SAFETY ASSESSMENT OF THE ARIES COMPACT STELLARATOR DESIGN

B. J. MERRILL,*a L. A. EL-GUEBALY,b C. MARTIN,b R. L. MOORE,a A. R. RAFFRAY,c D. A. PETTI,a and ARIES-CS TEAM

*aIdaho National Laboratory, Idaho Falls, Idaho 83415-3860
bUniversity of Wisconsin, Fusion Technology Institute, Madison, Wisconsin 53706
cUniversity of California, San Diego, Mechanical and Aerospace Engineering Department and Center for Energy Research, La Jolla, California 92093

Received May 3, 2007
Accepted for Publication September 10, 2007

ARIES-CS is a 1000 MW(electric) compact stellarator conceptual fusion power plant design. This power plant design contains many innovative features to improve the physics, engineering, and safety performance of the stellarator concept. ARIES-CS utilizes a dual-cooled lead lithium blanket that employs low-activation ferritic steel as a structural material, with the first wall cooled by helium and the breeding zone self-cooled by flowing lead lithium. In this paper we examine the safety and environmental performance of ARIES-CS by reporting radiological inventories, decay heat, and radioactive waste management options and 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). Our analyses demonstrate that the decay heat can be safely removed from ARIES-CS and the facility can meet the no-evacuation requirement.

KEYWORDS: ARIES-CS, safety assessment, stellarator

Note: Some figures in this paper are in color only in the electronic version.

I. INTRODUCTION

The goal of the ARIES compact stellarator (CS) design study was to develop an attractive power plant using the cost of electricity as the figure of merit while demonstrating that this power plant meets present-day environmental and public safety standards. One design requirement for the reference breeding blanket selected for this stellarator concept is that it be constructed of materials and by manufacturing techniques presently available to industry. The result is a 1000-MW(electric) power plant design that contains many innovative features to improve the physics, engineering, and safety performance of this power plant. ARIES-CS utilizes a dual-cooled lead lithium blanket concept that employs low-activation ferritic steel as a structural material, with the first wall cooled by helium and the breeding zone self-cooled by flowing lead lithium. A detailed design description of this power plant can be found in a companion article in this journal issue.1

In this paper we examine the safety and environmental performance of ARIES-CS by reporting radiological inventories, decay heat, and radioactive waste (radwaste) management options (disposal, recycling, and clearance) and 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). The ultimate concern regarding these accidents is the risks they pose to the public. Under the U.S. Department of Energy (DOE) Fusion Safety Standard,2 the maximum allowable dose during accident conditions at the site boundary must be less than the dose limit that requires a site evacuation plan. The
dose limit is 10 mSv and in the absence of available weather data must be based on conservative weather conditions. This dose is the early dose to the maximum exposed individual at the site boundary, which includes the dose contributions from cloud-shine during plume passage, seven days of ground-shine, and the 50-year adult dose commitment to human body organs due to inhalation of the plume passage and seven days of particle resuspension from the ground. To meet this requirement, the defense-in-depth confinement strategy has been adopted for ARIES-CS. This strategy employs multiple confinement barriers to prevent the release of confined radioactive materials.

The DOE Fusion Safety Standard also states that waste, especially high-level radioactive waste, shall be minimized. In this present work, our goal has been to minimize the production of waste that fails to qualify for class C, or low-level, land burial through the selection of reduced or low-activation structural and breeding materials for ARIES-CS. With tighter environmental controls and the political difficulty of building new repositories worldwide, the disposal option could be replaced with more environmentally attractive scenarios, such as recycling and clearance. For ARIES-CS, an effort is made to try to select materials (and their associated impurity levels) to allow for the potential of recycling of the activated material in either a hands-on or remote (shielded) mode. In addition, slightly irradiated materials containing traces of radionuclides were identified to be cleared for reuse within the nuclear industry or, preferably, released to the commercial market.

In the following sections we present a safety assessment of ARIES-CS based on these design requirements. Section II describes the system configuration and operating parameters of the ARIES-CS design and the confinement strategy adopted for ARIES-CS. Section III examines the safety-related source terms that must be contained by the design. Section IV assesses the response of ARIES-CS to accidents selected to challenge the confinement boundaries for this design. A characterization of the hazard and volume of radwaste produced from ARIES-CS is the subject of Sec. V. Finally, in Sec. VI we draw conclusions based on the results presented in this paper.

II. ARIES-CS DESIGN DESCRIPTION

II.A. Design Configuration and Parameters

Figure 1 shows the layout of the ARIES-CS power core, illustrating the complicated poloidal geometry of this stellarator. Called out in Fig. 1 are the major internal components, such as the blankets, divertor, shields, and coolant manifolds. Also illustrated are the vacuum vessel...
(VV) and field coils. There are two blanket regions in ARIES-CS, which have been designated as the uniform blanket and nonuniform blanket regions. When combined, these regions comprise ~90% of the first-wall (FW) surface area. Because stellarator physics requirements dictate that regions of the plasma remain very near to the field coils, the nonuniform blanket regions were developed to accommodate this requirement. The nonuniform blanket covers ~15% of the area occupied by the blanket. Figure 2 compares the radial builds associated with these two blanket regions.

A dual-coolant blanket configuration that has a self-cooled PbLi tritium breeding zone and a He-cooled reduced-activation ferritic steel (MF82H) structure was selected as the ARIES-CS reference concept based on this blanket’s relatively good combination of performance, design simplicity, safety features, and modest research and development needs. For this blanket concept, the use of helium is compatible with the divertor coolant, provides preheating of blankets, serves as guard heating, and provides independent and redundant afterheat removal during shutdown. In addition, cooling the first-wall region of the blanket (where the heat load is highest) with helium (instead of a liquid metal) avoids the need for an electrically insulating coating in this high-velocity region, eliminating the large magnetohydrodynamic (MHD) pressure drop associated with liquid metal flow at a velocity required to cool the first wall. This dual-coolant concept was originally developed as part of the ARIES-ST study and then at FZK in Germany. A modular concept was adapted for the ARIES-CS geometry with a particular focus on developing a more efficient coolant routing configuration and on optimizing the design performance of the blanket coupled to a Brayton power cycle through a heat exchanger system. The cycle working fluid (helium) is routed through the blanket helium heat exchanger first and then, in an optimized parallel arrangement, through the blanket PbLi heat exchanger and the divertor helium heat exchanger. The corresponding temperature distributions are illustrated in Fig. 3.

Figure 4 shows details of the module layout. The helium coolant is routed to first cool the first wall in a single pass; it is then routed in a combination of series and parallel flow channels to cool all other structural walls. The PbLi flows slowly in the large inner channels in a two-pass poloidal configuration. A free-floating silicon carbide fiber/silicon carbide (SiCf/SiC) composite structure resides inside these cooling channels to provide the electrical insulation needed to minimize the MHD effects on the PbLi pressure drop and to provide the thermal insulation function required to maintain the PbLi/ferritic steel (FS) interface temperature at the compatibility limit (~500°C) while allowing PbLi temperatures inside the insert to exceed 700°C. The major blanket parameters are listed in Table I. To help increase the coolant temperature at the first wall (and thermal efficiency) while maintaining the reduced-activation ferritic steel (RAFS) temperature within its 550°C limit, a layer of oxide dispersion-strengthened (ODS) ferritic steel (~4 mm thick) is diffusion bonded to the 1-mm RAFS module wall.

A key engineering parameter affecting the size of a compact stellarator is the minimum coil-plasma distance. A novel approach has been developed in which a thin
blanket combined with a highly efficient tungsten carbide (WC) shield is used in the critical area to minimize this radial standoff. For example, Fig. 2 illustrates the radial build of the in-vessel component in the critical area whereby the nominal radial build of 1.8 m can be reduced to 1.3 m by optimizing the blanket and shield. Loss of tritium breeding in these regions can be compensated by optimizing the breeding in other regions including behind

Fig. 3. Temperature rise of cycle working fluid (He) along flow paths in the different heat exchangers.

Fig. 4. Isometric and exploded views of the ARIES-CS dual-coolant blanket module.
TABLE I
Summary of ARIES-CS Blanket Parameters

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Typical module dimensions</td>
<td>~4 m² x 0.62 m</td>
</tr>
<tr>
<td>Tritium breeding ratio</td>
<td>1.1</td>
</tr>
<tr>
<td>Fusion thermal power in blanket</td>
<td>2480 MW</td>
</tr>
<tr>
<td>PbLi inlet/outlet temperatures</td>
<td>464/737°C</td>
</tr>
<tr>
<td>PbLi inlet pressure</td>
<td>1 MPa</td>
</tr>
<tr>
<td>Typical inner channel dimensions</td>
<td>0.26 m x 0.24 m</td>
</tr>
<tr>
<td>Average PbLi velocity in inner channel</td>
<td>~0.04 m/s</td>
</tr>
<tr>
<td>Fusion thermal power removed by PbLi</td>
<td>1323 MW</td>
</tr>
<tr>
<td>PbLi total mass flow rate</td>
<td>25.910 kg/s</td>
</tr>
<tr>
<td>PbLi pressure drop</td>
<td>~ 0.1 to 1 kPa</td>
</tr>
<tr>
<td>Maximum PbLi/ferritic steel temperature</td>
<td>472°C</td>
</tr>
<tr>
<td>He inlet/outlet temperatures</td>
<td>385/460°C</td>
</tr>
<tr>
<td>He inlet pressure</td>
<td>10 MPa</td>
</tr>
<tr>
<td>Typical first-wall channel dimensions</td>
<td>2 cm x 3 cm</td>
</tr>
<tr>
<td>He velocity in first-wall channel</td>
<td>46 m/s</td>
</tr>
<tr>
<td>Total blanket + heat exchanger He pressure drop</td>
<td>0.32 MPa</td>
</tr>
<tr>
<td>Total mass flow rate of blanket He</td>
<td>3559 kg/s</td>
</tr>
<tr>
<td>Maximum local ODS/RAFS temperature at first wall</td>
<td>654/550°C</td>
</tr>
</tbody>
</table>

The divertor so that the three-dimensional (3-D) tritium breeding ratio is 1.1. More details on the neutronics analysis in support of the power plant design can be found in Ref. 7.

