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FINAL REPORT

CODE DEVELOPMENT INCORPORATING ENVIRONMENTAL, SAFETY, AND
ECONOMIC ASPECTS OF FUSION REACTORS (FY 92-94)

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1. OVERVIEW

This is the Final Report for a three-year (FY 92-94) study of the Environmental, Safety, and Economic (ESE) aspects of fusion energy systems, emphasizing development of computerized approaches suitable for incorporation as modules in fusion system design codes.

1.1 PROGRESS SUMMARY

This reporting period marks the culmination of key items on the initial agenda of our program from its inception in FY 89.

First, as is reported in Section 2, we now have operating a simplified but complete environment and safety evaluation code, BESAFE. The code covers a comprehensive fusion system including the plasma, engineering, environmental and safety aspects, mostly by simplified computational schemes and installed databases. Specific applications to the assessment of radiological hazards for HT-9 first walls and tungsten divertors during the loss-of-coolant accidents have been performed.

The first tests of BESAFE as a module of the SUPERCODE, a design optimization systems code at LLNL, are reported in Section 3. These preliminary tests, applied to power-reactor parameters, are thus far too limited to draw any conclusions. However, they do serve to illustrate the power of on-line evaluation of environmental and safety characteristics along with engineering and physics constraints.

Secondly, as reported in Section 4, we have maintained a strong effort in developing fast calculational schemes for activation inventory evaluation. We have demonstrated the feasibility of an approximate scaling equation approach by data regression of a numerical neutronics database from the parametric scoping of various fusion reactor design parameters. We have also made much progress in expediting the computational speed of existing neutron transport and activation codes to evaluate radioactive inventories with sufficiently accuracy for environmental and safety calculations. Adaptation of these codes to BESAFE is expected in the near future.

In addition to these major accomplishments, considerable progress has been made on research on specific topics as follows.

A tritium modeling code TRIDYN was developed in collaboration with the TSTA group at LANL and the Fusion Nuclear Technology group at UCLA. This work contributed to part of a Ph.D. thesis for one of our students completed in March 1994.

A simplified algorithm has been derived to calculate the transient temperature profiles in the blanket during accidents. The scheme solves iteratively a system of non-linear ordinary differential equations describing about 10 regions of the blanket by preserving energy balance.

We have studied the physics and engineering aspects of divertor modeling for safety applications. Edge and divertor plasmas have been modeled with the UEDGE code to benchmark DIII-D data. The engineering and safety performance of silicon-carbide ceramic divertors has been investigated.

Several modifications in the automation and characterization of environmental and safety indices have been made. We have applied this work to the environmental and safety comparisons of stainless steel with alternative structural materials for fusion reactors.

A methodology in decision analysis utilizing influence and decision diagrams has been developed to model fusion reactor design problems. The methodology was applied to study the rationales and options of an optimal design of an experimental facility based, under uncertainty, on the attributes of technical performance, cost, safety, and time schedule.

We have been participating in the International Energy Agency Cooperative Program on the Environmental, Safety, and Economic Aspects of Fusion Power in the area of Fusion Safety Methodology. An invited paper on this collaborative work has been presented at the Eleventh Topical Meeting on the Technology of Fusion Energy in June, 1994.

Most of our work during this funding period has been reported in 26 publications including theses, journal publications, conference papers, and technical reports, as listed in Section 11.

1.2 PROGRAM INTERACTIONS

In addition to our involvement with the IEA collaboration on Development of Fusion Safety Methodology described in Section 10, our group continues to participate with others in the fusion environment and safety community.

In particular, during the past year, we have begun to interact more closely with the U.S. ITER safety team at INEL. While applying the empirical scaling equations for oxidation-driven volatility derived at INEL based on experimental data for our safety analysis, we discovered some uncertainties regarding initial responses and geometrical factors. As a result of our interactions, INEL has proceeded to upgrade their experiments to investigate these issues. Presently, we are carrying out a detailed comparison of our biological dose codes, FUSEDOSSE I and FUSEDOSSE II, with the FUSECRAC and MACCS codes in current use at INEL in order to discover differences and to attempt to remedy deficiencies.

The TRIDYN tritium modeling code was developed in collaboration with the TSTA group at LANL and the Fusion Nuclear Technology group at UCLA. This code incorporates simplified but accurate physicochemical models for dynamic simulation of fusion fuel-cycle components, with benchmarking performed against experimental data from TSTA.

Considerable progress was made by one of our students in benchmarking the UEDGE scrape-off-layer/gaseous divertor code against DIII-D experimental data. This project, which continues in FY 95, is expected to lead to a simplified but reliable code model of divertors in normal operation and during shutdown for safety applications.

In collaboration with LLNL, we have started to implement environmental and safety features and calculational capabilities in SUPERCODE. This work will ultimately provide a tool for an integrated reactor design optimization including the environmental and safety criteria.

We have regularly attended the ARIES, PULSAR, and STARLITE reactor design project meetings. Through these interactions, we have gained much experience in understanding the safety and environmental impacts on fusion reactor design from the perspective of reactor designers. We have also participated and contributed papers at international safety meetings, including the IAEA Technical Committee Meeting on the Development of Fusion Safety in June, 1993 and the IEA Workshop on Fusion Safety System Study Methodology in February, 1994.

On a wider front, we have kept abreast of the broader issues of the fusion program through Professor Holdren's membership on the Fusion Energy Advisory Committee.

During the past year, we have also interacted in the establishment on the Berkeley campus of a new Center for Nuclear and Toxic Waste Management. This Center, initiated by the Nuclear Engineering Department, now includes senior faculty and students from Engineering, Political Science, Public Policy and the Energy and Resources Group. Though initially focussed on high-level nuclear waste from fission reactors, the interdisciplinary approach of the Center is also expected to benefit our fusion environmental and safety research. In the near-term, the Center activities will be directly beneficial to an ongoing Ph.D. thesis project, under auspices of our group, to evaluate nuclear waste disposal from ITER in the context of specific host countries bidding to be the ITER site.

2. BESAFE--INTEGRATED ENVIRONMENTAL AND SAFETY SYSTEMS CODE

2.1 INTRODUCTION

Conventionally, environmental and safety (E&S) assessments are performed only for after-the-fact, point-design studies of specific, completed reactor designs. Our project goal is to develop computationally fast systems-code modules to carry out E&S analyses in parallel with other plasma and engineering analyses now incorporated in systems codes. Only then will E&S factors have a direct and interactive impact on fusion reactor design at the systems level.

As a result, we have developed a fast-running computer code, called the Berkeley Environmental and Safety Algorithm for Fusion Energy (BESAFE) [2.1-2.3], for performing E&S assessments of tokamak fusion reactor designs. The BESAFE code has many simplified calculational schemes to speed up the computational time in order to permit iterations on a timescale compatible with fusion reactor systems studies, while preserving important safety features of the reactor design, as well as maintaining the systems-level accuracy. The algorithms focus on a comprehensive overall fusion reactor system without emphasizing the component details.

The structure of BESAFE is highly modularized. The modularization is required for easy coupling and decoupling between the various submodules and the other modules of reactor design codes, as well as providing flexibility in modifying and upgrading the submodules.

The following list of components of BESAFE available for use at this writing illustrates our approach to providing a flexible systems code suitable for continuous upgrading. The code now consists of:

- (a) a simple plasma model including transient behavior (i.e. burn control and shutdown) specified by appropriate time constants provided as input parameters;
- (b) radioactive inventories provided by off-line calculations with the TART [2.4] and FORIG [2.5] codes, plus some on-line table-look-up and scaling data;
- (c) tritium inventories provided off-line by the TRIDYN [2.6] code (discussed below), plus an on-line simplified model that will use residence times provided by TRIDYN;
- (d) the INEL streamlining model of tritium permeation in the plasma-facing components (PFC) [2.7];
- (e) simple energy source calculations including plasma, magnetic and chemical;

- (f) a 1-D finite difference scheme to calculate PFC thermal transients including heat sources from the plasma, neutrons and decay heat;
- (g) mobilization of activation products applying the INEL scaling equations [2.8] based on experimental data for oxidation volatility;
- (h) on-line dose dispersion calculations for selected E&S indices at worst weather conditions, using the FUSEDISE [2.9] code; plus off-line calculations for other dispersion scenarios;
- (i) evaluation of life-cycle waste volume and associated waste indices.

A brief description of the modeling is presented in Section 2.2.

The code provides output including the evaluation of some characteristic radiological hazards of a given reactor design. The major results are translated to the various E&S indices, which contain condensed information regarding the E&S performance of the reactors. The continuous nature of these indices is highly desirable for systems analysis. Other figures of merit relevant to E&S assessment such as maximum temperature rise during accident are also investigated.

The BESAFE code is designed to be a stand-alone code, as it has been operated for the last year, or to be implemented as a module of a tokamak design code such as SUPERCORE [2.10], into which BESAFE has been installed recently. When operated as one of the modules in SUPERCORE, BESAFE can provide E&S results for the optimization process. Section 3 of this report is devoted to discussion of BESAFE as it operates within SUPERCORE.

2.2 MODELING

2.2.1 ACTIVATION INVENTORIES

We have three levels of evaluation of activation-induced radioactive inventories. The first is by full-blown neutronics and activation codes, TART [2.4] and FORIG [2.5], respectively. The long computational time for this approach is undesirable for systems analysis, even though it produces very accurate results. However, this approach provides a good source for benchmarking other simplified schemes.

The second approach is to modify and speed-up an existing discrete-ordinate neutron transport code ANISN [2.11] coupling with an activation code DKR [2.12]. The streamlining enables much faster calculations with potential for direct application to systems analysis.

The third level is an innovative scheme of using scaling equations [2.13] formulated by data regression of a numerical neutronics and activation database from parametric scoping runs by full-blown codes. This approach provides fast evaluation of activation inventories appropriate for repeated iterations, relative comparisons, and approximate assessments.

More detailed descriptions on these schemes will be discussed in Section 4.

2.2.2 TRITIUM FUEL CYCLE

One of the main purposes of studying the tritium fuel cycle in this work is to evaluate the tritium inventory in the reactor system. Our work involves two levels of modeling of the tritium fuel cycle as follows.

The first model is a stand-alone dynamic simulation code [2.6]. The model utilizes physicochemical equations of mass and energy balance and process kinetics. This results in an intermediate level of complexity, with all fuel-cycle units integrated dynamically in a consistent manner. The code requires a moderately long running time to process about hundreds of equations describing the fuel cycle system. The output shows time-dependent tritium inventories in all fuel-cycle units. A very constructive contribution of this level of analysis is to provide good estimations of fuel-cycle system data such as residence times for the systems-level modeling described below.

The second model is based on the flow-residence-time relationships. The system usually consists of about 10 units that are characterized by their respective residence times and leakage rates. The units are linked together through the inflows and outflows of tritium. Time-dependent inventory in each unit can be obtained easily from straight-forward numerical solutions. This model runs efficiently and can be incorporated into the E&S module. However, it can only predict gross inventory in the each unit and the accuracy is determined by that of the input residence times.

We have also adopted a simplified tritium permeation code [2.7] developed at INEL to evaluate tritium inventory in the plasma facing components (PFC). The code evaluates steady-state tritium inventory due to ion implantation and permeation from the plasma consistently with reactor parameters such as ion implantation flux and PFC coolant temperature.

2.2.3 THERMAL TRANSIENT ANALYSIS

The basic accident scenario employed in this work is a loss-of-coolant-accident (LOCA). The LOCA analyses are performed for the divertor and for the first wall, two critical reactor components in terms of generating radiological hazards during accidents. BESAFE performs a thermal analysis to find the temperature of these components as a function of time during accidents. The temperature evolution is important not only for determining whether the PFC will melt, but also for calculation of mobilization of the volatilized oxides of the PFC.

The temperature rise in a LOCA is driven by the energy sources within the machine and is diminished by radiative heat transfer across the vacant coolant gap into the blanket and other surrounding structures. The surface heat sources consist of plasma particle and radiation fluxes while the volumetric heat sources include activation decay heat and penetrating neutron deposition. The neutron deposition and surface heating are functions of the plasma behavior during the shutdown. The radioactive decay heat is based solely on the inventory of the component being analyzed. Ablation for the divertor was assessed as a mode of heat loss and found to be negligible for the tungsten divertor though it may prove to be important if other materials with lower melting points are employed.

We make the simplifying assumptions that the heat capacity and emissivity are not functions of the temperature of the material and that the heat sink temperature does not change with time. No conduction between components is considered. These assumptions do not change the results significantly for the materials treated here.

BESAFE utilizes an unusual combination of 1-D and 0-D analyses to cut the running time of the code. A 1-D finite differencing scheme is used to calculate the temperature profile of the PFC until the profile has flattened enough so that a 0-D analysis is sufficient. This procedure has been shown to yield accuracy to within a degree celsius and can reduce the running time from a minute on a Cray to a second on a workstation.

The 1-D analysis is necessary primarily for treatment of the LOCA scenario in which the plasma does not shut down immediately with the onset of the LOCA. In this case, the finite thermal conduction coefficient sustains a temperature profile driven mainly by the surface heat load. For slowly varying temperature well after plasma shutdown, the 0-D analysis is used for global power balance over the entire PFC region.

2.2.4 MOBILIZATION OF ACTIVATED MATERIALS

Mobilization of activated materials is calculated by two scenarios, namely, volumetric release fraction and surface release rate. For volumetric release, release fractions are assigned based on the chemical and physical properties of the elements being mobilized. The mobilized of a certain activation nuclide is calculated by the product of its release fraction and its inventory. We use the release fractions found in the ESECOM report. Release fractions reflect only the choice of composition of the components in question and do not account for design changes.

The surface release process involves scenario-based calculations, i.e. temperature history in accidents. We have adapted experimental results at INEL [2.8] that measured the oxidation volatility of the constituent elements of PCA and HT-9 steels and tungsten alloy. Mobilizations are calculated using an equation developed at INEL fitting the experimental results:

$$G = A t^B e^{-C/RT} \quad (2.1)$$

where G is the mobilization rate in $\text{g/m}^2\text{h}$, the empirical constants A , B and C have been established for each element by data regression (nine elements thus far), t is the time in hours, T is the temperature in Kelvin, and R is the gas constant.

These surface phenomena assume exposure to air or steam. The experiments involved small samples heated for several hours. We have applied the empirical equations to the LOCA temperature transients assuming that the equation represents instantaneous rates of volatilization. We integrate the rates calculated to find the total mobilization and then convert this into critical and chronic doses using the methods described for radiological dose conversions. Our only modification of the above equation concerns the time factor. There is no data for times less than one hour which may be important regarding plasma shutdown effects. We chose to maintain a constant oxidation rate for an initial time t_c of the accident and follow the t^B dependence thereafter, with the total release in 10 hours normalized to the experimental data. The parameter t_c can be varied typically in the range of half to one hour.

It should be noted at this point that the mobilized dose, calculated from the mass of mobilized material, is not the expected dose to the public in a LOCA. The mobilized dose is merely a figure of merit akin to a source term. No treatment is given here to the confinement or

containment barriers because these factors are highly disputed and their exclusion in no way detracts from the code's capacity to compare the relative merits of designs from a safety standpoint.

2.2.5 DOSE CONVERSION

A set of E&S indices is utilized to summarize the results of the E&S analysis. This set includes the critical and chronic doses to portray the accident hazard and the intruder dose and annualized intruder hazard potential to characterize the waste hazard, as defined in the ESECOM report [2.14]. The dose conversion is evaluated by the FUSEDLOSE [2.9] data and methodology with weather conditions used in ESECOM study.

In the dose conversion calculations, a Gaussian plume dispersion is used with meteorological conditions including decay, daughter production, plume rise, inversion layer, plume deposition and depletion. The critical (acute) dose includes both external irradiation by suspended and ground-deposited radionuclides and internal irradiation from radionuclides inhaled during plume passage. The chronic dose is evaluated by ground shine and inhalation of resuspended radionuclides from ground-deposited radionuclides after the plume passage.

We have stored a dose conversion database for a standard worst-case weather scenario with Pasquill class F stability, 1 m/s windspeed, inversion layer at 250 m, release at ground level with no thermal rise, and deposition velocity of 0.01 m/s. Other weather scenarios can be calculated off-line.

2.3 ILLUSTRATIVE RESULTS

We performed accident analyses for the first wall and divertor regions, as opposed to the blanket and shield regions, because they exhibit higher transient temperatures and vulnerability to the mobilization of activated materials in accidents.

2.3.1 HT-9 FIRST WALL

We performed our analysis with a reference case of a 2800-MW fusion power tokamak reactor. The tokamak has a 8-m major radius and a 2-m minor radius, with a 2-mm HT-9 first

wall. We used the material composition of HT-9 steel as described in the BCSS [2.15] report. The major constituents of the alloy are 86% of iron, 11% of chromium, 2.5% of tungsten by weight. The impurity contents include about 30 elements with concentration up to 0.1 ppm.

2.3.1.1 Radioactive Sources

For the activation inventory in an HT-9 first wall, we have generated a numerical activation database for 4 generic reference reactor cases. Reactor cases 1 to 4 representing gas-cooled without breeding, water-cooled without breeding, gas-cooled with breeding, and water-cooled with breeding, respectively, to account for the variations in the neutron flux spectrum.

Figure 2.1 shows the radioactivity of the major dose-contributing nuclides in the first wall of each of the 4 reference cases at fusion power of 1.4 GW and 2.8 GW. The dominant radioactivity is due to Mn56 (note the normalization in the figure). The first 4 nuclides contribute to the accident dose while Nb94 is of concern for waste disposal. We chose case 4 as the activation source for our subsequent analysis.

The impact of the alloy constituent and impurity content is illustrated in Fig. 2.2. The presence of 1% cobalt by weight in the HT-9 alloy can contribute to about the same level of critical dose as that of 80% iron. It is important for minimizing activation to limit the quantity of Co and Ta as impurity and Mn, W, Ni, and Mo as alloy constituents. The results in Fig. 2.2 are installed as a database for extrapolation to arbitrary alloy compositions and impurity levels.

2.3.1.2 LOCA Thermal Transient

The plasma shutdown/burn time after the onset of a LOCA is a critical parameter in the thermal response. Figure 2.3 shows the time evolution of the first wall surface temperature with various shutdown times. A prolonged shutdown time can raise the surface temperature by a few hundred degrees Kelvin. However, after the initial transient, the temperature is solely determined by the decay heat, which depends on the activation inventory in the first wall.

Next, we examine the burn time required in order for the first wall to reach the melting temperature of HT-9 in a LOCA as shown in Fig. 2.4. The required burn time is smaller for higher neutron wall loading and thinner first wall. For a 1-mm first wall, any plasma shutdown time longer than 20 s can cause melting for wall loading larger than 3 MW/m².

2.3.1.3 Accident Dose Mobilization

The volatility rate is larger at higher temperatures and decreases in time until saturation. Based on the temperature from the LOCA analysis, a typical cumulative critical dose as a function of time resulting from the mobilization of activated materials is shown in Fig. 2.5. The mobilization rate is higher at the early stage due to a higher temperature and the explicit time-dependent factor. The mobilization ends at about 5 hours when the temperature drops below the threshold for oxidation mobilization.

The critical dose of about 30 rem shown in Fig. 2.5 is due to slightly less than 0.1% mobilization of the total activation inventory in the first wall. On the other hand, a conservative estimation of volumetric mobilization following the ESECOM assumptions (1%, 3%, 10%, 30%, and 100% mobilization of 5 different categories of elements) will result in dose over the 200-rem threshold limit for early fatality.

2.3.1.4 Waste Hazard

The impact of first wall change-out time as a design input parameter on waste hazards is examined here. The effects of mechanical properties are not considered for the analysis. The total life-cycle volume is higher for a shorter change-out time. The intensity of the waste is characterized by the intruder dose which is dominated by the long-lived (half-life ~ 100 yr) nuclides. The radioactivity of these nuclides increases almost linearly with irradiation time, i.e. change-out time. The relationships of these two indices with the change-out time are displayed in Fig. 2.6, showing a trade-off in reducing one while increasing the other.

For change-out time less than 0.72 years, the intruder dose is below 0.5 rem, therefore, the waste qualifies for shallow burial under NRC 10CFR61. Hence, it will be beneficial to limit the change-out time to less than 0.72 years from the shallow-burial waste disposal perspective, while longer change-out times may be preferred for deep geological disposal due to the reduction in waste life-cycle volume.

2.3.2 TUNGSTEN DIVERTOR

The divertor is a critical component to the overall fusion reactor design. Its role as a plasma-facing component is characterized by very high heat and particle fluxes. It is also exposed to a large neutron wall loading and tritium permeation because of its proximity to the plasma.

Tungsten is one of the candidate materials for the divertor of fusion experimental reactors and power reactors. It has displayed good performance against sputtering and erosion effects, as well as having a high melting point and fairly good thermal properties. However, the radiological hazards of utilizing tungsten for divertors also need to be investigated. The safety and environmental consequences of these hazards will partially determine the acceptability of tungsten as divertor material.

We used a reference case of a 2800-MW fusion power tokamak, with a 8-m major radius, a 2-m minor radius, and a 1-mm thick tungsten divertor with no substrate. The divertor is assumed to have an area of 100 m² and a heat load of 6 MW/m² due to the impinging charged particles. The composition of the tungsten alloy follows the Blanket Comparison and Selection Study [2.15] and contains small quantities of Ni, Re, Fe, Co, Mn, Ta and other impurities.

2.3.2.1 Neutron Activation and Tritium Inventory

The dominant radionuclide contributing to the critical dose is W-187, generated by the thermal reaction (n, γ) from W-186. Other important dose-contributing radionuclides include Ta-182, W-181, W-185, Re-186, and Re-188. The inventory of W-187 reaches its maximum saturation value in a time-scale of several half-lives ($T_{1/2} = 23.9$ hr) and decreases gradually due to the burn-up of W-186. The production of Re-186 and Re-188 involves two-step activations so the resulting inventories depend strongly on the neutron flux and increase with irradiation time.

The choice of design configuration, such as water-cooling or gas-cooling, can affect significantly the neutron spectrum in the divertor, and hence, the resulting activation inventories. The water-cooled case produces higher activation inventory, especially from the thermal (n, γ) reactions, due to the strong moderation effect by the water. On the other hand, the presence of breeding material such as lithium simply reduces the activation inventory due to the attenuation of neutrons by lithium with its large thermal cross-section. We use the water-cooled configuration with no breeder as the worst-scenario case in terms of radiological dose inventory for our analysis.

Tritium generated from the (n,t) activation reaction is small. The parent nuclides undergoing the (n,t) reaction are mainly from the minority constituents such as Co-59, Mn-55, Fe-56, Ni-58, and Ni-60. For one-year irradiation of 2800 MW fusion power, the estimated tritium produced in the tungsten divertor by activation is about 2.9 mg.

The quantity of tritium inside the tungsten divertor due to implantation and permeation from the edge plasma is calculated by the INEL simplified permeation model [2.7], for typical reactor parameters, to be about 1.2 g and 2.2 g for the 1-mm and 2-mm thick divertors, respectively. This tritium inventory is usually conservatively assumed to be totally released in accidents due to the high mobility of tritium. This inventory is reasonably low and does not present a major safety concern.

2.3.2.2 LOCA Thermal Transient

The high heat load on the divertor necessitates constant cooling and rapid shutdown of the fusion reaction in a LOCA. The time of continuous plasma burn, after the onset of a LOCA, to reach the melting temperature of tungsten for a range of heat loads is shown in Fig. 2.7. The effects of varying the divertor thickness and adding a copper heat sink are also indicated. A thin divertor of 1 mm will melt in less than 0.5 s for an average heat load of 15 MW/m². Doubling the thickness can prolong the burn for another half second. A smaller heat load of 6 MW/m² can reach the melting point in less than 1.5 s and 3 s for the 1-mm and 2-mm cases, respectively. The inclusion of a copper heat sink, due its high conductivity and heat capacity, helps to extend the burn time to 5-7 s. Note that for these 2 cases, the copper heat sink reaches its melting temperature much faster than the tungsten divertor.

Figure 2.8 shows the maximum transient temperature of the divertor at a heat load of 6 MW/m² for plasma shutdown times of 0 and 1 s. For instantaneous shutdown, the divertor temperature rises gradually to reach the equilibrium temperature determined by the balance of the volumetric decay heat and surface radiation. The system stays in this equilibrium with the temperature dropping as the decay heat diminishes. With finite shutdown time, the temperature jumps to an extremely high value due to the large surface heating with no coolant for heat removal. For the 1-s shutdown case, the heat load due to the burning plasma spikes the temperature up thousands of degrees, though in the reference case the surface does not reach the melting point. After the plasma shuts down, the elevated temperature causes the heat to be quickly radiated away until the temperature reaches the same equilibrium mentioned above.

2.3.2.3 Safety Hazard from Mobilization of Activated Materials

We study the dose exposure due to a 1-day accident mobilization of the activated materials. The initial temperature surge in the first few minutes due to prolonged shutdown of the plasma has

a small impact to the total dose mobilization unless the mobilization rate is much higher initially. We use instantaneous plasma shutdown at onset of a LOCA for our analysis.

For typical ranges of power reactor parameters, such as 2800 MW fusion power (3.6 MW/m² neutron wall loading for our case here) and 1 year change-out time, the mobilized critical dose from a 1-mm thick divertor with an area of 100 m², triggered by a LOCA, is about 92 rem without confinement credit. The mobilized dose is about 5% of the dose inventory if all activated materials from the divertor are mobilized.

2.3.2.4 Waste Hazard

There are no long-lived activation products from tungsten that will impose severe problems for the waste disposal of tungsten divertors. The intruder dose is derived mainly from Fe-60 and Ni-59 by the activation of the minor constituents of nickel and iron in the alloy.

For our reference reactor case of 2800 MW fusion power and 1 year change-out time, the total intruder dose for the tungsten divertor is 0.21 rem which is smaller than the 0.5 rem-limit for shallow burial disposal.

The intruder dose will increase with fusion power and irradiation time, however, the power and time values used in the above analysis are at near the maximum of their ranges. Hence, in general, the tungsten divertor can be disposed of by shallow burial with its BCSS-specified alloy composition.

2.3.2.5 Parametric Scoping Analysis

The effects of changing the design parameters of operating scenarios (i.e., power level and change-out time) and size of the divertor are examined. Three safety figures-of-merits, namely, the maximum temperature achieved for instantaneous plasma shutdown, to characterize the thermal behavior; critical dose from total mobilization of activation inventory in the divertor, to characterize the inventory radiological hazard potential; and critical dose from the mobilized inventory, to characterize the volatilized inventory and its associated safety hazard.

As explained above, the inventory of W-187 decreases slowly, while those of Re-186 and Re-188 increase at a relative higher rate, with irradiation time (or, effectively, the change-out time of the divertor for 100% duty factor). As a result, the overall dose inventory and decay heat

increase with irradiation time. The higher decay heat subsequently induces a higher transient temperature of the divertor. This elevated temperature, in turn, causes a larger mobilization rate of the activation products, resulting in greater mobilized dose. The mobilized dose is enhanced by the increment in both the dose inventory and the temperature with longer change-out time. These variations are shown in Fig. 2.9.

Similar effects are observed with the changes in the fusion power as illustrated in Fig. 2.10. The dose inventory, dominated by W-187 especially at lower fusion power, varies almost linearly with the fusion power. The temperature drops from 1218 K to 930 K when the power changes from 2800 MW to 700 MW. The mobilization rate is a strong function of temperature, less than 1% of the activated materials will be mobilized if the temperature drops to about 1000 K. For lower fusion power, which may be relevant to experimental reactors, the mobilized dose is drastically reduced (to 0.5 rem for 700 MW fusion power). This can be understood by noting that at low fusion powers the maximum temperature achieved is only slightly above the threshold for the oxidation volatility reactions (875K) for a relatively short period of time.

In Fig. 2.11, the dose inventory increases with increasing divertor thickness, however, the dose inventory density (per unit volume) decreases due to the spatial attenuation of the neutron flux in the radial direction. The divertor temperature goes up with the thickness, for a fixed divertor surface area, as determined by the geometrical effect in the heat balance, in which, the decay heat is a volumetric source while the radiative heat sink is a surface effect. A thicker divertor causes a larger mobilized dose due to the elevated temperature despite the slight reduction in dose inventory density.

In summary, the activation products in tungsten divertors can contribute a significant amount of mobilized critical dose in a LOCA. The mobilized dose may be reduced by selecting shorter change-out time, lower fusion power, and smaller divertor thickness.

