先進核能安全分析技術之建立與應用

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106.09.11

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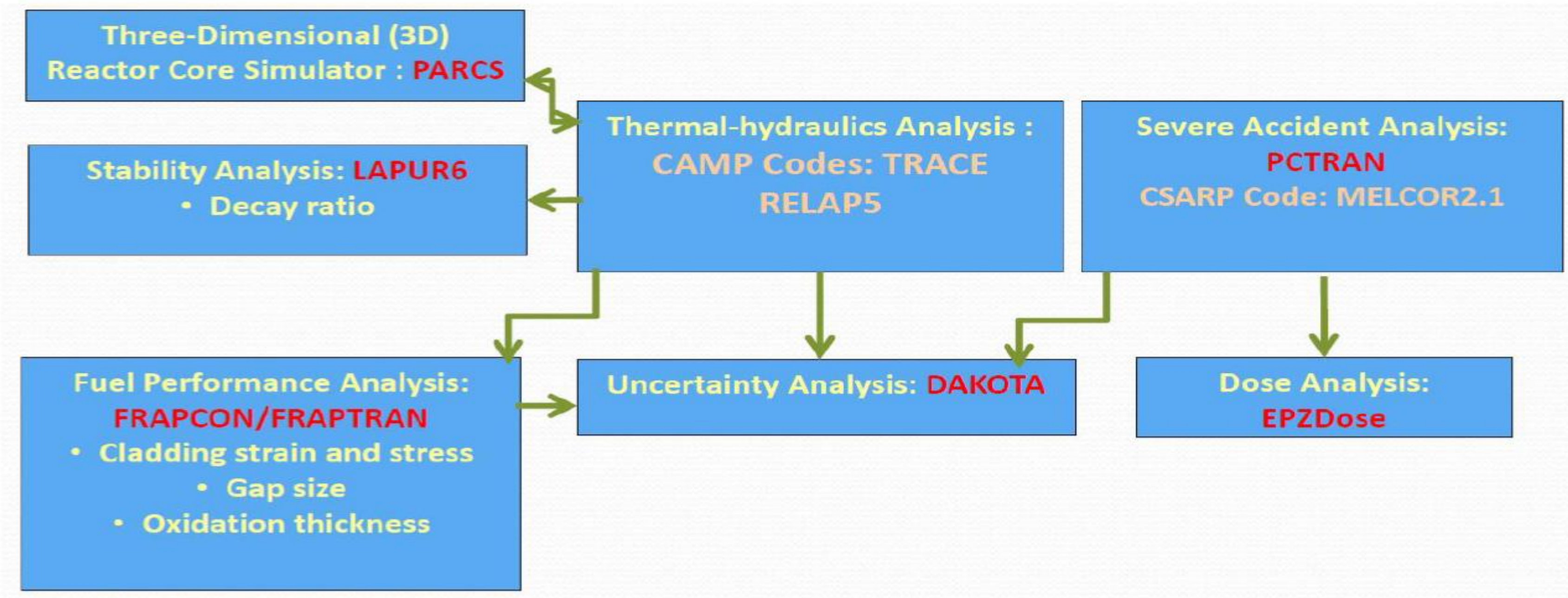
- 簡介
- 安全分析技術之建立與應用
- 結論
- 動畫展示

簡介

- 核能安全分析技術可幫助及提升核電廠之安全。
- 參與CAMP、CSARP、RAMP、DECOVALEX國際合作計畫，建立與世界各國學術團體的研究合作管道，引進相關程式 。
- 使用這些程式，發展與精進核能安全分析技術，投入相關研究與應用。

- LAPUR (Oak Ridge National Laboratory)
- TRACE (U.S. NRC)
- RELAP5 (U.S. NRC)
- PARCS (University of Michigan)
- FRAPCON/FRAPTRAN (Pacific Northwest National Laboratory)
- DAKOTA (Sandia National Laboratories)
- MELCOR (Sandia National Laboratories)
- PCTRAN (Micro-Simulation Technology)
- EPZDose
- CFD
- SAPHIRE (U.S. NRC)
- SCALE (Nuclear Energy Agency)
- TOUGH2 (Lawrence Berkeley National Laboratory)
- ALOHA (National Oceanic and Atmospheric Administration (NOAA))
- RESRAD(U.S. NRC)
- RAMP Codes



RAMP Codes

- 反應器氣體及液體放射性物質排放評估 (GALE)
- 生物圈之輻射劑量及風險評估 (GENII)
- 皮膚污染劑量評估 (VARSKIN)

- 放射性核種遷移、移除與劑量評估 (RADTRAD)
- 電廠人員與民眾的即時劑量評估 (Radiological Toolbox)
- 電廠除役後場址內殘留輻射評估 (DandD)

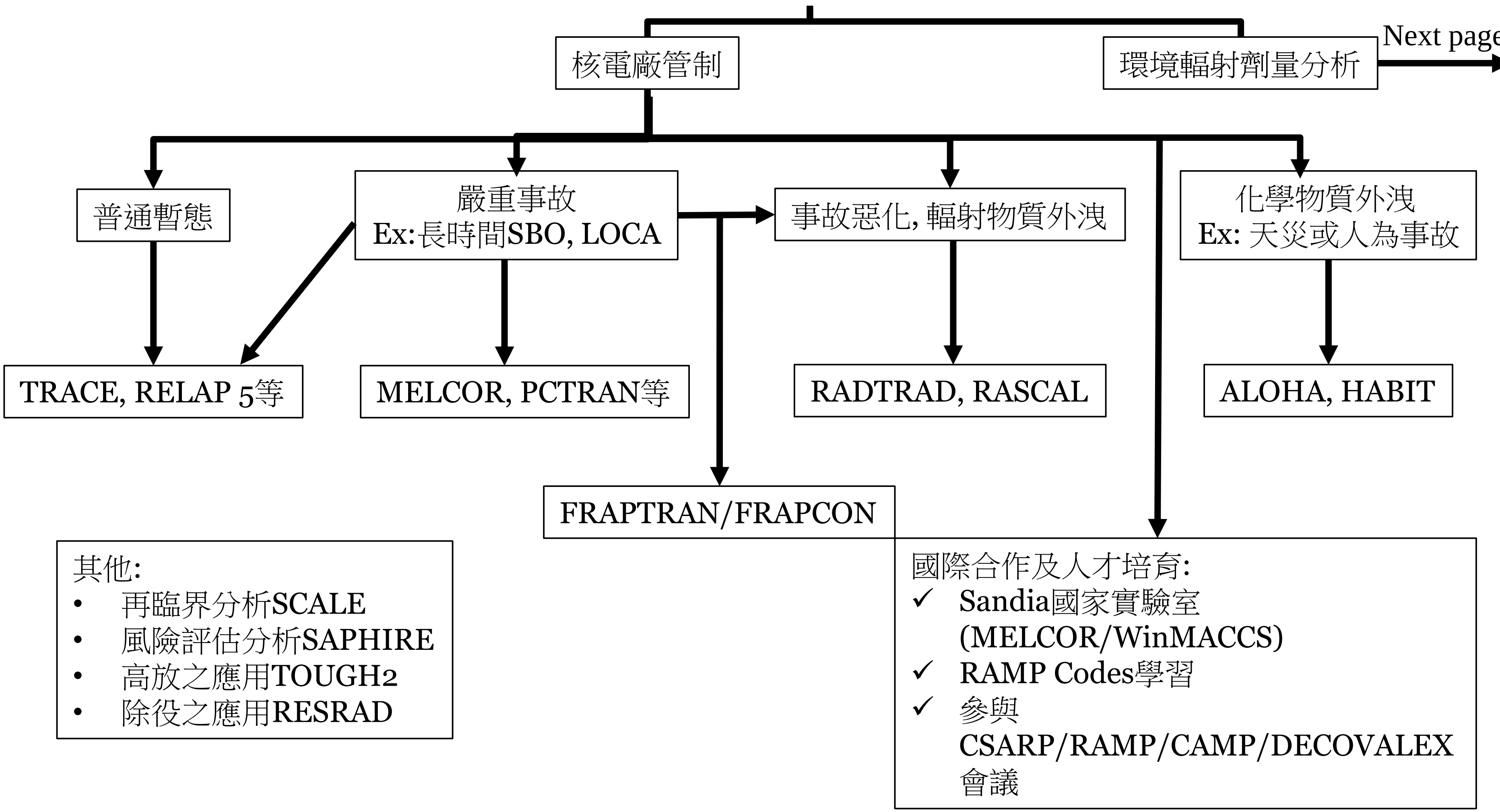
- 電廠與相關設備輻射外洩後果評估 (RASCAL)
- 人體器官輻射劑量估算工具 (PIMAL)
- 控制室適居性評估程式HABIT之介紹與應用

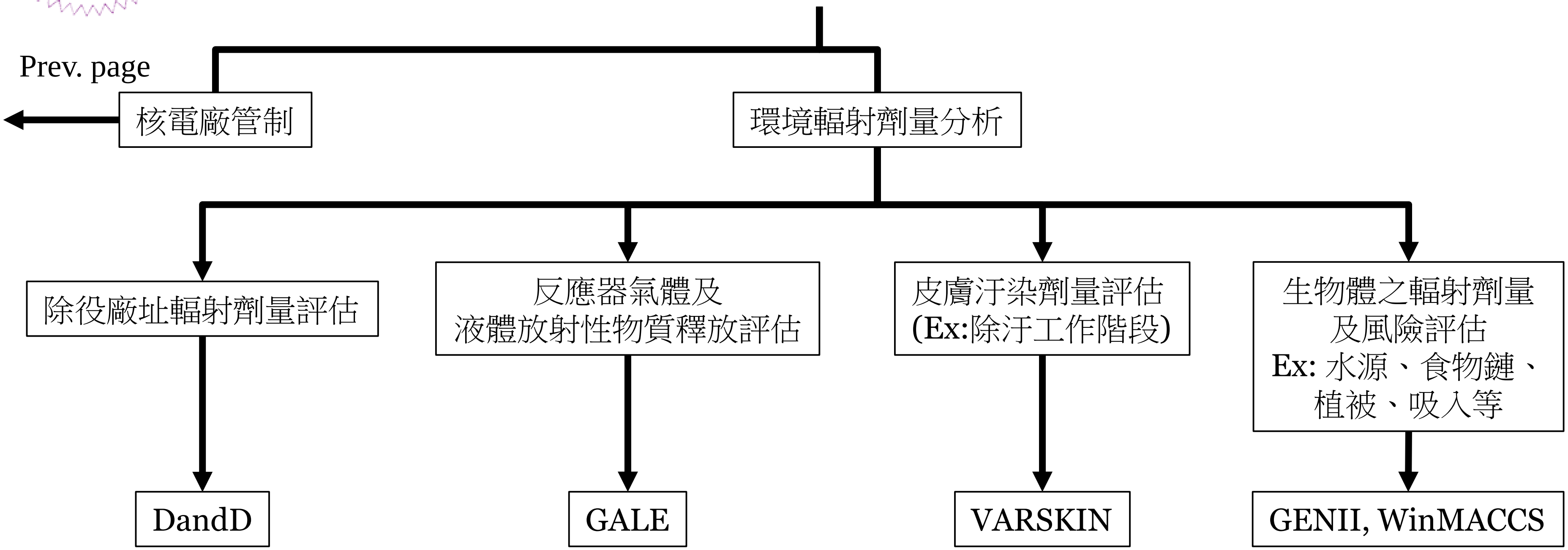
- SAPHIRE 風險評估分析模式
- SCALE 再臨界分析模式
- CFD 於核能安全之相關應用
- ALOHA 毒性化學物質之濃度評估
- RESRAD 除役之相關應用

DECOVALEX

- TOUGH2 於高放射性廢棄物最終處置之相關應用

安全分析技術之建立與應用




Prev. page
核電廠管制
環境輻射劑量分析
除役廠址輻射劑量評估
反應器氣體及
液體放射性物質釋放評估
皮膚汙染劑量評估
(Ex:除汙工作階段)
生物體之輻射劑量
及風險評估
Ex: 水源、食物鏈、
植被、吸入等
DandD
GALE
VARSKIN
GENII, WinMACCS

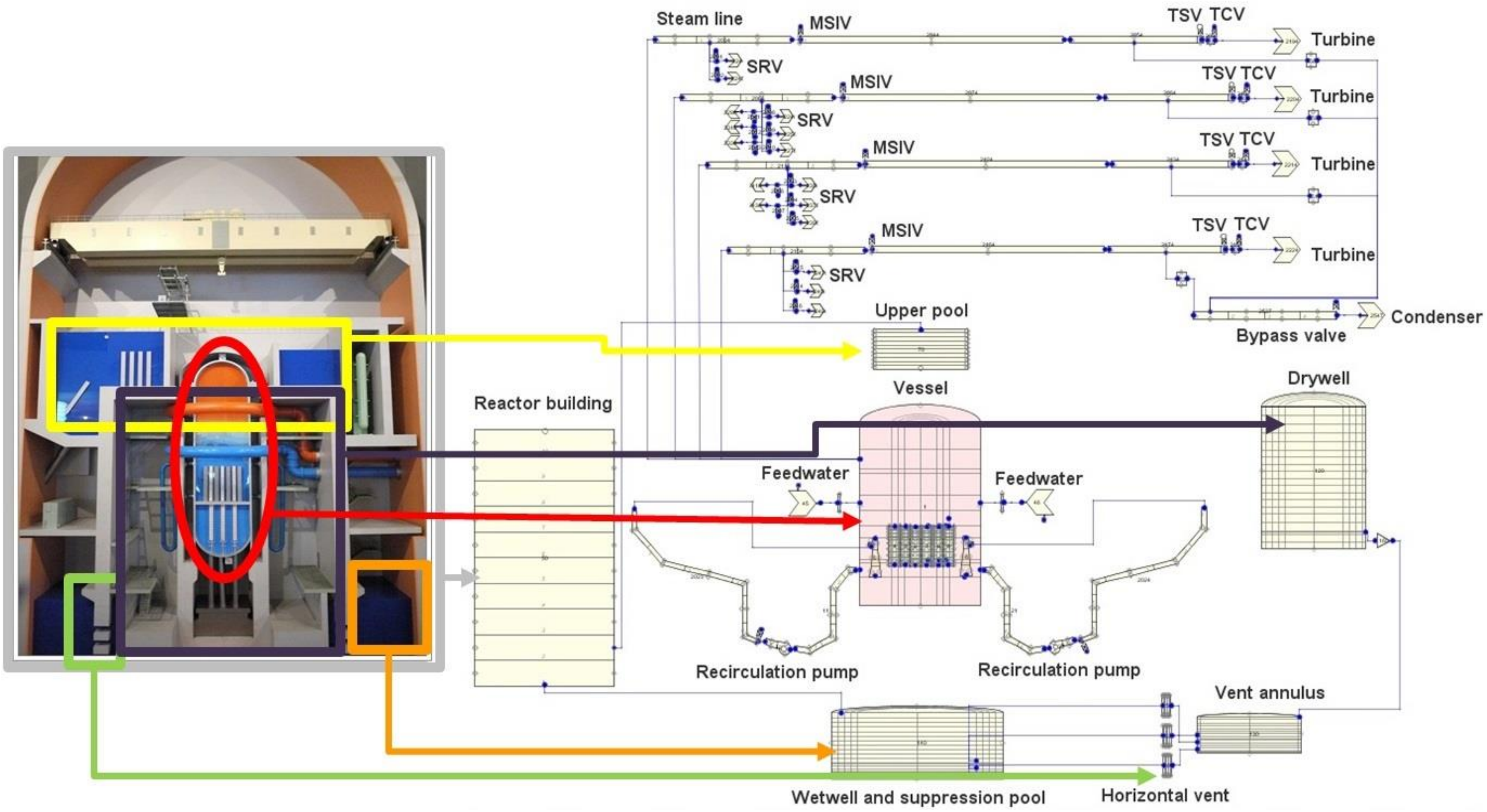
- 美國核管會（U.S. NRC）正在發展一套全新且先進的核電廠熱水流安全分析程式TRACE，並成立一國際合作計畫「CAMP」致力改良與發展TRACE程式
- 同時亦另發展一圖形化介面程式SNAP， 來處理TRACE的相關計算及動畫展示
- 本研究室已參與此項國際合作計畫，積極投入人力參與TRACE的相關研究

NUREG/IA's published (2009-2017)
Total of 116

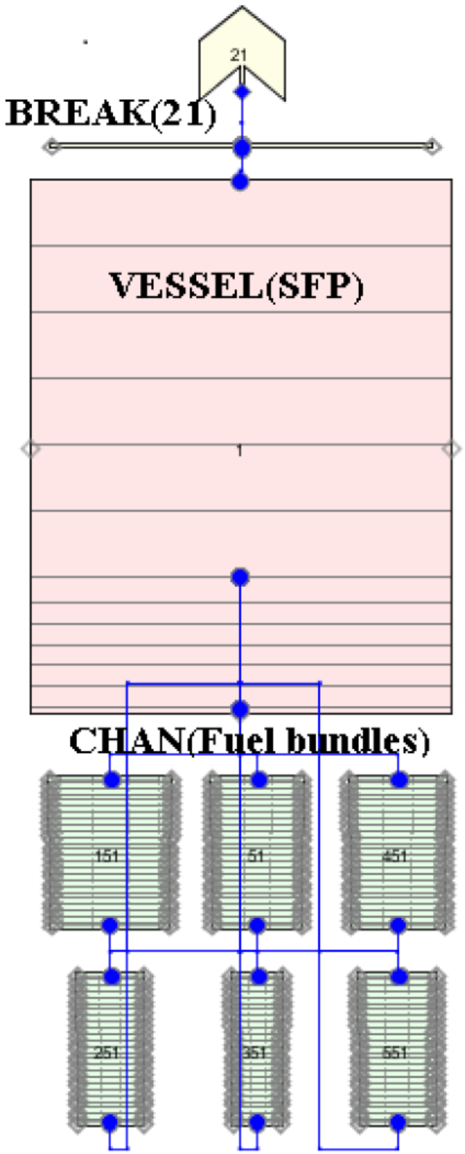
- Spain – 26
- Slovenia – 13
- Finland – 6
- Germany – 9
- Korea – 7
- Czech Republic – 6
- Republic of China – 27
- Sweden – 5
- Argentina – 1
- Canada – 4
- Italy - 3
- People's Republic of China -2
- Poland - 4
- United Kingdom – 1
- Switzerland – 2
- Croatia – 1
- Ukraine - 1

