Recent Accomplishments and Future Directions in US Fusion Safety & Environmental Program

David A. Petti¹, Brad J. Merrill¹, J. Phillip Sharpe¹, L. C. Cadwallader¹, L. El-Guebaly², and S. Reyes³

1. Fusion Safety Program Idaho National Laboratory, Idaho Falls, ID, USA
2. University of Wisconsin, Madison, WI, USA
3. Lawrence Livermore National Laboratory, Livermore, CA USA

Abstract. The US fusion program has long recognized that the safety and environmental (S&E) potential of fusion can be attained by prudent materials selection, judicious design choices, and integration of safety requirements into the design of the facility. To achieve this goal, S&E research is focused on understanding the behavior of the largest sources of radioactive and hazardous materials in a fusion facility, understanding how energy sources in a fusion facility could mobilize those materials, developing integrated state of the art S&E computer codes and risk tools for safety assessment, and evaluating S&E issues associated with current fusion designs. In this paper, recent accomplishments are reviewed and future directions outlined.

1. Introduction

Over the past five years, the US Fusion Safety Program (FSP) at the Idaho National Laboratory (INL) added a new research facility called the Safety and Tritium Applied Research (STAR) Facility; the US Fusion Program reentered the International Thermonuclear Experimental Reactor (ITER) project; and the National Ignition Facility (NIF) has early light in 2004. STAR is a US Department of Energy (DOE) National User Facility with the infrastructure to support both tritium and safety-related materials research activities important to the development of safe and environmentally friendly fusion power. With the US Fusion Program in the ITER Project, the US safety community is redirecting its research program to accommodate some of ITER’s safety needs. This research will utilize the unique capabilities of the STAR facility, and also require the application of US capabilities in the area of fusion safety codes, risk analysis and reliability methods. Beyond ITER, US safety activities continue in support of longer-range inertial fusion energy and magnetic fusion energy system studies. The US also participates in a series of international collaborative tasks under the International Energy Agency’s Implementing Agreement on the Economic, Safety, and Environmental Aspects of Fusion Power. US safety personnel are active in almost all of the eight task areas, including tritium and activation product source terms, thermofluid modeling, component failure rates, radioactive waste studies, magnet safety studies, and power plant design studies. A recent status of US S&E activities is found in Reference 1. In this paper we site recent achievements and outline plans for the next five years for US S&E activities.

2. Safety and Tritium Applied Research (STAR) Facility

Much of the US experimental work to support ITER and longer term fusion safety needs can be performed in the STAR Facility, a US DOE National User Facility at the INL.² STAR comprises capabilities and infrastructure to support both tritium and non-tritium research activities important to the development of safe and environmentally friendly fusion energy. STAR is classified as a Radiological, Low Hazard Facility, and it is restricted to a facility total tritium inventory of less than 16,000 Ci, the threshold limit for a DOE Category-3 Nuclear Facility.³ To assure compliance with this limit, the tritium inventory is administratively controlled to 15,000 Ci. The STAR facility exists within a multi-room complex in two adjoining buildings located at the INL Reactor Technology Complex. The facility provides about 400 m² of working area for bench scale experiments and limited engineering scale tests utilizing conventional laboratory infrastructure (power, water, pressurized air, HVAC, controlled ventilation hoods, and inert-gas gloveboxes) and tritium handling infrastructure. Figure 1 shows the STAR facility floor plan, indicating experiment and equipment layout. Dedicated tritium handling equipment consists of inert-gas gloveboxes, a tritium cleanup system (TCS) to
process glovebox and experiment tritiated gaseous effluent, a tritium storage and assay system (SAS) that allows distribution of tritium for use in experiments, glovebox purge control systems, and tritium monitoring instrumentation. Error! Reference source not found. and Error! Reference source not found. show photographs of the STAR TCS and SAS, respectively.