2.4 CONCLUSIONS

We have developed an E&S computational code BESAFE for performing systems-level evaluation and assessment of fusion reactor E&S characteristics. The modeling of BESAFE covers a comprehensive fusion E&S system, mostly by simplified computational schemes and installed database in order to perform fast iterations. The BESAFE code integrates the plasma, engineering,

and E&S aspects of the fusion reactor system, to obtain representative E&S indices for design characterization and optimization.

We have performed simple illustrative assessments of the E&S characteristics of an HT-9 first wall and a tungsten divertor in a LOCA scenario to illustrate the capabilities of the BESAFE code. The code runs sufficiently fast, thus, allowing broad scoping of input parameters. Some observations of the results are as follows.

The extremely high heat flux on the divertor imposes a very restrictive condition that the plasma must be shutdown rapidly (in about a second) at the onset of a LOCA in order to avoid melting of the divertor. It takes 10-100 seconds for the HT-9 first wall to reach the melting temperature depending on the neutron wall loading and first wall configuration.

The mobilization of activation products due to accident, without confinement, produces a critical dose on the order of about 30 rem and 100 rem for the HT-9 first wall and tungsten divertor, respectively. Both the HT-9 first wall and tungsten divertors have the potential to be disposed by shallow burial depending on the alloy impurity contents and operation change-out time.

Preliminary scoping analysis indicates the existence of design windows for potential E&S driven trade-offs and optimizations. The mobilized dose can be reduced by selecting shorter change-out times. However, the total life-cycle waste volume will be increased accordingly. A thinner PFC will produce less mobilized dose while suffering a more restrictive shutdown requirement. There is a large reduction in the mobilized dose for lower fusion powers. Thus, the experimental reactors will have a much lesser safety concern with dose mobilization.

There are still uncertainties in the modeling of BESAFE that have to be resolved. The fast approximate schemes for calculating activation and tritium inventories need to be upgraded to extend their range of validity to more general reactor design regimes. The uncertainty ranges in the mobilization model and the dose conversion data can change significantly the absolute values of the accident dose. Further work is required in all these areas.

In Section 3, we will describe the integration of BESAFE as one of the modules in SUPERCODE [2.10]. This integration will provide a tool to incorporate E&S criteria in design optimizations and help to focus E&S impacts in reactor design studies.

REFERENCES

- 2.1 S.K. Ho et al., Journal of Fusion Energy **12**, 397 (1993).
- 2.2 S.K. Ho and M.D. Lowenthal, "A Systems Module for Environmental and Safety Assessment in Fusion Reactor Studies--Illustrative Results for HT-9 Structures," Proc. 15th Symp. Fusion Engineering, Hyannis, Massachusetts, (1993).
- 2.3 S.K. Ho and M.D. Lowenthal, "Assessment of Radiological Hazards of Tungsten Divertors," Third Intl. Symp. on Fusion Nuclear Technology, Los Angeles, California, June (1994).
- 2.4 E.F. Plechaty and J.R. Kimlinger, "TARTNP: A Coupled Neutron-Photon Monte Carlo Transport Code," Lawrence Livermore National Laboratory, Report UCRL-50400 (1976).
- 2.5 J.A. Blink, "FORIG: A Computer Code for Calculating Radionuclides Generation and Depletion in Fusion and Fission Reactors," Lawrence Livermore National Laboratory, UCRL-53633 (1981).
- 2.6 D.A. Sarigiannis, "Modeling and Safety Optimization of the Fuel Processing System for Magnetic Fusion Reactors," Ph.D. Thesis, University of California, Berkeley (1994).
- 2.7 G.R. Longhurst, "Simplified Tritium Permeation Code," EGG-FSP-10960, Idaho National Engineering Laboratory (1992).
- 2.8 K.A. McCarthy, G.R. Smolik, and S.L. Harms, "A Summary and Assessment of Oxidation Driven Volatility Experiments at the INEL and Their Application to Fusion Reactor Safety Assessments," Report EGG-FSP-11193, Idaho National Engineering Laboratory (1994).
- 2.9 S.A. Fetter, "Radiological Hazards of Fusion Reactors: Models and Comparisons," PhD Thesis, University of California, Berkeley (1985).
- 2.10 S.W. Haney et al., Fusion Technol. **21**, 1749 (1992).
- 2.11 W.W. Engle, "A One-Dimensional Discrete Ordinates Transport Code System with Anisotropic Scattering," CCC-254, RSIC Computer Code Collection, Oak Ridge National Laboratory (1967).
- 2.12 D.L. Henderson and O. Yasar, "DKR-ICF: A Radioactivity and Dose Rate Calculation Code Package," UWFD-714, University of Wisconsin (1987).
- 2.13 S.K. Ho, F. Brechtel, and T.K. Fowler, Fusion Technol. **23**, 321 (1993).
- 2.14 J.P. Holdren et al., Fusion Technol. **13**, 7 (1988).
- 2.15 D.L. Smith et al., "Blanket Comparison and Selection Studies," Argonne National Laboratory, Report ANL/FPP-84-1, (1984).

Figure 2.1. Activation inventory for the major dose contributing nuclides (A: Mn56/4e8 Ci, B: Mn54/1e8 Ci, C: W187/1e8 Ci, D: Re188/1e8 Ci, E: Nb94/1e1 Ci) at two levels of fusion power for the 4 reference reactor cases.

Figure 2.2. Contribution to the critical dose from the alloy and impurity contents, normalized to the dose due to 80% by weight of Fe in HT-9.

Figure 2.3. Temperature history of the first wall during a LOCA for various plasma shutdown time.

Figure 2.4. Plasma burn time required for the first wall to reach the melting temperature of HT-9 during a LOCA as a function of the neutron wall loading.

Figure 2.5. Mobilized critical dose as a function of time.

Figure 2.6. Variations of first wall life-cycle volume and intruder dose with the change-out time.

Figure 2.7. The divertor inner surface temperature as a function of time after the onset of LOCA for shutdown times of 0 s and 1 s.

Figure 2.8. Time to reach melting temperature for the tungsten divertor after the onset of LOCA without plasma shutdown as a function of the heat flux for 1-mm and 2-mm divertors with and without copper heat sink.

Figure 2.9. The variations of the maximum divertor temperature, critical dose inventory, and mobilized critical dose with the divertor change-out time.

Figure 2.10. The variations of the maximum divertor temperature, critical dose inventory, and mobilized critical dose with the fusion power.

Figure 2.11. The variations of the maximum divertor temperature, critical dose inventory, and mobilized critical dose with the divertor thickness.

3. IMPLEMENTATION OF BESAFE INTO SUPERCODE

3.1 INTRODUCTION

The SUPERCODE [3.1] is a tokamak reactor design code developed at Lawrence Livermore National Laboratory. The code attempts to fill the gap between existing, comprehensive but simplified 0-D systems codes and highly sophisticated, multi-dimensional, specialized plasma performance codes. In spirit, the SUPERCODE is a systems code with greatly enhanced engineering and physics modules. Specifically, the code calculates the 1-1/2-D MHD-transport time evolution of a plasma in a realistic engineering environment.

A critical feature of the code is its fast execution time, a mandatory requirement if it is to be successfully used as a systems studies tool. The high speed is obtained by (1) the extensive use of variational techniques, (2) formulation that carefully chooses the input quantities so as to minimize the global iterations required, and (3) a set of semi-analytic approximations whenever computationally expensive procedures would dominate the execution time. Equally important attention has been devoted to the architecture of the code, resulting in a highly modularized and user-friendly code. Each module contains data and computational functions related by a single unifying abstraction. It provides flexibility for coupling and decoupling between various physics and engineering modules.

The implementation of BESAFE into SUPERCODE has several advantages. Our group can add environmental and safety relevant calculations into SUPERCODE. The plasma parameter input to BESAFE will be made much more realistic by extracting results from a better plasma model in SUPERCODE. Important safety events, such as emergency plasma shutdown, need to be investigated more accurately via a SUPERCODE-level plasma analysis. Our ultimate goal is to perform reactor design optimization by running SUPERCODE while using design constraints or figures-of-merit provided by BESAFE.

3.2 CURRENT IMPLEMENTATION STATUS

We have converted the BESAFE code as one of the modules in SUPERCODE. This conversion provides the feasibility of establishing information exchange pathways between the BESAFE module and other modules in SUPERCODE.

We can now execute BESAFE under SUPERCODE in three distinct modes, namely, (1) running BESAFE as a stand-alone code, (2) running BESAFE to obtain environmental and safety characterizations after a design point is derived by calculations with other physics and engineering modules in SUPERCODE, and (3) running BESAFE as one of the many modules to be processed in a global optimization scheme.

We have been successful in running the stand-alone mode under the execution shell of SUPERCODE. In this case, input data independent of SUPERCODE are required to initiate the BESAFE run. Benchmarking with previous BESAFE results has been performed.

For running BESAFE in the next two levels, we have to identify and establish specifically the coupling processes and associated parameters between BESAFE and the rest of SUPERCODE. At this time, for the purpose of demonstrating the feasibility of this approach and testing the programming logistics, we have selected fusion power as the only parameter linking BESAFE and other SUPERCODE modules. A complete coupling of all the relevant parameters will be implemented in the near future.

Hence, in this scheme, the SUPERCODE physics and engineering modules will iterate to calculate the plasma parameters such as plasma density and temperature, which in turn, determine the fusion power. This information is passed to BESAFE as its initial input. The fusion power has several critical correlations with the environmental and safety characterizations.

For our illustrative runs, we use the mobilized critical dose from the activated materials in the first wall during a loss-of-coolant accident (LOCA) as our safety index. The activation inventories and the corresponding decay heat are directly determined by the fusion power. The thermal transient temperatures in accidents are calculated from neutron and particle wall loadings and the decay heat, which are all dependent on the fusion power. The mobilization of activated materials by oxidation volatility is a strong non-linear function increasing with temperature. Therefore, due to all these effects, the fusion power has a strong influence on the activation inventories and the mobilization of these inventories.

3.3 PRELIMINARY RESULTS

We used one of the standard SUPERCODE reactor-case input parameter sets to obtain a reference reactor design, which has a fixed net electrical output of 1200 MW and optimized with

minimum cost-of-electricity (COE). We would like to investigate the changes in the design if safety restrictions are imposed in the optimization process.

In this reference SUPERCODE optimization run, the COE was used as the figure-of-merit to be minimized. There were 41 design variables to be calculated from the iterations. This set of variables included plasma parameters like major radius, minor radius, plasma density and temperature, and engineering parameters like radial build thickness, etc. There were 23 equality constraints and 16 inequality constraints to be satisfied. The inequality constraints may or may not be optimized at their extreme bound limits. In the optimization procedure, 46 calculational modules have been executed iteratively.

After obtaining the reference case parameters, we proceeded to evaluate the corresponding mobilized critical dose from the first wall in a LOCA scenario by running the BESAFE module with the reference case fusion power of 2766 MW. A mobilized dose of 23.6 rem was obtained. Then we ran three more optimization cases, with the inclusion of the constraint that the mobilized dose should be less than 20 rem, 15 rem, and 10 rem, respectively.

As expected, the fusion power has to be reduced in order to satisfy the dose-limit constraint. This constraint generates a reactor design with reduced fusion power and net electrical output power, and an elevated COE. A more restrictive constraint will produce a larger reduction in fusion and electrical power, with a steeper increase in COE. The results of these 4 optimization runs are summarized in Table 3.1. The relevant plasma parameters for the cases are shown in Table 3.2.

3.4 SUMMARY

We have installed the BESAFE code as an environmental and safety module in SUPERCODE, interlinking it with other physics and engineering modules. As a result, SUPERCODE will have the capability to perform environmental and safety assessments and design optimization with additional input from environmental and safety constraints.

We have benchmarked the implementation by running BESAFE as a stand-alone code under the execution shell of SUPERCODE. We have investigated the impacts of safety restrictions via BESAFE on the overall design optimization with emphasis on the variations in fusion power. The requirement of lowering the mobilized dose from the activation products in the first wall

results in lower fusion power and higher COE designs, however, we have not attempted to compensate this with other design modifications.

The search for a complete linkage and coupling of all relevant parameters in BESAFE and in other SUPERCODE modules is essential for a fully integrated optimization. Future work will focus on this issue and the subsequent optimization and trade-offs including environmental and safety features.

REFERENCES

- 3.1 S.W. Haney et al., Fusion Technol. **21**, 1749 (1992).

Table 3.1. Various reactor figures-of-merit for the reference reactor case and three dose-limited cases.

Reactor Case	1	2	3	4
Mobilized Dose (rem)	23.6	20.0	15.0	10.0
Fusion Power (MW)	2766	2596	2331	2031
Net Electrical Output Power (MW)	1200	1149	1025	885
COE (mills/kW-hr)	92.6	95.5	102.3	111.8

Table 3.2. Plasma parameters for the reference reactor case and three dose-limited cases.

Reactor Case	1	2	3	4
major radius (m)	5.25	5.43	5.34	5.22
minor radius (m)	1.65	1.67	1.63	1.56
plasma current (MA)	14.03	13.30	12.64	12.09
toroidal field (T)	5.32	5.55	5.54	5.54
plasma beta	0.096	0.086	0.084	0.084
average electron density (10^{20}m^{-3})	1.77	1.57	1.55	1.25
average ion density (10^{20}m^{-3})	1.47	1.30	1.29	1.25
Z-effective	1.82	1.84	1.84	1.84
average electron temperature (keV)	18.41	20.00	19.74	20.12
average ion temperature (keV)	18.14	20.05	19.74	20.17
energy confinement (s)	1.28	1.49	1.50	1.53

4. STREAMLINING NEUTRONICS AND ACTIVATION CALCULATIONS

Fusion reactor systems studies require iteratively scoping a large parameter space in order to locate an optimal design space. Thus, the modules in a systems code have to be computationally fast. The design community has been very successful in developing fast but simplified modules of plasma behavior and engineering design into systems codes, but thus far any E&S evaluation of the resulting designs has been carried out after the fact and was not incorporated directly into the iterative optimization procedure. One reason is that the calculation of activation-induced radioactivity inventories is a complicated and time-consuming procedure.

This section presents three innovative approaches in streamlining the calculation of activation-induced radioactivity for fusion reactor environmental and safety systems studies. The first approach, described in Section 4.1, is the formulation of scaling equations to directly calculate activation inventories. A streamlined neutronics and activation code-system derived from existing codes is described in Section 4.2. Finally, Section 4.3 describes the approach with a perturbation scheme.

4.1 FORMULATION OF SCALING EQUATIONS

4.1.1 INTRODUCTION

We have explored the feasibility of a scaling approach to estimate the radioactivity inventory with a simple fixed configuration of a silicon carbide (SiC) first wall in Ref. 4.1. The methodology has been extended and generalized to evaluate various radiological doses of a silicon carbide first wall coupled with a breeding blanket in an overall reactor system covering variations in typical reactor design parameters. An assessment of the accuracy in calculating the radiological doses for general reactor configurations has been performed.

The usefulness of this fast scaling calculational scheme requires the establishment of a complete database which can be reduced to sets of scaling equations to cover the range of design choices including materials, compositions, and configurations. As a next step towards this goal, an exploratory study for the ferritic steel structure in both the first wall and blanket regions is also investigated here. We focus our work here on the HT-9 ferritic steel which has a relatively

reduced-activation characteristic in the alloy composition. Various forms of steel with enhanced molybdenum and manganese will be briefly discussed.

We have also studied the formulation of scaling equations to approximate the variations of activation inventory with the operating conditions in power level and irradiation time. The changes in operating conditions will modify the neutron flux spectrum and the number density of the target nuclides. The impact on the neutron spectrum is relatively small because the spectrum is still dominated by the initial composition of material. However, the burn-up and build-up of target nuclides can significantly affect some activation reactions. As a result, we examine specifically the formulation of the burn-up and build-up modifications to the activation inventories.

4.1.2 MODELING AND ASSUMPTIONS

We have followed the ESECOM [4.2] computational methodology, which was also used in Ref. 4.1, in calculating the neutron activation radioactivity. In this scheme, a set of computer codes TART [4.3], ORLIB [4.4], and FORIG [4.5] are run in sequence. The code TART calculates the neutron transport by a Monte Carlo simulation. The resulting neutron spectra in various zones are then used as input into the ORLIB code, which integrates the neutron spectra to obtain spatial- and energy-averaged activation cross sections. The ACTL [4.6] activation library is used. Then, the FORIG code calculates, in each reactor zone, the nuclide generation and depletion by the nuclide decay, neutron activation, and neutron absorption processes with a point model utilizing the averaged reaction cross sections.

However, we would like to point out that there are uncertainties in the various activation libraries employed in existing activation codes. A comparison of the relative accuracy of various existing activation libraries and computer codes is a tremendous task that is out of the scope of this work. A recent report [4.7] which surveyed most major activation codes has shown the diverse results among the various codes. Our work is based on the FORIG code which uses the ACTL activation library. Hence, the accuracy of the scaling equations is limited by the accuracy of the ACTL library.

The tokamak is approximated in a cylindrical configuration. The radial-build includes first wall, blanket and shield regions. The blanket is divided into several zones. The spatial- and energy-averaged activation cross sections in a zone are used to calculate the total radioactivity of each activation product in that zone. The spatial resolution of the radioactive inventory is higher with smaller zone width as expected.

We specified the impurity content of silicon carbide as that in the ESECOM [4.2] report, namely, 11 ppm of iron and 0.3 ppm of cobalt by weight. We used the material composition of HT-9 steel as described in the BCSS [4.8] report. The major constituents of the alloy are 86% of iron, 11% of chromium, 2.5% of tungsten by weight. The impurity contents include about 30 elements with concentration up to 0.1 ppm.

We have fixed the neutron irradiation time at 3 years as being a representative maximum first-wall change out time. This will have little effect on the activation radioactivity of the short-lived nuclides while the radioactivity of long-lived nuclides will be overestimated or underestimated if the actual irradiation time is shorter or longer, respectively.

The environmental and safety indices used in this study are the same as those in the ESECOM report. The accident analysis is represented by the critical and chronic doses. The waste disposal problem is characterized by the intruder dose. The radiological doses from activities of various radionuclides are computed by a dose-conversion code FUSED0SE [4.9]. We assume class F weather conditions, with 1 m/s windspeed, deposition velocity of 0.01 m/s, with a release at ground level with no thermal rise. The mobility of various radionuclides is classified into five categories as in the ESECOM studies.

Recognizing at the outset that scaling laws cannot be expected to reproduce all details, we have specifically applied our results to calculate useful integral quantities, namely, the radiological hazard indices used by ESECOM. We take as a criterion of success the extent to which these indices calculated from scaling laws do or do not agree with those calculated directly from the computer code data. In this way, we focus quantitatively on just that level of information that we believe to be necessary and sufficient to provide an online evaluation of E&S characteristics during a systems code optimization procedure.

The recognition of the radiological doses as the relevant figures of merit for activation analysis is very essential for the scaling equation scheme. For both the silicon carbide and the HT-9 ferritic steel with the impurity contents as stated above, over 100 activation products are generated. From Ref. 4.1 in our previous work, we found that only 8 radionuclides from the activation of silicon carbide contributing to the relevant radiological doses. Similar results are found for the HT-9 steel. This greatly simplifies the quantity of data required and makes it practical in performing the data-regression procedure.

4.1.3 COMPLETION OF ACTIVATION SCALING FOR SILICON CARBIDE FIRST WALL

4.1.3.1 Review of Previous Work

From our previous work [4.1], we have formulated sets of scaling equations to calculate the radioactivity of the relevant dose-contributing radionuclides for a silicon carbide first wall. The scaling equations for the radioactivity of the one-step and two-step activations are given respectively, by:

$$A_{\text{one-step}} = K P_{\text{fus}} \text{CFB} \delta^x \rho^y (1 + \delta / 2 r_p) \text{BU} \quad (4.1)$$

$$A_{\text{two-step}} = K P_{\text{fus}}^2 \text{CFB} \delta^x \rho^y (1 + \delta / 2 r_p) \text{BU} / (r_p R) . \quad (4.2)$$

where A is the radioactivity, K is a data-fitting constant, P_{fus} is the fusion power, δ is the first wall thickness, r_p and R are the minor and major radii, ρ is the density.

The empirical coefficient (K) and exponents (x and y) for each radionuclide are obtained by data regression of numerous computer runs. The values of K, x, and y for the major dose-contributing nuclides for SiC are shown in Ref. 4.1. The factor CFB contains effects of both strong thermal neutron absorption by lithium and neutron moderation by water coolant in the blanket. The parameter BU corrects for the irradiation time-dependent effects such as burn-up. The correction in BU is small for the silicon carbide first wall.

4.1.3.2 Correction Factors for the Overall Reactor System

We focused here on the effects on first-wall activation due to the variations of blanket configurations as expressed in the higher-order correction factor CFB in the above scaling equations. The coupling of the first wall with the blanket is through the modification of the resulting neutron spectrum in the first wall. The magnitude of the neutron flux and the neutron-energy dependent reaction cross sections within the first wall are changed in the process.

In general, in the first wall region, the fast-neutron spectrum is not much affected by the blanket, while the thermal-neutron spectrum is very sensitive to the blanket configuration. As a result, fast-neutron activation reactions are almost independent of the blanket design while the thermal-neutron activation processes are strongly dependent on the geometry and composition of the blanket. Among the major dose-contributing nuclides in SiC, only C-14 and Co-60 are

produced mainly by thermal-neutron activations. Hence, modifications to the scaling equations are required for C-14 and Co-60 with respect to the blanket system parameters.

By analyzing an extensive neutronics and activation database covering the design ranges of lithium and water contents in blanket composition, we can formulate correction factors to account for the impacts of strong thermal neutron absorption by lithium and fast neutron moderation by water in the blanket on first wall radioactivity inventories. A simple scaling equation is constructed for the correction factor CF_B , with li being the lithium atom ratio and h being the water coolant fraction by volume, both treated as continuous varying parameters:

$$CF_B = C_0 (f_1 + f_2 h + f_3 h^2) \quad (4.3)$$

with $f_i = C_{i,1} + C_{i,2} li + C_{i,3} li^2 + C_{i,4} li^3 \quad i = 1, 2, \text{ and } 3.$

The values of the coefficients $C_{i,j}$ for the correction factor CF_B for C-14 and Co-60 from regression of the neutronics data are shown in Tables 4.1 as reported in Ref. 4.10. Thus, all major factors affecting the radioactive inventories of significant dose-contributing nuclides for a SiC first wall have been included in the scaling equations described in previous section.

4.1.3.3 Applicability

The above scheme is a simple approximation utilizing analytic insights and extensive data regression neutronics data for a predetermined reactor configuration. In this section, we examine the potential sources of error and estimate the accuracy of the scaling equations.

First, the regression scaling equations are based on the assumptions that $(\lambda + \phi\sigma) t \gg 1$ and $\lambda \gg \phi\sigma$. The former condition removes the explicit time dependency with respect to the irradiation time. The latter condition makes the radioactivity proportional linearly or to the square of the neutron flux, instead of a combination of the decay and activation rates. These are true for most of the nuclides involved here, except for the long-lived nuclides Al-26 and C-14 under very high flux irradiation condition.

Secondly, it is impractical to scope the variations of the radioactivity with all the reactor input parameters simultaneously. The neutronics data is established by varying the independent parameters one or two at a time around a reference case. For example, we studied the effects of

changing lithium breeder and water coolant together while keeping the first wall thickness at 2 cm, with a long irradiation time of 3 years.

Thirdly, the computational results are fitted by polynomial regression. If the radioactivity inventory of a particular nuclide changes drastically over some domain of an independent parameter, then the fitting is divided into various ranges. In this way, the discrete database is formulated in a continuous fashion through a set of polynomial equations.

As a check on these assumptions, we have compared the numerical calculations with those calculated by the approximate scaling equations for several cases. For these comparisons, we consider 12 reactor cases chosen at random from the range of parameters as those considered in the database for the formulation of the scaling equations. These ranges are shown in Table 4.2. We have then calculated numerically and by the scaling equations, the radioactivity for the eight major radionuclides for these 12 cases. We have not performed a complete analysis of the covariance of all the independent parameters. Instead, we would like to point out some general observations with respect to the most uncertain parameters and the ranges of uncertainty. The fast-neutron induced radioactivity of all nuclides are within 20% of the computational values. The largest errors are in the evaluation of the thermal-neutron-induced activation products, i.e. C-14 and Co-60, due to the changes in the blanket and coolant composition. The radioactivity of these nuclides may be off by about 80-100% in the extreme cases. Similar agreements have been obtained for other reactor cases during the course of our work.

In addition, we have calculated by both methods the doses obtained from the radioactive inventories from the 12 cases chosen for comparison. Comparisons of the critical, chronic, and intruder doses from neutronics-code calculations and radioactivity scaling equations are shown in Figs. 4.1-4.3, respectively. The two approaches agree very well in the evaluation of the critical dose. The critical dose is mainly due to the radionuclides Mg-27, Al-28, Al-29, and Na-24 generated by fast-neutron reactions. These reactions are less sensitive to variations in the reactor parameters, and hence the scaling equations result in less uncertainties and errors. On the other hand, the chronic dose, which is dominated by Co-60 from the thermal-neutron activation, exhibits a larger discrepancy between the scaling model and the neutronics-code calculations. The main reason for this discrepancy is the difference in thermal neutron population calculated by the neutronics code and that predicted implicitly by the scaling equations for various reactor configurations. The scaling equations seem to be conservative in that they tend to overestimate the chronic dose. Finally, the scaling equations produce good agreement with the numerical data for the waste disposal as characterized by the intruder dose.

Overall, the scaling approximation produces fairly good results for the radiological doses in the reactor parameter regime as shown in Table 4.2. The validity of the scaling equations is not restricted to this regime, except for the fusion power, first wall thickness, and water coolant content. The accuracy will be reduced if these parameters are increased beyond the extreme values indicated. The burn-up effect will be more apparent if higher fusion power level is used while higher values of first wall thickness and water coolant content have not been investigated.

4.1.4 COMPOSITION ASSESSMENT AND RADIAL SCALING FOR STEEL STRUCTURE

We have chosen four types of reactor designs representing the most common mainstream conceptual designs of tokamak as shown in Table 4.3. Reactor cases 1 to 4 represent gas-cooled without breeding, water-cooled without breeding, gas-cooled with breeding, and water-cooled with breeding, respectively. The helium gas coolant is treated as void for activation purpose. We used lithium oxide as the breeding material in which the effect of lithium is examined. The scaling equation formulation is performed separately for the 4 discrete types of reactor in order to avoid correlating the data in those different configurations.

For the HT-9 ferritic steel, the critical dose is mainly due to Mn-54 and Mn-56 from iron and W-187, Re-186, and Re-188 from tungsten. The chronic dose is produced again by Mn-54 from iron, and Co-60 from both iron and trace quantity of natural cobalt. On the other hand, the intruder dose is dominated by impurities of niobium and silver which produce Nb-94 and Ag-108m, respectively. These radionuclides, out of over 100 activation products induced, account for more than 90% of the total of the three above described radiological doses.

Instead of repeating the procedures in formulating the scaling equations with the associated empirical coefficients as for a SiC first wall, we look at issues that have not been fully addressed. In the following sections, we will first examine the changes in the activation inventories due to the variations in the steel alloy composition. Then, the formulation is extended to include the variations in radial location in the blanket regions.

4.1.4.1 Variations in Steel Composition

There is much uncertainty in the impurity levels in the steel alloy. We examine here the impacts of the impurities to the radiological doses and develop a scheme to incorporate them in the scaling formulation. There are also various forms of steel alloys different than our chosen HT-9

steel. We will also try to briefly estimate the qualitative differences of the major steel alloys in terms of environmental and safety implications.

A change in the contents of the material to be activated has two effects. It will modify the neutron flux spectrum and the available activation targets. The effect on neutron spectrum is very complicated and its solution requires a neutron transport analysis, for example, by one of the transport codes. On the other hand, for a given neutron flux spectrum, the induced-radioactivity from a target element is proportional to the number of atoms of the target element. These effects will be discussed below in reference to the impurity contents and other variants of steel alloys.

4.1.4.1.1 Role of Impurity Contents

The neutron spectrum in the HT-9 steel structure blanket is mainly determined by the major constituents in iron, chromium, and tungsten, together with the overall configuration with the coolant and breeder. The presence of small contents of impurity trace elements will not affect the resulting neutron spectrum. Hence, the activation radioactivity of each impurity element is directly proportional to its content. We can calculate the radioactivity of the impurities with some reference concentrations, and then use that as a reference to scale to other desired impurity levels. In this way, we can extend the scaling equations to include a large range of impurity levels.

