Probabilistic Risk Assessment of Cask Drop Accident during On-Site Spent Nuclear Fuel Transportation

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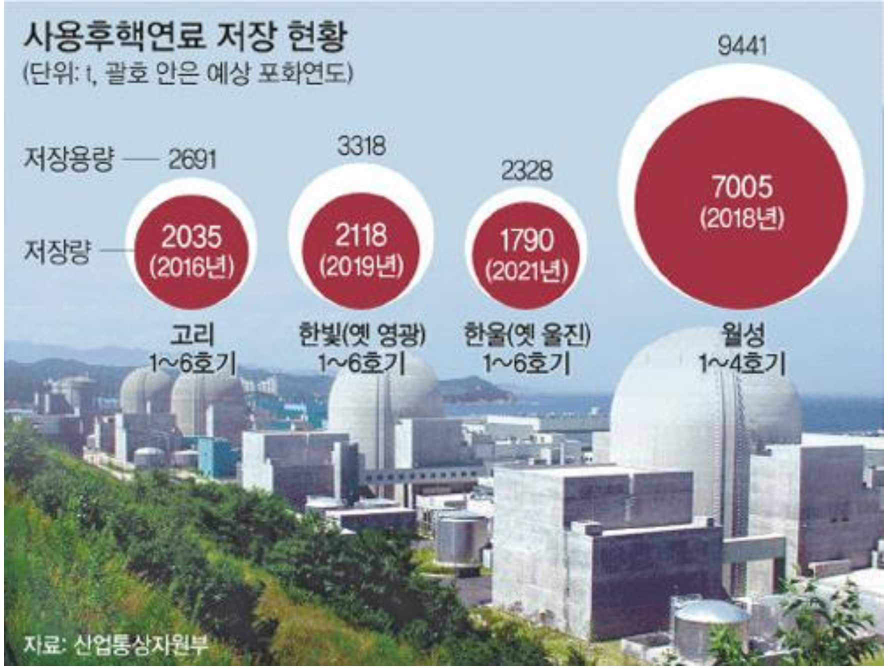
Introduction : Research overview

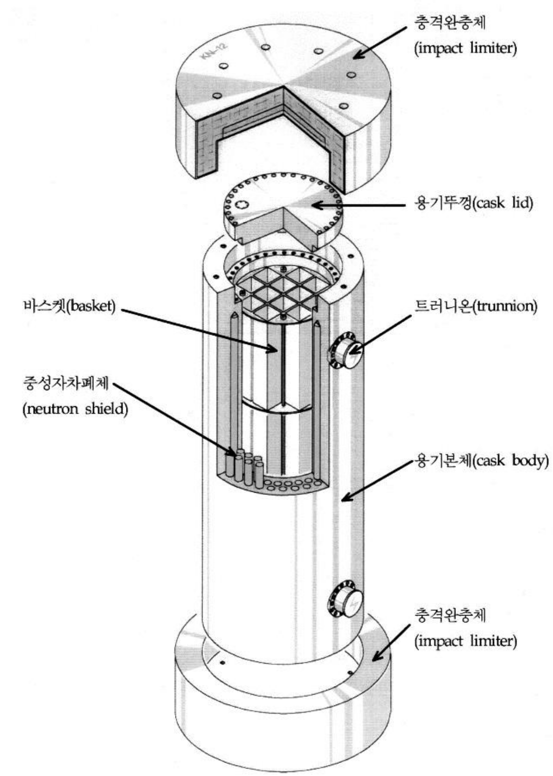
- SNF (Spent Nuclear Fuel) pool in Kori, Wolseong, Hanbit, Hanul sites will be saturated in 2016, 2018, 2019, 2021 each.
- Under the situation that solution of this problem is not determined, transferring the SNF temporarily from placeless pool to other pools can be an option.
- PRA (Probabilistic Risk Assessment) of on-site SNF transportation from SNF pool to wharf was done.
- Drop accident was only covered in this research.

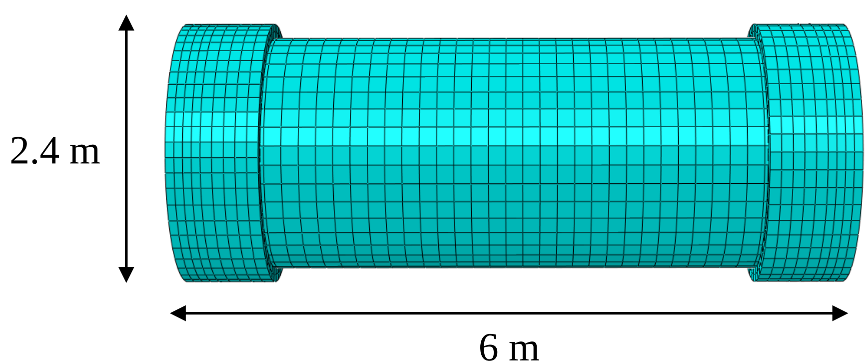


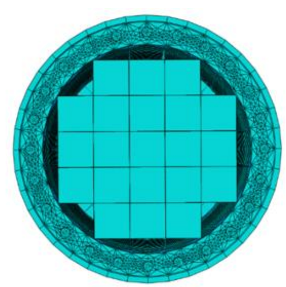
Introduction : Objective & Target cask

- Research objective : PRA of cask drop accident during on-site SNF transportation
- On-site transportation : SNF pool → Wharf
- Target cask : Bolted metal cask (Reference : KN-12)
  - 21 Fuel assemblies
  - 1 Cask body (Carbon Steel)
  - 2 Cask lids (Carbon Steel)
  - 2 Impact limiter (Balsa & Redwood)
  - Weight : about 75 ton (fuel + cask)

    about 85 ton (+ impact limiter)