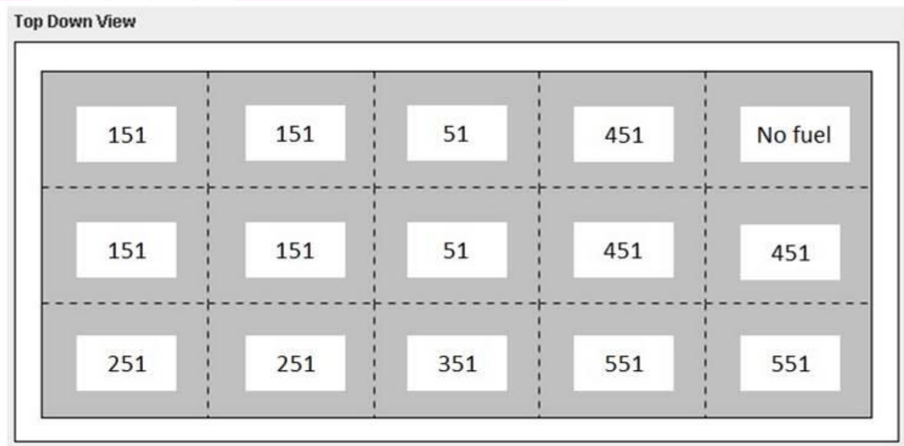
U.S.NRC

核二廠TRACE model(含圍阻體)

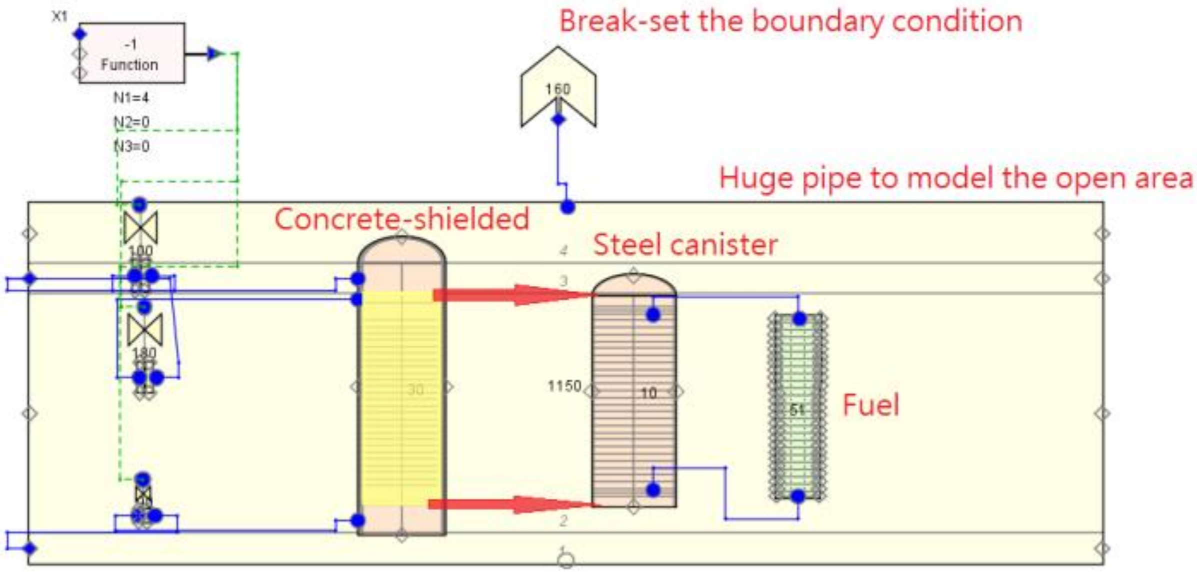


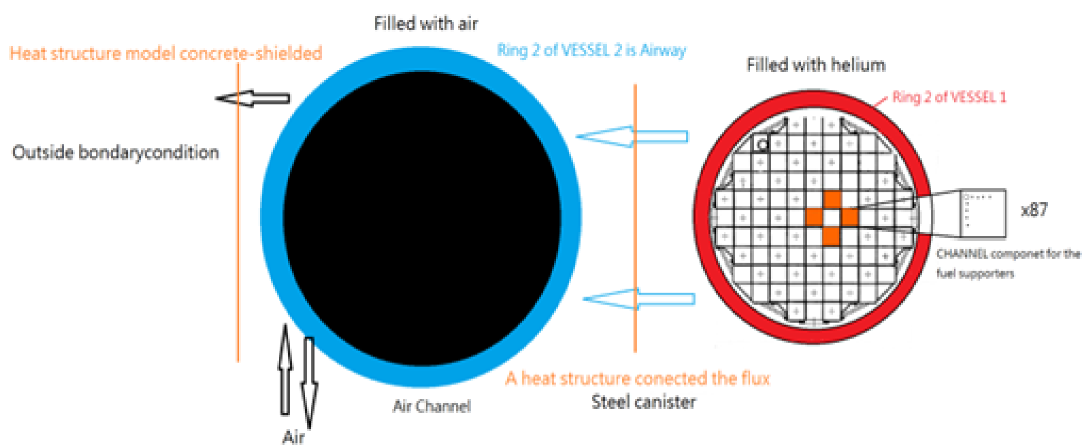
核一廠用過燃料池TRACE model



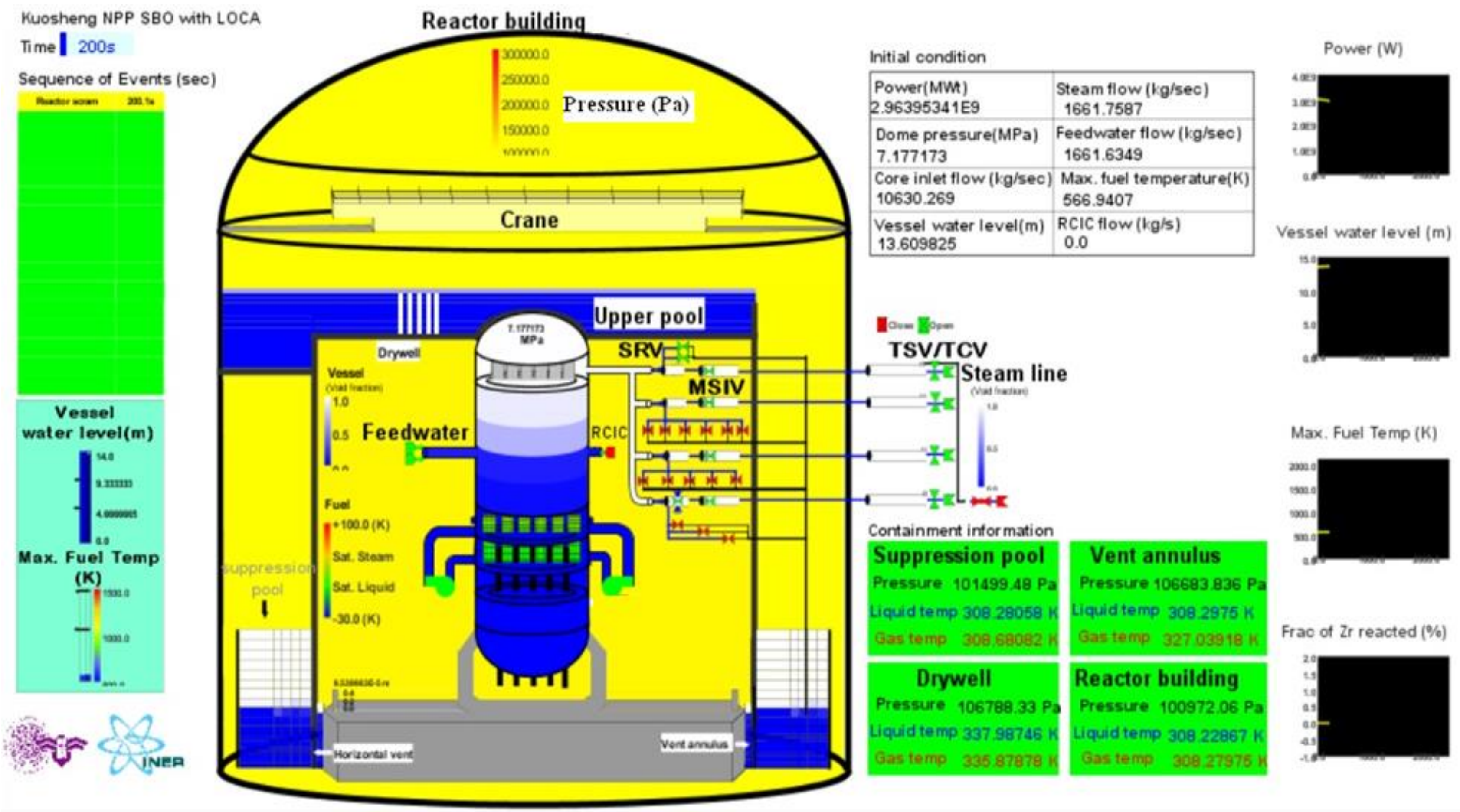


核二廠乾式貯存系統TRACE model

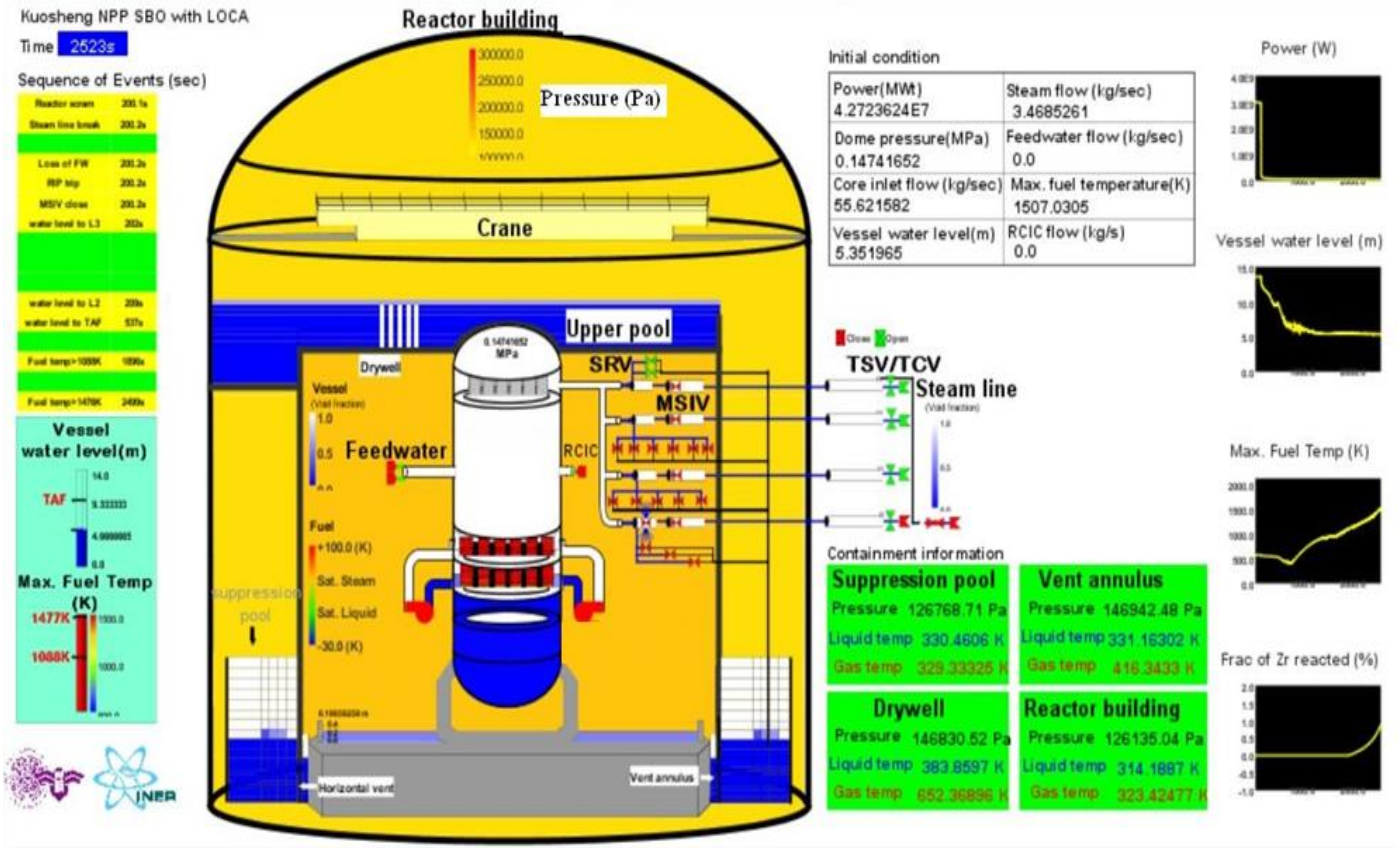


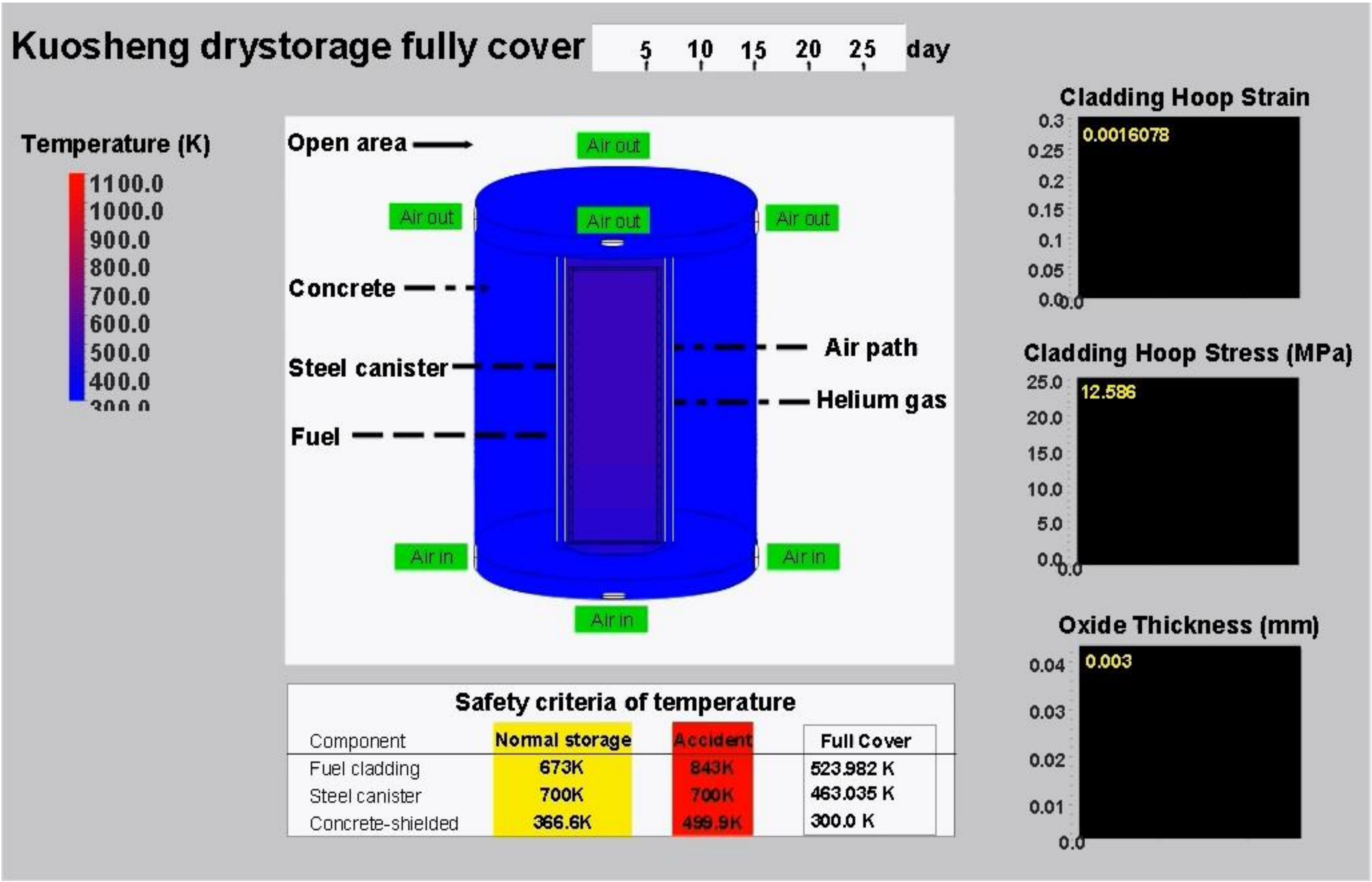


- The Ultimate Response Guideline Simulation and Analysis Using TRACE,MAAP5,and FRAPTRAN for the Chinshan Nuclear Power Plant, Annals of Nuclear Energy, Vol. 103, pp. 402-411, 2017.
- TRACE/FRAPTRAN Analysis of Kuosheng Nuclear Power Plant Dry-Storage System, International Journal of Chemical, Molecular, Nuclear, Materials and Metallurgical Engineering Vol:8, No:7, 2014.
- The Study of Ultimate Response Guideline of Kuosheng BWR/6 Nuclear Power Plant Using TRACE and SNAP, World Academy of Science, Engineering and Technology, International Journal of Chemical, Molecular, Nuclear, Materials and Metallurgical Engineering Vol:11, No:4, 2017.
- Analysis of A Postulated ELAP Event in Maanshan NPP Using TRACE Code, TopSafe2017, 12 - 16 February 2017 in Vienna, Austria.