Tritium is received in STAR in 1,000 Ci aliquots using Type A shipping containers. STAR became operational with tritium in January 2006. Experiments are currently being performed in STAR to evaluate the use of Be as
a REDOX agent for the molten salt, Flibe (2LiF-BeF$_2$), and to demonstrate the effectiveness of Be to inhibit corrosion in a molten salt/ferritic steel system as part of a collaboration with Japan.$^4$ Future plans include studying tritium permeation behavior in candidate fusion coolants (e.g., Flibe, LiPb, Li), structural materials, and various barrier coatings. Figure 4 illustrates a test configuration under development for evaluating the influence of thermal cycling on the permeation reduction factor for unirradiated and neutron-irradiated coatings.

3. **Dust**

Dust is produced in fusion devices by energetic plasma-surface interactions.$^5$ As the amount of dust increases, potential safety and operational concerns arise. The dust may contain tritium, may be radioactive from activation products, and may be chemically reactive and/or toxic. Off-normal events in fusion facilities like ITER could mobilize the dust, which must be considered in safety analyses. Thus, accurate characterization of the amount and behavior of dust generated during ITER operation in conjunction with a better understanding the mechanisms of dust generation and transport will permit more accurate estimates of dust quantities and distributions inside the ITER vacuum vessel. The goal of our dust research is to provide a defensible strategy to demonstrate that ITER dust safety limits are achievable with well-established guidelines and methodologies for ITER operation. Our plans include refinement of dust inventory estimates in ITER, development of dust mobilization and chemical reactivity data under off-normal conditions and development of a dust removal strategy that will demonstrate compliance against dust safety limits in ITER and not hamper operational flexibility of the machine. Addressing regulator concerns regarding dust throughout the ITER regulatory approval process is a top priority and involves coordination of dust R&D activities in the international fusion community.

Attributes of dust, such as size and mass distributions, shape, and chemical composition, are determined by formation mechanisms that may be different in various locations inside the ITER vacuum vessel. Dust produced by plasma-surface interactions during normal operation and off-normal events will be different than dust produced from co-deposited or plated materials subject to mechanical and/or thermal stresses. Studying dust collected from existing fusion devices, plasma experiments, and dedicated experiments will help identify these differences and provide a more complete view of the types of dust possible in ITER. A consistent methodology is required for comparative analysis of the dust, and analysis techniques must encompass the widely varying nature of the dust. Microscopy (optical and electronic) is suitable for count distributions of particles with regular shape, whereas mechanical methods (e.g. filters, impactors, electrostatic classifiers) are needed to provide mass distributions of particles with irregular shape. Surface area distributions are determined by adsorption and micro-porosity measurement techniques. Using a single technique and extrapolating to other distribution types insufficiently describes dust expected in ITER.

We have also developed a scaled experiment system to investigate mobilization of dust for anticipated system behavior during accidents. The experiment will approximately simulate geometry, flow conditions, temperature distributions, and structural components (e.g. divertor and first-wall grooves, and locations behind the divertor and other components) that effect mobilization of dust with different characteristics. The test plan also includes experimental investigation of the chemical reactivity of dust with various characteristics and exposure configurations (e.g. dust in grooves). Analyses of postulated accidents in ITER have been performed to initially determine the limiting quantity and location of dust in the vacuum vessel. As these limits are refined and better understood, emphasis should be placed on dust monitoring systems that assure the safety limits are not approached during ITER operation. Additionally, methods to remove dust are needed to counter activities that potentially increase dust levels and adversely affect ITER operation. This activity will also involve development and testing of systems and techniques for dust monitoring and removal that satisfy safety requirements and that may be incorporated into ITER.
4. Tritium Behavior in Fusion Materials

The Tritium Plasma Experiment (TPE) generates steady-state plasma consisting partially of tritium species. Plasma impingement onto a solid material target produces plasma-surface interactions by mechanisms equivalent to those expected in fusion reactors. TPE’s ion flux and energy at the target sample of nearly $10^{19} \text{cm}^{-2}\cdot\text{s}^{-1}$ and up to 200 eV, respectively, are suitable for simulating plasma exposure of material and structures placed in the divertor region of the future fusion reactors. TPE is unique in its capability of exposing material to tritiated plasmas with an intensity matching fusion plasmas, thereby providing a system with greatly enhanced sensitivity for understanding tritium migration and retention behavior in bulk components and debris found in fusion reactor chambers.