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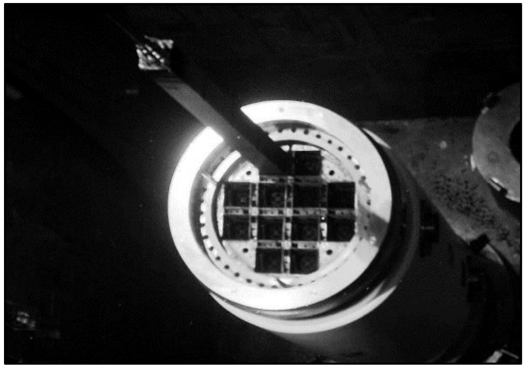
Method : Process of on-site transportation

Stages	Contents	Height (m)		State
		Before	After	
1	Loading fuel assemblies into the cask	4.8	0	SNF assemblies
2	Lifting the cask out of the cask pit	0	13	SNF assemblies + Cask
3	Moving the cask to the railing area	13	13	
4	Lowering the cask over a railing of the spent fuel pool	13	0.3	
5	Moving the cask to the preparation area	0.3	0.3	
6	Lowering the cask onto the preparation area	0.3	0	
7	Preparing the cask (draining, drying, inerting, and sealing)	0	0	
8	Lifting the cask	0	0.6	
9	Moving the cask to the equipment hatch	0.6	0.6	
10	Inspection and maintaining the cask	0.6	0.6	
11	Lowering the cask on to the equipment hatch	0.6	0	
12	Equipping the impact limiter to cask body	0	0	
13	Lifting the cask	0	0.6	SNF assemblies + Cask + Impact Limiter
14	Moving the cask to the shipment area	0.6	0.6	
15	Inspection and maintaining the cask	0.6	0.6	
16	Lifting the cask	0.6	3	
17	Moving the cask to the truck	3	3	
18	Lowering the cask on the truck	3	1	
19	Transferring the cask to wharf by truck	1	1	
20	Inspection and maintaining the cask	1	1	
21	Lifting the cask	1	5	
22	Moving the cask to the ship	5	5	
23	Lowering the cask on the ship	5	0	

Table 1. Stages of the Dry Cask Storage Operation [1]

Stages		Height (A) m	ft
1	Loading fuel assemblies into the MPC (B)	4.8	16
2	Placing the MPC lid onto the MPC and engaging the lift yoke on the transfer cask (C)	0	0
3	Lifting the transfer cask out of the cask pit	13	42.5
4	Moving the transfer cask over a railing of the spent fuel pool	0.9	3
5	Moving the transfer cask to the preparation area (1st segment)	0.3	1
6	Moving the transfer cask to the preparation area (2nd segment)	0.3	1
7	Moving the transfer cask to the preparation area (3rd segment)	0.3	1
8	Lowering the transfer cask onto the preparation area (D)	0.3	1
9	Preparing (draining, drying, inerting, and sealing) the MPC for storage	0	0
10	Installing the short stays and attaching the lift yoke (D)	0	0
11	Lifting the transfer cask	0.6	2
12	Moving the transfer cask to exchange bottom lids of the transfer cask (1st segment)	0.6	2
13	Moving the transfer cask to exchange bottom lids of the transfer cask (2nd segment)	0.6	2
14	Replacing the pool lid with the transfer lid	0.1	0.25
15	Moving the transfer cask near the equipment hatch	0.6	2
16	Holding the transfer cask	0.6	2
17	Moving the transfer cask to the equipment hatch	0.6	2
18	Lowering the transfer cask to the overpack through the equipment hatch	24.4	80
19	Preparing (remove short stays, disengage lift yoke, attach long stays) to lower the MPC	0	0
20	Lifting the MPC and opening doors of transfer lid	5.8	19
21	Lowering the MPC through the transfer cask into the storage cask	5.8	19
22	Moving the storage cask into the airlock on Helman rollers	0	0
23	Moving the storage cask out of the airlock on Helman rollers	0	0
24	Moving the storage cask away from the secondary containment on Helman rollers	0	0
25	Preparing (installing lid, vent shield cross-plates, vent screens) the storage cask for storage	0	0
26	Lifting the storage cask above the Helman rollers with the overpack transporter	0.1	0.25
27	Moving the storage cask above a cushion on the preparation area	<0.1	<0.25
28	Holding the storage cask above the cushion while attaching a Kevlar belt	<0.1	<0.25
29	Moving the storage cask above the concrete surface of the preparation area	0.3	1
30	Moving the storage cask above the asphalt road	0.3	1
31	Moving the storage cask above the gravel surface around the storage pads	0.3	1
32	Moving the storage cask above the concrete storage pad	0.3	1
33	Lowering the storage cask onto the storage pad	0.3	1
34	Storing the storage cask on the storage pad for 20 years	0	0