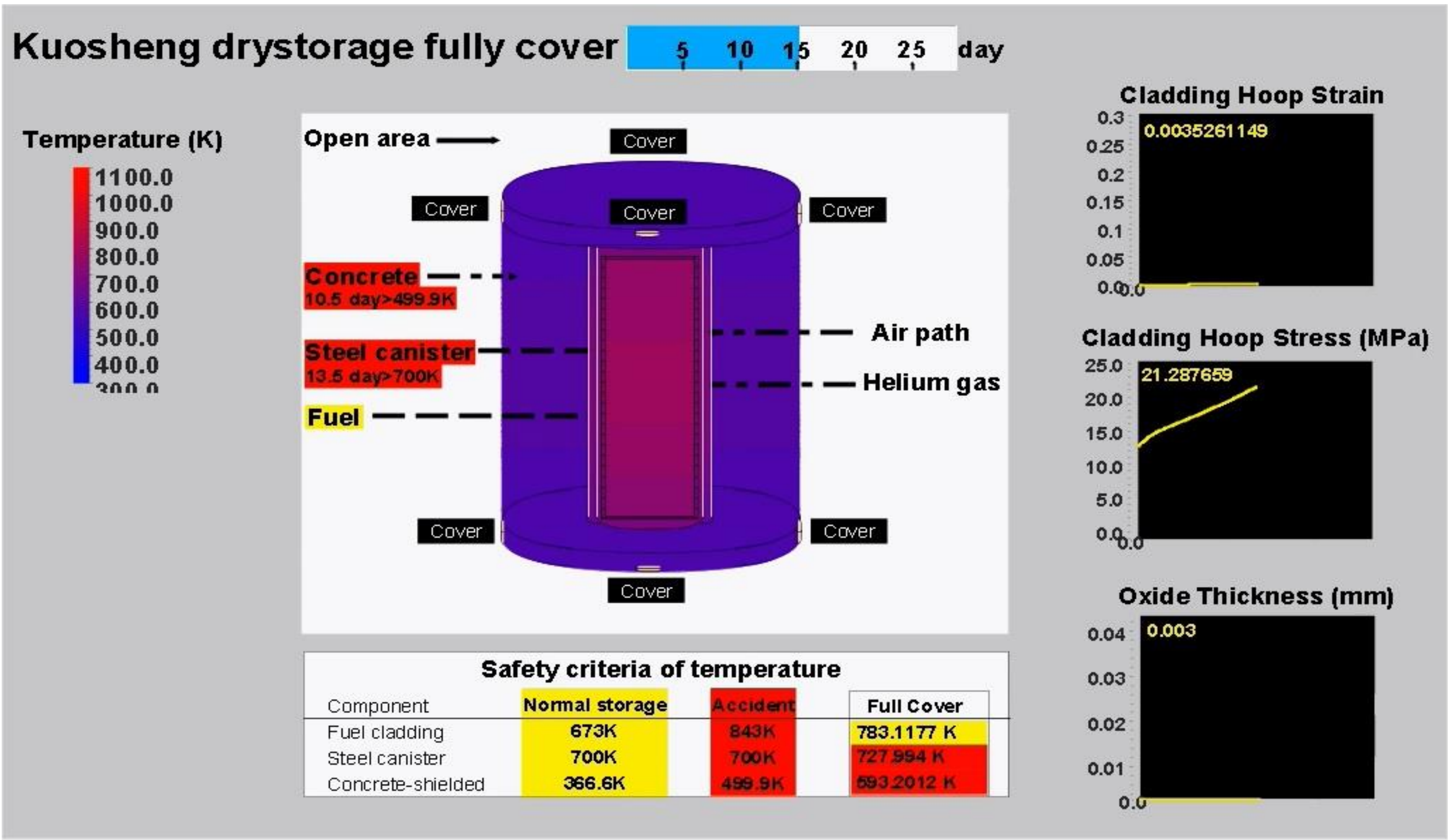


Kuosheng NPP SBO with LOCA
Time 200s
Sequence of Events (sec)
Vessel water level(m)
Max. Fuel Temp (K)
Reactor building
Pressure (Pa)
Crane
Upper pool
Drywell
SRV
MSIV
Feedwater
RCIC
Suppression pool
Horizontal vent
Vent annulus
TSV/TCV
Steam line
Initial condition
Power(MWt) 2.96395341E9
Steam flow (kg/sec) 1661.7587
Dome pressure(MPa) 7.177173
Feedwater flow (kg/sec) 1661.6349
Core inlet flow (kg/sec) 10630.269
Max. fuel temperature(K) 566.9407
Vessel water level(m) 13.609825
RCIC flow (kg/s) 0.0
Containment information
Suppression pool
Pressure 101499.48 Pa
Liquid temp 308.28058 K
Gas temp 308.68082 K
Vent annulus
Pressure 106683.836 Pa
Liquid temp 308.2975 K
Gas temp 327.03918 K
Drywell
Pressure 106788.33 Pa
Liquid temp 337.98746 K
Gas temp 335.87878 K
Reactor building
Pressure 100972.06 Pa
Liquid temp 308.22867 K
Gas temp 308.27975 K
Power (W)
Vessel water level (m)
Max. Fuel Temp (K)
Frac of Zr reacted (%)
INER

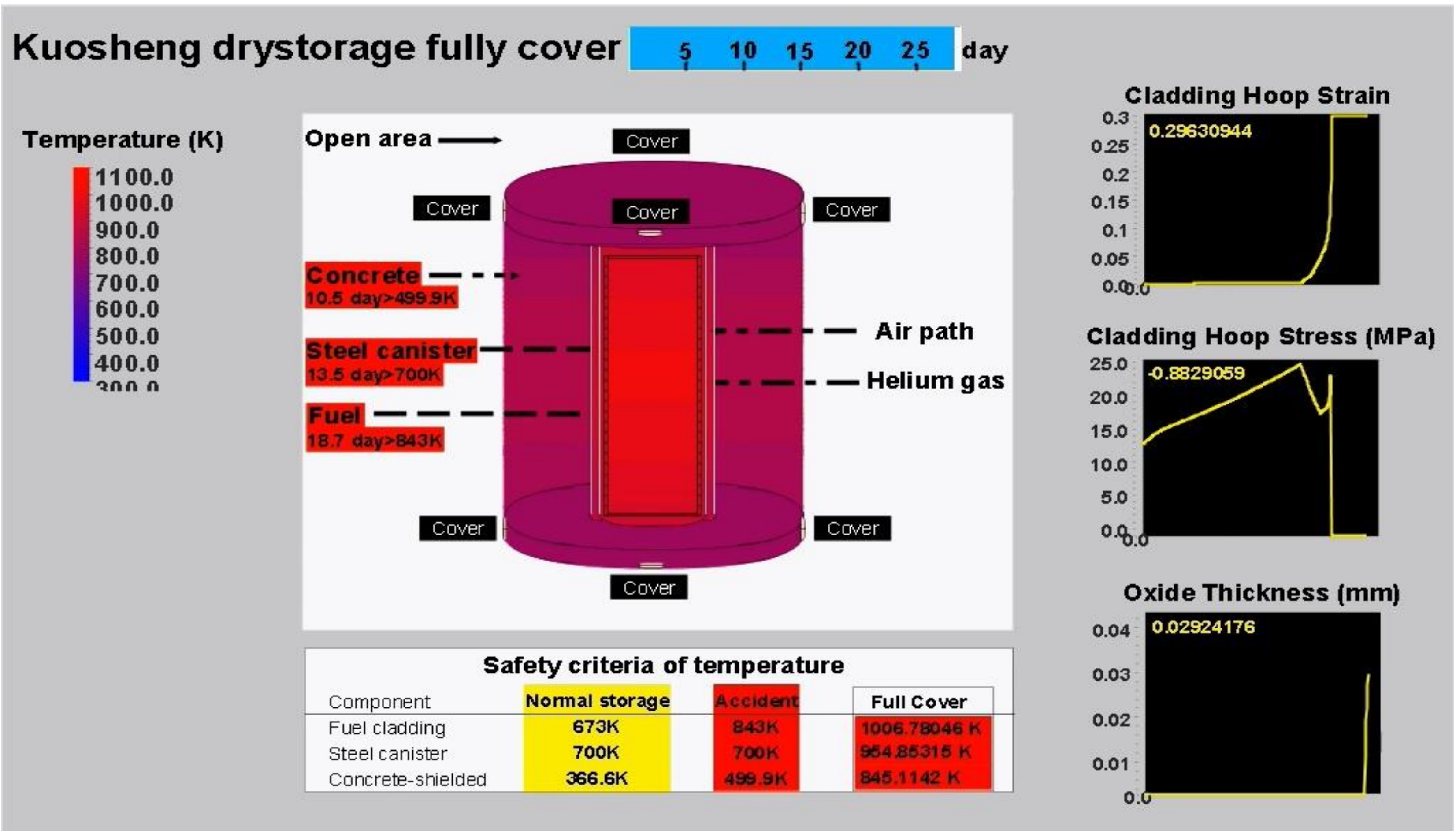

Kuosheng NPP SBO with LOCA
Time 2523s
Sequence of Events (sec)
Reactor building
Pressure (Pa)
Crane
Upper pool
Drywell
SRV
MSIV
Feedwater
RCIC
Vessel water level(m)
TAF
Max. Fuel Temp (K)
1477K
1088K
Initial condition
Power(MWt) 4.2723624E7
Steam flow (kg/sec) 3.4685261
Dome pressure(MPa) 0.14741652
Feedwater flow (kg/sec) 0.0
Core inlet flow (kg/sec) 55.621582
Max. fuel temperature(K) 1507.0305
Vessel water level(m) 5.351965
RCIC flow (kg/s) 0.0
TSV/TCV
Steam line
Containment information
Suppression pool
Pressure 126768.71 Pa
Liquid temp 330.4606 K
Gas temp 329.33325 K
Vent annulus
Pressure 146942.48 Pa
Liquid temp 331.16302 K
Gas temp 416.3433 K
Drywell
Pressure 146830.52 Pa
Liquid temp 383.8597 K
Gas temp 652.36896 K
Reactor building
Pressure 126135.04 Pa
Liquid temp 314.1887 K
Gas temp 323.42477 K
Power (W)
Vessel water level (m)
Max. Fuel Temp (K)
Frac of Zr reacted (%)
Horizontal vent
Vent annulus
suppression pool



Component	Normal storage	Accident	Full Cover
Fuel cladding	673K	843K	523.982 K
Steel canister	700K	700K	463.035 K
Concrete-shielded	366.6K	499.9K	300.0 K

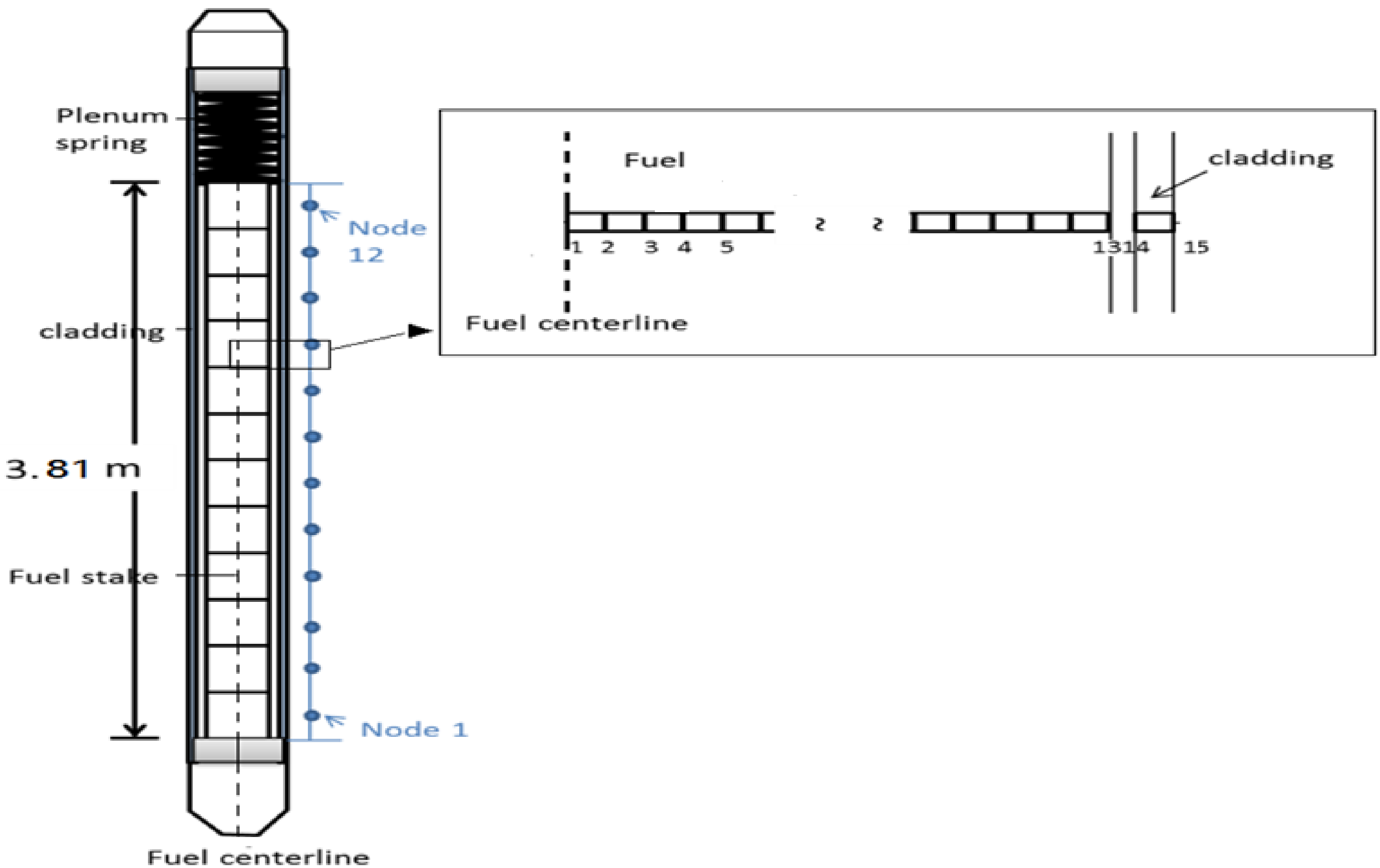


Component	Normal storage	Accident	Full Cover
Fuel cladding	673K	843K	783.1177 K
Steel canister	700K	700K	727.994 K
Concrete-shielded	366.6K	499.9K	593.2012 K


Kuosheng drystorage fully cover
5 10 15 20 25 day
Temperature (K)
1100.0
1000.0
900.0
800.0
700.0
600.0
500.0
400.0
Open area
Cover
Concrete
10.5 day>499.9K
Steel canister
13.5 day>700K
Fuel
18.7 day>843K
Air path
Helium gas
Cladding Hoop Strain
0.29630944
Cladding Hoop Stress (MPa)
-0.8829059
Oxide Thickness (mm)
0.02924176
Safety criteria of temperature
Component
Normal storage
Accident
Full Cover
Fuel cladding 673K 843K 1006.78046 K
Steel canister 700K 700K 954.85315 K
Concrete-shielded 366.6K 499.9K 845.1142 K

- 在FRAPCON/FRAPTRAN分析方面，可結合TRACE的分析結果，進行在特定暫態下之燃料棒性質分析
- FRAPCON/FRAPTRAN分析結果可得到相關之燃料棒參數如:護套溫度、燃料丸溫度、應力或應變等機械性質