Next, we examine the contributions of the impurities to the accident doses in the HT-9 steel alloy with concentrations as given in Ref. 4.8. There is no significant effect in the critical dose from the impurities. For the chronic dose, the only contributing impurity is cobalt, which accounts for about 3% of the total dose. However, if the impurity level of cobalt is increased from the specified 20 ppm to 200 ppm, then its contribution will be at slightly less than 30% of the total chronic dose.

The main constituents in ferritic steel have no significant impact in the waste disposal problem which is determined by the radioactivity at a hundred years after irradiation. Based on the levels of impurities used in Ref. 4.8, we find niobium and silver alone determine the acceptability for shallow burial under the 10CFR61 Class C waste disposal rating criterion. The resulting waste disposal rating is acceptable for shallow burial. However, any increase in the impurity levels of niobium and silver can lead to the violation of the shallow disposal limit.

4.1.4.1.2 Other Variants of Steel

There are other variants of steel such as the molybdenum- and manganese-enhanced versions. We examined the effects of replacing the chromium with either molybdenum or manganese in the alloy composition.

We found that there is no big impact on accident hazards from molybdenum. However, a 0.1% content of molybdenum by weight causes the structure to be marginal acceptable for shallow burial. Hence, the molybdenum-enhanced steel will have difficulty in shallow waste disposal.

For the manganese steel, a 10% by weight of manganese produces slightly larger values of critical and chronic doses than that of the iron and tungsten combined. Therefore, the manganese-enhanced steel is not a good option for the purpose of activation safety. The manganese imposes no further burden on the waste disposal problem.

It seems that the HT-9 steel with 11 % of chromium by weight as minor constituent is better than the molybdenum and manganese counterparts in terms of safety and environmental performance.

4.1.4.2 Radial Variations of Inventories in the Blanket

The radial-build of a tokamak reactor includes first wall, blanket and shield regions. The first wall thickness is so small that it can be treated as a single region. The blanket is divided into several zones. The spatial- and energy-averaged activation cross sections in a zone are used to calculate the total radioactivity of each activation product in that zone. The spatial resolution of the radioactive inventory is higher with smaller zone width as expected. We used 20 cm as the radial thickness of the blanket zones in this work.

The normalized (per 100 kg of HT-9 in each zone) radioactivities for Mn-54 mainly generated by fast-neutron activation and Co-60 mainly produced by thermal-neutron reaction are shown in Figs. 4.4 and 4.5, respectively. The data points are at the center of the various zones. From Figs. 4.4 and 4.5, we can see the radioactivity is largest for reactor case 1 and lowest for case 4 as expected from the roles of water and lithium. For Co-60, the radioactivity decreases very slowly in the radial direction in the absence of lithium, which has a high reaction cross section with thermal neutrons. In the first zone, case 2 has slightly higher radioactivity than case 1 due to the enhanced thermal neutron population resulted from fast neutron moderation by water. Similar

behaviors have been observed for other activation products characterized by the fast or thermal neutron activations.

Next, we turn to the formation of the empirical equations. The activation inventory scaling equations can be generalized with the inclusion of a radial function Rad. The radial dependence of each zone as represented by the term Rad can be put approximately in the form of:

$$\text{Rad}_{ij} = \exp(-X_{ij} r) \quad (4.4)$$

where i denotes the different nuclides and j for various types of reactor configurations, and r is the radial location of the center of a particular zone. The values of X for the four reference cases are shown in Table 4.4. A larger value of X represents a higher neutron flux attenuation rate. Case 4 has the steepest decline in radioactivity generation in the outward direction from the plasma.

4.1.5 ACTIVATION BUILD-UP AND BURN-UP SCALING FOR HT-9 STEEL

4.1.5.1 Analytic Solutions

We first briefly examine the neutron activation physics to have some qualitative understanding of the functional forms of the activation radioactivity. The neutron activation processes in general are schematically shown in the following sequences:

N is the radionuclide under consideration, generated from the activation of parent nuclide N_p . Our approach is to correct for the burn-up or build-up of N_p . The schematic reaction chains include possible production of N_p from the decay of N_d and activation of N_a . The usual notations are used, i.e., N is the number density, σ is the reaction cross-section, Φ is the neutron flux, λ is the decay constant, and the subscripts specify the nuclides as displayed in the reaction scheme. Activation reactions from multiple targets are considered while only a single decay process (β decay) is included.

The rate equations for N and N_p , and their general solutions are as follows:

$$\frac{dN}{dt} = \Sigma_p \sigma_p \Phi N_p(t) - \lambda N \quad (4.5)$$

$$N(t) = e^{-\lambda t} \Sigma_p [\sigma_p \Phi \int_0^t e^{\lambda t'} N_p(t') dt'] \quad (4.6)$$

$$\frac{dN_p}{dt} = \Sigma_a \sigma_a \Phi N_a + \lambda_d N_d - \sigma \Phi N_p \quad (4.7)$$

$$N_p(t) = e^{-\sigma \Phi t} \left\{ N_p(0) + \Sigma_a \sigma_a \Phi \int_0^t e^{\sigma \Phi t'} N_a(t') dt' + \lambda_d \int_0^t e^{\sigma \Phi t'} N_d(t') dt' \right\} \quad (4.8)$$

The complexity of the solutions increases if we trace higher order corrections of the parent nuclides up the reaction chains. For example, N is simply linearly dependent on Φ , unless N_p varies with time. In this work, we only try to account for the variations of N_p , assuming the contents of its parent nuclides are constant in time. For the trivial case of burn-up only in N_p ($N_d=0$ and $N_a=0$), solutions for N_p and N are,

$$N_p(t) = N_p(0) e^{-\sigma \Phi t} \quad (4.9)$$

$$N(t) = \Sigma_p \sigma_p \Phi N_p(0) \frac{e^{-\sigma \Phi t} - e^{-\lambda t}}{\lambda - \sigma \Phi} \quad (4.10)$$

If the production of N_p is only due to the decay of N_d , and N_d has reached its saturation value (i.e., $N_d = N_{dsat} = \sigma_i \Phi N_i / \lambda_d$), then N_p and N are given by Eqs. (4.11) and (4.12), respectively. To include the activation due to N_a , the term $\lambda_d N_{dsat}$ can be replaced by $(\lambda_d N_{dsat} + \sigma_a \Phi N_a)$, if N_a is constant in time. Analytic solutions can also be obtained for N_d and N_a as functions of time, however, the expression will be too complicated for the derivation of scaling equations.

$$N_p(t) = e^{-\sigma \Phi t} \left(N_p(0) - \frac{\lambda_d N_{dsat}}{\sigma \Phi} \right) + \frac{\lambda_d N_{dsat}}{\sigma \Phi} \quad (4.11)$$

$$N(t) = \sum_p \left\{ \frac{\sigma_p \Phi N_p(0)}{\lambda - \sigma \Phi} (e^{-\sigma \Phi t} - e^{-\lambda t}) + \frac{\sigma_p \lambda_d N_{dsat}}{\sigma} \left(\frac{1 - e^{-\lambda t}}{\lambda} - \frac{e^{-\sigma \Phi t} - e^{-\lambda t}}{\lambda - \sigma \Phi} \right) \right\} \quad (4.12)$$

4.1.5.2 Scaling Equations for Burn-up and Build-up of Target Nuclides

The factor BU, for measuring both the burn-up and build-up of the parent nuclide N_p , as a function of neutron wall loading Γ (Γ is used instead of Φ , as Γ is a more representative reactor design parameter and it is proportional to Φ for a given configuration) and irradiation time t , is defined as,

$$BU(\Gamma, t) = \frac{N_p(\Gamma, t)}{N_p(0)} \quad (4.13)$$

If $N_p(0) = 0$, we can assume an arbitrary small value of $N_p(0)$ for the calculation. The values of N_p and BU will be calculated consistently, in time, N_p should be much larger than the assumed initial value if the build-up of N_p is significant to the production of N .

Following Eqs. (4.9) and (4.11), the BU equations can be expressed as functions of Γ , t , and 1 or 2 constants for fixed reactor configurations. The constants are then solved analytically in terms of some BU values at certain operating conditions and substituted back to the equation. For the cases of (i) burn-up of N_p and (ii) build-up of N_p with $N_d = N_{dsat}$ and/or $N_a = N_a(0)$, the BU equations are given by Eqs. (4.14) and (4.15), respectively.

$$BU(\Gamma, t) = BU_{ref} \frac{\Gamma t}{(\Gamma t)_{ref}} \quad (4.14)$$

$$BU(\Gamma, t) = R + (1 - R) \left(\frac{1 - BU_{ref1}}{BU_{ref1} - BU_{ref2}} \right) \frac{\Gamma t}{(\Gamma t)_{ref1}} \quad (4.15)$$

where,

$$R = \frac{BU_{ref1}^2 - BU_{ref2}}{2BU_{ref1} - BU_{ref2} - 1} \quad (4.16)$$

BU_{ref} is a reference value of BU evaluated numerically by codes at a reference operating point (Γt)_{ref}. For the decay/activation production of N_p in Eq. (4.15), two equations are required to solve for the two constants in Eq. (4.11). Hence, we need to select two reference values in (Γ, t) (i.e., ref1 and ref2) and numerically calculate their corresponding BU values. A closed-form solution can be obtained only for the condition that one of the (Γ, t)_{ref} is exactly twice as that of the other one. We select (Γt)_{ref2} = 2(Γt)_{ref1} for our calculations. The BU values are calculated by the input of the ratio Γt/(Γt)_{ref}, thus, providing a convenient feature for design scoping studies.

4.1.5.3 Scaling Equations for Activation Inventory

Eventually, the BU values are used to calculate the number density N of the activation product investigated. Equations for activation inventory can be obtained by taking the ratio of analytic solutions and replacing all configuration constants in terms of Γ, t, and the BU parameters. The same two cases of (i) burn-up of N_p and (ii) build-up of N_p with constant N_d and/or N_a are considered. For radionuclides with short half-lives (T_{1/2}), i.e., λt >> 1, N for both cases can be expressed as in Eq. (4.17), with N_{ref} obtained numerically.

$$N(\Gamma, t) = N_{\text{ref}} \left(\frac{\Gamma}{\Gamma_{\text{ref}}} \right) \left(\frac{\text{BU}(\Gamma, t)}{\text{BU}_{\text{ref}}} \right) \quad (4.17)$$

For nuclides with relatively long half-lives, i.e., small λt, the general scaling equation is given by Eq. (4.18) with different forms of G(Γ, t) for case (i) and (ii) as shown in Eqs. (4.19) and (4.20), respectively. The scaling constant N_{ref}/G((Γt)_{ref}) is determined by numerical data. The input to these scaling equations are the ratios Γt/(Γt)_{ref} and t/t_{ref}. The absolute value of λ is replaced by the parameter τ = λ t_{ref} so that we can treat t in the ratio t/t_{ref}. The subscript "ref" in Eqs. (4.17) and (4.18) for case (ii) denotes "ref1".

$$N(\Gamma, t) = \frac{N_{\text{ref}}}{G((\Gamma t)_{\text{ref}})} \left(\frac{\Gamma t}{(\Gamma t)_{\text{ref}}} \right) G(\Gamma, t) \quad (4.18)$$

$$G(\Gamma, t) = \frac{\text{BU}(\Gamma, t) - e^{-\tau \left(\frac{t}{t_{\text{ref}}} \right)}}{\tau \left(\frac{t}{t_{\text{ref}}} \right) + \ln[\text{BU}(\Gamma, t)]} \quad (4.19)$$

$$G(\Gamma, t) = T (1 - R) + R \left(\frac{1 - e^{-\tau_1(\frac{t}{t_{ref1}})}}{\tau_1(\frac{t}{t_{ref1}})} \right) \quad (4.20)$$

where

$$T(\Gamma, t) = \frac{\left(\frac{1 - BU_{ref1}}{BU_{ref1} - BU_{ref2}} \right) \left(\frac{\Gamma t}{(\Gamma t)_{ref1}} \right) - e^{-\tau_1(\frac{t}{t_{ref1}})}}{\tau_1(\frac{t}{t_{ref1}}) - \ln \left[\frac{1 - BU_{ref1}}{BU_{ref1} - BU_{ref2}} \right] \left(\frac{\Gamma t}{(\Gamma t)_{ref1}} \right)} \quad (4.21)$$

For more than one N_p contributing to N , we can add an undetermined coefficient for each contribution as expressed by Eqs. (4.17) or (4.18). The number of numerical radioactivity data points required corresponds to the number of contributing N_p . The coefficients are then solved from this set of coupled linear equations relating the code data and values evaluated by the scaling equations.

4.1.5.4 Application To HT-9 First Wall

The validity of this scaling equation approach is examined by applying the scheme to calculate the activation inventories impacting the environmental and safety characterization, for a HT-9 first wall. The results are then compared with the numerical data from neutronics and activation computer codes.

A water-cooled reactor configuration is used to obtain the neutron spectrum. The reference operating condition is chosen as $\Gamma = 3 \text{ MW/m}^2$ and $t = 1 \text{ yr}$. For the case requiring 2 reference points, the second one is given by $\Gamma = 3 \text{ MW/m}^2$ and $t = 2 \text{ yr}$. The comparisons are made with 4 operating conditions, namely, the combinations of $\Gamma = 1$ or 5 MW/m^2 and $t = 0.5$ or 3 yr . The values are selected to represent the likely extreme ranges of reactor design.

The comparisons for the BU parameter and the radioactivity are shown in Tables 4.5 and 4.6, respectively. For the burn-up of target nuclides (Fe-54, Fe-56, Nb-93, Ag-107, Ag-109, and W-186) the agreements between the numerical and scaling results are excellent. The build-up of Mn-55, Co-59, Re-185, and Re-187 is due to the β -decay of Fe-55 ($T_{1/2} = 2.69 \text{ yr}$), Fe-59 (44.56 days), W-185 (75.1 days), and W-187 (23.85 hr), respectively. The BU values for Re-185

and Re-187 are large because their initial inventories are small compared with those produced by the two-step reactions from W-184 and W-186.

For large $T_{1/2}$, N_d may have not attained its saturation value in the time range considered, and the assumption of $N_d = N_{d\text{sat}}$ in the derivations becomes less valid. The accuracy of the BU estimates by scaling increases with shorter $T_{1/2}$ as shown by these 4 nuclides with decay production. Higher values and variations of BU occur at larger operating conditions of Γ and t . Hence, the deviations between the code and scaling results are relatively larger at the peak values of Γ and t .

The accuracy of evaluating the inventories of the activation products by the scaling scheme basically follows the accuracy of their corresponding BU factors. The scaling calculations for Mn-54 and Mn-56 involve both the build-up of Mn-55 and burn-up of Fe-54 and Fe-56, respectively. The nuclide Ag-108m is calculated by combining two burn-up processes involving Ag-107 and Ag-109.

Similar results are obtained for the neutron spectrum of a gas-cooled configuration. The comparisons indicate satisfactory agreements between the numerical code and scaling results for both the BU parameter and activation inventory for most nuclides. For the production of parent nuclides N_p with time-varying N_a or N_d , the accuracy suffers according to the degree of time variations.

4.1.6 SUMMARY

This section has described an exploratory study to devise a fast-running computational method by employing scaling equations to calculate radioactive inventories, from which various figures of merit can be readily determined that characterize environmental and safety features of reactor designs.

We have shown, for the case of a silicon carbide first wall that a scaling law fit to a numerically-generated database can capture the neutronics information needed to compute waste and safety indices such as those used by ESECOM to evaluate the environmental and safety characteristics of fusion reactor designs. Specifically, in Figs. 4.1-4.3, three radiological dose indices calculated from the scaling law were found to agree well with those calculated directly from the data used to generate the scaling law. Given its simple form, such a scaling law could be

incorporated in existing design codes to provide very fast on-line assessment of the environmental and safety impact of the design.

To make practical use of the method, it will be necessary to build up a library of scaling laws representing different system components and their couplings for a representative array of materials and geometries. In principle, such a library could serve as an environmental code module enabling the system code to search the library for best solutions as part of an overall optimization procedure.

We have also investigated the activation characteristics of HT-9 ferritic steel as structural material in a tokamak fusion reactor. The activation-induced radiological doses due to the radioactive inventories are relatively higher than those of SiC, implying their significant impacts on fusion environmental and safety consequences. Out of over 100 activation products, we have identified 8 major radionuclides that contribute to over 90% of the radiological doses used in the ESECOM report. This simplification makes the implementation of the scaling equation approach at a manageable level.

The scaling equations for the radioactivity have different regression coefficients for various discrete choices of reactor designs. The variations in alloy compositions and impurity contents have been studied. We have installed an exponential decay behavior for the spatial attenuation of the induced radioactivity for the relatively large size of the blanket.

We have not generated enough numerical activation data to cover a continuous typical reactor design window for ferritic steel structure reactors at this time. The feasibility of such an approach based on sufficient data is demonstrated in Ref. 4.10 and it is very likely the same can be done with the steel structure.

However, we have formulated a scaling-equation scheme regarding the selection of operating conditions in fusion power level and irradiation life time. The scheme utilizes closed-form equations, with one or two reference data points for each relevant radionuclide supplied from neutronics and activation codes, to scale the activation inventories for various operating conditions for a given reactor configuration.

The scheme has been applied to calculate the activation inventories of the dominant radiological dose-contributing nuclides for a HT-9 first wall. Reasonably good agreements with the numerical code results have been observed. Higher order corrections for relevant nuclides in the

reaction chains are still needed to extend the validity range of the scaling equations. The applicability of this scheme to other relevant reactor materials also needs to be explored.

4.2 REDUCTION OF RUNNING TIME FOR EXISTING CODES

4.2.1 STUDY GOALS AND SCOPE

This part of the work involved the investigation of ways to reduce the computational time required for getting inventories of radionuclides. The goals of this study are:

- (i) To determine typical calculational times of nuclide inventories using code systems in present use by the fusion community, and
- (ii) To investigate possibilities for reducing nuclide inventory calculational time in case some accuracy can be sacrificed.

The code systems, computers, and data libraries considered are as follows.

- The code systems:
 - The Monte-Carlo code TART [4.3]
 - The one-dimensional transport codes, ANISN [4.11] and ONEDANT [4.12]
 - The activation codes DKR-ICF [4.13] and FORIG [4.5]
- The computers:
 - The IBM 3090 Main Frame at UC Berkeley
 - The National Energy Research Super Computer (NERSC) A- and F- Crays
 - The Sun Sparc2 and Sparc10 at UC Berkeley
- The cross section libraries
 - The DABL69 [4.14] cross-section library
 - The University of Wisconsin's 67 group library
 - The ACTL [4.6] activation cross section library
 - The ENDL [4.15] cross-section library.

Two reference fusion reactor models were used for the calculations. Most of the computations so far were done for a simplified model (Model 1) of the low activation reactor designed in the ARIES project [4.16]. It features SiC as the structural and primary shield material. This simplified, one dimensional (1-D) cylindrical model consisted of a 161 cm plasma (i.e., fusion neutron source) radius followed by 9 cm of vacuum; 1.2 cm SiC first wall (FW); 20 cm solid breeder blanket (B); 10 cm SiC with Be reflector (1R); 68 cm SiC reflector/shield (2R); and

50 cm shield (S) of SiC with B₄C. This model has been used in other studies of the Berkeley Fusion Environmental and Safety Group.

The model for numerical calculations consisted of six zones; each zone represented a different material composition as follows:

- Zone 1, intervals 1 - 29, fusion plasma and vacuum
- Zone 2, interval 30, FW
- Zone 3, intervals 31 - 41, B
- Zone 4, intervals 42 to 59, 1R
- Zone 5, intervals 60 to 79, 2R
- Zone 6, intervals 80 to 99, S.

Another simplified model (Model 2) of the updated version [4.17] of the low activation reactor designed in the ARIES-I project was used for a later part of the studies (Section 4.3). It also features SiC as the structural and primary shield material, has a thicker lithium blanket, and an HT-9 vacuum vessel.

4.2.2 CONVENTIONAL APPROACHES FOR INVENTORY CALCULATIONS

4.2.2.1 The LLNL Approach: TART-ORLIB-FORIG

This is the computational system developed and used by the LLNL fusion programs. It has been adopted by our group and used for most of the nuclide inventory calculations so far. The neutron transport problem over all the zones in the reactor can be solved by a single run of TART [4.3], whereas the ORLIB [4.4] and FORIG [4.5] codes, which are used to process cross sections and to calculate inventories, respectively, can solve for one zone at a time. TART utilizes the ENDL [4.15] pointwise transport cross sections; ORLIB accesses the ACTL [4.6] pointwise activation cross section library. The computational time required for the reference fusion reactor consists of one run of TART and then successive runs of the ORLIB and FORIG codes for each of the five zones where activations were obtained. This procedure takes about 227 seconds on the F-Cray, as shown in Table 4.7. In addition, preparation of input files for the ORLIB and FORIG codes is required in between runs. This particular example calculated inventories for about 150 nuclides in five zones: first wall, blanket, two reflectors, and the shield (See Section 4.2.1, Model 1).

4.2.2.2 The ANISN or ONEDANT with DKR-ICF Approach

The other widely used method to calculate inventories is to couple a discrete ordinates transport code with an activation code. Two one-dimensional transport codes in wide use today are ANISN and ONEDANT. These types of codes run many times faster than Monte-Carlo codes. In the fusion community, the DKR-ICF activation code is being used with either ONEDANT or ANISN as one method to calculate nuclide inventories in fusion reactors.

Activation codes can use numerical or analytical methods; FORIG utilizes numerical methods and DKR-ICF utilizes analytical methods. It was felt that an activation code that utilizes an analytical method would probably run faster than FORIG. Therefore, combining a discrete ordinates transport code, such as ANISN or ONEDANT, with the DKR-ICF code showed the most promise of providing fast runtimes. DKR-ICF, ONEDANT, a sixty-seven group transport cross section library, a forty-six group ACTL activation cross section library, and a FORTRAN code to convert ONEDANT (or ANISN) fluxes into DKR-ICF input format were obtained from researchers at the University of Wisconsin.

Inquiries of typical running times for fusion applications using either the ONEDANT/DKR-ICF or the ANISN/DKR-ICF combinations did not provide definite answers. Nevertheless, the general opinion was that ONEDANT runs significantly faster than ANISN. Consequently, it was felt necessary to clarify these issues as part of the present work. The clarification of these issues is the subject of this section.

4.2.2.2.1 Transport Codes

ANISN and ONEDANT are one-dimensional (1-D) discrete ordinates transport code systems, allowing for anisotropic scattering. They solve the 1-D Boltzmann transport equation for neutrons or gamma rays in slab, sphere, or cylinder geometry. Unfortunately, the ANISN and ONEDANT codes acquired for our group operate on different platforms. This complicated the running time comparison, making it necessary to do multiple runs on different platforms. ANISN was run on the IBM 3090 Model 300J which uses the VM/XA CMS operating system. Some equivalent ANISN runs were made on the NERSC F- and A- Machines at LLNL because ONEDANT was only available on the F-Cray. The F Machine is a Cray 2S/4128 running UNICOS 7.0; the A Machine is a Cray 7-MP C90/16256, running UNICOS 7.C. The ANISN runs were repeated on the A-Cray which is (about three times) faster than the F-Cray to see the limits of runtimes that could be reached with the technology available to us today.

4.2.2.2.2 Cross Section Libraries

The DABL69 cross section library has forty-six neutron groups and twenty-three photon groups. The twenty-three photon groups were eliminated from the library for this study, and the library was modified by the addition of positions to hold activation cross sections for selected nuclides. The use of these additional cross sections will be addressed in later sections. DABL69 is based on ENDF/B-V [4.18] cross sections, except for the iron cross section which is not an official ENDF/B-V evaluation. The new iron data [4.19] has been shown to provide substantially better agreement with integral experimental data. A choice of cross sections is available which have been collapsed with three different weighting functions from the VITAMIN-E [4.20] 174-group library. The group constants used for this study were obtained using the standard weighting function consisting of a 300 K Maxwellian, a 1.4 MeV fission spectrum and a 14.07 MeV fusion peak spectrum overlaid on a $1/E$ slowing-down spectrum.

The other cross-section library is the University of Wisconsin "Sixty-Seven Group" transport cross-section library. This sixty-seven group library has forty-six neutron groups and twenty-one photon groups. It is also based on the VITAMIN-E library which is based on ENDF/B-V evaluations.

4.2.2.2.3 Activation Code

The DKR-ICF code utilizes the chain method to calculate radioactive inventories. In this method, every chain of reactions that can lead to the presence of a particular nuclide is independently evaluated for its contribution to the end nuclide. The burnup of the parent is always included; there are no branching chains. The sum of the results from each chain then represents the amount of the final nuclide present. The number of reactions in a chain can be controlled by setting in the input file the ratio of the last nuclide found to the one before it. That ratio is typically set to around $1E-4$. The other method that controls the number of reactions in a chain is to stop the chain after the a preselected number of stable nuclides is reached. Since it would be too time consuming to determine for every possible isotope, at every interval in the system, the number of chains that should be considered to calculate the inventory, a reference case is run first with a reference flux and a reference time which represent an approximation to the flux and time in the real problem. Using this reference case determines a priori how many chains will be considered to add to the production of each final nuclide when calculating the inventories.

The flux outputs from ANISN or ONEDANT can be converted to DKR-ICF input format using a small code called 1DSCALAR [4.21]. At present, this process of interfacing between the transport and activation code is not automated, but remains a possibility for the future. The DKR-ICF code uses activation cross sections from the ACTL library which have been collapsed into forty-six groups; the ACTL cross sections' group structure matches the group structure of the UW cross section library.

4.2.3 COMPUTATIONAL RUNNING TIMES

4.2.3.1 ANISN Versus ONEDANT Running Times

The evaluation of the differences in runtimes between the ANISN and ONEDANT codes was done as follows:

1. finding the fastest run time for ANISN that provides some minimal sacrifice in accuracy by varying the S_n , P_l , transport orders, convergence criteria, and the differencing scheme;
2. comparing runtimes between ANISN and ONEDANT running at the preferred transport order and convergence order as found in Step 1.

The accuracy of the lower transport order and reduced convergence criteria solution was judged by comparing group fluxes and/or selected activation-rates calculated with this approximation across a reference system, against the corresponding values calculated using the "exact" solution; i.e., the solution obtained with the highest transport orders and tightest convergence criteria.

The DABL69 library, which was modified by the addition of six positions to hold activation cross sections for selected nuclides, was used for this study. The activation cross sections were obtained in the group structure of the DABL69 library by weighting ACTL library cross sections with fluxes generated with the Monte Carlo code MCNP.

Twenty-three different activations were examined. They were grouped in three categories according to the energy dependence of their cross sections: (a) cross sections with high-energy thresholds, $E_{th} > 10$ MeV; (b) cross sections with mid-energy thresholds, $1 \text{ MeV} < E_{th} < 10 \text{ MeV}$; and (c) low-energy (including $1/v$) cross sections. In addition, the ${}^6\text{Li}$ and ${}^7\text{Li}$ absorption cross sections were included, making a total of twenty-five reactions. Both zone-averaged and space dependent activations were evaluated.

4.2.3.1.1 Optimizing the S_n , P_l , and Convergence Criteria Setup of ANISN

The "exact" flux and activation-rate distributions were first determined by solving with ANISN the all-transport equation for the Model-1 reference geometry of Section 4.2.1 using the S_{16} - P_5 transport approximation and .00001 convergence criterion. The CPU time required for this run was 44.4 seconds.

Repeated runs were made reducing the S_n and P_l orders and the convergence criterion. The fastest solution with acceptable accuracy was obtained using S_8 , P_3 , .01, respectively. The zonewise activation rates were all within a fraction of a per cent, excluding a couple of high threshold reactions which, in the shield, deviated by 1.6 per cent (see Fig. 4.6a). The pointwise activation rates are calculated to within 2 per cent accuracy through the system except for the shield. The maximum deviation in the shield is 9 per cent. The CPU running time for the S_n order, the P_l order and the convergence criterion set of " $S_8 / P_3 / .01$ " was only 5.8 seconds. The IFLU=0 option (see Section 4.2.3.1.2) and ninety-nine interval mesh were used for these calculations.

