

MONTE CARLO SIMULATION OF CADMIUM
CAPTURE-GATED NEUTRON DETECTOR

by

Christopher Lorin Busselberg

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DEPARTMENT APPROVAL

of a senior thesis submitted by

Christopher Lorin Busselberg

This thesis has been reviewed by the research advisor, research coordinator,
and department chair and has been found to be satisfactory.

Date

Lawrence Rees, Advisor

Date

Eric Hintz, Research Coordinator

Date

Ross L. Spencer, Chair

ABSTRACT

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Christopher Lorin Busselberg

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After 9/11/2001, the increase of security measures has made an important role for neutron detectors. The installation of detectors to scan for radioactive material has caused a shortage of helium-3. We have been working on making a cadmium capture-gated detector as a replacement for helium-3 detectors. Through using MCNP, a Monte Carlo simulation software, I have been working at optimizing the capture efficiency of the cadmium detector. By varying plastic scintillator and cadmium sheet thicknesses I have shown that doubling the thickness of the cadmium sheets has almost no effect on the efficiency. Variances in the plastic thickness have a much greater effect.

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Chapter 1

Introduction

1.1 Neutron Detectors

The purpose of a neutron detector is to detect radiation. Neutron detectors have important uses in nuclear reactors both in radiation safety and measuring power output of the reactor. They are used in linear accelerators both for safety and measuring results of experiments. Since the terrorist attack on September 11, 2001, they have played an even bigger role in the detection of special nuclear materials like uranium-233 and plutonium-239.

The U.S. government has been installing neutron detectors at sea ports, airports, and border crossings to monitor all incoming traffic for nuclear materials [1]. These efforts are in the interest of Homeland Security. The types of detectors being installed use helium-3 to detect the neutrons. Since the Department of Homeland Security is using all currently produced helium-3 in border detectors, there is a massive shortage of the gas for commercial and research purposes (See Figure 1.1). The amount of helium-3 is scarce to begin with and production has gone down since the cold war (see Appendix A) [2]. Because of this shortage the Department of Energy and the

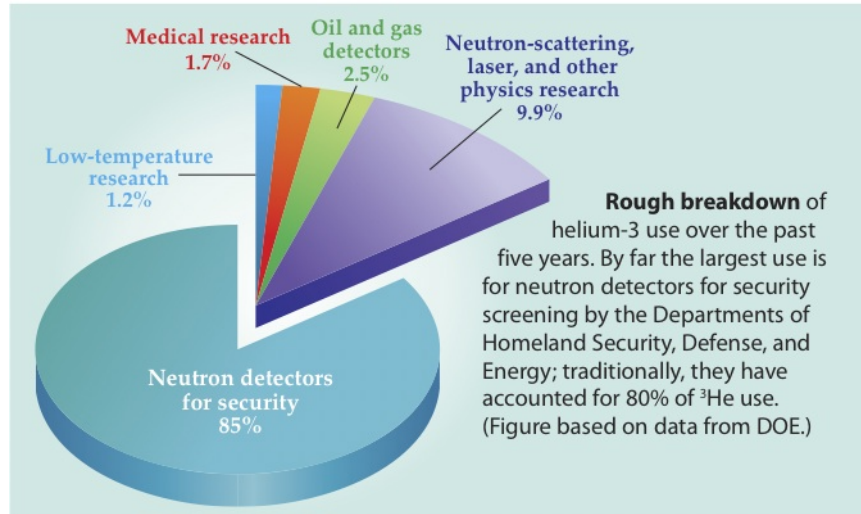


Figure 1.1 Helium-3 usage over the past five years [2]

Department of Homeland Security have been encouraging finding alternate methods of neutron detection.

Some research has been done into alternative methods already. Many groups have researched the effectiveness of lithium-6 glass and gadolinium-157 detectors [1]. Our group has been working on making and optimizing a cadmium-113 detector.

1.2 Monte Carlo and MCNP

Monte Carlo simulations have played an important role in optimizing our detector. They are named after the European gambling city. The Monte Carlo technique was developed because even with a computer it is impossible to consider every possible state or possibility of relatively simple systems. The method uses probabilities and random sampling instead to give an accurate picture of what can happen. Statistically the random samplings will most often demonstrate the most likely possibilities [3]. Due to the difference in experiments carried out by researchers, different areas of study use different versions of Monte Carlo simulations. Because of this answering

the question “what is a Monte Carlo simulation” difficult. However, random sampling is the one idea that each version of Monte Carlo has in common [4].

