2019 Symposium on Nuclear Data (Nov. 29, 2019, Chikushi Campus, Kyushu University)
(e-mail: mat@nucl.kyushu-u.ac.jp)

Performance of a gas-cooled reactor as a tritium production device for fusion reactors

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Collaborators

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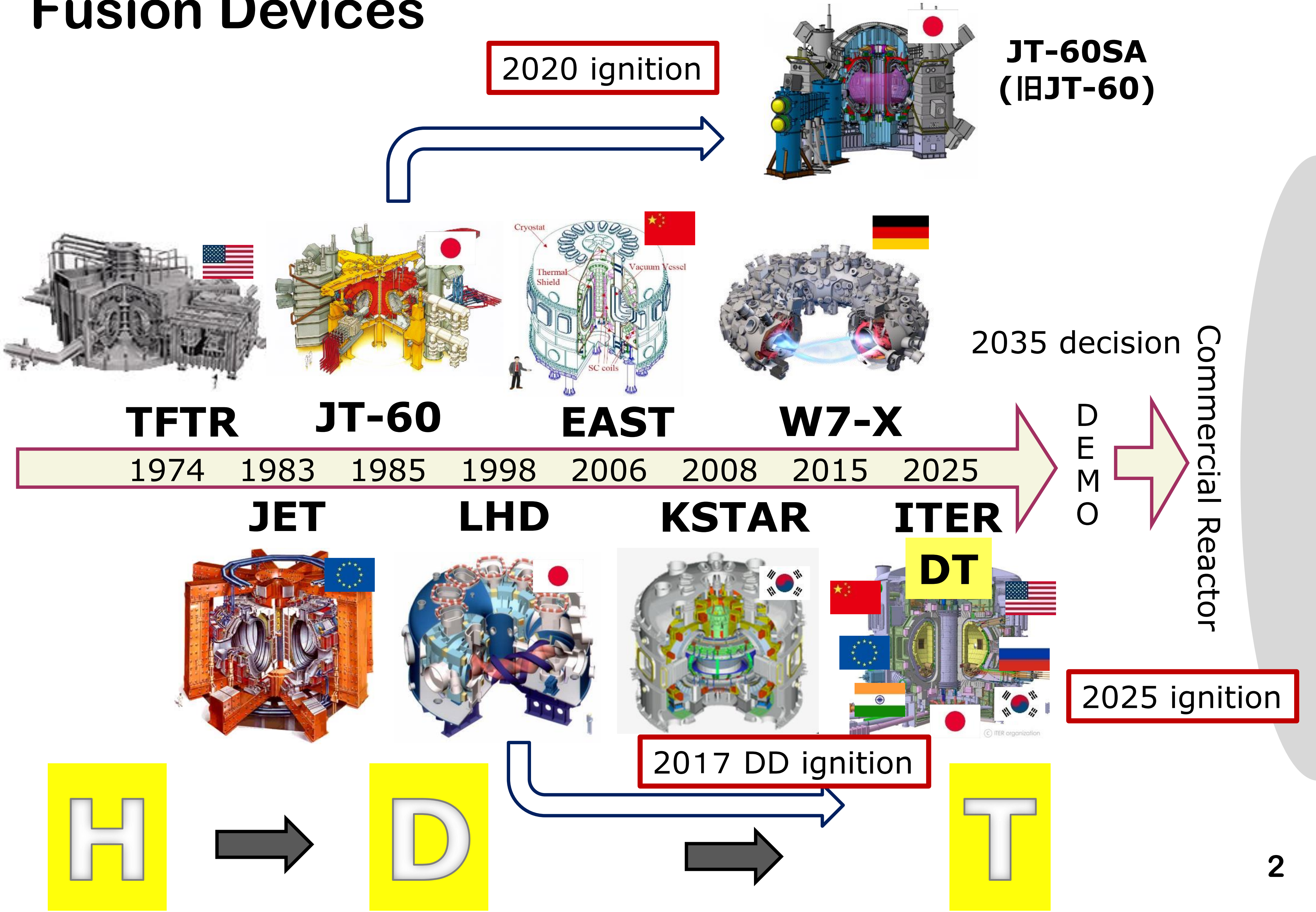
QST	**K. Tobita, R. Hiwatari, Y. Someya N. Sakamoto, Y. Udo**

supported by ***JSPS KAKENHI Grant-in-Aid for Scientific Research (B) JP18H1200***
Cooperative program with ITER/BA reactor design team

Fusion Devices



■ ITER-FEAT

Target :

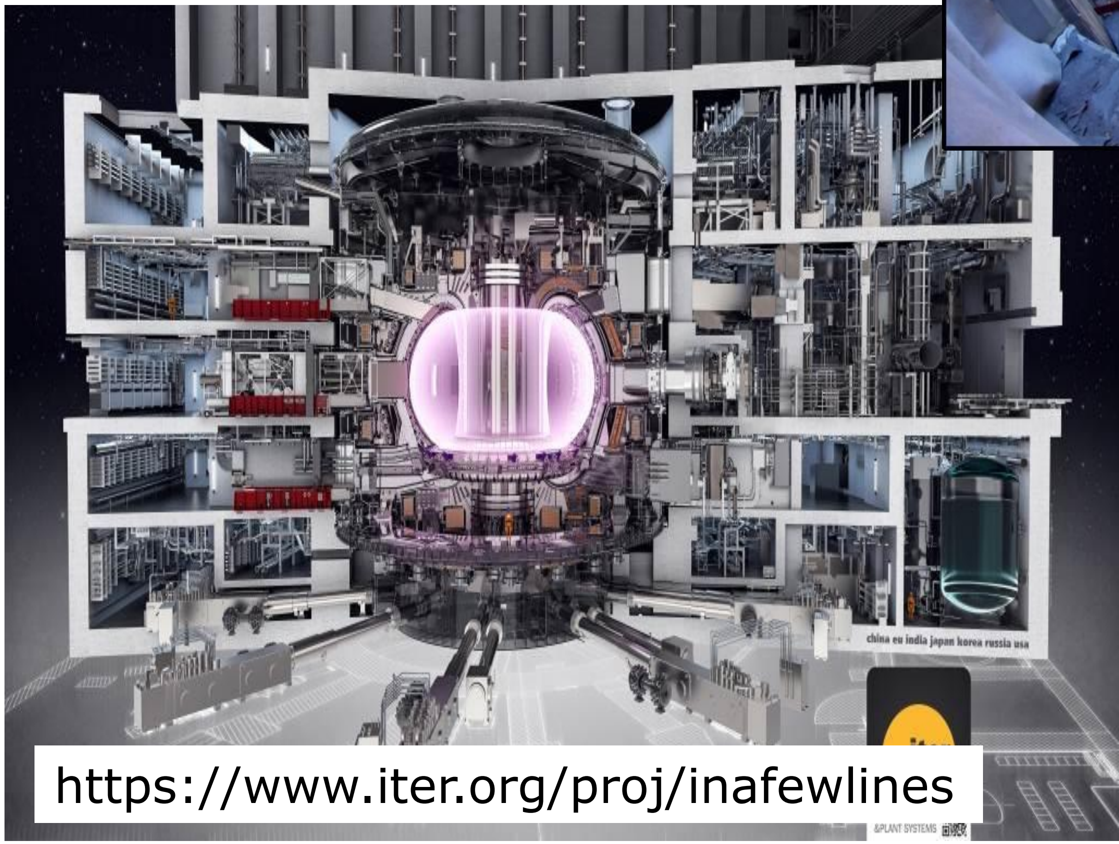
Fusion Power : 500 MW

Q value : > 10

Burning plasma sustainment :

400-500 s

Mar. 2019 (Cadarache, France)



https://www.iter.org/proj/inafewlines

Tritium requirement for fusion reactor and the development

$$\mathrm{D} + \mathrm{T} \longrightarrow \alpha\ (3.52\ \mathrm{MeV}) + n\ (14.07\ \mathrm{MeV})$$

12 years (Half life)

Existing in Nature

- In a 1.5 (3.0) GW fusion reactor, 200 (400) g/day of tritium is required.
- Although fusion reactor produces tritium in a blanket system, initial loading tritium must be provided from outside.
- For prior technical tests of tritium circulation and blanket systems, at least 100 g of tritium is necessary.

In fusion community, a method to supply sufficient amount of tritium for fusion reactor is not clarified.

Status of the CANDU reactors

IAEA Web Site:
https://nucleus.iaea.org/Pages/default.aspx.

