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RADIOLOGICAL ANALYSIS OF HYPOTHETICAL ACCIDENTS BY COMPUTER

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SUMMARY

This paper presents the modeling techniques used to analyze accidents hypothesized for FFTF. During the course of the regulatory review it became expeditious to hypothetically test the FFTF containment's ability to protect public health and safety from these extreme accidents. The scenario for these accidents began with an hypothetical core disruptive accident (HCDA) and further hypothesized that none of the provisions for decay heat removal survived the accident. The decay heat caused melt through of the reactor and guard vessels discharging the core debris onto the cavity floor. From this point two bounding cases were presented, the first; Case 718, assumed that the heat from the core debris caused the reactor cavity liner to fail whereupon the sodium attacked the concrete. The second, Case 719, assumed the core debris melted through the floor; at 30.5 hours the floor collapsed and fuel and sodium attacked the bare concrete of the subcavity. Both of these attacks were self limiting; but both would raise the containment vessel atmosphere past its design pressure. Therefore it was assumed that the containment was deliberately vented when the pressure rose to 10 psig. In addition to the original expulsion attributed to the HCDA all fission products which were both volatile and soluble were assumed to be fully dispersed within the sodium and were liberated as the sodium boiled. In addition to a description of the analyses of the various phenomena, the transition codes, those which edit data output by one Code into a form suitable for input to another, are described in this paper. Figure 1 presents a "flow chart" showing the relationship of the codes employed in these analyses. First it was necessary to determine the energies and inventories of the radionuclides present within the reactor. The RIBD Code [1] was used to determine these parameters for the fission products at the "End-of-Equilibrium-Cycle" condition. The table of time vs. decay heat for the noble cases and other fission products was used as input to the CACECO Code [2] which provided the thermophysical transient analysis for the containment.

This fission product inventory data from RIBD and similar data on coolant, fuel and transuranics provided by other organizations within HEDL were combined into a library by the code NEWLYB. This library was to be used later as input to the COMRADEX-H Code.[3] Also included in the library were "Gamma" values and "F" values for each isotope. The "F" values were based on adult inhalation dose commitment factors for various body organs in NUREG-0172.[4]

Since the accident scenario began with a postulated HCDA and the FFTF HCDA assumes a spray expulsion of sodium past the reactor head seals, the resultant sodium spray fire in the reactor head compartment was modeled by the SPRAY Code.[5] Mechanistic analyses have shown that this initial expulsion was limited to approximately 300 lb. The SPRAY analysis shows that although most of the energy of the burning sodium was used in the heating of the massive structures within the head compartment there was a noticeable increase in atmosphere temperatures and pressure. The SPRAY Pressure Transient is shown in Figure 2.

The CACECO analysis was performed concurrently with the SPRAY analysis. The details of the CACECO analysis were described in detail in the companion paper (See Reference 2). As CACECO examined the transient within containment it wrote instantaneous thermodynamic data to a file. Several CACECO runs were required to complete the analysis and each added to the file. When the sequence was complete a program called CCLST was used to prepare the input for the next code. The sodium boil-off rate from the reactor cavity, the leakrate of the containment vessel and the temperature of the containment vessel atmosphere temperature were presented as functions of time in another file which was used as input to the HAA-3C Code.[16]

The HAA-3C Code calculated the fallout within the containment vessel. In addition to the sodium the initial HCDA expulsion was assumed to contain all the noble gases and 1% of the fuel, transuranics, and fission products. The initial aerosol concentration included these materials and the initial atmosphere conditions were as specified by the SPRAY results. Thereafter the sodium boil-off rate from CACECO was used as the aerosol source rate. The containment vessel leakrate and atmosphere temperature from CACECO were applied to enhance the modeling accuracy. A conservative assumption made was that all sodium boiled-off was converted to sodium oxide; actually the oxygen is depleted and the humidity increases so that sodium hydroxide is formed. The hydroxide tends to agglomerate better than the oxide. HAA-3C writes an output file which contains the containment vessel leakrate (from CACECO) and the fallout rate within containment both, as functions of time. Figure 3 presents the suspended aerosol concentration for Case 719 as a function of time. (Case 719 is used for illustration herein since the curves are more dramatic.)

The TRIMIT program edits data from SPRAY, CACECO, and HAA-3C into a series of input files for COMRADEX-H. As the sodium in the reactor/reactor cavity boils, the dissolved volatile fission products are given off also. The TRIMIT program was used to predict the vaporization of cesium and sodium iodide from the sodium pool. The work of A. W. Castleman[7] was utilized to develop this algorithm. Figure 4 shows the fraction of fission product vaporized as a function of sodium vaporization. TRIMIT generates four input files for COMRADEX-H. The first three represent the reactor/reactor cavity leakrate for sodium vapor, cesium vapor, and sodium iodide vapor, respectively. Figure 5 shows the sodium vapor leakrate for Case 719. No fallout is considered within the reactor/reactor cavity; leakrates are expressed in instantaneous fractions of the quantity ultimately boiled away. The fourth file specifies the leakrates and fallout rates of the containment; these are illustrated for Case 719 in Figures 6 and 7.