Monte Carlo simulations are necessary in our research. Since neutrons are neutral charge, the only way to detect the presence of one is if it either collides with an atom or gets captured by an atom. These collisions and captures excite the particles and eventually give off photons. It isn't the neutron itself but the photons from the collisions which can be picked up by a photomultiplier tube which tell us that a neutron was even present. This secondary information makes things like determining the efficiency impossible to determine. Experimentally optimizing the detector is almost impossible to do. Efficiency is determined by

$$Efficiency = N_c/N$$

where N is the total number of neutrons that enter a cell and N_c is the number of neutrons that got captured.

Experimental verification would not only be difficult to accomplish but expensive as well. Monte Carlo allows us to simulate an actual physical experiment and look at what most likely will happen. This information helps us to optimize the detector without actually building it. To optimize detectors we use an software called A General Monte Carlo N-Particle Transport Code, or MCNP.

MCNP is a program which can take a geometric three dimensional configuration of materials and simulate particle transport through them. Using probabilities from known cross sections it counts things like particle flux, particle collisions, and particle captures. The extensive output includes all of this information, including calculating the efficiency of neutron capture inside the detector [5]. This software enables us to accurately predict the effectiveness of different variations on our cadmium detector.

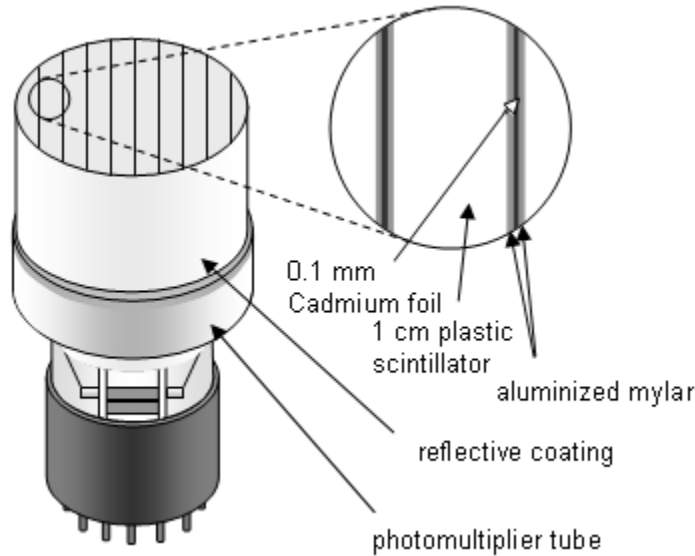


Figure 1.2 Cadmium Detector

1.3 Cadmium Capture-Gated Detector

Our detector, shown in Figure 1.2, is made from cadmium and plastic. The plastic is the scintillator, slowing down the neutrons. When the neutron collides with a hydrogen atom in the plastic, light is given off as the atom relaxes to a lower energy state. Each side of the cadmium is lined with aluminized Mylar to guide the light pulse down to a photomultiplier tube. The neutron is then captured in the cadmium-113 present between the plastic layers. The capture excites protons and neutrons to higher energy states and gives off another light pulse as they relax. These two light pulses are how we determine the presence of a neutron. If the first pulse is followed by the second in the right time interval, we know that a neutron was captured. This is also how we distinguish a neutron capture from a gamma ray. The gamma will only give one pulse, while a neutron capture always gives two.

My job was optimizing the detector. The detector was built with 1.0 cm thick plastic sheets and 0.01 cm thick cadmium sheets. I used MCNP to vary both cadmium thickness and plastic thickness to see how varying each changed the efficiency of

neutron capture inside the detector.

Chapter 2

Methods

The capture efficiency is one of the things that MCNP tells in its output. It records several different kinds of events as shown in Figure 2.1. It also records the total number of neutrons which enter a cell. MCNP uses all this information to calculate efficiencies. My responsibility is making the input files for MCNP to run. The input files need to be made as carefully as possible and as close to a real physical situation to get the most accurate results.

The first thing done when writing an input for MCNP is planning out specific geometries. The purpose of the simulation is to mimic what would actually happen if the detector were built. As such it is important to make the input as detailed as possible to imitate an actual detector in size and material. The input uses certain commands to “create” a 3D scenario. First surfaces of the detector, using planes and cylinders, need to be assigned. Each surface gets an arbitrary number assigned and no two surfaces can have the same number. Our simulations all use a cylindrical detector with an inside radius of 6.35 cm and a length of 12.7 cm. The surfaces which outline the plastic and cadmium layers are vertically oriented and run parallel to the axis of the cylinder.