[2]


[3]

[1] U.S.NRC, "A Pilot PRA of a Dry Cask Storage System at a Nuclear Power Plant", NUREG-1864, 2007.
[2] Sung-Hwan Chung, Chang-yeal Baeg, Byung-Il Choi, Ke-Hyung Yang and Dae-Ki Lee, "On-site Transport and Storage of Spent Nuclear Fuel at Kori NPP by KN-12 Transport Cask", J. of Korean Radioactive Waste Society, Vol.4, p.51-58, 2006.
[3] AREAVA, "Process of Dry Storage System".

Method : PRA methodology

- $\text{Risk} = \text{Probability} \times \text{Consequence}$
  - Probability : for each stage - *reference*
  - Consequence : Radiological consequence (man-mSv/transport) based on source term - *(1)*
- $\text{Source term} = \text{MAR} \times \text{FDR} \times RF_{R-C} \times RF_{C-E} \times \text{LPF}$
  - MAR : Material-At-Risk, the initial amount of radioactive materials in cask - *(2)*
  - FDR : Fuel Damage Ratio - *(3)*
  - $RF_{R\text{-}C}$: Release Fraction from rod to cask - *reference*
  - $RF_{C\text{-}E}$: Release Fraction from cask to environment – *(3)*
  - LPF : Leak Path Factor, dispersion factor - *reference*
- Calculation Tool

  (1) HOTSPOT 3.0.3

  : Atmospheric dispersion models

  (2) Scale 6.1.3 (ORIGEN-ARP)

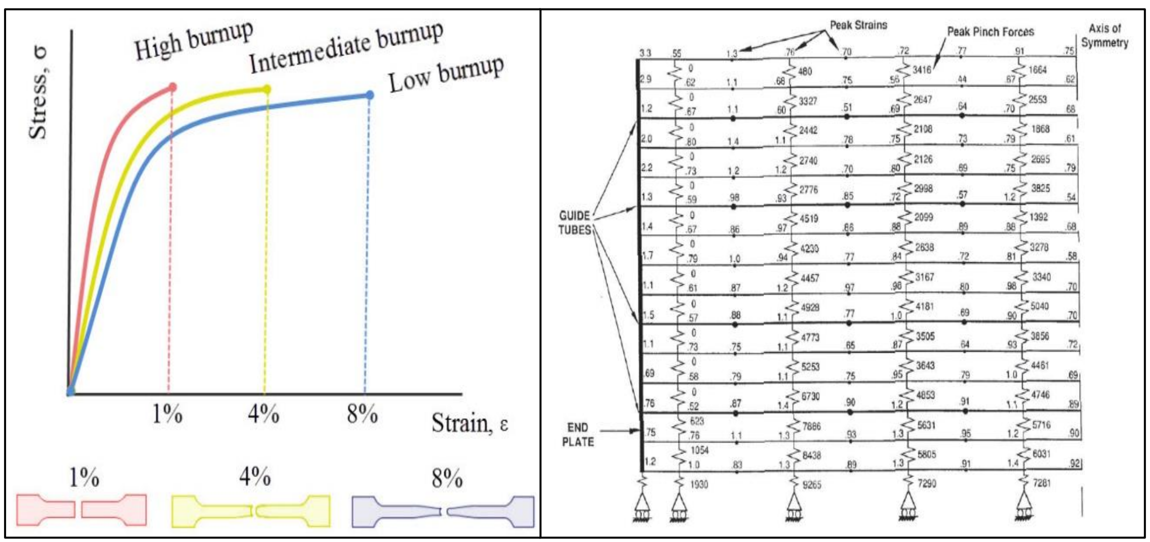
  : Buildup, decay, and processing of radioactive materials calculation

  (3) ABAQUS 6.13-1

  : FEM (Finite Element Method) simulation for impact analysis

Method : FDR (Fuel Damage Ratio)

- From FEM simulation, maximum accelerations for each SNF assembly was calculated.
- Using linear extrapolation, peak strain under 100 G for each SNF assembly was calculated.
- Failure criteria of peak strain by burn-up (GWDt/MTU)
  - High burn-up (55 ~ 60) : 1%
  - Intermediate burn-up (40 ~ 45) : 4%
  - Low burn-up (0 ~ 25) : 8%



