"A Revised ANS Standard for Decay Heat from Fission Products"

Invited Paper by

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ABSTRACT

Heating from the decay of radioactive nuclides in shutdown reactors plays an important role in the safety evaluation of nuclear power plants. Although there are many other important uses for this information, the need for more accurate data for the analysis of hypothetical reactor accident scenarios has been the main impetus for recent research activity that has led to a major revision of the Draft ANS 5.1 Standard "Decay Energy Release Rates Following Shutdown of Uranium Fueled Reactors" (published in 1971 and given minor revision in 1973). The basis of the first standard was the working curve developed by K. Shure[1] in 1961. Liberal uncertainties were assigned to the standard values because of lack of data for short cooling times and large discrepancies among experimental data.

This paper reviews the results of recent research programs and the application of our improved knowledge to a 1978 revision of the standard. Very accurate determination of the decay heat is now possible for light water reactors, especially within the first 10^4 seconds after shutdown, where the influence of neutron capture in fission products may be treated as a small correction to the idealized zero capture case. The new standard accounts for differences among fuel nuclides. It covers cooling times to 10^9 seconds, but provides only an "upper bound" on the capture correction in the interval from 10^4 to 10^9 seconds.

BACKGROUND

Decay power has been of interest from the very beginning of the development of fission reactors. The mechanisms involved in the production of decay heat have long been understood, however the quantitative assessment for particular situations has been subject to a considerable amount of uncertainty. The decay power in a shutdown reactor is a result of the decay of radioactive nuclides whose presence can be traced to either fission reactions or neutron induced transmutations in the fuel, moderator and structural material of the reactor. The major source is the decay of fission products which depends upon the reactor fuel, neutron flux history and neutron energy spectrum. The fission product

contribution has usually been kept separate in the evaluation of the total decay power.

Fission product decay power may be evaluated by so called summation calculations or it may be measured experimentally using radiometric or calorimetric techniques. The summation calculation approach consists of calculating the concentrations of all fission products as a function of time and employing their decay constants and decay schemes to calculate their individual contributions to the decay power. For finite reactor operation an exact calculation would consider the transmutations caused by neutron capture in fission products during reactor operation. Consequently the method requires very extensive nuclide data including direct yields, decay constants, branching ratios, beta and gamma energies, and cross sections. The data libraries suffer from lack of completeness and some uncertainty; therefore the calculated decay power is not exact.

Most of the radiometric techniques are independent detection methods for beta and gamma radiations. Some fraction of the gamma escapes the detector and sample self absorption causes some loss of beta. Detector efficiencies are energy dependent and the energy spectrum of the radiations is time dependent. Some fission gas may be lost from samples prepared for beta experiments. Finite counting times are necessary while the decay energy rate is dropping with time. These factors make accurate absolute data difficult to obtain.

Calorimetric techniques give the total decay power from a sample in one experiment, however the massive absorber needed to absorb most of the gamma energy tends to give a long time constant so that the instrument may not be capable of following the declining decay power at short cooling time. Both calorimetric and radiometric measurements require a companion measurement of the number of fissions experienced by the sample.

By 1960, rather extensive summation calculations had been done and radiometric experiments performed. Relatively little use had been made of the calorimetric method. Shure[1] made a careful assessment of the available data and developed "working" curves for the total and gamma decay heat powers for ^{235}U irradiated at constant power for an infinite time (a convenient idealization). The working curves gave decay power in units of MeV per fission which is a contraction of decay power (MeV/sec) divided by fission rate (fissions/sec). No assessment of the uncertainty was made. Shure's curve was based primarily on the work of Perkins and King[2] and Stehn and Clancy[3].

Shure's working curves were intended for use in calculations pertaining

to actual power reactors. Two important implications should be mentioned here. First, as discussed by Shure, based on data of Petrov[4] it was thought that the decay heat power from fission products produced by different fissioning nuclides would be practically the same as for ^{235}U. Second, though not discussed by Shure, it was assumed that neutron capture in fission products would have a negligible effect upon the decay heat power. Subsequent developments have shown that neither of these assumptions are adequate for a very accurate evaluation. Shure's working curve was widely accepted and used in reactor design analysis rather extensively for many years.

