Department of Energy
Washington, DC 20585

QA: N/A

November 5, 2009

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YUCCA MOUNTAIN - REQUEST FOR ADDITIONAL INFORMATION - VOLUME 2, CHAPTER 2.1.1.5, SET 2 (DEPARTMENT OF ENERGY'S SAFETY ANALYSIS REPORT SECTION 1.8) – Consequence Analyses

Reference: Ltr, Jacobs to Williams, dtd 07/22/09, "Yucca Mountain – Request For Additional Information – Volume 2, Chapter 2.1.1.5, Sets 1 & 2 (Department of Energy's Safety Analysis Report Section 1.8)"

The purpose of this letter is to transmit the U.S. Department of Energy's (DOE) response to a Request for Additional Information (RAI) identified in the above-referenced letter. The response for RAI number 6 from Set 2 is provided as Enclosure 1. RAI 3 was submitted on August 21, 2009; RAI 2 was submitted on September 2, 2009; RAIs 1 and 4 were submitted on October 8, 2009, and RAI 5 was submitted on October 15, 2009. This transmittal completes DOE's responses for Set 2.

Two DOE references, not previously submitted to the U.S. Nuclear Regulatory Commission (NRC), are provided on optical storage media (OSM) as Enclosure 2. Additionally, Enclosure 3, on OSM, contains electronic attachments associated with the reference. The electronic attachments are data files provided in their native file format, consistent with Sections 2.2 and 2.17 of the NRC guidance on electronic submissions. They are required by NRC staff in their native format to evaluate DOE's responses. The electronic attachments are not intended to be placed on or accessed through ADAMS, and will be made available to the public upon request.

There are no commitments in the enclosed RAI response. If you have any questions regarding this letter, please contact me at (202) 586-9620, or by email to jeff.williams@rw.doe.gov.

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Office of Technical Management

OTM: SEG-0086

NMSS25

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Enclosures (3):
1. Response to RAI Volume 2, Chapter 2.1.1.5, Set 2, Number 6
2. Optical Storage Media containing two references
   a) BSC (Bechtel SAIC Company) 2003. *BWR Source Term Generation and Evaluation.* 000-00C-MGR0-00200-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20030723.0001.
   b) BSC 2004. *PWR Source Term Generation and Evaluation.* 000-00C-MGR0-00100-000-00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040524.0007.
3. Optical Storage Media containing electronic data attachment to reference BSC 2003.

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EIE Document Components:

[164364]_000-00C-MGR0-00200-000-00A_wECN1.pdf	19,710 KB
[169061]_000-00C-MGR0-00100-000-00B_wCNs.pdf	5,137 KB

RAI Volume 2, Chapter 2.1.1.5, Second Set, Number 6:

Justify the use of SCALE/ORIGEN-S (SAR Section 1.10.3.2) in source term evaluations for high-burnup spent nuclear fuel (SNF) (e.g., maximum PWR CSNF assembly with 80 GWd/MTU, or maximum BWR CSNF assembly with 75 GWd/MTU (SAR Table 1.8-2)). The SCALE/ORIGEN-S cross-section library validation is currently limited to 47 GWd/MTU (NRC, 2003).

1. RESPONSE

The source terms evaluated for pressurized water reactor (PWR) spent nuclear fuel (SNF) at 80 GWd/MTU and for boiling water reactor (BWR) SNF at 75 GWd/MTU are used for shielding design and event sequence dose consequence analyses. They are intended to bound the source terms for existing and projected PWR and BWR SNF inventories, which have an average burnup less than 40 GWd/MTU and a maximum burnup less than 70 GWd/MTU.

The ORIGEN-S module of Versions 4.3 (ORNL 1997) and 4.4A (NRC 1999) of the SCALE code system is used to evaluate these source terms (gamma and neutron spectra as well as radionuclide concentrations) that are the basis for many of the shielding and dose analyses discussed in SAR Sections 1.8 and 1.10. The evaluations are performed using burnup-dependent macroscopic cross section libraries generated by the SAS2H module that reflect the changing composition of the fuel matrix with burnup. The validity of the underlying SCALE 44-group ENDF/B-V microscopic cross section library is not limited to a maximum burnup value because the microscopic cross section is a fundamental property of an isotope that is not burnup dependent.

