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NUCLEAR MODULES OF ITER TOKAMAK SYSTEMS CODE*

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Abstract

Nuclear modules were developed to model various reactor components in the ITER systems code.[1] These modules include first wall, tritium breeding blanket (or shield), bulk shield, reactor vault, impurity control, and tritium system. The function of these modules is to define the performance parameters for each component as a function of the reactor operating conditions. Several design options and cost algorithms are included for each component. The first wall, blanket and shield modules calculate the beryllium zone thickness, the disruptions results, the nuclear responses in different components including the toroidal field coils. Tungsten shield/water coolant/steel structure and steel shield/water coolant are the shield options for the inboard and outboard sections of the reactor. Lithium nitrate dissolved in the water coolant with a variable beryllium zone thickness in the outboard section of the reactor provides the tritium breeding capability. The reactor vault module defines the thickness of the reactor wall and the roof based on the dose equivalent during operation including skyshine contribution. The impurity control module provides the design parameters for the divertor including plate design, heat load, erosion rate, tritium permeation through the plate material to the coolant, plasma contamination by sputtered impurities, and plate lifetime. Several materials: Be, C, V, Mo, and W can be used for the divertor plate to cover a range of plasma edge temperatures. The tritium module calculates tritium and deuterium flow rates for the reactor plant. The tritium inventory in the fuelers, neutral beams, vacuum pumps, impurity control, first wall, and blanket is calculated. Tritium requirements are provided for different operating conditions. The nuclear models are summarized in this paper including the different design options and key analyses of each module.

Introduction

Several modules were developed to model various ITER reactor components. These modules include the first wall, tritium breeding blanket (or shield), bulk shield, reactor vault, impurity control system, and tritium system. The function of these modules is to define the performance parameters for each component as a function of the reactor operating conditions. Several design options and cost algorithms are included for each component. These modules were integrated successfully in the ITER systems code.[1]

First Wall Module

This module calculates first wall design parameters including the heating rates, radiation damage parameters, neutron fluence, tritium inventory, tritium permeation rate to the first wall coolant, and the tile erosion rate due to plasma disruptions. The module distinguishes between the inboard and the outboard sections of the first wall. Both sections have a PCA steel structure and a water coolant except for a 1-cm beryllium tile for the inboard section only. The first wall is integrated with the blanket and shield. A multivariable data set generated by several design codes is used to calculate all the above parameters except the tritium inventory and the tritium permeation rates. The tritium parameters are calculated through a simplified algorithm. This module has three sections to provide neutronics results (energy deposition, neutron fluences, and radiation damage analyses), tritium parameters, and disruption erosion rates.

Energy Deposition, Neutron Fluences, and Radiation Damage Analyses

The surface heat load and the nuclear heating were used to calculate the energy deposition rate in the first wall. The surface heat load is obtained from the impurity control module. The nuclear heating is calculated for both sections of the first wall using a toroidal cylindrical geometry. The one-dimensional discrete ordinates code ONEDANT[2] was used to perform the transport calculations with a P_5 approximation for the scattering cross sections and an S_8 angular quadrature set. A 67 coupled group nuclear data library (46 neutron and 21 gamma) based on ENDF/B-IV with corrected lithium-7 cross sections was employed for these calculations. VITAMEN-C[3] and MACKLIB-IV[4] libraries were used to obtain this library. In the geometrical model, it is assumed that the tile material does not have an active coolant and its energy is transmitted to the inboard first wall. The neutron fluence, the atomic displacement, and the helium and the hydrogen production rates are calculated based on the number of full power years of operation used in the analyses. The first wall coolant is integrated with the main coolant of the blanket/shield.

Plasma Disruptions

The energy deposited on part of the first wall during the plasma disruption results in melting and vaporization of wall materials.[5-6] An accurate calculation for the amount of vaporization losses and melt layer thickness resulting from the disruption is a very important parameter to define the first wall lifetime.

The time dependent heat conduction equation is solved subject to several boundary conditions. These boundary conditions include the surface heat flux and radiation to the surrounding surfaces. Also, possible material phase changes and the vaporization energy of target materials are considered explicitly in the solution. This system of equations is subject to two moving boundaries.[7-10] One moving boundary is the melt-solid interface because the surface heat flux may result in melting the surface of the exposed material. Another moving boundary is the receding surface as a result of evaporation of the wall material due to the continuous heating of the melted surface.

