

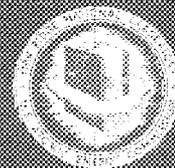
JUN 10 1974

DATE ISSUED

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DISTRIBUTION AND RELEASE OF TRITIUM IN  
HIGH-TEMPERATURE GAS-COOLED REACTORS  
AS A FUNCTION OF DESIGN, OPERATIONAL,  
AND MATERIAL PARAMETERS

E. L. Compere  
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**OAK RIDGE NATIONAL LABORATORY**

OPERATED BY UNION CARBIDE CORPORATION • FOR THE U.S. ATOMIC ENERGY COMMISSION

Printed in the United States of America. Available from  
National Technical Information Service  
U.S. Department of Commerce  
5285 Port Royal Road, Springfield, Virginia 22151  
Price: Printed Copy \$5.45; Microfiche \$1.45

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ORNL-TM-4303  
UC-77 - Gas Cooled Reactor Technology

Contract No. W-7405-eng-26

CHEMICAL TECHNOLOGY DIVISION

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JUNE 1974

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## PREFACE

This report summarizes work performed up to June 1973 relative to the description of the behavior of tritium in a high-temperature gas-cooled reactor; the main objective was the development of a computer code with which accurate assessment of tritium distribution in HTGRs could be made. The resulting code, TRITGO, is operational and, subject to the limitations discussed below, may be used for determining parameter sensitivity and for defining future research objectives. However, many extensions and amplifications that were evidently desirable were not included. Some of these are indicated in the text. (For example, the special role of the fuel element purge system in removing tritium released by the fuel in a reactor such as the Peach Bottom HTGR is not fully described.) In addition, it is now recognized that a slow, temperature-dependent release of tritium must occur from fuel particles, as well as from graphite and other solids. In the present version of TRITGO, this is handled by a partition factor corresponding to immediate release of a small fraction and permanent retention of the remainder. Similarly, fuel replacement in the core is not presently incorporated in the illustrative calculation.

Representative values of the input parameters have been taken from PSAR descriptions of the Fort St. Vrain Reactor. The relationship between code output and the performance of an HTGR remains to be established critically.



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ABSTRACT

Tritium will be produced in high-temperature gas-cooled reactors not only by fission, but also to a significant extent by reactions of neutrons with boron in burnable poison and control rods, with trace lithium impurity, and with  $^3\text{He}$  occurring naturally in the helium coolant flowing through the core. A portion of the tritium that is formed in solids or fuel may be released to the coolant. This tritium mingles with the hydrogen in the coolant and will, in part, chemisorb on graphite. In addition, it may leak with the coolant from the reactor vessel, be removed in the helium purification system, or permeate system walls into the steam generator, where as a result of removal (blowdown) or losses it is subject to release to the environment.

The differential equations describing the above behavior are developed. Assumption of steady conditions, including hydrogen concentration, for a period makes it possible to express the distribution as well as the production in the form of coupled sets of linear first-order ordinary differential equations. A generalized analytical solution of these is presented.

The above tritium generation and distribution model has been incorporated in a computer code, TRITGO. An illustrative calculation using parameters taken from the PSAR description of the Fort St. Vrain Reactor has been made. This indicates that most of the tritium emerging from the primary coolant will have been generated by the  $^3\text{He}(n,p)\text{T}$  reaction. A significant fraction enters the steam generator. The tritium in the steam generator blowdown, after proper dilution, may be considered for release to the environment. The effects of varying a number of system parameters are considered.

---

1.0 INTRODUCTION

Tritium<sup>1</sup> is expected to be the dominant contributor to the radiation dose received by the general population as the result of

nuclear reactor operation. Although fuel reprocessing comprises the major pathway to man, reactor operation also has a significant potential. A recently proposed regulation (10 CFR 50, Appendix I) for light-water-cooled reactors will permit only very low concentrations (5 pCi/ml) in liquids that are released to the environment (usually the turbine-condenser cooling tower release, in which appropriate other liquid effluents are diluted).

Tritium<sup>2</sup> (<sup>3</sup>H or T) is a radionuclide of 12.26-year half-life, decaying with the emission of a 18-keV beta particle. It enters the body readily, principally in the form of tritiated water, by ingestion, inhalation, or skin penetration; the biological half-life is about 12 days. In addition to the general whole body dose, detrimental effects may result<sup>2</sup> when <sup>3</sup>H is substituted for <sup>1</sup>H in a cell, due to mass differences or if lost by decay from a DNA or RNA molecule of which it was a part. No particular biological concentration processes have been noted in man or the environment.

High-temperature gas-cooled reactors differ from light-water reactors<sup>3-5</sup> in ways that affect tritium paths appreciably. The particle coating differs from the fuel cladding, and a small fraction of cracked particles may occur. In addition to formation in fuel by fission in HTGRs, tritium can be produced by reactions of neutrons with lithium impurities in the graphite, with added boron control materials, and with the <sup>3</sup>He occurring naturally in the primary coolant.

Tritium that enters or is formed in the helium primary coolant mingles with the hydrogen-containing species found there and can be chemisorbed on the graphite, as well as being steadily removed by the coolant purification plant. However, a gaseous mixture of hydrogen and tritium will dissolve in and diffuse through hot metals containing it, including steam generator surfaces.

Most of the hydrogen and tritium permeating into the steam generator can be assumed to exchange with, and thus become associated with, the steam generator water.

The major subsequent tritium pathway is then via steam generator water leakage or release (blowdown) into the cooling tower water that is expected to be evaporated or returned ultimately to natural source

waters (subject to possible regulatory limitations). These pathways are summarized in Fig. 1.

The following assumptions about the distributions are meaningful to our analysis:

1. The transient variation of diffusion profiles with time does not need to be taken into account. This eliminates position as a variable and reduces the problem to one of keeping an adequate set of inventory accounts characterizing the amount of tritium or precursor species in appropriate locations (compartments).
2. The distribution is successive; the path of tritium or its precursor atoms from any source to any sink follows a given sequence of downstream transfers and the rate does not depend on items excluded from the sequence or on feedback considerations. Distribution is linear so that a branched path can be treated as the sum of parallel linear paths.
3. Many aspects of the distribution pattern are essentially first order in terms of the resistances and capacitances associated with the system. This is true of leakages, blowdown removal, removal by purification plant, formation at steady power by neutron reaction, and radioactive decay. Although adsorption, transient diffusional transport, and metal permeation are not generally first order with respect to tritium amounts, certain assumptions permitting first-order representation appear to be applicable to HTGR systems. Consequently, we will justify and use a first-order approach.

It will be convenient to subdivide the reactor into a number of source regions having definite parameters (e.g., core, reflector, boronated reflector, and control rods) to each of which a uniform average slow and fast full power neutron flux can be assigned; the actual flux is proportional to the steady fraction of full reactor power that characterizes the particular period. Because of the additive property of the linear distribution, each fraction of the tritium entering the coolant and distributed from it can be associated with a particular source region. This permits calculation to be made region by region and chain by chain (and hence compartment by compartment) for both source and distribution aspects.

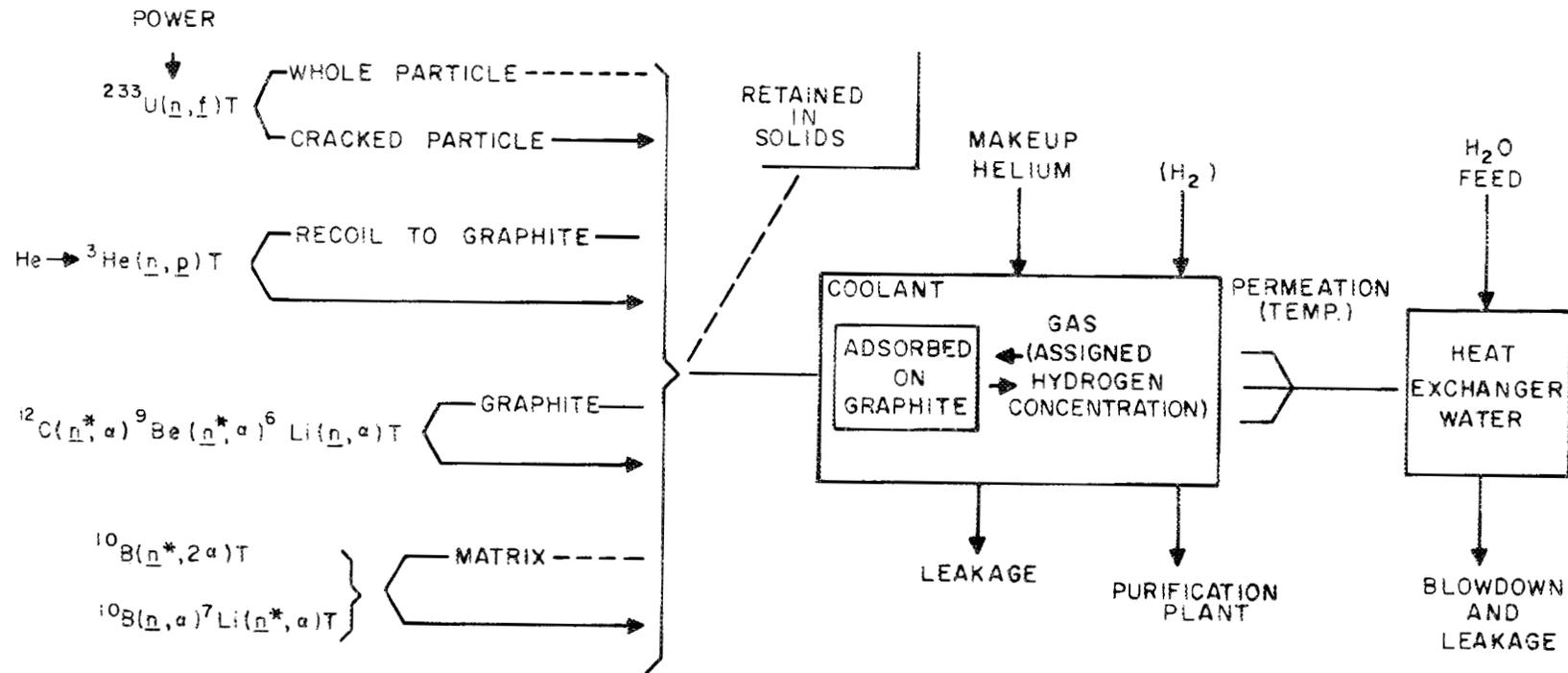


Fig. 1. Pathways by Which Tritium Is Transferred in HTGR Regions.

Thus for each region and chain we can write a coupled set of linear first-order differential equations of the type

$$\frac{dN_1}{dt} = I_0 - S_1 N_1$$

$$\frac{dN_j}{dt} = B_{(j-1,j)} N_{j-1} - S_j N_j \quad (j = 2, n),$$

which adequately describe the transfer from any source to any sink compartment. In these equations,  $N$  is the inventory of tritium or precursor atoms in a given compartment;  $S$  is the overall loss rate (including decay) from a given compartment;  $B_{(j-1,n)}$  is the transfer rate into a compartment from a prior compartment; and  $I_0$  (if not zero) is a steady input or source rate and, consequently, a "forcing" term.

It is necessary to identify the regions, chains, and transfer compartments of a given reactor and to develop the various source terms and first-order transfer or loss coefficients from the reactor parameters.

For a sequence of periods, we may then calculate region by region for each chain the amount of tritium or precursor that exists in each inventory compartment at the end of the period and was transferred to it under the operating conditions of the period from the previous inventory of a given prior compartment. Summation over all transfer sources gives the new inventory for each compartment. As will be shown later, a general analytical solution to the coupled sets of linear first-order differential equations has been developed, permitting explicit calculation of the results of each source-sink transfer across a set of compartments.

The model discussed above has been embodied in the computer code TRITGO, which calculates the inventory of source nuclides or tritium at various accumulation points (compartments) for a succession of periods subject to the constraint that changes in the pertinent parameters are made only between periods. When needed, overall tritium distribution rates can be obtained from successive inventory values in a straightforward manner.

The code can be used to evaluate tritium production and distribution under normal operating conditions, to provide source term data for

possible release to the environment, and to assess the adequacy of possible measures of relief if these are needed or desired.

## 2.0 SOURCES OF TRITIUM

### 2.1 Fission

Tritium is formed by ternary fission.<sup>2-6</sup> Tritium yields of  $0.8 \times 10^{-4}$  to  $1.2 \times 10^{-4}$  are given for the thermal-neutron fission of  $^{235}\text{U}$ .<sup>6-11</sup> Values for  $^{233}\text{U}$  do not differ significantly. Fast-neutron fission yields<sup>10</sup> of  $\sim 2.2 \times 10^{-4}$  for  $^{235}\text{U}$  are indicated. For an HTGR neutron energy spectrum we use a fission yield of  $1.0 \times 10^{-4}$  without specification of flux or fissile parent. Thus the rate of production of tritium by fission in region  $r$  can be expressed in terms of the associated reactor power:

$$\frac{d(N_T)_r}{dt} = KP_r Y - \lambda(N_T)_r,$$

where

- $K$  = fission rate at unit power,  $3.121 \times 10^{16}$  fissions  $\text{sec}^{-1} \text{MW}^{-1}$ ,
- $(N_T)_r$  = inventory of tritium atoms as produced in region  $r$ ,
- $P_r$  = power produced in region  $r$ , MW,
- $t$  = duration of period, sec,
- $Y$  = number of tritium atoms produced per fission,  $1.0 \times 10^{-4}$ ,
- $\lambda$  = tritium decay constant,  $1.793 \times 10^{-9} \text{sec}^{-1}$ .

The power produced in a given region containing fuel is calculated from the fuel element volume and the flux assigned to the region relative to the total, the full reactor power, and the relative power for the period. Hence

$$P_r = \pi P_{\text{total}} U_r \phi_r / \Sigma(U_r \phi_r),$$

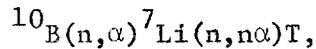
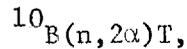
where

- $U_r$  = volume occupied by fuel elements in region  $r$ ,
- $\pi$  = fraction of full reactor power,
- $\Sigma$  = symbol for summation,
- $\phi_r$  = neutron flux for region.

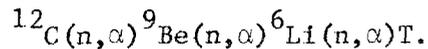
Because assigned region neutron fluxes are used here to subdivide the total power rather than calculate it, we do not need to take burnup into account. Even though it might affect relative region flux and power in the long run, this is not considered in the present model.

## 2.2 Neutron Reactions

Tritium is formed in appreciable yield by reactions of neutrons with boron,<sup>12</sup> which is intentionally present as a burnable poison in the fuel elements and as control rod material, and with lithium, which is a trace impurity in graphite and other solids. The reactions are summarized in the following reaction chains:



and



Because of the large amount of graphite in a high-temperature gas-cooled reactor, the latter reaction sequence eventually contributes to tritium production to a moderate extent even in the absence of lithium impurities.

The reaction rate equations for these reactions are:

for  $^{12}\text{C}(n,\alpha)^9\text{Be}(n,\alpha)^6\text{Li}(n,\alpha)\text{T}$ ,

$$\frac{dN_{12}}{dt} = - \pi \phi_{\text{fast}} \sigma_{12} N_{12}$$

$$\frac{dN_9}{dt} = \pi \phi_{\text{fast}} \sigma_{12} N_{12} - \pi \phi_{\text{fast}} \sigma_9 N_9$$

$$\frac{dN_6}{dt} = \pi \phi_{\text{fast}} \sigma_9 N_9 - \pi \phi_{\text{slow}} \sigma_6 N_6$$

$$\frac{dN_{\text{T}}}{dt} = \phi_{\text{slow}} \sigma_6 N_6 - \lambda_{\text{T}} N_{\text{T}};$$

and for  $^{10}\text{B}(n,\alpha)^7\text{Li}(n,n\alpha)\text{T}$  and  $^{10}\text{B}(n,2\alpha)\text{T}$ ,

$$\frac{dN_{10}}{dt} = \pi (\phi_{\text{slow}} \sigma_{10,\text{slow}} + \phi_{\text{fast}} \sigma_{10,\text{fast}}) N_{10}$$

$$\frac{dN_7}{dt} = \pi \phi_{\text{slow}} \sigma_{10, \text{slow}} N_{10} - \pi \phi_{\text{fast}} \sigma_7 N_7$$

$$\frac{dN_T}{dt} = \pi (\phi_{\text{fast}} \sigma_{10, \text{fast}} N_{10} + \phi_{\text{fast}} \sigma_7 N_7) - \lambda N_T,$$

where

N = number of given kinds of atoms in a region,

t = time, sec,

$\lambda$  = tritium decay constant,  $1.794 \times 10^{-9} \text{ sec}^{-1}$ ,

$\pi$  = fraction of full reactor power,

$\sigma$  = neutron cross section of given nuclide for fast or slow neutrons,  $\text{cm}^2$ ,

$\phi$  = region fast or slow neutron flux at full power, neutrons  $\text{cm}^{-2} \text{ sec}^{-1}$ .

Subscripts are:

fast = fast neutron,

slow = slow neutron,

6 = lithium-6,

7 = lithium-7,

9 = beryllium-9,

10 = boron-10,

12 = carbon-12,

T = tritium.

These constitute, for each chain in each region, coupled sets of linear first-order differential equations.

In addition, tritium is formed by the  ${}^3\text{He}(n,p)\text{T}$  reaction in the primary coolant.<sup>13</sup> Although the natural abundance of  ${}^3\text{He}$  in helium is given by most handbooks as  $1.3 \times 10^{-6}$ , it can be as low as  $2 \times 10^{-7}$  in helium obtained from wells, according to information furnished by the Bureau of Mines to the General Atomic Company (GAC). The above reaction of  ${}^3\text{He}$  has a relatively high thermal-neutron cross section ( $\sim 5300$  barns at 2200 m/sec) such that the reaction rate at a flux of  $6 \times 10^{13}$  neutrons  $\text{cm}^{-2} \text{ sec}^{-1}$  in a high-temperature gas-cooled reactor core corresponds to a loss of about 1% of the  ${}^3\text{He}$  in the core per day. Since about 6% of the helium is in the core, the tritium yield is appreciable. This reaction is particularly significant because about

half of the tritons from it do not recoil into solids but remain in the coolant helium. The coolant is the major distribution transfer compartment of the primary system.

The rate equations for production of tritium from  $^3\text{He}$  will be considered later.

Calculation of the production of tritium by neutron reactions is facilitated by the use of "effective" cross sections for an HTGR flux distribution obtained as follows. The neutron energy spectrum of an infinite homogeneous reactor with a C/ $^{233}\text{U}$  ratio of 5000, at a temperature of 900°K, was obtained\* for one source neutron  $\text{cm}^{-3} \text{sec}^{-1}$ . The energy spectrum of such an idealized reactor is believed to correspond usefully with that of actual high-temperature gas-cooled reactors, although obviously duplicating none with high precision. Using 2.38 eV as a cutoff, the calculated "slow" flux was 157 neutrons  $\text{cm}^{-2} \text{sec}^{-1}$  for one source neutron  $\text{cm}^{-3} \text{sec}^{-1}$ , and the corresponding calculated "fast" flux was 50 neutrons  $\text{cm}^{-2} \text{sec}^{-1}$ .

The values, along with the power density of a fueled region, could be used to calculate the corresponding estimated fast or slow flux.

$$\text{Estimated flux} = \left( \frac{\text{neutrons cm}^{-2} \text{sec}^{-1}}{\text{source neutrons cm}^{-3} \text{sec}^{-1}} \right) \times \left( \frac{\text{source neutron}}{\text{fission}} \right) \\ \times \left( \frac{\text{fission}}{\text{W-sec}} \right) \times \left( \frac{\text{W}}{\text{cm}^3} \right).$$

However, the results of more-refined reactor physics calculations are normally available for both fueled and unfueled regions. In the present code, such externally obtained values of slow and fast flux are assigned to their respective regions as input information.

For each reaction of interest the reaction rate for one source neutron  $\text{cm}^{-3} \text{sec}^{-1}$  was calculated using ENDF/B cross section - energy data and the above neutron energy spectrum. Table 1 shows the values obtained. Also shown are the corresponding effective cross sections, along with reaction rates (per day) in a core with a slow neutron flux of  $6.0 \times 10^{13}$  neutrons  $\text{cm}^{-2} \text{sec}^{-1}$  and a fast flux of  $3.6 \times 10^{13}$  neutrons  $\text{cm}^{-2} \text{sec}^{-1}$  (similar to the Fort St. Vrain Reactor core). The effective cross

\*Courtesy of A. M. Perry, ORNL Reactor Division.

Table 1. Reaction Rate Parameters for Nuclides of Interest

Reaction	For One Source Neutron $\text{cm}^{-3} \text{sec}^{-1}$			Calculated Effective Cross Section (barns)	Core <sup>a</sup> Reaction Rate ( $\text{days}^{-1}$ )
	Neutron Group	Flux, $\frac{\text{n cm}^{-2} \text{sec}^{-1}}{\text{n cm}^{-3} \text{sec}^{-1}}$	Reaction Rate, $\frac{\text{sec}^{-1}}{\text{n cm}^{-3} \text{sec}^{-1}}$		
${}^6\text{Li}(n,\alpha)\text{T}$	Slow	157	$6.40 \times 10^{-20}$	408	$2.1 \times 10^{-3}$
${}^7\text{Li}(n,n\alpha)\text{T}$	Fast	50	$3.59 \times 10^{-25}$	0.0072	$2.2 \times 10^{-8}$
${}^{10}\text{B}(n,2\alpha)\text{T}$	Fast	50	$6.76 \times 10^{-25}$	0.014	$4.2 \times 10^{-8}$
${}^{10}\text{B}(n,\alpha){}^7\text{Li}$	Slow	157	$2.56 \times 10^{-19}$	1630	$8.5 \times 10^{-3}$
${}^{12}\text{C}(n,\alpha){}^9\text{Be}$	Fast	50	$7.34 \times 10^{-27}$	0.00015	$4.6 \times 10^{-10}$
${}^9\text{Be}(n,\alpha){}^6\text{Li}$	Fast	50	$8.26 \times 10^{-25}$	0.17	$5.2 \times 10^{-7}$
${}^3\text{He}(n,p)\text{T}$	Slow	157	$3.58 \times 10^{-19}$	2280	$1.2 \times 10^{-2}$

<sup>a</sup>For a slow flux of  $6.0 \times 10^{13}$  neutrons  $\text{cm}^{-2} \text{sec}^{-1}$  or a fast flux of  $3.6 \times 10^{13}$  neutrons  $\text{cm}^{-2} \text{sec}^{-1}$ .

sections are, of course, appropriately less than the conventionally tabulated thermal values due to the increased average temperature among other factors.

The effective burnout half-life for original  ${}^6\text{Li}$  in such a reactor core is indicated to be about 330 days. However, the high mass of graphite in the core will result in an appreciable buildup of  ${}^9\text{Be}$  and subsequent formation of  ${}^6\text{Li}$  in the fast-neutron reactions  ${}^{12}\text{C}(n,\alpha){}^9\text{Be}(n,\alpha){}^6\text{Li}$ . With an original lithium concentration of 13 ppb ( ${}^{12}\text{C}/{}^6\text{Li} = 10^9$ ), the  ${}^6\text{Li}$  concentration would reach a minimum at full power in about four years; at 50 ppb of Li, the minimum would be attained in about five years. Thus,  ${}^6\text{Li}$  reactions can be expected to be of some significance before the end of fuel life, even in the initial absence of lithium in core graphite.

Boron will be added to fuel elements as a burnable control material. In the assumed core, the burnout half-life is about 82 days.

Helium-3 has an even higher core reaction rate (0.012 per day); however, only a small fraction (4 to 6%) of the primary coolant is in the core, so that the net removal rate is of the order of  $6 \times 10^{-4}$  per day. To some extent, leakage makeup will tend to offset this and to stabilize the rate of tritium formation from  ${}^3\text{He}$ .