The proposed divertor configuration consists of a “T-tube” illustrated in Fig. 5 (Ref. 1). The major design goal of the divertor component was to develop a design that is well suited to the compact stellarator with the capability to accommodate a peak heat flux of 10 MW/m². The T-tube is ~15 mm in diameter and ~100 mm long and is made of a W alloy inner cartridge and outer tube on top of which would be attached a W armor layer subject to the plasma heat flux. The design provides some flexibility in accommodating the divertor area since a variable number of such T-tubes can be connected to a common manifold to form the desired divertor target configuration.

II.B. Tritium Extraction System

A key technology issue regarding the success of the ARIES-CS blanket concept is the extraction of tritium from the PbLi to a level that avoids high tritium inventories in the PbLi-helium heat exchanger tubes and unacceptable permeation rates from the PbLi into the secondary helium, as well as from the primary and secondary systems into the confinement building, during normal operation. Because the PbLi temperature entering these heat exchangers is ~730°C, the materials most compatible with PbLi at this temperature are the refractory metal alloys. Alloys of niobium, tantalum, and vanadium are presently under consideration. Unfortunately, because the solubility of tritium is high and because the diffusivity of tritium is rapid for these alloys, the partial pressure of tritium above the PbLi breeder entering the heat exchanger must be kept below ~0.7 Pa in order to maintain ARIES-CS heat exchanger inventories below 200 g-T and permeation rates below the allowed
operational release guidelines for fusion facilities of 1 g·T/yr as tritiated water.8

An extraction method that appears promising for ARIES-CS is the vacuum permeator. This component contains a bank of small-diameter, thin-walled, niobium alloy tubes through which the entire PbLi primary coolant flows at 5 m/s. Outside the tube bundle is a high vacuum region. Because the PbLi flow is turbulent at this velocity, the mass transport of tritium to the PbLi/tube interface is greatly enhanced over that of ordinary diffusion. Once at this surface, the tritium readily diffuses through the niobium tube walls and into the vacuum region, where it is pumped away in the elemental form to the tritium processing plant.

II.C. Confinement Strategy

Radiological confinement is implemented for ARIES-CS to ensure that releases during normal operation are kept as low as reasonably achievable and that releases during accidents are below the no-evacuation release limits discussed in Sec. I. The approach taken for ARIES-CS is the same as that adopted for ARIES-AT (Ref. 9). Double confinement is implemented in ARIES-CS around all the large inventories of tritium and activation products. For in-vessel inventories, the vacuum vessel and its extensions are the primary confinement, and the cryostat and its extensions are the second confinement boundary. In the heat transfer systems, the coolant piping forms the primary boundary and the vaults or rooms that house the coolant systems form the second boundary. Demonstration of compliance with the no-evacuation safety requirements2 requires examination of a broad range of accidents that could challenge these radiological confinement boundaries to determine if any could lead to releases in excess of the no-evacuation limits. These accidents fall into three major categories:

1. events that directly breach a confinement boundary (e.g., loss of vacuum, overpressure failure of the vacuum vessel, in-vessel loss of coolant with bypass of vacuum vessel)
2. events related to loss of decay heat removal (e.g., complete loss of coolant or loss of flow)
3. events associated with the chemical reactivity of materials (e.g., ex-vessel spill of PbLi coolant).

In addition, segmentation of the ARIES-CS cooling system was adopted by allowing two separate cooling systems per each toroidal field period (six cooling systems or sectors in all) to minimize the at-risk radioactive source terms during an accident and to minimize the impetus for mobilizing these source terms during accidents such as pressure or chemical sources that can challenge the ARIES-CS containment boundaries.

III. ARIES-CS SOURCE TERMS

The safety-related sources that are a concern to ARIES-CS can be categorized as either radiological, nuclear (e.g., decay heating), mechanical (e.g., pressure sources), or chemical. The radiological sources include activation products in the breeding and structural material of the blanket and divertor components, tritium produced in the breeding material that diffuses into the structural material of these components and their cooling systems, and activated dust that forms inside the vacuum vessel as a result of plasma-facing component erosion. The following sections examine these source terms in more detail.

III.A. Radiological and Decay Heat Sources

The key design parameters for this assessment include an average neutron wall loading of 2.6 MW/m², a first-wall/blanket lifetime of 3 full power years (FPY) and a permanent component lifetime of 40 FPY. The activation model includes the PbLi blanket, followed by the shield and manifolds, then the vacuum vessel and magnets. Reference 8 provides the compositions of all components. The impurities of the MF82H ferritic steel structure, SiC/SiC composites, borated ferritic steel (3 wt% B) filler, and magnet structure have been included in the analysis.8 The constituents of all components and dimensions meet the ARIES-CS specific design requirements. For instance, 70% enriched PbLi offers an overall tritium breeding ratio of 1.1. All components provide a shielding function. The blanket protects the shield for the entire plant life (40 FPY), both components protect the manifolds and vacuum vessel, and all four components protect the superconducting magnets. The coil supporting tube surrounds the vacuum vessel completely. Superimposed on the tube are the grooves for winding the superconductor coils.

A poloidal cylindrical one-dimensional (1-D) geometry has been used to model the ARIES-CS system with an average minor radius of 1.7 m. Given the nature of the ARIES-CS design study, which is a conceptual design activity requiring rapid analyses in order to keep pace with numerous design changes, a 1-D poloidal calculation that employs an average neutron wall loading (2.6 MW/m²) is adequate for volumetric average neutron heating and activation predictions for ARIES-CS. The ALARA activation code,10 DANTSYS transport code,11 and their FENDL-2 data library have been used throughout the study.7 The operational history has been represented as 0.85 yr irradiation time followed by 0.15 yr downtime, and repeats for 4 or 47 yr, depending on the component lifetime.

aFor data about the alloying elements and impurities of the blanket and shield material, the reader is referred to http://fti.neep.wisc.edu/aries-cs/builds/build.html.
III.A.1. Structure Radiological Sources

The activity is used to evaluate the radiological hazards of the individual components. Figure 6 compares the results of the blanket, shields, vacuum vessel, magnet and divertor evaluated for fully compacted, 100% dense materials, excluding all voids. As can be seen, the activities of these components change slowly with time and the magnet exhibits a very low activity, presenting no radiological hazard to the design.

At shutdown the ODS ferritic steel of the blanket first wall has an activity level of $1.6 \times 10^{8}$ Ci/m$^3$. The top contributors to this level of activity are $^{55}$Fe (46%), $^{56}$Mn (28%), $^{54}$Mn (7.2%), and $^{51}$Cr (7.1%). In contrast, the divertor tungsten armor has an activity level of $5.3 \times 10^{8}$ Ci/m$^3$. Here the top contributors are $^{185}$W (36%), $^{181}$W (19%), $^{183m}$W (18%), $^{187}$W (13%), and $^{185m}$W (8%), with the $^{183m}$W and $^{185m}$W decaying to low levels by 1 day.

III.A.2. Mobilization Mechanisms

There are several mechanisms that could mobilize these source terms during accidents. Air could enter the vacuum vessel, an air ingress accident, while these components are at elevated temperatures, oxidize the base metals, and release the radioactive material as aerosols by oxide sublimation. Plasma erosion of the surface of the in-vessel components will occur during operation, forming radioactive metal dust particles on the floor of the vacuum vessel. Melting and evaporation of the base metal could occur as a result of continued plasma operation following a loss-of-cooling accident (LOCA), extreme surface heat loads from a rapid plasma thermal quench, or failure to remove the decay heat during a LOCA.

With respect to air ingress accidents, there is no credible accident for a stellarator during which the vacuum vessel boundary fails and internal components simultaneously lose cooling. For example, an air ingress could occur as a result of a vacuum vessel diagnostic or plasma heating window breach. However, the inrush of air would extinguish the plasma and since stellarator plasmas do not possess large induced currents, there is no viable mechanism for simultaneously producing large forces that would fail the blanket or divertor cooling systems. In addition, because the plasma’s magnetic field is primarily due to external field coils, the particle confinement of the plasma will not be lost, leaving the predominate mode of dissipating the plasma stored thermal energy by impurity radiation. This radiation will give a more or less uniform first-wall heating during the time of the quench. Because this process will take tens of milliseconds, the average first-wall heating is estimated to be less than 40 MW/m$^2$ over a 10-ms quench, which heating will not result in any significant first-wall damage. With that said, even if an air ingress accident were to occur, the first-wall oxidation rate at the peak first-wall operating temperature of 750°C is only $3.6 \times 10^{-7}$ (kg/m$^2$·s), which for a single sector, equal to half a field period, results in a surface mass loss of 0.14 kg/h based on the oxidation data of Ref. 12. Hypothetically speaking, if all of this mass were to be released to the environment, the dose rate would be 4.6 mSv/h (Ref. 13) at a 1-km site boundary if the release were to occur during conservative weather conditions at ground level. This would allow for more than 2 h to manually isolate the confinement building, which is more than enough time. After building isolation, any building leakage can easily be controlled by the continued cooling of in-vessel components by the intact cooling systems, and by building air chillers to maintain a negative pressure gradient relative to atmospheric conditions. However, as was demonstrated by Ref. 14, due to aerosol settling only one part in 3000 is likely to escape the confinement building anyway during such accidents.

