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# Source Term Estimation Using MENU-TACT

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**Oak Ridge National Laboratory**

**Prepared for  
U.S. Nuclear Regulatory  
Commission**

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## ABSTRACT

MENU-TACT is a computer code designed to be used for source term estimation during reactor emergencies. The code allows the user to define initial concentrations of the nuclides in the reactor and the release pathway and timing. Filtering and other in-plant removal processes may be included. MENU-TACT, a modification of the TACT-III code, is simple to use. To provide quick results of bounding calculations, it incorporates only those processes that can make major changes in reactor releases.

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## 1. INTRODUCTION

The major role of the U.S. Nuclear Regulatory Commission (NRC) during a severe reactor accident is to monitor actions being taken by the reactor licensee to ensure that these protective actions are appropriate. The NRC must first decide if the plant conditions warrant taking action, based on an assessment of core and containment states. If core damage exists or is projected, the population near the plant (within 2 to 5 miles) should be evacuated, and the remaining population out to 10 miles should be sheltered. This protective action strategy is based on consideration of the consequences of a wide range of core damage accidents. These immediate protective actions should be adequate for most core damage accidents.

This initial decision is not based on a dose assessment. However, determination of the need for additional protective actions will include consideration of dose assessments. In the past, these dose assessments have been based on licensee estimates of releases as measured by the stack monitor. This is not adequate because severe accident releases can bypass the stack and thus may not be characterized by the stack monitor. In addition, since effective protective action requires prompt implementation, an attempt should be made to project the magnitude of a release before it occurs (rather than waiting until it goes out the stack). Therefore, to perform the dose assessment, NRC needs to estimate possible off-site releases based on the probable range of accident conditions and not just on the relationship of stack monitor readings to doses. MENU-TACT is used to estimate these off-site releases.

The consequence estimates should take into consideration current and projected status of the core and possible release pathways as determined by the Reactor Safety Team (RST) or Reactor Safety Coordinator. If calculations are not based on RST assessments, discrepancies in information being provided to NRC decision makers could result. (Remember that the RST submits its assessment of core and containment condition directly to the decision makers.) An assessment of possible releases based on projected plant conditions may also be useful in determining if proposed actions at the plant (e.g., venting



the containment) are appropriate.

By their nature, core melt accidents result in conditions that make estimation of source terms very difficult. The plant is beyond design conditions, instrument readings are unreliable, accident progress is unpredictable, and specific in-plant conditions may be unknown. In addition, even if we had a complete understanding of plant conditions and accident progression, we would still be limited in our understanding of the physics and chemistry of the source term. The result is that, for a core melt accident, we have little hope of predicting "the source term" with much accuracy. Because of these great uncertainties, there is no need to perform complex detailed assessments that consider secondary or little-understood effects. Errors of a factor of 10 to 100 should be expected. The assessments must be based on bounding calculations that account for only dominant effects and can be performed quickly. If a change in assumptions does not change the calculated source term by a factor of 10, it is not worth considering.

For NRC decision makers to be able to use a source term estimate, uncertainties in the estimate must be bounded and the basis of the uncertainties must be explainable. To support any decision, the decision maker needs

1. the best estimate of the magnitude of any off-site release and the assumptions on which these estimates are based and
2. the upper and lower bounds of the off-site release based on reasonable possible courses of the accident or changes in key assumptions.

## 2. MENU-TACT DESCRIPTION

The basic tool used in the Operations Center to estimate source terms is MENU-TACT. NRC's Accident Evaluation Branch developed the TACT-III program (Killough et al., 1983) so that doses and releases from a variety of accidents could be estimated. TACT-III models transport of nuclides from containment through a series of volumes (which are also called nodes) to the environment. The number of nodes, the node volumes, and the pathways through the nodes and to the environment are all entered by the user. TACT uses a matrix inversion routine that solves simultaneous equations describing transport, decay, and removal processes in the volumes. Radionuclide daughter build-up is not considered and should not be important for the time involved in responding to an emergency. The mathematical framework of TACT-III is described in Appendix A. TACT-III has been modified to produce MENU-TACT; the modifications included simplifying the input required to run the model and simplifying the processes modeled. MENU-TACT is written in FORTRAN 77 to be run on a Data General MV 6000.

MENU-TACT has been tested for internal consistency and tested against TACT-III for correctness. Each type of input data was varied from its minimum to its maximum value, while all other input data were kept at their default values. The results were inspected to ensure that the expected changes occurred. These tests demonstrated that MENU-TACT was modeling its processes correctly, regardless of the correctness of TACT-III. More than 30 pairs of MENU-TACT/TACT-III runs were made to ensure that both codes produced the same results. (The greatest difficulty in performing these runs was ensuring that the two sets of input were equivalent. Also, the standard results of MENU-TACT and TACT-III are not directly comparable, so additional "write" statements had to be added to TACT-III.) Four types of tests were run:

1. A baseline case was run with default data.

2. One and two time-step cases were run, with shutdown time equal to process start time and with process start time greater than shutdown time.
3. Two volume cases were run with combinations of large and small volumes and large and small flow rates. (Note that TACT-III was somewhat more robust in handling cases that were numerically extreme.)
4. Large and small filter efficiencies were combined with large and small removal coefficients.

This document explains the use of MENU-TACT during an accident. A full description of data entry is included along with advice on selection of appropriate data values.

### 3. THE INTERMEDIATE DOSE ASSESSMENT SYSTEM MENU ENVIRONMENT

The Intermediate Dose Assessment System (IDAS) menu environment connects all Operations Center model software and all IDAS support software. Data file management and the recording of data entry and simulation results are provided through IDAS. To run a program, you simply select its name from a menu displayed on your terminal. (Currently MENU-TACT runs only on Data General terminals.) Within each program are other menu screens and template screens used to enter data or select options available in the model.

When a menu screen is displayed, you should choose one of the selections displayed. The current selection is the one that is highlighted. To choose a different option, use the arrow keys. The up arrow and the left arrow move the highlighted box up. The right arrow and the down arrow move the highlighted box down.

A menu screen consists of at most three parts: (1) one or two title lines, (2) between two and 20 selections, and sometimes (3) a hint line on the bottom of the screen. You can sometimes obtain further help about the highlighted selection by pressing the "H" key. Currently, only a few help items are available.

If you come to any IDAS screen and realize that you want to go back to a previous screen, press the ESCAPE key. If you are looking at the first screen in a program, the ESCAPE key will cause the program to stop.

A template screen requests a fill-in-the-blanks type of input. The template screens have no fixed format, but usually a title is at the top of the screen and a "helpful hint" is at the bottom. The data fields in between are labeled by name and have default units. Usually the fields already have data in them. If they do and you want to use the value shown, press the NEWLINE key when that item is highlighted. Otherwise you can type in the value you want. Remember that any characters in the data field that are past the cursor position are not read. If you catch a mistake before you press the NEWLINE key, you can correct it by backing up within that line using the arrow or DEL key. Once you have pressed the NEWLINE key, MENU-TACT will check that your entry is valid (i.e., it is of the proper type and within the acceptable range of

values). If your value is invalid, an error message will appear on the bottom of the screen for 5 sec and the cursor will return to the beginning of the item you are trying to enter.

No help messages are currently available for the data items on the template screens. Also, the IDAS "special input" commands are disabled.

The ESCAPE key works in the template screens similarly to the way it works in the menu screens. At any data item within a template screen, pressing the ESCAPE key will put you back at the previous data item. If you are positioned on the first item of a template screen, the ESCAPE key will put you back to the top of the previous screen.

#### 4. STEPPING THROUGH MENU-TACT

The major steps in performing a MENU-TACT analysis are:

1. reading this document;
2. preparing the release pathway form;
3. logging onto the system and branching into MENU-TACT;
4. selecting the plant site (and case);
5. specifying basic time-dependent information:
  - a. case description,
  - b. power level,
  - c. node names and volumes, and
  - d. shutdown time;
6. specifying core damage state, either
  - a. coolant release,
  - b. gap release,
  - c. grain boundary release, or
  - d. core melt release; and
7. specifying pathway data:

- a. event timing,
- b. removal coefficient for each node,
- c. portion of the release initially injected into each node, and
- d. transfer rates between nodes and between nodes and the environment and filter efficiencies assumed for transfers.

Refer to Fig. 1 while reading the following discussion about use of the code.

#### 4.1 LOGGING ONTO THE SYSTEM AND BRANCHING INTO MENU-TACT

When a terminal is turned on but is not logged into the system, the following message will generally be displayed on the screen:

```
**** MV6000B IMS/IDAS / Press NEW-LINE to begin logging on ****
```

To initiate the login sequence, press the NEWLINE key. The system will respond with the message:

```
AOS/VS 6.04.00.00 / EXEC 6.04.00.00      6-Aug-86 11:18:29      @CON10
Username: sourceterm
Password:
```

Enter your username and press the NEWLINE key. The system will prompt for your password. Enter your password and press NEWLINE. The system will respond with a series of messages and the display:

```
Last message change      6-Jun-86      12:22:00
```

```
*****
United States Nuclear Regulatory Commission
Data General Eclipse MV/6000 -- System B -- IDAS
AOS/VS Revision 6.04
```

```
*****
```

```
-----
Last previous logon      6-Aug-86      9:19:54
```

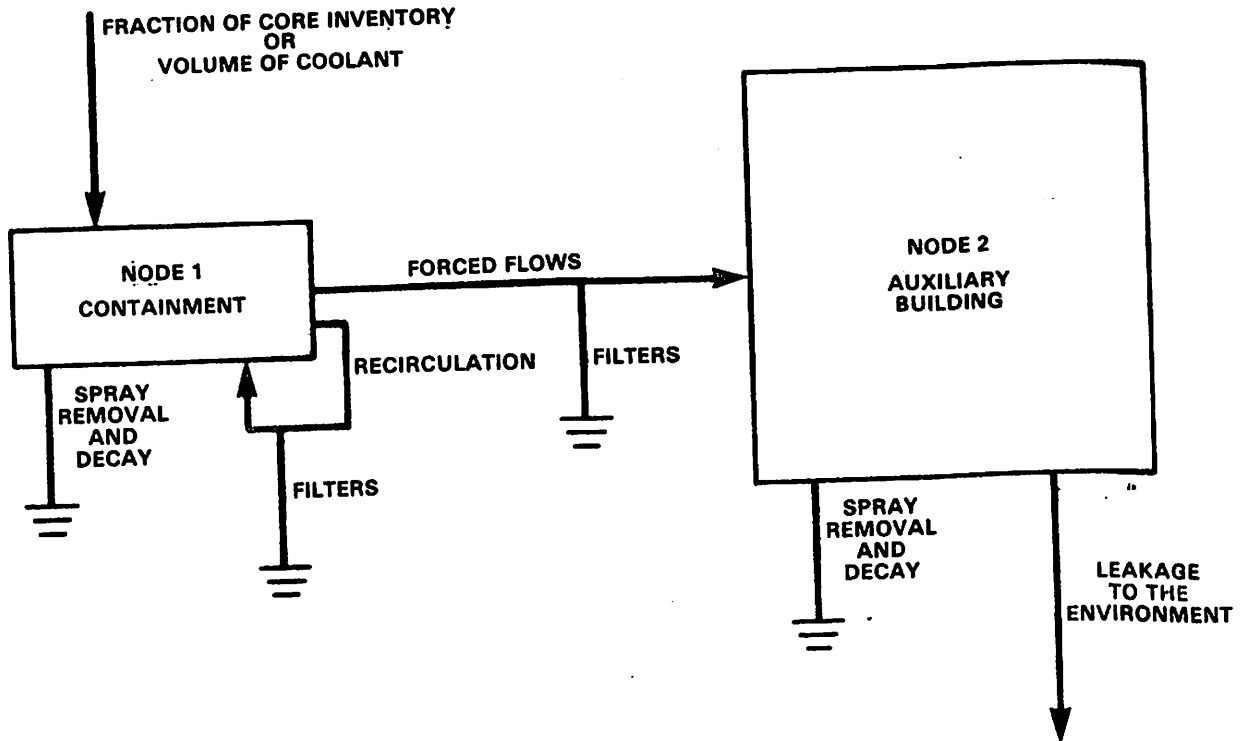


Fig. 1. Example MENU-TACT flow.



