Final Nuclear Analysis for ARIES-AT

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Web address:
http://fti.neep.wisc.edu/FTI/ARIES/JUN2000/final_lae.pdf

ARIES Project Meeting
19-21 June 2000
UW - Madison

Major Conclusions

Neutronics:

- Blanket satisfies breeding requirement (TBR ≥ 1.1) with minor adjustment to accommodate W shells
- Blanket segmented to lower replacement cost, reduce volume of waste, and increase repository capacity
- 4 FPY service lifetime for inner blanket and divertor system and 40 FPY for all other components
- 1.1 overall energy multiplication

Shielding:

- Shield and V.V. are well optimized for design constraints
- Compact radial build compared to previous ARIES designs
- Well protected magnets for 40 FPY
- Need higher reweldability limit for FS, relocate V.V. welds in low radiation zones, or avoid welding in front 15 cm of V.V.
- Remotely maintained components; no hands-on maintenance

Activation:

- Low activation materials for all components ⇒ low level waste (SiC, FS with impurity control, and LiPb with Bi purification system)
- Design solutions developed to reduce decay heat and mitigate effect during accident
- Relatively small volume of waste compared to previous ARIES designs
- Because of compactness, all components have clearance index > 1
  ⇒ no "Free Release" of metals even after 100 y storage period
- Waste management options:
  - Recycle and reuse in nuclear facilities (to increase repository capacity)
  - Dispose near surface as low level waste:
    - All components easily qualify as Class C LLW
    - Most components qualify as Class A LLW
    - Waste classification: 90% Class A and 10% Class C

Design Parameters[#] and Radiation Limits

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Fusion power	1719 MW
FW location at midplane – OB , IB	6.55 , 3.85 m
at top/bottom – OB , IB	4.7 , 3.85 m
Γ : Peak OB , IB , div.	4.75 , 3.1 , 2 MW/m^2
Average OB , IB , div.	4.0 , 2.2 , 1 MW/m^2
Machine lifetime	40 FPY
Availability	75%
SiC burnup limit	3% (1.5 atom% He[*])
FS dpa limit	200 dpa
Steel reweldability limit	1 He appm for FS 5 - 30 He appm for 316SS[**]
HT magnet fluence limit	10^{19} n/cm^2 (E_n> 0.1 MeV)

[#] 3/10/2000 Strawman
[*] up to 2 atom% He is acceptable, per R. Jones
[**] Ref.: W. Daenner (Germany), 1999 EU experimental results

Final Nuclear Parameters*

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- Key features of FW/Blanket:
  - 1.5 cm thick FW: 68% SiC, 32% LiPb
  - IB and OB blankets only (no blanket behind divertor):
    - 30 cm thick IB FW/blanket
    - 65 cm thick OB FW/blanket segmented into:
      - 30 cm FW/Blanket –I (replaceable)
      - 35 cm Blanket-II (permanent)
  - 90% enriched LiPb @ 770 °C
  - Penetrations:
    - 2.5 m^2 on OB for plasma control
    - 2 cm wide radial gaps between 16 blanket modules

- Nuclear parameters:

Overall TBR	1.11 w/o shells
	1.07 w/ 4 cm thick W shells**
Overall M_n	1.1
SiC Burnup rate	0.77% per FPY#
FW EOL Fluence	18.5 MWy/m^2
FW Lifetime	4 FPY

Thicken OB blanket-II by 10 cm to meet breeding requirement (1.1) in case of 4 cm thick W vertical stabilizing shells

- Comments:
  - Lower SiC content increases breeding
  - Thicker OB blanket (> 45 cm) does not increase breeding
  - Higher enrichment (> 90%) is expensive and has insignificant impact on breeding
  - More penetrations/gaps reduce breeding

Breeding level is marginal ⇒ reduce SiC in FW/blanket below 20%

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* Using FENDL-2 cross section data library

** Vertical stabilizing shells

\# 0.54% Si , 0.23% C

Nuclear Heat Load to In-vessel Components

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(P_f= 1719 MW , P_n= 1375 MW)

Nuclear Heating (MW)	Inboard	Outboard	Divertor**	Total	
FW or DP	41	100	39#	**180**	(12%)
Blanket:				**1150**	(76%)
B-I (28.5 cm)	280	710	---		
B-II (35 cm)	---	135	---		
W Shells (4 cm)	---	20	---		
16 Wedges*	---	5	---		
HT Shield/W Shells	50 / 4	20	113##	**187**	(12%)
Total	**375**	**990**	**152**	**1517**	
	(25%)	(65%)	(10%)		

Overall neutron energy multiplication is 1.1

Low Grade Heat:

Vacuum Vessel (MW)	7	3	1	11*** (< 1% of total htg)
Magnet (kW)	0.6	48	0.3	~50###

** upper and lower divertor regions
# 25 MW in dome, 8 MW in outer divertor plates, 6 MW in inner divertor plates
* 3% of B-II
58 MW in replaceable shield, 27 MW in vertical shield, 28 MW in IB shield above/below X point
*** does not include thermal heat leak from HT shield (5-10 MW)
requires 0.5 MW of cryogenic load @ 10 W/W