Event Log

1. Neutron scatter, photon production
2. Fission, photon production
3. Neutron capture
4. Neutron leakage
5. Photon scatter
6. Photon leakage
7. Photon capture

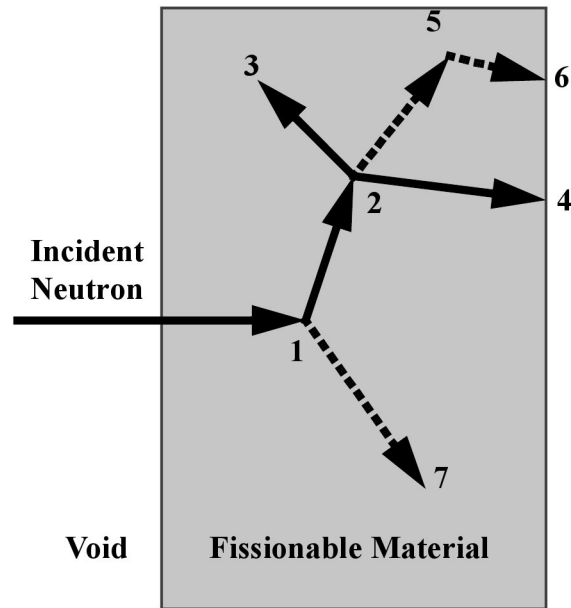


Figure 2.1 MCNP Event Log [5]

Once all the surfaces are defined where they need to be, the regions inside the surfaces need to be assigned cells. Each cell gets its own arbitrary number. When defining which surfaces enclose a cell, we assign a material number and a density to the cell. Like the size and spacing, the density needs to be as close to the density of the actual material as possible. All cells of the same material have the same material number and density. If the cell is empty we use “0” for the material number.

Once the cells and surfaces have been set, we define what material each material number is comprised of. Using a number called a Z A Identification number (or ZAID), the material’s atomic composition is set. The ZAID specifies both element and isotope of the element. Z is the standard atomic number of an element, and A specifies the isotope. If natural abundance of isotopes is desired then “000” is used for the last three numbers instead of a specific isotope [5]. I also specify how much of each element is in the material either by mass fraction or atomic ratios. For our detector we simulate layers made up of plastic and cadmium.

The next part of the input is defining a source for the neutrons which will strike the detector. A cell and surface is specified as the location for the source. Once a position is set a vector is set to show the direction that the source particles will be sent. An energy distribution is then specified to determine the type of source. For our purposes we placed a source with a distribution to imitate californium 0.1 cm away from and pointing towards the front of the detector.

The remainder of the input is specifying tallies, or counters. The tallies specify what needs to be kept track of. Different commands cause MCNP to record things like how many neutrons entered a cell, how many left, how many collisions there were, how many captures there were, and the overall capture efficiency. The efficiency we measure is how many neutrons were captured in a cell divided by the total number that entered. Once the tallies and run time are set the input is ready to be run through MCNP. Appendix B has an example of an input I made and ran. All inputs that I made for this have constant radius and length. They all use the californium source for the neutrons which enter.

Chapter 3

Results

3.1 Results and Discussion

When MCNP gets the input, it begins to run the simulation. When a neutron leaves the source, MCNP keeps track of when it hits a cell, and what cell it hits. The fact that it entered the cell is tallied. Then using the data tables and probabilities it will calculate whether the neutron gets captured, causes an atom to split, or bounces off a nucleus. The number of collisions in the cell is tallied as well as when the neutron leaves the cell, if it leaves the cell at all. The process is repeated for the next cell. If the neutron leaves the areas of interest it is no longer tracked. This happens over and over again for each neutron.

An output file is then generated and given by the program which tells all about what happened inside the object you described in the input. It details all the activity in each cell according to how you chose to keep tallies and which cells you specified interest in. Refer back to Figure 2.1 for a visual representation of simulated events.

