Shielding analysis for the In-Flight Fragment target facility of the RAON complex in Korea

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@ Institute for Basic Science

Contents

Analysis code system

Analysis	Code	Reaction model	Library
Source term	PHITS 2.64	JQMD + JAM + INCL	JENDL-HE 2007, ENDF/B VI
Bulk shield thickness	MCNPX 2.7	Bertini	JENDL-HE 2007, ENDF/B VI
Shielding design for prompt radiation field	MCNPX 2.7 + Advantg 3.1	Bertini	JENDL-HE 2007, ENDF/B VI
Activation analysis	PHITS 2.64 + MCNPX 2.7 + DCHAIN/SP(2011) or FISPACT*	JQMD + JAM + INCL or Bertini	JENDL-HE 2007, ENDF/B VI EAF2010

*FISPACT: modified version based on the FISPACT2010 for spallation reaction analysis (Chang Ho Shin, Hanyang univ.)

Evaluation of source term for shielding design and analysis

1) Evaluation of the charged particle distribution

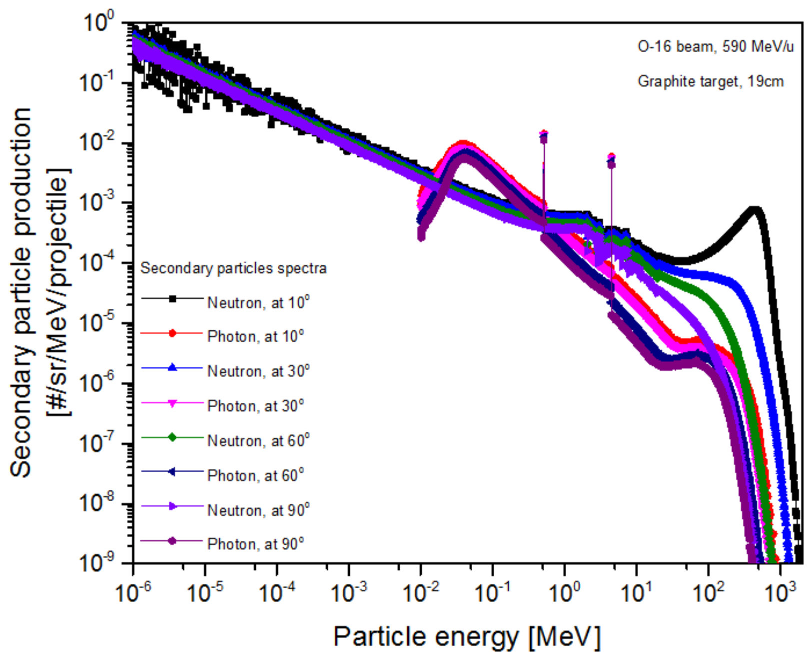
- Stopping rage of the energetic primary beam was calculated by SRIM code.
  - The stopping range of the Oxygen beam with 590 MeV/n is 6.84 cm in the iron.
  - The heavier beam particle has the shorter stopping range.

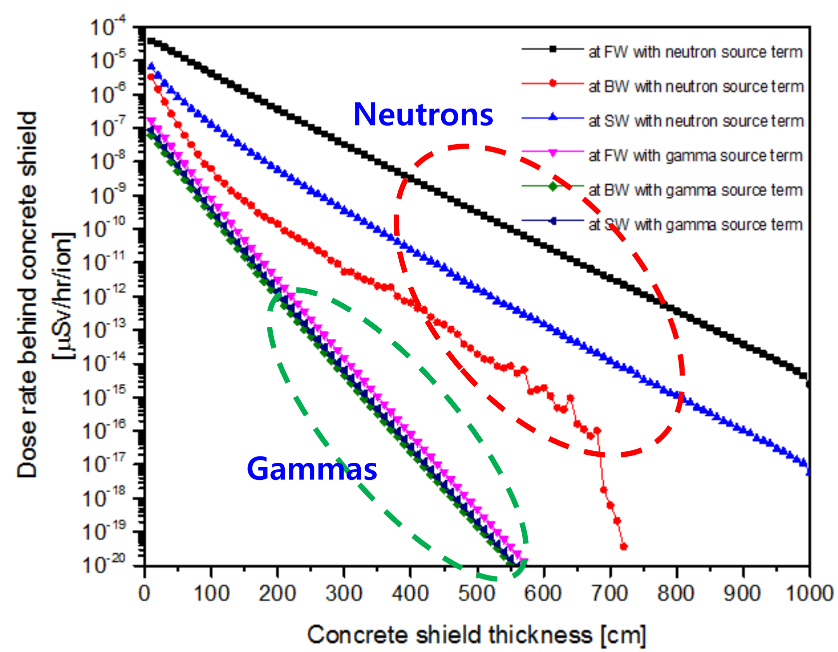
Primary beam	Energy [MeV/n]	Stopping range [cm]
O-16	590	6.84
Ca-48	550	2.97
Kr-86	550	1.64
Sn-112	550	1.12
U-238	400	0.47

Evaluation of source term for shielding design and analysis

2) Comparison of neutron and gamma

- Comparison of the neutron and gamma spectra produced from the IFF target: O-16 beam, 590 MeV/n
- Comparison of contribution to dose rate behind the concrete shield:
- The contribution of gammas to dose rate is less than 0.1%. Therefore the secondary neutrons produced by the primary beam is considered as the source term for the radiation shielding analysis