Components' Lifetimes

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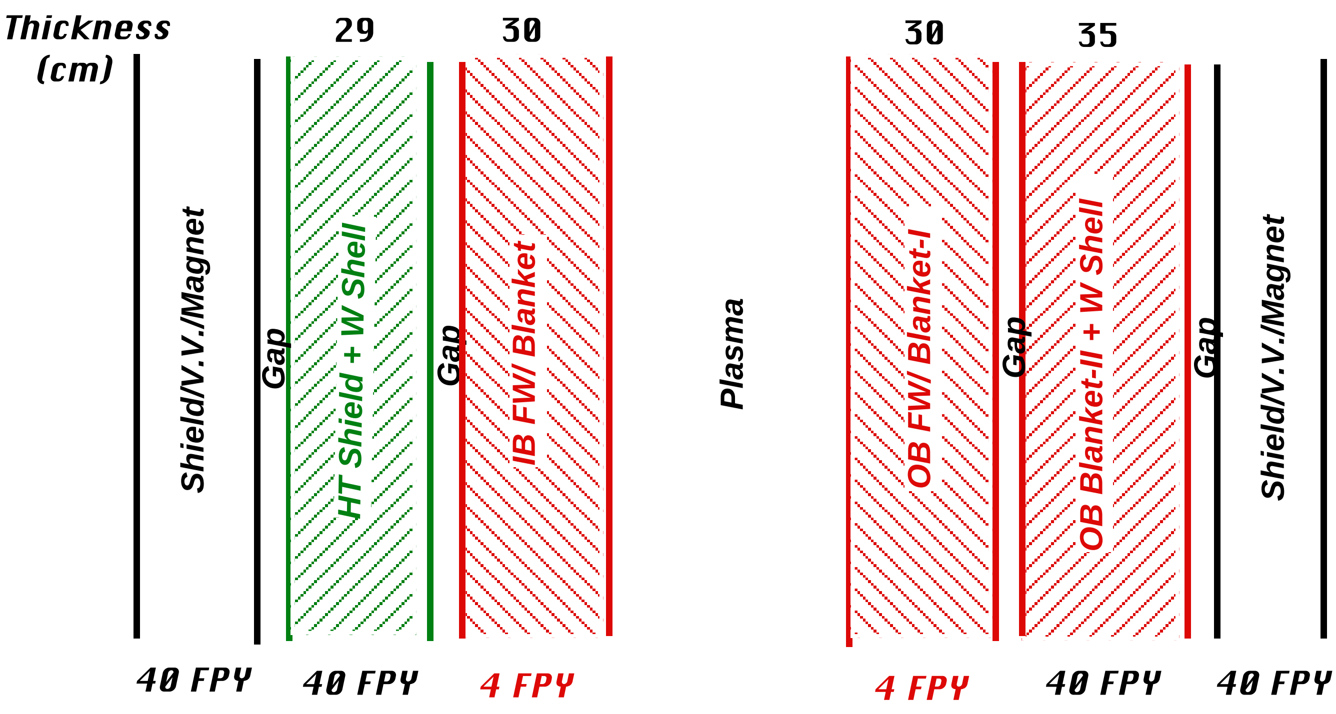
- Service lifetimes are based on:
  - – 3% burnup limit for SiC structure of FW, blanket, HT shield
  - – 200 dpa limit for FS structure of V.V.
  - – 10^{19} n/cm^2 fast n fluence to YBCO conductor of HT magnet



Inboard Radial Build*

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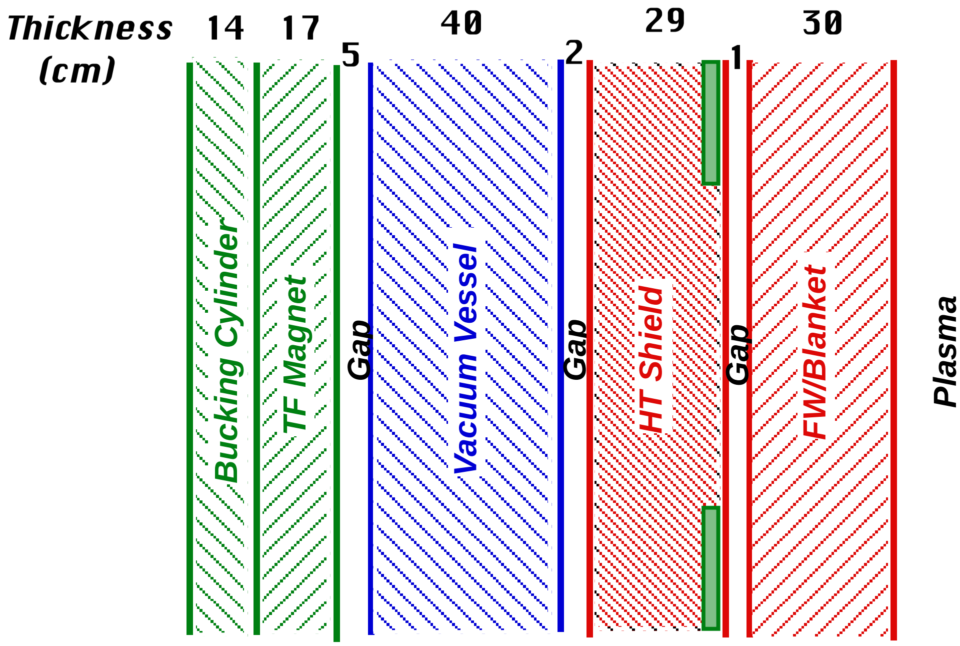

Component	Composition#
FW (1.5 cm)	68% SiC , 32% LiPb
Blanket (28.5 cm)	18% SiC , 82% LiPb
HT Shield	14% SiC, 10% LiPb , 72% B-FS, 4% W
Vacuum Vessel	13% FS, 22% H_2O, 65% WC
HT Magnet	87% SS, 10% LN, 2.5% $Y_1Ba_2Cu_3O_5$, 0.5% Ag
Bucking cylinder	95% SS, 5% LN

- 29 cm thick HT shield helps reduce heat leakage and V.V. decay heat
- FS 1 He appm reweldability limit is NOT met at outer surface of V.V. (10 He appm, 46 H appm, 20 dpa)
- Could higher reweldability limit for 316SS (5-30 He appm) apply to FS? If not, locate cut/weld areas away from high radiation zones
- Magnet composition does not contain CeO_2 insulator nor impurities

* Safety factor of 3 considered in all shielding calculations
# SiC and WC are 95% dense

Outboard Radial Build* without Shells

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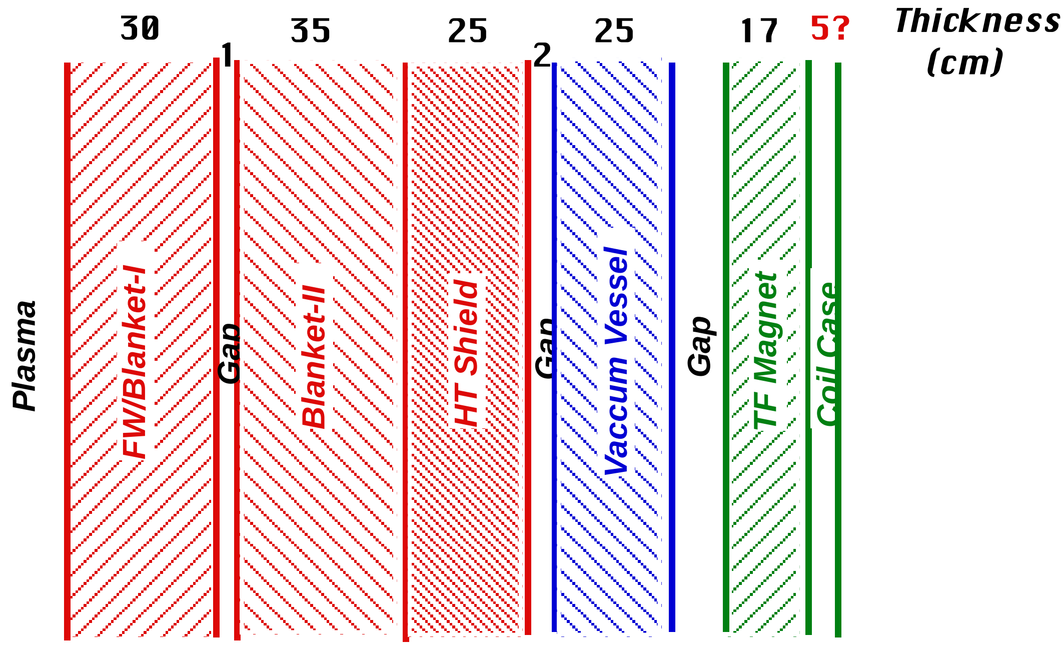

