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MCNP6 Fission Multiplicity

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Abstract
With the merger of MCNPX and MCNP5 into MCNP6, MCNP6 now provides all the capabilities of both codes allowing the user to access all the fission multiplicity data sets. Detailed in this paper is: (1) the new FMULT card capabilities for accessing these different data sets; (2) benchmark calculations, as compared to experiment, detailing the results of selecting these separate data sets for thermal neutron induced fission on U-235.

LA-UR 12-

MCNP6 Fission Multiplicity with FMULT Card

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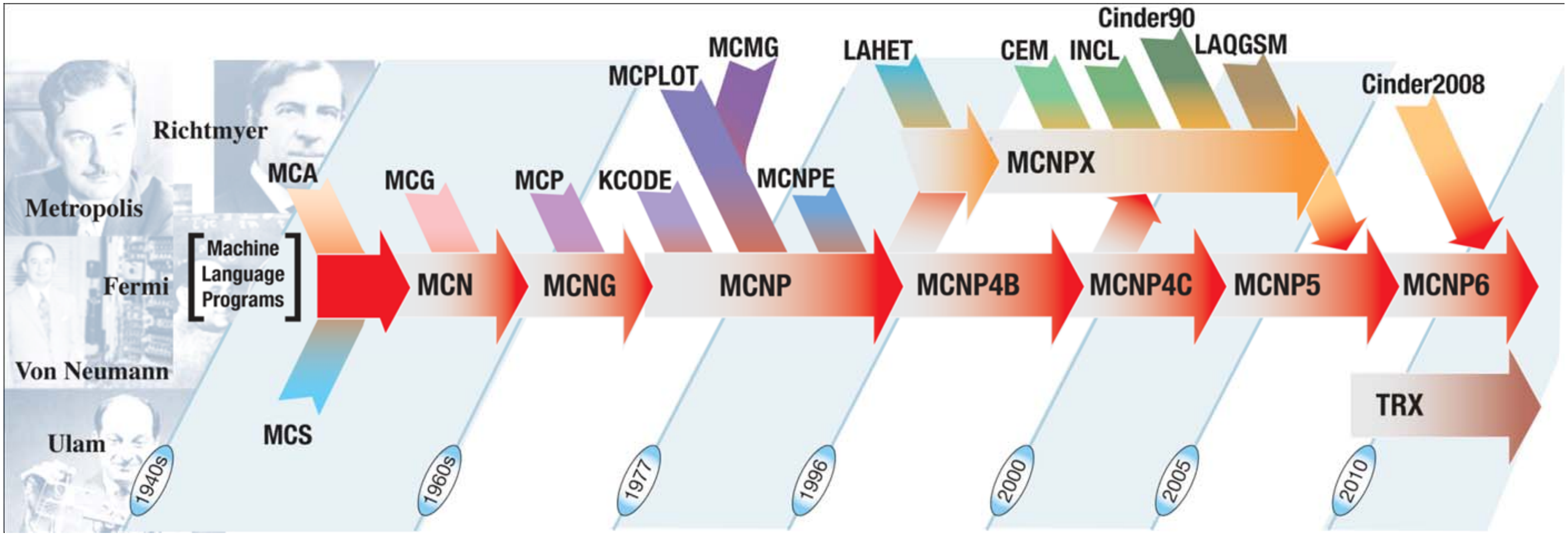
Outline

- **MCNP**
- **MCNPX**
  - PHYS:N 6th entry Nov. 2002
  - FMULT Aug. 2004
  - LLNL Jan 2010
- **MCNP5**
  - PHYS:N 5th entry Oct. 2005
- **MCNP6**
- **Some Details**
- **Test case**
- **Summary**

MCNP



- **Originally MCNP used binary integer sampling centered around nubar**
- **For nubar of 1.7**
  - One neutron is sampled 30% of the time
  - Two neutrons are sampled 70% of the time