The TPE and the associated STAR/INL facilities will be used to study tritium uptake, retention and permeation in plasma-facing components. Accountancy of tritium inventory is a substantial challenge for fusion reactors, and tritium transport properties are required for safe-handling practices and to prove inventories remain below regulatory limits. TPE will be used to obtain measurement of bulk tritium permeation parameters, such as mass transport coefficients for diffusion, solubility, and dissociation/recombination rates. Fusion mono-materials (i.e. non-mixed, singular modules) must be suitably characterized, and current materials of interest include beryllium, tungsten, carbon, and stainless steel. Sample disks of test material are mounted in a water-cooled copper sample holder. These disks will be exposed to intense plasma for different times, temperatures, and ion fluxes. After exposure, the samples are transferred to a furnace for thermal desorption spectroscopy. Mixed-materials effects on tritium transport will also be studied with properly prepared samples or seeded plasmas.

Fusion reactors will likely use bonded structures on all areas in contact with the plasma. These structures will typically be composed of beryllium, tungsten, or carbon brazed to a copper alloy. Permeation experiments in TPE will examine the effect of the front surface with its recombination rate coefficient, the effect of the braze material, the delaying effects of trapping, and the release of tritium into the coolant at the rear of the duplex structure. Such experiments will allow determination of important parameters affecting the uptake, retention and migration of tritium in the different materials and their interface, as well as directly measure the tritium permeation expected during operation of a fusion reactor. Post-exposure examinations of the duplex structures will used to determine the trapping or holdup in the different materials. In addition to studying hydrogen transport properties, TPE could also be used for engineering certification of these structures. The effects of neutron dose and irradiation temperature on trapping in PFCs could be examined with TPE. Because grain size and impurity levels may affect the trapping, it is necessary that realistic, reactor-relevant candidate materials be used in these experiments so the data can be used to project tritium inventory for the reactor. The effects of helium bubbles for the more highly irradiated materials may also be considered.

We plan to use tritium to accomplish the activities listed above because of its high detection sensitivity, allowing accurate accountancy in materials balance. Oftentimes measurements attempted with protium or deuterium are unreliable because instruments detecting these isotopes have limited sensitivity during the experiment time period, are subject to competing signals from other species, or have significantly high background concentrations. The ability to use tritium also allows investigation of isotopic effects on transport parameters. TPE has been re-assembled and is operational in STAR. Figure 5 is a photograph of the first plasma produced by TPE in STAR. Testing of fusion-relevant PFC is...
5. Fusion Safety Computer Codes

Our work during ITER Engineering Design Activities focused on developing fusion-specific safety models to be used in ITER safety studies (e.g., chemical reaction rates for fusion materials, cryogenic condensation of air and water, ice formation on cryogenic surfaces, and dust transport behavior). By using state-of-the-art fission safety codes and adapting them for fusion needs or developing fusion-specific computer codes where none existed, we developed the first self-consistent systems-level safety model of a D–T fusion facility (ITER), that was used to understand the integrated response of ITER to off-normal events, and to show that ITER could meet the no-evacuation safety objective. Major fusion safety computer codes developed in the US include:

- MELCOR - a fully integrated, engineering level computer code that models the progression of accidents in fission and now fusion power plants, including a spectrum of accident phenomena such as reactor cooling system and containment fluid flow, heat transfer, and aerosol transport,
- ATHENA - the advanced thermal hydraulic event network analyzer to analyze cooling transients,
- CHEMCON - a 1.5-D heat transfer code to analyze long-term loss of coolant accident behavior in fusion systems,
- MAGARC - a coupled electrical and heat conduction code developed to analyze magnet arcing accidents,
- TMAP - a tritium migration code that treats multi-species diffusion in composite materials with dislocation traps, plus the movement of these species from room to room by air flow within a given facility.