Events that could result in melting of the structure from loss-of-cooling events are reported in Sec. IV.A, with the result being that peak temperatures are not likely to exceed 740°C. This temperature is not only far below melting but is also below the temperature that would not allow reuse of the blanket. Again, because there is no plasma disruption during accidents, extreme surface heating that leads to melting is also unlikely.

This leaves surface erosion of the plasma-facing components as the most probable mobilization mechanism for these radioactive sources, with the divertor being the component most affected by this mobilization mechanism. For this study, it is assumed that over the three-year lifetime of the divertor components, 10% of the divertor surface area, that portion of the divertor with the highest particle heating, will experience erosion to a depth of 0.5 mm, which is the design life erosion depth. Assuming that every third year the divertor is replaced and the dust in that field period is cleaned up, then the maximum quantity of tungsten dust that can reside in the ARIES-CS will be $\sim$65 kg. This dust is readily mobilized by ingress...
III.A.3. PbLi Radiological Sources

The primary radiological concern for PbLi is from the radionuclides $^{210}$Po and $^{203}$Hg (Ref. 10). Both of these isotopes have well-defined primary production pathways beginning with the Pb isotopes $^{206}$Pb and $^{208}$Pb:

$$^{208}\text{Pb} \xrightarrow{n,\gamma} ^{209}\text{Pb} \xrightarrow{\beta^-} ^{209}\text{Bi} \xrightarrow{n,\gamma} ^{210}\text{Bi} \xrightarrow{\beta^-} ^{210}\text{Po}$$

(1)

and

$$^{206}\text{Pb} \xrightarrow{n,\alpha} ^{203}\text{Hg}$$

(2)

In addition to $^{208}$Pb and $^{206}$Pb, the PbLi coolant is assumed to contain a 43 wt ppm impurity of $^{209}$Bi that through a two-step neutron absorption and subsequent decay process leads to $^{210}$Po. A number of longer Pb-based pathways also contribute to the $^{210}$Pb concentration over the lifetime of the coolant.15

Based on the results presented in Ref. 16, the inventories of $^{210}$Po and $^{203}$Hg in the PbLi coolant can be limited to 1 wt ppb if the bismuth in the coolant is kept to 1 wt ppm. The ARIES-AT study10 suggests that the Bi removal could be even better, resulting in $^{210}$Po and $^{203}$Hg concentrations of below 0.1 wt ppb, which will be used in this safety assessment.

III.B. Nuclear and Decay Heating

As a source term, the decay heat is a safety concern during LOCAs. It is important that this heat be safely rejected without compromising the integrity of the ARIES-CS radioactive confinement systems. Of particular concern is the high decay heat level associated with the WC in the WC shields and the W of the divertor.

Decay heats were calculated for 20 discrete heating zones in the various components, at 18 time points. Examples of the decay heat in various full-blanket components are illustrated in Fig. 7. The blanket first wall has the highest heating. The average first-wall decay heat is 730 kW/m$^3$ immediately after shutdown of the plasma but drops an order of magnitude to 70 kW/m$^3$ within 1 day. The total decay heat burdens are $\sim 18$ and $\sim 2.4$ MW, respectively, for these same times. Heating falls off quickly through the radial thickness of the blanket, with the blanket back wall having a maximum decay heat of only 4 kW/m$^3$. For the nonuniform blanket, first-wall heats are similar to those of the full blanket, but other decay heats are generally higher largely because of the reduced breeding area of the nonuniform blanket and the presence of the WC filler in the shield. During plasma operation, the predicted fusion heating values range from 22 MW/m$^3$ at the first wall to 0.4 MW/m$^3$ at the back wall of the blanket.

III.C. Tritium Inventory

III.C.1. Method of Analysis

In order to estimate the tritium inventory and permeation rates for the ARIES-CS cooling systems, a TMAP code17 model of an entire sectors’ cooling system (one-sixth of the ARIES-CS) was developed. The TMAP code was developed at the Idaho National Laboratory (INL) to analyze tritium retention and loss in fusion reactor structures, systems, and confinement rooms during normal operation and accident conditions. TMAP incorporates 1-D thermal and mass-diffusion equations for predicting tritium permeation through walls of the fusion device, including trapping of the tritium on defects within the material of the walls. Movement of tritium across material surfaces can be modeled as being governed either by molecular dissociation/recombination or by solution laws such as Sievert’s or Henry’s laws. The flow of tritium between rooms of a fusion facility and the conversion of molecular form by chemical reactions within a room are also modeled by TMAP. In TMAP, room atmospheres are modeled as enclosure volumes and walls as diffusion structures.

The ARIES-CS TMAP model includes the structural material associated with the blanket and the major components of the cooling loops (e.g., pipes, heat exchangers, Brayton cycle, and tritium extraction system) and the fluid volumes containing PbLi and helium in this cooling system. A schematic of this model appears in Fig. 8. The ARIES-CS blanket modules for a single
sector are lumped in this TMAP model as five enclosure volumes and seven diffusion structures. Three of five enclosures are PbLi volumes that represent the combined volume of the breeder zones of the blanket. These volumes simulate the core blanket flow, the PbLi flow within the SiCf/SiC insert, and the flow in the gaps (i.e., PbLi-filled gaps) between the inserts and blanket structural walls. Of the ~7470 kg/s of PbLi that flows into this blanket sector, 95.6% of this flow is into the core volume, with the remainder flowing in the gap volumes. This flow split was imposed in this model to match two-dimensional MHD fluid flow results reported in Ref. 6. Whereas the PbLi flow has been split to match MHD calculations, the tritium bred in the PbLi is partitioned according to volume. The remaining two enclosure volumes simulate the first-wall helium coolant, both the radial and toroidal flow channels, and the helium cooling channels within the rib/divider, top plate, bottom plate, and back wall blanket structures. The helium flow rate in these volumes is 430 kg/s.

The seven diffusion structures represent the outer structures of the blanket, walls that have a vacuum on one side and helium coolant on the other (the first wall, side walls, top plate, bottom plate, and back wall), and the interior walls that have helium on one side and PbLi-filled gaps on the other (second wall, side walls, top plate, bottom plate, back wall, and ribs/divider). The total blanket module exterior wall area for one sector of ARIES-CS is estimated to be 750 m².

The ex-vessel components of the PbLi cooling loop are modeled by eight enclosure volumes and eight diffusion structures, representing the PbLi flow in the inlet and outlet pipes, permeator, and PbLi-helium Brayton cycle heat exchanger. The permeator consists of ~2060 1-cm inside diameter, 0.5-mm-thick, 5-m-long niobium tubes. The five enclosure volumes and diffusion structures used to model this component have a total surface area of 340 m². The PbLi flows in parallel through these tubes, resulting in a flow velocity of 5 m/s. The temperature of this component was set at 730°C.

The ex-vessel blanket helium cooling loop is modeled as three enclosure volumes and three diffusion structures, simulating the inlet pipe, outlet pipe, and helium-helium Brayton cycle heat exchanger. The Brayton cycle system was modeled as four enclosure volumes and two diffusion structures. The Brayton cycle is assumed to be a closed Brayton cycle similar to that examined in Ref. 18 for a fission reactor concept. For this type of Brayton cycle system, the turbines, intercoolers, and generators are enclosed in a thick pressure vessel. The walls of this vessel are made of thick steel (~8.5 cm) to contain the system pressure (15 MPa), this wall also represents an excellent barrier to tritium permeation. Based on the modeling performed in Ref. 18, the surface area of a system large enough to handle the 460 MW of a single sector of ARIES-CS would have an internal volume of 445 m³ (including the secondary side volumes of the heat exchangers), a vessel wall area of 2050 m², an intercooler surface area of 5050 m², and an internal flow rate of ~260 kg/s.

Fig. 8. Schematic of TMAP model of ARIES-CS.
III.C.2. Diffusion Boundary Conditions and Material Properties

For this analysis, the adopted structural material for the blanket, shields, manifolds, and PbLi loop piping is a reduced activation martensitic steel, that of the permeator and PbLi heat exchanger tubes is a niobium alloy, and that of all ex-vessel helium cooling system components is austenitic Type 316L stainless steel. An aluminum alloy is selected for the Brayton cycle helium-water intercooler tubes. Aluminum was chosen to reduce the permeation of tritium into the ARIES-CS heat rejection water cooling system. All metallic surfaces in contact with air, vacuum, or helium are modeled by the molecular disassociation/recombination boundary condition. This approach was taken to allow for the influence of possible thin oxide layers on tritium permeation, in particular in the permeator itself. The uptake of tritium by the aluminum tube surface in contact with helium was modeled by the Sievert’s law boundary condition, whereas the release of tritium into the water on the opposite side of the tube was modeled by a zero-tritium surface concentration boundary condition to add additional conservatism to this analysis.

Because the required constants for disassociation and recombination are not available for F82H martensitic steel, the measured constants for a similar martensitic steel, called MANET II, were adopted for this analysis. In particular, correlations for tritium disassociation and recombination, solubility, and diffusivity for MANET II steel with a natural oxide layer (∼5 nm) were obtained from Ref. 19. Correlations for the same material properties of Type 316 stainless steel, with a 50-nm oxide layer, were obtained from Ref. 20. Correlations for tritium solubility and diffusivity in aluminum were obtained from Ref. 21.

