US-JPN WS on Fusion Power Reactor
2014.3.13, UCSD

Concept Development of DEMO in Japan

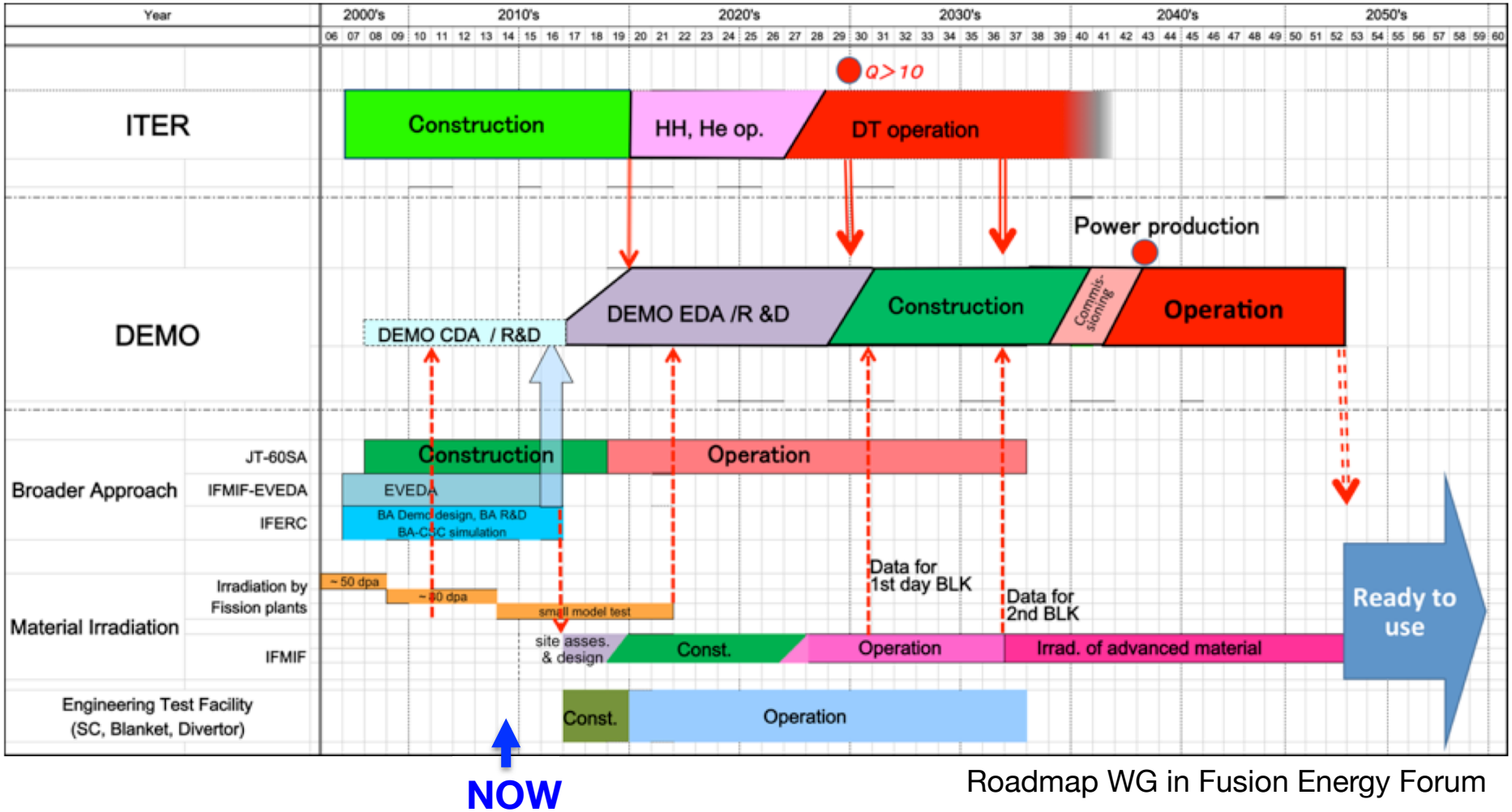
JAEA
Kenji Tobita

OUTLINE

1. **Circumstances Surrounding DEMO**
2. **DEMO Design Activity**
3. **Safety Study**

1. Circumstances Surrounding DEMO

- Conceptual design — To be completed in 2020
- Construction — To start in the early 2030s
- Operation — To be commenced in the early 2040s



Roadmap WG in Fusion Energy Forum

What to be done by the completion of CDA

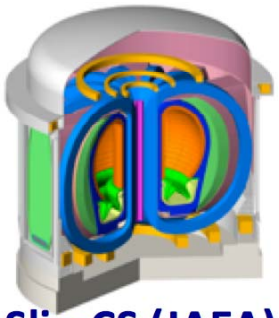
✓ Compelling vision of DEMO
in light of feasibility and public acceptance

✓ Systematically consistent design with fabrication feasibility

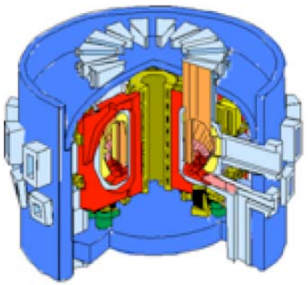
Previous DEMO concepts

too challenging

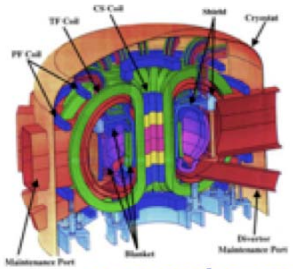
R_p = 5.8 – 7.3 m
P_{fus} ~ 3 GW



SlimCS (JAEA)



SSTR (JAEA)



Demo-CREST (CRIEPI)

Need to pursuit a more conservative concept

Recent situations

❖ **Most JA researchers have changed views on DEMO.**

Came to aware of

1) difficulty in power handling
2) importance of safety

❖ **Need to crystalize post-BA projects toward DEMO, asap**

- BA terminates in May 2017
- No R&D programs except ITER have been authorized so far.

"Joint Core Team" is at work on planning DEMO development

Details in Kasada-san's talk

- Organized under the Working Group of Fusion Research in MEXT
- Members: several in 40's from JAEA, NIFS and university.
- Tasks: 1) Review all kinds of on-going R&Ds toward DEMO,

2) Provide a practical interpretation of **AEC requirements**,

3) Define DEMO development programs after BA.