COMRADEX-H calculates the quantity of each radioactive isotope released from the containment as a function of time and then calculates the potential doses to receptors at specified locations. To model this event the part of the code which calculates releases was executed four times and the results summed. The first modeled the sodium released from the reactor cavity during the boilup phase. The volatile solids arsenic, selenium, and cadmium were assumed to be released at the same rate as the sodium. This run utilized the first and the fourth files from TRIMIT to describe a two chamber containment. The second run utilized the second and fourth TRIMIT files to model the release of cesium and rubidium (which act similarly) from the two chamber containment. The third run modeled the release of Na-I and Na-Br similarly, using the third and fourth TRIMIT files. The final run modeled the initial expulsion of 1000 lb of sodium, all the noble gases, and 1% of the fuel, transuranics, and solid fission products. This last run used only the fourth TRIMIT file to describe a single chamber containment. Figure 8 shows the cumulative release of radionuclides, by physical class, from containment for Case 719. Following release from containment each isotope was diminished by decay (but not by fallout) and the concentration was reduced by atmospheric dispersion enroute to the receptors. The atmospheric dispersion was made to reflect NRC estimates by selecting Pasquill stability classifications and wind speeds. The breathing rate of the receptors was as specified by Reg. Guide 1.4. The dose vs. distance curves for Case 719 are presented as Figure 9.

The doses are lower than might be expected considering the severity of the postulated event. The doses realized for Case 718 are within the guidelines for reactor siting given in 10 CFR 100. And although the thyroid dose for Case 719 is somewhat higher than the guideline value that value was not intended for application to this virtually impossible event.

This type of analysis has been developed by HEDL over the past 5 years and could be adapted to any LMFBR. It has, in fact, been applied to the Clinch River Breeder Reactor. The advantages inherent in this type of analysis are: 1) The relative ease with which modeling can be accomplished, 2) efficient employment of computer resources and manpower, 3) improved accuracy, and 4) better acceptance.

References

1. D. R. Marr, A User's Manual for Computer Code RIBD-II, A Fission Product Inventory Code, Hanford Engineering Development Laboratory, HEDL-TME-75-76, January 1975.

2. R. D. Peak, et. al., "FFTF Containment Transients Associated with Hypothetical Accidents," HEDL-SA-1736, presented at this conference.

3. G. W. Spangler, et. al., Description of the COMRADEX Code, Atomics International, AI-67-TDR-108, 1967. (COMRADEX-H denotes the current version in use at HEDL)

4. G. R. Hoenes and J. K. Soldat, Age-Specific Radiation Dose Commitment Factors for a One-Year Chronic Intake, Battelle Pacific Northwest Laboratories, NUREG-0172, November 1977.

5. P. R. Shire, Spray Code User's Report, HEDL, HEDL-TME-76-94, March 1977.

6. L. Baurmash, et. al., HAA-3 User Report, Atomics International, AI-AEC-13088, 1973. (HAA-3C denotes the current version in use at HEDL)

7. A. W. Castleman, Jr., Review of Fission Product and Aerosol Behavior for LMFBR Systems, Brookhaven National Laboratory, November 1972.