After a short time, the screen will clear and the highest-level IDAS menu will be displayed.

The highest-level IDAS menu allows you to branch to the various models within the system. To start MENU-TACT, use the arrow keys to move the selection window to the TACT line and press NEWLINE. The following message will be displayed:

STARTING MENU-TACT....

## 4.2 SELECTION PLANT SITE AND BRANCHING THROUGH MENU-TACT

### 4.2.1 Selecting the Plant Site

The first screen in MENU-TACT allows you to select the location of the site to be modeled. The following table is displayed.

PICK THE SITE THAT YOU WISH TO GENERATE SOURCE TERM FOR.

|                    |                     |                        |
|--------------------|---------------------|------------------------|
| 1. ARKANSAS        | 23. GINNA           | 45. QUAD CITIES        |
| 2. BEAVER VALLEY   | 24. GRAND GULF      | 46. RANCHO SECO        |
| 3. BIG ROCK POINT  | 25. HADDAM NECK     | 47. ROBINSON           |
| 4. BRAIDWOOD       | 26. HATCH           | 48. SALEM/HOPE CREEK   |
| 5. BROWNS FERRY    | 27. INDIAN POINT    | 49. SAN ONOFRE         |
| 6. BRUNSWICK       | 28. KEWAUNEE        | 50. SEABROOK           |
| 7. BYRON           | 29. LA CROSSE       | 51. SEQUOYAH           |
| 8. CALLAWAY        | 30. LASALLE         | 52. ST LUCIE           |
| 9. CALVERT CLIFFS  | 31. MAINE YANKEE    | 53. SUMMER             |
| 10. CLINTON        | 32. MCGUIRE         | 54. SURRY              |
| 11. COMANCHE PEAK  | 33. MILLSTONE       | 55. SUSQUEHANNA        |
| 12. COOK           | 34. MONTICELLO      | 56. THREE MILE ISLAND  |
| 13. COOPER STATION | 35. NINE MILE POINT | 57. TROJAN             |
| 14. CRYSTAL RIVER  | 36. NORTH ANNA      | 58. TURKEY POINT       |
| 15. DAVIS BESSE    | 37. OCONEE          | 59. VERMONT YANKEE     |
| 16. DIABLO CANYON  | 38. OYSTER CREEK    | 60. VOGTLE             |
| 17. DRESDEN        | 39. PALISADES       | 61. WATERFORD          |
| 18. DUANE ARNOLD   | 40. PALO VERDE      | 62. WATTS BAR          |
| 19. ENRICO FERMI   | 41. PEACH BOTTOM    | 63. WASHINGTON NUCLEAR |
| 20. FARLEY         | 42. PILGRIM         | 64. WOLF CREEK         |
| 21. FORT CALHOUN   | 43. POINT BEACH     | 65. YANKEE ROWE        |
| 22. FORT ST VRAIN  | 44. PRAIRIE ISLAND  | 66. ZION               |

Enter the number for the site you wish to use here: 1 \_\_\_\_\_

Specify the appropriate site by entering the number listed next to the site name. This selects a unique four-letter identifier, which is incorporated into the names of the MENU-TACT input and output files, which are both saved in the IDAS data base. The code also uses the site name selected to look up some site-specific data.

If the site you selected has multiple units and these units have different physical characteristics, you will then be asked which unit you wish to consider, as shown below.

Enter unit number for ARKANSAS : 1,2,

1\_\_\_\_\_

Unit-specific data for the unit you select will be displayed on subsequent data input screens.

After site selection, the program displays the MENU-TACT main menu.

#### 4.2.2 Selecting the Path Through MENU-TACT (Main Menu)

The main MENU-TACT menu is shown below:

```
NRC - IDAS TACT MAIN MENU
IDAS TACT - MENU VERSION
```

```
CREATE A NEW CASE
MODIFY AN EXISTING CASE
RETURN TO IDAS MAIN MENU
```

USE ARROW KEYS FOR SELECTION AND (CR) WHEN SELECTION IS THE CORRECT ONE

This menu has two forward paths and one path back to the main IDAS menu.

#### 4.2.3 Modifying an Existing Case

If you select the case modification option, the screen will list existing cases for the site (and unit) you select. If no cases exist for this site, an error message is printed and you return to the previous menu. To select the case you want to modify, scroll the names

up (with the down arrow) or down (with the up arrow) to the highlighted box and press NEWLINE. MENU-TACT will display the following message while it loads the case you have selected:

WORKING

After you make your selection, the case name you requested and its description will be shown. You will be asked if this is the case you want to use.

```

                                CONFIRM CASE TO BE USED
CASE NAME: ARK1_008
CASE DESCRIPTION : test
USE THIS CASE?      Y_ type Y or N

```

If you reply with an 'N', you will return to a previous menu. The next display is an internal menu, shown below.

```

NRC - IDAS TACT MODIFICATIONS OPTIONS
IDAS TACT - MENU VERSION

CHANGE TIME INDEPENDENT DATA
CHANGE TIME DEPENDENT DATA
CHANGE INITIAL SOURCE OF ACTIVITY
REVIEW MIX
RUN THIS CASE
RUN A DIFFERENT CASE
RETURN TO IDAS MAIN MENU

```

USE ARROW KEYS FOR SELECTION AND (CR) WHEN SELECTION IS THE CORRECT ONE

This menu provides three forward choices and two backward choices. You may run this case as it is. If you choose one of the data modification options, you will step through the same screens that are used in creating a new case, as described in Sect. 4.2.4.

#### 4.2.4 Creating a New Case

If you select the new case option, the system will request a case name as shown below:

```
Record name      =   ARK1_027      :
```

```
USE THIS CASE?   Y_  type Y or N
```

The case name you enter will replace the default case name. The prefix that identifies this site (e.g., BVRV for Beaver Valley) is assigned by the system. Note that this case name is also used to identify the MENU-TACT output file that can be read by MESORAD. Then the screen is cleared and the system prompts:

```
Do you wish to initialize this record with data
from a previous record?  N__
```

If you enter YES or Y, the screen will clear and display a list of previously defined cases that can be used to initialize this case. This is the same screen that appears in response to the case modification selection on the menu. This option allows you to create several cases, with unique names, from the same set of initial information without starting over for each case.

#### 4.3 SPECIFYING BASIC TIME-INDEPENDENT DATA

The code will prompt for basic data as shown below:

```
          BASIC TIME INDEPENDENT DATA FOR PWR ARKANSAS 1
BASIC CASE DESCRIPTION/CASE NAME:
test
NUMBER OF NODES (NODES <=4): 2
NAME OF NODES   VOLUME IN FT**3
NODE(1) = Primary   .1000E+07
NODE(2) = Contain.  .1900E+07
```

```
REACTOR POWER (MEGAWATTS THERMAL) : 2568.
REACTOR SHUTDOWN TIME - 24 HOUR CLOCK SITE TIME
enter 2 digits and NEWLINE 5 times: 87/08/04 14:00      (yr mo da hr mn)
```

```
=====
ARK1:REFERENCE VOLUMES (ft**3)
  Primary system   9.7E+03
  Containment      1.9E+06
```

```
ARE YOU DONE WITH THIS SCREEN ?   Y__
```

To change or enter data, move through the screen using the NEWLINE key to go forward (down) and the ESCAPE key to go back (up). If a case was initialized from a previous case or is being modified, the existing values will not be seen until you have stepped through to that entry.

You should always provide a basic description of the case by revising the default title shown. This description will be used later to identify the case. The default description is '...'. .

Next you must specify the number of nodes (or volumes) and node names to be considered for this case. MENU-TACT requires a name for each node. The volumes of the default nodes named for this unit are always displayed on the bottom of the screen. Note that the environment is not considered to be a node. Data relating to the release pathway to the environment will be requested later. You should include only those volumes that account for major changes in the amount of activity released. Since these are bounding calculations, if changes in the assumptions do not change the end result by more than a factor of 10, they are not worth considering. Although up to four nodes can be considered, you should usually use only two or three nodes. Tables 1 and 2 list the major volumes and release pathways. As an aid in identifying and showing the assumed release pathway, two generic forms have been developed (see Figs. 2 and 3). The assumed release pathway (along with any assumed release rates, etc.) should be clearly marked on this form with a colored marker. simple. It is a good idea to have the release pathway form marked before running MENU-TACT. The forms should be passed along with the rest of the analysis as an aid in discussing the results. You must specify the volume of each node in  $\text{ft}^3$ ; volumes must be greater than zero. The primary system, containment, and where appropriate, secondary containment volumes for the plant are shown at the bottom of the screen, for reference. In the case of boiling water reactor (BWR) plants, only one total primary containment volume is specified, representing the total of the components. If additional information is needed, consult the Final Safety Analysis Report (FSAR).

Reactor power level (thermal) at the time of the accident must be specified and will be used to determine the initial amount of isotopes (Ci) available in the core. The default value displayed on this screen is the rated thermal level of the plant.

Table 1. Pressurized water  
reactor major release pathways<sup>a</sup>

| Pathway                | Incident   |
|------------------------|--|
| A. Primary system      | <ol style="list-style-type: none"> <li>1. Primary system leak               <ol style="list-style-type: none"> <li>a. Small</li> <li>b. Large</li> </ol> </li> <li>2. Steam generator tube rupture</li> <li>3. Bypass accident (e.g., to low-pressure system outside containment)</li> </ol>         |
| B. Primary containment | <ol style="list-style-type: none"> <li>1. Leakage (0.1% to 0.5% per day)</li> <li>2. Failure to isolate (100% per day)</li> <li>3. Major failure (100% per hr)               <ol style="list-style-type: none"> <li>a. Overpressurization</li> <li>b. H<sub>2</sub> explosion</li> </ol> </li> </ol> |
| C. Secondary system    | <ol style="list-style-type: none"> <li>1. Secondary safety relief valve</li> <li>2. Turbine exhaust</li> </ol>   |

<sup>a</sup>See Fig. 1.