MCNPX

Spontaneous fission (SF) capability PAR=SF

- SF multiplicity uses Santi and Ensslin data
- Eighteen nuclides available for spontaneous fission
- Fissioning nuclide chosen proportionately to (atom fraction) * (SF yield)
- The 6th entry on the PHYS:N card automatically set to FISM=-1 for induced fission

Induce fission multiplicity

- Uses Ensslin data to create the Gaussian
  - PHYS:N 6Th entry FISM sets method for sampling the number of neutrons from fission
    - FISM=-1 same as FISM=1
    - FISM = 0 integer sampling used
    - FISM = 1 correct sampled nubar to preserve average (default)
    - FISM = 2 preserve multiplicity by increasing nubar threshold
    - FISM = 3 sample Gaussian without correction (tends to over predict)
    - FISM = 4 use integer sampling when using FMULT or PAR=SF
    - FISM = 5 LLNL fission model

MCNPX Continued

- **FISM = 5 LLNL capability (experimental data and models)**
  - Simulate neutron and gamma-ray distributions from fission reactions
    - Spontaneous
    - Neutron induced
    - Photon induced

J. Verbeke, C. Hagmann, D. Wright, "Simulation of Neutron and Gamma Ray Emission from Fission and Photofission", Lawrence Livermore National Laboratory, UCRL-AR-228518 (2010).

- **Spontaneous and induce fission multiplicity FMULT card**
  - Uses Santi and Ensslin or user specified data
    - FMULT zaid [KEYWORD=value(s) …]
      - zaid nuclide identifier
      - SFNU specify nubar for sampling SF multiplicity form a Gaussian
        provide a cumulative probability distribution of SF multiplicity (10 bins)
      - WIDTH Gaussian width for sampling nubar
      - SFYIELD SF yield (n/s-g)
      - WATT Watt energy spectrum parameters a and b

- **FMULT card can be used to specify SF source for isotopes with no transport cross sections**

MCNPX Continued

- **Default values for FMULT (Santi and Ensslin data)**

1fission multiplicity data. print table 38

zaid	width	watt1	watt2	yield	sfnu									
90232	1.079	.800000	4.00000	6.00E-08	2.140									
92232	1.079	.892204	3.72278	1.30E+00	1.710									
92233	1.041	.854803	4.03210	8.60E-04	1.760									
92234	1.079	.771241	4.92449	5.02E-03	1.810									
92235	1.072	.774713	4.85231	2.99E-04	1.860									
92236	1.079	.735166	5.35746	5.49E-03	1.910									
92238	1.230	.648318	6.81057	1.36E-02	0.048	.297	.722	.950	.993	1.00	1.00	1.00	1.00	1.
93237	1.079	.833438	4.24147	1.14E-04	2.050									
94236	0.000	.000000	0.00000	0.00E+00	0.080	.293	.670	.905	.980	1.00	1.00	1.00	1.00	1.
94238	1.115	.847833	4.16933	2.59E+03	0.056	.267	.647	.869	.974	1.00	1.00	1.00	1.00	1.
94239	1.140	.885247	3.80269	2.18E-02	2.160									
94240	1.109	.794930	4.68927	1.02E+03	0.063	.295	.628	.881	.980	.998	1.00	1.00	1.00	1.
94241	1.079	.842472	4.15150	5.00E-02	2.250									
94242	1.069	.819150	4.36668	1.72E+03	0.068	.297	.631	.879	.979	.997	1.00	1.00	1.00	1.
95241	1.079	.933020	3.46195	1.18E+00	3.220									
* 96242	1.053	.887353	3.89176	2.10E+07	0.021	.168	.495	.822	.959	.996	.999	1.00	1.00	1.
96244	1.036	.902523	3.72033	1.08E+07	0.015	.131	.431	.764	.948	.991	1.00	1.00	1.00	1.
96246	0.000	.000000	0.00000	0.00E+00	0.015	.091	.354	.699	.917	.993	1.00	1.00	1.00	1.
96248	0.000	.000000	0.00000	0.00E+00	0.007	.066	.287	.638	.892	.982	.998	1.00	1.00	1.
97249	1.079	.891281	3.79405	1.00E+05	3.400									
98246	0.000	.000000	0.00000	0.00E+00	0.001	.114	.349	.623	.844	.970	1.00	1.00	1.00	1.
98250	0.000	.000000	0.00000	0.00E+00	0.004	.040	.208	.502	.801	.946	.993	.997	1.00	1.
98252	1.207	1.180000	1.03419	2.34E+12	0.002	.028	.153	.427	.733	.918	.984	.998	1.00	1.
98254	0.000	.000000	0.00000	0.00E+00	0.000	.019	.132	.396	.714	.908	.983	.998	1.00	1.
100257	0.000	.000000	0.00000	0.00E+00	0.021	.073	.190	.390	.652	.853	.959	.993	1.00	1.
102252	0.000	.000000	0.00000	0.00E+00	0.057	.115	.207	.351	.534	.717	.863	.959	.997	1.

