Neutronics Issues and Inertial Fusion Energy: A Summary of Findings

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ABSTRACT

We have analyzed and compared five major inertial fusion energy (IFE) and two representative magnetic fusion energy (MFE) power plant designs for their environment, safety, and health (ES&H) characteristics. Our work has focussed upon the neutronics of each of the designs and the resulting radiological hazard indices. The calculation of a consistent set of hazard indices allows comparisons to be made between the designs. Such comparisons enable identification of trends in fusion ES&H characteristics and may be used to increase the likelihood of fusion achieving its full potential with respect to ES&H characteristics. The present work summarizes our findings and conclusions. This work emphasizes the need for more research in low-activation materials and for the experimental measurement of radionuclide release fractions under accident conditions.

I. INTRODUCTION

Nearly forty major conceptual IFE power plant designs have been published during the last 25 years.[1] Even more MFE designs have been published.[2,3] The Committee on Environmental, Safety, and Economic Aspects of Magnetic Fusion Energy (ESECOM) sought to compare various MFE designs using a standardized set of assumptions and a single set of computer codes and data libraries.[2] While most of the IFE studies have addressed ES&H issues, each has used its own assumptions, models, computer codes, and data libraries. Further, each study has defined and calculated a different set of radiological indices. In order for meaningful comparisons to be made between the various designs, a consistent set of radiological indices must be calculated. The present work fulfills this requirement.

II. METHODS AND MODELS

Each of the models used in this work is essential to the final product. Transport models are used to determine particle spectra; radiation-damage models are used to estimate a material's lifetime under irradiation; matrix exponential methods are used to calculate nuclide inventories; plume models are used to predict radionuclide concentrations on the ground and in the air during and after the accidental release of radiation; and dose-effect relationships are used to estimate the health effects of a given release of radioactivity. A brief description of the operation and application of these models is given below.

Three-dimensional Monte Carlo particle transport has been used to model the geometry and materials in each of the designs. Specifically, the TART neutron and photon transport code has been used.[4]

Radiation damage calculations are used to estimate material lifetimes. The International Atomic Energy Agency (IAEA) has recommended that displacement per atom (DPA) be accepted as the preferred method to characterize neutron irradiations.[5] Maximum allowable DPA values for a given material are not well-known. Nevertheless, recommended values do exist for some structural materials of interest to this study. Ref. 6 shows the limits assumed for this work. The limits are based upon knowledge of existing materials and reasonable extrapolation of existing materials to ones that might be available for use in a power plant of the future.

A recent study sponsored by the IAEA sought to identify neutron activation codes able to read standard cross-section libraries, predict accurately the quantity of radionuclides produced in multi-step pathways, calculate accurately light nuclide production, and treat isomeric states.[7] The IAEA study found that only the FISPACT and ACAB codes were "suitable and satisfactory" for detailed fusion calculations. Four other codes performed poorly in at least one of the areas, and five codes were found to be inadequate.[7] The present work utilizes an updated version of the ACAB code.[8]

The task of calculating release fractions is difficult and controversial. Past studies, such as ESECOM, devel-

oped simple, generic heat-transfer models in order to calculate time-temperature distributions within the various power plant components.[2] Piet, Cheng, and Porter[9] defined elemental mobility categories to account for the fact that some elements (or their oxides) have a greater likelihood of mobilization than others. Elements are categorized according to the lower boiling point of the pure element and its oxide. Figure 1 shows the mobility categories and the corresponding assumed release fractions.

The present work has used three different sets of release fractions. The first set, denoted the worst-case release fractions (WCRFs), uses the elemental release fractions shown in Figure 1. The WCRFs make no attempt

to account for the actual chemical form of any radionuclide. The second set of release fractions, the initial-case release fractions (ICRFs), use the same temperature scale shown in Figure 1, but it is applied to the actual chemical composition found in the power plant. Using Flibe (2LiF + BeF$_2$) as an example, the WCRFs assume that the ^{18}F would exist as F$_2$ and 100% would be released. The ICRFs, however, would consider the chemical constituents of Flibe, and would assume that only 1% of the ^{18}F within the LiF would be released (based upon its 1670 °C boiling point) along with 10% of that within the BeF$_2$ (based upon its 800 °C sublimation point). Overall, the ICRFs for Flibe would assume that 5.5% of the ^{18}F would be released.

