ARIES-ACT1 SAFETY DESIGN AND ANALYSIS

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ARIES-ACT1 is a 1000 MW(t) tokamak design featuring advanced plasma physics and divertor and blanket engineering. Some relevant features include an advanced SiC blanket with PbLi as coolant and breeder, a helium cooled steel structural ring and tungsten divertors, a thin-walled, helium cooled vacuum vessel, and a room temperature water-cooled shield outside the vacuum vessel. We consider here some safety aspects of the ARIES-ACT1 design, and model a series of design-basis and beyond design-basis accidents with the MELCOR code modified for Fusion. The presence of multiple coolants (PbLi, helium, and water) makes possible a variety of such accidents. We consider here a loss of flow accident (LOFA) caused by a long-term-station blackout (LTSBO), an ex-vessel helium break into the cryostat, and a beyond design-basis accident in which a LTSBO is aggravated by a loss of the coolant (LOCA) in ARIES-ACT1’s ultimate decay heat removal system, the water cooled shield. In all cases we find that secondary confinement boundaries are not challenged, and the structural integrity of in-vessel components is not threatened by high temperatures; decay heat can be passively removed in all cases.

I. INTRODUCTION

In light of advances in the understanding of tokamak physics and technology occurring in over a decade since the ARIES-AT (advanced tokamak) design study, the ARIES-ACT (Ref. 1) design study was undertaken to re-examine the possibilities for an attractive tokamak power plant design. While maintaining as a primary goal a competitive cost of electricity, the ARIES-ACT study investigates in particular the trade-offs inherent in certain design choices. Identification of these design points is made possible by the ARIES Systems Code, 2 which is equipped with costing algorithms that allow for determination of the effect of various design parameters on the cost of electricity generated by the reactor. This spectrum of design space is naturally bounded by “four corners” representing aggressive and conservative choices in the plasma physics and engineering/technology design. The ARIES-ACT1 design is based on both aggressive plasma physics and engineering, and is the subject of this article. A detailed tokamak design featuring more conservative design choices in both physics and engineering (ARIES-ACT2) is forthcoming.

In addition to a competitive cost of electricity, an attractive power plant design must also meet a high standard of safety. The U.S. Department of Energy (DOE) Fusion Safety Standard 3 stipulates that fusion reactor accident consequences should be sufficiently benign so as not to require a site evacuation plan; to meet this requirement the dose to a maximally exposed individual at the site boundary (under conservative weather assumptions, etc.) cannot exceed 10 mSv. So, as with previous ARIES designs, 4,5 we consider here a variety of accident scenarios that could lead to release of radioactive material to ensure that the ARIES-ACT designs meet this safety standard. Such accidents fall into one of 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 vacuum vessel bypass);
2. Events associated with the chemical reactivity of materials (e.g., loss of PbLi coolant and subsequent interaction with air or water);
3. Events related to loss of decay heat removal (e.g. complete loss of flow or loss of coolant); these could lead to possible structural damage or melting that could result in a release of radioactive material.

Events of the third type have received particular scrutiny in light of the Fukushima accident- not only for fission, but also for fusion reactors (see, for example, Ref. 6). The aggressive nature of both the ARIES-ACT1 physics and engineering design, which push the limits of temperature and heat flux on materials, make these concerns greater for this device relative to other designs. Thus, we focus here on accidents of this type, and consider three such cases: 1) a design-basis loss of flow accident (LOFA) in which forced convection is lost in all coolant loops as a result of a long-term-station blackout (LTSBO), and this LOFA coupled with both 2) an ex-vessel loss of helium coolant (helium LOCA) and 3) a loss of the coolant in ARIES-ACT1’s ultimate decay heat removal system, the water cooled shield. Both of the latter are beyond design-basis accidents. Since our present focus is on decay heat removal, we retain the PbLi in these accidents since it is itself an important contributor to the overall decay heat.

II. ARIES-ACT1 DESIGN DESCRIPTION

The layout of the ARIES-ACT1 power core is shown in Figure 1. We give here a brief description of some of the primary components included in the MELCOR model, and note that further detail on these is to be found in reference 7 and the other articles in this issue.