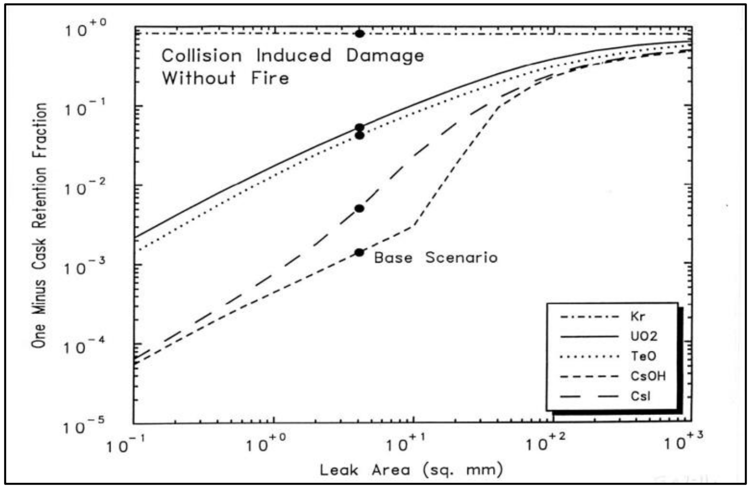
[4]

Fraction of PWR Rods	Peak Strain under 100 G (%)
1/15	3.3
2/15	2.9
3/15	2.2
4/15	2
5/15	1.7
6/15	1.5
7/15	1.4
8/15	1.4
9/15	1.4
10/15	1.3
11/15	1.3
12/15	1.2
13/15	1.2
14/15	1.1
15/15	1.1

[4] Sandia National Laboratories, “A Method for Determining the Spent-Fuel Contribution to Transport Cask Containment Requirements”, SAND90-2406, 1992.

Method : $RF_{C\text{-}E}$ (Release Fraction from Cask to Environment)

- Relationship between release fraction from cask to environment depends on leak area was considered for calculation.
- Cask is pressurized to 5 atm by the failure of all of the rods due to collision.
- Due to the leak area between lid and cask body, radioactive materials are released environment until pressure of the inner cask reaches to 1 atm.
- Calculation method of leak area : Lid gap analysis
- Maximum value : 0.8 for gas, 0.5 for volatile under > 1,000 mm^2 of leak area

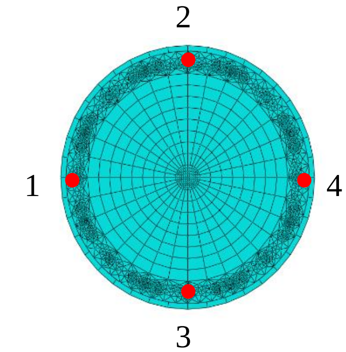


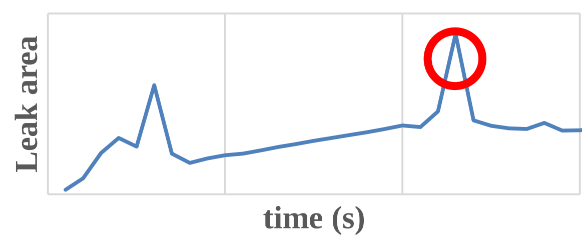
[5]

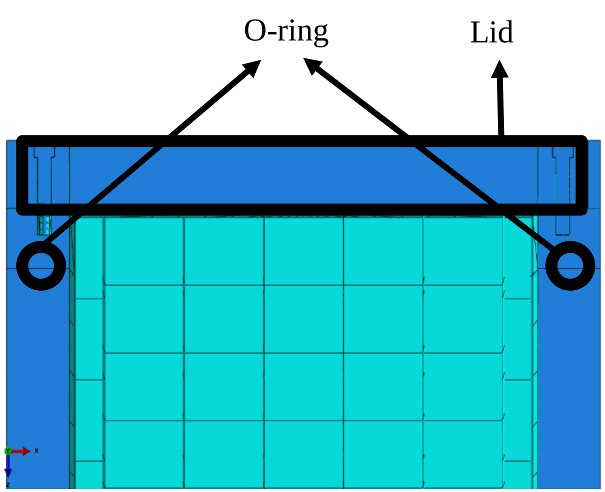
[5] U.S.NRC, "Reexamination of Spent Nuclear Fuel Shipment Risk Estimates", NUREG-6672, 2000.

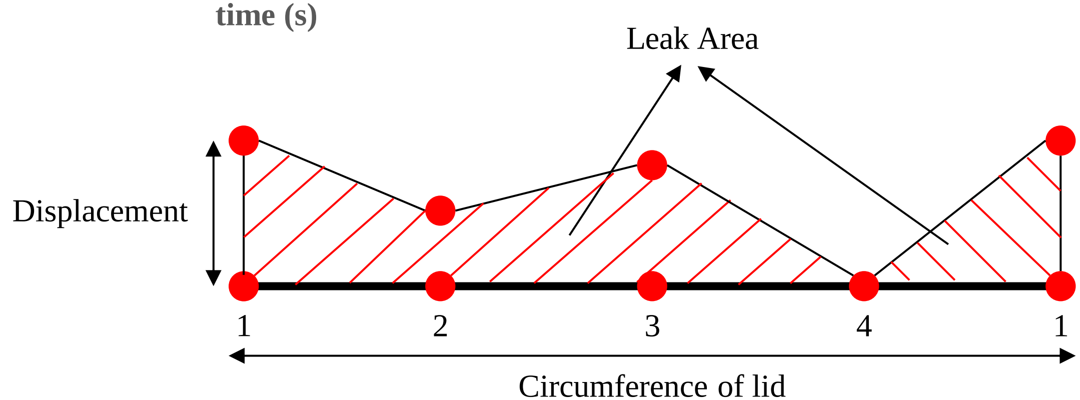
Method : Lid gap analysis

- 4 node pairs on each lid which are point of contact between each lid and cask body were selected.
- Average value of displacements in the outer direction between nodes in each pair is considered as lid gap. By multiplying the lid gap and circumference of lid, total leak area can be calculated.
- O-ring seal : coverage - 0.25 mm (metal), 2.5 mm (rubber)
- Leak area : The largest area during the impact situation