Canada

2018年（18 HWRs）

500～900 MWe

Electricity：**95037** GWh

Time on Line：**134838** h

Romania

2018年（2 HWRs）

~ 650 MWe

Electricity ：**10442** GWh

Time on Line ：**16578** h

Korea

2018年（3 HWRs）

~650 MWe

Electricity ：**13061** GWh

Time on Line ：**20976** h

Total

2018年（23 HWRs）

Electricity ：**119** TWh

Time on Line ：**172392** h

Fusion Engineering and Design 10 (1989) 399–403

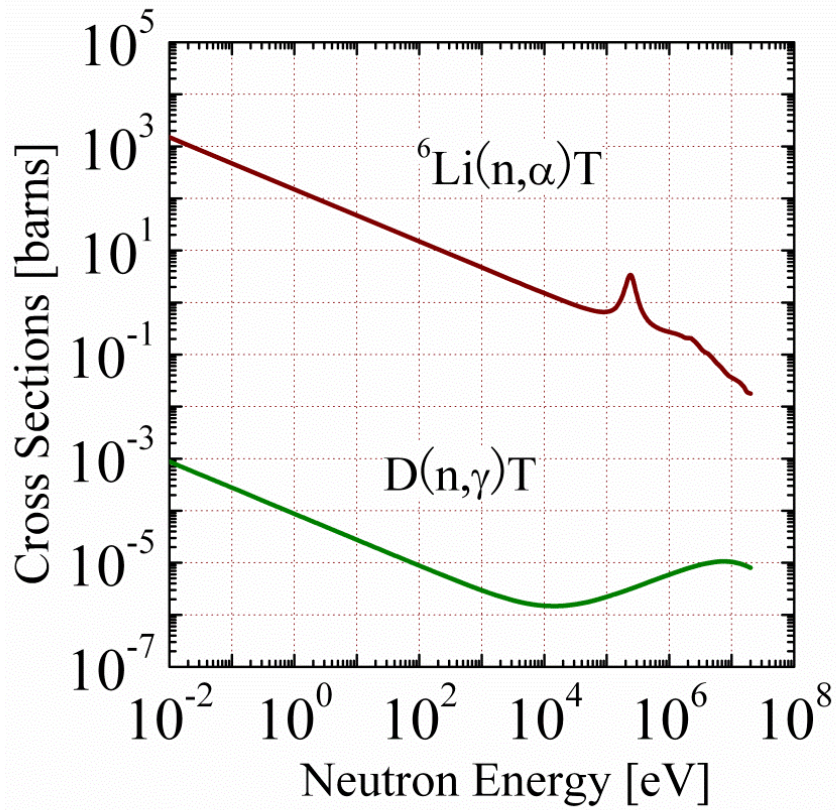
TRITIUM SUPPLY FOR NEAR-TERM FUSION DEVICES

P. GIERSZEWSKI

Canadian Fusion Fuels Technology Project, Mississauga, Ontario, Canada L5J 1K3

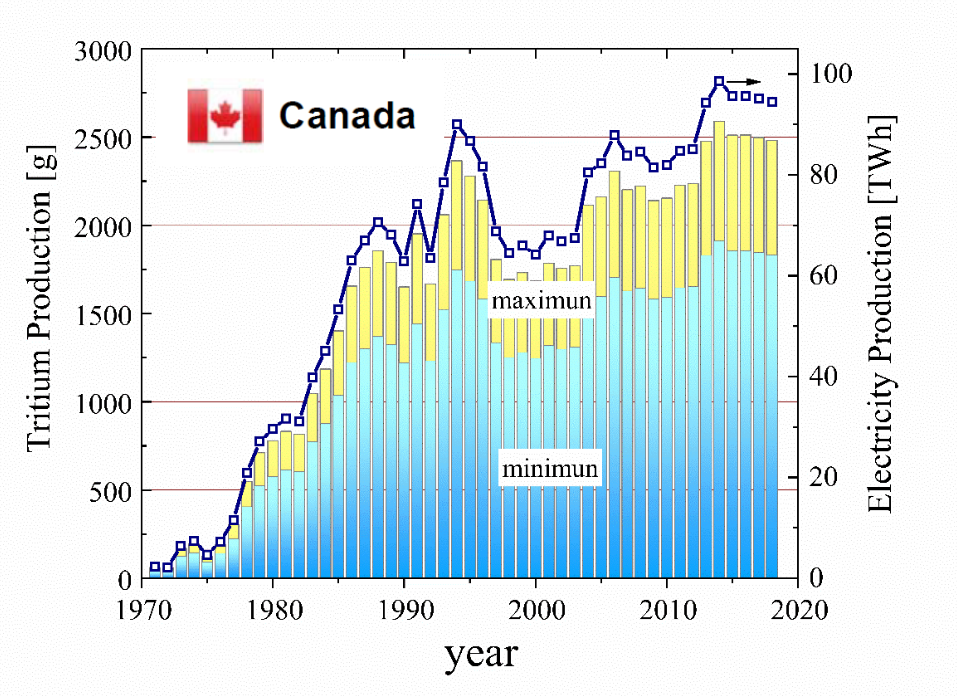
1.3. Heavy water absorption

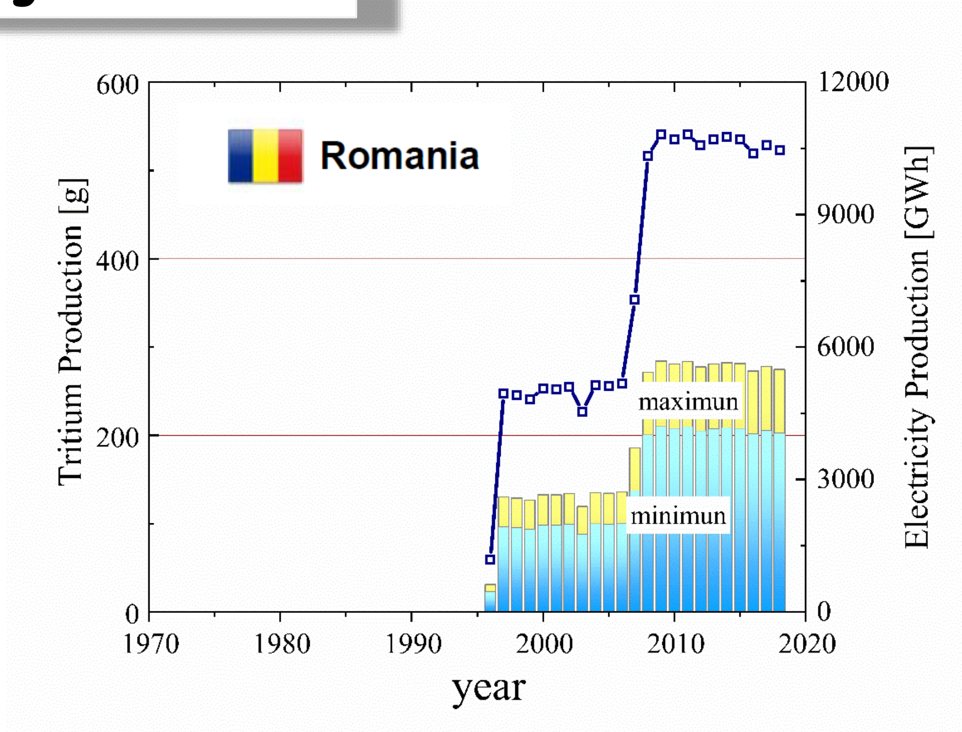
In heavy water reactors such as CANDUs, some neutrons are absorbed in the heavy water (D_2O) leading to production of tritium from (n, gamma) reactions. In present CANDUs, this occurs at about 0.17–0.23 kg T/GWe-y [4]. There is presently 12 GWe of CANDU reactor capacity in the Ontario Hydro electric grid and 3.5 GWe under construction at Darlington A. There is a further 2.5 GWe of CANDU operating in Canada and elsewhere internationally, and 3.1 GWe under construction (at Cernavoda, Romania). Other heavy-water reactors account for 1.6 GWe in operation and 2.5 GWe under construction [7]. For the 20 CANDU reactors within the Ontario Hydro grid (Pickering A & B, Bruce A & B, and Darlington A stations), the maximum tritium production is estimated as 2.5 kg T/y at 80% capacity. The total production rate for all heavy-water reactors could be 4 kg T/y.

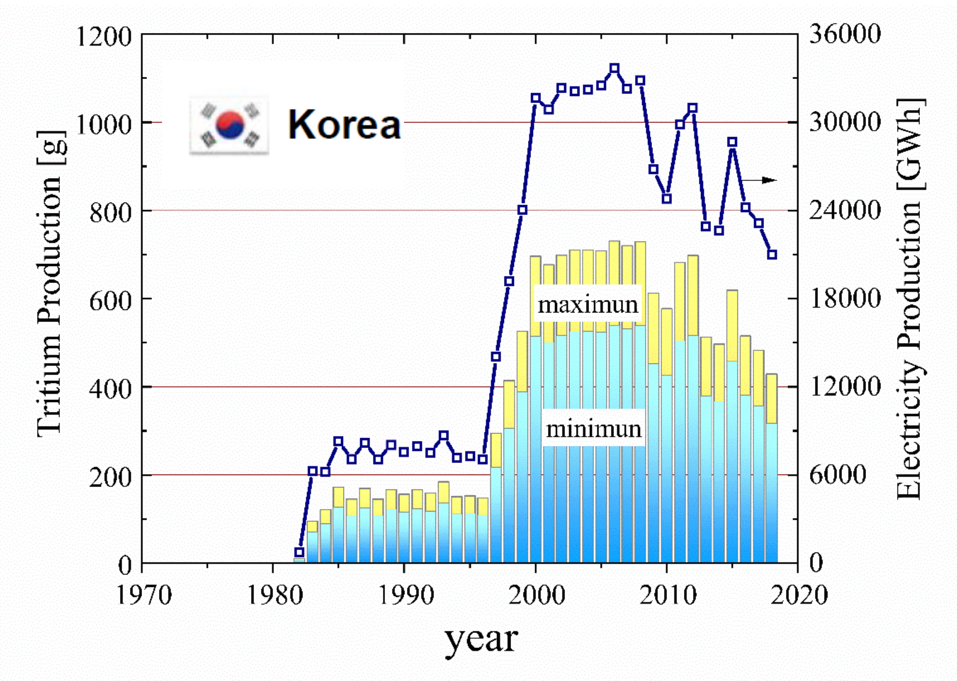


[4] K.Y. Wong et al., Canadian Tritium Experience, Ontario Hydro report (1984).

Estimated Tritium production by HWRs







Estimated Tritium production：

2.2～3.0 kg/y

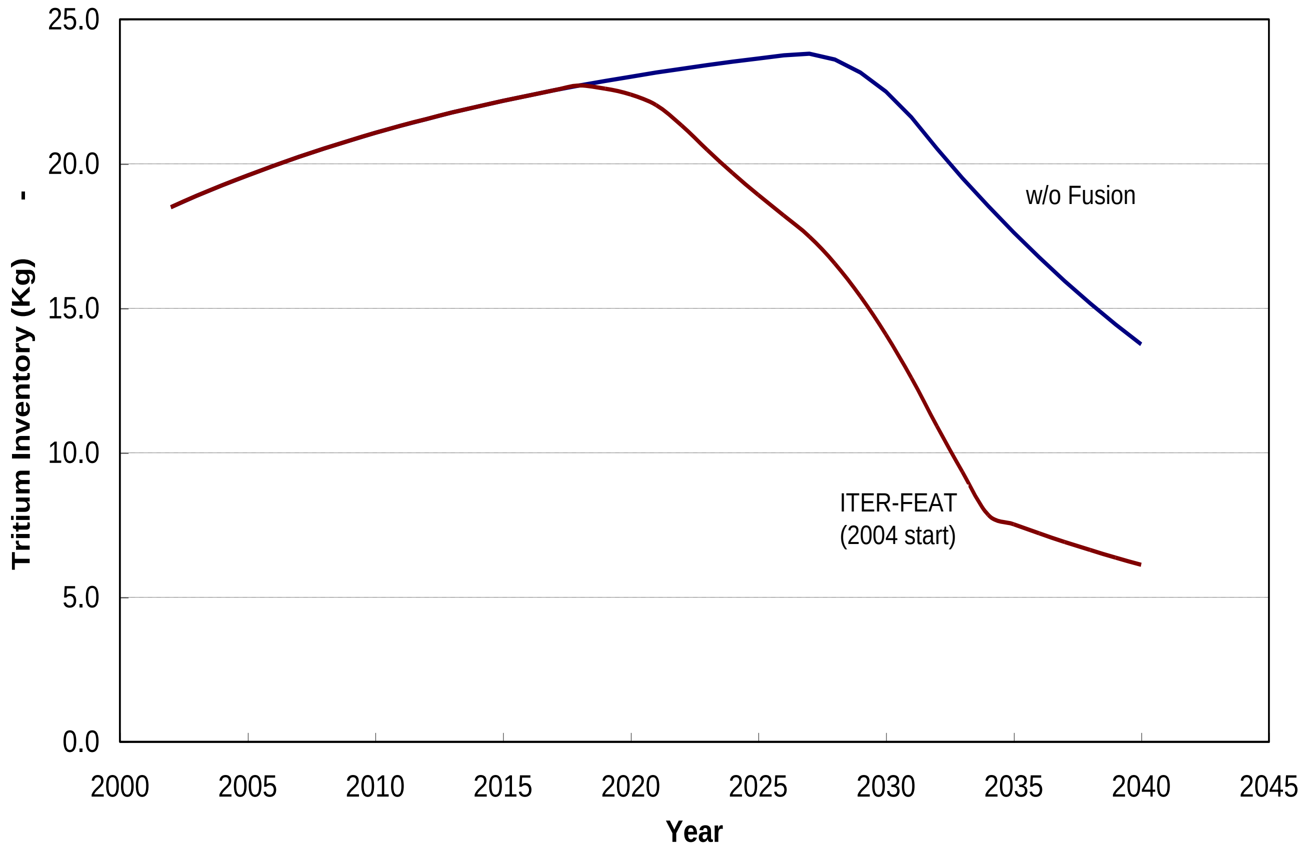
- 30～45-years operation (70% HWRs)
- ITER will use most of Tritium produced in Canada

Tritium in market

Assumption

Scott Willms, Tritium supply considerations

- Presently there are 20 operating Canadian CANDU reactors
- Reactors licensed for 40 years
- Tritium recovery rate was ~2.1 kg/y. Now it is ~1.5 kg/y.



How to supply a sufficient amount of tritium to
(1) DEMO fusion power generation reactor
(2) prior technical tests of tritium circulation and blanket systems of a DEMO fusion reactor.

- In 90s, required amount of tritium for startup of the fusion power reactor had been evaluated as 27.6 kg. There are several estimations at this stage.

  (Asaoka, et al., FT 1996)

- We do not have any actual plans to provide 20~30 kg of tritium to DEMO fusion reactor.

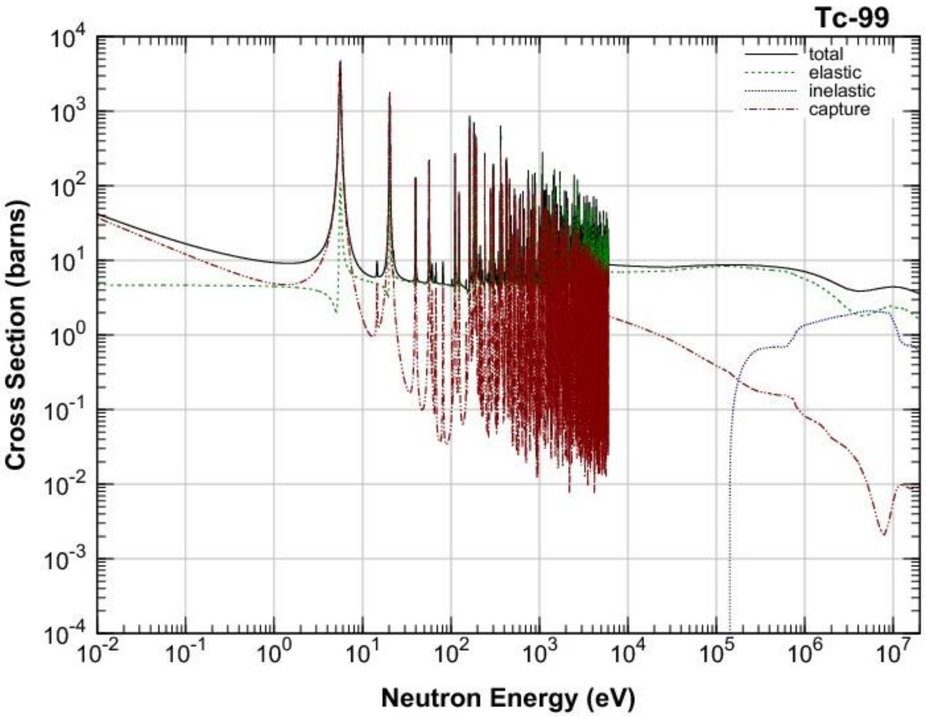
  Recently it was revealed that tritium retention in the plasma vessel have not been taken into account in the previous evaluations.