Component	Composition#
FW/Blanket-I:	
FW (1.5 cm)	68% SiC , 32% LiPb
B-I (28.5 cm)	18% SiC , 82% LiPb
Blanket-II	20% SiC , 80% LiPb
HT Shield	15% SiC , 10% LiPb , 75% B-FS
Vacuum Vessel**	30% FS , 70% H_2O
HT Magnet	87% SS, 10% LN, 2.5% $Y_1Ba_2Cu_3O_5$, 0.5% Ag
Coil Case	95% SS, 5% LN

- Along with blanket/shield/V.V., 10 cm thick port enclosures provide shielding for sides of magnets
- Wedges underneath magnets must be carefully designed to protect magnets
- FS 1 He appm reweldability limit is met at inner surface of V.V. (1 He appm, 5 H appm, 3 dpa)
- Is 5 cm thick outer coil case acceptable?

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* Safety factor of 3 considered in all shielding calculations
# SiC and WC are 95% dense
** Composition is slightly of-optimum to simplify port design

Outboard Radial Build*
With 4 cm W Vertical Stabilizing Shell

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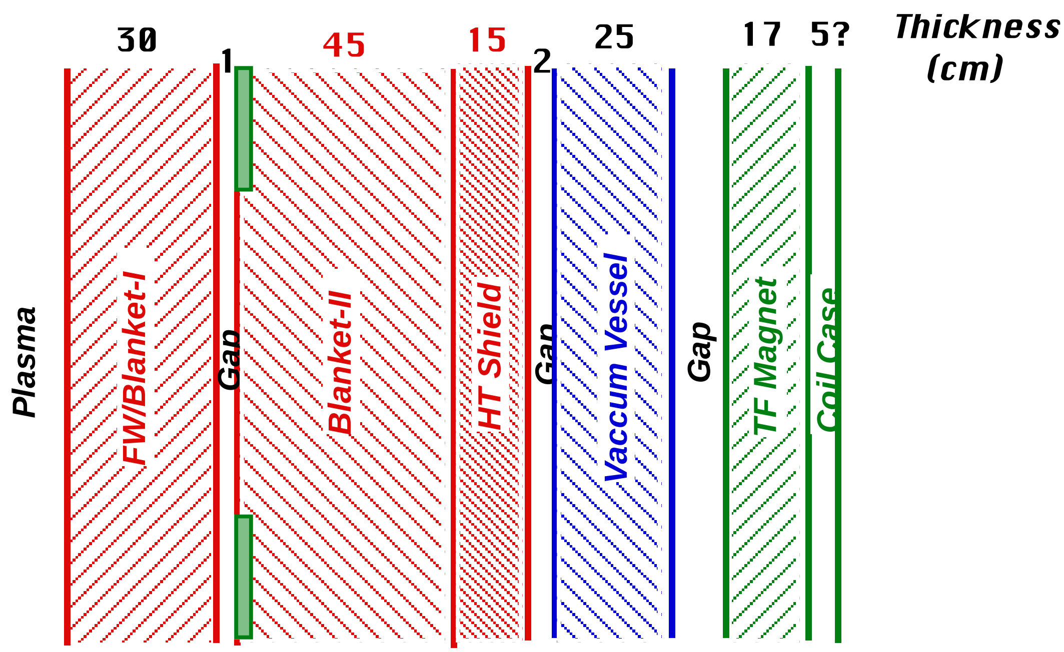

Component	**Composition#**
FW/Blanket-I:	
FW (1.5 cm)	68% SiC , 32% LiPb
B-I (28.5 cm)	18% SiC , 82% LiPb
Blanket-II	19% SiC , 78% LiPb , 3% W
HT Shield	15% SiC , 10% LiPb , 75% B-FS
Vacuum Vessel	30% FS , 70% H_2O
HT Magnet	87% SS, 10% LN, 2.5% $Y_1Ba_2Cu_3O_5$, 0.5% Ag
Coil Case	95% SS, 5% LN

- To meet breeding requirement, OB B-II should be 45 cm thick, trading shield for blanket
- Total OB radial standoff remains fixed

* Safety factor of 3 considered in all shielding calculations
\# SiC and WC are 95% dense

Vertical Build[*]

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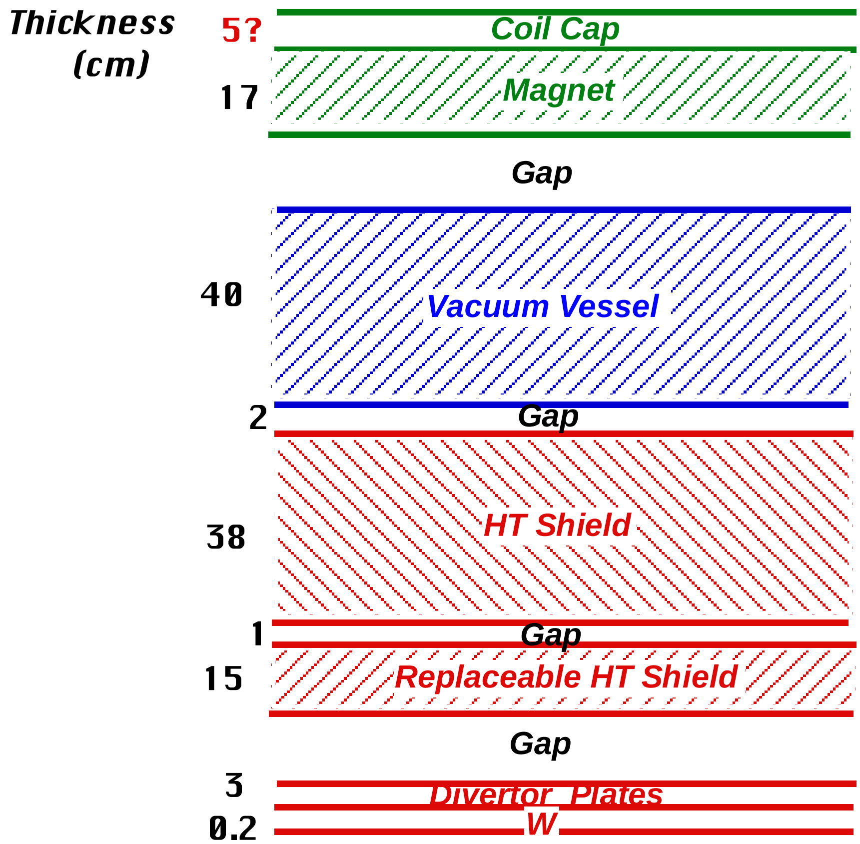