The information we wish to obtain from the simulation is the efficiency of neutron capture of the detector. That means the percent of neutrons which entered which

Efficiencies with:	0.01 cm cadmium	0.02 cm cadmium	Number of cadmium sheets
0.5 cm plastic	21.64%	21.71%	24
1.0 cm plastic	20.37%	20.36%	12
1.5 cm plastic	18.2%	–	8
2.0 cm plastic	16.2%	–	6

Figure 3.1 Efficiencies and number of cadmium sheets from varying cadmium and plastic sheet thickness

were actually captured. To get this number, only capture efficiencies for cadmium cells are specified for tallying. The results for the first few runs are shown in the table in Figure 3.1

The first two inputs compared cadmium thickness in a 1.0 cm thick plastic detector. I ran the first with 0.01 cm thick cadmium sheets and the second with 0.02 cm thick cadmium sheets, doubling the cadmium sheet width. As you can see from the table, the efficiencies have negligible difference.

I ran the next two inputs with 0.5 cm thick plastic sheets and varied cadmium thickness in the same way. Again, the difference in efficiencies from changing cadmium thickness was negligible. But in comparing the 0.5 cm plastic efficiency to that of the 1.0 cm plastic input, the efficiencies differ by about 1.3%. As making thicker cadmium sheets didn't alter the efficiency there was no need to look into it further. The 0.5 cm plastic detector has twice as many cadmium sheets and only a 1.3% increase in efficiency compared to the 1.0 cm plastic one. That makes the 0.5 cm plastic detector much more expensive so we decided that 1.0 cm thick plastic is as thin as we want to go.

I made two more inputs to see how making thicker plastic would change efficiency. The first had 1.5 cm plastic and the second had 2.0 cm plastic. Both had 0.01 cm