  (Roth, Nucl. Mater. 2009, Nishikawa, FST 2011)

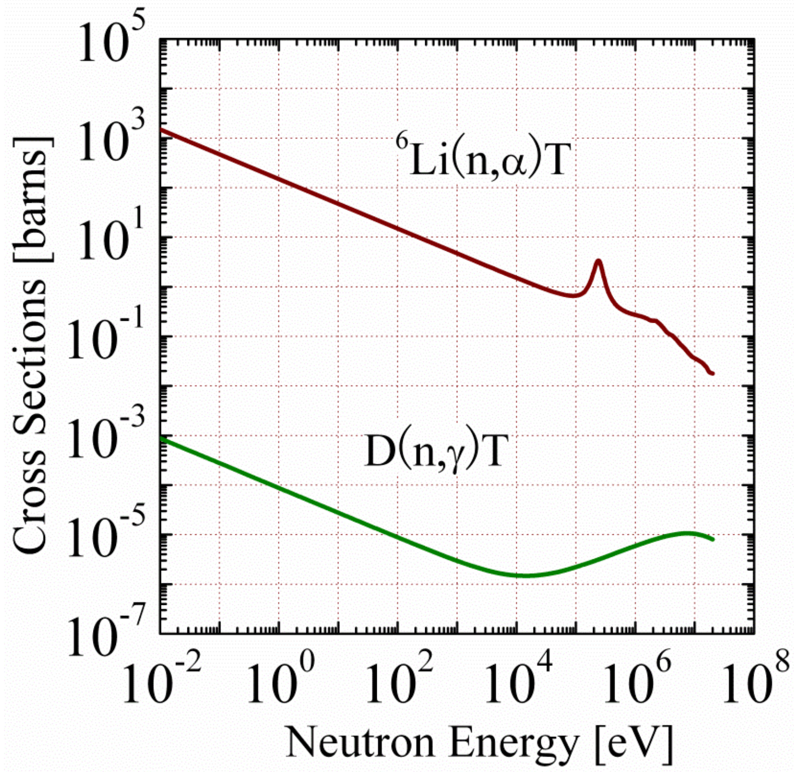
The uncertainties of the tritium-supply scenario seem to be still increasing.

To secure tritium supply to the DEMO fusion reactors, it is important to prepare an effective scenario to stably supply an adequate amount of tritium at this stage.

Transmutation of LLFP using high-temperature gas-cooled reactors



Tritium production using high-temperature gas-cooled reactors



(1)「Study on transmutation of long-lived fission product using high-temperature gas-cooled reactors」, H. Nakaya, et al., Proc. of GLOBAL2011, Paper No.391363 (2011).

(2)「Study on transmutation and storage of LLFP using a high-temperature gas-cooled reactor」, K. Kora, et al., JAEA-Conf 2014-003 (PHYSOR 2014, Kyoto, Japan) (2014).

(3)「A study on transmutation of LLFPs using various types of HTGRs」 K. Kora, et al., Nuclear Eng. Des. 300 (2016) 330-338.

Features of HTGR for nuclear transmutation

The number of LLFP atom N_{LLFP} （Initial： N^0_{LLFP} ）

$$N_{LLFP}(t) \cong N^0_{LLFP} \exp\left[-(\lambda+\sigma\phi)t\right]$$

（1） In generally

We consider how to increase the reactivity $\sigma\phi$

for short term nuclear transmutation

（2） HTGR (our opinion)

We can increase initial loading N^0_{LLFP}
and irradiation time t, in addition to the $\sigma\phi$

for long term nuclear transmutation and **storage**

(1)「Study on transmutation of long-lived fission product using high-temperature gas-cooled reactors」, H. Nakaya, et al., Proc. of GLOBAL2011, Paper No.391363 (2011).

Required conditions to be an effective production

- Established (reliable) technology
  From the viewpoint of cost and development period
  (to draw up a scenario to progress fusion system development)

- having an additional functions except tritium production

- It is desirable to use the system with an minimum modification

In the HTGR, the BP is usually used in a solid state (i.e., as B_4C), and thus the Li compound can be loaded into the reactor's core without significantly changing the original structural design.

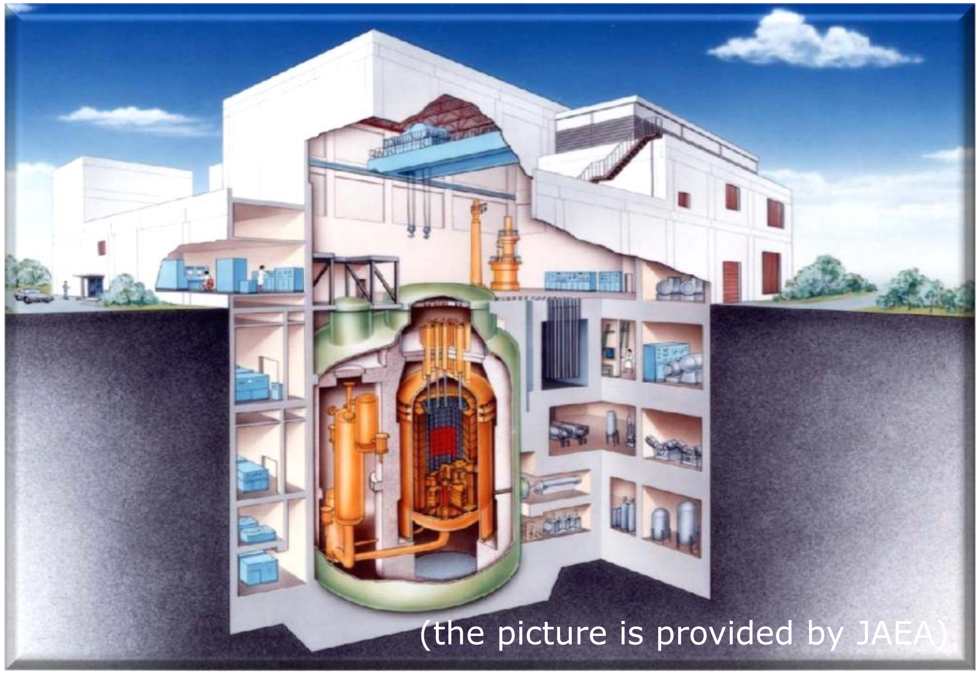
We are proposing a tritium production using high-temperature gas-cooled reactors

H. Matsuura, et al., Nucl. Eng. Des. 243 (2012) 95.
H. Matsuura, J. Plasma Fusion Res. 93 (2017) 457. [exposition, in Japanese]
H. Matsuura, et al., ATOMOΣ, 60 (2018) 567. [topics, in Japanese]

Development of gas-cooled reactor in Japan

- High efficiency electric-power generation using the ***gas turbine***.
- High temperature ***heat source***.

HTTR (High Temperature Engineering Test Reactor)
…… operated by Japan Atomic Energy Agency (JAEA)


(the picture is provided by JAEA)

Thermal Output	:	30 MW
Moderator	:	graphite
Coolant	:	helium gas

1991.03	Beginning of Construction
1998.11	First Criticality
2004.04	30MW thermal output 950 ℃ outlet coolant temperature
2007.05	30days continuous operation

⇨ conceptual design study for commercial-base gas-cooled reactor
GTHTR300 (Gas Turbine High Temperature Reactor of 300MWe)
(X.YAN, et al., Nucl. Eng. Des., **222**, 247 (2003).)

Merit of gas-cooled reactors for tritium production

1 Reactor Structure

Moderator：Graphite
Coolant：He

Graphite is chemically stable and does not react with the Li compound

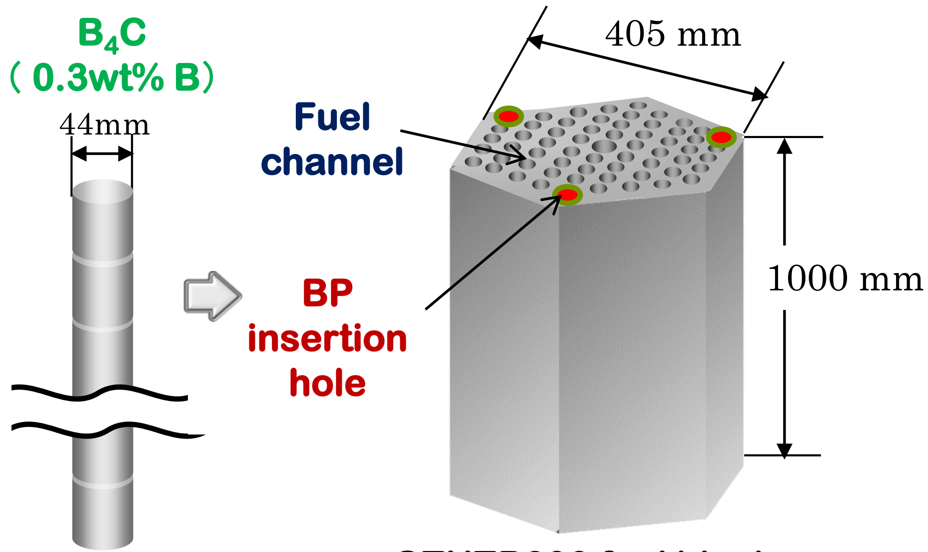
2 Use of solid BP (Burnable Poison)

- **Large mean-free path**

A lot of flexibility in the design

- **SOLID BP (B_4C)**

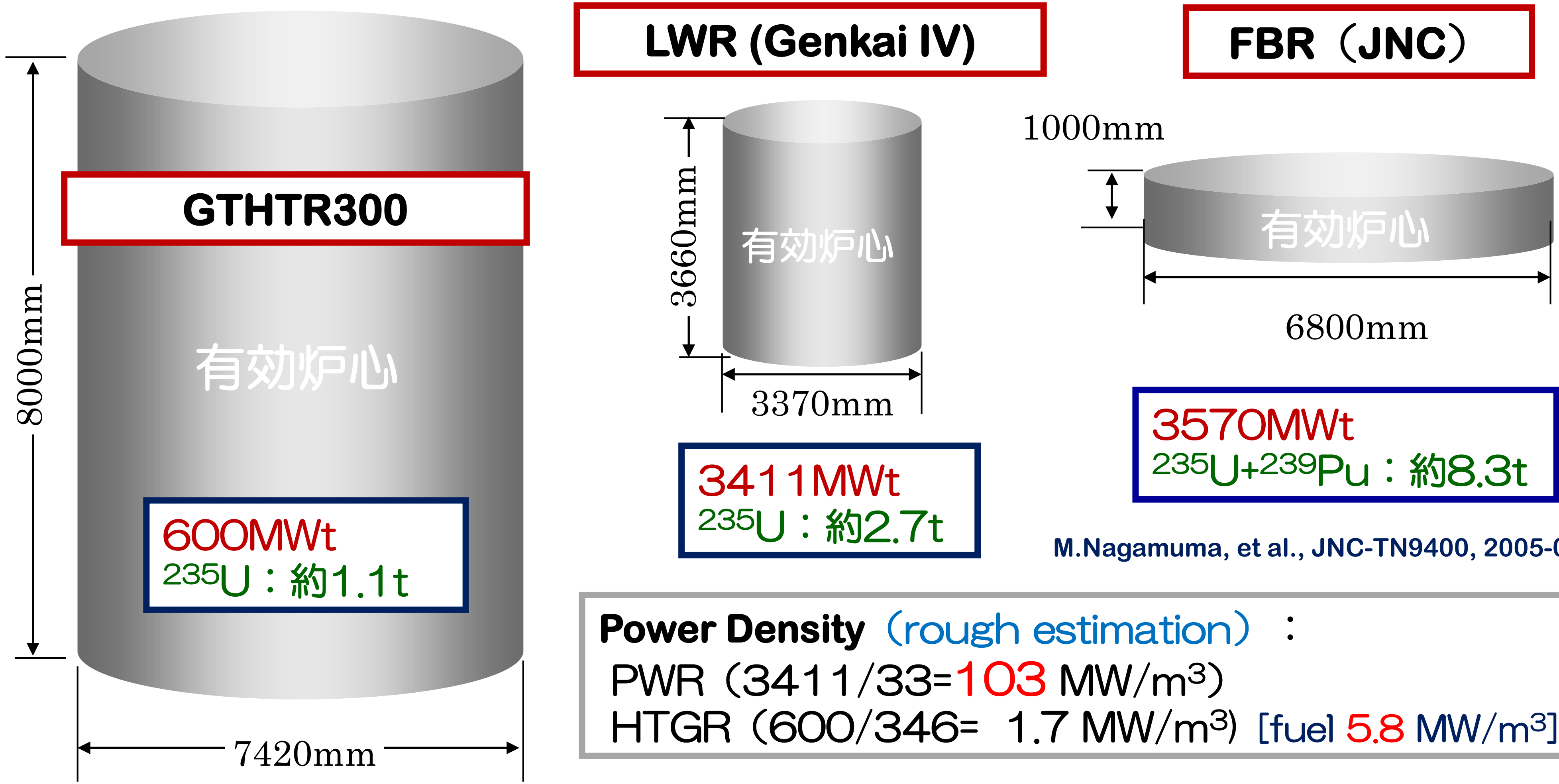
Replacement of ^{10}B with ^{6}Li is not difficult