Component	Composition[#]
W coating	100 W-0.2%TiC alloy
Divertor Plates	46% SiC , 54% LiPb
Replaceable HT Shield	15% SiC , 10% LiPb, 75% FS
HT Shield	15% SiC , 10% LiPb , 75% B-FS
Vacuum Vessel	13% FS, 22% H_2O, 65% B-FS
HT Magnet	87% SS, 10% LN, 2.5% $Y_1Ba_2Cu_3O_5$, 0.5% Ag
Coil Case	95% SS, 5% LN

- FS reweldability limit (1 He appm) is NOT met at front of V.V.
  ⇒ Locate cut/weld areas away from high radiation zones or adopt higher limits
- 15 cm thick local shield must be provided behind divertor pumping ducts. Cool it with LN to act as heat sink during LOCA/LOFA
- No shielding problem to inner legs of magnets behind inner divertor plates

[*] Safety factor of 3 considered in all shielding calculations
[#] SiC and WC are 95% dense

IB V.V. Optimum Composition

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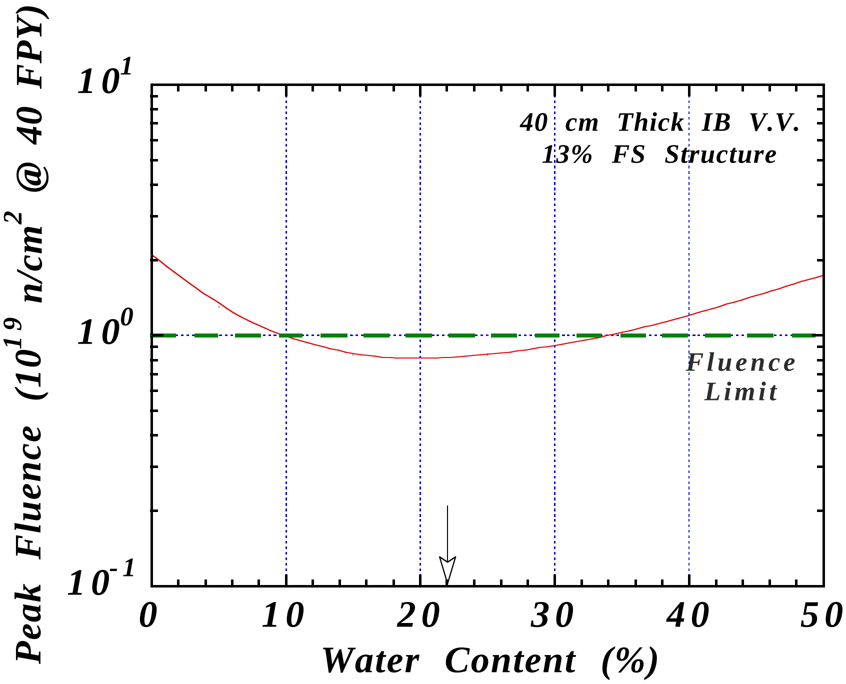

- 13% FS structural content dictated by structural requirements, per L. Waganer
- V.V. composition optimized by trading WC filler for water
- Optimal VV composition for minimum fluence is:
  13% FS structure, 22% H_2O, and 65% WC filler

Activation Analysis

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- Codes, parameters, and assumptions:
  - Activation: ALARA code; FENDL-2 activation library
  - Flux: 1-D DANTSYS code; FENDL-2 Xn data
  - 175 n and 42 g group structure
  - 3-D neutron flux used to re-normalize 1-D flux for all components
  - Average OB and IB Γ are 4 and 2.2 MW/m^2, respectively (20-30% lower Γ compared to previous strawman)
  - IB and OB sides as defined by radial builds (29 cm thick IB HT shield helps reduce V.V. decay heat)
  - 4 cm W vertical stabilizing shells embedded in IB HT shield and in 35 cm OB B-II
  - Operation time: 4 FPY for FW/B-I and 40 FPY for all other components
  - Latest magnet composition with impurities not included
  - 75% system availability included in analysis

- Reported results are for:
  - IB and OB at end of service life:
    - Activity
    - Decay heat
    - Radwaste level
  - 100% dense compacted waste (coolants and void excluded)
  - SiC, WC, W, and LiPb with impurities
  - FS with impurity control: 1 wppm Nb, 20 wppm Mo

- Activation results are posted on Wilson's web site: http://marie.ep.wisc.edu/~wilson/ALARA/aries_at_results/

Activity

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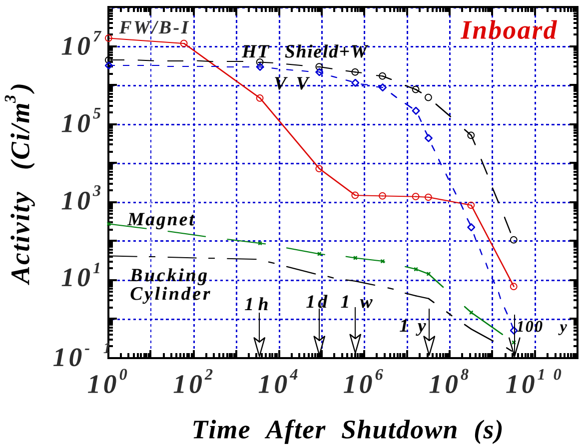

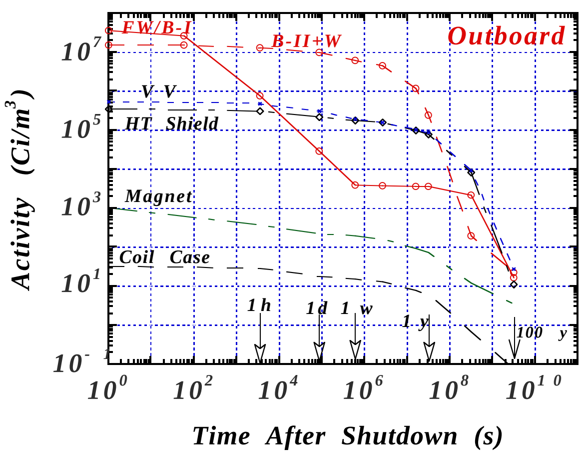

- Unlike metals, SiC activity drops by several orders of magnitude shortly after shutdown

