1 of 2

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Updated Neutron Spectrum Characterization of SNL Baseline Reactor Environments

Vol. 1: Characterization

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The authors thank the staff of the SPR and ACRR reactor facilities and the Radiation Metrology Laboratory for making it possible to gather the data summarized in this memorandum.

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Executive Summary

The neutron spectrum characteristics of the primary reactor environments are defined for use by facility customers and to provide an audit trail in support of current quality assurance initiatives. The neutron and gamma environments in the four primary customer environments at SPR-III and ACRR facilities are characterized in detail. Enough detail is provided on other frequently-used environments to support the definition of the 3-MeV and 1-MeV(Si) fluence provided on the Radiation Metrology Laboratory dosimetry reports.

Details are provided to enable customers to compare new dosimetry results in the four primary environments with dosimetry reports from previous irradiations. SAND-II and MANIPULATE output listings used to derive the new spectrum-averaged integral parameters are provided in appendices that appear in volume 2 of this document.

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Updated Neutron Spectrum Characterization of SNL Baseline Reactor Environments

1. Introduction

Effective April 1, 1994, changes have been implemented at the Sandia Pulsed Reactor (SPR) and Annular Core Research Reactor (ACRR) facilities in the reporting of neutron environments. These changes affect the 1-MeV(Si) equivalent fluence and the fluence greater than 3-MeV normally reported by the facilities. This report details the reasons for the change in the dosimetry reporting methodology, and in accordance with the new quality assurance (QA) program at the Radiation Metrology Laboratory (RML), provides a traceable characterization of the baseline Sandia National Laboratories (SNL) neutron environments.

2. Why the Change?

Several events have taken place that motivate SNL to change the reported reactor neutron fluences and to modify the reporting methodology. This change is designed to ensure that the facility users have the benefit of the highest quality radiation diagnostics. The events that motivated this change include

1. The ASTM E-722 Standard [1,2] for reporting equivalent monoenergetic neutron fluences for radiation-hardness testing of electronics has changed significantly. A new 1-MeV(Si) displacement damage function, based on a recent Oak Ridge National Laboratory (ORNL) ^{28}Si cross section evaluation [3] and experimentally validated for a wide range of fission spectra [4,5], has been adopted in the standard. The new standard also defined a 1-MeV(GaAs) equivalent fluence. To distinguish between the 1-MeV(Si) equivalent fluence based on this new 1993 version of the E-722 standard and the previous 1985 version of E-722, we have adopted the nomenclature $\Phi^{1\text{MeV}}_{\text{Si}}(93)$ and $\Phi^{1\text{MeV}}_{\text{Si}}(85)$.
2. The ASTM E-722 Standard has adopted a new definition for the hardness parameter HP [6] which has zero rather than 10 keV for the low energy integral limit. This means that the 1-MeV(Si) fluence is no longer equal to the product of the spectral shape parameter, SP, the hardness parameter, HP, and the fluence greater than 3-MeV, $\Phi_{3\text{MeV}}$. Note that the hardness parameter also depends upon the silicon displacement damage function that is used. For clarity in distinguishing the definition of the hardness parameter, we will use the nomenclature $\text{HP}_{10\text{keV}}(85)$ for the previously-reported hardness parameters, and $\text{HP}_{\text{Si}}(93)$ for the new hardness parameter, with a lower integration energy limit of 0 keV that corresponds to the ASTM definition.

3. Several new libraries of dosimetry-quality cross sections have become available. These libraries include the ENDF/B-VI library [7], the JENDL-3 Dosimetry library [8], a 1993 version of the GLUCS library [9], and the IRDF-90 library [10]. The SNL RML has compared the cross section evaluations and has prepared a compendium of dosimetry cross sections [11] that will be used in all future spectrum determinations. This new library reflects significant changes in some dosimetry reaction cross sections, such as for the $^{47}\text{Ti}(n,p)^{47}\text{Sc}$ reaction.
4. The recently-approved version of the ASTM E 265 Standard on radioactivation of sulfur [12] recommends a $^{32}\text{S}(n,p)^{32}\text{P}$ cross section that is significantly different than that in the previously-accepted ENDF/B-V Dosimetry File. Since most sulfur dosimetry systems are calibrated with a NIST ^{252}Cf -irradiation transfer calibration, the $^{32}\text{S}(n,p)^{32}\text{P}$ activity changes significantly ($\sim 8\%$), but the ^{252}Cf -equivalent fluence is left unchanged.
5. As part of the SNL RML quality program, we have increased the number of sensors used in a fission spectrum determination. We typically now use from 22 to 30 sensors for new spectrum characterizations. New sensor activities have been measured for most of the baseline reactor environments requested by facility users, including the SPR-III central cavity, the ACRR central cavity, and the ACRR Pb-B bucket.

SNL was an active participant in the development of the new ASTM E-722 and E-265 standards. Through involvement with the ASTM E-10 Committee, the IEEE Nuclear Space Radiation Effects Conference (NSREC), and the DNA-sponsored Space Parts Working Group, SNL has coordinated the implementation of these changes with other reactor facilities and with the electronic testing community.

3. $^{32}\text{S}(n,p)^{32}\text{P}$ Dosimetry Reporting Methodology

The $^{32}\text{S}(n,p)^{32}\text{P}$ sensor reporting methodology has been changed to reflect the latest sulfur cross sections and to take into account the transfer calibration to the NIST ^{252}Cf standard source. Since the $^{32}\text{S}(n,p)^{32}\text{P}$ cross section has a large uncertainty, over 40% for some energies near the reaction threshold and $\sim 8\%$ for a fission spectrum-averaged cross section, the SNL RML dosimetry reports will no longer reference a $^{32}\text{S}(n,p)^{32}\text{P}$ activity. The dosimetry reports will instead reference a ^{252}Cf -equivalent fluence, $^{\text{tot}}\text{Cf}^{S32}$, since this is the NIST-traceable quantity.

Table 1 presents the best available numbers to convert from a $^{32}\text{S}(n,p)^{32}\text{P}$ ^{252}Cf -equivalent fluence to $\Phi_{3\text{MeV}}$, $\Phi_{10\text{keV}}$, and $\Phi^{1\text{MeV}}_{\text{Si}(93)}$ for each of the baseline SNL reactor environments. The SNL RML now directly reports the $^{\text{tot}}\text{Cf}^{S32}$ and the $\Phi^{1\text{MeV}}_{\text{Si}(93)}$ on each ^{32}S dosimetry report. If the dosimetry reported value for the ^{252}Cf -equivalent fluence is multiplied by the entry from column 4 in Table 1, the result is the ASTM E-722(93) 1-

MeV-equivalent silicon fluence. This value is explicitly recorded in the dosimetry report as a convenience to the users. A similar procedure using columns 2 and 3 from Table 1 will yield values for the 10-keV and 3-MeV fluence values.

Table 1. Environment-dependent Conversions with SNL Sulfur Sensors

Reactor Environment	Conversion From ^{252}Cf -equivalent Fluence (n/cm ²) To:		
	$\Phi_{3\text{MeV}}$	$\Phi_{10\text{keV}}$	$\Phi^{1\text{MeV}}_{\text{Si}(93)}$
SPR-III Central Cavity	0.23228	2.1612	1.7773
SPR-III 17" Leakage at $\odot$	0.23394	1.9440	1.6469
ACRR Central Cavity	0.23068	2.6094	1.7746
ACRR "New" Pb-B Bucket	0.22282	3.8678	2.4392
The number of digits shown do not represent the precision of the quantities. The number of reproducible digits in the quantities is 3. The additional digits are given to aid in reproducibility and to provide an auditable record of the origin of the numbers.			

4. $^{58}\text{Ni}(\text{n,p})^{58}\text{Co}$ Dosimetry Reporting Methodology

The $^{58}\text{Ni}(\text{n,p})^{58}\text{Co}$ activity reporting methodology has not been changed. The recommended cross section for converting an activity to a $\Phi_{3\text{MeV}}$ has changed from the ENDF/B-V Dosimetry File to the ENDF/B-VI library. Changes in the recommended spectrum characterizations have also affected this conversion.

Table 2 presents the best available numbers to convert from a $^{58}\text{Ni}(\text{n,p})^{58}\text{Co}$ activity, $A_{\text{m}}^{58\text{Ni}}$, in units of Bq/g to $\Phi_{3\text{MeV}}$, $\Phi_{10\text{keV}}$, and $\Phi^{1\text{MeV}}_{\text{Si}(93)}$ for each of the baseline SNL reactor environments. The SNL RML currently reports the $A_{\text{m}}^{58\text{Ni}}$, in units of Bq/g. Some applications, e.g. SAND-II, require activities to be input in units of Bq/atom. To convert the RML-reported activity from Bq/g of nickel to Bq/atom, multiply by 9.620397×10^{-23} . This conversion represents the atomic weight of isotopic ^{58}Ni , 57.93534635 gm/mole (a mass excess of $\Delta = -60225 \text{ keV} = 0.064653645 \text{ amu}$), divided by Avogadro's number, $6.0221367 \times 10^{23} \text{ atoms/mole}$.

In the near future, the RML ^{58}Ni dosimetry will be changed to directly report the $\Phi^{1\text{MeV}}_{\text{Si}(93)}$, and $\Phi_{3\text{MeV}}$. If the current RML-dosimetry reported value for the $^{58}\text{Ni}(\text{n,p})^{58}\text{Co}$ activity is multiplied by the entry from column 4 in Table 2, the result is the ASTM E-722(93) 1-MeV-equivalent silicon fluence. Section 5 gives an example of how

to read and convert the RML ^{58}Ni dosimetry reports into fluence measures such as $\Phi_{3\text{MeV}}$, and $\Phi^{1\text{MeV}}_{\text{Si}(93)}$.

Table 2. Environment-dependent Conversions with Nickel Sensors

Reactor Environment	Conversion From Activity (Bq/g) To:		
	$\Phi_{3\text{MeV}}$	$\Phi_{10\text{keV}}$	$\Phi^{1\text{MeV}}_{\text{Si}(93)}$
SPR-III Central Cavity	$1.6158 \times 10^{+9}$	$1.5035 \times 10^{+10}$	$1.2363 \times 10^{+10}$
SPR-III 17" Leakage at $\text{\AA}$	$1.6986 \times 10^{+9}$	$1.4115 \times 10^{+10}$	$1.1958 \times 10^{+10}$
ACRR Central Cavity	$1.7055 \times 10^{+9}$	$1.9292 \times 10^{+10}$	$1.3119 \times 10^{+10}$
ACRR "New" Pb-B Bucket	$1.5564 \times 10^{+9}$	$2.7017 \times 10^{+10}$	$1.7038 \times 10^{+10}$
The number of digits shown do not represent the precision of the quantities. The number of reproducible digits in the quantities is 3. The additional digits are given to aid in reproducibility and to provide an auditable record of the origin of the numbers.			

5. 1-MeV(Si) Equivalent Fluence Determination

All reactor operations should use a monitor sensor to establish the neutron fluence produced at the location of interest. This monitor foil is typically either nickel or sulfur. The RML dosimetry reports are designed to allow easy conversion of the foil response into a 1-MeV(Si) equivalent fluence, $\Phi^{1\text{MeV}}_{\text{Si}(93)}$.

If the irradiation environment corresponds to one of the SNL well-characterized positions sulfur was used as a monitor foil, and the facility designation was correctly indicated by the experimenter's dosimetry reference card, then the RML dosimetry report will include a correct $\Phi^{1\text{MeV}}_{\text{Si}(93)}$. If the irradiation environment was incorrectly specified on the dosimetry reference card, or if a nickel monitor foil was used, then the user must use the following procedure to obtain a 1-MeV(Si) equivalent fluence:

- 1) If a nickel foil was used, and the irradiation environment appears in the first column of Table 2, then multiply the reported Co-58 activity (in units of Bq/g) in column 6 of the RML dosimetry report by the Table 2 entry from column 4 and in the appropriate row.

- 2) If an SNL sulfur pellet was used, and the irradiation environment appears in the first column of Table 1, then multiply the reported ^{252}Cf -equivalent fluence in column 6 of the RML dosimetry report by the Table 1 entry from column 4 and in the appropriate row.
- 3) If the irradiation environment is not one of the four baseline environments addressed in Sections 3 or 4 of this document, but the environment has been characterized by SNL, contact the facility operator to obtain the proper conversion number. (The facility operators can use the information provided in Table 7 in Section 7 to provide conversion numbers for a wide number of SNL available environments.)
- 4) If the irradiation environment has not been characterized by SNL, then contact J. (Jake) G. Kelly or Patrick J. Griffin to have a characterization made or to get approximate conversion numbers for your application.

The uncertainty in the 1-MeV(Si) equivalent fluence is obtained by adding, in quadrature, the uncertainty in the determination of the integral parameter, i.e. $\Delta_1 \Phi^{1\text{MeV}}_{\text{Si}(93)}$, as determined from the spectrum characterization process, to the uncertainty in the NIST-traceable RML reported quantity, $A^{\text{Ni}58}_{\text{m}}$ or $^{\text{tot}}\text{Cf}^{532}$, designated as $\Delta_2 \Phi^{1\text{MeV}}_{\text{Si}(93)}$. This uncertainty in $A^{\text{Ni}58}_{\text{m}}$ and $^{\text{tot}}\text{Cf}^{532}$ is irradiation-dependent and is printed on the RML dosimetry reports. This uncertainty is typically between 3-6%. The uncertainty, $\Delta_1 \Phi^{1\text{MeV}}_{\text{Si}(93)}$, is determined by sensitivity studies during the spectrum unfold process and is given in Table 3 for the baseline SNL environments. (Section 7 and Table 6 provides $\Delta_1 \Phi^{1\text{MeV}}_{\text{Si}(93)}$ for a variety of other well characterized environments.)

Table 3. Spectrum Characterization Component of $\Phi^{1\text{MeV}}_{\text{Si}(93)}$ and $\Phi_{3\text{MeV}}$ Uncertainty

Reactor Environment	$\Delta_1 \Phi^{1\text{MeV}}_{\text{Si}(93)}$ (%)	$\Delta_1 \Phi_{3\text{MeV}}$ (%)
SPR-III Central Cavity	5	3
SPR-III 17" Leakage at $\odot$	10	7
ACRR Central Cavity	6	4
ACRR "New" Pb-B Bucket	7	4

Figure 1 shows a typical RML ^{32}S dosimetry report. In this example, an irradiation was performed on January 1, 1994 in the "SPR-III Central Cavity" environment by experimenter John Doe from organization AAA. The SPR-III shot number was 9181, and four sulfur pellets were read from this irradiation. The 1-MeV(Si) equivalent fluence,

$\Phi^{1\text{MeV}}_{\text{Si}}(93)$, for pellet number 1 is seen to be $3.288 \times 10^{12} \text{ n/cm}^2\text{-1MeV(Si)}$ with a 1σ uncertainty of $\pm 5.8\%$. The pellet 1 3-MeV fluence, $\Phi_{3\text{MeV}}$, is $4.297 \times 10^{11} \text{ n/cm}^2\text{-3MeV}$ with a 1σ uncertainty of $\pm 2.9\%$.

Figure 1. Sample RML ^{32}S Dosimetry Report

```

SANDIA NATIONAL LABORATORIES
Radiation Metrology Laboratory
Nuclear Facilities and Diagnostics Department
Albuquerque, NM 87185-1142
(505) 845-3250

Sulfur Dosimetry
1-JAN-94 11:17:25

Experimenter: J. Doe                      Organization: AAA
Facility: SPR III central cavity           Shot Number: 9181
Shot Date: 01-JAN-1994 16:00

File: UD:[TEST.DATA]SUL_dath_03622.dat
Last edited: 16-FEB-94 12:57:53

Counter 1 Background 1.50 +/- 0.22 cpm
>3 Mev/fac_tot: 0.1073 +/- 0.00%
1-MeV(Si)/fac_tot: 0.8211 +/- 5.00%
Cf-252/fac_tot: 2.1644 +/- 0.00%

S-32      60.      11000 B

  Count      Seq Samp  Count  Counts  Cf equ.  Unc  Fluence  Unc  Fluence  Unc
Date  Time      length  gross  fluence  %    > 3 MeV  %    1-MeV(Si)  %
      (min.)
090793 1118    1   7   12.19  11000  1.850E+12  2.9  4.297E+11  2.9  3.288E+12  5.8
090793 1130    2   8   12.39  11000  1.821E+12  2.9  4.230E+11  2.9  3.236E+12  5.8
090793 1143    3   9   11.90  11000  1.897E+12  2.9  4.406E+11  2.9  3.371E+12  5.8
090793 1156    4  13   12.43  11000  1.817E+12  2.9  4.220E+11  2.9  3.229E+12  5.8

Statistical uncertainties are reported at a 1-sigma confidence
level, and include uncertainties in the counting, counter
calibration and response ratios, combined in quadrature. Any
zero uncertainties indicate that the value is not available.

```

Suppose John Doe failed to identify the irradiation facility or did not use one of the 20 baseline environments (discussed in Section 7 of this report) when he submitted his dosimetry to the RML operations personnel. In this case the RML dosimetry report would look like Figure 2. If, upon receiving this dosimetry report, John Doe checked with operations and confirmed that the SPR-III Central Cavity was the correct spectrum characterizing his irradiation, he could use the information in Table 1 to derive a 1-MeV(Si) fluence. In Figure 2 the ^{252}Cf -equivalent fluence for the first sulfur pellet is $1.850 \times 10^{12} \text{ n/cm}^2$. To convert this number to a 1-MeV(Si) fluence for an irradiation in the SPR-III Central Cavity, one should use the row 1 column 4 entry from Table 1, or 1.7773. Thus $\Phi^{1\text{MeV}}_{\text{Si}}(93) = 3.288 \times 10^{12} \text{ n/cm}^2\text{-1MeV(Si)}$ at the location of the fluence monitor represented in Figure 2.

SANDIA NATIONAL LABORATORIES
Radiation Metrology Laboratory
Nuclear Facilities and Diagnostics Department
Albuquerque, NM 87185-1142
(505) 845-3250

Sulfur Dosimetry
1-JAN-94 11:17:25

Experimenter: J. Doe
Facility: Uncharacterized
Shot Date: 01-JAN-1994 16:00

Organization: AAA
Shot Number: 9181

File: UD:[TEST.DATA]SUL_datb_03622.dat
Last edited: 16-FEB-94 12:56:25

```

Counter 1 Background  1.50 +/-  0.22 cpm
>3 Mev/fac_tot:      0.0000 +/-  0.00%
1-MeV(Si)/fac_tot:    0.0000 +/-  0.00%
Cf-252/fac_tot:       1.0000 +/-  0.00%

```

Count		Seq Samp		Count	Counts	Cf equ.	Unc	Fluence	Unc	Fluence	Unc
Date	Time			length	gross	fluence	%	> 3 MeV	%	1-MeV(SI)	%
				(min.)							
090793	1118	1	7	12.19	11000	1.850E+12	2.9	0.000E+00	0.0	0.000E+00	0.0
090793	1130	2	8	12.39	11000	1.821E+12	2.9	0.000E+00	0.0	0.000E+00	0.0
090793	1143	3	9	11.90	11000	1.897E+12	2.9	0.000E+00	0.0	0.000E+00	0.0
090793	1156	4	13	12.43	11000	1.817E+12	2.9	0.000E+00	0.0	0.000E+00	0.0

Statistical uncertainties are reported at a 1-sigma confidence level, and include uncertainties in the counting, counter calibration and response ratios, combined in quadrature. Any zero uncertainties indicate that the value is not available.