Release of the tritium bred from the blanket PbLi requires the modeling of the transport of tritium within the bulk of the liquid metal to the interface between the liquid metal and the steel structure and, once at this surface, release from the liquid metal and reabsorption by the steel surface. Transport in the liquid metal is by bulk diffusion, enhanced by turbulence within the liquid metal. The flux of tritium atoms, \( \Gamma_T \) (T/m² s), arriving at the liquid metal surface from the bulk can be estimated as follows:

\[
\Gamma_T = K_T (C_{T,B} - C_{T,S}), \tag{3}
\]

where

\[
K_T = \text{mass transport coefficient (m/s)}
\]

\[
C_{T,B} = \text{liquid metal bulk tritium concentration (T/m³)}
\]

\[
C_{T,S} = \text{liquid metal surface tritium concentration (T/m³)}.
\]

The correlation adopted to predict the tritium mass transport coefficient in PbLi is the following correlation from Ref. 22:

\[
K_T = \frac{D_T}{L} (0.0096 \text{Re}^{0.913} \text{Sc}^{0.346}), \tag{4}
\]

where

\[
D_T = \text{tritium diffusion coefficient in PbLi (m²/s)}
\]

\[
L = \text{characteristic dimension (m)}
\]

\[
\text{Re} = \text{Reynolds number} = \rho L \nu / \mu
\]

\[
\text{Sc} = \text{Schmidt number} = \mu / \rho D_T
\]

\[
\rho = \text{liquid metal density (kg/m³)}
\]

\[
\nu = \text{liquid metal bulk velocity (m/s)}
\]

\[
\mu = \text{liquid metal viscosity (kg/m·s)}.
\]

The diffusivity of tritium in PbLi was obtained for this study from the correlation developed by Ref. 23. Given the conditions expected in the permeator, the mass transport coefficient for tritium in the PbLi based on the above correlation is ∼1.2 mm/s.

Once tritium reaches the surface of the liquid metal, the tritium is modeled as being released according to Sievert’s law. However, once out of the liquid metal in gas form \( (T_2) \), this tritium should be immediately reabsorbed by the permeator niobium alloy. This reabsorption process is also modeled by Sievert’s law. If there is no holdup at the niobium/liquid metal interface, the result is a discontinuity in tritium concentration that is proportional to the ratio of the solubility coefficients for tritium in these materials, as follows:

\[
\frac{C_{T,S/Nb}}{C_{T,S/Pb-17Li}} = \frac{K_{S/Nb}}{K_{S/Pb-17Li}}. \tag{5}
\]

Tritium solubility coefficients for PbLi are obtained from the correlation developed in Ref. 24.

III.C.3. Permeation Results

Table II contains equilibrium results from this TMAP model for one sector of ARIES-CS. Two cases are presented in this table; the first assumes no implantation of tritium as a result of charge exchange in the scrape-off region of the plasma into the first wall and the second assumes that implantation does occur. Both assume the continuous production of tritium in the PbLi blanket of 152 g/day-1000 MW, which for one sector of ARIES-CS is ∼66 g/day. When implantation is considered, the implantation flux is set at 1.5 × 10²⁰ T/m²-s based on the value adopted in Ref. 25. The purpose for considering these two cases is not to demonstrate upper and lower bounds for the anticipated tritium inventory in ARIES-CS but to illustrate which tritium source contributes more to predicted tritium inventory for ARIES-CS.

There are several interesting results presented in Table II. The most striking result is that the inventory increases from 97 to 267 g when the implantation of...
FUSION SCIENCE AND TECHNOLOGY VOL. 54 OCT. 2008

Merrill et al. ARIES-CS SAFETY ASSESSMENT

Table II
Tritium Inventory and Permeation Results for One Sector of ARIES-CS

<table>
<thead>
<tr>
<th>Structure</th>
<th>No Implantation</th>
<th>First-Wall Implantation</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Inventory (g·T)</td>
<td>Permeation into Building (g/yr)</td>
</tr>
<tr>
<td>Blanket</td>
<td>1.23E-01</td>
<td>1.68E-01</td>
</tr>
<tr>
<td>High temperature shield</td>
<td>2.63E-03</td>
<td></td>
</tr>
<tr>
<td>Manifold</td>
<td>9.63E+01</td>
<td></td>
</tr>
<tr>
<td>PbLi outlet pipe</td>
<td>7.41E-03</td>
<td>1.25E-02</td>
</tr>
<tr>
<td>PbLi HTX tubes</td>
<td>5.12E-02</td>
<td>1.06E+00</td>
</tr>
<tr>
<td>PbLi inlet pipe</td>
<td>3.93E-01</td>
<td></td>
</tr>
<tr>
<td>Helium outlet pipe</td>
<td>3.60E-02</td>
<td>7.55E-01</td>
</tr>
<tr>
<td>Helium HTX tubes</td>
<td>1.59E-03</td>
<td>5.36E-04</td>
</tr>
<tr>
<td>Helium inlet pipe</td>
<td>1.01E-01</td>
<td></td>
</tr>
<tr>
<td>Brayton cycle wall</td>
<td>3.62E-01</td>
<td></td>
</tr>
<tr>
<td>Permeator</td>
<td>3.03E-02</td>
<td></td>
</tr>
<tr>
<td>Total</td>
<td>9.73E+01</td>
<td>1.71E+01</td>
</tr>
<tr>
<td>Release after 99% efficient cleanup</td>
<td>1.71E-01</td>
<td></td>
</tr>
</tbody>
</table>

*Read as $1.23 \times 10^{-01}$.

tritium atoms into the first wall is included in the calculation. This suggests that more tritium permeates through the blanket first wall from the plasma than is generated in the PbLi of the blanket. These inventories could be reduced by increasing the length of permeator tubes. However, since most of the inventory resides in the niobium tubes of the PbLi-helium heat exchanger (HTX), and because niobium behaves like a getter for tritium (e.g., tritium is very soluble in niobium and increasingly so as the niobium temperature drops), this tritium is relatively immobile during accidents unless the HTX tube temperature increases dramatically. As is demonstrated in the accident analyses of Sec. IV, the accidents analyzed do not indicated that this will be the case. It is also interesting to note that the total in-vessel ferritic steel structure inventory for all six sectors is only 1.5 to 53 g, depending on whether or not implantation is included. This demonstrates the very effective tritium removal from the PbLi by the permeator, keeping the inventory of tritium in the PbLi to less than 1 g per sector.

There is also a noted increase in permeation into the building from the ex-vessel components between the no implantation and implantation cases, from 17.1 to 43.5 g/yr. Since these numbers are for only one sector, the total would be six times higher. If the atmosphere of the vault that houses these components is detrinitiated before venting to the environment at an efficiency of 99%, then the annual release would be 1.0 to 2.6 g/yr, the latter of which is slightly above the allowable of 1 g/yr. One option for decreasing this permeation would be to use permeation barriers or aluminum shrouds around the ex-vessel components, which would virtually eliminate tritium permeation into the building.

A final safety concern is tritium permeation into the heat rejection cooling water of the Brayton cycle. Based on this model, the tritium permeation rate into this cooling system is $\sim 0.2$ mCi/day. If the volume of this system is taken as 100 m$^3$, then the tritium concentration in this water after a 30-yr lifetime is only $20 \mu$Ci/$\ell$. Although this is a very low value, it still exceeds by three orders of magnitude the safe drinking water limit of 20000 pCi/$\ell$ established by the Environmental Protection Agency. This water could either be reused at other fusion facilities until tritium concentrations reach levels that warrant tritium extraction or mixed with concrete and disposed of by shallow land burial.

III.D. Pressure Sources

There are two pressure sources in ARIES-CS that represent a safety concern: the helium of the first-wall/blanket cooling system and the helium of the Brayton cycle system. The helium in the first-wall/blanket cooling system has pressure of 10 MPa and an average temperature of 405°C. The volume of helium associated with this cooling system is 80 m$^3$, per sector. The Brayton cycle helium has an assumed pressure of 15 MPa and an average temperature 515°C. The volume of helium in the Brayton cycle is 295 m$^3$, per sector.

The vacuum vessel is designed to contain a $\pm 1.0 \times 10^5$ Pa pressure differential. To avoid overpressurization of this confinement barrier during design-basis accidents (DBAs, frequency $\approx 10^{-6}$/yr), a rupture disk between the vacuum vessel and the cryostat is set to open at a pressure of $2.0 \times 10^5$ Pa. This makes the cryostat the pressure confinement system for DBAs. The cryostat is
also part of the second confinement barrier and has a design pressure of $3.0 \times 10^5$ Pa. A rupture disk that connects the cryostat to the heat transport system (HTS) vault, which is also part of the second confinement barrier, is added to the ARIES-CS design to guarantee that overpressurization of the cryostat will not occur during beyond design-basis accidents (BDBAs, frequency < $10^{-6}$/yr). The HTS vault is the final line of defense for pressurization events in ARIES-CS. This vault is designed to withstand a $3 \times 10^5$ Pa internal pressure.

### III.E. Chemical Reactions

Chemical reactions of concern to ARIES-CS are the PbLi-water and PbLi-air reactions. PbLi-water reaction requires a postulated event that simultaneously spills PbLi breeder from the blanket and water from the vacuum vessel cooling systems in the ARIES-CS design. However, because the blanket and vacuum vessel cooling systems are separate and since the vacuum vessel is a lifetime component, there are no credible facility accidents (DBAs) that would produce simultaneous spills of PbLi and water in ARIES-CS.

Regarding PbLi-air reactions, as discussed in Ref. 10, the DBA of concern is the ex-vessel LOCA from the blanket PbLi cooling system. For this reaction, the Li in the PbLi is buffered to a large degree by the large heat capacity of the Pb. As a result, reaction with air is not a serious issue in terms of chemical energy production or combustible gas generation. At temperatures up to $900^\circ$C, no violent reaction was observed in experiments due to an oxide layer that develops on the surface of the PbLi pool. This leaves as the key concern for this event the release of $^{210}\text{Po}$ and $^{203}\text{Hg}$ from the PbLi during a spill. As a result, an ex-vessel spill of PbLi coolant in the room housing the coolant piping was examined to determine the potential for $^{210}\text{Po}$ and $^{203}\text{Hg}$ release. The results of this analysis are presented in the following section of this paper.