Our work over the past 5 years has focused on improving these safety codes to support the safety assessments of the ARIES, APEX and ALPS designs by accommodating different materials, coolants and geometric configurations. In addition, we have continued to verify and validate (V&V) our fusion safety codes by collaborating in international code benchmarking experiments being performed by Japan and Europe. Specifically, we have

- Developed a coolant property database for potential fusion coolants that can be used with ATHENA or MELCOR including properties for Flibe;
- Incorporated models in MELCOR for liquid metal fire assessments;
- Participated in V&V studies as part of the international fusion safety code benchmarking activity for the Japanese Ingress of Coolant Experiment (ICE) facility which studied the ITER pressure suppression system concept, and for Experimental Vacuum Ingress Test Apparatus (EVITA) cryogen experiments which studied ice formation on cryogenic surfaces.

We have continued our code development work in the area of magnet safety and dust oxidation. The MAGARC code was recently modified to analyze unmitigated quench events in ITER poloidal field (PF) magnets. These modifications include the electrical characteristics of the two-in-hand winding pair of the ITER PF coils and limits on the number of arcs that can form in the magnet during unmitigated quench events based on an energy minimization principle. In the future, MAGARC capabilities will be expanded to treat arcs in magnet busbars by including the magnetic effects of the arc that forms between the leads of a busbar. As part of an international collaboration, this new capability will be validated against data that has recently been obtained from the MOVARC experiment in Germany.
A beryllium dust layer oxidation model has recently been developed for the MELCOR code. This model was developed for ITER and simulates a layer of dust that has settled onto a given surface inside of a fusion device. This new model is based on measured oxidation reaction rates for fully dense beryllium and binary gaseous diffusion of oxygen or steam into the dust layer. Figure 6 contains a comparison of the predicted reaction rates from this new model compared to beryllium reaction data developed at the INL.

In the future, we will expand and generalize much of our codes’ features to improve their versatility and potential for use on ITER. Code quality assurance (QA) will be maintained within the ITER QA Standards program, presently under development, and subject to review by the regulatory authority. We also plan to continue our effort in the area of plasma safety by monitoring issues regarding plasma control, plasma shutdown, and plasma disruptions as they affect safety. Specifically, we will examine safe shutdown concepts, runaway electron generation and deposition, and liquid surface induced disruptions and their severity to support future reactor concept design studies.

6. Risk and Reliability

A part of the fusion mission is to demonstrate that a fusion power plant can be reliable, as well as safe and environmentally benign. Our work in this area is to collect and analyze operating experience data to indicate system reliability and facility availability as well as use the refined data in probabilistic safety assessment, risk and safety analysis, and reliability, availability, maintainability, and inspectability (RAMI) analyses. Initially, failure rate data were obtained by collecting data from published reports and handbook values that were applicable to fusion components. The advent of the International Energy Agency’s (IEA) cooperative agreement on fusion safety allowed more cooperation with other risk analysts and the opportunity to collect failure data from operating tokamaks, notably the DIII-D and JET tokamaks, and other facilities, such as the US Tritium Systems Test Assembly, Germany’s Tritium Laboratory Karlsruhe, and Japan’s Tritium Process Laboratory. IEA task participants have made comparisons between these independent data sets to validate the data and demonstrate that fusion does have reasonable component failure rate data values to apply to analysis needs. The analyses are now focusing on systems that have safety significance due to high energies controlled or flowing in the components, or due to confinement of hazardous radiological or toxicological materials. Components that affect machine availability are also analyzed.

The near-term future plans for the reliability data task are to complete work on large, pulsed power supplies, and then examine plasma heating systems, including microwave heating and neutral beam injection. More data will be collected from existing tokamaks to apply in a “stair-step” fashion to analogous systems on ITER and other future facilities. Prior to ITER operation, there will be an activity to establish a trouble reporting and machine engineering data collection system similar to other, existing tokamaks. That operating experience data collection provision will allow engineering data reduction and analysis. The ITER operations data are power plant relevant and will “stair-step” in application to future machines, such as the DEMO.