Table 2. Boiling water reactor MK I generic major release pathways<sup>a</sup>

| Pathway                                   | Incident   |
|---|--|
| A. Primary system                         | <ol style="list-style-type: none"> <li>1. Primary system break/leak (loss-of-coolant accident)</li> <li>2. Blowdown through pool</li> <li>3. Main steam isolation valve leakage</li> <li>4. Bypass accident (e.g., control rod drive break outside containment)</li> </ol> |
| B. Primary containment                    | <ol style="list-style-type: none"> <li>1. Leakage (0.5% per day)</li> <li>2. Failure (e.g., overpressurization) (100% per day)</li> <li>3. Nonisolation (100% per hr)</li> </ol>   |
| C. Reactor/refueling/<br>turbine building | <ol style="list-style-type: none"> <li>1. Leakage</li> <li>2. Blowout</li> <li>3. Standby gas treatment system <ol style="list-style-type: none"> <li>a. Filtered</li> <li>b. Blowout</li> </ol> </li> </ol>   |

<sup>a</sup>See Fig. 3.

| CORE CONDITIONS                                      | FUEL TEMPERATURE INDICATION | PRESENT CONDITION | PROJECTED CONDITION AT 2-3 HRS |
|--|-----------------------------|-------------------|--------------------------------|
| 1) Core Intact - Large Coolant Leak                  | 600 F                       | ---               | ---                            |
| 2) Clad Failure (Rupture/Oxidation 20% of Fuel Pins) | 1300 - 2000 F               | ---               | ---                            |
| 3) TRU Like (Grain Boundary Release)                 | > 2400 F for 10 Min.        | ---               | ---                            |
| 4) Core Melt   | > 4500 F                    | ---               | ---                            |

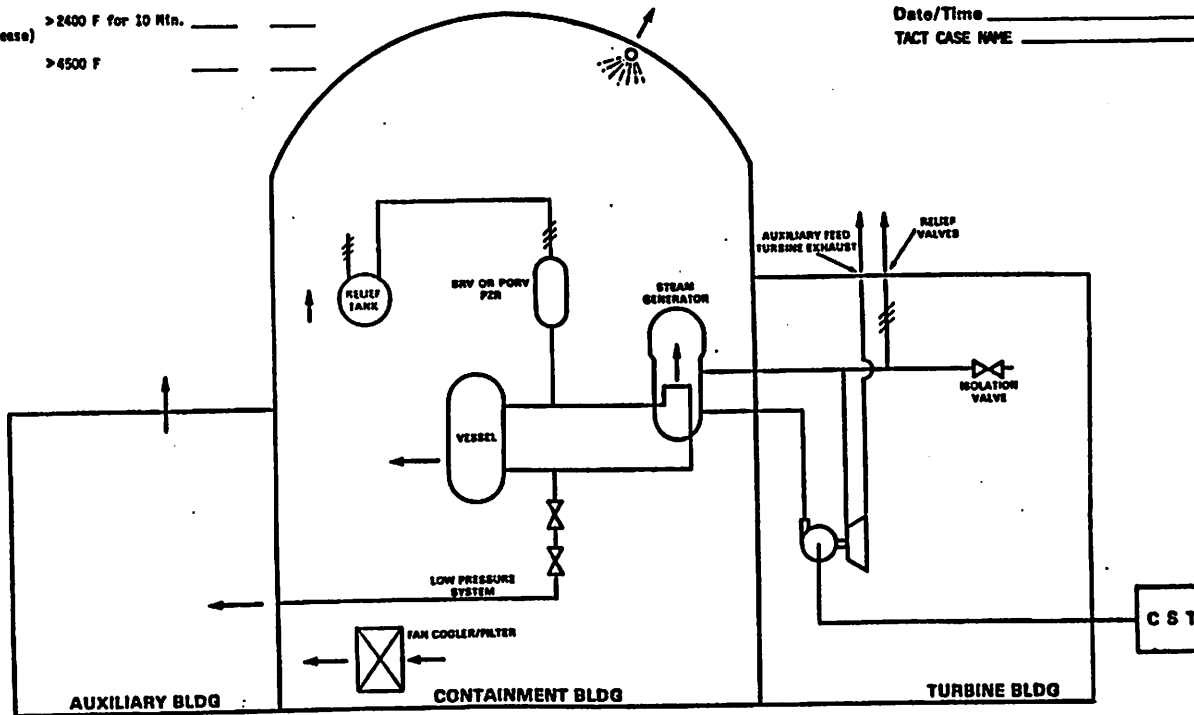
FORM 2.2P

PWR Release Pathway Worksheet

Prepared By \_\_\_\_\_

Date/Time \_\_\_\_\_

TACT CASE NAME \_\_\_\_\_



PWR Generic Major Release Pathways After Isolation

LEGEND

- ☉ Spray
- ⊠ Filter
- ⊠ Transfer Path

MAJOR NODES

- A Vessel (Primary System)
- B Primary Containment
- C Secondary System
- D Turbine Building
- E Auxiliary Building

Fig. 2. Pressurized water reactor release pathway form.



| <u>CORE CONDITIONS</u>                                     | <u>FUEL TEMPERATURE INDICATION</u> | <u>PRESENT CONDITION</u> | <u>PROJECTED CONDITION AT 2-3 HRS</u> |
|--|------------------------------------|--------------------------|---------------------------------------|
| 1) Core Intact-<br>Large Coolant Leak                      | 600 F                              | ---                      | ---                                   |
| 2) Clad Failure<br>(Rupture/Oxidation<br>20% of Fuel Pins) | 1300 - 2000 F                      | ---                      | ---                                   |
| 3) TMI Like<br>(Grain Boundary Release)                    | > 2400 F for 10 Min.               | ---                      | ---                                   |
| 4) Core Melt   | > 4500 F                           | ---                      | ---                                   |

FORM 2.2B

BWR Release  
 Pathway Worksheet  
 Prepared By \_\_\_\_\_  
 Date/Time \_\_\_\_\_  
 TACT CASE NAME \_\_\_\_\_

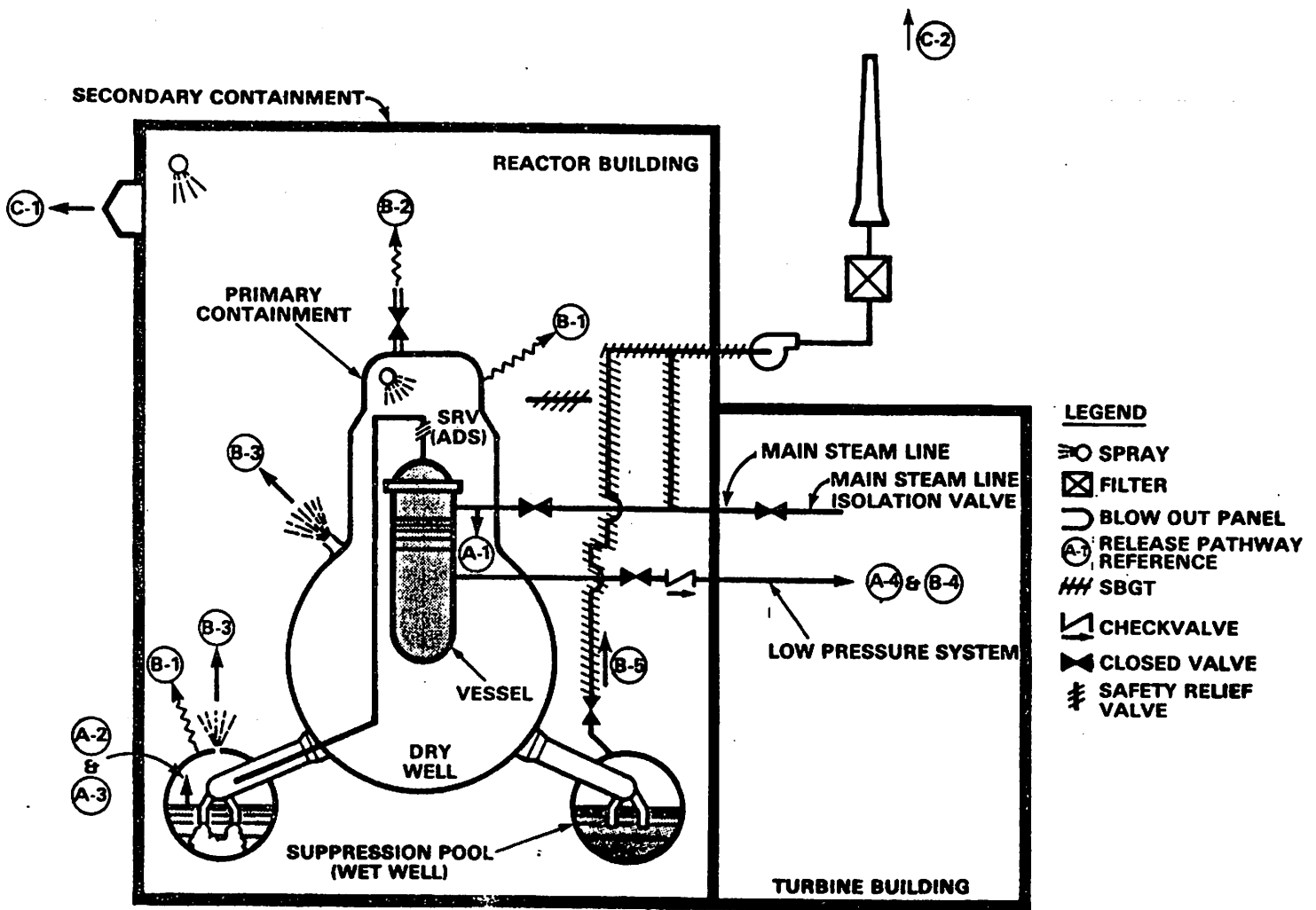


Figure 3. BWR Mark I release pathway form.

Finally, shutdown time must be specified. You should use local time at the reactor. The code asks for the date (Yr/Mo/Da) and time (Hr:Mn). The date and time consist of five numbers, so you must press NEWLINE five times to pass through these fields. Shutdown time can be less than process start time to account for decay prior to the beginning of release from the core.

#### 4.4 SPECIFYING CORE DAMAGE STATE

The next step is to define the amount and composition of material that will be released from the core or vessel. At this point, you are selecting the mix of isotopes that will be included in the analysis. This material will be acted on as it proceeds through the reactor until it is released to the environment, thereby becoming the "source term."

You must select one of four types of core damage: (1) coolant release, (2) gap release, (3) grain boundary release, or (4) core melt. In the following display, 'NG' stands for 'noble gases'. The revised coolant option ('REV. COOLANT') is explained below.

```

SELECT THE ACTIVITY SOURCE YOU DESIRE.
THE SELECTION SHOWN IS THE DEFAULT OR THE SELECTION LAST PICKED.
THE SELECTIONS ARE:
1) COOLANT (typical) (600 F)
2) GAP (5% Cs 3% NG, 2% I, 0.1% Te, Sb) (1300 - 2000 F)
3) GRAIN BOUNDARY (50% NG, I, Cs; 10% Te; 2% Sb; 1% Ba, Mo; .1% Sr; .01% Ru) (3000 F)
4) CORE MELT (100% NG, Cs, I, Rb; 30% Te; 20% Ba; 7% Ru; .01% La, Y, Np, Ce, Sb) (>4500 F)
5) REV. COOLANT (Enter Ci/gm of each isotope separately)

```

SOURCE TERM IS: 4

DO YOU WISH TO REVIEW THE MIX?(Y OR N):y\_\_

These levels of core damage were selected because they represent the major damage regimes that can be associated with plant conditions and release fractions. The core temperature associated with each of the damage states is shown as part of the prompt along with a summary of the assumed release fractions. The radionuclides that dominate bone marrow (whole body), lung, and thyroid dose are specified, along with noble gases. Whole body and lung doses were selected because WASH-1400 (USNRC, 1975) concluded that these dominate early health effects and deaths. Noble gases were included because these are most likely to be released during an accident. Table 3 shows the relationship between

Table 3. Overview of TACT release assumptions

| Core conditions                       | Fuel temperature (°F) | TACT assumed release fractions from core                                     |
|---------------------------------------|-----------------------|--|
| Core intact, large coolant leak       | <600                  | Typical  |
| Gap (clad failure, rupture/oxidation) | 1300-2000             | 5% Cs<br>3% NG<br>2% I<br>0.01% Te, Sb                                       |
| Grain boundary release                | 3000                  | 50% NG, I, Cs<br>10% Te<br>2% Sb<br>1% Ba, Mo<br>0.1% Sr<br>0.01% Ru         |
| Core melt                             | >4500                 | 100% NG, Cs, I, Rb<br>30% Te<br>20% Ba<br>0.7% Ru<br>0.01% La, Y, Np, Ce, Sb |