4.2.3.1.2 Setting the Differencing Scheme for ANISN

The "IFLU" parameter in ANISN selects which differencing scheme to be used for the solution of the discrete ordinates equation. The following choices are available for IFLU:

- 0 - step model used when linear extrapolation yields negative flux (mixed mode),
- 1 - use linear model only,
- 2 - use step model only,
- 3 - weighted difference model,
- 4 - linear model, weighted model if negative.

For a comparative time study, it was important to use the fastest time on ANISN that gave the most accurate flux profile. ONEDANT utilizes diamond differencing with synthetic diffusion acceleration [4.22]. The IFLU=1 option in ANISN provides the most accurate solution while using the diamond differencing scheme as in ONEDANT. Differences exist in the ANISN runtimes for each of the five differencing schemes used, as do slight variations in the fluxes. ANISN was run for each of the five options and the fluxes were plotted against ONEDANT fluxes for the same input files. Group-wise fluxes obtained using IFLU=3 had longer running time and varied by as much as ten per cent from IFLU=1 solutions. Running ANISN for the reference geometry with the original mesh of 99 intervals, using IFLU=1, resulted in excessive iterations (> 100) for one

energy group at one interval. Using IFLU=0 brought the iterations within that group down to an average value of 3-6 (.01 convergence criterion) or 8-13 (.00001 convergence criterion) for all intervals and improved the runtime by a slight margin. Group-wise fluxes obtained using IFLU=0, 2, and 4 were obtained in very close to the same time. The IFLU=0 option was selected for the following part of this study. The mesh used during the first part of the study, and to be considered as the default mesh, had ninety-nine intervals. In the later part of the study, we used a more optimal mesh having seventy-two intervals.

Both ANISN and ONEDANT were run for the Model-1 reference geometry, with the Sg / P3 / .01 mode. The same forty-six group DABL69 library was used with both codes. ANISN and ONEDANT ran at just under six seconds of CPU time. No increased speed of ONEDANT over ANISN was noted for this type of fusion reactor problem. The finding that ANISN ran as fast as ONEDANT was surprising since ONEDANT utilizes diffusion synthetic acceleration which should help to make it converge faster. It should be stated, however, that for problems using a tighter convergence criterion, such as .0001, and probably higher S_n and P_l orders, ONEDANT may start to have its time advantage over ANISN. This assumption is evaluated in a later section (Section 4.2.4.1.3).

4.2.3.2 DKR-ICF Running Times

Since there was hardly any detectable difference in the CPU times for ANISN and ONEDANT, we ran ONEDANT and interfaced the flux output to DKR-ICF using the flux conversion program 1DSCALAR. The UW sixty-seven group library was used instead of the DABL library because the group structure of the UW library matches the group structure of the ACTL library utilized by DKR-ICF. DKR-ICF requires only the forty-six neutron group flux spectrum as input, so running ONEDANT with the sixty-seven group library will not affect DKR-ICF runtimes. This procedure was done on the F-Cray to calculate inventories for the reference problem defined in Section 4.2.1, the same problem that had been modeled using TART-ORLIB-FORIG (to be referred to as TOF).

The CPU time to run DKR-ICF was approximately nine seconds for each zone. This time, though longer than FORIG, gives much more in-depth information than FORIG in that it calculates inventories over every interval. The number of intervals in the reference design varied from one interval per zone for the first wall, to about thirty intervals per zone in the second reflector. There were a total of ninety-nine intervals in all zones. Because of an error received when trying to run more than one zone at a time, DKR-ICF was run separately for each zone. This error message is

associated with the allocation of memory for calculations. We have been informed [4.23] that it can be easily changed in the source code to allow DKR-ICF to be run for the entire system in one run. DKR-ICF was not yet modified to eliminate this error message. We do not know yet the time it would take to run the DKR-ICF code for the entire system. It is not necessarily the number of zones times the run time of one zone, because it is likely that a great deal of the DKR-ICF run time is overhead associated with input and calculating the reference number of chains to be used for each nuclide. Our best estimate now is that DKR-ICF could probably be run in approximately thirty seconds for all zones and intervals for this reference case. DKR-ICF was typically calculating about twenty-five reactions per zone.

4.2.4 TRADING TIME WITH ACCURACY IN FLUX CALCULATIONS

Section 4.2.3.1.1 discussed the selection of the S_n , P_l , and convergence criterion for the ANISN and ONEDANT codes in sacrificing up to one per cent in the accuracy of the calculated activation rates. Here, we explored possibilities to further reduce the quadrature order, scattering order, and convergence criterion if some further acceptable sacrifice in accuracy could be traded for significant time savings. In Section 4.2.4.1, we examine the reduction of running times by using the diffusion approximation. The alternative of reducing the number of energy groups is briefly discussed in Section 4.2.4.2.

4.2.4.1 Use of the Diffusion Approximation

As the mean free path for collision of the 14 MeV neutrons is significantly larger than that of sub-MeV neutrons, it may be tolerable to use a higher order transport approximation for the higher energy groups, and a diffusion approximation for the lower energy groups. The first purpose of this portion of the study was to find whether this approach of using the diffusion approximation can provide accurately enough inventory data, and, if so, how much time it could save. The investigation was done using ANISN and ONEDANT. ANISN has an option to use a diffusion approximation for the calculation of the flux distribution in selected energy groups, while a higher order transport approximation is used to solve for the flux distribution in the rest of the energy groups.

Rather than using the diffusion approximation for lower energy groups, it is possible to use diffusion theory formulations to accelerate the convergence of the transport solution. The 1-D transport code, ONEDANT, does so using the Diffusion Synthetic Acceleration (DSA) [4.22] technique. A second purpose of this study, then, is to compare the time saving achievable using a

coupled diffusion-transport approximation in ANISN to that attainable using the DSA in ONEDANT.

The time saving attainable by setting ANISN to solve for the neutron flux in the lower energy groups using the diffusion approximation, and the corresponding accuracy, were studied parametrically; the variable parameter was the transition group from transport to diffusion calculations. A diffusion approximation for all energy groups was considered as a special case; it is expected to offer the maximum reduction in running time, along with the largest inaccuracy.

The accuracy of the coupled diffusion-transport approximation was judged by comparing group fluxes and selected activation-rates (see Section 4.2.3.1) calculated with this approximation across a reference system, against the corresponding values calculated using an all-transport representation.

4.2.4.1.1 Mixed Transport-Diffusion Calculations

Table 4.8 summarizes the running times and the sacrifice in accuracy of zone average fluxes obtained for selected transition groups in a coupled diffusion-transport solution. Results shown in the table are for transition groups 6, 10, and 20, corresponding to transition energies (lower limits) of 12.2 MeV, 8.8 MeV, and 1.4 MeV for the seventy-two interval mesh (the twenty-nine fusion plasma intervals were replaced by two intervals). Accuracy is expressed as the percentage deviation of the zone-averaged fluxes calculated using the diffusion-transport approximation from the corresponding transport-only solution in zones two through five of the reference reactor.

Figure 4.6c shows the zonewise accuracy of the activation rates for all twenty-five reactions calculated using 20T/26D, while Fig. 4.7a shows the corresponding pointwise accuracy for the low energy reactions. Figure 4.7a shows that better than 5 per cent accuracy is obtained from the FW through the 2R; however, in the shield, large excursions occur near the 2R/S interface, while further away from the interface the reactions agree with the all-transport case to within less than 10 per cent. As the transition energy is close to 1 MeV, the zone-wise activations for the 20T/26D case in Fig. 4.6c show the mid-threshold and the high-threshold activations are precisely predicted.

The zone-wise activations for the 10T/36D case are shown in Fig. 4.6b. Figure 4.7b shows the point-wise accuracy of mid threshold activations calculated using the 10T/36D solution.

These agree to within less than 1 per cent from the FW though the 1R, but are underpredicted in the 2R and S region by up to 8 per cent.

The accuracy obtained with the 20T/26D setup is better than the requirement anticipated for safety related studies. This approximation offers 31 per cent reduction in the CPU time. For applications which would accept the zone-wise activation rates to within, say, 5 per cent accuracy, an even higher transition energy (corresponding to, perhaps, group fifteen as the transition group) will be adequate. The required CPU time will be about 2.6 seconds for this seventy-two interval model. Applying the A-Cray .33 time-correction factor gives a CPU time of less than one second.

4.2.4.1.2 All Diffusion Approximation

Relative to the all-transport ($S_8 / P_3 / .01$) calculations, the all-diffusion calculations required only 46 per cent of the CPU time, or about 2 seconds for our seventy-two interval model. The corresponding ratio of activation rates are summarized in Table 4.9. It is concluded that the diffusion approximation has the potential for significant time saving. However, it can grossly mispredict the activation rates corresponding to all threshold reactions. It appears surprising that the prediction of the space-dependent activation rates corresponding to the low energy cross-sections is as good as found. Moreover, if one is interested in obtaining zone-wise activations to within 7 per cent accuracy for only the FW, or of better than 10 per cent accuracy for the low energy activations anywhere in the system, then the diffusion approximation would provide acceptable results with remarkable time saving.

However, if one is interested in the radioactive inventories in only the first wall and blanket (which dominate the inventories in the fusion reactor as a whole), then it might be possible to save even more time by using an infinite medium approximation. This option will be discussed in a later section (Section 4.2.4.1.4).

4.2.4.1.3 Diffusion Synthetic Acceleration

The DSA appears to be an attractive technique for using diffusion theory to cut down the CPU time required for solving a transport problem because it entails no accuracy penalty whatsoever. The question, however, is whether the CPU time saving offered by the DSA approach can be comparable (or possibly better) than the saving attainable using the diffusion-transport approximation. This question was investigated by comparing the CPU time required for the

ONEDANT solution (with its built-in DSA algorithm) with that required with ANISN using the coupled diffusion-transport approximation.

The comparison was done in two steps:

1. Using the lowest transport approximation and convergence criteria found (in Section 4.2.3.1) acceptable for our fusion safety applications.
2. Studying the dependence of the required number of iterations and CPU time on the S_N - P_1 orders and on the convergence criteria.

Table 4.10 summarizes the obtained results. It was found that for S_N - P_1 orders higher than S_8 - P_3 , and/or for convergence criteria tighter than 0.01, the convergence of ONEDANT requires fewer number of iterations and less CPU time. Thus, the relative efficiency of ONEDANT increases for high transport orders and tight convergence criteria. For example, with the S_{16} - P_5 / 0.00001 running mode, ONEDANT requires less than one third of the CPU time as ANISN. The ONEDANT CPU time is even significantly shorter than that required by ANISN using a diffusion-transport approximation. On the other extreme of S_2 - P_1 / 0.01, ONEDANT requires twice the CPU time as ANISN. For our fusion safety applications, applying ANISN with the diffusion-transport mode is more time efficient than ONEDANT; it offers approximately 30 per cent reduction in the CPU time.

4.2.4.1.4 Infinite Medium Approximation

In general, if the fusion reactor blanket is not designed to have a significant neutron multiplication, the blanket must be thick enough to reduce the fraction of neutrons leaking out to a few per cent to assure an effective tritium breeding ratio of at least 1.0. In such a case, if one is interested in the total inventory of different radioactive materials integrated over the blanket volume, it might be possible to get a reasonable estimate for these inventories by representing the blanket as an infinite medium and solving the neutron slowing down, rather than the transport equation. The blanket of our reference fusion reactor does not satisfy this condition because it contains a large volume fraction of beryllium (a very effective neutron multiplier) which enables the design to be as thin as 20 cm (and still provide ample tritium breeding). Approximately 30 per cent of the neutrons leak out from this thin blanket (and are being absorbed in the reflectors and shield).

When applicable (i.e., for low leakage blankets), the procedure for applying the infinite medium approximation is as follows:

- Run ANISN at S8 / P3 / .01 for a problem consisting of the plasma, the first wall, and the blanket (or part of it) only to obtain:
 - the per cent of neutrons absorbed in the first wall, and
 - the effective source of neutrons for the blanket.
- Run ANISN with the blanket modeled as an infinite medium, using the effective source of neutrons found above. The activation rates obtained from this run are expected to be a reasonable approximation to the corresponding activation rates calculated for the real blanket.

The validity of this approximation, and the time savings it can offer, will be investigated later.

4.2.4.2 Reducing the Number of Energy Groups

Previous work [4.24] described using an option in ANISN to collapse cross sections from the sixty-nine groups in DABL69 to three neutron energy groups and one photon group. The resulting four-group library was run on the F-Cray using the S8 / P3 / .00001 mode in .18 seconds. The accuracy was judged by comparing the total absorptions over all energy groups, in each zone, for five materials. More work is required to explore the feasibility of this option.

4.2.5 TRADING TIME WITH ACCURACY IN INVENTORY CALCULATIONS

A transport code must be interfaced with an inventory code to obtain inventories of radionuclides of relevance for safety evaluation. Section 4.2.2 had discussed the use of the Monte Carlo transport code TART, interfaced with the cross-section averaging code ORLIB to run the activation code FORIG. That combination took 227 seconds to calculate the inventories of approximately 150 nuclides in five zones of the reference geometry. Likewise, ANISN or ONEDANT must be interfaced with an activation code. For this study, we interfaced ONEDANT with DKR-ICF. The runtimes of DKR-ICF were discussed in Section 4.2.3.2 and were estimated to be about thirty seconds for the reference geometry of five zones for about twenty five nuclides per zone. Interfacing DKR-ICF with ONEDANT at S8 / P3 / .01 would add 5.5 seconds, for a total of 35.5 seconds to obtain the final inventories. This time is clearly too long for direct application in a systems code module.

We have discussed in Section 4.2.4 some ways to reduce the transport runtimes. This section will discuss the evaluation of methods to reduce the runtimes for the transmutation portion of the problem as follows:

- Determine if and how the DKR-ICF code could be applied in a mode which will reduce runtimes.
- Determine if approximations could be developed to estimate inventories significantly faster than even with the optimized mode of application of DKR-ICF. It is thought that a small code could be written to evaluate the expressions for the approximations in fractions of a second.
- Implant activation cross sections into the transport cross section library so that inventories for radionuclides with very short or very long half-lives can be obtained as part of the transport code run.

4.2.5.1 Application Modes Of DKR-ICF

The chain method utilized by the DKR-ICF code for radionuclide inventory calculations was described in Section 4.2.2.2.3. The runtime of nine seconds for one zone, or up to thirty seconds for five zones (thirty seconds is the estimated time if inventories for all zones were calculated in one run instead of in a separate run for each zone) for DKR-ICF seems a little long, considering that DKR-ICF uses analytical methods to solve essentially the Bateman [4.25] equation. It is theorized, therefore, that most of this time must be due to the solution of the reference case which must evaluate for each end-product nuclide, the number of nuclides in a chain and the total number of chains that should be considered. It is thought possible that for particular material/geometry combinations, this sort of information could be decided off-line, prior to the run of a systems code. Since DKR-ICF prints out the results of the nuclide inventories for every possible chain that could contribute to the final inventory, it is easy to review this list from a previous run and determine which chains can be discarded from the production runs to follow because their contribution is negligible. In this way, only the chains which had a non-negligible contribution to the inventory of a final nuclide need to be selected for calculation by DKR-ICF in the systems code. In the systems code, DKR-ICF would have only to solve the analytical expressions for each of the predetermined chains, which is thought to be done in fractions of a second. Regularly, DKR-ICF will make a trial run encompassing a large number of chains and nuclides in order to select those to include in the following round of more exact calculations. This method would require that a basic FW/B/S design has been decided, and that changes evaluated in the systems code would be of a size such that the number of nuclides per chain and number of

chains per nuclide would not vary. These parameters could be predetermined off-line for a number of different FW/B/S designs under consideration.

4.2.5.2 Approximations to the Bateman Equations

If it seems infeasible to sufficiently reduce the running time of DKR-ICF by optimizing the mode of its application, another possibility to decrease the runtime for transmutation calculations is to make approximations to the Bateman equations in a special purpose code that could be developed. DKR-ICF printouts will be very useful in selecting chains providing significant contributions and discarding chains with negligible contribution.

In order to check the feasibility of this approach, we developed analytical approximations for the inventory of twenty-two isotopes. Figure 4.8 compares the inventories of these twenty two radionuclides as calculated using the approximate analytical expressions with the corresponding inventories calculated using DKR-ICF. The comparison is at shutdown, after reactor operating times of one year and thirty years. It is observed that the results agree to within 5%.

A short code could be written to perform the calculation of these approximations. It is thought that such a code could make these types of calculations in fractions of a second.

4.2.5.3 Shortcut for Short- and Long-Lived Daughter Radionuclides

A fast computational method was implemented to calculate activation-induced radioactive inventories for very short-lived and for very long-lived isotopes which have negligible transmutation cross sections. The method takes advantage of the fact that for these types of isotopes, the inventories can be estimated by readily determined constants multiplying the activation cross sections of the parent nuclides. These constants transform the activation rate of the parent nuclide to the equilibrium inventory of the activation product.

In order to assess the feasibility of this shortcut approach, we had first to generate activation cross sections for a sample of reactions in the group structure used for the transport calculations. These activation cross sections were obtained from the ACTL library using the Monte-Carlo code, MCNP, for flux weighting the ACTL pointwise cross-section. The resulting multi-group cross-sections were implanted, after being multiplied by the necessary constants, into new positions of the transport cross section library which was used as input to the 1-D transport

code, ANISN. ANISN was then run with the modified cross section library to directly output inventories for these short- and long-lived isotopes.

Inventories estimated by this methodology using ANISN were compared to those obtained using the TOF as shown in Figs. 4.9 and 4.10. Agreement of better than 6 per cent was obtained for short-lived isotopes for irradiation times of up to seven years. For longer irradiation time, the agreement became worse as depletion and buildup effects became dominant, with the largest deviation being 15 per cent for ^{12}B after thirty years of operation. For the long-lived nuclides, the ratio of the ANISN approximations to the FORIG results for thirty years operating time were about 2 per cent and 12 per cent for ^{10}Be and ^{14}C , respectively. It should be noted that only part of the discrepancy between the ANISN and TOF results is due to approximations imbedded in the effective inventory cross sections used with ANISN. The other part is due to differences in the flux distribution as calculated by ANISN and TART.

The additional step of utilizing ANISN to calculate inventories this way adds only on the order of one second to any ANISN run. This appears to be much more than should be needed for performing activation calculations, once the group fluxes are known. It appears that using S8/P3/.01 transport/convergence criteria, with a coupled diffusion-transport solution and a transition of group fifteen, ANISN could provide radioactive inventories for these special case isotopes in about one second on the A-Cray.

4.2.5.4 Other Activation Codes

Other activation codes in use by the fusion community are RACC [4.26], FISPACT [4.27], REAC [4.28], THIDA [4.29], and ANITA [4.30]. We were not able to find any comparison of the relative running times of these codes. It will be useful to have such a comparison. This is, however, out of the scope of the present work. On the other hand, it is unlikely that any of these other codes will offer a shorter running time than DKR-ICF.

4.2.6 SUMMARY

This study examined a number of approaches for reducing the computer turn-around time for calculating the inventory of activation products in fusion reactors. If the turn-around time can be smaller than about 0.1 second, it will be possible to incorporate a safety-related module in fusion reactor systems codes, thus enabling safety-related characteristics to be accounted for within an overall fusion reactor optimization study.

The best runtimes for obtaining inventories for our 1-D reference fusion reactor with no sacrifice in accuracy were found to be approximately:

- 227 seconds using the Monte-Carlo transport code TART with the activation cross-section averaging code ORLIB and the inventory code FORIG.
- 35 seconds using the discrete ordinates codes ANISN or ONEDANT with inventory code DKR-ICF, in which the flux calculations taking an average of 5.5 seconds.

Trading time for accuracy in obtaining fluxes led to the investigation of mixed diffusion-transport solutions and all diffusion solutions using ANISN with the following results:

- The coupled diffusion-transport mode of running ANISN can provide activation-rates with acceptable accuracy for fusion reactor safety applications, with a significant saving in CPU time. For the ARIES-I design used for the reference case, a transition energy of approximately 1 MeV between the transport and diffusion approximations can save about 25 per cent in CPU time relative to the conventional (optimized) transport run, with no more than 5 per cent loss in accuracy in the zone-wise activations anywhere in the system. The flux calculation takes 2.6 seconds.
- The diffusion-only approximation can save 45 per cent in CPU time relative to the conventional (optimized) transport run, and give reasonable results for all activations in the FW, and/or for low energy zone-wise activations throughout the system, for the type of system considered. However, it highly underestimates the threshold activations in outer zones and the pointwise activations of low energy reactions near boundaries.
- ONEDANT can be significantly more efficient than ANISN for problems that need be solved using a high S_N - P_1 order and/or tight convergence criteria. On the other hand, ANISN was found to be more efficient than ONEDANT for problems which can be described using a very low transport approximation and low convergence criteria.
- The Diffusion Synthetic Acceleration (DSA) technique incorporated within ONEDANT does not offer any time-saving relative to ANISN for the level of accuracy required for safety problems of interest. This finding was surprising in view of the common notion that

ONEDANT is significantly more time efficient than ANISN for solving the transport equation.

- The diffusion-transport approximation first explored (at least for fusion applications) in this work appears more effective for CPU time saving than the DSA approach.

Another method to trade time for accuracy in flux calculations was to reduce the number of energy groups. Collapsing the forty-six neutron groups of DABL69 to three neutron groups, we could calculate fluxes in 0.18 seconds. Further evaluation is required to define the precision to which the activation rates can be predicted.

Two fast methods of calculating inventories of radionuclides (using fluxes calculated with ANISN or ONEDANT) were evaluated:

- Formulating approximations to the Bateman equations for the inventories of selected nuclides. Such approximations were derived for twenty two nuclides. The accuracy for predicting these inventories in the first wall were found to agree with DKR-ICF calculations to within 5 per cent. It is assumed a computer code could be written to evaluate the approximate expressions in a fraction of a second. However, this approach will become more complicated for radionuclides which are produced in the third and higher generation.
- Implanting activation cross sections in the transport cross section library such that inventories for special (short-lived and long-lived) nuclides could be calculated in one step as part of the transport run. Although very time efficient, this approach can not be applied to radionuclides characterized by an half life which is not much smaller or much longer than the fusion reactor operating time, as well as to radionuclides which are produced in higher generations. These calculations added an additional second to the transport run. Most likely they can be done much faster.

4.3 PERTURBATION THEORY FORMULATION

4.3.1 INTRODUCTION

The purpose of the present work is to investigate the feasibility of using a perturbation theory (PT) formulation to account for changes of inventories with design iterations in the course of an optimization. The perturbation coefficients obtained by the new formulation are the gradient of the inventory of a given nuclide with respect to a change in the value of one of the design variables. The product of a perturbation coefficient and the amplitude of the change in the design variable gives the estimated inventory in the perturbed system. The new formulation is also applicable for the calculation of the change in the total dose or other safety indices resulting from the change in the inventory of any number of activation products.

When the systems code is applied to the optimization of a fusion reactor, it starts from a well-defined reference design and "marches" in many small-step iterations towards the optimal design. It is anticipated that the design modifications between the steps would be of a size enabling perturbation theory to predict resulting inventory changes to acceptable accuracy. As the design evolves far enough from the reference, a new set of perturbation coefficients will be calculated and used for the overall system optimization study.

A specific goal of this study is to find the range of applicability of the new perturbation theory formulation. This is done by considering a number of special cases.

A simplified model of the low activation reactor designed in the ARIES-I project, described as Model 2 in Section 4.2.1, was used for all numerical calculations. It features SiC as the structural and primary shield material, has a relatively thicker lithium blanket, and an HT-9 vacuum vessel. A simplified 1-D planar model represented the fusion plasma in a one centimeter interval; a two centimeter FW; a 60 cm lithium-containing B; a 112 cm SiC S, and a 10 cm HT-9 vacuum vessel (VV). The model consisted of five zones; again each zone represented a new material, as follows:

- Zone 1, intervals 1 - 2, fusion plasma and vacuum
- Zone 2, intervals 3 - 6, FW
- Zone 3, intervals 7 - 126, B
- Zone 4, intervals 127 - 350, S
- Zone 5, intervals 351 - 370, VV.

The sixty-seven group transport library and the forty-six group ACTL activation library from the University of Wisconsin were used. All computations were performed on a SUN Sparc2 or Sparc 10 Workstation.

4.3.2 FORMULATION

The derivation of the perturbation theory formulation for nuclide inventory follows the methodology of Greenspan et al. [4.31]. The new formulation differs from that of Greenspan et al. in three ways:

1. For the purpose of solving the transport equation for the neutron flux and its adjoint, it assumes that the macroscopic cross sections for the different reactor components do not vary throughout the operation of the fusion reactor.
2. The variables of the nuclide density equations are only those radioactive isotopes of relevance for the safety and environmental impact of the fusion reactor under consideration (and their parents).
3. It ignores delayed neutrons and their precursors.

All equations referenced refer to those of Appendix A. Using the simplified, abbreviated notation of Greenspan et al., the coupled set of equations for the perturbation in the neutron flux and in the nuclides inventory distribution are given by Eqs. A.1 and A.2, where N_l is the atomic density of any of the isotopes the inventory of which is of interest, while C_l is the atomic density of any of the initial l system constituents (which is assumed constant, for the sake of solving the flux and adjoint equation).

A coupled set of equations adjoint to Eqs. A.1 and A.2 is then defined by Eqs. A.3 and A.4. Multiplying each of the equations for the distribution function perturbations (Eqs. A.1 & A.2) by the corresponding adjoint functions, then, multiplying each of the adjoint equations by the corresponding distribution function perturbation, and finally, integrating each equation over phase space and subtracting one set from the other, we obtain the, so called, importance balance condition. Imposing the boundary, time and source conditions of Eqs. A.5 - A.11, the importance balance condition reduces to the form of Eq. A.12. Assuming further that in Eq. A.12, $\delta S = 0$ and that $\delta C_l = \alpha_k C_k \delta(r \in r_k)$ for $t \geq t_i$, we can derive the basic perturbation theory expression for estimating the effect of design variations on the inventory of radioactive nuclides as:

$$\int_{(I \in I_S)} dI \delta N_S(t_f) = \alpha_k PC_{kS} \quad (4.22)$$

where PC_{kS} will be referred to as the "Perturbation Coefficient." The LHS of Eq. 4.22 represents the indirect effect of the perturbation on the total inventory of radionuclide N_S at a pre-selected desirable time (t_f), in a given zone (interval, region, the entire reactor etc.) defined by $(I \in I_S)$. The RHS offers a convenient formulation for calculating the indirect effect of the perturbation due to any combination of variations (as long as they are of a small enough amplitude to make the perturbation theory approximation valid) in the design of the system, as represented by variation at beginning of life ($t = t_i$) of the density C_k of a given constituent of the reference system in any zone ($I \in I_k$) in the system. α_k is the fraction of C_k , the initial concentration, which is being changed.

By feeding a set of the perturbation coefficients PC_{kS} to the systems code, along with the set of radioactive inventories for the reference system, N_S^0 (both sets of data are to be generated ahead of time; i.e., before their use in the systems code), the systems code can obtain an estimate of the inventory of radionuclides due to a perturbation of $\alpha_k C_k$ in the value of C_k by calculating $N_S = N_S^0 + \alpha_k [N_S^0 + PC_{kS}]$. The $\alpha_k N_S^0$ term is the direct effect of the perturbation; it is non zero only when N_S^0 happens to be produced from activation of the specific system constituent C_k (otherwise, it is zero). The term $\alpha_k PC_{kS}$ is the indirect effect of the perturbation.

4.3.3 APPLICATION PROCEDURE AND COMPARISON CRITERIA

A preliminary investigation of the validity and range of applicability of the new PT formulation has been initiated. This investigation is being done by applying this new formulation to a number of simple problems which represent a variety of situations to be encountered in fusion reactor safety analysis.

So far we concentrated on establishing the validity of the tools and procedures outlined in the preceding sections. This was done by applying these tools and procedures to the simplest possible types of inventory calculational problems, in which the depletion of the parent nuclides can be ignored.

The application of the inventory perturbation theory involves the following steps:

- a) solving the time-independent transport equation for obtaining the neutron flux distribution and flux-averaged activation cross sections,

- b) solving the time-dependent inventories of relevant nuclides in the reference system,
 - c) solving the time dependent inventory adjoint equations for the relevant nuclides,
 - d) calculating the source term for the adjoint flux equation,
 - e) solving the time-independent adjoint equation, and
 - f) calculating the bilinear integrals (or perturbation coefficients) for the effect being investigated.
- The flux, adjoint, and cross sections pertaining to the design variations are used for these calculations.

