Idaho National Engineering and Environmental Laboratory

Safety Related Design Issues

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ARIES Project Meeting, Wednesday, September 3rd, 2003

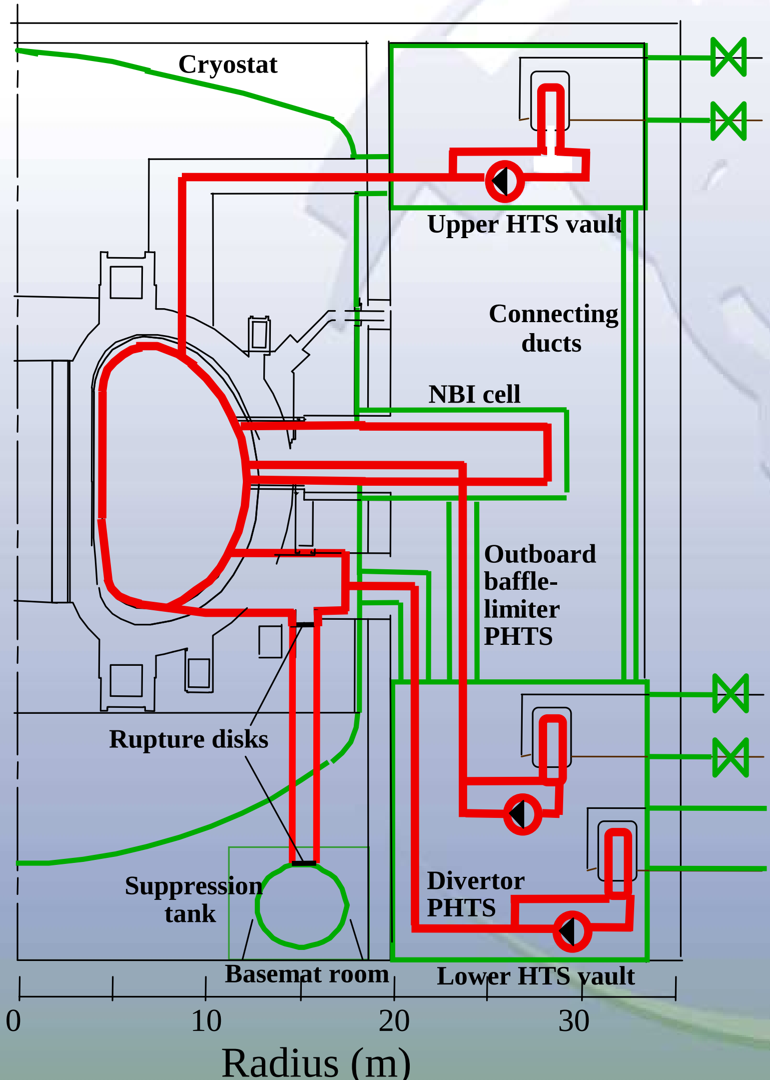
Safety Issues to be Addressed

- How many protection barriers are required on a reasonable safety basis?
- For a He-cooled ceramic breeder blanket with a He/steam heat exchanger (to drive a steam power cycle), a break in the steam generator (SG) pipe coupled with a break in the blanket coolant channel can lead to over-pressurization of the module and possible Be/steam reaction. Must the module be designed to take the steam pressure (with the penalty of thicker walls) or is it sufficient to assume that the coolant channels in the blanket are a sufficient barrier to take the pressure load?
- Is it acceptable to have a water-cooled shield in combination with a LiPb blanket?
- Safety issues associated with an external vacuum vessel and mitigating solutions. For example, for a liquid metal blanket, a rupture would lead to a spill of hot liquid which when touching the coils could lead to over-pressurization as He gets vaporized. Also, Brad has mentioned a possible concern with arcing of the coil.

How many protection barriers are required on a reasonable safety basis?

