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THE USE OF CRITICALITY CODES IN NUCLEAR SAFETY CONSIDERATIONS  
AT THE OAK RIDGE GASEOUS DIFFUSION PLANT

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THE USE OF CRITICALITY CODES IN NUCLEAR SAFETY CONSIDERATIONS  
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C. E. Newlon  
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Introduction

The criticality codes which have been developed for use with modern high speed computing machines may be considered as very powerful tools for the solution of complex nuclear problems and may, therefore, be used to good advantage in the nuclear safety evaluation of such problems. So far, at the Oak Ridge Gaseous Diffusion Plant, criticality codes have been used primarily in the re-evaluation of "older" problems, particularly those where some of the necessarily simplifying assumptions may not have been appropriately conservative. However, it appears quite reasonable that the future use of such codes will not be limited only to the long-range criticality studies, but may possibly include the more or less routine problems of criticality control as well.

In the past, of course, the primary concern at ORGDP has been the determination of basic nuclearly safe variables such as the U-235 masses at various enrichments for the relatively simple geometries of the sphere, slab, and cylinder. Here the principal calculational schemes used were the Water Boiler Theory,<sup>1</sup> and in some cases a two-group method,<sup>2</sup> for the well-moderated systems in which the various minima occur. It should be noted that, in general, these methods gave results which were in reasonably good agreement with experimental critical mass data over a wide range of U-235 enrichments and in addition could be readily reduced to routine computational forms such as nomographs or critical mass slide rules.

However, since it is perhaps obvious that the usefulness of such methods for systems other than thermal,\* is somewhat limited both by accuracy and difficulty of computation, some interest has been generated in the more recently developed criticality codes in order to ascertain if they could be used in the evaluation of nuclear safety problems wherein both the geometrical configurations and the fissionable materials themselves were considered to be rather complex from a nuclear standpoint.

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- 1 Greuling, E., Theory of Water-Tamped Water Boiler, 9-27-45 (LA-399)  
2 Henry, H. F., et al., Studies in Nuclear Safety, 8-14-58 (K-1380)

\* A thermal system may be considered as one where fission is induced primarily by neutrons of such energy that they are in thermal equilibrium with the core material; the representative energy of the neutrons is often taken as 0.025 ev. A moderator such as hydrogen is an essential element of a thermal system.

### Available Criticality Codes

Of the approximately 20 criticality codes which are available for use with the IBM-704 in the ORGDP computing laboratory, table 1 lists seven of the more frequently used codes.\* The development report numbers are also identified underneath each code name where available. Under Hydrogen Scattering, it will be noted that the treatment of hydrogen scatter is left to the discretion of the investigator in four of the codes by permitting the use of input cross section data; even now there appear to be differences of opinion, both experimental and theoretical, as to the proper value of the neutron age in water.

The last two columns indicate the dimension and geometry features of the codes. Although the majority of the codes developed so far can handle only one-dimensional configurations such as the sphere, or the slab and cylinder which can be readily reduced to one-dimensional problems, it will be noted that both the CURE and PDQ codes are two-dimensional and thus can handle some of the more difficult nuclear configurations.

Table 2 lists some additional features of the codes under consideration. It will be noted that both the CURE and PDQ codes can handle a fairly large number of regions and materials. A region may be defined as a sub-division of a nuclear system consisting of homogeneous material bounded on at least one surface by material of different composition; this region may or may not be a part of the nuclear system. The material itself may be either fissionable or non-fissionable and may in some cases be composed of as many as 90 different elements.

The last column lists the number of neutron energy groups which naturally raises the question as to the optimum number of groups needed to attain the highest degree of accuracy consistent with the computing or running time of a problem. Although there are obviously many factors to be considered, it appears that, in general, few groups will be adequate for survey studies while the multigroup calculations may be used for special nuclear studies where a more rigorous approach is considered desirable. In this regard, it might be noted that COBRAS\*\* has a maximum of 390 groups. This particular code was developed several years ago at ORGDP, primarily for the nuclear safety evaluation of a problem involving UF<sub>6</sub> gas coolers which will be discussed more in detail later. Although COBRAS is somewhat limited since it permits the evaluation of only one region and one fuel material, the code did point out that one of the recognizably non-conservative simplifying assumptions used in an early study of the gas cooler problem was, in fact, too non-conservative for comfort.