GTHTR300 fuel block

→ **Li compound can be loaded into the reactor's core without significantly changing the original structural design.**

Large space to load a large amount of Li compounds



M.Nagamuma, et al., JNC-TN9400, 2005-051

Power Density （rough estimation）：

PWR（3411/33=103 MW/m^3）

HTGR（600/346= 1.7 MW/m^3） [fuel] 5.8 MW/m^3]

The unique structure of the reactor to reduce the power (heat) density and to contain fission products provides sufficient space to load a large amount of Li compounds almost uniformly close to the ^{235}U fuel region with large surface area per unit volume.

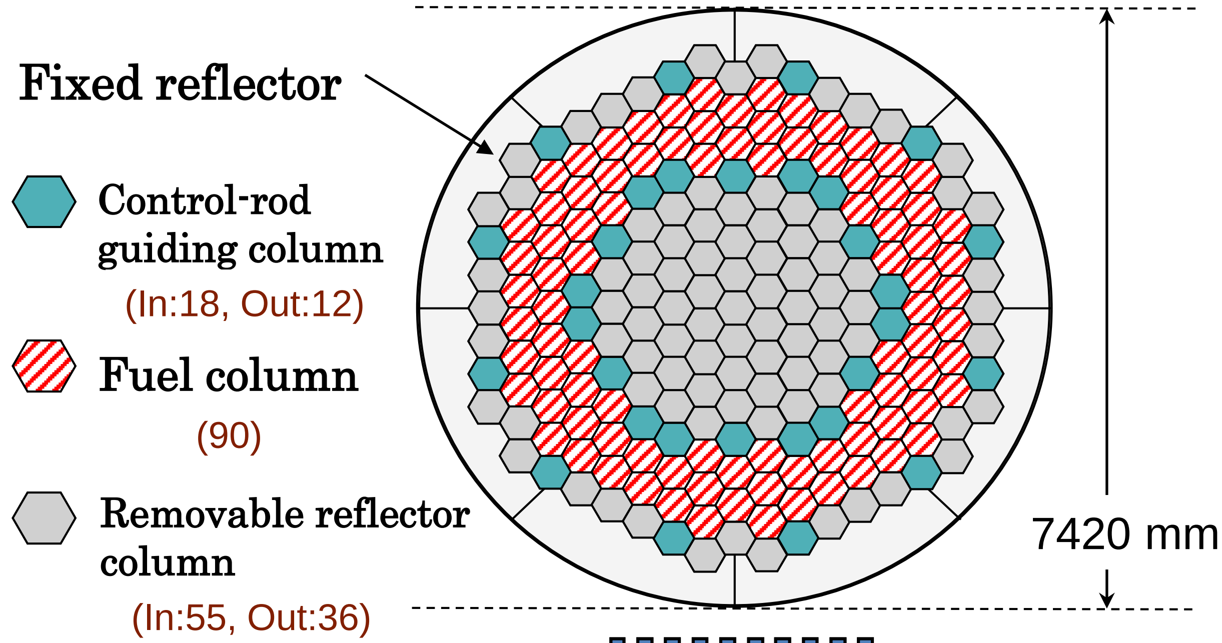
Core configuration

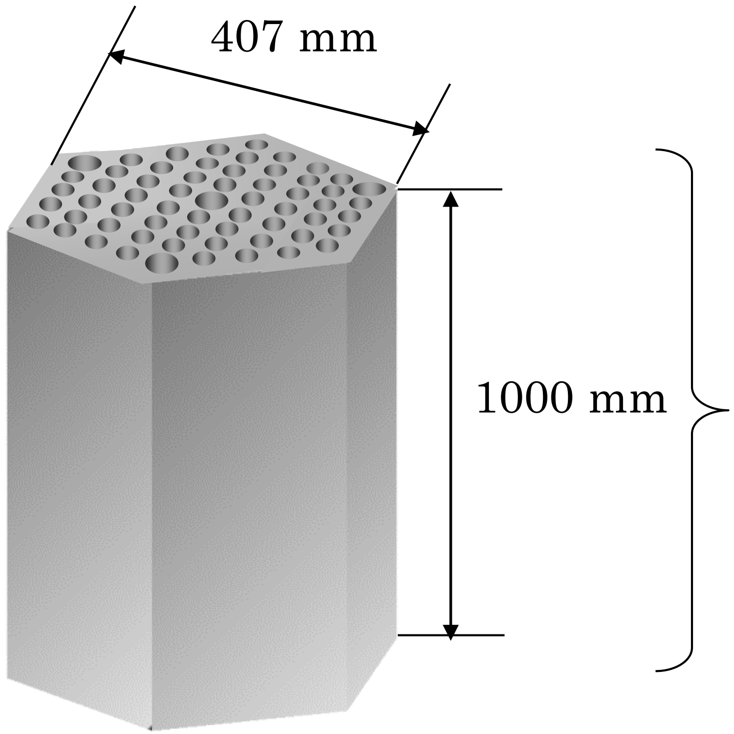
GTHTR300

(X.YAN, et al., Nucl. Eng. Des.,**222**, 247 (2003).)

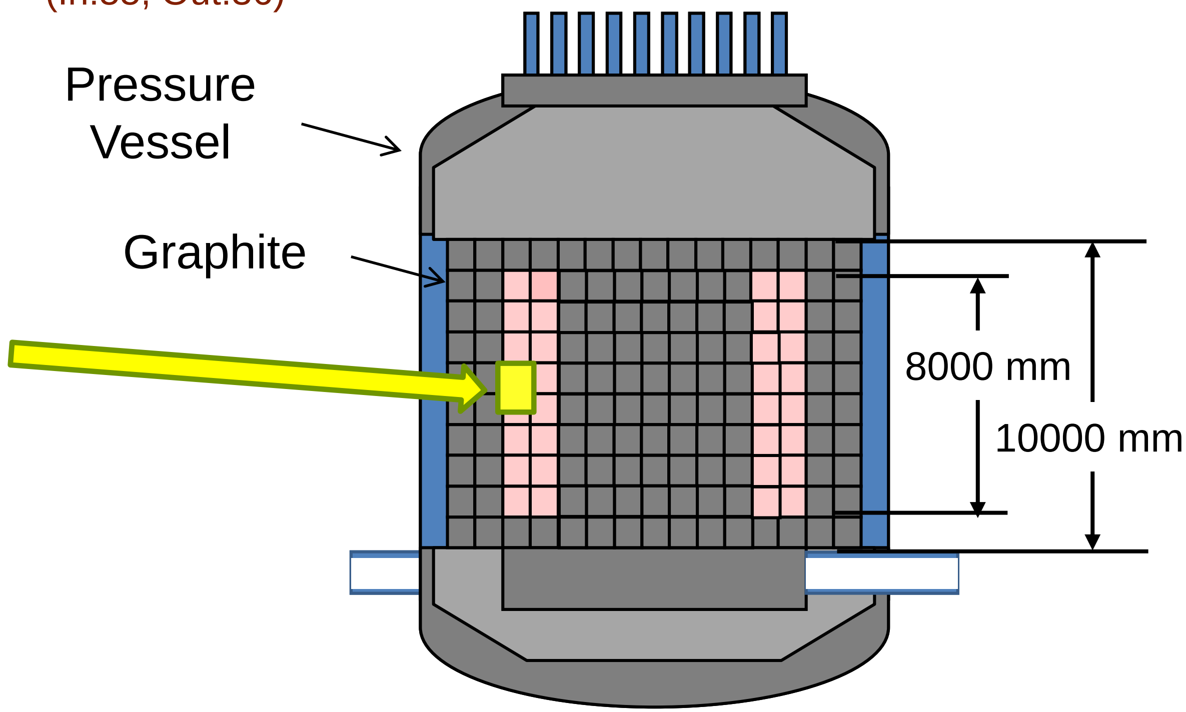
Thermal Output	: 600 MWt
^{235}U enrichment	: 14 wt%
Fuel Temp.	: 1350 K
Moderator Temp.	: 1200 K

Horizontal cross section of GTHTR300 core.



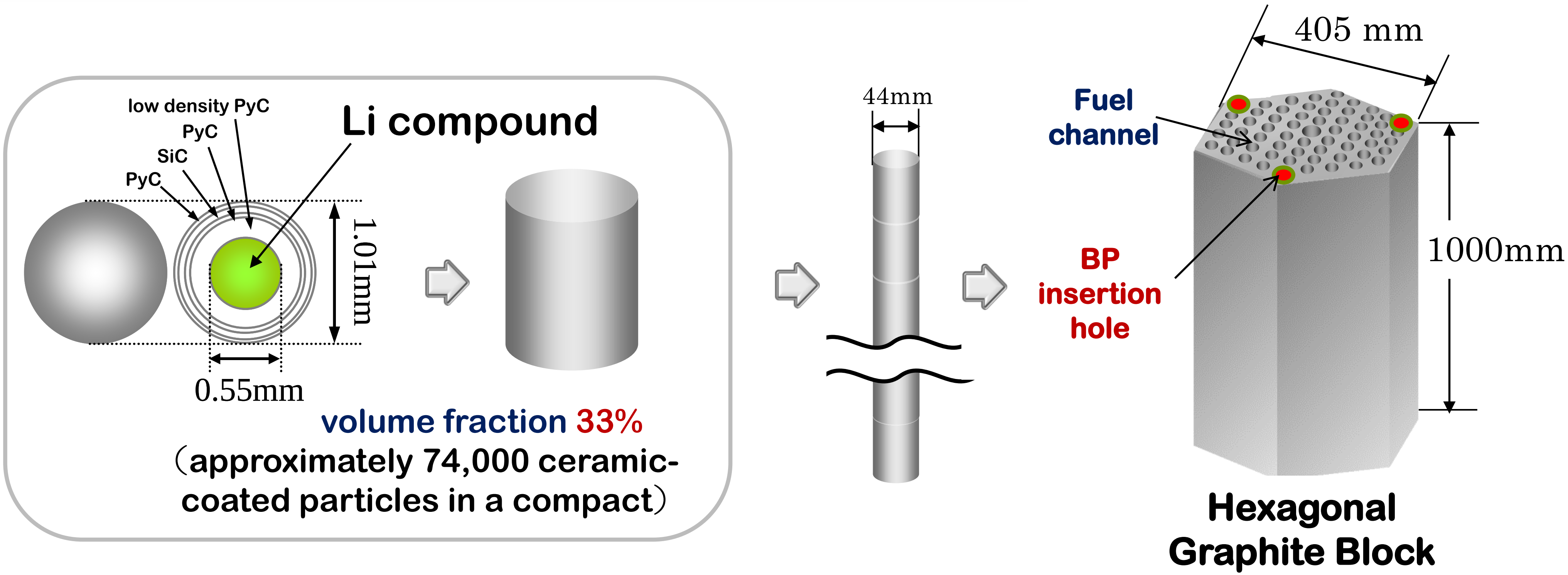


Hexagonal Graphite Block

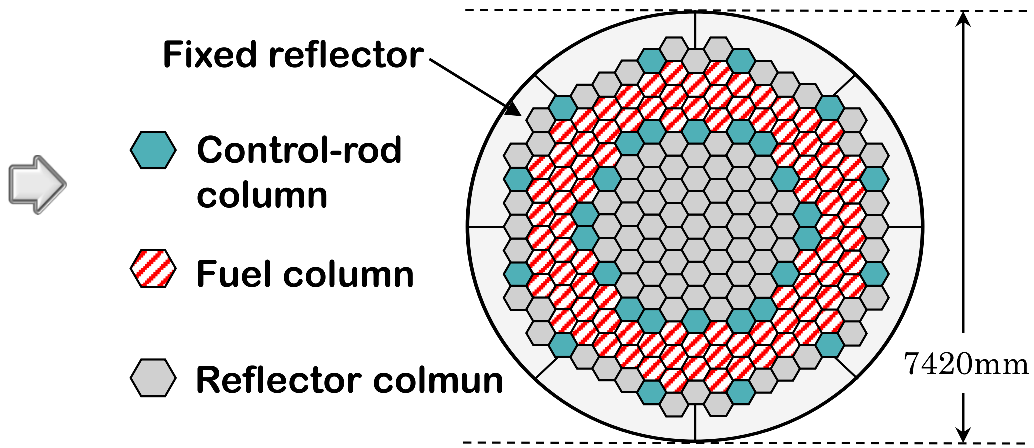


Vertical cross section of GTHTR300 core.