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The general time-dependent one-dimensional heat conduction is given by:[10]

$$\rho_s(T)\, C_{p_s}(T) \frac{\partial T_s}{\partial t} = \frac{\partial}{\partial x}\left[K_s(T) \frac{\partial T_s}{\partial x}\right] \qquad 0 < x < L,\ t > 0 \quad (1)$$

$$T(x,o) = f(x) \qquad 0 < x < L,\ t = 0 \quad (2)$$

where f(x) is the initial temperature distribution function.

The solution process requires partitioning of the incident energy into conduction, melting, evaporation, and radiation. To account for a phase change when the temperature of a node reaches the melting temperature of the material, this node temperature is fixed until all the heat of fusion is absorbed. Then the temperature of this node is allowed to change. During the phase change the material properties of the node are given by a combined value from both solid and liquid properties according to the ratio of the transformation at this time step.

The velocity of the receding surface is a highly non-linear function of temperature. A review of the model used to calculate the evaporation losses is given in Ref. 6. A parametric study was performed to generate a disruption data base as a function of the disruption energy and the deposition time for the first wall analysis. The first wall module interpolates from this data base to calculate the first wall erosion due to vaporization only, assuming no vapor protection.

Hydrogen Isotopes Permeation and Inventory in Fusion Reactor Components

Tritium permeation into the coolant and the tritium inventory in the first wall and the divertor are key parameters in reactor design scoping studies because of their large impact on both economic and safety aspects of the fusion device. To compute the permeation and inventory of hydrogen isotopes in fusion reactor components, a steady state is assumed. The permeation model assumes a hydrogen isotope atom implantation flux J_i at a depth δ in the surface which is less than the surface thickness d. This implantation depth depends primarily on the energy of D-T particles which is usually in the 100's of eV range.

It is assumed that gas molecules leave either wall surface (front or back) by recombination-limited desorption according to

$$J = 2\, K_r\, C^2 \quad , \qquad (3)$$

where C is the dissolved hydrogen isotope concentration near the surface, and K_r is the recombination coefficient given by:[11]

$$K_r = \frac{4\,\alpha}{K_{so}^2 \sqrt{\pi K\, MT}} \exp\left(\frac{2\,E_s - E_x}{KT}\right) \quad , \qquad (4)$$

where

α = sticking coefficient (= 1 for clean surfaces)

K_{so} = pre-exponential Sievert's solubility constant

K = Boltzmann constant

M = mass of the molecule formed by recombination

T = absolute temperature

E_s = energy of solution for hydrogen in metal

E_x = $E_s + E_d$ (E_d = diffusion energy)

and

$$E_x = \begin{cases} E_s + E_d & \text{if } E_s + E_d > 0 \\ 0 & \text{otherwise.} \end{cases} \qquad (5)$$

The governing equations for the surface currents can then be written as:

$$J_1 = 2\, K_{r_1}\, C_1^2 = \frac{D(C_m - C_1)}{\delta} \quad , \qquad (6)$$

$$J_2 = 2\, K_{r_2}\, C_2^2 = \frac{D(C_m - C_2)}{(d - \delta)} \quad , \qquad (7)$$

$$J_i = J_1 + J_2 \qquad (8)$$

where the subscripts 1 and 2 refer to the front (plasma) and back (coolant) surface, respectively. The value C_m refers to the maximum concentration obtained at the implantation depth δ from the surface. The total first wall material thickness is assumed to be d and J_i is the implantation flux. The surface current J_2 then represents the net flux going to the coolant, i.e., the permeation flux. The above equations are solved numerically using efficient iteration techniques to yield accurate solutions.[12] The steady-state tritium inventory is calculated from the established concentrations at both surfaces along with the maximum concentration at depth δ. The physical properties of the candidate materials are stored in a separate subroutine in the code.