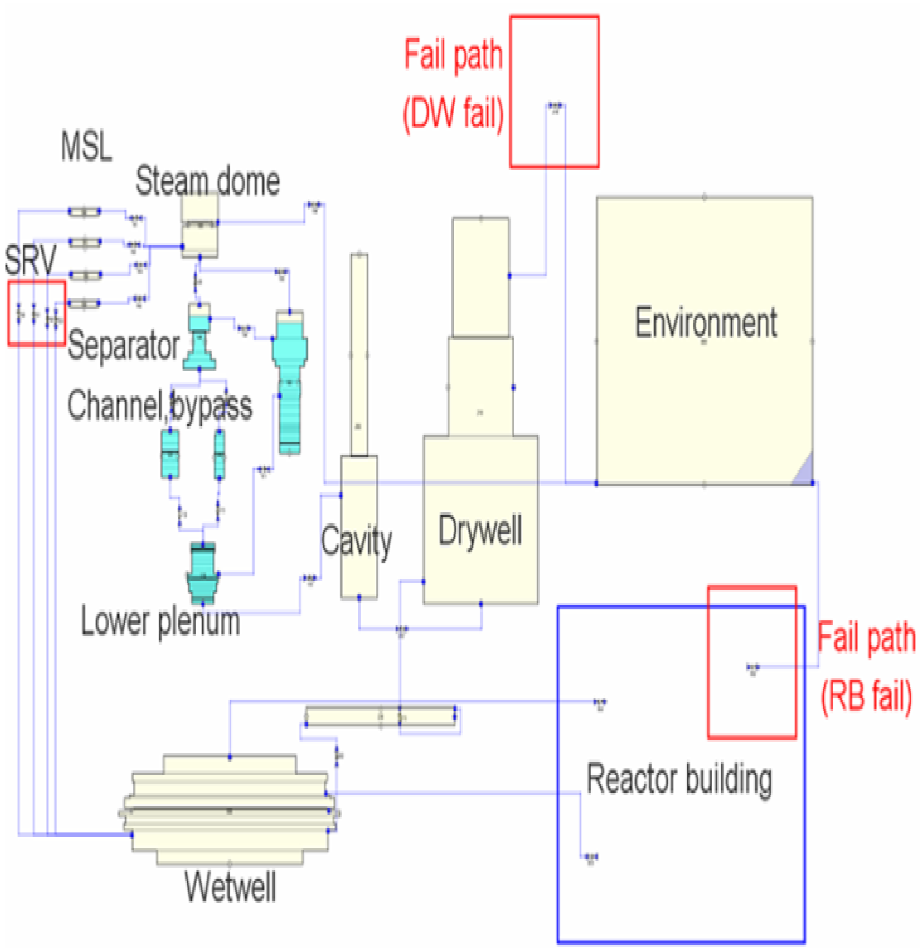
FRAPTRAN fuel rod model

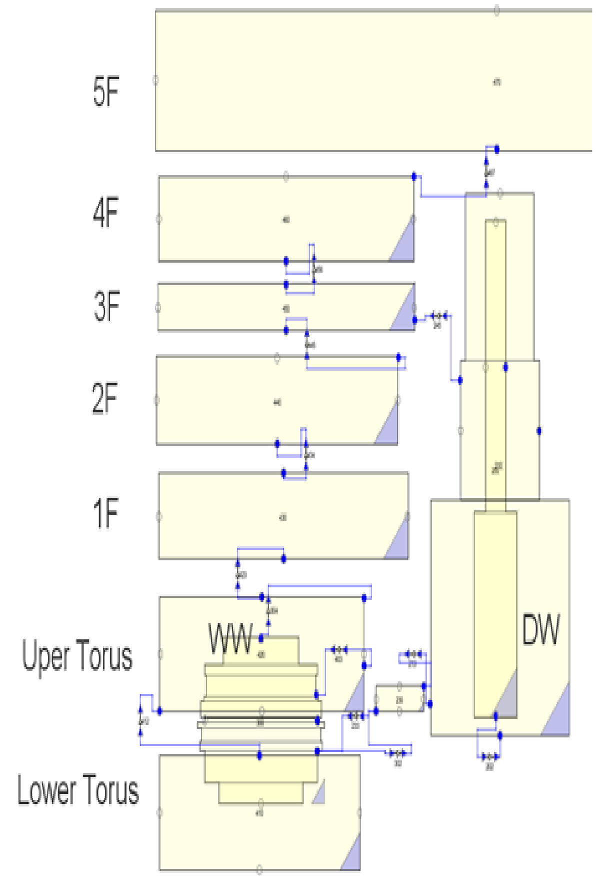


- The Analysis of Kuosheng Nuclear Power Plant Spent Fuel Pool by Using FRAPTRAN-2.0, ICAPP 2017, April 24-28, 2017 / FUKUI and KYOTO,JAPAN.
- Application of TRACE and FRAPTRAN in the Spent Fuel Pool of Chinshan Nuclear Power Plant, Applied Mechanics and Materials, Vols. 479-480, pp. 543-547, 2014.
- The Analysis of TRACE/FRAPTRAN in the Fuel Rods of Maanshan PWR for LBLOCA, World Academy of Science, Engineering and Technology, Vol. 85, pp. 52-57, 2014.
- Fuel Rod Behavior and Uncertainty Analysis by FRAPTRAN/TRACE/DAKOTA Code in Maanshan LBLOCA, NUREG/IA-0471, 2016.

- 國際合作計畫「CSARP」致力改良與發展MELCOR程式
- 圖形化介面程式SNAP可處理MELCOR的相關計算及動畫展示
- 本研究室已參與此項國際合作計畫，積極投入人力參與MELCOR的相關研究

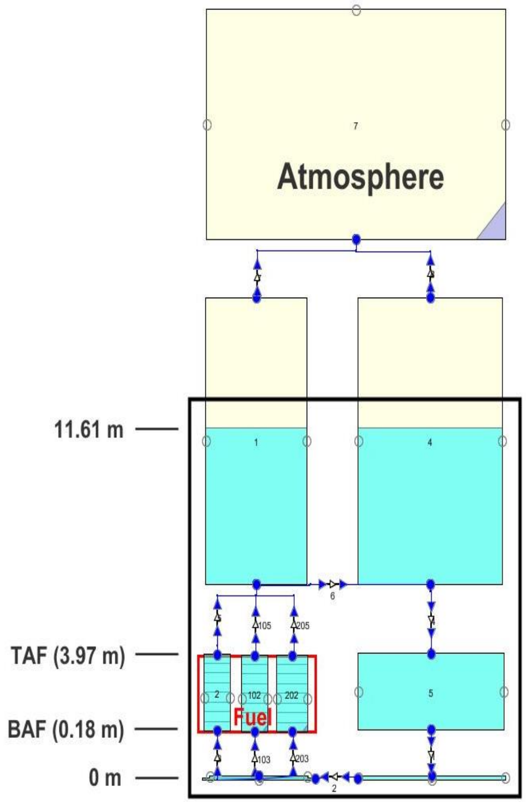
核一廠MELCOR model(含圍阻體)



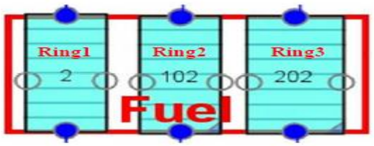


Pressure settings of failure path
Drywell to reactor building 3F
5 Atm
Reactor building 5F to environment
1.5 Atm

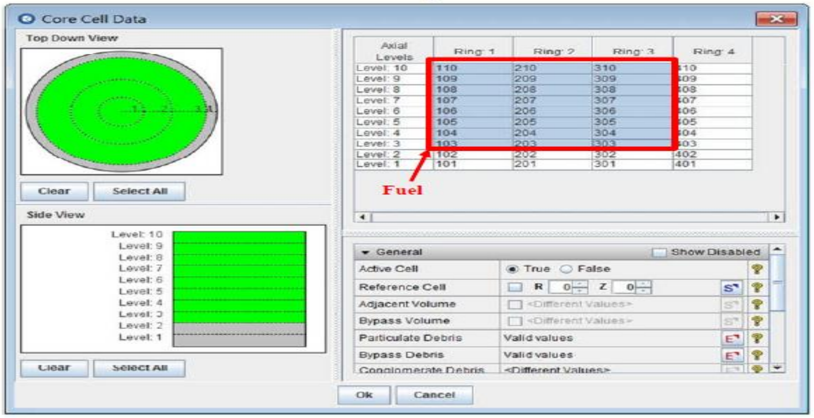
核一廠用過燃料池MELCOR model

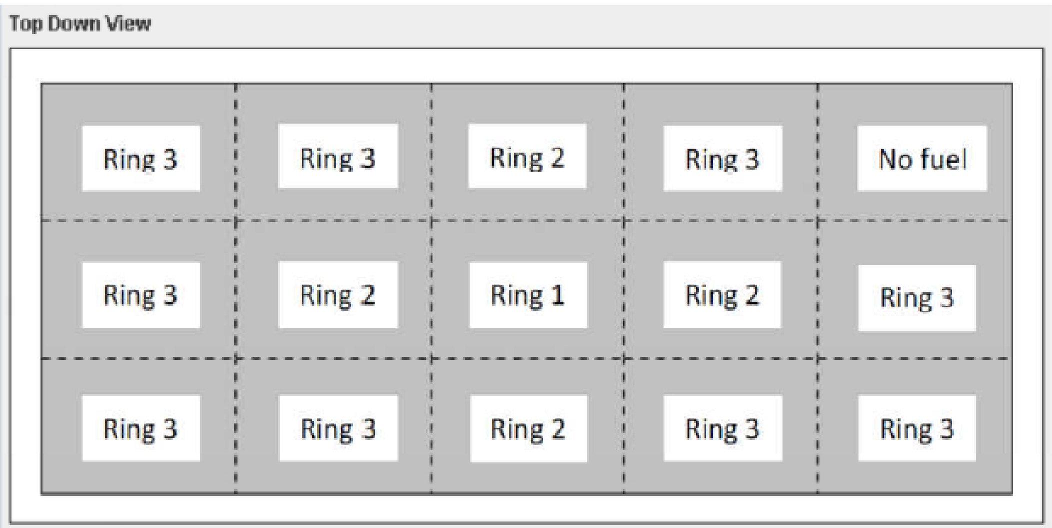


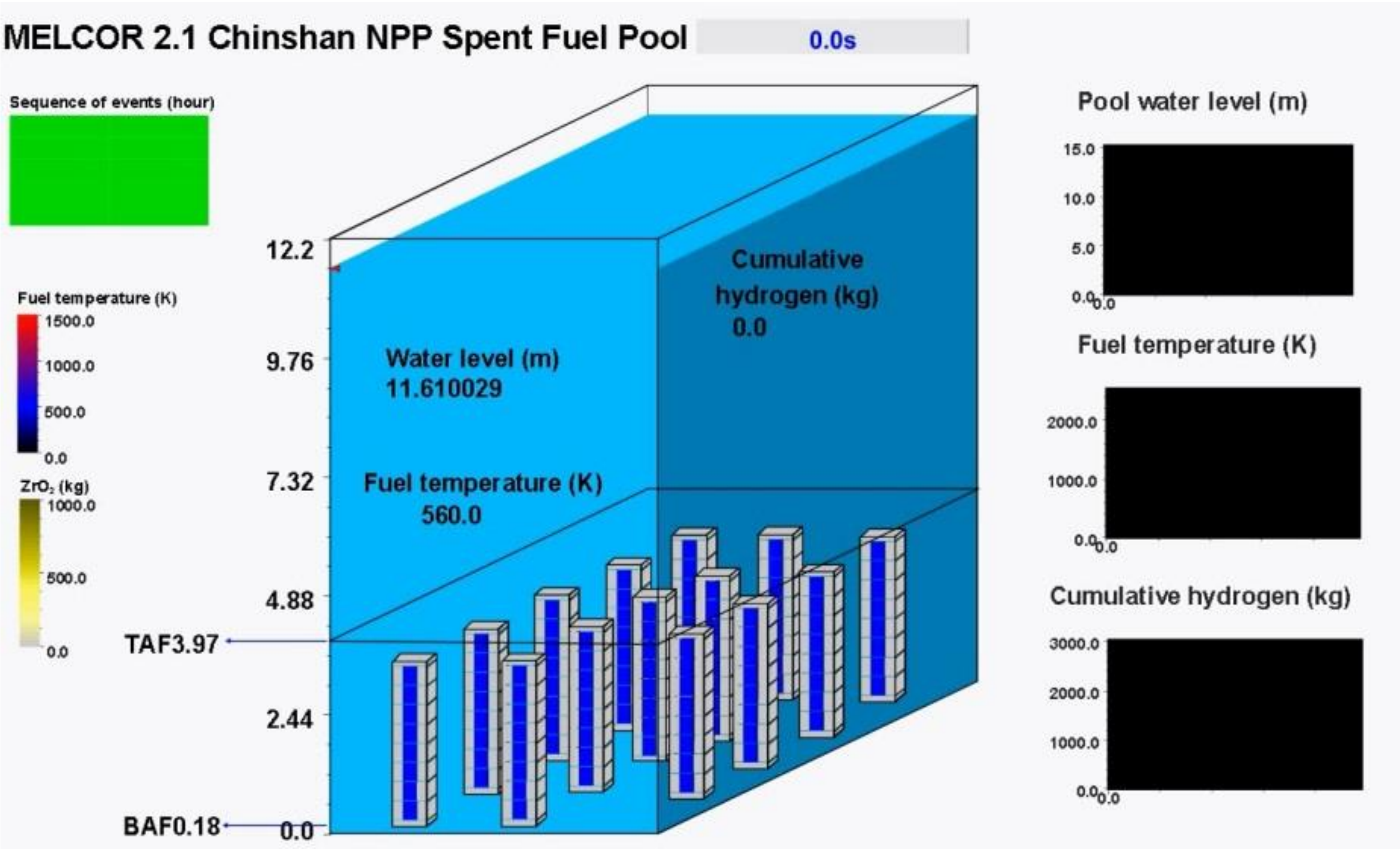
Spent fuel pool



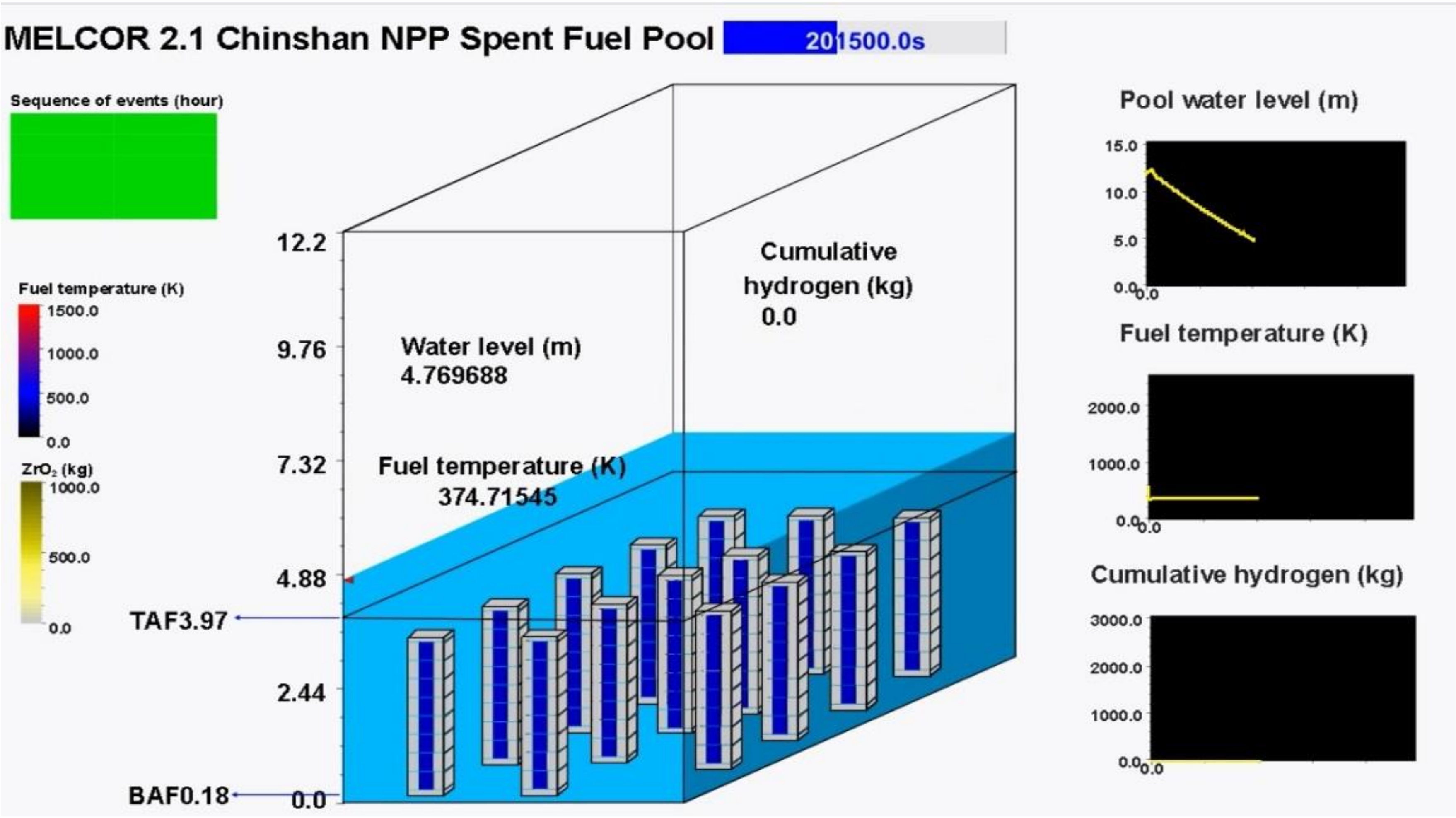
CVH component



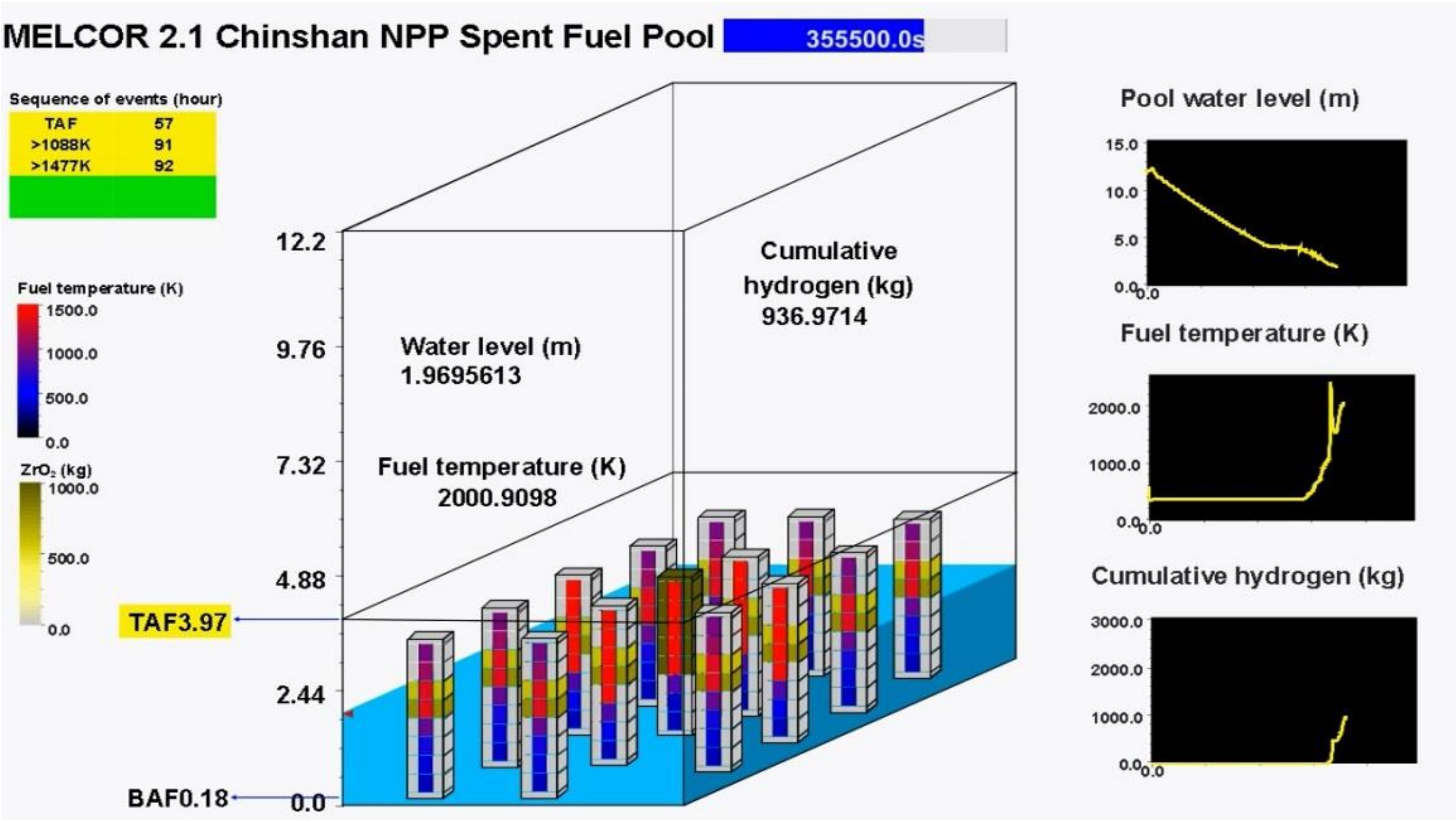




MELCOR 2.1 Chinshan NPP Spent Fuel Pool
0.0s
Sequence of events (hour)
Fuel temperature (K)
1500.0
1000.0
500.0
0.0
ZrO2 (kg)
1000.0
500.0
0.0
12.2
9.76
7.32
4.88
2.44
0.0
TAF3.97
BAF0.18
Water level (m)
11.610029
Fuel temperature (K)
560.0
Cumulative
hydrogen (kg)
0.0
Pool water level (m)
15.0
10.0
5.0
0.0
0.0
Fuel temperature (K)
2000.0
1000.0
0.0
0.0
Cumulative hydrogen (kg)
3000.0
2000.0
1000.0
0.0
0.0