Table 4. Specific release assumptions

| ISOTOPE<br># NAME | DECAY<br>1/HR | SOURCE<br>CI/MWT | PWR---COOL---BWR       |         | GAP<br>FRACT | G.B.<br>FRACT | MELT<br>FRACT |
|-------------------|---------------|------------------|------------------------|---------|--------------|---------------|---------------|
|                   |               |                  | CONCENTRATION<br>CI/GM |         |              |               |               |
| 1 I-131           | 3.6E-03       | 2.8E+04          | 5.0E-09                | 5.0E-09 | 2.0E-02      | 5.0E-01       | 1.0E+00       |
| 2 I-132           | 3.0E-01       | 4.1E+04          | 3.0E-08                | 3.0E-08 | 2.0E-02      | 5.0E-01       | 1.0E+00       |
| 3 I-133           | 3.3E-02       | 5.9E+04          | 2.0E-08                | 2.0E-08 | 2.0E-02      | 5.0E-01       | 1.0E+00       |
| 4 I-134           | 7.9E-01       | 6.5E+04          | 5.0E-08                | 5.0E-08 | 2.0E-02      | 5.0E-01       | 1.0E+00       |
| 5 I-135           | 1.1E-01       | 5.5E+04          | 2.0E-08                | 2.0E-08 | 2.0E-02      | 5.0E-01       | 1.0E+00       |
| 6 KR-85           | 7.4E-06       | 1.9E+02          | 0.0E-01                | 0.0E-01 | 3.0E-02      | 5.0E-01       | 1.0E+00       |
| 7 KR-85M          | 1.5E-01       | 7.9E+03          | 0.0E-01                | 0.0E-01 | 3.0E-02      | 5.0E-01       | 1.0E+00       |
| 8 KR-87           | 5.4E-01       | 1.5E+04          | 0.0E-01                | 0.0E-01 | 3.0E-02      | 5.0E-01       | 1.0E+00       |
| 9 KR-88           | 2.4E-01       | 2.2E+04          | 0.0E-01                | 0.0E-01 | 3.0E-02      | 5.0E-01       | 1.0E+00       |
| 10 XE-131M        | 2.4E-03       | 3.7E+02          | 0.0E-01                | 0.0E-01 | 3.0E-02      | 5.0E-01       | 1.0E+00       |
| 11 XE-133M        | 1.3E-02       | 2.1E+03          | 0.0E-01                | 0.0E-01 | 3.0E-02      | 5.0E-01       | 1.0E+00       |
| 12 XE-133         | 5.5E-03       | 5.7E+04          | 0.0E-01                | 0.0E-01 | 3.0E-02      | 5.0E-01       | 1.0E+00       |
| 13 XE-135         | 7.6E-02       | 1.2E+04          | 0.0E-01                | 0.0E-01 | 3.0E-02      | 5.0E-01       | 1.0E+00       |
| 14 XE-138         | 2.9E+00       | 1.2E+04          | 0.0E-01                | 0.0E-01 | 3.0E-02      | 5.0E-01       | 1.0E+00       |
| 15 CS-134         | 3.8E-05       | 2.3E+03          | 3.0E-11                | 3.0E-11 | 5.0E-02      | 5.0E-01       | 1.0E+00       |
| 16 CS-136         | 2.2E-03       | 9.2E+02          | 2.0E-11                | 2.0E-11 | 5.0E-02      | 5.0E-01       | 1.0E+00       |
| 17 CS-137         | 2.6E-06       | 2.0E+03          | 7.0E-11                | 7.0E-11 | 5.0E-02      | 5.0E-01       | 1.0E+00       |
| 18 TE-129M        | 8.6E-04       | 6.4E+02          | 4.0E-11                | 4.0E-11 | 1.0E-04      | 1.0E-01       | 3.0E-01       |
| 19 TE-131M        | 2.3E-02       | 4.1E+03          | 1.0E-10                | 1.0E-10 | 1.0E-04      | 1.0E-01       | 3.0E-01       |
| 20 TE-132         | 8.9E-03       | 4.1E+04          | 1.0E-11                | 1.0E-11 | 1.0E-04      | 1.0E-01       | 3.0E-01       |
| 21 SB-127         | 7.5E-03       | 3.5E+03          | 0.0E-01                | 0.0E-01 | 1.0E-04      | 2.0E-02       | 2.0E-02       |
| 22 SB-129         | 7.4E-02       | 9.0E+03          | 0.0E-01                | 0.0E-01 | 1.0E-04      | 2.0E-02       | 2.0E-02       |
| 23 SR-89          | 5.7E-04       | 3.4E+04          | 1.0E-10                | 1.0E-10 | 0.0E-01      | 1.0E-03       | 7.0E-02       |
| 24 SR-90          | 2.8E-06       | 1.7E+03          | 6.0E-12                | 6.0E-12 | 0.0E-01      | 1.0E-03       | 7.0E-02       |
| 25 SR-91          | 7.3E-02       | 3.6E+04          | 4.0E-09                | 4.0E-09 | 0.0E-01      | 1.0E-03       | 7.0E-02       |
| 26 MO-99          | 0.0E-01       | 5.2E+04          | 2.0E-09                | 2.0E-09 | 0.0E-01      | 1.0E-02       | 1.0E-01       |
| 27 RU-103         | 7.3E-04       | 4.2E+04          | 2.0E-11                | 2.0E-11 | 0.0E-01      | 1.0E-04       | 7.0E-03       |
| 28 RU-106         | 7.8E-05       | 1.1E+04          | 3.0E-12                | 3.0E-12 | 0.0E-01      | 1.0E-04       | 7.0E-03       |
| 29 BA-140         | 2.3E-03       | 5.1E+04          | 4.0E-10                | 4.0E-10 | 0.0E-01      | 1.0E-02       | 2.0E-01       |
| 30 Y-91           | 4.9E-04       | 3.7E+04          | 4.0E-11                | 4.0E-11 | 0.0E-01      | 0.0E-01       | 1.0E-04       |
| 31 LA-140         | 1.7E-02       | 5.4E+04          | 0.0E-01                | 0.0E-01 | 0.0E-01      | 0.0E-01       | 1.0E-04       |
| 32 CE-144         | 1.0E-04       | 3.0E+04          | 3.0E-12                | 3.0E-12 | 0.0E-01      | 0.0E-01       | 1.0E-04       |
| 33 NP-239         | 0.0E-01       | 5.5E+05          | 7.0E-09                | 7.0E-09 | 0.0E-01      | 0.0E-01       | 1.0E-04       |

core damage, assumed radionuclide release fractions, and core temperature. Specific release fractions assumed for each radionuclide are shown in Table 4. Assumed inventories [Ci/MW(t)] were for a 22,000 MWd/ton burn-up case. This is typical of normal burn up. Coolant levels are based on American National Standards Institute/American Nuclear Society (ANSI/ANS) 18.1-1984, "Radioactive Source Term for Normal Operation of Light Water Reactors." These are typical of normal coolant levels.

Actual coolant sample concentrations may be known and could be considerably different than those assumed. Therefore, the assumed levels may require modification to correspond to a known coolant sample. This can be done by multiplying (1) the assumed iodine levels by a scaling factor, RELFRI, and (2) the noble gases by RELFRN as discussed below. Using this method, actual coolant concentrations normally are within a factor of 2 of those defined in MENU-TACT for all isotopes. This is well within the uncertainties associated with coolant release and transport. This is particularly true if the sample was taken in front of the vent and an iodine spike has occurred or core damage has progressed. An iodine spike is characterized by a rapid increase in iodine concentration in the coolant by as much as three orders of magnitude. Iodine spikes have been observed to occur following shutdowns, start-ups, rapid power changes, and coolant depressurizations. During 1983, spikes occurred at 10 pressurized water reactors. Reliance on a pre-event sample has obvious shortcomings. You should expect to revise the coolant concentrations used in your analysis as the event proceeds.

A second method of specifying the concentrations in the coolant allows you to enter specific coolant concentrations for each isotope. This is the fifth option on the menu. When you select it, a screen appears with all available nuclide names and their default coolant concentrations in Ci/g. You may select the default value or enter a new one for each isotope. These data are stored with the rest of the data for this case, so that data entered here will appear as the default values in subsequent modifications of this case.

REVISED COOLANT MIX (Ci/gm)  
FOR ARKANSAS 1

|            |          |            |          |           |          |
|------------|----------|------------|----------|-----------|----------|
| 1 I-131    | .220E-08 | 14 XE-138  | .000E+00 | 27 RU-103 | .200E-10 |
| 2 I-132    | .220E-07 | 15 CS-134  | .300E-10 | 28 RU-106 | .300E-11 |
| 3 I-133    | .150E-07 | 16 CS-136  | .200E-10 | 29 BA-140 | .400E-09 |
| 4 I-134    | .430E-07 | 17 CS-137  | .800E-10 | 30 Y-91   | .400E-10 |
| 5 I-135    | .220E-07 | 18 TE-129M | .400E-10 | 31 LA-140 | .400E-09 |
| 6 KR-85    | .000E+00 | 19 TE-131M | .100E-09 | 32 CE-144 | .300E-11 |
| 7 KR-85M   | .000E+00 | 20 TE-132  | .100E-10 | 33 NP-239 | .800E-08 |
| 8 KR-87    | .000E+00 | 21 SB-127  | .000E+00 |           |          |
| 9 KR-88    | .000E+00 | 22 SB-129  | .000E+00 |           |          |
| 10 XE-131M | .000E+00 | 23 SR-89   | .100E-09 |           |          |
| 11 XE-133M | .000E+00 | 24 SR-90   | .700E-11 |           |          |
| 12 XE-133  | .000E+00 | 25 SR-91   | .400E-08 |           |          |
| 13 XE-135  | .000E+00 | 26 MO-99   | .200E-08 |           |          |

DO YOU WANT TO MODIFY THESE VALUES? N

The gap and grain boundary release fraction are based on NUREG-0772 (USNRC, 1981). These relationships are based on very sparse data and are subject to great uncertainties. These release fractions are estimated to be correct only to within one order of magnitude, and these values are single-point estimates of a continuum of release fractions as a function of temperature. Therefore, even if you predict the core damage state accurately, your estimate of initial material released is very rough (as is all of the subsequent analysis).

The core melt release fractions are typical of the results of core melt accidents discussed in NUREG-0957 (USNRC, 1985).

For gap, grain boundary, or core melt, the specific amount (Ci) of material released from the core is in terms of Ci/MW(t) and is scaled by the code based on the actual reactor power level. These amounts may not be adjusted individually. The assumed inventory is for an "old" core (33,000 MWd/ton burn up). If the core is only a few months old (e.g., Three Mile Island at the time of the accident), the long half-life radionuclides will be at a much lower level. This can be seen from Table 5, which shows an example of the buildup of Sr-90 and Cs-137 as a function of core life. While this can have a major effect on activity, the long-half-life radionuclides are not the most important contributors to whole body dose. The variation between these concentrations and the actual concentrations should be well within the uncertainties of the analysis and is not considered in the code.

Table 5. Strontium and cesium inventories vs core age<sup>a</sup>

| Core age<br>(months) | Inventory<br>(Ci) |                   |
|----------------------|-------------------|-------------------|
|                      | Sr-90             | Cs-137            |
| 2                    | $4.5 \times 10^5$ | $5.4 \times 10^5$ |
| 3                    | $6.8 \times 10^5$ | $8.2 \times 10^5$ |
| 12                   | $2.5 \times 10^6$ | $3.0 \times 10^6$ |
| 24                   | $4.0 \times 10^6$ | $5.0 \times 10^6$ |

<sup>a</sup>Based on Case 1.B of  
NUREG/CR-3108.

## 4.5 REVIEWING THE ASSUMED MIX

The mix assumed for each core damage state or coolant concentration can be reviewed. If you decide to review the mix, the following screen is shown.