Analyses of commercial SNF with burnup values up to 70 GWd/MTU show the same level of agreement between SCALE/ORIGEN-S calculations and measurements of isotopic concentrations as do previous comparisons accepted by the NRC for burnup values up to 47 GWd/MTU. For the radionuclides of interest in shielding design and dose consequence analyses, there is no divergence of the differences between calculations and measurements with increasing burnup beyond 47 GWd/MTU (Figures 1 and 2) and no indication that divergence should be expected at burnup values as high as 80 GWd/MTU. The concentrations of the most important radionuclides used for shielding design and dose consequence analyses are conservative with respect to fuel modeling and burnup to an extent greater than or equal to the observed differences between calculations and measurements in the burnup range greater than 47 GWd/MTU. In addition, the process and criteria for acceptance of commercial SNF, as discussed in Section 1.4 of this response, will ensure that each shipment accepted will lie within the safety bases established by the shielding and dose consequence analyses.

Therefore, the use of SCALE/ORIGEN-S to generate the source terms for the preclosure shielding and dose analyses is justified.

1.1 COMPARISONS OF SCALE/ORIGEN-S ESTIMATES OF SPENT FUEL ISOTOPIC CONCENTRATIONS TO MEASUREMENTS

Uncertainties in SCALE/ORIGEN-S calculated radionuclide concentrations are dependent upon (NRC 2000):

- uncertainties in the system parameters being modeled (e.g., initial fuel compositions, assembly power history, effective fuel temperature, water density or void fraction, burnup at discharge, etc.);
- uncertainties due to modeling approximations (e.g., simplification of geometry and power history, etc.); and
- uncertainties attributed to the calculational methods and data used by the codes.

The totality of these uncertainties can be captured by comparisons of SCALE/ORIGEN-S estimates of spent fuel isotopic concentrations to measured values. Until recently, the highest burnup at which such comparisons had been performed was 47 GWd/MTU (NRC 2003). Comparisons of calculated (SAS2H/ORIGEN-S) and measured isotopic concentrations were reported for 4.11 wt % enriched PWR fuel samples with burnups from about 14 to 47 GWd/MTU and indicate good agreement with the code predictions (NRC 2003, Section 4) described in *Validation of the SCALE System for PWR Spent fuel Isotopic Composition Analysis* (Hermann et al. 1995) and *An Extension of the Validation of SCALE (SAS2H) Isotopic Predictions for PWR Spent Fuel* (DeHart and Hermann 1996).

Recent comparisons between SCALE/ORIGEN-S calculations and measurements have been reported for 4.3 wt % enriched commercial SNF with burnup values from about 46 to 70 GWd/MTU (Ilas and Gauld 2008, Section 4) and indicate that agreement between calculation and measurement is generally good and consistent with results obtained from previous evaluations of experimental data obtained from different programs for lower burnup fuel. The burnup-dependent cross sections for this comparison were generated using the TRITON control module of SCALE V5.1 (ORNL 2006) and again were based upon the 44-group ENDF/B-V SCALE cross section library. Although the unit cell models of XSDRNPM employed by SAS2H are replaced by the two-dimensional model of the NEWT transport module when TRITON is used (ORNL 2006, Section T1.2.3), the resonance cross section processing was still performed by the BONAMI and NITAWL modules and the changes in fuel composition were performed by ORIGEN-S as when SAS2H was used. The differences between the TRITON/ORIGEN-S calculations and measurements for the 46 GWd/MTU fuel (Ilas and Gauld 2008, Section 5) are comparable to the differences between the SAS2H/ORIGEN-S calculations and measurements for the 47 GWd/MTU fuel (NRC 2003, Tables 23 and 24). Therefore, use of burnup-dependent cross section libraries generated by SAS2H yields calculated results with accuracy similar to results based on libraries generated by TRITON.

1.2 SUITABILITY OF SOURCE TERMS BY APPLICATION

Because a source term depends on a relatively small number of dose-significant radionuclides, the suitability of each source term calculated by ORIGEN-S can be assessed on the basis of its application to shielding, airborne release, and dose consequence analyses.

1.2.1 Source Terms for Shielding Applications

The maximum gamma and neutron sources provided in SAR Table 1.10-18 are for PWR SNF with 5.0 wt % initial enrichment, 80 GWd/MTU burnup, and 5 years cooling time and are used for geologic repository operations area shielding analyses in which commercial SNF is the bounding source.