Figure 3 shows a typical RML ^{58}Ni dosimetry report. In this example an irradiation was performed on January 1, 1994 by experimenter John Doe. The irradiation environment/facility is not identified on the RML dosimetry report because it does not impact the reported nickel activity measurement. The experimenter is responsible for knowing the environment used on the irradiation. In this case, the environment was the SPR-III Central Cavity. The RML-measured decay-corrected $^{58}\text{Ni}(\text{n,p})^{58}\text{Co}$ activity is $8.6888 \times 10^{+4} \text{ Bq/g}$.

The 1-MeV(Si) equivalent fluence, $\Phi^{1\text{MeV}}_{\text{Si}}(93)$, in the SPR-III Central Cavity is obtained by multiplying $A^{\text{Ni58}}_{\text{m}}$, in units of Bq/g, by the entry in Table 2, row 1, column 4, 1.2363×10^{10} . Thus, for the case in Figure 3, $\Phi^{1\text{MeV}}_{\text{Si}}(93) = 1.0742 \times 10^{15}$. The statistical counting component of the nickel activity uncertainty is seen from the Figure 3 RML dosimetry report to be $\pm 2.97\%$. The spectrum determination component of the $\Phi^{1\text{MeV}}_{\text{Si}}(93)$ uncertainty is seen from Table 3 to be $\Delta_1 \Phi^{1\text{MeV}}_{\text{Si}}(93) = \pm 5\%$. Thus, the total uncertainty in $\Phi^{1\text{MeV}}_{\text{Si}}(93)$ is given by $\Delta \Phi^{1\text{MeV}}_{\text{Si}}(93) = \pm 5.8\%$

Figure 3. Sample RML ⁵⁸Ni Dosimetry Report

```

*****
*                                     *
*          GAMMA SPECTRUM ANALYSIS          *
*                                     *
*****

CANBERRA APOGEE V2.0A

Canberra Industries, Inc.                      05-JAN-94   13:18:21

          ANALYSIS PARAMETERS

Spectrum file number      :24829.0
MCA unit number           :    1      ADC unit number :    2.0
Detector number           :    2      Geometry number  :    2

Search discrimination level #1 : 2.0          level #2 : 4.0
Search FROM channel : 100          TO channel : 8191
Identification energy tolerance : 1.5      order of background : parabola

Confidence levels      LLD : 1.645 (95.0%)      MDA : 1.645 (95.0%)

Analysis library        : SPFLIBRARY:SPFANL.002
Background subtract     : disabled      Random sum corr : 0.0000

Sample description      : NI58-4 3G-9299 J. Doe   Sample no. :24829
Analyzed by            : APOGEE2 24829 D02G02

Sample size             : 1.464300E-01 GM
Conversion factor       : 0.674
Standard size           : 1.000000E+00 EA

Sample taken on         : 01-JAN-94   at 14:07:00
Collect started on      : 04-JAN-94   at 08:11:57
Decay time              : 5405.0 minutes

live time               : 590.0 seconds
real time               : 594.0 seconds
dead time               : 0.67 %

Energy calibration used done on 1 / 23 / 1992
Efficiency calibration used done on 1 / 23 / 1992

*** P E A K   F I T   R E P O R T ***      05-JAN-94   13:18:21

peak  nuclide(s) centroid energy FWHM net area error gammas error
no.      channel   channel   keV   keV  counts  %   per second %
1       TL-208 NA-22  178.28  44.61  1.29    275.5  7.99    69.5 31.30
2       ANN-RD  2044.16  511.14  2.79   7058.5  1.34   1855.5  2.60
3       I-132  CO-58  3242.79  810.93  1.57   20012.5  0.71   8214.3  1.80
4       I-132  CO-58  3455.37  864.10  1.93    112.1 10.87    48.9 10.97

*** R A D I O N U C L I D E   R E P O R T ***      05-JAN-94   13:18:21

Sample description      :NI58-4 3G-9299 J. Doe
Analyzed by            :APOGEE2 24829 D02G02

number nuclide conf.value ----- Activity ( Bq/GM ) -----
                        measured      decay corrected
1 CO-58 0.9967 8.3753E+04 +- 2.97% 8.6888E+04 +- 2.97%
                        -----
                        total 8.3753E+04 total 8.6888E+04

These peaks were not identified

number channel energy peak intensity (gps)
1 178.3 44.61 6.9514E+01 +- 51.64%
2 2044.2 511.14 1.8555E+03 +- 4.29%
4 3455.4 864.10 4.8885E+01 +- 18.10%

Errors quoted at 1.650 sigma ( 90.1% )

```

The 3-MeV equivalent fluence, $\Phi_{3\text{MeV}}$, in the SPR-III Central Cavity is obtained by multiplying $A_{\text{m}}^{58\text{Ni}}$, in units of Bq/g, by the entry in Table 2, row 1, column 2, 1.6158×10^9 . Thus, for the case in Figure 3, $\Phi_{3\text{MeV}} = 1.4039 \times 10^{14}$. The statistical counting component of the $\Phi_{3\text{MeV}}$ uncertainty is the same as the uncertainty in the ^{58}Co activity measurement, which is $\pm 2.97\%$ according to the Figure 3 RML dosimetry report. The spectrum determination component of the $\Phi_{3\text{MeV}}$ uncertainty is given in Table 3 column 3 as 3%. This number is fairly small due to the very good multiple sensor coverage in the determination of the high energy part of the spectrum. Thus, the total uncertainty in $\Phi_{3\text{MeV}}$ is given by the activity statistical counting uncertainty of $\pm 4.22\%$.

6. Characterization of Baseline Environments

Table 4 shows the date and facility shot number of the most recent spectrum characterizations for the primary neutron irradiation environments at SNL. The SPR-III 80 inch and 120 inch leakage spectra have been added to the baseline environments described in this section. These environments were added to aid the user in interpolating dosimetry conversion numbers for positions further from the reactor than 17 inches. SPR-III users often are forced to choose an irradiation position based on experiment size and neutron uniformity requirements that does not correspond to the well-characterized central cavity or 17 inch leakage location.

The dosimetry conversion numbers given in the previous section are all based on SAND-II spectrum unfolding of activities from these reactor shots. The detailed SAND-II outputs are provided in Appendix A to ensure a clean audit trail for this work. Figure 4 shows some of the spectra that resulted from the SAND-II characterization.

Table 5 provides a detailed characterization of these environments. The entries in the column headings are explained in detail in the Nomenclature section of this report.

The quantities in Table 5 were chosen to cover all conversions that may be needed by facility users or may be needed to provide comparison with any future updated spectral characterization or new dosimetry cross section library. The entries are not all independent. Often an entry can be expressed as the product of two other entries. The redundant entries are intended to aid the users who want a fast lookup of a quantity and do not wish to study the mathematical definitions of the quantities. Entries are provided on the neutron monitor foil response and on the response of ionizing dose sensors.

Table 4. Description of Irradiation Conditions for Environments

Reactor Environment	Spectrum Name	Shot Number	Shot Date
SPR-III Central Cavity	SPR3CC	8827	12/1/92
SPR-III 17" Leakage at $\odot$	SCR4	6338	10/5/88
SPR-III 80" Leakage	SPR_80A3	7700	10/26/90
SPR-III 120" Leakage	SPR_120	7282	7/14/92
ACRR Central Cavity	ACF9	5407	7/14/92
ACRR "New" Pb-B Bucket	TPB13	4681	6/6/90

Figure 4. Spectral Characterization of the Baseline SNL Environments

Since mixed-field dosimetry is very difficult and standards are only beginning to be developed Table 5 provides information on the response of the RML TLD-400, radiochromic film, alanine, and teflon dose sensors. Work with teflon dose sensors is just now starting and entries are not yet mature enough for public distribution. The appropriate teflon table column headings are left in the table, but the entries are labeled "-tbd-" to indicate work-in-progress. All ionizing dose sensors have some neutron sensitivity. Work is in progress at Sandia National Laboratories to better define the neutron sensitivity of several of the dose monitors. Our best estimate of the spectrum-averaged neutron sensitivities of the ionizing dose sensors is indicated in the table. The Nomenclature section provides a definition or a reference to the source of these estimates.

Table 5. Characterization of Primary SNL Neutron Fields

Spectrum Parameter	SPR-III Central Cavity	SPR-III 17" Leakage	SPR-III 80" Leakage	SPR-III 120" Leakage	ACRR Central Cavity	ACRR "New" Pb-B Bucket
	SPR3CC	SCR4	SPR_80A3	SPR_120	ACF9	TPB13
Shot	8827	6338	7700	7282	5407	4681
Energy (MJ)	0.25	10.0	54.0	17.52	16.66	8.0
$\Phi_{3\text{MeV}}(\text{n/cm}^2)$	1.4701×10^{12}	3.2029×10^{12}	7.14939×10^{11}	4.94126×10^{11}	1.94650×10^{13}	3.31004×10^{13}
Spectrum-averaged Quantities						
${}^{\text{tot}}\Phi_{10\text{keV}}$	0.99854	0.98582	0.79985	0.69782	0.62815	0.84094
${}^{\text{tot}}\Phi_{3\text{MeV}}$	0.10732	0.11863	0.0685361	0.049987	0.055532	0.048445
${}^{\text{tot}}\Phi_{\text{th}}$	1.81198×10^{-4}	4.46780×10^{-3}	7.26805×10^{-2}	0.11356	0.11694	2.26787×10^{-4}
${}^{\text{tot}}\Phi_{1\text{MeV}}^{\text{Si (85)}}$	0.89524	0.91175	0.62514	0.51400	0.46522	0.56900
${}^{\text{tot}}\Phi_{1\text{MeV}}^{\text{Si (93)}}$	0.82113	0.83516	0.57821	0.47912	0.42719	0.53032
${}^{\text{tot}}\Phi_{1\text{MeV}}^{\text{GaAs (93)}}$	0.86384	0.87134	0.62696	0.52893	0.47154	0.58773
E_{avg} (MeV)	1.2580	1.3380	0.84921	0.66338	0.65944	0.73309
$SP_{10\text{keV}}$	9.3039	8.3099	11.671	13.960	11.312	17.359
SP_{tot}	9.3175	8.4294	14.591	20.005	18.008	20.642
$HP_{\text{Si}(85)}$	0.89524	0.91175	0.62515	0.51400	0.46522	0.56899
$HP_{\text{Si}(93)}$	0.82113	0.835153	0.57787	0.47878	0.42671	0.52980
$HP_{10\text{keV}}(85)$	0.89655	0.92486	0.78158	0.73658	0.74062	0.67661
$HP_{10\text{keV}}(93)$	0.82233	0.84714	0.72247	0.68610	0.67931	0.63001

The number of digits shown do not represent the precision of the quantities. The digits are given to aid in reproducibility and to provide an auditable record of the origin of the numbers.

Table 5. Characterization of Primary SNL Neutron Fields (Continued)

Spectrum Parameter	SPR-III Central Cavity	SPR-III 17" Leakage	SPR-III 80" Leakage	SPR-III 120" Leakage	ACRR Central Cavity	ACRR "New" Pb-B Bucket
	SPR3CC	SCR4	SPR_80A3	SPR_120	ACF9	TPB13
TP	1.6883×10^{-3}	3.7659×10^{-2}	1.0605	2.2718	2.1059	4.6854×10^{-3}
$^{58}\text{Ni}_{\text{flu}}$	6.3842×10^{-11}	7.0114×10^{-11}	4.0800×10^{-11}	2.9862×10^{-11}	3.2634×10^{-11}	3.0467×10^{-11}
$^{32}\text{S}_{\text{sand}}$ [Bq/atom]	1.8258×10^{-32}	2.0040×10^{-32}	1.1643×10^{-32}	8.5024×10^{-33}	9.5134×10^{-33}	8.5923×10^{-33}
$3\text{MeV} \langle \sigma^{32}\text{S} \rangle_{\text{G93}}$	302.39	300.27	301.97	302.34	304.51	315.26
$3\text{MeV} \langle \sigma^{58}\text{Ni} \rangle_{\text{E6}}$	505.49	502.23	505.88	507.66	499.39	534.43
$\text{tot} \langle \sigma^{32}\text{S} \rangle_{\text{G93}}$	32.454	35.621	20.696	15.113	16.910	15.273
$\text{tot} \langle \sigma^{58}\text{Ni} \rangle_{\text{E6}}$	54.252	59.581	34.671	25.376	27.732	25.890
$^{58}\text{Ni}_{\text{m}}$ [(Bq/g)/(n/cm ²)]	6.6415×10^{-11}	6.9840×10^{-11}	4.1567×10^{-11}	3.1209×10^{-11}	3.2561×10^{-11}	3.1126×10^{-11}
$^{32}\text{S}_{\text{m}}$ [(dpm/g)/(n/cm ²)] (ENDF/B-V)	2.2326×10^{-8}	2.4023×10^{-8}	1.4040×10^{-8}	1.0273×10^{-8}	1.1212×10^{-8}	1.0431×10^{-8}
$^{32}\text{S}_{\text{Cf}}$	0.46202	0.50710	0.29463	0.21514	0.24073	0.21742
η^{m}	0.3100	- tbd -	- tbd -	- tbd -	0.2848	0.3220
η^{calc}	0.2955	0.2961	0.2951	0.2942	0.2900	0.2893
R_{kelly} [1-MeV(Si)/ (Gy(TLD))] $\cdot 10^{-11}$	2.8715	2.9571	2.7206	2.3539	0.7570	5.1618

The number of digits shown do not represent the precision of the quantities. The digits are given to aid in reproducibility and to provide an auditable record of the origin of the numbers.

The number of digits shown do not represent the precision of the quantities. The digits are given to aid in reproducibility and to provide an auditable record of the origin of the numbers.

7. Other SNL Reactor Environments

Many other reactor environments are available at the SNL facilities. This document addresses the new neutron characterizations for irradiation geometries used frequently by customers. The reactor operations staff can readily provide comparable information on any of about 20 other well-characterized environments that were evaluated for specific experimenters. Table 6 provides a description of some of these environments. All of the environments in Table 6 are recognized by the RML and are valid inputs to the sulfur and nickel monitor foil analysis programs as well-characterized environments. Table 7 shows the dosimetry-related conversion numbers that are being used to convert from the NIST-traceable quantity to the user-requested quantities in the RML dosimetry reports for these environments. If a facility user desires a spectrum characterization at a new location/environment that will be used for several future irradiations, we can provide a spectrum characterization with about a one month notice.

It should be noted that a large test object may affect the neutron environment. A facility user can test if the neutron spectrum is changed by the test article by fielding a sensor consistent with the damage measure of interest, such as a silicon 2N2222 transistor to measure the $\Phi^{1\text{MeV}}_{\text{Si}(93)}$, in both a free-field exposure and in an exposure with the test article present. If the spectrum is significantly perturbed by a user test article, we can work with the user to obtain a spectrum characterization consistent with his test requirements.

Table 6. Description of Some Available SNL Environments

Spec- trum Name	Run Characterization							Environment Description
	Shot	Date	Energy (MJ)	$\Phi_{3\text{MeV}}$ (n/cm ²)	D_m^{TLD} Gy(TLD)	$\Delta_1 \Phi_{\text{Si}}^{1\text{MeV}}$ (93) (%)	$\Delta_1 \Phi_{3\text{MeV}}$ (%)	
SPR-III								
spr3cc	8827 8835 8843 8869	12/1/92 12/7/92 12/7/92 12/15/ 92	0.25 0.25 2.50 36.0	1.470×10^{12}	39.17 -- -- --	5	3	Central cavity at fuel centerline at a radius of 2.25 inches. Spacers were used to assure centering of the fixture. This experiment uses a cardboard fixture and aluminum honeycomb to attain the vertical position.
s12l8	5636	11/12/ 82	15.4	1.064×10^{13}	232.0	15	10	12 inch leakage along a line through the Cu control rod.
scr4	6338	10/5/88	10.0	3.203×10^{12}	76.25	10	7	17 inch leakage along the line through the Cu control rod
s45a7	6439	11/2/88	10.0	2.770×10^{12}	53.79	10	7	17 inch leakage 45 degrees from the Cu control rod
spr_80a3	7700	10/26/ 90	54.0	7.149×10^{11}	22.17	10	7	80 inch leakage along track of the cart. The experiment fixture had its back to the kiva door.
spr_120	7282	4/26/90	17.52	4.941×10^{11}	20.12	10	7	120 inch leakage from the core center on the opposite side of the reactor from the beam port.

Table 6. Description of Some Available SNL Environments (Continued)

Spec- trum Name	Run Characterization							Environment Description
	Shot	Date	Energy (MJ)	$\Phi_{3\text{MeV}}$ (n/cm ²)	D_{TLD_m} Gy(TLD)	$\Delta_1\Phi_{1\text{MeV}}^{\text{Si}}$ (93) (%)	$\Delta_1\Phi_{3\text{MeV}}$ (%)	
polyca48	3037	5/21/84	2.10	$8.517 \times 10^{+12}$	401.1	15	8	2.54 cm of Cd-poly in cavity.
	3041	5/21/84	2.10		--			
	3074	6/7/84	2.10		--			
flatcb24	3441	11/14/ 84	6.33	$1.814 \times 10^{+11}$	90.52	15	8	30 inches from core sandwiched between a 10 cm. Cd-loaded polyethylene block and a 0.020 inch Cd metal sheet in front and with a 5 cm Cd-loaded polyethylene block behind.
pbshot10	4675	8/21/86	36.0	$2.696 \times 10^{+12}$	168.5	20	10	Leakage at 22 inches behind a 6 inch thick Pb shot wall
pb23	5946	3/29/88	10.0	$1.458 \times 10^{+11}$	9.49	15	8	Leakage in Pb box with front surface at 12 inches. The front wall is 4 inch thick, the other walls are 2 inches thick.
ACRR								
acf9	5407	7/14/92	16.66	$1.946 \times 10^{+13}$	1978.	6	4	Central cavity at fuel centerline using a cardboard fixture in the small aluminum frame. Run done in two parts due to operation problem, power obtained by scaling nickel activity from run 5408. TLD readings also scaled from run 5408 due to dose delivered by 10 minute wait between run segments.