Li-loading rod（by using Li particles）



GTHTR300



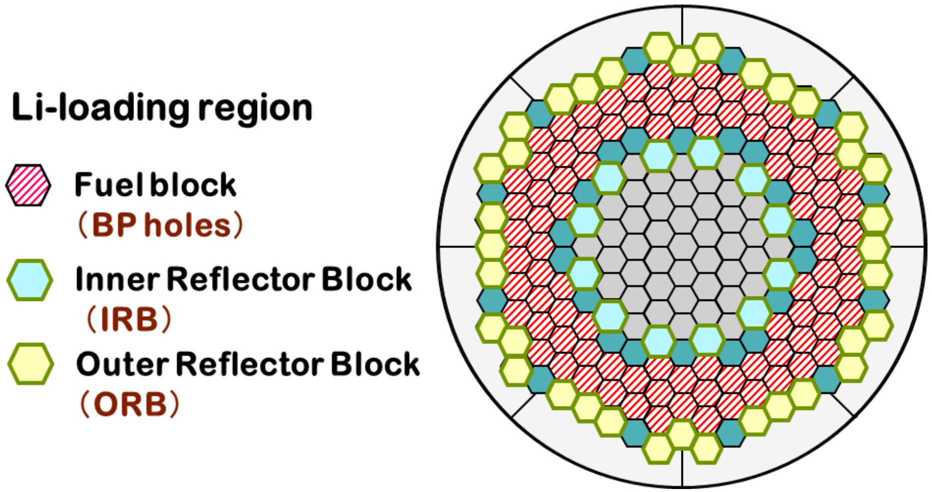
Tritium production using excess neutrons by inserting a Li compound as a burnable poison (BP) instead of a boron compound

Tritium production using ceramic-coated Li particles

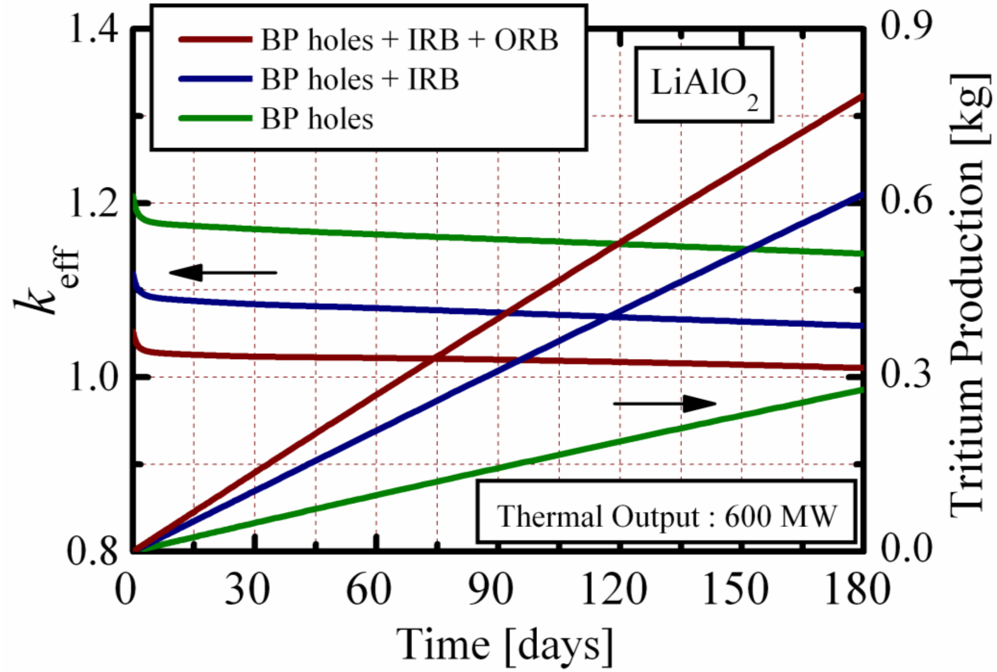
BP holes ＋ IRB: inner reflector block / ORB: outer reflector block

「$LiAlO_2$ compound is assumed」

「standard ceramic-coated particles used in GTHTR300」

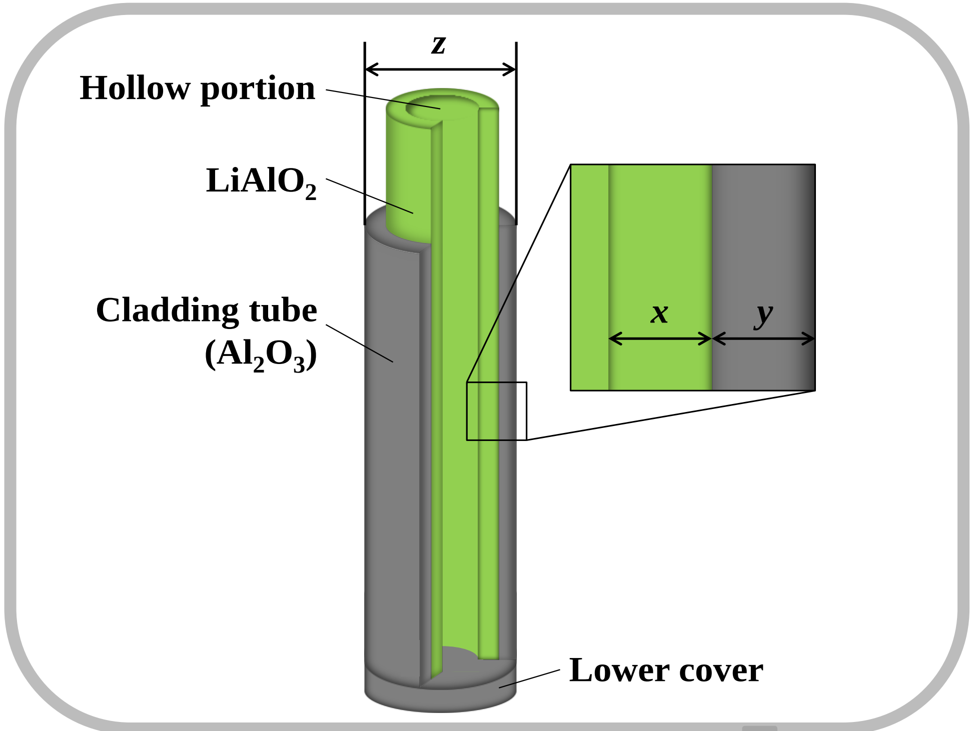


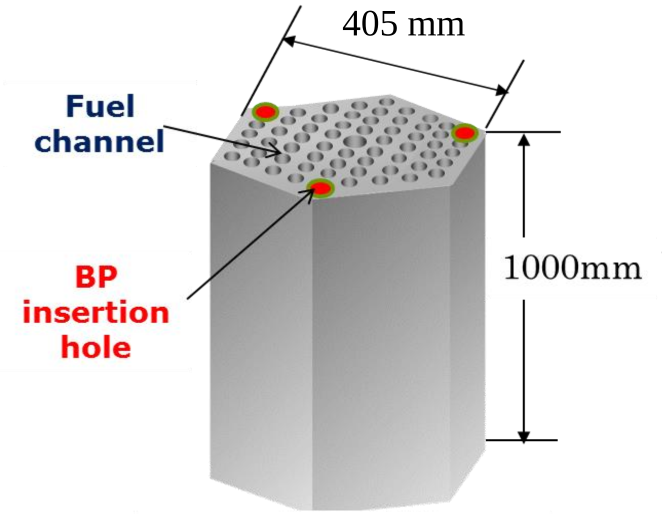
GTHTR300 horizontal cross section



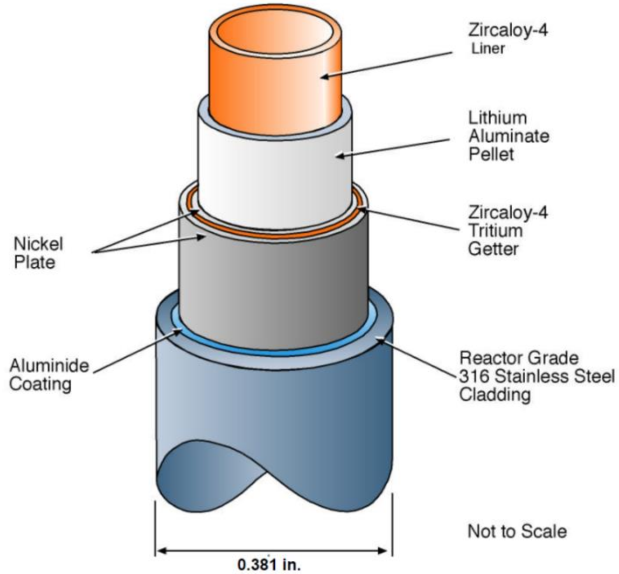
- 「only BP holes」
  → ～280 g of tritium can be produced（600 MWt、180 days）
- 「BP holes」 ＋ 「IRB」 ＋「ORB」
  → ～700 g of tritium can be produced（600 MWt、180 days

Li-loading rod（by using columnar Li compound）





Hexagonal Graphite Block



TPBAR (Tritium Producing Burnable Absorber Rod)[1]

[1] Recommendations for Tritium Science and Technology Research and Development in Support of the Tritium Readiness Campaign, TTP-7-084, PNNL-22873 (2013)

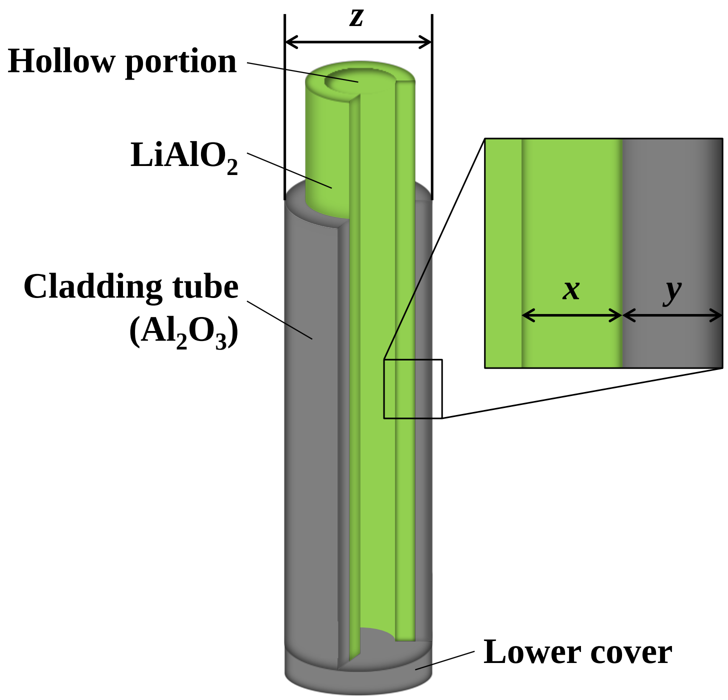
(1) **Reduction of inner pressure by hollow portion**