In the mid 1960's, the American Nuclear Society became committed to the development of Standards including one on decay heat in shutdown reactors. The Working Group on decay heat of ANS-5 Subcommittee of the ANS Standards Committee reviewed the state-of-knowledge and concluded that the summation calculation itself was too complex and not well enough developed to serve alone as the Standard. They selected the Shure working curve as the standard decay power from ^{235}U fission products and assigned uncertainties from an unpublished study by May and Witt[5]. Although neutron absorption in fission products was not addressed in the Standard, just as in the Shure work, it was implicit that the effect was negligible.

The first working draft of the ANS Standard was presented to the ANS-5 Subcommittee in June 1968 and after editorial work it was published by the American Nuclear Society in October 1971 as a Draft Standard.

The uncertainty specified for the Standard was

Cooling Time, Seconds	Uncertainty, %
$t < 10^3$	+20, -40
$10^3 > t > 10^6$	+10, -20
$10^6 > t > 10$	+25, -50

Unfortunately these numbers were merely brackets around the Shure curve that enclosed all the data reviewed. As such they have no statistical basis. However, in such terms they undoubtedly represented a "high" confidence level.

In 1970, England[6] completed a thesis on an advanced summation calculation using his CINDER code. The data library included data for >350 nuclides and although a "point reactor" model it could trace the evolution of fuel composition, i.e., account for depletion and conversion during reactor operation. Neutron capture in fission products was fully accounted for. The results showed that

- There are large differences between decay heat from fission products of different fissioning nuclides.
- ^{235}U results were significantly above the Shure curve.
- Neutron capture in fission products increased decay power even at short cooling time.

Shure[7] reviewed this work and found a programming error and a misinterpretation concerning the fission rate. He concluded that the corrected result agreed reasonably with the Shure working curve and that neutron capture has a small effect for time after shutdown $t < 10^7$ sec.

Perry, et. al.[8] in 1972 completed a review of available data sources. They were particularly interested in the short time cooling time, 0 to 10^3 seconds, because it is the range important to LOCA analysis and also the range where the ANS Standard (Shure working curve) "was based upon limited and uncertain evidence". They reviewed recent radiometric data of a number of experimenters and made an assessment of the "best-estimate" experimental afterheat for beta and gamma (separately and in combination) and they estimated the standard deviation in the cooling time range of 1 to 5×10^5 seconds. The result ranged between 0.98 and 1.08 times the Shure working curve and the standard deviation was estimated to vary with time between 10 and 15%. "Upper bounds" chosen to encompass all the data are somewhat above the ANS5.1 upper bounds. The same is true for the best estimate plus two standard deviations (2σ).

Perry's evaluation of the effect of neutron capture in fission products was accomplished using the Oak Ridge National Laboratory code ORIGEN[9] by "switching off" the capture cross sections for all fission products while maintaining other conditions the same. The results with neutron capture are higher than without by less than 2% when $t < 10^3$ seconds, 4% higher at $t = 10^4$ seconds and 7% higher at $t = 10^5$ seconds, for the power history studied.

The U. S. AEC used the proposed Standard in the regulatory process, most importantly in the Code of Federal Regulations, Title 10-Energy, Part 50-Licensing of Production and Utilization Facilities, Appendix K - ECCS Evaluation Models. This Federal Regulation specifies the technical conditions that must be used in the "evaluation model" (licensing calculation method) concerning the performance of the emergency core cooling system in limiting the temperature reached by fuel cladding during the hypothetical loss of coolant accident (LOCA). Appendix K requires the use of the ANS 5.1 Standard fission product decay heat

times 1.2. By this requirement AEC essentially agreed that the 20% uncertainty in the Standard represents a "high" confidence level. However, writing this number into law has not served to give it a more precise meaning.

This background was also reviewed with somewhat different emphasis by Lott[10] in 1973, by Tobias[11] in 1973, and by Bjerke, et. al.[12] in 1977. DeVillers[13] gave a good discussion of the importance of decay heat and other fission product properties in the design and safety analysis of power reactors.

Because of the strong dependence of the peak clad temperature during LOCA on decay heat power there became a strong incentive to improve our knowledge, especially at short cooling times. New research was sponsored in the United States by the Nuclear Regulatory Commission, the Department of Energy and the Electric Power Research Institute. The ANS Working Group was reconstituted with the aim of improving the Standard, particularly for LOCA applications.