- The adopted ITER confinement concept is based on the safety concept termed Defense in Depth, in which multiple passive and active measures are used to confine radioactive inventories
- A passive measure is the use of physical confinement barriers (such as double walled vacuum vessel (VV), guard pipes, room walls, etc)
- An active measure is the use of safety systems to isolate or remedy a failure, such as isolation valves, detritiation systems, etc
- The number of barriers will depend on the hazard (tritium, structure activation products, FW/divertor erosion dust, coolant activation products, etc) of radiological and toxicological inventories, mobility of these inventories, and energy sources available to mobilize these inventories
- Simple answer: as many confinement barriers as are required to ensure that releases are well below site limits during Design Basis Accidents (DBA, frequency $> 10^{-6}$/yr)
- In addition, design features or mitigation measures must be in place to ensure that radiological site limits are not exceeded during Beyond Design Basis Accidents (BDBA, frequency $< 10^{-6}$/yr) to meet the no-evacuation requirement called out in the Fusion Safety Standard

Schematic of ITER Confinement Barriers



- Confinement of radioactive inventories by multiple barriers (defense in depth), *primary boundary*, *secondary boundary*
- Vacuum vessel (VV) is part of primary confinement boundary

How many protection barriers are required on a reasonable safety basis? (cont.)

- ITER EDA adopted two confinement barriers, a strong primary barrier (failure probability < 10^{-3} per challenge and a leak rate of 1%/day at a design limit pressure of 500 kPa) and a weak secondary barrier (failure probability < 10^{-1} per challenge and a leak rate of 20%/day at a design pressure limit of 200 kPa) to confine the tritium and activation products inside of the tokamak
- First barrier consisted of the outer wall of the VV, pumping ducts, pressure suppression system, and primary heat transport system (PHTS)
- Second barrier consisted of the PHTS vault and cryostat

How many protection barriers are required on a reasonable safety basis? (cont.)

- The 500 kPa design pressure for the ITER EDA primary barrier was established by several criterion one of which was the pressure produced by an in-vessel loss-of-coolant accident (LOCA)
- Peak pressure, even with pressure suppression system, ~400 kPa
- The re-design of the pressure suppression system for ITER FEAT, based on results from ITER safety group experiments and analyses, lowered this criterion to 200 kPa
- The secondary barrier design pressure of 200 kPa was also established by a number of criteria, one of which was the pressure produced by an ex-vessel LOCA into the PHTS vault (~150 kPa)
- Active air detritiation, stacked ventilation, and room ventilation isolation systems were adopted to minimize BDBA releases to meet no-evacuation limits

How many protection barriers are required on a reasonable safety basis? (cont.)

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Table 5.5.1-1 Reference Events Analysed in Detail

Plasma events
Loss of plasma control/exceptional plasma behaviour (i, a)
Loss of electrical power
Loss of off-site power for up to 1 h (i)
Loss of off-site power for up to 32 h (a)
Loss of off-site power and on-site class III power for up to 1 h (a)
In-vessel events
In-vessel first wall pipe or coolant channel leak (i)
Multiple first-wall pipe or coolant channel damage (a)
Loss of vacuum through a vacuum vessel penetration line (a)
Ex-vessel HTS events
Loss of heat sink in divertor HTS (i)
Pump trip/loss of flow in divertor HTS (i)
Pump seizure in divertor HTS (a)
Vacuum vessel HTS break (a)
Large ex-vessel divertor HTS break (a)
Heat exchanger leakage (i)
Heat exchanger tube rupture (a)
Maintenance events
Stuck divertor cassette in transport cask (a)
Maintenance accident on vacuum vessel (a)
Tritium plant and fuel cycle events
Tritium process line leakage (i)
Transport hydride bed mis-handling (a)
Isotope separation system failure (a)
Fuelling line with impaired confinement (a)
Magnet Events
TF short (a)
Magnet arc (a)
Cryostat Event
Air ingress (a)
Water/air/helium ingress (a)
Hot Cell Events
Failure of confinement (a)

(i) Incidents are deviations from normal operation, event sequences or conditions not planned but likely to occur during the life of the plant (see Table 5.2.3-2).

(a) Accidents are event sequences or conditions not likely to occur during the plant life but are postulated to demonstrate the safety of the plant (see Table 5.2.3-2).