### IV. ARIES-CS ACCIDENT ANALYSIS

In this section we examine the response of ARIES-CS to credible accidents that challenge the safety function of radioactive confinement. The accidents analyzed were selected because they challenge the integrity of ARIES-CS confinement barriers. We examine LOCAs and loss-of-flow accidents (LOFAs) in the blanket and divertor to ensure that decay heating can be safely removed from these components of ARIES-CS. We examine in-vessel helium LOCAs to study the impact that this pressurization source has on the vacuum vessel and cryostat confinement barriers. We examine the consequences of an ex-vessel spill of PbLi to estimate the quantity of $^{210}\text{Po}$ and $^{203}\text{Hg}$ that is mobilized during this accident. Finally, we examine a noncredible or BDBA designated as the double tube breach event to determine ultimate safety margins of the ARIES-CS confinement strategy.

#### IV.A. Blanket and Divertor LOCA and LOFA

The presence of three coolants makes possible a variety of LOCA and LOFA events. Because the water coolant system of the vacuum vessel is a safety system, only a loss of forced flow in the vacuum vessel is considered, with the consequence being that this cooling system enters the designed natural convection decay heat removal mode. The vacuum vessel cooling water is assumed to flow by natural convection out of the nuclear island to an air/water heat exchanger on the roof of the confinement building. For the helium in the blanket and shield, only LOCA conditions are considered. In a helium LOFA event, the presence of nonflowing gas would have minimal thermal effect and results would be very similar to a LOCA. For the PbLi in the blanket, the analyses described here focus on the LOFA condition, in which PbLi was assumed to be nonflowing with afterheat. Simultaneous LOCAs in both systems were not considered because the breach of two separate and independent cooling systems is not a credible event (e.g., a DBA). It is also assumed that the loss of coolant or flow occurs in all 200 modules. The modular nature of the ARIES-CS design would make this scenario less likely, and some of the heat from a failed module could be removed through the adjacent cooled modules, but the systemwide LOCA/LOFA events represent a potential worst-case scenario.

Additionally, about 5% of the plasma-facing surface is covered by divertor plates. The response of the divertor regions to a LOCA/LOFA event was also analyzed. The divertor plates, designed to withstand local heat fluxes as high as 10 MW/m$^2$, use a concept based on a modular arrangement of helium-cooled T-tubes made of tungsten alloy, as illustrated in Fig. 5 (Ref. 1). Behind the T-tubes are ferritic steel manifolds that distribute the coolant. The divertor plates and manifolds are separated from the blanket by a vacuum gap. The blanket behind the divertor has a reduced thickness (the PbLi channels in the blanket are reduced to 15 cm), whereas the shield, manifold, and vacuum vessel are the same as in the full-blanket design. A schematic of the radial build is also illustrated in Fig. 5. Decay heats and finite element thermal models were developed for the divertor radial build in a manner similar to that for the full-blanket regions. The decay heat for the divertor plate was approximately 14% higher than for the first wall in the full blanket.

#### IV.A.1. Method of Analysis

There are a variety of assumptions required for these analyses that are important to detail. The finite element modeling and analysis was performed using the ANSYS code. The models used axisymmetric elements to model the details of components. The finite element models
used for the full- and nonuniform blanket analyses are illustrated in Fig. 9. Because of the repeating nature of the build geometry, the top and bottom edges of the model may be treated with symmetry boundary conditions, and the first-wall surface is treated as being adiabatic because the facing surfaces in the plasma chamber will be at a similar temperature. Although these models do not capture the toroidal nature and varying cross section of the complex compact stellarator geometry, they do provide a good estimate of overall system performance at reasonable modeling and computational costs. The modeling also assumes that the outside of the vacuum vessel is adiabatic and that heat is removed from the system by only natural convection to the water inside the vacuum vessel. This convection occurs with a coefficient of 500 W/m²·C, and the water remains at 140°C. Also, in the event of an accident, the plasma is assumed to be terminated by a plasma shutdown system. The reaction time for this shutdown system was that adopted by the International Thermonuclear Experimental Reactor (ITER), which is 3 s. The 3-s shutdown time of Ref. 28 is ITER’s estimate of how fast an accident condition can be accurately detected and assessed and plasma terminated by impurity injection. We have adopted this shutdown time for ARIES-CS because we believe it to be a conservative value for future reactors, given the experience of operating the ITER device and advances in diagnostics that will occur in the future. In addition to conduction, radiation heat transfer is modeled in the empty coolant passages and the gap between the manifold and vacuum vessel (VV). An emissivity of 0.3 is used for all radiating surfaces.

IVA.2. LOCA and LOFA Results

The finite element models, heat loads, and conditions previously described were used to calculate the transient temperature response from the accident onset until the analysis termination time of 30 days. Results for the full-blanket design are plotted in Fig. 10 for a LOCA in the He and a LOFA in the PbLi and water. The maximum temperature, 725°C, occurs in the first wall immediately after shutdown of the plasma (3 s). This indicates that the critical factor for LOCA/LOFA first-wall temperature is the time between loss of coolant and the plasma quench as the first wall rises from 571 to 725°C in the 3 s assumed in this analysis. After plasma shutdown, first-wall temperatures fall briefly then begin to rise again. A second peak in first-wall temperature occurs at a time of 4500 s (75 min) when the temperature reaches 688°C. The other components remain significantly cooler; the maximum temperatures of the back wall (545°C) and shield (524°C) are significantly below the 740°C reuse temperature of the ferritic steel. Temperatures in the vacuum vessel remain below 180°C and approach the 140°C water temperature with increasing time. The maximum temperatures in all the components occur within the first 12 h after shutdown, and all temperatures fall continuously after 12 h, indicating that longer-term LOCA/LOFA conditions would not be a concern as long as natural convection in the water VV loop is maintained.

Because the maximum front wall temperature occurs immediately after the shutdown of the plasma, it is highly dependent on the heat inputs during this time. Heat is applied to the first wall as volumetric nuclear heating and
surface heating from the plasma, with the surface heating representing 90% of the total input. A value of 0.76 MW/m² was applied in the analyses, and the response is very sensitive to this value. When the surface flux is reduced to 0.5 MW/m², the first-wall temperature after 3 s of fusion power is 676°C and the overall maximum temperature is 687°C. So in the future more attention will need to be paid to the assumption regarding the time required to quench the plasma fusion heating because this assumption is crucial to predicting LOCA/LOFA temperatures.

For the nonuniform blanket regions, the maximum temperature occurs again in the first wall immediately after shutdown of the plasma at 3 s, as illustrated in Fig. 11. The maximum temperature of 730°C in this case is higher than that of the uniform blanket. This is most likely due to the higher average nuclear heating of the reduced-thickness PbLi layer. After this peak, first-wall temperatures fall then begin to rise with the rest of the component temperatures. The second first-wall peak reaches 701°C, and temperatures reach 678°C in the ferritic steel shield and 661°C in the WC shield. These maxima occur approximately one day after loss of coolant, and then all component temperatures fall continuously. The differences in the behaviors of the two blanket regions can be attributed to the significantly higher decay heats of the WC material in the nonuniform blanket.

The transient results for the divertor LOCA are presented in Fig. 12. The temperatures in the W T-tubes rise after LOCA for the 3 s prior to plasma shutdown to a temperature of 1074°C and then decline. The divertor manifolds reach a maximum of 804°C 132 min after loss of coolant while the blanket reaches a maximum temperature of 621°C. The maximum tungsten temperature of 1074°C should not present a problem, but the 804°C predicted manifold temperature exceeds the reuse temperature for ferritic steel, and hence a LOCA in the divertor lasting more than 1600 s may require replacement of this component. Significantly, blanket temperatures remain well below the 740°C ferritic steel reuse temperature.
IV.B. In-Vessel Helium Pressurization Events

The in-vessel helium pressurization events were analyzed for ARIES-CS using the fusion version of the MELCOR 1.8.5 code. MELCOR (Ref. 29) is currently being developed at the Sandia National Laboratories for the U.S. Nuclear Regulatory Commission (NRC) to analyze severe accidents in fission reactors. MELCOR tracks the flow of water during normal reactor operations as well as the flow of two-phase water during severe accidents. It also tracks any radioactive aerosols that may be present in either fluid phase. Structure temperatures are determined by the code using 1-D heat conduction equations with appropriate boundary conditions. Heat transfer to both phases is considered during forced, natural, boiling, and condensation heat transfer modes. Modifications to MELCOR were made by the INL for fusion-specific analyses. These modifications are discussed in Ref. 30. One of the major modifications pertinent to the analysis of ARIES-CS is the capability of the fusion version of MELCOR 1.8.5 to use different primary coolants such as PbLi in place of water.

To analyze helium pressurization events as well as other safety-related events, an ARIES-CS MELCOR model was developed. The model represents one-half a field period segment of ARIES-CS (one-sixth of the reactor). A radial view\(^\text{b}\) of the full-blanket and shield portion of the model is shown in the schematic presented in Fig. 13. The helium and PbLi coolant loops, toroidal headers, and He and PbLi heat exchangers are also shown in the schematic.

The helium coolant loop operating at a pressure of 10 MPa is used to help cool the first wall and the blanket and shield regions of both the uniform and nonuniform blanket regions. The helium flows from the helium heat exchanger through the helium toroidal header piping and then enters the manifold region, where it is split into two streams. One stream flows to the nonuniform blanket region (not shown in the figure) with the other stream flowing to the uniform blanket region. The helium coolant that flows to the uniform blanket is split again into two streams, one flowing to the first wall of the uniform blanket and the other flowing through the shield region. After flowing through the first wall and blanket and shield regions, the He coolant streams recombine and flow back to the manifold region, where the coolant from the uniform blanket is recombined with the coolant from the nonuniform blanket region. The helium flow paths through the nonuniform blanket mirror those shown for the uniform blanket. After exiting the manifold the helium flows back through the toroidal header (concentric pipes) to the heat exchanger, where the excess heat is removed. During steady-state operation the helium coolant enters the first-wall region at \(379^\circ\text{C}(652 \text{ K})\) and exits the blanket region at \(455^\circ\text{C}(728 \text{ K})\). The helium loop mass flow used in the MELCOR model was determined to be 492 kg/s in order to maintain the loop \(\Delta T\) at 76°C during steady-state operation.