RECENT ADVANCES

Advances have been made in recent years in all aspects of the decay heat problem. Experimental techniques have been improved for both radiometric and calorimetric measurements. Nuclide data libraries have been vastly improved by the sustained effort in the United States and internationally to evaluate existing and new data, particularly the Evaluated Nuclear Data File[14]. Summation codes have been developed to a greater level of sophistication. The effect of neutron capture in fission products has been studied extensively. Statistical methods have been developed for evaluation of the uncertainty in summation results and least squares methods, developed for statistically combining data sets to obtain a best estimate and its uncertainty.

Calorimetry was advanced in precision by Johnston[15] in 1965. Lott, et.al.[16] did the first calorimetric measurements using a transient method which could obtain data as early as 70 seconds after irradiation. Yarnell and Bendt[17] at Los Alamos Scientific Laboratory (LASL) designed a transient calorimeter having a time constant on the order of 1 second. Schrock, et.al.[18] at the University of California, Berkeley (UCB) developed another fast response calorimeter, based on a different principle, that also has a time constant of about one second. These instruments are able to measure decay heat at times as short as 10 seconds (determined by sample transfer time).

Dickens, et.al.[19] at Oak Ridge National Laboratory (ORNL) improved the classical radiometric techniques and Friesenhahn, et.al.[20] at IRT Corporation (IRT) developed a novel experimental method they called the "nuclear calorimeter".

These radiometric methods have obtained data at 1 second after irradiation.

The irradiation histories for all these experiments are short and therefore there is no significant capture effect in any of the results. Gunst, et.al.[21] used longer irradiations in a quasi-steady calorimeter but obtained data only for long cooling times.

Data for ^{235}U were obtained by LASL, ORNL, IRT, UCB and the French (Lott) to cooling times of 10^5 seconds. Of these, the LASL data showed exceptional agreement with the summation calculations using the ENDF/B-IV data file. Data for ^{239}Pu were reported by ORNL[22], LASL[17] and Fiche, et.al.[23] using the French calorimeter. In this case the ORNL data are in best agreement with summation calculations while LASL and French data are higher (so far no cause has been identified). ^{233}U data were reported by LASL[17] and ^{241}Pu data were reported by ORNL[24]. Most of the new data are within their estimated error (1σ) of the summation calculation. ^{239}Pu data are the principal exception.

Spinrad[25] and his group at Oregon State University (OSU), Schmittroth and Schenter[26] at Hanford Engineering Development Laboratory (HEDL) and Prussin and Nuh[27] at UCB made statistical evaluations of the error in the summation calculation results. Schmittroth[28] developed a statistical approach for combining data sets to obtain a best estimate and its uncertainty and he and Schenter (HEDL) collaborated with England[29] (LASL) to complete the application of this methodology to the new experimental data and summation calculations (and their uncertainties) to develop the functions that have been adopted for the 1978 Revised Standard. The capture effect has been studied by the LASL[29] and OSU[30] groups. The Standard functions pertain to the idealized zero capture case and the capture effect has been treated as a multiplier (correction). For $t < 10^4$ seconds, Spinrad and Tripathi[30] developed an accurate correlation for the correction factor (based on ^{235}U):

$$G(t) = 1 + (3.24\times10^{-6} + 5.23\times10^{-10}t)T^{0.4}\psi \qquad (1)$$

where t is time after shutdown, seconds

T is reactor operating time, seconds

ψ is the number of fissions per initial fissile atom in the reactor.

For $t < 10^9$ seconds, England[31] developed an upper bound correction factor based on extreme operating conditions for a LWR.

The above data comprise the basis of the 1978 Revised Standard. The summary by England, Schenter and Schmittroth[29] is the most comprehensive documentation

of the Standard development. Other summaries have been provided by Bjerke et.al.[12] and Schrock[32].

The Standard provides nominal values for the zero capture functions in the form of the sum of 23 exponentials which were fit to the best estimate values for ^{235}U and ^{239}Pu fissioned by thermal neutrons and fast fission of ^{238}U. The functions, whose relations are valid only in the absence of neutron capture in fission products, are:

Fission Pulse

$$f(t) = \sum_{i=1}^{23} \alpha_i e^{-\lambda_i t} \text{ , MeV/fission-sec} \qquad (2)$$

Irradiation at Constant Fission Rate for T Seconds

$$F(t,T) = \sum_{i=1}^{23} \frac{\alpha_i}{\lambda_i} e^{-\lambda_i t}(1-e^{-\lambda_i T}), \frac{\text{MeV}}{\text{fission}} \qquad (3)$$