- Highly irradiated SiC components generate lower intermediate activity (1d-5y) than well protected FS and WC components

Decay Heat

(Coolants Excluded)

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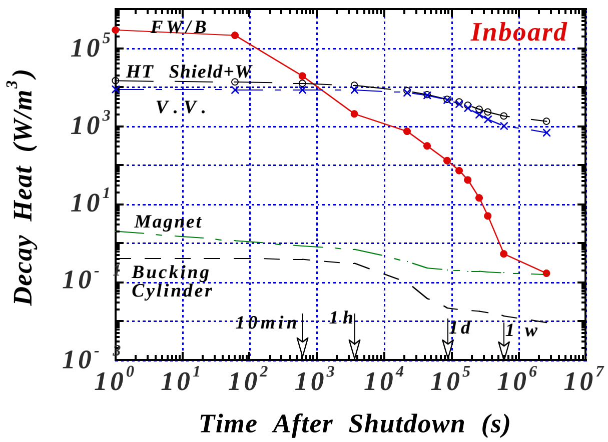

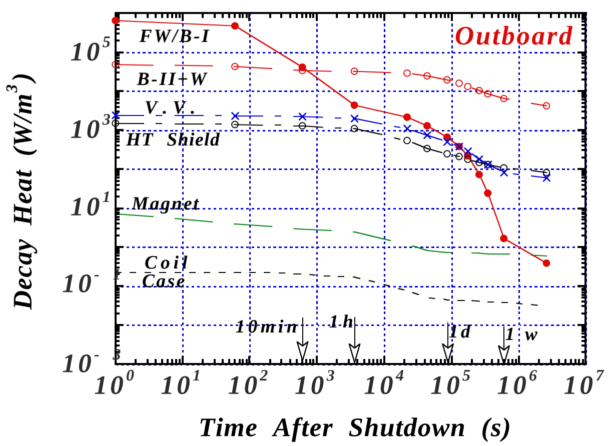

- Unlike metals, SiC decay heat drops fast after one minute, meaning slight increase in temperature of SiC components during LOCA/LOFA

- Detailed decay heat for individual constituents of all components (including coolants) provided for LOCA/LOFA analysis

Impact of 75% Availability on Decay Heat

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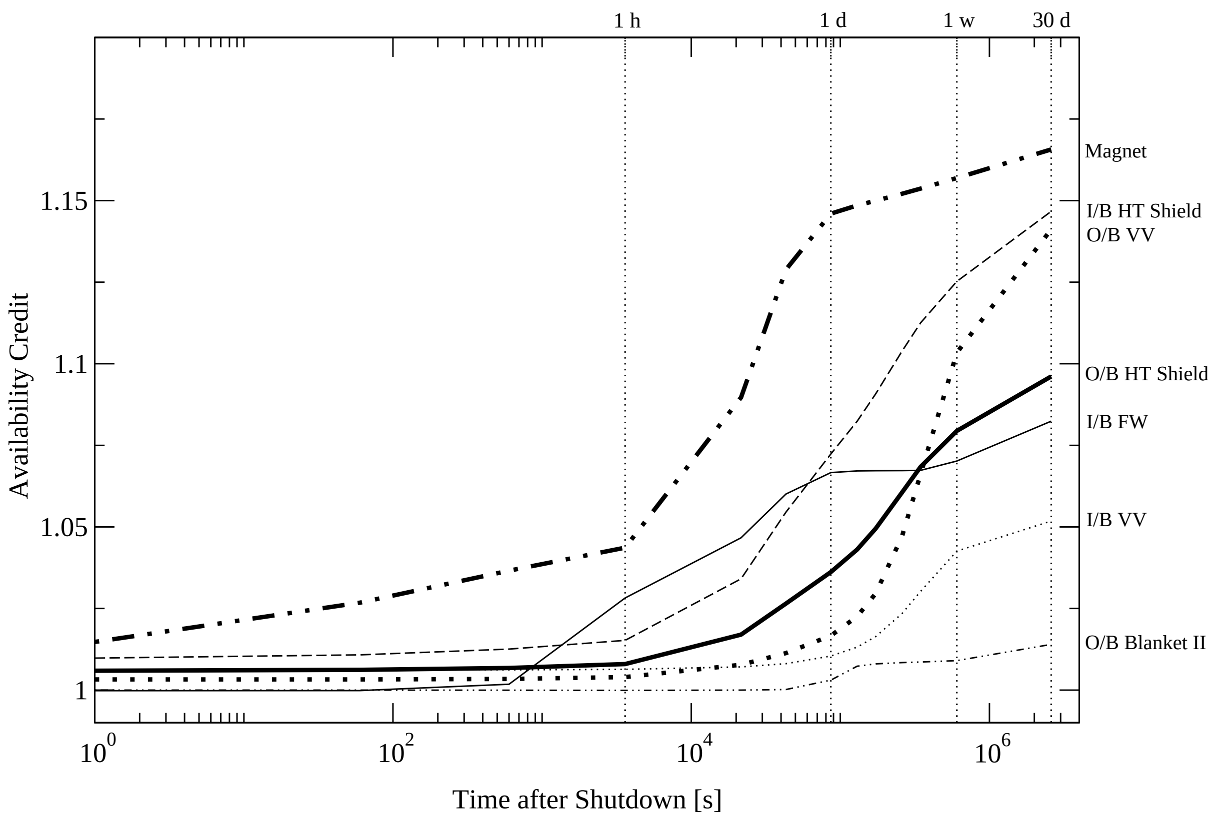

- Availability credit = decay heat with 100% availability / decay heat with 75% availability

- Assumption: 9 months of operation and 3 months of down time

- Results for 1h-1w time period:
  - 1-7% effect on SiC components
  - 1-12% effect on FS filler and structure
  - 5-16% effect on magnet

System availability will be included in future activation analysis

LiPb Decay Heat for LOFA Analysis

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- Continuous irradiation overestimates decay heat of flowing LiPb by more than factor of 10 ⇒ consider LiPb residence time in torus and in outer loop

- Assumptions:
  - Same LiPb used for 40 FPY (Li can be refurbished if needed)
  - On-line removal of all tritium generated
  - Per René, LiPb residence times for in-vessel components are:
    - 1 s in IB FW
    - 2 s in OB FW
    - 3 s in DP
    - 35 s in channel of IB Blanket
    - 70 s in channel of OB Blanket-I
    - 240 s in channel of OB Blanket-II
    - 10 s in side and back walls of blankets
    - 60 s in HT shield
  - LiPb spends ~2 min in outer loop for heat recovery, T extraction, and Po/Bi/Hg purification
  - LiPb returns to same location inside torus (conservative results)
  - 75% system availability

- If LOFA temperature is excessive, less conservative assumptions will be considered to reduce LiPb decay heat

LiPb Decay Heat for LOFA Analysis (cont.)