Examination of the intensities of the gamma source spectra by energy group in the ORIGEN-S run for the maximum fuel in the PWR source term calculation (BSC 2004, Appendix X) shows that the principal contributors to the gamma sources are ^{60}Co (1.17 and 1.33 MeV gammas), the ^{137m}Ba daughter (0.67 MeV gamma) in equilibrium with its ^{137}Cs parent, and ^{134}Cs (0.6 to 0.8 MeV gamma rays). These principal contributors are consistent with expectations for radionuclides important to shielding applications (Gauld and Ryman 2001, Figures 22, 24, and 26). Examination of "Analysis of Isotopic Data from the MALIBU Program" (Ilas and Gauld 2008, Figure 4) reveals that the differences between calculated and measured concentrations of ^{137}Cs (less than 0.5%) are among the smallest differences between calculated and measured concentrations observed for high-burnup fuel. Although the differences between calculated and measured concentrations of ^{134}Cs (11.5% to 15.5%) are greater than those for ^{137}Cs, the differences are actually decreasing with increasing burnup, and are generally consistent with the differences observed at lower burnup values (Hermann et al. 1995; DeHart and Hermann 1996). The ^{60}Co is produced from neutron absorption by ^{59}Co in cladding, corrosion products, and structural materials whose compositions change very little during irradiation and whose reaction rate depends only upon the neutron flux spectrum in those materials. The change in fuel composition with burnup has only an insignificant secondary effect on the neutron spectrum in the cladding and structural materials where ^{60}Co is produced. Furthermore, the nuclear decay properties of ^{137m}Ba, ^{134}Cs, and ^{60}Co are unaffected by burnup and are well known from extensive studies because of their recognized importance in SNF.

Examination of the neutron sources in the ORIGEN-S run for the maximum fuel in *PWR Source Term Generation and Evaluation* (BSC 2004) shows that the principal contributor to the neutron source is ^{244}Cm, consistent with expectations (Gauld and Ryman 2001, Figures 22, 24, and 26). The differences between calculated and measured concentrations for ^{244}Cm at burnup values up to 70 GWd/MTU are within 9% (Ilas and Gauld 2008, Section 5) and are comparable to the differences at burnup values less than 47 GWd/MTU. There is no divergence of the differences between calculations and measurements with increasing burnup and no indication that divergence should be expected at burnup values as high as 80 GWd/MTU.

Since the differences between calculated and measured concentrations of the principal radionuclides contributing to gamma and neutron spectra for high-burnup fuel are consistent with the differences accepted for low- and medium-burnup fuel (NRC 2003), those differences are

also acceptable for high-burnup fuel. Therefore, the gamma and neutron spectra in the ORIGEN-S runs for the maximum burnup fuels are justified for use as shielding source terms.

1.2.2 Concentrations of Radionuclides Released During Normal Operations

For potential releases associated with normal operations, the source terms generated for representative PWR and BWR SNF given in SAR Table 1.8-3 are used to calculate airborne release doses from surface facilities (SAR Section 1.8.2.2.1). The characteristics of the representative SNF are 50 GWd/MTU burnup and 10 years decay time for both PWR and BWR SNF with initial enrichments of 4.2 wt % for the PWR SNF and 4.0 wt % for the BWR SNF (BSC 2007, Table 12). The burnup values for the representative SNF are conservative given that the average SNF burnup values in the existing and projected PWR and BWR SNF waste streams described in SAR Table 1.5.1-5 are 36.2 GWd/MTU and 28.6 GWd/MTU, respectively.

The 50 GWd/MTU burnup of the representative SNF is only 6% greater than the 47 GWd/MTU burnup for which SCALE/ORIGEN-S calculations have previously been accepted (NRC 2003). Figures 2 through 6 of "Analysis of Isotopic Data from the MALIBU Program" (Ilas and Gauld 2008) show no divergence in the differences between calculations and measurements with increasing burnup, consistent with previous studies (NRC 2003, Section 3). As noted in Section 1.1, the magnitude of observed differences for burnup values up to 70 GWd/MTU is well within the acceptable range. Therefore, the use of SCALE/ORIGEN-S to calculate source terms for the 50 GWd/MTU representative SNF is justified for application to releases during normal operations.