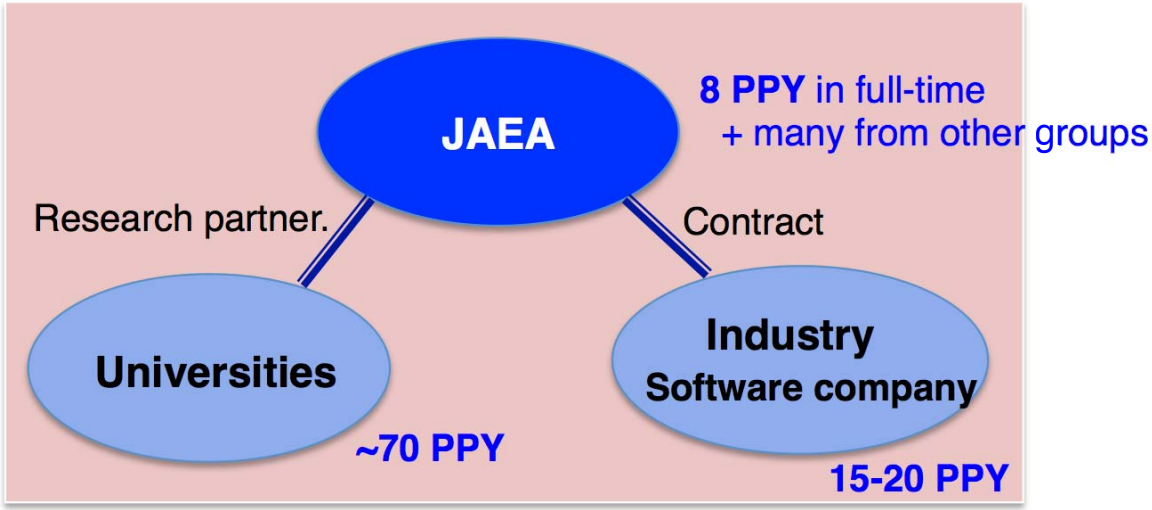
demanding

Requirements for DEMO ← AEC Report (2005)

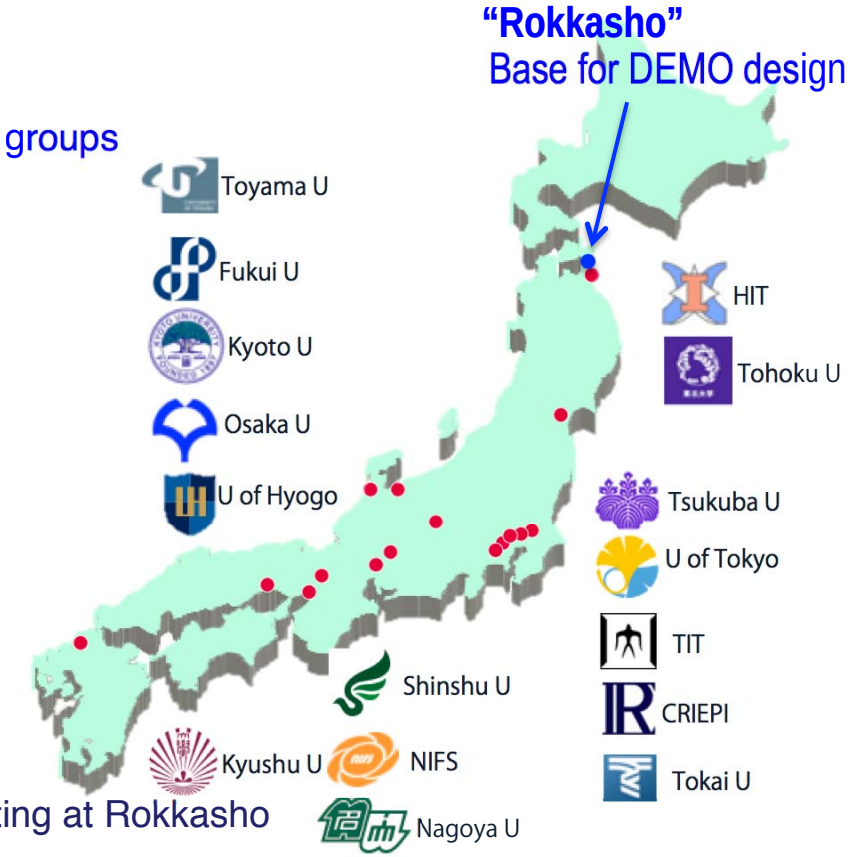
1) Size — as small as ITER
2) Power — as large as 1,000 MWe
3) Operation — 1 year-long continuous operation
4) Fuel production — self-sufficienct production of T

2. DEMO Design Activity

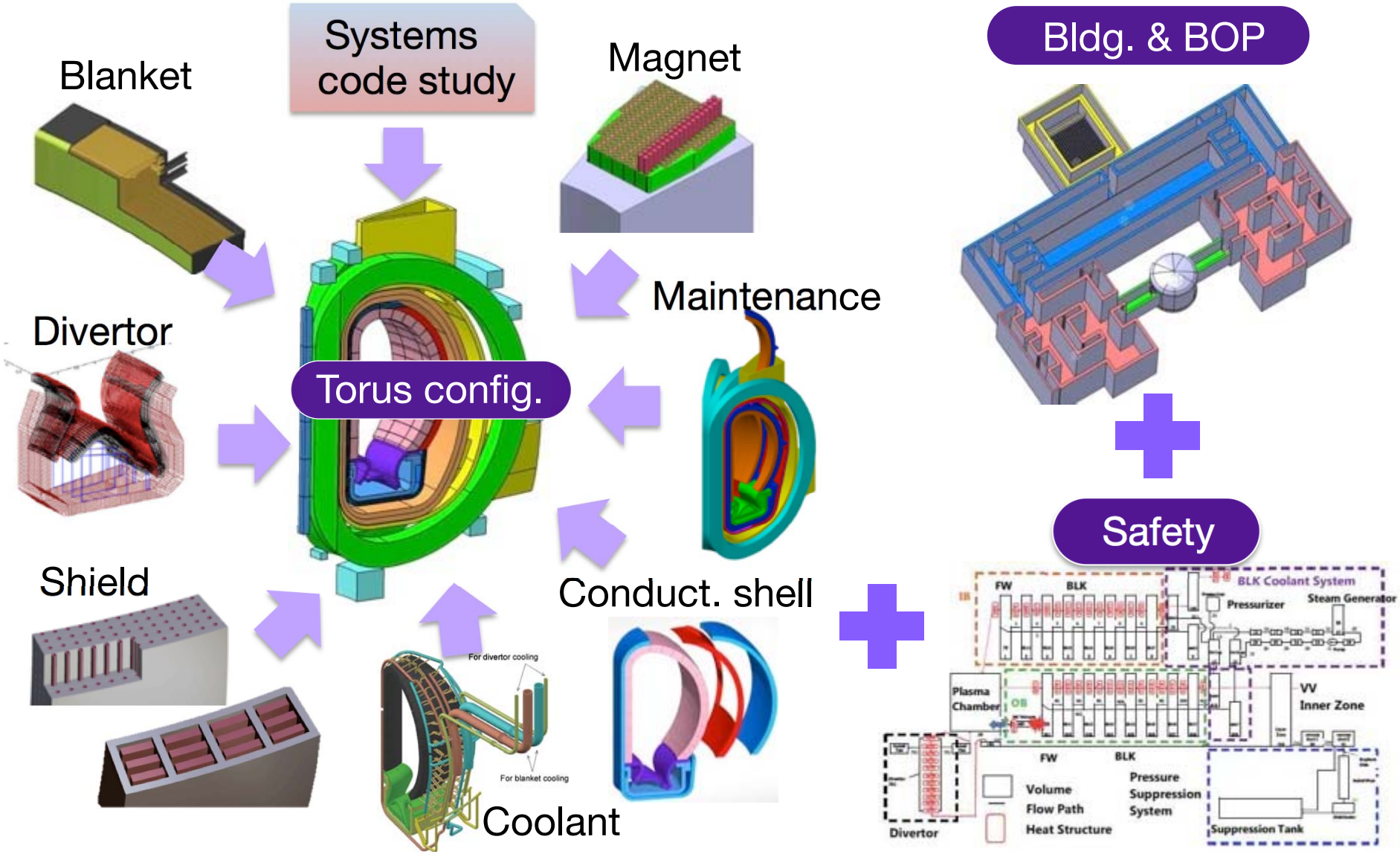
- Requested to carry out the activity on a Japan-wide basis.
- Tokamak design groups have been consolidated into a framework centered around the BA.
- 70 researchers of univ. and labs., involved in DEMO design.
  Met 30 times for cooperation on DEMO design in 2013.