MELCOR 2.1 Chinshan NPP Spent Fuel Pool
201500.0s
Sequence of events (hour)
Fuel temperature (K)
1500.0
1000.0
500.0
0.0
ZrO₂ (kg)
1000.0
500.0
0.0
12.2
9.76
7.32
4.88
2.44
0.0
TAF3.97
BAF0.18
Water level (m)
4.769688
Fuel temperature (K)
374.71545
Cumulative
hydrogen (kg)
0.0
Pool water level (m)
15.0
10.0
5.0
0.0
0.0
Fuel temperature (K)
2000.0
1000.0
0.0
0.0
Cumulative hydrogen (kg)
3000.0
2000.0
1000.0
0.0
0.0

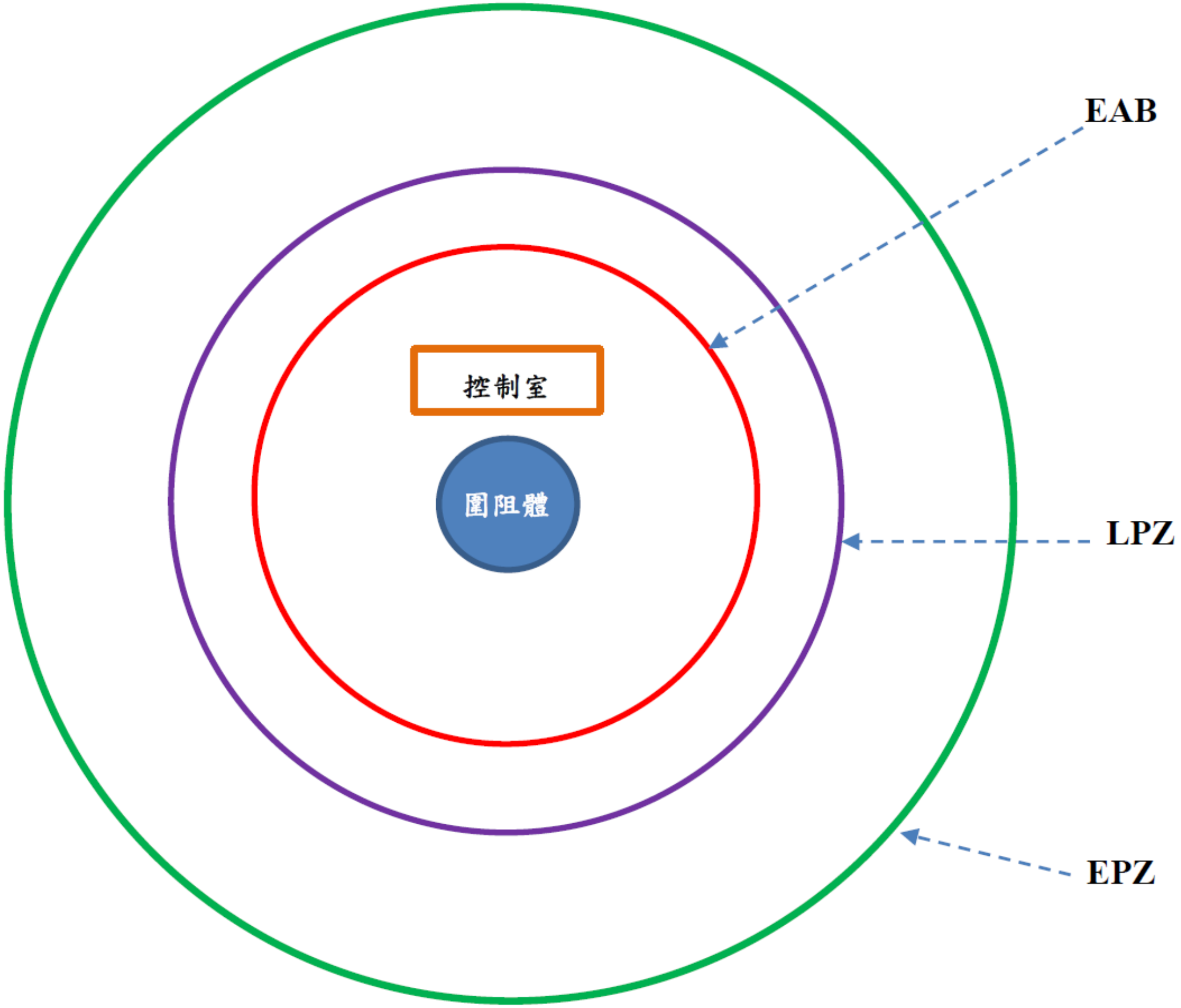

MELCOR 2.1 Chinshan NPP Spent Fuel Pool
355500.0s
Sequence of events (hour)
TAF 57
>1088K 91
>1477K 92
Fuel temperature (K)
1500.0
1000.0
500.0
0.0
ZrO2 (kg)
1000.0
500.0
0.0
TAF3.97
BAF0.18
12.2
9.76
7.32
4.88
2.44
0.0
Water level (m)
1.9695613
Fuel temperature (K)
2000.9098
Cumulative hydrogen (kg)
936.9714
Pool water level (m)
15.0
10.0
5.0
0.0
0.0
Fuel temperature (K)
2000.0
1000.0
0.0
0.0
Cumulative hydrogen (kg)
3000.0
2000.0
1000.0
0.0
0.0

- The Model Establishment and Analysis of TRACE/MELCOR for Kuosheng Nuclear Power Plant Spent Fuel Pool, International Journal of Chemical, Molecular, Nuclear, Materials and Metallurgical Engineering Vol:10, No:10, 2016.
- The Mitigation Strategy Analysis of Kuosheng Nuclear Power Plant Spent Fuel Pool Using MELCOR2.1/SNAP, International Journal of Chemical, Molecular, Nuclear, Materials and Metallurgical Engineering Vol:11, No:4, 2017.
- Uncertainty Analysis for Chinshan (BWR/4) Spent Fuel Pool Severe Accident By MELCOR2.1/SNAP and DAKOTA, in 2015 Top Fuel. 2015:Zurich.
- MELCOR Analysis on Hydrogen Behaviors of Chinshan NPP Spent Fuel Pool, in PSAM-13. 2016: Korea.

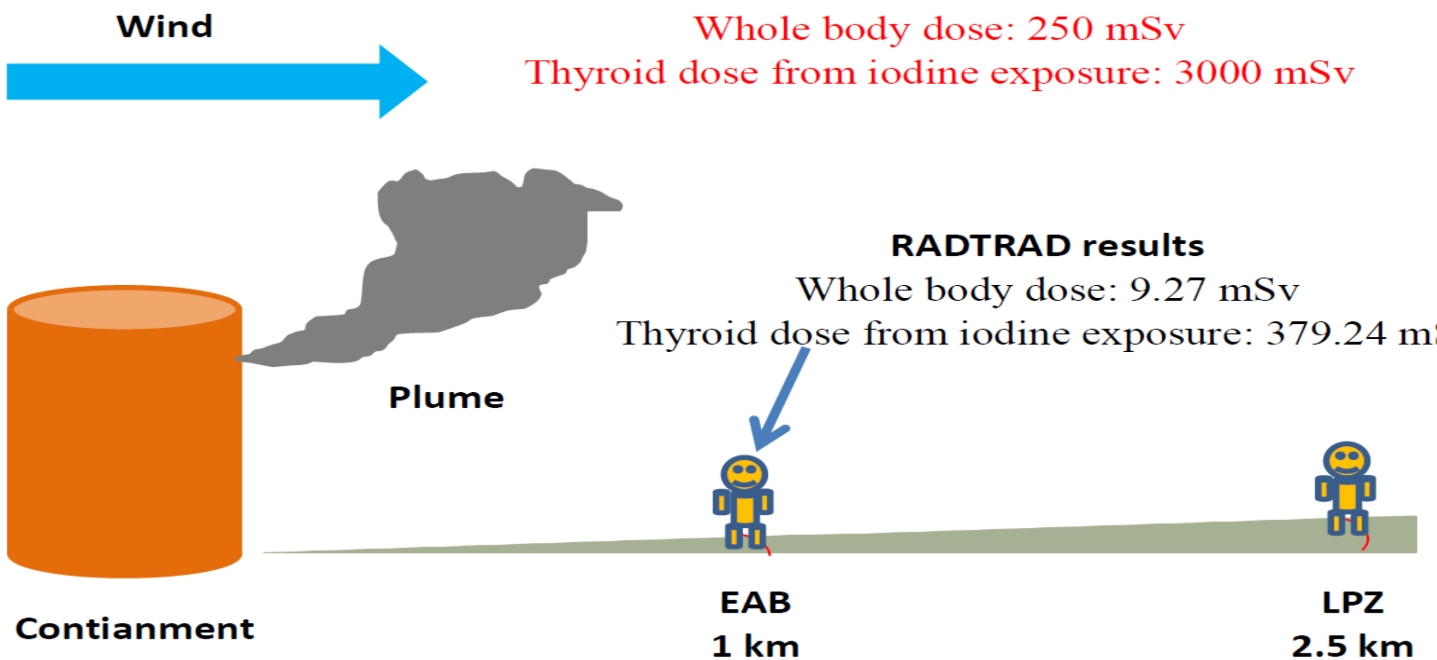
- 本研究室於2016年開始加入RAMP國際合作計畫，積極投入相關人力，使用RAMP的相關程式，進行核能電廠之輻射劑量評估、除役、大氣擴散因子等方面的相關研究
- RAMP相關的程式如下：

RADTRAD、HABIT、RASCAL、DAND、GENII、GALE、PIMAL、VARSKIN、ARCON、PAVEN等。

- RADTRAD (RADionuclide, Transport, Removal And Dose Estimation Computer Code, 放射性物質遷移、移除及劑量評估簡算程式)是由美國聖地亞國家實驗室(Sandia National. Laboratory, SNL)所研發，可進行放射性物質遷移、移除及劑量評估之計算，主要用於評估廠界劑量，包括EAB、LPZ，以及核電廠內控制室的職業輻射曝露。
- 圖形化介面程式SNAP可處理RADTRAD的相關計算


EAB
控制室
圍阻體
LPZ
EPZ

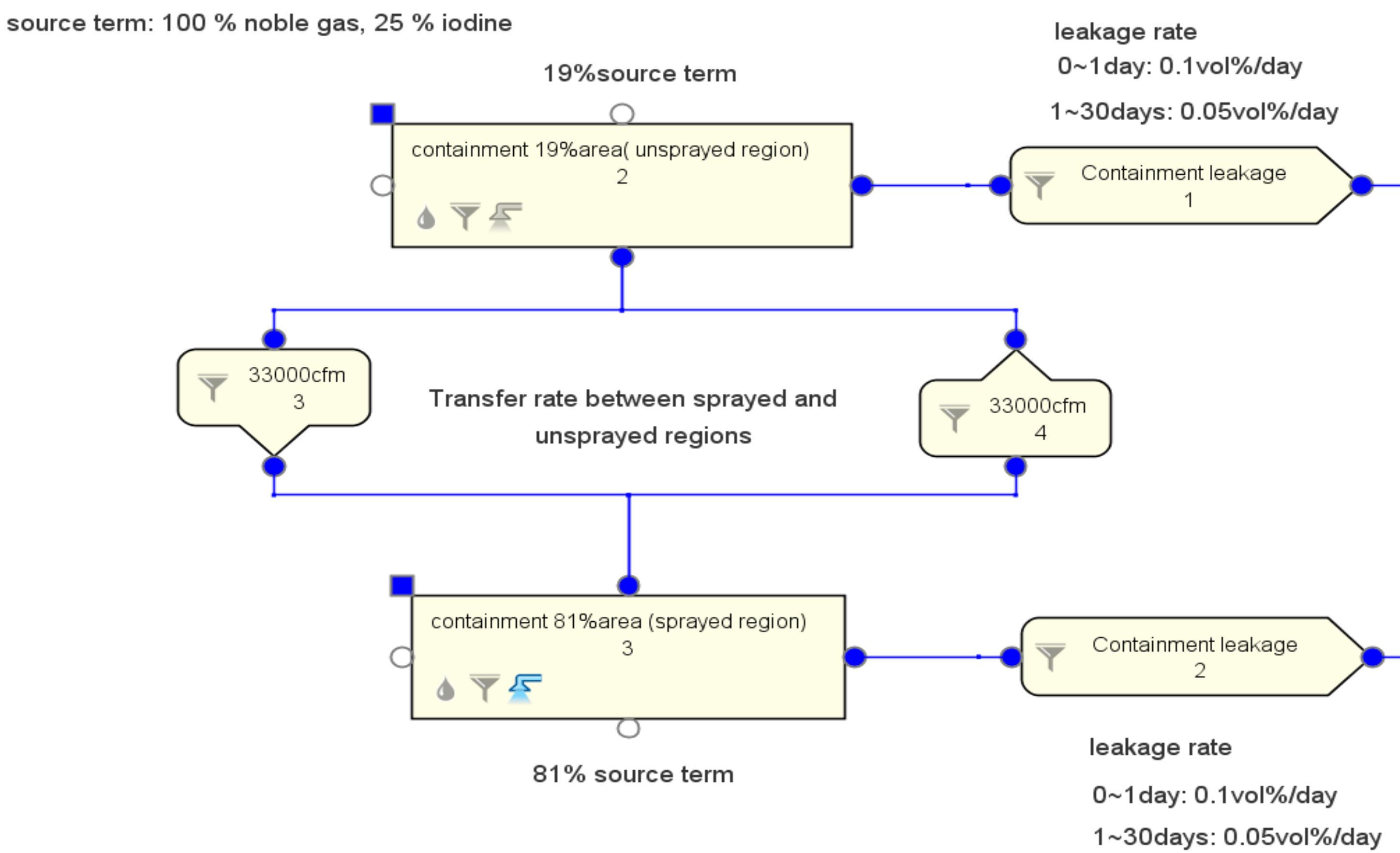
Failure criteria:
Whole body dose: 250 mSv
Thyroid dose from iodine exposure: 3000 mSv



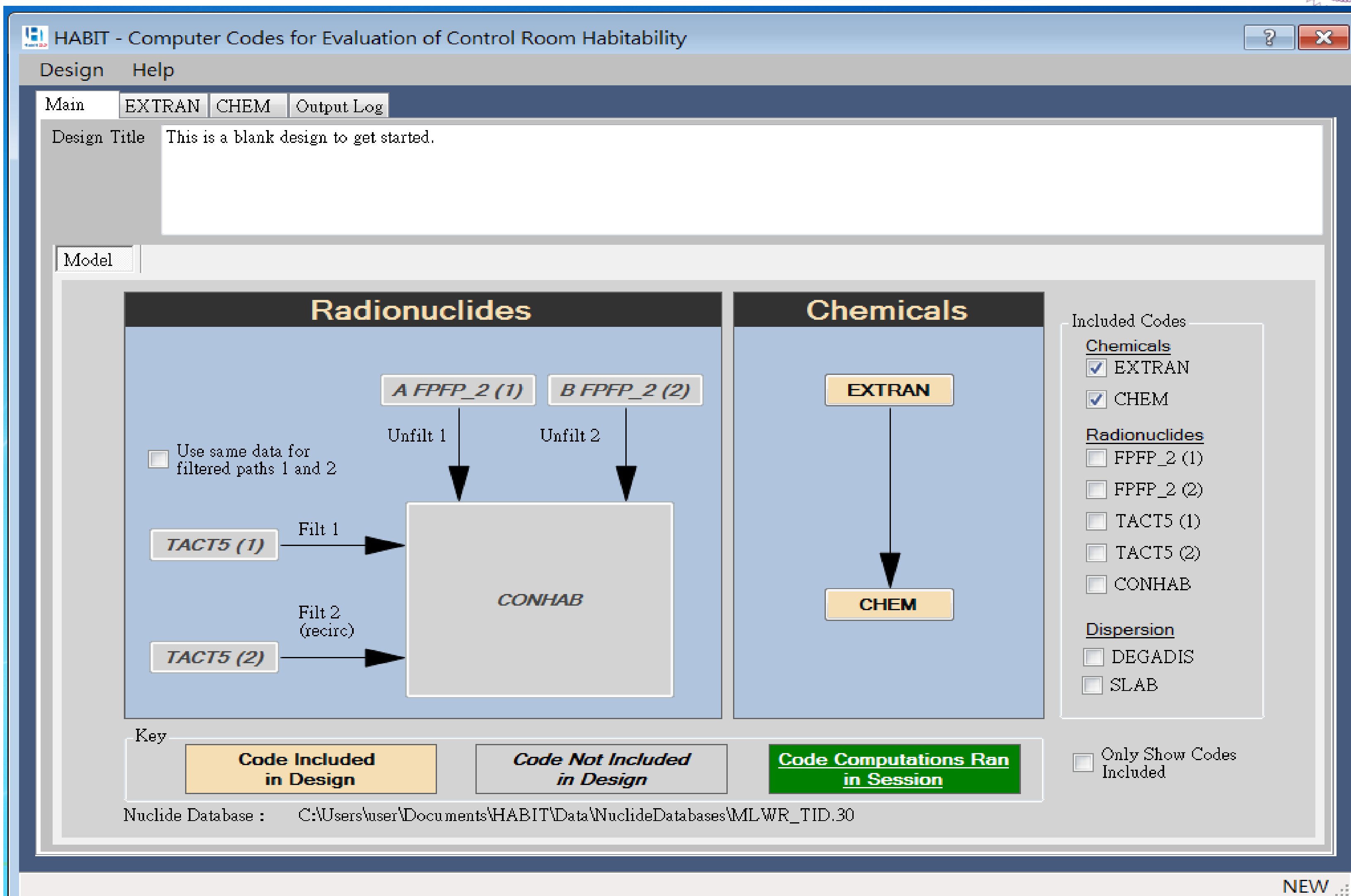
RADTRAD results
Whole body dose: 9.27 mSv
Thyroid dose from iodine exposure: 379.24 mSv