Table 6. Description of Some Available SNL Environments (Continued)

Spec- trum Name	Run Characterization							Environment Description
	Shot	Date	Energy (MJ)	$\Phi_{3\text{MeV}}$ (n/cm ²)	D ^{TLD} _m Gy(TLD)	$\Delta_1\Phi^{1\text{MeV}}_{\text{Si}}$ (93) (%)	$\Delta_1\Phi_{3\text{MeV}}$ (%)	
tpb13	4681	6/6/90	8.0	3.310×10^{13}	702.	7	4	"New" Pb-B bucket. The bucket has 1.5 inches of Pb and 0.5 inches of powdered B ₄ C.
wb7	3905 3910	11/29/ 88 12/5/88	30.0 10.0	$2.376 \times 10^{+13}$	5378. --	15	8	Water bucket in ACRR central cavity. The annular thickness of the water channel is 7/8 inch.
bfrec9	3228	6/29/87	30.0	$8.098 \times 10^{+12}$	905.0	15	8	Bare FREC-I, a 15 inch diameter external fuel ringed cavity. An 8 inch diameter rotating disk with a horizontal rotation axis along the line-of-sight to the ACRR reactor was located in the center of the FREC-I cavity and used to field the dosimetry (rotation was in the vertical plane).
bfii5	3753	7/25/88	60.0	$1.218 \times 10^{+13}$	1807.0	15	8	Bare FREC-II cavity. An 8 inch diameter rotating disk with a horizontal rotation axis along the line-of-sight to the ACRR reactor was located in the center of the FREC-II cavity and used to field the dosimetry (rotation was in the vertical plane). The FREC-II cavity is a 20 inch external fuel ringed cavity.

Table 6. Description of Some Available SNL Environments (Continued)

Spec- trum Name	Run Characterization							Environment Description
	Shot	Date	Energy (MJ)	$\Phi_{3\text{MeV}}$ (n/cm ²)	D_m^{TLD} Gy(TLD)	$\Delta_1 \Phi_{\text{Si}}^{1\text{MeV}}$ (93) (%)	$\Delta_1 \Phi_{3\text{MeV}}$ (%)	
freccd7	3199	5/21/87	30.0	$1.710 \times 10^{+12}$	812.0	15	8	FREC-I with Cd-polyethylene bucket. An 8" diameter rotating disk with a horizontal rotation axis along the line-of-sight to the ACRR reactor was located in the center of the FREC cavity and used to field the dosimetry (rotation was in the vertical plane).
freciip6	5212	9/26/91	30.0	$3.273 \times 10^{+12}$	1005.0	12	6	FREC-I Cd-loaded polyethylene bucket inside FREC-II. An 8 inch diameter rotating disk with a horizontal rotation axis along the line-of-sight to the ACRR reactor was located in the center of the FREC cavity and used to field the dosimetry (rotation was in the vertical plane).
f2b6	3811	9/19/88	50.0	$4.269 \times 10^{+12}$	110.5	15	8	FREC-I Pb-B ₄ C bucket inside FREC-II. An 8" diameter rotating disk with a horizontal rotation axis along the line-of-sight to the ACRR reactor was located in the center of the FREC cavity and used to field the dosimetry (rotation was in the vertical plane).

Table 6. Description of Some Available SNL Environments (Continued)

Spec- trum Name	Run Characterization							Environment Description
	Shot	Date	Energy (MJ)	$\Phi_{3\text{MeV}}$ (n/cm ²)	D_m^{TLD} Gy(TLD)	$\Delta_1\Phi_{1\text{MeV}}^{\text{Si}}$ (93) (%)	$\Delta_1\Phi_{3\text{MeV}}$ (%)	
frecpb6	3354	10/1/87	50.0	$5.078 \times 10^{+12}$	128.0	15	8	FREC-1 Pb-B4C bucket inside FREC-I. An 8 inch diameter rotating disk with a horizontal rotation axis along the line-of-sight to the ACRR reactor was located in the center of the FREC cavity and used to field the dosimetry (rotation was in the vertical plane).
REFERENCE FIELDS								
Cf252vi	NIST	---	---	---	---	5	3	Free-field ²⁵² Cf spontaneous fission spectrum as taken from ENDF/B-VI Tape 200, MAT=9861, MT=18, MF=5. D refers to NIST transfer calibration of sulfur counting system. The spectral uncertainties characteristics are estimated from the extensive foil activation studies by Manhart [13, 14, 15]

Table 7. Spectral Characteristics for Some Available SNL Environments

Spectrum Name	Spectrum Characteristic					
	$^{tot}\Phi_{3MeV}$	$^{tot}\Phi_{1MeV}^{Si(93)}$	$^{tot}\langle\sigma_{Ni58}\rangle_{E6}$	$^{tot}\langle\sigma_{S32}\rangle_{G93}$	$^{tot}\Phi_{th}$	R_{kelly}
SPR-III						
spr3cc	0.10732	0.82113	54.252	32.454	0.01812×10^{-2}	2.8715
s1218	0.12293	0.83138	61.697	37.262	0.15135×10^{-2}	3.1000
scr4	0.11863	0.83516	59.581	35.621	0.44676×10^{-2}	2.9571
s45a7	0.11461	0.82046	57.491	34.402	0.48511×10^{-2}	3.6857
spr_80a3	0.06854	0.57821	36.671	20.696	7.26804×10^{-2}	2.7206
spr_120	0.049987	0.47912	25.376	15.113	11.3559×10^{-2}	2.3539
polyc48	0.053177	0.31475	26.241	15.810	38.4841×10^{-2}	1.2568
flatcb24	0.099000	0.50645	49.871	29.470	6.95181×10^{-2}	0.10252
pbshot10	0.075390	0.73717	38.726	23.145	1.04461×10^{-2}	1.5648
pb23	0.051332	0.76286	28.746	16.374	0.37933×10^{-2}	22.840
ACRR						
acf9	0.055532	0.42719	27.732	16.910	11.694×10^{-2}	0.7570
tpb13	0.048445	0.53032	25.890	15.273	0.02270×10^{-2}	5.16179
wb7	0.036262	0.27628	18.735	11.140	38.137×10^{-2}	0.33656
bfrec9	0.049709	0.34443	25.300	14.988	27.685×10^{-2}	0.62002
bfi5	0.049110	0.34422	24.851	14.513	27.612×10^{-2}	0.46986
freccd7	0.071983	0.43921	36.094	21.594	4.2289×10^{-2}	0.21560

The number of digits shown do not represent the precision of the quantities. The digits are given to aid in reproducibility and to provide an auditable record of the origin of the numbers.

Table 7. Spectral Characteristics for Some Available SNL Environments (Continued)

Spectrum Name	Spectrum Characteristic					
	$\text{tot } \Phi_{3\text{MeV}}$	$\text{tot } \Phi_{1\text{ MeV}}^{\text{Si}(93)}$	$\text{tot } \langle \sigma^{\text{Ni58}} \rangle_{\text{E6}}$	$\text{tot } \langle \sigma^{\text{S32}} \rangle_{\text{G93}}$	$\text{tot } \Phi_{\text{th}}$	R_{kelly}
freciip6	0.026957	0.15045	13.317	8.0477	67.637×10^{-2}	0.18179
f2b6	0.054538	0.54561	29.067	16.952	0.01444×10^{-2}	3.8650
frecpb6	0.052177	0.51898	27.559	16.179	0.06984×10^{-2}	3.9457
REFERENCE FIELDS						
Cf252vi	0.23694	1.05783	115.32	70.244	0.0	----
The number of digits shown do not represent the precision of the quantities. The digits are given to aid in reproducibility and to provide an auditable record of the origin of the numbers.						

8. Comparison With Previous Irradiations

If users have testing requirements that span the time in which this change in dosimetry reporting takes place, they may need to compare irradiations characterized by old and new dosimetry reports. Table 5 provides the information to allow the user to convert between currently-reported fluence measures and the previous fluence measures for the baseline environments. Table 8 summarizes some of the traditional spectrum integral parameters for the latest environment characterizations. Table 9 defines the previous (pre-10/1/93) characterizations for these environments. These previous characterization numbers were taken from Reference 6 and the latest facility users memorandum.* The data from Tables 5, 8, and 9 are combined in Table 10 to provide an easy conversion between old and new SNL-reported spectral quantities.

Since the mathematical equations defining the hardness parameter and the 1-MeV(Si)-equivalent fluence have changed, many of the entries for the quantities in Table 10 appear as ratios. This format permits the reader to explicitly convert one measure to another. In the case of the 3-MeV fluence, $\Phi_{3\text{MeV}}$, and the shape parameter, $SP_{10\text{keV}}$, the mathematical definition, and hence the nomenclature used in this document is unchanged. Thus the heading in the tables for the 3-MeV conversion is labeled " $\Phi_{3\text{MeV}}$," and does not appear as a ratio. Likewise, the heading for the shape parameter is labeled as " $SP_{10\text{keV}}$," and not as a ratio. Some of the quantities in the table [$SP_{10\text{keV}}$, $HP_{10\text{keV}}(85)$, $HP_{10\text{keV}}(93)$, and $HP_{\text{Si}}(93)$] depend only upon the spectrum shape. Other parameters [$\Phi_{3\text{MeV}}$, $\Phi_{\text{Si}}^{1\text{MeV}}(85)$, and $\Phi_{\text{Si}}^{1\text{MeV}}(93)$] depend upon the sensor used as a run monitor.

To avoid any ambiguity the exact method employed to obtain the conversion quantities in Table 10 is made explicit. The change in $SP_{10\text{keV}}$ and the HP ratios are derived by taking the ratio of the respective entries in Tables 8 and 9. The conversion from new to old $\Phi_{3\text{MeV}}$ was obtained by forming an intermediate quantity that was the 3-MeV fluence divided by the monitor foil (nickel or sulfur) activity. The ratio of these intermediate quantities, as drawn from the information in Tables 8 and 9, formed the $\Phi_{3\text{MeV}}$ entry in Table 10. The $\Phi_{\text{Si}}^{1\text{MeV}}(85)/\Phi_{\text{Si}}^{1\text{MeV}}(93)$ entry in Table 10 was obtained by multiplying the Table 10 entries for $SP_{10\text{keV}}$, $HP_{10\text{keV}}(85)/HP_{10\text{keV}}(93)$ and $\Phi_{3\text{MeV}}$. This conversion for the 1-MeV(Si) fluence implicitly assumes that a monitor foil activity was used to characterize the irradiation, and that the change in the reported 3-MeV fluence directly impacted the reported 1-MeV(Si) fluence.

An example may clarify usage of the Table 10. Suppose a user wants to convert from a new silicon 1-MeV-equivalent fluence, $\Phi_{\text{Si}}^{1\text{MeV}}(93)$, to the previously reported quantity,

*T. F. Lüera, "Adjustments in Dosimetry Reporting at SPR," SNL Memorandum to Facility Users, January 25, 1988.

$\Phi^{1\text{MeV}}_{\text{Si}}(85)$. If the exposure was in the SPR-III central cavity with a nickel monitor foil, he should multiply the new RML reported value by the entry in column 6 row 1 of Table 10, namely 1.04468. This quantity can now be compared with previous 1-MeV(Si)-equivalent fluence values. Note that use of the $\Phi^{1\text{MeV}}_{\text{Si}}(85)$ values is not in accordance with the new ASTM standards. Its usage may, however, be justified for systems whose radiation hardness specification pre-dates the new ASTM standard. SNL operations and dosimetry experts will be glad to discuss with you the appropriate unit for your intended application. The final decision for the reported quantity rests with the vendor or program office that levied the radiation hardness requirement. We encourage users to fully define the quantity used in their written reports to their sponsor. This will minimize any confusion in the future interpretation of the results. A very detailed nomenclature section is provided in this report to aid the reader in understanding and reproducing the quantities in the tables.

Note that to convert an old silicon-equivalent fluence, $\Phi^{1\text{MeV}}_{\text{Si}}(85)$, to the new ASTM silicon-equivalent fluence, $\Phi^{1\text{MeV}}_{\text{Si}}(93)$, follow a procedure similar to that described above, but divide (instead of multiply) the old value by the appropriate entry from Table 10.

Reactor Environment (spectrum file)	SP _{10kev}	HP _{10kev} (85)	HP _{10kev} (93)	HP _{Si} (93)	Φ _{3MeV}	A ^{Ni58} _m	A ^{S32} _m (GLUCS-93)
SPR-III Central Cavity (spr3cc)	9.3039	0.89655	0.82233	0.82113	1.4701x10 ⁺¹²	8.752x10 ⁻²⁰	2.508x10 ⁻¹⁹
SPR-III 17" Leakage at OE (scr4)	8.3099	0.92486	0.84714	0.83513	3.2029x10 ⁺¹²	1.814x10 ⁻¹⁹	5.319x10 ⁻¹⁹
ACRR Central Cavity (acf9)	11.312	0.74062	0.67931	0.42671	1.9465x10 ⁺¹³	1.098x10 ⁻¹⁸	3.223x10 ⁻¹⁸
ACRR "New" Pb-B Bucket (tpb13)	17.359	0.67661	0.63001	0.52980	3.3100x10 ⁺¹³	2.046x10 ⁻¹⁸	5.845x10 ⁻¹⁸

The number of digits shown do not represent the precision of the quantities. The digits are given to aid in reproducibility and to provide an auditable record of the origin of the numbers.

Table 9. Dosimetry Results from Previous SNL Neutron Field Characterization Irradiations

Reactor Environment (spectrum file)	SP_{10keV}	HP_{10keV}(85)	$\Phi_{3\text{MeV}}$	A^{Ni58}_m	A^{S32}_m (ENDF/B-Vd)
SPR-III Central Cavity (barnob6)	8.46*	0.889*	$3.588 \times 10^{+13}$	2.010×10^{-18}	6.678×10^{-18}
SPR-III 17" Leakage at $\text{\textcircled{E}}$ (scr4)	8.30	0.890	$3.267 \times 10^{+12}$	1.814×10^{-19}	5.756×10^{-19}
ACRR Central Cavity (acrrcf28)	12.46	0.726	$1.027 \times 10^{+13}$	5.910×10^{-19}	1.927×10^{-18}
ACRR "New" Pb-B Bucket (lbacrr12)	16.83	0.659	$3.140 \times 10^{+13}$	1.879×10^{-18}	6.208×10^{-18}
* The product of these numbers has been typically reported as 7.57 due to rounding.					

Table 10. Conversion From New Spectral Quantities to Old Spectral Quantities

Reactor Environment	Conversion (Old/New)						
				Depends on use of $^{58}\text{Ni}(n,p)^{58}\text{Co}$ Sensor		Depends on use of $^{32}\text{S}(n,p)^{32}\text{P}$ Sensor	
	SP _{10kev}	HP _{10kev} (85)/ HP _{10kev} (93)	HP _{10kev} (85)/ HP _{Si} (93)	$\Phi_{3\text{MeV}}$	$\Phi_{1\text{MeV}}^{\text{Si}(85)}/$ $\Phi_{1\text{MeV}}^{\text{Si}(93)}$	$\Phi_{3\text{MeV}}$	$\Phi_{1\text{MeV}}^{\text{Si}(85)}/$ $\Phi_{1\text{MeV}}^{\text{Si}(93)}$
SPR-III Central Cavity	0.9093	1.0811	1.0827	1.0627	1.0447	0.9890	0.9721
SPR-III 17" Leakage at CE	0.9988	1.0506	1.0657	1.0200	1.0703	1.0170	1.0671
ACRR Central Cavity	1.1015	1.0687	1.7014	0.9802	1.1539	0.9521	1.1208
ACRR "New" Pb-B Bucket	0.9695	1.0460	1.2439	1.033	1.0476	0.9636	0.9773

The number of digits shown do not represent the precision of the quantities. The digits are given to aid in reproducibility and to provide an auditable record of the origin of the numbers.

9. Audit Trail for Analysis Output Listings

To provide an audit log of the numbers that went into the current characterization of the SNL baseline environments, a set of three appendices are published as a second volume. All the calculations in the appendices were performed on a SUN Sparc 2 running SunOS 4.1.2 and OPENWINDOWS Ver. 3.

The first appendix provides the short version of the SNL-SAND-II listing which documents the final spectrum characterization. SNL-SAND-II is the SNL version of W. McElroy's SAND-II code [16]. The SNL-SAND-II code is being made available for external distribution through the Radiation Shielding Information Center at ORNL. Version 1.0 of SNL-SAND-II was used to produce the listing in Appendix A. The expanded outputs in the SNL-SAND-II analysis summaries allow easy identification of the spectral integral parameters that appear in Tables 5 and 7.

The MANIPULATE code is a simple data manipulation code that transforms spectra into different representations and folds sensor response functions and differential number spectra to provide spectrum integral parameters. Appendix B provides the MANIPULATE analysis of the baseline SNL environments. The spectra for these four environments are provided in five different forms: differential number, differential energy, number fraction, energy fraction, and integral number.

Appendix C provides some MANIPULATE listing for folding the SNL-SAND-II spectrum into various response functions. The MANIPULATE listings in this appendix represent the integral of a differential number spectrum with a response function. The response functions represent various energy-dependent cross sections and dosimetry efficiencies that affect the characteristics reported in Tables 5 and 7. For example, "s-32g3.103" represents the 1993 version of the GLUCS $^{32}\text{S}(n,p)^{32}\text{P}$ cross section [9]. The 103 designation corresponds to the ENDF designation [17] for the (n,p) reaction. Another example of a response function is "silicon.kerma-rad" which represents the silicon energy-dependent neutron kerma in units of $\text{rad}(\text{Si})/(\text{n}/\text{cm}^2)$. The "alanine-eff-kerma" represents the effective alanine neutron kerma, that is, it represents the convolution of the alanine neutron kerma with the energy-dependent alanine efficiency [18] for recording neutron-induced ionizing dose. When this quantity is folded with a spectrum and divided by the spectrum-averaged alanine neutron kerma, the resulting quantity is ξ^{Aln}_n . The "co-58m_sigpopx" response function [19] represents the convolution of the ENDF/B-VI $^{58}\text{Ni}(n,p)^{58}\text{Co}$ cross section [7] with the energy-dependent isomer production ratio of $^{58\text{m}}\text{Co}$ to $^{58\text{tot}}\text{Co}$ for nickel foils. When this quantity is folded with a spectrum and divided by the spectrum-averaged $^{58}\text{Ni}(n,p)^{58}\text{Co}$ total cross section, the resulting quantity is η^{calc} . Sometimes the spectral integral parameters from MANIPULATE differ slightly from those in the SNL-SAND-II analysis. The difference is due to numerical round-off in the processing and the precision of the ASCII file transfer from SNL-SAND-II to MANIPULATE. In all cases where spectral quantities can be derived from both sources, the SNL-SAND-II number, being a more basic quantity, is used in this analysis. Some of

the MANIPULATE integral parameters are used in determining the neutron efficiency of various dosimeters. Other integrals are of important kerma and spectrum-averaged cross sections.

10. Conclusions

This report has provided a detailed characterization of the baseline SNL reactor environments. It has also provided a look at the diversity of neutron environments available to users of the facilities. This report forms part of an audit trail for the Area V Reactor Facilities QA program for radiation hardness testing purposes.

If facility users have any questions on the changes in neutron environments detailed in this report, SNL personnel in the areas of facility operations, dosimetry, and test fidelity will be glad to provide additional information. Some of the primary contacts are listed below.

Table 11. SNL Contacts for Dosimetry Questions

Name	Phone	Fax	E-mail Address	Expertise
	Area Code 505			
D. T. Berry	845-3142	844-5901	----	SPR Operations
J. W. Bryson	845-3210	845-3115	----	Facilities Manager
P. J. Griffin	845-9121	845-3117	pjgriff@sandia.gov	Dosimetry, Theoretical
J. G. Kelly	845-3021	845-3117	----	Dosimetry, Experimental
M. F. McCrory	845-3031	845-3115	----	ACRR Operations
D. W. Vehar	845-3414	845-3115	dwvehar@sandia.gov	RML Dosimetry

11. Nomenclature

Spectrum Descriptors

$\phi^x(E)$ Differential number distribution of the fluence, typical units are $n/(cm^2 \cdot MeV)$. The argument "E" represents the neutron energy. The superscript "x" represents a spectrum. When this symbol refers to a generic spectrum, the superscript is omitted.