The constants α_i and λ_i are given in Table 1. Tabular data were generated from these exponential fits for f(t) and $F(t,\infty)$ where for computational purposes $F(t,\infty) = F(t,10^{13})$. These tabular values are given in the Standard together with the assigned uncertainty (1σ in a normal distribution). An abbreviated table of $F(t,\infty)$ and uncertainties for the three fuel nuclides is given in Table 2. The uncertainty on ^{235}U is seen to be very small, i.e., only about 2%. The higher uncertainties on ^{239}Pu reflect the greater uncertainty in summation results and the poorer agreement with experiment as compared to the ^{235}U case. Results for ^{238}U depend entirely upon the summation method.

The Standard prescribes equivalent alternate methods for using these functions to obtain the decay power, following shutdown, from fission products for any arbitrary power history. The user provides the power history and the partition of the total fission rate among the fissioning nuclides, ^{235}U, ^{238}U and ^{239}Pu. Since the data for ^{241}Pu were not ready for inclusion in the Standard it and other small contributors are lumped with the ^{235}U fission rate. This choice reflects the knowledge that the decay heat from ^{241}Pu is higher than that from ^{239}Pu.

Denoting the "uncorrected" decay heat power by P_d' we have:

$$P'_d = \sum_{j=1}^{3} P'_{dj} \tag{4}$$

where j is an index identifying the fissioning nuclide. The P'_{dj} may be obtained by one of three alternative but equivalent methods:

1. Using the exponential fit constants

$$P'_{dj}(t,T) = \sum_{\alpha=1}^{M} \frac{P_{j\alpha}}{Q_j} \sum_{i=1}^{23} \frac{\alpha_i}{\lambda_i} e^{-\lambda_i t_\alpha}(1-e^{-\lambda_i T_\alpha}) \tag{5}$$

where

P_j is the power from fissioning of nuclide j

Q_j is the total recoverable energy associated with one fission of nuclide j.

$$t_1 = t, \quad t_2 = t+T_1, \quad t_M = t+\sum_{\alpha=1}^{M-1} T_\alpha. \tag{6}$$

and

$$T = \sum_{\alpha=1}^{M} T_\alpha \tag{7}$$

and α is an index to identify the periods of constant fission rate in a histogram representation of the fission rate.

2. Numerically integrating the pulse function tabular data, i.e.,

$$P'_{dj}(t,T) = \int_0^T \frac{P_j(T')}{Q_j} f_j(t+T-T')dT' \tag{8}$$

3. Using the tabular data for $F_j(t,\infty)$ for each period of constant power (and fission rate of nuclide j), i.e.,

$$P'_{dj} = \sum_{\alpha=1}^{M} \frac{P_{j\alpha}}{Q_j}\left[F_j(t_\alpha,\infty) - F_j(t_\alpha+T_\alpha,\infty)\right] \tag{9}$$

where α, t_α and T_α have the same meaning as in Equations (5), (6) and (7).

The third method is subject to interpolation and round off error for short irradiation time combined with long cooling time.

The uncertainty may be calculated from one of two equivalent methods.

1. Using tabular uncertainties, Δf_j, on the pulse function, the following integral is evaluated numerically

$$\left(\frac{\Delta P'_{dj}}{P'_{dj}}\right)^2 = \left(\frac{\Delta Q_j}{Q_j}\right)^2 + \left[\frac{\int_o^T P_j(T')\Delta f_j(t+T-T')dT'}{Q_j P'_{dj}}\right]^2 \qquad (10)$$

in which Q_j and ΔQ_j are the recoverable energy release per fission of nuclide j during reactor operation and its uncertainty, respectively. The Standard does not specify the values of Q_j and ΔQ_j to be used since they must be evaluated for each specific reactor. These values must be provided and justified by the user.