A new aspect of the task is addressing fusion worker safety. By US DOE direction, fusion facility workers shall be protected such that the risks to which they are exposed at a fusion facility are no greater than those to which they would be exposed at a comparable industrial facility, and risks to
workers shall be maintained as low as reasonably achievable. Some initial steps in worker safety assessment were to construct a logic diagram for the safety of personnel in fusion facilities, and a method called the Worker Exposure Failure Modes and Effects Analysis has been developed to assess the risks of component failures that could endanger personnel by exposure to released energies or hazardous materials. An initial quantification of worker injury rates in fusion and accelerators has been performed. Future work in this area includes construction worker and operations personnel injury rate estimates to support ITER safety planning, and then a room-by-room assessment of ITER equipment, with identification of the energies and hazardous materials that are utilized or controlled by the equipment. The room-by-room assessment will support industrial safety and other aspects of ITER design activities. As the reliability data work continues, attention will be given to component failures that produce local area effects that might present hazards to personnel (e.g., arcs, fires, pressure releases, hazardous material releases). These local area effects data will be recorded in the component failure rate data reports to support worker safety assessments. Another part of the personnel safety task is to analyze failure rates of protective equipment, such as room oxygen monitors, radiation monitors, and other personnel safety monitors.

7. Radioactive Waste Management

Radioactive waste management is an issue that is significant not only to the fusion community, but also to the global environment. As fusion energy becomes a contributor to the world’s energy supply by the mid 21st century, managing the many thousand cubic meters of active materials after plant decommissioning represents a real challenge to the fusion industry and cannot be relegated to the back-end as only a disposal issue. With increasing costs, tighter environmental controls and the political difficulty of building new repositories worldwide, the disposal option may have to be replaced with more attractive scenarios, such as:

1. Recycle and reuse within the nuclear industry
2. Clear or release to the commercial market if materials contain traces of radioactivity

There is a growing international effort in support of this new trend. Over the past 10 years, we have introduced the clearance option to various design concepts, such as tokamaks and stellarators. In general, none of the in-vessel components (blanket, shield, vacuum vessel, and magnets) can be cleared from regulatory control even after an extended storage period of 100 y, according to the IAEA and U.S. Nuclear Regulatory Commission clearance guidelines. Only the bioshield (and perhaps the cryostat) qualifies for clearance. Representing 75-80% of the total waste volume, the bioshield release saves a substantial disposal cost and, more importantly, frees ample space in geological repositories for other radwaste.

Just recently, we have applied the recycling approach to the non-clearable in-vessel components of the most recent ARIES design – a compact stellarator. As Figure 7 indicates, all ARIES-CS components can potentially be recycled using a combination of advanced remote handling (RH) equipment and conventional equipment. Target dose rates for RH techniques are presently evaluated as 3,000 Sv/h for the advanced limit and 0.01 Sv/h for conservative limit, consistent with practical dose minimization. Continual removal of the radionuclides during recycling will shorten the required cooling period to meet the radiological limit for advanced remote handling.
equipment. Transmutation of these byproducts in fusion devices\textsuperscript{33} is one of the means being explored to avoid their geological burial.

8. Magnetic Fusion Safety Design Support

The Fusion Safety Program has historically provided safety design support to a number of national and international conceptual design activities. Most recently, these include ITER, ARIES, ALPS, APEX, FIRE, and US inertial fusion design activities including HYLIFE-II and ARIES-IFE.

Since the US reentered the ITER project in 2004, the FSP has been able to once again provide safety support to ITER in the areas of code and magnet safety. We are presently working with the ITER Joint Work Site (JWS) International Team (IT) to establish the quality assurance documentation for the MELCOR code and safety analyses performed with the MELCOR code needed to prepare ITER’s Report on Preliminary Safety (RPrS), and to provide the ITER JWS with the latest version of MELCOR. Unmitigated quench analyses have just been completed for ITER’s large PF coils using the MAGARC code which show that this event is not a safety concern for the ITER device, although analyses for the central solenoid coils still need to be undertaken.

