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CHARACTERIZATION OF NEUTRON SOURCES FROM SPENT FUEL CASKS

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In the interim period prior to the acceptance of spent fuel for disposal by the USDOE, utilities are beginning to choose dry cask storage as an alternative to pool re-racking, transshipments, or new pool construction.[1] In addition, the current MRS proposal calls for interim dry storage of consolidated spent fuel in concrete casks.[2] As part of the licensing requirements for these cask storage facilities, calculations are typically necessary to determine the yearly radiation dose received at the site boundary. Unlike wet storage facilities, neutron skyshine can be an important contribution to the total boundary dose from a dry storage facility. Calculation of the neutron skyshine is in turn heavily dependent on the source characteristics and source model selected for the analysis. This summary will present the basic source characteristics of the spent fuel stored in dry casks and discuss factors that must be considered in evaluating and modeling the radiation sources for the subsequent skyshine calculation.

The design fuel for most storage casks is PWR spent fuel with a burnup of 35 GWD/MTU and a minumum cooling time of 5 years. The isotopes Cm-242 and Cm-244 characteristically produce all except a few percent of the spontaneous fission and (alpha,n) neutron source in spent PWR fuel over a ten-year decay time. The next largest contribution is usually from the (alpha,n) reaction of alphas from Pu-238, which is approximately 1-2% of the source. The (alpha,n) source component is a relatively small fraction of the total neutron source for the decay times of interest in cask storage (about 20 years). It is important to notice that the total neutron source at discharge increases almost exponentially with burnup while the gamma source changes linearly with burnup. Also, the long half-life of the Cm-244 causes the neutron source to decrease slowly with cooling time. Therefore it follows that the overall importance of the neutron source relative to the gamma source increases with burnup and cooling time. This is an important point in light of the trend towards fuel cycles that deplete PWR fuel up to 60 GWD/MTU. Table 1 compares calculated neutron source strengths and energy spectra for 35 and 60 GWD/MTU fuel cooled for 5 years. These sources were calculated with the SAS2H/ORIGEN-S sequence of the SCALE system.[3]

Based on the specified spent fuel source, a cask shielding analysis should be performed to obtain the magnitude and the energy, space, and angular distribution of the surface flux from a single cask. The characteristics of the flux leakage are dependent on both the cask design and the spent fuel contents. Storage casks have been and are being designed with a variety of shield materials - lead, cast iron, stainless steel, reinforced concrete, polyethylene, and a host of other solid light-element neutron shielding materials.

Although the surface dose for most casks are designed to have approximately the same magnitude, the selection of cask materials determines the relative neutron component of the total surface flux. Therefore, a general characterization of the neutron flux leakage from casks is not possible. However, using a typical cask iron storage/transport cask[4] and the proposed concrete cask for the MRS facility,[2] analyses have been performed to compare characteristics of the neutron flux leakage. Both the 35 GWD/MTU and the 60 GWD/MTU sources of Table 1 were used in the analyses to demonstrate the effect of burnup. The characteristics of the neutron flux for these cask and burnup specifications will be presented.

After characterizing the surface flux from a single cask, an appropriate radiation source model for the neutron skyshine analysis needs to be developed. Potential options include

1) Development of an equivalent point source for a single cask. Neutron skyshine analysis would then be performed as for a single cask and results modified to consider the multiple cask field.
2) Consideration of multiple cask interaction by performing a multidimensional radiation transport analysis that provides flux leakage from the surface bounding the array of casks.

The first option is the simplest and most expendient, however the potential beneficial effects of using option 2 needs to be further studied. Depending on the size of the cask array at the storage facility, the cask separation distance, and the distance from the site boundary, the source modeling of option 2 may provide a substantial reduction in the neutron dose at the site boundary.

REFERENCES

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3. "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," Vols. 1-3, NUREG/CR-0200, U. S. Nuclear Regulatory Commission (originally issued July 1980, reissued January 1982, Revision 1 issued July 1982, Revision 2 issued June 1983, Revision 3 issued December 1984).

4. "The Castor-V/21 PWR Spent-Fuel Storage Cask: Testing and Analyses," EPRI NP-4887, Electric Power Research Institute, 1986.

Table 1. TYPICAL NEUTRON SOURCE STRENGTH AND ENERGY SPECTRA FOR FIVE-YEAR-COOLED PWR FUEL

	NEUTRONS/SEC/ASSEMBLY	
ENERGY RANGE, MEV	35 GWD/MTU [a]	60 GWD/MTU [b]
1.42E+01 - 1.73E+01	0.0	0.0
1.22E+01 - 1.42E+01	1.738E+04	8.213E+04
1.00E+01 - 1.22E+01	1.517E+05	7.171E+05
8.61E+00 - 1.00E+01	2.342E+05	1.107E+06
7.41E+00 - 8.61E+00	7.435E+05	3.514E+06
6.07E+00 - 7.41E+00	2.248E+06	1.062E+07
4.97E+00 - 6.07E+00	4.486E+06	2.120E+07
3.68E+00 - 4.97E+00	1.172E+07	5.520E+07
3.12E+00 - 3.68E+00	9.239E+06	4.309E+07
2.73E+00 - 3.12E+00	7.842E+06	3.627E+07
2.47E+00 - 2.73E+00	6.727E+06	3.119E+07
2.37E+00 - 2.47E+00	2.892E+06	1.343E+07
2.35E+00 - 2.37E+00	5.785E+05	2.689E+06
2.23E+00 - 2.35E+00	3.374E+06	1.568E+07
1.92E+00 - 2.23E+00	1.006E+07	4.688E+07
1.65E+00 - 1.92E+00	9.790E+06	4.580E+07
1.35E+00 - 1.65E+00	1.270E+07	5.964E+07
1.00E+00 - 1.35E+00	1.694E+07	7.978E+07
8.21E-01 - 1.00E+00	8.564E+06	4.040E+07
7.43E-01 - 8.21E-01	4.092E+06	1.932E+07
6.08E-01 - 7.43E-01	7.307E+06	3.451E+07
4.98E-01 - 6.08E-01	5.855E+06	2.766E+07
3.69E-01 - 4.98E-01	6.782E+06	3.204E+07
2.97E-01 - 3.69E-01	3.522E+06	1.664E+07
1.83E-01 - 2.97E-01	2.111E+03	6.744E+03
1.11E-01 - 1.83E-01	1.334E+03	4.262E+03
0.0 1.11E-01	2.053E+02	6.558E+02
TOTAL	1.359E+08	6.375E+08

[a] Depletion analysis assumes no reactor downtime, 3.2% enriched fuel.

[b] Depletion analysis assumes 20% reactor downtime, 5.0% enriched fuel.