2. Using tabular data on the infinite irradiation function and a histogram representation of the power history a summation is used as follows

$$\left(\frac{\Delta P'_{dj}}{P'_{dj}}\right) = \left(\frac{\Delta Q_j}{Q_j}\right)^2 + \left[\frac{\sum_{\alpha=1}^{M} P_{j\alpha}\Delta F_j(t_\alpha, T_\alpha)}{Q_j P'_{dj}}\right]^2 \qquad (11)$$

where $\Delta F_j(t,T) = \Delta F_j(t,\infty) - \Delta F_j(t+T,\infty)$ (12)

The uncertainty in P_d is given by

$$\left(\frac{\Delta P_d}{P_d}\right)^2 = \left(\frac{\sum_{j=1}^{3}|\Delta P'_{dj}|}{P'_d}\right)^2 + \frac{\Delta P}{P} \qquad (13)$$

where P is total power and ΔP is user supplied. The final decay heat power is calculated from

$$P_d(t) = P'_d(t) \cdot G(t) \qquad (14)$$

G(t) is given by Equation (1) for $t < 10^4$ seconds. For $t < 10^4$ seconds the user may use the upper bound $G_{max}(t)$ given in the Standard or use the option of calculating and justifying a better value. G_{max} is shown in Figure 1.

CONCLUDING REMARKS

The 1978 Revised ANS Standard on decay heat can give very precise values for LWRs. This has been made possible by the results of recent research programs and by changing the format to include a detailed evaluation that accounts for the difference between decay heat from the various fissioning nuclides. A simplified approach is provided that uses only the ^{235}U data but this approach is known to be conservative. At present the Standard explicitly accounts for ^{235}U, ^{238}U and ^{239}Pu fission products.

Future improvements will:

1. Improve the capture effect specification for $10^4 \leq t \leq 10^9$ seconds.
2. Address other thermal reactor fuel cycles.
3. Address fast reactor fuel cycles.

It is anticipated that the 1978 Revised Standard will be published about January 1979.

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Table 1

Constants for the Exponential Representation
of ANS Nominal Decay Heat Power Functions

^{235}U Thermal		^{238}U Fast		^{239}Pu Thermal	
α	λ	α	λ	α	λ
6.5057E-01	2.2138E+01	1.2311E+00	3.2881E+00	2.0830E-01	1.0020E+01
5.1264E-01	5.1587E-01	1.1486E+00	9.3805E-01	3.8530E-01	6.4330E-01
2.4384E-01	1.9594E-01	7.0701E-01	3.7073E-01	2.2130E-01	2.1860E-01
1.3850E-01	1.0314E-01	2.5209E-01	1.1118E-01	9.4600E-02	1.0040E-01
5.5440E-02	3.3656E-02	7.1870E-02	3.6143E-02	3.5310E-02	3.7280E-02
2.2225E-02	1.1681E-02	2.8291E-02	1.3272E-02	2.2920E-02	1.4350E-02
3.3088E-03	3.5870E-03	6.8382E-03	5.0133E-03	3.9460E-03	4.5490E-03
9.3015E-04	1.3930E-03	1.2322E-03	1.3655E-03	1.3170E-03	1.3280E-03
8.0943E-04	6.2630E-04	6.8409E-04	5.5158E-04	7.0520E-04	5.3560E-04
1.9567E-04	1.8906E-04	1.6975E-04	1.7873E-04	1.4320E-04	1.7300E-04
3.2535E-05	5.4988E-05	2.4182E-05	4.9032E-05	1.7650E-05	4.8810E-05
7.5595E-06	2.0958E-05	6.6356E-06	1.7058E-05	7.3470E-06	2.0060E-05
2.5232E-06	1.0010E-05	1.0075E-06	7.0465E-06	1.7470E-06	8.3190E-06
4.9948E-07	2.5438E-06	4.9894E-07	2.3190E-06	5.4810E-07	2.3580E-06
1.8531E-07	6.6361E-07	1.6352E-07	6.4480E-07	1.6710E-07	6.4500E-07
2.6608E-08	1.2290E-07	2.3355E-08	1.2649E-07	2.1120E-08	1.2780E-07
2.2398E-09	2.7213E-08	2.8094E-09	2.5548E-08	2.9960E-09	2.4660E-08
8.1641E-12	4.3714E-09	3.6236E-11	8.4782E-09	5.1070E-11	9.3780E-09
8.7797E-11	7.5780E-10	6.4577E-11	7.5130E-10	5.7300E-11	7.4500E-10
2.5131E-14	2.4786E-10	4.4963E-14	2.4188E-10	4.1380E-14	2.4260E-10
3.2176E-16	2.2384E-13	3.6654E-16	2.2739E-13	1.0880E-15	2.2100E-13
4.5038E-17	2.4600E-14	5.6293E-17	9.0536E-14	2.4540E-17	2.6400E-14
7.4791E-17	1.5699E-14	7.1602E-17	5.6098E-15	7.5570E-17	1.3800E-14