* = used in problem.

MCNP5

- **No direct spontaneous fission option**
  - Uses Watt Fission Spectrum
  - PAR=n
  - EGR set to SP -3 with bias a b
    - Watt spectrum
  - Default a=0.965 b=2.29

	a(MeV)	b(MeV^{-1})
240Pu	0.799	4.903
242Pu	0.833668	4.431658
242Cm	0.891	4.046
244Cm	0.906	3.848
252Cf	1.025	2.926

Data from T-16 LANL

- **Focus is on induced fission multiplicity**
  - Use Terrell or Lestone data to create the Gaussian
    - PHYS:N 5Th entry FISNU method for sampling the number of neutrons from fission
      - FISNU <0 Terrell data and sets Gaussian width
      - FISNU =0 integer sampling (default)
      - FISNU =1 Lestone data
      - FISNU =2 Terrell data
      - FISNU >2 gives fatal error (if fatal option is on, Terrell data is used)
      - non-zero not allowed in a KCODE or adjoint multigroup calculation

MCNP5 Continued

- **PHYS:N 5Th Entry FISNU=1**
  - Lestone data is used with the following widths

Isotope	Width
233U	1.070
235U	1.088
238U	1.116
239Pu	1.140
241Pu	1.150

  - For isotopes not listed

    width = 0.007338*(na+1.0)-0.64

    na = atomic weight of fissioning isotope

- **PHYS:N 5Th Entry FISNU=2**
  - Terrell data is used with the following widths

Isotope	Width
233U	1.041
235U	1.072
238U	1.230
236Pu	1.110
238Pu	1.115
239Pu	1.140
240Pu	1.109
242Pu	1.069
242Cm	1.053
244Cm	1.036
252Cf	1.207

  - For isotopes not listed

    width = 1.079

MCNP6

- **MCNP5 PHYS:N 5th entry and MCNPX PHYS:N 6th entry have been moved to the FMULT card using new keywords**
  - FMULT zaid [KEYWORD=value(s) …]
    - zaid nuclide identifier
    - SFNU specify nubar for sampling SF multiplicity form a Gaussian
      provide a cumulative probability distribution of SF multiplicity (10 bins)
    - WIDTH Gaussian width for sampling nubar
    - SFYIELD SF yield (n/s-g)
    - WATT Watt energy spectrum parameters a and b
    - METHOD
    - DATA
    - SHIFT

New Keywords

MCNP6 Continued

- **METHOD determines Gaussian sampling algorithm**
  - METHOD = 0 sine cosine method
  - 0 < METHOD < 5 polar method
  - METHOD = 5 LLNL fission model
- **SHIFT sets how nu is changed to preserve nubar**
  - SHIFT = 0 uses MCNP5 method
  - SHIFT = 1 corrects sampled nubar to preserve average multiplicity
  - SHIFT = 2 increases nubar threshold to preserve average multiplicity
  - SHIFT = 3 samples the Gaussian without correction (tends to over predict)
  - SHIFT = 4 uses integer sampling