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\* The data in tables 1, 2, and 3 were kindly furnished by V. E. Anderson of the ORGDP Operations Analysis Division.

\*\* An abbreviation of "Criticality of Bare Reactors and Systems."

Attention is called also to one of the features of GNU-II, a somewhat newer and more versatile code. GNU-II contains a sub-routine which is used to compute flux weighted average values of the macroscopic cross sections which can be used as input data for the few groups codes such as CURE or PDQ. This procedure was actually used in the evaluation of another "older" problem involving pipe connections which will also be discussed later.

Table 3 lists some approximate running times for the various criticality codes which have been used in the CRGDP computing laboratory, obviously an important factor. Although it is rather difficult to make an exact comparison, an attempt was made to illustrate the general trend in running times by selecting representative sample problems for the various codes. Thus, for one-dimensional codes, the sample problem consisted of 100 mesh points, the maximum permissible number of groups specified by the code, and five iterations on the neutron source; the sample problem for two-dimensional codes consisted of 600 mesh points, three energy groups, and five iterations on the neutron source. For the SNG code, the problem selected was a fixed "Q" type (one-group calculation), 50 mesh points, with an  $S_n$  approximation of four. Obviously, there may be some particular feature of a code which would dictate its use in preference to another; however, it is apparent from the table that, in general, a two-dimensional code would not be used for a sphere since this simple geometry could be handled in far shorter time by the one-dimensional codes. Thus, the table does emphasize the need for careful selection of the criticality code to be used for the problem under consideration.

### The Gas Cooler Problem

Tables 4-8 show some results obtained with criticality codes in recent nuclear safety studies at CRGDP involving  $UF_6$  gas coolers<sup>3</sup> and pipe connections.<sup>4</sup> Obviously, there are many  $UF_6$  gas coolers in a gaseous diffusion plant, all of which are so large that criticality could occur in any one cooler, should a significant quantity of uranium accumulate therein under the proper conditions of hydrogen moderation. It should be noted, however, that any such accumulation of uranium would be essentially unmoderated and thus would be considered safe under prevailing cascade conditions.

In one of the early studies of the gas cooler problem,<sup>5</sup> the rather simplifying assumption was made that uranium-238 would act as a spacer only. Although it was recognized at the time that this assumption was non-conservative, it was the consensus of all concerned that several other

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3 Knight, J. R., Critical Conditions in  $UF_6$  Gas Coolers, 2-13-59 (KSA-158)

4 Newlon, C. E., A Nuclear Safety Evaluation of Standard Pipe Connections (to be issued)

5 Macklin, R. L., Critical Mass Limitations on K-25 Process Stream Pressures and Temperatures, 10-25-50 (K-675)

rather conservative assumptions used in the evaluation method would more than compensate for the apparent non-conservatism in the uranium-238 assumption.<sup>6</sup> The results of the recent study<sup>3</sup> made with the GNU-II code have validated the opinion that when this earlier method was applied to "doughnut-shaped" coolers, the compensating factors do in effect cancel each other. However, the GNU-II calculations also indicated that the factor of non-conservatism in the simplified method was such that it should be applied very cautiously to the nuclear safety of cooling units, such as interstage coolers, which are not "doughnut-shaped."

Two other rather interesting conclusions were drawn from results of the GNU-II calculations, which are given in tables 4 and 5 and may be stated as follows:

- a. In the gas coolers handling slightly enriched uranium, the introduction of  $\text{Cd}(\text{NO}_3)_2$  solution, into the cooler coils, effectively removes the possibility of criticality in any uranium materials that might accumulate therein, under any conditions of hydrogen moderation, and,
- b. The use of  $\text{Cd}(\text{NO}_3)_2$  solution in the cooler coils of units handling highly enriched uranium would in most cases provide some measure of protection, although not completely adequate. In some circumstances, the use of a  $\text{Cd}(\text{NO}_3)_2$  solution might actually increase the hazard.