The one-dimensional discrete ordinates transport code ANISN, a module of SWAN [4.32], is used to obtain the flux and adjoint flux in steps (a) and (e) above. The module SWIF of SWAN then calculates the bilinear integrals of step (f) using input (hand calculations at this stage) from steps (b), (c), and (d).

The validity of the formulation, procedure and tools used for the perturbation theory calculations was assessed by comparing the indirect effect of a change in design variable on selected inventories (for definition of indirect and direct effect, see Section 4.3.2). The direct effect of the perturbation can be calculated accurately and straightforwardly and is therefore excluded from the comparison. The perturbation theory prediction of the indirect effect of the perturbation was obtained with SWAN as described above. The exact indirect effects (against which the perturbation theory results are being compared) are obtained from the difference between the inventory calculated with ANISN for the perturbed system and that calculated with ANISN for the reference (unperturbed) system.

The only type of design variables considered in this work is the initial concentration of the fusion reactor material constituents. These can also represent a change, or an uncertainty, in a zone thickness. The perturbation theory formulation can, nevertheless, be applied to other design variables as well. Results to date were obtained for density changes of reactor materials comprising the first wall, blanket and shield from a fraction of a per cent up to two times the original density

The performance characteristics considered in this assessment are the inventories of radioactive nuclides. The nuclides selected for the assessment represent cross-sections (for the generation of these nuclides) which exhibit a wide variety of energy dependent cross-sections - from $1/v$ like dependent to high energy threshold reactions. In the following section, we report upon two of the activation products studied so far.

4.3.4 ILLUSTRATIVE RESULTS

4.3.4.1 Tritium Production from Li-6 (n, α) Reaction

The first illustration is the tritium production rate from neutron capture in ^6Li in the blanket of the reference fusion reactor. This reaction closely represents activation exhibiting a $1/v$ cross-section. The inventory of the resulting radionuclide is, in many cases, proportional to the activation rate. The design variables considered were of the concentration of Be, Si, and C in the blanket.

Figure 4.11 shows the ratio of the indirect effect calculated by SWAN compared to the exact indirect effect calculated by ANISN. Some roundoff errors occur for very small density increases of the C in the blanket, which account for the oscillation of SWAN/ANISN ratio in that location. These roundoff errors are errors in the calculation of the "exact" result; they are due to the very small difference between the reaction rates in the reference and perturbed system calculated with ANISN. It is observed that the perturbation theory predictions are accurate for small density variations. For example, for a density change of 10 per cent in the Be, C, and Si, the SWAN/ANISN ratios are 1.006, 1.012, and 1.067, respectively. The inaccuracy in the perturbation theory prediction increases almost linearly with the amplitude of the perturbation. It should be realized, nevertheless, that for fusion reactor optimization application, a 10% change in design variables is more than is necessary for a single iteration.

4.3.4.2 Al-28 Production from Si-28 (n,p) Reaction

The reaction used for this illustration has a high energy threshold. We consider the ^{28}Al production rate in the blanket due to an increase in the concentration of C and Si in the blanket. Figure 4.12 shows that the perturbation theory predictions are of very good accuracy up to relatively large (more than 20%) composition variations. Even for a density change of 50 per cent, the SWAN/ANISN ratio is 1.026 and 1.070 for the C and Si cases, respectively.

4.3.5 SUMMARY

A new formulation of perturbation theory, which should enable to predict the effect of changes in fusion reactor design variables on the inventory of any nuclide of interest for safety analysis at any given time after a prescribed operating time, has been derived. If found accurate enough over a reasonable range of variation of the design variables, this perturbation theory formulation can provide a very fast and efficient way to account for safety related characteristics

within an overall fusion reactor optimization study. In fact, it appears that of all the computational approaches examined, only the perturbation theory based methods can predict inventory changes in less than 0.1 second as desirable for systems code applications.

So far we have been investigating the feasibility of a special case of the above mentioned perturbation theory - changes in activation rates, not accounting for the depletion of the parent nuclide. This special case is, nevertheless, of practical interest in itself. It can be used to replace two out of the three phases in the conventional process used for inventory calculations, namely, the phase of solution of the flux equation (the TART phase of Section 4.2.2.1) and the phase of calculating an effective one group activation cross sections (the ORLIB phase of Section 4.2.2.1). It will have to be supplemented by the phase of solution of the Bateman equations or some approximations to these equations (the FORIG phase of Section 4.2.2.1). Based on the limited experience gained so far it appears that this perturbation theory for reaction rates and its implementation tools are valid. Hence, it appears justified to continue to explore the feasibility of the more general perturbation theory for nuclide inventory.

Finally, it should be realized that although we have limited our consideration to the prediction of the inventory of specific radionuclides, the new perturbation theory can be applied to predict the effect of design variations on integral safety characteristics, such as total activity, dose, safety indices, etc., in specific zones of the fusion reactor.

REFERENCES

- 4.1 S.K. Ho, Fusion Technol. 19, 1368 (1991).
- 4.2 J.P. Holdren, et al., "Exploring the Competitive Potential of Magnetic Fusion Energy: The Interaction of Economics with Safety and Environmental Characteristics," Fusion Technol. 13, 7 (1988).
- 4.3 E.F. Plechaty and J.R. Kimlinger, "TARTNP: A Coupled Neutron-Photon Monte Carlo Transport Code," Lawrence Livermore National Laboratory, Report UCRL-50400 (1976).
- 4.4 J.A. Blink, R.E. Dye, and J.R. Kimlinger, "ORLIB: A Computer Code that Produces One-Energy-Group, Time and Spatially Averaged Neutron Cross Sections," Lawrence Livermore National Laboratory, UCRL-53262 (1981).
- 4.5 J.A. Blink, "FORIG: A Computer Code for Calculating Radionuclides Generation and Depletion in Fusion and Fission Reactors," Lawrence Livermore National Laboratory, UCRL-53633 (1981).
- 4.6 M.A. Gardner and R.J. Howerton, "ACTL: Evaluated Neutron Activation Cross Section Library--Evaluation Techniques and Reaction Index," Lawrence Livermore National Laboratory, UCRL-50400 (1978).
- 4.7 E.T. Cheng et al., "International Fusion Activation Calculation Comparison Study," TSI Research, Report TSIR-12 (1991).
- 4.8 D.L. Smith et al., "Blanket Comparison and Selection Studies," Argonne National Laboratory, Report ANL/FPP-84-1, (1984).
- 4.9 S.A. Fetter, "Radiological Hazards of Fusion Reactors: Models and Comparisons," PhD Thesis, University of California, Berkeley (1985).
- 4.10 S.K. Ho, F. Brechtel, and T.K. Fowler, Fusion Technol. 23, 321 (1993).
- 4.11 W.W. Engle, "ANISN: A One-Dimensional Discrete Ordinates Transport Code System with Anisotropic Scattering," CCC-254, RSIC Computer Code Collection, Oak Ridge National Laboratory (1967).
- 4.12 R. Douglas O'Dell, F. W. Brinkley, Jr., D. R. Marr, and R. E. Alcouffe, "ONEDANT, A Code Package for One-Dimensional, Diffusion-Accelerated, Neutral-Particle Transport," Los Alamos National Laboratory, LA-9184-M (1982).
- 4.13 D. L. Henderson and O. Yasar, "DKR-ICF: A Radioactivity and Dose Rate Calculation Code Package," University of Wisconsin, UWFD-714, Vol. 1 (1987).
- 4.14 D. T. Ingersoll, R. W. Roussin, C. Y. Fu, and J. E. White, "DABL69: A Broad-Group Neutron/Photon Cross-Section Library for Defense Nuclear Applications," Oak Ridge National Laboratory, ORNL/TM-10568 (1989).

- 4.15 R. J. Howerton, R. E. Dye, and S. T. Perkins, "Evaluated Nuclear Data Library," Lawrence Livermore National Laboratory, UCRL-50400, Vol. 4 (1978).
- 4.16 "The ARIES-I Tokamak Reactor Study--Final Report," University of California at Los Angeles, UCLA-PPG-1323, (1991).
- 4.17 Private communication with Charles Bathke of Los Alamos National Laboratory, June 1994.
- 4.18 R. Kinzey, "Data Formats and Procedures for the Evaluated Nuclear Data File, ENDF," Brookhaven National Laboratory, BNK-NCS-50496 (ENDF 102), 2nd Edition (ENDF/B-V) (1979).
- 4.19 C. Y. Fu and D. M. Hetrick, "Update of ENDF/B-V Mod-3 Iron: Neutron-Producing Reaction Cross Sections and Energy-Angle Correlations," Oak Ridge National Laboratory, ORNL/TM-9964 (1986).
- 4.20 C. R. Weisbin et al., "VITAMIN-E: An ENDF/B-V Multigroup Cross-Section Library for LMFBR Core and Shield, LWR Shield, Dosimetry, and Fusion Blanket Technology," Oak Ridge National Laboratory, ORNL-5505 (1979).
- 4.21 1DSCALAR is a FORTRAN code for converting ANISN or ONEDANT flux into the input format for DKR-ICF, obtained from H. Khater of the University of Wisconsin.
- 4.22 R. E. Alcouffe, "Diffusion Synthetic Acceleration Methods for the Discrete Ordinates Equations," Proc. of Top. Meeting on Adv. in Reactor Computations, Amer. Nucl. Soc., Salt Lake City, March 28-31, 1982.
- 4.23 Private communication with H. Khater of the University of Wisconsin.
- 4.24 S.K. Ho, Ed., "Status Report: Code Development Incorporating Environmental, Safety, and Economic Aspects of Fusion Reactors," UC-BFE-039, University of California, Berkeley, July 1993.
- 4.25 H. Bateman, "Solution of a System of Differential Equations Occurring in the Theory of Radioactive Transformations," Proc. Cambridge Phil. Soc. 15, 423 (1910).
- 4.26 J. Jung, "Theory and Use of the Radioactivity Code RACC," ANL/FPP/TM-122, Argonne National Laboratory (1979).
- 4.27 R. A. Forrest and J-Ch Sublet, "FISPACT3 User Manual," Euratom/UKAEA Fusion Association, AEA/InTec 1293, AEA/FUS 227 (1993).
- 4.28 F. M. Mann, "REAC*2: Users Manual and Code Description," Westinghouse Hanford Company, WHC-EP-0282 (1989).
- 4.29 Y. Seki et al., "THIDA-2: An Advanced Code System for Calculation of Transmutation, Activation, Decay Heat and Dose Rate," Japan Atomic Energy Research Institute, JAERI-1301 (1985).

- 4.30 C. Ponti et al., "ANITA: Analysis of Neutron Induced Transmutation and Activation," JRC-Ispira, EUR 12622 EN (1989).
- 4.31 E. Greenspan, M. L. Williams, and J. H. Marable, "Time-Dependent Generalized Perturbation Theory for Coupled Neutron-Nuclide Problems," Nuc. Sci. and Eng., 73 (1980).
- 4.32 E. Greenspan, W. G. Price, Jr., and H. Fishman, "SWAN: A Code for the Analysis and Optimization of Fusion Reactor Nucleonic Characteristics," Princeton Plasma Physics Laboratory, MATT-1008 (1973).

Table 4.1. Coefficients for the correction factor CF_B for lithium breeder and water coolant in the blanket for C-14 and Co-60. We have $CF_B = C_0 (f_1 + f_2 h + f_3 h^2)$, where f_i takes the form $f_i = C_{i,1} + C_{i,2} li + C_{i,3} li^2 + C_{i,4} li^3$. Note that the value of C_0 for C-14 and Co-60 are 6.93 and 7.00×10^{-3} , respectively, for all range of li . Any other unlisted coefficients are zero.

Nuclide		$C_{1,1}$	$C_{1,2}$	$C_{1,3}$	$C_{2,1}$	$C_{2,2}$
C-14	$0 \leq li \leq 0.02$	1.21	-40.8	0	26.2	-575.0
	$0.02 < li \leq 0.1$	0.501	-6.30	37.50	14.9	-14.65
	$0.1 < li \leq 0.2$	0.390	-2.00	5.25	13.0	12.5
	$0.2 < li \leq 0.4$	0.110	0.60	-0.75	13.0	12.5
	$0.4 < li \leq 0.6$	0.110	0.60	-0.75	9.50	6.05
	$li > 0.6$	0.204	0	0	13.0	0
Co-60	$0 \leq li \leq 0.02$	904.77	-2.65×10^4	0	1.63×10^3	-8.75×10^4
	$0.02 < li \leq 0.04$	410.64	-2.32×10^3	7.00×10^3	147.0	-1.33×10^4
	$0.04 < li \leq 0.2$	410.64	-2.32×10^3	7.00×10^3	-300.0	-2.45×10^3
	$0.2 < li \leq 0.6$	308.87	-477.0	350.0	-861.0	2.10×10^3
	$li > 0.6$	144.71	0	0	-93.0	0

Nuclide		$C_{2,3}$	$C_{3,1}$	$C_{3,2}$	$C_{3,3}$	$C_{3,4}$
C-14	$0 \leq li \leq 0.02$	0	39.7	250.0	0	0
	$0.02 < li \leq 0.1$	0	46.7	-163.0	595.0	-606.25
	$0.1 < li \leq 0.2$	-35.0	47.0	-163.0	595.0	-606.25
	$0.2 < li \leq 0.4$	-35.0	47.0	-163.0	595.0	-606.25
	$0.4 < li \leq 0.6$	0	47.0	-163.0	595.0	-606.25
	$li > 0.6$	0	32.0	0	0	0
Co-60	$0 \leq li \leq 0.02$	0	-2.60×10^3	2.25×10^5	0	0
	$0.02 < li \leq 0.04$	0	1.70×10^3	1.15×10^4	0	0
	$0.04 < li \leq 0.2$	7.50×10^3	2.20×10^3	-2.10×10^3	0	0
	$0.2 < li \leq 0.6$	-1.40×10^3	2.20×10^3	-2.10×10^3	0	0
	$li > 0.6$	0	950.0	0	0	0

Table 4.2. Ranges of variations for major parameters in the assessment of the accuracy of the radiological-dose calculation using the scaling equations.

Parameter	Minimum Value	Maximum Value
Fusion power (MW)	300	3815
Minor radius (cm)	50	250
Major radius (cm)	400	955
First wall thickness (cm)	0.5	2.875
Blanket thickness (cm)	20	96
Lithium fraction (%)	0	100
Water fraction (%)	0	50

Table 4.3. Reactor cases studied with the composition given in volume fraction.

case	steel	water	lithium-oxide	void
1	0.3	0	0	0.7
2	0.3	0.3	0	0.4
3	0.3	0	0.4	0.3
4	0.3	0.3	0.4	0

Table 4.4. The coefficient X_{ij} in Eq. 4.4 which characterizes the spatial dependency of the averaged radioactivity in the zonal approximation for the major dose-contributing nuclides in the four different reactor cases studied.

	case 1	case 2	case 3	case 4
Mn-54	-5.01e-2	-7.63e-2	-7.63e-2	-1.17e-1
Mn-56	-5.60e-2	-8.33e-2	-9.54e-2	-1.26e-1
Co-60	-7.56e-3	-6.68e-2	-2.66e-2	-1.00e-1
Nb-94	-6.93e-3	-6.27e-2	-3.20e-2	-9.99e-2
Ag-108m	-9.56e-3	-5.92e-2	-3.70e-2	-1.01e-1
W-187	-6.67e-3	-5.53e-2	-2.84e-2	-9.31e-2
Re-186	-1.84e-2	-1.20e-1	-6.71e-2	-2.05e-1
Re-188	-1.32e-2	-1.19e-1	-5.80e-2	-1.95e-1

Table 4.5. Comparison of the BU factors for the relevant parent nuclides calculated by codes and scaling equations for 4 sample operating conditions.

	$\Gamma=1\text{MW/m}^2$ $t=0.5\text{yr}$		$\Gamma=1\text{MW/m}^2$ $t=3\text{yr}$		$\Gamma=5\text{MW/m}^2$ $t=0.5\text{yr}$		$\Gamma=5\text{MW/m}^2$ $t=3\text{yr}$	
	code	scaling	code	scaling	code	scaling	code	scaling
Mn-55	2.11	2.10	13.96	9.06	6.47	7.44	61.51	132.1
Fe-54	1.00	1.00	0.99	0.99	0.99	0.99	0.95	0.95
Fe-56	1.00	1.00	0.99	0.99	0.99	0.99	0.96	0.96
Co-59	1.02	1.02	1.19	1.16	1.09	1.13	1.91	2.40
Nb-93	1.00	1.00	0.99	0.98	0.99	0.99	0.93	0.93
Ag-107	0.98	0.98	0.89	0.89	0.91	0.91	0.57	0.57
Ag-109	0.89	0.89	0.49	0.49	0.55	0.55	0.033	0.029
W-186	0.96	0.96	0.79	0.79	0.82	0.82	0.32	0.31
Re-185	18.22	21.72	138.4	111.9	72.87	96.07	218.7	300.9
Re-187	329.6	335.8	1631	1623	1400	1410	3140	3602

Table 4.6. Comparison of the radioactivity (Ci) of major dose-contributing radionuclides calculated by codes and scaling equations for 4 sample operating conditions.

	$\Gamma=1\text{MW/m}^2$ $t=0.5\text{yr}$		$\Gamma=1\text{MW/m}^2$ $t=3\text{yr}$		$\Gamma=5\text{MW/m}^2$ $t=0.5\text{yr}$		$\Gamma=5\text{MW/m}^2$ $t=3\text{yr}$	
	code	scaling	code	scaling	code	scaling	code	scaling
Mn-54	4.58e6	3.74e6	1.29e7	8.79e6	2.31e7	2.20e7	7.19e7	2.56e8
Mn-56	5.54e7	5.54e7	5.75e7	5.65e7	2.80e8	2.81e8	3.27e8	3.99e8
Co-60	8.28e3	8.19e3	4.59e4	4.47e4	4.22e4	4.30e4	2.99e5	3.31e5
Nb-94	8.56e-2	8.15e-2	4.94e-1	4.89e-1	4.15e-1	4.13e-1	2.06	2.40
Ag-108m	3.93e-1	4.00e-1	2.00	2.00	1.73	1.73	5.33	5.39
W-187	3.43e7	3.43e7	2.83e7	2.83e7	1.47e8	1.47e8	5.65e7	5.58e7
Re-186	2.98e5	3.52e5	2.07e6	1.75e6	6.04e6	7.51e6	1.69e7	2.34e7
Re-188	1.35e6	1.37e6	6.56e6	6.59e6	2.87e7	2.86e7	6.31e7	7.31e7

Table 4.7. The total run time (in seconds) to calculate inventories of radionuclides using TART, ORLIB and FORIG for a five zone system is 227 seconds on the F-Cray.

TART	ORLIB	FORIG
102	100	25
	(5 Zones x 20 seconds / zone)	(5 Zones x 5 seconds / zone)

Table 4.8. Percentage deviation of zone-average fluxes calculated with the coupled diffusion-transport approximation from the all-transport solution (72 Intervals; $S_8 - P_3 / .01$).

Per Cent Difference Diffusion-Transport Compared to All-Transport Solution							
No. of Transport (T) / Diffusion (D) Groups	Run time (sec)	Zone*	Energy Groups 7-10	Energy Groups 11-16	Energy Groups 17-25	Energy Groups 26-31	Energy Groups 35-46
6T/40D	2.2	FW	20	12	10	5	2.5
		B	20	12	10	5	2.5
		1R	4	8	8	4	4
		2R	45	33	35	4	5
10T/36D	2.4	FW		10	7.5	4	2
		B		10	7	4	2
		1R		5	4	3	3
		2R		22	1	6	3
20T/26D	2.9	FW			2	2	1
		B			2	2	5
		1R			0.5	1	0.8
		2R			1.5	1.5	2.5

* First wall (FW), blanket (B), first reflector (1R), second reflector (2R)

Table 4.9. Range of ratio of all-diffusion to all-transport calculated activation rates.

Zone*	Cross Section Category		
	Hi Threshold	Mid Threshold	Non Threshold
FW	0.93	0.93	0.99 - 1.03
B			
near FW	1.15	1.15	1.01 - 1.05
near R1	0.65	0.90	0.97 - 1.03
zone avg.	1.05	1.05	1.03
1R+2R			
near B	0.65	0.92	1.03
end of 2R	0.10	0.20	0.92

* First wall (FW), blanket (B), first reflector (1R), second reflector (2R)

Table 4.10. ANISN vs ONEDANT runtimes/number-of-iterations for different transport orders and convergence criteria (99 intervals).

S _n	P ₁	Convergence Criteria	ONEDANT		ANISN		ANISN 22T/24D CPU Time
			CPU Time/ Avg. No. of Iterations		CPU Time/ Avg. No. of Iterations		
S ₁₆	P ₅	.00001	13.3	5-6	44.4	8-13	23.2
S ₁₆	P ₅	.01	9.2	2-3	21.6	3-6	12.9
S ₁₆	P ₃	.00001	12.3	5-6	26.6	8-13	14.1
S ₁₆	P ₃	.01	8.3	2-3	12.9	3-6	7.9
S ₈	P ₃	.00001	7.1	5-6	11.2	8-13	7.5
S ₈	P ₃	.01	5.7	2-3	5.8	3-6	4.2
S ₂	P ₁	.01	4.4	NA	2.1	NA	NA

Figure 4.1. Comparison of critical dose evaluated by neutronics codes and scaling equations. The squares denote data points.

Figure 4.2. Comparison of chronic dose evaluated by neutronics codes and scaling equations. The squares denote data points.

Figure 4.3. Comparison of intruder dose evaluated by neutronics codes and scaling equations. The squares denote data points.

Figure 4.4. Radioactivity of Mn-54 at various blanket zones for the 4 reactor cases studied. The location of the data point denotes the center of the zone.

Figure 4.5. Radioactivity of Co-60 at various blanket zones for the 4 reactor cases studied. The location of the data point denotes the center of the zone.

Figure 4.6. Ratio of activation rate from (a) all-transport $S_g-P_3 / .01$ to all-transport $Si_6-P_3 / .00001$ solution, (b) $10T(S_g-P_3 / .01) / 36D$ to all-transport $S_g-P_3 / .01$ solution, and (c) $20T(S_g-P_3 / .01) / 26D$ to all-transport $S_g-P_3 / .01$ solution by zones.

Figure 4.7. Pointwise ratios of activation rates of low energy reactions for (a) 20T($S_8 - P_3 / .01$)/26D and (b) mid threshold reactions for 10T($S_8 - P_3 / .01$)/36D compared to the all-transport ($S_8 - P_3 / .01$) solution.

ACTIVATION APPROXIMATIONS

Inventory Approximations
Agree to approximately $\pm 5\%$
in FW

Figure 4.8. Ratio of radionuclide inventories obtained using analytical approximations relative to DKR-calculated inventories corresponding to one year and to thirty years operating times.

Figure 4.9. Long-lived radionuclide inventories as calculated by ANISN using effective inventory cross-sections relative to the ToF calculated inventories.

Figure 4.10. Short-lived radionuclide inventories as calculated by ANISN using effective inventory cross-sections relative to the ToF calculated inventories.

Figure 4.11. Perturbation theory results comparing the SWAN/ANISN ratios for the indirect effect on tritium production rates in the blanket from the ${}^6\text{Li}(n,\alpha)$ reaction.

Figure 4.12. Perturbation theory results comparing the SWAN/ANISN ratios for the indirect effect on ${}^{28}\text{Al}$ production rates in the blanket from the ${}^{28}\text{Si}(n,p)$ reaction.

Appendix A

Perturbation Theory Formulation

Perturbed parameters for the flux equation:

$$\left[\Omega \cdot \nabla + \sum_l C_l(\underline{r}, t) (\sigma_{t_l} - \sigma_{s_l}) \right] \delta \Phi = \delta S - \sum_l (\sigma_{t_l} - \sigma_{s_l}) \Phi \delta N_l \quad (\text{A-1})$$

Perturbed nuclide density equation:

$$\begin{aligned} \left[\frac{\partial}{\partial t} + \lambda_i + \sigma_{a_i} \Phi \right] \delta N_i - \sum_{j \neq i} [\sigma_{j \rightarrow i} \Phi + \lambda_{j \rightarrow i}] \delta N_j \\ + \left[N_i \sigma_{a_i} - \sum_{j \neq i} N_j \sigma_{j \rightarrow i} \right] = \delta M_i \end{aligned} \quad (\text{A-2})$$

Adjoint flux equation:

$$\left[-\Omega \cdot \nabla + \sum_l C_l(\underline{r}, t) (\sigma_{t_l} - \sigma_{s_l}) \right] \Phi^* + \sum_i \left[N_i \sigma_{a_i} N_i^* + \sum_{j \neq i} N_i \sigma_{i \rightarrow j} N_j^* \right] = S^* \quad (\text{A-3})$$

Adjoint nuclide density equation:

$$\left[-\frac{\partial}{\partial t} + \lambda_i + \sigma_{a_i} \Phi \right] N_i^* - \sum_{j \neq i} [\sigma_{i \rightarrow j} \Phi + \lambda_{i \rightarrow j}] N_j^* = M_i^* \quad (\text{A-4})$$

Boundary, time, and source conditions:

$$\delta \Phi(\underline{r}, E, \Omega, t) = 0 \quad \Omega \cdot \hat{n} \geq 0 \quad (\text{A-5})$$

$$\Phi^*(\underline{r}, E, \Omega, t) = 0 \quad \Omega \cdot \hat{n} \leq 0 \quad (\text{A-6})$$

$$S^* = M_i^* = \delta M_i = 0 \quad (\text{A-7})$$

$$\delta \Phi(t_i) = 0 \quad (\text{A-8})$$

$$\Phi^*(t_f) = 0 \quad (\text{A-9})$$

$$\delta N_i(t_i) = 0 \quad (\text{A-10})$$

$$N_i^*(t_f) = \delta_{is} \delta(\underline{r} \in \underline{r}_s) \delta(t - t_f) \quad (\text{A-11})$$

Importance balance condition:

$$\int_{(r \in r_s)} dr \delta N_s(t_f) = \iiint \Phi^* \left(\delta S - \sum_l \delta C_l (\sigma_{tl} - \sigma_{sl}) \right) \quad (\text{A-12})$$

Assuming:

$$\begin{aligned} \delta S &= 0 \\ \delta C_l &= \alpha_k C_k \delta(r \in r_k) \quad \text{for } t \geq t_i \end{aligned}$$

gives:

$$\begin{aligned} \int_{(r \in r_s)} dr \delta N_s(t_f) &= -\alpha_k C_k \int_{(r \in r_k)} dr \int dE \int d\Omega \Phi^*(r, E', \Omega') \quad (\text{A-13}) \\ &\quad \left[\sigma_{tk} - \int dE' d\Omega' \sigma_{sk}(E\Omega \rightarrow E'\Omega') \Phi(r, E, \Omega) \right] \end{aligned}$$

which can be written as:

$$\int_{(r \in r_s)} dr \delta N_s(t_f) = \alpha_k PC_{ks} \quad (\text{A-14})$$

where PC_{ks} is referred to as the "Perturbation Coefficient."

5. TRITIUM FUEL-CYCLE MODELING

Earlier progress reports under this contract have described in some detail our work in dynamic modeling of tritium inventories and flows in the fuel-processing systems of magnetic fusion reactors (see especially Ref. 5.1). This work was conducted mainly by UCB Nuclear Engineering doctoral student Dimosthenis Sarigiannis; his results are presented in his doctoral dissertation, which was filed in March 1994 [5.2]. The following excerpts from his abstract summarize the scope of this work.

The focal point of this work has been the development of a comprehensive, multi-step methodology in order to incorporate the safety concerns associated with the radiologically hazardous tritiated inventories present in the fusion power plant into the system design process. This methodology includes:

(a) the development of a dynamic process-simulation code (TRIDYN, for TRItium DYNamics) in order to assess both the transient and steady-state behavior of the fuel processing system,

(b) identification of the reactor-plant subsystems that are most vulnerable to release of tritium, and calculation of the tritium inventories that could be mobilized in the case of a catastrophic failure, and

(c) optimization of the design of the vulnerable sub-systems to minimize releasable tritium inventories.

This methodological framework is, more generally, applicable to industrial plants characterized by the presence of significant amounts of hazardous materials. In this work, it has been applied to the fuel processing system envisaged for the first generation of power-producing magnetic fusion reactors.