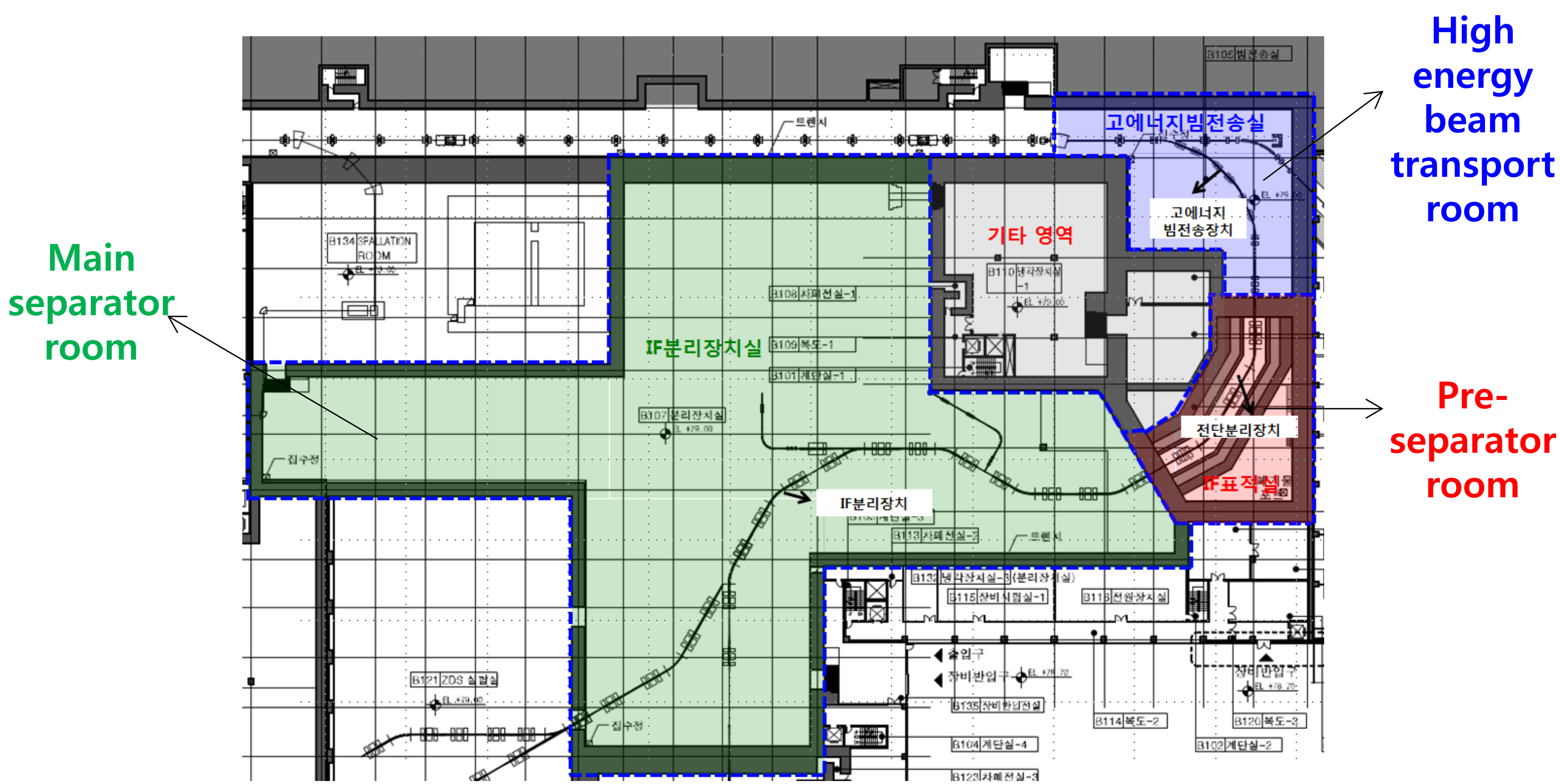




Evaluation of source term for shielding design and analysis

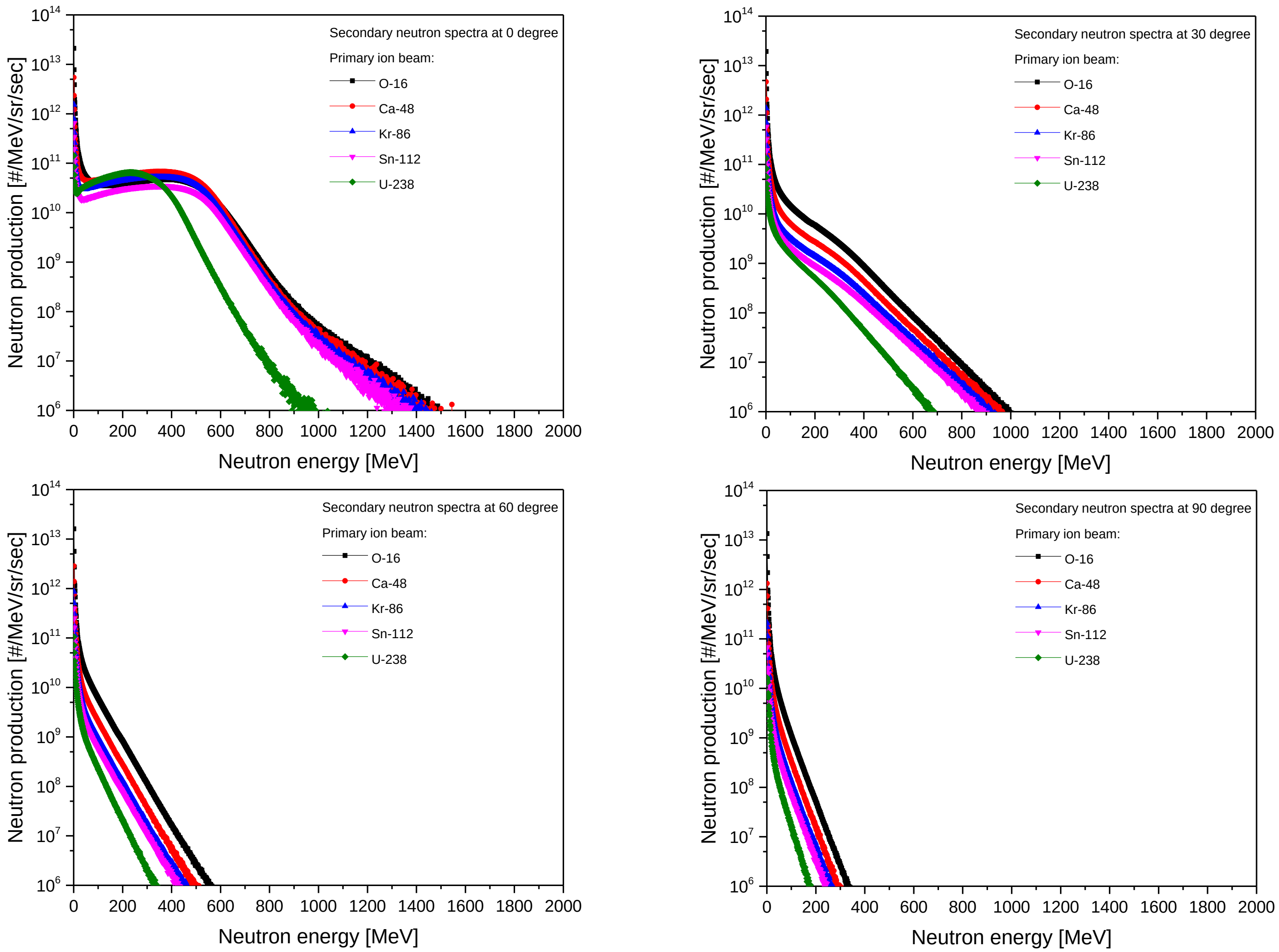
- Neutron source term in each area of the IFF target facility

Area	Neutron production	Material	Beam loss (intensity)
High energy beam transport room	@ beam line	SS beam line	1 W/m beam loss
Pre-separator room	@ target	Graphite target	20% beam loss
	@ beam dump	Water dump	80% beam loss
	@ pre-separator	Pre-separator material	Full beam loss (100%)
Main separator room	@ main separator	Magnet material	Isotope loss data



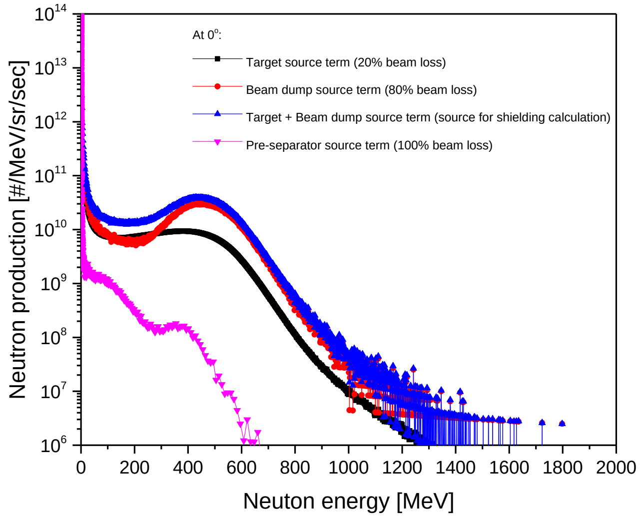
Evaluated neutron source terms

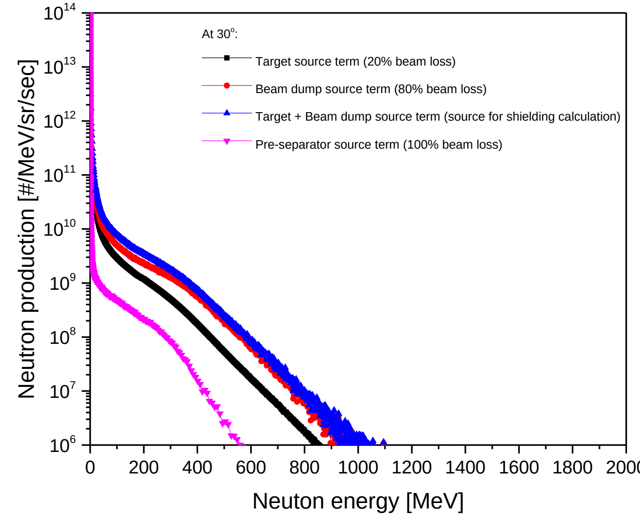
- Neutron spectra produced at each angle
  - It is expected that O-16 beam produces most conservative source term.