RADTRAD results
Whole body dose: 6.09 mSv
Thyroid dose from iodine exposure: 785.95 mSv

核三廠RADTRAD model



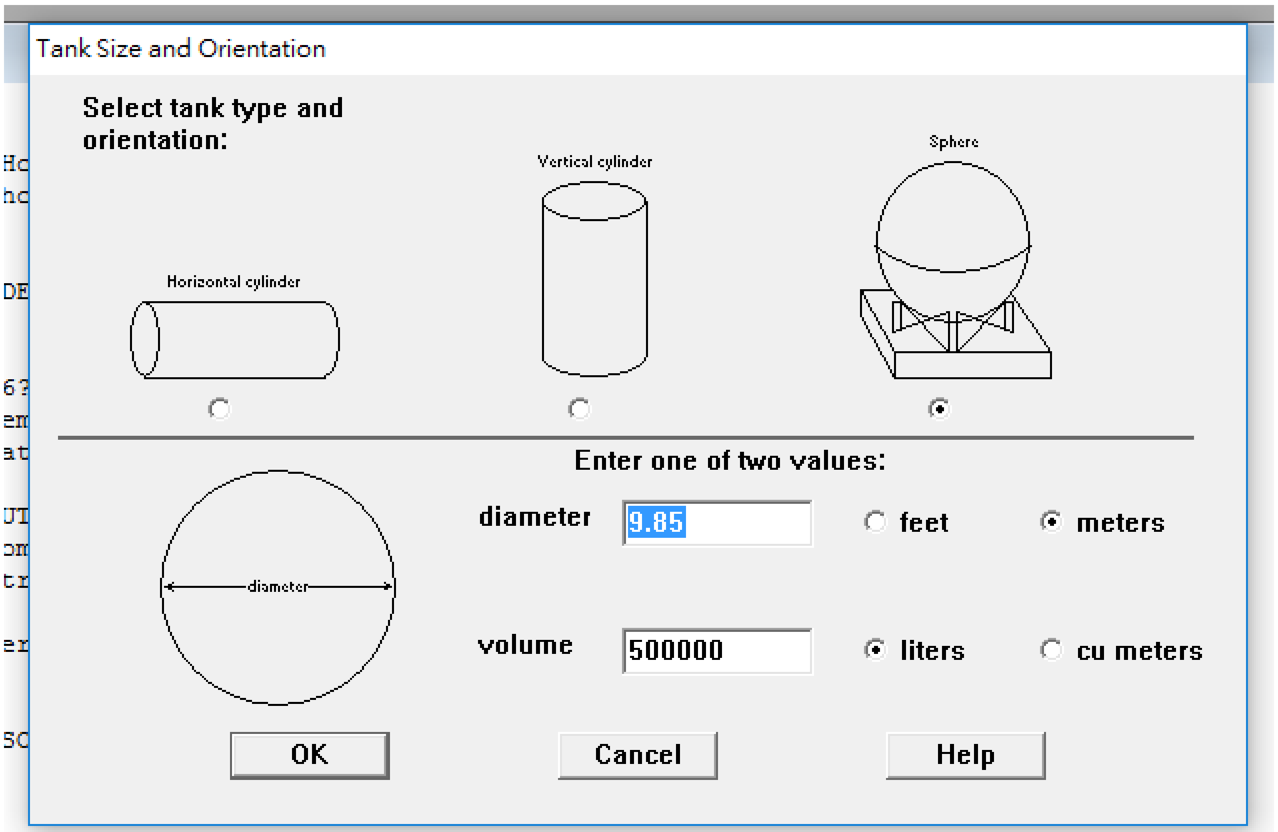
- HABIT程式為U.S. NRC所發展的控制室適居性評估程式，此程式假設核電廠發生事故，造成化學物質(氣態或液態)釋放時，評估其對於控制室適居性之影響
- HABIT程式目前最新的版本為2.0，並持續進行更新中，根據其簡報資料，未來將移除radionuclides的分析，只保留化學物質的分析

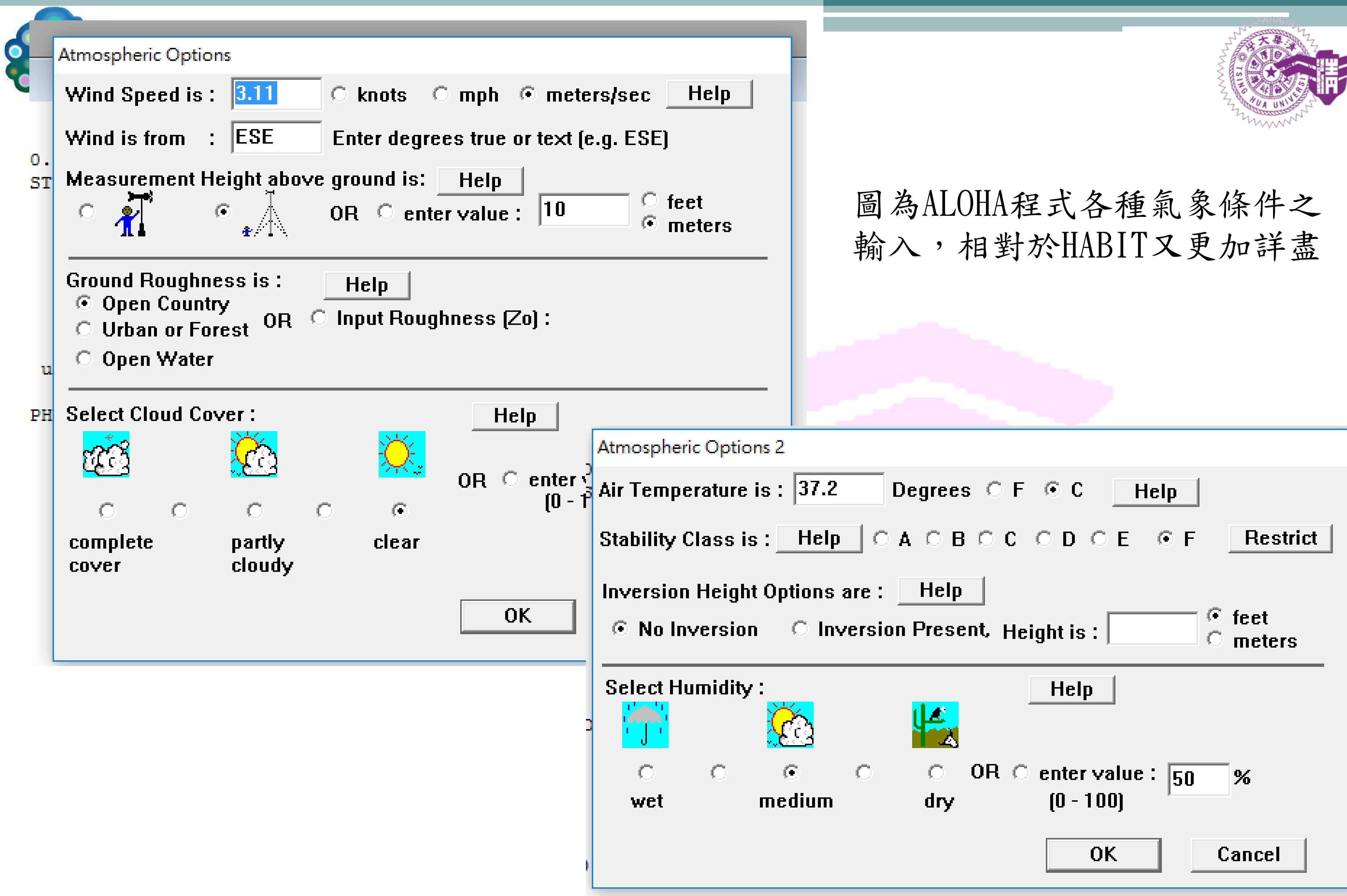

HABIT - Computer Codes for Evaluation of Control Room Habitability
Design
Help
Main
EXTRAN
CHEM
Output Log
Design Title
This is a blank design to get started.
Model
Radionuclides
Chemicals
A FPFP_2 (1)
B FPFP_2 (2)
Unfilt 1
Unfilt 2
Use same data for filtered paths 1 and 2
TACT5 (1)
Filt 1
CONHAB
Filt 2 (recirc)
TACT5 (2)
EXTRAN
CHEM
Included Codes
Chemicals
EXTRAN
CHEM
Radionuclides
FPFP_2 (1)
FPFP_2 (2)
TACT5 (1)
TACT5 (2)
CONHAB
Dispersion
DEGADIS
SLAB
Key
Code Included in Design
Code Not Included in Design
Code Computations Ran in Session
Only Show Codes Included
Nuclide Database :
C:\Users\user\Documents\HABIT\Data\NuclideDatabases\MLWR_TID.30
NEW

ALOHA程式之化學毒物擴散計算

- ALOHA程式為美國用於工業界的化學物質擴散事故之分析程式，NRC正在嘗試HABIT與ALOHA程式之驗證及比較，用以強化控制室適居性之安全分析。
- 目前ALOHA已有初步分析結果用以比對HABIT分析。