A simplified approach for modeling packed cryogenic distillation columns such as the ones used for separation of the hydrogen isotopes in fusion reactors has been introduced, and other short-cut methods for calculating the tritiated inventories in each subsystem are presented. The results of the TRIDYN code have been checked for accuracy against experimental and simulation data from existing pilot-plant facilities. Parametric studies using TRIDYN have identified the

relationship between vulnerable tritium inventories and reactor system parameters such as fusion power, tritium burnup fraction in the plasma, and the purity requirement on the tritium product of the separation system. Finally, the application of the safety optimization algorithm to critical parts of the system, such as the isotope separation system, has shown how different system design configurations may impact the overall safety of the fuel processing plant.

The following excerpts from the final chapter of the dissertation capture some of Sarigiannis's main conclusions:

Dynamic simulation showed that pulsed operation or semi-continuous operation followed by long torus maintenance periods may have destabilizing effects on the evolution of the fuel species inventories, setting, thus, limits to the envisageable modes of plasma burn.

Parametric studies of different system parameters have shown that the tritium inventory in the vacuum pumping system, as well as the overall tritium inventory in the fuel cycle, can be significantly reduced by increasing the fuel burnup fraction in the plasma.

The thermal power output of the plasma has been found to be nonlinearly correlated with the active tritiated inventories in the impurity removal and processing system. This behavior shows that the minimum values of hazardous inventories in this subsystem are constrained by the required loading capacity of the molecular sieve beds in the system.

Very high purity of the T₂ product (purity > 99%) results in increased tritium residence times in the distillation columns and, consequently, in high tritiated inventories.

For further details, the reader may consult the Sarigiannis dissertation. Dr. Sarigiannis is now employed at the European Joint Fusion Research Centre at Ispra, Italy.

REFERENCES

- 5.1 S.K. Ho, Ed., "Status Report: Code Development Incorporating Environmental, Safety, and Economic Aspects of Fusion Reactors," UC-BFE-039, Berkeley Fusion Engineering, University of California, Berkeley, July 1993.

- 5.2 D.A. Sarigiannis, "Modeling and Safety Optimization of the Fuel Processing System in Magnetic Fusion Reactors", PhD Dissertation, Department of Nuclear Engineering, University of California, Berkeley, March 1994.

6. SIMPLIFIED THERMAL TRANSIENT ANALYSIS

6.1 INTRODUCTION

In normal operation, a fusion reactor is cooled by a stream of fluid that carries heat away from the plasma. Should the coolant stop flowing, the blanket surrounding the plasma will increase in temperature, possibly damaging or even melting its component materials. This work investigates the temperature profile of the reactor blanket after a Loss of Flow Accident (LOFA). In particular, we seek an approximation method that is simpler to implement than a standard finite-difference scheme.

Modern fusion reactors are designed to quench their plasmas within a few seconds should the plasma-facing surface of the blanket overheat. Even after the plasma has shut down, however, neutron-activated isotopes in the blanket will continue to generate heat by radioactive decay. With the coolant not flowing, the blanket is assumed to transfer heat only by radiation from its outermost surface to an external heat sink. Conduction and natural convection of heat to the external sink are neglected in this work.

In our model, a fusion reactor is approximated as a long, hollow cylinder. The wall of this cylinder is the blanket, and it is assumed that the temperature varies only in the radial direction. We can further simplify the geometry by neglecting the curvature of the cylinder's wall and treating the blanket as a one-dimensional plane slab. One side of the slab faces the plasma, and the other sees an external heat sink. This plane-slab approximation is not strictly valid, since the thickness of the cylinder's wall is not much smaller than the cylinder's inner radius, but a previous study has concluded that the plane-slab approximation leads to an acceptably small overestimation of the blanket temperature [6.1].

After the coolant stops flowing, the plasma will continue to radiate energy into the blanket for several seconds until it shuts down. During shutdown, the plasma will transfer heat to the plasma-facing surface, or "first wall," and send heat-generating neutrons into the blanket. The plasma is assumed gone after shutdown. We expect no subsequent heat transfer through the first wall because this self-viewing surface is at a uniform temperature, and thus in radiative equilibrium with itself. The blanket temperature rises in response to the transient heat transfer at the surface of

the first wall, the transient neutron heating within the blanket, and the radioactive decay of neutron-activated isotopes. It is assumed to cool only by radiation to the external heat sink.

Analyzing a one-layer blanket would be straightforward, though laborious. We would solve for the transient temperature profile of a one-dimensional slab with non-uniform generation, variable properties, and a radiative boundary condition. Real blankets, unfortunately, typically contain eight or nine different layers, including a vacuum gap. As a first step toward studying a multi-layer slab, we shall consider the simpler case of a two-layer slab.

6.2 CALCULATIONAL SCHEMES

There are different ways to seek the transient temperature profile of a finite, two-layer composite slab with heat generation, including purely analytical solution, numerical solution, and analytical solution following spatial discretization of the blanket. Exact analysis is virtually impossible for this problem. The problem has been solved numerically by finite differences, but efficient finite-difference codes are complex. Our approach to this problem shall be to solve a set of ordinary differential equations that arise from the finite-difference description of the blanket.

6.2.1 FINITE DIFFERENCE SCHEME

Consider the one-dimensional energy equation for a plane slab with constant heat generation and material properties:

$$\frac{\partial^2 T}{\partial x^2} + \frac{g'''}{k} = \frac{1}{\alpha} \frac{\partial T}{\partial t}, \quad \alpha = \frac{k}{\rho c_p}. \quad (6.1)$$

Expressed dimensionlessly, the equation is dependent on the rate of heat generation, but independent of k and ρc_p :

$$\bar{T}_{\bar{x}} + G = \bar{T}_i, \quad (6.2)$$

where $\bar{T} = \frac{T - T_\infty}{T_i - T_\infty}$, $\bar{x} = \frac{x}{L}$, $\bar{t} = \frac{\alpha}{L^2} t$, $G = \frac{g''' L^2}{k(T_i - T_\infty)}$, L is a characteristic length, T_i is the initial temperature, and T_∞ is the boundary temperature.

There exists a family of dimensionless equations over the domain of G . After solving these equations over some domain of G , we can obtain $\bar{T}(\bar{x}, \bar{t}, G)$, and then $T(x, t, k, \rho c_p, g''')$. Unfortunately, this technique is complex for layers with different material properties and generation rates. Consider a slab with layers 1 and 2 that have properties $k_1, (\rho c_p)_1, g_1'''$ and $k_2, (\rho c_p)_2, g_2'''$. We write separate dimensionless equations for each layer,

$$\bar{T}_{1, \bar{x}_1 \bar{x}_1} + G_1 = \bar{T}_{1, \bar{t}_1} \quad \text{and} \quad \bar{T}_{2, \bar{x}_2 \bar{x}_2} + G_2 = \bar{T}_{2, \bar{t}_2}, \quad (6.3)$$

because the variables will be defined differently in each layer. At the boundary between the two layers, continuity of temperature and heat flux requires that

$$T_1(x, t) = T_2(x, t) \quad (6.4)$$

$$k_1 T_{1,x}(x, t) = k_2 T_{2,x}(x, t). \quad (6.5)$$

When the variables that appear in the continuity equations are non-dimensionalized, the constants $k_1, (\rho c_p)_1, k_2, (\rho c_p)_2$ will reappear. Even if the finite-difference scheme can simultaneously solve for $\bar{T}_1$ and $\bar{T}_2$, $\bar{T}_1$ must be plotted against $\bar{x}_1, \bar{t}_1, G_1, k_1, (\rho c_p)_1, k_2, (\rho c_p)_2$ and $\bar{T}_2$ against $\bar{x}_2, \bar{t}_2, G_2, k_1, (\rho c_p)_1, k_2, (\rho c_p)_2$. in order to solve for the solutions. Thus non-dimensionalizing the heat equation for a composite slab can not provide general results.

6.2.2 ANALYTICAL SOLUTION OF THE FINITE-DIFFERENCE EQUATIONS

We can approximate the partial-differential energy equation as a set of ordinary differential equations by solving the finite-difference equations analytically. We divide the slab into N regions, each region n wholly at a single temperature u_n . Region 1 contains the plasma-facing surface, and region N is the radiation heat sink. Let $q_{m,n}$ = heat transfer from region m to adjacent region n , in W/m^2 ; $\dot{E}_n^g$ = rate of energy generation in the region n , in W/m^2 ; and $\dot{E}_n^s$ = rate of energy storage in the region n , in W/m^2 . Conservation of energy then requires

$$q_{n-1,n} + \dot{E}_n^g = q_{n,n+1} + \dot{E}_n^s; \quad (6.6)$$

We can express the heat transfer q by analogy with the current in electrical circuits. Each region can be considered to have a thermal resistance, thermal capacitance, and thermal generation. The thermal resistance comes from rearranging expressions of heat flow. In a solid, Fourier's law gives

$$q = -k \frac{du}{dx} \approx k \frac{u_1 - u_2}{\Delta x} = \frac{u_1 - u_2}{\Delta x / k} = \frac{u_1 - u_2}{R} \quad (6.7)$$

where Δx is arbitrarily taken to be the center-to-center distance between a pair of adjacent regions 1 and 2, and $R = \frac{\Delta x}{k}$ is the "thermal resistance." If the regions have different thermal conductivities, the thermal resistance across the center-to-center distance is taken to be half the resistance of the first region plus half the resistance of the second region, with l being the region width:

$$R = \frac{1}{2} R_1 + \frac{1}{2} R_2 = \frac{1}{2} \frac{l}{k_1} + \frac{1}{2} \frac{l}{k_2}, \quad (6.8)$$

Using gray-body approximations, the net heat radiated across a gap between two surfaces is

$$q = \frac{\sigma(u_1^4 - u_2^4)}{\epsilon_1^{-1} + \epsilon_2^{-1} - 1} = \frac{\sigma(u_1 - u_2)(u_1 + u_2)(u_1^2 + u_2^2)}{\epsilon_1^{-1} + \epsilon_2^{-1} - 1} = \frac{u_1 - u_2}{R}, \quad (6.9)$$

where

$$R = \frac{\epsilon_1^{-1} + \epsilon_2^{-1} - 1}{\sigma(u_1 + u_2)(u_1^2 + u_2^2)}, \quad (6.10)$$

σ is the Stefan-Boltzmann constant, and ϵ is emissivity. If one of the surfaces is a radiation heat sink, we can treat it as a black body of emissivity.

The rates of energy generation and energy storage in a solid region are given, respectively, by:

$$\dot{E}^s = l g''' = G, \quad (6.11)$$

$$\dot{E}^s = \rho c_p l \dot{u} = C \dot{u}, \quad (6.12)$$

where g''' is the rate of energy generation per unit volume, ρ is density, c_p is specific heat, and $C = \rho c_p l$ is the thermal capacitance. A vacuum gap has no thermal capacitance or generation, of course.

Each of the N regions may include more than one physical layer of the blanket. We may, for example, divide the eight physical layers of a typical blanket into three regions. Region 1 would be the layer containing the first wall; region 2 would include the remaining 7 layers; and region 3 would be the external heat sink. When a region contains more than one layer, its resistance, capacitance, or generation is the sum of the resistances, capacitances, or generations of each layer included.

It is pointed out that a physical layer of the blanket may contain more than one material component. Let f_i represent the volume fraction of the layer's i -th component. We shall assume that the composition of the layer is homogenous in the direction of heat flow, and that f_i equals the fraction of the conduction path area occupied by the layer's i -th component; i.e., the components are assumed to conduct heat in parallel. The resistance of each component is then

$$R_i = \frac{l}{k_i f_i} \quad (6.13)$$

and the layer's total resistance is

$$R = \left(\sum_i \frac{1}{R_i} \right)^{-1} \quad (6.14)$$

The capacitance of each component is

$$C_i = (\rho c_p)_i l f_i, \quad (6.15)$$

and the capacitance of the layer is the sum of the component capacitances. Each layer's generation G is $g'''l$, since the volumetric generation rate g''' takes into account the material composition of the layer.

We assume that the heat flux across the plasma-facing surface is $q_{0,1} = q_{op} r(t)$, where q_{op} is the surface heat (in W/m^2) transferred from plasma to first wall during normal operation and $r(t)$ is the plasma shutdown profile. The radiation heat sink remains at some constant temperature. With energy balance equations and initial temperature values for all regions, we have a complete system of ordinary differential equations to solve for the temperatures $u_n(t)$:

$$q_{0,1} = q_{op} r(t) \quad (6.16)$$

$$u_N(t) = u_{sink} \quad (6.17)$$

$$q_{n-1,n} + \dot{E}_n^z = q_{n,n+1} + \dot{E}_n^z, \quad n = 1..(N-1) \quad (6.18)$$

$$u_n(0) = u_n^0, \quad n = 1..(N-1). \quad (6.19)$$

These ODE's can be rearranged in the form:

$$\dot{\mathbf{u}} = \mathbf{A} \mathbf{u} + \mathbf{B}, \quad (6.20)$$

where,

$$\mathbf{u} = \begin{bmatrix} u_1 \\ \vdots \\ u_M \\ \vdots \\ u_{N-1} \end{bmatrix}, \quad \mathbf{A} = \begin{bmatrix} a_{1,1} & L & a_{1,(N-1)} \\ & M & O & M \\ & a_{(N-1),1} & L & a_{(N-1),(N-1)} \end{bmatrix}, \quad \mathbf{B} = \begin{bmatrix} b_1 \\ \vdots \\ b_M \\ \vdots \\ b_{N-1} \end{bmatrix}. \quad (6.21)$$

Unfortunately, some of the coefficients in our system of ODE's vary with temperature, and other with time. The temperature dependencies of R and C make the ODE's non-linear, and thus quite difficult to solve analytically.

We may solve the temperature-dependent ODE's by iteration. The ODE's are composed with resistances and capacitances evaluated at the regions' initial temperatures. These equations are solved to find the blanket's temperature profile at some later time. The ODE's are regenerated with resistances and capacitances evaluated at the new temperatures, then solved again. We iterate to obtain the blanket's temperature profile over time.

An alternative to iteration is to simplify the problem by seeking only an upper limit to the temperature of the first wall. This will allow us to write linear, temperature-independent ODE's. First, we guess a temperature range for this blanket, bounded firmly below by the temperature of the radiative heat sink, and limited physically by the melting or structural failure point of some material. We then remove the temperature variation of the thermal resistances and capacitances by maximizing R and minimizing C over the expected temperature range. The principle is that the flow of heat to the sink varies inversely with resistance, while the rate of temperature rise in the

blanket varies inversely with capacitance. The combination of high R and low C impedes the flow of heat to the sink, thereby maintaining the heat in the blanket. The blanket temperature rises as it absorbs the generated heat.

If the region farthest away from the heat sink contains only a thin first wall, solving the temperature-independent ODE's should yield an upper bound to the first-wall temperature profile. However, the solution of these ODE's will not necessarily bound the temperatures of regions between the first wall and heat sink. When we maximize the conduction resistance in the blanket, we reduce the flow of heat from the first wall to other blanket regions. Thus, we tend to underestimate the temperature of regions closer to the heat sink.

It should also be noted that our conservative choices of R and C serve only to bound the temperature of the "first wall" described in our simplified, one-dimensional, spatially-coarse model of the blanket. The "upper-bound" solution does not necessarily limit the temperature of the real first wall. It has been found, for example, that the temperatures predicted by the linear ODE's rise asymptotically as the number of regions in the model is increased. Thus, other things being equal, the crude spatial resolution of our model tends to yield temperatures lower than those obtained from a more spatially-refined finite-difference scheme.

When we speak of $g'''(t)$ in a region, we have ignored the fact that heat generation drops off within each region as we move away from the plasma. We remove this spatial dependence by using the spatially-averaged $g'''(t)$ in each region. It is more difficult to remove the time dependence of generation, however. Using the rate of heat generation at shutdown is a gross overestimate, because the rate of generation from isotopic decay falls by several orders of magnitude over the course of a day. We can let generation vary with time and note that the problem is not too difficult because $g'''(t)$ appears only in B . We now have a non-homogenous set of linear ODE's with constant coefficients to solve for the upper bound to the blanket temperature.

6.3 TEST CASES

This work considered three blankets that were previously studied by J.E. Massidda and M.S. Kazimi [6.1] for the Environmental, Safety, and Economics Committee (ESECOM) report [6.2]. Massidda has modeled these blankets as plane slabs, and computed their temperature profiles by the method of finite differences. We shall refer to Massidda's results as the "finite-difference solutions." The material properties, plasma neutron-heating rates, isotopic-decay

heating rates, heat sink temperatures, initial conditions, and blanket geometries used in our model were taken from this study.

Blanket #1 comes from a self-cooled liquid metal (Li/Li/V) tokamak. It has eight layers. Moving from the plasma to the heat sink, they are layer 1, the front face of the first wall; layer 2, the first wall coolant channel; layer 3, the back structural slab of the first wall; layer 4, the main breeding area; layer 5, the manifold; layer 6, a second structural wall; layer 7, an insulating vacuum gap; and layer 8, the thermal and radiation shield for the superconducting magnets. Heat radiates to the magnets, which are conservatively estimated to lie at the shield's operating temperature of 100°C. When operating, the plasma delivers 5 MW/m² of power to the blanket, with 20% in the form of surface heat and 80% in the form of neutron heat.

Blanket #2 comes from a helium-cooled, solid breeder (Li₂O/He/MT-9) tokamak. Its nine layers are layer 1, the front slab of the first wall; layer 2, the path for the helium coolant; layer 3, the back slab of the first wall; layer 4, where the coolant is distributed into the breeder channels; layer 5, the main breeder region; layer 6, the second breeder region; layer 7, the manifold; layer 8, the vacuum gap, and layer 9, the magnet shield. We assume that the helium coolant is at rest, and treat it as a solid. Heat sinks to the magnets, which are conservatively assumed to lie at 100°C.

Blanket #3 comes from a self-cooled liquid metal (Li/Li/V) Reversed-Field-Pinch reactor. Its first seven layers are analogous to those of Blanket#1. There is no magnet shield, so layers 8 and 9 are the copper magnet coils. Heat sinks to the room at 20°C. When operating, the plasma delivers 15 MW/m² of power to the blanket, with 20% in the form of surface heat and 80% in the form of neutron heat.

6.4 RESULTS

A *Mathematica* 2.0.3.0 program was written to compute the many resistances, capacitances, and generations required to set up the problem. It solves a system of at least two, and more likely on the order of ten, non-homogeneous ordinary differential equations. Given expressions for the material properties and generation rates, descriptions of the blanket geometry and materials, and initial conditions, the *Mathematica* program finds the upper-bound solution by producing a set of temperature-independent ODE's and solving them with a *Mathematica* routine. A second *Mathematica* program solves the temperature-dependent ODE's by iteratively composing and solving the ODE's as the blanket temperature profile evolves with time.

A typical blanket is divided into 10 regions, with one region assigned to the heat sink and one to each solid layer. The vacuum gap within each blanket is included into the region of an adjacent solid layer. Volumetric generation rates are spatially averaged over each layer, and blankets start at their normal operational temperatures. Two temperature profiles are computed for each blanket. The first profile, or "upper-bound solution," is obtained by solving the temperature-independent ODE's. The second profile, or "iterative solution," approximates the actual temperature by iteratively solving the temperature-dependent ODE's. Each temperature profile is computed for 48 hours following the LOFA.

As an illustration, results for test case #1 will be shown and discussed in detail in the following sections. The findings for test cases #2 and #3 are summarized in Section 6.5.

6.4.1 BLANKET #1: FIRST WALL

In response to transient plasma heating during shutdown, the finite-difference solution rose from 530°C to 650°C within five seconds. The finite-difference solution declined to 500°C by 1 hour as the transient heat diffused (Fig. 6.1). Isotopic decay raised the finite-difference solution to 650°C by 48 hours (Fig. 6.2).

The iterative solution was very close to the finite-difference solution, differing by no more than 40°C at any time during the first 48 hours. The upper-bound solution remained within 80°C of the finite-difference solution during the first 10 hours, but diverged from the finite-difference solution thereafter, climbing to 840°C by 48 hours. All solutions remained below VCrTi's structural failure point of 1200°C.

Several factors made the upper-bound solution diverge from the finite-difference solution. Conduction resistances played a small role: when maximized over the assumed temperature range of 100°C to 1200°C, the total resistance of the solid layers is only 20% higher than when minimized over this range. Underestimating capacitance can have a greater effect, for the total maximized capacitance is twice as high as the total minimized capacitance. However, the main reason that the upper-bound did not level off closer to 670°C is related to how the resistances of the vacuum gaps next to the shield and heat sink were determined. To make the ODE's linear, the gap resistances were not allowed to vary with temperature, and were maximized by assuming that each gap face was at the temperature of the heat sink, or 100°C. The resistance across a gap decreases with the cube of the face temperatures, so using the heat sink temperature can grossly overestimate the true gap resistance. When calculated this way for Blanket#1, the two gap resistances accounted

for more than 90% of the total resistance between the first wall and the heat sink, even when the solid resistances were maximized.

The finite-difference scheme predicted that the shield temperature reaches a maximum value of approximately 500°C. Recomputing the shield gap resistance with each face at 500°C reduces its value by a factor of 10; recalculating the heat sink resistance with the shield at 500°C and the sink at 100°C reduces the sink resistance by a factor of 4. Together, they then account for only 60% of the total resistance between first wall and heat sink.

6.4.2 BLANKET #1: SHIELD

The finite-difference solution reached 480°C by 5 hours, then rose to 520°C by 48 hours (Fig. 6.2). The iterative solution reached 360°C at 10 hours, then leveled off. The upper-bound solution rose to 400°C by 10 hours, declining to 360°C by 48 hours. Both iterative and upper-bound solutions underestimated the finite-difference solution by approximately 100°C. All temperature solutions are well below Fe1422's structural failure point of 900°C.

6.5 CONCLUSIONS

The iterative solutions to the temperature-dependent ODE's closely approximated the temperature profiles predicted by Massidda's finite-difference scheme. At any given time in the first 48 hours, the iterative solutions never varied by more than 150°C from the finite-difference solutions, and were often within 50°C.

The iterative solution will generally approximate the finite-difference solution to within 150°C. Hence, the iterative solution can sometimes be used to determine if the first wall of a blanket will get hot enough to fail. In the cases studied, iterative solutions predicted that

- the temperature of the first wall of Blanket #1 will not exceed 680°C. This is significantly more than 150°C below VCrTi's structural failure point of 1200°C;
- the temperature of the first wall of Blanket #3 will not exceed 1020°C. This is slightly more than 150°C below VCrTi's structural failure point of 1200°C;

- the temperature of the first wall of Blanket #2 will reach 1000°C. This exceeds HT-9's structural failure point of 900°C. If we assume that the margin of error is 150°C, this tells us that the wall may or may not fail.

The upper-bound solutions did not approximate the finite-difference solutions as closely as the iterative solutions. Considering that the upper-bound solution overestimated the finite-difference solution by about 500°C in the first wall of Blanket #2, and by about 1500°C in the first wall of Blanket #3, we can conclude that upper-bound solutions are not generally useful. Finally, it is pointed out that within ten seconds of the LOFA, the shutdown plasma transient can raise the first wall to temperatures comparable to those later achieved by decay heating. Thus, the plasma transient's brief heating effect may be as important as that due to longer-term isotopic decay.

REFERENCES

- 6.1 John E. Massidda and Mujid S. Kazimi, "Thermal Design Considerations for Passive Safety of Fusion Reactors, " MIT Plasma Fusion Center, October 1987.
- 6.2 J.P. Holdren, D.H. Berwald, R.J. Budnitz, J.G. Crocker, J.G. Delene, R.D. Endicott, M.S. Kazimi, R.A. Krakowski, B.G. Logan, and K.R. Schultz, "Report of the Senior Committee on Environmental, Safety, and Economic Aspects of Magnetic Fusion Energy," Report: UCRL-53766; Lawrence Livermore National Laboratory (1989).

Figure 6.1. Finite-difference and ODE solutions for the first-wall and shield temperatures of Blanket#1 for 0-5 hours after LOFA. Thick lines are first-wall temperatures; thin lines are shield temperatures.

- = finite-difference solution (Massidda)
- - - - = iterative solution to ODE's (Levinson)
- = upper-bound solution to ODE's (Levinson)

Figure 6.2. Finite-difference and ODE solutions for the first-wall and shield temperatures of Blanket#1 for 0-48 hours after LOFA. Thick lines are first-wall temperatures; thin lines are shield temperatures.

- = finite-difference solution (Massidda)
- - - - = iterative solution to ODE's (Levinson)
- = upper-bound solution to ODE's (Levinson)

7. DIVERTOR MODELING

In this section, we will investigate some physics and engineering issues relating to the utilization of divertors in tokamaks. A good understanding of the physics and engineering aspects of divertor operation is an essential starting point for safety analysis. The divertor region is closely related to safety-relevant events such as plasma disruption, erosion of activated materials, creation of release pathway for activated dusts, etc.

Our group is contributing to the improvement of SUPERCODE [7.1] in areas of environmental and safety relevance, specifically the divertor model. During last year, considerable progress has been made in benchmarking the UEDGE scrape-off-layer/gaseous divertor code against DIII-D experimental data. This work is expected to lead to a simplified but reliable code model of divertors in normal operation and during transient shutdown.

We have also examined the engineering implications of utilizing low-activation silicon carbide (SiC) ceramic as divertors. Technological performance of a proposed ITER technology phase divertor design and a hypothetical SiC ceramic design are compared in terms of critical heat flux, burnout margins, disruption erosion, and activation-induced radiological hazard. Results indicate that the SiC ceramic merits further study as a potential divertor material.

7.1 MODELING OF EDGE PLASMA AND GASEOUS DIVERTOR

7.1.1 INTRODUCTION

The tokamak divertor is still one of the chief engineering obstacles in the design of a reactor size machine. The divertor plates are the components of the tokamak subjected to the highest particle and heat flux, by far. Reliability, physics, and safety issues need to be resolved in any divertor design. Thus, systems design codes, such as SUPERCODE [7.1], require divertor models that are as accurate as possible in predicting the peak power flux to the divertor and peak divertor temperatures. We propose to improve the present 0-D simple power balance divertor model in SUPERCODE with the aid of UEDGE [7.2], a 2-D fluid code that more realistically models the complex physics of the plasma scrape off layer (SOL). We report on the first stage of the development of a better divertor model for SUPERCODE by benchmarking the UEDGE code with data from the DIII-D tokamak.

7.1.2 THE UEDGE CODE

UEDGE is a 2-D fully implicit fluid code that simulates the edge region of a tokamak plasma. This code was developed at Lawrence Livermore National Laboratory primarily by Rognlien, Milovich, and Rensink. UEDGE can run either in a time-dependent mode or search for a steady-state solution. Both a one-group and a two-group neutrals model are available to the user. UEDGE contains a fixed fraction, spatially uniform atomic physics package to model impurity radiation.

7.1.2.1 The Grid

UEDGE solves a set of fluid equations in the poloidal plane of the tokamak plasma. The code can handle either single null or double null divertor configurations. Toroidal symmetry is assumed, and flux coordinates are employed. UEDGE generates realistic grids by accepting actual tokamak equilibrium values as input. These equilibrium values come from the equilibrium solver code, EFIT, which generates its results from DIII-D's diagnostic data.

The x coordinate points in the poloidal direction (along the flux surface), and the y coordinate points in the radial direction (perpendicular to the flux surface) (Fig. 7.1). The UEDGE grid includes a small portion of the core plasma inside the separatrix. Part of the private flux region, the area under the X-point between the two divertor "legs", is also contained in the grid.

UEDGE allows for some flexibility in setting the grid resolution. There are four parameters which specify the resolution of the grid (there are other parameters that customize the spacing of the grid). Two parameters determine the resolution in the poloidal direction by specifying the number of grid points in the region above the X-point (the main body of the plasma) and the number of grid points in the region below the X-point (the "legs"). The other two parameters determine the radial direction resolution: the number of grid points inside the separatrix (the core) and outside the separatrix (the SOL). The number of radial grid points in the private flux region is set to be the same as that of the core. In addition, it is possible to set the poloidal resolution for the left half of the plasma to be different from the right half.