MCNP6 Continued

- **DATA determines data used to construct Gaussians**
  - DATA = 1 Lestone data
  - DATA = 2 Terrell data
  - DATA = 3 Ensslin data
    - DATA = 0 will be reset to DATA = 1
    - Lestone data adjusts Gaussian widths to fit 2nd and 3rd measured fission multiplicity moments
    - The Ensslin data is loaded by default
      - Portions of the Ensslin data are over written by Lestone or Terrell
- **If FMULT card is present but METHOD, SHIFT, or DATA are not**
  - The unspecified key words default to
    - METHOD = 3 polar method
    - SHIFT = 1 corrects sampled nubar to preserve average multiplicity
    - DATA = 3 Ensslin data
- **If METHOD = 5 LLNL fission model is used and DATA and SHIFT are ignored**

MCNP6 Continued

- **FMULT Settings for reproducing MCNPX PHYS:N 6th entry or MCNP5 5th entry**

MCNPX phys:n 6th entry	MCNP6 FMULT setting	
0	method=0, shift=0, data=0	
-1,1	method=3, shift=1, data=3	Default
2	method=3, shift=2, data=3	
3	method=3, shift=3, data=3	
4	method=3, shift=4, data=3	
5	method=5	
MCNP5 phys:n 5th entry		
0	method=0, shift=0, data=0	
1	method=0, shift=0, data=1	
2	method=0, shift=0, data=2	

Some Details

- **METHOD = 0 sine cosine method**

$$X_1 = \left(-2\ln R_1\right)^{1/2}\cos 2\pi R_2$$
$$X_2 = \left(-2\ln R_1\right)^{1/2}\sin 2\pi R_2$$

- If R_1 and R_2 are independent random variables on (0,1)
  - Then X_1 and X_2 are independent random variables from the same normal distribution
- MCNP6 use half of the method

$$N(w) = w\left(-2\ln R_1\right)^{1/2}\cos 2\pi R_2$$

  - w is the Gaussian width
- The sampled number of neutrons are

$$\nu_x = N(w) + \overline{\nu}$$

- The number of neutrons emitted from a fission event is

$$\nu = \nu_x - \nu_{shift}$$

Some Details Continued

- **0 < METHOD < 5 polar method**

$$U_1 = 2R_1 - 1$$
$$U_2 = U_1^2 + R_2^2$$
$$N(w) = wU_1\left(-2\ln(U_2)/U_2\right)^{1/2}$$

  - Where R_1 and R_2 are independent random variables on (0,1) and w is the Gaussian width
  - The sampled number of neutrons are

$$\nu_x = N(w) + \overline{\nu}$$

  - The number of neutrons emitted from a fission event is

$$\nu = \nu_x - \nu_{shift}$$

- **Details for ν_{shift} (SHIFT) can be found in**
  - J. S. Hendricks, “Monte Carlo Sampling of Fission Multiplicity”, Monte Carlo 2005 Topical Meeting.

Test Case

```
U-235 FMULT test case for mcnp6
c CELL CARDS
1000 1 -1 -101 imp:n=1
1001 0      101 imp:n=0

c SURFACE CARD
101 sph 0  0  0  0.01

c DATA CARDS
m1 92235.70c 1
sdef par=n erg=2.53e-8
print
nps 1e8
fmult method=0 shift=0 data=0
```