Blanket

The tritium breeding function of the blanket is integrated in the bulk shield by using an aqueous solution of a lithium compound in the water coolant of the shield. The other blanket function of producing recoverable heat in suitable conditions for power generation is accomplished only in the blanket test module of the device. The first wall/blanket/shield is cooled by room temperature water (20-40°C) at low pressure. The coolant flow rate is adjusted to achieve a 20°C increase in the coolant temperature. The aqueous solution has 16 g $LiNO_3$/100cc based on neutronics considerations to maximize the tritium breeding ratio. A beryllium zone behind the first wall in the outboard section of the reactor is also employed as a neutron multiplier to enhance the tritium breeding ratio.[13] The neutron transport code and data library described in the first wall section are employed for the calculations. It should be noted that the results of the first wall/blanket/shield modules assume a 100% coverage of the plasma by the first wall. An adjustment should be considered to account for different penetrations in the first wall due to plasma heating options and impurity control systems.

The first wall/blanket/shield parameters of this version of the code are defined to include the TIBER design as given in Table 1. A parametric study was performed to generate a three-dimensional data set to provide the blanket performance parameters as a

TABLE 1

First Wall/Blanket/Shield Parameters

Zone Description	Zone Thickness (cm)	Zone Composition (Vol. %)
Inboard blanket/shield	(x)[a]	10% PCA steel 20% H_2O (16 g $LiNO_3$/100cc)[b] 70% W
Inboard first wall	1	50% PCA steel 50% H_2O (16 g $LiNO_3$/100cc)[b]
Inboard tile	1	100% Be
Inboard scrapeoff layer	6	--
Plasma	86	--
Outboard scrapeoff layer	11	--
Outboard first wall	1	50% PCA steel 50% H_2O (16 g $LiNO_3$/100cc)[b]
Neutron Multiplier	(y)[a]	5% PCA steel 35% H_2O (16 g $LiNo_3$/100cc)[b] 60% Be
Outboard blanket/shield	(z)[a]	65% PCA steel 35% H_2O (16 g $LiNO_3$/100cc)[b]

[a] x, y and z are variables
[b] a 90% lithium -6 enrichment is used

function of the zone thickness of the inboard blanket/shield, the beryllium neutron multiplier, and the outboard blanket/shield. This module calculates first the beryllium zone thickness to achieve the required tritium breeding ratio by interpolation in the data set. A normalized bicubic spline algorithm is used to perform the interpolation. Also, the nuclear heating and the coolant mass flow rate for both sections of the reactor are calculated based on a 20°C increase in the coolant temperature.

If the required tritium breeding ratio is zero, the blanket module eliminates the beryllium multiplier zone and the $LiNO_3$ salt from the water coolant by using a different data set.

Bulk Shield

The nuclear responses in the superconducting toroidal field coils and the dose equivalent in the reactor vault one day after shutdown are calculated in this module based on the input shield thicknesses. These responses define the allowable DT neutron wall loading for a specific operating scenario, the total DT neutron fluence, or the required shield thicknesses to satisfy the design goals.

Irradiation of superconducting coils tends to lower their performance.[14,15] For superconductor materials, neutron irradiation reduces the critical current density J_c and the critical temperature T_c. For Nb_3Sn material it has been shown that J_c generally increases, reaching a maximum, and then decreases as the fast neutron fluence increases.[16] Irradiation experiments[15,17] at 6 K with a fast neutron fluence of 4×10^{18} n/cm^2 show the maximum value and the increased rate of J_c increases with the magnetic field. At a magnetic field of 5T, the 6 K experiment resulted in a 16% increase in J_c at a fast neutron fluence of 2×10^{18} n/cm^2 without reaching a peak.

Irradiation experiments at higher temperatures (~350°K) show the same behavior for J_c. The J_c peak, however, was lower than the corresponding value at lower temperature for the same magnetic field. For example, the 350 K irradiation only gave an 8% increase compared to the 16% mentioned before at 6 K. Other experiments at 400 K and a 10-T field[18] resulted in a 90% increase for J_c at a fast neutron fluence of 4.4×10^{18} n/cm^2 and dropped to the original value of J_{c2} as the fast neutron fluence increased to 10^{19} n/cm^2. Based on these experimental results[15,16,18] and a comparison between the room temperature and the 6 K irradiation results, and on a maximum field of ~11 T, it is possible to achieve a fast neutron fluence above 10^{19} n/cm^2 for Nb_3Sn without a decrease in the critical current density. At this fluence the T_c value is ~0.9 the original value.[18] The module calculates the fast neutron fluence in the Nb_3Sn superconductor to help define the allowable J_c value for the toroidal field coils.