1.2.3 Concentrations of Radionuclides Released During Potential Event Sequences

As stated in SAR Section 1.8.1.3.1, the maximum radionuclide inventories in SAR Table 1.5.1-12 (80 GWd/MTU for PWR and 75 GWd/MTU for BWR) are used as input to the preclosure consequence analyses for Category 2 event sequences. There are no Category 1 event sequences. The principal contributors to the total effective dose equivalent dose for Category 2 event sequences are ^{241}Am, ^{244}Cm, ^{134}Cs, ^{137}Cs, ^{3}H, ^{85}Kr, and ^{238}Pu from SCALE/ORIGEN-S calculations (BSC 2008, Table III-1) and ^{60}Co from crud, for which a bounding concentration is based on measured activity (SAR Table 1.5.1-6). Although there appear to be few or no comparisons of SCALE/ORIGEN-S calculated and measured values for ^{3}H and ^{85}Kr at any level of burnup, their predicted concentrations are considered acceptable because both nuclides have very small absorption cross sections and their calculated concentrations depend primarily on fission yields and half-lives, both of which are well known. The use of the fission product ^{134}Cs, ^{137}Cs, and the actinide ^{244}Cm radionuclide concentrations at high burnup values is justified in Section 1.1.1. For ^{241}Am, the magnitude of the differences between calculated and measured concentrations (less than 13% for burnup values from 46 to 70.8 GWd/MTU) in "Analysis of Isotopic Data from the MALIBU Program" (Ilas and Gauld 2008, Figure 3) is consistent with or less than the differences for ^{241}Am in previous comparisons at burnup values up to 47 GWd/MTU (NRC 2003; Hermann et al. 1995). Similarly, the differences between calculated and measured concentrations of ^{238}Pu are less than 10% and decreasing with increasing burnup from 46 to 70.8 GWd/MTU (Ilas and Gauld 2008, Figure 2). These differences exhibit the same level of agreement as in earlier comparisons for lower burnup SNF (NRC 2003; DeHart and

Hermann 1996). Similar to the predictions for the radionuclides ^{134}Cs, ^{137}Cs, and ^{244}Cm, the SCALE/ORIGEN-S predictions of radionuclide concentrations for ^{241}Am and ^{238}Pu at burnup values up to 70 GWd/MTU are acceptable based on comparisons with values acceptable at burnup values less than 47 GWd/MTU. Therefore, the use of SCALE/ORIGEN-S to predict the concentrations of radionuclides important to dose consequence is justified for PWR and BWR SNF having the maximum burnup values.

1.2.4 Summary of Dose-Significant Radionuclides as a Function of Burnup

The differences between calculated and measured radionuclide concentrations as a function of burnup are shown in Figure 1 for dose-significant cesium fission products and in Figure 2 for dose-significant actinides. Comparisons of calculated and measured isotopic concentrations (Hermann et al. 1995; Dehart and Hermann 1996; NRC 2003; Ilas and Gauld 2008) show good agreement between predicted and measured values for dose-significant radionuclides over a wide range of burnup values from 11 to 70 GWd/MTU. There is no divergence of the differences between calculations and measurements with increasing burnup and no indication that divergence should be expected at burnup values as high as 80 GWd/MTU.

1.3 CONSERVATISM OF SOURCE TERMS AND THEIR RELATIONSHIP TO THE COMMERCIAL SPENT NUCLEAR FUEL WASTE STREAM

The PWR and BWR SNF source terms (BSC 2004; BSC 2003) were calculated for assemblies modeled with greater initial uranium content than expected. For the PWR source terms, the PWR assembly model has 475 kg uranium (BSC 2004, Section 5.2), which is 10% greater than the average initial uranium content of 431 kg (SAR Table 1.5.1-5). For the BWR source terms, the BWR assembly model has 200 kg uranium (BSC 2003, Section 5.2), which is 12% greater than the average initial uranium content of 179 kg (SAR Table 1.5.1-5). The compositions of the cladding and structural materials include the maximum amounts of cobalt impurities from the expected ranges for those materials and the flux scaling factors for the hardware regions of the PWR and BWR assemblies were taken to be 150% of the expected values (BSC 2004, Section 5.2; BSC 2003, Section 5.2), which produces conservative estimates of ^{60}Co concentrations.

The expected average and maximum burnup values for the existing and projected waste stream are 36.2 GWd/MTU and 69.5 GWd/MTU respectively for PWR SNF (SAR Table 1.5.1-5) and for BWR SNF, the expected average and maximum burnups are 28.6 GWd/MTU and 65.1 GWd/MTU (SAR Table 1.5.1-5). Thus, no commercial SNF expected to be received at the repository has characteristics approaching those used for the shielding design and event sequence dose consequence analyses, which use the SCALE/ORIGEN-S calculation of the maximum commercial SNF source terms.