7.1.2.2 Model Equations

UEDGE solves Braginskii-type fluid equations of the form

$$\frac{\partial \psi_k}{\partial t} + \nabla \cdot (v \psi_k - D_k \cdot \nabla \psi_k) = S_k \quad (7.1)$$

where the dependent variable, Ψ_k , represents a set of plasma parameters including density n , parallel momentum density $nmv_{||}$, temperature density nT , for the ion and electron species, neutral gas density n_g , and electrostatic potential, ϕ . The parameter v is the convection velocity from parallel momentum and drifts, D_k is the diffusion tensor, and S_k is the source term. The convection velocity's poloidal and radial components are

$$v_x = \frac{B_\phi}{B^2} \left(\frac{\partial \phi}{\partial y} + \frac{1}{Z_i e n_i} \frac{\partial p_i}{\partial y} \right) + \frac{B_x}{B} v_{||} \quad (7.2)$$

$$v_y = -\frac{B_\phi}{B^2} \left(\frac{\partial \phi}{\partial x} + \frac{1}{Z_i e n_i} \frac{\partial p_i}{\partial x} \right) - \frac{D_\alpha}{n_i} \frac{\partial n_i}{\partial y} + v_\alpha \quad (7.3)$$

where D_α and v_α are anomalous diffusion and pinch terms. Energy transport parallel to the magnetic field is treated by classical collisional theory. Anomalous diffusion coefficients, adjusted to satisfy experimental results, dictate the perpendicular momentum and energy transport.

When the electrostatic potential equation is solved, parallel currents are allowed and cross field drifts ($\mathbf{E} \times \mathbf{B}$) can also be computed. The equation for the electrostatic potential, ϕ , comes from the divergence-free property of the current: $\nabla \cdot \mathbf{J} = 0$. The parallel component of the current comes from the electron momentum equation

$$0 = -\frac{1}{n_e e} \frac{\partial p_e}{\partial x} + \frac{\partial \phi}{\partial x} - \frac{\alpha}{e} \frac{\partial T_e}{\partial x} + \frac{B}{B_p} \eta_{||} J_{||} \quad (7.4)$$

where $p_e = n_e T_e$, B_p is the poloidal magnetic field, $\eta_{||}$ is the classical resistivity, and $\alpha = 0.71$ if the ion charge, Z_i , is 1. There are two terms that make up the perpendicular component of the current.

The classical term is given by $J_{\perp c} = \mathbf{B} \times \nabla_{\perp} p / B^2$. Anomalous contributions to the perpendicular current appear in the term of the form

$$J_{ya} = \sigma_{\perp a} \left(\frac{1}{n_e e} \frac{\partial p_e}{\partial y} - \frac{\partial \phi}{\partial y} \right) \quad (7.5)$$

where $\sigma_{\perp a}$ is an anomalous conductivity. Collecting all the currents and setting a false time dependence gives the resulting potential equation

$$\varepsilon \frac{\partial \phi}{\partial t} + B_p \frac{\partial}{\partial x} \left(\frac{1}{B} J_{\parallel} \right) + \frac{\partial J_{ya}}{\partial y} = -\nabla_{\perp} \cdot \mathbf{J}_{\perp c} \quad (7.6)$$

where B_p is the poloidal component of the magnetic field. The false time dependence (the first term in the above equation) is introduced in order to conform the potential equation to the general form of the other Braginskii-type fluid equations. This allows the potential equation to be included in the matrix formulation. The factor ε is chosen to be small as compared to other processes of interest in order to re-establish the steady state form of the potential equation.

UEDGE can be specified to use either a one-group or two-group neutrals fluid model. In the one-group model, the neutrals are specified to originate at the Franck-Condon energy of 10 eV. The diffusion coefficient is given by $D_g = T_i [em_i(v_i + v_{cx})]$ where $v_{i,cx}$ are the ionization and charge exchange rates. In the two-group model, both a Franck-Condon group and a charge exchange group with energy at the local ion temperature are considered. UEDGE's impurity model assumes spatially uniform impurity concentration and neglects impurity flow velocity. Each density of state has sources and sinks in the forms of ionization, recombination, charge-exchange, and flow loss. Radiative loss rates for a specified impurity specie are obtained from a database created by the MIST code.

7.1.2.3 Numerical Methods

UEDGE performs volume differencing discretization in space while retaining time dependent terms. This method, known as the methods of lines, yields a system of coupled nonlinear ordinary differential equations (ODEs) that are advanced in time for each variable at each spatial location. The ODEs have the form

$$\frac{\partial \eta}{\partial t} + \nabla \cdot \left(v \eta - D \cdot \frac{\partial \eta}{\partial x} \right) = S \quad (7.7)$$

Isolating the time dependent term and spatially discretizing the equations, the ODEs can be rewritten as

$$\frac{\partial \eta_i}{\partial t} = f_i(\eta_1, \eta_2, \dots, \xi_1, \xi_2, \dots) \quad (7.8)$$

where f_i is the sum of the discretized spatial derivatives and the source term. A Taylor expansion of f_i gives

$$\frac{\partial \eta_i}{\partial t} = f_i \approx f_{i0} + \frac{\partial f_i}{\partial \eta_j} (\eta_j - \eta_{j0}) \quad (7.9)$$

Letting $\partial \eta_i / \partial t = \Delta \eta_i / \Delta t$, and rewriting in matrix form, the problem becomes

$$\left(\frac{\mathbf{I}}{\Delta t} - \mathbf{J} \right) \Delta \eta = \mathbf{A} \Delta \eta = f_0 \quad (7.10)$$

where $\mathbf{I}$ is an identity matrix and $\mathbf{J}$ is the Jacobian ($J_{ij} = \partial f_i / \partial \eta_j$). UEDGE uses a Krylov method which does not require finding the inverse of $\mathbf{A}$ directly, but rather, iterates on an initial approximate solution. For steady state (equilibrium) solutions, Δt is set to be large so as to eliminate the identity matrix term. A simple Newton scheme can then be used to solve $\Delta \eta = \mathbf{J}^{-1} f_0$.

7.1.3 BENCHMARKING RESULTS

We compare UEDGE simulation results with experimental data for two typical single-null, H-mode DIII-D shots. Generally, we use the one-group neutrals model without impurities, and do not solve for parallel currents or drifts. These modeling simplifications lead to faster run-times and aid in avoiding convergence failures. Experimental data were provided by Thomson scattering, CER, IRTV, Langmuir probes, photodiodes, and bolometry systems. Not all diagnostics were available for each shot, however.

7.1.3.1 Shot 75915

We model shot 75915 at time 2600 ms. This shot is a single-null plasma with 7 MW of injected beam power, 1.4 MA of plasma current, and 2.1 T on axis. Available diagnostics that were used to compare with UEDGE simulation results include Thomson scattering, IRTV at the divertor floor, photodiodes, and the divertor Langmuir probe arrays. The grid size for this simulation is 28 (poloidal) by 14 (radial). We used the one-group neutrals model and included impurity radiation from Carbon.

From the midplane temperature and density profiles provided by Thomson scattering, we specified the anomalous particle and electron energy diffusion coefficients to be $D_{\perp} = 0.15 \text{ m}^2/\text{s}$, $\chi_e = 0.05 \text{ m}^2/\text{s}$. CER data for the edge plasma was poor for this shot, thus we just assumed $T_i \approx T_e$ at the core, and set χ_i equal to χ_e . This greatly affects our confidence in comparing the power crossing the separatrix, and ultimately radiating or depositing on the divertor plates, since to compute the power flow in the ion channel properly, we need the information provided by the midplane ion temperature profile. We set the recycling coefficient to unity at the divertor plates. This represents a saturated condition at the strike points where for every ion lost to the plate, a neutral hydrogen is born. Neutral albedo at the walls is set at 0.95.

Figures 7.2 and 7.3 indicate that UEDGE matches the Thomson scattering data well with the chosen $D_{\perp}$ and χ_e . UEDGE computes the power crossing the separatrix to the SOL to be 5.24 MW. Compared to the measured injected power of 7.05 MW, that leaves 1.81 MW as the power radiated by the core. Although we can not directly compare the core radiation power values for this time slice, previous bolometer measurements indicate that 1.81 MW is a reasonable value for core radiation.

As we travel from the midplane to below the x-point and towards the divertor, UEDGE becomes less able to simulate the more complicated flows and physical phenomena. Figure 7.4 shows the divertor flux comparisons for the inner and outer strikepoints. Typical of most runs, UEDGE overestimates the power flowing to the divertor, especially at the inner strikepoint. The extremely low value of the experimental divertor power flux at the inner strike point indicates possible detachment of the inner leg, a common phenomenon in many shots.

Figure 7.5 shows a comparison between UEDGE computed H_{α} brightness and experimental values. This measurement is made along the divertor floor, and the two peaks, at 1.1 and 1.6 m, are roughly the strike point locations. H_{α} signals correlate with hydrogen recycling,

and we see that the experiment shows more recycling in the private flux area than does UEDGE. This could account for some of the disparity in divertor power flux, as the higher experimental recycling values would indicate more power being lost through ionization.

7.1.3.2 Shot 79300

Our second modeling example is shot 79300 at time 2250 ms. This shot is also a single-null plasma shot with 10 MW of beam power, 1.6 MA plasma current, and 2.1 T on axis. Out of the 10 MW of injected beam power, approximately 8 MW is accounted for by diagnostics. The grid size for this simulation is 34 (poloidal) by 19 (radial). The recycling for the divertor plates were set to 0.985, and wall albedo were set to 0.95 as in the previous case. We neglected impurities and used the one-group neutral model in the simulation.

The core boundary conditions were different than the previous case. Instead of specifying the core ion and electron temperatures, we specified the power flowing across the core instead. We divided the core power equally between the electron and ion channel, and set the sum equal to the experimentally measured values. By specifying the energy diffusion coefficients (χ_e , χ_i), UEDGE would adjust the core temperature so that the specified core power is satisfied. We still needed to specify the core density and $D_{\perp}$ so as to match the density profile given by Thomson scattering data.

Figures 7.6 and 7.7 show the midplane profile comparisons for electron density and temperature. The poorer match as compared to the previous case may be attributed to the different core boundary condition used in this particular modeling run. However, the ion temperature profiles between UEDGE and experiment, as provided by CER data, match well (Fig. 7.8).

As in the previous case, the UEDGE-computed divertor peak power fluxes are quite a bit higher than the experiment data for both the outer and inner legs (Fig. 7.9). UEDGE's integrated divertor power is also twice as large as experimental values. Introducing impurities in the modeling runs would have reduced this disparity somewhat, but generally impurity radiation does not reduce the peak power flux by more than 25%. We could have also increased the recycling coefficient from 0.985, but as Fig. 7.10 shows, UEDGE is correctly computing recycling at the outer strike point, and overestimating recycling at the inner strike point. Figure 7.10 also shows that, as in the previous case, there is more recycling in the private flux region experimentally.

7.1.4 CONCLUSION

We have successfully performed several UEDGE runs for some of the DIII-D shots. In general, the UEDGE simulation matches well with the Thomson scattering data at the midplane with the chosen transport coefficients in predicting the density and temperature profiles. The results become less accurate as the simulation moves from the midplane to below the x-point and towards the divertor. The UEDGE code overestimates the heat flux to the inner strikepoint of the divertor. The code also underestimates the recycling in the private flux region.

Possible explanations for the discrepancies between UEDGE simulation results and experiment data include toroidal asymmetries, incorrect wall boundary conditions, non-fluid-like phenomena in the SOL, and the modeling accuracy of UEDGE regarding the neutrals and the MARFE effects. Continued work is in progress to improve the modeling capabilities and simulation accuracy.

It is anticipated this work will provide a realistic divertor plasma model to facilitate the study of plasma-driven safety events such as plasma emergency shutdown and erosion of divertor materials.

7.2 ASSESSMENT OF TECHNOLOGICAL PERFORMANCE OF CERAMIC (SiC DOPED WITH BeO) DIVERTOR

7.2.1 INTRODUCTION

Ceramic materials have long been regarded as desirable fusion reactor materials because of their low radiological hazard characteristics. Only very recently have ceramic materials been developed which could potentially withstand the demanding erosion, heat flux, and stress conditions expected within a fusion reactor. In this study we compare the thermal-hydraulic, disruption erosion, and neutron-induced activation radiological hazard properties of the proposed tungsten-niobium technology phase divertor design of ITER, operating in a steady-state condition, with a hypothetical divertor designed from silicon carbide (SiC) ceramic doped with beryllium oxide (BeO). The SiC is doped with BeO in order to increase its thermal conductivity by a factor of about five over conventional SiC.

In Section 7.2.2 of this study, we examine the steady-state heat transfer characteristics of the two divertor designs during normal operation of the fusion reactor. This is done by evaluating the peak or critical heat flux (CHF) which the system can dissipate and comparing it to the steady-state heat flux. Section 7.2.3 outlines the erosion response of the two materials when they are exposed to the intense heating of plasma disruptions. The analysis of disruption erosion presented here is very simplistic and attempts only to provide a relative gauge of the effect of disruptions upon the operating lives of the two divertors. The next section presents a comparison of tungsten and SiC from a radiological hazard standpoint.

There are, of course, many other important issues which would need to be considered in a comprehensive comparison of the ability of tungsten and SiC ceramic to function as divertor plate materials. A more complete study would include high helium production rates in SiC compared to that in tungsten (30-65 times), higher hydrogen production rates in SiC (5-6 times), tritium retention and decay into helium, brittleness of welds in tungsten, helium embrittlement, fatigue, thermal creep, irradiation creep, corrosion by coolant, steady-state erosion by plasma, compatibility of materials, manufacturability, cost, and potential synergistic effects of any of the above.

The general divertor operating parameters used in this study were taken from Kuroda et al. [7.3]. A drawing of the cross-sections of the two designs is given in Fig. 7.11.

7.2.2 EVALUATION OF CRITICAL HEAT FLUX AND BURN-OUT MARGINS

Heat transfer analyses of systems with expected boiling are complicated by variable fluid property values, the dependency upon surface morphology, changing flow regimes due to acceleration of the flow as vaporization occurs, and several other factors. The precise mechanism responsible for the critical heat flux (CHF) phenomenon is not completely understood. In general, when the wall temperature of a heated surface reaches a point where the fluid can exist on and near the surface only as vapor, a rapid increase in wall temperature to the melting temperature of the material may occur, resulting in system failure.

The analysis of CHF begins by determining appropriate fluid properties, the length over which boiling is to be expected within the cooling channel, the flow regime, and an appropriate CHF correlation. Once the critical heat flux has been determined, the burnout margin, defined as the ratio of CHF to applied heat flux (assumed to be 15 MW/m^2), may be evaluated.

Several correlations were examined in order to predict the CHF for expected fluid conditions at the outer divertor strike point. Two correlations with parameter ranges encompassing or reasonably close to the divertor operating point produced results in reasonable agreement with those from the ITER conceptual design activity (CDA) [7.3]. Once the CHF was determined, the burnout margin, which can be thought of as a "factor of safety" for boiling heat transfer equipment, could be calculated. Using the properties at the outer divertor strike point, the burnout margins are presented in Fig. 7.12. Burnout margins for the ceramic divertor design are approximately 10% higher than those for the W-Nb divertor due to the increased surface roughness of the ceramic, even though the thermal conductivity is much lower.

It is important to keep in mind that all of the CHF correlations presented here are empirically derived and can provide only rough estimates of actual divertor behavior. One interesting result of the burnout margin analysis is that burnout during a loss-of-flow accident predicted to occur before a zero flow velocity condition. According to Fig. 7.12, burnout will occur when the burnout margin is unity at flow velocities between 5 m/s and 7.5 m/s. Two key ramifications of this result are that the time before failure of the divertor will be reduced if the divertor fails at higher flow velocities, and fouling or debris within the coolant channel could cause an upstream reduction in flow velocity resulting in burnout. The fact that there will be a total of 1024 divertor coolant channels places special importance upon such fouling considerations.

7.2.3 DISRUPTION EROSION

Of all the different issues treated in this study, the impact of plasma disruptions upon divertor life is perhaps the most critical because disruptions are expected to induce very severe divertor surface temperatures and the plasma-divertor interactions during disruptions are not very well understood.

It is not the intent here to develop a model for predicting absolute disruption lifetimes of the two different divertor materials. Instead, we present two methods for determining the relative disruption erosion of tungsten and SiC ceramic.

We have examined the disruption erosion using two simple methods. In the first method, it is assumed that the heat transfer at the divertor plasma facing surface is conduction limited and the high temperatures developed at the divertor surface will drive vaporization characterized by the partial pressure of the material at the disruption temperature. The second method assumes that disruptions occur over such a small time period that conduction is of little help and the heat transfer is latent heat limited. Here, the vaporization erosion depends simply upon the ratio of disruption energy and the latent heat of vaporization for the particular material multiplied by the mass per unit area of divertor material available to be vaporized. According to Ref. 7.3, the expected number of disruptions to which the divertor will be exposed is 500. Therefore, the per-pulse erosion values calculated by the two methods were multiplied by 500 in order to determine the total erosion expected over the life of the divertor. When the calculated total erosion equaled the divertor armor thickness (2 mm tungsten; 4 mm SiC), it was assumed that a divertor replacement or repair would be necessary. The simple scaling relations used account only for vaporization erosion, therefore, the loss of melt layer material and vapor shielding effects are not included.

In order to investigate the relative disruption tolerance in a realistic fashion, we compared the replacement or maintenance requirements for the SiC and tungsten divertors versus disruption energy. Results from the calculations of evaporative erosion for the conduction and latent heat limited models are plotted as number of divertor replacements or repairs in Figs. 7.13 and 7.14, respectively, for the expected range of disruption energies.

In Fig. 7.13, the relatively poorer performance of the SiC during a disruption is due primarily to its poor thermal conductivity. The disruption induced erosion of tungsten is limited by its heat capacity. For comparison purposes, data from the ITER CDA are also plotted in Figs. 7.13 and 7.14. The ITER data were calculated assuming half of the melt layer was lost and no vapor

shielding effect was included. The results in Fig. 7.14 from the latent-heat limiting model show surprising agreement with the results from the ITER CDA.

Although not treated here, the structural integrity of the divertor material is a major concern, particularly in the case of a brittle ceramic. For a 2 MJ/m^2 disruption, a cursory examination indicates that the SiC could potentially withstand the induced strains, although the values are dangerously close to the yield strength. For the tungsten quite a different situation exists, if the rapid expansion during the disruption is constrained by some other structural member, sub-surface crack propagation and yielding will almost assuredly occur since the high temperature tensile and yield strength values are very low and the elastic modulus is still relatively high. It is almost certain that the SiC will not be able to withstand the high strain rates induced by a 1 ms, 10 MJ/m^2 disruption, without some material or design innovations. The thermal stresses induced in tungsten by the 1 ms, 10 MJ/m^2 pulse are expected to exceed the yield stress of tungsten after only 0.1 ms.

7.2.4 ACTIVATION INDUCED RADIOLOGICAL HAZARD

All other design issues being equal, the SiC+BeO material would certainly be chosen over the tungsten simply due to its much better neutron-induced activation characteristics. The potential radiological doses resulting from a three year irradiation of each divertor plate material were determined. The reference tokamak design used in the radiological analyses had a fusion power of 292 MW and physical dimensions similar to those proposed for ITER. The blanket and shield were both assumed to be SiC. The activation induced radioactive hazard attributable to the different divertor materials was calculated keeping all other tokamak parameters constant.

The doses were evaluated using the methodology outlined by Holdren et al. [7.4]. The results from the analysis are presented in Table 7.1 for the tungsten and SiC divertor plates.

7.2.5 SUMMARY AND CONCLUSIONS

The steady-state and transient operating characteristics, as well as the radiological hazard potential, were examined for two divertor designs, namely, ITER technology phase divertor design and a hypothetical SiC+BeO ceramic design. Here we present a brief summary of the results from each of the sections and the major conclusions from the work.

It was determined that in steady-state operation the tungsten and SiC-BeO divertor plate materials could satisfactorily dissipate the 15 MW/m^2 applied heat flux with a margin of safety of

about 1.5. However, reactor start-up, shut-down, and variations in applied heat flux during steady-state operation were not taken into account. Surface roughness and material thermal conductivity were accounted for in order to delineate the behavior of SiC ceramic and tungsten during nucleate boiling.

It was determined that currently available SiC ceramic would require between 6 and 50 replacements when exposed to 500 disruption pulses with energies up to 3 MJ/m^2 and duration of 1 ms. For the same disruption conditions, tungsten is expected to require approximately 7 replacements. The higher levels of vaporization erosion of SiC would lead to increased divertor replacement, maintenance, and potentially greater contamination of the plasma by lost material. The ability of the SiC to withstand disruptions of higher energy is questionable. Development of ceramics with higher heat capacity and thermal conductivity is necessary in order to improve disruption resistance.

The radiological doses resulting from the full mobilization of activation products in the tungsten divertor plate exceeded characteristic threshold limits of 200 rem critical dose and 25 rem chronic dose by factors of 3 and 16, respectively. All dose calculations were performed assuming a 3 year irradiation period at a fusion power of only 292 MW. The doses calculated here will scale with fusion power, therefore, the actual doses for ITER will be higher since the design goal for fusion power during the technology phase is 1000 MW. The critical and chronic doses resulting from the total release of activation products in the SiC divertor were insignificant compared to threshold limits. Intruder doses for both materials were well below the 0.5 rem limit for shallow burial waste disposal.

It is critically necessary that the disruption behavior of tungsten and SiC be verified by experiments. The disruption erosion calculations are based upon material vapor pressure data which are very sparse for the SiC ceramic. The CHF calculations must also be verified by experimentation, as the correlations used generally did not encompass the divertor operating conditions.

REFERENCES

- 7.1 S.W. Haney et al., Fusion Technol. **21**, 1749 (1992).
- 7.2 T.D. Rognlien et al., "Simulations of Tokamak Divertor Plasmas including Cross-Field Drifts," Contributions to Plasma Physics **32**, 485 (1992).

- 7.3 T. Kuroda et al., "ITER Plasma Facing Components," *ITER Documentation Series*, No. 30, International Atomic Energy Agency, (1991).
- 7.4 J.P. Holdren et al., "Exploring the Competitive Potential of Magnetic Fusion Energy: The Interaction of Economics with Safety and Environmental Characteristics," Fusion Technology, 13, 7 (1988).

Table 7.1. Radiological doses expected from tungsten (W) and silicon carbide (SiC) divertor plates.

	W	SiC	Dose _W / Dose _{SiC}
Critical Dose (rem)	625	1.6×10^{-2}	3.8×10^4
Chronic Dose (rem)	775	9.75×10^{-2}	7.9×10^3
Intruder Dose (rem)	7.6×10^{-6}	5.4×10^{-5}	1.4×10^{-1}

Figure 7.1. Schematic of UEDGE's modeling region.

Figure 7.2. Midplane electron density for shot 75915.

Figure 7.3. Midplane electron temperature for shot 75915.

Figure 7.4. Divertor power flux for shot 75915.

Figure 7.5. H_{α} brightness at divertor for shot 75915.

Figure 7.6. Midplane electron density for shot 79300.

Figure 7.7. Midplane electron temperature for shot 79300.

Figure 7.8. Midplane ion temperature for shot 79300.

Figure 7.9. Divertor power flux for shot 79300.

Figure 7.10. H_α brightness at divert r for shot 79300.

(a) Cross-section at outboard strike point of ITER design (from Kuroda et al.,1991)

(b) Cross-section at outboard strike point of theoretical SiC+BeO divertor plate.

Figure 7.11. Cross-section of (a) the ITER Technology Phase divertor plate design and (b) a potential SiC+BeO ceramic divertor plate design.

Predicted Burnout Margin

Figure 7.12. Predicted burnout margin at the outer divertor strikepoint vs. flow velocity. Results are for the tungsten-niobium technology phase design of ITER. An applied heat flux of 15 MW/m² is assumed at the strikepoint. The burnout margins for the hypothetical ceramic divertor are approximately 10% higher.

Figure 7.13. Number of divertor replacements as a function of disruption energy for a 1 ms disruption duration. Results calculated using the conduction limited model.

Figure 7.14. Number of divertor replacements as a function of disruption energy for a 1 ms disruption duration. Results calculated using the latent heat limited model.

8. ENVIRONMENTAL AND SAFETY INDICES

8.1 AUTOMATION AND GRAPHICAL REPRESENTATION

We have developed a computational program called *Hazard* to translate radioactive inventories in fusion reactors into various environmental and safety indices studied in our previous reports (for example, see Ref. 8.1). A brief description of the features of the program *Hazard* and some illustrative results are presented here.

8.1.1 FEATURES OF THE HAZARD PROGRAM

The reactor design tool *Hazard*, developed by our group, translates and interprets various measures of the environmental and safety consequences of a reactor accident in the forms of tables and charts. The program accepts the isotopic inventory of each component in a reactor as input. As part of the output of *Hazard*, the activity, biological hazard potential in air, biological hazard potential in water, power generation, and energy deposition are plotted chronologically. The time-integrated indices relating to accidents and waste disposals such as critical dose, chronic dose, intruder dose, and deep disposal index can also be charted. The various indices can be compared by reactor cases, as well as disaggregated by reactor components, mobility categories, or isotopes.

Hazard was written as a Mathematica package because Mathematica is high-level, portable programming environment suited to computation, data manipulation, and graphical output. Earlier spreadsheet programs that measured environmental and safety (E&S) indices were inflexible and user-unfriendly. *Hazard* accepts English-like reactor descriptions and charting instructions. It also permits comparisons among arbitrary groups of reactors.

An overview of the execution of the program is described as follows:

(i) Input

A reactor component is described by a text file that lists its properties, including zone, material, volume, density, and isotopic inventory. A reactor with all of its components is described by a group of such files placed in the same directory.

(ii) Execution Commands

As a Mathematica package, Hazard accepts simple functional commands and returns charts and tables to the screen. Instructions, with options allowing for fine-tuning the output, are either of the form:

Hazard[reactor, index, options]

or

Hazard[{ reactor1, reactor2, ... }, index, options].

(iii) Computations

Most indices are calculated by multiplying the matrices of hazard-per-isotope-per-curie by an isotopic inventory developed by Ann Kinzig [8.1 and 8.2]. Some data manipulation follows to prepare the charts and tables.

(iv) Output

We can, for example, compare the biological hazard potential in air of the four reactors PCASS-Li/TOK, ESE1, ESE2, and SiC-He/TOK with the instruction:

Hazard[{"PCASS-Li/TOK", "ESE1", "ESE2", "SiC-He/TOK"}, "BHP in Air"].

Other forms of tables and charts are displayed. Some of them will be discussed in the next section to demonstrate the output capability of the program.

8.1.2 ILLUSTRATIVE RESULTS

Indices can be charted, tabulated, or both. Examples of *Hazard* input and output are shown in Figs. 8.1 through 8.5.

In Fig. 8.1, the top ten isotopic contributions to the biological hazard potential versus time for a single reactor are shown. The graph is produced by the command *Hazard*["PCASS-Li/TOK", "BHP in Air"].

A chart of the critical dose at 1 km (maximum plausible release) of a single reactor, broken down by component and mobility factor is shown in Fig. 8.2. The results are generated by the command *Hazard["PCASS-Li/TOK", "Critical (MPR)"]*.

Similarly, Fig. 8.3 displays a chart of the chronic dose (full release) of four reactors, broken down by zone. The command for this run is *Hazard[{"PCASS-Li/TOK", "ESE1", "ESE2", "SiC-He/TOK"}, "Chronic (Full)"]*.

Figure 8.4 shows the first-wall power density as a function of time for four different types of reactors. The execution is performed by the command *Hazard[{"PCASS-Li/TOK", "ESE1", "ESE2", "SiC-He/TOK"}, "Power"]*.

Various disposal indices for a single reactor, broken down by component, can be generated by the command *Hazard["PCASS-Li/TOK", "Waste"]*. The resulting chart is shown in Fig. 8.5.

8.2 APPLICATION TO COMPARISON OF STAINLESS STEEL WITH ALTERNATIVE STRUCTURAL MATERIALS FOR FUSION REACTORS

The examples in the preceding section were taken from an extensive set of safety and environmental comparisons we conducted, under this project, of stainless steel and alternative structural materials for fusion reactors [8.1]. This work augmented and extended that of the USDOE's Senior Committee on Environmental, Safety, and Economic Aspects of Magnetic Fusion Energy (ESECOM) [8.3], which calculated a diverse array of E&S indices for seven different fusion reactor designs as well as, for comparison, a liquid-metal fast-breeder fission reactor (LMFBR). None of the cases for which ESECOM calculated E&S indices employed stainless steel as a primary structural material, despite the use of this material in a number of earlier fusion-reactor conceptual designs (most notably the vintage-1980 STARFIRE reactor). ESECOM's neglect of stainless-steel designs resulted from the Committee's perception that work in the early 1980s had so convincingly demonstrated the E&S liabilities of stainless steel as a fusion-reactor structural material that its use in commercial fusion reactors would no longer be seriously considered.