Design Team meeting at Rokkasho
30-31 Sept, 2013



Critical design issures are being addressed for concept development

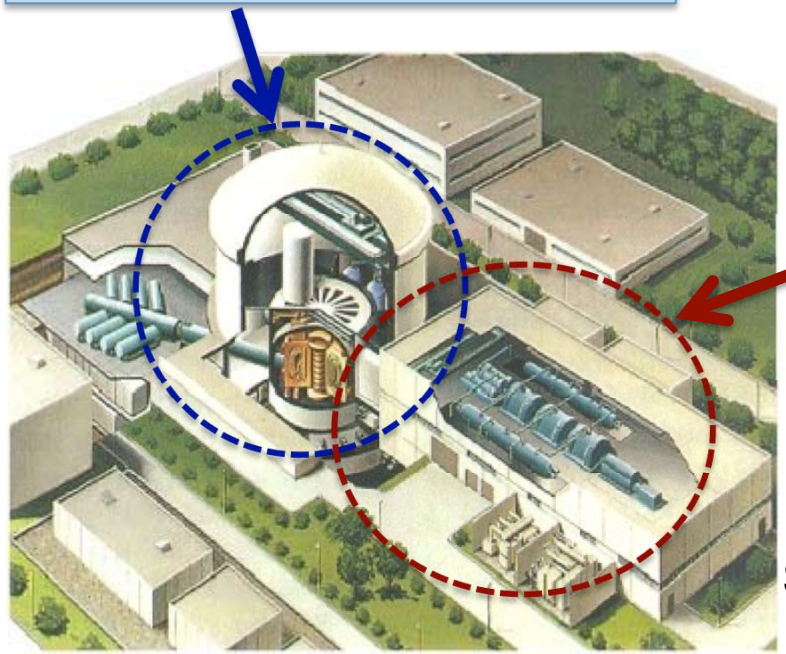


Strawman plan of DEMO *under negotiation in Japan*

- ❖ Foreseeable from ITER and on-going JA programs
- ❖ Pursuit a low power DEMO for divertor heat removal

Reactor

- R_p ~ 8.2 m or more
- P_{fus} ~ 1.5 GW
- Continuous operation
- T self-sufficiency