## \*\* ISOTOPE DATA USED IN CALCULATIONS \*\*

| ISOTOPE<br># NAME         | DECAY<br>1/HR | SOURCE<br>CI/MWT | PWR---COOL---BWR       |                        | GAP<br>FRACT | G.B.<br>FRACT | MELT<br>FRACT |
|---------------------------|---------------|------------------|------------------------|------------------------|--------------|---------------|---------------|
|                           |               |                  | CONCENTRATION<br>CI/GM | CONCENTRATION<br>CI/GM |              |               |               |
| 1 I-131                   | 3.6E-03       | 2.8E+04          | 4.5E-08                | 2.2E-09                | 2.0E-02      | 5.0E-01       | 1.0E+00       |
| 2 I-132                   | 3.0E-01       | 4.1E+04          | 2.1E-07                | 2.2E-08                | 2.0E-02      | 5.0E-01       | 1.0E+00       |
| 3 I-133                   | 3.3E-02       | 5.9E+04          | 1.4E-07                | 1.5E-08                | 2.0E-02      | 5.0E-01       | 1.0E+00       |
| 4 I-134                   | 7.9E-01       | 6.5E+04          | 3.4E-07                | 4.3E-08                | 2.0E-02      | 5.0E-01       | 1.0E+00       |
| 5 I-135                   | 1.1E-01       | 5.5E+04          | 2.6E-07                | 2.2E-08                | 2.0E-02      | 5.0E-01       | 1.0E+00       |
| 6 KR-85                   | 7.4E-06       | 1.9E+02          | 4.3E-07                | 0.0E-01                | 3.0E-02      | 5.0E-01       | 1.0E+00       |
| 7 KR-85M                  | 1.5E-01       | 7.9E+03          | 1.6E-07                | 0.0E-01                | 3.0E-02      | 5.0E-01       | 1.0E+00       |
| 8 KR-87                   | 5.4E-01       | 1.5E+04          | 1.5E-07                | 0.0E-01                | 3.0E-02      | 5.0E-01       | 1.0E+00       |
| 9 KR-88                   | 2.4E-01       | 2.2E+04          | 2.8E-07                | 0.0E-01                | 3.0E-02      | 5.0E-01       | 1.0E+00       |
| 10 XE-131M                | 2.4E-03       | 3.7E+02          | 7.3E-07                | 0.0E-01                | 3.0E-02      | 5.0E-01       | 1.0E+00       |
| 11 XE-133M                | 1.3E-02       | 2.1E+03          | 7.0E-08                | 0.0E-01                | 3.0E-02      | 5.0E-01       | 1.0E+00       |
| 12 XE-133                 | 5.5E-03       | 5.7E+04          | 2.6E-06                | 0.0E-01                | 3.0E-02      | 5.0E-01       | 1.0E+00       |
| 13 XE-135                 | 7.6E-02       | 1.2E+04          | 8.5E-07                | 0.0E-01                | 3.0E-02      | 5.0E-01       | 1.0E+00       |
| 14 XE-138                 | 2.9E+00       | 1.2E+04          | 1.2E-07                | 0.0E-01                | 3.0E-02      | 5.0E-01       | 1.0E+00       |
| 15 CS-134                 | 3.8E-05       | 2.3E+03          | 7.1E-09                | 3.0E-11                | 5.0E-02      | 5.0E-01       | 1.0E+00       |
| 16 CS-136                 | 2.2E-03       | 9.2E+02          | 8.7E-10                | 2.0E-11                | 5.0E-02      | 5.0E-01       | 1.0E+00       |
| 17 CS-137                 | 2.6E-06       | 2.0E+03          | 9.4E-09                | 8.0E-11                | 5.0E-02      | 5.0E-01       | 1.0E+00       |
| 18 TE-129M                | 8.6E-04       | 6.4E+02          | 1.9E-10                | 4.0E-11                | 1.0E-04      | 1.0E-01       | 3.0E-01       |
| PRESS NEWLINE TO CONTINUE |               |                  |                        |                        |              |               |               |
| 19 TE-131M                | 2.3E-02       | 4.1E+03          | 1.5E-09                | 1.0E-10                | 1.0E-04      | 1.0E-01       | 3.0E-01       |
| 20 TE-132                 | 8.9E-03       | 4.1E+04          | 1.7E-09                | 1.0E-11                | 1.0E-04      | 1.0E-01       | 3.0E-01       |
| 21 SB-127                 | 7.5E-03       | 3.5E+03          | 0.0E-01                | 0.0E-01                | 1.0E-04      | 2.0E-02       | 2.0E-02       |
| 22 SB-129                 | 7.4E-02       | 9.0E+03          | 0.0E-01                | 0.0E-01                | 1.0E-04      | 2.0E-02       | 2.0E-02       |
| 23 SR-89                  | 5.7E-04       | 3.4E+04          | 1.4E-10                | 1.0E-10                | 0.0E-01      | 1.0E-03       | 7.0E-02       |
| 24 SR-90                  | 2.8E-06       | 1.7E+03          | 1.2E-11                | 7.0E-12                | 0.0E-01      | 1.0E-03       | 7.0E-02       |
| 25 SR-91                  | 7.3E-02       | 3.6E+04          | 9.6E-10                | 4.0E-09                | 0.0E-01      | 1.0E-03       | 7.0E-02       |
| 26 MO-99                  | 1.0E-02       | 5.2E+04          | 6.4E-09                | 2.0E-09                | 0.0E-01      | 1.0E-02       | 1.0E-01       |
| 27 RU-103                 | 7.3E-04       | 4.2E+04          | 7.5E-09                | 2.0E-11                | 0.0E-01      | 1.0E-04       | 7.0E-03       |
| 28 RU-106                 | 7.8E-05       | 1.1E+04          | 9.6E-08                | 3.0E-12                | 0.0E-01      | 1.0E-04       | 7.0E-03       |
| 29 BA-140                 | 2.3E-0        |                  |                        |                        |              |               |               |

Note that this listing is for the initial assumed mix for each core damage state and does not reflect any changes resulting from multiplication by the factors RELFRI or RELFRN.



#### 4.6 SPECIFYING THE FRACTION OF INITIAL ACTIVITY RELEASED FROM THE CORE

Once the initial core damage state is chosen, you can adjust the assumed release fractions as mentioned above. MENU-TACT will request the fraction of noble gases (Xe, Kr) and particulates and aerosols (other radionuclides), which should be used as shown below:

##### ADJUSTMENT OF DEFAULT INVENTORY

FRACTION OF NOBLE GASES = 1.000  
 FRACTION OF PARTICULATE/AEROSOLS = 1.000 \_\_\_\_\_

These scaling factors are used to increase or decrease the default number of Curies available in the inventory for release from the core or in the coolant.

ASSUMED MIX IS FOR CORE MELT

Note that the core damage state selected earlier (e.g., core melt) is displayed.

The initial core release inventories assumed will be multiplied by these fractions. You must specify two factors, one for noble gases (RELFRN) and one for the other radionuclides (RELFRI). These factors can be greater or less than 1. They can be used to

1. adjust the default coolant activities to match actual measured coolant concentrations,
2. account for plate-out or other removal mechanisms in the vessel, or
3. adjust the default damage states up or down.

You should not change these values from the default value (i.e., 1), unless vessel removal or actual coolant sample results are being considered; otherwise, the relationship to a known damage state will be lost.

#### 4.7 SPECIFYING TIME/NODE-DEPENDENT DATA

Three times must be entered to define process starts and stop. For each node, the following data must be provided:

1. a particulate/aerosol (non-noble gas) removal coefficient,  $\lambda$  ( $\text{ft}^3/\text{hr}$ );
2. the portion of the released isotopes injected into each volume (node) at the start of the problem ( $\text{ft}^3/\text{hr}$  for coolant releases and fraction/hr for other releases);
3. the rate at which material is transferred from one node to another (fraction/hr or  $\text{ft}^3/\text{hr}$ ); and
4. the filter efficiency of any filter mechanism encountered during transfer between or within each node ( $\text{hr}^{-1}$ ).

These are discussed in detail in Sects. 4.7.1 through 4.7.4.

4.7.1 Specifying Process Start and Release Start and Stop Times, and Particulate/Aerosol Removal Coefficients ( $\lambda$ )

The prompt for these data follows:

TIME INDEPENDENT DATA, SHUTDOWN TIME IS 87/08/04 14:00

PROCESS START TIME: 87/08/04 14:00 24 HOUR CLOCK SITE TIME  
 START OF RELEASE: 87/08/04 14:00 (Yr/Mo/Da Hr:Mn)  
 RELEASE END TIME: 87/08/04 15:00

PARTICULATE/AEROSOL/IODINE REMOVAL COEFFICIENTS (LAMBDA)  
 NODE NAME COEFFICIENT(LAMBDA/HR)  
 1 Primary .0000E+00  
 2 Contain. .0000E+00\_\_\_\_\_

=====

LAMBDA FOR LARGE VOLUMES

| HOLD UP TIME | SPRAY ON | SPRAY OFF |
|--------------|----------|-----------|
| 1 hr         | 3.5      | 2.0       |
| 4 hr         | 1.0      | 0.87      |
| 6 hr         | 0.7      | 0.58      |

#### 4.7.1.1 Specifying process start and release start and stop times

The process start time refers to the beginning of the release from the core. The release start time refers to the beginning of the release from the nodes to the environment. Shutdown, start of the process, and start of the release may be at the same time. As can be seen in Fig. 4, all removal and transfer processes defined to operate in-plant act from the start of the process to the end of the release. Processes allowing release to the environment act only from the start of the release to the end of the release. Radioactive decay of isotopes occurs from shutdown until the end of the release.

#### 4.7.1.2 Specifying volume removal coefficients ( $\lambda$ )

This coefficient implements the processes that remove a fraction per hour of the particulates and aerosols (non-noble gases) suspended inside a large volume. This applies to containment and auxiliary buildings where sprays or natural processes act on any material held. These processes can be very important if they have sufficient time to act.

Removal coefficients may depend on the chemical or physical form of the radionuclide. However, the current consensus is that, during a severe accident, all non-noble gases will form particulates or aerosols. Therefore, MENU-TACT allows you to specify only one removal coefficient. These removal coefficients will be applied to all non-noble gases. You cannot specify radionuclide-specific removal coefficients.

The removal coefficient is  $\lambda$  in the equation:

$$\frac{\text{Final concentration}}{\text{Original concentration}} = e^{-\lambda t}$$

Test results have shown that sprays are very effective initially. Later, natural processes appear to dominate. Figure 5 shows a typical experimental result. Note that a single  $\lambda$  does not characterize this process over time. The internal plant  $\lambda$  is valid initially and, for the remainder of the time, the natural coefficients would be valid.

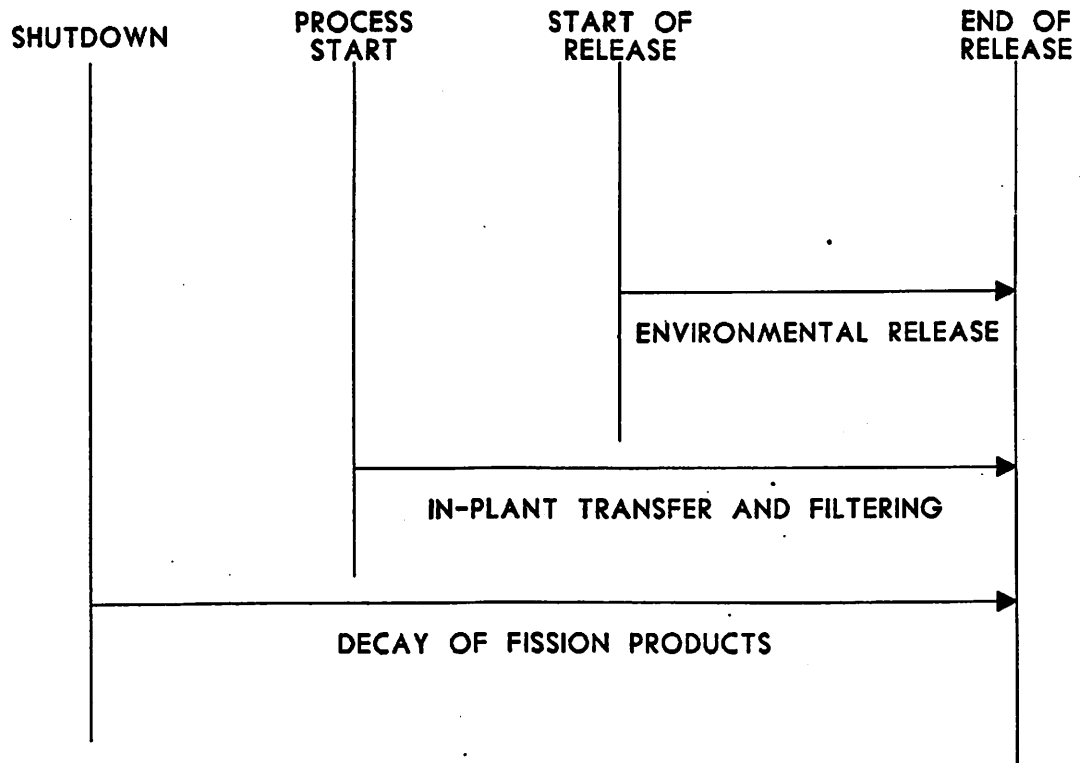


Fig. 4. Relationship among times entered.