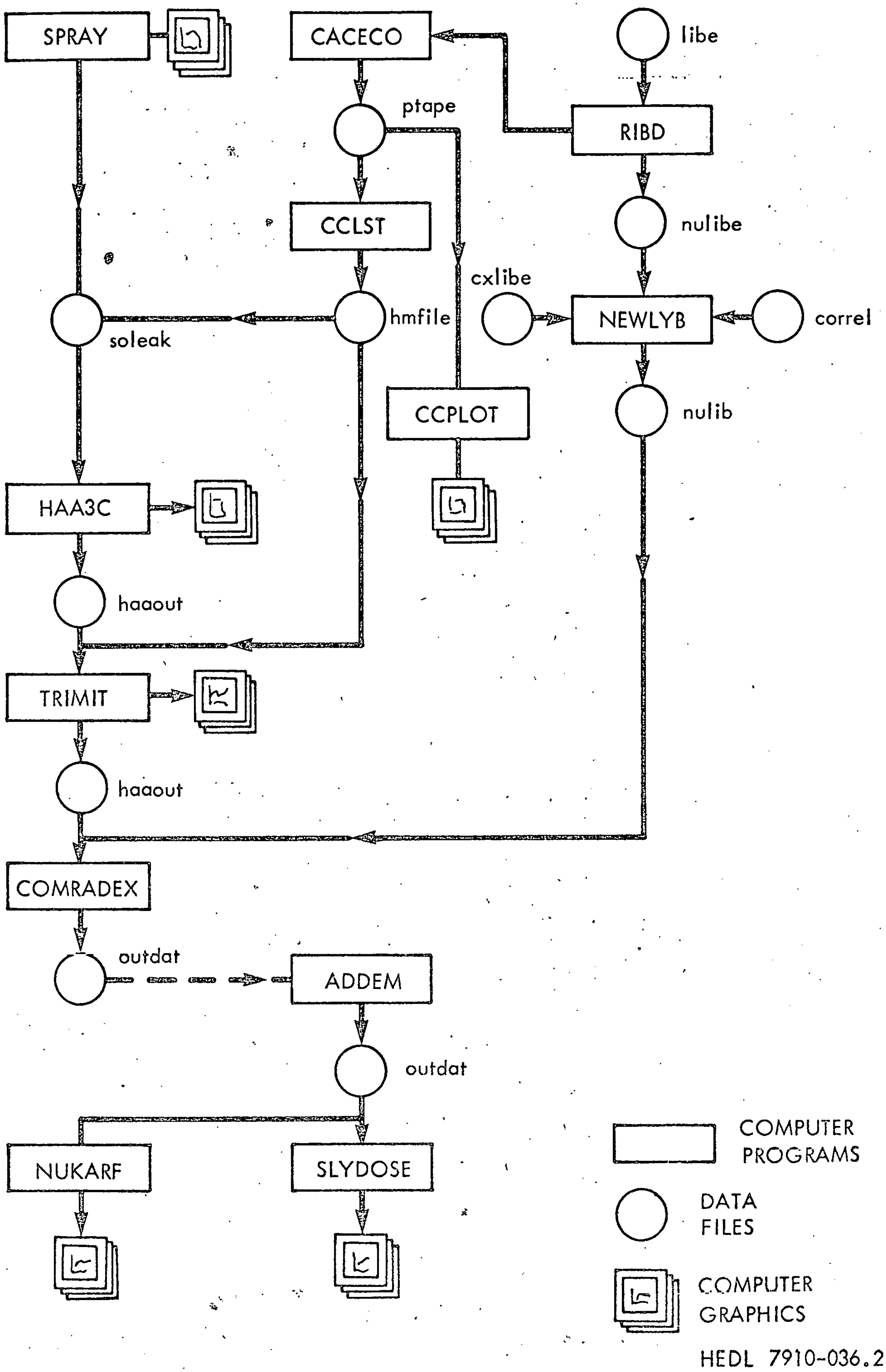


Figure 1 - Relationship of Codes & Files