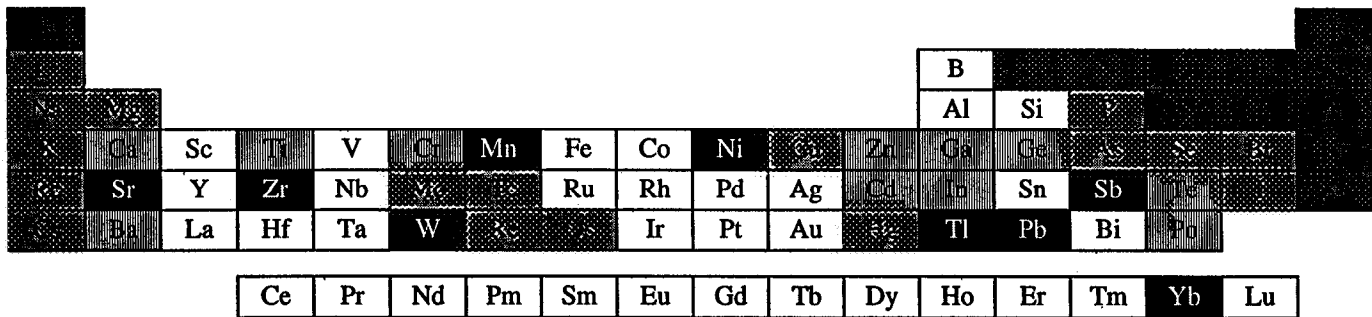| Volatile temperature (°C)	< 20	< 500	<1000	< 1500	> 1500
Release category	1	2	3	4	5
Release fraction (%)	100	30	10	3	1

Fig. 1. Elements assigned to release categories.[9]

The final set of release fractions includes an analysis of likely energy sources in an attempt to establish an upper limit to the radionuclide inventories that might be released in an actual accident. These release fractions, dubbed the mechanistic-case release fractions (MCRFs), assume that all available energy is channeled into the mobilization of the largest contributing radionuclide(s) identified in the worst-case and initial-case analyses. The MCRFs consider energy provided by fusion yield (a capsule is ignited just as the accident begins), radioactive afterheat, and obvious chemical sources.

The Gaussian plume model has been used with release and atmospheric parameters consistent with those used in the ESECOM study.[2] Using the Gaussian plume model in conjunction with radionuclide decay constants, release rates, diffusion coefficients, and atmospheric parameters, it is a straightforward task to obtain radionuclide concen-

trations and deposition rates. Each is adjusted to account for movement away from the plume centerline. The radionuclide concentrations are used to obtain cloudshine and inhalation doses, and the deposition rates are used for the calculation of groundshine, ingestion, and resuspension doses. A dose library containing about 250 radionuclides has been generated using the MACCS2 code.[10]

Estimates of the effects resulting from radiation doses are made using risk analysis and epidemiological data from the irradiation of humans and animals. Acute and somatic effects are estimated using the results presented in NUREG/CR-4214, Rev. 1.[11]

The effective acute dose (early dose) is defined as "that dose which if delivered entirely in one day would induce the same health effects as the actual dose that was delivered over many days".[10] The early dose is reported on

the plume centerline at a distance of 1 km from the power plant. It is obtained by applying effective acute dose reduction factors to the dose received during the various time intervals following the radiation exposure. These are not only time-dependent, they are organ-dependent. Reduction factors have been taken from Ref. 10 for the bone marrow, the gastrointestinal (GI) tract, and the lungs.

Early doses include contributions from inhalation, cloudshine, and groundshine. Inhalation and cloudshine are assumed to occur only during plume passage. Groundshine occurs for the entire 7 day period during which the affected population is relocated. No corrections have been made for evacuation or sheltering. Corrections have been made for added shielding during time spent indoors. Early doses do not include contributions from the ingestion of contaminated food or water.

Early doses are used in a set of two-parameter Weibull functions to obtain hazards for death associated with injury to the bone marrow, the GI tract, and the lungs.[11] The cumulative hazard from all three effects is simply the sum of the individual hazards. The risk of lethality is calculated from this cumulative hazard, and the number of early fatalities is calculated as the risk in a given spatial interval times the population subject to that risk. A population density of 100 people/km^2 has been used. Naturally, corrections are made to account for population away from the plume centerline. The reader is referred to Ref. 6 or Ref. 11 for the details of these calculations.

The chronic effective dose (chronic dose) is a 50-year dose commitment and is reported on the plume centerline at a distance of 10 km from the power plant. It includes contributions from inhalation and cloudshine during plume passage, groundshine after plume passage, and cloudshine and inhalation from resuspended material. The chronic dose also includes contributions from the ingestion of contaminated food and water. Corrections are made to account for surface roughness and for added shielding during time spent indoors.