This latter conclusion, which is seemingly paradoxical, might be attributed to the fact that the moderation at which criticality is possible in the highly enriched cooling units is so low that the cadmium would be relatively ineffective as a poison. However, it is well-known that only relatively small amounts of hydrogen are required for a significant reduction in the overall critical mass. Apparently similar conclusions might be drawn for other potentially poisoning materials.

#### The Pipe Connection Problem

Another of the "older" problems re-evaluated by use of criticality codes is the one of permissible pipe connections which are of considerable interest to designers of enriched uranium processing plants. The GNU-II code was employed in this study to obtain the input parameters for the few group two-dimensional code, PDQ, which was then used to evaluate the pipe connections of interest. In table 6, the calculated results are compared with the available critical mass experiments,<sup>7</sup> and it will be noted that the results agree to within about 4% in k, the multiplication factor for finite systems.

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6 Henry, H. F., Minutes of Special Hazards Committee Meeting, December 29, 1950, 1-15-51 (KS-169)

7 Fox, J. K., Gilley, L. W., and Callihan, A. D., Critical Mass Studies, Part IX Aqueous U-235 Solutions, 2-5-58 (ORNL-2367)

In the chart shown in table 7, the infinite multiplication factor,  $k_{\infty}$ , for various symmetrical pipe connections and cylinders, is plotted as a function of the pipe diameter, D. Although the pipe diameters are finite, the symbol  $k_{\infty}$  is used since all pipe connections, including cylinders, are considered to be infinite in length. In addition, they are considered to be completely water reflected and to contain highly enriched uranyl fluoride solution fuel. It may be noted from the chart that a self-consistent set of nuclear safety criteria may be established simply by selecting pipe connections with identical multiplication factors. Thus, for example, the 5" pipe would be approximately equivalent in nuclear safety to the 4.4" cross (+), the 4.7" tee (T), or the 4.9" ell (L). Since these values are somewhat larger than those of the current ORGDP pipe connection criteria shown in table 8, it would appear that in this case the use of criticality codes indicates that a slight upward revision of the present criteria might be possible.

### Conclusions

The experience of the ORGDP nuclear safety group with criticality codes, which include the COBRAS, GNU-II, and the PDQ codes, might be summarized by stating that, so far, criticality codes have proven useful in the evaluation of nuclear safety problems which otherwise would have been extremely laborious and in some cases essentially impossible without the use of simplifying assumptions which are only qualitative at best. In general, the results of the machine calculations have been in fairly good agreement with the results of critical mass experiments, certainly when the uncertainties of the basic nuclear input data are considered. Thus, it would appear reasonable to anticipate that criticality codes will play an increasingly important role, not only in nuclear safety evaluations of the ever present criticality problems in a gaseous diffusion plant, but in facilitating the continued development and improvement of criticality control criteria in general.

TABLE I

## CRITICALITY CODES USED IN THE ORGDP COMPUTING LABORATORY

<u>Code Identification</u>	<u>Theory</u>		<u>Hydrogen Scattering</u>	<u>Dimensions</u>	<u>Geometry</u>
	<u>Diffusion</u>	<u>Transport</u>			
I-2		x	Input Cross Sections	1	Cylinder
SNG (T-1-159)		x	Transfer Coefficient	1	Sphere, Slab, Cylinder
WANDA (WAPD-TM-28)	x		Input Cross Sections	1	Sphere, Slab, Cylinder
GNU-II (GMR-101)	x		Goertzel-Selengut	1	Sphere, Slab, Cylinder
COBRAS (K-1260)	x		Goertzel-Selengut	1	Sphere, Slab, Cylinder
CURE (KAPL-1724)	x		Input Cross Sections	2	xy, rz, r $\theta$
PDQ (WAPD-TM-70)	x		Input Cross Sections	2	xy, rz