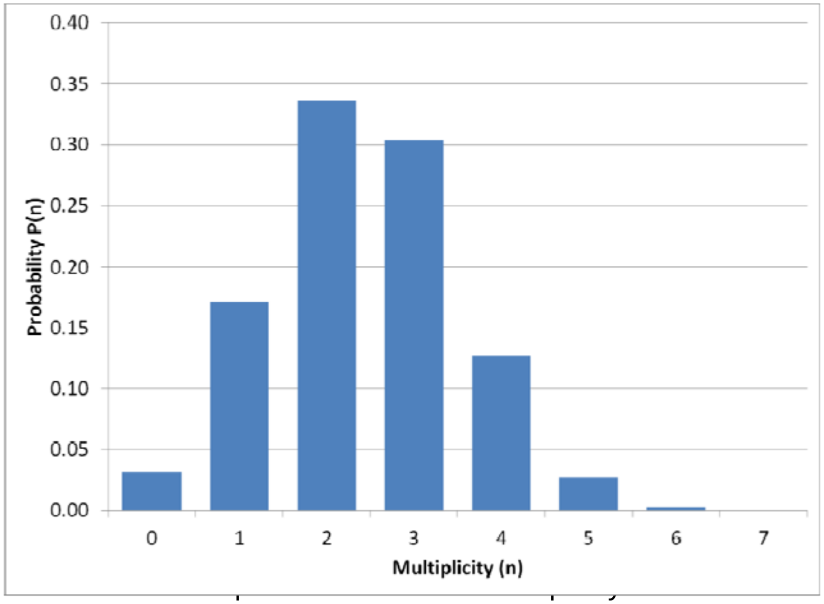
- **METHOD=0 SHIFT=0 DATA=1 (001)**
- **METHOD=0 SHIFT=0 DATA=2 (002)**
- **METHOD=3 SHIFT=1 DATA=3 (313)**
- **METHOD=3 SHIFT=2 DATA=3 (323)**
- **METHOD=3 SHIFT=3 DATA=3 (333)**
- **METHOD=5 SHIFT=0 DATA=0 (500)**

Test Case Experimental Data

- **N. E. Holden, M. S. Zucker, "A Reevaluation of the Average Prompt Neutron Emission Multiplicity (Nubar) Values from Fission of Uranium and Transuranium Nuclides", Brookhaven National Laboratory, BNL-NCS-35513**

Thermal neutrons on U-235

	Gwinn '84	Boldeman '84	Avg.	Std. Dev.
P(0)	0.0306776	0.0327670	0.0317223	0.00148
P(1)	0.1707282	0.1726860	0.1717071	0.00138
P(2)	0.3384084	0.3339897	0.3361991	0.00312
P(3)	0.3036614	0.3042775	0.3039695	0.00044
P(4)	0.1295224	0.1243698	0.1269461	0.00364
P(5)	0.0248181	0.0285404	0.0266793	0.00263
P(6)	0.0019968	0.0032675	0.0026322	0.00090
P(7)	0.0001872	0.0001206	0.0001539	0.00005
nubar	2.414	2.414	2.414	
nu(nu-1)	4.6172	4.6592	4.638	0.0297
nu(nu-1)/2!	2.3086	2.3296	2.319	
nu(nu-1)(nu-2)	6.6985	6.9366	6.8176	0.1683
nu(nu-1)(nu-2)/3!	1.1164	1.1561	1.1363	



Test Case Results

- **Percent Error** $100(Sim - Exp)/Exp$

Thermal neutrons on U-235

(MSD)	001	002	313	323	333	500
P(0)	19.3	12.5	28.7	41.2	27.7	7.1
P(1)	-8.4	-9.3	-7.2	-10.1	-7.6	-7.7
P(2)	-2.2	-1.0	-3.8	-3.9	-3.9	-0.9
P(3)	2.8	3.8	1.2	1.4	1.4	3.3
P(4)	6.4	5.0	8.3	8.8	8.8	4.9
P(5)	-1.7	-7.3	6.3	7.2	7.2	-6.2
P(6)	-11.4	-21.8	3.1	4.7	4.7	-19.1
P(7)	-41.8	-56.0	-23.0	-23.0	-23.0	-51.9
nubar	1.0	1.0	1.0	1.0	1.2	1.0
nu(nu-1)/2!	2.4	1.7	3.4	3.8	3.8	1.7
nu(nu-1)(nu-2)/3!	2.5	0.3	5.6	6.2	6.2	0.5

