A Comparison of the TRITON and ORIGEN2 Source Generation Programs

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ABSTRACT

The Battelle Energy Alliance Research Reactor (BRR) Package has been designed to ship spent high-enriched aluminum-plate fuels from several research reactors. The reactor sites have historically used the ORIGEN2 [1] computer program for generating neutron and gamma source terms. However, ORIGEN2 is an older computer program that is no longer supported by the code developers. In addition, the reactor libraries used with ORIGEN2 are typically developed for low-enriched pressurized water reactors (PWR). To study the accuracy of the source term generated for spent research reactor aluminum plate fuel by ORIGEN2, the source term was regenerated using the TRITON [2] sequence of the SCALE6 code package. TRITON allows a two-dimensional representation of the fuel elements, and generates the data libraries in a problem-specific manner. Using TRITON, the gamma source was similar to the gamma source developed by ORIGEN2. However, the neutron source increased by 1 to 3 orders of magnitude. This paper compares the neutron and gamma source terms for both ORIGEN2 and TRITON, and suggests possible reasons for the differences.

INTRODUCTION

The BRR package is designed to transport spent fuel from three different research reactors, the Advanced Test Reactor (ATR), the University of Missouri Research Reactor (MURR), and the Massachusetts Institute of Technology Research Reactor (MITR-II). The fuel from each of these three reactors is similar. These reactors all utilize high-enriched uranium mixed in a matrix of aluminum, and clad with aluminum as either curved or flat plates. The neutron and gamma source terms for each of the fuel types was provided by staff at the host reactors, and each independently used ORIGEN2 to generate the source terms. Although ORIGEN2 is no longer supported by the developer, the program is still in use throughout the nuclear industry in the United States.

ORIGEN2 was originally released in 1980 and has gained widespread use. It is a very simple computer program to run, as it requires no geometry input. A number of cross-section libraries are provided with the program, and these libraries are collapsed based on specific reactor types. ORIGEN2 users typically select the library that is most appropriate to the problem at hand, although differences between the actual reactor and the reactor used to generate the library is inevitable. Reactor-specific libraries may also be developed. All three reactors in this study used different cross section libraries in their calculations. MURR used the THERMAL library, MITR-II used the PWRUS library, and ATR used an ATR-specific library.

It was decided to use the TRITON sequence of the SCALE6 code package to regenerate the ORIGEN2 source terms. TRITON is not a program per se, but a module that controls program flow between the two major components of the sequence, NEWT and ORIGEN-S. NEWT is a two-dimensional transport program that is used to compute the flux across the fuel element. Therefore, each of the three reactor types could be modeled explicitly, and the neutron spectrum determined uniquely for each reactor. The detailed flux solution determined in NEWT is then collapsed to three groups and used in the subsequent ORIGEN-S depletion calculation. This method has a greater computational rigor than ORIGEN2 because each library used in the depletion calculation is problem-specific. However, the computational rigor requires a much longer run time. While ORIGEN2 runs in 1 second, a TRITON model may take from several minutes to hours to execute, depending on the various control parameters selected and the desired rigor of the transport solution, which tends to be the limiting step.

GAMMA SOURCE COMPARISON

The ORIGEN2 and TRITON gamma source terms (γ/s) are provided in Table 1. The MURR source is for a burnup of 180 MWD and decay time of 180 days. The MITR-II source is for a burnup of 225 MWD and a decay time of 930 days. The ATR source is for a burnup of 350 MWD and a decay time of 1280 days. Agreement is quite good, considering that TRITON uses a much more advanced calculational method, and considering the ORIGEN2 data libraries for MURR and MITR-II are default libraries included with the program and not specifically generated for these reactors.

Because the BRR package is heavily shielded for gamma radiation, the effect on the dose rate is not readily apparent simply by examining the source terms in Table 1. Therefore, the maximum gamma dose rate on the surface of the cask is computed with the MCNP computer program for each source term and is reported in Table 2. For MURR and ATR, the increase in the gamma dose rate is negligible. For MITR-II, the increase in the gamma dose rate is 11.4%, which is not significant considering the uncertainties and conservatism of a dose rate calculation. It is concluded that ORIGEN2 and TRITON compare quite well for computing the gamma source term for these fuel types.

NEUTRON SOURCE COMPARISON

The neutron source may be extracted from the same output files used to compute the gamma source. The ORIGEN2 neutron source is provided only as a magnitude (no spectral information). The neutron source is comprised of spontaneous fission and (α,n) components.