Similar to the ARIES-AT (advanced tokamak) design, we adopt an advanced SiC blanket concept with PbLi as coolant and breeder. The PbLi flows upward through a thin outer annular shell at high velocity to provide sufficient cooling for the hot first wall, and then returns flowing downward at lower velocity through a larger center channel (Figure 2). There is one such blanket on the inboard and two on the outboard side, and the two outboard blankets are separated by a thin tungsten kink shell; it is noteworthy here since, though small, its decay heat is high. The PbLi is fed to the blankets from a toroidal ring header visible at lower right in Figure 1.

The other two ring headers supply helium coolant to all the in-vessel components. These include the divertor, which comprises solid tungsten plasma facing material (armor) and back plates constructed of a tungsten alloy support structure and oxide dispersion strengthened (ODS) steel manifold structure. This helium also cools the ferritic (ODS or nanostructured) steel structural ring, which supports the in-vessel components and acts as a high temperature (700 °C) heat shield for the magnets and cryostat, and a thin-walled (10 cm total, double-walled) vacuum vessel.

The latter represents a departure from previous ARIES concepts. The ARIES-AT vacuum vessel was a thick (25-40 cm), water-cooled, low activation ferritic steel structure that also served as a neutron shield for the magnets. For ARIES-ACT, concerns about tritium permeation through the vacuum vessel wall at high temperature and the resulting possibility of a large inventory of tritium accumulating in the vacuum vessel water coolant prompted an investigation of helium-cooled designs. Since this new (thin, helium cooled) vacuum vessel does not provide
adequate shielding for the magnets, an additional water-cooled shield is included outside the vacuum vessel for that primary purpose. This water-cooled shield also represents the ultimate heat sink for the reactor’s decay heat during most design-basis (DBA) and beyond design-basis

Figure 1. Layout of ARIES-ACT1 power core.

Figure 2. ARIES-ACT1 outboard blanket and PbLi flow paths.
(BDBA) accidents. While the helium-cooled vacuum vessel is intended to operate at high (350-500 °C) temperature and high helium pressure, the water-cooled shield operates near room temperature and low pressure water. Both the vacuum vessel and water-cooled shield are constructed of 3Cr-3WV bainitic steel (Ref. 10).

The arrangement of these components is shown in Figure 1. Worth noting, and not obvious in that figure, is the fact that most of the outboard surface (outside the vacuum vessel) is surrounded not by the water-cooled shield, but by a series of helium-cooled shield blocks that reside in the maintenance ports. These play a passive heat removal role during the final accident analyzed in this paper. This is more easily recognized in the top view in Figure 3 below, and it becomes important in the accident analysis that follows.

![Figure 3. Top view of ARIES-ACT1 power core illustrating the position of the helium cooled shield blocks within the maintenance ports.](image)

In addition to providing shielding for the magnets, and similarly to previous ARIES designs\(^5\), the water-cooled shield also provides passive decay heat removal by natural circulation in the event of a loss of power or loss of flow accident. The heat exchanger for this system is located on the roof of the confinement building, and is thus able to transfer heat from the circulating water to ambient air. The ability of this system to remove decay heat in just such a scenario (or inability, in the event of its malfunction) is the focus of the analyses described subsequently.

