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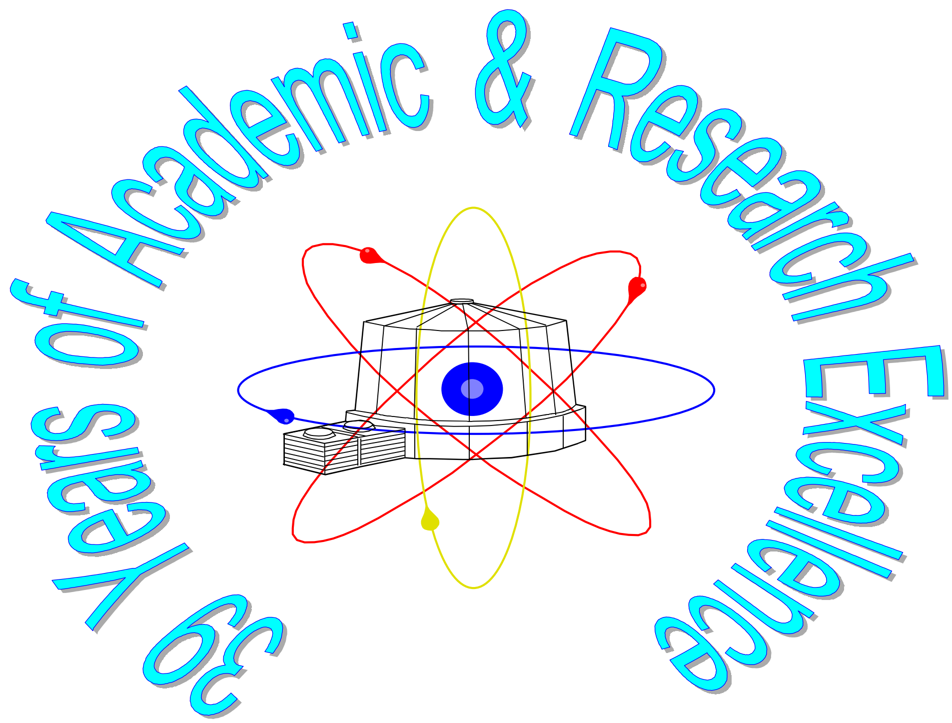
Technical Report 1998-03

Decay Heat Estimates for MNR



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February 23, 1999

Decay Heat Estimates for MNR

1 Introduction

Decay heat removal is essential in preventing fuel overheating and possible fission product release. This report provides estimates of decay heat production for use in MNR safety analysis.

2 A Simple Expression for Decay Heat

Todreas and Kazimi [Todreas 1990] (based on [Glasstone 1967]) give the approximate decay power for beta heating as

$$\frac{P_\beta}{P_0} = 0.035\left[(\tau-\tau_s)^{-0.2} - \tau^{-0.2} \right] \qquad (1)$$

and the approximate decay power for gamma heating as

$$\frac{P_\gamma}{P_0} = 0.031 \left[(\tau-\tau_s)^{-0.2} - \tau^{-0.2} \right] \qquad (2)$$

for a total of

$$\frac{P}{P_0} = 0.066 \left[(\tau-\tau_s)^{-0.2} - \tau^{-0.2} \right] \qquad (3)$$

where P is the decay power, P_0 is the nominal reactor power, τ is the time since reactor startup and τ_s is the time of reactor shutdown measured from the time of startup. The above expressions are valid for times between 10 seconds and 100 days (8.6 x 10^8 seconds) after shutdown.

To cast the expressions with a shutdown as the time reference point, we define

$$\tau_{elapsed} = \tau-\tau_s \qquad (4)$$

and hence

$$\frac{P}{P_0} = 0.066 \left[\tau_{elasped}{}^{-0.2} - (\tau_s + \tau_{elapse})^{-0.2} \right] \qquad (5)$$

Equation 5 is plotted in figure 1 assuming the reactor has been operating continuously for 10 years (ie τ_s = 10 years). Normally, the β heat is deposited locally in the fuel and much of the γ heat is deposited in the coolant. However, if the core should be voided, as would occur under a LOCA accident with no long term ECC, the γ heat would be deposited in the fuel assemblies for the most part. To be conservative, one could assume that all decay heat is deposited locally in the fuel meat.