FIGURE 2

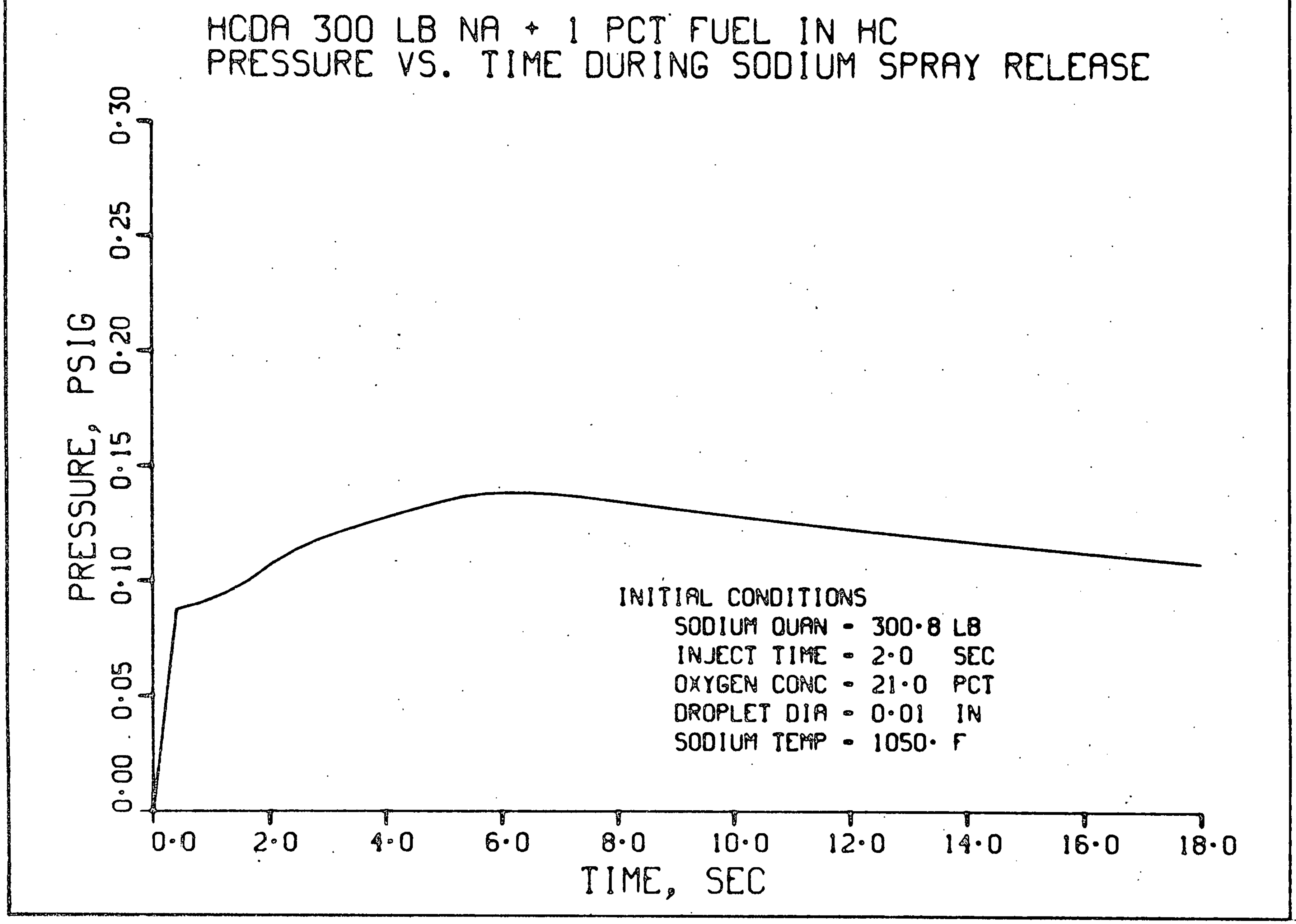
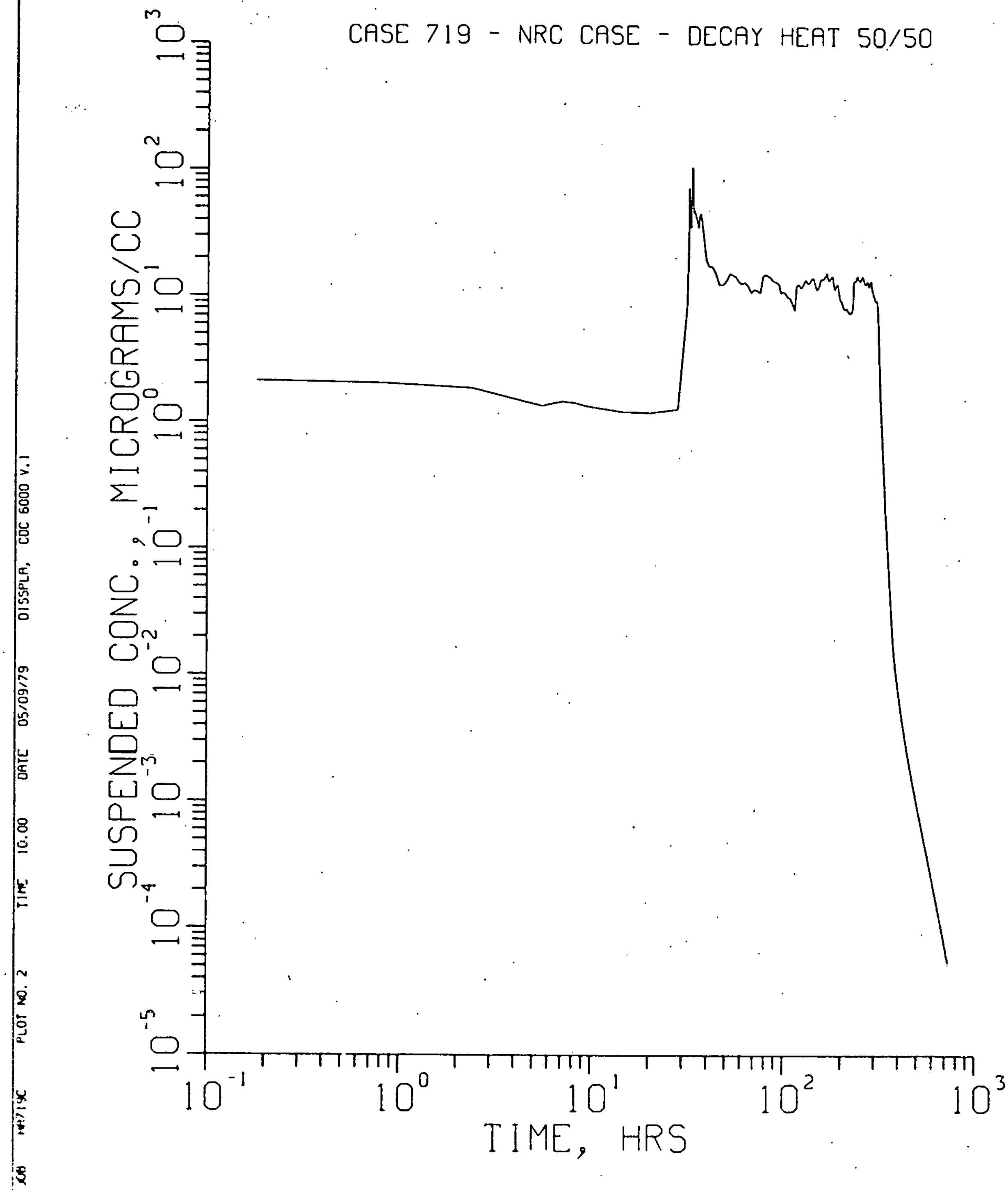