### 2.3 Helium-3

Helium primary coolant circulates through all reactor regions. In order to fit the neutron reactions of  ${}^3\text{He}$  into a linear, region-by-region calculation scheme, each region must be assigned its reaction - rather than its mass fraction of the total, as shown below.

Helium is contained in all fabricated channels or holes, annuli, and clearance tolerances, as well as in the pores of graphite or other solids in the reactor system. The graphite is porous and cross flow is expected. Therefore, all space accessible to helium in a region is treated as a single, well-mixed volume with respect to tritium generation. However, the fraction of tritons which recoil to adjacent solids will vary with different classes of space, and an estimate of the overall fraction retained in each region will be presented later. It is assumed that a triton recoiling to a solid will enter and be retained interstitially.

First, consider the overall dynamics of  $^3\text{He}$  in the primary system.

For any given time,  $t$ :

- $N_4$  = atoms of  $^4\text{He}$  in primary coolant system,
- $N_3$  = atoms of  $^3\text{He}$  in primary coolant system,
- $X_3 = N_3/N_4 = ^3\text{He}$  fraction in coolant,
- $X_3^0$  = abundance of  $^3\text{H}$  in input helium,
- $M$  = makeup rate constant for primary coolant,  $\text{sec}^{-1}$ ,
- $\pi$  = relative reactor power (actual power/full power),
- $\sigma$  = effective cross section of  $^3\text{He}$  for tritium production.

For each region  $r$ :

- $V_r$  = region space accessible to helium,
- $C_r$  = atoms of helium per unit volume at region temperature and pressure,
- $\phi_r$  = assigned neutron flux for region at full power.

Relatively rapid mixing in and among all parts of all regions is assumed. The change in the number of  $^3\text{He}$  atoms in the reactor as a function of time is given by:

$$\frac{dN_3}{dt} = MN_4X_3^0 - MN_4X_3 - \Sigma(V_r C_r X_3 \sigma \pi \phi_r).$$

Now define

$$\begin{aligned} Q_r &= V_r C_r \sigma \phi_r, \\ Q_s &= \Sigma(Q_r), \\ F_r &= Q_r / Q_s. \end{aligned}$$

Then

$$\frac{dN_3}{dt} = MN_4X_3^0 - MN_4X_3 - MN_4X_3 - \pi X_3 Q_s \Sigma(F_r).$$

Since  $\Sigma(F_r) = 1$ , this factor may be inserted in any desired term. Hence

$$\Sigma(F_r) \left( \frac{dN_3}{dt} \right) = MN_4X_3^0 \Sigma(F_r) - MN_4X_3 \Sigma(F_r) - \left( \frac{\pi Q_s}{N_4} \right) N_4 X_3 \Sigma(F_r).$$

Since, by definition, none of the terms of  $Q_r$  or  $F_r$  is time-dependent,

$$\Sigma \frac{d}{dt} (N_4 X_3 F_r) = \Sigma [MN_4X_3^0 F_r - M(N_4 X_3 F_r) - \left( \frac{\pi Q_s}{N_r} \right) (N_4 X_3 F_r)].$$

We are thus interested in the fraction of the total  $^3\text{He}$  reactions that are to be associated with each region, rather than the volume fraction of

the  $^3\text{He}$  contained in the region. Our primary variable may be defined:

$$(N_3)_r = N_4 X_3 F_4;$$

the equation for  $^3\text{He}$  reaction for a region then becomes:

$$\frac{d(N_3)_r}{dt} = MN_4 X_3^o F_4 - \left(M + \frac{\pi Q_s}{N_4}\right) (N_3)_r.$$

It follows also that the rate of tritium production associated with region r from  $^3\text{He}$  is given by:

$$\frac{d(N_T)_r}{dt} = \left(\frac{\pi Q_s}{N_4}\right) (N_3)_r - \lambda (N_T)_r.$$

These equations are now linear in each region and may be incorporated in the general set of system equations. Consequently, the number of  $^3\text{He}$  atoms to be associated with a given region is given by the product of the total number in the primary coolant system and the fraction of the  $^3\text{He}$  reactions occurring in that region.

The formation of  $^3\text{He}$  in the system as a result of tritium decay is neglected for the following reasons. Tritium entering the primary coolant is removed fairly rapidly relative to the decay rate, and solids (especially fuel particles) retaining tritium are expected to also retain its  $^3\text{He}$  decay product. Thus, attempts to describe the contribution of tritium decay to the  $^3\text{He}$  content of the system would not improve the accuracy of our model significantly, and would unduly complicate the present equations and their solution.

Some modification of this approach is undoubtedly necessary for reactors using purged fuel elements, such as the Peach Bottom Reactor. However, such changes will not be considered in detail here. In principle, the distribution remains linear and first order, with an additional path to be included. The total production should not be significantly changed in such a calculation, although a somewhat larger fraction would be diverted to the purification system instead of following other distribution paths.

#### 2.4 Organization of Source Reactions

Examination of the rate equations for the inventory of the nuclides in the various chains in each region leading to tritium production indicates

that, for each chain, these constitute coupled sets of linear first-order differential equations of the form

$$\begin{aligned} dN_1 &= I_0 - S_1 N_1 \\ \frac{dN_j}{dt} &= B_{(j-1,j)} N_{j-1} - S_j N_j \quad (j = 2, m), \end{aligned}$$

where

$I_0$  = fixed input rate for period, atoms/sec, may be zero,  
 $B_{(j-1,j)}$  = rate constant for transfer from (j-1)th to jth compartment,  
sec<sup>-1</sup>

$N_j$  = atoms in jth compartment,

$S_j$  = rate constant for loss from jth compartment, sec<sup>-1</sup>.

Subscripts indicate the compartment number in a chain sequence of m compartments. Compartments are the successive atom inventory accounts for the particular chain and region.

We shall next show that these sets of equations can be extended to cover tritium distribution.

### 3.0 DISTRIBUTION OF TRITIUM

#### 3.1 Transfer Compartments

As indicated in Fig. 1, all the various production chains lead to either retention of tritium in solids or entry of tritium into the primary coolant system. We shall assume that, in the latter system, rapid equilibration occurs between the tritium in the circulating primary helium coolant and that chemisorbed on the graphite.

Tritium may leave the system via leaks of primary system helium; it is also removed from the stream of coolant which is passed through the helium purification plant. In addition, it may permeate the metal walls of the system along with hydrogen, particularly to enter the steam generator. Tritium entering the steam generator is removed in blowdown and leakage. Thus it is necessary to keep a number of inventory accounts with respect to the distribution of tritium. These distribution compartments include, in succession: overall production, bound in solids, adsorbed on solids, primary gas, primary leakage, purification plant removal, steam generator entry, steam generator content, and blowdown removal and loss.

It may be seen that the inventory of all the sinks named above may be defined by specification of only a few actual transfer compartments: (1) production or primary gas content, (2) primary gas losses or steam generator entry, and (3) steam generator content or blowdown, plus appropriate partition ratios for a particular period. We shall now consider these in detail.

### 3.2 Tritium Bound in Solids

After formation, tritons may come to rest in solids or in the helium primary coolant gas. Further distribution occurs from the primary coolant system. In the present computations, an arbitrary factor is used to calculate the fraction of tritium produced by a given chain in a given region that is retained or bound in solids; by difference, the fraction not retained and thereby entering the primary coolant is obtained.

Although the generalized analytical equations presented later have been shown, in principle (but with complexity), to be susceptible to extension to include slow release from solids (see Appendix I), release is likely to be quite rapid or quite slow. Thus, a useful approach is to treat release as either being prompt or not occurring at all. When valid, this approach has the merit of avoiding the need to consider additional compartments and to have considerably more equations handled in the code. Furthermore, release constants for tritium are not readily available. If it should become desirable to consider release rates from solids, reprogramming to extend the code to include this step appears feasible.

Thus, we assign a steady-state retention fraction for each chain that produces tritium. In the case of fuel, we note that tritium formed in cracked particles or from contaminants on particles may be assumed to escape readily, while that formed within intact TRISO (and probably also BISO) particles is strongly retained. Further, the fraction of cracked particles should be below 1% originally and should not greatly exceed 1% after extended operation. Consequently, an arbitrary retention fraction in fuel particles of 0.99 is used.

The chains that involve reaction of nuclides in the solid phase to produce tritium depend on the content and location of possible boron and lithium impurities in graphite, which are poorly known, and of boron in

burnable poison rods. The release rate of tritium which is bound interstitially in graphite is obscure but doubtless quite slow since a temperature at least several hundred degrees higher than reactor operating temperature is required to drive out residual hydrogen from graphite.<sup>17</sup> The use of a retention fraction of 0.99 appears reasonable and not unconservative. There is no present reason to use different factors for different chains.

The boron burnout poison rods are treated as if the boron were dispersed uniformly throughout the core. Such treatment may cause the average reaction rate for this boron to be too high.

In the case of control rods, the major lithium content is in the metal; tritium formed from this source might escape more readily than tritium that had been formed and retained in graphite. The boron fraction is obtained by calculating the total boron mass and dividing by the total control rod mass.

The control rods will, of course, not be fully inserted. An insertion fraction of about 11% is frequently cited. The relationship between extra gas space under rods, degree of rod insertion, etc. has not been treated as a variable. Of the gross rod mass, 11% is assumed to be in the core and thus subject to reaction. The introduction of additional lengths of rod because of depletion of the existing insertion has not been included in the calculations since the depletion is not great.

### 3.3 Tritium Recoil from $^3\text{He}$ Reaction

Tritons from the  $^3\text{He}(n,p)\text{T}$  reaction in the primary coolant will come to rest in solids in fractions which vary from region to region because of the varying proportion of space associated with channels and fabricated holes, clearance annuli, and pores. We will now consider the calculation of the average fraction of such tritium.

The recoil energy of a triton is about 0.2 MeV\* leading to an estimated range of 0.05 cm (500  $\mu$ ) in helium at 47.6 atm and 600°C.<sup>14</sup>

Because pores generally are smaller than 0.05 cm, the fraction bound is taken as unity.

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\*Personal communication from David Kocher, ORNL Nuclear Data Group.

Clearance annuli are the tolerances between fuel element blocks, and between the fuel rods and the blocks that contain them; for simplicity, we use a planar approximation. If the width exceeds the range,

Fraction bound = range/(2 x width), where range < width.

If the width is less than the range, the relationship is:

Fraction bound = (range - width/2)/range, where range > width.

For large cylindrical channels, or holes:

Fraction bound = range/diameter of channel.

The average fraction of retained recoil tritium for a given region is calculated as follows:

Avg. fraction =  $\Sigma(\text{vol} \times \text{fraction bound})/\Sigma(\text{vol})$ ,

where the summation is made over the various spaces of the region in which helium and tritium may be found (i.e., pores, channels and other holes, internal and external annuli). If the characteristics of the fuel or reflector blocks comprising a region are given and the mass and density of a region are known, the desired fractions may be calculated.

The present calculations with respect to the contribution of the  $^3\text{He}$  reactions are based on the volume and spacing and other characteristics of a single fueled element, with extension to a particular region being based on the mass of the region. An allowance for the one element out of seven that contains control rod passages is included. While the inserted part of the rod (taken as 11% of the possible full insertion) is treated separately as an individual region, the void in the remainder of the element is assumed to be part of the array of core fuel elements. It appears possible that a future revision could take into account the special characteristics of the control rod channels without undue difficulty. With values of triton recoil retention for the several chains available for the various regions, the amounts retained in a given region as "bound in solids" can be calculated from production values. The present calculations do not consider release of this bound tritium.

#### 3.4 Primary Coolant Compartment

The amount of tritium not bound in solids is directly calculable as the supplemental fraction of production. This material is considered to immediately enter the primary coolant compartment, which consists of the tritium present in any form in the helium coolant, as well as the

considerably larger amount chemisorbed on the carbon surfaces of the system. Tritium is lost from this compartment by four mechanisms: primary coolant leakage, removal in the purification plant, permeation of containment surfaces (in particular, the steam generator tubing), and decay.

### 3.5 Tritium and Hydrogen

Tritium entering the primary coolant compartment becomes equilibrated with the relatively large amount of natural hydrogen in this part of the system, and its chemical behavior then becomes indistinguishable from that of the hydrogen. Consequently, we shall determine the rate behavior for hydrogen, expressing it in "first-order" rate terms (rate/amount present), and assume that tritium acts similarly in proportion to its amount and consequently has the same rate constant. This approach is valid as long as the hydrogen concentration for a particular period is known and remains steady. The first-order loss rate terms allow the equation for tritium accumulation in the primary coolant compartment (here, index  $i$ ) to assume the usual form:

$$\frac{dN_i}{dt} = B_i N_{i-1} - S_i N_i,$$

where  $B_i$  is the rate of transfer into the coolant,  $S$  is the overall loss and removal rate from the coolant,  $N_i$  is the number of atoms of tritium in the coolant, and  $N_{i-1}$  is the number of atoms in the inventory compartment prior to the primary coolant compartment. This expression thus continues into the first distribution compartment the coupled set of linear first-order ordinary differential equations previously obtained for tritium generation.

Hydrogen concentrations in the primary coolant are input data for each period and are assumed to remain constant within the period. The source of this hydrogen is not considered. Levels of 10 to 30 ppm are anticipated for a reactor similar to the Fort St. Vrain Reactor.

### 3.6 Chemisorption

Hydrogen is chemisorbed by nuclear-grade graphite,<sup>15-18</sup> the equilibrium being essentially complete in the order of 1 to 2 hours.<sup>17</sup> Since we do not consider such short time periods here, it is adequate to regard hydrogen that is present on the surfaces as being in adsorptive and isotopic equilibrium with hydrogen in the gas. The chemisorption of hydrogen on graphite is indicated to follow a Temkin isotherm in that, at intermediate values of  $\theta$ , the amount adsorbed is linear with the logarithm of pressure.

The Temkin isotherm is as follows:<sup>19</sup>

$$\frac{\theta}{1 - \theta} = a p e^{q(1-\alpha\theta)/RT},$$

where

$a$  = adsorption constant, atm<sup>-1</sup>,

$p$  = gas pressure, atm,

$q$  = heat of adsorption at zero coverage, cal/mole,

$R$  = gas constant, 1.987 cal mole<sup>-1</sup> deg<sup>-1</sup>,

$T$  = °K,

$\alpha$  = factor for change in heat of adsorption with coverage,

$\theta$  = fraction of available sites on which chemisorbed atoms are found.

It is necessary to use the full expression in order to consider adsorption at lower pressures. The expression above fits literature data satisfactorily with respect to pressure, but no satisfactory values of  $q$  and  $\alpha$  could be extracted to show temperature dependence. As data from several sources had about the same  $\theta$ -vs- $p$  behavior at a given temperature, an expression independent of temperature was obtained. This implicitly expresses the amount of hydrogen,  $A$ , in cm<sup>3</sup>(STP)/m<sup>3</sup>, chemisorbed on a unit (B.E.T.) surface of graphite as a function of hydrogen partial pressure ( $p$ , torr) and an assumed monolayer saturation value,  $A_0$  (where  $\theta = A/A_0$ ):

$$A = (A_0/17.97) \ln [16.54 p(A_0 - A)/A].$$

This expression may be solved iteratively without difficulty.

The ratio between numbers of adsorbed and gasborne hydrogen atoms is then calculated using system carbon masses, B.E.T. surface, and hydrogen partial pressure in the total quantity of primary coolant gas. This

assumes reasonably rapid interchange of coolant gas with all pore gas and at least similarly rapid adsorption<sup>17,18</sup> or exchange processes. Although plausible, the validity of this assumption is not established. In particular, the extent of hydrogen and tritium penetration into the outer reflector blocks, which have much more graphite between surfaces in contact with flowing gas than do fuel elements, may be somewhat lower than assumed. Adjustment could be made for this possible discrepancy in the present calculation scheme by introducing an ad hoc reduction in the total B.E.T. surface of a region if access to it is restricted.

In any event, the ratio of adsorbed/gasborne hydrogen for the entire primary coolant system is assumed to hold for tritium throughout a given period. In a new time period, the adsorbed and gasborne tritium would be redistributed on the basis of the possibly new hydrogen concentration before other items are considered.

In a reactor similar to the Fort St. Vrain which contains 6140 lb of helium at 47.6 atm as the primary coolant and a total of 1,400,000 lb of graphite (core, reflector, and boronated reflector) with a B.E.T. surface area of  $0.2 \text{ m}^2/\text{g}$ , 30 ppm of hydrogen in the gas will result in about 25 times as much hydrogen being adsorbed as is gasborne. If exceptional temperature excursions should occur, considerable desorption of tritium and hydrogen could result, but the temperature range involved and the extent of desorption to be expected are not at all clear from the rather limited information available in the literature. Laboratory investigation in this regard should be of value.

Chemisorption thus acts as a coupled capacitance term for the hydrogen and tritium in the coolant system. Correspondingly, the removal terms from the coolant system are conductances in parallel and may be lumped where appropriate.

### 3.7 Coolant System Removal Terms

Hydrogen and tritium are removed from the primary coolant system via (1) leakage of coolant, (2) passage through the purification plant, and (3) permeation through various walls of the system (in particular, the heat transfer surfaces of the steam generator). By regarding the leakage, purification plant removal, and steam generator entry as final

accumulation sink compartments (subject only to decay of tritium), we can obtain their total content for any desired time. Each of the removal rate coefficients from the primary coolant system is expressed as a fraction of the content of primary coolant gas lost per unit time, since loss values in the various paths are most directly related in this manner to parameters and mechanisms determined independently.

Only 75 to 80% of the total primary system helium is in active circulation. It appears unnecessary, even though it is possible in principle, to account for this considering the noncirculating and circulating parts to be separate segments with a defined turnover rate. We will take the mass of actively circulating helium to be the inventory basis, thereby treating turnover as zero. This will give proper answers for tritium generation and removal. Leakage (or makeup) is actually a fixed mass per day. In the case where turnover between noncirculating and circulating helium is more rapid than makeup or burnup, the use of the circulating instead of the total helium inventory will simply modify the time required to reach a burnout-vs-makeup steady state, rather than affecting the ultimate  $^3\text{He}$  concentration. This is thought to be tolerable.

Leakage from the primary system is assumed to be equal to the average expected helium makeup rate. The leakage rate coefficient is then obtained by dividing the makeup rate by the circulating helium content of the primary system.

The rate coefficient for purification plant removal of hydrogen is given as the product of the flow rate of primary system coolant gas to the purification plant and a removal efficiency factor, divided by the amount of helium circulating in the primary system.

### 3.8 Permeation of Steam Generator Surfaces

Tritium will also diffuse through the walls of the various components of the steam generator system. (Moreover, although some diffusion will occur through all metallic components, their combined surface area, divided by thickness, is negligible compared with that of the steam generator system; thus this mode of transport is ignored.)

Hydrogen dissolves in metal in proportion to the square root of its partial pressure, according to Sieverts law. If the activity of hydrogen differs between two points within the metal, diffusion will occur down the activity gradient in accord with Fick's first law. These effects are combined (for constant temperature) so that the hydrogen permeation rate for a steady gradient is given by:<sup>20,21</sup>

$$\frac{dD}{dt} = \frac{\text{area}}{\text{thickness}} \times (p_1^{1/2} - p_2^{1/2}) \times C \times e^{-E/RT},$$

where D is the amount of hydrogen [in cm<sup>3</sup>(STP)] permeating through metal of a given area (cm<sup>2</sup>) and thickness (mm) with a metal permeation pre-exponential term C and activation energy E (cal mole<sup>-1</sup>), at a temperature T (°K). Upstream and downstream hydrogen partial pressures (in atmospheres) are indicated by p<sub>1</sub> and p<sub>2</sub>, respectively. The value of R is 1.987 cal mole<sup>-1</sup> deg<sup>-1</sup>.

At present, we assume the downstream hydrogen partial pressure (p<sub>2</sub>) to be negligible and the upstream pressure of hydrogen to remain the same throughout the system. Conceivably, the downstream hydrogen pressure in the steam generator could be more than trivial, as a result of water corrosion, hydrazine decomposition, or other processes. However, it is possible to show that tritium permeation would not be strongly affected by the downstream hydrogen pressure as long as the downstream tritium partial pressure was negligible.

The amount of tritium dissolved in metal in the presence of a dominant excess of hydrogen will be proportional to the square root of the partial pressure of total hydrogen (including tritium), and to the atom fraction of tritium in the total hydrogen. Thus, since hydrogen and tritium have similar chemistries, the fractional loss rate of hydrogen (the amount permeating in a unit time, divided by the amount in the primary coolant system) should hold for tritium. The permeation rate of tritium<sup>20</sup> is actually about 60% of that for hydrogen (for stainless steel, and doubtless for these metals also).

Thus, at a reference hydrogen pressure, the hydrogen permeation rates for the different steam generator components (economizer, evaporators, superheaters, reheaters, etc.) can be summed if we know the respective areas, thicknesses, average temperatures, and permeation characteristics

of the metals. The total hydrogen permeation rate for all components is divided by the amount of hydrogen in the primary system gas to obtain the fraction of hydrogen in primary system gas permeating in unit time. This fraction is then corrected (60%) to apply to tritium. The value may be adjusted to other system hydrogen pressures by multiplying by the ratio of the square roots of reference and actual hydrogen pressures.

Fractional loss rates by decay and by leakage, purification, or permeation, all now "first order," can then be summed to provide the overall removal rate for the primary coolant compartment.

### 3.9 Steam Generator Content

The steam generator content is the next compartment in sequence. It is the final compartment which we consider, since its output (blowdown and loss) can be obtained as the difference between the amount of tritium entering the steam generator and its tritium content and need not be treated as an additional compartment. The removal rate is the ratio of water makeup rate to water content of the steam generator.

The description of the paths by which tritium is distributed in the reactor system is now complete.

### 4.0 SOLUTION OF SYSTEM EQUATIONS

The various coupled differential equations appropriate to the system are set up and solved as shown below. However, several points need to be established before the solutions are described:

We deal with a succession of time periods, throughout each of which all rate terms, partition fractions, and other coefficients are assumed to be constant. The reactor is treated as a set of regions (core or core segments, reflector, rods, etc.) in each of which an average fast neutron flux and an average slow neutron flux characteristic of the region apply. Moreover, the fluxes are proportional to the relative reactor power. Each region has characteristic mass, dimensions, substances, and physical conditions.

The regions contain similar sets of source and sink compartments, each generally characterized by the number of atoms of a particular source nuclide or tritium. The loss from a given compartment, or the

transfer to another, is given by the product of the first power of the amount present, an appropriate "first order" removal or transfer rate factor, and possibly a partition fraction.

The system of equations that emerges is a coupled set of linear first-order ordinary differential equations with constant coefficients. Such sets may be solved numerically using Runge-Kutta or other techniques if the coefficient values are not unduly disparate. Analytical solutions using Laplace transform techniques are likewise possible. (The Bateman equation system, for decay and transmutation chains involving a number of steps, is a classic case of such a method of solution.) Analytical solutions ordinarily result in faster computation times and are, of course, inherently more accurate.

#### 4.1 General Solution Algorithm: Functions G and H

A solution algorithm was developed which gives the same results as other analytical solutions when dealing with a linear system of compartments with only downstream transfer and has the advantage of being more adaptable to automatic computation. The contents of a given compartment at the end of a period must have been in some upstream compartment at the beginning of the period, if the compartment itself and the system input to the chain are included. Consequently, the inventory of the sink compartment at the end of the period may be calculated as the summation of the possible source-sink transfers, based only on period transfer conditions and the inventories at the beginning of the period. The calculation of an individual source-sink transfer is facilitated by use of one or the other of two special functions. Function G is used if the source compartment inventory is subject to depletion only (no input), while function H applies if the source is a steady nonzero system input. It was established earlier that, for the HTGR system, transfers in a set of chain compartments may be described with a set of linear first-order differential equations of the type

$$\frac{dN_j}{dt} = B_{(j-1,j)} N_{j-1} - S_j N_j$$

with the addition, where appropriate, of an equation describing system input to the first compartment:

$$\frac{dN_1}{dt} = I_0 - S_1 N_1,$$

where

$N_j$  = inventory of compartment, number of atoms,

$B_{(j-1,j)}$  = first-order transfer rate coefficient from compartment j-1 to compartment j,  $\text{sec}^{-1}$ ,

$S_j$  = overall loss rate coefficient of compartment,  $\text{sec}^{-1}$ ,

$I_0$  = system input rate to first compartment, atoms/sec.