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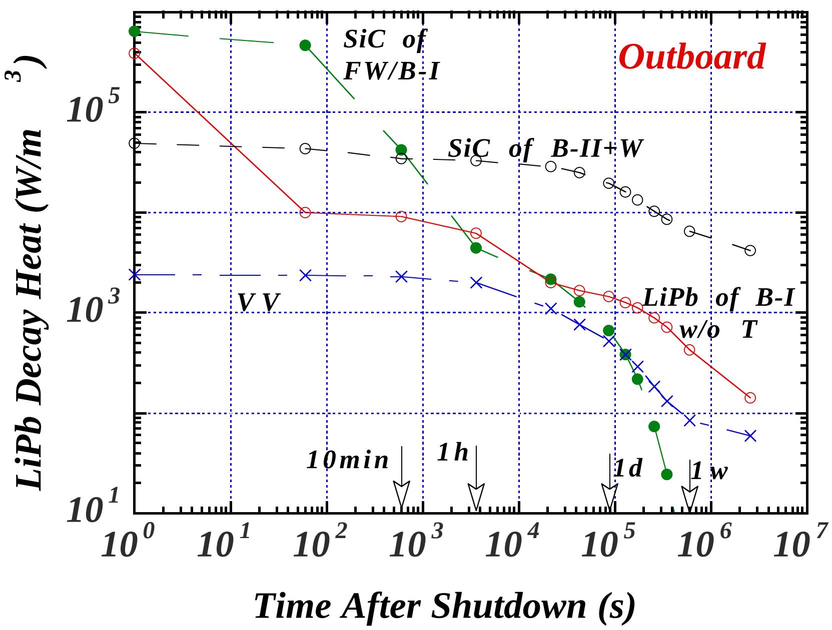

- After one hour, LiPb of Blanket-I generates higher decay heat than SiC

In LiPb/SiC system, LOFA is more critical than LOCA

LiPb Purification System

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- Bi and Po inventories continue to rise during operation

- For safety reasons, Bi and Po should be controlled during operation

- Pb and Bi impurity (43 wppm) generate 90% and 10% of Bi^{208}, respectively, via the following reactions:
  Pb^{208} (n,γ) Pb^{209} (β^- decay) Bi^{209} (n,2n) Bi^{208}
  Bi^{209} (n,2n) Bi^{208}

- Bi generates Po^{210}:
  Bi^{208} (n,γ) Bi^{209} (n,γ) Bi^{210} (β^- decay) Po^{210}

- Purification system should be designed to keep average Bi^{208} and Po^{210} inventories below permissible level (~25 Ci)

Waste Management

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- **Options:**
  - – Clear or “free release” of materials to industrial facilities
  - – Dispose near surface as Class A or C low level waste (LLW)
  - – Recycle waste and reuse in nuclear facilities

- Clearance and disposal options have been addressed in details in ARIES-AT study

- Waste could be recycled at a cost (unknown). No effort devoted to recycling of ARIES-AT waste

Clearance*

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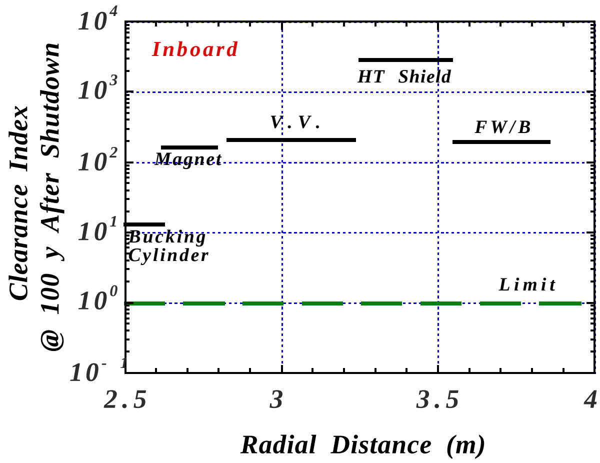

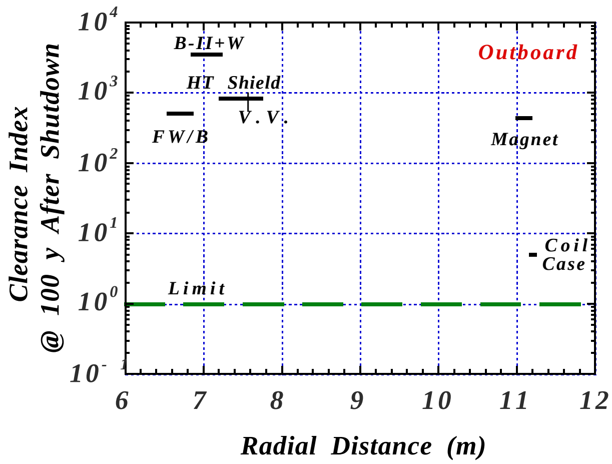

- In absence of national NRC standards, IAEA clearance limits are used
- All ARIES-AT components have clearance index > 1 based on IAEA clearance limits
  ⇒ **Components/constituents cannot be released as cleared metals**
- At present, US industry has no tolerance for slightly contaminated materials
  ⇒ No market for cleared metals
- NRC limits could be more restrictive than IAEA's (dose ~1 mrem/y)

ARIES-AT waste will be disposed near-surface as Class A or C LLW or could be recycled

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* Defined as unrestricted release of items and materials from radiologically controlled areas

Waste Disposal Rating

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- WDR reported for compacted waste (void excluded)

- WDR < 1 means component qualifies as LLW

- WDR remains constant for 100's of years after shutdown, unless indicated

- All components should meet BOTH Fetter's and NRC-10CFR61 WD limits for Class A or C LLW