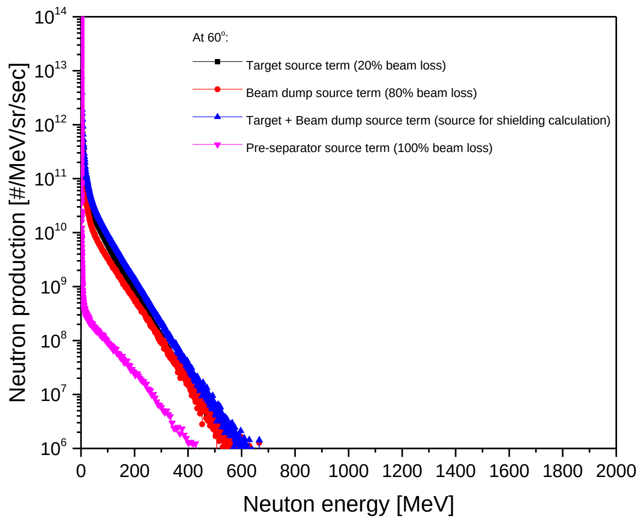


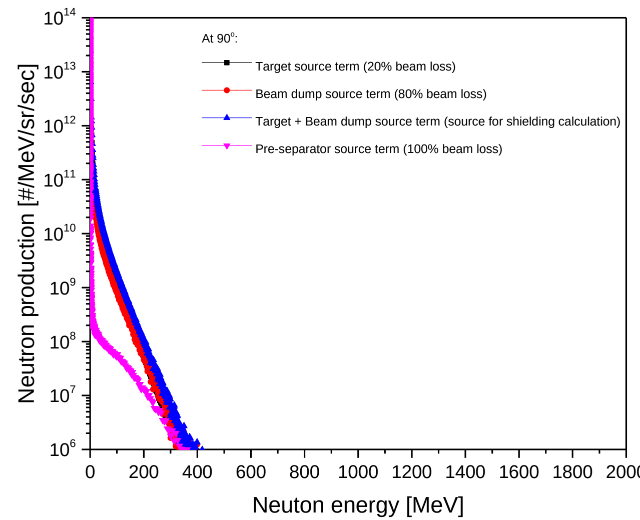
Evaluated neutron source terms

- Neutron spectra at each beam loss point
  - At Target, dump and pre-separator



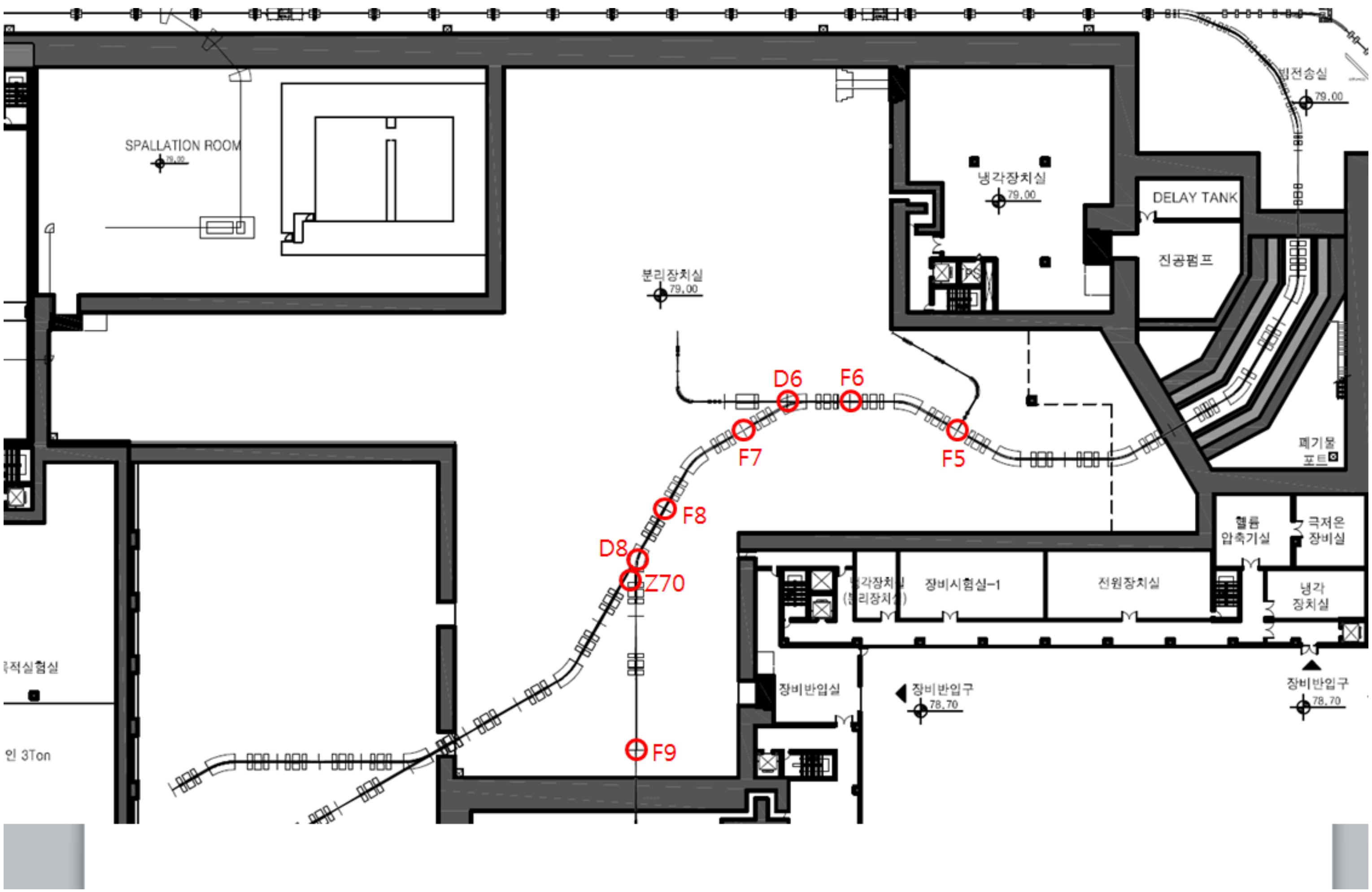






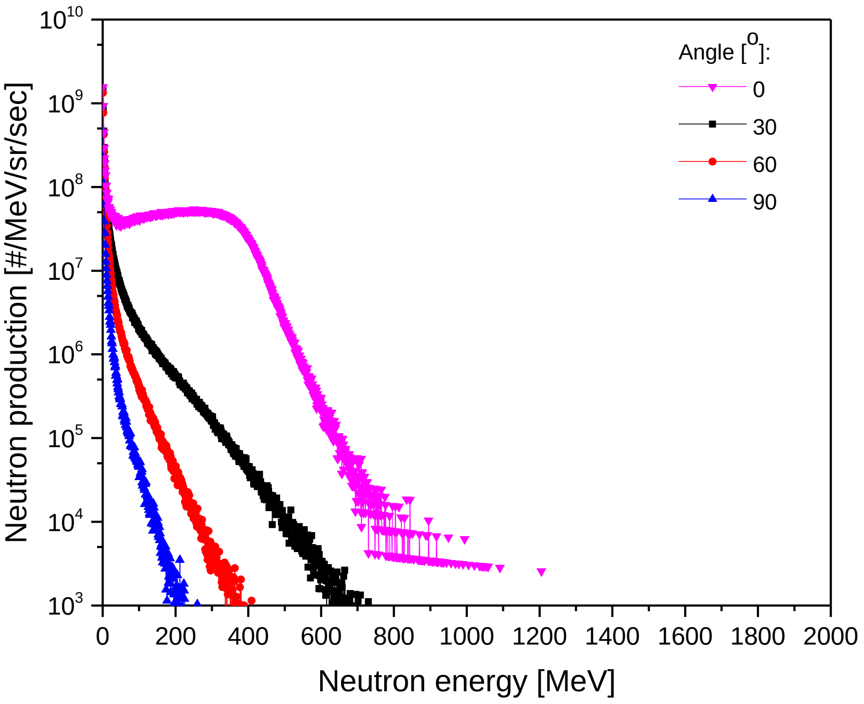
Evaluated neutron source terms

- Beam loss positons in the main separator room



Evaluated neutron source terms

- In main separator room, many isotopes produced from target are distributed. About 400 nuclides will be loss.
- It is impossible to applying all isotopes loss.
- Most conservative neutron spectra was produced with the Tm-173 isotope, and it is applied as the source term.
- Neutron spectra produced by Tm-173

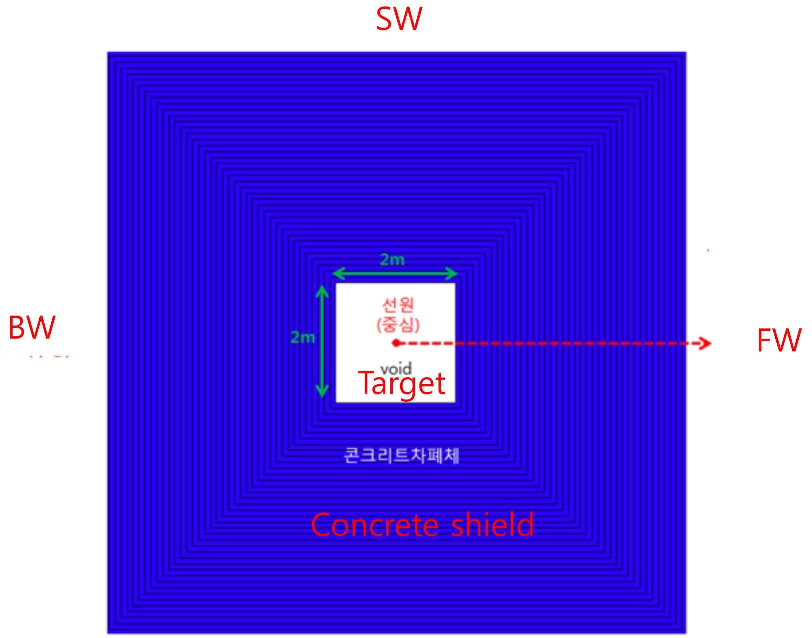


Evaluation of the bulk shield thickness

- With the source terms, the bulk shied thicknesses satisfying the requirement of exposure limit t be used initial facility design.

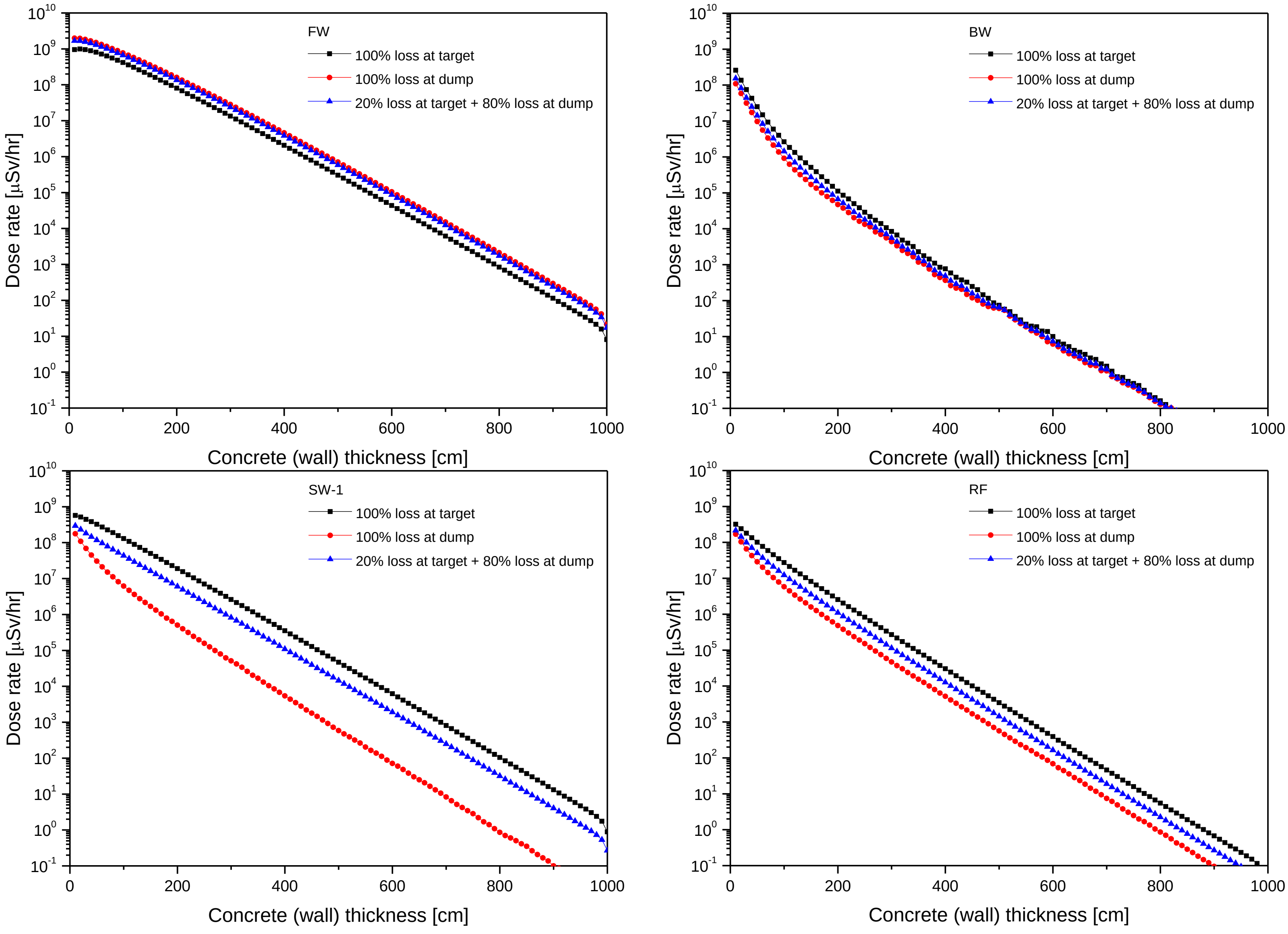
- Source term:
  - O-16 beam, 590 MeV/n + target & dump

- Requirement of exposure limit:
  - 5 uSv/hr for radiation workers
  - 0.25 uSv/hr for the public