The solution to these equations is:

$$\Delta N_{\text{source-sink compartment}} = Z_{\text{source}} \times B_{(\text{source}, \text{source}+1)} \times (\dots) \times B_{(\text{sink}-1, \text{sink})} \times G(t, S, q)$$

and

$$\Delta N_{\text{input-sink}} = I_0 \times B_{(1,2)} \times (\dots) \times B_{(\text{sink}-1, \text{sink})} \times H(t, S, q),$$

where

$\Delta N$  = the inventory contribution to the sink compartment from given source, atoms,

$Z$  = inventory of the source compartment at the beginning of the period, atoms,

$B_{(j-1,j)}$  = transfer rate coefficient from preceding compartment to jth compartment,  $\text{sec}^{-1}$ ,

$I_0$  = system input rate, atoms/ $\text{sec}^{-1}$

and

$G(t, S, q)$  and  $H(t, S, q)$  are functions defined below, for a period of duration  $t$ , for a chain span of  $q$  compartments with overall compartment loss rate coefficients  $S$  ( $\text{sec}^{-1}$ ).

One or the other of the equations for  $\Delta N$  is applicable to each source-sink transfer to a particular compartment. The function  $G(t, S, q)$  is defined as:

$$G(t, S, q) = \sum_1^q [e^{-S_j t} \times \prod_{\substack{i=1 \\ i \neq j}}^q \left( \frac{1}{S_i - S_j} \right)];$$

and  $H(t, S, q)$  is defined as:

$$H(t, S, q) = \sum_1^q \left[ \left( \frac{1 - e^{-S_j t}}{S_j} \right) \times \prod_{\substack{i=1 \\ i \neq j}}^q \left( \frac{1}{S_i - S_j} \right) \right],$$

where

$S_i, S_j$  = loss rate coefficients,  $\text{sec}^{-1}$ ,  
q = number of compartments in transfer chain,  
t = duration of period, sec.

Because a difference of loss terms ( $S_i - S_j$ ) occurs in the denominator, no pair of loss terms in a given chain should be exactly the same since such a situation would result in division by zero.

For any chain in any region we will consider a system input rate and a succession of atom inventory compartments of tritium precursors, with transfer by neutron reaction, followed by three tritium compartments: (1) production and coolant; (2) coolant losses, steam generator entry; and (3) steam generator content. The transfer coefficients are calculated from reactor parameters and operating conditions as described above. Compartment inventories are calculated chain by chain and region by region for a given period. The results are completely additive because all steps are linear and first order. The full set of inventory values, period by period, is the fundamental output of the program.

## 4.2 Code Structure

The code proper contains statements that define and give the units of all input and output variables and major interim or internal variables. Furthermore, as the input is read in, a verbatim labeled copy in the input format [Appendix III(a)] is written as output. As this should be sufficient, the details of the definition of variables or input data format are omitted. With few exceptions, integers are input with a five-character format and real numbers are input with a ten-character format. Double precision computation is used throughout.

### 4.2.1 Input

The sequence of data input is as follows. The number of cases to be calculated for each and the numerical sequence of cases are given, followed by a title line giving the name of the reactor. The reactor power and core power density are followed by the coolant parameters, which include helium weight, pressure, temperature,  $^3\text{He}$  abundance, makeup rate, and rate of input into the purification plant. The mass of water in the steam generators and the water makeup rate are then given and the fraction replaced per second is calculated.

We then indicate the number of regions in which tritium can be produced. For each region we read in the mass, density, B.E.T. surface area, temperature, and assigned retention fraction for tritium formed in the solids. The fraction of the solids having coolant and control rod channels and the width of internal and external annuli are read in next, followed by a fuel presence flag, the fraction carbon in the solids, and the concentrations (ppm) of Li, B, and Be. A statement of the assigned fast- and slow-neutron flux at full power completes the input parameters of a region. At present we use a composite single region core. The average region boron concentration is calculated prior to input, with the assumption that each fuel element contains six boron burnable poison rods and each control element contains four.

Only a slight modification of the code should be required for a multiregion case, or one in which parts of the core are replaced at particular times. Response to an event flag in the history list (described later) could be used to simulate the effect of fuel removal or addition.

The reflector is a second region; this includes both the permanent and the replaceable reflector up to the boronated reflector region, with coolant and rod holes above and below the core. The boronated reflector is a third region, and control rods a fourth. Under normal conditions, the control rods occupy about 11% of their full insertion, and the input mass is this fraction of the rod total mass.

At this point, the mass of carbon and the number of atoms of the various source nuclides are calculated for each region. Fuel, if present is given a value of 1.0. Helium-3 is calculated later.

We next provide input to describe the properties of the major heat transfer surfaces and their hydrogen permeation characteristics. In each case, we supply the name, area, thickness, mean surface temperature, and the preexponential and energy term in the permeation rate equation. This completes the description of reactor properties.

#### 4.2.2 History - Subroutine HISTRY, ADATE

We next input three lists of dates and associated reactor operation parameters; these should be of particular value when the code is to be validated by comparison with actual operation of a particular reactor.

Mock dates may be used if actual dates are not appropriate. Each date is expressed internally after entry as time in days in order to put it on a continuous scale (in subroutine ADATE, the scale begins arbitrarily 1-1-67). The first list shows the fraction of full reactor power during the period following the date. The second list shows the hydrogen concentration and the purification plant efficiency. After the first entry in these lists, only changes need to be shown. An integer 1, rather than a blank, in the first column of a record indicates that the list ends with that record. A third list contains event flags. A zero causes the printout for the ensuing period to be omitted, and other values might be assigned, for example, to initiate calculation to account for fuel replacement. The dates at which the computation is to start and stop complete the lists. These lists are merged into a single sequence of operating periods, for use in calculations, by subroutine HISTORY. At this point the data input is complete.

#### 4.2.3 Preliminary Computation - Subroutines RECOIL, CROSS, HTEX

A description of the computation sequence follows. Preliminary calculations related to the original amount of  $^3\text{He}$  assigned to various regions and to the fraction of triton recoils striking adjacent solids (subroutine RECOIL) are followed by branching into several cases in which operating parameters are varied. Any number of cases may be executed in any sequence, as given by the appropriate input data. Reactor parameters are output, and the steam generator coefficient is calculated (subroutine HTEX) with descriptive output. A description of each region is output, including amounts and standard reaction rates for the various source nuclides (subroutine CROSS) and the fast and slow neutron flux for the region. Tritium carryforward for each source, sink, and nuclide is initially set to zero.

At this point the calculations for the sequence of periods covered by the computation span begin. After the computation for all regions has been completed for a given period, the results showing the status at the end of that period are output. The computation is terminated on completion of the calculations for all periods.

#### 4.2.4 Cases

The cases presently programmed include: (1) base case, with anticipated reactor parameters; (2) low hydrogen concentration (3 ppm) in coolant; (3) high hydrogen concentration (300 ppm) in coolant; (4) all tritium from  $^3\text{He}$  retained in gas and none retained in solids; (5) purification plant efficiency set at 50%; (6) all tritium formed in solids irreversibly bound; (7) heat exchanger permeation reduced<sup>21</sup> by a factor of 1/10; and (8) an increase of coolant leakage by a factor of 10. Possible modification to the present code would include the handling of case parameters in a subroutine rather than in the main program. None of the cases is to be taken as necessarily representing a case that is properly conservative for licensing purposes.

#### 4.2.5 Period Characteristics

The tasks for a given case prior to calculation of inventory changes for a period are considered next. The length of the period is calculated. The total number of tritium atoms in a sink, to be summed later, is set equal to zero. The total B.E.T. surface area for all of the reactor graphite is calculated, and a (volume·flux·fuel presence) sum for all regions is taken for subsequent use in apportioning reactor power among the various regions. (This is actually needed only if more than one core region is used.) At this point, we begin calculation for each region in succession.

The neutron reaction rate terms for the various source nuclides, including fuel, are adjusted for the relative power of the period. For  $^3\text{He}$ , the rate term is simply the overall reaction rate adjusted for the relative power for the period, divided by the amount of coolant helium. The rate at which fission tritium is produced is based on the total reactor power for the period, the region fraction (based on region volume and flux), the fission rate at unit power, and the yield of tritium from fission.

The purification plant removal rate and steam generator entry rates are adjusted so as to comply with conditions for the period. The chemisorption of hydrogen is next calculated as a function of pressure, and the partition ratio between adsorbed and gasborne hydrogen is obtained.

The distribution of carryforward tritium (from the prior period) between adsorbed and gas phase in the primary coolant compartment is adjusted for the current partition ratio for each source in each region.

#### 4.2.6 Production and Distribution

The calculation of tritium production and distribution for the period is considered next. Calculations for total tritium production and the amount of tritium in the coolant gas follow similar patterns except for branching fractions, partition ratios, and final compartment loss terms. By altering these appropriately on a second pass down the list of sources, the amounts produced from each source to these sink compartments in each region are calculated.

For all the sources, a major fraction of the tritium will recoil into the solids and is treated as being retained. Thus the "bound in solids" sink is a fixed fraction of production for each source.

Similarly, "adsorbed on solids" is in a fixed ratio to the tritium in the primary coolant gas, depending on the hydrogen concentration in the gas for the period.

The next sequence of calculations is similar, except that it includes an additional compartment; the coolant gas is now the next-to-last compartment. A final compartment of "emerged from gas system" or "retained in steam generator" is used. From these we calculate the accumulation in the purification plant removal, in the total gas leakage, and in the steam generator entry and contents; by difference, we obtain the accumulation in steam generator blowdown and loss.

Following this, the new inventory of the respective depletable source atoms associated with the region is calculated. Carryforward tritium is calculated for each source, sink, and region.

The above calculations are repeated for each region. This completes the calculations for the new atom inventories from all sources to all sinks for the period. In the case of the final period, the total distributed tritium for each region is compared with production. Agreement between these two values has always been exact to at least 10 significant figures (usually 15), confirming that the inventory accounts are in good balance.

#### 4.2.7 Output

Output is organized as follows. The dates covered by the period, the relative power, the purification plant efficiency, and the hydrogen concentration applicable to the period are stated, along with the overall steam generator permeation rate factor and the ratio of adsorbed to gas-phase hydrogen in the coolant.

Then, for each region, the calculated recoil fraction from the  ${}^3\text{He}(n,p)\text{T}$  reaction is stated, followed by a listing of the number of atoms of each source nuclide and the number of atoms in each sink of the region that have accumulated as a result of transfer during each successive period from the source nuclide.

The numbers of tritium atoms in each sink are totaled for the region and listed. After values for all regions are complete, the overall reactor total of tritium atoms in each unit is obtained and printed. Tritium inventories are expressed as curies. Furthermore, concentrations are calculated for those regions (sinks) corresponding to tritium remaining in the reactor, (1) "bound in solids" including fuel, (2) "adsorbed", (3) in primary coolant gas, and (4) in steam generator water. In particular, the concentration in effluent fluids may be compared with applicable regulatory emission limits.

The calculation scheme outlined above is contained in code TRITGO, a current version of which is presented in Appendix II; an output copy of an input data file for a reactor with parameters similar to those of the Fort St. Vrain Reactor is given in Appendix III(a).

The code is written in IBM FORTRAN IV, and was run on an IBM 360/91 computer at the Oak Ridge National Laboratory. Compilation and computation for eight cases for one reactor, over six time periods, required a total of 16 cpu sec and used a maximum of 270K core space. For convenience, the program and data files were prepared on the ORNL-DEC-PDP-10 time-sharing computer and submitted from it to the 360/91 system.

### 4.3 Code Data and Assumptions

#### 4.3.1 Block Data

In the block data section, the number of source neutrons produced per megawatt second, as well as the neutron reaction rate constants for one source neutron per  $\text{cm}^3$  per second, are listed for the various reactions considered. These values were obtained as a summation of flux-cross section products over a neutron energy spectrum characteristic of an HTGR. The cross sections as a function of energy were obtained from ENDF/B compilations. The flux per source neutron as a function of energy was obtained from a separate calculation for an infinite homogeneous reactor with a  $\text{C}/^{233}\text{U}$  ratio of 5000, at a temperature of  $900^\circ\text{K}$ . Though these assumptions do not precisely fit the Fort St. Vrain or any other particular reactor, it is believed that reliable results can be obtained using data from this base without particular modification. Results were expressed in terms of a reaction rate constant per source neutron  $\text{cm}^{-3} \text{sec}^{-1}$ . The calculation also showed a slow flux of 157 neutrons  $\text{cm}^{-2} \text{sec}^{-1}$  per source neutron  $\text{cm}^{-3} \text{sec}^{-1}$ , and a fast flux of 50 neutrons  $\text{cm}^{-2} \text{sec}^{-1}$  per source neutron  $\text{cm}^{-3} \text{sec}^{-1}$ .

#### 4.3.2 Data Section of Main Program

The decay constant for tritium,  $1.792 \times 10^{-9} \text{sec}^{-1}$ , corresponds to a half-life of 12.26 years.

The value of the retention fraction of tritium in fuel (FTRAP) was arbitrarily set as 0.99. It is based on the postulate that intact fuel particles retain tritium perfectly, cracked particles permit its release quite readily, and that the fraction of particles cracked for a long-term run is of the order of 1%, much of which was cracked before the fuel use began. None of these assumptions is precise, but all are thought to be reasonable. Modification to take into account experimental findings may be necessary.

Three coefficients associated with the empirical Temkin isotherm for the chemisorption of hydrogen on carbon are given in the data statements. The first is the monolayer saturation value,  $A_0$ , taken as  $0.2 \text{cm}^3$  (STP) of  $\text{H}_2$  per  $\text{m}^2$  of B.E.T. surface. We note that only two literature values are available; Redmond and Walker<sup>17</sup> found  $0.27 \text{cm}^3(\text{STP})/\text{m}^2$  and Thomas<sup>16</sup> cited values corresponding to 0.07

$\text{cm}^3(\text{STP})/\text{m}^2$ . Chemisorption corresponding to  $0.045 \text{ cm}^3(\text{STP})/\text{m}^2$  at 1 torr and  $0.065 \text{ cm}^3(\text{STP})/\text{m}^2$  at 10 torr permits evaluation of the other two constants in the empirical Temkin equation:

$$A = (A_0/17.94) \ln (16.54 p(A_0 - A/A)).$$

This equation is solved iteratively for A in the program.

Handbook values (Handbook of Chemistry and Physics, 49th ed., 1968) were used for the natural abundance of lithium, boron, beryllium, and carbon isotopes and the respective atomic weights for the elements.

For 12.26-year tritium, the number of curies is equal to the number of tritium atoms multiplied by  $4.84 \times 10^{-20}$ .

In subroutine RECOIL, we use a tritium recoil energy of 0.2 MeV in the reaction  ${}^3\text{He}(n,p)\text{T}$ .\* In this subroutine we also consider the average mass and space relationships of graphite elements with coolant and control rod channels, fuel rod annuli, and external clearance annuli. The number and sizes of coolant and control rod holes, and the relative number of each type of element, are those characteristic of the Fort St. Vrain Reactor. However, they are used as ratios to total carbon volume and thus should be applicable to other similar reactors.

Except for the coefficients of the Padé approximants used in the evaluation of function H, and of accepted ordinary unit conversion terms and scientific constants, these are the total of arbitrary constants and data contained in the code.

#### 4.3.3 Input Data: Fort St. Vrain Reactor Parameters

Data based on parameters of the Fort St. Vrain Reactor have been used in the illustrative calculation. These data were generally taken from the PSAR and FSAR documentation (Docket 50-267).

The weight of helium is required in accounting for  ${}^3\text{He}$ , hydrogen, and tritium. About 6170 lb is found in the primary coolant proper, and about 8040 lb if that in the purification plant holdup (680 lb) and in the PCRV penetration are included. A single value, 6170 lb, has been used here. It is suitable for hydrogen and, as discussed earlier, will only alter the burnout and loss time factors for  ${}^3\text{He}$  to a moderate extent.

\*David Kocher, ORNL Nuclear Data Group, personal communication.

Helium pressure is always given as 47.6 atm (700 psia) even though it is actually somewhat lower when the reactor is at less than full power. The average gas temperature is taken as the mean of core outlet and steam generator return gas temperatures at full power.

The abundance of  $^3\text{He}$  in the input helium is given as  $2 \times 10^{-7}$ , as it is expected that this will be a purchase specification.

The helium makeup rate is set at 1% per year. It is regarded as unlikely that it would rise as high as 14%/year, but 1% is not thought surprising, based on PSAR considerations. A value of 10% per year is considered as case 8.

The purification plant rate constant is based on a throughput of 935 lb/hr and a circulating gas mass of 6170 lb.

The mass of water in each of two of six-module steam generators is given as 4400 lb, for a total steam generator content of 8800 lb. The makeup rate is given as 20 gpm or 240,000 lb/day.

Helium is contained in channels, internal and external annuli, and pores. The description of coolant and control rod channel volumes is contained in subroutine RECOIL; it is sufficient to furnish as input the fraction of fuel or reflector element mass having such channels as opposed to none: 1.0 for core, 0.17 for mixed reflector elements (including elements with such holes above and below the core, and those without holes around the core) and 0.0 for the boronated reflector and the control rod regions. Then the width, in inches, for internal (fuel or poison rod) and for external (between elements) annuli is also input. Pore volume, which is not input, is estimated from density and mass. Modifications would be required for nonprism elements, in particular for Peach Bottom elements.

Values of the average concentrations of lithium, boron, and beryllium in a given region are expressed as parts per million. For core and reflectors, the detection limit, frequently stated as 0.05 ppm of lithium, is used. This still results in  $^6\text{Li}$  being the second highest source of tritium production; however, since most is retained by solids, the largest fraction of distributed tritium comes from  $^3\text{He}$  reaction. (Note that recently obtained values of 0.010 ppm of lithium on a specimen of "Peach Bottom" graphite and  $< 0.005$  ppm for lithium on a specimen of H-327 graphite indicate that the 0.050 ppm value cited above may be unduly high.)

Lithium is known to be present in the control rod metal cans. This concentration value, averaging 0.27 ppm of the total control rod mass, is evidently real rather than a detection limit. Boron has been assumed to be present in the core in all six corner positions of each regular fuel element, and in four for control elements. The total amount is expressed as parts per million of the core mass.

The detection limit for boron (1 ppm) is used for the reflector proper. The boron content of the boronated reflector was calculated to be equivalent to 6000 ppm. The boron concentration of the control rods was obtained by dividing the total weight of boron in the rods by the total weight of the rods.

No beryllium is assumed to be originally present in any region; however, it does "grow in" from the reaction  $^{12}\text{C}(n,\alpha)^9\text{Be}$  and is thus included in the input as 0.0.

For each region the average slow and fast neutron flux for full-power operation is given. Some of the data are from General Atomic Company as supplied in the PSAR, and other consistent values were from internal documents supplied by GAC as a personal communication.

The six components of the steam generator are: economizer, evaporators I and II, superheaters I and II, and reheater. Lumped values for the two steam generator loops are used for each component. The area ( $\text{cm}^2$ ) and thickness (mm) of the transfer surfaces of each are followed by the average gas side temperature ( $^\circ\text{C}$ ) and the preexponential and energy terms of the standard hydrogen permeation coefficient for the metal.

The hydrogen permeation coefficient for the Croloys used in the first four of the steam generator components is:<sup>21</sup>

$$\text{cm}^3(\text{STP}) \text{H}_2 \cdot \text{mm} \text{cm}^{-2} \text{hr}^{-1} \text{atm}^{-1/2} = 144 e^{-9100/\text{RT}}.$$

The coefficient for the Incoloy 800 used in Superheater II and the reheater is:<sup>21</sup>

$$\text{cm}^3(\text{STP}) \text{H}_2 \cdot \text{mm} \text{cm}^{-2} \text{hr}^{-1} \text{atm}^{-1/2} = 1210 e^{-17,500/\text{RT}}.$$

An assigned fractional efficiency (assumed to be 1.0) of the purification plant for hydrogen removal is used. The hydrogen concentration in the circulating helium is likely to fall in the range 10 to 30 ppm; we have assumed 30 ppm. Hydrogen concentrations of 3 and 300 ppm and a purification plant efficiency of 0.5 are used as alternative parameter values.

#### 4.3.4 Code Listing

A copy of the code as described above is given in Appendix II. The output listing of the input data used, based on the Fort St. Vrain Reactor, is given in Appendix III(a). Calculated reactor characteristics are given in Appendix III(b), followed by the output for the first [Appendix III(c)] and sixth [Appendix III(d)] years of the "base case," where the source inventories of a region at the end of the period are given along with the tritium in each sink resulting from all direct transfers within a region between that particular source and sink, cumulated for the successive periods. The sum over all source members of the chain to a given sink as of a particular time, although not specifically printed, will be invariant with respect to how the period might be subdivided; however, the contribution of individual sources of sink segments will shift, depending on the subdivision of the period.

### 5.0 RESULTS OF CALCULATION

Table 2 presents a listing of the source atom inventories for the respective chains in the various regions and for tritium production from each chain, at the end of six successive years, as calculated for the base case, based on Fort St. Vrain Reactor parameters.

All values for tritium inventory are corrected for decay. To estimate a decay-corrected production during a given interval, the prior inventory should be decay-corrected (a factor of 0.945 per year).

The  $^3\text{He}$  values in the table are based on primary coolant leakage and makeup rates of 1% per year. It is possible that higher makeup rates (case 8) would tend to offset  $^3\text{He}$  burnup more completely.