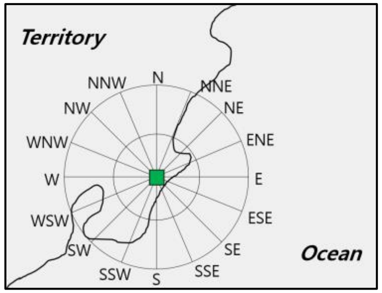




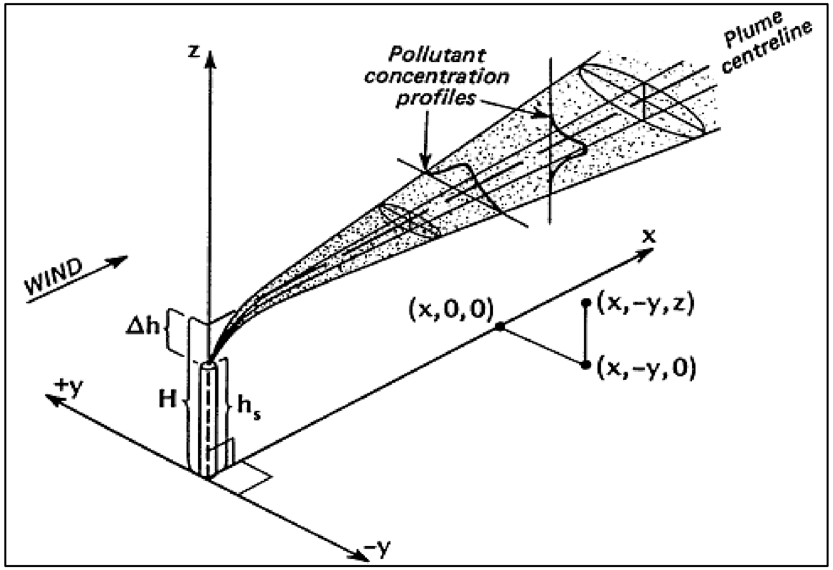


Method : Radiological consequence

- Gaussian dispersion is used to model the plume dispersal that developed in hotspot code. The hotspot code is designed for short range (less than 10 km), and short-term (less than few hours prediction).
- Values to calculate the radiological consequence were considered like below as a sample case.



Radionuclides		Release Height	Wind Speed Ref. Height	Breathing height & breathing rate
Kr (Gas)	Cs (Volatile)	0 m	80 m	1.5 m & 3.47E-04 m3/sec



Wind Speed Groups	Group probability
Group 0: 0.10 <= u <= 0.80	3.21 %
Group 1: 0.80 < u <= 1.60	9.66 %
Group 2: 1.60 < u <= 2.40	17.35 %
Group 3: 2.40 < u <= 3.20	22.03 %
Group 4: 3.20 < u <= 4.00	21.16 %
Group 5: 4.00 < u <= 4.80	13.72 %
Group 6: 4.80 <= u <= 5.60	6.15 %
Group 7: 5.60 < u <= 5.61	0.50 %
Group 8: u > 5.61	6.21 %
	Total Sum: 100.00 %

Method : Other conditions

- Probability
  - Cask drop during a single crane action : 5.6E-05 (Stage 2 ~ 18, 20 ~ 23)
  - Cask drop during transferring by truck : 3.3E-08 (Stage 19)[6]
- Source term $= \mathrm{MAR} \times \mathrm{FDR} \times \mathrm{RF}_{\mathrm{R-C}} \times \mathrm{RF}_{\mathrm{C-E}} \times \mathrm{LPF}$
  - $\mathrm{RF}_{\mathrm{R\text{-}C}}$: 3.0E-05 for Cs (volatile) / 0.12 for Kr (gas)[7]
  - Leak Path Factor (LPF) = 1
- Impact analysis
  - Drop accident was only covered.
  - Floor was assumed as a rigid body.
  - The most conservative drop angle was applied to each state of cask.
- Consequence of stage 1 was not covered in this research. (Loading fuel assemblies into the cask under the water)

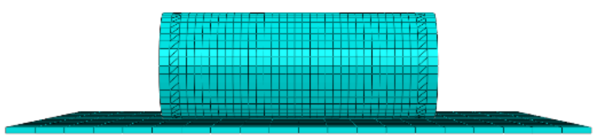
Item	Value
SNF type	CE 16 × 16
Initial enrichment	4.5 wt%
Cooling period	10 years
Amount of uranium (per assembly)	450 kg
Number of fuel assembly	21
Burn up rate	Intermediate (45 GWD/MTU)
Target distance	5.7 km (LPZ)
Wind direction	Average

[6] EPRI, "Probabilistic Risk Assessment of Bolted Storage Casks", EPRI-1009691, 2004.
[7] U.S.NRC, "Spent Fuel Transportation Risk Assessment", NUREG-2125, 2012.