3 A Less Simple Expression

Todreas [Todreas 1990] and [Rust 1979]quote Glasstone and Sesonske (1967, 2nd edition) [Glasstone 1967] as giving the decay heat as

$$\frac{P}{P_0} = 0.1 \left[\left(\tau-\tau_s + 10\right)^{-0.2} - \left(\tau + 10\right)^{-0.2} + 0.87 \left(\tau + 2x10^7\right)^{-0.2} - 0.87 \left(\tau-\tau_s + 2x10^7\right)^{-0.2} \right] \quad (6)$$

This is also plotted in figure 1. Equation 6 appears to lose accuracy beyond 10^6 seconds.

4 ANS 5.1 / N18.6

The Rhode Island Safety Analysis Report - Part B, Thermal Hydraulic Analysis [RI-SAR] refers to the ANS reference curve of 1968 and presents a tabulated list of values. This is plotted in figure 1. This curve was adopted by the ANS as a standard in 1971 and is known as the ANS-5.1 / N18.6 standard [ANS-5.1 1973].

Glasstone (3rd edition 1981) [Glasstone 1981] suggests the use of the following approximation of the ANS curve:

$$\frac{P}{P_0} = 5\times10^{-3} a\left[\tau_{elapse}^{-b} - (\tau_s + \tau_{elapse})^{-b}\right] \quad (7)$$

where a and b are constants which depend on τ_s as follows

Time after shutdown (s)	a	b
0.1 to 10	12.05	0.0639
10 to 150	15.31	0.1807
150 to 8 x 10^8	27.43	0.2962

The formula is said to fit the ANS curve to within ±6%.

This standard is applicable as a general estimate of the decay heat. Errors up to 50% can result for short (< 1000 seconds) and long (> 10^7 seconds) times [Todreas 1990]. In the mid range, errors are of the order of +10% and -20%. To get more accurate estimates, one would need to perform detailed inventory scenario calculations using a code such as ORIGEN as discussed in Appendix 1.

5 Example Results

Appendix 2 tabulates the ANS curve values and some typical results for equations 5 and 6. Also given are the prorated power and maximum heat flux values for MNR at 2 MW with a maximum power eighteen plate assembly at 125 kW based on the ANS curve.

6 Conclusion

The simple expression (equation 5) is adequate for times between 10 seconds and 100 days. Beyond that, it is recommended that the ANS curve or its approximation (equation 7) be used. It should not be necessary to perform detailed ORIGEN calculations given the large safety margins of MNR and the varied history of typical cores. Equation 6 does not appear to be useful or preferable to the above alternatives.

References

ANS-5.1 1973 Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors, Draft ANS-5.1 / N18.6, October 1973.

Glasstone 1967 Samuel Glasstone and Alexander Sesonske, *Nuclear Reactor Engineering*, Van Nostrand Reinhold, 1967, ISBN ?? (2rd edition).

Glasstone 1981 Samuel Glasstone and Alexander Sesonske, *Nuclear Reactor Engineering*, Krieger Publishing Company, Malabar, Florida 32950, 1981, ISBN 0-89464-567-6 (3rd edition).

RI-SAR 1991 Rhode Island Safety Analysis Report - Part B, Thermal Hydraulic Analysis, internal technical report, Rhode Island Nuclear Science Center, 1991.

Rust 1979 James H. Rust, *Nuclear Power Plant Engineering*, Haralson Publishing Company, P.O. Box 20366, Atlanta, Georgia 30325, 1979.