Summary

- MCNP6 contains the aggregate of multiplicity options from MCNPX and MCNP5
- Additional keywords allow finer control of multiplicity options
- PAR=SF is the preferred option for spontaneous fission
- Users should select METHOD, SHIFT, and DATA based on moments and multiplicities of interest
- 500 and 002 gives low errors for the 2nd and 3rd moments as well as the multiplicity centered around 3
- Other isotopes are currently being run and compared to experimental data

Questions

- **P. Santi, D. H. Beddingfield, D. R. Mayo, "Revised Prompt Neutrons Emission Multiplicity Distributions for ^{236,238}Pu", Nuclear Physics A 756 (2005) 325-332.**
- **N. Ensslin, et. al., "Application Guide to Neutron Multiplicity Counting", LA-13422-M (1998).**
- **J. P. Lestone, "Energy and Isotope Dependence of Neutron Multiplicity Distributions", LA-UR-05-0288 (2005).**
- **J. Terrell, "Distributions of Fission Neutron Numbers", *Phys. Rev.* 108, 783 (1957).**
- **J. S. Hendricks, "Monte Carlo Sampling of Fission Multiplicity", Monte Carlo 2005 Topical Meeting.**

Backup Slides Setting ifisnu

```
! integer variable ifisnu determines the fission multiplicity method to be used:
!
!   ifisnu = 1 selects improved Gaussian width to fit 2nd and 3rd
!            neutron multiplicity moments for all fissionable isotopes.
!      reference: John Lestone, X-5 research note.
!
!   ifisnu = 2 selects terrell data with Gaussian sampling.
!      reference: James Terrell, "Distribution of Fission Neutron Numbers,"
!               Phys. Rev. 108, 783 (1957).
```

```
case( 54 )
 ! >>>>> physics parameters                          phys
 if( nqp(1)/=0 ) then
  if( nwc==1 )  emx(1) = ritm
  if( nwc==2 )  emcf(1) = ritm
  if( nwc==3 .and. iitm/=0 )  iunr = 100000
  if( nwc==4 .and. nsr/=71 .and. ritm>=0.)  dnb = ritm
  if( nwc==4 .and. nsr==71 .and. ritm==0.)  dnb = ritm
  if( nwc==5 )  then
   if( nsr==71 ) then
    if( ritm/=0. )call erprnt(2,2,0,0,0,0,0,0,0, &
     & "'phys:n fission multiplicity 5th entry ignored in eigenvalue calculation.'")
   elseif( mcal==2 ) then
    if( ritm/=0. )call erprnt(2,2,0,0,0,0,0,0,0, &
     & "'phys:n fission multiplicity 5th entry ignored in adjoint calculation.'")
   else
     if( ritm<0. ) then
      ! use Gaussian nu sampling for user specified width.
      width_nu = -ritm
      ! warn the user of possible bad width values.
      if( width_nu<0.9 ) call erprnt(2,2,0,0,0,0,0,0,0, &
       & "'width for gaussian sampling of fission multiplicity appears small.'")
      if( width_nu>1.4 ) call erprnt(2,2,0,0,0,0,0,0,0, &
       & "'width for gaussian sampling of fission multiplicity appears large.'")
      ifisnu = 2  !  use Terrell data flag to sample width.
     elseif( ritm>0. ) then
      ifisnu = nint(ritm)
      if( ifisnu>mnudat ) then
       call erprnt(2,1,2,ifisnu,mnudat,0,0,0, &
        & "'5th phys:n entry requests nu set",i2,": only",i2," sets exist.'")
       ifisnu = 1  !  set ifisnu = 1 in case fatal option is used.
      endif
    endif
   endif
  endif
 elseif( nqp(2)/=0 ) the
```