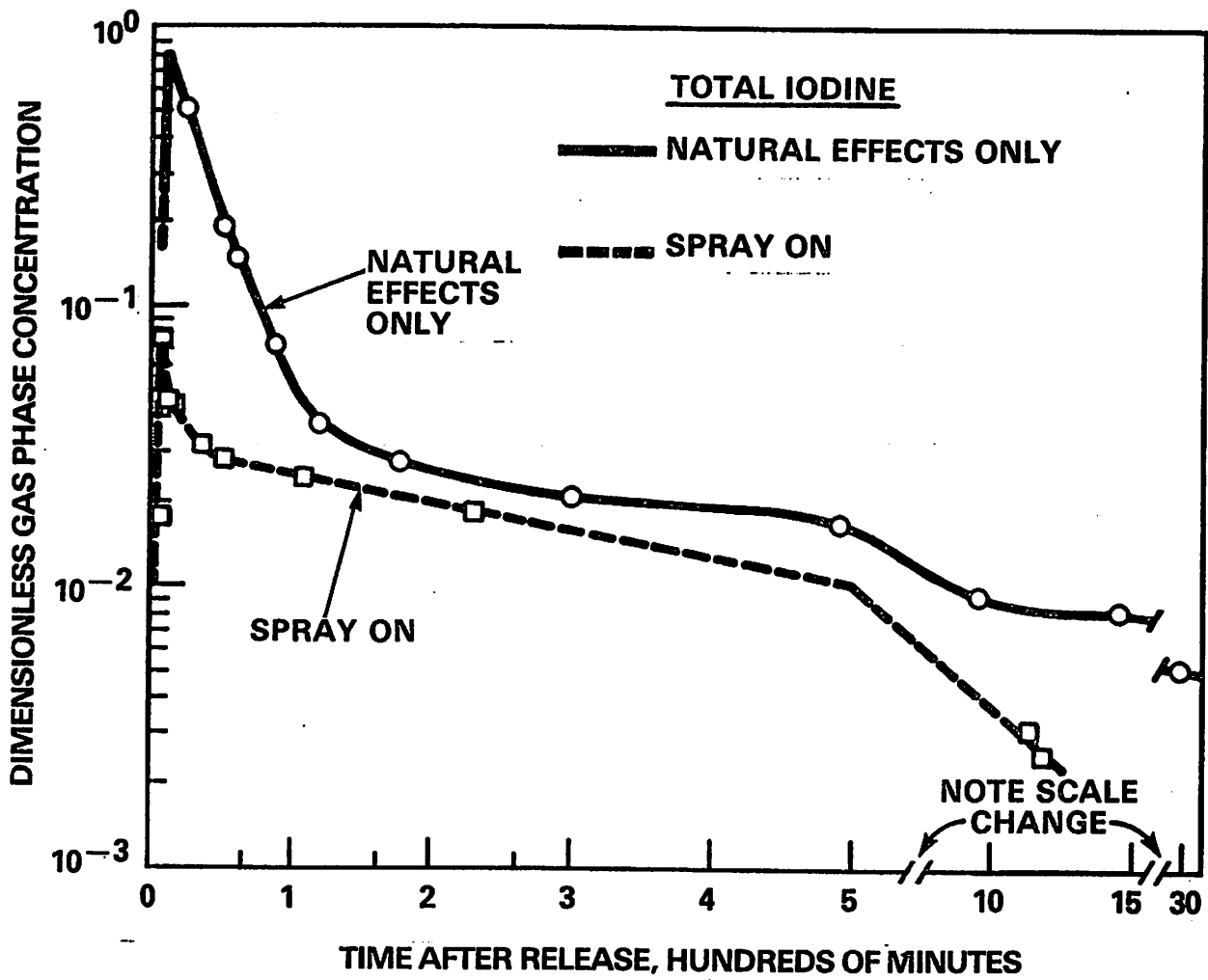


Fig. 5. Comparison of volume removal effects.

However, MENU-TACT allows only one coefficient to be specified. Consequently, the appropriate  $\lambda$ 's to account for this effect as a function of time have been calculated based on Fig. 5; these are shown in Table 6. The  $\lambda$ 's for 1, 2, 4, and 6 hr of hold-up time are shown at the bottom of the screen for reference. The hold-up time is the period between the start of the process and the start of the release.

#### 4.7.2 Specifying the Fraction of Initial Activity Released to Various Nodes

The second set of time- (and node-) dependent data requested is the initial distribution of inventory in the various nodes. The prompt for this information is shown below.

INITIAL DISTRIBUTION OF MATERIAL AT START OF PROBLEM  
INITIAL DISTRIBUTION OF CORE FRACTIONS  
FRACTION OF CORE MELT ACTIVITY RELEASED.

IN Primary .8  
IN Contain. .2

(AND/OR)

INITIAL RELEASE RATE FROM CORE  
FRACTION/HR OF CORE MELT ACTIVITY RELEASED.

TO Primary .0000E+00  
TO Contain. .0000E+00 \_\_\_\_\_

THESE RELEASE FRACTIONS ARE COMBINED IN THE NODES TO START THE PROBLEM  
\*\*NOTE: DEFAULT ACTIVITY ALREADY MULTIPLIED BY -  
1. FOR NOBLE GASES; 1. FOR PARTICULATE/AEROSOLS

MENU-TACT allows you to place any portion of the coolant (ft<sup>3</sup>) or gap, grain boundary, or core melt (fraction) specified directly into all nodes at the start of the release but not directly into the environment. Be careful because coolant is expressed in terms of ft<sup>3</sup>, while gap, grain boundary, and core melt are in terms of fraction of core inventory. The code prompt indicates the units (fraction or ft<sup>3</sup>) and type of damage (e.g., core melt). You must start the problem by placing material in one or more nodes. You may not transfer material directly to the environment. Remember that, earlier in MENU-TACT, you may have decided to reduce or increase the initial assumed release fraction. If

Table 6. Volume removal coefficient,  $\lambda$ 

| $\Delta t$<br>(hr) <sup>a</sup> | With spray<br>( $\gamma$ ) | Without spray<br>( $\gamma$ ) |
|---------------------------------|----------------------------|-------------------------------|
| 1                               | 3.5                        | 2.0                           |
| 4                               | 1.0                        | 0.87                          |
| 6                               | 0.7                        | 0.58                          |
| 24                              | 0.22                       | 0.18                          |

<sup>a</sup> $\Delta t$  - Time the material is held  
in volume before release.

you did, this is a fraction of those fractions. As a reminder, the bottom of the screen shows the damage state selected (e.g., core melt) followed by the factors (RELFN and RELFRI) used to adjust initial gas and non-noble gas release fractions.

Material can be transferred once (e.g., fraction of gap), at a rate (fraction of gap per hour), or both. If this is not a coolant release, the code checks to see if the amount of material injected exceeds 100% of the inventory and prints a warning message if it does. This is an informational warning only. You may proceed with the analysis regardless of the amount of material injected.

#### 4.7.3 Specifying In-Plant Transfer Rates and Filter Efficiencies

The next data requested are in-plant transfer rates and filter efficiencies as shown below.

| IN-PLANT TRANSFERS |          |                    |   |
|--------------------|----------|--------------------|---|
| FROM               | TO       | TRANSFER RATE(CFM) | PARTICULATE/AEROSOL/IODINE<br>FILTER EFFICIENCY (% 0.0 - 100.0) |
| Primary            | Primary  | .0000E+00          | .0000E+00   |
|                    | Contain. | 2.                 | 99.   |
| Contain.           | Primary  | .0000E+00          | .0000E+00   |
|                    | Contain. | .0000E+00          | .0000E+00_____  |

```
=====
REFERENCE FILTER DATA: FILTER CONDITIONS  EFFICIENCY
                        DRY/SLOW           99%
                        BLOW OUT/BY PASS   0%
```

##### 4.7.3.1 Specifying in-plant transfer rates of material

This is the rate (ft<sup>3</sup>/hr) at which material is assumed to be passed from one volume (node) to another. This could be based on pump or fan capacities, leak rates, etc. Typical transfer rates are listed in Table 7. The basis for selecting these rates is given in Appendix B. Note that you can recirculate material from a node to itself. The code will dilute the material as it goes from one node to the next to account for



Table 7. Typical transfer rates

| Type               | volume<br>(ft <sup>3</sup> ) | Design rate<br>(ft <sup>3</sup> /min) | Containment leak rates   |   |
|--------------------|------------------------------|---------------------------------------|--|---|
|                    |                              |                                       | 100%/day<br>(failure of<br>isolation seal)<br>(ft <sup>3</sup> /min) | 100%/hr<br>(catastrophic)<br>(ft <sup>3</sup> /min) |
| PWR-large dry      | 2.6 x 10 <sup>6</sup>        | 2 (0.1%/day)                          | 1.8 x 10 <sup>3</sup>  | 4 x 10 <sup>4</sup>                                 |
| PWR-subatmospheric | 1.8 x 10 <sup>6</sup>        | 1 (0.1%/day)                          | 1.2 x 10 <sup>3</sup>  | 3 x 10 <sup>4</sup>                                 |
| PWR-ice condenser  | 1.6 x 10 <sup>6</sup>        | 3 (0.25%/day)                         | 1.1 x 10 <sup>3</sup>  | 3 x 10 <sup>4</sup>                                 |
| BWR Mark I         | 1.6 x 10 <sup>5</sup>        | 1 (0.5%/day)                          | 1.1 x 10 <sup>2</sup>  | 3 x 10 <sup>3</sup>                                 |
| BWR Mark II        | 2.0 x 10 <sup>5</sup>        | 1 (0.5%/day)                          | 1.4 x 10 <sup>2</sup>  | 3 x 10 <sup>3</sup>                                 |
| BWR Mark III       | 2.7 x 10 <sup>5</sup>        | 1 (0.4%/day)                          | 1.9 x 10 <sup>2</sup>  | 4 x 10 <sup>3</sup>                                 |

Steam generator tube rupture (SGTR)

1 tube at pressure = 75 ft<sup>3</sup>/min (500 g/min)

Evolution of iodine from a pool (leakage)

1%/hr - relation to ft<sup>3</sup>/min will be  
based on leakage rate

refilling of the first node and dilution in the second. This can have a major effect on the release concentrations. For example, specifying a transfer rate ( $\text{ft}^3/\text{hr}$ ) equivalent to 100% of the volume in 1 hr will not result in transfer of 100% of the radioactive material in the volume in 1 hr. The magnitude of this effect can be seen in Table 8. If you do not want this effect considered, specify either a small volume or large transfer rate.

#### 4.7.3.2 Specifying in-plant filter efficiencies

The code allows a percentage of the particulate/aerosol to be removed as it passes from one node to another (or as it recirculates within a node). You must be careful to ensure that all major filter mechanisms are considered (e.g., suppression pool). Filter efficiencies for the Standby Gas Treatment System (SGTS), filters, primary system, suppression pool, ice condenser, and steam generator partitioning are summarized in Table 9. The basis for these efficiencies is discussed in Appendix C. Note that these filter efficiencies, like the removal coefficients entered earlier, are not nuclide-specific.