Todreas 1990 Neil E. Todreas and Mujid S. Kazimi, *Nuclear Systems I, Thermal Hydraulic Fundamentals*, Hemisphere Publishing Corporation, 1990, ISBN 0-89116-935-0 (v. 1)

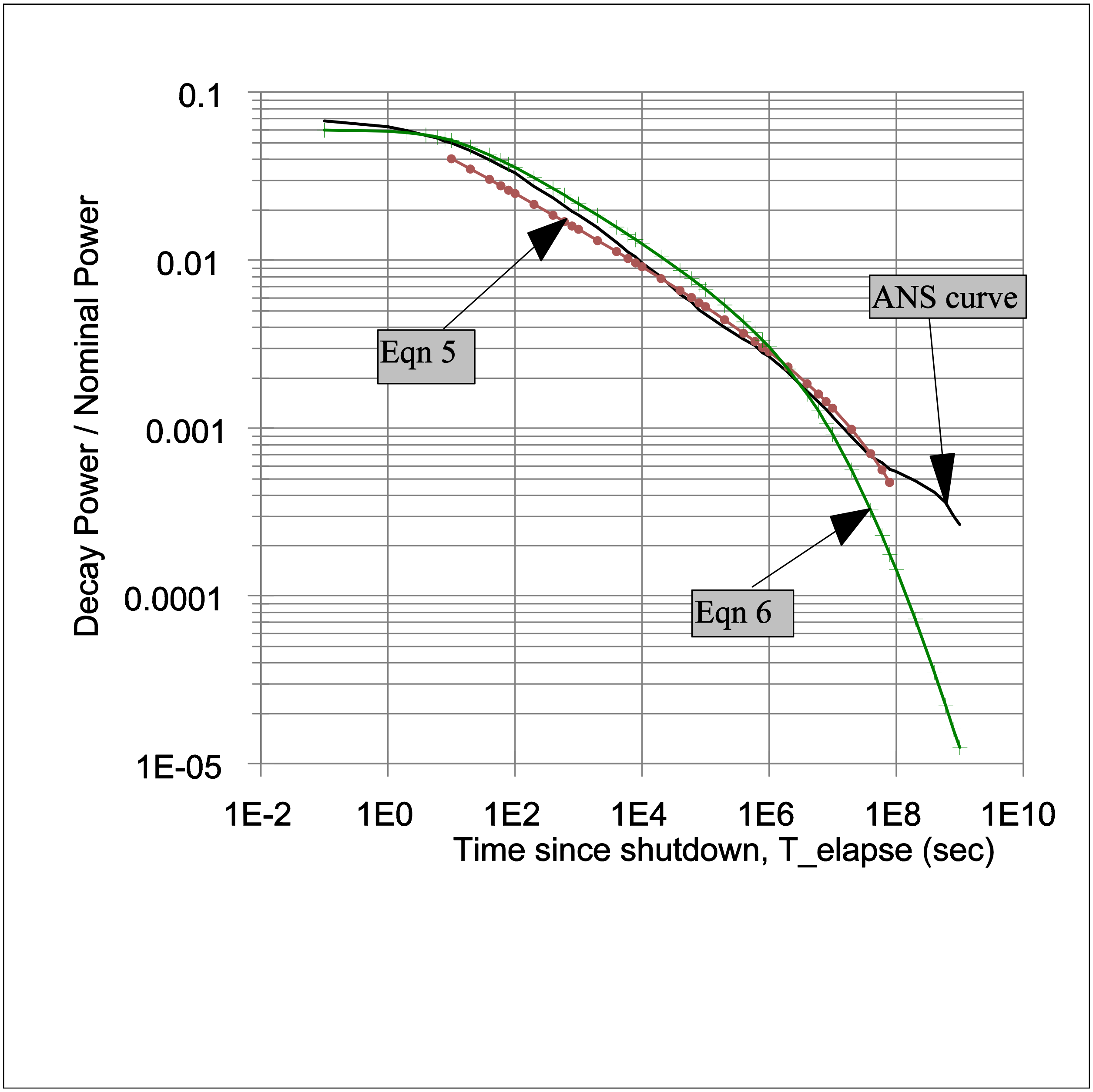

Figure 1 Relative decay heat from β's and γ's as a function of time since shutdown after extended operation [source: Corel Quattro file: d:\mnr-anal\thanal\decayhe\decayhe1.wb3].