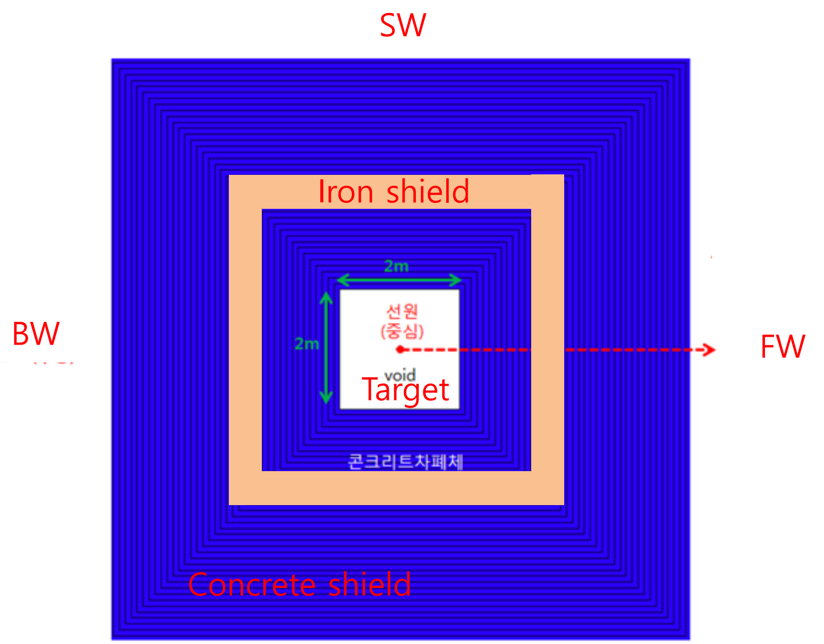
Attenuation of dose rate by the shield

- Using the concrete shield only



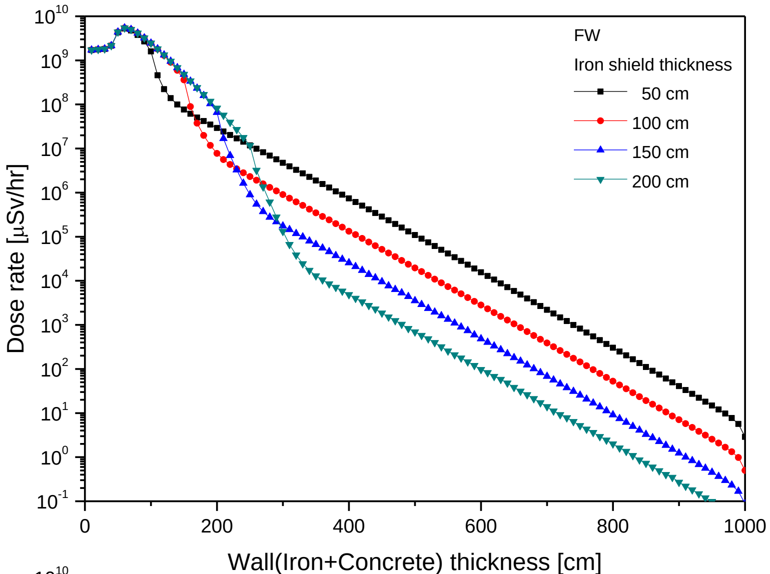
Attenuation of dose rate by the shield

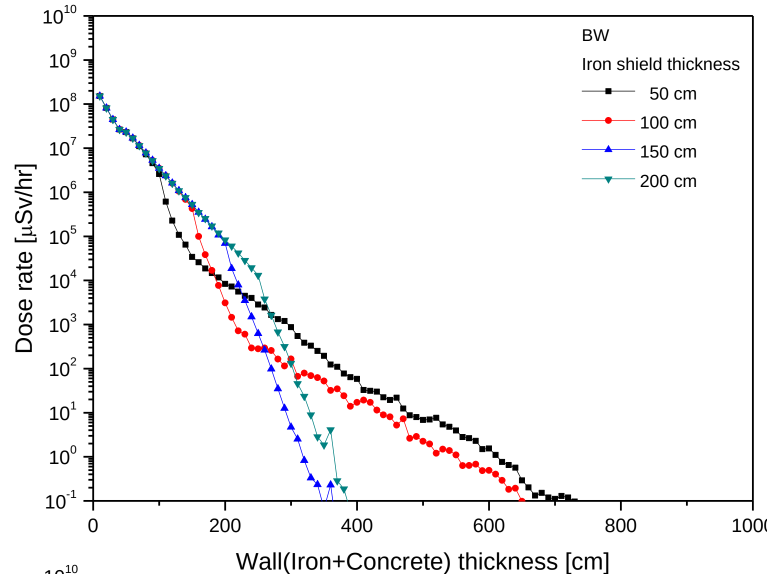
- Using the combination of concrete and iron as the shield
  - To reduce the shield thickness, iron shield was inserted into the concrete shield.
  - The high energetic neutrons loss more energy by the inelastic reactions with the high-Z materials such as iron than the light elements.

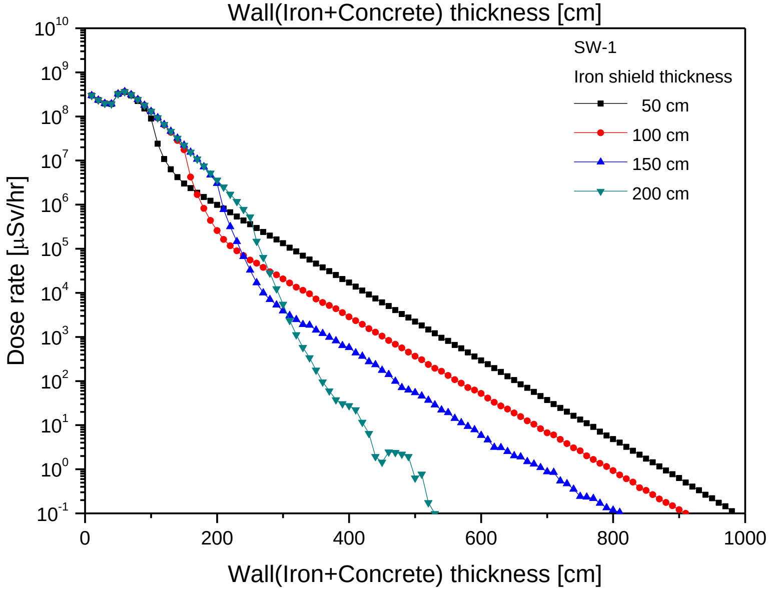


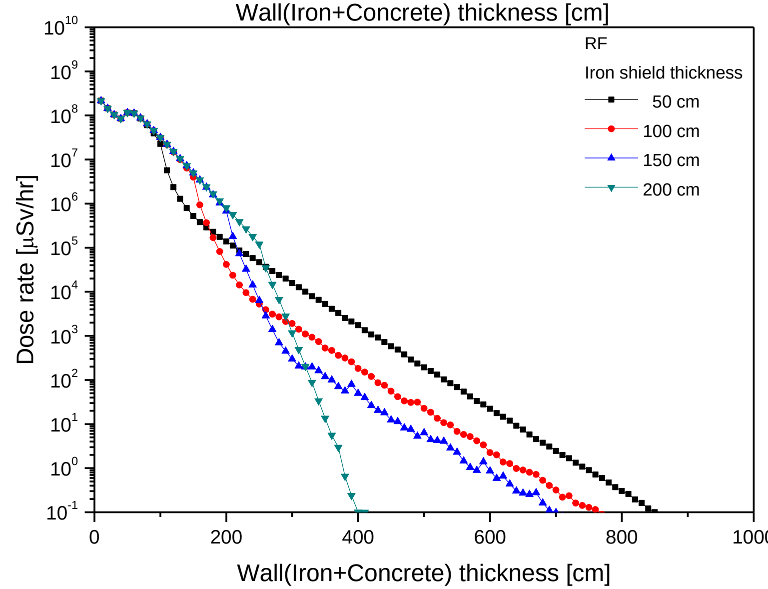
Attenuation of dose rate by the shield

- Using the combination of concrete and iron as the shield









Determination of the bulk shield thickness

- The bulk shield thickness was reflected in the facility design.
  - The BW is the high energy beam transport room and a prohibited area.
  - In the simplified calculation model, the height of the room is considered as 2m, but the real design value is more higher and removal shield was added.

Position	Bulk shield thickness [cm]			Dose rate behind shield [uSv/hr]		Designated wall thickness [cm]		
	Iron	concrete	Total	Calculation	Requirement	iron	concrete	total
FW	150	680	830	4.10	5.00	300	800	1100
BW	0	630	630	4.11	-	0	180	180
SW-R	150	650	800	4.82	5.00	173	689	862
SW-L	150	430	590	4.88	5.00	173	589	762
Roof	150	510	660	4.53	-	150	100	250

Radiation area of RAON

- According to the domestic regulatory based on the ICRP recommendation, the radiation area was consisted and dose rate limit was applied.

<table>
<tr><th colspan="2" rowspan="2">Area</th><th colspan="2">Regulatory</th><th rowspan="2">Designated dose rate
(Safety factor 2)</th><th rowspan="2">Entrance condition</th></tr>
<tr><th>Regulations</th><th>Conversion to dose rate (2000 hr/yr)</th></tr>
<tr><td rowspan="3">Radiation area</td><td>Worker's area</td><td rowspan="3">< 20 mSv/yr</td><td rowspan="2">< 10 uSv/hr</td><td rowspan="2">< 5 uSv/hr</td><td>Not limited</td></tr>
<tr><td>Limited area</td><td>Limited</td></tr>
<tr><td>High level radiation area</td><td>> 10 uSv/hr</td><td>-</td><td>Prohibition</td></tr>
<tr><td>Outside of the facility</td><td>Public area</td><td>< 1 mSv/yr</td><td>< 0.5 uSv/hr</td><td>< 0.25 uSv/hr</td><td>Not limited</td></tr>
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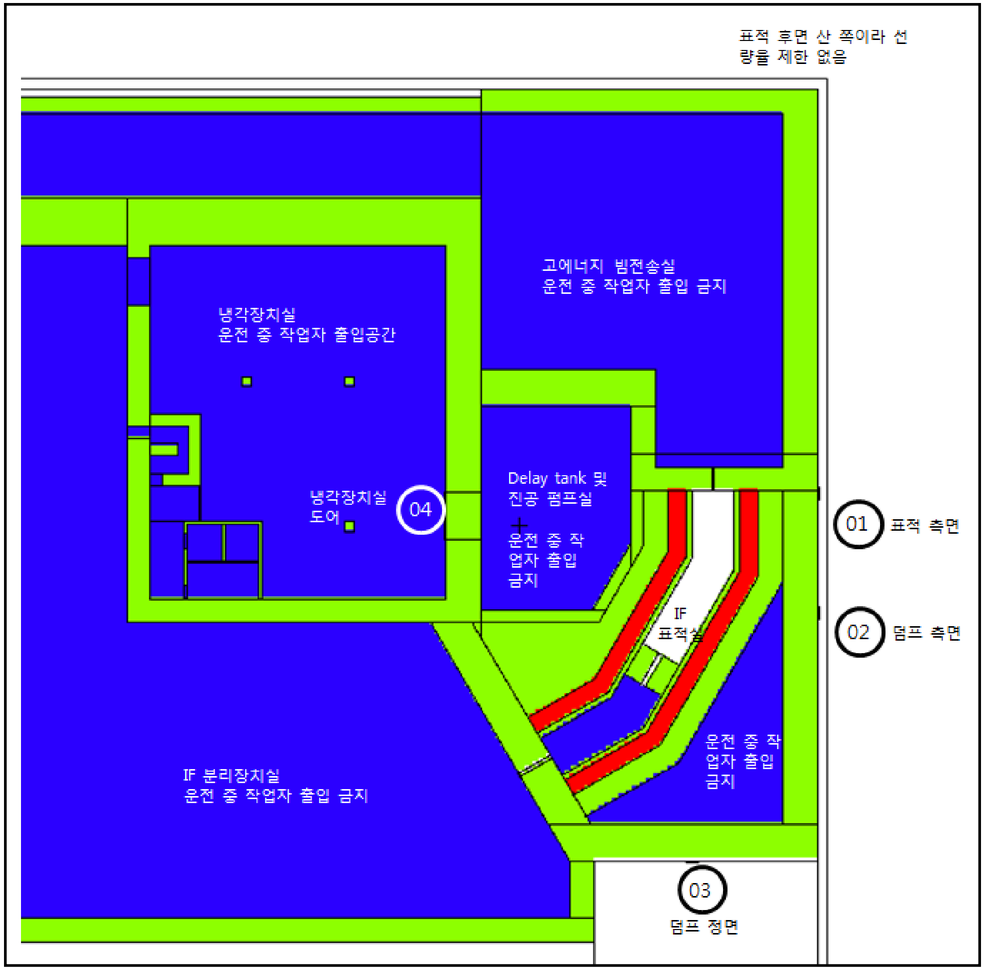
Evaluation of dose rate distribution

- For the facility design, dose rate distribution at each interesting positions was evaluated.
- Total 16 calculation points were selected considering penetration holes, entrance area, door etc.