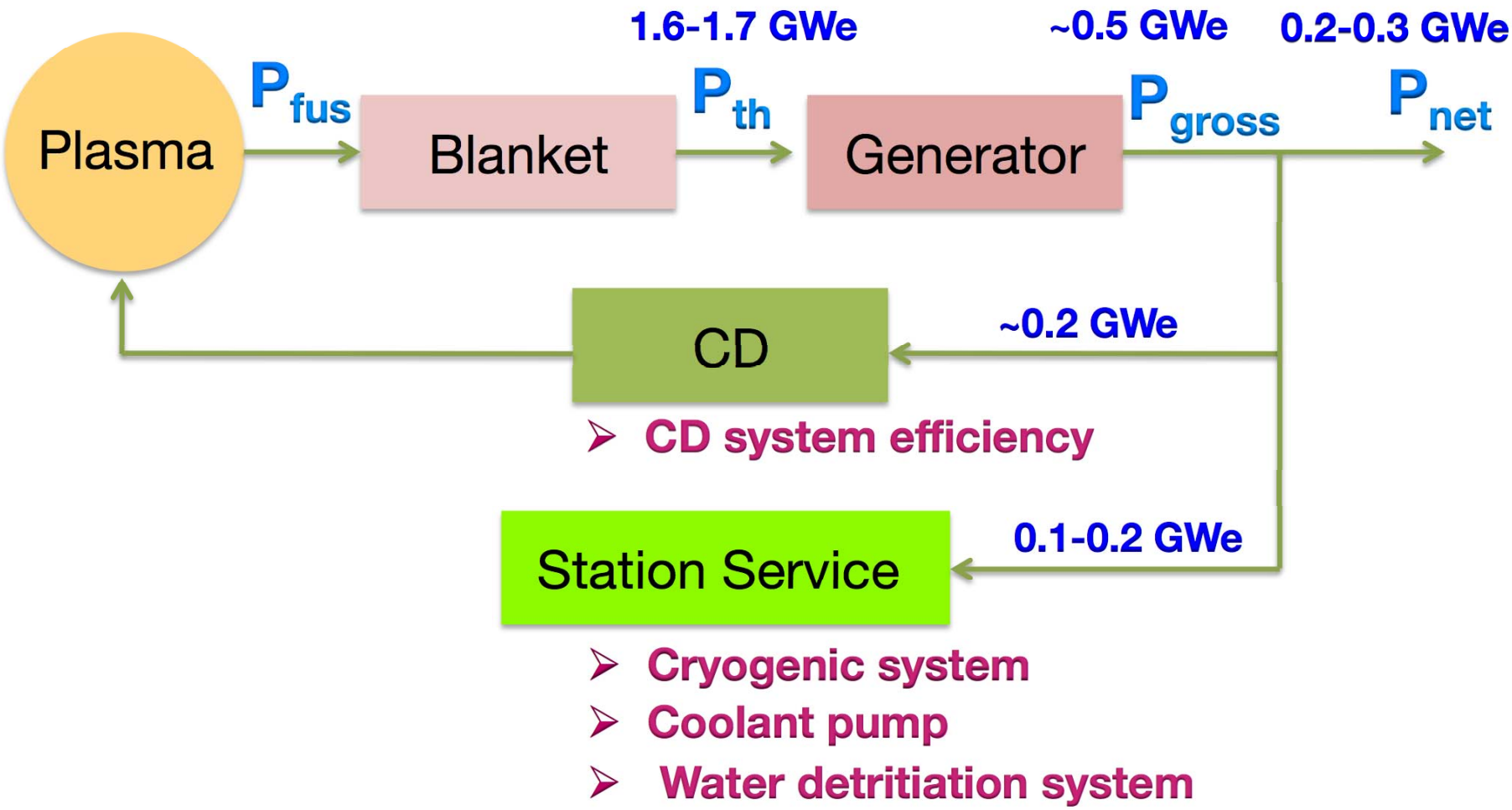


SSTR
to be replaced

BoP (Balance of plant)

- PWR water
  15.5 MPa, 290-325°C
- P_{th} = 1.6 – 1.7 GW
- P_{net} = 0.2-0.3 GWe
  η_{th}: ~30%（steam turbine）

Power Flow

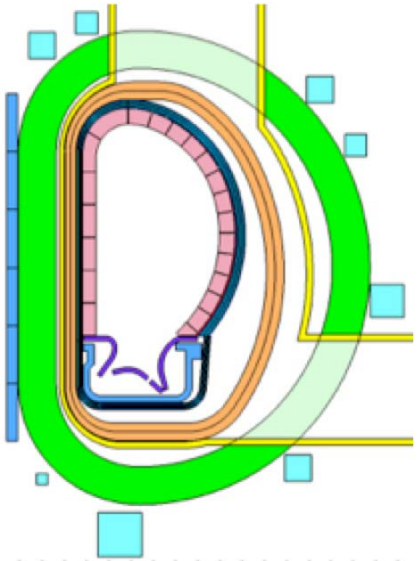


In a low power DEMO,

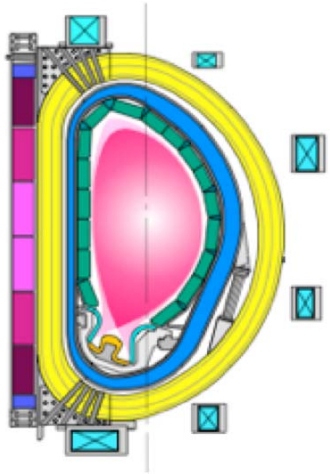
it is of importance to assess the station service power.

Magnet (TF)

DEMO



ITER

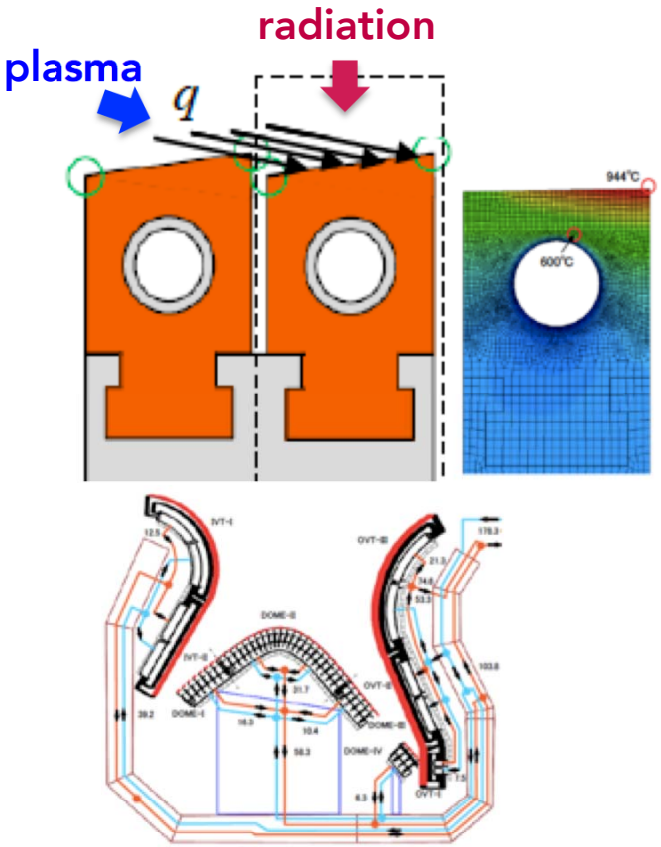


	DEMO	ITER
SC strand	Nb_3Sn or Nb_3Al	Nb_3Sn
Height Width	~20 m ~14 m	13.5 m 9 m
N of TF	16	18
B_{max}	~13.3 T	11.8 T
Cond. current	tbd	68 kA
Voltage proof	20 kV	20 kV
T_{op}	5K	5K
Design stress	~800 MPa	667 MPa
Magnetic energy	~170 GJ	41 GJ

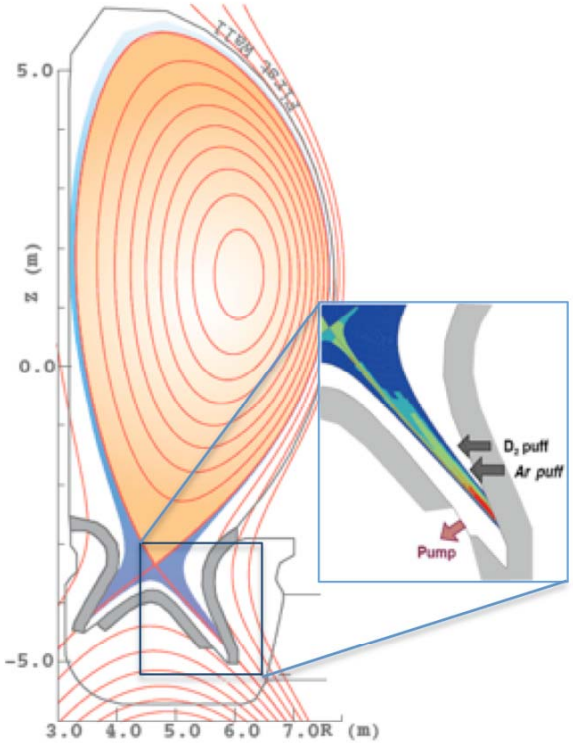
Divertor design

Three appreaches are ongoing to develop DEMO divertor concept.