Table 1 gives the increments in concentrations of the principal contributors to the calculated normal operations doses due to assuming a representative burnup of 50 GWd/MTU rather than the expected waste stream average burnup values for PWR and BWR SNF of 36.2 GWd/MTU and 28.6 GWd/MTU, respectively. Tables 2 and 3 give the incremental increases in concentrations of the principal contributors to the calculated shielding and event sequence doses

due to assuming a maximum burnup of 80 GWd/MTU for PWR SNF and 75 GWd/MTU for BWR SNF rather than the projected waste stream maximum values of 69.5 GWd/MTU for PWR SNF and 65.1 GWd/MTU for BWR SNF.

Table 1 shows that the concentrations of radionuclides important to normal operations doses overpredict those of the average waste stream from 31% to 68% due to the conservative representative burnup values. Table 2 illustrates that the concentrations of radionuclides important to shielding doses overpredict those of the average waste stream from 12% to 62% due to the selection of the conservative maximum burnup values. Table 3 demonstrates that the concentrations of radionuclides important to event sequence dose consequences overpredict those of the average waste stream by as much as 65% because of the use of conservative maximum burnup values. This conservatism in the dose-significant radionuclide concentrations is greater than any potential differences between calculated and measured values associated with using SCALE/ORIGEN-S radionuclide concentrations with increased burnup, with the exception of ^{241}Am. In that case, the combination of the conservatism in initial uranium mass with the conservatism of the ^{241}Am concentration is about 13% (1.10 × 1.03) for PWR SNF and 12% (1.12 × 1.004) for BWR SNF when the largest difference between calculation and measurement is less than 12% for burnup values greater than 47 GWd/MTU.

Thus, the concentrations of the most important radionuclides used for shielding design and dose consequence analyses are conservative with respect to fuel modeling and burnup to an extent greater than or equal to the observed differences between calculations and measurements in the burnup range greater than 47 GWd/MTU.

1.4 ADMINISTRATIVE CONTROLS ON WASTE ACCEPTANCE CRITERIA

As noted in Section 1, the dose consequence analyses and the shielding analyses for which commercial SNF is the bounding source are based on PWR SNF having a burnup of 80 GWd/MTU and BWR SNF having a burnup of 75 GWd/MTU. The discussions in Sections 1.1 through 1.3 demonstrate that use of SCALE/ORIGEN-S to calculate these source terms is justified for the expected waste stream. Since the actual waste stream may include commercial SNF that varies from the expected waste stream, the acceptability of each commercial SNF shipment to the repository will be evaluated with respect to the safety bases established by the shielding and dose consequence analyses. SAR Table 5.10-3 states that the probable administrative controls identified in the Technical Requirements Manual include "Acceptance criteria and designation of waste forms and waste packages approved for emplacement with the waste form and waste package qualification program." These acceptance criteria will include the appropriate provisions to ensure that waste accepted in the geologic repository operations area is within the safety bases established by the dose consequence analyses and shielding design. These safety bases include important SNF parameters such as initial fuel composition and assembly power history used to evaluate the source terms as well as the doses and dose rates based on the maximum PWR and BWR source terms. If it is necessary to evaluate commercial SNF with burnup values beyond those of the expected waste stream, additional comparisons of SCALE/ORIGEN-S calculations and measurements will be acquired as needed. Commercial SNF with such characteristics will still be acceptable if (1) measurements of dose-significant radionuclides in high-burnup SNF continue to exhibit the same level of agreement with

calculations as do existing measurements available at lower burnup values, or (2) the concentrations of dose-significant radionuclides in this SNF are less than the concentrations that form the bases for the dose consequence analyses and shielding design. If needed, additional analyses will be performed to demonstrate compliance with the performance objectives specified in 10 CFR 63.111 or changes to the Technical Requirements Manual will be evaluated using the criteria of 10 CFR 63.44 and implemented accordingly.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

BSC (Bechtel SAIC Company) 2003. *BWR Source Term Generation and Evaluation.* 000-00C-MGR0-00200-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20030723.0001.

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NRC 2003. *Isotopic Analysis of High-Burnup PWR Spent Fuel Samples from the Takahama-3 Reactor.* NUREG/CR-6798. Washington, DC: U.S. Nuclear Regulatory Commission. ML030570346.