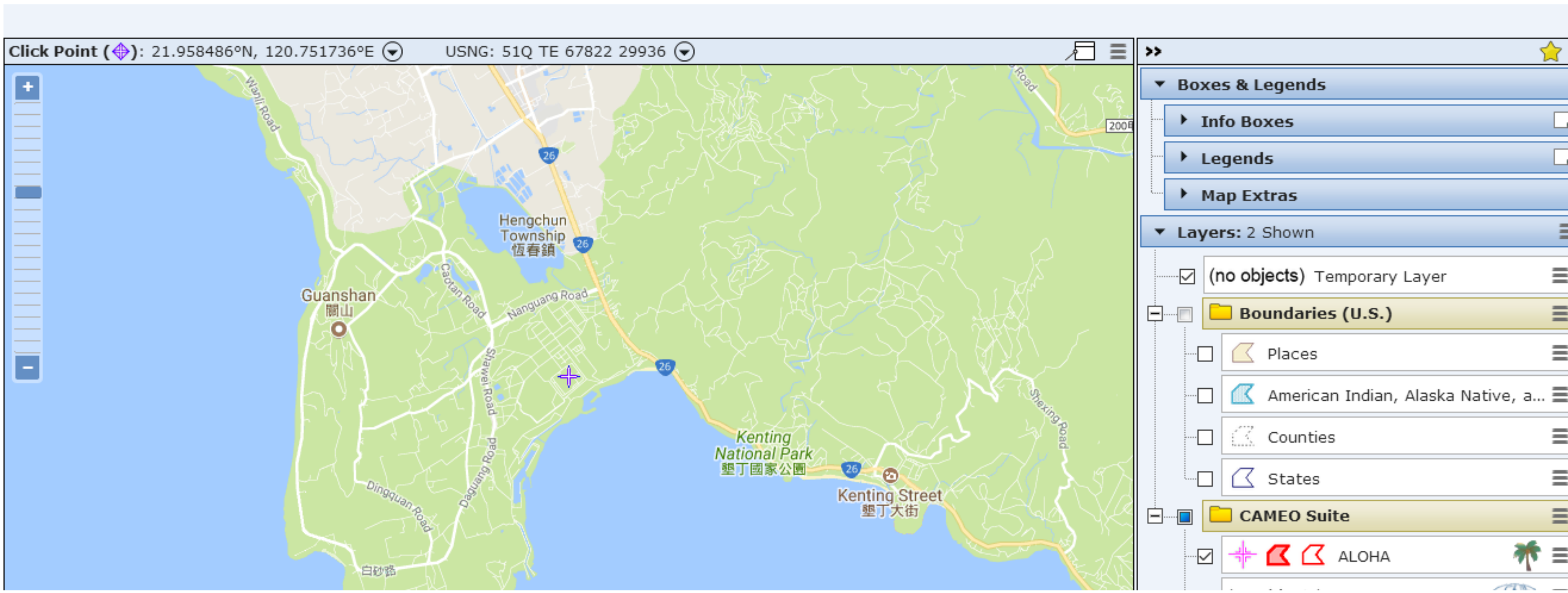
ALOHA可模擬不同貯存方式之化學物質事故，包含洩漏或爆炸



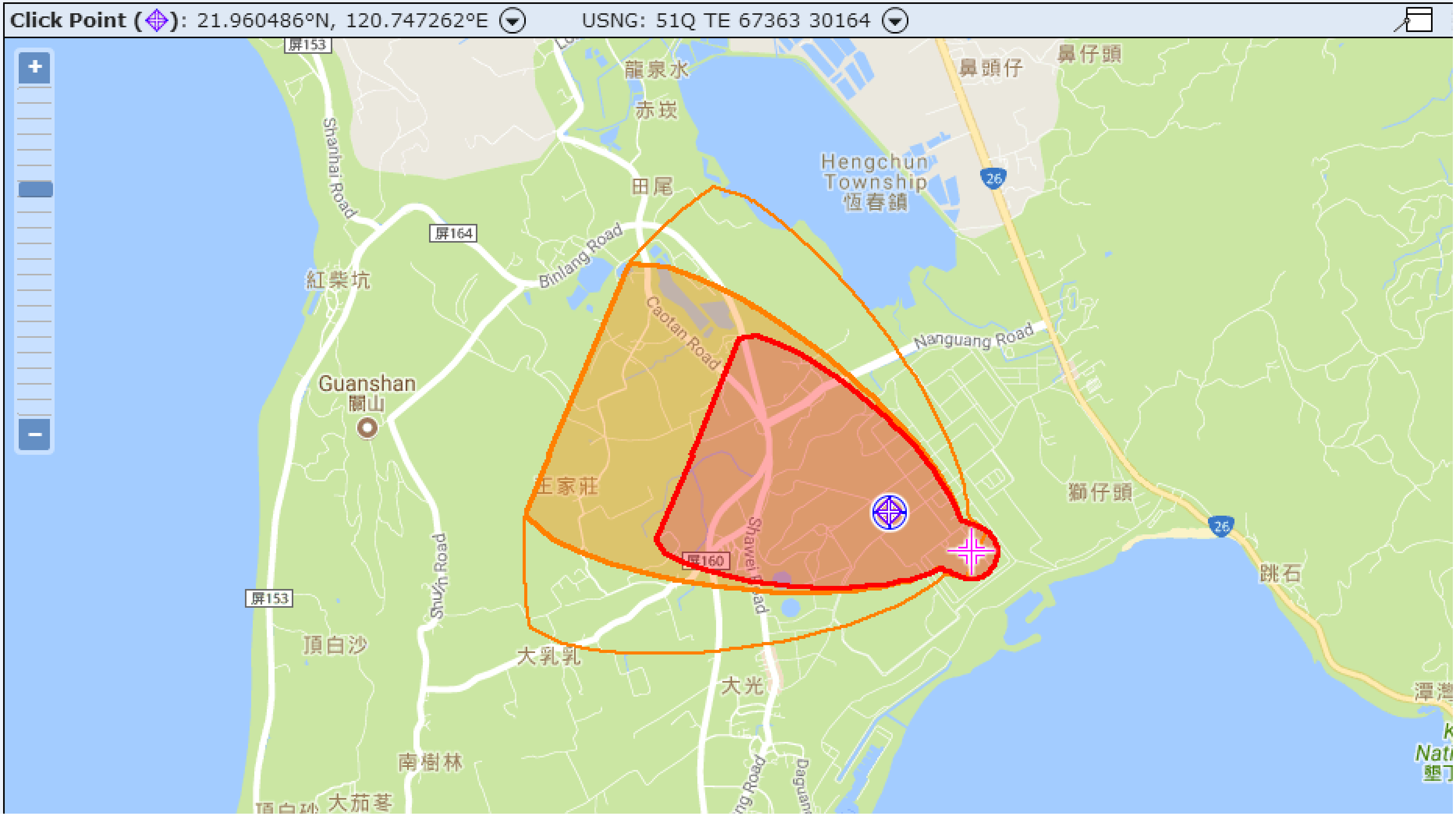


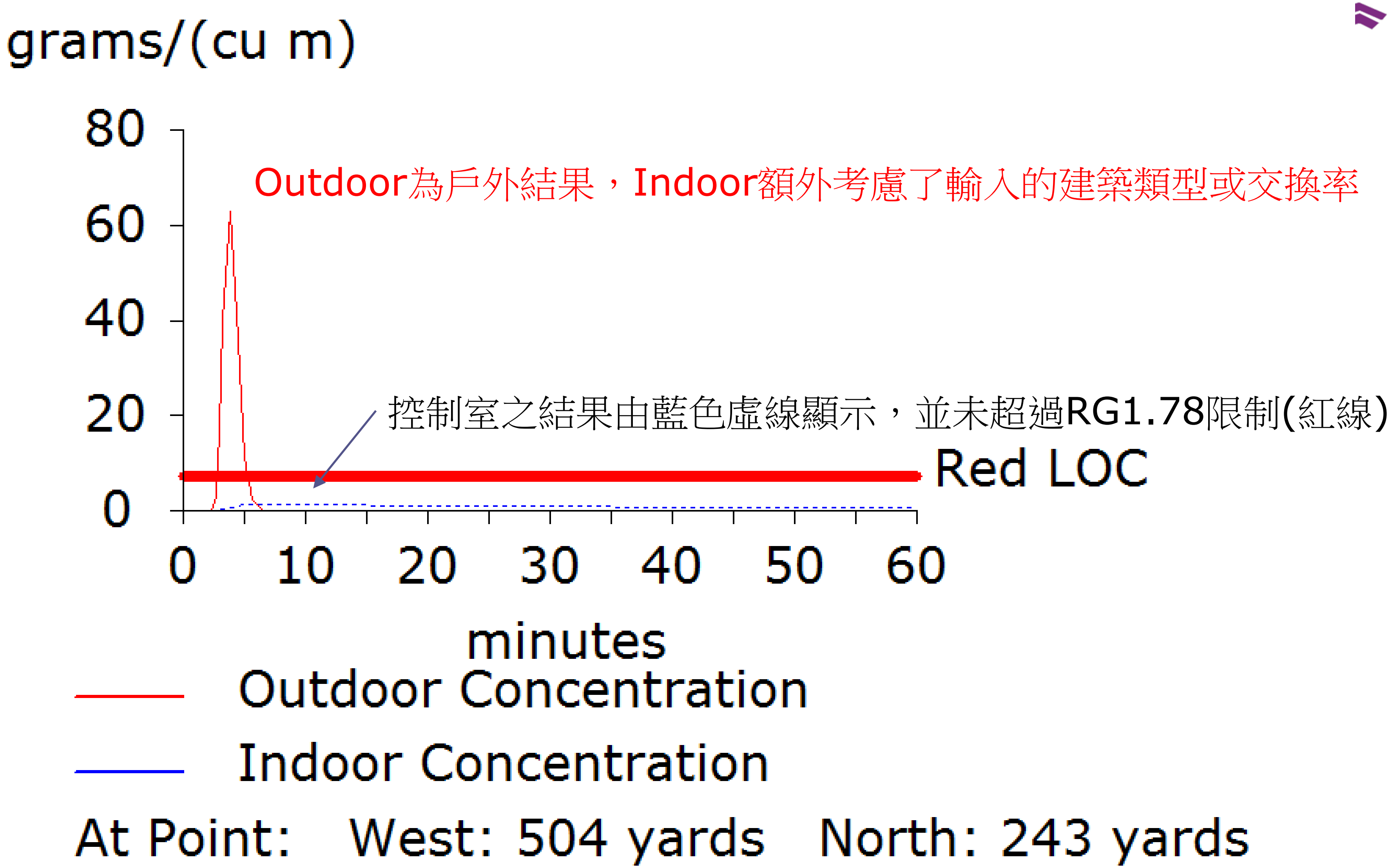
圖為ALOHA程式各種氣象條件之輸入，相對於HABIT又更加詳盡

ALOHA可結合程式MARPLOT輸出世界各地之地圖，圖為核三廠



ALOHA分析核三廠二氧化碳槽爆炸事故之戶外濃度結果

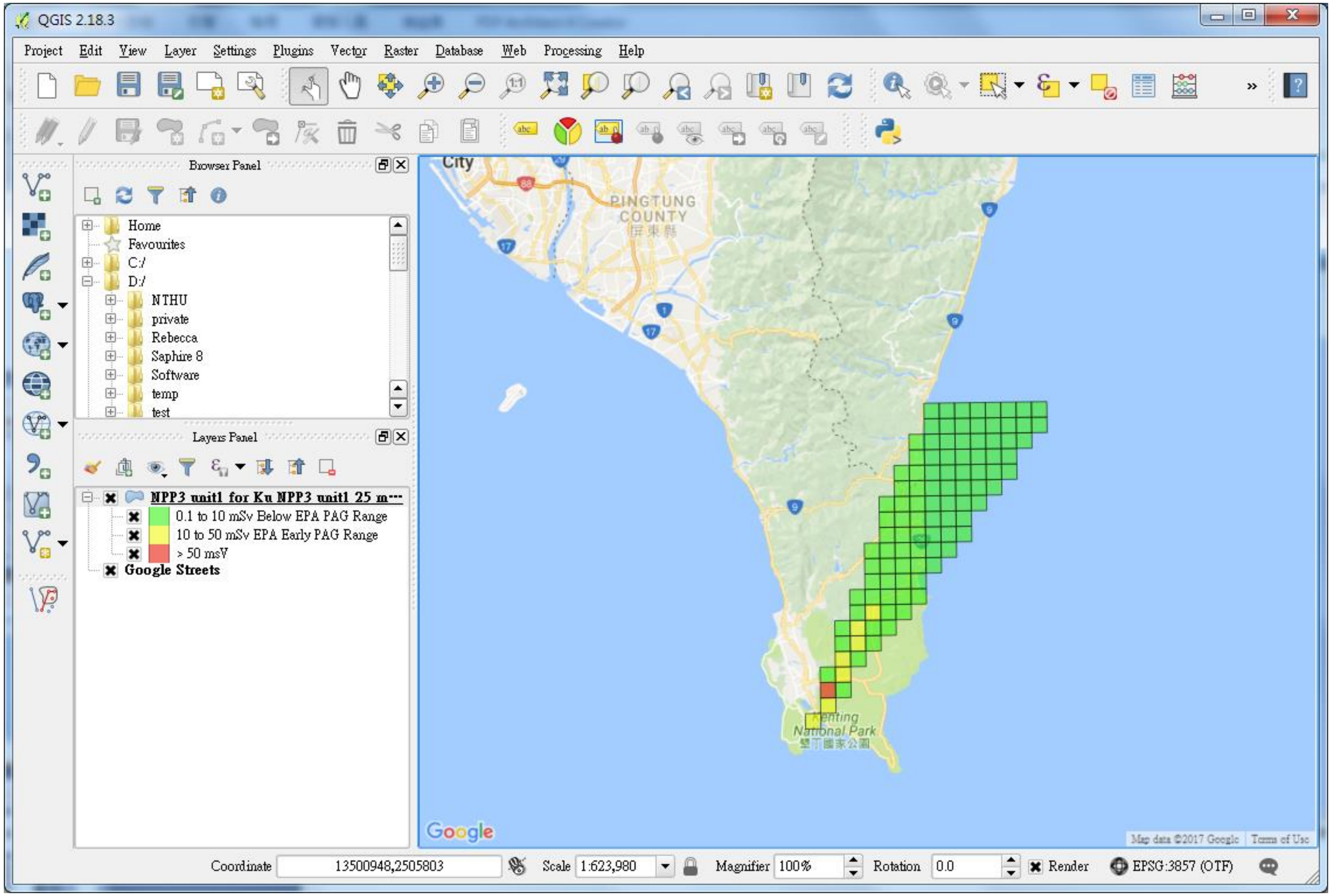



grams/(cu m)
80
60
40
20
0
Outdoor為戶外結果，Indoor額外考慮了輸入的建築類型或交換率
控制室之結果由藍色虛線顯示，並未超過RG1.78限制(紅線)
Red LOC
0
10
20
30
40
50
60
minutes
Outdoor Concentration
Indoor Concentration
At Point: West: 504 yards North: 243 yards

- Using RADTRAD and SNAP to Establish the Analysis Model for Maahshan PWR Plant , World Academy of Science, Engineering and Technology International Journal of Nuclear and Quantum Engineering Vol:4, No:7, 2017.
- Using HABIT to Establish the Chemicals Analysis Methodology for Maanshan Nuclear Power Plant, World Academy of Science, Engineering and Technology, International Journal of Chemical, Molecular, Nuclear, Materials and Metallurgical Engineering Vol:11, No:6, 2017.
- The Establishment of the Analysis Methodology for Maanshan Nuclear Power Plant Using RADTRAD, HABIT, and ALOHA, KERNTECHNIK 2017.(已投稿)

RASCAL程式

RASCAL可計算用過燃料池及乾式貯存設施事故下之放射產物釋出結果



GALE程式

- ALE中文名稱為輕水式核電廠氣液態排放物輻射估算程式(Gaseous And Liquid Effluent from commercial light water nuclear power plants)，其主要使用FORTRAN程式語言編寫，用來計算核能電廠(包含PWR及BWR)所排出氣體及液體放射性物質的活性含量。
- 其根據電廠模式可分為BWR-GALE與PWR-GALE兩種
- 可下載程式版本有兩種
  - GALE86
  - GALE-2.0 BETA（最新版本）
- 參考文件
  - NUREG-0016 Revision 1, *Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-GALE Code)*
  - NUREG-0017 Revision 1, *Calculation of Releases of Radioactive Materials in Gaseous and Liquid Efluents from Pressurized Water Reactors (PWR-GALE Code)*

GALE BWR主畫面


BWR GALE-2.0
建立新檔名或使用舊存檔
GALE 2.0
Input File Name
BWRGALE.in
Type of Analysis
Browse
Gas
Liquid
分析模式選擇
Output Files:
Gas
BWRGE.out
Liquid
BWRLE.out
輸出檔名
Legacy Input
Read Legacy Input
輸入舊格式檔案
Gas Input
BWRGE09.inp
Gas Output
BWRGE09.out
Liquid Input
BWRLE09.inp
Liquid Output
BWRLE09.out
OK
Cancel

GALE2.0 Output file

maanshan_MUR PWR

GASEOUS RELEASE RATE - CURIES PER YEAR

	PRIMARY COOLANT (MICROCI/GM)	SECONDARY COOLANT (MICROCI/GM)	GAS STRIPPING SHUTDOWN	GAS STRIPPING CONTINUOUS	BUILDING VENTILATION REACTOR	BUILDING VENTILATION AUXILIARY	BUILDING VENTILATION TURBINE	BLOWDOWN VENT OFFGAS	AIR EJECTOR EXHAUST	TOTAL
KR-85M	1.855E-02	4.806E-09	0.0E+00	0.0E+00	1.4E+01	0.0E+00	0.0E+00	0.0E+00	0.0E+00	1.4E+01
KR-85	1.657E-01	4.181E-08	6.0E+01	6.5E+02	3.0E+02	4.0E+00	0.0E+00	0.0E+00	2.0E+00	1.0E+03
KR-87	1.987E-02	1.425E-08	0.0E+00	0.0E+00	6.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	6.0E+00
KR-88	2.095E-02	5.394E-09	0.0E+00	0.0E+00	1.2E+01	0.0E+00	0.0E+00	0.0E+00	0.0E+00	1.2E+01
XE-131M	5.509E-01	1.381E-07	1.4E+01	1.6E+02	9.7E+02	1.3E+01	0.0E+00	0.0E+00	6.0E+00	1.2E+03
XE-133M	7.229E-02	1.889E-08	0.0E+00	0.0E+00	1.2E+02	2.0E+00	0.0E+00	0.0E+00	0.0E+00	1.2E+02
XE-133	2.634E-02	6.647E-09	0.0E+00	0.0E+00	4.5E+01	0.0E+00	0.0E+00	0.0E+00	0.0E+00	4.5E+01
XE-135M	1.524E-01	3.859E-08	0.0E+00	0.0E+00	1.1E+01	4.0E+00	0.0E+00	0.0E+00	2.0E+00	1.7E+01
XE-135	7.671E-02	1.955E-08	0.0E+00	0.0E+00	8.2E+01	2.0E+00	0.0E+00	0.0E+00	0.0E+00	8.4E+01
XE-137	3.987E-02	1.015E-08	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
XE-138	7.149E-02	1.858E-08	0.0E+00	0.0E+00	5.0E+00	2.0E+00	0.0E+00	0.0E+00	0.0E+00	7.0E+00
TOTAL NOBLE GASES										2.5E+03

0.0 APPEARING IN THE TABLE INDICATES RELEASE IS LESSTHAN 1.0 CI/YR FOR NOBLE GAS, 0.0001 CI/YR FOR I

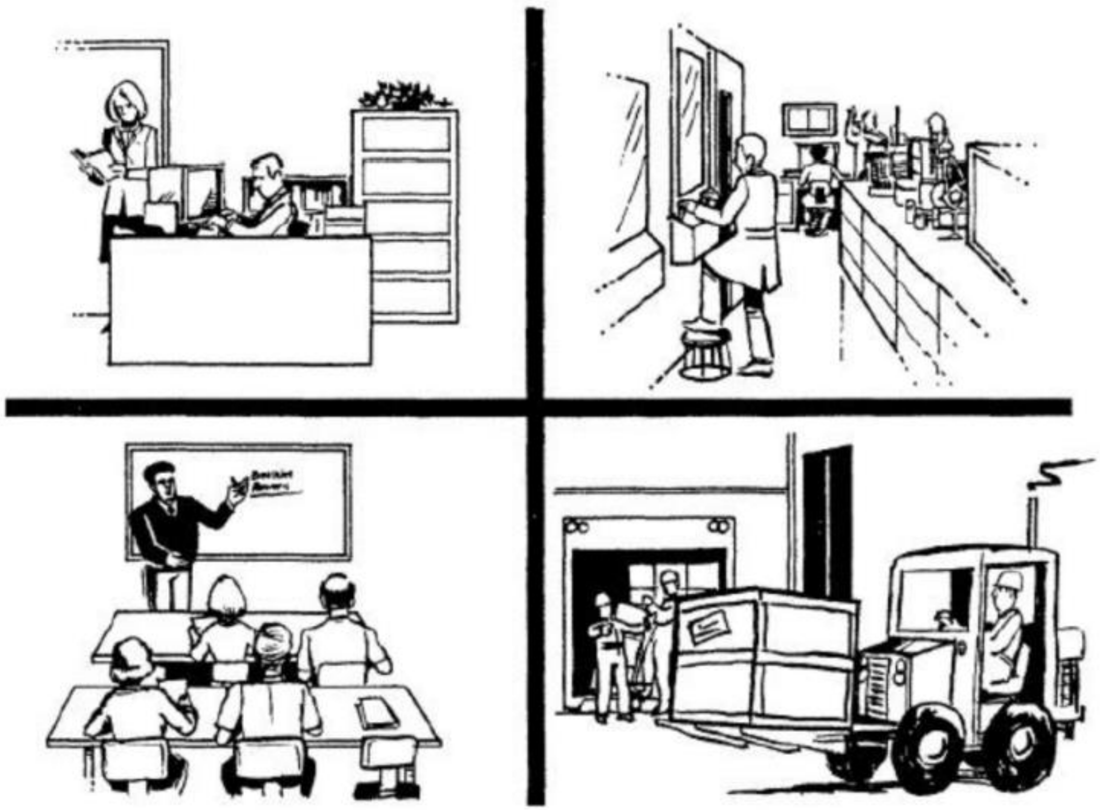
DandD電廠除役廠址劑量評估

- According to 10CFR part 20, the dose should lower than 0.25mSv/yr for releasing.
- DandD is the code which can simplify the regularity process of U.S.NRC.