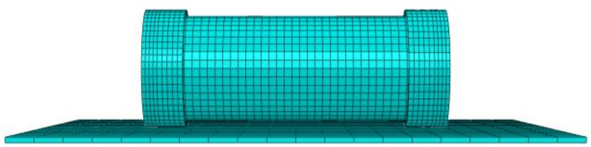
Results : Drop angle analysis

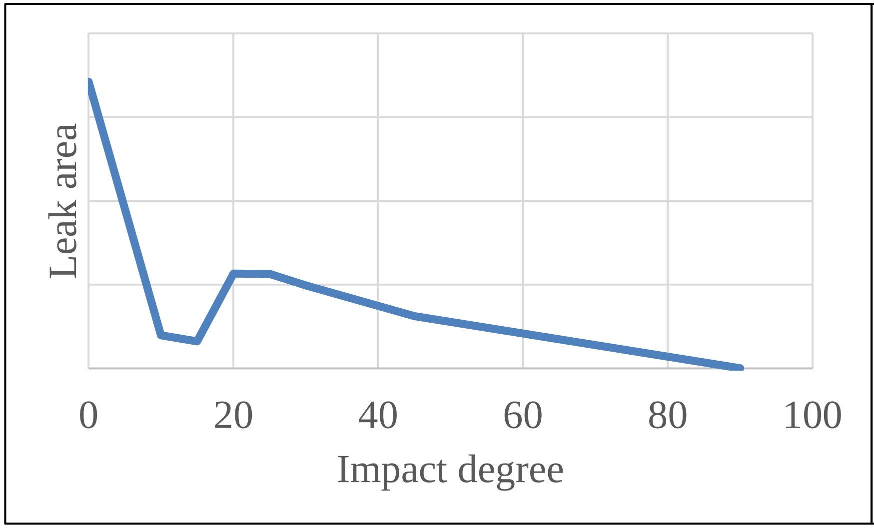
- Two states of cask were considered during the whole process.
- Specific drop angles were considered for each state conservatively.
- Considered drop angle : Angle which shows the largest leak area
  - Cask without impact limiter : 0° (side drop)
  - Cask with impact limiter : 0° (side drop)