Cancer fatalities have been estimated using the chronic doses. Conservative assessments of induced cancer fatalities assume a linear model without a threshold dose. The rate constant for cancer fatalities resulting from the irradiation of humans is approximately 3×10^{-2}/person–Sv. This value is the mean of the predicted range reported in the literature.[11]

Fetter points out that radiological hazard indices vary widely in both their usefulness and ease of calculation.[12] While activity is quite simple to calculate and is widely cited, it is often of little use. The number of fatalities that might occur in an accident, however, would be an invaluable result. Unfortunately, such a number is much more difficult to calculate and even harder to defend.

In an effort to strike a balance between meaningfulness and simplicity, a large number of radiological hazard indices have been calculated. These fall into three classes of hazards: accidents, routine and occupational exposures, and waste disposal. Accident indices that have been calculated include activity, biological hazard potential (BHP) in air, early and chronic doses, threshold-dose release fractions, and the number of early and cancer fatalities. For occupational exposures, the contact dose rates have been calculated for all major components in each design. The chronic dose from routine tritium releases also has been estimated. Waste disposal indices include life-cycle waste volume (LCWV), BHP in water, waste-disposal rating (WDR)/intruder dose, deep disposal index (DDI), and annualized intruder hazard potential (AIHP). Details of the calculation of these indices are given in Ref. 2, 3, 6, and 12.

III. POWER PLANT DESIGNS

No single design has been, or probably ever will be, universally accepted as the best fusion energy power plant. Since one of the goals of this work is to learn something of the possibilities in the safety and environmental characteristics of IFE, it is prudent to consider a wide variety of design choices. Five different IFE designs have been analyzed. These include Cascade, HYLIFE-II, Osiris, Prometheus-L, and SOMBRERO.[13-16] These designs include a wide variety of possible design features. They include indirect- and direct-drive targets, solid and liquid first-walls, and traditional and advanced structural materials. Ref. 1 and 6 summarize the major features of each design.

For comparisons between IFE and MFE power plant designs to be done on equal footing, two MFE designs have been analyzed using the same methodology and codes. These designs were selected as they proably represent "bookends" in the full range of MFE designs. The first is a silicon carbide, helium-cooled tokamak (dubbed SiC-He/TOK). The second is a stainless steel tokamak with liquid lithium coolant (PCA-Li/TOK). Additional details for several MFE designs are given in the ESECOM report

and in Kinzig et al.[23] Inclusion of the two MFE cases is intended to highlight key ES&H differences between the two technologies.

IV. SUMMARY OF RESULTS

A brief summary of our results is now given for each of the IFE and MFE designs. For the most part, in the interest of space, our discussion of doses (early and chronic) is limited here to those that would result from the MCRFs. Also, we discuss the hazards associated with a typical target factory. For more detailed results, the reader may consult Ref. 6.

A. Reference target design and factory

Tritium inventories vary widely from one IFE design to another, but the average is about 100 g for modern indirect-drive designs and 200 g for direct-drive designs. Woodworth and Meier give a detailed description of how a target fabrication facility might be operated.[17] Cascade, the oldest of the IFE designs analyzed, conservatively assumed a 1 kg tritium inventory. The MFE designs, which undoubtedly made different assumptions about tritium storage and use, average about 500 g. During an accident, early dose contributions from tritium would be small compared to those from activation products for any of the designs. Tritium, however, would dominate the chronic doses for the Osiris and SiC-He/TOK designs.

Activated, high-Z target materials would make a negligible contribution to the early doses during an accident. Even for the worst-case release fractions, the highest of the early doses would be only 0.13 Sv at the site boundary. High-Z materials could, however, potentially make significant contributions to the chronic doses of the Osiris and SiC-He/TOK designs. It is worth noting that only worst-case release fractions were used for high-Z target materials. Additionally, the high-Z materials seldom would be located in large quantities, and thus, a large-scale release is unlikely.

Contact dose rates from any of the candidate high-Z target materials would be quite high. Since only small quantities would be handled at any time, however, detailed analyses would be required to determine the feasibility of hands-on maintenance.

Routine operation of IFE power plants would include the release of minute amounts of tritium. For the Osiris/SOMBRERO designs, this release would result in chronic doses of 7×10^{-4} Sv at 1 km. Such releases would produce less than a single cancer fatality during 30 years of operation.