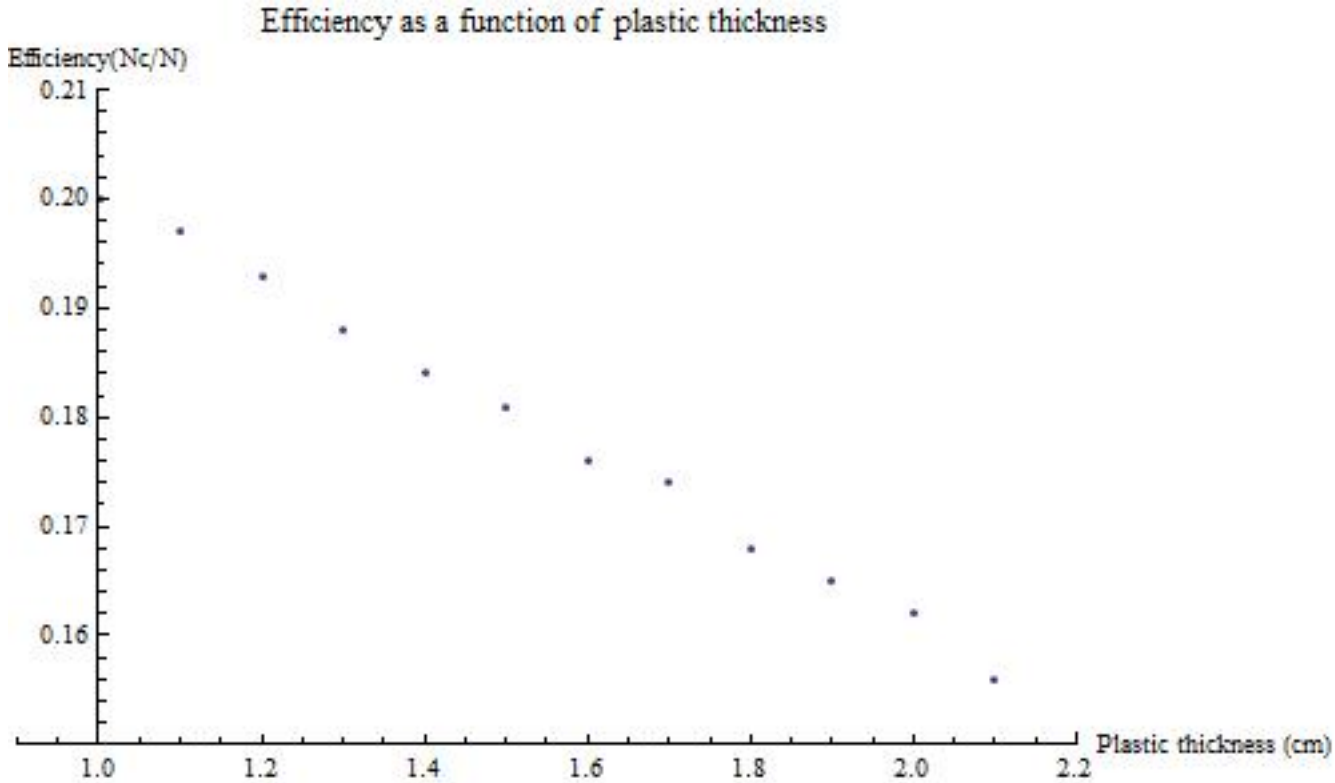


Figure 3.2 Efficiency vs. plastic thickness

thick cadmium. As shown in Figure 3.1, the efficiency dropped by about 2% each time the plastic thickness was increased by 0.5 cm. This seemed to be a linear relationship between plastic thickness and efficiency. To verify a linear relationship between 1.0 cm and 2.0 cm I ran a series of eleven inputs incrementing the plastic thickness of each consecutive input by 0.1 cm. Figure 3.2 is a plot of the efficiencies of thicknesses ranging from 1.0 cm to 2.1 cm.

Over the range of possible plastic thicknesses for our detector there is a linear relationship between efficiency and plastic thickness as shown by Figure 3.2. At first I thought that the reason that efficiency dropped was due to fewer sheets of cadmium inside the detector. Fewer sheets of cadmium means less total cadmium surface area exposed to neutrons. However, if that were a major factor there would be large jumps in efficiency where the differences in sheet numbers occur. The arrangements which

had the same number of cadmium sheets (for example 1.9 cm, 2.0 cm, and 2.1 cm thick plastic all have 6 sheets of cadmium) would have about the same efficiency. Instead the graph shows a relatively constant decline even when the number of cadmium sheets is constant. This suggests that the most important factor in our detector's arrangement is the thickness of the plastic between two cadmium sheets.

3.2 Conclusion

Based on the data we have from Monte Carlo we have decided that the optimal cadmium thickness is 0.01 cm. Making it thicker did not appreciably change the efficiency and 0.01 cm is the thinnest commercially available thickness. We also decided that the optimal plastic thickness is 1.0 cm. Using 0.5 cm thick plastic gave a 1.3% better efficiency but had twice as many cadmium sheets as the 1.0 cm thick detector. This not only makes it a more expensive detector but internal reflections of the light pulses play a part in the overall neutron detection efficiency of the detector. Half as thick plastic means twice as many reflections and intensity is lost with each reflection.

The total efficiency for neutron detection is determined by both capture efficiency and by how much light makes it from the collision and capture events to the photomultiplier tube. The thinner the plastic layers, the more capture efficient but less light reaches the photomultiplier tube per capture. The thicker the layers have smaller capture efficiency but fewer reflections so more light makes it to the photomultiplier tube. Future research will be balancing these two factors to maximize total detection efficiency.

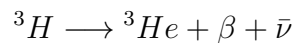
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Appendix A

Helium-3 Creation

Helium-3 is a product of tritium decay and comes from the following reaction:



Helium-3 is harvested when nuclear weapons are refurbished or dismantled. Tritium produces extra neutrons which boost the explosive power of plutonium. Since the cold war ended, the number of nuclear weapons available has gone down, reducing helium-3 production.

Appendix B

Sample MCNP Input Code

1.0cm Plastic scintillator, 0.01cm Cd sheets, no cover

```
1 1 -0.86 -1 -2 52 -53
2 2 -8.65 -1 2 -3 52 -53
3 1 -0.86 -1 3 -4 52 -53
4 2 -8.65 -1 4 -5 52 -53
5 1 -0.86 -1 5 -6 52 -53
6 2 -8.65 -1 6 -7 52 -53
7 1 -0.86 -1 7 -8 52 -53
8 2 -8.65 -1 8 -9 52 -53
9 1 -0.86 -1 9 -10 52 -53
10 2 -8.65 -1 10 -11 52 -53
11 1 -0.86 -1 11 -12 52 -53
12 2 -8.65 -1 12 -13 52 -53
13 1 -0.86 -1 13 -14 52 -53
14 2 -8.65 -1 14 -15 52 -53
15 1 -0.86 -1 15 -16 52 -53
```