As part of the US participation in the ITER Project, the US FSP is participating in the licensing process for the US Test Blanket Module (TBM). The US TBM is based on a dual coolant liquid lead-lithium (DCLL) blanket concept. This blanket is constructed of reduced activation ferritic steel (RAFS). Helium is used to cool the blanket first wall and internal wall structures, and a self-cooled liquid breeder, Pb–17Li, is circulated for cooling the interior of the blanket and for tritium breeding and removal. A SiCf/SiC composite insert inside of the breeding zone is used as an electrical insulator to reduce the impact of the magnetohydrodynamic pressure drop associated with the circulating Pb–17Li and as a thermal insulator to separate the high temperature Pb–17Li from the helium cooled RAFS structure. The FSP has been supporting the US TBM design team by assessing the impact this TBM has on overall ITER safety.\textsuperscript{34} This safety assessment, based in part upon a generalized failure modes and effects analysis, addressed a number of concerns that are directly caused by TBM system failures, such as coolant leaks from the TBM directly into the ITER vacuum vessel (VV). These events create safety concerns, such as VV pressurization, confinement building pressure build-up, tritium and activation products release from the TBM system, and hydrogen and heat production from chemical reactions. In the Spring of 2006, a TBM Safety Workshop was held in Cadarache, France by the ITER IT to review the safety analyses performed for all TBMs planned for ITER, and to relay to the TBM Parties the licensing strategy for TBMs and the TBM input required for writing ITER’s RPrS. In the future, ITER is requesting a more complete Dossier on Safety per TBM concept from the participating parties.

The ARIES team is presently examining a compact stellarator conceptual fusion power plant design called ARIES-CS. ARIES-CS is a 1000 MW\textsubscript{e} power plant design that contains many innovative features to improve the physics, engineering, and safety performance of this power plant. The ARIES design team has also adopted the DCLL blanket design for ARIES-CS. The FSP is supporting this design effort by assessing the safety and environmental performance of ARIES-CS by examining the response of ARIES-CS to accident conditions. These accidents include conventional loss of coolant and loss of flow events, an ex-vessel loss of coolant event, and an in-vessel loss of coolant with bypass event that mobilizes in-vessel radioactive inventories (e.g., tritium and erosion dust from plasma facing components). We are also providing a systematic assessment of the design to address key safety functions such as confinement, decay heat removal, chemical energy control and plasma shutdown. Our analyses demonstrate that the decay heat can be safely removed from ARIES-CS and the facility can meet the no-evacuation requirement.\textsuperscript{35} This result was obtained in large part by segmenting the reactor into six separate cooling loops to reduce the quantity of helium that can be released into the VV during loss-of-coolant accidents, and by venting VV pressures in excess of 1.5 atmospheres into the cryostat, which serves as an expansion volume to confine this helium and mobilized radioactive inventories from the VV.
As dictated by specific project needs, we use our suite of safety computer codes to assess critical safety issues. As each design evolves, we help improve these designs from a safety perspective, and evaluate how well each design concept implements these safety functions to demonstrate the S&E potential of fusion.

9. Inertial Fusion Energy Safety Design Activities

Excellent progress has been made in the understanding of S&E aspects of inertial fusion energy (IFE) which has allowed for a better understanding of the behavior of radioactive sources and hazardous materials within the power plant, identification of the energy sources that could mobilize those materials in case of an accident, and assessment of waste management options for IFE. Currently, ongoing IFE S&E studies are focusing on emerging design concepts, which include support to the US High Average Power Laser (HAPL) program for development of a dry-wall, laser-driven IFE power plant, and collaboration with the Z-Pinch IFE program for the production of an economically-attractive power plant using high-yield z-pinch-driven targets.

The HAPL program is carrying out a coordinated multi-institutional effort to develop IFE based on lasers, direct-drive targets and a dry wall chamber. The armor/FW configuration is currently based on tungsten and ferritic steel F82H as preferred armor and structural materials, respectively. A potential breeder concept that can be integrated with the chosen FW protection scheme is based on a self-cooled liquid lithium blanket.

Figure 8 shows a cross-section of the HAPL chamber.

Figure 8. Cross-section of HAPL conceptual chamber using a self-cooled Li blanket.