#### 4.7.4 Specifying Releases to the Environment

The next screen is shown below.

```

ENVIRONMENTAL RELEASE DATA
START OF RELEASE 87/08/04 14:00; END OF RELEASE 87/08/04 15:00

FROM      TO      TRANSFER RATE(CFM)    PARTICULATE/AEROSOL/IODINE
Primary   ENVIRON  1.                    FILTER EFFICIENCY (% 0.0 - 100.0)
Contain.  ENVIRON  10.                   99.
                                                99.00E+00 _____

=====
REFERENCE FILTER DATA: FILTER CONDITIONS    EFFICIENCY
                           DRY/SLOW          99%
                           BLOW OUT/BY PASS   0%

ARK1: CONTAINMENT FAILURE/LEAK RATE DATA
      DESIGN LEAK RATE: 150. FT**3/M
      PREDICTED FAILURE PRESSURE: 0. PSIG

```

This screen allows you to enter rates of transfer from nodes to the environment and filter efficiencies for these transfers. These transfers begin at the time entered as the release time. Clearly, at

Table 8. Example TACT 1-hr transfer rates

| Percentage<br>of volume specified<br>in TACT to be<br>transferred in 1 hr | Percentage<br>of radioisotope in<br>volume calculated to be<br>transferred by TACT |
|---|--|
| >200  | 100  |
| 100   | 63   |
| 50  | 38   |
| 20  | 17   |
| 10  | 9  |
| 1   | 1  |

Table 9. Filter efficiency (particulate/aerosols)

| Filter location                      | Conditions  | Efficiency (%) |
|--------------------------------------|---|----------------|
| Filters/standby gas treatment system | Case 1: Dry, low pressure                           | 99.9           |
|                                      | Case 2: Wet, high pressure (system blowout)         | 0              |
| Primary system retention             | Short residence time, fast flow                     | 15             |
|                                      | Long residence time, slow flow                      | 70             |
|                                      | Assumed   | 40             |
| Suppression pools                    | Violent flashing of steam (greater than decay heat) | 15             |
|                                      | Slow steady flow (decay-heat)                       |                |
|                                      | Subcooled   | 99             |
|                                      | Saturated   | 95             |
| Ice condenser                        | No recirculation                                    | 50             |
|                                      | With recirculation                                  | 75             |
| Steam generator partitioning         | Dry steam (1000 partitioning factor)                | 98             |
|                                      | Solid water (coolant)                               | 0              |

least one of these transfer rates should be greater than zero or no release to the environment will be generated. This is the final data input screen. At the bottom of the screen are shown the design leak rates and projected failure pressure, if known. Table 10 shows representative containment capabilities, and Table 7 shows typical transfer rates for these various containments. The basis for these rates is discussed in Appendix C.

Table 10. Representative containment structural capabilities<sup>a</sup>

| Typical<br>plant/type            | Design<br>Pressure<br>(psig) | Capability (failure)<br>pressure<br>(psig) |
|----------------------------------|------------------------------|--|
| Dry Pressurized<br>water reactor | 47                           | 134  |
| Subatmospheric                   | 45                           | 119  |
| Ice condenser                    | 12                           | 50   |
| Mark I                           | 56                           | 117  |
| Mark II                          | 55                           | 140  |
| Mark III                         | 15                           | 60   |

<sup>a</sup> From NUREG-1037 (USNRC, 1985).

## 5. EXECUTING TACT

Once all data have been entered, a MENU-TACT internal menu is displayed, as shown below.

```
NRC - IDAS TACT MODIFICATIONS OPTIONS
IDAS TACT - MENU VERSION

CHANGE TIME INDEPENDENT DATA
CHANGE TIME DEPENDENT DATA
CHANGE INITIAL SOURCE OF ACTIVITY
REVIEW MIX
RUN THIS CASE
RUN A DIFFERENT CASE
RETURN TO IDAS MAIN MENU
```

USE ARROW KEYS FOR SELECTION AND (CR) WHEN SELECTION IS THE CORRECT ONE

At this point you can go back and revise (or review) the data you have selected, or you may decline to run TACT.

If you select the "Run this Case" option, TACT will begin. A summary of the input will be displayed, followed by a table of the initial concentrations released into each node. Press the NEWLINE key as requested to step through these screens. The ESCAPE key will not allow you to back up once you have begun executing TACT.

```
Site is:ARKANSAS 1   Record is: ARK1_008
TIME INDEPENDENT INPUT
RUN DATE 5/ 26/ 87
TITLE: test
NUMBER OF NODES:           2
POWER (MWT):                2.568E+03
TIME OF SHUTDOWN:          87/08/04 14:00
RELEASE FRACTION:
PARTICULATES:               1.000E+00
NOBLE GASES:                 1.000E+00
SOURCE OPTION:              4-CORE
PLANT OPTION:               1-PWR
```

Node names:

Primary Contain.

Press newline to continue = :

```
TIME DEPENDENT INPUT
PROCESS START TIME   END OF RELEASE
87/08/04 14:00      87/08/04 15:00
```

### VOLUMES

| NODES                       | VOLUME (FT**3) |             |
|-----------------------------|----------------|-------------|
| Primary                     | 1.000E+06      |             |
| Contain.                    | 1.900E+06      |             |
| REMOVAL COEFFICIENTS (1/HR) |                |             |
| NODE                        |                | NOBLE GASES |
| Primary                     | 0.000E-01      | 0.000E-01   |
| Contain.                    | 0.000E-01      | 0.000E-01   |

Press newline for Transfer Rates. =

| INTER-NODE TRANSFERS |            |                         |                                |
|----------------------|------------|-------------------------|--------------------------------|
| FROM<br>NODE         | TO<br>NODE | TRANSFER RATES<br>(CFM) | FILTER EFFICIENCY<br>(PERCENT) |
|                      |            | PARTICULATES            | -NOBLES--                      |
| Primary              | Contain.   | 2.00                    | 99.0 0.000E-01                 |

| RELEASE TO ENVIRONMENT                |                         |                                |           |
|---------------------------------------|-------------------------|--------------------------------|-----------|
| FROM 87/08/04 14:00 TO 87/08/04 15:00 |                         |                                |           |
| FROM<br>NODE                          | TRANSFER RATES<br>(CFM) | FILTER EFFICIENCY<br>(PERCENT) |           |
|                                       |                         | PARTICULATES                   | -NOBLES-- |
| Primary                               | 1.00                    | 99.0                           | 0.000E-01 |
| Contain.                              | 10.0                    | 99.0                           | 0.000E-01 |

\*\*\* ALL TRANSFER RATES NOT LISTED ARE ZERO \*\*\*  
 Press newline to continue = :

ACTIVITY (CURIES) IN EACH NODE AT 87/08/04 14:00  
 (BEGINNING OF PROCESS)

| ISOTOPE | ENVIR.    | Primary   | Contain.  |
|---------|-----------|-----------|-----------|
| I-131   | 0.000E-01 | 5.752E+07 | 1.438E+07 |
| I-132   | 0.000E-01 | 8.423E+07 | 2.106E+07 |
| I-133   | 0.000E-01 | 1.212E+08 | 3.030E+07 |
| I-134   | 0.000E-01 | 1.335E+08 | 3.338E+07 |
| I-135   | 0.000E-01 | 1.130E+08 | 2.825E+07 |
| KR-85   | 0.000E-01 | 3.903E+05 | 9.758E+04 |
| KR-85M  | 0.000E-01 | 1.623E+07 | 4.057E+06 |
| KR-87   | 0.000E-01 | 3.082E+07 | 7.704E+06 |
| KR-88   | 0.000E-01 | 4.520E+07 | 1.130E+07 |
| XE-131M | 0.000E-01 | 7.601E+05 | 1.900E+05 |
| XE-133M | 0.000E-01 | 4.314E+06 | 1.079E+06 |
| XE-133  | 0.000E-01 | 1.171E+08 | 2.928E+07 |
| XE-135  | 0.000E-01 | 2.465E+07 | 6.163E+06 |
| XE-138  | 0.000E-01 | 2.465E+07 | 6.163E+06 |
| CS-134  | 0.000E-01 | 4.725E+06 | 1.181E+06 |
| CS-136  | 0.000E-01 | 1.890E+06 | 4.725E+05 |
| CS-137  | 0.000E-01 | 4.109E+06 | 1.027E+06 |
| TE-129M | 0.000E-01 | 3.944E+05 | 9.861E+04 |
| TE-131M | 0.000E-01 | 2.527E+06 | 6.317E+05 |
| TE-132  | 0.000E-01 | 2.527E+07 | 6.317E+06 |
| XE-138  | 0.000E-01 | 2.465E+07 | 6.163E+06 |
| CS-134  | 0.000E-01 | 4.725E+06 | 1.181E+06 |
| CS-136  | 0.000E-01 | 1.890E+06 | 4.725E+05 |
| CS-137  | 0.000E-01 | 4.109E+06 | 1.027E+06 |
| TE-129M | 0.000E-01 | 3.944E+05 | 9.861E+04 |
| TE-131M | 0.000E-01 | 2.527E+06 | 6.317E+05 |
| TE-132  | 0.000E-01 | 2.527E+07 | 6.317E+06 |

It is possible to set up your pathways and transfer rates so that the numerical routines in TACT will not be able to find a solution in terms of the released amounts. For example, this can happen if all transfer rates are equal. When the TACT numerical routines cannot find a solution, TACT will print a (fairly cryptic) error message and return you to the main MENU-TACT menu. You can then modify your transfer rates and try again.



TACT can produce detailed tables of results by node and time step. In general, these tables are not helpful in emergency situations. The default summary tables should be sufficient for your needs. You respond with Y or N at each request for a detailed table listing.

After the summary table of final concentrations is displayed, you will be asked if you want to save your results.

```

ACTIVITY RELEASED TO THE ENVIRONMENT FROM 87/08/04 14:00 TO 87/08/04 15:00
ISOTOPE      RELEASED
              CURIES
I-131        7.977E+01
I-132        1.010E+02
I-133        1.656E+02
I-134        1.282E+02
I-135        1.490E+02
KR-85        5.424E+01
KR-85M       2.089E+03
KR-87        3.301E+03
KR-88        5.571E+03
XE-131M      1.055E+02
XE-133M      5.955E+02
XE-133       1.623E+04
XE-135       3.299E+03
XE-138       1.102E+03
CS-134       6.564E+00
CS-136       2.623E+00
CS-137       5.708E+00
TE-129M      5.478E-01
TE-131M      3.470E+00
Press newline to continue =
TE-132       3.495E+01
SB-127       1.990E-01
SB-129       4.952E-01
SR-89        6.791E+00
SR-90        3.396E-01
SR-91        6.936E+00
MO-99        1.476E+01
RU-103       8.388E-01
RU-106       2.198E-01
BA-140       2.908E+01
Y-91         1.056E-02
LA-140       1.528E-02
CE-144       8.562E-03
NP-239       1.560E-01
Do you want to use these results in MESORAD? =

```

Y :

In general, you will want to save your TACT results because you cannot use them as input to MESORAD unless you save them at this point.

When TACT finishes, the MENU-TACT internal menu is displayed and you can run another case or exit MENU-TACT.

## 6. REFERENCES

- P. Easley and W. Pasedag, "NRC Source Term Assessments for Incident Response Dose Projections," pp. 6.19-6.25 in Computer Applications in Health Physics, 1984.
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- U.S. Nuclear Regulatory Commission, Technical Bases for Estimating Fission Product Behavior During LWR Accidents, NUREG-0772, Office of Nuclear Regulatory Research and Office of Nuclear Reactor Regulation, Washington, D.C., June 1981.
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- U.S. Nuclear Regulatory Commission, Containment Performance Working Group (Nuclear Regulatory Commission) Report: Draft Report for Comment, NUREG-1037, Washington, D.C., May 1985 (see Table 9).