(2) **Increased Li-loading by columnar Li compound**

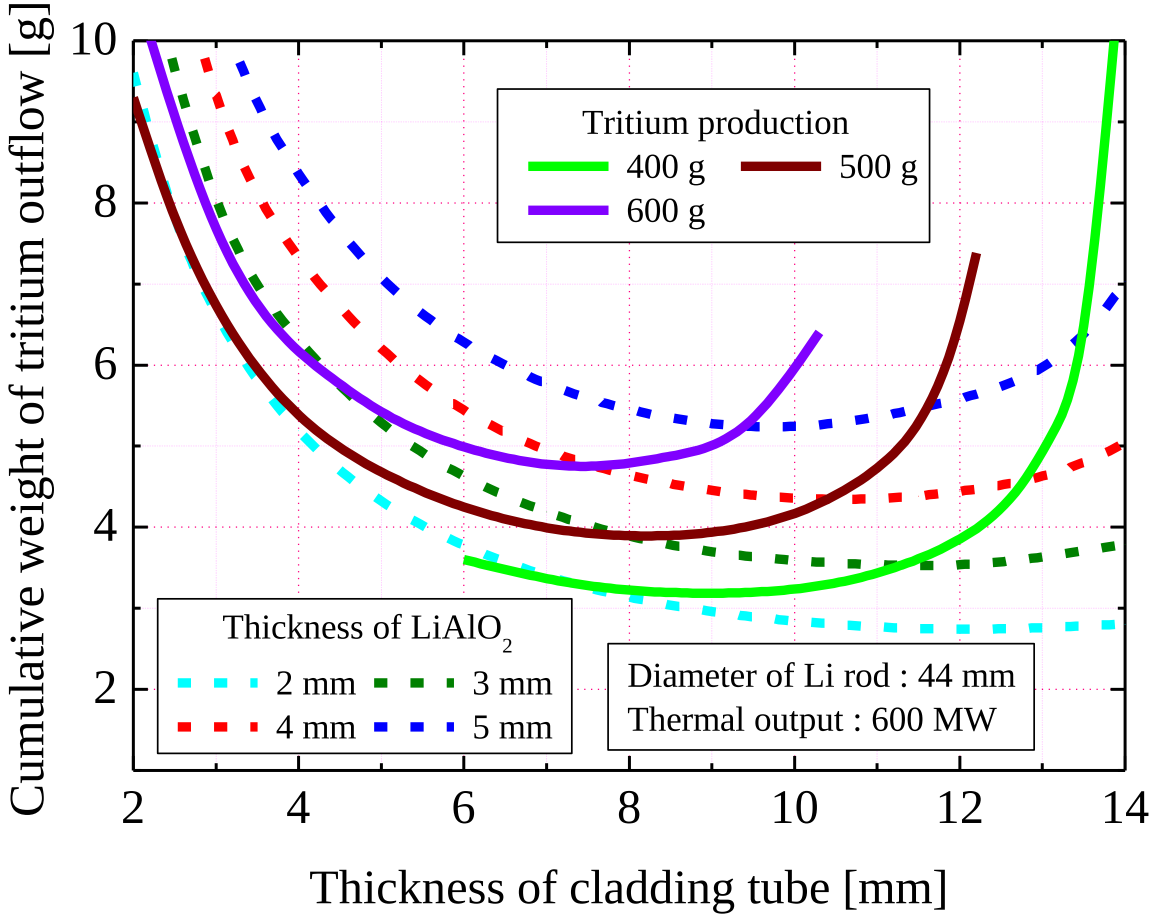
(3) **Decreased out flow rate by decreased surface area**

T production and flow out from rods

Rod Temp.：800 K



GTHTR300 is assumed



- 180-day operatio
- Diameter of BP hole is fixed as 44 mm

continuous-energy Monte Carlo transport Code MVP-BURN

Diffusion coefficient is taken from
Katayama, et al., Fusion Sci. Tech. (2015)

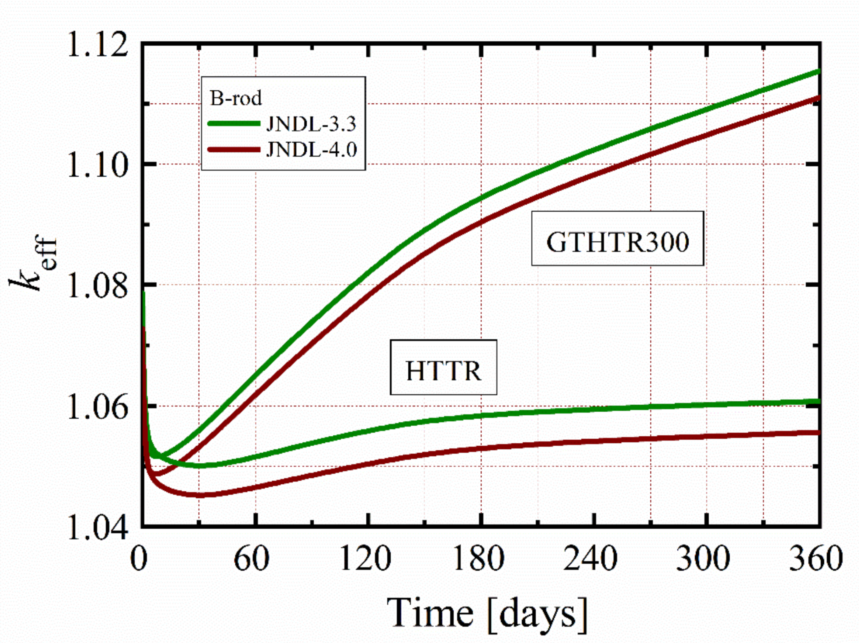
T production： 600～900 g/year

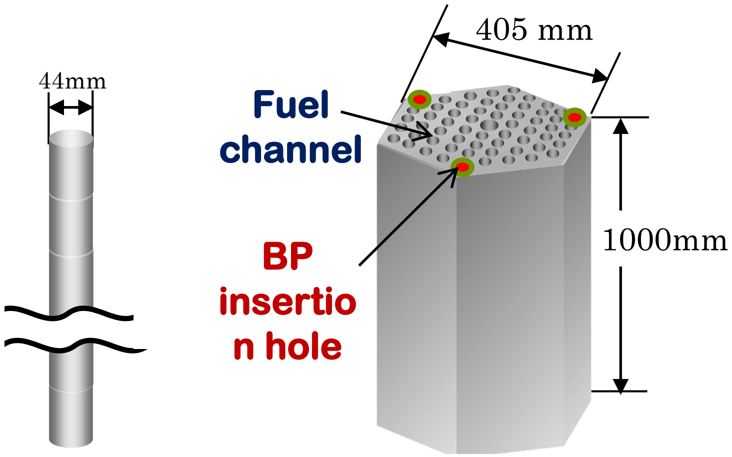
T flown out from rods：1 % of production

Nakaya, et al., Nucl. Eng. Des. (2015)

Influence of Nuclear Data on T production

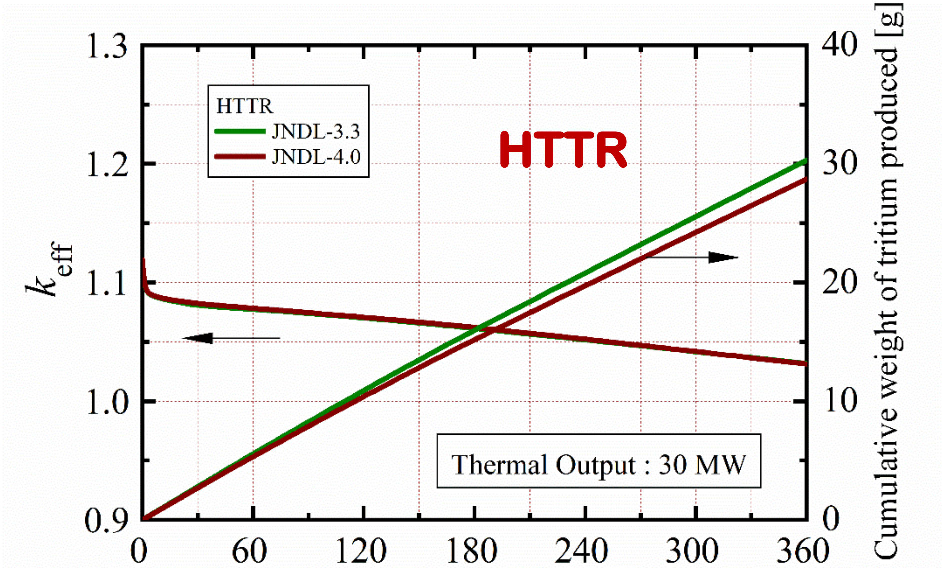
Standard Simulation
(B_4C is assumed as BP)

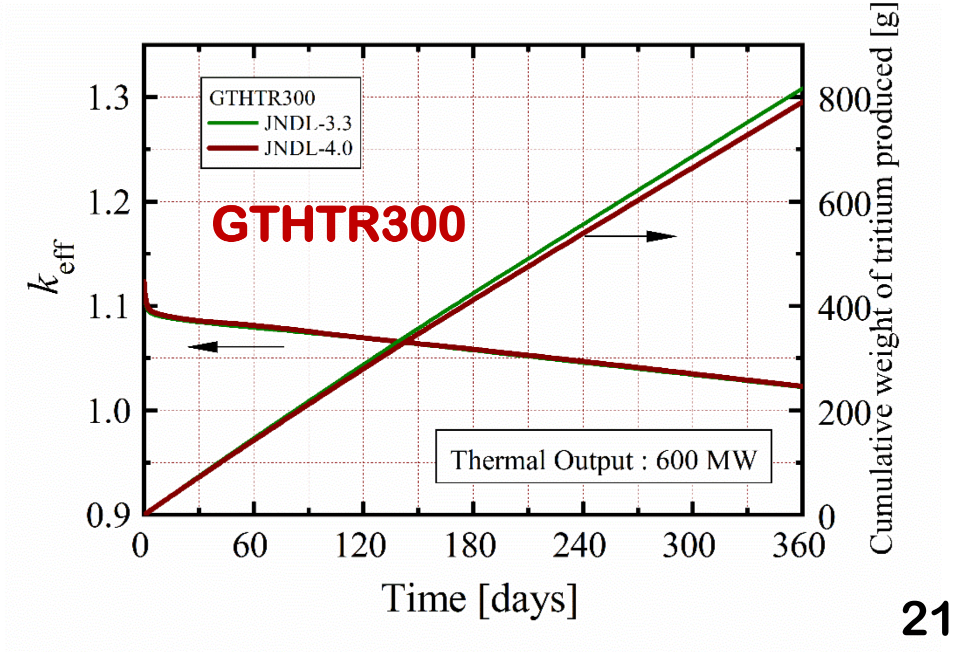




Hexagonal Graphite Block

Reactors for T-production
($LiAlO_2$ is assumed as BP compound)



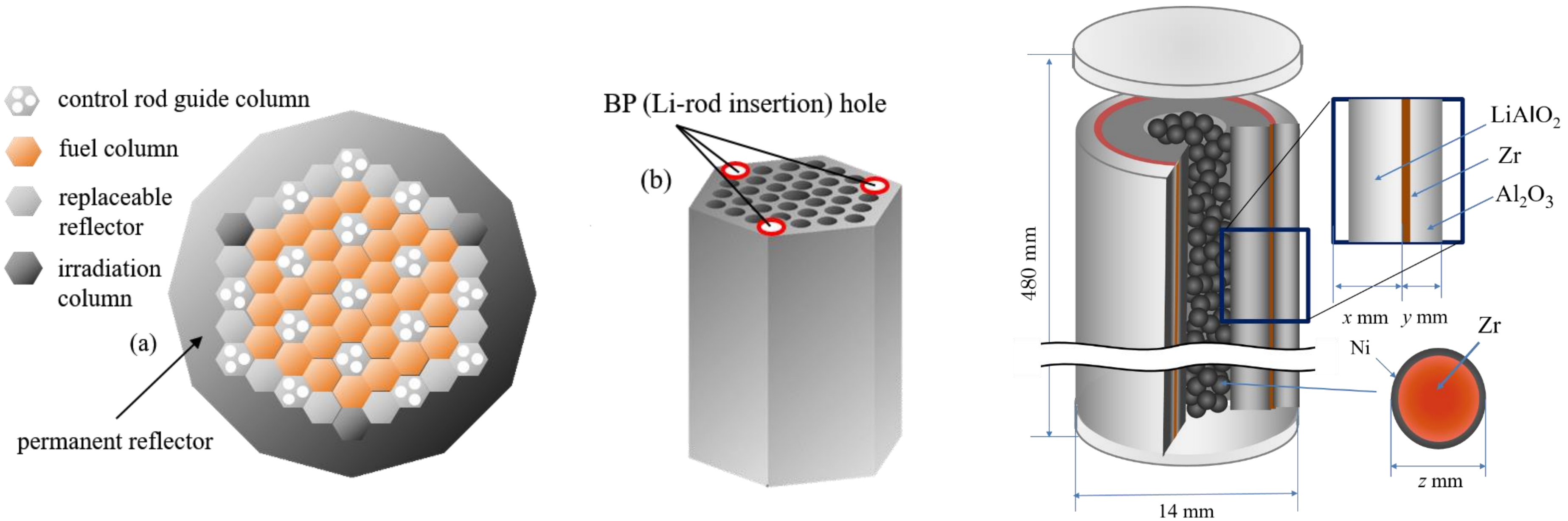


New Li rod using Zr

- Numerical simulations have predicted that if we could operate the HTGR in a low temperature range, keeping the rod temperature below 520 ℃, the tritium leaking from the Li rod can be suppressed to less than 1% of the amount produced