The neutron sources for each of the fuel types for both ORIGEN2 and TRITON are summarized in Table 3. The spectral information for the TRITON source is not provided because equivalent information is not available from ORIGEN2. The neutron source is significantly larger using TRITON compared to ORIGEN2. The neutron source magnitude for MURR, MITR-II, and ATR increases by a factor of 1612, 35, and 12, respectively. The neutron dose rate scales proportionally with the neutron source magnitude, so the neutron dose rate will increase significantly.

The best agreement is for the ATR source. The spontaneous fission component is different by only a factor of 1.3, although the (α,n) component is different by a factor of 20. ATR is the only fuel type that utilized a user-generated cross-section library specific to ATR. This is the likely reason why the ATR results for spontaneous fission track reasonably well between ORIGEN2 and TRITON. The primary reason why the (α,n) component is significantly different is because ORIGEN2 uses an oxygen target with number densities typical of commercial light water reactor fuel, while TRITON uses an aluminum target consistent with the fuel element description. Only the small-abundance oxygen isotopes O-17 and O-18 are used as targets in ORIGEN2, while Al-27 is a target isotope that is present in the fuel at a high number density. No oxygen is present in the fuel matrix for any of the fuel types studied, which makes the ORIGEN2 methodology poor for this application.

The poorest agreement is for the MURR source. The ORIGEN2 input file used the THERMAL cross section library, which only considers thermal reactions. However, much of the neutron source is driven by daughter products of reactions with U-238 (such as plutonium), which is a higher energy reaction. Therefore, most of the neutron producing isotopes were neglected in the ORIGEN2 calculation. In fact, the ORIGEN2 MURR neutron source magnitude is only 30 neutrons/s per fuel element. By inspection it may be concluded that such a low number of neutrons is non-physical for a high-burnup fuel element. For this reason, the TRITON neutron source is a factor of ~1600 greater than the ORIGEN2 neutron source.

The results for MITR-II fall between MURR and ATR. The MITR-II ORIGEN2 calculations utilized the PWRUS library. However, the total neutron source magnitude is under predicted by a factor of 35 when comparing ORIGEN2 to TRITON. Based on the results from MURR, it may be inferred that the PWRUS library uses a spectrum that is not as hard as the actual MITR-II spectrum. A harder spectrum leads to more U-238 reactions and a larger neutron source. Also, as indicated in the MURR discussion, the ORIGEN2 (α,n) calculation is not applicable to this fuel type.

The maximum neutron dose rate on the surface of the cask computed with the MCNP computer program is summarized in Table 4 for each source term. The ratio of the dose rates between the ORIGEN2 and TRITON sources tracks well with the ratio of the neutron source magnitudes, as expected. Since the original neutron dose rates based on the ORIGEN2 source were quite low, increasing these dose rates by 1 to 3 orders of magnitude did not result in dose rates that exceeded the limits.

Although the neutron results from TRITON are significantly different than ORIGEN2, it is assumed that ORIGEN2 may accurately predict the neutron source for light water reactors, since the program was designed for this purpose.

CONCLUSIONS

ORIGEN2 and TRITON are two programs that may be used to generate gamma and neutron source terms for dose rate calculations. ORIGEN2 is a simple program that uses standard collapsed cross-section libraries based primarily on light water reactors. TRITON uses the geometry of the fuel element to perform explicit two-dimensional depletion calculations based on the actual fuel element spectrum. Based on calculations performed for MURR, MITR-II, and

ATR, it is concluded that the ORIGEN2 program works reasonably well for gamma source generation for these fuel types, regardless of the library selected. However, the neutron source term calculation is very sensitive to spectral effects, which are not captured properly in ORIGEN2 unless a reactor-specific library is created. Also ORIGEN2 cannot properly calculate the (α,n) neutron source for an aluminum target. Because agreement between ORIGEN2 and TRITON is poor for neutrons, TRITON is the recommended program to use for source generation for these fuel types.

REFERENCES

1. ORIGEN 2.1, *Isotope Generation and Depletion Code, Matrix Exponential Method,* CCC-371, Oak Ridge National Laboratory, August 1996.
2. *TRITON: A Two-Dimensional Transport and Depletion Module for Characterization of Spent Nuclear Fuel*, ORNL/TM-2005/39, Oak Ridge National Laboratory, January 2009.