That perception subsequently became questionable, however, given the choice of stainless steel as the primary structural material for the International Thermonuclear Experimental Reactor (ITER) and given the possibility that satisfactory alternative materials might not be developed and certified in time for service in first-generation fusion reactors. These considerations, and recognition that clarifying the E&S characteristics of stainless steel vis-a-vis those of alternative materials may be helpful in deciding on the priority to be accorded development of such alternatives, motivate our attention here to comparing stainless steel with alternative materials using our current models.

In the work described here, we used the FuseDose II computer code (on which the HAZARD program described in the previous section is also based) to calculate and compare several indices of E&S hazards for conceptual fusion-reactor designs based on a variety of structural materials -- stainless steel, ferritic steel, vanadium-chromium-titanium alloy, and silicon-carbide -- and, for comparison, the fuel of a liquid-metal fast-breeder fission reactor (LMFBR). FuseDose II is a second-generation code derived from the FuseDose code used in the ESECOM study in the late 1980s. The new comparisons update and extend those of the ESECOM study by adding the stainless-steel case, some new indices, graphical representations of the results, and other refinements.

In almost every class of E&S hazard considered here (the exception being energy deposition/temperature increase during accident), stainless steel poses significant liabilities when compared to other fusion reactor designs. Most importantly, the use of stainless steel greatly reduces the potential advantages of D-T fusion over fission with respect to public exposure due to accidental release of radioactive material. While still offering an improvement over fission with respect to the magnitude of the waste disposal problem, the use of stainless steel would make it impossible for fusion wastes to qualify for shallow burial under current U.S. regulations, while the other fusion reactor materials considered here offer great promise of being able to meet that standard. None of the fusion reactors considered is attractive with respect to the possibility of remote maintenance, though the stainless steel reactor is the least attractive.

Dose hazards are usually dominated by only a few isotopes. Of the fifteen combinations of materials, components, and hazard types sampled here, only 18 precursor isotopes are listed as being ultimately responsible (through production of hazardous isotopes) for 10% or more of a given dose potential or hazard. Of those 18 precursor isotopes, only 5 have natural abundances that are less than 10%, and 10 have natural abundances in excess of 50%. In addition, the precursor isotopes responsible for most of the production of hazardous activation products are frequently found only in minute quantities within the structural material. Of the 18 isotopes listed here, 10 are always found to be less than 1% (by weight) of the material, with an additional 3 isotopes found to be less than 5% by weight of the material. These numbers do not offer much hope for reduction of hazard through isotopic tailoring.

The PCA first wall fares dismally when compared to dose potentials from other first wall materials. An understanding of the material constituents and radioactive isotopes responsible for this performance may help in developing and selecting alloys in the future. The main cause of the difference in the critical and chronic doses between PCA and other materials in this analysis is the relatively large presence of nickel and manganese in PCA (producing the hazardous isotopes Co58, Co60, and Mn54). PCA is 16% nickel (by weight) compared to 0.01% for VCrTi, 0.05% for FeCrV, and 0.006% for RAF, and 2% manganese by weight compared to 0% for VCrTi, 0.35% for FeCrV, and 0.53% for RAF. The PCA intruder dose is dominated by isotopes produced by nuclear reactions with molybdenum; molybdenum is ultimately responsible for 85% of the intruder dose in the PCA first wall considered here. Again, PCA has a higher weight fraction of molybdenum than the other materials considered here; 2% for PCA compared to 0.001% (VCrTi), 0.02% (FeCrV), and 0.00027% (RAF). A first step toward selecting materials that could be

attractive from an E&S perspective would be to avoid materials with significant amounts of nickel, manganese, and molybdenum.

These findings are reported in more detail in Reference 8.2, which is provided together with this report as an attachment. They reinforce the conclusion, which we and others have reached previously, that, while fusion reactors have the potential for large E&S advantages over fission reactors, the realization of those advantages depends in substantial part on the choice of materials used in the reactor structure. The use of high-activation-hazard materials such as stainless steel can seriously reduce or perhaps even eliminate some of fusion's advantages with respect to severe-accident potential, in particular, which could be critical for public and industry acceptance.

REFERENCES

- 8.1 S.K. Ho, T.K. Fowler, and J.P. Holdren, Eds., "Final Report: Code Development Incorporating Environmental, Safety, and Economic Aspects of Fusion Reactors," UC-BFE-027, Berkeley Fusion Engineering, University of California, Berkeley, November 1991.
- 8.2 A.P. Kinzig, J.P. Holdren, and P. Hibbard, " Safety and Environmental Comparisons of Stainless Steel Versus Alternative Structural Materials for Fusion Reactors," Fusion Technol. **26**, 79 (1994).
- 8.3 J.P. Holdren et al., Fusion Technol. **13**, 7 (1988).

Figure 8.1. Chart of the top ten isotopic contributions to the biological hazard potential in air versus time of a single reactor, generated by the instruction Hazard["PCASS-Li/TOK", "BHP in Air"].

Critical Dose (MPR)

Figure 8.2. Chart of the critical dose at 1 km (maximum plausible release) of a single reactor, broken down component and mobility factor, generated by the instruction Hazard["PCASS-Li/TOK", "Critical (MPR)"].

Chronic Dose (Full Release)

Figure 8.3. Chart of the chronic dose (full release) of four reactors, broken down by zone, generated by the instruction Hazard[("PCASS-Li/TOK", "ESE1", "ESE2", "SiC-He/TOK"), "Chronic (Full)"].

Power Density

Figure 8.4. Chart of the first-wall power density versus time of four reactors, generated by the instruction Hazard[("PCASS-Li/TOK", "ESE1", "ESE2", "SiC-He/TOK"), "Power"].

Waste Disposal Indices

Figure 8.5. Chart of waste disposal indices for a single reactor, broken down by component, generated by the instruction Hazard["PCASS-Li/TOK", "Waste"].

9. APPLICATION OF DECISION ANALYSIS -- CONSTRUCTION OF THE STEADY-STATE EXPERIMENTAL TOKAMAK

9.1 BACKGROUND AND PROBLEM STATEMENT

One of the main experimental facilities recommended to advance the fusion energy program by both the Fusion Policy Advisory Committee [9.1] and its successor, the Fusion Energy Advisory Committee [9.2], is the Steady State Experimental (SSX) Tokamak, also currently referred to as the Tokamak Plasma Experiment. The arguments for and against construction of the SSX focus, roughly speaking, are on the tradeoff between its estimated cost (and the equivalent lost opportunity) and the marginal expected benefits from providing early¹ experimental data regarding the physics and technology of steady state operation to the ITER project. The arguments have been qualitative, even when relatively detailed conceptual designs of the SSX accompanied them [9.3]. Here we briefly describe an attempt to analyze the decisions related to the construction and funding of SSX using modern tools of decision analysis.

The decision problem is inherently difficult because complex and developing technologies, as well as domestic and international economics and politics are involved and highly intertwined. More specifically, the following characteristics of the problem complicate its analysis:

- Large uncertainties: technological, economical, and political, partially due to the very long time into the future for which the decisions are relevant.
- Multiple objectives: cost, technical performance (itself of multiple components), time to first successful operation, safety, and environmental impact.
- Multiple decision makers: government, fusion scientific community, utilities, and general public.

The approach taken here is to construct a *probabilistic* model for the decision problem, simplified by aggregating similar details, carefully assess "reference data," and conduct extensive *sensitivity analysis* with respect to those data that are most uncertain or subjective. The model is

¹ At the end of the EDA or the beginning of ITER's construction stage.

constructed such that future extensions, refinements, and relaxation of restrictive assumptions are easy to make.

The main goal of the analysis is to identify parameters, such as those related to costs, conditional probability distributions of the uncertain events, and preference structures of the decision makers, that significantly affect the optimal strategies; as well as those parameters or their value ranges, to which the optimal strategy for the reference case is *insensitive*. The results of a first-round analysis will allow us to identify some of the parameters, that require a better assessment or that need to be agreed upon by "decision makers" differing in their opinions.

9.2 BASIC MODEL AND ASSUMPTIONS

9.2.1 The Decisions

In the basic model, the decisions are about the design, construction, and funding of the SSX. There are two sequential decisions:

- D_1 : to be made "now," among the alternatives: *abolish* the SSX, *design*² it (but not necessarily construct later), and *postpone* decision (for about one year).
- D_2 : to be made about one year later, among: *abolish*, *fully fund* (in which case construction is assumed to take 4 years), and *stretch funding* (in which case construction is assumed to take 6 years).

9.2.2 Chance and Deterministic Variables

The main uncertain events are:

DS , the outcome of the SSX design, if carried out;

RS , the outcome of the SSX experiments³, if indeed it is constructed;

DI , the status of the *steady-state* aspects of ITER's EDA at the time decision D_2 is made.

² The detailed design stage is assumed to take 1 year.

³ Mainly in terms of power and particle handling at divertors, disruption control, and steady-state current drive.

The outcomes of the above chance events have been roughly characterized in a binary way. For example, the outcome of R_S is said to be *favorable* when results of experiments are convincing most of the fusion community that they will extrapolate favorably to ITER (and potentially induce early and intensive preparation for steady-state in ITER); or *unfavorable*, when most of the fusion community regard the results as poor or seriously doubt their extrapolability to ITER. Here, we define P as the steady-state performance of the emerging ITER. In the simplest model, P is a measure of the achievement of the two steady-state goals as they appear in the relevant documents establishing ITER [9.4]: continuous pulse > 2 weeks and neutron fluence > 1 MW/m² yr, then:

$$P = \begin{cases} 2, & \text{if both goals are achieved} \\ 1, & \text{if one goal is achieved} \\ 0, & \text{if none is achieved} \end{cases}$$

The overall cost, C , of SSX and ITER is modeled as a deterministic function of the decisions and the times T_S and T where T_S is the SSX testing duration in the range of 1 – 3 years, and T is the time to ITER's decommissioning in 20 – 30 years.

Figure 9.1 shows the sequential diagram for the basic model, depicting its sequential and asymmetric structure. It shows all possible time sequences of variable realization in the problem. Figure 9.2 shows an influence diagram for the model, depicting the assumed relationships and conditional independencies among its variables (of type: decision, chance, utility). It shows, for instance, that the probability distribution for T is conditioned on the alternatives chosen at D_1 and D_2 and on the outcomes of T_S (if $D_2 = f$ or s) and of P .

A **Sequential Diagram** is an acyclic, directed graph, whose nodes correspond to the variables in the decision problem (chance, deterministic, decision, and value variables denoted by circles, double circles, squares, and diamonds, respectively) and whose directed paths correspond to all sequences of variable realizations in the problem. When more than one arc emanates from a node, each is denoted by a corresponding range of realization values (outcomes or alternatives) for that node. Thus, a sequential diagram depicts the asymmetric and sequential structure of a problem as effectively as a decision tree, but does so much more compactly (a decision tree corresponding to the sequential diagram in Fig. 9.1 would have 594 terminal nodes).

A **Decision Diagram** is an acyclic, directed graph, whose nodes correspond to the variables in the decision problem, and whose arcs denote possible probabilistic dependence when they enter chance nodes, functional dependence, when they enter deterministic or value nodes, and information flow when they enter decision nodes. Lack of arcs among chance nodes implies conditional independence. Thus, for example, the influence diagram in Fig. 9.2 implies that the utility U is a function of the three attributes C , T , and P , and that given T 's direct predecessors, T is independent of R_S .

9.2.3 The Attributes and Preference Structure

Three attributes were chosen for the basic model: cost, C , steady-state performance, P , and time to ITER's decommissioning, T . In the basic model a multi-attribute utility function was used, assuming the attributes are additive independent⁴ and that the decision maker is risk neutral. Rather than scaling the linear multi-attribute utility function, the following function was used:

$$u(C, T, P) = -C - w_T T + w_P P \quad (\$B)$$

where w_T (\$B/year) and w_P (\$B/performance unit) are the (assumed) linear tradeoffs between time and cost and between performance and cost, respectively. Other approaches and assumptions for modeling preferences are discussed in Section 9.4.

9.2.4 Cost and Time Assumptions

The ITER assumed total cost (in today's dollars), if completed in 25 years, is \$10B. Extending (reducing) the testing period costs (saves) \$270M /year (estimate from Ref. 9.6). c_S is the cost of design and construction of SSX, if the latter is completed in 4 years. The reference value for it was .5B\$ [9.3], but it is a sensitivity analysis parameter in the basic model. Related assumptions in the analysis are: design alone costs .1 c_S ; additional year of construction costs $0.25c_S/4 = 1/16c_S$; and each year of testing costs: 2.5 ($0.25c_S/4$).

The minimum time till decommissioning of ITER is assumed to be 20 years and the maximum time is 30 years. If SSX is constructed and its experiments run, the earliest time

⁴ A set of attributes is said to be additive independent if ranking of alternatives based on the attributes only depends on their marginal probability distributions. It can be shown that this property implies a linear multiattribute utility function [6.5].

complete experimental data could be supplied to ITER is within 6 years and the latest is within 11 years. Figure 9.3 shows all 6 timing possibilities that were considered, contingent on the decisions and the outcome of T_S .

9.2.5 Assessment of the Probabilistic Reference Data

Conditional probability distributions that were used as the reference point for sensitivity analysis, were assessed as our personal probabilities. This was not an easy task, since for T only, for example, 30 different conditional probability distributions were used. In most cases, in scenarios when conditioning variables are not realized,⁵ the probability distributions of conditioned random variables was calculated by marginalization.

9.3 SENSITIVITY ANALYSIS AND SOME PRELIMINARY RESULTS

To allow our model to apply for decision makers differing in their attribute tradeoffs, all sensitivity analysis was performed with respect to w_T and w_P . Some typical preliminary result are given below in the (w_T, w_P) space. The parameters for this sensitivity analysis were, in addition to w_T and w_P , c_S , the assumed SSX cost, and $P\{D_S = \text{credible}\}$, the probability that the SSX design is credible⁶. Our model was programmed into ADAM2, a PC-based decision analysis tool [9.7], and the latter used for all computations.

9.3.1 Reference Case ($c_S = \$5B$ and $P\{D_S = c\} = .5$)

- Decision D_1 the *only* optimal alternative is *design* for all plausible ranges of w_T and w_P .
- Decision D_2 the optimal alternative is *fully fund*, except when $D_S = \text{disputable}$ and $D_I = \text{promising}$; in this case, the optimal strategy space is that of Fig. 9.4.

⁵ For example, D_S , the design of SSX, when $D_1 \neq d$ is not realized; D_S is, however, a conditioning variable for R_S, T_S, D_I , and P .

⁶ Since many random variables are conditioned on D_S (see previous footnote), results are potentially highly sensitive to this probability; a "credible" design is defined as one that, in the view of a majority of the fusion community, can be accomplished in construction and has the potential to lead to "favorable" experimental results.

An interpretation of Fig. 9.4 could be that, when the steady state EDA looks better than the SSX design, it is better to abolish when the performance attribute is extremely important, and much more so than the time attribute.

9.3.2 Sensitivity to $P\{D_S = c\}$

- Decision D_1 : the only optimal alternative is still *design* for $P\{D_S = c\} \in [0, 1]$ and for all plausible ranges of w_T and w_P ! Somewhat surprisingly, it seems that even when one is certain that the SSX design will be *disputable*, one should go ahead with it and, possibly (depending on the observed D_S and D_I outcomes) with subsequent construction.
- Decision D_2 : optimal strategy must be the same as in reference case.

9.3.3 Sensitivity to c_S and $P\{D_S = c\}$

Given the insensitivity of the optimal D_1 to $P\{D_S = c\}$, it seems natural to suspect that the cost of SSX, and hence its design cost relative to ITER's cost, is so low that it is optimal to design even for a slim chance of potential benefits. This motivated a sensitivity analysis with respect to the following values of c_S : 1, 1.5, and 2B\$. Figures 9.5 and 9.6 show results in the (w_T, w_P) space for the optimal D_1 and D_2 , respectively.

9.4 DISCUSSION AND FURTHER ANALYSIS

Results of a preliminary sensitivity analysis, some of which were described above, indicate that the optimal strategy is to design the SSX now and fully fund it subsequently, *unless* $D_S = \text{disputable}$ and $D_I = \text{promising}$ are observed, in which case it should be abolished when, roughly speaking, performance is much more valued than time. This strategy seems to be very insensitive not only to the values of the pairwise tradeoffs between the three attributes in the problem, but also, for the reference cost of SSX ($c_S = .5B\$$), to the prior probability that the design will be *credible*. It is only for much higher assumed costs, c_S , that *postpone* becomes optimal as a first decision and *abolish* becomes optimal as a second decision for all combinations of observed D_S and D_I , for some regions in the tradeoff space. Given such a high c_S value, optimal D_1 is only slightly sensitive to the prior probability $P\{D_S = c\}$. The conclusion from this analysis is that, if estimated cost of SSX is not expected to increase significantly, most decision makers should agree on the above optimal strategy.

We need, however, to remember that relatively restrictive assumptions about the preferences of the decision maker (in particular: risk neutrality and attribute additive independence), and that the reference conditional probability distributions may also be controversial. Further efforts, designed to make results of analysis more robust, may include the following approaches, extensions, and relaxation of assumptions:

- Modeling the preference structure using an "independent" attribute approach⁷, allowing for risk aversion in the individual utility functions, and conducting sensitivity analysis with respect to risk aversion coefficients.
- Conducting further sensitivity analysis with respect to other probability and cost (in particular cost of ITER) parameters and, possibly, with respect to time parameters.
- Refining the basic model by more detailed attributes and the addition of uncertain budget variables.

REFERENCES

- 9.1 Fusion Policy Advisory Committee, Final Report, US DOE, Washington, D.C. (1990).
- 9.2 Fusion Energy Advisory Committee, "Report on Program Strategy for US Magnetic Fusion Research", US DOE, Washington, D.C. (1992).
- 9.3 K. Thomassen, R. Goldston, W. Nevins, H. Neilson, T. Shannon, and B. Montgomery, "Steady State Advanced Tokamak: The Mission and the Machine", Report UCRL-ID-109807, Lawrence Livermore National Laboratory, Report UCRL-ID-109807 (1992).
- 9.4 "Establishment of International Thermonuclear Experimental Reactor (ITER): Relevant Documents", IAEA, Vienna, Austria (1988).
- 9.5 R. Keeney and H. Raiffa, "Decisions with Multiple Objectives: Preferences and Value Tradeoffs", Wiley, New York (1976).
- 9.6 "ITER Conceptual Design Activities: Final Report" IAEA, Vienna, Austria (1992).
- 9.7 C.W. Kirkwood, "ADAM2: An Algebraic Decision Analysis Modeling System for Research", Technical Report DIS 91/92-9, Department of Decision and Information Systems, Arizona State University, Tempe, AZ (1992).

⁷ The "independent attribute" approach uses the individual-attribute utility functions, rather than a multiattribute utility function, and makes tradeoffs "transparent" to the decision maker by working in a non-inferior (Pareto) frontier. It also requires far less restrictive assumptions than all tractable multiattribute utility functions.

Figure 9.1. Sequential diagram for a basic model of the SSX problem.

Figure 9.2. Influence diagram for a basic model of the SSX problem.

Figure 9.3. Timing of the SSX project.

Figure 9.4. The optimal strategy for D_2 when $D_S = d$ and $D_I = p$.

Figure 9.5. The optimal strategy for D_1 for various values of c_S and $P\{D_S = c\}$.

Figure 9.6. The Optimal Strategy for D_2 for various Values of c_S

10. IEA COLLABORATION ON DEVELOPMENT OF FUSION SAFETY METHODOLOGY

We have been participating in the International Energy Agency (IEA) Cooperative Program on the Environmental, Safety, and Economic Aspects of Fusion Power [1] in the area of Fusion Safety Methodology. The other groups participating in this work include Japan Atomic Energy Research Institute (JAERI) and the Tokyo Institute of Technology in Japan, and Bologna University and ENEA Frascati Energy Research Centre (ENEA) in Italy.

Recently, an IEA workshop on this subject was held at Naka, Japan in February, 1994. Agreements have been reached by the participating groups to continued the collaborative efforts. The coordination and interrelations between the various fusion safety modeling tasks studied by the collaborating parties have been established.

The scope of this fusion safety systems-study methodology includes the aspects of schematic classification of overall fusion safety system, qualitative assessment of fusion system for identification of critical accident scenarios, quantitative analysis of accident consequences and risk for safety design evaluation, and system-level analysis of accident consequences and risk for design optimization.

A long-term goal is the formulation of a comprehensive methodology integrating these schemes focusing on all aspects of fusion safety. However, most of these schemes are currently being utilized for some specific near-term applications. Safety assessments for ITER and other experimental and power reactors have been performed by these tools.

Shu K. Ho from our group, with collaborators from Japan and the EU, presented an invited paper [2] on this collaborative work at the Eleventh Topical Meeting on the Technology of Fusion Energy in June, 1994. More detailed information about the methodology development can be obtained from this paper appended to the end of this report.

REFERENCES

- 10.1 See, G. Nardella, "Record of Discussion of Meetings of the International Energy Agency Executive Committee on a Cooperative Program on the Environmental, Safety and Economic Aspects of Fusion Power," November 24, 1993.

- 10.2 S.K. Ho, G. Cambi, S. Ciattaglia, Y. Fujii-e, and Y. Seki, "Progress in the Development of Methodology for Fusion Safety Systems Studies," ANS 11th Topical Meeting on the Technology of Fusion Energy, New Orleans, Louisiana, June 1994.

11. PUBLICATIONS AND THESES

11.1 THESES COMPLETED

1. F. Brechtel, "A Comparative Study of the ITER Technology Phase Divertor Design and a Hypothetical Ceramic Divertor Constructed from SiC Doped with BeO," M.S. Thesis, University of California, Berkeley, March 1992.
2. R. Levinson, "An Ordinary Differential Equation Solution for the Transient Temperature Profile of a Fusion Reactor Blanket Following a Loss of Flow Accident" M.S. Thesis, University of California, Berkeley, May 1992.
3. Z. Covaliu, "Representation and Solution of Decision Problems using Sequential Diagrams," Ph.D. Thesis, University of California, Berkeley, November 1993.
4. D.A. Sarigiannis, "Modeling and Safety Optimization of the Fuel Processing System for Magnetic Fusion Reactors," Ph.D. Thesis, University of California, Berkeley, March 1994.

11.2 PAPERS AND REPORTS

1. S.K. Ho, T.K. Fowler, and J.P. Holdren, Eds., "Final Report: Code Development Incorporating Environmental, Safety, and Economic Aspects of Fusion Reactors," UC-BFE-027, Berkeley Fusion Engineering, University of California, Berkeley, November 1991.
2. S.K. Ho, "Controlled Central Fueling for ARIES-III by Compact-Toroid Injection," Proc. 14th Symp. Fusion Engineering, San Diego, California, (1991).
3. S.K. Ho, "Comparison of Activation of Ceramic and Ferritic Structural Materials with D-T and D-He3 Fuel Cycles," Proc. 14th Symp. Fusion Engineering, San Diego, California, (1991).

4. S.K. Ho, in F. Najmabadi and R.W. Conn, et al., "The ARIES-III D-3He Tokamak-Reactor Study," Proc. 14th Symp. Fusion Engineering, San Diego, California, (1991).
5. S.K. Ho, in "The ARIES-I Tokamak Reactor Study--Final Report," University of California at Los Angeles, UCLA-PPG-1323, (1991).
6. J.P. Holdren, "The Role of Safety and Environmental Issues in the Strategy for Fusion Development," invited presentation at the 10th Topical Meeting on the Technology of Fusion Energy, Boston, June 1992.
7. S.K. Ho and T.K. Fowler, "Fast, Approximate Scheme of Activation-Induced Radioactivity Evaluation for Fusion Reactor Systems Studies," Proc. 17th Symp. of Fusion Technology, Rome, Italy, September (1992).
8. D. Sarigiannis, T.K. Fowler, S.K. Ho, and J.P. Holdren, "Tritium Process Modeling: A System Approach," Proc. 17th Symp. of Fusion Technology, Rome, Italy, September (1992).
9. R.W. Conn, D.E. Baldwin, K.H. Berkner, F.L. Culler, R.C. Davidson, S.O. Dean, D.A. Dreyfus, J.P. Holdren, R.L. McCrory, N.F. Ness, D.O. Overskei, R.R. Parker, B.H. Ripin, M.N. Rosenbluth, J. Sheffield, R.E. Siemon, P. Staudhammer, and H. Weitzner, "Report on Program Strategy for U.S. Magnetic Fusion Energy Research," Report of the Fusion Energy Advisory Committee of the U.S. Department of Energy, DOE/ER-0572T, September 1992.
10. Z. Covaliu, "A Multi-attribute Analysis of a Fusion-Energy-Program Decision under Large Uncertainties", 15th Annual Industrial Liaison Program, University of California, Berkeley, California, March 1993.
11. S.K. Ho, C. Annese, T.K. Fowler, E. Greenspan, J.P. Holdren, R. Levinson, M. Lowenthal, and D. Sarigiannis, "Integrated Fusion Environmental and Safety Systems Studies," International Atomic Energy Agency Technical Committee Meeting on Developments in Fusion Safety, Toronto, Canada, June 1993; Journal of Fusion Energy 12, 397 (1993).

12. S.K. Ho, F. Brechtel, and T.K. Fowler, "A Simple Scaling Approach to Calculate Activation Radiological Hazards in a Silicon Carbide First Wall," Fusion Technol. **23**, 321 (1993).
13. S.K. Ho, Ed., "Status Report: Code Development Incorporating Environmental, Safety, and Economic Aspects of Fusion Reactors," UC-BFE-039, Berkeley Fusion Engineering, University of California, Berkeley, July 1993.
14. S.K. Ho and M.D. Lowenthal, "A Systems Module for Environmental and Safety Assessment in Fusion Reactor Studies--Illustrative Results for HT-9 Structures," Proc. 15th Symp. Fusion Engineering, Hyannis, Massachusetts, (1993).
15. S.K. Ho, "Status of Work at UCB in Methodology and Code Development for Fusion Reactor Safety," International Energy Agency Workshop on Fusion Safety Methodology Studies, Naka, Japan, February 1994.
16. S.K. Ho and M.D. Lowenthal, "Assessment of Radiological Hazards of Tungsten Divertors," Third Intl. Symp. on Fusion Nuclear Technology, Los Angeles, California, June 1994.
17. S.K. Ho, G. Cambi, S. Ciattaglia, Y. Fujii-e, and Y. Seki, "Progress in the Development of Methodology for Fusion Safety Systems Studies," ANS 11th Topical Meeting on the Technology of Fusion Energy, New Orleans, Louisiana, June 1994.
18. S.K. Ho, "Scaling Equations to Model Variations in Operating Conditions for Ferritic Steel First Wall Activation Calculations," ANS 11th Topical Meeting on the Technology of Fusion Energy, New Orleans, Louisiana, June 1994.
19. C.E. Annese and E. Greenspan, "Coupled Diffusion-Transport Calculations of Fusion Reactor Activations," ANS 11th Topical Meeting on the Technology of Fusion Energy, New Orleans, Louisiana, June 1994.
20. C.E. Annese and E. Greenspan, "Perturbation Theory to Calculate Inventories for Fusion Reactor Designs," ANS 11th Topical Meeting on the Technology of Fusion Energy, New Orleans, Louisiana, June 1994.

21. A.P. Kinzig, J.P. Holdren, and P. Hibbard, "Safety and Environmental Comparisons of Stainless Steel Versus Alternative Structural Materials for Fusion Reactors," Fusion Technol. 26, 79 (1994).
22. Z. Covaliu and R.M. Oliver "Formulation and Solution of Decision Problems using Sequential Diagrams", to be published in Management Science.
23. S.K. Ho and L.J. Perkins, "Incorporating Safety and Environmental Aspects in Fusion Reactor Systems Studies--the Prerequisites, Impacts and Applications," to be published in Fusion Technol.

12. ATTACHMENTS

- 12.1 "Safety and Environmental Comparisons of Stainless Steel Versus Alternative Structural Materials for Fusion Reactors," Fusion Technol. 26, 79 (1994) by A.P. Kinzig, J.P. Holdren, and P. Hibbard.
- 12.2 "Progress in the Development of Methodology for Fusion Safety Systems Studies," ANS 11th Topical Meeting on the Technology of Fusion Energy, New Orleans, Louisiana, June 1994 by S.K. Ho, G. Cambi, S. Ciattaglia, Y. Fujii-e, and Y. Seki.