\(^{b}\)The radial dimensions and structural composition of the radial build were obtained from the University of California, San Diego, ARIES-CS Web site at http://aries.ucsd.edu/ARIES/MEETINGS (see Fig. 2).
The PbLi coolant loop, which operates at a pressure of 4 MPa, is modeled in a manner similar to that for the He loop. During steady-state operation the PbLi coolant enters the blanket region at 462°C (735 K) and exits the blanket region at 727°C (1000 K). The PbLi loop mass flow rate used in the MELCOR model was determined to be 3088 kg/s in order to maintain the PbLi loop ΔT at 265°C during steady-state operation.

To ensure that the plasma chamber will not be overpressurized, two analyses were conducted, one assuming a small in-vessel LOCA associated with failure of a single first-wall helium coolant tube and the other assuming a large in-vessel LOCA associated with the failure of a cooling manifold. The breach of these components results in high-pressure helium being discharged into the plasma chamber, thus rapidly pressurizing the plasma chamber. The break sizes associated with the analyses were 0.0012 and 0.04 m², respectively.

The plasma chamber (volume = 566 m³) is protected from overpressurization by a rupture disk that allows venting of the plasma chamber to the cryostat (volume = 7000 m³) once the plasma chamber pressure exceeds 0.2 MPa.

IV.B.1. Pressurization Results

The results from the analyses are presented in Fig. 14. The accident is assumed to occur at time corresponding to zero in the figure. Prior to this time the MELCOR model was run until steady-state conditions were achieved. By time zero it is assumed that the helium flow into the plasma chamber causes termination of the plasma. The coolant pumps are shut down and the vacuum vessel coolant (water) enters natural convection mode. As shown in the figure, the plasma chamber pressure reaches 0.2 MPa for the small-area break (0.0012 m²) and 0.208 MPa for the large-area break (0.04 m²). Both peak pressures occur within 10 s after the breach in the tubes. Once the rupture disk breaks, the pressure in the cryostat and plasma chamber equalize at 0.15 MPa. The final pressure of 0.15 MPa occurs within 10 s for the large break because of the large mass flow from the break; however, it requires ~300 s for the pressure in the two volumes to equalize for the small break. This is due to the much smaller mass flow rate from the small break. The equilibrium pressure is slightly higher for the small break case due to the longer resident time of the helium in the first wall, i.e., higher helium temperatures exiting from the break.

IV.C. Ex-Vessel PbLi Spill Event

As mentioned in Sec. III.B, the key safety concern for this event is the release of 210Po and 203Hg from the PbLi as a consequence of the ex-vessel spill. The magnitude of this release strongly depends on the temperature of the pool formed during this event. To model this pool, two additional control volumes were added to the MELCOR model described in Sec. IV.B. These volumes simulate the cooling system vault and a single pipe chase that leads into the vault. The physical characteristics of these control volumes are those of the ITER-FEAT design, with their volumes being 38 345 and 900 m³, respectively. The reason for adopting these vault characteristics is because the ITER-FEAT confinement building is the most developed fusion building presently available, including specifications of isolation times, building leak rates, and atmospheric chiller capacities, just to name a few of the safety characteristics defined for the ITER-FEAT building. This assumption will result in conservative building pressure estimates because the ARIES-CS vault will be larger than that for ITER-FEAT in order to accommodate the larger power extraction system anticipated for the ARIES-CS reactor. Concrete ceilings, floors, and walls of these enclosures were also added to this model to simulate the heat transfer between the PbLi pool and the ARIES-CS confinement building. These structures were modeled as being 1 m thick. The boundary condition at the back of the pipe chase and HTS vault walls is convection to ambient temperature. Heat transfer from the front of the pipe chase walls to the pool was modeled by the MELCOR heat transfer package. The spill was assumed to occur at the bottom of the pipe chase to generate the deepest PbLi pool possible for this event.

Figure 15 contains the predicted break mass flow rate of PbLi into the pipe chase following the pipe break. The pipe break flow initially reaches a peak of about 45 500 kg/s and then decays to near zero by 100 s. A second peak of 63 200 kg/s occurs once the pump stops and the flow in the loop reverses direction. As result of these flows, an ~0.9-m-deep pool (with a surface area of 110 m² and a total volume of 98 m³) forms on the floor of the pipe chase for this sector.

![Fig. 14. Pressurization of plasma chamber.](image-url)
Figure 16 shows how the temperature of this pool evolves during the course of this accident. The peak temperature is 640°C, down from the outlet temperature of 730°C due to contact cooling of the PbLi with the pipe chase floor and walls. Because PbLi shows no energetic chemical reaction with air even at 900°C (Ref. 26), chemical reactions will not be a concern for this pool. However, there is decay heat in the PbLi that must be considered. The pool temperature drops to 580°C after 1.5 h due to heat conduction into the walls of the pipe chase. By 2.8 days the pool reaches a minimum temperature of 480°C, when the heat conduction through the wall comes into equilibrium with the decay heat produced within the pool. After 7 days, the pipe chase concrete wall reaches a maximum temperature of 460°C at the inner wall surface in contact with the PbLi pool, and the temperature gradient through the wall is 380°C. At these temperatures, water will be driven out of the concrete, requiring a steel liner to separate the pool from the concrete surface to avoid PbLi pool-water reactions.

For a spill of ~100 m³, 325 Ci of $^{210}$Po and 1.6 $\times$ 10⁵ Ci of $^{203}$Hg will be in the pool. Two key processes are involved in the release of these isotopes into the atmosphere of the pipe chase: diffusion from the pool to the pool surface and vaporization off the pool surface. Diffusion to the pool surface is modeled by the following 1-D diffusion equation:

$$\frac{\partial C_i}{\partial t} = D_i \frac{\partial^2 C_i}{\partial z^2},$$

where

- $C_i$ = concentration of isotope $i$ in the pool
- $D_i$ = diffusion coefficient (m²/s)
- $z$ = depth of pool (m)
- $i$ = isotope designation, either $^{210}$Po or $^{203}$Hg.

The diffusion coefficient in the liquid metal can be estimated using the Scheibel modification of the Wilke-Change correlation.31 It relates diffusivity $D$ (cm²/s) to the viscosity ($\eta$) of the liquid in Poise, the temperature of the pool in Kelvin, and the molar volumes of the species ($V_a$) and the liquid ($V_m$) in cm³/gmole:

$$D = 8.2 \times 10^{-10} \left[ 1 + \left( \frac{3V_a}{V_m} \right)^{2/3} \right] \frac{T}{V_m^{1/3}}.$$

For Hg and Po, the values of $D$ range from 1.5 $\times$ 10⁻⁵ cm²/s at 200°C to 1.5 $\times$ 10⁻⁴ cm²/s at 1000°C.

Once at the surface the Hg and Po behave quite differently. The Hg will be well above its boiling point of 360°C for most of the transient and thus we assume that it completely vaporizes once it reaches the surface; that is, the concentration of $^{203}$Hg ($C_{203\text{Hg}}$) equals zero at the pool surface in solving Eq. (6). For Po, the vaporization is based on an assessment of release data from laboratory experiments in Russia, Germany, and the United States over the past two decades.30 Because release of Po as PbPo should be in aerosol form, we use the vaporization rate $\Gamma$ (Ci/cm²-h), recommended by Schipakin:

$$\Gamma_{^{210}\text{Po}} = 3.0 \times 10^5 P_{sat}(T) (1000/T)^{1/2} \times (1.95 \times 10^{-11} x/x_o),$$

where $P_{sat}(T)$ is the saturation vapor pressure for PbPo in mm of Hg and the ratio $x/x_o$ is a linear correction factor, equal to 4.7, to account for the fact that the Po concentration in the PbLi in ARIES-CS is 4.7 times higher than the 1.95 $\times$ 10⁻¹¹ mole fraction in Schipakin’s experiments. Equation (8) was used as a surface boundary.
condition for Eq. (6) through the Von Neumann–type boundary condition as follows:

$$-D_{210\text{Po}} \left. \frac{\partial C_{210\text{Po}}}{\partial z} \right|_{\text{surface}} = \Gamma_{210\text{Po}}.$$  (9)

Based on this model, the predicted quantity of $^{210}\text{Po}$ and $^{203}\text{Hg}$ released from the pool is given in Fig. 17. As can be seen, 17.0 Ci of $^{210}\text{Po}$ and 98.7 Ci of $^{203}\text{Hg}$ are mobilized from the pool in 7 days. Based on the dose conversion factors calculated by Ref. 13 for a ground release during conservative weather conditions, $^{210}\text{Po}$ has a dose of 3.9 mSv/Ci and $^{203}\text{Hg}$ has a dose of 0.41 mSv/Ci (both evaluated for a 1-km site boundary). Since the allowable dose for the maximally exposed individual at the site boundary is limited to 10 mSv, the ARIES-CS building will have to provide a confinement factor of at least 11 to guarantee that this dose limit is not exceeded after 7 days. Figure 18 contains the curies of $^{210}\text{Po}$ and $^{203}\text{Hg}$ released to the environment through the HTS vault heating/ventilating/air conditioning (HVAC) system and by leakage from the HTS vault. By 7 days, the quantity of $^{210}\text{Po}$ released reaches 0.13 Ci and the quantity of $^{203}\text{Hg}$ released reaches 0.56 Ci. This represents a confinement factor of more than 100, and as a consequence the dose at the site boundary after 7 days is only 0.51 mSv, which is below the site boundary limit for evacuation of 10 mSv. This confinement is due in large part to the chiller in the HTS vault that maintains the air in the HTS vault at atmospheric pressure, thereby reducing leakage from the vault to the environment. Eventually, corrective action will have to be taken to drain the pool from the pipe chase in order to stop the mobilization of $^{210}\text{Po}$ and $^{203}\text{Hg}$, but according to these results there is more than enough time to accomplish this corrective action during this event.