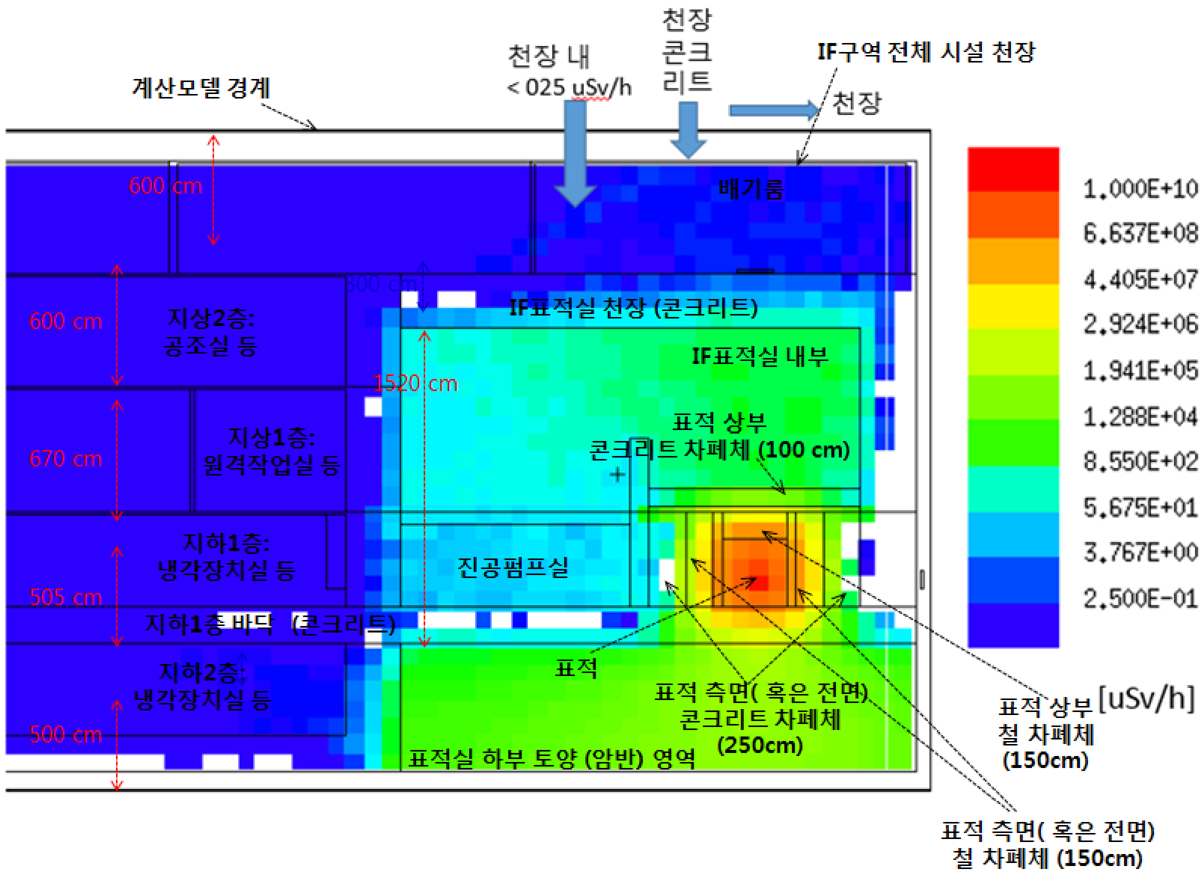
Dose rates in the IFF target facility

- All calculated dose rate satisfied with the designated limit.

#	Floor	위치 개요	Radiation area	Limit [uSv/hr]	**Calculated dose rate [uSv/hr]**
1	B1	Side of the IFF target	Limited area	< 5	**3.8**
2	B1	Side of the beam dump			**4.5**
3	B1	Front of the beam dump			**4.2**
4	B1	Door of cooling system room			**1.2E-02**
5	B2	Entrance are of the drained water processing room			**5.2E-04**
6	B2	Entrance of the rad-waste storage room			**2.3E-05**
7	1F	Inside of the Window station room			**8.7E-03**
8	1F	Inside of the remote control room			**5.4E-05**
9	1F	Side of the remote control room			**2.4E-04**
10	1F	Entrance of the remote control room			**3.4E-03**
11	2F	Entrance of the 2F			**1.7E-10**
12	2F	HEBT room inside	Worker's are		**2.4E-12**
13	2F	Electricity room inside			**2.4E-12**
14	2F	UPS room inside			**4.1E-13**
15	2F	Aisle	Limited area		**6.6E-17**
16	3F	Ventilation room entrance			**3.2E-08**

Dose map

- With the final facility design, dose map was produced.
- From the dose map, it's expected the dose rates distributed inside the facility satisfy with the limit in the all area.



Evaluation of air activation

- Induced activities inside the facility were evaluated with the neutron source term and geometric model.
- Source term:

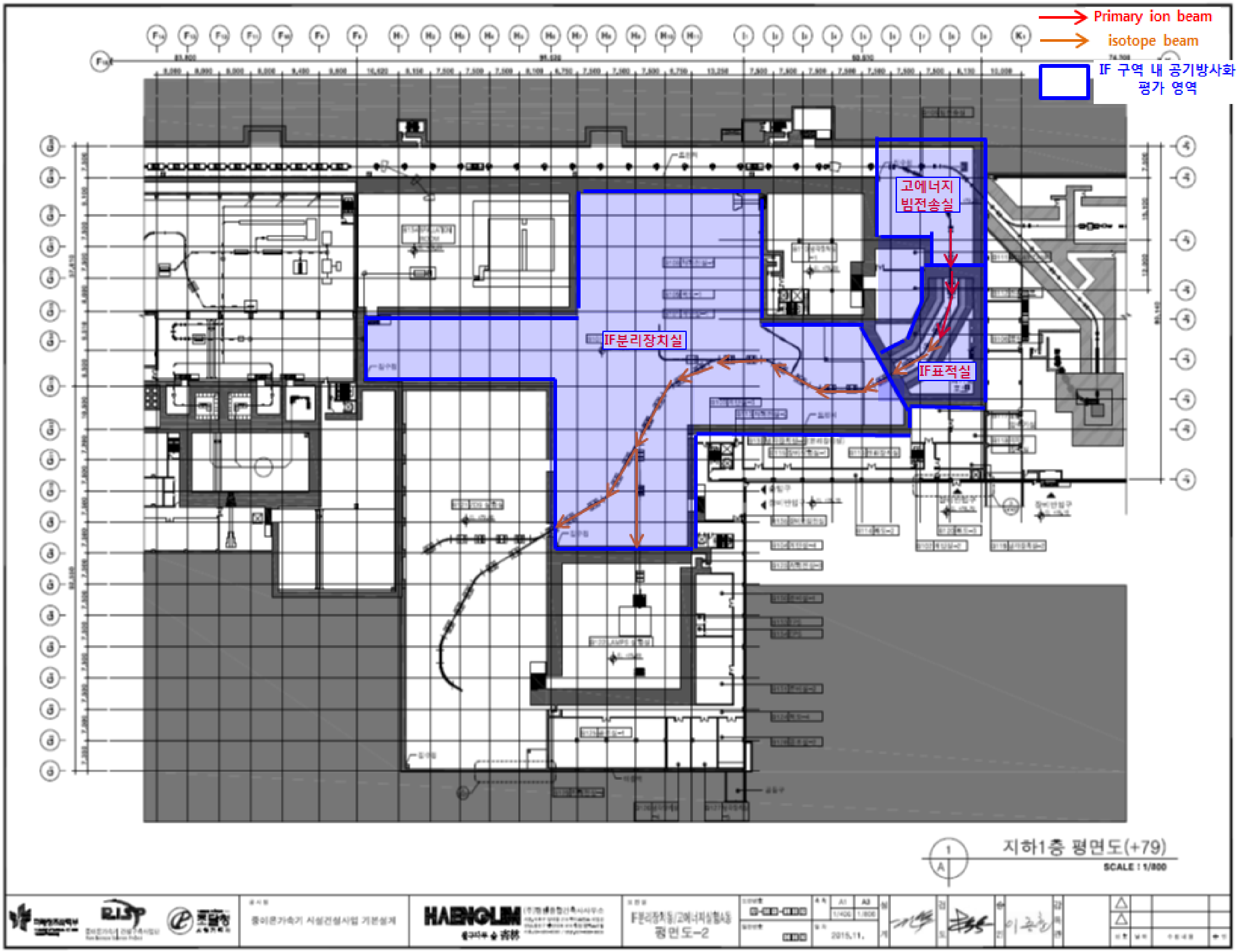
  1) High energy beam transport room: 1 W/m beam loss

  2) Pre-separator room: 20% beam loss at the target and 80% beam loss at the dump

  3) Main separator room: beam loss at the every loss position

Evaluation of air activation

- Calculation area for air activation



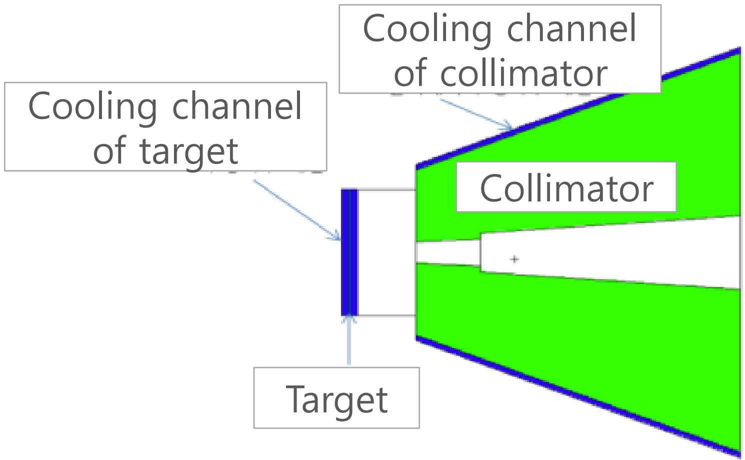
Long-lived radio-active nuclide release

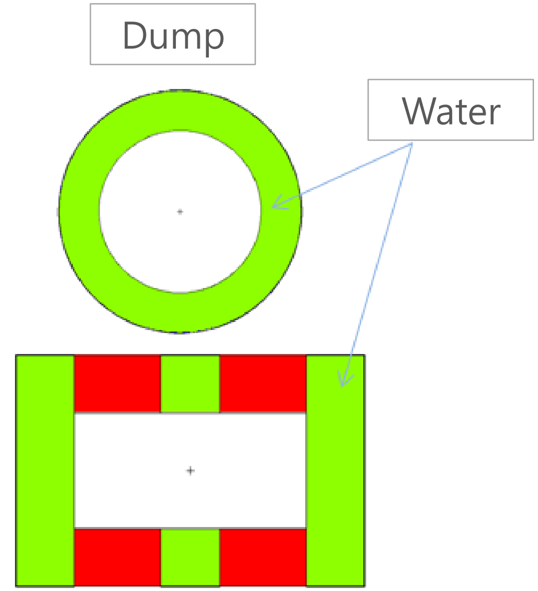
- The release of the long-live radio-active nuclides, H-3 and C-14, were estimated less than 0.3 mg
- In this estimation, the operation time of 5000 hour per year is assumed, conservatively.