The superconducting coil is designed to remove the generated I^2R heat so that the normal region does not propagate. The resistance R of the copper stabilizer is the key parameter for this process. The total resistivity ρ of the copper stabilizer can be described as the sum of three components: the initial resistivity, ρ_0; the magnetoresistivity, ρ_μ; and the irradiation-induced resistivity, ρ_{irr}. Magnetoresistance is a function of ρ_0, ρ_{irr}, and the magnetic field which complicates the evaluation of ρ. Few experimental studies[20-24] have focused on the change of the copper resistivity as a function of magnetic field, neutron fluence, and number of cycles of alternate neutron irradiation at 4 K and annealing at 300 K. The change in the copper resistivity can be accommodated by using more copper stabilizer, which increases the thickness of the coils, can be partially annealed out by warming the coils, or can be reduced by improving the shielding performance through an increase in shielding thickness or through the use of better materials. The module calculates the maximum induced resistivity in the copper stabilizer for the toroidal field magnet design module.

The most sensitive component in the coils is the insulator material because the radiation damage is irreversible and this damage limits the operating life of the coils. The properties of interest for the coil designs are the electrical resistivity, dielectric strength, mechanical strength, and thermal insulation. Experimental results[25-28] from neutron irradiation at 5 K suggest that polyimides can withstand a radiation dose of 10^{10} rad and retain high resistivity and mechanical strength. Glass cloth reinforced epoxy-type G10-CR or G11-CR shows a serious degradation at 2×10^9 rad. The module calculates the maximum insulator dose based on the DT full power years of operation.

The nuclear energy deposited in the coils impacts the refrigeration power required since ~500W of electrical power is consumed to remove 1 W from the superconducting coils at 4 K. This low removal efficiency calls for minimizing the nuclear energy deposited in the superconducting coils. The module calculates the total nuclear heating in the

superconducting coils and the maximum value for the magnet design module. For personnel protection, the module calculates the dose equivalent in the reactor vault one day after shutdown based on the neutron flux at the outer shield surface[27].

Vault Shield

The shield system of a fusion reactor consists of two parts, the different materials around the vacuum chamber and the concrete walls of the reactor vault. The first part of the shield has to reduce the neutron and photon leakage intensities at the outer shield surface. This reduction ensures several design criteria: a) the different reactor components are protected from radiation damage and excessive nuclear heating, b) the neutron reaction rates in the reactor components which produce undesirable radioactive isotopes are reduced, and c) the workers are permitted in the reactor vault one day after shutdown. The second part of the shield has to protect the workers and the public from radiation exposure during the reactor operation. The latter part of the shield is calculated in this module.

The International Commission on Radiological Protection Recommendations and the U.S. Federal Regulations limit the occupational exposure to 5 rem/y with a maximum of 3 rem/quarter. The occupation exposure based on regular working hours is 2.5 mrem/h. However, the current practice in the nuclear industry, the exposure policy of the U.S. Department of Energy (DOE), and the National Laboratories' guidelines are to reduce radiation exposure as low as reasonably achievable. Specifically, DOE order 5980.1, Chapter XI states: "Exposure rates in work areas should be reduced as low as reasonably achievable by proper facility design layout. Design efforts to consider are: occupancy time, source terms, spacing, processes, equipment, and shielding. On-site personnel exposure level less than one-fifth of the permissible dose equivalent limits, prescribed in this chapter should be used as a design objective". This guideline limits on-site workers to <1 rem/y (0.5 mrem/h).

The calculation of this module is concerned with the total dose equivalent outside the reactor building during operation to satisfy the 0.5 mrem/h design criterion. A parametric study was performed to define the dose equivalent outside the reactor building as a function of the roof thickness including the contribution from neutrons and photons scattered back by collision with air nuclei (skyshine). A three-dimensional model was employed to generate a data set for this module.[28] The MCNP general Monte Carlo code[29] for neutron and photon transport was used to perform all the calculations. Variance reduction schemes were employed for the calculations. The energy distribution of the neutron source was used explicitly in the calculations with a nuclear library based on ENDF/B-V data.