HCDA 300 LB NA + 1 PCT FUEL IN HC
PRESSURE VS. TIME DURING SODIUM SPRAY RELEASE
PRESSURE, PSIG
0·00
0·05
0·10
0·15
0·20
0·25
0·30
0·0
2·0
4·0
6·0
8·0
10·0
12·0
14·0
16·0
18·0
TIME, SEC
INITIAL CONDITIONS
SODIUM QUAN - 300·8 LB
INJECT TIME - 2·0 SEC
OXYGEN CONC - 21·0 PCT
DROPLET DIA - 0·01 IN
SODIUM TEMP - 1050· F

FIGURE 3

HAA SUSPENDED CONC. VS. TIME

CASE 719 - NRC CASE - DECAY HEAT 50/50



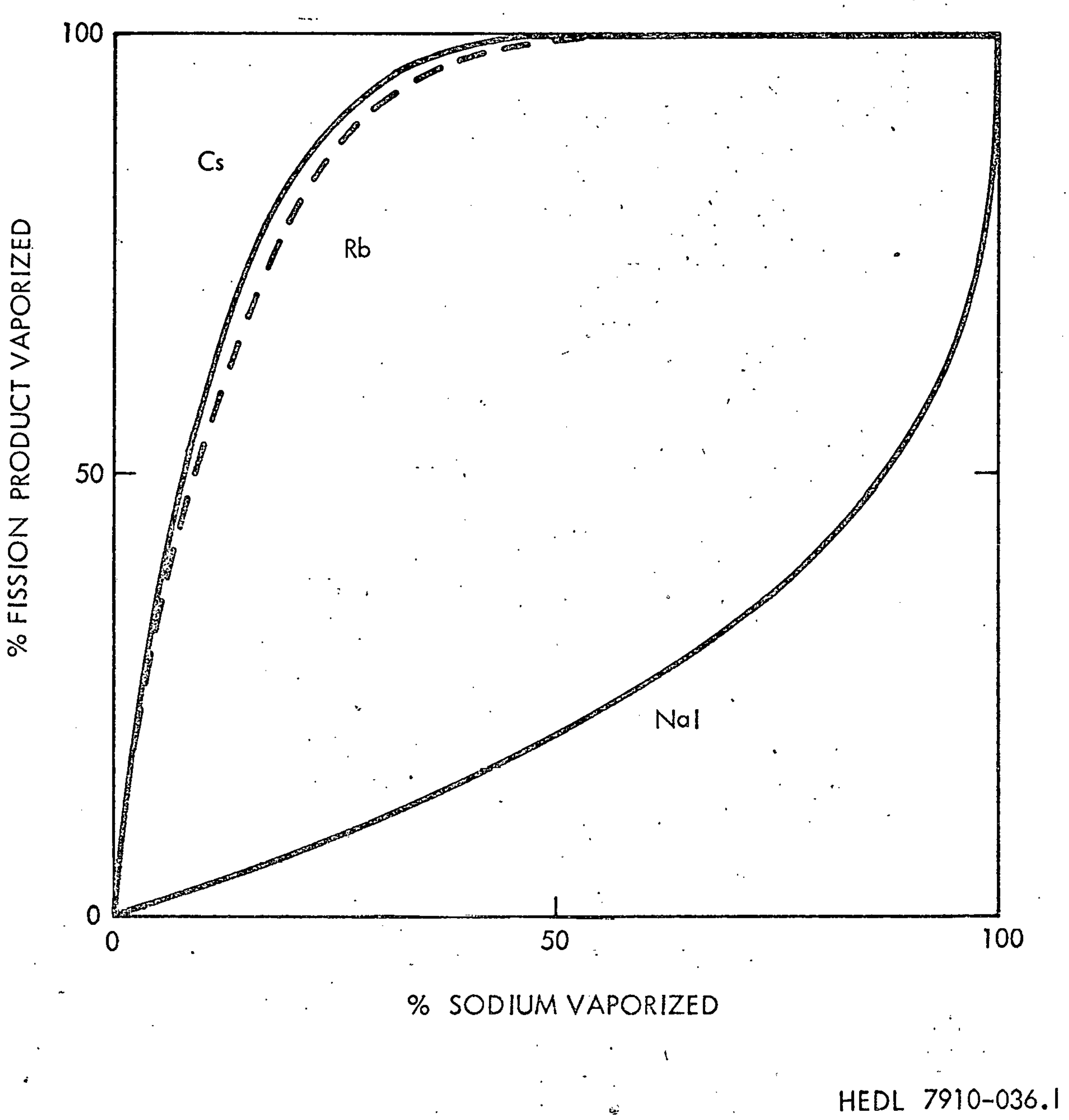
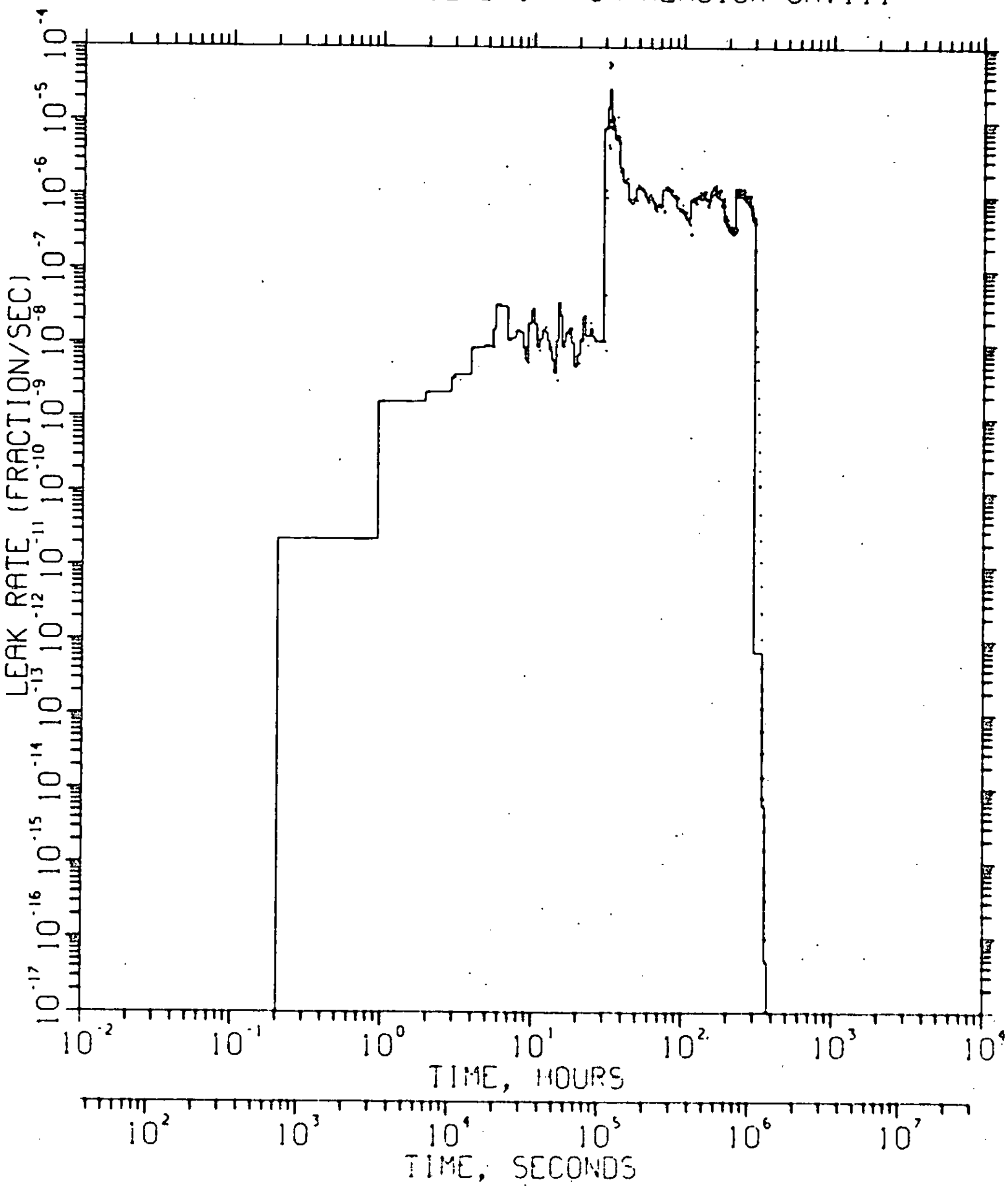