Table 2

ANS 1978 Decay Heat Function, $F(t,\infty)$ for Three Nuclides

Time After Shutdown t,sec.	^{235}U		^{239}Pu		^{238}U	
	$F(t,\infty)$ MeV/fission	Uncertainty (1σ)%	$F(t,\infty)$ MeV/fission	Uncertainty (1σ)%	$F(t,\infty)$ MeV/fission	Uncertainty (1σ)%
1	12.31	3.3	10.27	5.6	14.19	12.0
10	9.494	2.0	8.243	4.2	10.29	9.5
10^2	6.198	1.8	5.685	4.2	6.217	5.9
10^3	3.796	1.8	3.516	4.2	3.598	4.9
10^4	1.908	1.7	1.727	4.2	1.777	4.4
10^5	0.9691	2.0	0.9421	5.0	0.9383	3.9
10^6	0.5509	2.0	0.5097	5.0	0.5171	3.9
10^7	0.2457	2.0	0.2282	5.0	0.2296	4.4
10^8	0.1165	2.0	0.08931	5.0	0.09280	5.0
10^9	0.05678	2.0	0.04195	5.0	0.04321	5.0

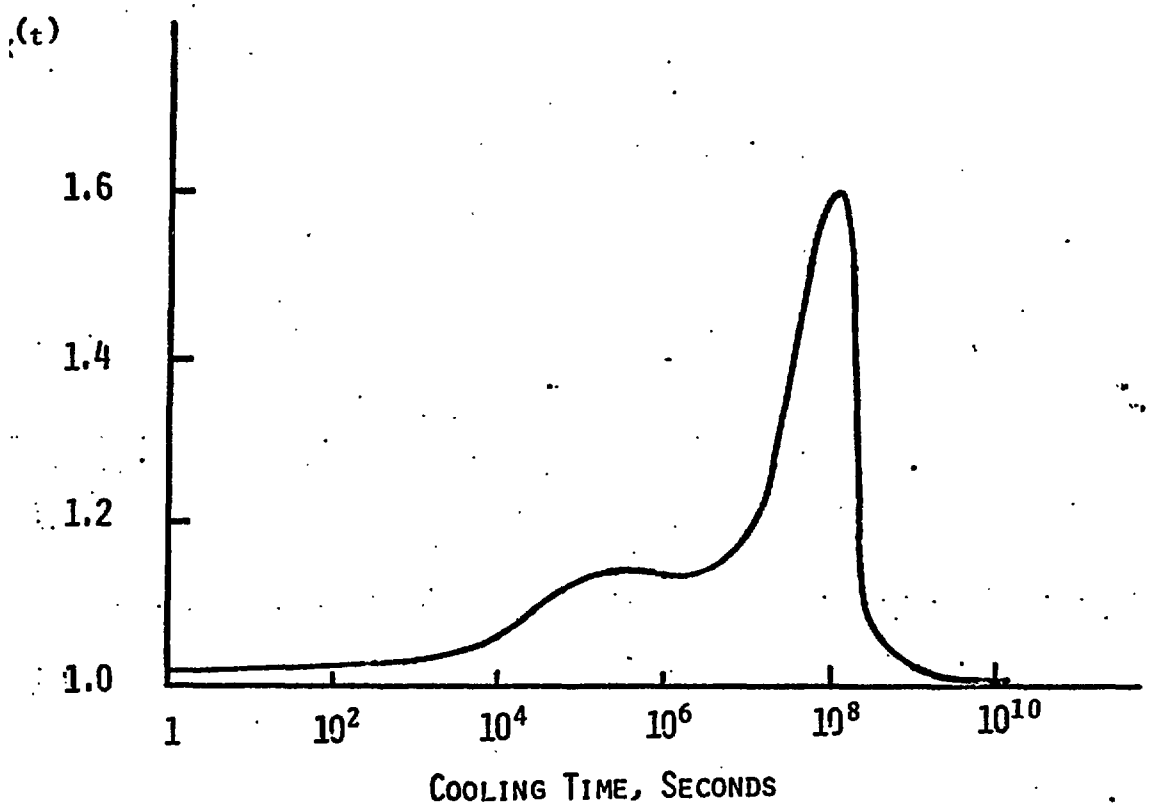

FIGURE 1 UPPER BOUND CORRECTION FOR NEUTRON CAPTURE IN FISSION PRODUCTS 1978 REVISED ANS STANDARD

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