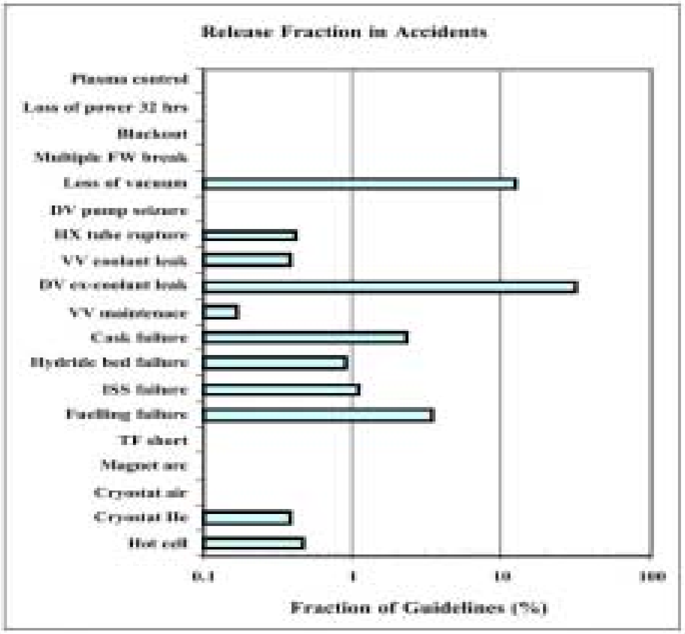


FIGURE VII.1-1
Margins against Project Environmental Release Guidelines for Reference Events. Shown is the Sum of Tritium, Activated Dust and Activation Corrosion Product Releases

- Bottom line – ITER met all operation and accident safety targets using these physical barriers, active air detritiation and room ventilation isolation systems

How many protection barriers are required on a reasonable safety basis? (cont.)

European Power Plant Conceptual Study (PPCS) Helium Cooled FW/Blanket (BL) Concept

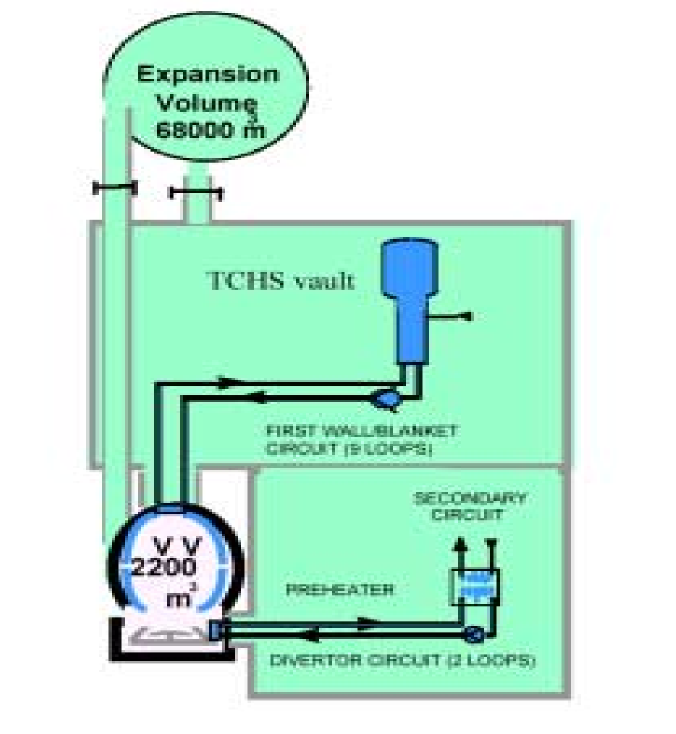


Fig. 1 – Model B Helium Cooled

- PHTS helium pressure (8 MPa) handled during LOCA in this design by subdividing FW/BL into nine toroidal cooling loops (segmented to minimize helium inventory), and using rupture disks to relieve pressure to an expansion volume
- How many confinement barriers exist for in-vessel radioactive inventories?
- Depends on accident scenario
  - In-vessel LOCAs – one barrier
  - Ex-vessel LOCAs – two barriers
- A single barrier represents a major issue for some accidents scenarios

How many protection barriers are required on a reasonable safety basis? (cont.)

How Well Does This Confinement Concept Work?