TABLE II

CODE SPECIFICATIONS

<u>Code Identification</u>	<u>Regions (No.)</u>	<u>Materials (No.)</u>	<u>Neutron Energy Groups (No.)</u>
I-2	50	10	1
SNG	$\alpha$	32	1
WANDA	25	25	4
GNU-II	19	19	32*
COBRAS	1	1	390
CURE	576	40	$\alpha$
PDQ	2400	35	4

$\alpha$  No upper limit specified by code.

\* Includes four thermal groups.

TABLE III

APPROXIMATE RUNNING TIMES OF SAMPLE PROBLEMS  
(IBM-704)

<u>Code Identification</u>	<u>Problem (No.)</u>	<u>Time (minutes)</u>	<u>Code Dimensions</u>
I-2	A	2-3	1
SNG	B	2-3	1
WANDA	A	1.5	1
GNU-II	A	6	1
COBRAS	A	4	1
CURE	C	~ 30	2
PDQ	C	~ 20	2

## Notes:

- Problem A One dimensional problem consists of 100 mesh points, maximum permissible number of groups, and 5 iterations on the neutron source.
- Problem B SNG problem is for a fixed "Q" type (one-group) calculation, 50 mesh points,  $S_n$  approximation = 4.
- Problem C Two dimensional problem consists of 600 mesh points, 3 groups, and 5 iterations on the neutron source.

TABLE IV

SUMMARY OF RESULTS OF GNU-II CALCULATIONS FOR  
PARTIALLY REFLECTED GAS COOLERS

<u>Cooler Size*</u>	<u>Moderation at Criticality** (H/U) Atomic Ratio</u>
A	9.2
B	4.4
C	4.4
D	4.2
E	1.3
F	0.7

\* In order of decreasing size.

\*\* Lower moderation limit only, upper limit not calculated.

TABLE V

SUMMARY OF RESULTS OF GNU-II CALCULATIONS FOR  
PARTIALLY REFLECTED GAS COOLERS WITH  $\text{Cd}(\text{NO}_3)_2$

<u>Cooler Size*</u>	<u><math>\text{Cd}(\text{NO}_3)_2</math> Solution in Tubes (Concentration)</u>	<u>Remarks</u>
A	14%	Safe for any moderation.
B	25%	Safe for any moderation.
C	50%	Safe for any moderation.
D	50%	Not safe at H/U ratios above $\sim 6$ .
E	50%	Not safe to add solution.
F	50%	Not safe to add solution.

\* In order of decreasing size.

TABLE VI

k CALCULATED FOR CRITICAL EXPERIMENTS

Fuel: Highly Enriched Uranyl Fluoride Solutions

Reflector: Water

<u>Geometry</u>	<u>Diam. (inches)</u>	<u>Height (inches)</u>	<u>Moderation, H/U-235 Atomic Ratio</u>	<u>k</u>	
				<u>GNU Code</u>	<u>PDQ Code</u>
Cylinder	6.0	35.4	27.1	1.039	-
Cylinder	6.0	27.6	44.3	1.023	0.9841
Cylinder	6.5	16.6	73.4	1.011	-
Cross	5.0	5.7*	44.3	-	0.9757
Cross	5.0	7.8*	73.4	-	0.9668
Y (30°)	5.0	15.6*	73.4	-	0.9641

\* Above intersection of center lines.

Note: The four arms of the cross and three arms of the Y were at least 24 in. long measured from the center of the intersection.  
(See CRNL 2367.)