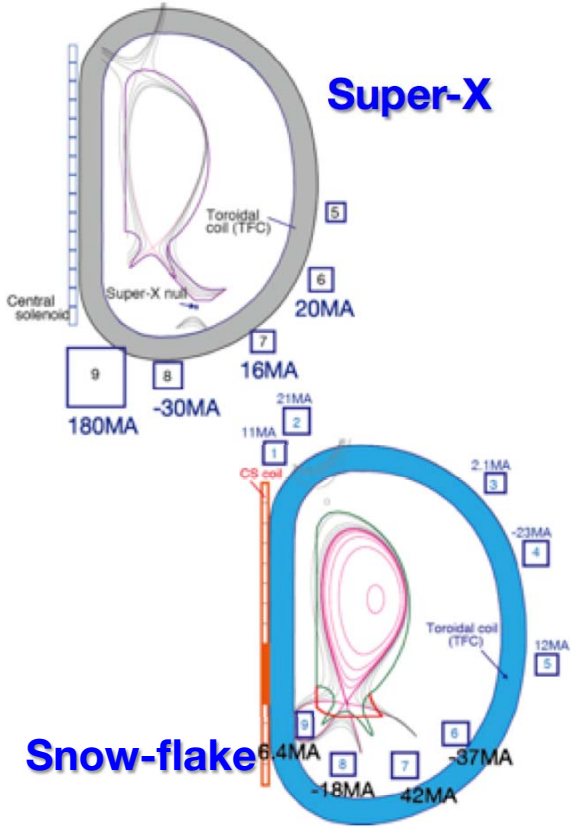
Target design



Detach. simulation

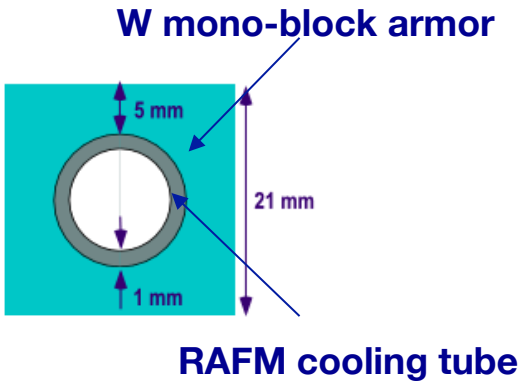


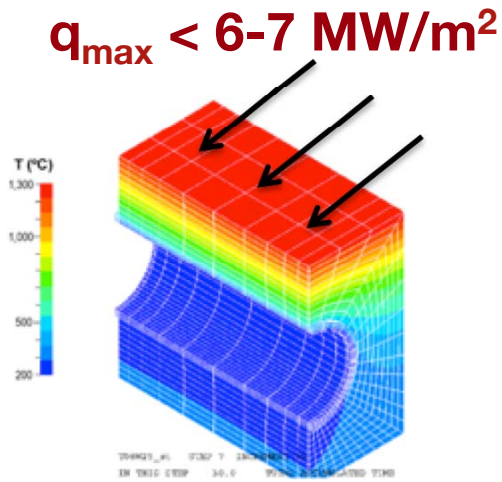
Alternative config.



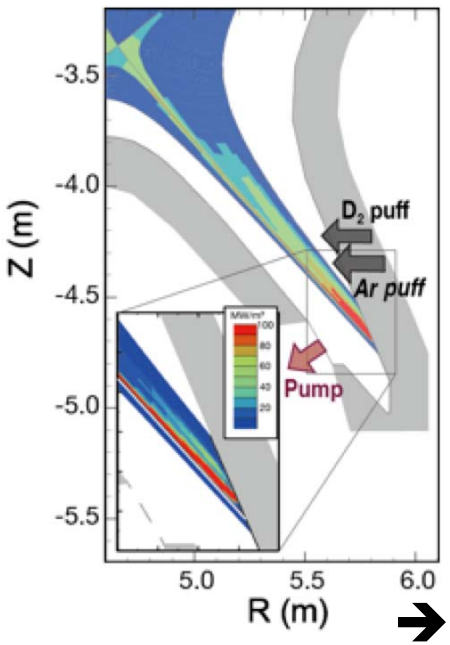
Divertor heat removal

Allowable heat flux





	Requirement
W	T ≤ 1,200°C
RAFM	290°C ≤ T ≤ 550°C
water	~2900°C



Divertor simulation for conventional divertor:

- P_{fus} = 3 GW ➔ q_{max} ≥ 10 MW/m² f_{rad} = 92%
- P_{fus} = 2 GW ➔ q_{max} ~ 7 MW/m² f_{rad} = 80%

➔ P_{fus} < 2 GW seems to have reality in light of Div. heat load.

Advanced divertor configurations

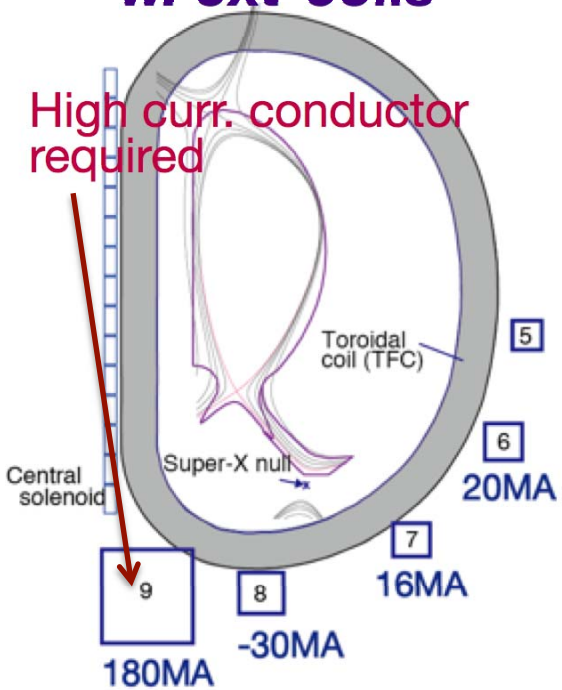
- Applicability of “super-X” and “snowflake” have been studied.
- Application of “super-X” may be possible when either of two challenging PF coil technologies are developed:

  1) high current conductor of >150 kA, 2) inter-linked divertor coils

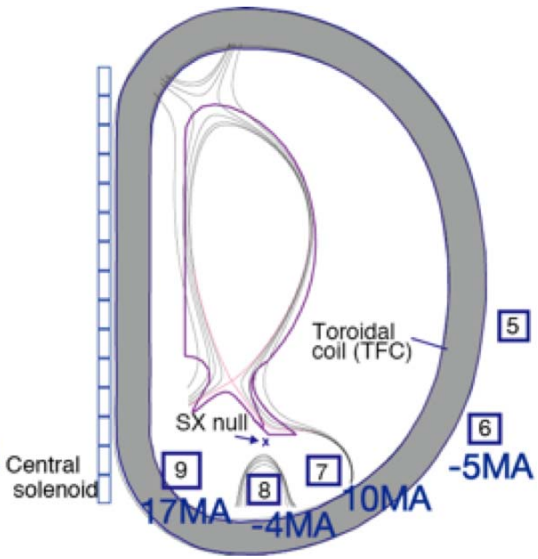
- “Snowflake” is likely to be ruled out because of required coil currents.