Appendix 1 Letter from the USNRC

As downloaded from http://www.nrc.gov/NRC/FEDWORLD/NRC-GC/IN/1996/in96039.txt on 98-05-26:

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

July 5, 1996

NRC INFORMATION NOTICE 96-39: ESTIMATES OF DECAY HEAT USING ANS 5.1 DECAY HEAT STANDARD MAY VARY SIGNIFICANTLY

Addressees

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees to the sensitivity of analytical results to input parameters used with American Nuclear Society standard 5.1 on decay heat (Reference). It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Background

American Nuclear Society standard 5.1 (ANS 5.1 standard) for decay heat generation in nuclear power plants provides a simplified means of estimating nuclear fuel cooling requirements that can be readily programmed into computer codes used to predict plant performance. The ANS 5.1 standard models the energy release from the fission products of U-235, U-238, and Pu-239 using a summation of exponential terms with empirical constants. Corrections are provided to account for energy release from the decay of U-239 and Np-239, and for the neutron activation of stable fission products. Although the empirical constants are built into the standard, certain data inputs are left to the discretion of the user. These options permit accounting for differences in power history, initial fuel enrichment, and neutron flux level.

Description of Circumstances

During a review of decay heat estimates calculated using various codes for the same plant, the staff found that the predicted decay heat varied considerably. This was unexpected because the analyses were made using the 1979 ANS 5.1 standard and were to be used in "best-estimate" thermal-hydraulic analyses.

The staff compared calculations of decay heat using MELCOR, TRAC, RELAP, and a vendor code for a pressurized water reactor operating at 1933 megawatts thermal (MWT). The staff made 3 calculations using RELAP but varied certain inputs. These calculations appear as RELAP1, RELAP2, and RELAP3 in the following tables. The decay heat estimates in MWT calculated for 2 hours

after shutdown of the reactor using the various codes and allowable inputs were as follows:

MELCOR -	20.70 MWT
Vendor code -	22.15 MWT
RELAP1 -	22.02 MWT
RELAP2 -	26.13 MWT
RELAP3 -	20.82 MWT
TRAC -	22.02 MWT
ORIGEN -	20.90 MWT

It should be noted that the differences in predicted decay heat resulted from the parameters selected for input and not from the codes themselves.

The last entry in the table was not calculated by the ANS 5.1 standard but was calculated by the ORIGEN computer code. ORIGEN does not use empirical methods to calculate decay heat but tracks the buildup and decay of the individual fission products within the reactor core during operation and shutdown. ORIGEN also includes the effect of element transmutation from neutron capture, both in fissile isotopes and fission products. Because ORIGEN is a rigorous calculation of all decay heat inputs, it was used in the calculations for decay heat in attached Figure 1 and is contrasted with attached Figure 2 using decay heat internally calculated by RELAP to the ANS 5.1 standard.

Discussion

The staff found that the different decay heat estimates occurred because the ANS 5.1 standard was not fully utilized in the selection of inputs. Attached Table 1 shows the various inputs to the codes to estimate decay heat. Attached Table 2 shows the effect of different assumptions on best estimate calculations of decay heat 2 hours after shutdown of the reactor.

The variation in peak cladding temperatures with various predictions of decay heat may be seen by comparing the attached Figures 1 and 2. These figures are plots of the peak cladding temperatures for a hypothetical beyond-design-basis loss-of-feedwater event. For Figure 1, values of decay heat predicted by the ORIGEN code were input directly into RELAP. Figure 2 is the same event with the decay heat internally calculated by RELAP using the ANS 5.1 standard. Input assumptions to the standard were those that produced the highest values of decay heat denoted in Table 1 as RELAP2. The difference in predicted peak core cladding temperatures (approximately 250ø K [630ø F]) demonstrates the importance of carefully selecting required input parameters when using ANS 5.1 because analytical results may be significantly affected. Depending on the input parameters selected the results may be conservative or nonconservative.