**III. MELCOR MODEL**

In order to assess the consequences of a series of accident scenarios, a model of ARIES-ACT1 has been developed using the MELCOR for fusion code. MELCOR is a code developed at Sandia National Laboratory\(^1\), originally for analysis of light water fission reactor accidents. A series of modified versions have been developed at INL for application to fusion accidents. MELCOR 1.8.5 (Ref. 12) was unique among these code versions in that the default water coolant could be replaced by a number of others relevant for fusion, including PbLi, a feature as yet unavailable in the other fusion versions of the code.
Though MELCOR 1.8.5 can use a variety of working fluids, as with all versions of MELCOR, only one such multi-phase fluid may be used in a given problem (numerous non-condensible ideal gases can be used). This presents some obvious challenges when modeling the present ARIES-ACT1 design, since there is no way to include both PbLi and water coolants within the same model. In order to overcome this, the present model is actually a scripted coupling of two separate MELCOR models, one containing the PbLi and helium cooled components and loops, and the other containing the water cooled shield and associated heat transfer system. The two models run concurrently, and information is passed between the two in internally in the code; the outer surface temperatures of the vacuum vessel and water cooled shield from each model are used to update the surface heat fluxes between these at the start of a time step. Both models wait at the end of the time step to exchange the latest information before proceeding to the next time step. Conservation of energy is verified through output information. An additional benefit of this method is that it essentially parallelizes the problem, and a modest improvement in the wall clock time required to complete the simulation is realized.

The MELCOR model of ARIES-ACT1 comprises 1/3 of the tokamak. The primary heat structures and fluid volumes included in the PbLi/helium system model are:

- Three blankets (one inboard and two outboard) each with:
  - Inboard, outboard, lower and upper SiC blanket structures and PbLi flow channels;
- Upper and lower divertors, including:
  - Tungsten armor
  - ODS steel structure
  - Helium coolant
- The steel structural ring (high temperature shield) and helium coolant;
- The vacuum vessel segments representing the inboard, outboard, top, and bottom of the vessel.

These components are shown in the schematic in Figure 4.

Figure 4 does not show the water-cooled shield, which is part of a different input file as described above. The model of the water-cooled shield parallels the helium-cooled vacuum vessel, with four structures, each with a corresponding control volume, to which heat can be radiatively transferred from the vacuum vessel. The vacuum vessel, which operates at significantly lower temperature than the other helium cooled components (high temperature shield and divertor), is cooled by a separate helium loop as shown in Figure 4.

All the accident models begin with a short (1000 s) initialization period, which includes surface heat fluxes on the FW and divertors as well as nuclear heating in all components based on the analysis in Ref. 13, in order to establish reasonable initial reactor temperature and flow conditions. After this time, an accident is initiated and after a plasma disruption the reactor is shutdown to decay heat levels, in all of the components and PbLi coolant. Decay heats are added for each component, as a function of time, based on the analysis in Ref. 13; the total decay heat in the ARIES-ACT1 MELCOR model is shown in Figure 5.
Figure 4. A schematic of the primary components of the PbLi and He cooled systems in ARIES-ACT1 as modeled in MELCOR. Three coolant loops are present: two helium loops (one for the highest temperature components, the divertors and structural ring, and one for the VV), and the PbLi loop through the blankets. Solid heat structures are labeled with the prefix “HS” and coolant volumes with “CV”; arrows indicate the flow direction. Note that the ex-VV heat transfer systems and the water-cooled shield and associated water systems are not shown here.
In an accident scenario, the decay heat removal paths in the model include 1) heat transfer from the coolant loops (water, helium, or PbLi) to the respective heat exchangers (provided these are intact), and 2) heat transfer radially outward via conduction, convection, or radiation as applicable. We do not presently account for heat transfer from the water-cooled shield to other structures in the cryostat (e.g. the magnets); while such heat transfer paths are well-insulated during normal operation, this insulation (e.g. multi-layer insulation designed to protect against radiative heat transfer) would likely be compromised by high temperatures experienced during a severe accident. Thus, the assumption of an adiabatic boundary here represents a considerable conservatism in the present model. As noted previously, the shield blocks, which reside in the maintenance ports (which are part of the vacuum vessel volume) cover most of the outboard surface area. These are assumed to radiate from the back side to a room temperature (300 K) boundary. This may not be an appropriate assumption if this becomes the only heat removal mechanism, a point we will comment further on in subsequent sections.