Super-X

w. ext-coils

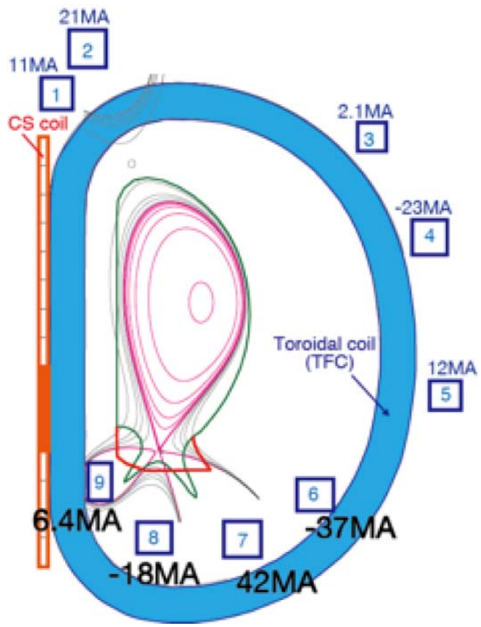


w. inter-linked



Need feasibility study on IL winding

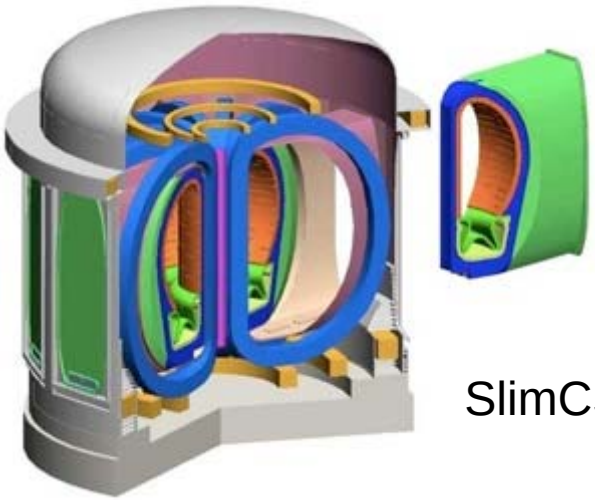
Snowflake



Too high current even for inter-linked

Remote maintenance

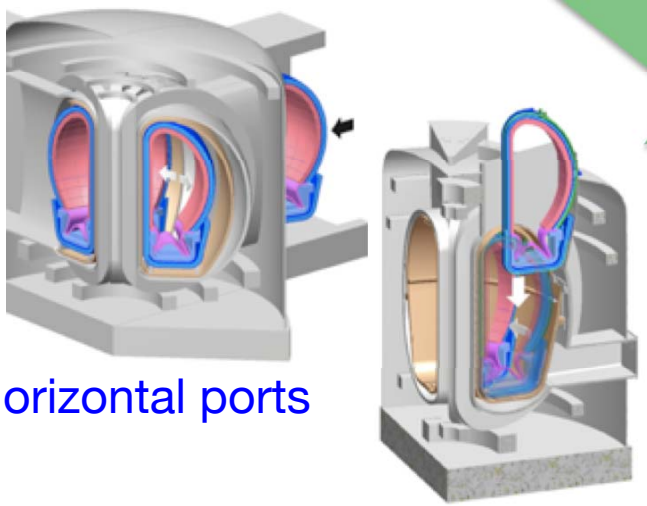
Sector scheme, originally



SlimCS

- separated maintenance of divertor and blanket

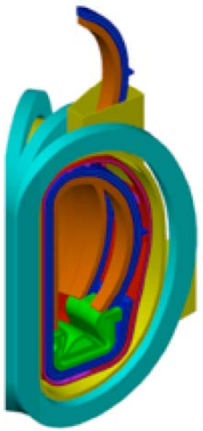
  divertor – 1 yr, blanket – 4 yrs
- Minimize radwaste
- No merits in sector scheme for a larger reactor (R_p~ 8 m)



Horizontal ports

Vertical ports

➔ Need to revisit banana-segment

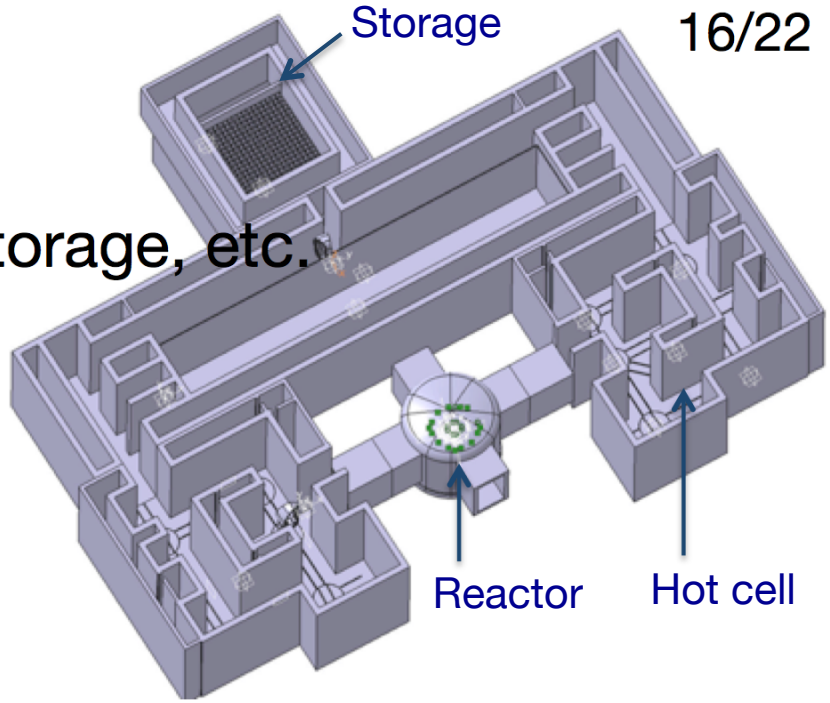


like SSTR, EU designs

New findings on RM

- Area required for hot cell, waste storage, etc.