Area	Activity [Bq/cc]		Activity [Bq]	
	H-3	C-14	H-3	C-14
High energy beam transport room	1.92E-04	4.57E-05	1.44E+06	3.43E+05
Pre-separator room	1.63E-04	1.31E-04	2.00E+06	1.61E+06
Main separator room	7.59E-08	1.72E-07	4.44E+03	1.01E+04
Activity after 200 hour operation			3.45E+06	1.96E+06
Total release during an year [Bq]			8.62E+07	4.91E+07
Total release during an year [mg]			2.41E-04	2.97E-01

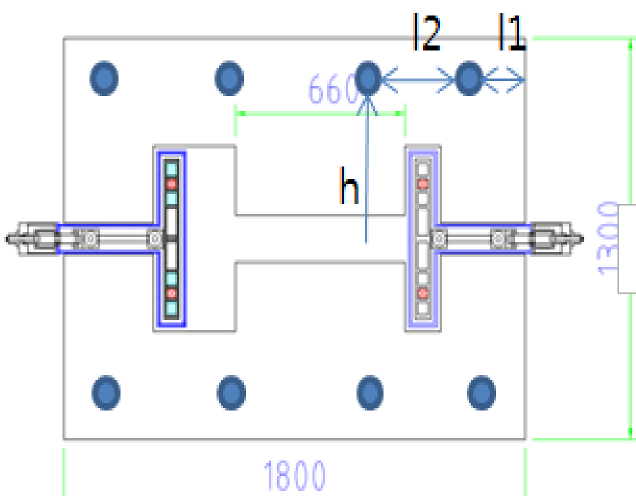
Evaluation of coolant activation

- Target, Collimator and dump

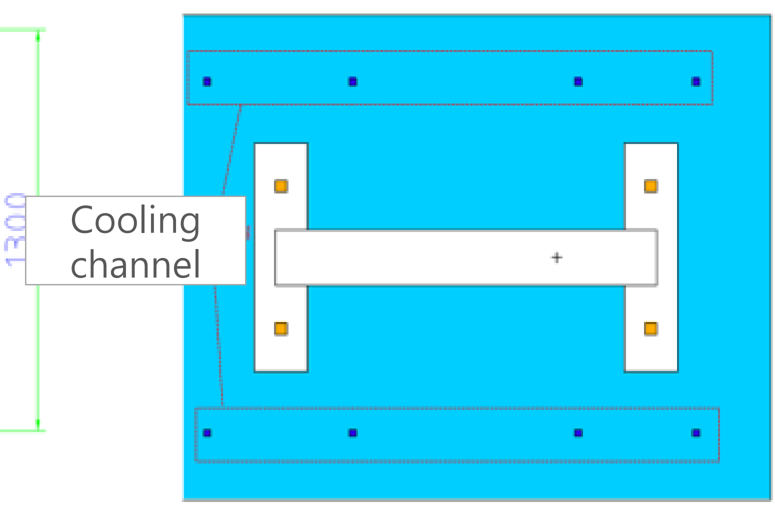




- Dipole

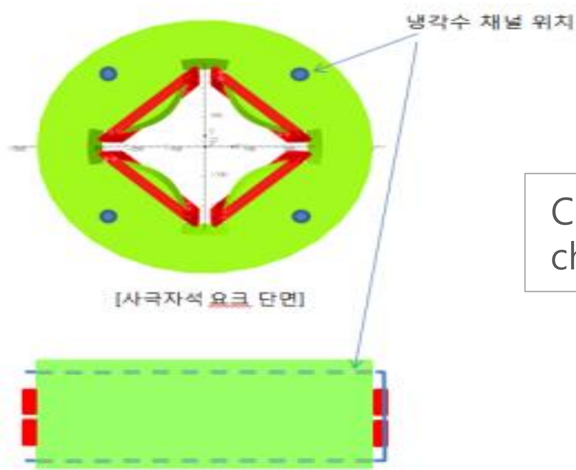


[이극자석 단면]



Evaluation of coolant activation

- Quadrupole:



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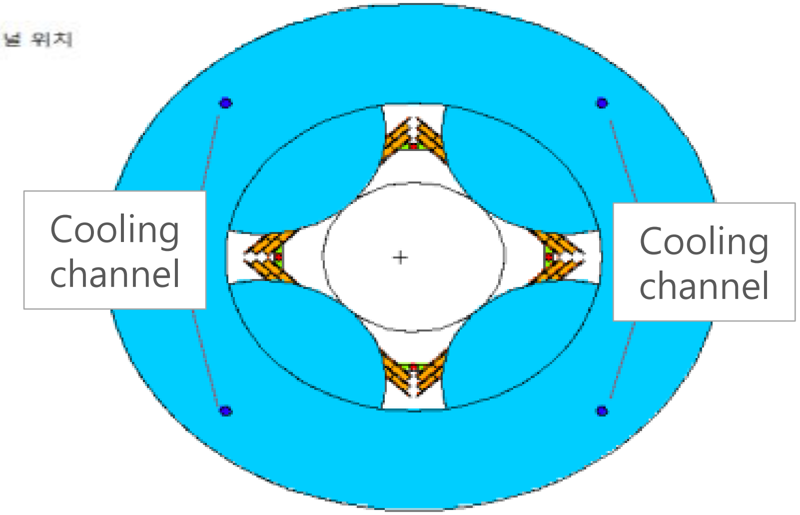
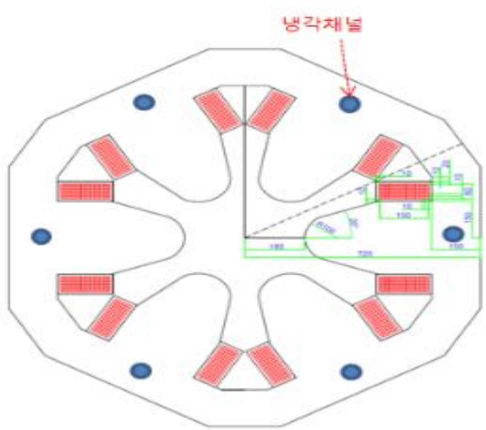
[사극자석 요크 상부]



- Hexapole:



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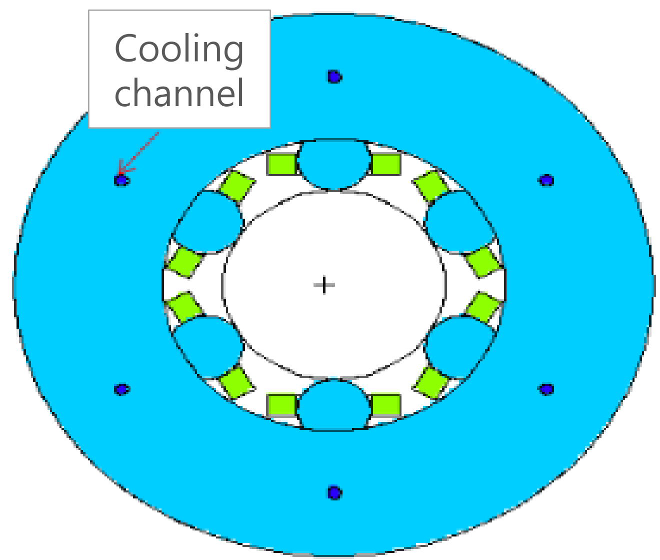


Evaluation of coolant activation

- Various radio-active nuclides are produced in the water.
- Most of them are produced primary beam spallation.
- The cooling system of the closed loop should be installed.

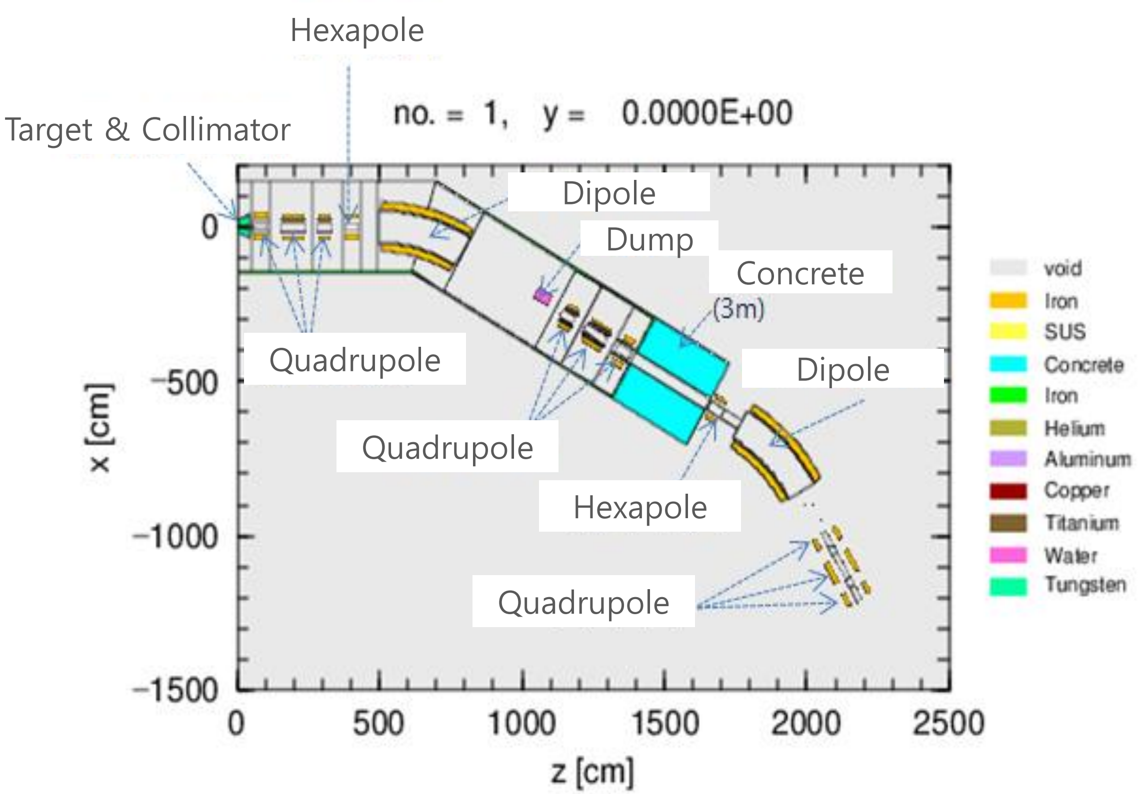
Nuclide		Saturation concentration [Bq/cc]	Half-life [sec]
H	3	1.23E+10	3.89E+08
Be	7	7.64E+09	4.60E+06
Be	10	1.92E+03	4.77E+13
Be	11	6.41E+08	1.38E+01
C	10	2.97E+09	1.93E+01
C	11	1.36E+10	1.22E+03
C	14	4.48E+05	1.81E+11
C	15	4.94E+08	2.45E+00
N	13	7.88E+09	5.98E+02
N	16	2.90E+09	7.12E+00
N	17	4.26E+07	4.17E+00
N	18	1.21E+07	6.30E-01
O	13	5.98E+07	8.58E-03
O	14	1.52E+10	7.06E+01
O	15	4.38E+10	1.22E+02
O	19	3.99E+07	2.69E+01
F	17	5.98E+07	6.45E+01
F	18	1.59E+08	6.59E+03
F	20	1.99E+07	1.10E+01
F	21	3.99E+07	4.32E+00
F	22	3.99E+07	4.24E+00
Ne	23	1.99E+07	3.72E+01
Ne	24	1.99E+07	2.03E+02
Na	24	5.98E+07	5.39E+04
Na	25	3.99E+07	5.96E+01
Mg	27	1.29E+03	5.68E+02
Mg	28	1.99E+07	7.53E+04
Al	28	9.97E+07	1.35E+02
Al	29	3.99E+07	3.94E+02
Al	30	3.99E+07	3.60E+00