- ENEA Report, "Safety Analysis of the PPCS FW/BL Helium Cooled with the ATHENA Code," F. Mattioda, P. Meloni, M. Poli

  Analyzed pump seizure in a single loop, leading to FW melt by continued plasma power operation, two coolant channels fail producing an in-vessel LOCA, rupture disk to expansion volume (EV) opens, and flow into EV gives a final pressure of ~140 kPa (could be a DBA because these two failures, the pump seizure ~10^{-3}/yr plus plasma shutdown system failure ~10^{-3}/demand, gives ~ 10^{-6}/yr for this accident)

- ENEA Report, "ECART Analysis of an In-vessel Break in the First Wall of the Power Plant Conceptual Design Study," S. Paci

  Given the EV leak rate of 75%/day at the rated design pressure of 160 kPa (possibly a weak barrier), this accident results in the release of 255 g of T_2 over a one day time period

- While the EV did not fail from this LOCA, the ground release of 255 g of T_2 as HTO (U.S. assumption) would result in a site boundary dose of 195 mSv, which is in excess of the allowed 10 mSv

How many protection barriers are required on a reasonable safety basis? (cont.)

How Well Does This Confinement Concept Work?

- To be fair, S. Paci showed that allowing for uptake in EV walls reduced the T_2 release to 50 g => 39 mSv (an assumption not applicable to HTO), and down to 30 g if air detritation system (ADS) is activated => 23 mSv
- S. Paci also gave results for reduce EV leak rates of 1%/day and 10%/day
- This raises the question of what leak rate is an upper limit for a confinement building (adopted approach for fusion and next generation fission) ?
- Typical PWR fission containment buildings have leak rates of ~1.5%/day at 45-60 psig and a free volume of ~ 57,000 m^3
- For tritium handling buildings, DOE-HDBK-1129 ("Tritium Handling and Safe Storage", March 1999) states that a confinement room or building (small volumes) could leak at 5%/day
- So an EV that leaks at 1%/day would be similar to a containment building (a difficult goal), and even a 10%/day could be an aggressive design target for a volume the size of the EV
- Bottom line is that Helium may be more difficult than even H_2O

What Does This Mean to ARIES Compact Stellarator Blankets

- Base on limited resources we would like to rely on ITER results and analyze only a select few events, that is those accidents we think will be worst case accidents (as per ARIES-AT)
- Usually confinement bypass accidents (i.e. an accident that is postulated to fail the primary confinement barrier and bypass the remaining barriers) are those worst case accidents, which means ARIES-CS will need at least two confinement barriers to rely on ITER results
- A helium cooled blanket design represents a particularly difficult design to develop a safety case for, because a confinement strategy similar to that of the PPCS design means that an in-vessel LOCA (an anticipated event) could be a worst case accident due to only having a single confinement (weak or strong) barrier
- For a weak barrier, an in-vessel LOCA would fail this single confinement barrier with a frequency = LOCA frequency (single pipe break) x barrier failure frequency = 10^{-5}/yr x 10^{-1}/demand = 10^{-6}/yr (borderline DBA/BDBA event)
- This would mean that based on the PPCS results an in-vessel LOCA (DBA) could exceed the Fusion Safety Standard public dose limit at the site boundary of 10 mSv based on the the tritium release alone
- This is the primary reason low pressure or condensable coolant designs are much easier to develop a safety case for

Must a helium cooled module be designed to take the secondary cycle steam pressure?

- If the desired safety goal for a SG tube failure accident is low hydrogen generation then it must be demonstrated that the module fails in a manner that does not lead to the failure of the Be multiplying zone at the anticipated accident temperatures
- We may be dealing with a DBA since this accident would only require two failures, the steam generator (SG) tube break (frequency ~ 10^{-2}/yr) coupled with the failure of the pressure relief valve for the PHTS (frequency ~ 10^{-3}/demand) giving a frequency of 10^{-5}/yr for this accident
- But this accident could be put into a BDBA category by adding a redundant relief valve (or rupture disk) on the helium side of the SG
- In addition, the amount of Be in the largest blanket module of the European PPCS helium cooled FW/BL concept is ~870 kg, which translates into 195 kg of H_2 being produced if all of the Be reacts with the H_2O from the SG
- The hydrogen lower flammability limit in air (4%) is about 0.01 kg-H_2/m^3, which for the combined volume of the VV and EV of the PPCS design requires 700 kg of H_2 to achieve flammable concentrations
- However, to determine the consequences more accurately requires more design details. For example, will the resulting Be steam reaction produce enough heat to become self-sustaining or lead to the failure of adjacent modules?