Backup Slides Data

iacenus.F90

```
if( width_nu==zero ) then
  !  use reevaluation fitting first three factorial multiplicity moments
  if( ifisnu==1 ) then
    ! select fissionable isotope id to be sampled for number of fission neutrons.
    select case( id )
    case( 92233 )
      w = 1.070d0
    case( 92235 )
      w = 1.088d0
    case( 92238 )
      w = 1.116d0
    case( 94239 )
      w = 1.140d0
    case( 94241 )
      w = 1.150d0
    case default
      na = id-(id/1000)*1000  !  get the atomic weight of the fissioning isotope id.
      w = 0.007338d0*(na+1.)-0.64d0
    end select
  elseif( ifisnu==2 ) then  !  sample using the terrell width values.
    ! select fissionable isotope id to be sampled for number of fission neutrons.
    select case( id )
    case( 92235 )
      !  ignore energy-dependent (1.25 and 4.8 mev) widths because of large errors (0.06).
      w = 1.072d0  !  + or - 0.021  (80 kev)
    case( 92238 )
      w = 1.23d0   !  + or - 0.08  (1.5 mev)
    case( 94239 )
      w = 1.14d0   !  + or - 0.07   (80 kev)
    case( 94240 )
      w = 1.109d0  !  + or - 0.012
    case( 92233 )
      w = 1.041d0  !  + or - 0.041  (80 kev)
    case( 94236 )
      w = 1.11d0   !  + or - 0.11
    case( 94238 )
      w = 1.115d0  !  + or - 0.023
    case( 94242 )
      w = 1.069d0  !  + or - 0.035
    case( 96242 )
      w = 1.053d0  !  + or - 0.013
    case( 96244 )
      w = 1.036d0  !  + or - 0.018
    case( 98252 )
      w = 1.207d0  !  + or - 0.012
    case default
      w = 1.079d0  !  + or - 0.007
    end select
  endif
else
  ! use user input width for Gaussian sampling for all fissionable isotopes (b=0).
  w = width_nu
endif
```