  > 17,000m^2, being 4-6 times as large as the hot cell of ITER (4,000 m^2)



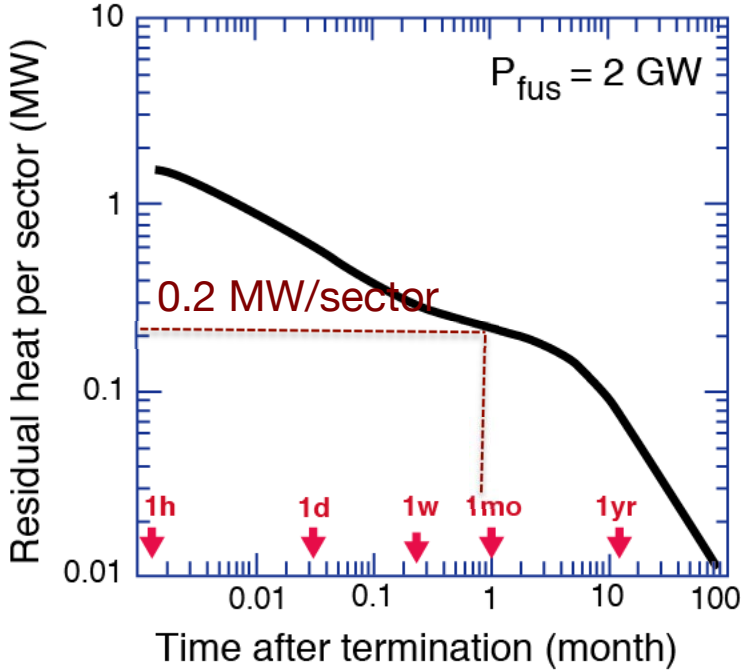
- Residual heat

  No coolant is supplied during maintenance although the residual heat per sector is as high as 0.1-0.2 MW.

  ➔ considerable temperature rise of components during maintenance

  ⬇

  Residual heat removal can be a critical issue in RM scenario.



3. Safety Study

Started in 2013. At present, analysis only.

❖ Long-term targets

- Radioactive protection for public and workers
- Prevention and mitigation of accidents

  ← To meet evacuation-free dose (IAEA : ≤ 50 mSv in 7days)
- Management strategy of radioactive waste

❖ Mid-term targets

- Characterize safety on DEMO
- Define measures to ensure the safety of DEMO
- Foster young experts

Reference DEMO

	DEMO	ITER
Operation	Steady state	Pulsed
Major radius (m)	~ 8 m	6.2m
Fusion power	1.3-1.5 GW	500 MW
Enthalpy of coolant	200 GJ/loop 290-325°C, 15.5MPa	53-76 GJ/loop 100-150°C, 3MPa
Magnetic E of TF	~160 GJ	~50 GJ
Chemical E	Be: 780 ton ($Be_{12}Ti$)	-
Residual heat	~5 MW (1day later) ~1.3 MW (1 mo. later)	0.6 MW (1day later) 0.3 MW (1 mo. later)

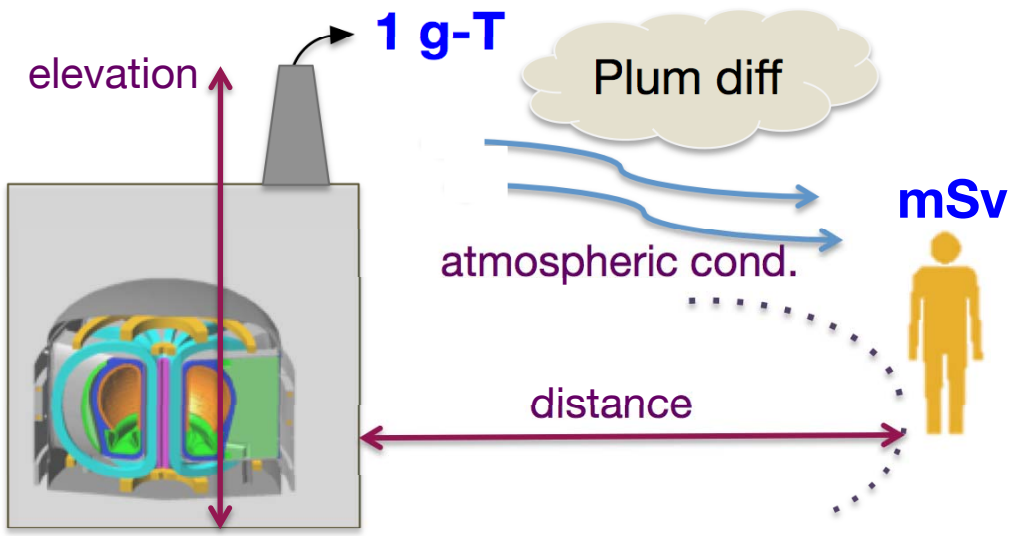
Management of the following energies are key for safety.

- Coolant enthalpy ➔ concept development of confinement
- Chamical E ➔ suppression of Be-steam reactions
- Residual heat ➔ Torus config. for removing the heat
- Magnetic E

Recent results (1)

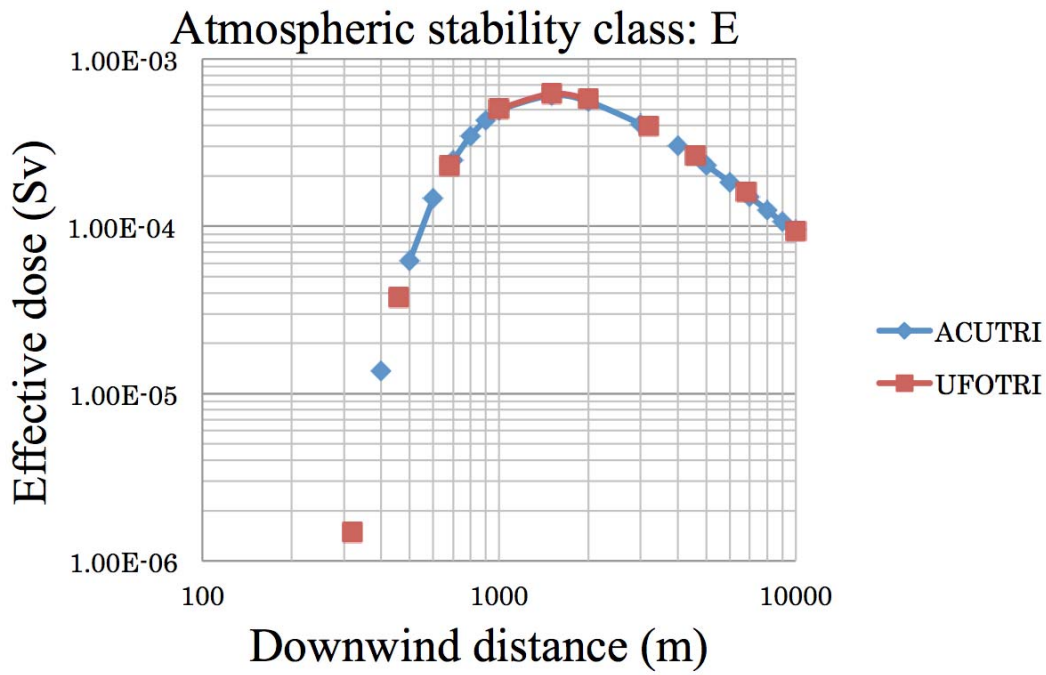
Environmental release of T

- Necessary to relate T release to public dose
- Calclation under various conditions