Is it acceptable to have a water-cooled shield in combination with a LiPb blanket?

- Safety issue is the hydrogen and heat produced by the chemical reaction between the Li in the LiPb and any water that comes into contact with the LiPb

$$\mathrm{Li} + \mathrm{H_2O} \Rightarrow \mathrm{LiOH} + \frac{1}{2}\mathrm{H_2} + 204\,\mathrm{kJ}\,;\mathrm{T} < 450\,^{\circ}\mathrm{C}$$

$$2\,\mathrm{Li} + \mathrm{H_2O} \Rightarrow \mathrm{Li_2O} + \mathrm{H_2} + 186\,\mathrm{kJ}\;\;;\mathrm{T} > 450\,^{\circ}\mathrm{C}$$

- Experiments have shown that the amount of hydrogen generated is strongly dependent on the contact mode
  - Injection – pressurized injection of water into liquid metal (LM)
  - Pouring – pouring of LM into water
  - Layered – pouring of water onto LM
  - Pool – steam environment over LM pool
  - Spray – steam environment present during LM spray
- For the layered, pool, and spray contact modes the reaction is self-limiting by the formation of solid LiOH and Li_2O that shields the LM from the water/steam interface, but other contact modes may be a problem

Is it acceptable to have a water-cooled shield in combination with a LiPb blanket? (cont.)

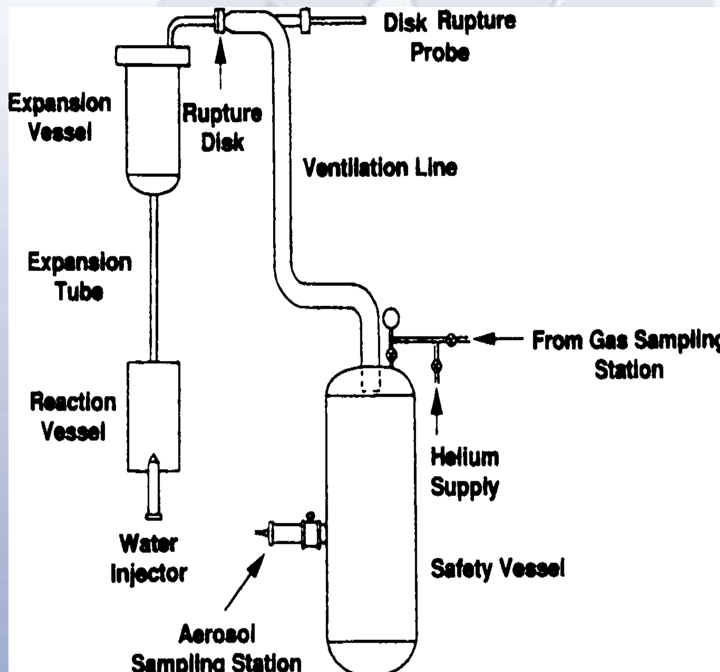
Possible accident scenario of concern

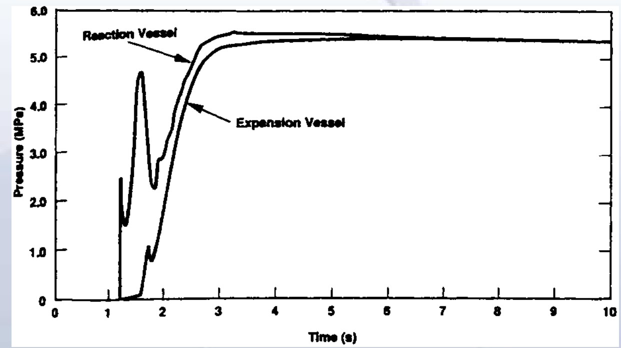
- A divertor tube breaks, jetting water into plasma causing a rapid bootstrap current quench
- The combination of induced eddy currents in the blanket and the water jet quenching the FW fails the blanket
- LiPb pours into the VV and forms a pool at the bottom of the VV covering the failed divertor
- The water jet continues under the pool, jetting water though pool and into the free space in the VV