### IV.D. Double-Tube Breach Pressurization Event

Bounding events are used in safety assessments to establish the margins of safety that a confinement design possesses. One of the more severe events in this category for the ARIES-CS design is the double-tube breach pressurization event. The postulated initiating event for this accident is the failure of a first-wall cooling channel. This failure rapidly depressurizes the first-wall helium cooling system, inducing a large pressure gradient on, and vibrations in, the tubes of the Brayton cycle helium-helium heat exchanger, leading to the failure of a single heat exchanger tube. The consequence is that the high-pressure helium inventory from the Brayton cycle adds to the pressurization of the vacuum vessel and cryostat already imposed on these structures by the blowdown of the first-wall helium cooling system. The resulting cryostat pressure will exceed the setpoint for opening the rupture disk between the cryostat and the HTS vault, allowing high-pressure helium to vent into the vault. As the pressure rises in the vault, the vault HVAC system will automatically isolate the vault. The end result is an accident that bypasses the first confinement boundary of ARIES-CS, which is the vacuum vessel, and challenges the confinement function of the second confinement boundary, which is the cryostat-vault combination. This means that activated dust and tritium from within the vacuum vessel is carried by the helium stream into the vault, where it can now leak to the environment through various pathways, such as door seals.

To model this accident, the MELCOR model presented in Sec. IV.B was modified by adding two additional volumes that simulate the HTS vault and the Brayton cycle helium volume. The characteristics for the vault are those of the ITER-FEAT design. The combined

---

**Fig. 17.** Quantity of $^{210}\text{Po}$ and $^{203}\text{Hg}$ released from PbLi pool during an ex-vessel pipe break accident.

**Fig. 18.** Quantity of $^{210}\text{Po}$ and $^{203}\text{Hg}$ leaked from ARIES-CS confinement building during an ex-vessel pipe break accident.
volume of the vault plus pipe chases is 39 240 m³. The walls, ceiling, and floor were modeled as 1-m-thick concrete heat structures. The vault HVAC system is designed to give a one volume exchange per day and to isolate at a pressure of 120 kPa. The leak rate from the vault to the environment is estimated to be a volume exchange per day for a vault overpressure (with respect to the environment) of 400 Pa. This leak rate is scaled by the model as the square root of the predicted overpressure divided by 400 Pa, times one volume exchange per day. The Brayton cycle helium volume was estimated to be 445 m³ based on scaling the volume associated with the closed Brayton cycle examined by Ref. 18 to the total thermal load of one ARIES-CS sector. The temperature and pressure of this volume were set at 515°C and 15 MPa, respectively.

Figures 19, 20, and 21 present the pressure response of the ARIES-CS confinement boundaries to this event. Figure 19 shows the rather slow depressurization of the Brayton cycle system as a result of the single heat exchanger tube failure, taking more than 2000 s to come into an equilibrium with the HTS vault. This can be seen more clearly in the pressures of Fig. 20, which also contains the predicted pressure in the cryostat during this accident. The cryostat pressure reaches the setpoint for the rupture disk within 75 s of the initiation of this accident. However, what is of ultimate concern regarding leakage to the environment is the overpressure history of the vault, as presented in Fig. 21. As can be seen, it takes about 8000 s (2.2 h) for the vault to leak the helium to the environment.

The source terms at risk during this accident are the activated tungsten dust and tritium inventories in the vacuum vessel. As stated in Sec. III.A, the present estimate is that the maximum quantity of dust available will be kept to 65 kg. This dust is expected to be small, micron size in diameter, and should be immediately resuspended by the first-wall helium jetting into the vacuum vessel from the initial first-wall breach. As detailed in Sec. III.C, most of the tritium resides in the structure of the blanket and cooling system components of ARIES-CS, in particular the niobium tubes of the PbLi-helium heat exchanger. The release of this inventory will be governed by the rate at which this tritium can diffuse out of these structures. To simulate this process, the TMAP model of Sec. III.C was modified by adding two additional flow paths representing the first-wall breach and the heat exchanger tube break. The volumetric flow rate required by TMAP to calculate the convection of tritium through the failed system was set at the break flow predicted by MELCOR as it appears in Fig. 22. Because this accident will result in a loss of cooling for the PbLi cooling system, the
heat exchanger tube temperature was set at the outlet temperature for the PbLi of 730°C. Figure 23 shows the mobilized tritium inventory (released into the vacuum vessel) over a period of one week. By the end of this week less than 60 g of the entire ARIES-CS inventory has been mobilized. A much smaller tritium inventory of \( \sim 54 \) g will reside in the divertor tungsten of the divertor component.\(^{10}\) The mobilization of this inventory was not modeled and for conservatism is assumed to be released at the start of the accident.

These radioactive source terms were added to our MELCOR model through the use of the aerosol package of the MELCOR code and the accident analysis rerun to determine the quantity of material released to the environment. The results are shown in Figs. 24 and 25. By the time the vault depressurizes, 27 g of tritium and 37 g of tungsten dust has been released to the environment. Based on the predicted dose conversion factors presented in Ref. 13, the dose at the site boundary associated with a ground release of 27 g of tritium during conservative weather conditions is 8.5 mSv. In addition, the tungsten dust would add another 0.6 mSv of dose, bringing the total to 9.1 mSv. This dose is below the 10-mSv no-evacuation limit, demonstrating that some margin is provided by the ARIES-CS confinement system even for this severe accident. It should be pointed out that this dose estimate is primarily due to the tritium release, and because this release is dominated by the divertor...
inventory, which was conservatively assumed to be released at the start of the accident, the actual margin is much larger than this dose result suggests.

V. RADIOLOGICAL WASTE MANAGEMENT

In general, ARIES plants generate only low-level waste (class A or C), which requires only near-surface, shallow land burial. However, with the recent introduction of the clearance category for slightly radioactive materials and the development of radiation-hardened remote-handling equipment, the possibility of clearing and recycling the majority of the radwaste can also be considered for ARIES-CS. Historically, the ARIES project has been committed to the goal of radwaste minimization. For ARIES-CS, we have examined the radwaste issue as follows:

1. minimize volume of active materials by design
2. recycle in-vessel components, if economically and technologically feasible
3. clear slightly irradiated materials
4. burn long-lived radionuclides in fusion devices.

Discussed below are the three scenarios for ARIES-CS radwaste management: disposal in geological repositories, recycling and reuse within the nuclear industry, and clearance or release to the public sector. The neutronics analysis for this waste management assessment uses the same models described in Sec. III.

V.A. Disposal

There are two categories of materials that are candidates for disposal according to the official U.S. criteria: high-level waste (HLW) and low-level waste (LLW). The NRC has defined two more categories for LLW: class C and class A. Class C waste requires engineered intruder barriers and a minimum disposal depth of 8 m. The lower-level class A waste must meet only the minimum packaging requirements. We evaluated the waste disposal rating (WDR) for a compacted waste using the most conservative waste disposal limits developed by Fetter, Cheng, and Mann and NRC-10CFR61 (Ref. 39). Here, we report the class C WDR at 100 yr after shutdown, allowing the short-lived radionuclides to decay. A WDR $< 1$ means LLW and a WDR $> 1$ means HLW. The ARIES-CS WDRs are $< 1$, meaning that all components qualify as class C LLW. The WDRs of the vacuum vessel and external components are very low ($< 0.1$), to the extent that these components could qualify as class A LLW. Excluding the clearable components (cryostat and bioshield), $\sim 70\%$ of the waste (blanket, shield, divertor, and manifolds) is class C LLW. The remaining $\sim 30\%$ (vacuum vessel and magnet) would fall under the class A LLW category. Table III identifies the class A and class C components along with the clearable components.

V.B. Clearance

Clearance (unconditional, unrestricted release) means that the material can be handled as if it is no longer radioactive. Solid materials can be reused without restrictions, recycled into a consumer product, or disposed of in a nonnuclear landfill, with no controls. After plant decommissioning, individual materials could be stored for a specific period ($< 100$ yr), segregated, and then released to the commercial market if the clearance index (CI) falls below 1. The CI is the ratio of the activity (in Ci/g) to the allowable limit summed over all radioisotopes. Over the past five years, clearance guidelines have been issued by the NRC (Ref. 40), the European Commission, and the International Atomic Energy Agency (IAEA). Based on a detailed technical study, the 2003 NUREG-1640 document contains estimates of the total effective dose equivalent (from which the CI can be derived) for 115 radionuclides that could be present in activated steel, copper, aluminum, and concrete from decommissioning of nuclear facilities. The NRC has not yet issued an official policy on the unconditional release of specific materials. Therefore, the proposed annual doses reported in the NUREG-1640 document will be referred to as the proposed U.S. limits.

The clearance limits developed by the IAEA (Ref. 41) over the past two decades are for 257 radionuclides, and these limits have been used worldwide for a diverse range of fusion concepts. Even though both standards recommend an annual dose of 10 $\mu$Sv as the basis for clearance of solids from regulatory control, there are still notable differences in the approach to clearance, and the specific materials considered for clearance.