16 2 -8.65 -1 16 -17 52 -53

17 1 -0.86 -1 17 -18 52 -53

18 2 -8.65 -1 18 -19 52 -53

19 1 -0.86 -1 19 -20 52 -53

20 2 -8.65 -1 20 -21 52 -53

21 1 -0.86 -1 21 -22 52 -53

22 2 -8.65 -1 22 -23 52 -53

23 1 -0.86 -1 23 -24 52 -53

24 2 -8.65 -1 24 -25 52 -53

25 1 -0.86 -1 25 52 -53

52 2 -8.65 1 -54 57 -58

53 0 54:-57:58 -56

54 0 -54 -57 55

55 0 -1 -52 57

56 0 -1 53 -58

57 0 56

1 cz 6.35

2 px -5.35

3 px -5.34

4 px -4.34

5 px -4.33

6 px -3.33

7 px -3.32

8 px -2.32

9 px -2.31

10 px -1.31

11 px -1.3

12 px -0.3

13 px -0.29

14 px 0.71

15 px 0.72

16 px 1.72

17 px 1.73

18 px 2.73

19 px 2.74

20 px 3.74

21 px 3.75

22 px 4.75

23 px 4.76

24 px 5.76

25 px 5.77

52 pz 0.0

53 pz 12.7

54 cz 6.45

55 pz -0.101

56 so 50

57 pz -0.100

58 pz 12.8

mode n

m1 1001. 7.06

```
6012. 3.73
m2 48000 1
imp:n 1 25r 0 1 1 1 0
tmp1 2.55e-008 30
mt1 benz.60t
sdef POS=0 0 -0.101 CEL=54 ERG=D2 WGT=1 TME=0
RAD=D1 SUR=55 vec=0 0 1 dir=1
si1 H 0 6.35
sp1 D -21 1
si2 H 0.015 0.035 0.055 0.075 0.095 0.115 0.135 0.165 0.195
0.225 0.255 0.305 0.355 0.405 0.455 0.505 0.555 0.605
0.655 0.705 0.755 0.805 0.855 0.905 0.955 1.05 1.15
1.25 1.35 1.45 1.55 1.65 1.75 1.85 1.95 2.15
2.35 2.55 2.75 2.95 3.25 3.55 3.85 4.15 4.45
4.75 5.05 5.55 6.05 6.55 7.05 7.55 8.05 8.55
9.05 9.55 10.05 10.55 11.05 11.55 12.05 12.55
13.05 13.55 14.05 14.6 15.9 16.9 17.9 19.1 20.0
sp2 D 0 1.98e-3 2.64e-3 3.13e-3 3.54e-3
3.88e-3 4.18e-3 6.75e-3 7.25e-3 7.67e-3
8.04e-3 1.41e-2 1.48e-2 1.54e-2 1.58e-2
1.61e-2 1.64e-2 1.66e-2 1.67e-2 1.68e-2
1.68e-2 1.68e-2 1.68e-2 1.67e-2 1.66e-2
3.12e-2 3.23e-2 3.15e-2 3.07e-2 2.98e-2
2.88e-2 2.77e-2 2.67e-2 2.56e-2 2.46e-2
4.60e-2 4.20e-2 3.81e-2 3.44e-2 3.10e-2
4.07e-2 3.44e-2 2.89e-2 2.42e-2 2.02e-2
```

1.68e-2 1.39e-2 1.80e-2 1.31e-2 9.84e-3
6.83e-3 4.90e-3 3.52e-3 2.52e-3 1.81e-3
1.29e-3 9.26e-4 6.62e-4 4.73e-4 3.38e-4
2.41e-4 1.72e-4 1.23e-4 8.75e-5 6.21e-5
4.78e-5 2.12e-5 1.06e-5 1.06e-5 6.01e-6
2.15e-6
sd4 1. 1. 1.
f4:n (2 4 6 8 10 12 14 16 18 20 22 24) (52 55) T
fm4 -1 2 102
t4 50 100 200 400 800 1600 3200 6400 12800
25600 51200 1.0e+37
e4 0.025E-6 0.05E-6 0.1E-6 0.15E-6 0.2E-6 0.25E-6 0.3E-6
0.4E-6 1.0E-6 1.0E-5 1.0E-4 1.0E-3 1.0E-2 1.0E-1 1.0 15.0
phys:n 15
thtme 1.0
prdmp 100000000 100000000
nps 100000000
ctme 30
dbcn 0 0 0 0 0 0 0 1 0 10 0 0 0 0
print -20 -32 -35 -80 -85 -86 -90 -98 -110 -120 -128 -130 -160 -161 -162