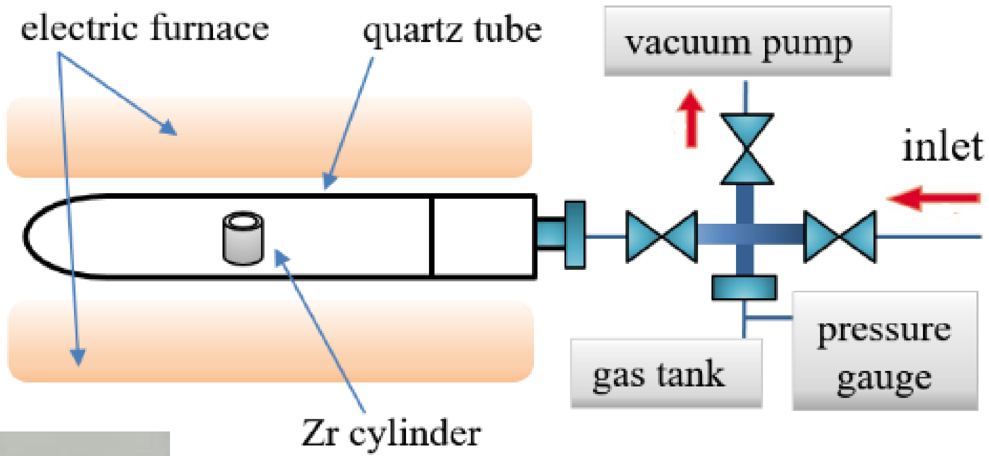
- However, if we intended to operate the HTGR in a much higher temperature range (i.e., the rod temperature reaching 800 ℃ –900 ℃) so as to increase the electricity generation efficiency, the leakage of the tritium would rapidly increase.

New rod structure using Zr has been proposed



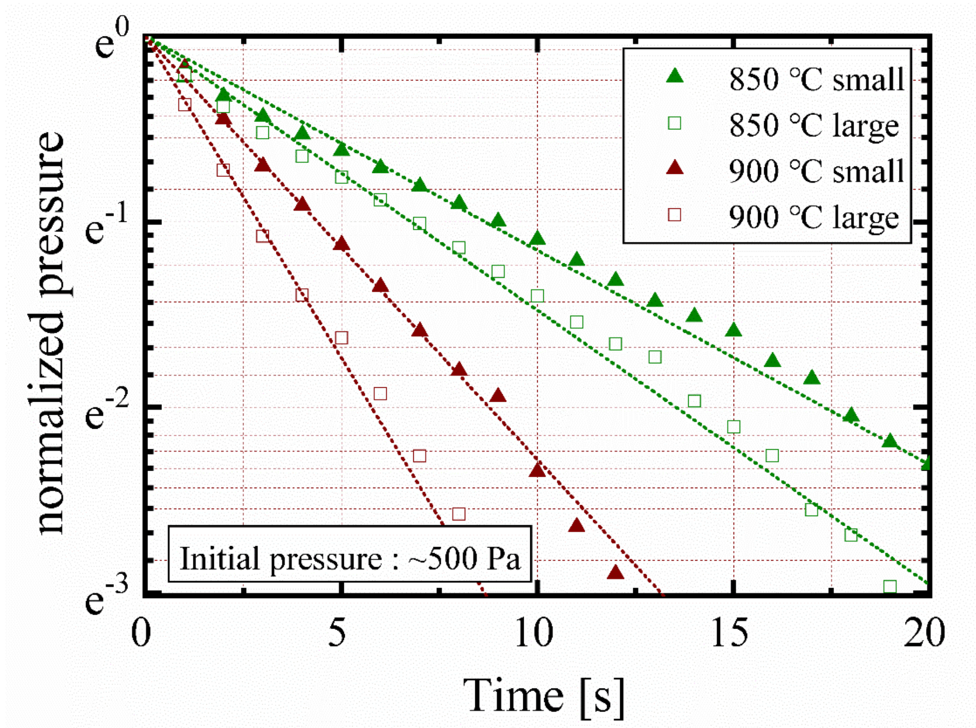
Matsuura, et al., presented at SOFT2018

Measurement of H-absorbing speed



	Small	Large
Diameter (mm)	9.5	15.8
Height (mm)	15	15
Thickness (mm)	1	1
Impurity		
C (wt%)	0.015	
FeCr (wt%)	0.083	
H (wt%)	< 0.0003	
N (wt%)	0.006	
O (wt%)	0.143	
Zr+Hf (wt%)	99.5	

Matsuura, et al., Fusing Eng. Des. (2019)



$$\hat{P} \propto \exp\left(-t/\tau_a\right) + P_{equilibrium}$$

$$\tau_a^{large} = 8.7\ (4.4)\ \mathrm{s}\ ,$$

$$\tau_a^{small} = 6.8\ (2.9)\ \mathrm{s}\ \text{for } 850\ (900)°\mathrm{C}$$

$$\tau_a^* \equiv \tau_a^{small} A^{small} / V_{ex} \approx \tau_a^{large} A^{large} / V_{ex}$$

Analysis model for tritium outflow from Zr-included Li-loading rod

Tritium density in the hollow portion

Matsuura, et al., presented at SOFT2018

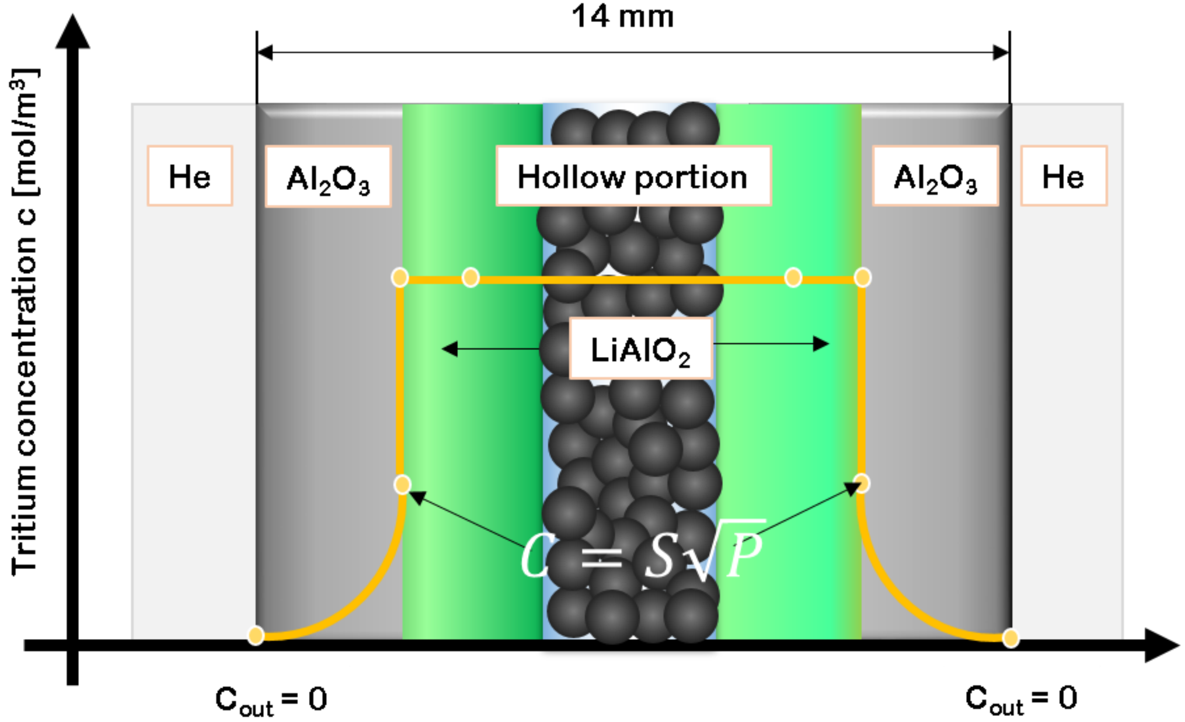
$$\frac{dc}{dt} = \frac{S_T^{rod}}{V_{hp+Li(15)}} - \frac{cA_{pebble}}{\tau_a^* V_{hp+Li(15)}} + \frac{AD}{V_{hp+Li(15)}} \left.\frac{\partial c}{\partial r}\right|_{r=7\,\mathrm{mm}}$$

Tritium diffusion equation in Al_2O_3 region

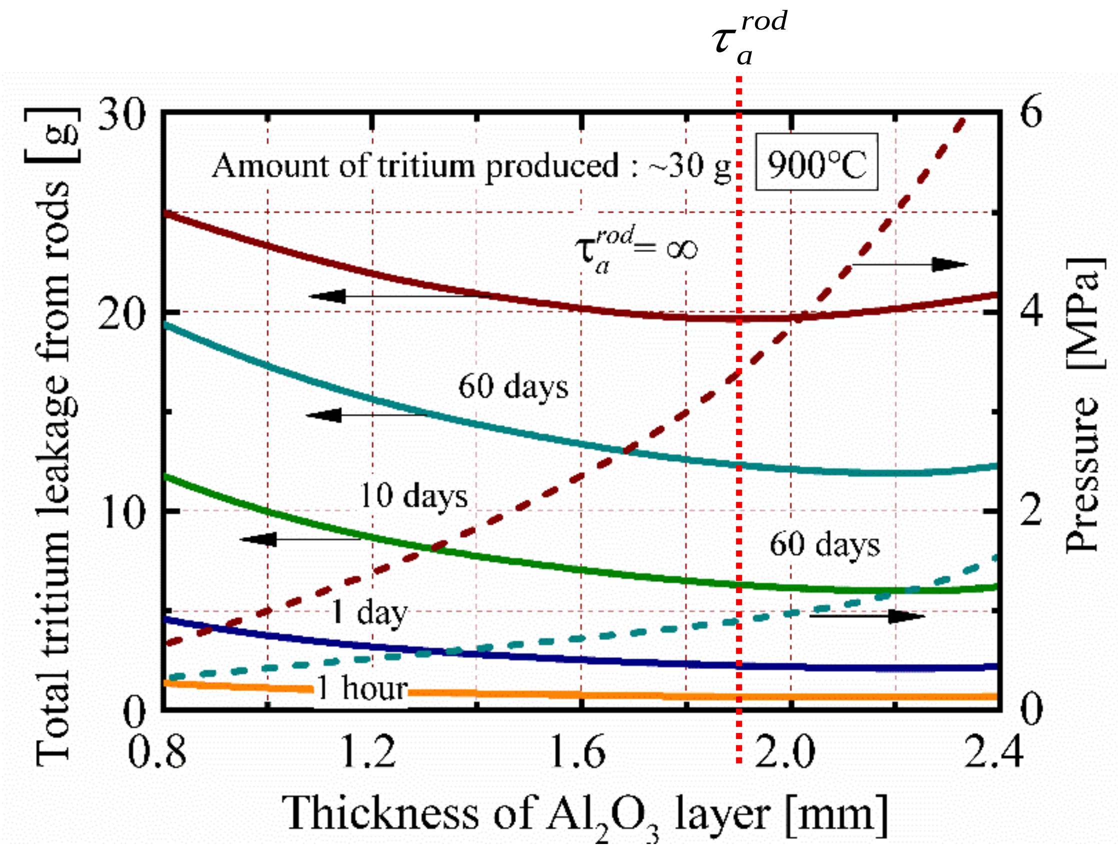
$$\frac{\partial c}{\partial t} = D\nabla^2 c$$