Appendix A

**MATHEMATICAL DESCRIPTION OF TACT**

# NRC SOURCE TERM ASSESSMENT FOR INCIDENT RESPONSE DOSE PROJECTIONS

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## ABSTRACT

The NRC provides advice and assistance to licensees and State and local authorities in responding to accidents. The TACT code supports this function by providing source term projections for two situations during early (15 to 60 minutes) accident response: (1) Core/containment damage is indicated, but there are no measured releases. Quantification of a predicted release permits emergency response before people are exposed. With TACT, response personnel can estimate releases based on fuel and cladding conditions, coolant boundary and containment integrity, and mitigative systems operability. For this type of estimate, TACT is intermediate between default assumptions and time-consuming mechanistic codes. (2) A combination of plant status and limited release data are available. For this situation, iterations between predictions based on known conditions which are compared to measured releases gives reasonable confidence in supplemental source term information otherwise unavailable: nuclide mix, releases not monitored, and trending or abrupt changes. The assumptions and models used in TACT, and examples of its use, are given in this paper.

## BACKGROUND - THE ROLE OF TACT

The Nuclear Regulatory Commission established the Incident Response Center to advise and assist the licensee and State and local authorities in responding to radiological emergencies. The Intermediate Dose Assessment System (IDAS) is a group of integrated computer codes designed to provide offsite dose projections during the NRC's initial response in a radiological emergency. This paper

discusses one of the computer codes, TACT (Transport of Activity), which can be used to make projections of releases from accidents, and to interpret stack, dome, and offsite dose measurements. The IDAS version of the TACT code is a modified version of a code used in licensing evaluations. It provides input to RADPUR, which together with MESODIF will determine offsite doses corresponding to the source term and dispersion projections. RADPUR and MESODIF will be discussed separately.

TACT provides a mathematical framework for the solution of the equations arising from the consideration of fission product release, transport, and attenuation in the plant's systems and structures.

By their nature, accidental releases cannot be pre-defined. Accidents considered in source term assessment should not be limited to design basis accidents, or severe accident scenarios currently under study, or past accidents. Any means for transport of radionuclides from where they are at the start of the accident (the core, coolant, spent fuel pool, or waste handling system) to the environment is a candidate for analysis. If the transport path and the removal processes are simple, hand calculations of source terms may suffice. If not, the TACT user can create a simple mathematical model, via the TACT data input scheme, which will describe the initial source of radionuclides, their transport, their decay before release to the environment, and in-plant removal processes. Transport from place to place prior to release to the environment is modeled via nodes, which are designated volumes or compartments that hold radioactive material. The processes associated with a given node can include: transfer in and out, decay during residence, first-order removal processes (e.g., plate-out, spray wash-out, etc.) and the filtration of material transferred.

The flexibility of TACT permits the estimation of potential source terms before any releases actually take place. A series of hypotheses regarding releases can be developed and tested to support emergency response as the accident develops.

## MATHEMATICAL FRAMEWORK

The basic system of equations that TACT solves is:

$$(1) \frac{dA_{n,i}}{dt} = \sum_{j=i}^L K_{n,i,j} A_{n,j} + P_{n,i} - (\lambda_n + \mu_i) A_{n,i}$$

where:

- $A_{n,i}$  = curies of radionuclide  $n$  in node  $i$
- $n$  = index of radionuclides
- $i$  = index of nodes
- $t$  = time
- $\lambda_n$  = sum of all first-order removal coefficients, including radioactive decay-rate coefficient (inverse time)
- $K_{n,i,j} = (\dot{V}_{i,j}/V_j)(1 - f_{n,i,j})$ , which is the rate coefficient governing the flow of nuclide  $n$  from node  $j$  to node  $i$
- $V_j$  = volume of node  $j$
- $\dot{V}_{i,j}$  = volumetric flow rate from node  $j$  to node  $i$
- $j$  = index for other nodes that are not  $i$
- $f_{n,i,j}$  = filter efficiency (curies retained/curies in) for flow from node  $j$  to node  $i$ . Includes any phenomenon that removes a radionuclide in a fluid stream involved in internode transport or release to the environment.
- $L$  = total numbers of nodes, not counting the environment
- $P_{n,i}$  = user defined constant rate of activity insertion into a node
- $\dot{V}_{j,i}$  = volumetric flow rate from node  $i$  to node  $j$
- $\mu_i = \sum_{j=i}^L \dot{V}_{j,i}/V_i$

All terms in the equation have the units of curies/time. Also, note that the IDAS version of TACT includes radioactive decay, but does not account for the formation of radioactive daughter products. The equations are solved for any number of user-defined time steps. The initial conditions for the first time step are:

$$(2) A_{n,i}(t_0) = R_{n,i} \\ (n=1, \dots, N; \\ i=1, \dots, L+1)$$

where:

$R_{n,i}$  = user-defined initial insertion of an amount of nuclide  $n$  into node  $i$  (or the environment) at time  $t_0$ .

For each time step after the first, the initial conditions are:

$$(3) A_{n,i}^m(t_{m-1}) = A_{n,i}^{m-1}(t_{m-1}) + R_{n,i}^m$$

where:

$m$  = the time step index and

$A_{n,i}(t_{m-1})$  = the calculated activity at the end of the previous time step.

The equations presented by (1) comprise  $L+1$  simultaneous equations describing curies in each node and the environment, for each radionuclide, for each of the user-defined time steps. They are solved explicitly by standard matrix inversion subroutines. An equivalent set of subroutines is documented (Killough 83).

The solution to this system of equations provides the activity of each isotope in each node as a function of time. The result of interest for dose assessment, i.e., the curies transported to the environment, is calculated as follows:

$$(4) A_n^m = \sum_{j=2}^{L+1} K_{n,i,j} \int_{t_{m-1}}^{t_m} A_{n,j} dt \\ + (t_m - t_{m-1})P_{n,1} + R_{n,1}$$

where:

$A_n^m$  = total activity released to the environment from all nodes during time step  $m$

$t_m - t_{m-1}$  = duration of the time interval

$P_{n,1}$  = rate of radioactivity insertion into the environment, defined by the user. Decay is accounted for prior to insertion.

$R_{n,1}$  = instantaneous, one-time (per time step) insertion of radioactivity into the environment, defined by the user

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**Appendix B**

**BASES FOR SPECIFYING TRANSFER RATES**



## APPENDIX B

### BASES FOR SPECIFYING TRANSFER RATES

#### B.1 LEAKAGE VS CONTAINMENT LEAKAGE RATES

Containment leakage rates may be of three types:

1. Design leakage rates of 0.1% to 0.5% per day at design pressure.
2. Leakage from purge and vent system isolation valve seal failures. (WASH-1400 assumed that a failure to isolate corresponded to 1000 times design leakage or 100% per day. This would result from an approximately 8-in.<sup>2</sup> hole in the valve seals. Therefore, the 100% per day assumption for failure of the isolation valve seals appears to be reasonable.)
3. Major containment failure. It must be assumed that secondary containment will also fail and that 100% of the containment volume will be released in a few (e.g., 2) hours. This would be expected at the capability pressure of containment.

#### B.2 STEAM GENERATOR TUBE RUPTURE LEAK RATES AND PARTITION FRACTIONS

At operating pressure, the failure of a single tube corresponds to about 500 gal/min or 75 ft<sup>3</sup>/min. This decreases rapidly as the pressure falls.

Actual measurements have indicated partition factors in steam generators of 1000 to 1500 for dry steam where

$$\text{partition factor} = \frac{\text{Mass I}}{\text{Unit Vol of Water}} / \frac{\text{Mass I}}{\text{Unit of Vol of steam}}$$

Since the volume of water is about 1/20 that of steam, it is assumed that only about 1/50 (2%) of the primary iodine moves to secondary

containment. Therefore, a filter efficiency of 99% can be assumed for dry steam. However, if water is carried with the steam and primary coolant is leaking directly to secondary containment, partitioning will decrease.

### B.3 EVOLUTION OF IODINE FROM A POOL (LEAKAGE)

One case you may need to evaluate is the escape of iodine from contaminated water to the environment (e.g., leakage outside containment). Studies have shown that iodine evolves very slowly and that only 10% has the potential for release from a pool of water. The actual release rate is a function of chemistry, pool surface area, and other factors that will not be known at the time of event. Calculations based on reasonable assumptions have shown about 1%/hr of the iodine would enter the gas phase, while experiments have indicated a loss of about 0.5%/hr. Therefore, an assumption of a 1%/hr transfer rate from the pool is reasonable.

**Appendix C**

**BASES FOR SPECIFYING FILTER EFFICIENCIES**

## APPENDIX C

### BASES FOR SPECIFYING FILTER EFFICIENCIES

#### C.1 STANDBY GAS TREATMENT SYSTEM/OTHER FILTERS

In containment, filter systems and the Standby Gas Treatment System (SGTS) are not considered to be very effective during severe accidents, because they plug and rupture after accumulating a large amount of aerosols.

The purpose of the SGTS, generally found only in boiling water reactors (BWRs), is to collect and filter the release from primary containment [and in some cases the downstream leakage from the main steam isolation valves (MSIV)] and direct the release out the plant stack. The design-rated capacities vary widely from 8000 ft<sup>3</sup>/min to 25,000 ft<sup>3</sup>/min per train; typically there are three trains. In general, smaller-capacity systems incorporate some recirculation of filtered containment air.

The SGTS will perform effectively only for accidents where dry aerosol loading is minimal because the filter will plug and rupture if there is too much aerosol (<1 kg). In addition, the SGTS will be bypassed for severe accidents if the secondary containment blowout panels fail due to excessive secondary containment pressure, as would happen for any type of primary containment failure.

#### C.2 ICE CONDENSER

Only preliminary estimates have been made of the effectiveness of ice condensers in removing fission products. Preliminary model analysis concluded that "the ice condenser was found to have significant potential for the removal of fission products passing through it. For cases where the air return fans were not available and the released radioactivity made only a single pass through the ice condenser, approximately one-half of the fission products released to the containment were removed by the ice bed. With the air return fans available to continually recirculate the containment atmosphere, even

greater retention by the ice bed can be seen."

### C.3 BWR SUPPRESSION POOLS

All BWRs have pressure-suppression pools designed to condense steam following loss-of-coolant accidents. The suppression pool is a large mass of water through which the BWR primary system blows down. The ability of the system to remove fission products will depend on (1) the energy of the blowdown and (2) pool temperature. Tests have shown that pools may have a low decontamination factor for the violently flashing steam that may be present if more than decay heat is being removed by the blowdown. However, for most postulated BWR accidents, the release of fission products will be associated with removal of decay heat and is expected to be relatively slow and steady. Usually the removal rate will be high if the pool is subcooled or unsaturated. If the residual heat removal (RHR) system is not operational, the pool could become saturated. The condition of the suppression pool should be provided by the Reactor Safety Team (RST).

### C.4 DEPOSITION IN PRIMARY SYSTEM

Experimental data on deposition of materials in the primary system are very uncertain and limited. In addition, actual deposition will depend on the specific release pathway, flow rate, temperature, chemical form of the material, etc. None of these specifics will be known when the NRC Operations Center staff is analyzing the situation. Reviews of the results of various model runs have indicated that there are considerable differences in retention for CsI, aerosols, and Te. However, for the purpose of our analysis, CsI and Te will be assumed to act like aerosols or particulates.

Removal efficiency is primarily influenced by flow rate and residence time of the material in the system. The following two bounding cases show the range of primary system deposition. However, because of the uncertainties, 40% deposition would normally be assumed.

**CASE 1 - FAST FLOW AND SHORT RESIDENCE TIME**

This is typified by the pressurized water reactor (PWR) AB sequence, which is a large pipe break directly into the containment with minimum contact in the primary system.

**CASE 2 - LOW FLOW WITH LONG RESIDENCE TIMES**

This is typified by the PWR TMLB or V sequence with its release through the pressurizer and relief tank.

**BIBLIOGRAPHIC DATA SHEET**

NUREG/CR-4722  
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SEE INSTRUCTIONS ON THE REVERSE.

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MENU-TACT is a computer code designed to be used for source term estimation during reactor emergencies. The code allows the user to define initial concentrations of the nuclides in the reactor and the release pathway and timing. Filtering and other in-plant removal processes may be included. MENU-TACT, a modification of the TACT-III code, is simple to use. To provide quick results of bounding calculations, it incorporates only those processes that can make major changes in reactor releases.

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