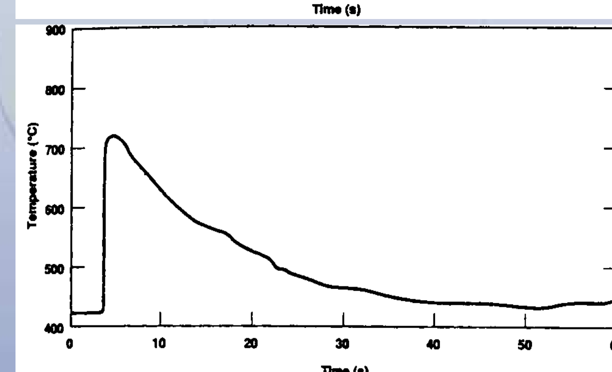
An experiment with a similar configuration to this postulated accident was performed at the BLAST Facility at Ispra by D. W. Jeppson and C. Savatteri

Is it acceptable to have a water-cooled shield in combination with a LiPb blanket? (cont.)

D. W. Jeppson and C. Savatteri BLAST Experiment (1991)







- Based on mass spec analysis, ~8% of the lithium in the LM reacted to form H_2
- Given the volume of LM associated with one quadrant of ARIES-AT (~140 m^3 containing 9000 kg Li), the amount of H_2 generated by this divertor pipe break scenario could be as high as ~210 kg, with the flammability limit for a 1000 m^3 VV at ~ 10 kg

Is it acceptable to have a water-cooled shield in combination with a LiPb blanket? (cont.)

- Even higher Li oxidation fractions can occur for more violent injection scenarios, as evidenced by University of Wisconsin Experiments conducted by L. S. Nelson (UWFDM-1031, 1996)

Figure 3. Photographs of the vertical water column apparatus.
Figure 3a. Overall view of the apparatus. (B-59-3)

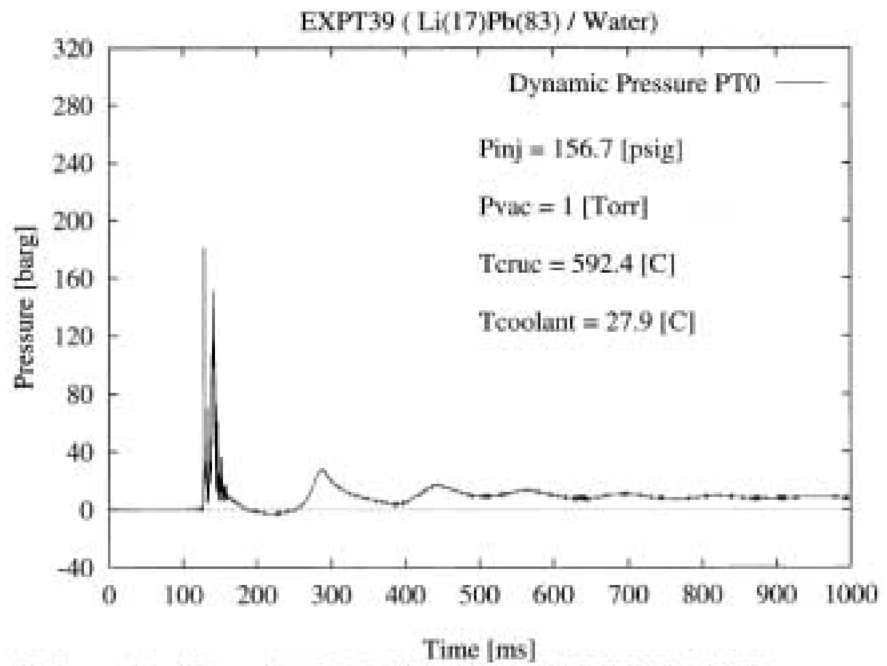


Figure 15a. Instrumental records from experiment B-39-1. Crucible contained molten 83a/o Pb-17a/oLi alloy at 592.4°C, water was at 27.9°C. Transducer PT0, 1000 ms trace.