#### 5.1 Production

Production of tritium is highest in the core, with the fuel being the principal source. Most of this tritium will be formed in intact fuel particles and will therefore be retained. The next-highest certain yield in the core is from  $^3\text{He}$ ; only about half recoils to solids, the remainder remaining in the primary coolant. The traces of  $^6\text{Li}$  thought to be in graphite at the arbitrarily assigned lithium content of 50 ppb also comprise a tritium source similar in magnitude to the yield from

Table 2. Calculated Inventory of Atoms of Source Nuclide Chains and Tritium Production by Given Chain at End of Six Years of Reactor Operation

Conditions: base case, 80% service factor, Fort St. Vrain reactor parameters

CHAIN:	Fission Atoms T	$^3\text{He}(n,p)\text{T}$		$^{10}\text{B}(n,2\alpha)\text{T}$		$^{10}\text{B}(n,\alpha)^7\text{Li}(n,\alpha)\text{T}$		$^{12}\text{C}(n,\alpha)^9\text{Be}(n,\alpha)^6\text{Li}(n,\alpha)\text{T}$				Total Atoms T
		Atoms $^3\text{He}$	Atoms T	Atoms $^{10}\text{B}^a$	Atoms T	Atoms $^7\text{Li}$	Atoms T	Atoms $^{12}\text{C}$	Atoms $^9\text{Be}$	Atoms $^6\text{Li}$	Atoms T	
<u>Region: Core</u>												
0	0.0	7.89E22	0.00	6.54E26	0.0	7.52E23	0.0	9.28E30	0.0	6.03E22	0.0	0.0
1	6.45E22	6.62E22	1.24E22	5.54E25	2.87E21	6.00E26	2.63E21	9.28E30	1.24E24	3.25E22	2.69E22	1.09E23
2	1.25E23	5.56E22	2.21E22	4.70E24	2.95E21	6.50E26	6.51E21	9.28E30	2.48E24	1.76E22	4.00E22	1.97E28
3	1.83E23	4.69E22	2.97E22	3.95E23	2.81E21	6.55E26	1.03E22	9.28E30	3.72E24	9.50E21	4.56E22	2.71E23
4	2.38E23	3.96E22	3.55E22	3.35E22	2.66E21	6.55E26	1.39E22	9.28E30	4.96E24	5.18E21	4.74E22	3.37E23
5	2.89E23	3.36E22	3.98E22	2.84E21	2.51E21	6.55E26	1.73E22	9.28E30	6.19E24	2.86E21	4.72E22	3.96E23
6	3.37E23	2.86E22	4.29E22	2.40E20	2.37E21	6.55E26	2.05E22	9.28E30	7.43E24	1.62E21	4.59E22	4.49E23
<u>Region: Reflector</u>												
0	0.0	5.18E21	0.0	2.99E24	0.0	1.10E24	0.0	1.36E31	0.0	8.81E22	0.945	0.0
1	0.0	4.35E21	8.14E20	2.62E24	1.39E17	1.47E24	3.40E16	1.36E31	7.54E21	8.53E22	2.77E21	3.58E21
2	0.0	3.65E21	1.45E21	2.30E24	2.54E17	1.79E24	7.65E16	1.36E31	1.51E22	8.25E22	5.30E21	6.75E21
3	0.0	3.08E21	1.95E21	2.01E24	3.47E17	2.07E24	1.22E17	1.36E31	2.26E22	7.98E22	7.61E21	9.56E21
4	0.0	2.60E21	2.33E21	1.76E24	4.21E17	2.32E24	1.74E17	1.36E31	3.02E22	7.72E22	9.71E21	1.10E22
5	0.0	2.21E21	2.61E21	1.55E24	4.80E17	2.43E24	2.28E17	1.36E31	3.77E22	7.47E22	1.16E22	1.42E22
6	0.0	1.88E21	2.82E21	1.36E24	5.26E17	2.73E24	2.86E17	1.36E31	4.52E22	7.23E22	1.33E22	1.61E22
<u>Region: Boronated Reflector</u>												
0	0.0	1.17E20	0.0	1.23E28	0.0	7.57E23	0.0	9.34E30	0.0	6.06E22	0.0	0.0
1	0.0	9.83E19	1.84E19	1.23E28	7.75E17	6.15E25	1.06E15	9.34E30	6.57E18	6.06E22	7.27E19	9.19E19
2	0.0	8.27E19	3.29E19	1.22E28	1.50E18	1.22E26	4.07E15	9.34E30	1.31E19	6.05E22	1.41E20	1.75E20
3	0.0	6.96E19	4.41E19	1.22E28	2.19E18	1.82E26	8.96E15	9.34E30	1.97E19	6.04E22	2.06E20	2.52E20
4	0.0	5.89E19	5.27E19	1.21E28	2.83E18	2.42E26	1.56E16	9.34E30	2.63E19	6.03E22	2.67E20	3.22E20
5	0.0	4.99E19	5.91E19	1.20E28	3.44E18	3.02E26	2.39E16	9.34E30	3.29E19	6.03E22	3.25E20	3.88E20
6	0.0	4.25E19	6.37E19	1.20E28	4.01E18	3.61E26	3.36E16	9.34E30	3.94E19	6.02E22	3.79E20	4.47E20
<u>Region: Control Rods</u>												
0	0.0	1.51E18	0.0	7.61E26	0.0	1.04E22	0.0	2.38E28	0.0	8.36E20	0.0	0.0
1	0.0	1.26E18	2.37E17	7.44E26	3.49E21	1.70E25	2.13E19	2.38E28	1.24E21	8.31E20	4.58E18	3.51E21
2	0.0	1.06E18	4.23E17	7.27E26	6.72E21	3.37E25	8.29E19	2.38E28	2.47E21	8.26E20	8.89E18	6.81E21
3	0.0	8.95E17	5.67E17	7.11E26	9.69E21	5.00E25	1.82E20	2.38E28	3.71E21	8.22E20	1.29E19	9.79E21
4	0.0	7.56E17	6.77E17	6.95E26	1.24E22	6.59E25	3.15E20	2.38E28	4.95E21	8.17E20	1.67E19	1.27E22
5	0.0	6.41E17	7.59E17	6.79E26	1.49E22	8.14E25	4.80E20	2.38E28	6.18E21	8.12E20	2.03E19	1.54E22
6	0.0	5.46E17	7.19E17	6.64E26	1.72E22	9.66E25	6.73E20	2.38E28	7.42E21	8.08E20	2.36E19	1.79E22
<u>Sum Over All Regions to End of Period</u>												
0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
1	6.45E22	1.32E22	1.32E22	6.36E21	2.66E21	6.59E21	2.66E21	2.97E22	4.55E22	2.10E23	1.16E23	1.16E23
2	1.25E23	2.36E22	2.36E22	9.67E21	1.25E22	1.05E22	1.05E22	4.55E22	5.34E22	2.91E23	2.91E23	2.91E23
3	1.83E23	3.17E22	3.17E22	1.51E22	1.42E22	1.78E22	1.78E22	5.74E22	5.91E22	3.63E23	3.63E23	3.63E23
4	2.38E23	4.25E22	4.25E22	1.74E22	1.96E22	2.11E22	2.11E22	5.91E22	5.96E22	4.26E23	4.26E23	4.26E23
5	2.89E23	4.58E22	4.58E22	1.96E22	2.11E22	2.11E22	2.11E22	5.96E22	5.96E22	4.83E23	4.83E23	4.83E23
6	3.37E23	4.58E22	4.58E22	1.96E22	2.11E22	2.11E22	2.11E22	5.96E22	5.96E22	4.83E23	4.83E23	4.83E23

<sup>a</sup>Includes both  $^{10}\text{B}(n,2\alpha)\text{T}$  and  $^{10}\text{B}(n,\alpha)^7\text{Li}$ .

<sup>3</sup>He. Tritium yields in the control rods from the reaction of <sup>10</sup>B with fast neutrons, and in the reflector from assumed <sup>6</sup>Li impurities (Li ~ 50 ppb), are about 4% of that from the core. Helium-3 yields in the reflector are also lower but remain significant. The lithium impurity in the control rod cans is only a minor contributor (0.1%) to the tritium yield from this region because of the relatively low thermal flux and the considerably higher amounts of <sup>10</sup>B source material. The boronated reflector has a gross tritium yield of about 0.1% of the core; again, the assumed lithium impurity provides the major source.

Table 3 shows the production of tritium by chains and regions over the six-year term.

Table 3. Six-Year Tritium Production by Chains and Regions

Source	Tritium Production		Fraction
	Total Number of T Atoms	Ci of Tritium	
<u>Chain</u>			
Fuel	3.37E23	16,335	0.698
<sup>3</sup> He	4.58E22	2,216	0.095
<sup>10</sup> B	4.07E22	1,973	0.084
<sup>12</sup> C( <sup>6</sup> Li)	5.96E22	2,882	0.123
Total	4.836E23	23,406	1.000
<u>Region</u>			
Core	4.491E23	21,736	0.929
Reflector	1.614E22	781	0.033
Boronated reflector	4.472E20	22	0.001
Control rods	1.793E22	867	0.037
Total	4.836E23	23,406	1.000

## 5.2 Comments on Sources

Unambiguously the fuel is the dominant source. Our information about the release of tritium from the fuel is not well developed. In the absence of other information we would, of course, expect it to be released in a fashion similar to that of a rare gas with a comparable half-life - especially  $^{85}\text{Kr}$ . This, however, remains to be considered at a future date. Because of the low fraction presently assumed to be released, and the linearity of our expressions, the amounts in various sinks from fuel will be essentially proportional to the fraction of tritium released from the fuel.

Helium-3 is possibly the next most important contributor to tritium production (about 10% of the total), and very likely the major contributor to the tritium that enters the coolant and is thereby subject to distribution. The tritium yield from  $^3\text{He}$  is, of course, proportional to the abundance of  $^3\text{He}$  in the input and makeup helium. Conceivably, values different from the assigned ( $2 \times 10^{-7}$ ) abundance would be appropriate. Because we set makeup at a minimum (1%/year), burnup losses are greater than leakage. Higher makeup (and leakage) rates would result in somewhat higher tritium production, even though this is not a dominant effect.

Boron included in the core graphite as a burnable poison and used as control rod material is also an appreciable contributor. Use of substitute absorbing materials, although not proposed or expected, would remove this source.

We find that about 12% of the tritium comes from the  $^{12}\text{C}-^9\text{Be}-^6\text{Li}$  chain. Almost all of this yield results from the arbitrarily assigned lithium impurity of 50 ppb. Consequently, verification of the lithium concentration of core graphite is important because the tritium yield from this source is of significant magnitude and the currently assigned value is arbitrary. As mentioned earlier, lower values may be in order.

## 5.3 Comments on Distribution

Table 4 shows the distribution of the accumulated inventory of tritium (in curies) after six years of operation at a 80% service factor, as calculated for a reactor with Fort St. Vrain characteristics. These values will average somewhat lower per year than first-year values

Table 4. Distribution of Tritium (Ci) After Six Years of Reactor Operation

	Source of Tritium				Total
	Fuel	<sup>3</sup> He	<sup>10</sup> B( <sup>7</sup> Li)	<sup>12</sup> C( <sup>6</sup> Li)	
<u>Tritium Produced</u>	16,334.83	2216.52	1972.94	2882.30	23,406.59
<u>Contained:</u>					
Bound in core	16,171.49	1046.58	1095.18	2195.94	20,509.19
Bound in reflector	0	84.97	0.04	638.23	723.14
Bound in boronated reflector	0	2.17	0.19	18.18	20.54
Bound in rods	0	0.03	857.80	1.13	858.96
Total bound	16,171.49	1133.75	1953.21	2853.48	22,111.93
Distributed	163.34	1082.77	19.73	28.82	1,294.66
<u>Tritium Contained:</u>					
Adsorbed on solids	2.27	(0.0035 μCi/g)			
In primary coolant	0.09	(0.000006 μCi/cm <sup>3</sup> (STP))			
In steam generator water	0.005	(0.001235 μCi/g)			
<u>Cumulative six-year releases:</u>					
Leakage from primary	0.007				
Purification plant removal	907.99				
Steam generator blowdown and loss	384.32				

because of various burnout effects, etc. Possibly, release rates from fuel or from solids could become higher as exposure was prolonged; however, as noted earlier, this consideration has not been included in the present code.

It is evident that most (83%) of the distributed tritium (i.e., that not retained "permanently" by solids including fuel, comes from the  $^3\text{He}(n,p)\text{T}$  chain; less is retained by solids in comparison with other sources.

It is further evident that of the gross tritium produced in the reactor over a six-year period (23,406 Ci contained and distributed), most (22,111 Ci) is bound in solids and is presumably susceptible to release only at much higher temperatures. A total of 1295 Ci remains to be distributed throughout the system. About 2.3 Ci is contained in the primary coolant system, largely adsorbed on graphite. (The concentration in the primary and secondary coolants is discussed later.) Major distribution sinks include the purification plant (908 Ci) and the steam generator blowdown (and loss)(384 Ci).

As may be seen from the discussion of parameter variation to follow, some modification of the steam generator concentration and blowdown amounts is possible; on the other hand, complete elimination of entry of significant levels of tritium into the steam generator cannot reasonably be expected.

Table 5 shows the effects of varying a number of operating parameters on the tritium inventories and concentrations. The major effects (as compared with the base case) are described below.

A tenfold increase in hydrogen concentration(to 300 ppm) in the primary coolant results in a reduction of the amount going to the steam generator by a factor of 2.5, and a reduction in the amount of tritium that is adsorbed. Purification plant removal is consequently increased.

Conversely, reduction of hydrogen concentration by a factor of 10 (to 3 ppm) increases the amount going to the steam generator approximately twofold, diminishes the amount going to the purification plant, and increases the amount of adsorbed tritium.

If all the tritium produced by the  $^3\text{He}(n,p)\text{T}$  reaction were retained by the coolant, the amounts lost to the steam generator, to the purification plant, and to adsorption processes would be almost doubled.

Table 5. Effect of Operating Parameter Variations on Tritium Distribution and Inventory (in Ci) After One Year and Six Years of Reactor Operation at 80% Service Factor<sup>a,b</sup>

	Case							
	Base (H <sub>2</sub> , 30 ppm)	High H <sub>2</sub> (300 ppm)	Low H <sub>2</sub> (3 ppm)	All <sup>3</sup> He(n,p)T to Gas	Purif. Plt. Eff. = 0.5	All T Formed in Graphite Permanently Bound	Steam Gen. Perm. Coeff. = 10%	He Makeup, 10%/year
Total produced, Ci	5637.37 (23,406.6)	5637.37 (23,406.6)	5637.37 (23,406.6)	5637.37 (23,406.6)	5637.37 (23,406.6)	5637.37 (23,406.6)	5637.37 (23,406.6)	5639.04 (23,603.4)
<u>Removed or Released</u>								
Primary leakage, Ci	0.002 (0.007)	0.002 (0.009)	0.001 (0.004)	0.004 (0.013)	0.003 (0.011)	0.002 (0.007)	0.003 (0.010)	0.019 (0.075)
Purification plant, Ci	251.5 (938.0)	319.1 (1141.4)	147.8 (550.1)	478.6 (1703.2)	192.4 (699.2)	238.5 (873.9)	341.7 (1239.0)	252.1 (975.1)
Blowdown and leakage, Ci	106.45 (384.3)	42.7 (152.8)	197.8 (736.3)	202.6 (720.9)	162.9 (591.9)	100.9 (36.9)	14.5 (52.4)	106.7 (412.7)
<u>Contained</u>								
Bound in solids, Ci	5274.49 (22,111.9)	5274.49 (22,111.9)	5274.49 (22,111.9)	4946.85 (20,978.2)	5274.49 (22,111.9)	5293.26 (22,160.48)	5274.49 (22,111.9)	5275.34 (22,212.6)
Adsorbed, Ci	4.70 (2.27)	0.85 (0.41)	17.06 (8.21)	8.95 (4.10)	7.26 (3.49)	4.49 (2.19)	6.42 (3.09)	4.73 (2.84)
In primary He, Ci	0.19 (0.09)	0.23 (0.11)	0.11 (0.05)	0.35 (0.16)	0.29 (0.14)	0.18 (0.09)	0.25 (0.12)	0.19 (0.11)
<u>Conc. in He, μCi/std cc, 1 year</u>	0.000012	0.000015	0.000007	0.000023	0.000018	0.000011	0.000016	0.000012
In steam gen. H <sub>2</sub> O, Ci	0.010 (0.005)	0.004 (0.002)	0.020 (0.009)	0.019 (0.009)	0.016 (0.008)	0.010 (0.005)	0.0014 (0.0007)	0.010 (0.006)
<u>Conc. in Steam Gen., μCi/g, 1 year</u>	0.002563	0.001016	0.004953	0.004876	0.003954	0.002499	0.000350	0.002580

<sup>a</sup> Reactor characteristics are assumed to be those of the Fort St. Vrain Reactor.

<sup>b</sup> Values for six years of operation are given in parentheses. The remaining values are for one year of operation.

If the purification plant efficiency is reduced by 50%, the purification plant removal is about 75% of the base case since primary coolant concentrations increase. This also results in the permeation of about 50% more tritium into the steam generator.

The effect of permanent retention of all tritium formed from the  $^{10}\text{B}$ - $^7\text{Li}$  and  $^{12}\text{C}$ - $^9\text{Be}$ - $^6\text{Li}$  chains, rather than the base value of 0.99 retention, is relatively trivial. Use of a helium makeup rate of 10% per year, rather than the base value of 1% per year, results in somewhat higher values after extended exposure (six years) due to the more rapid replacement of  $^3\text{He}$  burnup. Also, of course, there is a corresponding increase in the relatively trivial leakage term.

The case in which steam generator permeation coefficients were reduced to 10% of the base value is significant, in that losses to the steam generator appear to be reduced about sevenfold relative to the base case. This case may be closer to reality than the base case in which permeation coefficients for the pure metals have been used. It has been shown<sup>21</sup> that, under some conditions, oxide films formed on metal surfaces can significantly inhibit hydrogen and tritium permeation. However, whether the type of films formed on steam generator surfaces will provide any significant and assured resistance to hydrogen and tritium permeation is not established at present.

#### 5.4 Release of Tritium to the Environment

The tritium in the steam generator blowdown, after proper dilution, may be considered for release to the environment. The concentration of any such release would of course depend on system factors affecting concentration in the blowdown stream as well as the amount of water used to dilute the blowdown prior to release.

The major system factor that could make release higher is the assumption of a release rate from graphite or from fuel particles greater than the 1% used in the present calculation, or lack of retention of the tritons that recoil from the  $^3\text{He}(n,p)\text{T}$  reaction. Inadequate tritium removal by the purification plant would also be unfavorable because a higher fraction of the tritium would enter the steam generator.

The principal system factor that could diminish release from the steam generator is a lower overall permeation coefficient, as this would reduce entry of tritium into the steam generator in almost direct proportion. The base case used permeation values for clean metal; consequently, somewhat lower permeation coefficients could reasonably be expected under more realistic conditions.

It may be possible to release all blowdown tritium to the cooling tower water, which is the major emergent liquid stream. As an alternative, some or all of it can be evaporated into a major emergent gas stream. Tentative calculations suggest that inclusion of all blowdown tritium in the main stack gas should result in discharge concentrations well below limits for gasborne emissions.

6.0 LIST OF SYMBOLS

A	Quantity adsorbed, $\text{cm}^3(\text{STP})/\text{m}^2$
B	Rate constant for transfer to given compartment, $\text{sec}^{-1}$
C	Atoms per unit volume at region temperature and pressure
D	Quantity of hydrogen permeating a given surface, $\text{cm}^3(\text{STP})$
E	Activation energy for permeation, $\text{cal mole}^{-1}$
F	Reaction fraction of $^3\text{He}$ (in region)
$G(t, S, q)$	Transfer function from depletable source through to sink, for time interval $t$ , with loss term $S$ for $q$ compartments (including source and sink)
$H(t, S, q)$	Transfer function from constant source through to sink, for time interval $t$ , with loss term $S$ for $q$ compartments (including sink)
I	Steady system input rate, $\text{atoms sec}^{-1}$
K	Rate of fission at unit power, $3.121 \times 10^{16}, \text{sec}^{-1} \text{MW}^{-1}$
L	Loss rate of tritium from compartment, $\text{sec}^{-1}$
M	Primary coolant makeup or loss rate as fraction of circulating gas, $\text{sec}^{-1}$
N	Number of atoms of a species in or assigned to a compartment of a region
n	Neutron
P	Region (or total) power, MW
p	Partial pressure, atm (for chemisorption, torr)
Q	$^3\text{He}$ reaction rate factor in region at full power, at unit $^3\text{He}$ concentration
q	Number of compartments in calculation span
R	Gas constant, $1.987 \text{ cal mole}^{-1} \text{ deg}^{-1}$
S	Overall loss rate from compartment, including decay, $\text{sec}^{-1}$
T	Temperature, $^{\circ}\text{K}$
t	Duration of period, sec
U	Volume occupied by fuel elements in region $r$
V	Volume of region in which reaction can occur
X	Mole fraction ( $^3\text{He}$ in He)
Y	Fractional fission yield

Z	Amount (atoms) in compartment at start of period
$\theta$	Fraction of adsorption sites occupied by adsorbed atoms
$\lambda$	Decay constant, $\text{sec}^{-1}$
$\Pi$	Mathematical symbol for product
$\pi$	Relative reactor power (vs "full" power)
$\Sigma$	Mathematical symbol for summation
$\sigma$	Cross section, $\text{cm}^2$ (effective)
$\phi$	Neutron flux, $\text{cm}^{-2} \text{sec}^{-1}$

Subscripts

i or j	Compartment index
r	Region index
s	Total: thus $Q_s = \sum Q_r$
T	Tritium
3	Helium-3
4	Helium-4
6	Lithium-6
7	Lithium-7
9	Beryllium-9
10	Boron-10
12	Carbon-12
fast	Region of neutron flux above 2.38 eV
slow	Region of neutron flux below 2.38 eV

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## 8.0 ADDITIONAL SOURCES

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3. Preliminary and Final Safety Analysis Reports for Fort St. Vrain Nuclear Generating Station (Unit 1) are contained in the document series Docket-50-267-\_\_\_\_\_.



9.0 APPENDIXES

### 9.1 Appendix I: Possible Extension of Code

The assumption of a fixed prompt release ratio for tritium formed in fuel particles as a fission product (and no slow release), which we have used in the present version of the TRIGO code, is adequate for normal conditions but would not give acceptable results for abnormal conditions. However, the linear attack we have used in the present code appears to be capable of extension to slow release, which would permit consideration of reactor cores with different temperature zones, release rates that vary with temperature and fraction of fuel particles cracked, and fractions of cracked particles which are believed to vary with time, temperature, and burnup.

In seeking to extend the code, we first note that the general approach has been to use first-order transfer rates between compartments or inventory accounts.

We now wish to add to the fuel accounts (prior to entry to the coolant), inventory accounts for the fraction of intact particles, a partition fraction of tritium in whole particles between fuel kernel and gas phase (two accounts here), and an account for the tritium in kernels of cracked particles. After its release from fuel particles, the tritium then undergoes the same distribution pattern as before.

In this way, properly operating portions of programs for tritium production from fuel have been written but have not been coupled with the coolant distribution processes. With the present code, this would require an accommodation of the code to extra chains and accounts, as well as a restatement of the equations pertaining to the fuel chain. If these provisions were added, failure rates and release rates for each period could be set by the temperature and other factors associated with the period. Since conditions are regarded as constant within a period, expressions would be required for transients relating time step and temperature. Such expressions could be used, if needed, to control the length of the time step as well as the system responses to the transient and thus keep the responses reasonably linear from step to step.

The question of penetration of tritium into pores of thick graphite material, such as reflector blocks, and its movement into such materials appears to require a distributed system attack that is different from that

used for the present code. However, some approaches to this question appear to offer useful approximations within the present code structure, and these should be examined.

In order to facilitate the addition of either sources or distribution compartments in particular chains without radical reprogramming, the organization of transfer terms into an array which will supply the needed values to the calculation steps is suggested. If this could be done, the flexibility of this code for a variety of uses would be considerably enhanced.