Benchmark of

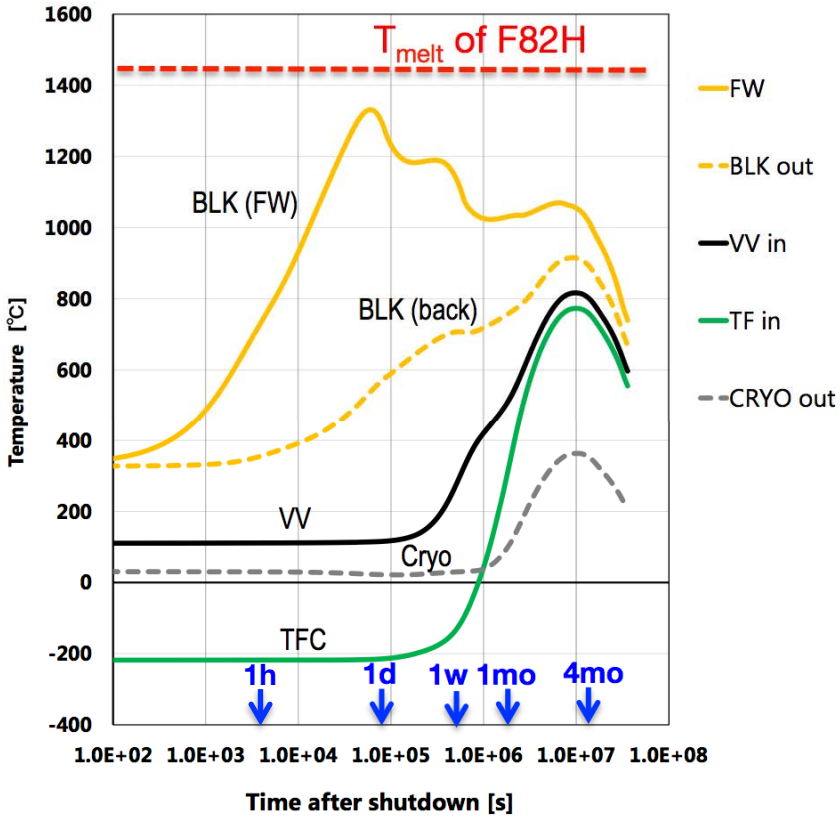
- ✓ ACUTRI (JAEA)
- ✓ UFOTRI (KIT)



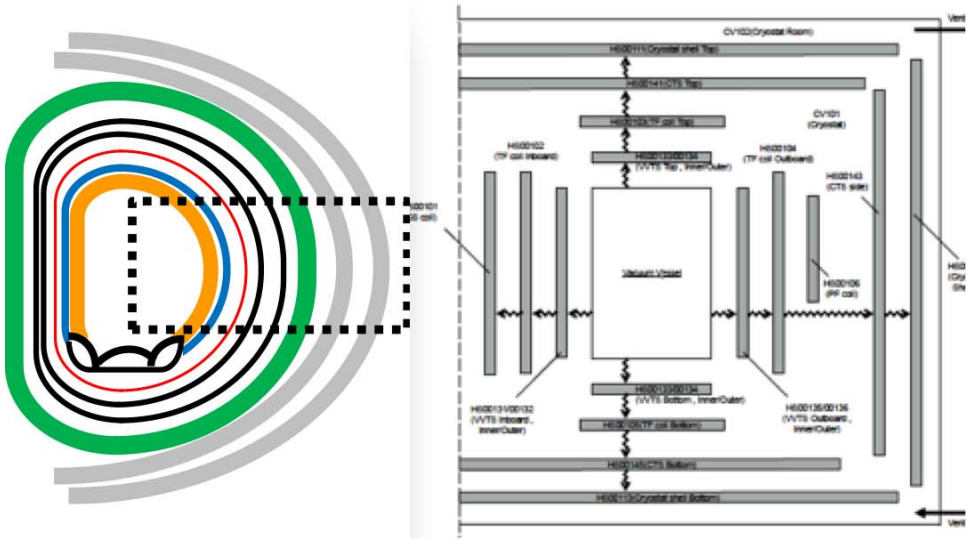
Recent results (2)

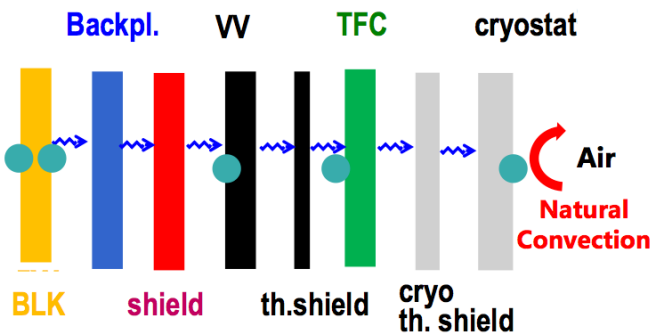
Temperature after LOCA

- Total LOCA in a moment
- No power supply & no actions to enhance heat transfer



MELCOR model





Domestic and international partnership on safety

Accident analysis

- MELCOR analysis / Feedback to design TOSHIBA Leading Innovation >>>
- Modelling / safety sys Kyoto U Tokyo Inst. Tech

Idaho National Lab

Safety system

- Confinement / detritiation sys JGC
- T concentration in water Kyushu U
- Passive heat removal sys Hachinohe Inst. Tech.

Generic safety issues

U of Tokyo CRIEPI

FUSION FOR ENERGY

F4E

Public dose assessment

- Environmental release of tritium and dust

Radioactive waste

- Waste management scenario / hot cell /storage MITSUBISHI HEAVY INDUSTRIES, LTD. Our Technologies, Your Tomorrow
- Disposal Tokai U Fukui U Tokyo Inst. Tech NIFS

Summary

- Tokamak design groups have been consolidated into a framework centered around the BA.
- Strawman plan of JA DEMO is medium-sized (R_p ~8.2 m or more) and low power (P_{fus} = 1.3-1.5 GW) for ease of divertor heat removal and ensuring safety.
- DEMO safety study stared with severe accident analysis.
- Planning on DEMO development (post-BA program) is under full consideration.