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This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

signed by

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Office of Nuclear Reactor Regulation

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Attachments:
1. Table 1, "Decay Heat Options Using ANS 5.1, 1979"
2. Table 2, "Relative Importance of Options 7200 Seconds (2 hours) After Reactor Shutdown"
3. Figure 1, "Peak Core Cladding Temperature Calculated by ORIGEN" and Figure 2, "Peak Core Cladding Temperature Calculated by ANS 5.1"
4. List of Recently Issued NRC Information Notices

Reference:
"American National Standard for Decay Heat Power in Light Water Reactors," American Nuclear Society Standards Committee Working Group, ANS 5.1, Approved August 29, 1979.

Attachment 1

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Table 1. Decay Heat Options Using ANS 5.1, 1979

Codes
Actinides
R-Factor
G-Factor
Si Power
HistoryFissile
ElementsVendor

Yes 0.6 Not ANS 1.347 3 Yrs. 3 RELAP1 No NA MAX NA Infinite All
U235RELAP2 Yes 1.0 MAX NA Infinite All
U235RELAP3
No NA No NA Infinite All
U235MELCOR
Yes 0.526 Yes 0.713 1.6 Yrs. 3TRAC No NA Yes 1.0 Infinite All
U235
Notes:

Actinides: Neutron capture by U238 produces U239 which decays into Np239 which also decays, adding to the total decay heat. Not all the above code inputs included actinide decay.

R-factor: The actinide production multiplier. The standard states that the value of R shall be supplied and justified by the user.

G-factor: A decay heat multiplier to account for the effect of neutron capture in fission products. The standard provides the option of using a maximum value table for the G-factor or a best estimate equation for the first 10,000 seconds.

Si: Fissions per initial fissile atom. Si is a multiplier applied to the G-factor equation.

Power History: Length of full-power operation before shutdown.

Fissile Elements: The standard permits decay power to be fractionally attributed to the fission products of 3 fissile isotopes U235, U238, and Pu239. The vendor input attributed the fractional fission product power as 0.487 from U235, 0.069 from U238, and 0.443 from U239. The fractional fission product power input to MELCOR was 0.647 from U235, 0.0425 from U238, and 0.31 from Pu239. The other code inputs assumed all fission product power came from U235.
Attachment 2

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Table 2. Relative Importance of Options
7200 Seconds (2 hours) After Reactor Shutdown

Complete Set For:	Options as Input and Varied	Decay Heat in MWT
MELCOR	MELCOR assumptions (see Table 1)	20.70
	Increase R from .526 to .6	21.03
	Increase Si from .713 to 1.347	21.13
	Operation time from 1.6 years to 3 years	21.40
	Vendor fissile isotopes	21.12
	Vendor heavy element equation	21.86
Vendor code	Vendor G-factor equation	22.15

	ANS 5.1 G-factor and heavy element eqs.	21.12
	No actinides	18.65
	Gmax rather than equation	19.44
	Infinite operation	20.53
RELAP1	All fission from U235	22.02
RELAP2	Add actinides with R=1	26.13
RELAP3	Remove actinides and G-factor	20.82
TRAC	Include G-factor equation with Si=1	22.02

Note: This table illustrates the relative importance of the various options contained in the 1979 ANS 5.1 standard. Starting with the decay heat rate calculated by MELCOR using the inputs listed in Table 1, the options were changed in sequence to those used in the vendor computer code. The options were then changed to those used in the RELAP1 analyses, the RELAP2 analyses, the RELAP3 analyses, and the TRAC analyses.