9.2. Appendix II: TRITGO Code Listing

```

COMPILER OPTIONS - NAME= MAIN,OPT=00,LINECNT=60,SIZE=0000K,
                  SOURCE,EBCDIC,NOLIST,NODECK,LOAD,MAP,NORBIT,NOID,NOXREF
C
C PROGRAM TRITGO - E. L. COMPRE, CHEMICAL TECHNOLOGY DIVISION, ORNL,
C S. H. FREID, BECHTEL CORPORATION
C AND C. W. NESTOR, COMPUTER SCIENCES DIVISION,
C OAK RIDGE NATIONAL LABORATORY
C
C TRITIUM PRODUCTION AND DISTRIBUTION IN AN HTGR
C
ISN 0002 BLOCK DATA
ISN 0003 IMPLICIT REAL*8(A-H,O-Z)
C
ISN 0004 COMMON/A1/ANAME(9),SAG(10,15),ENS,TMW,WTC1X(15),TRAPX(15)
& ,SIGFL(9),SA(10,15),ENC(15),THE,FISDEN,FFST(15),PSLW(15)
& ,FLXPSL,FLXPFS
& ,ISKIP(15),MSKIP(15)
C
ISN 0005 ENS=NEUTRONS PER MEGAWATT SEC
DATA ENS/7.80D16/
C
ISN 0006 LISTING OF NUCLIDES CONSIDERED FOR TRITIUM PRODUCTION
DATA ANAME/' LI-6 ',' LI-7 ',' FB-10 ',' SB-10 ',' C-12 ','
& ' BE-9 ',' HE-3 ',' FUEL ',' TRITIUM '/
C
C LISTING OF THEORETICAL FLUX-CROSS-SECTION INTEGRAL OVER ENERGY,
C PER SOURCE NEUTRON/CC-SEC, BASIS ENERGY SPECTRUM BELOW OR ABOVE
C 2.38 EV FOR INFINITE HOMOGENEOUS REACTOR AT 900 DEG K WITH
C C/U-233=5000, AND USING ENDF/B CROSS SECTION DATA
ISN 0007 DATA SIGFL/6.40D-20,3.59D-25,6.76D-25,2.56D-19,7.34D-27,8.26D-25,
& 3.58D-19, 1.0D-4, 0.0 /
C
ISN 0008 FLXPFS=FAST NEUTRON FLUX/(SOURCE NEUTRON/CC-SEC)--ABOVE 2.38 EV
FLXPSL=SLOW NEUTRON FLUX/(SOURCE NEUTRON/CC-SEC)--BELOW 2.38 EV
DATA FLXPFS/50./,FLXPSL/157./
C
ISN 0009 END

```

```

COMPILER OPTIONS - NAME= MATN,OPT=00,LTNECNT=60,SIZE=0000K,
SOURCE,FRCDIC,NOLIST,NODECK,LOAD,MAP,NOEDIT,NOID,NOXPEF
C
C MATN PROGRAM TRITGO, WITH SUBROUTINES HISTRY, ADATE, RECOIL, HTEX,
C CROSS, AND FUNCTIONS G AND H.
C
TSN 0002      IMPLICIT REAL*8 (A-H,O-Z)
TSN 0003      REAL*4 TITLE, RNAME, XNAME
TSN 0004      DIMENSION TITLE(18), RNAME(10,15), RGNLRS(15), FHOLES(15),
& WEXTAN(15), DENS(15), BET(15), RGTMP(15), HEXX(15), SCC(3),
& FRACC(15), PPMLI(15), PPMR(15), FPMBF(15), SIG(10), S(4), ST(8),
& A(50), Z(50), TMP(8,2), X(8,9,15), Y(8,9,15), TATOM(8), SO(10,15),
& ATCC(15), WINTAN(15), V(15), HERECI(15), R(15), FR(15), FUEL(15),
& CATOM(8), XATOM(8),
& JFLAG(8,15), IP(8), TC(7)
C
C
TSN 0005      COMMON/A1/ANAME(9), SAG(10,15), ENS, TMW, WTCIX(15), TRAPX(15)
& , SIGFL(9), SA(10,15), ENC(15), THE, FISDEN, FEET(15), FSLW(15)
& , FLXPSL, FLXPEF
& , ISKIP(15), MSKIP(15)
C
TSN 0006      COMMON/B/PEMH2(100), PRESS, CEATX, WTC, GAMM(100), DT(100),
& PWR(100), EVNT(100), TEND, TBEGIN, WTHE, TEMP, NEND
C
TSN 0007      COMMON/HIST/ESTART(50), TH2(50), CODEF(50), GAM(50), PM2(50), TEV(50),
& SRELPWR(50), JZ, IZ, KZ
C
TSN 0008      COMMON/HX/ HEAT(10), APEA(10), THICK(10), STMP(10), CONS(10),
& SENERGY(10), XNAME(5,10), NCOMP
C
C DCAY=TRITIUM DECAY RATE, PEP SECOND
C FTRAP=FRACTION OF TRITIUM PRODUCED IN FUEL NOT RELEASED TO COOLANT
C SCC(1), SCC(2), SCC(3) ARE CONSTANTS IN EMPIRICAL TEMKIN EQUATION
C FOR CHEMISORPTION OF HYDROGEN ON GRAPHITE. SEE MAIN STATEMENT 186,+2
TSN 0009      DATA DCAY/1.792D-9/, FTRAP/.99/, SCC/.2,16.54,17.97/
C JFLAG (1 OR 0) SPECIFIES PRINTOUT OF ALL TRITIUM SINKS FROM A GIVEN
C SOURCE AND GIVEN REGION. EIGHT SOURCES REGION 1, THEN EIGHT REGION
C 2, ETC. FIFTEEN REGIONS TOTAL, ZERO IF NO PRINTOUT DESIRED
TSN 0010      DATA JFLAG/8*1,7*1,1,7*1,1,8*1,8*0,80*C/
C IP IS AN INDEX USED TO DEFINE THE LISTING SEQUENCE OF SOURCE
C NUCLIDES AND TRITIUM ATOMS IN THE VARIOUS SINKS OF A REGION FROM
C THESE SOURCES WHEN PRINTING RESULTS FOR A PERIOD
TSN 0011      DATA IP/8,7,3,4,2,5,6,1/
C
C
C BEGIN READING INPUT--A REPLICAS OF INPUT RECORDS LABELLED WITH THE
C ASSOCIATED VARIABLES IS OUTPUT AT THE SAME TIME
C
TSN 0012      READ(1,10012) NCASES
TSN 0013      10012 FORMAT(I5)
TSN 0014      READ(1,10013) (TC(N),N=1,NCASES)
TSN 0015      10013 FORMAT(8I5)
TSN 0016      WRITE(5,20013) NCASES, (TC(N),N=1,NCASES)
TSN 0017      20013 FORMAT(1H1,' COPY OF INPUT VARIABLE NAMES AND DATA RECORDS',
& ' FOLLOWS, ALONG'/
& ' WITH RELATIVE TIMES (1/1/67=1) ASSOCIATED WITH DATES'/
& ' NCASES'/1H ,15/' TC'/1H ,8I5)

```

```

ISN 0018          READ(1,1001) TITLE
ISN 0019          1001  FORMAT(18A4)
ISN 0020          WRITE(5,20C1) TITLE
ISN 0021          2001  FORMAT(1H0,'TITLE'/1H,18A4)
                  C  TMW=REACTOR THERMAL MEGAWATTS AT FULL POWER//
                  C  FTSDEN= MEAN CORE FISSION POWER DENSITY,WATTS PER CC//
ISN 0022          READ(1,1002) TMW,FISDEN
ISN 0023          1002  FORMAT(2F10.2)
ISN 0024          WRITE(5,20C2) TMW,FISDEN
ISN 0025          2002  FORMAT(1H0,6X,'TMW   FISDEN'/1H,2F10.2)
                  C  WTWE=WEIGHT OF PRIMARY COOLANT HELIUM, LBS// PRESS=PRESSURE
                  C  OF PRIMARY COOLANT HELIUM,ATMOSPHERES// TEME=AVERAGE TEMPERATURE OF
                  C  PRIMARY COOLANT HELIUM, DEG C// ABUND3=ABUNDANCE OF HELIUM-3 IN
                  C  INPUT HELIUM TO PRIMARY COOLANT SYSTEM// GSCUT=REPLACEMENT RATE OF
                  C  PRIMARY SYSTEM HELIUM,PER SECCND// CPUR=THE FRACTION OF PRIMARY
                  C  SYSTEM HELIUM PUT THROUGH THE PURIFICATION SYSTEM PER SECOND
ISN 0026          READ(1,1003) WTWE,PRESS,TEMP,ABUND3,GSOUT,CPUR
ISN 0027          1003  FORMAT(3F10.2,3D10.2)
ISN 0028          WRITE(5,20C3) WTWE,PRESS,TEMP,ABUND3,GSOUT,CPUR
ISN 0029          2003  FORMAT(1H0,5X,'WTWE   PRESS   TEMP   ABUND3   GSOUT',
                  & '   CPUR'/1H,3F10.2,1P3D10.2)
                  C  H2O=MASS OF WATER IN SECONDARY (STEAM) SYSTEM,LBS
                  C  H2OM=MAKEUP RATE FOR STEAM SYSTEM BLOWDOWN AND LOSSES, LBS/DAY
ISN 0030          READ(1,1004) H2O,H2OM
ISN 0031          1004  FORMAT(2D10.2)
ISN 0032          WRITE(5,20C4) H2O,H2OM
ISN 0033          2004  FORMAT(1H0,6X,'H2O   H2OM'/1H,1P2D10.2)
                  C  SLOSS=FRACTION OF STEAM SYSTEM WATER REPLACED PER SECOND
ISN 0034          SLOSS=H2OM/H2O/86400.
                  C
                  C  NREG=NUMBER OF REGIONS IN WHICH TRITIUM GENERATION IS CONSIDERED
ISN 0035          WTC=0.0
ISN 0036          READ(1,1005) NREG
ISN 0037          1005  FORMAT(I5)
ISN 0038          WRITE(5,20C5) NREG
ISN 0039          2005  FORMAT(1H1,' NREG'/1H,I5)
ISN 0040          DO 91 N=1,NREG
                  C  RNAME=REGION NAME
ISN 0041          READ(1,1006) (RNAME(I,N),I=1,10)
ISN 0042          1006  FORMAT(10A4)
ISN 0043          WRITE(5,20C6) (RNAME(I,N),I=1,10)
ISN 0044          2006  FORMAT(1H0,'RNAME'/1H,10A4)
                  C  RGNLBS=MASS OF REGION IN LBS// FHOLES=WT FRACTION OF REGION WITH
                  C  CHANNELS AND ROD HOLES// WEXTAN=WIDTH OF EXTERNAL ANNULAR
                  C  CLEARANCE BETWEEN REGION ELEMENTS,INCHES// DENS=BULK DENSITY OF
                  C  REGION SOLIDS,G/CC// BET=B.E.T SURFACE AREA, SQ METERS PER GRAM//
                  C  WINTAN=WIDTH OF INTERNAL ANNULUS BETWEEN ROD MATRIX AND GRAPHITE//
                  C  RGTEMP=AVERAGE TEMPERATURE OF REGION SOLIDS,DEG C//
                  C  TRAPX=FRACTION OF TRITIUM FORMED IN SOLIDS WHICH DOES NOT RECOIL INTO
                  C  GAS PHASE; PRESUMED INTERSTITIAL AND FIRMLY BOUND
ISN 0045          READ(1,1007) RGNLBS(N),DENS(N),BET(N),RGTEMP(N),TRAPX(N)
ISN 0046          1007  FORMAT(1F10.0,4F10.3)
ISN 0047          WRITE(5,2007) RGNLBS(N),DENS(N),BET(N),RGTEMP(N),TRAPX(N)
ISN 0048          2007  FORMAT(1H0,3X,'RGNLBS   DENS   BET'
                  & '   RGTEMP   TRAPX'/1H,1F10.0,4F10.3)
ISN 0049          READ(1,10071) FHOLES(N),WINTAN(N),WEXTAN(N)
ISN 0050          10071  FORMAT(3F10.4)
ISN 0051          WRITE(5,20071) FHOLES(N),WINTAN(N),WEXTAN(N)

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ISN 0052 20071 FORMAT(1H0,' PHOIPS WINTAN WEXTAN'/1H ,3F10.4)
C FUEL=PRESENCE=1.,OR ABSENCE=0.,OF FUEL IN REGION
C FRACC=FRACTION CARBON IN SOLIDS// PPMLI=AVERAGE LITHIUM CONTENT
C OF SOLIDS,PPM// PPMB=AVERAGE BORON CONTENT OF SOLIDS,PPM//
C PPMBE=AVERAGE BERYLLIUM CONTENT OF SOLIDS,PPM//
ISN 0053 READ(1,1008) FUEL(N), FRACC(N),PPMLI(N),PPMB(N),PPMBE(N)
ISN 0054 1008 FORMAT(F10.0,F10.7,F10.3,F10.1,F10.3)
ISN 0055 WRITE(5,2008) FUEL(N), FRACC(N),PPMLI(N),PPMB(N),PPMBE(N)
ISN 0056 2008 FORMAT(1H0,4X,' FUEL FRACC PPMLI PPMB PPMBE'/
& 1H ,F10.0,F10.7,F10.3,F10.1,F10.3)
C FFST=AVERAGE FAST NEUTRON FLUX OF REGION AT FULL POWER//
C FSLW=AVERAGE SLOW NEUTRON FLUX OF REGION AT FULL POWER//
ISN 0057 READ(1,1009) FFST(N),FSLW(N)
ISN 0058 1009 FORMAT(2D10.2)
ISN 0059 WRITE(5,2009) FFST(N),FSLW(N)
ISN 0060 2009 FORMAT(1H0,5X,'FFST FSLW'/1H ,1P2D10.2)
ISN 0061 WTC1X(N)=RGNLBS(N)*FRACC(N)
ISN 0062 WTC=WTC+WTC1X(N)
ISN 0063 FACTOR=WTC1X(N)*453.59*6.02D23*1.D-6
C SA(*,N)=NUMBER OF ATOMS OF SOURCE NUCLIDE * IN REGION N
C 1=LI-6,2=LI-7,3=B-10(FAST NEUTRONS),4=B-10(SLOW NEUTRONS),5=C-12,
C 6=BE-9,7=HE-3. 8 REFERS TO FUEL IN REGION. INPUT AT 1. OR 0., NOT
C DEPLETED AS BURNUP NEGLECTED IN CALCULATION.
ISN 0064 SA(1,N)=FACTOR*PPMLI(N)*.0742/6.939
ISN 0065 SA(2,N)=FACTOR*PPMLI(N)*.9258/6.939
ISN 0066 SA(3,N)=FACTOR*PPMB(N)*.1960/10.811
ISN 0067 SA(4,N)=SA(3,N)
ISN 0068 SA(5,N)=FACTOR*.9889D6/12.01115
ISN 0069 SA(6,N)=FACTOR*PPMBE(N)*1.00/9.0122
C SA(7,N) WILL BE CALCULATED LATER
ISN 0070 SA(7,N)=0.DC
ISN 0071 SA(8,N)=FUEL(N)
ISN 0072 91 CONTINUE
C
C NCOMP=NUMBER OF COMPONENTS PROVIDING PERMEATION AND/OR HEAT TRANSFER
C SURFACES FROM THE PRIMARY SYSTEM
ISN 0073 READ(1,1010) NCOMP
ISN 0074 1010 FORMAT(I3)
ISN 0075 WRITE(5,2010) NCOMP
ISN 0076 2010 FORMAT(1H1,'NCOMP'/1H ,I3)
C
C FOR EACH COMPONENT
ISN 0077 DO 4 I=1,NCOMP
C XNAME=COMPONENT NAME// AREA=SQ CM HEAT (OR TRITIUM) TRANSFER SURFACE
C THICK=THICKNESS,MM// STMP=SURFACE TEMPERATURE,DEG C,MEAN OF INLET
C AND OUTLET// CONS=PREEXPONENTIAL TERM IN ARRHENIUS FORM OF PERMEATION
C EQUATION,STD CC H2*MM/HR/SQ CM AT 1 ATM H2// ENERGY=ACTIVATION ENERGY
C FOR PERMEATION, CAL/GMOL/DEG K
ISN 0078 READ(1,1011) (XNAME(J,I),J=1,5),AREA(I),THICK(I),STMP(I),
& CONS(I),ENERGY(I)
ISN 0079 1011 FORMAT(5A4,D10.2,2F10.2,2D10.2)
ISN 0080 WRITE(5,2011) (XNAME(J,I),J=1,5),AREA(I),THICK(I),STMP(I),
& CONS(I),ENERGY(I)
ISN 0081 2011 FORMAT(1H0,'XNAME',20X,'AREA',5X,'THICK',6X,'STMP',6X,'CONS',
& 4X,'ENERGY'/1H ,5A4,1E10.2,0E2F10.2,1P2D10.2)
ISN 0082 4 CONTINUE
C
C REACTOR DESCRIPTION COMPLETE

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C
C HERE BEGIN INPUT OF THREE LISTS OF DATE AND LEVEL OF VARIOUS
C REACTOR CONDITIONS, AND FLAGS
C
ISN 0083      WRITE(5,11011)
ISN 0084      11011 FORMAT(1H1,'HISTORY LISTS--VALUE OF 1 IN FIRST COLUMN INDICATES ',
                & 'END OF LIST'/1H,'POWER HISTORY LIST'/1H,'JX JMS JDS JYS',
                & ' RPLPWR DSTART(NOT INPUT)')
ISN 0085      J=0
ISN 0086      T=0
ISN 0087      K=0
ISN 0088      5 J=J+1
C JX=FLAG// JMS,JDS,JYS=MONTH,DAY,YEAR// RPLPWR=RELATIVE
C REACTOR POWER BEGINNING AT THIS TIME
ISN 0089      READ(1,1101) JX,JMS,JDS,JYS,RPLPWR(J)
ISN 0090      1101 FORMAT(I1,3I5,F10.3)
C ADATE CHANGES DATE TO TIME IN DAYS FROM 1/1/67=1 IF FINAL ARGUMENT
C IS 1, AND VICE VERSA IF 2
ISN 0091      CALL ADATE(JMS,JDS,JYS,DSTART(J),1)
ISN 0092      WRITE(5,2101) JX,JMS,JDS,JYS,RPLPWR(J),DSTART(J)
ISN 0093      2101 FORMAT(1H,I1,3I5,F10.3,F10.1)
ISN 0094      IF(JX.NE.1) GO TO 5
C
ISN 0096      WRITE(5,11021)
ISN 0097      11021 FORMAT(1H0,'CHEMICAL PARAMETER LIST(HYDROGEN CONC IN PPM,AND ',
                & 'PURIFICATION PLANT EFFICIENCY)'/1H,'IX IM ID IY',
                & ' PM2 GAM TH2(NOT INPUT)')
ISN 0098      6 I=I+1
C IX=FLAG// IM, ID, IY=MONTH,DAY,YEAR// PM2=HYDROGEN CONCENTRATION IN
C PRIMARY COOLANT IN VOLUME PPM// GAM=PURIFICATION PLANT EFFICIENCY
C BEGINNING THIS DATE
ISN 0099      READ(1,1102) IX,IM,ID,IY,PM2(I),GAM(I)
ISN 0100      1102 FORMAT(I1,3I5,2F10.3)
C TH2=DAYS FOR PERIOD DATE
ISN 0101      CALL ADATE(IM,ID,IY,TH2(I),1)
ISN 0102      WRITE(5,2102) IX,IM,ID,IY,PM2(I),GAM(I),TH2(I)
ISN 0103      2102 FORMAT(1H,I1,3I5,2F10.3,F10.1)
ISN 0104      IF(IX.NE.1) GO TO 6
C
ISN 0106      WRITE(5,11051)
ISN 0107      11051 FORMAT(1H0,'LIST OF EVENTS INDICATED BY CODE. 0=NO PRINTOUT,',
                & '9.=DEFAULT ENTRY.'/1H,' OTHER VALUES MAY BE ASSIGNED TO ',
                & 'ACTIVATE STATEMENTS NOT PRESENTLY PROGRAMMED'/1H,'KX KM KD',
                & ' KY CODE TEV(NOT INPUT)')
ISN 0108      7 K=K+1
C KX=FLAG// KM,KD,KY=MONTH,DAY,YEAR OF EVENT FLAGGED BY CODE(*)
ISN 0109      READ(1,1105) KX,KM,KD,KY,CODE(K)
ISN 0110      1105 FORMAT(I1,3I5,F10.1)
C TEV=DAYS CORRESPONDING TO PERIOD DATE
ISN 0111      CALL ADATE(KM,KD,KY,TEV(K),1)
C
ISN 0112      WRITE(5,2105) KX,KM,KD,KY,CODE(K),TEV(K)
ISN 0113      2105 FORMAT(1H,I1,3I5,2F10.1)
ISN 0114      IF(KX.NE.1) GO TO 7
C
C KSM,KSD,KSX AND KEM,KED,KEY ARE DATES AT WHICH COMPUTATION STARTS AND
C ENDS IN A POWER HISTORY WHICH EXTENDS BEYOND COMPUTATION SPAN
ISN 0116      WRITE(5,21031)

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ISN 0117 21031 FORMAT (1H0,'DATES STARTING AND STOPPING COMPUTATION SPAN'/1H ,
& ' KSM KSD KSY KEM KED KEY TBEGIN',' TEND')
ISN 0118 READ (1,1103) KSM,KSD,KSX,KEM,KED,KEY
ISN 0119 1103 FORMAT (6I5)
C TBEGIN =BEGINNING TIME OF COMPUTATION SPAN
ISN 0120 CALL ADATE (KSM,KSD,KSX,TBEGIN,1)
C TEND=ENDING TIME OF COMPUTATION SPAN
ISN 0121 CALL ADATE (KEM,KED,KEY,TEND,1)
ISN 0122 WRITE (5,2103) KSM,KSD,KSX,KEM,KED,KEY,TBEGIN,TEND
ISN 0123 2103 FORMAT (1H ,6I5,2F10.1)
C
ISN 0124 JZ=J
ISN 0125 IZ=I
ISN 0126 KZ=K
ISN 0127 DSTART (JZ+1)=99999.
ISN 0128 TH2 (IZ+1)=99999.
ISN 0129 TEV (KZ+1)=99999.
C
C SUBROUTINE HISTRY MERGES THE DATA TABLES FOR POWER HISTORY,
C CHEMICAL (PPM HYDROGEN AND PURIFICATION PLANT EFFICIENCY) AND EVENT
C LOGS INTO A SINGLE HISTORY, WITH NEW VARIABLE NAMES FOR THESE
ISN 0130 CALL HISTRY
C ATHE=ATOMS HELIUM IN THE PRIMARY SYSTEM
C HEORIG=ATOMS OF HELIUM-3 IN PRIMARY COOLANT AT ORIGINAL CONCENTRATION
ISN 0131 ATHE=WTHE*453.6*6.02D23/4.0026
ISN 0132 HEORIG=ABUND3*ATHE
C IN THE FOLLOWING LOOP WE CALCULATE FOR EACH REGION THE CARBON PORE
C VOLUME FROM DENSITY, THEN TOTAL REGION GAS SPACE, ATOMS PER CC AT
C REGION CONDITIONS ,THE FRACTION OF TRITIUM FROM HE3(N,P)T REACTION
C RECOILING TO SOLIDS IN THE REGION, AND THE ORIGINAL NUMBER OF HE-3
C ATOMS IN THE REGION
ISN 0133 RT=0.
ISN 0134 DO 2202 N=1,NREG
ISN 0135 CALL RECOIL (WTC1X(N),DENS(N),PRESS,RGTEMP(N),WINTAN(N),
& WEXTAN(N),PHOLES(N),V(N),HERFCL(N))
ISN 0136 ATCC(N)=6.02D23/22410.*PRESS*273.1/(273.1+RGTEMP(N))
ISN 0137 R(N)=V(N)*ATCC(N)*SIGFL(7)*FSLW(N)/FIXESI
ISN 0138 RT=RT+R(N)
ISN 0139 2202 CONTINUE
C
ISN 0140 DO 8 N=1,NREG
ISN 0141 FR(N)=R(N)/RT
ISN 0142 SA(7,N)=HEORIG*FR(N)
ISN 0143 DO 8 I=1,8
ISN 0144 SO(I,N)=SA(I,N)
ISN 0145 8 CONTINUE
C
C DATA INPUT COMPLETED
C
C HERE VARY THE CASES- THE EXPRESSIONS BETWEEN HERE AND STATEMENT 12
C SHOULD BE REPROGRAMMED TO SUIT THE NEED OF THE USER. SHOWN HERE ARE
C SEVEN CASES IN WHICH REACTOR OR OPERATION PARAMETERS ARE VARIED
ISN 0146 HXTRM=1.0
C LOOP HERE THE NUMBER OF CASES TO CONSIDER BEFORE RETURN
C TO STATEMENT 1 TO BEGIN INPUT TO RESTART CALCULATION FOR NEW REACTOR
ISN 0147 DO 14 ICASE=1,NCASES
ISN 0148 WRITE (5,20110)
ISN 0149 20110 FORMAT (14I)

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ISN 0150          DO 11 I=1,8
ISN 0151          DO 11 N=1,NREG
ISN 0152          SA(T,N)=SO(T,N)
ISN 0153          11 CONTINUE
ISN 0154          ICC=IC(ICASE)
ISN 0155          GO TO (201,202,203,204,205,206,207),ICC
ISN 0156          201 CONTINUE
ISN 0157          WRITE(5,20111)
ISN 0158          20111 FORMAT(1H1,'EASE CASE')
ISN 0159          GO TO 12
ISN 0160          202 CONTINUE
ISN 0161          DO 2021 I=1,NEND
ISN 0162          PPMH2(I)=3.
ISN 0163          2021 CONTINUE
ISN 0164          WRITE(5,2022)
ISN 0165          2022 FORMAT(1H1,'CASE = LOW HYDROGEN')
ISN 0166          GO TO 12
ISN 0167          203 CONTINUE
ISN 0168          DO 2031 I=1,NEND
ISN 0169          PPMH2(I)=300.
ISN 0170          2031 CONTINUE
ISN 0171          WRITE(5,2032)
ISN 0172          2032 FORMAT(1H1,'CASE = HIGH HYDROGEN')
ISN 0173          GO TO 12
ISN 0174          204 CONTINUE
ISN 0175          DO 2041 I=1,NEND
ISN 0176          PPMH2(I)=30.
ISN 0177          2041 CONTINUE
ISN 0178          DO 2043 N=1,NPEG
ISN 0179          HEXX(N)=HERECL(N)
ISN 0180          HERECL(N)=0.
ISN 0181          2043 CONTINUE
ISN 0182          WRITE(5,2042)
ISN 0183          2042 FORMAT(1H1,'CASE = ALL TRITIUM FROM HELIUM-3 RETAINED IN GAS,
& NONE TO SOLIDS')
ISN 0184          GO TO 12
ISN 0185          205 CONTINUE
ISN 0186          DO 2053 N=1,NREG
ISN 0187          HERECL(N)=HEXX(N)
ISN 0188          2053 CONTINUE
ISN 0189          DO 2051 I=1,NEND
ISN 0190          GAMM(I)=.5
ISN 0191          2051 CONTINUE
ISN 0192          WRITE(5,2052) GAMM(1)
ISN 0193          2052 FORMAT(1H1,'CASE = PURIFICATION PLANT EFFICIENCY SET TO ',F5.3)
ISN 0194          GO TO 12
ISN 0195          206 CONTINUE
ISN 0196          DO 2061 N=1,NREG
ISN 0197          TRAPX(N)=1.C
ISN 0198          2061 CONTINUE
ISN 0199          DO 2062 N=1,NEND
ISN 0200          GAMM(N)=1.C
ISN 0201          2062 CONTINUE
ISN 0202          WRITE(5,2063)
ISN 0203          2063 FORMAT(1H1,'CASE = ALL TRITIUM FORMED IN SOLIDS (IRREVERSIBLY
& BOUND')
ISN 0204          GO TO 12
ISN 0205          207 CONTINUE

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ISN 0206          DO 2071 N=1,NREG
ISN 0207          TRAPX(N)=.99
ISN 0208          2071 CONTINUE
ISN 0209          HXTERM=.1
ISN 0210          WRITE(5,2072)HXTERM
ISN 0211          2072 FORMAT(1H1,'CASE = HEAT EXCHANGER PERMEATION COEFFICIENTS REDUCED
& BY FACTOR = ',F5.3)
ISN 0212          GO TO 12
C
C END OF CASES. BEGIN CALCULATION ON GIVEN CASE
C
ISN 0213          12 CONTINUE
ISN 0214          WRITE(5,2080)
ISN 0215          2080 FORMAT(1H , 'TRITIUM GENERATION AND DISTRIBUTION IN A HIGH',
& ' TEMPERATURE GAS COOLED REACTOR')
ISN 0216          WRITE(5,20801)TITLE
ISN 0217          20801 FORMAT(1H ,18A4)
C
ISN 0218          DPUR=CPUR*86400.
ISN 0219          DGSOUT=GSOUT*86400.
ISN 0220          WRATE=86400.*SLOSS
ISN 0221          TFLXSL=FISCFN*ENS*1.D-6*FLXPSL
ISN 0222          TFLXFS=FISCFN*ENS*1.D-6*FLXPFS
ISN 0223          WRITE(5,2081) TMW,FISDEN,TEMP,PRESS,DPUR,WTHE,DGSOUT,ABUND3,
& H2O,WRATE,TFLXSL,TFLXFS
ISN 0224          2081 FORMAT(1H0,'FULL REACTOR POWER ',F6.0,' MWTM, ',F3.1,' KW/L, ',
& ' AVERAGE GAS TEMP ',F4.0,' DEG C, PRESSURE, ',F4.1,' ATM'//
& 1X,'PURIFICATION RATE ',OFF9.6,
& '/DAY '
& 1X,'HE AMOUNT',1PD9.2,' LBS HE MAKEUP RATE ',D9.2,
& '/DAY ',6X,'HE3 ABUNDANCE',D9.2/
& 1X,'WATER ',D9.2,' LBS WATER MAKEUP RATE ',OFF6.3,3X,
& '/DAY ',6X,'THEC CORE FLUX-SLOW',1PD9.2,' , FAST',D9.2)
C
C
ISN 0225          CALL HTEX(EXTERM)
ISN 0226          NEND1=NEND-1
C
ISN 0227          DO 13 N=1,NREG
ISN 0228          ISKIP(N)=0
ISN 0229          MSKIP(N)=0
ISN 0230          WRITE(5,2082) (RNAME(J,N),J=1,10)
ISN 0231          2082 FORMAT(1H0,10A4)
ISN 0232          WRITE(5,2083) WTC1X(N),PPMLI(N),FPMP(N),PPMBE(N),PPT(N),
& TRAPX(N)
ISN 0233          2083 FORMAT(1H , 'WEIGHT'1PD10.3,' LBS FPMLI=',OFF6.3,' PPMP=',
& F8.1,' PPMBE=',F6.3,' SURFACE AREA',F6.3,' SQ M/G,'
& , ' T BOUND IN SOLIDS',F7.4)
C
ISN 0234          CALL CROSS(N)
ISN 0235          IF(MSKIP(N).GT.0) GO TO 2201
ISN 0237          WRITE(5,2200) (SA(I,N),I=1,7)
ISN 0238          2200 FORMAT(1H , 'ORIGINAL NUMBER OF ATOMS IN REGION'/1X,7(3X,1PD10.4)
& , ' (REACTION FRACTION OF TOTAL HE-3) ')
ISN 0239          2201 CONTINUE
ISN 0240          MSKIP(N)=MSKIP(N)+1
C
ISN 0241          DO 13 I=1,8

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TSN 0242      DO 13 K=1,9
C Y=ATOMS TRITIUM FROM SOURCE I, IN SINK K, REGION N, CARRIED
C FORWARD FROM PRIOR PERIOD. HERE INITIALLY ZERO
TSN 0243      Y (I, K, N) = 0.0
TSN 0244      13 CONTINUE
C
C HERE BEGIN CALCULATIONS FOR INDIVIDUAL PERIODS
C
C INDEX L INDICATES PARTICULAR PERIOD
TSN 0245      DO 14 L=1, NEND1
C CALCULATE PERIOD LENGTH IN SECONDS
TSN 0246      T=8.64E4*(DT(L+1)-DT(L))
TSN 0247      TSAREA=0.0
C VFFSUM IS VOL*FLUX*FUEL SUM FOR REACTOR
TSN 0248      VFFSUM=0.
TSN 0249      DO 142 N=1, NREG
TSN 0250      TSAREA=TSAREA+WTCIX(N)*453.6*BET(N)
TSN 0251      VFFSUM=VFFSUM+V(N)*FSLW(N)*FUEL(N)
TSN 0252      142 CONTINUE
C
C INDEX K INDICATES PARTICULAR TRITIUM SINK
TSN 0253      DO 15 K=1,8
TSN 0254      15 TATOM(K)=0.00
C
C INDEX N INDICATES REGION FOR WHICH CALCULATION IS BEING MADE
TSN 0255      DO 16 N=1, NREG
C
TSN 0256      DO 18 T=1,6
C ADJUST SOURCE REACTION RATE FOR PERIOD RELATIVE POWER
TSN 0257      18 SIG(I) = SAG(I, N) *PWR(L)
TSN 0258      SIG(7) =PWR(L)*RT/ATHE
TSN 0259      FIS=SIGPL(8)*PWR(L)*TMW*3.1210E16*V(N)*FSLW(N)*FUEL(N)/VFFSUM
C
TSN 0260      PUR=GAMM(L)*CPUR
TSN 0261      CHANGE=PPMH2(1)/PPMH2(L)
TSN 0262      HEATX=CEATX*DSQRT(CHANGE)
TSN 0263      ATGAS=ATHE*PPMH2(L)*1.E-6
TSN 0264      PMM=PRESS*760.*PPMH2(L)*1.E-6
C
C ITERATIVE SOLUTION TO TEMPERATURE NON-DEPENDENT TEMKIN ISOTHERM
C FOR HYDROGEN CHEMISCRPTION ON GRAPHITE
TSN 0265      XX=.1
TSN 0266      NX=0
TSN 0267      186 YX0=XX
TSN 0268      NX=NX+1
TSN 0269      XX=SCC(1)/SCC(3)*DLOG(SCC(2)*PMM*(SCC(1)-XX0)/XX0)
TSN 0270      IF(((DABS((XX-YX0)/(XX+XX0))).GT..0002).AND.(NX.LT.20)) GO TO 186
TSN 0272      SCCSQM=XX
C
TSN 0273      ADATOM=TSARPA*SCCSQM*5.373E19
TSN 0274      SAD=ADATOM/ATGAS
TSN 0275      ADS=1.0+SAD
C
C INDEX I INDICATES SOURCE NUCLIDE
TSN 0276      DO 17 I=1,8
C REDISTRIBUTE THE CARRYFORWARD TRITIUM BETWEEN GAS AND ADSORBED
C PHASES FOR POSSIBLY CHANGED HYDROGEN CONCENTRATION IN PERIOD
TSN 0277      Y(I,4,N)=(Y(I,4,N)+Y(I,3,N))/ADS

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ISN 0278      Y(I,3,N)=Y(I,4,N)*SAD
ISN 0279      DO 17 K=1,9
ISN 0280      X(I,K,N)=0.0
ISN 0281      17 CONTINUE
C
C CALC RELATED TO GAS SYSTEM
C
C CALCULATE EFFECTIVE LOSS CONSTANT FROM PRIMARY COOLANT NODE
C (GAS AND ADSORBED)
ISN 0282      CLAM=(PUR+HEATX+GSOHT)/ADS+DCAY
C
C THROUGH STATEMENT 23,J IS AN INDEX INDICATING WHETHER WE CALCULATE
C TRITIUM REMAINING IN=1 OR EMERGING FROM=2 A GIVEN SINK NODE
ISN 0283      DO 19 J=1,2
ISN 0284      IF(J.EQ.2) GO TO 20
ISN 0285      AA=1.0
ISN 0287      BB=1.0
ISN 0288      TT=DEXP(-DCAY*T)
ISN 0289      VV=1.0
ISN 0290      FF=1.0
ISN 0291      S(1)=DCAY
ISN 0292      GO TO 21
ISN 0293      20 AA=ADS
ISN 0294      BB=1.0-TRAFX(N)
ISN 0295      TT=DEXP(-CLAM*I)
ISN 0296      VV=1.0-HERECL(N)
ISN 0297      FF=1.0-FTRAP
ISN 0298      S(1)=CLAM
ISN 0299      21 IF(L.EQ.1) TT=0.0
C
C X=ATOMS TRITIUM FROM SOURCE I,IN SINK K OF REGION N AT END
C OF CURRENT PERIOD
ISN 0301      S(2)=SIG(1)
ISN 0302      TMP(1,J)=(BB/AA)*SIG(1)*SA(1,N)*G(T,S,2)
ISN 0303      IF(J.EQ.1) X(1,1,N)=TMP(1,J)+Y(1,1,N)*TT
ISN 0305      IF(J.EQ.2) X(1,4,N)=TMP(1,J)+Y(1,4,N)*TT
C
ISN 0307      S(2)=SIG(2)
ISN 0308      TMP(2,J)=(FF/AA)*SIG(2)*SA(2,N)*G(T,S,2)
ISN 0309      IF(J.EQ.1) X(2,1,N)=TMP(2,J)+Y(2,1,N)*TT
ISN 0311      IF(J.EQ.2) X(2,4,N)=TMP(2,J)+Y(2,4,N)*TT
C
ISN 0313      S(2)=SIG(3)+SIG(4)
ISN 0314      TMP(3,J)=(BB/AA)*SIG(3)*SA(3,N)*G(T,S,2)
ISN 0315      IF(J.EQ.1) X(3,1,N)=TMP(3,J)+Y(3,1,N)*TT
ISN 0317      IF(J.EQ.2) X(3,4,N)=TMP(3,J)+Y(3,4,N)*TT
C
ISN 0319      S(2)=SIG(2)
ISN 0320      S(3)=SIG(3)+SIG(4)
ISN 0321      TMP(4,J)=(BB/AA)*SIG(2)*SIG(4)*SA(4,N)*G(T,S,3)
ISN 0322      IF(J.EQ.1) X(4,1,N)=TMP(4,J)+Y(4,1,N)*TT
ISN 0324      IF(J.EQ.2) X(4,4,N)=TMP(4,J)+Y(4,4,N)*TT
C
ISN 0326      S(2)=SIG(1)
ISN 0327      S(3)=SIG(6)
ISN 0328      TMP(5,J)=(BB/AA)*S(2)*SIG(6)*SIG(5)*SA(5,N)*H(T,S,3)
ISN 0329      IF(J.EQ.1) X(5,1,N)=TMP(5,J)+Y(5,1,N)*TT
ISN 0331      IF(J.EQ.2) X(5,4,N)=TMP(5,J)+Y(5,4,N)*TT