IV. LOSS OF FLOW ACCIDENT

We first consider a loss of flow accident (LOFA) resulting from long-term station blackout (LTSBO). In this scenario, all offsite power (and as a result forced convection in the water, helium, and PbLi loops) is lost. It is desired that decay heat can be passively removed (indefinitely) in this scenario, and all structures adequately cooled, by the water-cooled shield operating by natural convection. At the initiation of the accident, the flow rate in the water loop (Figure 6) drops abruptly from its operating value (~86 kg/s), then increases (driven by natural convection) over the first half-day to around 10 kg/s, where it remains relatively constant for the duration of the modeled accident (~30 days). The flow rate shown here is from the exit of the water cooled shield (WCS) to the heat transfer system (flow path 701), see Figure 7. The temperature of the water increases over the first day to almost 100 °C (but does not boil), remains constant over a couple days, and slowly decreases thereafter for the duration of the accident (Figure 8).
Figure 6. Mass flow rate from the water-cooled shield outlet to the natural convection heat transfer system.

Figure 7. Schematic of the water heat transfer system.
Temperatures of the various inboard (Figure 9) and outboard (Figure 10) structures are shown below. The highest temperature component in the model is the kink shell; this, coupled with the fact that the primary heat sink (the water cooled shield) is on the inboard side, results in higher outboard than inboard temperatures. Heat is also transferred away from the outboard to be radiated away by the shield blocks. The transfer of heat to components successively further from the plasma is evident in the temperature increases on both the inboard and outboard sides in the figures below. The structures farthest from the plasma peak in temperature at about 2.9 days. Here it can be seen that the temperatures of the divertors and first wall (inboard and outboard) do not increase significantly above their normal operating temperatures. The inboard vacuum vessel and water cooled shield also experience minimal temperature increases. The corresponding outboard structures (including the shield blocks) experience a larger temperature increase later in the transient, but do not reach levels that raise concerns about their structural integrity. The structural ring (both inboard and outboard) experiences the most dramatic temperature transient; ~1000 seconds in to the accident, a substantial temperature gradient develops across it, and the hot side reaches temperatures in excess of 860 °C (inboard) and 890 °C (outboard). The acceptability of these temperatures depends on the nature of the ODS steel used that makes up the structural ring. 9Cr ODS steels undergo a ferrite-austenite transformation at ~850 °C (Ref. 14) that limits their use to temperatures of ~800 °C (Ref. 7). ODS steels with a higher Cr content (12-15%) that do not experience this phase transition may be suitable for use to higher temperatures (~1000 °C, Ref. 15). In the event ARIES-ACT1 is held to the lower temperature limit, the structural ring would have to be replaced following an accident of this kind, and a more detailed structural analysis of this component is probably warranted. For a more detailed discussion of these materials issues as they pertain to ARIES-ACT1, see Ref. 7.
Some fluctuations in the temperatures of the blanket structures, kink shell, and structural ring are apparent, particularly on the outboard side. These are an artifact of the simplified PbLi cooling system model (Figure 11). The control volumes in the PbLi cooling system have no corresponding heat structures (which would represent pipe walls), i.e. they have no mechanism for heat removal by conduction and subsequent convection to exterior volumes. The temperatures in these volumes increase (as a result of PbLi decay heating) throughout the transient. This would normally establish some flow by natural convection, but in this version of the MELCOR model the loop was not sufficiently elevated above the vessel to do so. As a result, some of the PbLi temperatures in the coolant loop become unphysically high, which drives some flow reversals in the blankets and results in the abrupt changes visible near the end of the transient.
Figure 11. PbLi heat transfer system.

V. HELIUM LOSS OF COOLANT ACCIDENT

Helium is used to cool all structures other than the blankets and water cooled shield in ARIES-ACT1. In our MELCOR model we consider two separate high and low temperature helium cooling systems, each with three distinct loops that cool 1/3 of the tokamak. The lower temperature system cools the vacuum vessel; the high temperature system cools the structural ring ($T_{\text{out}} \sim 680 ^\circ \text{C}$) and divertors ($T_{\text{in}} \sim 700 ^\circ \text{C}$). The helium that exits the divertors and re-enters the high temperature helium coolant loop does so at 10 MPa and $\sim 800 ^\circ \text{C}$, and as such represents a pressure source that may challenge confinement boundaries. The design limits of those boundaries set the allowable helium inventory for each single loop cooling 1/3 of the tokamak.