ORNL (Oak Ridge National Laboratory) 1997. *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation.* NUREG/CR-0200, Rev. 5. Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 235920.

ORNL 2006. *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation.* ORNL/TM-2005/39, Version 5.1, Vols. I, II, and III. Oak Ridge, Tennessee: Oak Ridge National Laboratory. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-732.

Table 1. Increases in Concentrations of Principal Contributors to Normal Operation Doses due to Use of Representative Burnup Fuel

Nuclide	PWR SNF Concentration Increment	BWR SNF Concentration Increment
^{3}H	35%	59%
^{85}Kr	31%	52%
^{129}I	39%	68%
^{137}Cs	37%	66%

NOTE: These increments represent the relative increased concentrations in the SNF due to assuming a representative burnup of 50 GWd/MTU rather than expected waste stream average burnup values for PWR and BWR SNF of 36.2 GWd/MTU and 28.6 GWd/MTU, respectively.

Source: Calculated from radionuclide concentrations in BSC 2004, Attachment X; and BSC 2003, Attachment VII.

Table 2. Increases in Concentrations of Principal Contributors to Shielding Doses due to Use of Maximum Burnup Fuel

Nuclide	PWR SNF Concentration Increment
^{60}Co	12%
^{134}Cs	19%
^{137}Cs	12%
^{244}Cm	62%

NOTE: These increments represent the relative increased concentrations in the PWR SNF due to assuming a maximum burnup of 80 GWd/MTU rather than the projected waste stream maximum burnup of 69.5 GWd/MTU.

Source: Calculated from radionuclide concentrations in BSC 2004, Attachment X.

Table 3. Increases in Concentrations of Principal Contributors to Event Sequence Dose Consequences due to Use of Maximum Burnup Fuel

Nuclide	PWR SNF Concentration Increment	BWR SNF Concentration Increment
^{3}H	13%	11%
^{85}Kr	7%	4%
^{134}Cs	21%	20%
^{137}Cs	13%	12%
^{238}Pu	26%	22%
^{241}Am	3%	0.4%
^{244}Cm	65%	64%

NOTE: These increments represent the relative increased concentrations in the SNF due to assuming a maximum burnup of 80 GWd/MTU for PWR SNF and 75 GWd/MTU for BWR SNF rather than the projected waste stream maximum burnup values of 69.5 GWd/MTU for PWR SNF and 65.1 GWd/MTU for BWR SNF.

Source: Calculated from radionuclide concentrations in BSC 2004, Attachment X; and BSC 2003, Attachment VII.