```

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C
ISN 0333      TMP(6,J) = (FP/AA) * SIG(1) * SIG(6) * SA(6,N) * G(T,S,3)
ISN 0334      IF (J.EQ.1) X(6,1,N) = TMP(6,J) + Y(6,1,N) * TT
ISN 0336      IF (J.EQ.2) X(6,4,N) = TMP(6,J) + Y(6,4,N) * TT

C
ISN 0338      S(2) = GSOUT + SIG(7)
ISN 0339      TMP(7,J) = (VV/AA) * SIG(7) * (SA(7,N) * G(T,S,2) + GSOUT * HEORIG *
& FR(N) * H(T,S,2))
ISN 0340      IF (J.EQ.1) X(7,1,N) = TMP(7,J) + Y(7,1,N) * TT
ISN 0342      IF (J.EQ.2) X(7,4,N) = TMP(7,J) + Y(7,4,N) * TT

C
ISN 0344      TMP(8,J) = (FP/AA) * FIS * H(T,S,1)
ISN 0345      IF (J.EQ.1) X(8,1,N) = TMP(8,J) + Y(8,1,N) * TT
ISN 0347      IF (J.EQ.2) X(8,4,N) = TMP(8,J) + Y(8,4,N) * TT
ISN 0349      19 CONTINUE

C
ISN 0350      DO 22 I=1,8
ISN 0351      X(I,3,N) = X(I,4,N) * SAD
ISN 0352      IF (I.LE.6) BB = TRAPX(N)
ISN 0354      IF (I.EQ.7) BB = HERECL(N)
ISN 0356      IF (I.EQ.8) BB = FTRAP
ISN 0358      X(I,2,N) = X(I,1,N) * BB
ISN 0359      22 CONTINUE

C
C CALC RELATED TO LOSSES FROM GAS SYSTEM AND WATER SYSTEM
ISN 0360      S(2) = CIAM
ISN 0361      AA = ADS
ISN 0362      PP = 1.0 - TRAPX(N)
ISN 0363      VV = 1.0 - HERECL(N)
ISN 0364      WF = 1.0 - FTRAP

C
ISN 0365      DO 23 J=1,2
ISN 0366      IF (J.EQ.2) GO TO 24
ISN 0368      S(1) = PCAY
ISN 0369      GO TO 25
ISN 0370      24 S(1) = DCAY + SICSS
ISN 0371      25 GK = G(T,S,2)
ISN 0372      IF (L.EQ.1) GK = 0.0

C
ISN 0374      S(3) = SIG(1)
ISN 0375      TMP(1,J) = (EB/AA) * SIG(1) * SA(1,N) * G(T,S,3) + Y(1,4,N) * GK

C
ISN 0376      S(3) = SIG(2)
ISN 0377      TMP(2,J) = (PB/AA) * SIG(2) * SA(2,N) * G(T,S,3) + Y(2,4,N) * GK

C
ISN 0378      S(3) = SIG(3) + SIG(4)
ISN 0379      TMP(3,J) = (EE/AA) * SIG(3) * SA(3,N) * G(T,S,3) + Y(3,4,N) * GK

C
ISN 0380      S(3) = SIG(2)
ISN 0381      S(4) = SIG(3) + SIG(4)
ISN 0382      TMP(4,J) = (EE/AA) * SIG(2) * SIG(4) * SA(4,N) * G(T,S,4) + Y(4,4,N) * GK

C
ISN 0383      S(3) = SIG(1)
ISN 0384      S(4) = SIG(6)
ISN 0385      TMP(5,J) = (FR/AA) * SIG(1) * SIG(6) * SIG(5) * SA(5,N) * H(T,S,4) + Y(5,4,N) * GK