- Waste disposal limits:
  - NRC (10CFR61):
    - **Official** U.S. WD limits
    - NRC has developed Class A and Class C WD limits for 9-10 isotopes beside actinides.
    - NRC limits not available for ~90 isotopes of interest to fusion
    - Class A has low limit for tritium ($T_{1/2}$~12.3 y)
  - Fetter's:
    - **Not** in regulations form
    - Approved by U.S. Fusion Safety Standing Committee
    - NRC has not endorsed Fetter's limits
    - No limits available for Class A LLW
    - Fetter developed Class C WD limits for 101 isotopes of interest to fusion. 19 isotopes have range of limits rather than single value due to uncertainties in corrosion assumptions. Those beta emitters are: C^{14}, Si^{32}, Cl^{36}, Ca^{41}, Ni^{63}, Se^{79}, Sr^{90}, Tc^{97}, Tc^{98}, Tc^{99}, Pd^{107}, I^{129}, Sm^{151}, Gd^{148}, Gd^{150}, Dy^{154}, Pb^{210}, Ra^{226}, Ac^{227}
    - Fetter-L and Fetter-H WDRs are calculated using Fetter's low and high limits, respectively.
    - Fetter-L limits were not considered in previous ARIES designs
    - Which limit should we continue to report, Fetter-H or Fetter-L? (Fetter-L is more conservative)

Fetter's Waste Disposal Rating

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	Fetter-H Class C Limits w/o Shells	Fetter-H Class C Limits with W Shells*	Fetter-L Class C Limits w/o Shells	Fetter-L Class C Limits with W Shells
Inboard Components:				
FW/B	0.015		0.017	
HT Shield	0.45	0.47	0.68	0.73
V.V.	0.03		0.06	
Magnet	0.015		0.024	
Bucking Cylinder	0.005		0.01	
Outboard Components:				
FW/B-I	0.072		0.076	
B-II	0.001	0.18	0.01	0.35
HT Shield	0.15		0.23	
V.V.	0.03		0.04	
Magnet	0.03		0.04	

- Al^{26} is dominant nuclide for Fetter's WDR of SiC components
  Si^{28} (n , np) Al^{27} (n , 2n) Al^{26}

Based on Fetter's limits, all components qualify as Class C LLW @ EOL

* 4 cm thick vertical stabilizing shell

NRC Waste Disposal Rating

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	NRC Class A Limits w/o Shells	NRC Class A Limits with W Shells	NRC Class C Limits w/o Shells	NRC Class C Limits with W Shells
Inboard Components:				
FW/B	3.8		0.017	
HT Shield	63	61	0.39	0.41
V.V.	0.08		0.006	
Magnet	0.08		0.007	
Bucking Cylinder	0.05		0.004	
Outboard Components:				
FW/B-I	10		0.03	
B-II	1.3	3.1	0.07	0.23
HT Shield	4.7		0.1	
V.V.	1.9		0.02	
Magnet	0.3		0.009	

- NRC-A WDR reported at shutdown and varies with time after shutdown
- For SiC components, T and C^{14} are dominant nuclides for NRC-A WDR and C^{14} is dominant nuclide for NRC-C WDR

$$C^{12}\,(n,\gamma)\;C^{13}\;(n,\gamma)\;C^{14}$$

Based on NRC limits, all components qualify as Class C LLW @ EOL

NRC-A Waste Disposal Rating

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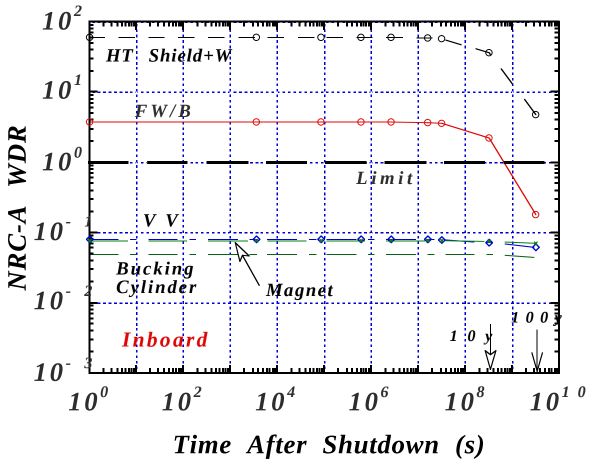

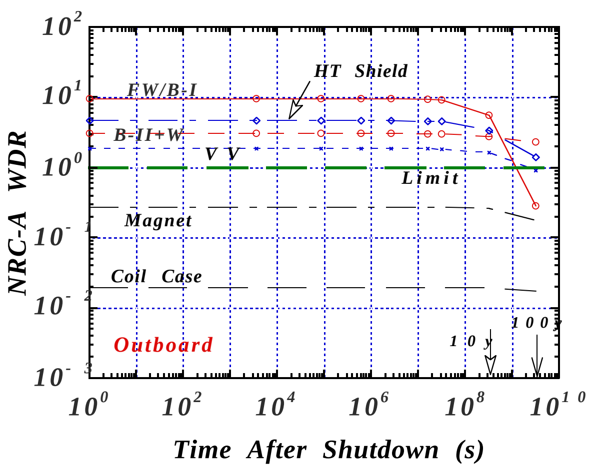

- Thicker OB blanket (45 cm) will reduce OB HT shield WDR by factor > 2

Based on NRC limits, all IB & OB components
except IB HT shield and OB B-II
qualify as Class A LLW after 50-100 y storage period

IB & OB Compacted Waste Volumes*

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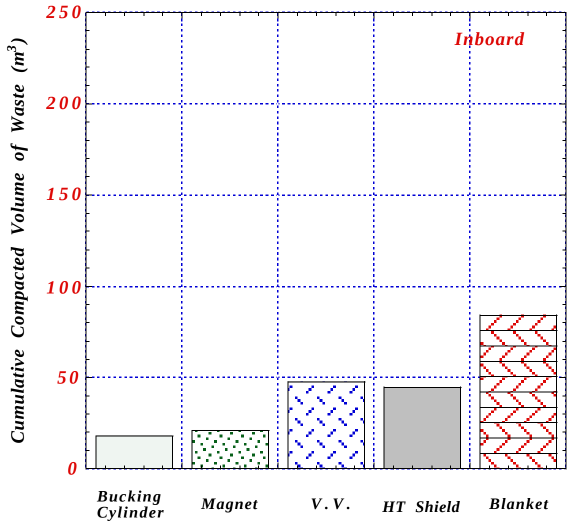