Φ_x Neutron fluence, typically in units of n/cm^2 . The argument "x" can be used to provide a further qualification on the type of neutron fluence. Specific subscripts used in this document include "tot," "10kev," "3MeV," and "th" which are defined in the following entries.

Φ_{tot} Total neutron fluence in units of n/cm^2 .

$$\Phi_{tot} = \int_0^{\infty} \phi(E) \cdot dE \quad (1)$$

Φ_{10kev} Neutron fluence greater than 10 keV in units of n/cm^2 .

$$\Phi_{10kev} = \int_{10kev}^{\infty} \phi(E) \cdot dE \quad (2)$$

Φ_{3MeV} Neutron fluence greater than 3 MeV in units of n/cm^2 .

$$\Phi_{3MeV} = \int_{3MeV}^{\infty} \phi(E) \cdot dE \quad (3)$$

Φ_{th} Neutron fluence less than 1 eV in units of n/cm^2 .

$$\Phi_{th} = \int_0^{1eV} \phi(E) \cdot dE \quad (4)$$

$\Phi_{Si}^{1MeV}(85)$ 1-MeV(Si) equivalent fluence using the ASTM E722-85 standard [1].

$$\Phi_{Si}^{1MeV}(85) = \frac{\int_0^{\infty} \phi(E) \cdot \sigma_{Si}^{dpa-85}(E) \cdot dE}{K_{ref}^{Sidpa}} \quad (5)$$

$\Phi_{Si(93)}^{1MeV}$ 1-MeV(Si) equivalent fluence using the ASTM E722-93 standard [2].

$$\Phi_{Si}^{1MeV}(93) = \frac{\int_0^\infty \phi(E) \cdot \sigma_{Si}^{dpa-93}(E) \cdot dE}{K_{ref}^{Sidpa}} \quad (6)$$

$\Phi_{GaAs(93)}^{1MeV}$ 1-MeV(GaAs) equivalent fluence using the ASTM E722-93 standard.

$$\Phi_{GaAs}^{1MeV}(93) = \frac{\int_0^\infty \phi(E) \cdot \sigma_{GaAs}^{dpa-93}(E) \cdot dE}{K_{ref}^{GaAsdpa}} \quad (7)$$

$^x\Phi_y$ The fluence Φ_y normalized by the fluence Φ_x . This produces a spectrum characteristic that is independent of the magnitude of the irradiation. Typical usages of this form include:

$$^{tot}\Phi_{3MeV} = \frac{\Phi_{3MeV}}{\Phi_{tot}} \quad (8)$$

$$^{tot}\Phi_{10keV} = \frac{\Phi_{10keV}}{\Phi_{tot}} \quad (9)$$

$$^{tot}\Phi_{Si}^{1MeV}(93) = \frac{\Phi_{Si}^{1MeV}(93)}{\Phi_{tot}} \quad (10)$$

$E_{avg}(F)$ Average energy of a neutron spectrum “F,” in units of MeV. If used in a generic sense, the facility designation may be omitted.

$$E_{avg} = \frac{\int_0^\infty E \cdot \phi(E) \cdot dE}{\int_0^\infty \phi(E) \cdot dE} \quad (11)$$

SP_{10keV} 10 keV spectral shape parameter. This parameter is traditionally called the SI or spectral index by the electronics testing community.

$$SP_{10keV} = \frac{\Phi_{10keV}}{\Phi_{3MeV}} \quad (12)$$

SP_{tot} Total spectral shape parameter. Due to the new ASTM E722-93 definition of the hardness parameter, this quantity was defined in order to preserve a methodology for converting from a 3-MeV fluence to a silicon 1-MeV-equivalent fluence.

$$SP_{tot} = \frac{\Phi_{tot}}{\Phi_{3MeV}} \quad (13)$$

LP Low energy spectrum form parameter. This parameter is used to relate the new and old definitions of the hardness parameter and spectral parameter.

$$LP = \frac{\Phi_{10keV}}{\Phi_{tot}} \quad (14)$$

Note that

$$SP_{10kev} = SP_{tot} \cdot LP \quad (15)$$

TP Thermal spectrum form parameter.

$$TP = \frac{\Phi_{th}}{\Phi_{3MeV}} \quad (16)$$

$\Delta_1 \Phi^{1MeV}_{Si(93)}$ Spectrum characterization component of the uncertainty in the 1-MeV(Si) equivalent fluence using the ASTM E722-93 standard. This uncertainty component depends upon the fidelity of the spectrum characterization. In a least squares spectrum determination the standard deviation of $\Phi^{1MeV}_{Si(93)}$ is a direct output from the analysis. In a SAND-II iterative spectrum determination, this uncertainty term represents an empirical estimate of the uncertainty in the spectrum shape component of the 1-MeV(Si) equivalent fluence based on experience with variation of experimental conditions and dosimetry cross sections.

This term is about twice as large as the uncertainty in the 1-MeV(Si) equivalent fluence derived just from a given set of sensors by perturbing the trial function with the outer iteration methodology [20,6].

- $\Delta_1 \Phi_{3\text{MeV}}$ Spectrum characterization component of the uncertainty in the 3-MeV fluence. This uncertainty component depends upon the fidelity of the spectrum characterization. In a least squares spectrum determination the standard deviation of $\Phi_{3\text{MeV}}$ is a direct result of the methodology. In a SAND-II iterative spectrum determination, this term represents an empirical estimate of the uncertainty in the spectrum shape component of the 3-MeV fluence based on experience with variation of experimental conditions and dosimetry cross sections.
- $\Delta_2 \Phi_{\text{Si}}^{1\text{MeV}}(93)$ Monitor sensor component of the uncertainty in the 1-MeV(Si) equivalent fluence using the ASTM E722-93 standard. This uncertainty component applies to the NIST-traceable quantity reported by the dosimetry laboratory and contains such terms as the detector efficiency and counting statistics.
- $\Delta \Phi_{\text{Si}}^{1\text{MeV}}(93)$ Uncertainty in the 1-MeV(Si) equivalent fluence using the methodology defined in the ASTM E722-93 standard.

$$\Delta \Phi_{\text{Si}}^{1\text{MeV}}(93) = \sqrt{\left[\Delta_1 \Phi_{\text{Si}}^{1\text{MeV}}(93)\right]^2 + \left[\Delta_2 \Phi_{\text{Si}}^{1\text{MeV}}(93)\right]^2} \quad (17)$$

Cross Section Descriptors

- $\Delta\sigma/\sigma$ Cross section standard deviation, typically expressed as a percentage. This energy-dependent quantity is typically defined by the cross section evaluator and is found in File 33 of ENDF-6 format [17] evaluations.
- $\sigma_y(E)$ Microscopic neutron cross section, typical units are b. The subscript “y” can be used to indicate a reaction identifier.
- $\langle\sigma^y\rangle_x$ Spectrum-averaged cross section for reaction “y,” where the neutron spectrum denoted by “x” is used as a weighting function.

$$\langle\sigma^y\rangle_x = \frac{\int_0^\infty \phi^x(E) \cdot \sigma_y(E) \cdot dE}{\int_0^\infty \phi^x(E) \cdot dE} \quad (18)$$

$HP_{Si}(93)$ Silicon hardness parameter, based on ASTM E722-93. Note this is a new definition, defined in ASTM E722-93, and does not correspond to the traditional definition employed in the fast burst reactor community.

$$HP_{Si}(93) = \frac{\Phi_{Si}^{1MeV}(93)}{\Phi_{tot}} = {}^{tot}\Phi_{1MeV}^{Si}(93) \quad (19)$$

$HP_{10kev}(85)$ Silicon 10kev hardness parameter, based on ASTM E722-85. This was traditionally called the hardness parameter by the electronic testing community.

$$HP_{10kev}(85) = \frac{\Phi_{Si}^{1MeV}(85)}{\Phi_{10kev}} = {}^{10kev}\Phi_{1MeV}^{Si}(85) \quad (20)$$

$HP_{10kev}(93)$ Silicon 10kev hardness parameter, based on ASTM E722-93 kerma with the lower energy integration limit of 10 keV. This corresponds to the traditional hardness parameter, as the electronic testing community, with an updated energy-dependent model for the silicon displacement kerma.

$$HP_{10kev}(93) = \frac{\Phi_{Si}^{1MeV}(93)}{\Phi_{10kev}} = {}^{10kev}\Phi_{1MeV}^{Si}(93) \quad (21)$$

A_{m}^{Ni58} $^{58}Ni(n,p)^{58}Co$ measured activity from RML dosimetry reports in units of (Bq/g).

${}^{tot}A_{m}^{Ni58}$ $^{58}Ni(n,p)^{58}Co$ measured activity from RML dosimetry reports normalized by Φ_{tot} in units of (Bq/g)/(n/cm²).

$${}^{tot}A_{m}^{Ni58} = \frac{A_m^{Ni58}}{\Phi_{tot}} \quad (22)$$

$^{tot}C^{Ni58}_{flu}$

Ratio of $^{58}Ni(n,p)^{58}Co$ activity to the total fluence as determined from a SNLSAND-II spectral characterization. This quantity uses the ENDF/B-VI $^{58}Ni(n,p)^{58}Co$ cross section and has units of (Bq/atom)/(n/cm²).

$$^{tot}C^{Ni58}_{flu} = \frac{A^{Ni58}_{sand}}{\Phi_{tot}} \quad (23)$$

$\eta^m(F)$

The measured ratio of the isomer production ratio of ^{58m}Co to $^{58tot}Co$ for nickel foils fielded in an environment "F." If used in a generic sense, the facility designation may be omitted. This number is used to permit the reading of nickel activity at short decay times. The methodology is documented in Reference 19.

$\eta^{calc}(F)$

The calculated ratio of the isomer production ratio of ^{58m}Co to $^{58tot}Co$ for nickel foils fielded in an environment "F." If used in a generic sense, the facility designation may be omitted. This number is used to permit the reading of nickel activity at short decay times in cases where a measured isomer production ratio is not available. The methodology is documented in Reference 19.

A^{S32}_m

$^{32}S(n,p)^{32}P$ measured activity from old format RML dosimetry reports in units of (dpm/g). The RML sulfur counting system uses a transfer calibration with the NIST ^{252}Cf benchmark standard source. This activity, when reported on old-format RML dosimetry reports, assumed the ENDF/B-V Dosimetry File cross section for the $^{32}S(n,p)^{32}P$ reaction. To avoid confusion, when this quantity is used, a cross section library should be indicated if it is other than the ENDF/B-V Dosimetry Library.

$^{tot}A^{S32}_m$

$^{32}S(n,p)^{32}S$ measured activity from old format RML dosimetry reports normalized by Φ_{tot} in units of (dpm/g)/(n/cm²).

$$^{tot}A^{S32}_m = \frac{A^{S32}_m}{\Phi_{tot}} \quad (24)$$

A^{S32}_{sand}

$^{32}S(n,p)^{32}P$ SNL SAND-II calculated activity in units of (dps/atom). The SNLRML dosimetry cross sections are used in the SAND-II spectrum determination.

$^{tot}A^{S32}_{sand}$

$^{32}S(n,p)^{32}P$ SNL SAND-II calculated activity in units of (dps/atom) divided by the total fluence. The SNLRML dosimetry cross sections are used in the SAND-II spectrum determination.

$$^{tot}A^{S32}_{sand} = \frac{A^{S32}_{sand}}{\Phi_{tot}} \quad (25)$$

$Cf^{S32}(F)$

The $^{32}S(n,p)^{32}P$ ^{252}Cf -equivalent total fluence in a specified environment "F." If used in a generic sense, the environment designator may be omitted. This quantity represents the specified total fluence in field "F" times the sulfur spectrum-averaged cross section in the specified field divided by the sulfur spectrum-averaged cross section of the ENDF/B-VI ^{252}Cf spectrum. This number is the $^{32}S(n,p)^{32}P$ NIST-traceable dosimetry quantity at the SNL RML.

$$Cf_x^{S32}(F) = \frac{\Phi_{tot} \cdot \langle \sigma^{S32} \rangle_x}{\langle \sigma^{S32} \rangle_{252Cf}} \quad (26)$$

$^{tot}Cf^{S32}(F)$

The RML-reported $^{32}S(n,p)^{32}P$ ^{252}Cf -equivalent total fluence normalized by the total fluence in a specified environment "F." If used in a generic sense, the environment designator may be omitted.

$$^{tot}Cf_x^{S32}(F) = \frac{Cf_x^{S32}(F)}{\Phi_{tot}} \quad (27)$$

$^{tot}Cf^{S32}_{flu}(F)$

Ratio of $^{32}S(n,p)^{32}P$ ^{252}Cf -equivalent total fluence, as determined by a SNLSAND-II spectral characterization, to the total fluence. This quantity has units of (n/cm²)/(n/cm²) and differs from $^{tot}Cf^{S32}(F)$ only in that the SNLSAND-II ^{252}Cf -equivalent fluence is used rather than the RML-reported ^{252}Cf -equivalent fluence.

Cf^{Gl}_{E5}

A ratio of the ^{252}Cf -spectrum-averaged $^{32}S(n,p)^{32}P$ cross sections from the GLUCS-93 dosimetry library [9] and the ENDF/B-V Dosimetry Library. The ENDF/B-VI ^{252}Cf fission spectrum is used as the weighting function.

$$Cf^{Gl}_{E5} = \frac{\langle \sigma^{S32} \rangle_{GLUCS} \cdot 252Cf}{\langle \sigma^{S32} \rangle_{ENDFVI} \cdot 252Cf} = \frac{70.45 \text{ mb}}{76.01 \text{ mb}} = 0.92685 \quad (28)$$

$K_{ref}^{Si\ dpa}$ Silicon displacement damage reference kerma as stated in ASTM E-722(93). This silicon kerma was obtained by using the NJOY code [21] to process the latest ORNL pre-ENDF-6 ^{28}Si evaluation [3].

$$K_{ref}^{Sidpa} = 95\ \text{MeV} \cdot \text{mb} \quad (29)$$

$K_{ref}^{GaAs\ dpa}$ GaAs displacement damage reference kerma as stated in ASTM E-722(93).

$$K_{ref}^{GaAsdpa} = 70\ \text{MeV} \cdot \text{mb} \quad (30)$$

$\sigma_{Si}^{dpa-85}(E)$ Silicon displacement damage kerma, in units of $\text{MeV} \cdot \text{mb}$, as taken from ASTM standard E-722(85).

$\sigma_{Si}^{dpa-93}(E)$ Silicon displacement damage kerma, in units of $\text{MeV} \cdot \text{mb}$, as taken from ASTM standard E-722(93).

$\sigma_{GaAs}^{dpa-93}(E)$ GaAs displacement damage function, in units of $\text{MeV} \cdot \text{mb}$, as taken from ASTM standard E-722(93). This quantity is not a displacement damage kerma, but a displacement damage function. The displacement kerma is multiplied by a quality factor in order to model the energy dependence of displacements in GaAs devices.

$\langle \sigma^y \rangle_x^z$ The spectrum-averaged cross section for reaction “y” in the specified reactor field “x” and with cross sections from the “z” dosimetry cross section library.

The 1993 GLUCS dosimetry cross section, which was adopted in the SNLRML [11], is used for the $^{32}\text{S}(n,p)^{32}\text{P}$ reaction unless otherwise indicated.

An example of the use of this nomenclature is $\langle \sigma^{s32} \rangle_{252\text{Cf}}^{E5}$ which indicates the ENDF/B-V Dosimetry File $^{32}\text{S}(n,p)^{32}\text{P}$ spectrum-averaged cross section in an ENDF/B-VI ^{252}Cf spontaneous fission spectrum.

${}^{3\text{MeV}}\langle\sigma^y\rangle_x^z$

Spectrum-averaged cross section for the reaction “y” from the dosimetry cross section library “z” in the specified reactor field “x,” divided by the 3-MeV fluence. This quantity is expressed in units of (mb/[n/cm²])). If used in a generic sense, the reactor field descriptor “x” may be omitted. If the cross section library descriptor “z” is omitted, the SNLRML library with the descriptor “SNL” should be assumed.

$${}^{3\text{MeV}}\langle\sigma^y\rangle = \frac{\langle\sigma^y\rangle}{\Phi_{3\text{MeV}}} \quad (31)$$

${}^{\text{tot}}\langle\sigma^y\rangle_x^z$

Spectrum-averaged cross section for the reaction “y” from the dosimetry cross section library “z” in the specified reactor field “x,” divided by the total fluence. This quantity is expressed in units of (mb/[n/cm²])). If used in a generic sense, the reactor field descriptor “x” may be omitted. If the cross section library descriptor “z” is omitted, the SNLRML library with the descriptor “SNL” should be assumed.

$${}^{\text{tot}}\langle\sigma^y\rangle = \frac{\langle\sigma^y\rangle}{\Phi_{\text{tot}}} \quad (32)$$

Ionizing Dose Related Quantities

$C_{\text{equil}}(\text{F})$

Conversion from rad(TLD-400) in an RML normal thickness aluminum equilibrator to rad(TLD-400) in the RML normal thickness Lucite equilibrator in facility F. For a fast fission gamma spectrum, this value is 1.0395. If a facility is not indicated, then a fast fission gamma spectrum is assumed. This quantity is needed to relate alanine ionizing dose to TLD ionizing dose. The TLD uses an aluminum equilibrator and alanine a Lucite equilibrator to ensure a good match between the “Z” of the dosimeter and the equilibrator. A more detailed discussion of the intercomparison of RML gamma dosimeters is provided in Reference 18.

$C_{\text{rad}}^{\text{Gy}}$

Conversion from rad(material) to Gy(material).

$$C_{\text{rad}}^{\text{Gy}} = 10^{-2} \cdot \frac{\text{Gy}(\text{material})}{\text{rad}(\text{material})} \quad (33)$$

$C_{\gamma}^{\text{Dose}}(\text{F})$

Photon kerma (calculated) per photon for a gamma spectrum from facility F in a Lucite equilibrator. For a fast fission gamma spectrum this value is 4.213×10^{-10} rad(TLD-400)/(γ/cm²). If a facility is not indicated, then a fast fission gamma spectrum is assumed.