C
ISN 0386      S(4) = SIG(2)
ISN 0387      TMP(6,J) = (EE/AA) * SIG(1) * SIG(6) * SA(6,N) * G(T,S,4) + Y(6,4,N) * GK

```

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C
ISN 0388      S(3)=GSOUT+SIG(7)
ISN 0389      TMP(7,J)=(VV/AA)*SIG(7)*(SA(7,N)*G(T,S,3)+GSOUT*HEORIG*
              F*FR(N)*H(T,S,3))+Y(7,4,N)*GK

C
ISN 0390      TMP(8,J)=(FF/AA)*FIS*H(T,S,2)+Y(8,4,N)*GK
ISN 0391      23  CONTINUE

C
ISN 0392      DO 25 I=1,8
ISN 0393      ZUK=DEXP(-DCAY*T)
ISN 0394      XUK=DEXP(-(DCAY+SLOSS)*T)
ISN 0395      IF(L.NE.1) GO TO 27
ISN 0397      ZUK=0.0
ISN 0398      XUK=0.0
ISN 0399      27  Y(I,6,N)=TMP(I,1)*PUR+Y(I,6,N)*ZUK
ISN 0400      X(I,5,N)=TMP(I,1)*GSCUT+Y(I,5,N)*ZUK
ISN 0401      X(I,9,N)=TMP(I,1)*HEATX+Y(I,9,N)*ZUK
ISN 0402      X(I,7,N)=TMP(I,2)*HEATX+Y(I,7,N)*XUK
ISN 0403      X(I,8,N)=X(I,9,N)-X(I,7,N)
ISN 0404      26  CONTINUE

C
C UPDATE THE SOURCE ATOM CONCENTRATIONS
ISN 0405      S(1)=SIG(1)
ISN 0406      S(2)=SIG(5)
ISN 0407      SA(1,N)=SA(1,N)*DEXP(-S(1)*T)+S(2)*SA(6,N)*G(T,S,2)+S(2)*SIG(5)*
              F*SA(5,N)*H(T,S,2)

C
ISN 0408      S(1)=SIG(2)
ISN 0409      S(2)=SIG(3)+SIG(4)
ISN 0410      SA(2,N)=SA(2,N)*DEXP(-S(1)*T)+SIG(4)*SA(4,N)*G(T,S,2)

C
ISN 0411      SA(3,N)=SA(3,N)*DEXP(-(SIG(3)+SIG(4))*T)

C
ISN 0412      SA(4,N)=SA(3,N)

C
ISN 0413      S(1)=SIG(6)
ISN 0414      SA(6,N)=SA(6,N)*DEXP(-S(1)*T)+SIG(5)*SA(5,N)*H(T,S,1)
ISN 0415      S(1)=GSOUT+SIG(7)
ISN 0416      SA(7,N)=SA(7,N)*DEXP(-S(1)*T)+GSCUT*HEORIG*FR(N)*H(T,S,1)

C
ISN 0417      DO 28 I=1,8
ISN 0418      DO 28 K=1,9
C ATOMS TRITIUM CARRYFORWARD FOR EACH (SOURCE,SINK,REGION)
ISN 0419      Y(I,K,N)=X(I,K,N)
C EXPRESS PRESENT ATOMS AS CURIES FOR OUTPUT
ISN 0420      X(I,K,N)=X(I,K,N)*4.84D-20
ISN 0421      28  CONTINUE

C
C CALC HERE COMPLETED FOR PERIOD
C
ISN 0422      IF(EVNT(L).EQ.0.) GO TO 14
ISN 0424      IF(N.GT.1) GO TO 29
ISN 0426      ATE=DT(L)
C CALCULATE DATE OF START OF CURRENT PERIOD
ISN 0427      CALL ADAIE(JM,JD,JY,ATE,2)
C CALCULATE DATE OF END OF CURRENT PERIOD
ISN 0428      DATE=DT(L+1)
ISN 0429      CALL ADATE(IM,ID,IY,DATE,2)

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C
C OUTPUT RESULTS FOR CURRENT PERIOD. HERE DO HEADINGS FOR PERIOD
ISN 0430 WRITE(5,2085) JM,JD,JY,IM,IP,IY,EWR(L),GAMM(L),HEATX,SAD,PPMH2(L)
ISN 0431 2085 FORMAT(1H1,'DURING THE PERIOD FROM ',3I3,' TO ',3I3,' THE POWER LE
&VEL WAS ',F5.2,' AND THE PURIFICATION PLANT EFFICIENCY WAS ',F5.2,
&,' THE HEAT EXCHANGER RATE OF ',1PF10.4,' (SEC-1) AND THE ADSORPT
&ION RATIO OF ',0FF9.4,' WERE BASED ON ',F7.2,' PPM H2')
C
ISN 0432 WRITE(5,2086)
ISN 0433 2086 FORMAT(1H0,'CUMULATIVE TRITIUM INVENTORY IN CURIES AS OF
&PERIOD END FOR PRODUCTION, AND DISTRIBUTION SINKS'/1X,
&'SOURCE -ATOMS',10X,'TRITIUM',8X,'BOUND IN',2X,
&' ADSORBED ON',3X,' IN PRIMARY',2X,' LEAKAGE FROM',2X,
&' PURIF*N',2X,' H2O IN STEAM ',3X,' PLOWDOWN+LOSS'/24X,' PRODUCTION',
& 5X,' SOLIDS ',2X,' GR. SURFACE',4X,' COOLANT',5X,' PRIMARY',7X,
& ' PLANT',8X,' GENERATOR',4X,' STEAM GENERATOR')
ISN 0434 29 CONTINUE
C
ISN 0435 WRITE(5,2084) (RNAME(J,N),J=1,10),HERECL(N)
ISN 0436 2084 FORMAT(1H0,10A4,' FRACTION OF HE3(N,P) T RECOILS BOUND IN SOLIDS',
&' =',
& F8.5)
ISN 0437 DO 30 I=1,8
ISN 0438 IF(JFLAG(I,N).EQ.0) GO TO 30
C OVERALL SUM OF TRITIUM CURIES FOR THE VARIOUS SINKS
ISN 0440 DO 31 K=1,8
ISN 0441 31 TATOM(K)=TATCM(K)+X(I,K,N)
C
ISN 0442 IF((SA(8,N).EQ.0.).AND.(IF(I).EQ.8)) GO TO 30
ISN 0444 WRITE(5,2087) ANAME(IP(I)),SA(IP(I),N),(X(IP(I),K,N),K=1,8)
ISN 0445 2087 FORMAT(1H ,A8,1PC12.4,CP8F13.6)
ISN 0446 30 CONTINUE
C DO SUM OF TRITIUM CURIES FOR THE VARIOUS SINKS OF GIVEN REGION
ISN 0447 DO 32 K=1,8
ISN 0448 ST(K)=0.
ISN 0449 DO 32 I=1,8
ISN 0450 ST(K)=ST(K)+X(I,K,N)
ISN 0451 32 CONTINUE
ISN 0452 WRITE(5,2088) (ST(K),K=1,8)
ISN 0453 2088 FORMAT(1H0,'REGION TOTALS ',8F13.6)
ISN 0454 IF(L.NE.NFND1) GO TO 16
ISN 0456 33 CONTINUE
C TD=SUM OF TRITIUM DISTRIBUTED IN REGION, TO CHECK AGAINST PRODUCTION
ISN 0457 TD=0.
ISN 0458 DO 34 K=2,8
ISN 0459 TD=TD+ST(K)
ISN 0460 34 CONTINUE
ISN 0461 WRITE(5,2089) ST(1),TD
ISN 0462 2089 FORMAT(1H ,',TRITIUM PRODUCED IN REGION ',3X,1PD22.15,10X,
&' TRITIUM DISTRIBUTED IN REGION ',E22.15)
ISN 0463 16 CONTINUE
C HERE COMPLETED OUTPUT FOR GIVEN REGION FOR CURRENT PERIOD.
C OUTPUT TOTALS FOR EACH SINK AT END OF PERIOD
C
ISN 0464 WRITE(5,2090) (TATOM(K),K=1,8)
ISN 0465 2090 FORMAT(1H0,'TOTAL CURIES',8X,8F13.6)
ISN 0466 DO 35 K=1,8
ISN 0467 35 CATOM(K)=TATCM(K)

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C
C REEXPRESS IN MICROCURIE, CONCENTRATION UNITS AS OF PERIOD END
ISN 0468 XATOM(2)=CATCM(2)*1.D6/WTC/453.6
ISN 0469 XATOM(3)=CATCM(3)*1.D6/WTC/453.6
ISN 0470 XATOM(4)=CATCM(4)*1.D6*4.0026/(WIHE*453.59*22410.)
ISN 0471 XATOM(7)=CATCM(7)*1.D6/H2O/453.6
ISN 0472 WRITE(5,2092) XATOM(2),XATOM(3),XATOM(4),XATOM(7)
ISN 0473 2092 FORMAT(1H0,'AVERAGE ACTIVITY',19X,1P3D13.4,26X,D13.4/
8' OF GIVEN REGION',22X,'MICROCURIES',2X,'MICROCURIES',2X,
5'MICROCURIES',28X,'MICROCURIES'/38X,'PER GRAM GR',2X,
8'PER GRAM GR',2X,'PER SCC HE',29X,'PER GRAM H2O')
ISN 0474 14 CONTINUE
C
ISN 0475 STOP
ISN 0476 END

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COMPILE OPTIONS - NAME= MAIN,OPT=00,LINECNT=60,SIZE=0000K,  
SOURCE,RECDIC,NOLIST,NODECK,LOAD,MAP,NOEDIT,NOID,NOXREF

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C END OF MAIN PROGRAM
C
C
C SUBROUTINE HISTORY MERGES THE SEVERAL INPUT TABLES OF TIME AND
C (1) RELATIVE POWER, (2) HYDROGEN CONCENTRATION AND PURIFICATION PLANT
C EFFICIENCY, AND (3) PARTICULAR EVENTS, AND STIPULATED SPAN OF
C COMPUTATION INTO A SINGLE TABLE. IT WILL RECEIVE 50 PERIODS IN ANY
C INPUT TABLE PROVIDED THE OUTPUT TABLE DOES NOT EXCEED 100 PERIODS
ISN 0002 SUBROUTINE HISTORY
ISN 0003 IMPLICIT REAL*8 (A-H,O-Z)
C
ISN 0004 COMMON/HIST/DSTART(50),TH2(50),CCDE(50),GAM(50),PM2(50),TEV(50),
ERELEPWR(50),JZ,IZ,KZ
C
ISN 0005 COMMON/B/PPMH2(100),PRESS,CEATX,WTC,GAMM(100),DT(100),
EPWR(100),EVNT(100),TEND,TBEGIN,WIHE,TEMP,NEND
C
ISN 0006 JB=0
ISN 0007 IP=0
ISN 0008 KB=0
ISN 0009 DO 20 J=1,JZ
ISN 0010 IF (DSTART(J).LT.TBEGIN) GO TO 20
ISN 0012 IF (JB.NE.0) GO TO 20
ISN 0014 JB=J
ISN 0015 20 CONTINUE
ISN 0016 DO 40 I=1,IZ
ISN 0017 IF (TH2(I).LT.TBEGIN) GO TO 40
ISN 0019 IF (IP.NE.0) GO TO 40
ISN 0021 IB=I
ISN 0022 40 CONTINUE
ISN 0023 DO 60 K=1,KZ
ISN 0024 IF (TEV(K).LT.TBEGIN) GO TO 60
ISN 0026 IF (KB.NE.0) GO TO 60
ISN 0028 KB=K
ISN 0029 60 CONTINUE
ISN 0030 J=JB
ISN 0031 I=IB
ISN 0032 K=KB
ISN 0033 X1=0.
ISN 0034 X2=0.
ISN 0035 X3=0.
ISN 0036 X4=0.
ISN 0037 IF (JB.GT.1) X1=RELEPWR(JB-1)
ISN 0039 IF (IB.GT.1) X2=PM2(IB-1)
ISN 0041 IF (IB.GT.1) X3=GAM(IB-1)
ISN 0043 IF (KB.GT.1) X4=CCDE(KB-1)
ISN 0045 N=0
ISN 0046 TX=TBEGIN
ISN 0047 301 N=N+1
C DT=TIME AT START OF CURRENT PERIOD
ISN 0048 DT(N)=DMIN1(DSTART(J),TH2(I),TEV(K),TX)
ISN 0049 TX=TEND
C PWR=RELATIVE POWER FOR CURRENT PERIOD
ISN 0050 PWR(N)=X1
C PPMH2=HYDROGEN CONCENTRATION (VOL PPM) IN PRIMARY COOLANT, FOR
C CURRENT PERIOD

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TSN 0051      PPMH2(N)=X2
              C  GAMM=PURIFICATION PLANT EFFICIENCY FOR CURRENT PERIOD
TSN 0052      GAMM(N)=X3
              C  EVNT=EVENT FLAG VALUE FOR CURRENT PERIOD
TSN 0053      EVNT(N)=X4
TSN 0054      IF (DSTART(J).NE.DT(N)) GO TO 302
TSN 0056      PWR(N)=RELPWR(J)
TSN 0057      X1=RELPWR(J)
TSN 0058      J=J+1
TSN 0059      302  IF (TH2(I).NE.DT(N)) GO TO 303
TSN 0061      PPMH2(N)=PM2(I)
TSN 0062      X2=PM2(I)
TSN 0063      GAMM(N)=GAM(I)
TSN 0064      X3=GAM(I)
TSN 0065      I=I+1
TSN 0066      303  IF (TEV(K).NE.DT(N)) GO TO 304
TSN 0068      EVNT(N)=CODE(K)
TSN 0069      X4=CODE(K)
TSN 0070      K=K+1
TSN 0071      304  CONTINUE
TSN 0072      IF (DT(N).LT.TEND) GO TO 301
TSN 0074      280  NEND=N
TSN 0075      RETURN
TSN 0076      END

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COMPILER OPTIONS - NAME= MAIN,OPT=00,LINECNT=60,SIZE=0000K,  
SOURCE,EBCDIC,NOLIST,NODECK,LOAD,MAP,NOEDIT,NOTD,NOYREF

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C
C  SUBROUTINE ADATE CONVERTS A DATE (MCNTH, DAY, YEAR) INTO TIME IN DAYS
C  FROM 1/1/67=1 IF FINAL ARGUMENT IS 1, AND VICE VERSA IF 2
      SUBROUTINE ADATE (IM, ID, IY, DAYS, IFLAG)
      IMPLICIT REAL*8 (A-H, O-Z)
      DIMENSION DMC (13)
      DATA DMC / 0., 31., 59., 90., 120., 151., 181., 212., 243., 273., 304., 334.,
& 365. /
      IF (IFLAG.EQ.2) GO TO 20
      DAYS=-365.
      DO 10 I=67, IY
      DAYS=DAYS+365.
      IF ((4*(I/4)).EQ.I) DAYS=DAYS+1.
10     CONTINUE
      IF (IM.LE.2.AND.(4*(IY/4)).EQ.IY) DAYS=DAYS-1
      DAYS=DAYS+DMC (IM)+ID
      RETURN
      IY=67
      DAY=DAYS
21     IF (4*(IY/4).NE.IY) GO TO 22
      X=1.
      IF (DAY-366.) 3, 3, 4
4      DAY=DAY-366.
      IY=IY+1
      GO TO 21
      X=0.
22     IF (DAY-365.) 3, 3, 5
      DAY=DAY-365.
      IY=IY+1
      GO TO 21
      DO 6 I=1, 12
      DX=DMC (I+1)
      IF (I.GT.2) DX=DX+X
      IF (DAY.LE.DX) GO TO 7
6      CONTINUE
7      IF (I.IT.3) X=0.
      IM=I
      ID=DAY-DMC (I)-X
      RETURN
      END
ISN 0002
ISN 0003
ISN 0004
ISN 0005
ISN 0006
ISN 0008
ISN 0009
ISN 0010
ISN 0011
ISN 0013
ISN 0014
ISN 0016
ISN 0017
ISN 0018
ISN 0019
ISN 0020
ISN 0022
ISN 0023
ISN 0024
ISN 0025
ISN 0026
ISN 0027
ISN 0028
ISN 0029
ISN 0030
ISN 0031
ISN 0032
ISN 0033
ISN 0034
ISN 0036
ISN 0038
ISN 0039
ISN 0041
ISN 0042
ISN 0043
ISN 0044

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COMPILER OPTIONS - NAME= MAIN,OPT=00,LINECNT=60,SIZE=0000K,  
SOURCE,EPCDIC,NOLIST,NODECK,LOAD,MAP,NORDIT,NOID,NOXREF

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C
C SUBROUTINE RECOIL CALCULATES (1) THE CM RANGE IN HELIUM AT REACTOR
C TEMPERATURE AND PRESSURE OF TRITON RECOILS FROM HE-3(N,P)T REACTION,
C (2) THE REGION HELIUM SPACE IN CUBIC CENTIMETERS,
C INCLUDING ANY CHANNELS AND HOLES, INTERNAL AND EXTERNAL ANNULI, AND
C PORES, AND (3) THE OVERALL FRACTION OF RECOILS WHICH WILL STRIKE
C ADJACENT SOLIDS.
C VALUES OF RATIO CONSTANTS ARE BASED ON FORT ST. VRAIN REACTOR BUT
C SHOULD APPLY ADEQUATELY TO OTHER REACTORS WITH SIMILAR CONFIGURATION
C
ISN 0002     SUBROUTINE RECOIL(WTC1X,DENS,PRESS,RGTEMP,WINTAN,WEXTAN,
& FHOLES,V,HFRECL)
ISN 0003     IMPLICIT REAL*8(A-H,O-Z)
C RANGE BASED ON FORMULAS IN ETHERINGTON 57 NUC ENG HANDBOOK, P.2-34,
C WITH TRITIUM RECOIL ENERGY OF .2 MEV
ISN 0004     ET=.2
ISN 0005     RANGEA=.56*ET
ISN 0006     RANGEB=.56*{(4.0026)**.3333333333)*RANGEA
ISN 0007     DENHE=4002.6/22410.*PRESS*273.1/(273.1+RGTEMP)
ISN 0008     RANGE=4./3.*RANGEB/DENHE
C
ISN 0009     VC=4246.513
ISN 0010     VT=5430.32
ISN 0011     VCARB=WTC1X*453.6/DENS
ISN 0012     VPORES=VCARB*(1.-DENS/2.25)
ISN 0013     RINTAN=211.859*WINTAN/.02/VC
ISN 0014     RHOLES=.2562287*FHOLES
ISN 0015     VNET=VCARB*(1.+RINTAN+RHOLES)
ISN 0016     REXTAN=940.282*WEXTAN/.05/VT
ISN 0017     V=VCARB*(RHOLES+RINTAN)+VNET*REXTAN+VPORES
C
ISN 0018     HRCL1=.02823*RANGE/.0508
ISN 0019     HRCL2=1.-WINTAN/2./RANGE
ISN 0020     HRCL3=RANGE/2./WEXTAN/2.54
ISN 0021     HERBCL=(VCARB*(RHOLES*HRCL1+RINTAN*HRCL2)+VNET*REXTAN*HRCL3
& +VPORES*1.)/V
ISN 0022     RETURN
ISN 0023     END

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COMPILER OPTIONS - NAME= MAIN,OPT=00,LINECNT=60,SIZE=0000K,
                  SOURCE,EBCDIC,NOLIST,NODECK,LOAD,MAP,NORDIT,NOID,NOXREF
C
C SUBROUTINE HTEX,PRESENTLY SET UP FOR UP TO 10 COMPONENTS OR
C SURFACES, CALCULATES THE FRACTIONAL ICSS RATE OF TRITIUM IN THE
C PRIMARY COOLANT BY PERMEATION IN ASSOCIATION WITH HYDROGEN THROUGH
C THE SYSTEM SURFACES. HXTERM IS THE RATIO OF OVERALL PERMEATION
C RATE ASSUMED FOR A PARTICULAR CASE TO THAT RESULTING IF ALL REGIONS
C USED PURE METAL VALUES FOR PERMEATION COEFFICIENTS
ISN 0002      SUBROUTINE HTEX(HXTERM)
ISN 0003      IMPLICIT REAL*8 (A-H,O-Z)
ISN 0004      REAL*4 XNAME
ISN 0005      DIMENSION SCCHR(10)
C
ISN 0006      COMMON/HX/ HEAT(10),AREA(10),THICK(10),STMP(10),CONS(10),
& ENRGY(10),XNAME(5,10),NCOMP
C
ISN 0007      COMMON/R/PPMH2(100),PRESS,CEATX,WTC,GAMM(100),DT(100),
& PWR(100),EVNT(100),TEND,TEEGIN,WIHE,TEMP,NEND
C
C H3H1=RATIO OF TRITIUM PERMEATION RATE TO THAT OF NATURAL HYDROGEN
ISN 0008      H3H1=.60
ISN 0009      TOTHT2=PPMH2(1)*1.D-6*WTC*453.6/4.0026*22410.
ISN 0010      PH2STD=PRESS*PPMH2(1)*1.D-6
ISN 0011      CEATX=0.0
ISN 0012      DO 50 I=1,NCOMP
C CALCULATE THE STANDARD NATURAL HYDROGEN PERMEATION COEFFICIENT
C FOR GIVEN TEMPERATURE AND COMPONENT PURE METAL
ISN 0013      SCCHR(I)=CCNS(I)*DEXP(-ENERGY(I)/(1.987*(273.1+STMP(I))))
C HEAT=FRACTION OF PRIMARY COOLANT GAS PHASE TRITIUM PERMEATING
C A PARTICULAR COMPONENT PER SECOND FOR PURE METAL, AT REFERENCE INITIAL
C HYDROGEN CONCENTRATION
ISN 0014      HEAT(I)=H3H1*AREA(I)/THICK(I)*DSQRT(PH2STD)*SCCHR(I)/TOTHT2/3600.
C SUM OVER ALL COMPONENTS AND CORRECT FOR ALTERED PERMEATION
C COEFFICIENT EMBEDDED IN HXTERM
ISN 0015      CEATX=CEATX+HEAT(I)*HXTERM
ISN 0016      50 CONTINUE
C
C OUTPUT THE RESULTS OF ABOVE CALCULATIONS
ISN 0017      WRITE(5,60) CEATX,PPMH2(1)
ISN 0018      60  FORMAT(1H0,5X,'TOTAL STEAM GENERATOR RATE CONSTANT=',1PD12.4,
& ' SEC-1 WAS BASED ON ',0PF7.2,' EPM H2')
ISN 0019      DO 56 J=1,NCOMP
ISN 0020      WRITE(5,61) (XNAME(K,J),K=1,5),HEAT(J),AREA(J),THICK(J),
& STMP(J),SCCHR(J)
ISN 0021      61  FORMAT(1H ,5A4,1FD12.4,' SEC-1,'
& D12.4,' SQCM,',0PF8.2,' MM,',F7.1,' DEG C,',1PD10.3,
& ' STD CC H2,MM/SQCM,HR AT 1 ATM')
ISN 0022      56  CONTINUE
C
ISN 0023      RETURN
ISN 0024      END

```

```

COMPILER OPTIONS - NAME= MAIN,OPT=00,LINECNT=60,SIZE=0000K,
                  SOURCE,FECDIC,NOLIST,NODECK,LOAD,MAP,NOEDIT,NOID,NOXREF
C
C SUBROUTINE CROSS PRODUCES FULL POWER REACTION RATES OF THE VARIOUS
C SOURCE NUCLIDES IN EACH REGION AT THE INPUT SLOW OR FAST FLUX VALUES,
C BASED ON THE VALUES GIVEN IN THE BLOCK DATA SECTION FOR THE
C CROSS SECTION FOR SLOW OR FAST NEUTRONS PER SOURCE NEUTRON PER CC
C PER SECOND
ISN 0002      SUBROUTINE CROSS(N)
ISN 0003      IMPLICIT REAL*8(A-H,O-Z)
C
ISN 0004      COMMON/A1/ANAME(9),SAG(10,15),FNS,IMW,WTC1X(15),TRAPX(15)
& ,SIGFL(9),SA(10,15),ENC(15),THE,FISDEN,FFST(15),FSLW(15)
& ,FLXPSL,FLXFFS
& ,ISKIP(15),MSKIP(15)
C
ISN 0005      COMMON/B/PEMH2(100),PRESS,CEATX,WTC,GAMM(100),DT(100),
&PWR(100),EVNT(100),TEND,TBEGIN,WTHE,TEMP,NEND
C
C FLXPSL=PERRY SLOW FLUX PER SOURCE NEUTRON PER CC-SEC
C FLXFFS=PERRY FAST FLUX PER SOURCE NEUTRON PER CC-SEC
C SNCS=SOURCE NEUTRONS PER CM**3-SEC//
C TFLXSL=THEORETICAL SLOW FLUX //TFLXFS=THEORETICAL FAST FLUX
ISN 0006      SNCS=FISDEN*ENS*1.0E-6
ISN 0007      TFLXSL=FLXPSL*SNCS
ISN 0008      TFLXFS=FLXFFS*SNCS
ISN 0009      CFSTT=FFST(N)/TFLXFS
ISN 0010      CSLW =FSLW(N)/TFLXSL
C CORRECTION TO SIGFL FOR POWER FACTOR AND DEVIATION FROM THEO. FLUX
C SAG=EFFECTIVE FULL POWER REACTION RATE FOR A GIVEN SOURCE
C NUCLIDE AT THE ASSIGNED FLUX OF A GIVEN REGION
ISN 0011      DO 10 I=1,7
ISN 0012      SAG(I,N)=SIGFL(I)*SNCS
ISN 0013      GO TO(8,9,9,8,9,9,8),I
ISN 0014      8 SAG(I,N) = SAG(I,N)*CSLW
ISN 0015      GO TO 10
ISN 0016      9 SAG(I,N) = SAG(I,N)*CFSTT
ISN 0017      10 CONTINUE
C VALUE FOR FUEL IS SIMPLY THE TRITIUM FISSION YIELD. IN MAIN PROGRAM
C IT WILL BE MULTIPLIED BY THE FISSION RATE RATHER THAN FLUX
ISN 0018      SAG(8,N)=SIGFL(8)
ISN 0019      IF(ISKIP(N).NE.0) GO TO 78
C
ISN 0021      WRITE(5,77) FSLW(N),FFST(N)
ISN 0022      77 FORMAT(1H,'ASSIGNED SLOW FLUX=',1E10.2,6X,'ASSIGNED FAST FLUX=',
&D10.2)
C
ISN 0023      WRITE(5,102) (ANAME(I),I=1,7)
ISN 0024      102 FORMAT(1H,'EFFECTIVE CROSS SECTION FLUX PRODUCTS ',/,7(3X,A8,2X),
& 3X,'T FISSION YIELD')
C
ISN 0025      WRITE(5,101) (SAG(I,N),I=1,8)
ISN 0026      101 FORMAT(8(3X,1PE10.4))
ISN 0027      78 ISKIP(N)=ISKIP(N)+1
ISN 0028      RETURN
ISN 0029      END

```

```

COMELER OPTIONS - NAME= MAIN,OPT=00,LINECNT=60,SIZE=0000K,
                  SOURCE,EPCDIC,NOLIST,NODECK,LOAD,MAP,NOPDIT,NOTD,NOXREF
C
C FUNCTIONS G AND H USE AS ARGUMENTS THE INTERVAL DURATION, THE
C OVERALL LOSS TERMS FROM EACH NODE AND THE NUMBER OF LOSS NODES.
C FUNCTION G IS USED FOR A SOURCE WHICH IS DELETED DURING THE PERIOD.
C IT IS TO BE MULTIPLIED BY THE SOURCE AMOUNT AT THE BEGINNING OF THE
C PERIOD AND THE TRANSFER RATE TERMS FROM NODE TO NODE.
C FUNCTION H IS USED FOR A STEADY INPUT OR FORCING FUNCTION FOR
C THE PERIOD. IT IS TO BE MULTIPLIED BY THE INPUT RATE TERM, AND THE
C NODE TO NODE TRANSFER TERMS.
C THESE FUNCTIONS ARE TO BE APPLIED TO ALL PATHS BETWEEN ALL SOURCE
C AND ALL SINK NODES. ( NODE = COMPARTMENT )
C FUNCTIONS G AND H BY C. W. NESTOR, COMPUTER SCIENCES DIVISION, ORNL
C
ISN 0002      FUNCTION G(T, S, N)
ISN 0003      IMPLICIT REAL*8 (A-H, O-Z)
C
C              DOUBLE-PRECISION CALCULATION OF
C
C              SUMJ=1N EXP(-S(J)*T) * PRCD 1/(S(I)-S(J))
C              I=1
C              I.NE.J
C
ISN 0004      DIMENSION S(N)
ISN 0005      SUM=0.D0
ISN 0006      DO 105 J=1,N
C
C              COMPUTE ARGUMENT OF EXPONENTIAL AND SEE IF IT IS IN RANGE.
C              IF NOT, SKIP CALCULATION OF THE PRODUCT.
C
ISN 0007      X = S(J)*T
ISN 0008      IF(X .GT. 174.D0) GO TO 105
C
C              COMPUTE PRODUCT, SKIPPING THE I = J TERMS.
C
ISN 0010      PROD = 1.D0
ISN 0011      DO 100 I=1,N
ISN 0012      IF(I.EQ.J) GO TO 100
ISN 0014      PROD = PROD / (S(I) - S(J))
ISN 0015      100 CONTINUE
ISN 0016      SUM = SUM + PROD*DEXP(-X)
ISN 0017      105 CONTINUE
ISN 0018      G = SUM
ISN 0019      RETURN
ISN 0020      END

```

COMPILE OPTIONS - NAME= MAIN,OPT=00,LINECNT=60,SIZE=0000K,  
SOURCE,EBCDIC,NOLIST,NODECK,LOAD,MAP,NORDIT,NOID,NOXREF

```

C
ISN 0002      FUNCTION H(T, S, N)
ISN 0003      IMPLICIT REAL*8 (A-H, O-Z)
ISN 0004      DIMENSION S(N)

C
C      DOUBLE-PRECISION CALCULATION OF
C
C      N
C      SUM ((1 - EXP(-S(J)*T)) / S(J)) * EPROD 1/(S(I)-S(J))
C      J=1                                I=1
C                                          T.NF.J
C
C      CONSTANTS FOR PADE' APPROXIMANT FOR SMALL ARGUMENT ( .LF. 0.2)
C
ISN 0005      DATA A1, A2, A3, A4, A5, A6 / -2.97619E-4, .0125D0, -.125D0,
& 3.D0, -4.5D0, 99.D0/
ISN 0006      DATA B1, B2, B3, B4, B5, B6 / -1.7857143E-3, .0625D0, 1.D0,
& 9.D0, 45.D0, 99.D0/

C
ISN 0007      SUM = 0.D0
ISN 0008      DO 105 J=1,N
ISN 0009      PROD = 1.D0
ISN 0010      DO 100 I=1,N
ISN 0011      IF(I.EQ.J) GC TO 100
ISN 0012      PROD = PROD / (S(I) - S(J))
ISN 0013      100 CONTINUE
ISN 0014      X = S(J)*T
ISN 0015      P = 0.D0
ISN 0016      IF(X .GT. (.2D0)) GO TO 101
ISN 0017      Q = (((B1*X + E2)*X + B3)*X + E4)*X + E5)*X + B6
ISN 0018      Q = T* (((A1*X + A2)*X + A3)*X + A4)*X + A5)*X + A6) / Q
ISN 0019      GO TO 102
ISN 0020      101 IF(X .LT. 174.D0) P = DEXP(-X)
ISN 0021      Q = (1.D0 - P) / S(J)
ISN 0022      102 SUM = SUM + EPROD*Q
ISN 0023      105 CONTINUE
ISN 0024      H = SUM
ISN 0025      RETURN
ISN 0026      END