Any of the candidate high-Z target materials would qualify for disposal via shallow land burial (SLB). Impure lead, however, would not. It is reasonable to expect that, with impurities, none of the high-Z materials would qualify for SLB. Fortunately, the total LCWV is quite small — far less than 1 m^3.

B. Cascade

Due primarily to activation of the LiAlO$_2$ granules and their Ar impurity, the Cascade power plant design did not fare well compared to the other designs. Since Ar reacts with O$_2$ in an exothermic reaction to produce oxides that are gaseous below Cascade's normal operating temperatures, an argument could not be made for a reduction in the ^{76}As release fraction. As a result, Cascade could produce early doses in excess of the threshold doses for early fatalities associated with injuries to all three organs that have been considered in the present work. The MCRFs would lead to about 20 early fatalities.

The chronic dose from Cascade, using MCRFs, is the highest of the designs that have been analyzed. The chronic dose of about 1.1 Sv, which includes contributions from ingestion of contaminated food and water, would produce about 100 cancer fatalities within a 100 km radius of the accident.

The contact dose rates from Cascade components are about average when compared with components from the other designs. As might be expected, the contact dose rates from the carbon granules and the SiC first-wall are the lowest. Those of the LiAlO$_2$ granules are dominated by contributions from activation products created from impurities. Finally, the contact dose rates from the aluminum 5083 vacuum vessel and the concrete shield suggest that some redesign would be required for limited access to be possible.

In terms of waste-disposal issues, Cascade ranks in the middle of the pack. Although it has the second largest LCWV, it would qualify entirely for SLB after 30 years of operation. Due primarily to its large waste stream, Cascade has the second highest AIHP.

C. HYLIFE-II

Once energy considerations were taken into account, the Flibe release fraction fell substantially, and early doses from HYLIFE-II fell below the thresholds for early fatalities. Apart from the rapid oxidation of the stainless steel structure or an energetic accident, such as an airplane crash, it is unreasonable to expect that a large quantity of Flibe could be mobilized. Although HYLIFE-II would not cause any early fatalities, it could produce as many as 40 cancer fatalities in a population of over three million people within 100 km of the accident.

It is likely that the presence of ^{60}Co in activated steel components will preclude all but the most limited access to regions near the first wall, blanket, and vacuum vessel. In addition, it appears that all Flibe coolant pipes would need to be drained prior to any maintenance due to the high contact dose rates from Flibe at early decay times.

Due to its first wall and blanket being lifetime components, HYLIFE-II would have the second smallest LCWV. In fact, since the Flibe dominates the existing waste stream, it is conceivable that the LCWV could be reduced to about 30 m^3 if the Flibe were reused in another such facility. This would give HYLIFE-II the smallest LCWV by more than 20×. Although the first wall, deflectors, and Flibe trays would not qualify separately for SLB, it is conceivable that these components and the blanket could be considered to be of the same waste stream. If so, then the entire mixture would qualify for SLB. The AIHP from HYLIFE-II is average among the designs.

D. Osiris

Despite a large quantity of stored chemical energy in the carbon composite chamber and blanket, it is unlikely that enough Flibe could be mobilized to cause any early fatalities in the Osiris design. Using the MCRFs, Osiris has the lowest early doses of any of the designs. The bone marrow dose is more than 3× below the early fatality threshold, while the lung and GI tract early doses are each 12× below their respective thresholds for early fatality. The chronic dose from Osiris is the lowest of all designs at just over 0.01 Sv. This chronic dose would cause only a single cancer fatality.

Due to its use of low-activation composites, the Osiris design offers a real opportunity for near-chamber hands-on maintenance. The contact dose rates from the Osiris first wall, blanket, and vacuum vessel are, in fact, lower than those from the Flibe coolant and the concrete shielding. After one week of cooling, the contact dose rate from Osiris' blanket has fallen to about 25 mSv/hr. Once geometric effects are considered (the semi-infinite medium approximation significantly overestimates the true dose rate), limited access near the blanket may be possible.

The Osiris LCWV is only about 720 m^3 -- the lowest of any of the designs. As was true of HYLIFE-II, the majority of this volume would be Flibe. If the Flibe were recycled in another facility, the LCWV would drop to about 250 m^3. Not only would all components qualify for SLB, but they would do so by a large margin. Due to the combination of a low LCWV and a low-activation material, Osiris has the second best AIHP.