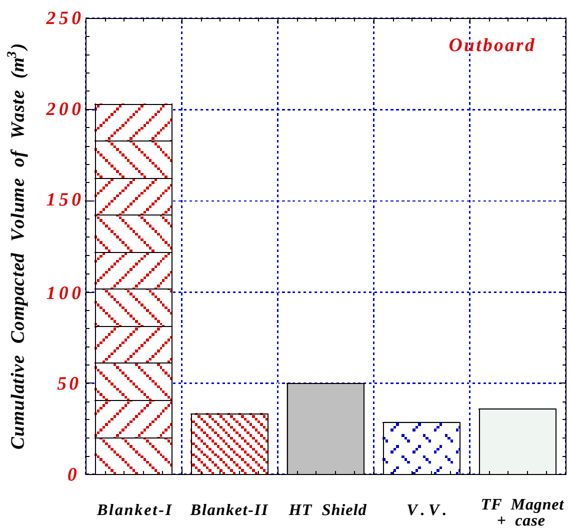

* 45 cm thick OB B-II and 15 cm thick OB HT shield. 8m height for all components except IB blanket (6 m)

Total Compacted Waste Volumes*

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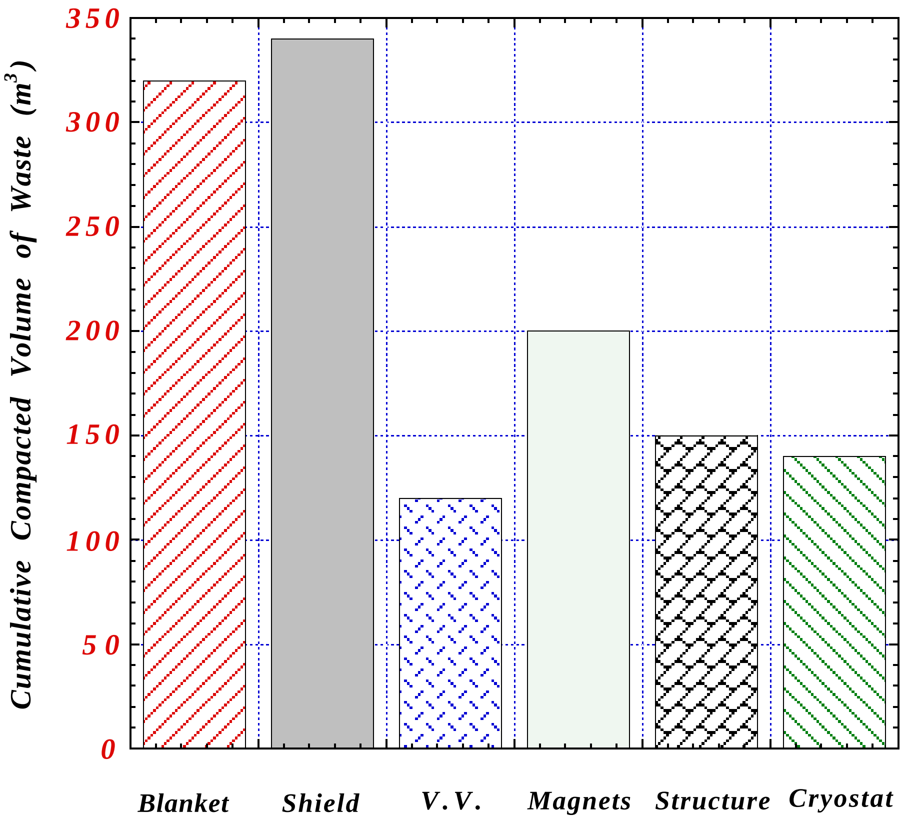
	Cumulative Compacted Waste Volume (m^3)		
	IB & OB	All Components	
IB & OB Blanket-I	287	287	(22%)
OB Blanket-II	33	33	(3%)
Shield	95	340	(27%)
V.V.	75	120	(9%)
Magnets & SS	75	200	(16%)
Structure		150	(12%)
Cryostat		140	(11%)
		1270	



* 45 cm thick OB B-II and 15 cm thick OB HT shield

Class A and C Waste Volumes*
for All Components

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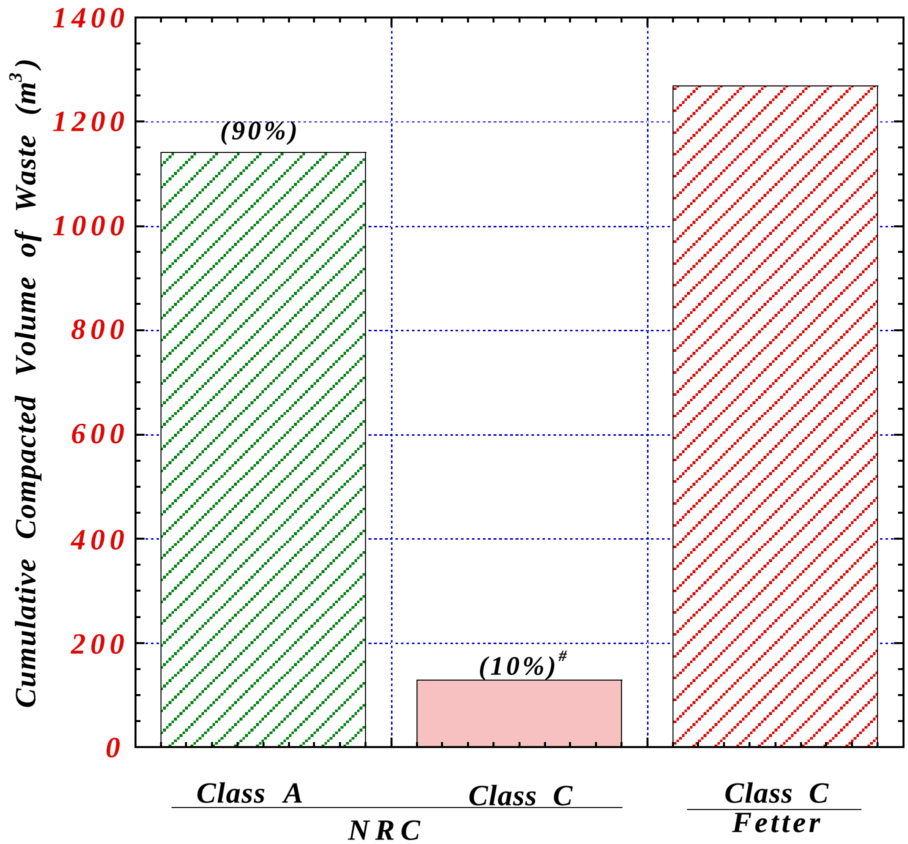

ARIES-AT generates
90% Class A LLW and 10% Class C LLW
according to NRC limits

- 45 cm thick OB B-II and 15 cm thick OB HT shield

\# OB B-II, IB HT shield, and divertor HT shield