The vacuum vessel is the primary confinement boundary. As the vacuum vessel is not a pressure vessel (it is designed for a pressure difference of 0.1 MPa), it can only withstand 0.2 MPa (absolute). To avoid over-pressurization of the vacuum vessel it is equipped with a rupture disk to allow venting to the cryostat, the ultimate confinement system for design-basis accidents. The cryostat has a design limit of 0.3 MPa.

In order to ensure that this pressure is not exceeded, we consider here a pipe break accident in one of the structural ring/divertor helium coolant loops (27 m$^3$ in-vessel, 26 m$^3$ ex-vessel). A one meter diameter double ended offset shear break of the pipe occurs in flow path 703, the helium outlet line at the exit from the vacuum vessel, and opens both volumes 702 and 704 to the cryostat (see Figure 12).
The blowdown occurs in ~1 second, at which point the peak pressure of 0.29 MPa is reached, i.e. it does not reach the design pressure of the cryostat. Over the next 500 seconds, the pressure drops as the gas cools through heat transfer to the cryostat wall and its thermal shield, and a minimum of 0.067 MPa is reached at around 1500 seconds. At that point a slow increase in pressure begins as the decay heat from the reactor is ultimately transferred by natural convection of the helium through the pipe break into the cryostat, which continues for ~3 days into the accident at which point it plateaus around 0.163 MPa. The initial portion of the transient is shown in Figure 13.
If this helium break were to occur during a LTSBO (a beyond design-basis accident), we do not anticipate higher reactor temperatures than occurred during the LTSBO alone (considered in section IV). This is because, in the design-basis LOFA, the helium is neither a heat source nor sink (it is not a significant heat removal mechanism), nor is it a barrier to heat transfer. When lost to the cryostat, it provides additional heat removal paths in the form of convection through the break, and conduction and convection within the cryostat, which is no longer at vacuum.

VI. COMBINED LOSS OF FLOW AND LOSS OF WATER COOLANT ACCIDENT

While a helium LOCA may not result in increased temperatures relative to the LTSBO, the situation is clearly different for the water coolant. In addition to its primary functions as a coolant and neutron shield, as discussed previously it serves a safety function as the primary decay heat removal system in the event of complete loss of forced convection. The decay heat not just from structures, but also from the PbLi (provided that the PbLi loops remain intact) must be removed, and a loss of power coupled with a water LOCA leaves few possibilities for heat removal. From a heat transfer perspective this is a worst case scenario. We consider the implications of such a beyond design-basis accident here.

Without the aid of the water coolant, heat can only be removed across the vacuum boundary to the cryostat via radiation. Of course, normal operation requires that the hot structures be well-insulated against this very thing, and we expect all exterior surfaces of the water cooled shield to be covered with multi-layer insulation. This insulation contains multiple layers across which heat must be successively radiated, resulting in a very low effective emissivity. While designed for use in cryogenic systems, in principle it could provide the same insulating effect during an accident. While it seems unlikely that the insulation will survive the high temperatures encountered in this case, lacking any detailed information about its composition (and as a measure of conservatism), we neglect radiative heat transfer across these boundaries.

Most of the outboard side of the vessel, however, does not radiate to this insulation, but rather to the shield blocks that occupy the maintenance ports (see Figure 3). These do not constitute a vacuum boundary, but simply provide the necessary neutron shielding for the field coils. The back side of the shield blocks will in turn radiate to the vacuum vessel maintenance port doors, which form the vacuum boundary and separate the ports from the bioshield and maintenance corridor (Figure 14). This is the primary heat removal path for this accident, and we assume the VV maintenance port doors to be fixed at 300 K. We also assume an emissivity ($\varepsilon$) of 0.9 for most surfaces; this is a high value, which in practice will have to be generated by an appropriate surface treatment or coating. In MELCOR the effective emissivity ($\varepsilon_{\text{eff}}$) between any two successive structures (with emissivity $\varepsilon$) is specified:

$$\varepsilon_{\text{eff}} = \frac{\varepsilon}{2 - \varepsilon} = 0.818$$

The MELCOR model for this accident breaks the water cooled shield pump outlet line (FL715, see Figure 7), and water begins to drain into an auxiliary room (CV143), which fills with water. Almost 42,000 kg of water drain from the system over the course of 2500 seconds;
the mass draining from all sections of the water-cooled shield and accumulating in room 143 is shown in Figure 15.