Appendix 2 Spreadsheet Printout

[source: Corel Quattro file: d:\mnr-anal\thanal\decayhe\decayhe1.wb3]

Decay Heat Calculations

Input data

P0	2e+06	watts	Initial Power	
T	varies	seconds	Time since reactor startup	
T_s	315360000	seconds	Time of reactor shutdown	(10 years)
T_elapse	varies	seconds	Time since reactor shutdown = T-T_s	
P_ass	125000	watts	Max assembly power	
q_max	154199.92	W/m^2	max surface heat flux	

T_elapse	ANS P/P0	Eqn 5 P/P0	Eqn 6 P/P0	ANS Reactor Power	ANS Max Assembly Power	ANS Surface heat flux
1e-01	0.0675		0.05967	135000	8437.5	10408.5
1e+00	0.0625		0.05861	125000	7812.5	9637.5
2e+00	0.059		0.05754	118000	7375.0	9097.8
4e+00	0.0552		0.05569	110400	6900.0	8511.8
6e+00	0.0533		0.05414	106600	6662.5	8218.9
8e+00	0.0512		0.05280	102400	6400.0	7895.0
1e+01	0.05	0.04033	0.05163	100000	6250.0	7710.0
2e+01	0.045	0.03493	0.04735	90000	5625.0	6939.0
4e+01	0.0396	0.03024	0.04243	79200	4950.0	6106.3
6e+01	0.0365	0.02778	0.03946	73000	4562.5	5628.3
8e+01	0.0346	0.02616	0.03736	69200	4325.0	5335.3
1e+02	0.0331	0.02496	0.03576	66200	4137.5	5104.0
2e+02	0.0275	0.02156	0.03102	55000	3437.5	4240.5
4e+02	0.0235	0.01860	0.02673	47000	2937.5	3623.7
6e+02	0.0211	0.01704	0.02443	42200	2637.5	3253.6
8e+02	0.0196	0.01602	0.02290	39200	2450.0	3022.3
1e+03	0.0185	0.01526	0.02177	37000	2312.5	2852.7
2e+03	0.0157	0.01311	0.01855	31400	1962.5	2420.9
4e+03	0.0128	0.01125	0.01573	25600	1600.0	1973.8
6e+03	0.0112	0.01027	0.01425	22400	1400.0	1727.0
8e+03	0.0105	0.00962	0.01327	21000	1312.5	1619.1
1e+04	0.00965	0.00914	0.01255	19300	1206.3	1488.0
2e+04	0.00795	0.00779	0.01050	15900	993.8	1225.9

Decay Heat Calculations

4e+04	0.00625	0.00661	0.00872	12500	781.3	963.7
6e+04	0.00566	0.00599	0.00778	11320	707.5	872.8
8e+04	0.00505	0.00558	0.00716	10100	631.3	778.7
1e+05	0.00475	0.00528	0.00671	9500	593.8	732.5
2e+05	0.004	0.00443	0.00542	8000	500.0	616.8
4e+05	0.00339	0.00368	0.00429	6780	423.8	522.7
6e+05	0.0031	0.00330	0.00371	6200	387.5	478.0
8e+05	0.00282	0.00304	0.00333	5640	352.5	434.8
1e+06	0.00267	0.00285	0.00304	5340	333.8	411.7
2e+06	0.00215	0.00231	0.00225	4300	268.8	331.5
4e+06	0.00166	0.00184	0.00159	3320	207.5	256.0
6e+06	0.00143	0.00160	0.00127	2860	178.8	220.5
8e+06	0.0013	0.00144	0.00107	2600	162.5	200.5
1e+07	0.00117	0.00132	0.00092	2340	146.3	180.4
2e+07	0.0009	0.00099	0.00056	1780	111.3	137.2
4e+07	0.0007	0.00070	0.00032	1360	85.0	104.9
6e+07	0.0006	0.00056	0.00023	1240	77.5	95.6
8e+07	0.0006	0.00047	0.00018	1140	71.3	87.9
1e+08	0.0006		0.00014	1100	68.8	84.8
2e+08	0.0005		0.00007	970	60.6	74.8
4e+08	0.0004		0.00004	830	51.9	64.0
6e+08	0.0004		0.00002	720	45.0	55.5
8e+08	0.0003		0.00002	606	37.9	46.7
1e+09	0.0003		0.00001	534	33.4	41.2