TABLE VII  
 $k_{\infty}$  FOR VARIOUS GEOMETRIES

Code - PDQ  
 Fuel - Highly Enriched Urany Fluoride Solution  
 Reflector - Water  
 Moderation - H/U-235 = 44.3

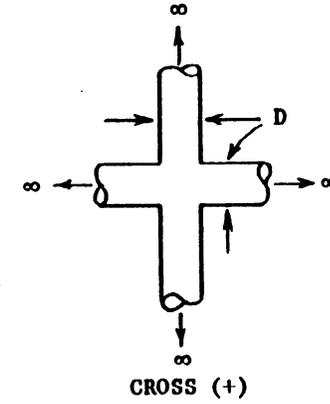
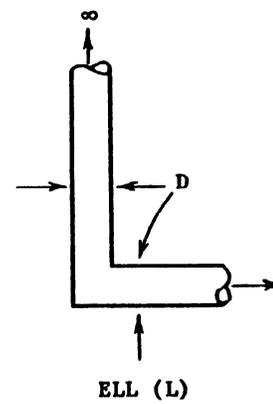
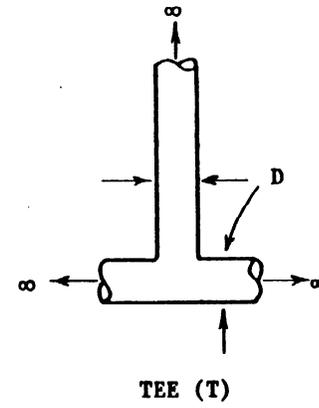