$C_{\gamma}^{\text{TLD-Aln}}(F)$	Calculated conversion from rad(TLD-400) to rad(alanine) for a gamma spectrum in facility F. This conversion assumes use of a Lucite equilibrators. For a fast fission gamma spectrum, this value is 1.104. For a ^{60}Co field this value is 1.1223. If a facility is not indicated, then a fast fission gamma spectrum is assumed.
$C_n^X(F)$	Spectrum-averaged neutron kerma, calculated with the NJOY code and averaged over an environment "F," for material "X" in units of rad(X)/(n/cm ²). The range of the material designator is described in the Nomenclature section under Dosimetry Material Designators. If used in a generic sense, the facility designation may be omitted.
$D_m^X(F)$	Dose as measured in environment "F" with an RML dosimeter of material "X" in units of Gy("X"), where the dosimeter has been calibrated in a ^{60}Co field. The environment "F" would typically include both neutron and gamma incident radiation. The range of the material designator is described in the Nomenclature section under Dosimetry Material Designators. If used in a generic sense, the facility designator may be omitted. Users are warned that measurements with gamma-sensitive dosimeters will vary for a given neutron exposure based on how long the dosimeter was left in the reactor and on how "cool" the reactor was at the start of the irradiation. The time-integrated delayed-fission gamma dose can be comparable to the prompt fission gamma dose.
$^{\text{tot}}D_m^X(F)$	Dose per incident n/cm ² as measured in environment "F" with an RML dosimeter of material "X" in units of Gy(X)/(n/cm ²), where the dosimeter has been calibrated in a ^{60}Co field. The environment "F" would typically include both neutron and gamma incident radiation. The range of the material designator is described in the Nomenclature section under Dosimetry Material Descriptors. If used in a generic sense, the facility designator may be omitted.
$\xi_n^{\text{TLD}}(F)$	Spectrum-averaged neutron efficiency of the RML TLD-400 thermoluminescent detectors in an environment "F." If used in a generic sense, the facility designation may be omitted. This estimate is based on discussions in Reference 22 which indicates a good approximation of TLD neutron efficiency, in percent, is given by the neutron energy, in MeV, divided by 100.

$\xi^{Aln}_n(F)$	Spectrum-averaged neutron efficiency of the RML alanine dosimeters in an environment "F." If used in a generic sense, the facility designation may be omitted. Neutron energy-dependent efficiencies are given by Reference 23 and have been provided in a private communication from Gerstenberg.* The current estimate used in this report is based on the Gerstenberg work since he provided a very fine energy grid tabulation in addition to a plot. His work is based on the methodology of Caswell and Coyne [24] but uses updated cross sections.
$\xi^{Rch}_n(F)$	Spectrum-averaged neutron efficiency of the RML radiochromic film in an environment "F." If used in a generic sense, the facility designation may be omitted. This data is not yet determined, but H. M. Gerstenberg is in the process of performing calculations using track structure theory [25] to provide this data. Since the radiochromic film is impregnated in nylon, which has a similar "average-Z" to alanine, preliminary estimates suggest that $\xi^{Rch}_n = \xi^{Aln}_n$.
$\xi^{Tef}_n(F)$	Spectrum-averaged neutron efficiency of the RML teflon dosimeters in an environment "F." If used in a generic sense, the facility designation may be omitted. This data is not yet determined, but H. M. Gerstenberg is in the process of performing calculations using track structure theory [25] to provide this data. Since the teflon consists of chains of CF_2 , which has a similar "average-Z" to alanine, preliminary estimates suggest that $\xi^{Tef}_n = \xi^{Aln}_n$.
$R_{kelly}(F)$	Traditional measure of the neutron-to-gamma ratio at the SNL RML, in units of $1\text{-MeV}(\text{Si})/[\text{Gy}(\text{TLD-400})] \cdot 10^{-11}$, for an environment "F." If used in a generic sense, the environment designation can be omitted. This quantity uses the measured TLD-400 dose and the SAND-II calculated $\Phi^{1\text{MeV}}_{\text{Si}}(93)$.

$$R_{kelly} = \frac{{}^{tot}\Phi_{Si}^{1\text{MeV}}(93)}{{}^{tot}D_m^{TLD} \cdot 10^{11}} \quad (34)$$

*Personal communication from Henry M. Gerstenberg, AFRRRI, on September 23, 1993 which included a tabulation of calculated relative effectiveness numerical values for the relative effectiveness of alanine to neutron ionizing dose.

$R_n^{TLD}(F)$

Gamma-to-neutron ratio in a specified irradiation field, "F," in units of photons/neutron. If used in a generic sense, the field designation "F" can be omitted. This number is derived from R_{kelly} and depends only on the measured TLD dose,

$$R_n^{TLD}(F) = \frac{10^{-11} \cdot {}^{tot}\Phi_{Si}^{1MeV}(93)}{R_{kelly}(F) \cdot C_{equil} \cdot C_{rad}^{Gy} \cdot C_{\gamma}^{Dose}(F)} \quad (35)$$

or

$$R_n^{TLD}(F) = \frac{{}^{tot}D_m^{TLD}}{C_{equil} \cdot C_{rad}^{Gy} \cdot C_{\gamma}^{Dose}(F)} \quad (36)$$

The user is warned that the measured gamma-to-neutron ratio is affected by the fraction of the delayed-fission gamma dose seen by the dosimeter. Thus gamma-to-neutron ratio may vary for pulsed versus steady-state irradiations. The ratio will also be affected by the time between irradiation and retrieval of the dosimeter.

$R_n^{TLD-Aln}(F)$

Gamma-to-neutron ratio in a specified field, "F" in units of photons/neutron. This number is derived by using ξ_n^{TLD} , ξ_n^{Aln} , D_m^{TLD} , D_m^{Aln} , C_n^{TLD} , and C_n^{Aln} to solve a system of two simultaneous equations for a best-fit single unknown, $R_n^{TLD-Aln}(F)$. This number is believed to be an improvement over the $R_n^{TLD}(F)$ characterization, but it is dependent upon our model of the energy-sensitivity of the response of the dosimeters to neutrons.

Reaction Descriptors

Si	Refers to the ${}^{nat}\text{Si}(n,X)\text{dpa}$ reaction, that is, the silicon displacement reaction.
Ni58	Refers to the ${}^{58}\text{Ni}(n,p){}^{58}\text{Co}$ reaction.
S32	Refers to the ${}^{32}\text{S}(n,p){}^{32}\text{P}$ reaction.
252Cf	The ${}^{252}\text{Cf}$ free-field spontaneous fission spectrum. This spectrum is an National Institute of Standards and Technology (NIST) reference benchmark field [26,13]. Unless otherwise indicated the ${}^{252}\text{Cf}$ fission spectrum is taken from the ENDF/B-VI ${}^{252}\text{Cf}$ material description.

Dosimeter Material Descriptors

Aln	Alanine, an amino acid. ESR techniques can be used with alanine to obtain a measure of ionizing dose.
Rch	Radiochromic film. This film changes light transmission properties when exposed to ionizing radiation and is used as a dosimeter in high dose applications.
Si	Natural silicon. Silicon is not typically used directly as a dosimeter, but the facility user is often most interested in a dose measure to silicon-based electronics.
Tef	Teflon™, a trademarked material consisting of chains of CF ₂ . ESR techniques can be used with teflon to obtain a measure of ionizing dose.
TLD-400	A specific formulation of CaF ₂ :Mn TLDs produced and sold by Harshaw Inc.

Dosimetry Cross Section Library Descriptors

G93	1993 GLUCS cross sections [9].
E6	ENDF/B-VI cross sections [17].
E5	ENDF/B-V Dosimetry Library cross sections.
SNL	SNLRML cross section [11].
IRDF	1990 IRDF cross sections [10].
J3	JENDL-3 dosimetry cross sections [8].

Miscellaneous Abbreviations

ACRR	Annular Core Research Reactor
ASTM	American Society for Testing and Materials
CSEWG	Cross Section Evaluation Working Group
dpa	displacements per atom
DNA	Defense Nuclear Agency
ESR	Electron Spin Resonance
IEEE	Institute of Electrical and Electronics Engineers
NIST	National Institute of Standards and Technology
NSREC	Nuclear Space Radiation Effects Conference
ORNL	Oak Ridge National Laboratory
QA	Quality Assurance
RML	Radiation Metrology Laboratory at SNL
SNL	Sandia National Laboratories
SNLRML	Sandia National Laboratories' Radiation Metrology Library
SPR	Sandia Pulsed Reactor

tb	To be determined This symbol is used for table entries of spectral parameters that are not well known but are the focus of current work.
TLD	Thermoluminescent dosimeter
⌘	Symbol for the centerline of the reactor configuration.

12. References

- [1] Standard Practice for Characterizing Neutron Energy Fluence Spectra in Terms of an Equivalent Monoenergetic Neutron Fluence for Radiation-Hardness Testing of Electronics, ASTM E 722-85, Annual Book of ASTM Standards, Vol. 12.02, 1993.
- [2] Standard Practice for Characterizing Neutron Energy Fluence Spectra in Terms of an Equivalent Monoenergetic Neutron Fluence for Radiation-Hardness Testing of Electronics, ASTM E 722-93, Annual Book of ASTM Standards, Vol. 12.02, 1994.
- [3] D. C. Larson, D. M. Hetrick, S. J. Epperson, N. M. Larson, Evaluation of $^{28,29,30}\text{Si}$ Neutron Induced Cross Sections for ENDF/B-VI, Oak Ridge National Laboratory, Report ORNL-TM-11825. This cross section is likely to be considered for inclusion in the next ENDF/B-VI release
- [4] J. G. Kelly, P. J. Griffin, Comparison of Measured Silicon Displacement Damage Ratios with ASTM E-722 and NJOY Calculated Damage, in *Proceedings of the Seventh ASTM-Euratom Symposium on Reactor Dosimetry*, held in Strasbourg, France, 27-31 August 1990, G. Tsotridis, R. Dierckx, P. D'Hondt editors, Kluwer Academic Publishers, 1990, pp. 711-718.
- [5] M. H. Sparks, T. M. Flanders, J. G. Williams, J. G. Kelly, W. W. Sallee, M. Roknizadeh, J. L. Meason, Energy Dependence of Neutron Damage in Silicon Bipolar Transistors, in *IEEE Trans. on Nuclear Science*, NS-36, No. 6, pp. 1904-1911, 1989.
- [6] J. G. Kelly, D. W. Vehar, *Measurement of Neutron Spectra in Varied Environments by the Foil-Activation Method With Arbitrary Trials*, SAND87-1330, Sandia National Laboratories, Albuquerque, NM, December 1987.
- [7] ENDF-201, ENDF/B-VI Summary Documentation, edited by P. F. Rose, Brookhaven National Laboratory Report BNL-NCS-1741, 4th Edition, October 1991.
- [8] T. Asami, T. Nakagawa, M. Mizumoto, T. Narita, K. Shibata, S. Chiba, T. Fukahori, A. Hasegawa, S. Igarasi, Status of Japanese Evaluated Nuclear Data Library Version 3, in *Nuclear Data for Science and Technology*, pp. 533-536, JAERI, 1988.
- [9] C. Y. Fu, D. M. Hetrick, Experience in Using the Covariances of Some ENDF/B-V Dosimetry Cross Sections: Proposed Improvements and Addition of Cross-Reaction Covariance, in *Proceedings of the Fourth ASTM-Euratom Symposium on Reactor Dosimetry: Radiation Metrology Techniques, Data Bases, and Standardization*, Volume II, conference held at the National Bureau of Standards, Gaithersburg, Maryland on March 22-26, 1982, pp. 877 - 887, Report number NUREG/CP-0029, CONF-820321/V2.
- [10] *International Reactor Dosimetry File (IRDF-90)*, assembled by N. P. Kocherov et al., International Atomic Energy Agency, Nuclear Data Section, IAEA-NDS-141 Rev. 0, August 1990.

- [11] P. J. Griffin, J. G. Kelly, T. F. Luera, J. VanDenburg, *SNL RML Recommended Dosimetry Cross Section Compendium*, SAND92-0094, Sandia National Laboratories, November 1993.
- [12] Standard Test Method for Measuring Reaction Rate and Fast-Neutron Fluences by Radioactivation of Sulfur-32, ASTM E 265-93, Annual Book of ASTM Standards, Vol. 12.02, 1993.
- [13] W. Mannhart, Status of CF-252 Neutron Spectrum as a Standard, in *Reactor Dosimetry: Methods, Applications, and Standardization*, ASTM STP 1001, Harry Farrar IV and E. P. Lippincott, Eds., American Society for Testing and Materials, Philadelphia, 1989, pp. 340-347.
- [14] W. Mannhart, The Neutron Spectrum of Spontaneous Fission of Californium-252, in International Atomic Energy Agency Technical Reports Series No. 273, *Handbook on Nuclear Activation Data*, pp. 163 - 185, 1987.
- [15] W. Mannhart, Data Fitting and Evaluation Techniques for Neutron Spectra, in *International Symposium on Nuclear Data Evaluation Methodology* held at Brookhaven National Laboratory, Upton, NY, on 12 - 16 October 1992, pp. 247 - 256, C.L. Dunford, editor, World Scientific, 1993.
- [16] W. N. McElroy, S. Berg, T. Crockett, and R. Hawkins, *A Computer-Automated Iterative Method for Neutron Flux Spectral Determination by Foil Activation*, AFWL-TR-67-41, Vol. 1, Air Force Weapons Laboratory, Kirtland AFB, New Mexico, July 1967.
- [17] ENDF-102, Data Formats and Procedures for the Evaluated Nuclear Data File ENDF-6, edited by P. F. Rose and C. L. Dunford, Brookhaven National Laboratory Report BNL-NCS-44945, July 1990, revised October 1991.
- [18] P. J. Griffin, D. W. Vehar, J. W. VanDenburg, Characterization of SNL Gamma Dosimeters, SAND93-1421, Sandia National Laboratories, (to be published May 1994).
- [19] P. J. Griffin, C. V. Holm, J. G. Kelly, D. W. Vehar, Neutron Dosimetry with the $^{58}\text{Ni}(n,p)^{58}\text{Co}$ Reaction at Short-Decay Times, in *Reactor Dosimetry*, ASTM STP 1228, Harry Farrar IV, E. Parvin Lippincott, John Williams, and David W. Vehar, Eds., American Society for testing and Materials, Philadelphia, 1994.
- [20] J. G. Kelly, Neutron Spectrum Adjustment with SANDII Using Arbitrary Trial Functions, in *Reactor Dosimetry: Methods, Applications, and Standardization*, ASTM STP 1001, Harry Farrar IV and E. P. Lippincott, Eds., American Society for Testing and Materials, Philadelphia, 1989, pp. 460-468.
- [21] R. E. MacFarlane, D. W. Muir, R. M. Boicourt, The NJOY Nuclear Data Processing System, Volume 1: User's Manual, report LA-9303-M, ENDF-324, May 1982.
- [22] F. H. Attix, *Introduction To Radiological Physics and Radiation Dosimetry*, John-Wiley & Sons, 1986, pg 493.

- [23] M. P. R. Waligorski, G. Danialy, Kim Sun Loh, R. Katz, The Response of the Alanine Detector After Charged-particle and Neutron Irradiations, in *Appl. Radiat. Isot.*, Vol. 40, No. 10-12, pp. 923-933, 1989.
- [24] R. S. Caswell, J. J. Coyne, Interactions of Neutrons and Secondary Charged Particles With Tissue: Secondary Particle Spectra, in *Radiat. Res.* Vol. 52, pp. 418-470, 1972.
- [25] R. Katz, Track Structure Theory in Radiobiology and in Radiation Detection, in *Nuclear Track Detection*, Vol. 2, No. 1, pp. 1-28, 1978.
- [26] J. A. Grundl, C. M. Eisenhauer, *Compendium of Benchmark Neutron Fields for Reactor Dosimetry*, U.S. Department of Commerce, National Bureau of Standards, NBSIR-85-3151, January 1986.

APPENDIX A

**SNL-SAND-II SPECTRA CHARACTERIZATION of
PRIMARY RADIATION ENVIRONMENTS**

SNL-SAND-II Run 1:

Environment:	SPR-III Central Cavity
File Name:	SPR3CC
Shot Date:	12/1/92
Facility Shot Number:	8827
Run Conditions:	250 sec @ 1 kwatt
Dosimetry Position:	Position in central cavity with RCC cardboard fixture and aluminum honeycomb to attain vertical centerline position. Used cart for experiment insertion.

[illegible]

```

1 cases
t spr-iii central cavity, shot 8827, 12/1/92
t sand ii spr iii foil set for shot 8827, spr3cav.
t (dec. 1, 1992),fitted trial spectrum, cavity free field.
t
t modification of spr3cav28-nf to fix bump in trial function
t   that can not be substantiated by MCNP transport calculations
t   and looks to be unacceptable if a spectrum differential energy
t   representation
t
t
t convert s32p activity to GLUCS-93 with ENDF/B-VI cf-252 spectrum
t SAND-II activity ratio for conversion is  $3.954e-32/4.278-32 = 0.9242636746$ 
t MANIPULATE xsec ratio for conversion is  $70.45055/76.01238 = 0.9268299453$ 
t the difference is due to numerical roundoff and interpolation of flux
t we select the SAND-II activity ratio for RML conversion
t
t
t
t
t add new fission foil data - normalized to Ni foil in
t Boron ball of original shot
t
t use boron ball data only doe u235
t include uncovered data on old foils since data was not taken
t with new foils
t
t original data

```

```

t u235f      1.755-11 cadmium 4.705-3
t u235f      1.555-11 cadmium 4.705-3 b4c 0.1481
t u238f      2.317-12 cadmium 4.705-3
t u238f      2.169-12 cadmium 4.705-3 b4c 0.1481
t pu239f     2.233-11 cadmium 4.705-3
t pu239f     2.061-11 cadmium 4.705-3 b4c 0.1481
t
t conversion factors for new foils
t rmlpu nickel ratio (this/test) = 0.401726, test fission activity = 6.0744-
t rmldu nickel ratio (this/test) = 0.40758, test fission activity = 5.3432-
t rmlcu nickel ratio (this/test) = 0.40758, test fission activity = 3.6010-
t mass conversion for rmlcu foil = 0.8805085 (U par of UO2)
t mass conversion for rmldu foil = 0.88176613 (U part of UO2)
t mass conversion for rmlpu foil = 1.14916 (Pu part where 1.0 = Pu239 pa
t
t use 10% reduction in ni58 inside/outside B4C ball to form adjustment for
t fission foils. Note the current SAND model provides -5% with its
t exponential attenuation model. The second 5% is applied by hand
t to the activities to account for scattering effects. The boron ball run
t uses a ni58 foil both inside and outside to confirm identical
t placement of the dosimetry.
t In reality cover corrections are foil/sensor and spectrum dependent.
t
t
t iteration time integrated
t 24 foils jpg
t au197g      6.574-18 cadmium 2.587-3
t au197g      7.414-18
t mg24p       1.242-19 cadmium 2.587-3
t fe56p       5.408-19 cadmium 2.587-3
t al27a       6.135-20 cadmium 2.587-3
t fe54p       1.400-20 cadmium 2.587-3
t ni58p       8.752-20 cadmium 2.587-3
t zr902       1.616-21 cadmium 2.587-3
t s32p        2.508-19
t rmlcu       1.5003-11 cadmium 4.705-3 b4c 0.1481
t u238f       2.317-12 cadmium 4.705-3
t rmldu       2.2228-12 cadmium 4.705-3 b4c 0.1481
t pu239f      2.233-11 cadmium 4.705-3