Table 1. Gamma Source Terms per Fuel Element

Mean Photon Energy (MeV)	MURR Gamma Source (γ/s)		MITR-II Gamma Source (γ/s)		ATR Gamma Source (γ/s)	
	ORIGEN2	TRITON	ORIGEN2	TRITON	ORIGEN2	TRITON
1.00E-02	3.322E+14	3.334E+14	5.357E+13	5.586E+13	8.557E+13	8.623E+13
2.50E-02	7.122E+13	6.966E+13	1.167E+13	1.187E+13	1.868E+13	1.839E+13
3.75E-02	8.163E+13	8.513E+13	1.335E+13	1.465E+13	2.117E+13	2.240E+13
5.75E-02	6.650E+13	5.847E+13	1.076E+13	9.871E+12	1.717E+13	1.524E+13
8.50E-02	4.752E+13	4.401E+13	7.521E+12	7.366E+12	1.200E+13	1.134E+13
1.25E-01	7.077E+13	8.001E+13	9.086E+12	1.089E+13	1.438E+13	1.662E+13
2.25E-01	3.866E+13	4.044E+13	6.179E+12	6.672E+12	9.878E+12	1.033E+13
3.75E-01	1.873E+13	1.960E+13	3.156E+12	3.354E+12	5.090E+12	5.251E+12
5.75E-01	6.015E+13	7.095E+13	4.251E+13	5.041E+13	5.375E+13	5.533E+13
8.50E-01	3.184E+14	3.389E+14	1.125E+13	1.648E+13	8.026E+12	8.995E+12
1.25E+00	3.547E+12	4.045E+12	1.624E+12	1.995E+12	1.711E+12	1.561E+12
1.75E+00	4.426E+11	8.629E+11	7.615E+10	1.511E+11	1.162E+11	2.075E+11
2.25E+00	2.282E+12	2.173E+12	2.911E+11	2.880E+11	4.708E+11	4.547E+11
2.75E+00	8.308E+09	8.769E+09	1.225E+09	1.392E+09	2.057E+09	2.104E+09
3.50E+00	5.794E+08	4.661E+08	1.266E+08	1.061E+08	2.151E+08	1.586E+08
5.00E+00	5.166E-01	1.193E+02	9.235E+01	1.720E+03	1.335E+02	1.757E+02
7.00E+00	5.697E-02	1.355E+01	9.919E+00	1.968E+02	1.506E+01	1.990E+01
9.50E+00	6.390E-03	1.544E+00	1.093E+00	2.256E+01	1.708E+00	2.267E+00
Total (γ/s)	1.112E+15	1.148E+15	1.710E+14	1.898E+14	2.480E+14	2.524E+14

Table 2. Comparison of Cask Surface Maximum Gamma Dose Rates

Fuel Type	ORIGEN2 Gamma Dose Rate (mrem/hr)	TRITON Gamma Dose Rate (mrem/hr)	% Change
MURR	9.90	9.93	0.3%
MITR-II	2.85	3.17	11.4%
ATR	1.73	1.78	3.3%

Table 3. Neutron Source Terms per Fuel Element, ORIGEN2 and TRITON

MURR			
	ORIGEN2 Neutron Source (n/s)	TRITON Neutron Source (n/s)	Ratio, TRITON/ORIGEN2
Spontaneous fission	8.5	2.326E+03	274
(α,n)	21.5	4.604E+04	2,141
Total	30.0	4.837E+04	1,612
MITR-II			
	ORIGEN2 Neutron Source (n/s)	TRITON Neutron Source (n/s)	Ratio, TRITON/ORIGEN2
Spontaneous fission	1.527E+03	3.708E+04	24
(α,n)	6.867E+03	2.575E+05	38
Total	8.394E+03	2.946E+05	35
ATR			
	ORIGEN2 Neutron Source (n/s)	TRITON Neutron Source (n/s)	Ratio, TRITON/ORIGEN2
Spontaneous fission	2.687E+03	3.515E+03	1.3
(α,n)	3.190E+03	6.506E+04	20
Total	5.878E+03	6.858E+04	12

Table 4. Comparison of Cask Surface Maximum Neutron Dose Rates

Fuel Type	ORIGEN2 Neutron Dose Rate (mrem/hr)	TRITON Neutron Dose Rate (mrem/hr)	Ratio
MURR	7.51E-04	1.15	1537.3
MITR-II	0.30	10.20	33.5
ATR	0.12	1.32	11.0