E. PCA-Li/TOK

The mechanistic-case early doses for the PCA-Li/TOK design are nearly 40× lower than the initial-case early doses. Although they cannot easily be compared to one another due to different release assumptions, the PCA-Li/TOK early doses are lower than those calculated for Prometheus and Cascade. Using the MCRFs, the PCA-Li/TOK design would not cause any early fatalities. As for chronic dose and cancer fatalities, the PCA-Li/TOK design performs better than Cascade and about the same as HYLIFE-II.

Due to extremely high contact dose rates, there is little chance that any PCA-Li/TOK components could be maintained manually. The PCA components, in particular, have contact dose rates greater than 50 Sv/hr, more than 30 years after irradiation ends. The contact dose rate from the PCA first-wall is more than ten orders of magnitude greater than that from the Osiris first wall.

The LCWV for the PCA-Li/TOK design is average. The intensity of the wastes, however, would be much greater than that for the other designs. The PCA first wall and blanket would exceed the intruder dose limit by 120× and 50×, respectively. The manifold, composed of PCA and FeCrV, would have an intruder dose of 4.6 mSv, and thus, would just qualify for SLB. The total DDI for the PCA-Li/TOK design would be more than 2.0×10^4 m^3, which far exceeds the HYLIFE-II value of 30 m^3. Finally, the PCA-Li/TOK would have the highest total AIHP by about an order of magnitude.

F. Prometheus

Even if the WCRFs were realized, the Prometheus design would produce only 1-2 early fatalities. Using the ICRFs or MCRFs, we find that Prometheus would not produce early fatalities. The chronic dose from Prometheus, although it would be nearly 20× that of Osiris, would still be lower than the doses from the Cascade, HYLIFE-II, and PCA-Li/TOK designs. Using the MCRFs, we estimate that Prometheus would produce 25 cancer fatalities.

Contact dose rates in Prometheus would be lower than those in the PCA-Li/TOK design, but they would still rule out hands-on maintenance. The HT-9 vacuum vessel would be particularly problematic due to its contact dose rate of 640 Sv/hr even 1 year after shutdown. Although dose rates near the first wall and blanket would be lower due to use of SiC, the Pb first-wall coolant would have a contact dose rate of 23 Sv/hr 1 week after shutdown.

The LCWV from Prometheus would be about 3400 m^3. Most of this would be from the blanket. Due to a large inventory of ^{208}Bi, generated from the Bi impurity, the Pb coolant would fail to qualify for SLB by more than 60×. The HT-9 vacuum vessel would exceed the 5 mSv intruder dose limit by 20×. The total DDI for Prometheus would be about 1.7×10^3 m^3 -- 60× that of HYLIFE-II and 10× lower than PCA-Li/TOK. The total AIHP for Prometheus would be the second highest among the IFE designs at 0.35 Sv-m^3/yr.

G. SiC-He/TOK

The SiC-He/TOK design would not produce any early fatalities even when the WCRFs are used. The chronic dose would range from about 0.15 Sv for the WCRFs to 0.04 Sv for the MCRFs. Using the same release fractions, the number of cancer fatalities ranges from 15 to 4.

Contact dose rates in the SiC-He/TOK design would be dominated by ^{24}Na at times of less than 1 week. For SiC components, the long-term contact dose rates would be dominated by ^{60}Co generated from the Fe and Co impurities. Despite contact dose rates at 1 week in excess of 1 Sv/hr for the first wall, some limited hands-on maintenance may be possible near the manifold where contact dose rates would be near 10 mSv/hr at 1 week.

The SiC first wall qualifies for SLB by 60×. The total LCWV, however, is the highest of all designs by nearly 4×. The large LCWV results from the need to replace the entire first wall and blanket structure after only 1 year of irradiation due to SiC's relatively low radiation damage limit. Despite the large waste volume, its low activity gives the SiC-He/TOK design a competitive total AIHP -- equal to that of HYLIFE-II.

H. SOMBRERO

The SOMBRERO design would be fairly benign with respect to accident consequences. Even assuming the WCRFs, SOMBRERO would produce no early fatalities and about 20 cancer fatalities. Due to materials composition and plentiful energy sources, however, the ICRFs and MCRFs vary little from the WCRFs. Thus, the doses change little from one case to another. The early bone marrow dose, for example, only drops from 0.75 Sv with the WCRFs to 0.70 Sv using the MCRFs. The chronic dose remains constant at 0.20 Sv for all three sets of release fractions. Nonetheless, only SiC-He/TOK and Osiris would produce fewer cancer fatalities.