Si	31	7.97E+07	9.26E+03
Si	32	4.58E+04	5.43E+09
Si	33	1.99E+07	6.11E+00
Si	34	1.99E+07	2.77E+00
P	30	1.99E+07	1.50E+02
P	32	2.19E+08	1.23E+06
P	33	1.39E+08	2.20E+06
P	34	3.99E+07	1.24E+01
P	35	5.98E+07	4.73E+01
P	37	1.99E+07	2.31E+00
S	35	1.77E+08	7.56E+06
S	37	3.99E+07	2.99E+02
Cl	36	3.14E+02	9.50E+12
Cl	38	9.97E+07	2.23E+03
Cl	39	1.99E+07	3.34E+03
Cl	42	1.99E+07	6.80E+00
Ar	37	1.32E+08	3.03E+06
Ar	39	2.91E+05	8.49E+09
Ar	41	7.97E+07	6.58E+03
Ar	42	4.76E+05	1.04E+09
K	42	1.40E+08	4.45E+04
K	43	5.98E+07	8.03E+04
K	44	1.99E+07	1.33E+03
K	45	1.20E+08	1.04E+03
K	46	3.99E+07	1.05E+02
K	47	1.99E+07	1.75E+01
Ca	45	1.17E+08	1.41E+07
Ca	47	3.99E+07	3.92E+05
Ca	49	3.99E+07	5.23E+02
Ca	50	1.99E+07	1.39E+01

Sc	43	1.99E+07	1.40E+04
Sc	44	5.98E+07	1.41E+04
Sc	44m	7.03E+02	2.11E+05
Sc	46	1.80E+08	7.24E+06
Sc	46m	5.73E+04	1.87E+01
Sc	47	2.39E+08	2.94E+05
Sc	48	1.59E+08	1.57E+05
Sc	49	1.40E+08	3.43E+03
Sc	50	3.95E+07	1.03E+02
Sc	51	7.97E+07	1.24E+01
Sc	52	1.99E+07	8.20E+00
Ti	45	5.98E+07	1.11E+04
Ti	51	1.79E+08	3.46E+02
Ti	52	5.98E+07	1.02E+02
Ti	53	1.99E+07	3.27E+01
V	48	2.19E+08	1.38E+06
V	49	9.18E+07	2.85E+07
V	52	1.99E+08	2.25E+02
V	53	1.40E+08	9.66E+01
V	54	1.99E+07	4.98E+01
V	55	7.97E+07	6.54E+00
Cr	49	3.99E+07	2.54E+03
Cr	51	2.38E+08	2.39E+06
Cr	55	1.99E+08	2.10E+02
Cr	56	1.40E+08	3.56E+02
Cr	57	1.99E+07	2.11E+01
Cr	58	5.98E+07	7.00E+00
Mn	51	3.99E+07	2.77E+03
Mn	52	5.98E+07	4.83E+05
Mn	52m	3.78E+02	1.27E+03
Mn	54	1.85E+08	2.70E+07
Mn	56	4.19E+08	9.28E+03

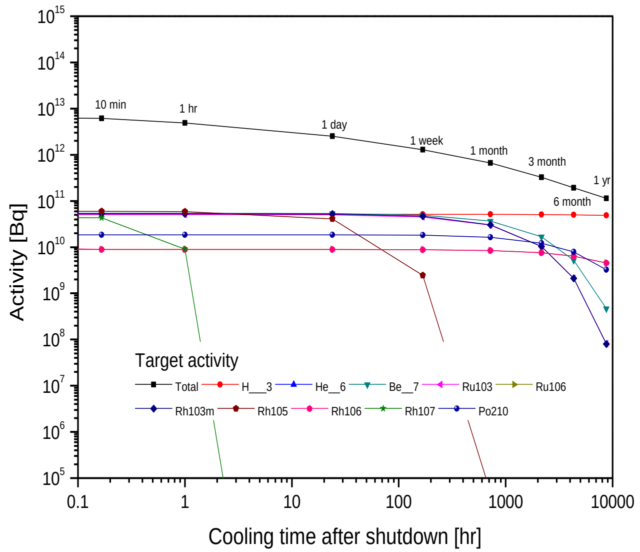
Activation of pre-separator

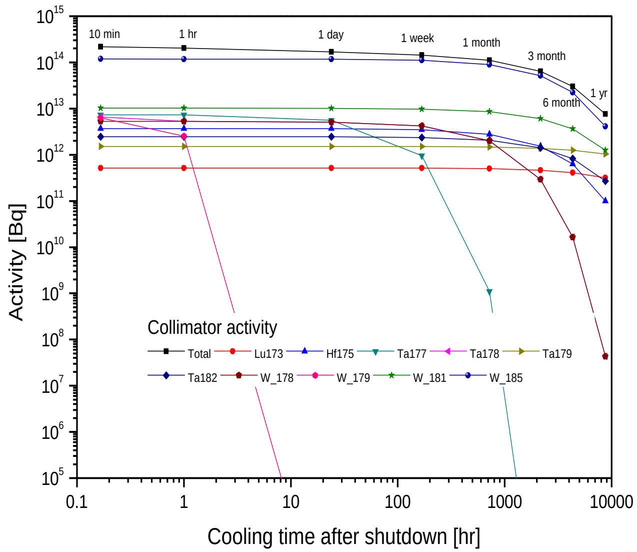
- For the all component of the pre-separator, induced activity was calculated and dose rate around the pre-separator due to the decay gamma.
- The operation time of 5000 hour (in a year) was assumed, conservatively.
- U-238 beam with 400 MeV/u was applied as the primary beam.