### Table III

<table>
<thead>
<tr>
<th>Structure</th>
<th>Class C LLW</th>
<th>Class A LLW</th>
<th>Could Be Cleared?</th>
</tr>
</thead>
<tbody>
<tr>
<td>First wall/blanket/back wall</td>
<td>✓</td>
<td></td>
<td>No</td>
</tr>
<tr>
<td>Divertor system</td>
<td></td>
<td></td>
<td>No</td>
</tr>
<tr>
<td>Shield/manifolds</td>
<td></td>
<td>✓</td>
<td>No</td>
</tr>
<tr>
<td>Vacuum vessel</td>
<td></td>
<td>✓</td>
<td>No</td>
</tr>
<tr>
<td>Magnet</td>
<td></td>
<td>✓</td>
<td>No</td>
</tr>
<tr>
<td>Nb$_3$Sn</td>
<td></td>
<td>❌</td>
<td>✓</td>
</tr>
<tr>
<td>Cu stabilizer</td>
<td></td>
<td>✓</td>
<td>✓</td>
</tr>
<tr>
<td>JK2LB steel</td>
<td></td>
<td>✓</td>
<td>✓</td>
</tr>
<tr>
<td>Insulator</td>
<td></td>
<td>✓</td>
<td>✓</td>
</tr>
<tr>
<td>Cryostat</td>
<td></td>
<td>✓</td>
<td>✓</td>
</tr>
<tr>
<td>Bioshield</td>
<td></td>
<td>✓</td>
<td>✓</td>
</tr>
</tbody>
</table>

FUSION SCIENCE AND TECHNOLOGY VOL. 54 OCT. 2008
differences between the limits for radionuclides these standards have in common. This inconsistency is due to the differences in the adopted approach to radwaste clearance.

The CI of each component depends strongly on the neutron flux level, neutron spectrum, composition, and service lifetime. Because of the compactness of ARIES-CS, the CIs of all internal components (blanket, shield, manifolds, and vacuum vessel) exceed the clearance limit by a wide margin even after an extended period of 100 yr (refer to Fig. 26). This means that the in-vessel components should be recycled or disposed of in repositories as LLW according to the U.S. waste classification guidelines. Examining the magnet constituents indicates that the JK2LB steel structure is preferred over Incoloy H23041 and that both the JK2LB structure and the Cu stabilizer are clearable within 100 yr based on the U.S. guidelines. Of interest is the 2-m-thick external concrete building (bioshield) that surrounds the torus. It represents the largest single component of the decommissioned waste. Fortunately, the bioshield along with the 5-cm-thick cryostat and some magnet constituents qualify for clearance, representing ~80% of the total active material volume.

Since the ultimate goal is to separate the constituents of the component for recycling and reuse by industry, the ARIES-CS approach for handling the cleared components (CI < 1) is to reevaluate the CIs for the component constituents. The entire component could have a CI < 1, but the individual constituents may not, and vice versa, requiring further segregation of the activated materials based on component constituents rather than components. Figure 27 confirms that the magnet Nb3Sn superconductor cannot be cleared because of 94Nb (from Nb) and the magnet polyimide insulator cannot be cleared because of 14C (from N). The remaining magnet constituents can be cleared, however, within 20 to 300 yr.

We propose the same approach to deal with other sizable components, such as the 2-m-thick bioshield. It should be segmented and reexamined. Therefore, the bioshield was divided into four segments (0.5 m each) and the CIs reevaluated for the constituents (85% type-04 ordinary concrete, 10% mild steel, and 5% He, by volume). The results indicate that the innermost segment has the highest CI, whereas the outer three segments meet the clearance limit within a few days after decommissioning. As Figs. 28 and 29 indicate, the mild steel is a major

![Fig. 26. Decrease of ARIES-CS CI with time after shutdown.](image1)

![Fig. 27. Variation of IAEA CI of winding pack constituents with time after shutdown. The U.S. CI for JK2LB steel and Cu stabilizer are included for comparison.](image2)

![Fig. 28. Comparison of U.S. and IAEA CIs for steel of innermost segment of bioshield.](image3)
contributor to the CI even though its volume fraction is only 10%. The recommended storage periods are given in Table IV along with the dominant radionuclides in descending order. Note that the inconsistencies in the $^{14}\text{C}$, $^{54}\text{Mn}$, and $^{63}\text{Ni}$ clearance standards result in a wide variation in the required storage period for the Cu stabilizer, coil structure, and mild steel.

V.C. Recycling

We examined the possibility of recycling the non-clearable in-vessel components (blanket, shield, divertor, and vacuum vessel) of ARIES-CS. All components can potentially be recycled using conventional and advanced remote-handling (RH) equipment that can handle 10 mSv/h (1000 times the hands-on dose limit) and high doses of 3000 Sv/h or more, respectively. The variation with time of the recycling dose shows a strong material dependence (refer to Fig. 30). The first wall is an integral part of the blanket. It is shown in Fig. 30 as a separate component to provide the highest possible dose to the remote-handling equipment. Manganese-54 (from Fe) is the main contributor to the dose of ferritic steel–based components (first wall, blanket, shield, manifolds, and vacuum vessel). Storing the first wall/blanket temporarily for several years helps drop the dose by a few orders of magnitude before recycling. After several life cycles, advanced remote-handling equipment could handle the shield, manifolds, and vacuum vessel. For the SiC inserts, the main contributors to the dose are $^{58,60}\text{Co}$, $^{54}\text{Mn}$, and $^{65}\text{Zn}$, originating from SiC impurities. More strict impurity control may allow hands-on recycling of SiC.

V.D. Magnet Structure: Incoloy® 908 or JK2LB?

Two candidates steels were proposed for the ARIES-CS magnet structure: Incoloy® 908 and JK2LB (Ref. 42). The former, a nickel-based alloy developed by the United States, contains 3 wt% Nb as an alloying element that raised an activation concern. The materials of the alternate JK2LB structure, low-carbon steel developed in Japan for the ITER central solenoid, were carefully chosen to minimize the long-lived radwaste products.


---

**TABLE IV**

<table>
<thead>
<tr>
<th>Constituent</th>
<th>U.S.</th>
<th>IAEA</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cu stabilizer</td>
<td>20 yr</td>
<td>$^{60}\text{Co}$, $^{63}\text{Ni}$</td>
</tr>
<tr>
<td>Intercoil structure (JK2LB)</td>
<td>10 yr</td>
<td>$^{54}\text{Mn}$, $^{63}\text{Ni}$</td>
</tr>
<tr>
<td>Cryostat (Type 304 stainless steel)</td>
<td>64 yr</td>
<td>$^{60}\text{Co}$, $^{63}\text{Ni}$, $^{65}\text{Co}$</td>
</tr>
<tr>
<td>Bioshield</td>
<td>3.5 yr</td>
<td>$^{54}\text{Mn}$, $^{55}\text{Fe}$, $^{65}\text{Fe}$</td>
</tr>
<tr>
<td>Mild steel</td>
<td>0.6 yr</td>
<td>$^{22}\text{Na}$, $^{54}\text{Mn}$, $^{55}\text{Fe}$, $^{45}\text{Ca}$, $^{53}\text{Fe}$</td>
</tr>
<tr>
<td>Type-04 ordinary concrete</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

---

Fig. 29. Comparison of U.S. and IAEA CIs for concrete of innermost segment of bioshield.

Fig. 30. Variation of dose with time at end of service lifetime of ARIES-CS in-vessel, nonclearable components.
We applied the three waste management approaches to both steels and evaluated the WDR, CI, and recycling dose to select the most favorable steel from the radiological viewpoint. Our results show that both Incoloy® and JK2LB qualify as class A LLW with WDRs of 0.1 and 0.0003, respectively. According to the clearance guidelines, the JK2LB steel can be recycled within the nuclear industry or preferably released to the commercial market for reuse after ~1 yr following shutdown, but the Incoloy® steel cannot because of its high Nb content. Figures 31 and 32 display the clearance results, showing the very slow decay rate of Incoloy®. The recycling dose of Fig. 33 suggests that JK2LB can be recycled hands-on within a few months after shutdown, whereas Incoloy® requires remote-handling equipment to handle the high dose from Nb. Based on its favorable environmental and economic characteristics, JK2LB was selected as the preferred steel for the ARIES-CS magnet structure. The transition to the JK2LB steel involved several changes to the magnet design and engineering system. For more details, the reader is directed to Ref. 42.

VI. CONCLUSIONS

In this paper, we examine the safety and environmental performance of ARIES-CS by reporting radiological inventories, decay heat, and radioactive waste (radwaste) management options (disposal, recycling, and clearance) and by examining the response of ARIES-CS to accident conditions. These accidents included 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). The goal of this safety study was to demonstrate that the high-level requirements of the DOE Fusion Safety Standard2 are met by ARIES-CS, which are that a site evacuation plan is not needed and that radioactive waste shall be minimized.

We have demonstrated through the use of reduced activation materials, a passive decay heat removal system, and the defense-in-depth confinement strategy that the radioactive source terms of ARIES-CS are contained during the DBAs analyzed, to the degree that no site evacuation plan is needed for ARIES-CS. In general, for the accidents analyzed, except the accident divertor LOCA/LOFA, the temperatures reached by the in-vessel...
components of ARIES-CS did not exceed levels that would prevent their reuse upon recovery from these accidents. We have also demonstrated, during a severe bypass accident scenario, that this confinement strategy provides enough safety margin that the dose to the maximally exposed individual at the site boundary is less than the 10-mSv dose limit for public evacuation, even during conservative weather conditions.

In this present work, our goal has been to minimize the production of waste that fails to qualify for class C, or low-level land burial through the selection of reduced or low-activation structural and breeding materials for ARIES-CS. We also examined more environmentally attractive scenarios, such as recycling and clearance. The result of our investigation is that internal components of ARIES-CS will have to be recycled or disposed of in LLW repositories but that portions of the magnets, cryostat, and bioshield could meet the national and international clearance criteria for release to private industrial markets.

ACKNOWLEDGMENT

This work was prepared for the U.S. DOE, Office of Fusion Energy Sciences, under DOE Idaho Field Office contract DE-AC07-05ID14517.

REFERENCES