Figure 14. Layout of the tokamak building, showing vacuum boundary in relation to the maintenance ports and corridor.

Figure 15. Water mass in the water-cooled shield and auxiliary room as a function of time.
Figure 16. Inboard (and divertor) temperatures during loss of flow and loss of water coolant.

Figure 17. Outboard temperatures during loss of flow and loss of water coolant.

Structure temperatures initially progress similarly to those in the loss of flow accident. The divertor and blanket structures do not increase substantially in temperature. As previously, the structural ring experiences a high temperature gradient and high temperatures on the hot side that may necessitate its replacement. The temperature increases in the outer structures (VV, shield blocks, and water-cooled shield) are far more dramatic in this case. The water-cooled shield reaches a temperature of nearly 550 °C, the shield blocks 520-620 °C, and the vacuum vessel ~575 °C, all around 21 days into the accident, after which they decrease (more slowly than in the LOFA case). These are high temperatures, but not sufficient to cause concern about catastrophic failure of the components. Thus, even in this worst-case, beyond design-basis accident, no significant releases of radiation from the power core can be expected.
One word of caution is in order regarding this result. Recall that the shield blocks radiate to a boundary that we have fixed at room temperature. As the shield blocks heat up, the temperature difference between them and this fictitious boundary becomes large, and that drives the radiative heat transfer away from the vessel. When temperatures peak, the heat removal by this mechanism is nearly 10 MW (see Figure 18). With this amount of heat transfer through the vacuum boundary, it (along with the bioshield and other structures in the maintenance corridor beyond) will increase in temperature, reducing the temperature difference from the shield blocks and decreasing the effectiveness of the radiative heat transfer. In other words, the room temperature boundary assumption is not conservative. This point is worthy of some further analytical scrutiny.

![Figure 18. Power radiated from the shield plugs to the 300 °C maintenance corridor boundary.](image)

**VII. SUMMARY AND CONCLUSIONS**

The multiple coolants and coolant loops present in ARIES-ACT imply a number of different possible accident scenarios involving loss of flow or coolant. We have considered here three Fukushima-like scenarios involving long term station blackout (or loss of forced convection in all loops), and shown that no releases from the ARIES-ACT1 power core can be expected. The LTSBO/LOFA scenario alone constitutes a design-basis accident, and it has been demonstrated that the water coolant (which functions as both a neutron shield and emergency cooling system) adequately removes decay heat in this scenario. The only structure temperature that perhaps requires further attention is that of the structural ring, which may exceed allowable temperatures depending on the properties of the ODS steel specified in the design.

We also considered two beyond design-basis accidents in which the loss of power is accompanied by a LOCA, in the helium and water loops respectively. The helium LOCA does not represent an extreme for structure temperatures (helium in the cryostat provides another decay heat removal path), but it is necessary to ensure resulting pressure increases in the cryostat do not exceed the design limits of this confinement boundary, which has been demonstrated in section V.

In the final case, a water LOCA occurs during the power outage. Since the water is the intended emergency heat removal mechanism, and the intact PbLi heat transfer system
contributes considerable decay heat in this scenario as well, this represents a worst-case scenario from a decay heat removal standpoint. The only mechanism for its removal is radiation through the maintenance ports to the vacuum boundary. While the MELCOR simulations do not indicate any risk of catastrophic failures in the case, and there are some conservatisms built in (e.g. no heat transfer through insulated surfaces in the cryostat), the assumption of a fixed room temperature boundary for radiative heat transfer may not be justified, and a more thorough analysis should investigate this point further.

REFERENCES