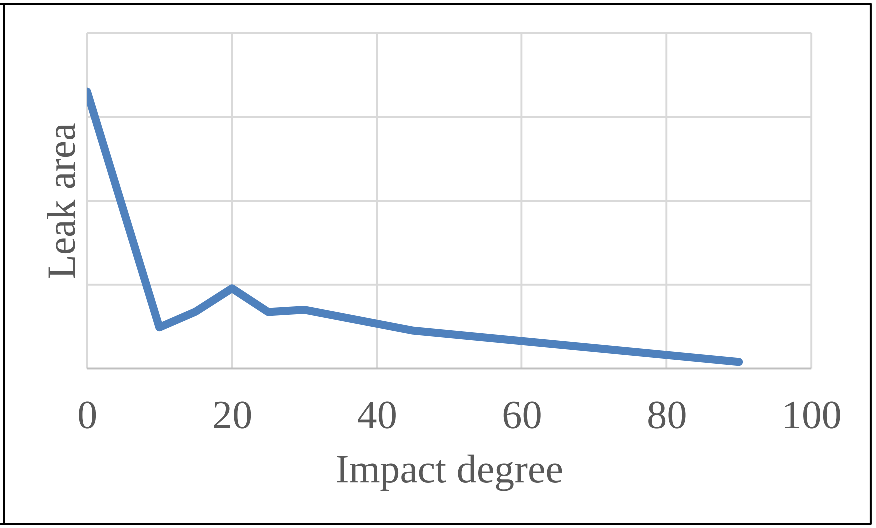
Cask without impact limiter



Cask with impact limiter



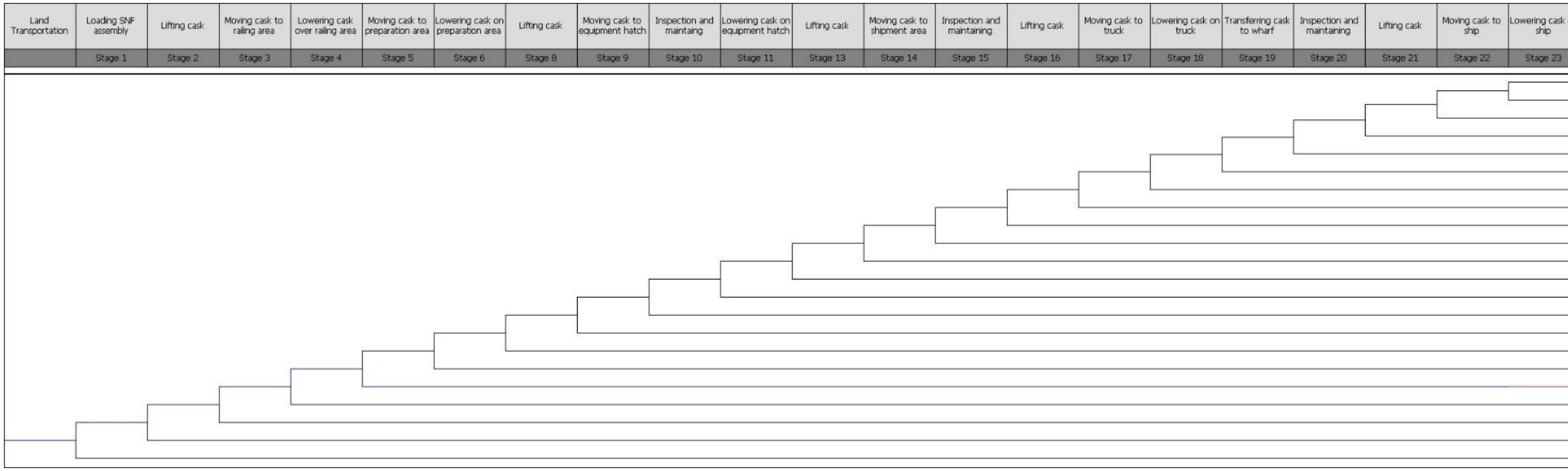




Results : Release fraction

Case	State	Height (m)	FDR	Leak Area (mm^2)		Release fraction		Release fraction	
				Rubber O-ring	Metal O-ring	Rubber O-ring		Metal O-ring	
						Cs	Kr	Cs	Kr
1	Without impact limiter	1	1	0	1310	**0**	**0**	**2.4E-06**	**9.6E-02**
2		4	1	203	5390	**1.54E-06**	**9.6E-02**	**2.4E-06**	**9.6E-02**
3		7	1	1160	9430	**2.40E-06**	**9.6E-02**	**2.4E-06**	**9.6E-02**
4		10	1	1920	13000	**2.40E-06**	**9.6E-02**	**2.4E-06**	**9.6E-02**
5		13	1	2700	17100	**2.40E-06**	**9.6E-02**	**2.4E-06**	**9.6E-02**
6	With impact limiter	1	1	0	0	**0**	**0**	**0**	**0**
7		3	1	0	2960	**0**	**0**	**2.4E-06**	**9.6E-02**
8		5	1	0	5330	**0**	**0**	**2.4E-06**	**9.6E-02**