Backup Slides SHIFT

```
real(DKND), parameter :: TV(NB) = &  !  values of (nubar+0.5)/width_nu.
  & (/ 1.9857_DKND, 2.0022_DKND, 2.0192_DKND, 2.0369_DKND, 2.0552_DKND, 2.0743_DKND, 2.0943_DKND, &
  &    2.1151_DKND, 2.1369_DKND, 2.1599_DKND, 2.1841_DKND, 2.2097_DKND, 2.2369_DKND, 2.2659_DKND, &
  &    2.2971_DKND, 2.3307_DKND, 2.3673_DKND, 2.4075_DKND, 2.4521_DKND, 2.5023_DKND, 2.5600_DKND, &
  &    2.6281_DKND, 2.7114_DKND, 2.8199_DKND, 2.9785_DKND, 3.2980_DKND, 3.3597_DKND, 3.4379_DKND, &
  &    3.5455_DKND, 3.7227_DKND /)

real(DKND), parameter :: BS(NB) = &  !  nu shift values for Gaussian multiplicity sampling.
  & (/ 0.0255_DKND, 0.0245_DKND, 0.0235_DKND, 0.0225_DKND, 0.0215_DKND, 0.0205_DKND, 0.0195_DKND, &
  &    0.0185_DKND, 0.0175_DKND, 0.0165_DKND, 0.0155_DKND, 0.0145_DKND, 0.0135_DKND, 0.0125_DKND, &
  &    0.0115_DKND, 0.0105_DKND, 0.0095_DKND, 0.0085_DKND, 0.0075_DKND, 0.0065_DKND, 0.0055_DKND, &
  &    0.0045_DKND, 0.0035_DKND, 0.0025_DKND, 0.0015_DKND, 0.0005_DKND, 0.0004_DKND, 0.0003_DKND, &
  &    0.0002_DKND, 0.0001_DKND /)

real(dknd), parameter :: c1(31) = & !  nu shift by MCNPX method 1
   & (/1.00528E-1_DKND, 8.02249E-2_DKND, 6.38562E-2_DKND, 5.06562E-2_DKND, 4.00222E-2_DKND, &
   &   3.14746E-2_DKND, 2.46252E-2_DKND, 1.91590E-2_DKND, 1.48167E-2_DKND, 1.13856E-2_DKND, &
   &   8.69035E-3_DKND, 6.58690E-3_DKND, 4.95634E-3_DKND, 3.70147E-3_DKND, 2.74301E-3_DKND, &
   &   2.01670E-3_DKND, 1.47075E-3_DKND, 1.06379E-3_DKND, 7.63034E-4_DKND, 5.42685E-4_DKND, &
   &   3.82671E-4_DKND, 2.67508E-4_DKND, 1.85374E-4_DKND, 1.27331E-4_DKND, 8.66889E-5_DKND, &
   &   5.84945E-5_DKND, 3.91172E-5_DKND, 2.59241E-5_DKND, 1.70256E-5_DKND, 1.10803E-5_DKND, &
   &   7.14548E-6_DKND /)

real(dknd), parameter :: c2(31) = & !  nu shift by MCNPX method 2
   & (/ .302631_DKND, .443487_DKND, .579851_DKND, .712453_DKND, .841865_DKND,    &
   &    .968544_DKND, 1.09286_DKND, 1.21512_DKND, 1.33556_DKND, 1.45442_DKND,    &
   &    1.57186_DKND, 1.68803_DKND, 1.80308_DKND, 1.91710_DKND, 2.03021_DKND,    &
   &    2.14248_DKND, 2.25400_DKND, 2.36483_DKND, 2.47502_DKND, 2.58463_DKND,    &
   &    2.69372_DKND, 2.80231_DKND, 2.91045_DKND, 3.01818_DKND, 3.12552_DKND,    &
   &    3.23249_DKND, 3.33914_DKND, 3.44547_DKND, 3.55151_DKND, 3.65728_DKND,    &
   &    3.76280_DKND /)
```

Backup Slides SHIFT Continued

```
case(0)
  ! Shift 0 - MCNP5
  vb = ( nubar + HALF ) / width_nu
  if(    vb <= TV(1) ) then
    ! below the table
    ! being below the table will cause a slight over prediction of nu-bar.
    nubar_shift = BS(1)
  elseif( vb <= TV(NB) ) then
    ! within the table
    do i = 2, NB-1
      if( vb <= TV(i) ) then
        exit
      endif
    enddo
    nubar_shift = BS(i-1) + ( BS(i) - BS(i-1) ) *
     &              ( vb - TV(i-1) ) / ( TV(i) - TV(i-1) )
  endif

 case(1)
  ! Shift1 - MCNPX method 1 - shift nubar.
  x = nubar/width_nu
  j = int(ten*x)-9
  if( j <= 30 ) then
    nubar_shift = (c1(j)+(c1(j+1)-c1(j))*(ten*x-j-9))*width_nu
  endif

case(2)
  ! Shift 2 - MCNPX method 2 - increase cutoff above nu = 0 to compensate
for nu < 0.
  if( xnu < nubar-c2(1)*width_nu ) then
    x = nubar / width_nu
    j = int(TEN*x) - 9
    if( j <= 30 ) then
      c = c2(j) + ( c2(j+1) - c2(j) ) * ( TEN*x - j - 9)
      if( xnu < nubar - c*width_nu ) then
        nubar_shift = xnu + nubar
      endif
    endif
  endif

 case(3)
  ! no correction - MCNPX method 3
  continue

 case (4)
  ! no fission multiplicity: MCNP4C method.
  nubar_shift = xnu - nubar
```