SOMBRERO contact dose rates would be dominated by its Xe gas and the high-density concrete shield. With dose rates of nearly 10 Sv/hr 1 week after shutdown, even limited hands-on maintenance will be difficult.

The SOMBRERO LCWV is slightly lower than that for Cascade, but it could be cut by as much as 2× if the lifetime of the blanket could be extended beyond the 5 year first-wall lifetime. All components would easily qualify for disposal via SLB. The first wall becomes the most activated, and its intruder dose is more than 600× below the limit. Due primarily to its low intruder doses, SOMBRERO has the lowest AIHP.

V. CONCLUSIONS AND FUTURE WORK

The previous section compared the results for each design and each class of hazard individually, but to be meaningful, comparisons must consider all three classes of hazard. The first step in this process is to rank each design within each of the classes of hazard. These rankings by class can then be used to draw qualitative conclusions regarding the overall attractiveness of a design with respect to its ES&H characteristics. Quantitative comparisons have not been attempted as they would require that some form of weighting function be applied to the rankings within each class of hazard — a process that is inherently subjective.

Although quantitative integration of the classes of hazard is not attempted in the present work, some qualitative trends seem quite obvious without the need for weighting function. Osiris has the best overall performance as the best (or tied for best) performer in each of the classes of hazard. The PCA-Li/TOK and Prometheus designs appear at or near the bottom of each of the classes and are the worst overall performers. The remaining designs, Cascade, HYLIFE-II, SiC-He/TOK, and SOMBRERO are in the middle of the pack, and thus, their overall performance depends on subjective decisions about the relative importance of the three classes of hazard.

The results of the present work suggest directions for future research. Such recommendations are based not only upon directions that could yield the best possible designs, but they are based upon directions which have demonstrated the possibility of offering a large return for a given investment. That is, the issues with the highest leverage should be of the most interest to the fusion community.

The accident doses and the resulting early and cancer fatalities that were calculated for the three sets of radionuclide release fractions highlight the tremendously important task of obtaining accurate release fractions. Over the last decade, a series of experiments, performed at INEL, have measured oxidation-driven mobilization of materials as a function of temperature.[18] These experiments have been performed for only a very limited set of materials under a limited set of conditions. Materials of potentially great importance to fusion power plants such as SiC, carbon composites, Pb, Flibe, concrete, Na, and Li have not been included in the experiments. A better understanding of release fractions requires that experiments such as those conducted at INEL be expanded to include many more materials. Such experiments would enable future comparative analyses like the present work to calculate release fractions using time-dependent temperature models.

Not only must the experiments be expanded to include additional materials, they also must include the mobilization of activation products. For example, if SiC is tested, it should include some quantity of Na to simulate the ^{22}Na and ^{24}Na that would be present in a component that has been activated. One possible method for simulating the mobilization of activated materials would be to use samples that have been doped with trace quantities of stable isotopes. The addition of stable ^{23}Na to SiC, for example, could reveal valuable information regarding the mobilization of ^{22}Na and ^{24}Na.

Another high-leverage area of research is in the development and characterization of materials. Neutron irradiation experiments of key fusion materials to fluences relevant to power plants will provide essential data. Structural materials must be irradiated and tested for changes in their strength, brittleness, and swelling. Not only are material lifetimes a function of integrated fluence and DPA, but they may depend upon dose rates effects as well. Irradiation of materials will also aid in the determination of impurity concentrations and can potentially lead to the development of high-lifetime, low-activation materials. Finally, such experiments would supply information that could be used to validate data libraries such as those used for neutron transport and activation calculations.

The largest single improvement to the methodology can be made in the area of radionuclide release fractions. Although accurate release fractions will require much more experimental data than are currently available, significant advances could be made in this area through the use of heat-transfer calculations. A transient heat-transfer calculation, in conjunction with experimental mobilization data, could be used to estimate release fractions more accurately and thus to obtain a more accurate estimate of accident consequences. Eventually, the transient heat-transfer calculation could be linked to some type of chemical-reaction kinetics package, which could provide time-dependent inventories of important chemical compounds (such as potentially volatile oxides).

The number of cancer fatalities has been estimated using the Linear No-Threshold (LNT) model. Recent work in the field, however, suggests not only that the LNT model may overestimate the risk of exposure to low levels of radiation but that some amount of radiation may actually be beneficial.[19-21] Future work should carefully consider use of the LNT model. At the very least, future work should utilize an additional model(s) and represent the uncertainties in our understanding of the effects of exposure to low-level radiation.

ACKNOWLEDGMENTS

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