```

```

ralpu      1.9116-11  cadmium 4.705-3 b4c 0.1481
np237f     1.234-11  cadmium 4.705-3
np237f     1.182-11  cadmium 4.705-3 b4c 0.1481
tisc46     7.325-21  cadmium 2.587-3
tisc47     2.882-19  cadmium 2.587-3
ti48p      8.684-21  cadmium 2.587-3
zn64p      3.882-18  cadmium 2.587-3
co59g      6.923-22  cadmium 2.587-3
cu63g      3.302-18  cadmium 2.587-3
in115n     6.204-17
u235f      1.755-11  cadmium 4.705-3
spectrum tabular
53 points
ener       5.00-8  7.20-8  1.00-7  1.50-7  2.00-7
ener       3.00-7  5.00-7  7.20-7  1.00-6  1.50-6  2.00-6
ener       3.00-6  5.00-6  7.20-6  1.00-5  1.50-5  2.00-5
ener       3.00-5  5.00-5  7.20-5  1.00-4  1.50-4  2.00-4
ener       3.00-4  5.00-4  7.20-4  1.00-3  1.50-3  2.00-3
ener       3.00-3  5.00-3  7.20-3  1.00-2  1.50-2  2.00-2
ener       3.00-2  5.00-2  7.20-2  1.00-1  1.50-1  2.00-1
ener       3.00-1  5.00-1  7.20-1  1.00-0  1.50-0
ener       2.00-0  3.00-0  4.00-0  5.00-0  6.00-0
ener       7.00-0  8.00-0
flux       1.05+3  8.20+2  6.00+2  3.60+2  2.35+2
flux       9.00+1  2.80+1  1.60+1  1.10+1  7.50-0  5.80-0
flux       4.40-0  3.30-0  2.40-0  2.00-0  1.60-0  1.35-0
flux       1.10-0  8.50-1  7.00-1  5.50-1  4.10-1  3.20-1
flux       2.50-1  1.70-1  1.35-1  1.00-1  7.60-2  6.50-2
flux       6.80-2  1.00-1  1.30-1  1.70-1  2.40-1  3.10-1
flux       4.20-1  6.60-1  8.20-1  9.60-1  1.02-0  9.80-1
flux       8.60-1  6.20-1  4.30-1  3.10-1  1.85-1
flux       1.35-1  7.30-2  3.50-2  1.87-2  1.20-2
flux       5.50-3  1.90-3
limit      25
deviation  3.5
discard    3.0
low end thermal
high end fission
norm       0.010

```

plot cards
smooth 5

interim results after 0 iterations

foil reaction	nominal 5.00 percent activity limits (mev)		ratio measured to calculated activities	deviation of measured from calculated activity (percent)
	lower	upper		
s32 (n,p) p32	2.300E+00	7.200E+00	1.0081	0.81
ni58 (n,p) co58 Cd	1.900E+00	7.400E+00	1.0469	4.69
mg24 (n,p) na24 Cd	6.500E+00	1.160E+01	0.9541	-4.59
al27 (n,a) na24 Cd	6.400E+00	1.200E+01	1.0231	2.31
tinat (n,x) sc46 Cd	3.800E+00	9.300E+00	1.0673	6.73
ti48 (n,p) sc48 Cd	5.900E+00	1.230E+01	1.0788	7.88
tinat (n,x) sc47 Cd	1.600E+00	7.500E+00	0.9462	-5.38
fe54 (n,p) mn54 Cd	2.300E+00	7.400E+00	0.9809	-1.91
fe56 (n,p) mn56 Cd	5.500E+00	1.120E+01	1.0100	1.00
co59 (n,g) co60 Cd	1.200E-04	1.700E+00	1.0650	6.50
cu63 (n,g) cu64 Cd	2.100E-02	2.400E+00	0.9504	-4.96
zn64 (n,p) cu64 Cd	2.400E+00	7.400E+00	0.9686	-3.14
zr90 (n,2n) zr89 Cd	1.270E+01	1.750E+01	1.0154	1.54
in115(n,n)in115m	1.000E+00	5.900E+00	1.0169	1.69
au197(n,g) au198 Cd	4.750E-06	1.600E+00	0.9556	-4.44
au197(n,g) au198	1.100E-07	1.600E+00	1.0104	1.04
u235 (n,f) fp Cd	6.300E-02	4.000E+00	0.9902	-0.98
u238 (n,f) fp Cd	1.400E+00	6.600E+00	1.0346	3.46
np237(n,f) fp Cd	5.250E-01	5.200E+00	0.9983	-0.17
np237(n,f) fp Cd B4C	5.250E-01	5.200E+00	0.9989	-0.11
pu239(n,f) fp Cd	8.800E-02	4.200E+00	0.9410	-5.90
u235 (n,f) rml Cd B4C	8.400E-02	4.200E+00	0.9886	-1.14
u238 (n,f) rml Cd B4C	1.400E+00	6.600E+00	1.0248	2.48
pu239(n,f) rml Cd B4C	1.100E-01	4.400E+00	0.9260	-7.40

standard deviation of measured activities (percent) 4.18

average total flux (above 1.00E-02 mev) 1.399E+13
loop 1 iteration 0 4.1840

loop 1 iteration 1 3.7032
loop 1 iteration 2 3.5102

solution has been achieved.

standard deviation of measured-to-calculated activity ratios is less than 3.50percent./
solution results obtained after 3 iterations

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foil reaction	measured activity (dps/nucleus)	calculated activity (dps/nucleus)	nominal 5.00 percent activity limits (mev)		ratio measured to calculated activities	deviation of measured from calculated activity (percent)
			lower	upper		
s32 (n,p) p32	2.508E-19	2.501E-19	2.300E+00	7.300E+00	1.003	0.2825
ni58 (n,p) co58 Cd	8.752E-20	8.413E-20	1.900E+00	7.400E+00	1.040	4.030
mg24 (n,p) na24 Cd	1.242E-19	1.324E-19	6.500E+00	1.160E+01	0.9383	-6.170
al27 (n,a) na24 Cd	6.135E-20	6.107E-20	6.400E+00	1.200E+01	1.005	0.4599
tinat (n,x) sc46 Cd	7.325E-21	6.981E-21	3.800E+00	9.300E+00	1.049	4.923
ti48 (n,p) sc48 Cd	8.684E-21	8.210E-21	5.900E+00	1.230E+01	1.058	5.769
tinat (n,x) sc47 Cd	2.882E-19	3.061E-19	1.600E+00	7.500E+00	0.9416	-5.844
fe54 (n,p) mn54 Cd	1.400E-20	1.437E-20	2.300E+00	7.400E+00	0.9739	-2.605
fe56 (n,p) mn56 Cd	5.408E-19	5.465E-19	5.500E+00	1.120E+01	0.9895	-1.051
co59 (n,g) co60 Cd	6.923E-22	6.721E-22	1.150E-04	1.700E+00	1.030	3.006
cu63 (n,g) cu64 Cd	3.302E-18	3.374E-18	2.100E-02	2.400E+00	0.9786	-2.140
zn64 (n,p) cu64 Cd	3.882E-18	4.032E-18	2.400E+00	7.400E+00	0.9629	-3.712

zr90 (n,2n) zr89 Cd	1.616E-21	1.619E-21	1.270E+01	1.750E+01	0.9984	-0.1608
in115(n,n)in115m	6.204E-17	6.094E-17	1.000E+00	5.900E+00	1.018	1.810
au197(n,g) au198 Cd	6.574E-18	6.699E-18	4.750E-06	1.600E+00	0.9813	-1.870
au197(n,g) au198	7.414E-18	7.190E-18	8.400E-08	1.600E+00	1.031	3.119
u235 (n,f) fp Cd	1.755E-11	1.734E-11	6.300E-02	4.100E+00	1.012	1.210
u238 (n,f) fp Cd	2.317E-12	2.245E-12	1.400E+00	6.600E+00	1.032	3.191
np237(n,f) fp Cd	1.234E-11	1.222E-11	5.250E-01	5.300E+00	1.010	0.9581
np237(n,f) fp Cd B4C	1.182E-11	1.170E-11	5.500E-01	5.300E+00	1.010	1.022
pu239(n,f) fp Cd	2.233E-11	2.325E-11	8.800E-02	4.300E+00	0.9604	-3.956
u235 (n,f) rml Cd B4C	1.500E-11	1.485E-11	8.400E-02	4.200E+00	1.010	1.026
u238 (n,f) rml Cd B4C	2.223E-12	2.174E-12	1.400E+00	6.600E+00	1.023	2.254
pu239(n,f) rml Cd B4C	1.912E-11	2.024E-11	1.150E-01	4.500E+00	0.9445	-5.552

standard deviation of measured activities (percent) 3.390

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response analysis - loop = 1 iteration = 3

response function

dose

fluence (n/cm**2)	0.1369761E+14
kerma (rad-tissue)	29237.31
cp dose to elem-57 (rad-tissue)	32953.93
dose eqv. (rem)	348054.5
dose to elem-57 from h-1(n,g)h-2 (rad-tissue)	3470.135
ansi and ncrp dose (rem)	364748.0

spectrum characterization

total fluence (n/cm**2)	1.3697612E+13	
fluence > 10 kev	1.3677593E+13	(0.9985385)

fluence > 3 Mev	1.4700893E+12	(0.1073245)
fluence < 1 eV	2.4819794E+09	(1.8119796E-04)
avg. energy (mev)	1.257954	
HP-ASTM E722-85	0.8952391	
(ASTM Si damage function)		
HP-ASTM E722-93	0.8211282	
(ENDF-6 Si damage function)		
HP-10kev E722-85	0.8965495	
(ASTM Si damage function)		
HP-10kev E722-93	0.8223301	
(ENDF-6 Si damage function)		
SP	9.303920	
(F[>10 kev]/F[>3 MeV])		
SP	9.317537	
(F[tot]/F[>3 MeV])		
sulfur cross section (mb)	302.3892	
(spectrum averaged, > 3 mev)		
nickel cross section (mb)	505.4932	
(spectrum averaged, > 3 mev)		
sulfur cross section (mb)	32.45376	
(spectrum averaged, > total)		
nickel cross section (mb)	54.25180	
(spectrum averaged, total)		
gaas 1 mev fluence	1.1832524E+13	(0.8638384)
(ref-sig[gaas] = 70 mev-mb)		
silicon 1 mev fluence	1.1247595E+13	(0.8211354)
(ref-sig[si] = 95 mev-mb)		
old ASTM Si 1 mev flu.	1.2262639E+13	(0.8952392)
(ref-sig[si] = 95 mev-mb)		
S32 Act. -dps/gm meas-ENDF	305819.5	(2.2326480E-08)
S32 Act. - Bq/atom meas.	2.5080000E-19	(1.8309760E-32)
S32 Act. - Bq/atom calc.	2.5009335E-19	(1.8258172E-32)
252-Cf Eq. Flu. S32 Meas.-n/cm2	6.3463542E+12	
252-Cf Eq. Flu. S32 Calc.-n/cm2	6.3284723E+12	
Mi58 Act. - Bq/atom meas.	8.7520000E-20	(6.3894348E-33)
Mi58 Act. - Bq/atom calc.	8.4129878E-20	(6.1419374E-33)

spr-iii central cavity, shot 8827, 12/

Figure A-1: SPR3CC Differential Number Distribution

SNL-SAND-II Run 2:

Environment:	SPR-III 17" Leakage
File Name:	SCR4
Shot Date:	10/5/88
Facility Shot Number:	6338
Run Conditions:	1000 sec @ 10 kwatt
Dosimetry Position:	Dosimetry on curved aluminum plate located at reactor centerline. The azimuthal position is fixed to along the line through the copper control rod.

A-12

```
Job name = scr4
Library = snlrml
```

```

1 cases
t SPR-III 17" leakage opposite cu control rod, shot 6338, 10-5-88
t 17" leakage opposite cu control rod
t (mar. 27, 1984) with spr317 trial spectrum, free field.
t
t convert s32p activity to GLUCS-93 with ENDF/B-VI cf-252 spectrum
t SAND-II activity ratio for conversion is  $3.954e-32/4.278-32 = 0.9242636746$ 
t MANIPULATE xsec ratio for conversion is  $70.45055/76.01238 = 0.9268299453$ 
t the difference is due to numerical roundoff and interpolation of flux
t we select the SAND-II activity ratio for RML conversion
t
iteration      time integrated
18 foils
au197g      cadmium 2.587-3
au197g
fe56p      cadmium 2.587-3
mn55g      cadmium 2.587-3
mg24p      cadmium 2.587-3
al27a      cadmium 2.587-3
ni58p      cadmium 2.587-3
s32p
na23g      cadmium 4.705-3
na23g
u235f      b4c 0.1481 cadmium 4.705-3
u238f      b4c 0.1481 cadmium 4.705-3
pu239f     b4c 0.1481 cadmium 4.705-3
np237f     b4c 0.1481 cadmium 4.705-3

```

fe54p cadmium 2.587-3
 co59g cadmium 2.587-3
 sc45g cadmium 2.587-3
 ti48p cadmium 2.587-3
 acts 1.208-16 1.410-16 1.132-18 3.784-17 2.807-19
 acts 1.340-19 1.814-19
 acts 5.319-19
 acts 2.358-19 7.022-19
 acts 3.029-11 4.604-12 4.171-11 2.505-11
 acts 3.191-20 7.982-21 5.687-20 1.902-20
 spectrum tabular
 41 points

ener	3.00-7	5.00-7	7.00-7	1.00-6	2.00-6	
ener	3.00-6	5.00-6	7.00-6	1.00-5	2.00-5	
ener	3.00-5	5.00-5	7.00-5	1.00-4	2.00-4	
ener	3.00-4	5.00-4	7.00-4	1.00-3	2.00-3	
ener	3.00-3	5.00-3	7.00-3	1.00-2	2.00-2	
ener	3.00-2	5.00-2	7.00-2	1.00-1	2.00-1	
ener	3.00-1	5.00-1	7.00-1	1.00-0	1.50-0	
ener	2.00-0	3.00-0	4.00-0	5.00-0	6.00-0	8.00-0
flux	3.40+3	1.75+3	1.20+3	8.50+2	4.40+2	
flux	2.90+2	1.75+2	1.20+2	9.10+1	4.50+1	
flux	2.90+1	1.75+1	1.30+1	9.10-0	4.90-0	
flux	3.40-0	1.90-0	1.40-0	9.80-1	5.20-1	
flux	4.10-1	3.50-1	3.40-1	3.50-1	4.20-1	
flux	5.00-1	7.00-1	9.30-1	1.00-0	9.50-1	
flux	8.00-1	5.20-1	3.60-1	2.80-1	2.05-1	
flux	1.45-1	7.40-2	4.10-2	2.40-2	1.30-2	2.25-3

 limit 50
 deviation 2.00
 discard 3.0
 low end thermal
 high end fission
 norm 0.010
 plot cards
 smooth 5

interim results after 0 iterations

foil reaction	nominal 5.00 percent activity limits (mev)		ratio measured to calculated activities	deviation of measured from calculated activity (percent)
	lower	upper		
s32 (n,p) p32	2.300E+00	7.300E+00	0.9736	-2.64
ni58 (n,p) co58 Cd	2.000E+00	7.400E+00	0.9924	-0.76
na23 (n,g) na24 Cd	7.200E-07	2.100E+00	0.9435	-5.65
na23 (n,g) na24	7.600E-09	8.400E-01	1.0622	6.22
mg24 (n,p) na24 Cd	6.500E+00	1.170E+01	0.9968	-0.32
al27 (n,a) na24 Cd	6.400E+00	1.210E+01	1.0170	1.70
sc45 (n,g) sc46 Cd	4.750E-07	1.000E+00	0.9806	-1.94
ti48 (n,p) sc48 Cd	5.900E+00	1.240E+01	1.0865	8.65
mn55 (n,g) mn56 Cd	6.000E-07	8.000E-01	0.9938	-0.62
fe54 (n,p) mn54 Cd	2.300E+00	7.400E+00	1.0166	1.66
fe56 (n,p) mn56 Cd	5.400E+00	1.130E+01	0.9870	-1.30
co59 (n,g) co60 Cd	7.600E-07	2.800E-01	0.9113	-8.87
au197(n,g) au198 Cd	4.000E-06	2.100E-01	0.9890	-1.10
au197(n,g) au198	3.200E-08	1.700E-01	1.0060	0.60
u235 (n,f) fp B4CCd	6.900E-02	4.400E+00	0.9741	-2.59
u238 (n,f) fp B4CCd	1.400E+00	6.600E+00	1.0080	0.80
np237(n,f) fp B4CCd	5.500E-01	5.400E+00	1.0705	7.05
pu239(n,f) fp B4CCd	9.600E-02	4.600E+00	0.9911	-0.89

standard deviation of measured activities (percent) 4.22

average total flux (above 1.00E-02 mev) 2.686E+13

loop 1 iteration 0 4.2186

loop 1 iteration 1 3.0098

loop	1	iteration	2	2.6975
loop	1	iteration	3	2.5465
loop	1	iteration	4	2.4468
loop	1	iteration	5	2.3722
loop	1	iteration	6	2.3130
loop	1	iteration	7	2.2639
loop	1	iteration	8	2.2222
loop	1	iteration	9	2.1855
loop	1	iteration	10	2.1528
loop	1	iteration	11	2.1228
loop	1	iteration	12	2.0952
loop	1	iteration	13	2.0693
loop	1	iteration	14	2.0449
loop	1	iteration	15	2.0218

solution has been achieved.

standard deviation of measured-to-calculated activity ratios is less than 2.00percent./
 solution results obtained after 16 iterations

foil reaction	measured activity (dps/nucleus)	calculated activity (dps/nucleus)	nominal 5.00 percent activity limits (mev)		ratio measured to calculated activities	deviation of measured from calculated activity (percent)	

			lower	upper		-----	
s32 (n,p) p32	5.319E-19	5.411E-19	2.300E+00	7.300E+00	0.9831	-1.691	
ni58 (n,p) co58 Cd	1.814E-19	1.821E-19	1.900E+00	7.400E+00	0.9961	-0.3912	
na23 (n,g) na24 Cd	2.358E-19	2.364E-19	6.900E-07	2.200E+00	0.9973	-0.2660	
na23 (n,g) na24	7.022E-19	7.020E-19	7.200E-09	8.000E-01	1.000	23.019E-03	
mg24 (n,p) na24 Cd	2.807E-19	2.868E-19	6.500E+00	1.180E+01	0.9788	-2.124	
al27 (n,a) na24 Cd	1.340E-19	1.355E-19	6.400E+00	1.220E+01	0.9891	-1.088	
sc45 (n,g) sc46 Cd	5.687E-20	5.636E-20	4.750E-07	1.100E+00	1.009	0.8979	
ti48 (n,p) sc48 Cd	1.902E-20	1.799E-20	5.900E+00	1.260E+01	1.057	5.745	
mn55 (n,g) mn56 Cd	3.784E-17	3.787E-17	5.750E-07	8.400E-01	0.9993	-74.446E-03	
fe54 (n,p) mn54 Cd	3.191E-20	3.117E-20	2.300E+00	7.500E+00	1.024	2.373	

fe56 (n,p) mn56 Cd	1.132E-18	1.162E-18	5.500E+00	1.150E+01	0.9741	-2.593
co59 (n,g) co60 Cd	7.982E-21	7.989E-21	6.900E-07	3.400E-01	0.9992	-81.384E-03
au197(n,g) au198 Cd	1.208E-16	1.208E-16	4.000E-06	2.100E-01	1.000	37.909E-03
au197(n,g) au198	1.410E-16	1.412E-16	2.800E-08	1.500E-01	0.9989	-0.1145
u235 (n,f) fp B4CCd	3.029E-11	3.081E-11	7.200E-02	4.400E+00	0.9830	-1.703
u238 (n,f) fp B4CCd	4.604E-12	4.650E-12	1.400E+00	6.600E+00	0.9901	-0.9907
np237(n,f) fp B4CCd	2.505E-11	2.434E-11	5.750E-01	5.300E+00	1.029	2.897
pu239(n,f) fp B4CCd	4.171E-11	4.207E-11	1.000E-01	4.600E+00	0.9914	-0.8567

standard deviation of measured activities (percent) 2.000

response analysis - loop = 1 iteration = 16

response function

dose

fluence (n/cm**2)	0.2699855E+14
kerma (rad-tissue)	59096.36
cp dose to elem-57 (rad-tissue)	67198.90
dose eqv. (rem)	696092.5
dose to elem-57 from h-1(n,g)h-2 (rad-tissue)	6712.248
ansi and ncrp dose (rem)	729588.6

spectrum characterization

total fluence (n/cm**2)	2.6998548E+13	
fluence > 10 kev	2.6615822E+13	(0.9858242)
fluence > 3 Mev	3.2028880E+12	(0.1186319)
fluence < 1 eV	1.2062399E+11	(4.4677956E-03)
avg. energy (mev)	1.337955	
HP-ASTM E722-85	0.9117525	