Recently, the fusion safety group at LLNL led a safety analysis for the HAPL conceptual design using a liquid Li blanket. First, a loss-of-flow accident (LOFA) was simulated in order to assess the decay heat removal capability of the proposed chamber concept. It was found that the decay heat rapidly transferred through radiation to the cooler structures (shielding and confinement building). Also, a version of the thermal-hydraulics code MELCOR modified by INL and capable of assessing Li fires, was used to simulate a Li spill in the HAPL main cooling system. The results showed that even in the case of a major spill with simultaneous air ingress (300-tonnes spill and a 1-m diameter break in the confinement building), the temperatures of all the structures would remain below melting. In such circumstances, the maximum hazard would be the release of the tritium contained in the liquid Li, and therefore, it is critical for the safety of the design to minimize the tritium inventory in the coolant loop. Possible approaches may include: gas recovery systems, getters, cold traps, and molten salt and permeation technologies. Future work by the HAPL team should include the detailed design of the tritium recovery system for the HAPL concept using a self-cooled liquid lithium blanket.

Another area of ongoing IFE S&E studies is related to the Z-pinch driven IFE (Z-IFE) concept. A thick-liquid-wall chamber has been proposed to absorb the target emission (X-rays, debris and neutrons) and mitigate the blast effects on the chamber wall. The target is attached to the end of a conical shaped recyclable transmission line (RTL) made from a solid coolant (e.g., frozen Flibe), or a
material that is easily separable from the coolant (e.g., steel). The RTL/target assembly is inserted through a single opening at the top of the chamber for each shot. Preliminary neutronics calculations and S&E assessments have been performed for the proposed Z-IFE chamber that uses a Flibe thick-liquid-wall protection and a carbon composite first wall. Neutronic and waste analyses of what types of materials would be acceptable as RTLs were completed due to the requirement that the RTL materials must be recycled in order to control the amount of activated waste. It was found that iron was an acceptable material; however, there were also other materials that performed very well. Frozen Flibe, if it can handle power transmission, would be a very attractive candidate for the RTLs, as separation from the protecting Flibe pocket would not be necessary, and the waste disposal rating and the contact dose rates are very low compared with other materials. We also applied the concept of “clearance indexes” to determine if the used RTLs could be cleared from regulatory control for unconditional re-use. In the case of Flibe, the RTLs could qualify for regulatory clearance (clearance index < 1) after only 3 days of cooling. As an outcome of this work it was recommended to continue with iron as the baseline material for the RTL but have a parallel effort to evaluate the feasibility of using frozen Flibe RTLs.

Future work in the IFE S&E area will continue supporting the HAPL and Z-IFE programs and additional analyses will have to be performed as design details for these power plant concepts become available. Also, as the National Ignition Facility (NIF) project in LLNL reaches completion and prepares for operation, more attention is being paid to NIF S&E aspects. It is predicted that a significant effort will be devoted towards the development of integrated radiation safety assessments for NIF in the next few years. In addition to fusion ignition, NIF will provide data that can benchmark and improve the predictive capability of various neutronics and safety computer codes that will be needed to design future IFE power plants. Finally, NIF will be able to demonstrate the safe and environmentally benign operation that is important for IFE, and the proper quality assurance in minimizing both occupational and public exposures to radiation.

10. Summary and Conclusions

US S&E research continues to help improve fusion facility design in terms of accident safety, worker safety, and waste disposal. The R&D underway and currently planned in the areas of dust and tritium source terms will answer important questions for ITER and future machines. Ongoing verification and validation activities for fusion safety codes, as well as risk and reliability methodology, will provide enhanced confidence in the evaluation of public and worker safety in fusion facility designs. The resurgence of nuclear fission reactor construction activities worldwide will cause increased attention to waste management issues associated with nuclear power which in turn should assist fusion as it develops a long term waste management strategy consistent with on-going governmental regulation. Finally, safe and environmentally sound operation of both ITER and NIF will be important public demonstrations of the S&E potential of fusion.

REFERENCES