```



9.3 Appendix III: Output (Partial)

9.3.1 Appendix III(a): Replica of Input

COPY OF INPUT VARIABLE NAMES AND DATA RECORDS FOLLOWS, ALONG  
WITH RELATIVE TIMES (1/1/67=1) ASSOCIATED WITH DATES

NCASES

1

IC

1

TITLE

PORT SAINT VRAIN REACTOR

THW	FISDEN
842.00	6.30

WTHE	PRESS	TEMP	ABUND3	GSOUT	CPUR
6170.00	47.60	589.00	2.00D-07	3.17D-10	4.10E-05

H2O	H2OM
8.80D 03	2.40D 05

NREG  
4

RNAME  
CORE

RGNLBS	DENS	BET	RGTEMP	TRAPX
413000.	1.770	0.200	732.000	0.990
FHOLES	WINTAN	WEXTAN		
0.6250	0.0050	0.0400		
FUEL	FRACC	PPMLI	PPMB	PPMBE
1.	1.0000000	0.050	320.0	0.0
FFST	FSLW			
3.60D 13	6.00D 13			

RNAME  
REFLECTOR

RGNLBS	DENS	BET	RGTEMP	TRAPX
603500.	1.770	0.200	600.000	0.990
FHOLES	WINTAN	WEXTAN		
0.1700	0.0	0.0400		
FUEL	FRACC	PPMLI	PPMB	PPMBE
0.	1.0000000	0.050	1.0	0.0
FFST	FSLW			
1.50D 11	3.20D 12			

RNAME  
BORONATED REFLECTOR

RGNLBS	DENS	BET	RGTEMP	TRAPX
415300.	1.770	0.200	600.000	0.990
FHOLES	WINTAN	WEXTAN		
0.0	0.0	0.0400		
FUEL	FRACC	PPMLI	PPMB	PPMBE
0.	1.0000000	0.050	6000.0	0.0
FFST	FSLW			
1.90D 08	1.20D 11			

RNAME  
CONTROL RODS

RGNLBS	DENS	BET	RGTEMP	TRAPX
1060.	1.770	0.200	600.000	0.990
FHOLES	WINTAN	WEXTAN		
0.0	0.0	0.0500		
FUEL	FRACC	PPMLI	PPMB	PPMBE
0.	1.0000000	0.270	145000.0	0.0
FFST	FSLW			
1.40D 13	5.50D 11			

NCOMP  
6

XNAME	AREA	THICK	STMP	CCNS	ENERGY
ECONOMIZER	1.63D 07	3.50	332.00	1.44D 02	9.10D 03
XNAME	AREA	THICK	STMP	CCNS	ENERGY
EVAPORATOR 1	5.50D 06	3.50	410.00	1.44D 02	9.10D 03
XNAME	AREA	THICK	STMP	CCNS	ENERGY
EVAPORATOR 2	5.00D 06	5.71	440.00	1.44D 02	9.10D 03
XNAME	AREA	THICK	STMP	CCNS	ENERGY
SUPERHEATER 1	5.10D 06	5.71	472.00	1.44D 02	9.10D 03
XNAME	AREA	THICK	STMP	CCNS	ENERGY
SUPERHEATER 2	1.00D 07	5.21	583.00	1.21D 03	1.75D 04
XNAME	AREA	THICK	STMP	CCNS	ENERGY
REHEATER	5.60D 06	3.56	632.00	1.21D 03	1.75D 04

HISTORY LISTS--VALUE OF 1 IN FIRST COLUMN INDICATES END OF LIST  
POWER HISTORY LIST

JX	JMS	JDS	JYS	RELPR	DSTART (NOT INPUT)
1	1	1	74	0.800	2558.0

CHEMICAL PARAMETER LIST (HYDROGEN CONC IN PPM, AND PURIFICATION PLANT EFFICIENCY)

IX	IM	ID	IY	PM2	GAM	TH2 (NOT INPUT)
1	1	1	74	30.000	1.000	2558.0

LIST OF EVENTS INDICATED BY CODE. 0=NO PRINTOUT, 9.=DEFAULT ENTRY.

OTHER VALUES MAY BE ASSIGNED TO ACTIVATE STATEMENTS NOT PRESENTLY PROGRAMMED

KX	KN	KD	KY	CODE	TEV (NOT INPUT)
0	1	1	74	9.0	2558.0
0	1	1	75	0.0	2923.0
0	1	1	76	0.0	3288.0
0	1	1	77	0.0	3654.0
0	1	1	78	0.0	4019.0
0	1	1	79	9.0	4384.0
1	1	1	80	9.0	4749.0

DATES STARTING AND STOPPING COMPUTATION SPAN

KSM	KSD	KSY	KEM	KED	KEY	TBEGIN	TEND
1	1	74	1	1	80	2558.0	4749.0



9.3.2 Appendix III(b): Reactor Characteristics:  
Base Case

EASE CASE  
 TRITIUM GENERATION AND DISTRIBUTION IN A HIGH TEMPERATURE GAS COOLED REACTOR  
 PORT SAINT VRAIN REACTOR

FULL REACTOR POWER 842. MWTH, 6.3 KW/L, AVERAGE GAS TEMP 589. DEG C, PRESSURE, 47.6 ATM

PURIFICATION RATE 3.542800/DAY  
 HE AMOUNT 6.17D 03 LBS HE MAKEUP RATE 2.74D-05/DAY HE3 ABUNDANCE 2.00D-07  
 WATER 8.80D 03 LBS WATER MAKEUP RATE 27.273 /DAY THEO CORE FLUX-SLOW 7.71D 13, FAST 2.46D 13

TOTAL STEAM GENERATOR RATE CONSTANT= 1.7354D-05 SEC-1 WAS BASED ON 30.00 PPM H2  
 ECONOMIZER 4.6399D-06 SEC-1, 1.6300D 07 SQCM, 3.50 MM, 332.0 DEG C, 7.436D-02 STD CC H2,MM/SQCM,HR AT 1 ATM  
 EVAPORATOR 1 3.7154D-06 SEC-1, 5.5000D 06 SQCM, 3.50 MM, 410.0 DEG C, 1.765D-01 STD CC H2,MM/SQCM,HR AT 1 ATM  
 EVAPORATOR 2 2.7450D-06 SEC-1, 5.0000D 06 SQCM, 5.71 MM, 440.0 DEG C, 2.340D-01 STD CC H2,MM/SQCM,HR AT 1 ATM  
 SUPERHEATER 1 3.6892D-06 SEC-1, 5.1000D 06 SQCM, 5.71 MM, 472.0 DEG C, 3.083D-01 STD CC H2,MM/SQCM,HR AT 1 ATM  
 SUPERHEATER 2 1.0552D-06 SEC-1, 1.0000D 07 SQCM, 5.21 MM, 583.0 DEG C, 4.103D-02 STD CC H2,MM/SQCM,HR AT 1 ATM  
 REHEATER 1.5093D-06 SEC-1, 5.6000D 06 SQCM, 3.56 MM, 632.0 DEG C, 7.161D-02 STD CC H2,MM/SQCM,HR AT 1 ATM

CORE  
 WEIGHT 4.130D 05 LBS PPMLI= 0.050 PPMB= 320.0 PPMBE= 0.0 SURFACE AREA 0.200 SQ M/G, T BOUND IN SOLIDS 0.9900  
 ASSIGNED SLOW FLUX= 6.00D 13 ASSIGNED FAST FLUX= 3.60D 13  
 EFFECTIVE CROSS SECTION FLUX PRODUCTS  
 LI-6 LI-7 PB-10 SB-10 C-12 BE-9 HE-3 T FISSION YIELD  
 2.4459D-08 2.5848D-13 4.8672D-13 9.7834D-08 5.2848D-15 5.9472D-13 1.3682D-07 1.0000D-04  
 ORIGINAL NUMBER OF ATOMS IN REGION  
 6.0296D 22 7.5232D 23 6.5426D 26 6.5426D 26 9.2849D 30 0.0 7.8887D 22 (REACTION FRACTION OF TOTAL HE-3)

REFLECTOR  
 WEIGHT 6.035D 05 LBS PPMLI= 0.050 PPMB= 1.0 PPMBE= 0.0 SURFACE AREA 0.200 SQ M/G, T BOUND IN SOLIDS 0.9900  
 ASSIGNED SLOW FLUX= 3.20D 12 ASSIGNED FAST FLUX= 1.50D 11  
 EFFECTIVE CROSS SECTION FLUX PRODUCTS  
 LI-6 LI-7 PB-10 SB-10 C-12 BE-9 HE-3 T FISSION YIELD  
 1.3045D-09 1.0770D-15 2.0280D-15 5.2178D-09 2.2020D-17 2.4780D-15 7.2968D-09 1.0000D-04  
 ORIGINAL NUMBER OF ATOMS IN REGION  
 8.8108D 22 1.0993D 24 2.9876D 24 2.9876D 24 1.3568D 31 0.0 5.1811D 21 (REACTION FRACTION OF TOTAL HE-3)

FORNATED REFLECTOR  
 WEIGHT 4.153D 05 LBS PPMLI= 0.050 PPMB= 6000.0 PPMBE= 0.0 SURFACE AREA 0.200 SQ M/G, T BOUND IN SOLIDS 0.9900  
 ASSIGNED SLOW FLUX= 1.20E 11 ASSIGNED FAST FLUX= 1.90E 08  
 EFFECTIVE CROSS SECTION FLUX PRODUCTS  
 LI-6 LI-7 PB-10 SB-10 C-12 BE-9 HE-3 T FISSION YIELD  
 4.8917D-11 1.3642D-18 2.5688D-18 1.9567D-10 2.7892D-20 3.1388D-18 2.7363D-10 1.0000D-04  
 ORIGINAL NUMBER OF ATOMS IN REGION  
 6.0632D 22 7.5651D 23 1.2336D 28 1.2336D 28 9.3366D 30 0.0 1.1719D 20 (REACTION FRACTION OF TOTAL HE-3)

CONTROL RODS  
 WEIGHT 1.060D 03 LBS PPMLI= 0.270 PPMB=145000.0 PPMBE= 0.0 SURFACE AREA 0.200 SQ M/G, T BOUND IN SOLIDS 0.9900  
 ASSIGNED SLOW FLUX= 5.50D 11 ASSIGNED FAST FLUX= 1.40E 13  
 EFFECTIVE CROSS SECTION FLUX PRODUCTS  
 LI-6 LI-7 PB-10 SB-10 C-12 BE-9 HE-3 T FISSION YIELD  
 2.2420D-10 1.0052D-13 1.8928D-13 8.9682D-10 2.0552D-15 2.3128D-13 1.2541D-09 1.0000D-04  
 ORIGINAL NUMBER OF ATOMS IN REGION  
 8.3567D 20 1.0427D 22 7.6089D 26 7.6089D 26 2.3831D 28 0.0 1.5058D 18 (REACTION FRACTION OF TOTAL HE-3)

9.3.3 Appendix III(c): First Year of Reactor Operation

DURING THE PERIOD FROM 1 1 74 TO 1 1 75 THE POWER LEVEL WAS 0.80 AND THE PURIFICATION PLANT EFFICIENCY WAS 1.00  
 THE HEAT EXCHANGER RATE OF 1.7354E-05 (SEC-1) AND THE ADSORPTION RATIO OF 25.2670 WERE BASED ON 30.00 PPM H2

CUMULATIVE TRITIUM INVENTORY IN CURIES AS OF		PERIOD END FOR PRODUCTION, AND DISTRIBUTION SINKS						BLOWDOWN+LOSS
SOURCE - ATOMS	TRITIUM PRODUCTION	BOUND IN SOLIDS	ADSORBED ON GR. SURFACE	IN PRIMARY COOLANT	LEAKAGE FROM PRIMARY	PURIF** PLANT	H2O IN STEAM GENERATOR	STEAM GENERATOR
CORE		FRACTION OF HE3(N,F) T RECOILS BOUND IN SOLIDS = 0.50389						
PURF	1.0000D 00	3119.852553	3088.653872	0.440224	0.017423	0.000167	21.598798	9.141111
HE-3	6.6187D 22	600.223413	302.447686	3.857124	0.152655	0.001596	206.401365	87.354595
PB-10	5.5437D 25	138.707856	137.320777	0.004685	0.000185	0.000008	0.971146	0.411044
SB-10	5.5437D 25	127.109075	125.837985	0.025841	0.001023	0.000007	0.874200	0.369964
LI-7	5.9957D 26	0.230863	0.228554	0.000033	0.000001	0.000000	0.001598	0.000676
C-12	9.2849D 30	0.078640	0.077853	0.000030	0.000001	0.000000	0.000530	0.000224
BE-9	1.2379D 24	0.0	0.0	0.0	0.0	0.0	0.0	0.0
LI-6	1.2539D 22	1302.767023	1285.739353	0.134474	0.005322	0.000070	9.055082	3.832429
REGION TOTALS		5288.969424	4944.306080	4.462412	0.176610	0.001947	238.902720	101.110045
REFLECTOR		FRACTION OF HE3(N,F) T RECOILS BOUND IN SOLIDS = 0.62290						
HE-3	4.3470D 21	39.421284	24.555456	0.192559	0.007621	0.000080	10.304155	4.360995
PB-10	2.6191D 24	0.006736	0.006668	0.000001	0.000000	0.000000	0.000047	0.000020
SB-10	2.6191D 24	0.000243	0.000241	0.000000	0.000000	0.000000	0.000002	0.000001
LI-7	1.4678D 24	0.001406	0.001392	0.000000	0.000000	0.000000	0.000010	0.000004
C-12	1.3568D 31	0.000000	0.000000	0.000000	0.000000	0.000000	0.000000	0.000000
BE-9	7.5374D 21	0.0	0.0	0.0	0.0	0.0	0.0	0.0
LI-6	8.5255D 22	134.208033	132.865953	0.018639	0.000738	0.000007	0.929338	0.393318
REGION TOTALS		173.637702	157.429710	0.211199	0.008359	0.000087	11.233551	4.754337
BORONATED REFLECTOR		FRACTION OF HE3(N,F) T RECOILS BOUND IN SOLIDS = 0.70304						
HE-3	9.9319D 19	0.891624	0.626851	0.003430	0.000136	0.000001	0.183525	0.077673
PB-10	1.2275D 28	0.037527	0.037151	0.000005	0.000000	0.000000	0.000260	0.000110
SB-10	1.2275D 28	0.000050	0.000049	0.000000	0.000000	0.000000	0.000000	0.000000
LI-7	6.1501D 25	0.000001	0.000001	0.000000	0.000000	0.000000	0.000000	0.000000
C-12	9.1366D 30	0.000000	0.000000	0.000000	0.000000	0.000000	0.000000	0.000000
BE-9	6.5700D 18	0.0	0.0	0.0	0.0	0.0	0.0	0.0
LI-6	5.0557D 22	3.518989	3.483800	0.000496	0.000020	0.000000	0.024362	0.010311
REGION TOTALS		4.448191	4.147853	0.003931	0.000156	0.000002	0.208148	0.088094
CONTROL RODS		FRACTION OF HE3(N,F) T RECOILS BOUND IN SOLIDS = 0.64005						
HE-3	1.2634D 18	0.011457	0.007333	0.000053	0.000002	0.000000	0.002859	0.001210
PB-10	7.4387D 26	169.046292	167.355829	0.023594	0.000934	0.000009	1.170494	0.495381
SB-10	7.4387D 26	1.029105	1.018814	0.000283	0.000011	0.000000	0.007024	0.002972
LI-7	1.7033D 25	0.001244	0.001232	0.000000	0.000000	0.000000	0.000009	0.000004
C-12	2.3831D 28	0.000000	0.000000	0.000000	0.000000	0.000000	0.000000	0.000000
BE-9	1.2356D 21	0.0	0.0	0.0	0.0	0.0	0.0	0.0
LI-6	8.3096D 20	0.221803	0.219585	0.000031	0.000001	0.000000	0.001536	0.000650
REGION TOTALS		170.309902	168.602793	0.023962	0.000948	0.000009	1.181921	0.500216
TOTAL CURIES		5637.365218	5274.486436	4.701503	0.186073	0.001945	251.526319	106.452692
AVERAGE ACTIVITY OF GIVEN REGION			8.1153D 00 MICROCURIES PER GRAM GR	7.2337D-03 MICROCURIES PER GRAM GR	1.1875D-05 MICROCURIES PER SCC H2		2.5628D-03 MICROCURIES PER GRAM H2O	

9.3.4 Appendix III(d): End of Sixth Year of Reactor Operation

DURING THE PERIOD FROM 1 1 79 TO 1 1 80 THE POWER LEVEL WAS 0.80 AND THE PURIFICATION PLANT EFFICIENCY WAS 1.00  
 THE HEAT EXCHANGER RATE OF 1.7354E-05 (SEC-1) AND THE ADSORPTION RATIO OF 25.2670 WERE BASED ON 30.00 PPM H2

CUMULATIVE TRITIUM INVENTORY IN CURIES AS OF PERIOD END FOR PRODUCTION, AND DISTRIBUTION SINKS  
 SOURCE - ATOMS TRITIUM BOUND IN ADSORBED ON IN PRIMARY LEAKAGE FROM PURIF\*H2O IN STEAM BLOWDOWN\*LOSS  
 PRODUCTION SOLIDS GR. SURFACE COOLANT PRIMARY PLANT GENERATOR STEAM GENERATOR

CORE FRACTION OF HE3(N,F) T RECOILS BOUND IN SOLIDS = 0.50389  
 PUEL 1.0000D 00 15334.840001 16371.490783 0.440224 0.017423 0.000885 114.448399 0.000958 48.441329  
 HE-3 2.8606D 22 2076.985829 1046.576233 1.566664 0.065962 0.005588 722.753376 0.003626 305.914380  
 PB-10 2.4050D 20 114.845511 113.697056 0.000000 0.000000 0.000006 0.806910 0.000000 0.341539  
 SB-10 2.4050D 20 105.244744 104.192297 0.000000 0.000000 0.000006 0.719455 0.000000 0.312987  
 LI-7 6.5498D 26 886.151007 877.289497 0.028361 0.001122 0.000048 6.205424 0.000062 2.626493  
 C-12 9.2849D 30 0.412082 0.407961 0.000030 0.000001 0.000000 0.002873 0.000000 0.001216  
 BE-9 7.4308D 24 3.122907 3.091678 0.000281 0.000011 0.000000 0.021736 0.000001 0.009200  
 LI-6 1.6178D 21 2214.587185 2192.441313 0.006369 0.000252 0.000120 15.555144 0.000014 6.581973

REGION TOTALS 21736.189266 20509.186817 2.141930 0.084772 0.006653 860.533317 0.004661 364.231116  
 TRITIUM PRODUCED IN REGION 2.173618926612284D 04 TRITIUM DISTRIBUTED IN REGION 2.173618926621225D 04

REFLECTOR FRACTION OF HE3(N,F) T RECOILS BOUND IN SOLIDS = 0.62290  
 HE-3 3.6548D 21 70.362910 43.828946 0.161890 0.006407 0.000143 18.524644 0.000352 7.840527  
 PB-10 2.2961D 24 0.012270 0.012148 0.000001 0.000000 0.000000 0.000086 0.000000 0.000036  
 SB-10 2.2961D 24 0.000443 0.000438 0.000000 0.000000 0.000000 0.000003 0.000000 0.000001  
 LI-7 1.7909D 24 0.003205 0.003173 0.000000 0.000000 0.000000 0.000022 0.000000 0.000009  
 C-12 1.3568D 31 0.000000 0.000000 0.000000 0.000000 0.000000 0.000000 0.000000 0.000000  
 BE-9 1.5075D 22 0.000000 0.000000 0.000000 0.000000 0.000000 0.000000 0.000000 0.000000  
 LI-6 8.2495D 22 256.697072 254.130101 0.018036 0.000714 0.000014 1.790393 0.000039 0.757776

REGION TOTALS 327.075901 297.974807 0.179927 0.007121 0.000157 20.315144 0.000391 8.598350  
 TRITIUM PRODUCED IN REGION 3.270759013602299D 02 TRITIUM DISTRIBUTED IN REGION 3.270759013602297D 02

POROSATED REFLECTOR FRACTION OF HE3(N,F) T RECOILS BOUND IN SOLIDS = 0.70304  
 HE-3 8.2663D 19 1.591456 1.118865 0.002883 0.000114 0.000003 0.329939 0.000006 0.139646  
 PB-10 1.2214D 28 0.072807 0.072079 0.000005 0.000000 0.000000 0.000508 0.000000 0.000215  
 SB-10 1.2214D 28 0.000096 0.000095 0.000000 0.000000 0.000000 0.000001 0.000000 0.000000  
 LI-7 1.2195D 26 0.000101 0.000100 0.000000 0.000000 0.000000 0.000001 0.000000 0.000000  
 C-12 9.3366D 30 0.000000 0.000000 0.000000 0.000000 0.000000 0.000000 0.000000 0.000000  
 BE-9 1.3140D 19 0.000000 0.000000 0.000000 0.000000 0.000000 0.000000 0.000000 0.000000  
 LI-6 6.0482D 22 6.840287 6.771884 0.000496 0.000020 0.000000 0.047698 0.000001 0.020188

REGION TOTALS 8.504746 7.963022 0.003384 0.000134 0.000003 0.378146 0.000007 0.160050  
 TRITIUM PRODUCED IN REGION 8.504746382759695D 00 TRITIUM DISTRIBUTED IN REGION 8.504746382759695D 00

CONTROL RODS FRACTION OF HE3(N,F) T RECOILS BOUND IN SOLIDS = 0.64005  
 HE-3 1.0622D 18 0.020450 0.013089 0.000045 0.000002 0.000000 0.005139 0.000000 0.002175  
 PB-10 7.2722D 26 325.021663 321.771446 0.023066 0.000913 0.000018 2.266770 0.000050 0.959400  
 SB-10 7.2722D 26 1.978636 1.958852 0.000276 0.000011 0.000000 0.013700 0.000001 0.005798  
 LI-7 3.3674D 25 2.033833 2.013495 0.000287 0.000011 0.000000 0.014080 0.000001 0.005959  
 C-12 2.3831D 28 0.000001 0.000001 0.000000 0.000000 0.000000 0.000000 0.000000 0.000000  
 BE-9 2.4712D 21 0.000001 0.000001 0.000000 0.000000 0.000000 0.000000 0.000000 0.000000  
 LI-6 8.2629D 20 0.430170 0.425868 0.000031 0.000001 0.000000 0.003000 0.000000 0.001270

REGION TOTALS 329.484755 326.182751 0.023705 0.000938 0.000018 2.302689 0.000052 0.974602  
 TRITIUM PRODUCED IN REGION 3.294847550721681D 02 TRITIUM DISTRIBUTED IN REGION 3.294847550721679D 02

TOTAL CURIES 22401.254669 21141.307398 2.348947 0.092965 0.006831 883.529300 0.005111 373.964117

AVERAGE ACTIVITY OF GIVEN REGION 3.2528D 01 3.6141D-03 5.9330D-06 1.2804D-03  
 MICROCURIES MICROCURIES MICROCURIES MICROCURIES  
 PER GRAM GR PER GRAM GR PER SCC HE PER GRAM H2O

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