The reactor wall thickness and the reactor roof thickness are calculated in this module based on the 0.5 mrem/h outside the reactor building during operation including the skyshine contribution by interpolation from the stored data set.

Impurity Control Module

The impurity control module models plasma/material interactions related to a divertor. The module provides an estimate of the following: divertor plate design and heat load, divertor plate lifetime due to erosion by sputtering and disruptions, sputtered impurity content in the core plasma, and tritium permeation through the divertor plate into the coolant. Inputs to the module are the reactor geometry, frequency of disruptions, plasma edge temperature, and plasma heating powers.

The code is based on the extensive analysis of impurity control issues for tokamak fusion reactors, particularly for the INTOR design.[30] Due to the complexity of most impurity control analysis, the intent of the module is to provide approximate parameter estimates for reactor design scoping purposes, and to identify general trends; for example how erosion might vary with plasma edge temperature.

Sputtering erosion of the divertor plate is computed using a simplified version of the REDEP erosion/redeposition code[31] applied to the divertor plate center. The divertor model is a single or double null type operated in the high or medium recycling regime. Erosion is computed for plate materials of beryllium, carbon, vanadium, molybdenum, and tungsten, with plasma edge temperature (near the plate center) in the range of from 10 to 150 eV. Sputtering due to hydrogen, helium, oxygen, and self-sputtering is computed. Self-sputtering, and associated redeposition is modeled as arising from ionization and transport in the scrapeoff zone and plasma. The general features of sputtered impurity transport identified in Ref. 32 for a high recycling divertor are used for the module. The disruption model is based on the disruption results of the TIBER design[33] using disruption parameters of 1 ms energy deposition time and 5 MJ/m^2 thermal load. The main assumption is that the melt layer is not lost; erosion is due to vaporization only.

The sputtering calculation predicts both the gross and net (sputtering minus redeposition) erosion rates. Because this calculation depends on uncertain estimates of redeposited material properties (e.g., adhesion), the results must be cautiously applied whenever high gross erosion rates are predicted. This is particularly true for carbon surfaces.

The divertor plate lifetime calculation uses a design thickness in the range of 0.5 to 2.0 cm, depending on the plate material, and on the disruption and net sputtering erosion rates. Plate design information (e.g. area) is scaled from the INTOR and TIBER designs for single and double null divertors respectively, and for different major radii.

The core plasma impurity content due to sputtering depends on the net sputtering rates, transport through the scrapeoff zone and transport in the core plasma. Data for the latter is taken from 1-D transport calculations[34] for a tokamak fusion reactor, under the assumption of non-neoclassical impurity particle transport. In this case, the core impurity content is approximately equal to one fourth of the effective edge DT sputtering coefficient (ratio of the impurity current entering the plasma edge to the DT current leaving the edge).

The heat and particle loads to the divertor and adjacent first wall (necessary estimates for the tritium permeation calculations) are based primarily on scaling of data from the INTOR analysis. Charge exchange flux to the first wall is modeled as occurring in a small region adjacent to the divertor about equal to the divertor area. The total charge exchange current is equal to half of the ion current to the divertor. Energy of the charge exchange neutrals is scaled from the edge temperature, based on INTOR analysis. The tritium permeation calculations

use the particle fluxes as inputs in a manner similar to that which was discussed for the first wall tritium permeation module.

Tritium Module

A magnetic fusion reactor fueled with tritium and deuterium has four main tritium processing systems - the plasma processing system, the blanket processing system, the water processing system and the atmospheric processing system. A computer module which provides information on costs, tritium inventory, power requirements and sizes for these systems was developed. The tritium module has a main section and two subroutines. The two subroutines provide detailed information on the plasma processing system and the water processing system.

Main Section

The main section of the tritium module has six functions. First, it determines the tritium and deuterium feedrates for the plasma processing units as was done previously.[35] Second, it determines the tritium inventory and capital cost for blanket options other than the aqueous lithium salt. Third, it determines the tritium inventory in the tritium processing components and in all major units in the fusion plant; this includes tritium in storage or in high heat components. Fourth, it determines the tritium supply needed at startup and for each year. Fifth, it determines the total capital cost, operating cost, size and power requirements for all tritium systems, using information supplied by the two subroutines. Sixth, it assesses the tritium loss to the environment as a function of cleanup time and base tritium concentration.