Results : Event tree & Risk assessment

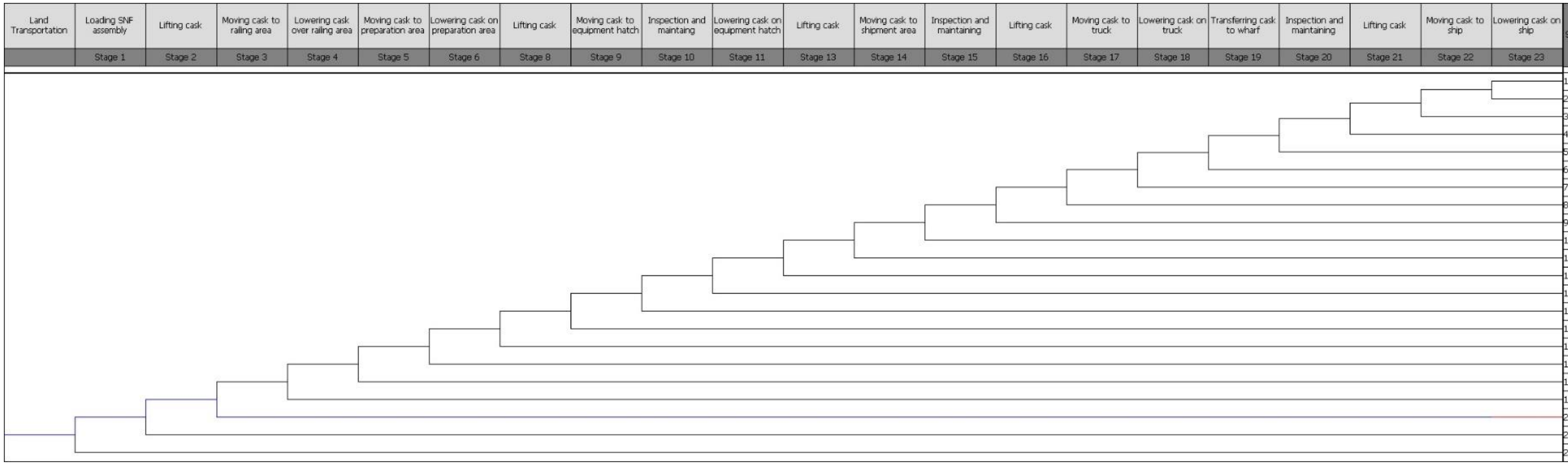


Seq#	State	Frequency	Consequence	Risk
1	OK	9.989E-001	0	0.000E+000
2	ND	5.594E-005	0	0.000E+000
3	ND	5.594E-005	0	0.000E+000
4	ND	5.595E-005	0	0.000E+000
5	ND	5.595E-005	0	0.000E+000
6	ND	3.297E-008	0	0.000E+000
7	ND	5.595E-005	0	0.000E+000
8	ND	5.596E-005	0	0.000E+000
9	ND	5.596E-005	0	0.000E+000
10	ND	5.596E-005	0	0.000E+000
11	ND	5.597E-005	0	0.000E+000
12	ND	5.597E-005	0	0.000E+000
13	ND	5.597E-005	0	0.000E+000
14	ND	5.598E-005	0	0.000E+000
15	ND	5.598E-005	0	0.000E+000
16	ND	5.598E-005	0	0.000E+000
17	ND	5.599E-005	0	0.000E+000
18	ND	5.599E-005	0	0.000E+000
19	F	5.599E-005	1.127E-002	6.310E-007
20	F	5.600E-005	1.230E-002	6.888E-007
21	F	5.600E-005	1.046E-002	5.858E-007
22	ND	3.200E-005	0	0.000E+000

Event tree for rubber O-ring case

- Event tree was constructed for on-site SNF transportation (23 stages) including probabilities and consequence which is based on the interpolation of FEM simulation table on previous page.
- State of each sequence : OK / ND (No Damage) / F (Failure)
- 'OK' state can occur only in 1st sequence which is success of whole on-site transportation without any drop accident, and the probability is 99.89 %.

Results : Event tree & Risk assessment



Seq#	State	Frequency	Consequence	Risk
1	OK	9.989E-001	0	0.000E+000
2	F	5.594E-005	8.183E-003	4.578E-007
3	F	5.594E-005	1.230E-002	6.881E-007
4	F	5.595E-005	9.820E-003	5.494E-007
5	ND	5.595E-005	0	0.000E+000
6	ND	3.297E-008	0	0.000E+000
7	F	5.595E-005	8.167E-003	4.569E-007
8	F	5.596E-005	1.230E-002	6.883E-007
9	F	5.596E-005	8.167E-003	4.570E-007
10	ND	5.596E-005	0	0.000E+000
11	ND	5.597E-005	0	0.000E+000
12	ND	5.597E-005	0	0.000E+000
13	F	5.597E-005	6.150E-003	3.442E-007
14	F	5.598E-005	1.230E-002	6.886E-007
15	F	5.598E-005	1.230E-002	6.886E-007
16	F	5.598E-005	6.150E-003	3.443E-007
17	F	5.599E-005	6.150E-003	3.443E-007
18	F	5.599E-005	1.230E-002	6.887E-007
19	F	5.599E-005	1.230E-002	6.887E-007
20	F	5.600E-005	1.230E-002	6.888E-007
21	F	5.600E-005	1.142E-002	6.395E-007
22	ND	3.200E-005	0	0.000E+000

Event tree for metal O-ring case

- Total risk for LPZ (Low Population Zone, 5.7km)
  - Rubber O-ring : 1.906E-06 man-mSv/transport
  - Metal O-ring : 8.413E-06 man-mSv/transport
- Most risky sequence : 3 (Moving the cask to the railing area on 13 m)
- Consequence verification : Total risk (distance)
  - NUREG-1864 : ~ 3.3E-08 man-mSv/transport (16 km)

Conclusion & Future work

- PRA of cask drop accident during on-site SNF transportation was done in this research.
- For all states of SNF cask during whole process, side drop was applied as the most conservative drop condition.
- Total risk of rubber O-ring case can be reduced more than 4 times than that of metal O-ring case.

- Floor can be applied with the realistic model instead of rigid body.
- Detailed model of fuel assembly and O-ring can be applied.
- Human error which can influence the consequence can be applied.
- Fire accident can be analyzed.

References

[1] U.S.NRC, "A Pilot PRA of a Dry Cask Storage System at a Nuclear Power Plant", NUREG-1864, 2007.

[2] Sung-Hwan Chung, Chang-yeal Baeg, Byung-Il Choi, Ke-Hyung Yang and Dae-Ki Lee, "On-site Transport and Storage of Spent Nuclear Fuel at Kori NPP by KN-12 Transport Cask", J. of Korean Radioactive Waste Society, Vol.4, p.51-58, 2006.

[3] AREAVA, "Process of Dry Storage System".

[4] Sandia National Laboratories, "A Method for Determining the Spent-Fuel Contribution to Transport Cask Containment Requirements", SAND90-2406, 1992.

[5] U.S.NRC, "Reexamination of Spent Nuclear Fuel Shipment Risk Estimates", NUREG-6672, 2000.

[6] EPRI, "Probabilistic Risk Assessment of Bolted Storage Casks", EPRI-1009691, 2004.

[7] U.S.NRC, "Spent Fuel Transportation Risk Assessment", NUREG-2125, 2012.

Thank You

Q & A

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