Is it acceptable to have a water-cooled shield in combination with a LiPb blanket? Maybe Not

- Summary table from L. S. Nelson (UWFDM-1031)

Table 3. Comparisons of Hydrogen Generated and Lithium Removed During Three Different Sets of Experiments in Which Molten 83a/o Pb-17a/o Li at 600°C Interacted with Liquid Water

Experiment No.		T_{H_2O} (°C)	Wt. of Alloy (g)	No. mmoles of Li	H_2 Generated (mmole) Δp	ms	OH^-	H_2 Generated/Wt. Alloy (mmole/g) Δp	ms	OH^-	Lithium Removed (%) Δp	ms	OH^-
Biney (1995)	L11	60	44.0	44.37	8.05	----	----	0.183	----	----	36.3	----	----
	L12	60	38.0	38.32	10.7	----	----	0.282	----	----	55.8	----	----
	L14	60	49.0	49.42	12.6	----	----	0.257	----	----	51.0	----	----
	L21	60	41.0	41.35	8.76	----	----	0.214	----	----	42.4	----	----
	L22	60	35.0	35.30	11.1	----	----	0.317	----	----	62.9	----	----
					10.24 ±1.837			0.251 ±±0.053			49.7 ±10.6		
Herzog (1987)	16	60	47.0	47.40	7.01	----	----	0.149	----	----	29.58	----	----
	17	60	56.3	56.78	6.48	----	----	0.115	----	----	22.82	----	----
	20	90	39.8	40.14	7.47	----	----	0.188	----	----	37.22	----	----
	21	70	67.9	68.48	6.97	----	----	0.103	----	----	20.36	----	----
	22	80	51.3	51.74	4.84	----	----	0.094	----	----	18.71	----	----
	29	90	25.9	26.12	6.94	----	----	0.268	----	----	53.14	----	----
	30	70	19.0	19.16	6.12	----	----	0.322	----	----	63.88	----	----
	42	60	35.5	35.80	8.68	----	----	0.245	----	----	48.49	----	----
	43	60	32.8	33.08	6.40	----	----	0.195	----	----	38.69	----	----
					6.768 ±1.039			0.187 ±0.079			36.988 ±15.754		
This Work	B-39-1	27.9	119.57	120.6	----	33.7	45.3	----	0.282	0.379	----	55.9	75.0
	B-50-1	27.4	116.59	117.6	----	42.0	42.9	----	0.360	0.368	----	71.5	73.0
	B-45-1	61.0	115.91	116.9	----	29.3	37.6	----	0.253	0.324	----	50.1	64.4
	B-52-1	59.6	122.42	123.5	----	34.4	45.5	----	0.281	0.372	----	55.8	73.7
						34.85 ±5.27	42.83 ±3.68		0.294 ±0.046	0.361 ±0.025		58.33 ±9.19	71.45 ±4.85

Experimental Techniques: Δ p = pressure measurements; ms = quadrupole mass spectrometry; and OH^- = acid-base titration of hydroxyl ion.

Safety issues associated with an external vacuum vessel and mitigating solutions.

- For a LM blanket, a rupture would lead to a spill of hot liquid which when touching the coils could lead to over-pressurization as He gets vaporized
  - Siegfried suggests possibly draining LM from VV, but an analysis would have to be performed to demonstrate that the LM doesn't freeze or that the coils don't quench
- Also, for LM blanket a possible concern is with the arcing of the coil if coil quenches
- Regardless of the blanket concept, decay heat removal will be an issue because of the continuous 'cold' coil support structure which like the coils is insulated to protect it from hot blanket structures. This insulation will also limit radial heat conduction to ambient
  - Possible solution – a natural convection system similar to ARIES-RS decay heat removal system. Coolant choice is a problem.

Summary

- We should try for at least two confinement barriers, an easily achievable goal for low vapor pressure coolants
- If the module can not be designed to accommodate the secondary steam pressure in a helium cooled blanket design, then at least the region containing the beryllium should be designed to withstand this pressure and prevent propagation of other module failures
- Large quantities of hydrogen could be generated for a LiPb blanket in combination with a water cooled shield, but additional investigation of this issue should be undertaken because some authors do not seem to come to this same conclusion

  D. A. Petti, B. J. Merrill, …"Safety and Environment Assessment of ARIES-AT", 2000

  L. Giancarli, et al, "Status of European breeding blanket technology", FED, 36,1997
- A passive decay heat removal system may be required for ARIES-CS because of the insulation needed for the magnet cold support structure