The gas feed rate is the sum of the plasma exhaust, the fueler exhaust and the blanket exhaust. The magnitude of the plasma exhaust depends on the plasma fractional burn. The fueler options are: a neutral beam system, a pellet fueler system, a combination of these two systems, or an alternate system.

The blanket options are: flibe, lithium-lead, solid oxide, lithium, lithium salt/water. Generic algorithms are used to determine the tritium inventory in the breeding blanket, the tritium inventory in the blanket processing system, and the capital cost of the blanket processing system. The blanket inventory is a function of blanket mass.

The tritium inventory for plasma processing units or water processing units is calculated in the two subroutines. Inventories in other areas are calculated in the main section of the module.

The tritium supply needs are defined as a function of breeding ratio, processing losses during a year and decay losses for the onsite tritium inventory.

The capital costs of the plasma processing system and the water processing system are determined in the two subroutines. A multiplicative factor of four is used with specific plasma processing support equipment; monitors, inventory control instrumentation, secondary containment, etc, to account for needed equipment in the tritium area, in the hot cell area, in the reactor hall area, and in the neutral beam area or in the heat exchanger area. In addition, additional storage beds are needed to handle the reserve storage of a 24 hour day of input to the plasma. A separate algorithm[35] is used to determine the capacity of the needed atmospheric processing systems in the four areas.

Tritium losses in the four areas are assessed as a function of cleanup time, inleakage rate, amount of tritium release, and tritium base concentration.

The input variables are: burntime, dwell time, ramp time, fractional burn-up, fueler options, fuel cleanup option, cost of tritium, type of blanket, blanket mass, water processing option, cleanup time for the four areas, inleakage rate for the four areas, tritium release for the four areas, tritium base concentration in the four areas.

The major assumptions are: 1) the reactor runs continuously or for one burn time/dwell time/ramp time combination; 2) the neutral beam efficiency is 0.07 to 0.3; 3) the pellet fuelers use fuel at levels 1, 2 or 3 times the rate needed to maintain the plasma; 4) the pump regeneration times are < 2 hr; 5) the cost of tritium is $< \$3$ US per curie; 6) the net breeding ratio is 0-1.5; and 7) a day is a 24 hour operational day.

The output is total tritium and total deuterium flow rates, tritium inventory, capital costs, operating costs, size of equipment, power requirements, and tritium cleanup for unplanned releases.

Plasma Processing System

The first subroutine (TSTA), which is based on the operating experience at the Tritium Systems Test Assembly, provides information on units in the plasma processing system. The processing units included are a palladium diffuser or a mole sieve unit for fuel cleanup, a cryogenic distillation unit, storage beds, gas analysis instrumentation, monitors, secondary containment units, a gas effluent unit, emergency air cleanup units, data acquisition units, a solid waste unit, and inventory control unit.

The TSTA subroutine calculates the tritium inventory, power requirements, size and capital cost (1986 US$) of a given plasma processing unit as a power function of the tritium and deuterium feedrate. Based on the experience of the chemical process industry, the power function used for large units is 0.6[36] and 0.3 to 0.5 for very small installations or for processes employing extreme conditions of temperature or pressure.[37] For certain fixed costs, the power used is 0.

The original capital and installation costs of the subsystems at TSTA have been published[38]; these costs are summarized in Table 2. For the gas analysis system, the cost was increased 0.5M$ to include a mass spectrometer. For the gas effluent system, a recombiner for 0.12M$ was added.

The tritium inventory in each component at TSTA is given in Table 3.

The input variables to the TSTA subroutine are the mass flowrate of tritium and deuterium, reactor hall volume and time to clean the reactor hall.