(ASTM Si damage function)		
HP-ASTM E722-93	0.8351308	
(ENDF-6 Si damage function)		
HP-10kev E722-85	0.9248631	
.(ASTM Si damage function)		
HP-10kev E722-93	0.8471395	
(ENDF-6 Si damage function)		
SP	8.309945	
(F[>10 kev]/F[>3 MeV])		
SP	8.429439	
(F[tot]/F[>3 MeV])		
sulfur cross section (mb)	300.2650	
(spectrum averaged, > 3 mev)		
nickel cross section (mb)	502.2347	
(spectrum averaged, > 3 mev)		
sulfur cross section (mb)	35.62100	
(spectrum averaged, > total)		
nickel cross section (mb)	59.58104	
(spectrum averaged, total)		
gaas 1 mev fluence	2.3524815E+13	(0.8713363)
(ref-sig[gaas] = 70 mev-mb)		
silicon 1 mev fluence	2.2547985E+13	(0.8351554)
(ref-sig[si] = 95 mev-mb)		
old ASTM Si 1 mev flu.	2.4615995E+13	(0.9117525)
(ref-sig[si] = 95 mev-mb)		
S32 Act. -dps/gm meas-ENDF	648586.1	(2.4022997E-08)
S32 Act. - Bq/atom meas.	5.3190004E-19	(1.9701061E-32)
S32 Act. - Bq/atom calc.	5.4105191E-19	(2.0040038E-32)
252-Cf Eq. Flu. S32 Meas.-n/cm2	1.3459433E+13	
252-Cf Eq. Flu. S32 Calc.-n/cm2	1.3691011E+13	
Ni58 Act. - Bq/atom meas.	1.8140001E-19	(6.7188803E-33)
Ni58 Act. - Bq/atom calc.	1.8211238E-19	(6.7452658E-33)

Figure A-2: SCR4 Differential Number Distribution

SNL-SAND-II Run 3:

Environment:	ACRR Central Cavity
File Name:	ACF9
Shot Date:	7/14/92
Facility Shot Number:	5407
Run Conditions:	100sec @ 150 kwatt
Dosimetry Position:	Dosimetry fixed on cardboard fixture with in small aluminum frame. Aluminum honeycomb was used to place the dosimetry along the reactor centerline. This run was performed in two parts, 65 and 35 seconds respectively.

*** snl 6522 endf/b-vi specialized version of sand-ii ***

control parameters: 0 0 4 1 0
spline knot control file : fission

Job name = acf9
Library = snlrml

1 cases

t acrr central cavity, shot #5407, 7/14/92

t (jul. 14, 1992), acf.

t

t convert s32p activity to GLUCS-93 with ENDF/B-VI cf-252 spectrum

t SAND-II activity ratio for conversion is $3.954e-32/4.278-32 = 0.9242636746$

t MANIPULATE xsec ratio for conversion is $70.45055/76.01238 = 0.9268299453$

t the difference is due to numerical roundoff and interpolation of flux

t we select the SAND-II activity ratio for RML conversion

t

t fe58g from jef

t suppress fe53gjef cadmium 2.587-3

t acts 2.698-18

t

t

iteration time integrated

24 foils

silmev

au197g

au197g cadmium 2.587-3

mn55g cadmium 2.587-3

pu239f b4c 0.1481 cadmium 4.705-3

u235f b4c 0.1481 cadmium 4.705-3

np237f b4c 0.1481 cadmium 4.705-3

u238f b4c 0.1481 cadmium 4.705-3

fe54p cadmium 2.587-3

ni58p cadmium 2.587-3

s32p
 mg24p cadmium 2.587-3
 fe56p cadmium 2.587-3
 al27a cadmium 2.587-3
 zr902 cadmium 2.587-3
 co59g cadmium 2.587-3
 na23g
 na23g cadmium 2.587-3
 sc45g cadmium 2.587-3
 ti46p cadmium 2.587-3
 ti47p cadmium 2.587-3
 ti48p cadmium 2.587-3
 zn64p cadmium 2.587-3
 cu63g cadmium 2.587-3
 acts 1.468+14
 acts 5.891-14 5.183-14 9.503-15
 acts 3.448-10 3.230-10 1.633-10 2.986-11
 acts 1.832-19 1.098-18
 acts 3.223-18
 acts 1.423-18
 acts 6.017-18 7.187-19 2.111-20 2.500-18
 acts 1.906-16 4.319-17 1.307-17 8.659-20
 acts 4.103-18 1.021-19 5.043-17 8.368-16
 spectrum tabular
 45 points

ener	5.00-8	7.00-8	1.00-7	2.00-7		
ener	3.00-7	5.00-7	7.00-7	1.00-6	2.00-6	
ener	3.00-6	5.00-6	7.00-6	1.00-5	2.00-5	
ener	3.00-5	5.00-5	7.00-5	1.00-4	2.00-4	
ener	3.00-4	5.00-4	7.00-4	1.00-3	2.00-3	
ener	3.00-3	5.00-3	7.00-3	1.00-2	2.00-2	
ener	3.00-2	5.00-2	7.00-2	1.00-1	2.00-1	
ener	3.00-1	5.00-1	7.00-1	1.00-0	1.50-0	
ener	2.00-0	3.00-0	4.00-0	5.00-0	6.00-0	8.00-0
flux	8.90+5	6.50+5	4.45+5	2.20+5		
flux	1.50+5	9.00+4	6.60+4	4.70+4	2.45+4	
flux	1.60+4	9.80+3	6.80+3	4.70+3	2.10+3	
flux	1.25+3	7.50+2	5.20+2	3.20+2	1.75+2	
flux	1.20+2	7.50+1	5.70+1	4.20+1	2.20+1	

flux 1.65+1 1.15+1 8.40-0 6.50-0 4.00-0
flux 3.00-0 2.10-0 1.70-0 1.30-0 8.00-1
flux 6.00-1 4.00-1 3.10-1 2.30-1 1.70-1
flux 1.25-1 7.00-2 3.30-2 1.30-2 5.50-3 1.50-3
limit 50
deviation 4.0
discard 3.0
low end thermal
high end fission
norm 0.010
plot cards
smooth 5

interim results after 0 iterations

foil reaction	nominal 5.00 percent activity limits (mev)		ratio measured to calculated activities	deviation of measured from calculated activity (percent)
	lower	upper		
	-----	-----		
s32 (n,p) p32	2.300E+00	7.100E+00	0.9520	-4.80
ni58 (n,p) co58 Cd	1.900E+00	7.300E+00	0.9875	-1.25
na23 (n,g) na24	6.600E-09	2.550E-03	0.9771	-2.29
na23 (n,g) na24 Cd	4.500E-07	3.400E-03	0.9381	-6.19
mg24 (n,p) na24 Cd	6.600E+00	1.180E+01	0.9782	-2.18
al27 (n,a) na24 Cd	6.600E+00	1.220E+01	1.0402	4.02
si-28 (n,1mev) x	1.900E-01	4.700E+00	0.9548	-4.52
sc45 (n,g) sc46 Cd	4.250E-07	4.250E-03	0.9840	-1.60
ti46 (n,p) sc46 Cd	3.600E+00	9.500E+00	1.1602	16.02
ti47 (n,p) sc47 Cd	1.600E+00	7.300E+00	0.9900	-1.00
ti48 (n,p) sc48 Cd	6.000E+00	1.250E+01	1.1562	15.62
mn55 (n,g) mn56 Cd	5.000E-07	1.150E-03	0.9183	-8.17
fe54 (n,p) mn54 Cd	2.200E+00	7.300E+00	0.9777	-2.23
fe56 (n,p) mn56 Cd	5.500E+00	1.160E+01	1.1068	10.68
co59 (n,g) co60 Cd	6.000E-07	1.425E-04	0.9524	-4.76

cu63 (n,g)	cu64	Cd	5.250E-07	7.600E-02	1.0157	1.57
zn64 (n,p)	cu64	Cd	2.400E+00	7.300E+00	0.9419	-5.81
zr90 (n,2n)	zr89	Cd	1.270E+01	1.750E+01	1.0158	1.58
au197 (n,g)	au198		3.600E-08	5.750E-06	1.0121	1.21
au197 (n,g)	au198	Cd	4.000E-06	6.000E-06	1.0055	0.55
u235 (n,f)	fp	B4CCd	1.800E-03	3.400E+00	1.0156	1.56
u238 (n,f)	fp	B4CCd	1.400E+00	6.000E+00	0.9633	-3.67
np237 (n,f)	fp	B4CCd	5.500E-01	4.700E+00	1.0097	0.97
pu239 (n,f)	fp	B4CCd	3.800E-03	3.800E+00	0.9468	-5.32

standard deviation of measured activities (percent) 6.28

loop	1	iteration	0	average total flux (above 1.00E-02 mev)	2.247E+14
				6.2822	
loop	1	iteration	1	4.4663	
loop	1	iteration	2	4.1068	

solution has been achieved.

standard deviation of measured-to-calculated activity ratios is less than 4.00percent./

solution results obtained after 3 iterations

foil reaction	measured activity (dps/nucleus)	calculated activity (dps/nucleus)	nominal 5.00 percent activity limits (mev)		ratio measured to calculated activities	deviation of measured from calculated activity (percent)
			lower	upper		
s32 (n,p) p32	3.223E-18	3.335E-18	2.300E+00	7.200E+00	0.9665	-3.347
ni58 (n,p) co58 Cd	1.098E-18	1.100E-18	1.900E+00	7.400E+00	0.9977	-0.2263

na23 (n,g) na24	1.906E-16	1.908E-16	6.600E-09	2.400E-03	0.9992	-81.474E-03
na23 (n,g) na24 Cd	4.319E-17	4.422E-17	4.500E-07	3.400E-03	0.9766	-2.337
mg24 (n,p) na24 Cd	1.423E-18	1.559E-18	6.500E+00	1.170E+01	0.9129	-8.708
al27 (n,a) na24 Cd	7.187E-19	7.407E-19	6.500E+00	1.210E+01	0.9703	-2.971
si-28 (n,lmev) x	1.468E+14	1.497E+14	1.900E-01	4.800E+00	0.9804	-1.963
sc45 (n,g) sc46 Cd	1.307E-17	1.283E-17	4.250E-07	4.500E-03	1.018	1.836
ti46 (n,p) sc46 Cd	8.659E-20	7.822E-20	3.700E+00	9.600E+00	1.107	10.70
ti47 (n,p) sc47 Cd	4.103E-18	4.080E-18	1.600E+00	7.500E+00	1.006	0.5657
ti48 (n,p) sc48 Cd	1.021E-19	9.491E-20	6.000E+00	1.250E+01	1.076	7.577
mn55 (n,g) mn56 Cd	9.503E-15	9.610E-15	5.000E-07	1.150E-03	0.9889	-1.111
fe54 (n,p) mn54 Cd	1.832E-19	1.860E-19	2.200E+00	7.400E+00	0.9847	-1.531
fe56 (n,p) mn56 Cd	6.017E-18	5.864E-18	5.500E+00	1.160E+01	1.026	2.607
co59 (n,g) co60 Cd	2.500E-18	2.501E-18	6.000E-07	1.425E-04	0.9994	-58.925E-03
cu63 (n,g) cu64 Cd	8.368E-16	8.278E-16	5.250E-07	7.200E-02	1.011	1.082
zn64 (n,p) cu64 Cd	5.043E-17	5.295E-17	2.400E+00	7.400E+00	0.9524	-4.763
zr90 (n,2n) zr89 Cd	2.111E-20	2.117E-20	1.270E+01	1.750E+01	0.9974	-0.2629
au197(n,g) au198	5.891E-14	5.862E-14	3.800E-08	5.500E-06	1.005	0.4907
au197(n,g) au198 Cd	5.183E-14	5.207E-14	4.000E-06	5.750E-06	0.9953	-0.4663
u235 (n,f) fp B4CCd	3.230E-10	3.122E-10	1.800E-03	3.400E+00	1.035	3.459
u238 (n,f) fp B4CCd	2.986E-11	3.010E-11	1.400E+00	6.200E+00	0.9920	-0.8017
np237(n,f) fp B4CCd	1.633E-10	1.577E-10	5.500E-01	4.800E+00	1.036	3.552
pu239(n,f) fp B4CCd	3.448E-10	3.564E-10	3.800E-03	3.800E+00	0.9676	-3.242

standard deviation of measured activities (percent) 3.908

response analysis - loop = 1 iteration = 3

response function

dose

fluence (n/cm**2)	0.3505201E+15
kerma (rad-tissue)	401297.4
cp dose to elem-57 (rad-tissue)	453061.4
dose eqv. (rem)	4801317.

dose to elem-57 from h-1(n,g)h-2 (rad-tissue)
ansi and ncrp dose (rem)

95531.10
4913262.

spectrum characterization

total fluence (n/cm**2)	3.5052017E+14		
fluence > 10 kev	2.2017974E+14	(0.6281514	)
fluence > 3 Mev	1.9464955E+13	(5.5531628E-02	)
fluence < 1 eV	4.0990933E+13	(0.1169432	)
avg. energy (mev)	0.6594403		
HP-ASTM E722-85	0.4652211		
(ASTM Si damage function)			
HP-ASTM E722-93	0.4267116		
(ENDF-6 Si damage function)			
HP-10kev E722-85	0.7406195		
(ASTM Si damage function)			
HP-10kev E722-93	0.6793134		
(ENDF-6 Si damage function)			
SP	11.31160		
(F[>10 kev]/F[>3 MeV])			
SP	18.00776		
(F[tot]/F[>3 MeV])			
sulfur cross section (mb)	304.5098		
(spectrum averaged, > 3 mev)			
nickel cross section (mb)	499.3938		
(spectrum averaged, > 3 mev)			
sulfur cross section (mb)	16.90993		
(spectrum averaged, > total)			
nickel cross section (mb)	27.73215		
(spectrum averaged, total)			
gaas 1 mev fluence	1.6528472E+14	(0.4715413	)
(ref-sig[gaas] = 70 mev-mb)			
silicon 1 mev fluence	1.4973991E+14	(0.4271934	)
(ref-sig[si] = 95 mev-mb)			
old ASTM Si 1 mev flu.	1.6306939E+14	(0.4652211	)
(ref-sig[si] = 95 mev-mb)			

S32 Act. -dps/gm meas-ENDF	3930048.	(1.1212046E-08)
S32 Act. - Bq/atom meas.	3.2229999E-18	(9.1949056E-33)
S32 Act. - Bq/atom calc.	3.3346244E-18	(9.5133595E-33)
252-Cf Eq. Flu. S32 Meas.-n/cm2	8.1556211E+13	
252-Cf Eq. Flu. S32 Calc.-n/cm2	8.4380807E+13	
Ni58 Act. - Bq/atom meas.	1.0980000E-18	(3.1324871E-33)
Ni58 Act. - Bq/atom calc.	1.1004907E-18	(3.1395930E-33)

Figure A-3: ACF9 Differential Number Distribution

SNL-SAND-II Run 4:

Environment:	ACRR New Pb-B₄C Bucket
File Name:	TPB13
Shot Date:	6/6/90
Facility Shot Number:	4681
Run Conditions:	160 sec @ 50 kwatt
Dosimetry Position:	Dosimetry placed on a cardboard ring and aluminum honeycomb was used to vertically align it with the reactor centerline. The cardboard ring diameter was 2.5 inches. This bucket is composed of 3.8 cm of lead inside 1.27 cm of natural B₄C powder. Top and bottom bucket caps have 2.5 cm of lead inside 1 cm of B₄C. When this bucket is used the experimnetal package can be up to 12.7 cm in diameter.