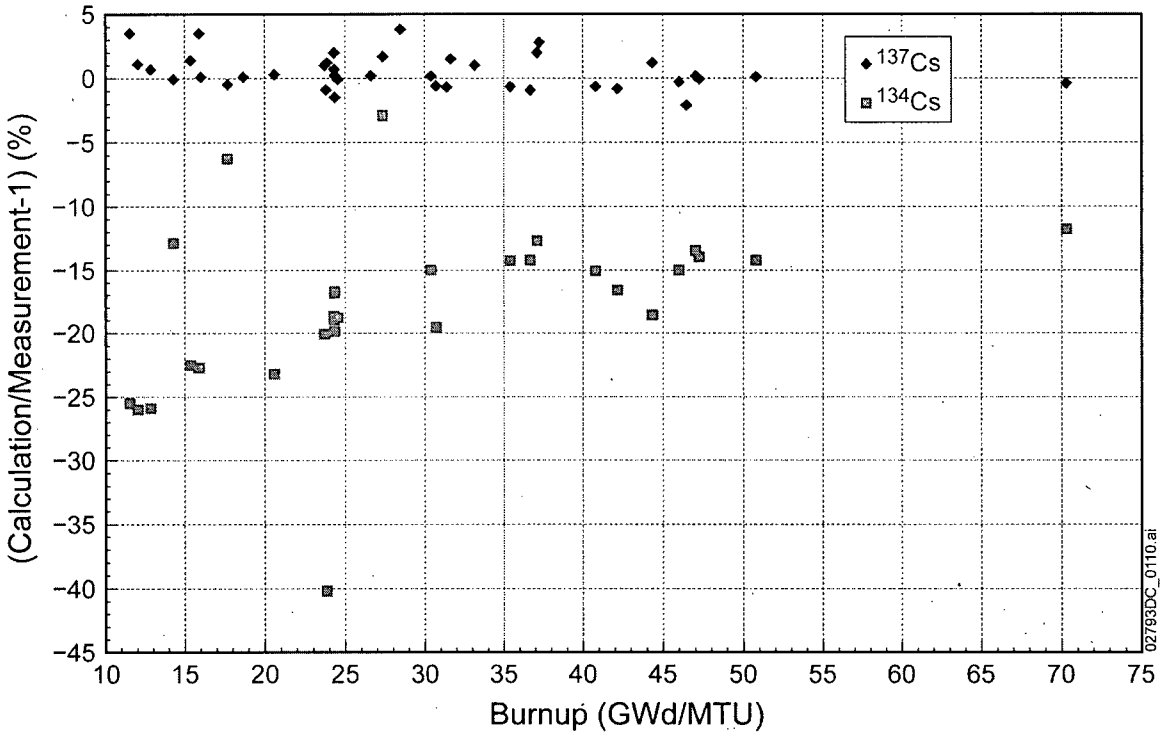

Figure 1. Comparison of Calculated and Measured Radionuclide Concentrations for Dose-Significant Cesium Fission Products

Source: Hermann et al. 1995; DeHart and Hermann 1996; NRC 2003; Ilas and Gauld 2008.

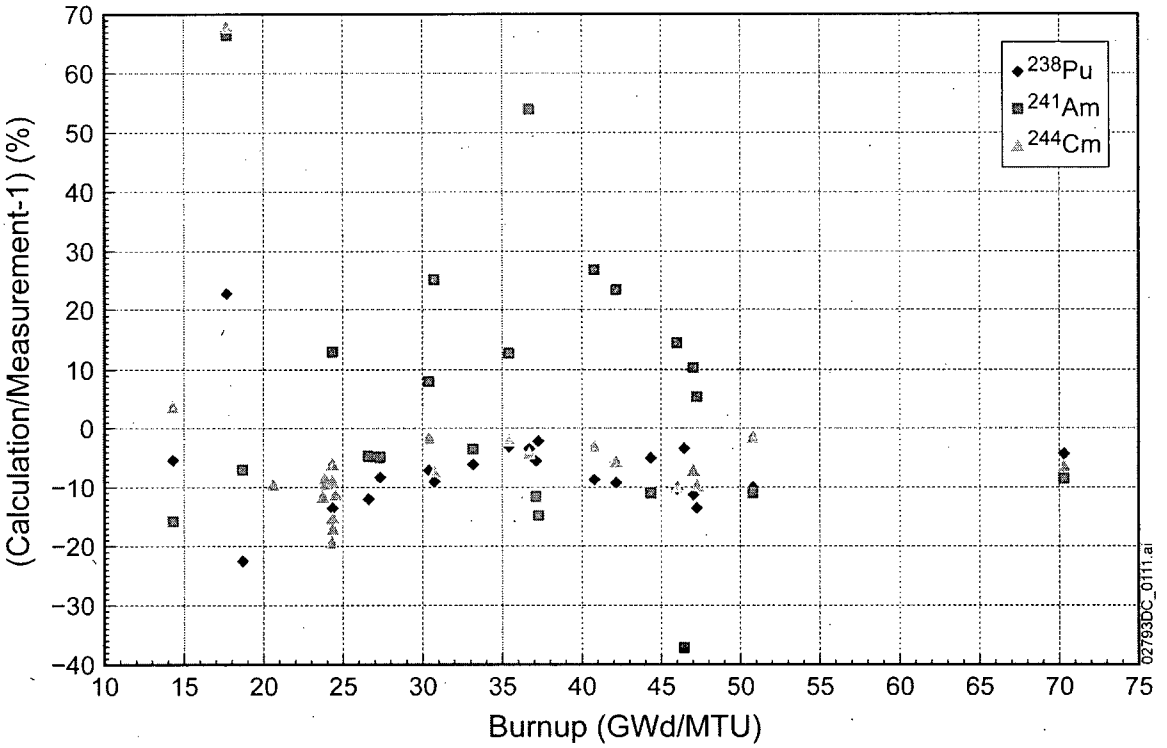

Figure 2. Comparison of Calculated and Measured Radionuclide Concentrations for Dose-Significant Actinides

Source: Hermann et al. 1995; DeHart and Hermann 1996; NRC 2003; Ilas and Gauld 2008.