The major assumptions made are: 1) the emergency room cleanup (ERC) unit is sized for the reactor hall; 2) flow through the ERC is greater than the inleakage rate; 3) cleanup time with the ERC is less than 5 days; 4) the ERC decontamination factor is greater than 10,000; 5) the target concentration after decontamination is 20 microcuries/m^3; 6) the volume

Table 2

Summary of TSTA Plasma Processing Units' Capital Costs

Subsystem	Cost,K$ (Cap.)	Cost,K$ (Inst.)	Year of Cap. Exp.	Design Variable	Power
Transfer Pumps	111	112	77	F	0.3
Fuel Cleanup	1000	70	80	F	0.3
Cryo. Distil.	1237	63	78	F	0.3
Storage Beds	60	10	81	T	0.6
Gas Analysis	469	26	79	-	0.0
Tritium Moni.	193	33	78-82	F	0.3
Sec. Contain.	182	30	78-82	F	0.3
Gas Effl. Detrit.	443	60	80-81	F	0.6
Emerg. Cleanup	382	357	79-80	Complex, see program	
Data Adq/Control	1379	531	79-81	-	0.0
Unit. Power	95	44	80	Complex, see program	
Emerg. Gen.	100	168	80	Complex, see program	
Solid Waste Dis.	23	0	80	F	0.3
Inv. Control	25	13	--	-	0.0

Note: F - Flowrate (2.08×10^{-5} kg DT/sec); T - Tritium inventory at TSTA (0.130 kg); V - Test cell volume at TSTA (3000 m^3); D - Room decontamination time (86400 sec - 1 day).

Table 3

Summary of TSTA Tritium Inventory Data

Subsystem	Tritium Inventory	Design Variable	Power
Isotope Sep.	100	F	0.6
Fuel Cleanup Diffuser	3	F	0.3
Mol. Sieve	30	F	0.3
Gas Effl. Detrit.	2	F	0.3
Other	1	F	0.3

Note: F - Flowrate (2.08×10^{-5} kg DT/Sec); A diffuser is the only option if turbomolecular pumps are used.

needed for the plasma processing equipment is an independent parameter; 7) power functions are useful algorithms used for modeling the plasma processing units; and 8) a storage bed holds 100 g of tritium.

Water Processing System

The second subroutine (TWCS) is based on the operating experience at Ontario Hydro. It provides tritium inventory, capital cost, size, and power requirements for several options used to process water or an aqueous lithium salt breeder.

The two main processing options for extracting tritium from water are: vapor phase catalytic exchange coupled with cryogenic distillation (VPCE/CD) and direct electrolysis coupled with cryogenic distillation (DE/CD). Other processing options available for specific purposes are: water distillation to pre-enrich the tritiated water; flashing to separate the lithium salt from the water; ion exchange to remove neutron activation products from the water (this last option has not yet been added to the subroutine).

System cost, size, and power consumption are calculated as a function of feed concentration and flowrate. Cost correlations are based on published data.[38-39] The costs are calculated by upgrading known costs to 1987 Canadian $ and then converting to 1987 US$.

The input variables to the Ontario Hydro subroutine include: tritiated water feed, tritium concentration in the feed, pre-enrichment choice, front end choice (VPCE or DE), Li salt concentration as fraction of solubility limit, and mole fraction of light water (1 or 0).

The major assumptions made are: 1) water distillation is only economic for light water cleanup; 2) water is either light water or heavy water (no intermediate mixtures); 3) separate correlations are used for light and heavy water; 4) lithium salt solutions are flashed to leave the lithium concentration in the remaining liquid at 90% of the lithium salt solubility limit; 5) in VPCE, 5 stages are used to achieve a detritiation factor of 10; 6) in DE, 25 kA electrolytic cell modules are used; 7) in an electrolysis cell, the tritium concentration in the electrolyte is 12 times higher than the feedwater concentration for H/T and 2 times higher for D/T; 8) above 70 Ci/kg, double containment of the electrolysis cells is recommended but not included in the cost correlation; 9) the largest cost component in CD is the first column and its cost is correlated to the cryogenic hydrogen refrigeration requirement; 10) the CD contains a catalytic equilibrator to break up HT and DT; and 11) the water concentration is between 0.01 and 34 Ci/L.

Conclusions

Nuclear systems modules have been developed for a general ITER systems code developed by the FEDC. The available design options at this time include TIBER-II design. These modules have been integrated into an operating systems code.[1]

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