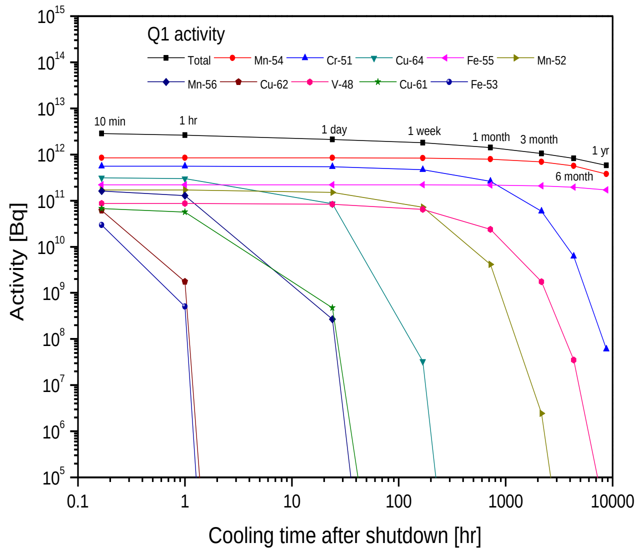


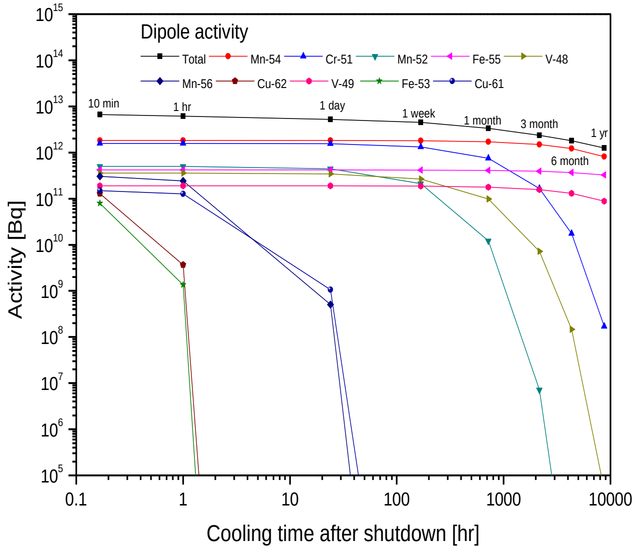
Activation of pre-separator

- Activities and dominant nuclides in each component



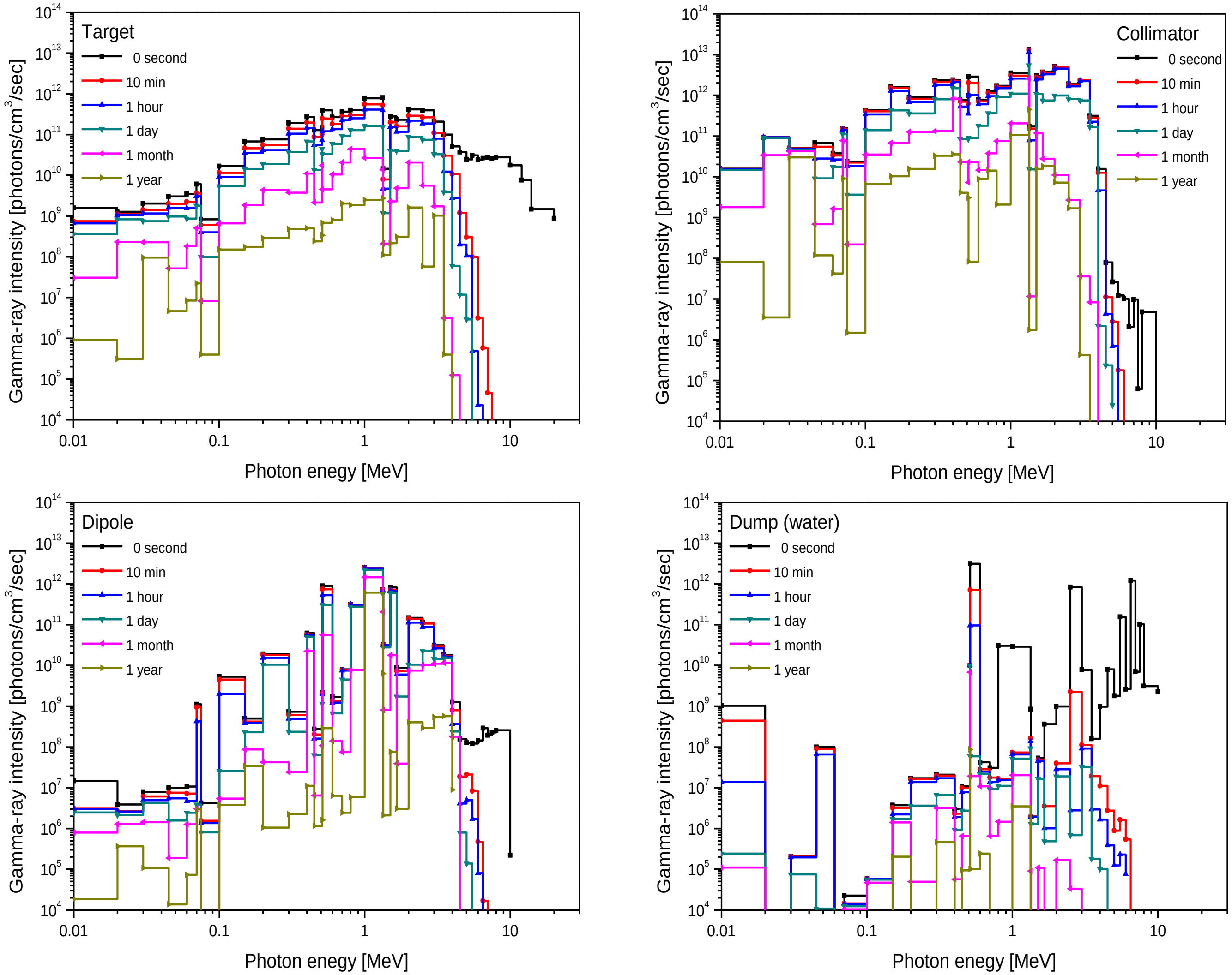






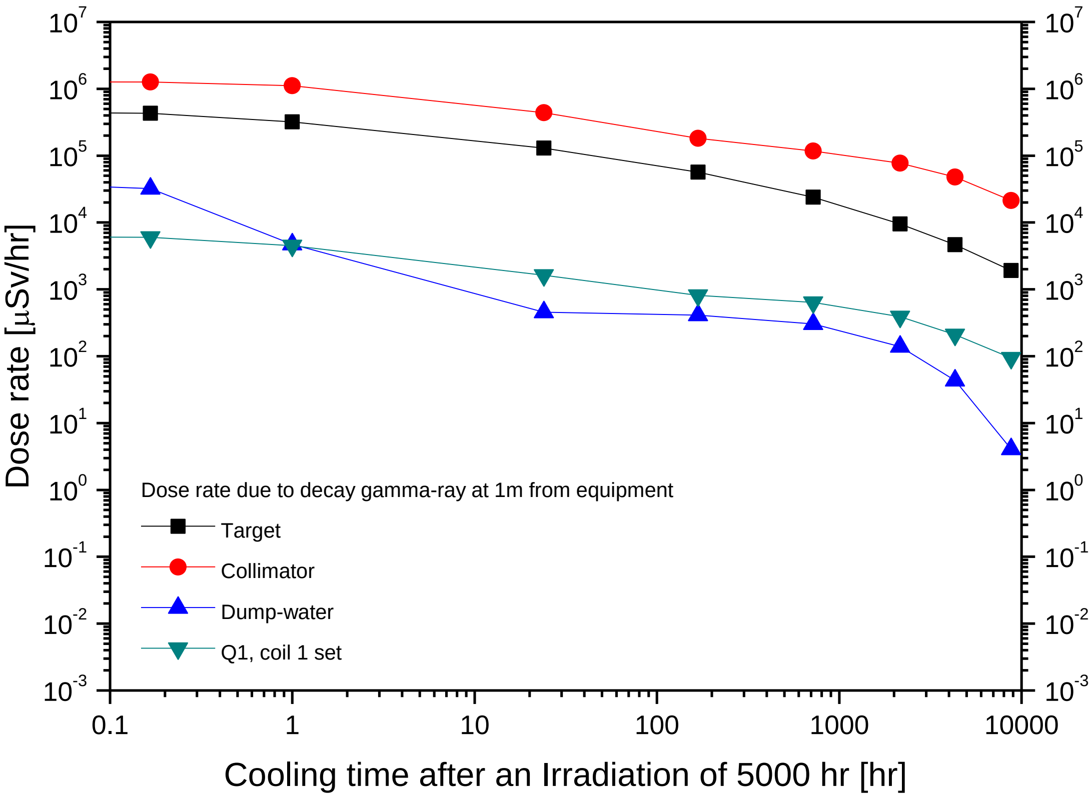
Activation of pre-separator

- Decay gamma spectra were estimated at each component and cooling time step for evaluating shutdown dose rate.



Activation of pre-separator

- Dose rate was evaluated by the decay gamma spectra from the irradiated pre-separator.
- The target and collimator are dominant components contributing to the exposure after a shutdown.



Concrete activation

- It is difficult to predict the composition of the concrete including impurities at the design stage.
- The density and composition of the ANSI/ANS 6.4-1977 was applied in the shielding calculation and activation analysis for the IFF target facility.
- Additionally, impurities was assumed based on the experiment data of other facilities.

Concrete composition

원소	weight %
H	7.6433E-03
O	4.9724E-01
Si	3.1507E-01
Ca	8.2378E-02
Na	1.6985E-02
Mg	2.5478E-03
Al	4.5435E-02
S	1.2739E-03
K	1.9108E-02
Fe	1.2314E-02
Co	1.00E-03
Eu	1.00E-04
Li	2.00E-03
Ni	2.00E-03
Cs	1.00E-03

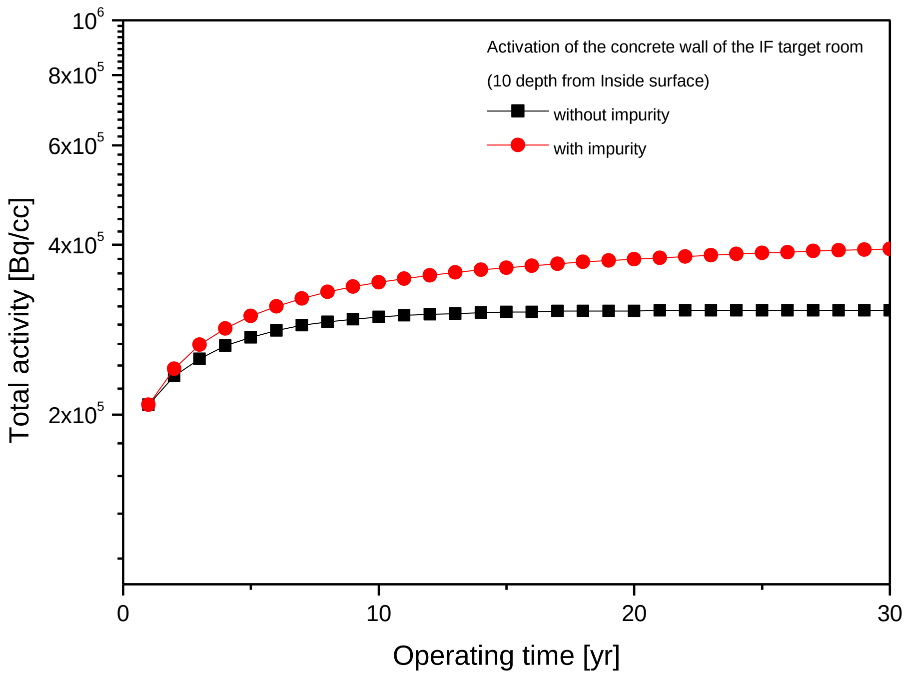
Impurities

모핵종	핵반응	생성핵종	불순물 함량, [ppm]				
			Ref. 1	Ref. 2	Ref. 3	Ref. 4	적용값
Li-6	n, α	H-3	20.00	-	-		20.00
Co-59	n, γ	Co-60	10.00	3.92	10.00	2.20	10.00
Ni-58	n, p	Co-58	30.00		-		30.00
Ni-62	n, γ	Ni-63			-	2.20	-
Cs-133	n, γ	Cs-134	-	1.20	1.20	0.03	1.2
Eu-151	n, γ	Eu-152	0.37	0.17	0.80	0.00	0.80
Eu-153	n, γ	Eu-154	0.40			0.0019	0.4

Ref.1 : KOPEC internal material
Ref.2 : T. Zager et. Al., "Long-lived activation products in TRIGA Mark II research reactor concrete shield: Calculation and experiment," Journal of Nuclear Materials, 335 (2004) 379-386
Ref.3 : K. Masumoto et. Al., "Evaluation of reactivity induced in the accelerator building and its application to decontamination work," Journal of Radioanalytical and Nuclear Chemistry, Vol. 255, No. 3 (2003) 465-469
Ref.4 : G. Hampel et. Al., "Calculation of the activity inventory for the TRIGA reactor at the medical university of Hannover(MHH) in preparation for dismantling the facility," WM'02 Conference, Feb. 24-28, 2002, Tuson, AZ, U.S.

Concrete activation

- Operation condition: 30 years operation, 5000 hours per year is applied as the operation time.
- Most reaction of impurities, producing radio-active nuclides, is capture reaction, (n,γ). Because the hard neutron spectra (high energy neutrons dominant) of the source term, the increase in the activity is about 1.5 times.



Summary, experience and discussion

- The radiation shielding analysis for the IFF target facility of RAON was performed in this study.
- At first, the source terms and bulk shield thickness were evaluated. Secondly, the dose rate distribution during an operation was estimated and produce the dose map. Finally induced activities and shutdown dose distribution were evaluated.

- In the activation calculation, impurities may affect the calculation results. But it is difficult to predict the composition and contents of impurity during the design step.
- For the large-scale facility, obtaining enough low rel. error is not easy in the MC calculation. In this study, Advantg code producing ww file was used. It was very useful in the reduction of the computation time.
- I also expect more powerful modelling tool will be developed.