Figure 4 - Fission Product - Sodium Equilibrium Vaporization

FIGURE 5

CASE 719 - DECAY HEAT 50/50

CACECO LEAKRATE DATA FOR REACTOR CAVITY



TIME 10.03 DATE 05/09/79 DISSPLA, CDC 6000 V.1 PLOT NO. 1

FIGURE 6

CASE 719 - NRC CASE - DECAY HEAT 50/50

CONTAINMENT BUILDING LEAKRATE

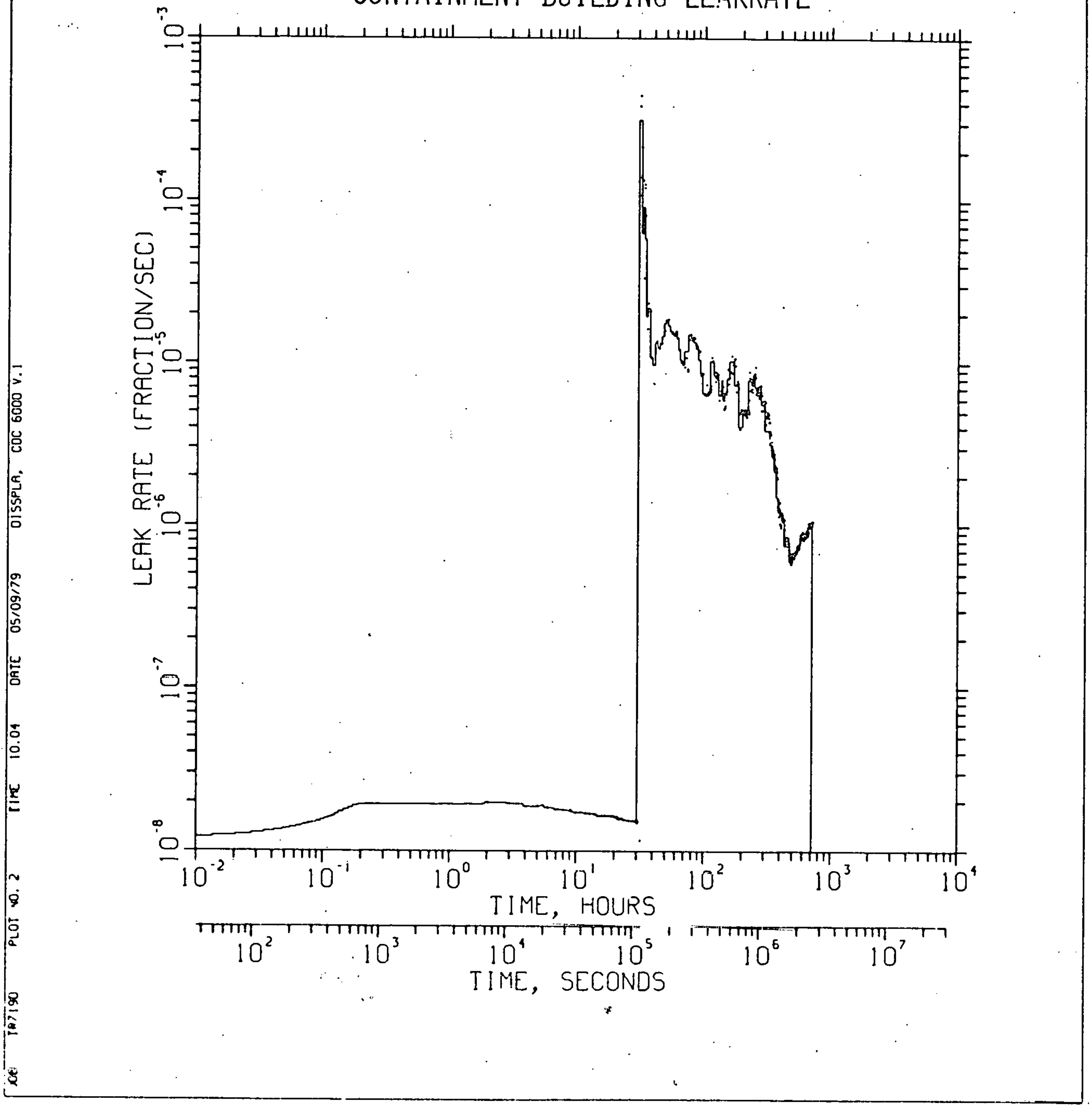
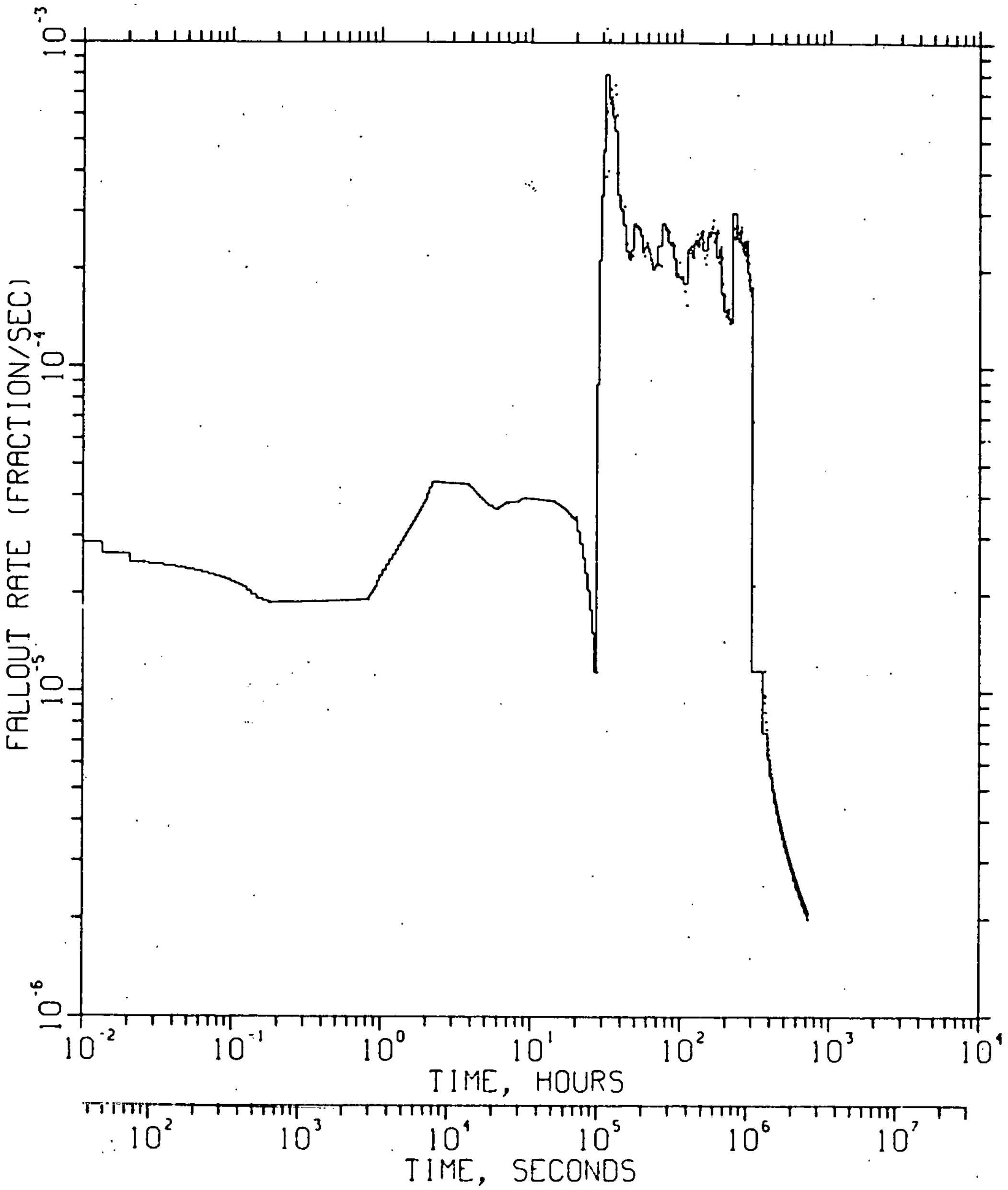


