S.M. Bowman, Scale 6: Comprehensive Nuclear Safety Analysis Code System, Nucl. Technol. 174, 126 – 148 (2011).
2. M. Pusa, Rational Approximations to the Matrix Exponential in Burnup Calculations, Nucl. Sci. Eng. 169, 155 – 167 (2011).
3. S. Choi, C. Lee, D. Lee, Resonance Treatment using Pin-based Pointwise Energy Slowing-down Method, J. Comput. Phys. 330 (2017).
4. SKB Report R-05-62, "Measurements of Decay Heat in Spent Nuclear Fuel at Swedish Interim Storage Facility, CLAB," Svensk Kärnbränslehantering AB (SKB), Swedish Nuclear Fuel and Waste Management Co. (2006).