A-29

```
Job name = tpb13
Library  = snlrml
```

iteration	time integrated
23 foils	
silmev	
au197g	cadmium 2.587-3
au197g	
mn55g	cadmium 2.587-3
ni58p	cadmium 2.587-3
al27a	cadmium 2.587-3
mg24p	cadmium 2.587-3
fe56p	cadmium 2.587-3
fe54p	cadmium 2.587-3
na23g	cadmium 4.705-3
na23g	
s32p	
zr902	cadmium 2.587-3

u235f b4c 0.1481 cadmium 4.705-3
u238f b4c 0.1481 cadmium 4.705-3
pu239f b4c 0.1481 cadmium 4.705-3
np237f b4c 0.1481 cadmium 4.705-3
co59g .cadmium 2.587-3
sc45g cadmium 2.587-3
tisc46 cadmium 2.587-3
tisc47 cadmium 2.587-3
ti48p cadmium 2.587-3
zn64p cadmium 2.587-3
acts 3.736+14
acts 9.208-15 8.715-15 9.575-15 2.046-18 1.114-18
acts 2.363-18 1.004-17 3.237-19 3.144-17
acts 3.263-17 5.845-18 3.569-20
acts 8.218-10 6.691-11 8.472-10 3.608-10
acts 1.885-18 4.414-18 1.485-19 7.183-18 1.601-19
acts 8.777-17
spectrum tabular
45 points
ener 5.00-8 7.20-8 1.00-7 2.00-7
ener 3.00-7 5.00-7 7.20-7 1.00-6 2.00-6
ener 3.00-6 5.00-6 7.20-6 1.00-5 2.00-5
ener 3.00-5 5.00-5 7.20-5 1.00-4 2.00-4
ener 3.00-4 5.00-4 7.20-4 1.00-3 2.00-3
ener 3.00-3 5.00-3 7.20-3 1.00-2 2.00-2
ener 3.00-2 5.00-2 7.20-2 1.00-1 2.00-1
ener 3.00-1 5.00-1 7.20-1 1.00-0 1.50-0
ener 2.00-0 3.00-0 4.00-0 5.00-0 6.00-0
ener 8.00-0
flux 7.80+2 1.10+3 1.60+3 2.60+3
flux 3.40+3 4.70+3 5.80+3 6.50+3 7.80+3
flux 8.20+3 8.30+3 8.20+3 8.00+3 6.90+3
flux 5.60+3 4.20+3 3.20+3 2.50+3 1.50+3
flux 1.18+3 8.70+2 6.80+2 5.30+2 3.35+2
flux 2.50+2 1.75+2 1.40+2 1.12+2 7.00+1
flux 5.30+1 3.75+1 2.95+1 2.35+1 1.45+1
flux 1.03+1 7.00-0 5.20-0 3.80-0 2.60-0
flux 1.80-0 7.30-1 3.40-1 1.72-1 7.00-2
flux 1.40-2

limit 50
 deviation 4.0
 discard 3.0
 low end e
 high end fission
 norm 3.000
 plot cards
 smooth 5

interim results after 0 iterations

foil reaction	nominal 5.00 percent activity limits (mev)		ratio measured to calculated activities	deviation of measured from calculated activity (percent)

	lower	upper		
s32 (n,p) p32	2.200E+00	6.800E+00	0.9772	-2.28
ni58 (n,p) co58 Cd	1.700E+00	7.000E+00	1.0135	1.35
na23 (n,g) na24 Cd	1.350E-05	1.900E-01	0.8794	-12.06
na23 (n,g) na24	1.275E-05	1.900E-01	0.9039	-9.61
mg24 (n,p) na24 Cd	6.500E+00	1.170E+01	1.0496	4.96
al27 (n,a) na24 Cd	6.500E+00	1.210E+01	1.0521	5.21
si-28 (n,1mev) x	1.900E-01	4.500E+00	1.0383	3.83
sc45 (n,g) sc46 Cd	3.000E-06	2.300E-01	1.0433	4.33
tinat (n,x) sc46 Cd	3.700E+00	9.200E+00	1.1139	11.39
ti48 (n,p) sc48 Cd	5.900E+00	1.240E+01	1.1483	14.83
tinat (n,x) sc47 Cd	1.500E+00	7.000E+00	0.9328	-6.72
mn55 (n,g) mn56 Cd	4.750E-05	6.900E-03	0.9438	-5.62
fe54 (n,p) mn54 Cd	2.100E+00	7.100E+00	0.9663	-3.37
fe56 (n,p) mn56 Cd	5.400E+00	1.140E+01	1.1115	11.15
co59 (n,g) co60 Cd	7.600E-05	1.500E-04	0.7600	-24.00
zn64 (n,p) cu64 Cd	2.300E+00	7.000E+00	0.9267	-7.33
zr90 (n,2n) zr89 Cd	1.270E+01	1.750E+01	1.1713	17.13
au197(n,g) au198 Cd	4.500E-06	6.600E-03	0.9862	-1.38
au197(n,g) au198	4.500E-06	6.600E-03	0.9310	-6.90

u235 (n,f) fp	B4CCd	2.300E-03	2.900E+00	1.0323	3.23
u238 (n,f) fp	B4CCd	1.400E+00	5.700E+00	1.0838	8.38
np237(n,f) fp	B4CCd	5.250E-01	4.400E+00	0.9910	-0.90
pu239(n,f) fp	B4CCd	4.750E-03	3.300E+00	0.9438	-5.62

standard deviation of measured activities (percent) 9.48

loop	1	iteration	0	average total flux (above 3.00E+00 mev)	3.390E+13
				9.4812	
loop	1	iteration	1	4.8170	
loop	1	iteration	2	4.3540	
loop	1	iteration	3	4.1248	

solution has been achieved.

standard deviation of measured-to-calculated activity ratios is less than 4.00percent./

solution results obtained after 4 iterations

foil reaction	measured activity (dps/nucleus)	calculated activity (dps/nucleus)	nominal 5.00 percent activity limits (mev)		ratio measured to calculated activities	deviation of measured from calculated activity (percent)		

			lower	upper				
s32 (n,p) p32	5.845E-18	5.871E-18	2.200E+00	7.000E+00	0.9956	-0.4388		
ni58 (n,p) co58 Cd	2.046E-18	2.003E-18	1.700E+00	7.100E+00	1.022	2.164		
na23 (n,g) na24 Cd	3.144E-17	3.221E-17	1.150E-05	2.000E-01	0.9762	-2.376		
na23 (n,g) na24	3.263E-17	3.252E-17	1.050E-05	2.000E-01	1.003	0.3299		
mg24 (n,p) na24 Cd	2.363E-18	2.458E-18	6.500E+00	1.180E+01	0.9615	-3.851		
al27 (n,a) na24 Cd	1.114E-18	1.157E-18	6.500E+00	1.220E+01	0.9629	-3.714		

si-28 (n,lmev) x	3.736E+14	3.623E+14	1.900E-01	4.500E+00	1.031	3.105
sc45 (n,g) sc46 Cd	4.414E-18	4.243E-18	3.000E-06	2.400E-01	1.040	4.029
tinat (n,x) sc46 Cd	1.485E-19	1.387E-19	3.700E+00	9.300E+00	1.070	7.033
ti48 (n,p) sc48 Cd	1.601E-19	1.526E-19	5.900E+00	1.250E+01	1.049	4.927
tinat (n,x) sc47 Cd	7.183E-18	7.657E-18	1.500E+00	7.100E+00	0.9381	-6.185
mn55 (n,g) mn56 Cd	9.575E-15	9.607E-15	4.000E-05	7.600E-03	0.9967	-0.3325
fe54 (n,p) mn54 Cd	3.237E-19	3.317E-19	2.100E+00	7.200E+00	0.9758	-2.419
fe56 (n,p) mn56 Cd	1.004E-17	9.849E-18	5.400E+00	1.150E+01	1.019	1.935
co59 (n,g) co60 Cd	1.885E-18	1.891E-18	4.750E-05	1.600E-04	0.9968	-0.3209
zn64 (n,p) cu64 Cd	8.777E-17	9.326E-17	2.300E+00	7.200E+00	0.9412	-5.883
zr90 (n,2n) zr89 Cd	3.569E-20	3.575E-20	1.270E+01	1.750E+01	0.9984	-0.1635
au197(n,g) au198 Cd	9.208E-15	8.971E-15	4.500E-06	8.000E-03	1.026	2.640
au197(n,g) au198	8.715E-15	8.994E-15	4.500E-06	8.000E-03	0.9690	-3.097
u235 (n,f) fp B4CCd	8.218E-10	8.018E-10	2.550E-03	2.900E+00	1.025	2.491
u238 (n,f) fp B4CCd	6.691E-11	6.207E-11	1.400E+00	5.900E+00	1.078	7.793
np237(n,f) fp B4CCd	3.608E-10	3.660E-10	5.250E-01	4.500E+00	0.9857	-1.432
pu239(n,f) fp B4CCd	8.472E-10	9.035E-10	5.500E-03	3.300E+00	0.9377	-6.234

standard deviation of measured activities (percent) 3.966

response analysis - loop = 1 iteration = 4

response function	dose
fluence (n/cm**2)	0.6832604E+15
kerma (rad-tissue)	955125.2
cp dose to elem-57 (rad-tissue)	1056944.
dose eqv. (rem)	0.1158082E+08
dose to elem-57 from h-1(n,g)h-2 (rad-tissue)	198960.2
ansi and ncrp dose (rem)	0.1194448E+08

spectrum characterization

total fluence (n/cm**2)	6.8326004E+14	
fluence > 10 kev	5.7458140E+14	(0.8409410)
fluence > 3 Mev	3.3100392E+13	(4.8444796E-02)
fluence < 1 eV	1.5495437E+11	(2.2678681E-04)
avg. energy (mev)	0.7330868	
HP-ASTM E722-85	0.5689850	
(ASTM Si damage function)		
HP-ASTM E722-93	0.5297971	
(ENDF-6 Si damage function)		
HP-10kev E722-85	0.6766051	
(ASTM Si damage function)		
HP-10kev E722-93	0.6300051	
(ENDF-6 Si damage function)		
SP	17.35875	
(F[>10 kev]/F[>3 MeV])		
SP	20.64205	
(F[tot]/F[>3 MeV])		
sulfur cross section (mb)	315.2602	
(spectrum averaged, > 3 mev)		
nickel cross section (mb)	534.4255	
(spectrum averaged, > 3 mev)		
sulfur cross section (mb)	15.27271	
(spectrum averaged, > total)		
nickel cross section (mb)	25.89014	
(spectrum averaged, total)		
gaas 1 mev fluence	4.0157240E+14	(0.5877299)
(ref-sig[gaas] = 70 mev-mb)		
silicon 1 mev fluence	3.6234814E+14	(0.5303224)
(ref-sig[si] = 95 mev-mb)		
old ASTM Si 1 mev flu.	3.8876470E+14	(0.5689850)
(ref-sig[si] = 95 mev-mb)		
S32 Act. -dps/gm meas-ENDF	7127252.	(1.0431243E-08)
S32 Act. - Bq/atom meas.	5.8449997E-18	(8.5545756E-33)
S32 Act. - Bq/atom calc.	5.8707622E-18	(8.5922811E-33)
252-Cf Eq. Flu. S32 Meas.-n/cm2	1.4790445E+14	
252-Cf Eq. Flu. S32 Calc.-n/cm2	1.4855636E+14	

Ni58 Act. - Bq/atom meas. 2.0460000E-18 (2.9944675E-33)
Ni58 Act. - Bq/atom calc. 2.0026680E-18 (2.9310480E-33)

Figure A-4: TPB13 Differential Number Distribution

SNL-SAND-II Run 5:

Environment:	NIST ^{252}Cf Reference Benchmark Source
File Name:	CF252VI
Shot Date:	---
Facility Shot Number:	NIST
Run Conditions:	Variable run times to establish a $^{32}\text{S}(\text{n,p})^{31}\text{P}$ β^- counting calibration curve
Dosimetry Position:	Dosimetry results from opposite ends of the radial arms were averaged to compensate for any misalignment. Corrections are made for neutron scattering and dosimeter size.

*** snl 6522 endf/b-vi specialized version of sand-ii ***

control parameters: 0 0 4 1 0 0 0 0 0 1 0
spline knot control file : fission

Job name = cf252vi
Library = snlrml

1 cases
t
t special generation of cf252 spectrum (ENDF/B-VI source)
t dummy ni58p and s32p sensors to fill monitor foil measured
t activity integral quantities
t dummy au197g sensor to inhibit action of low energy extrapolation
t which occurs below the energy of lowest/highest sensor response
t
iteration time integrated
3 foils pjg
s32p 3.954-12
ni58p 1.306-12
au197g 2.213-11
spectrum low

**** SNL combined energy/flux input spectrum format invoked ****
Energy-Flux Listing

122 points

0.000000E+00	0.000000E+00	0.100000E-10	0.203300E-05	0.200000E-10	0.287500E-05	0.500000E-10	0.454600E-05
0.100000E-09	0.643000E-05	0.200000E-09	0.909300E-05	0.500000E-09	0.143800E-04	0.100000E-08	0.203300E-04
0.200000E-08	0.287500E-04	0.500000E-08	0.454600E-04	0.100000E-07	0.643000E-04	0.200000E-07	0.909300E-04
0.500000E-07	0.143800E-03	0.100000E-06	0.203300E-03	0.200000E-06	0.287500E-03	0.500000E-06	0.454600E-03
0.100000E-05	0.643000E-03	0.200000E-05	0.909300E-03	0.500000E-05	0.143800E-02	0.100000E-04	0.203300E-02
0.200000E-04	0.287500E-02	0.500000E-04	0.454600E-02	0.100000E-03	0.642900E-02	0.200000E-03	0.909200E-02

0.500000E-03	0.143700E-01	0.100000E-02	0.203200E-01	0.200000E-02	0.287100E-01	0.500000E-02	0.453000E-01
0.100000E-01	0.638500E-01	0.250000E-01	0.998900E-01	0.300000E-01	0.109000E+00	0.350000E-01	0.117400E+00
0.450000E-01	0.132200E+00	0.550000E-01	0.145200E+00	0.700000E-01	0.162200E+00	0.850000E-01	0.176900E+00
0.100000E+00	0.190000E+00	0.130000E+00	0.212300E+00	0.160000E+00	0.230800E+00	0.200000E+00	0.251200E+00
0.250000E+00	0.271600E+00	0.300000E+00	0.287700E+00	0.370000E+00	0.304900E+00	0.460000E+00	0.320100E+00
0.500000E+00	0.324800E+00	0.600000E+00	0.332900E+00	0.700000E+00	0.336300E+00	0.850000E+00	0.335200E+00
0.100000E+01	0.328900E+00	0.120000E+01	0.315200E+00	0.150000E+01	0.287500E+00	0.200000E+01	0.235200E+00
0.240000E+01	0.195000E+00	0.270000E+01	0.167700E+00	0.300000E+01	0.143100E+00	0.330000E+01	0.121300E+00
0.350000E+01	0.108200E+00	0.370000E+01	0.963600E-01	0.390000E+01	0.855900E-01	0.410000E+01	0.758800E-01
0.430000E+01	0.671600E-01	0.450000E+01	0.593700E-01	0.470000E+01	0.524200E-01	0.490000E+01	0.462500E-01
0.510000E+01	0.407800E-01	0.530000E+01	0.359200E-01	0.550000E+01	0.316300E-01	0.570000E+01	0.278200E-01
0.590000E+01	0.244600E-01	0.610000E+01	0.214900E-01	0.630000E+01	0.188700E-01	0.650000E+01	0.165500E-01
0.670000E+01	0.145100E-01	0.690000E+01	0.127200E-01	0.710000E+01	0.111400E-01	0.730000E+01	0.975500E-02
0.750000E+01	0.854000E-02	0.770000E+01	0.747500E-02	0.790000E+01	0.654100E-02	0.810000E+01	0.572500E-02
0.830000E+01	0.501100E-02	0.850000E+01	0.438500E-02	0.870000E+01	0.383800E-02	0.890000E+01	0.336000E-02
0.910000E+01	0.294000E-02	0.930000E+01	0.257300E-02	0.950000E+01	0.225200E-02	0.970000E+01	0.197000E-02
0.990000E+01	0.172300E-02	0.101000E+02	0.150600E-02	0.103000E+02	0.131700E-02	0.105000E+02	0.115100E-02
0.107000E+02	0.100700E-02	0.109000E+02	0.880200E-03	0.111000E+02	0.769700E-03	0.113000E+02	0.672900E-03
0.115000E+02	0.588300E-03	0.117000E+02	0.514200E-03	0.119000E+02	0.449400E-03	0.121000E+02	0.392700E-03
0.123000E+02	0.343000E-03	0.125000E+02	0.299600E-03	0.127000E+02	0.261700E-03	0.129000E+02	0.228400E-03
0.131000E+02	0.199400E-03	0.132000E+02	0.186200E-03	0.134000E+02	0.162500E-03	0.135000E+02	0.151700E-03
0.137000E+02	0.132300E-03	0.139000E+02	0.115400E-03	0.141000E+02	0.100700E-03	0.142000E+02	0.939900E-04
0.144000E+02	0.819500E-04	0.146000E+02	0.714500E-04	0.153000E+02	0.442000E-04	0.160000E+02	0.273300E-04
0.167000E+02	0.168800E-04	0.174000E+02	0.104200E-04	0.181000E+02	0.642200E-05	0.188000E+02	0.395000E-05
0.195000E+02	0.242700E-05	0.200000E+02	0.171300E-05				

limit 1
 deviation 5.0
 discard 3.0
 low end e
 high end fission
 norm 0.010
 plot cards
 smooth 5

interim results after 0 iterations

nominal

deviation

foil reaction	5.00 percent activity limits (mev)		ratio measured to calculated activities	of measured from calculated activity (percent)
	lower	upper		
s32 (n,p) p32	2.400E+00	7.800E+00	1.0001	0.01
ni58 (n,p) co58	2.100E+00	7.900E+00	0.9999	-0.01
au197(n,g) au198	5.250E-02	2.800E+00	1.0000	0.00

standard deviation of measured activities (percent) 0.01

loop 1 iteration 0 average total flux (above 1.00E-02 mev) 1.000E+20
11.9011E-03

solution has been achieved.
standard deviation of measured-to-calculated activity ratios is less than 5.00percent./
solution results obtained after 1 iterations

foil reaction	measured activity (dps/nucleus)	calculated activity (dps/nucleus)	nominal 5.00 percent activity limits (mev)		ratio measured to calculated activities	deviation of measured from calculated activity (percent)
			lower	upper		
s32 (n,p) p32	3.954E-12	3.954E-12	2.400E+00	7.800E+00	1.000	11.206E-03
ni58 (n,p) co58	1.306E-12	1.306E-12	2.100E+00	7.900E+00	0.9999	-12.040E-03
au197(n,g) au198	2.213E-11	2.213E-11	5.250E-02	2.800E+00	1.000	83.447E-05

2 of 2

standard deviation of measured activities (percent) 11.645E-03

response analysis - loop = 1 iteration = 1

response function	dose
fluence (n/cm**2)	0.1000423E+21
kerma (rad-tissue)	0.2881116E+12
cp dose to elem-57 (rad-tissue)	0.3389861E+12
dose eqv. (rem)	0.3201368E+13
dose to elem-57 from h-1(n,g)h-2 (rad-tissue)	0.2137501E+11
ansi and ncrp dose (rem)	0.3354969E+13

spectrum characterization

total fluence (n/cm**2)	1.0004228E+20		
fluence > 10 kev	9.9999553E+19	(0.9995729)	
fluence > 3 Mev	2.3703971E+19	(0.2369395)	
fluence < 1 eV	0.	(0.)	
avg. energy (mev)	2.122686		
HP-ASTM E722-85	1.195567		
(ASTM Si damage function)			
HP-ASTM E722-93	1.057829		
(ENDF-6 Si damage function)			
HP-10kev E722-85	1.196078		
(ASTM Si damage function)			
HP-10kev E722-93	1.058281		
(ENDF-6 Si damage function)			
SP	4.218684		
(F[>10 kev]/F[>3 MeV])			

SP	4.220486	
(F[tot]/F[>3 MeV])		
sulfur cross section (mb)	296.4655	
(spectrum averaged, > 3 mev)		
nickel cross section (mb)	486.7026	
(spectrum averaged, > 3 mev)		
sulfur cross section (mb)	70.24439	
(spectrum averaged, > total)		
nickel cross section (mb)	115.3191	
(spectrum averaged, total)		
gaas 1 mev fluence	1.0596234E+20	(1.059176)
(ref-sig[gaas] = 70 mev-mb)		
silicon 1 mev fluence	1.0582797E+20	(1.057832)
(ref-sig[si] = 95 mev-mb)		
old ASTM Si 1 mev flu.	1.1960721E+20	(1.195567)
(ref-sig[si] = 95 mev-mb)		
S32 Act. -dps/gm meas-ENDF	4.8214122E+12	(4.8193748E-08)
S32 Act. - Bq/atom meas.	3.9539999E-12	(3.9523291E-32)
S32 Act. - Bq/atom calc.	3.9535567E-12	(3.9518859E-32)
252-Cf Eq. Flu. S32 Meas.-n/cm2	1.0005376E+20	
252-Cf Eq. Flu. S32 Calc.-n/cm2	1.0004242E+20	
Ni58 Act. - Bq/atom meas.	1.3060000E-12	(1.3054482E-32)
Ni58 Act. - Bq/atom calc.	1.3061573E-12	(1.3056054E-32)

Figure A-5: CF252VI Differential Number Distribution

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