FIGURE 7

CASE 719 - NRC CASE - DECAY HEAT 50/50

FALLOUT RATE DATA BY HAA3C



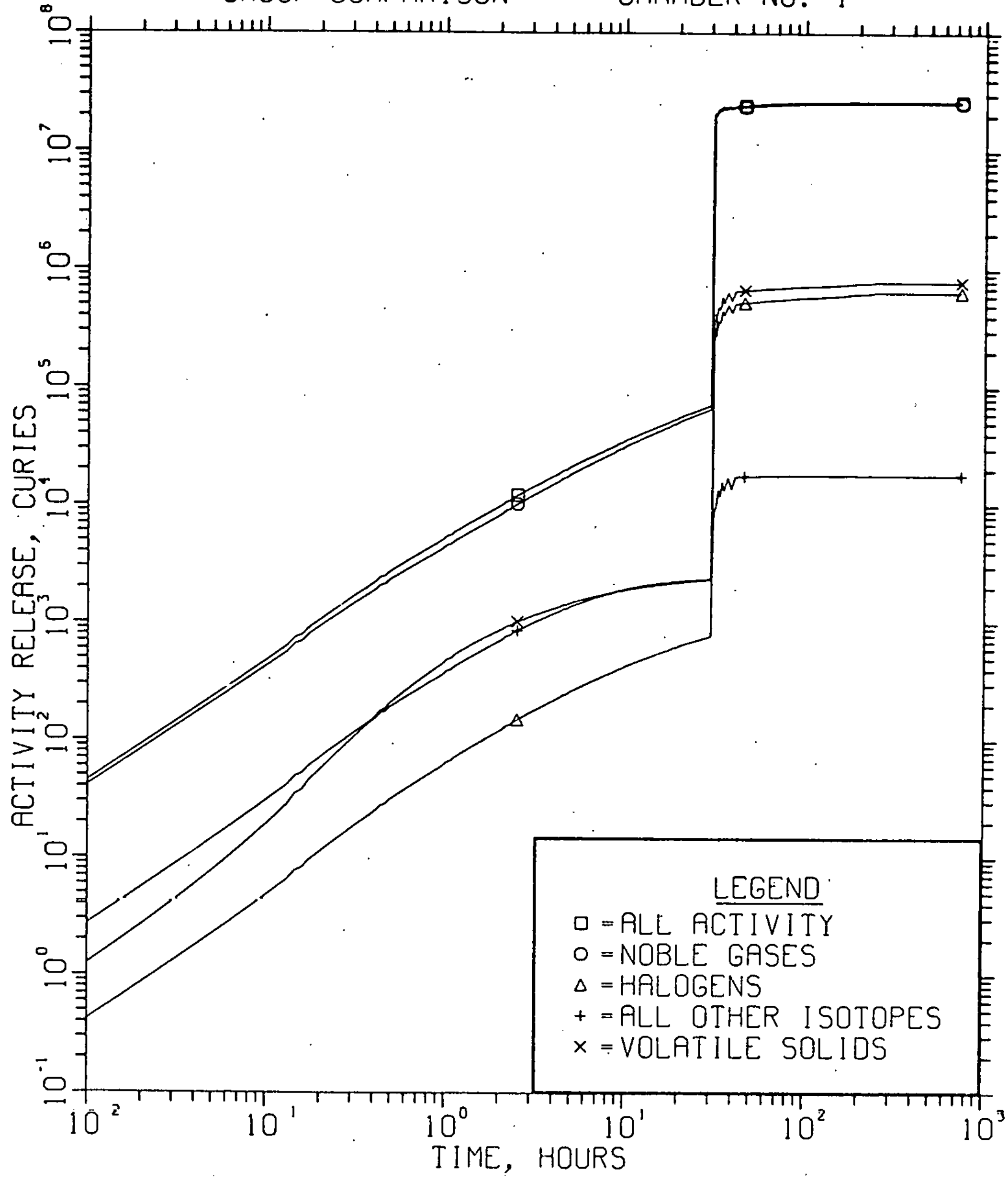
JOB TR7190 PLOT NO. 3 TIME 10.04 DATE 05/09/79 DISSPLA, CDC 6000 V.1

FIGURE 8

CASE 719 - NRC CASE - DECAY HEAT DISTRIBUTED 50/50

RELEASE OF RADIOACTIVITY FROM CONTAINMENT BUILDING

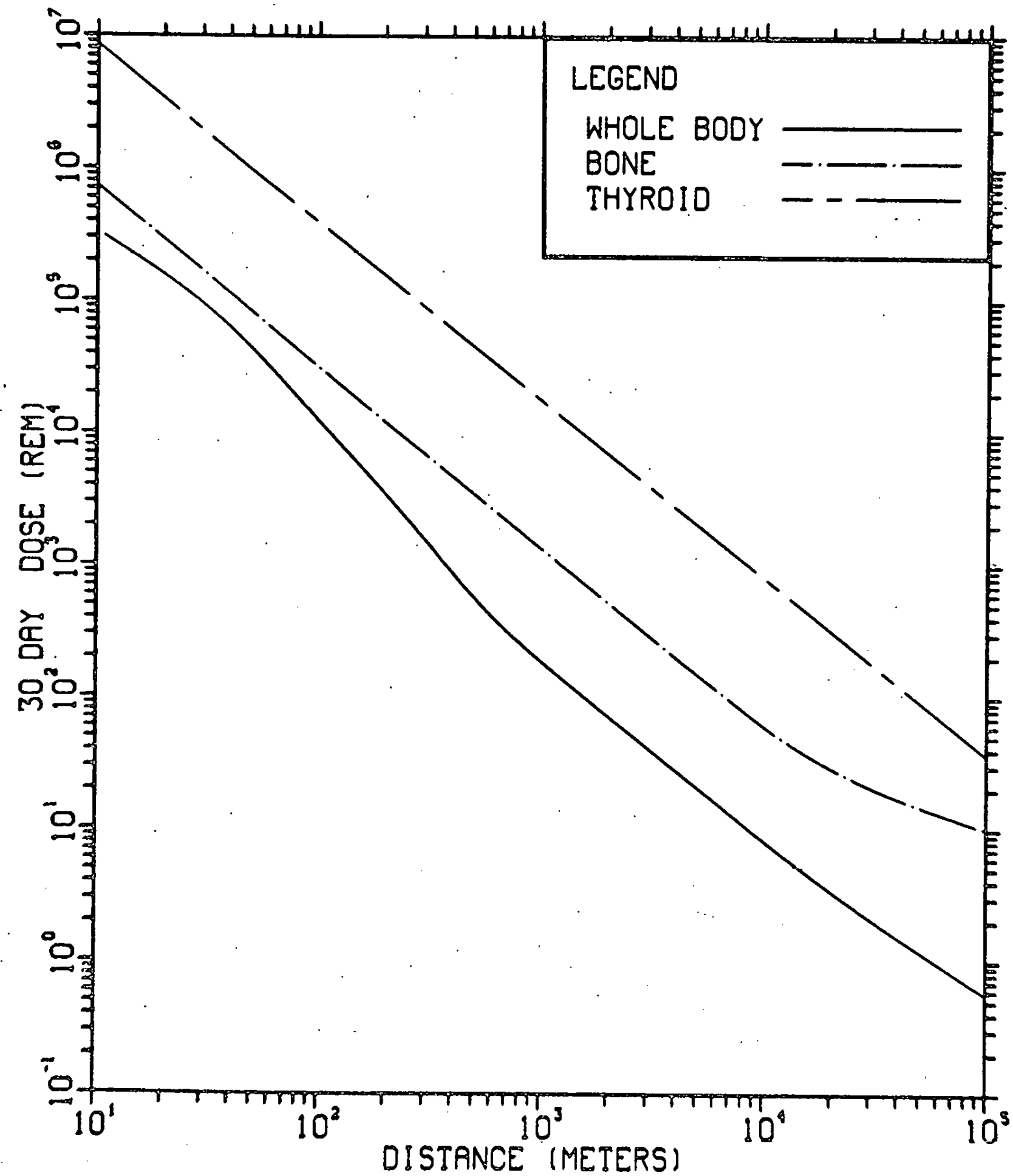
GROUP COMPARISON CHAMBER NO. 1



JOB CR7190 PLOT NO. 2 TIME 10.16 DATE 05/09/79 DISSPLA, CDC 6000 V.1

FIGURE 9

CASE 719 - DOSE VS DISTANCE CURVES



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