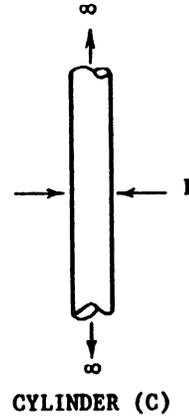
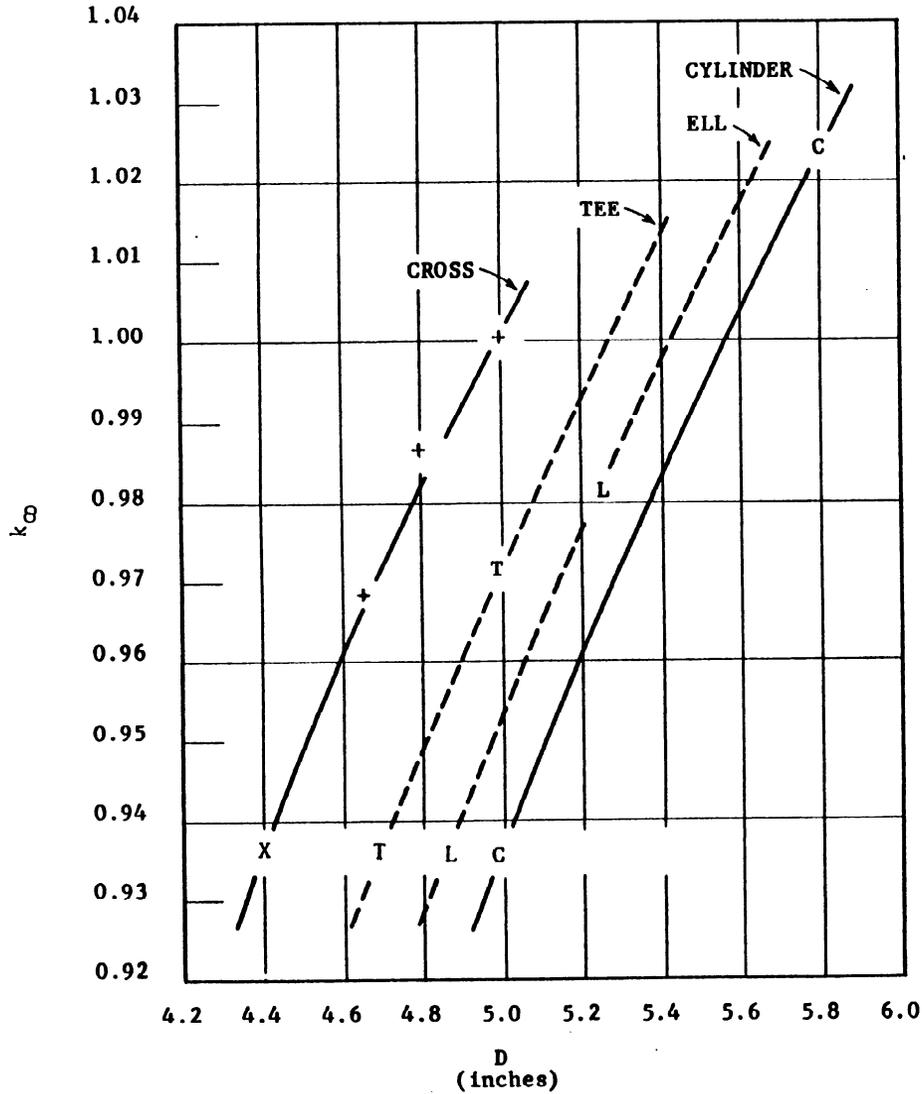
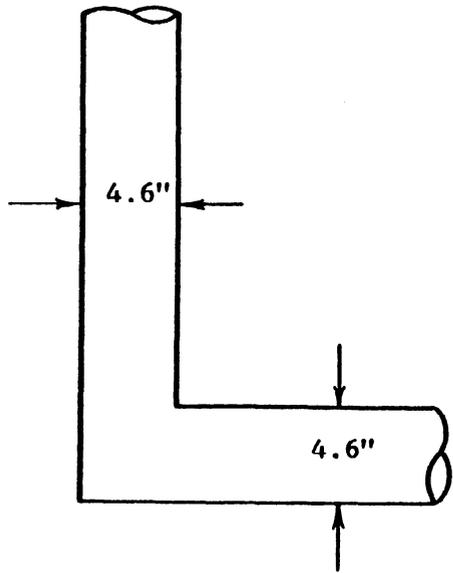
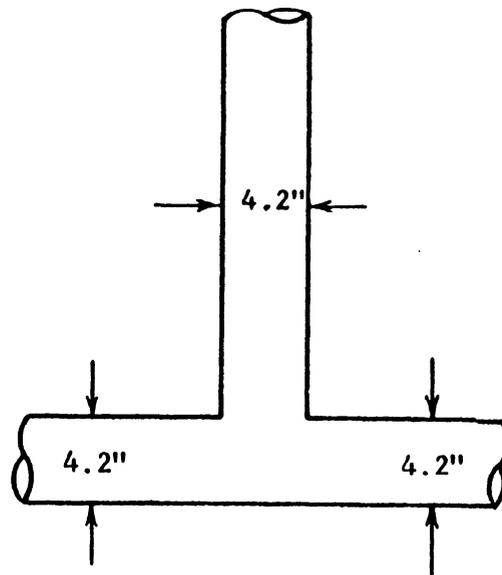


TABLE VIII

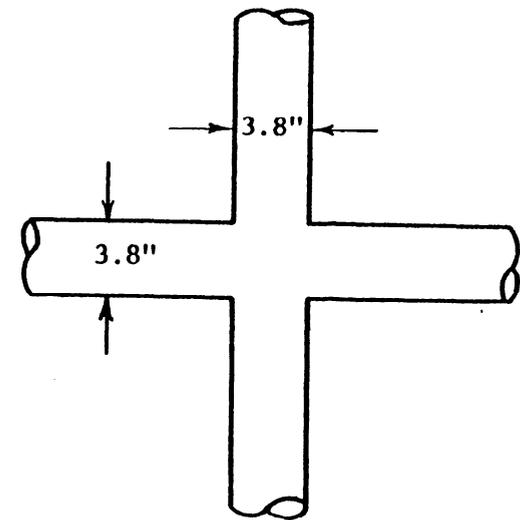
PERMISSIBLE PIPE JUNCTIONS



MAXIMUM I.D. FOR AN "L"



MAXIMUM I.D. FOR A "T"



MAXIMUM I.D. FOR A "+"