Tritium outflow into He region

$$J = -AD \left.\frac{\partial c}{\partial r}\right|_{r=7\,\mathrm{mm}}$$



Tritium outflow from Li rod with Zr layer

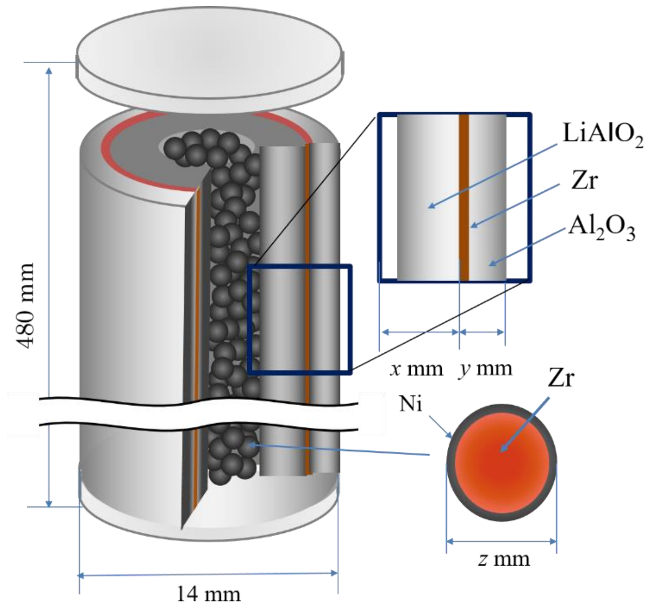


Rod Temp. : 900 ℃

Thickness of $LiAlO_2$ layer is adjusted so that the tritium production is kept as 30 g

Thickness of Zr layer is fixed as 0.1mm

τ_a^{rod} : tritium adpsoption time when thickness of Al_2O_3 layer is 1.9 mm



- $\tau_a^{rod} = \infty$ （w/o Zr） : ~2/3 of tritium is flown out
- $\tau_a^{rod} = 1$ day : tritium outflow is ～2 g

（Experimet : τ_a^{rod} =~ 1.2 ms ）

Comparison with other reactor types

HTGR

about 1.5-2.4 kg T/GWe-y

(when $LiALO_2$ with columnar rods are used in GTHTR300)

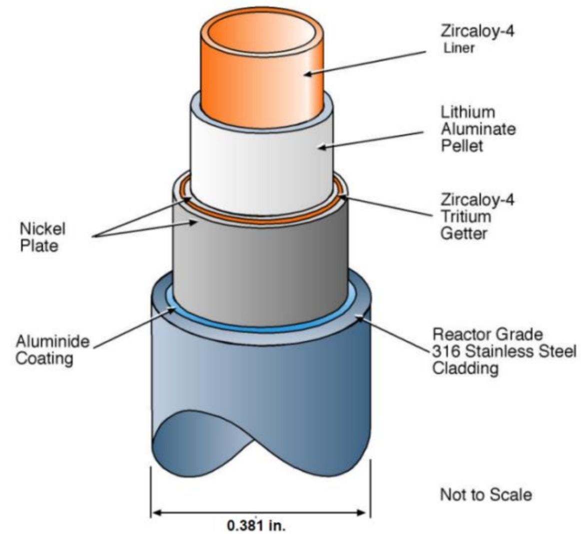
500-800 g/y by GTHTR300

LWR

Watts Bar Nuclear Power Plant (Tennessee Valley Authority)

about 1.0-1.6 kg T/GWe-y

(may not be maximum value)



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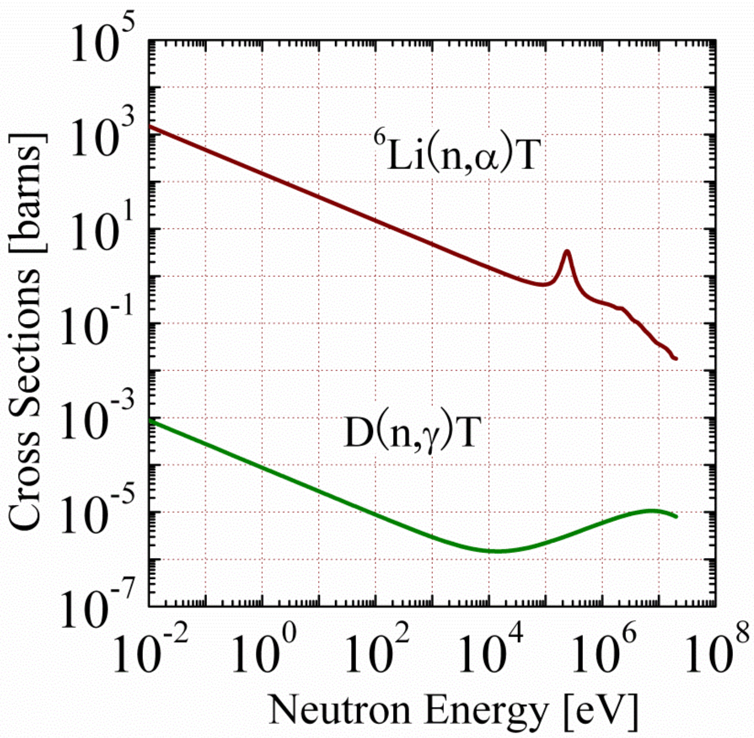
CANDU reactor

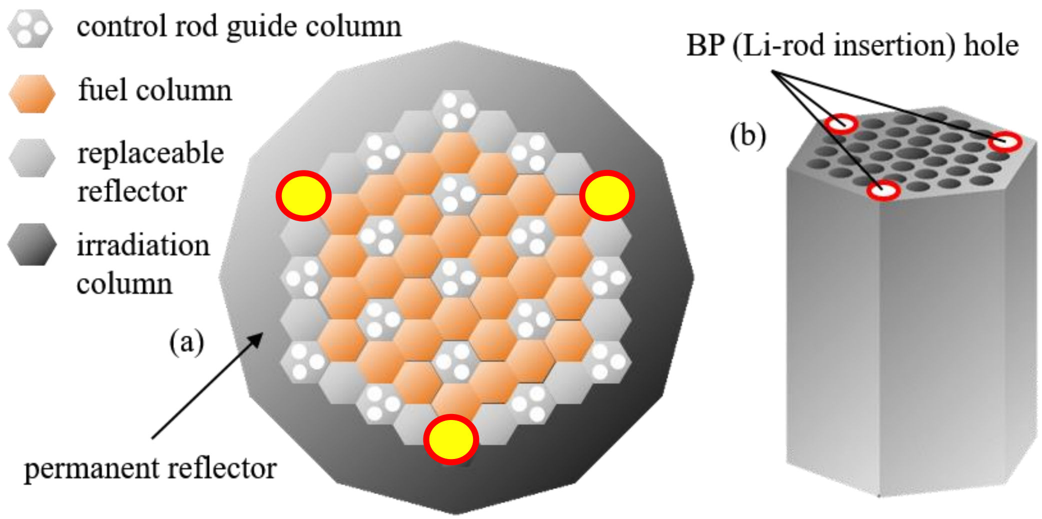
about 0.17-0.23 kg T/GWe-y[3]

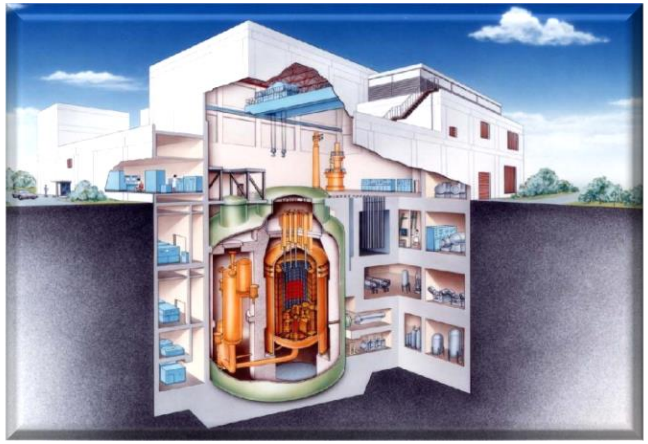


[2] K.Y. Wong et al., Canadian Tritium Experience, Ontario Hydro report (1984).

[3] P. Gierszewski, “Tritium Supply for Near-Term Fusion Devices”, FED 10 (1989) 399

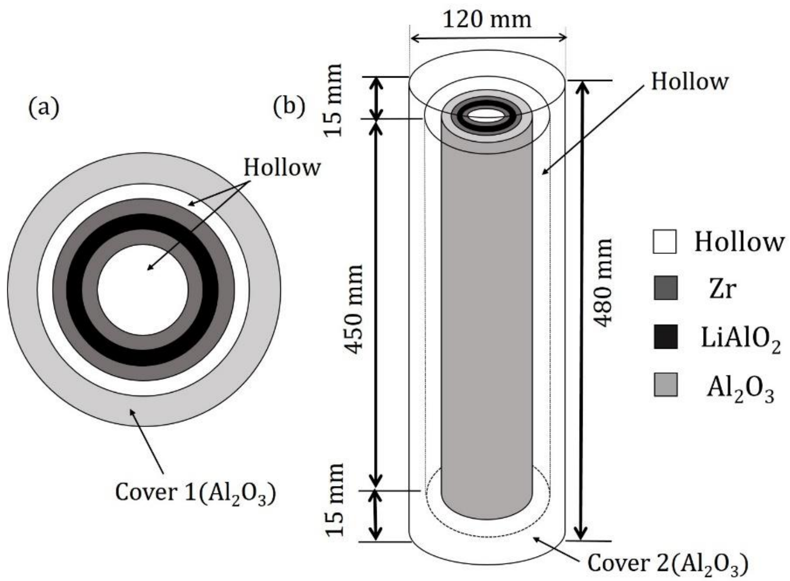
Demonstration test using HTTR



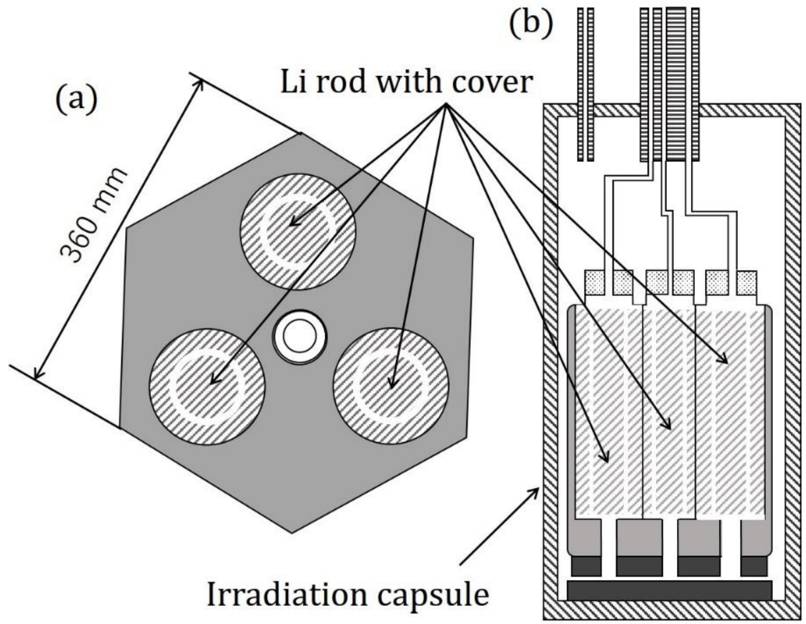


Temp. : 400～800℃
Neutron Flus : ～7×10^{17} n/m^{2}s
Licensed T production: less than 0.6 g/y (batch operation)

Li-rod structure



Schematic view of irradiation capsule



Summary

- For prior technical tests of tritium circulation and blanket systems of a demonstration fusion reactor and for the startup operation, it is necessary to provide a sufficient amount of tritium from an outside source.

- We have proposed the tritium production using the HTGR by inserting a Li compound as a burnable poison (BP) instead of a boron compound.

- In Japan, the HTGR is one of the most promising method to produce more than 100 g orders of tritium.

- The technical understanding of the Li-loading rod structure, together with the evaluation of the basic properties of the structural materials of the rod (i.e., Al_2O_3 and Zr [4-8]), has been progressed to efficiently produce a sufficient amount of tritium and to hold the produced tritium inside of the rod during the reactor's operation period.

- In order to confirm the performance of the Li-loading rod and to demonstrate the tritium production process using the HTGR, we now plan an irradiation test on a high-temperature engineering test reactor (HTTR).