- Reference:
  NUREG-1549
  NUREG/CR-5512

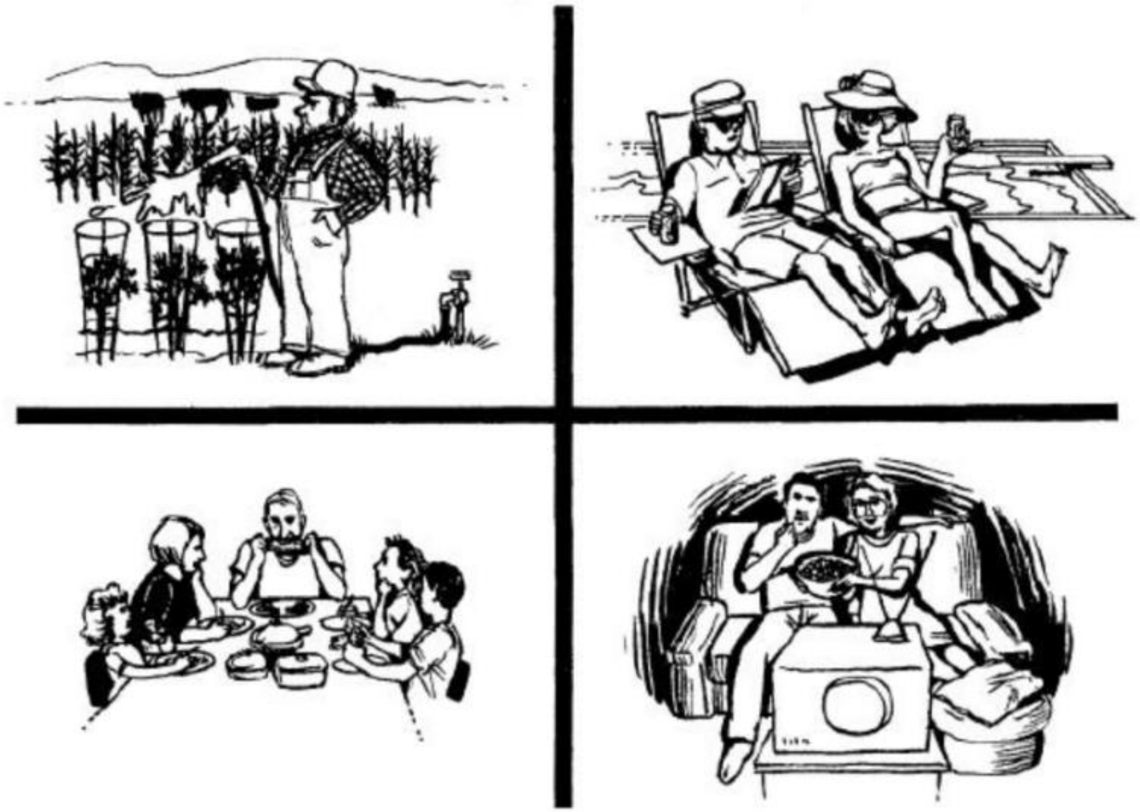
Status of RAMP codes in NTHU-DandD

Pathway-building occupancy



Status of RAMP codes in NTHU-DandD

Pathway-residential



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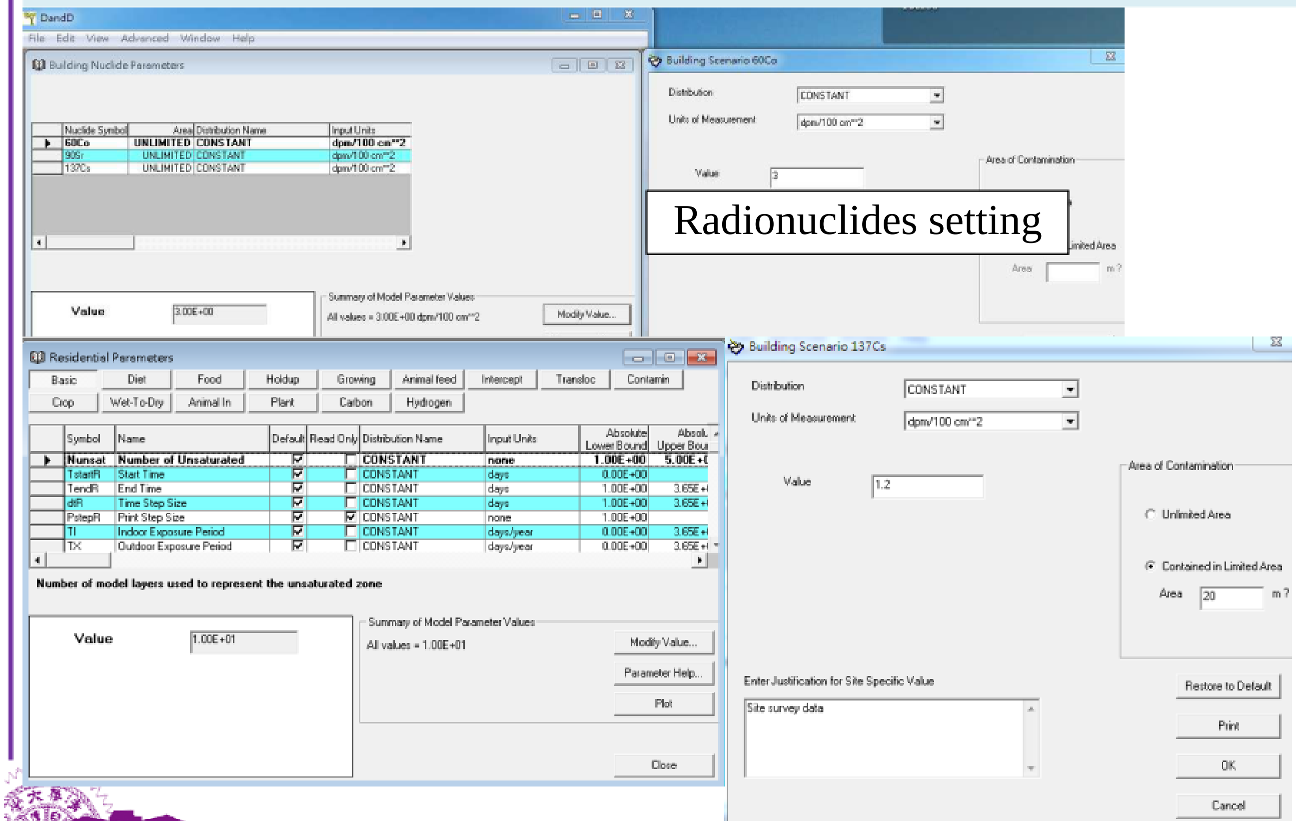
Status of RAMP codes in NTHU-DandD

Building Nuclide Parameters



10

Status of RAMP codes in NTHU-DandD

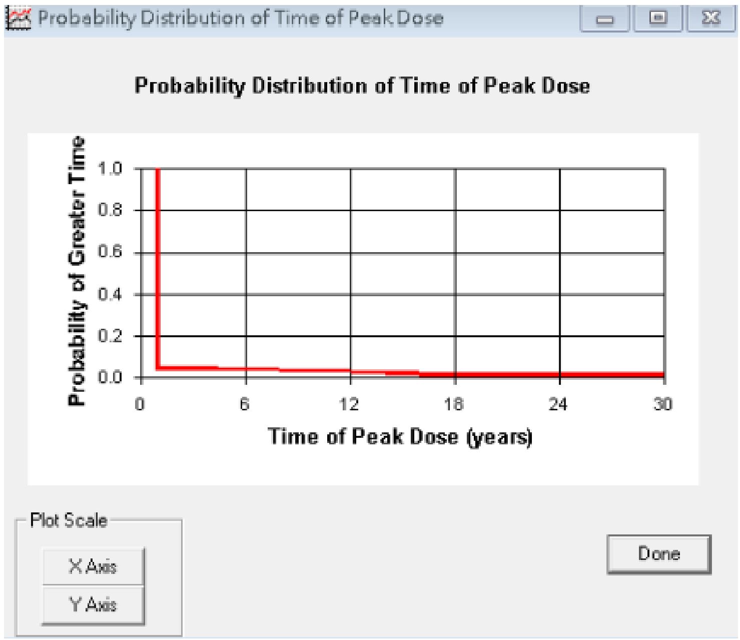
Case calculations:

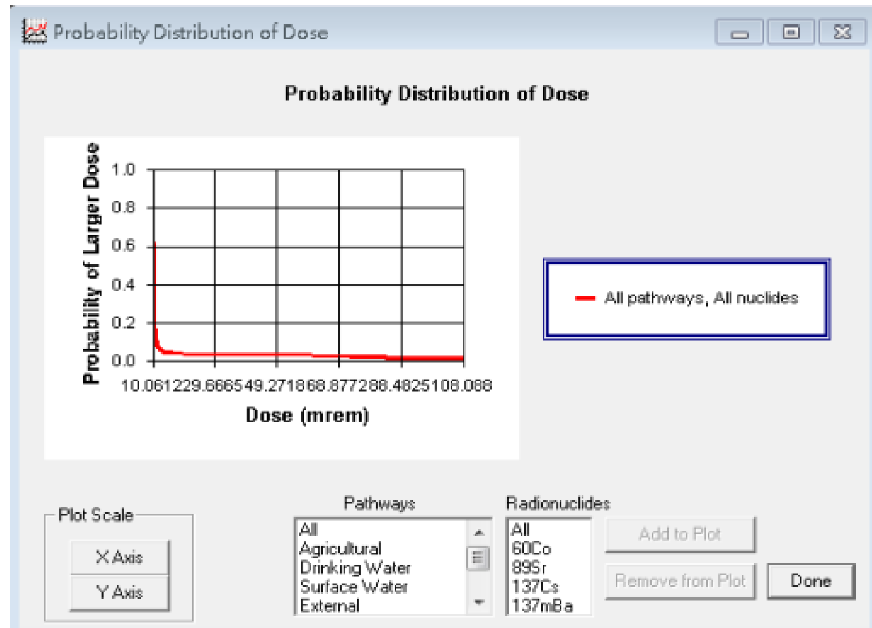
- A decommission NPP with Co-60, Sr-90, Cs-137 for Residential
- 7 path included: External, Inhalation, Secondary ingestion, Agricultural, Drinking water, Irrigation, Surface water

Nuclide	Concentration (pCi/g)	Area of contamination (m^2)	Distribution
Co-60	1	40	Constant
Sr-90	1.5	30	Constant
Cs-137	2	20	Constant

Status of RAMP codes in NTHU-DandD

Results





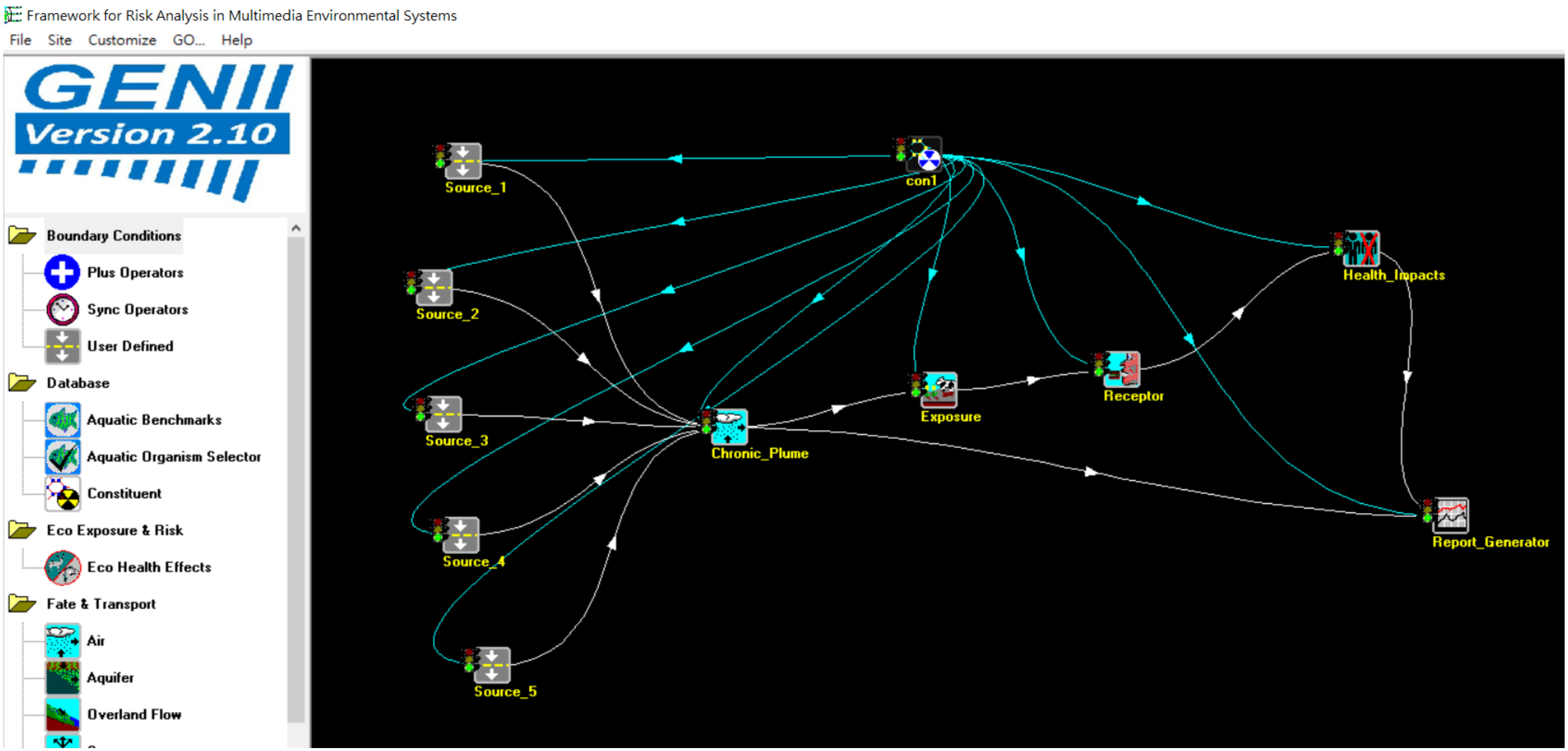
TEDE was 11.3mRem, it's lower than the 10CFR part 20(25mRem).

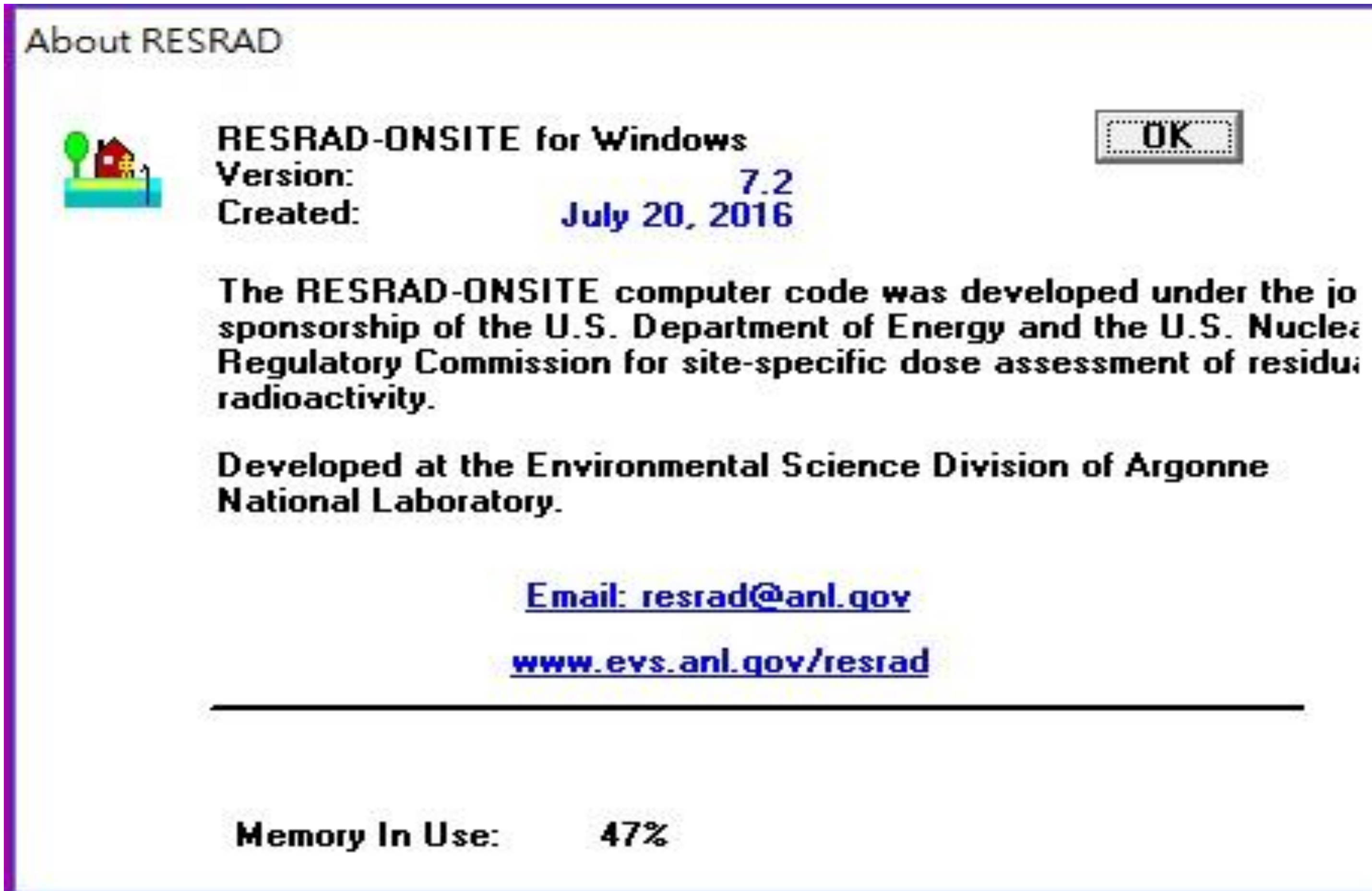
DandD可以有效評估除役後場址的輻射劑量，用於後續除役過程之規劃

GENII

- 生物圈之輻射劑量及風險評估程式(GENII)為美國Pacific Northwest National Laboratory(PNNL)所發展之分析軟體，該軟體可模擬當放射性核種外釋時，其傳播途徑可能造成的輻射劑量，可計算之傳播途徑包括地表水、大氣擴散、生物累積(動植物、水中生物)等，並評估當人類經由曝露、吸入與攝入等途徑接觸到放射性核種，對其造成的健康危害與風險，且ICRP-20、ICRP-26、ICRP-30、ICRP-60等資料亦已納入程式模擬選項，可根據法規要求進行體內劑量的保守評估。GENII曾應用於美國Hanford場址復原及Yucca Mountain計畫之環境輻射劑量評估。

GENII之操作介面，囊括多項環境及生物劑量評估的子程式




About RESRAD
RESRAD-ONSITE for Windows
Version: 7.2
Created: July 20, 2016
OK
The RESRAD-ONSITE computer code was developed under the jo
sponsorship of the U.S. Department of Energy and the U.S. Nucle
Regulatory Commission for site-specific dose assessment of residu
radioactivity.
Developed at the Environmental Science Division of Argonne
National Laboratory.
Email: resrad@anl.gov
www.evs.anl.gov/resrad
Memory In Use: 47%

RESRAD-ONSITE 7.2 (7/20/16)

- Improved ICRP-107 radionuclide decay chain threads condensing routine to reduce computation time.
- Provided options to choose between the ICRP-38 radionuclide decay database and the ICRP107 radionuclide decay database.
  - Support to use either ICRP-26/30 or ICRP-60/72 based dose coefficients with the first option.
  - Support to use ICRP60 based dose coefficients from DCFPAK 3.02 with the second option.

RESRAD 7.0 (4/4/14)

- Extend DCF Database and software capability for ICRP 107; added Reference Person DCF's as an option
- Use of different cover and depth factors for photosynthesis and root uptake for C-14
- Updated Help for RESRAD and for DCF Editor
- Fix problem in graphics with multiply threaded progeny

RESRAD 6.5 (10/30/09)

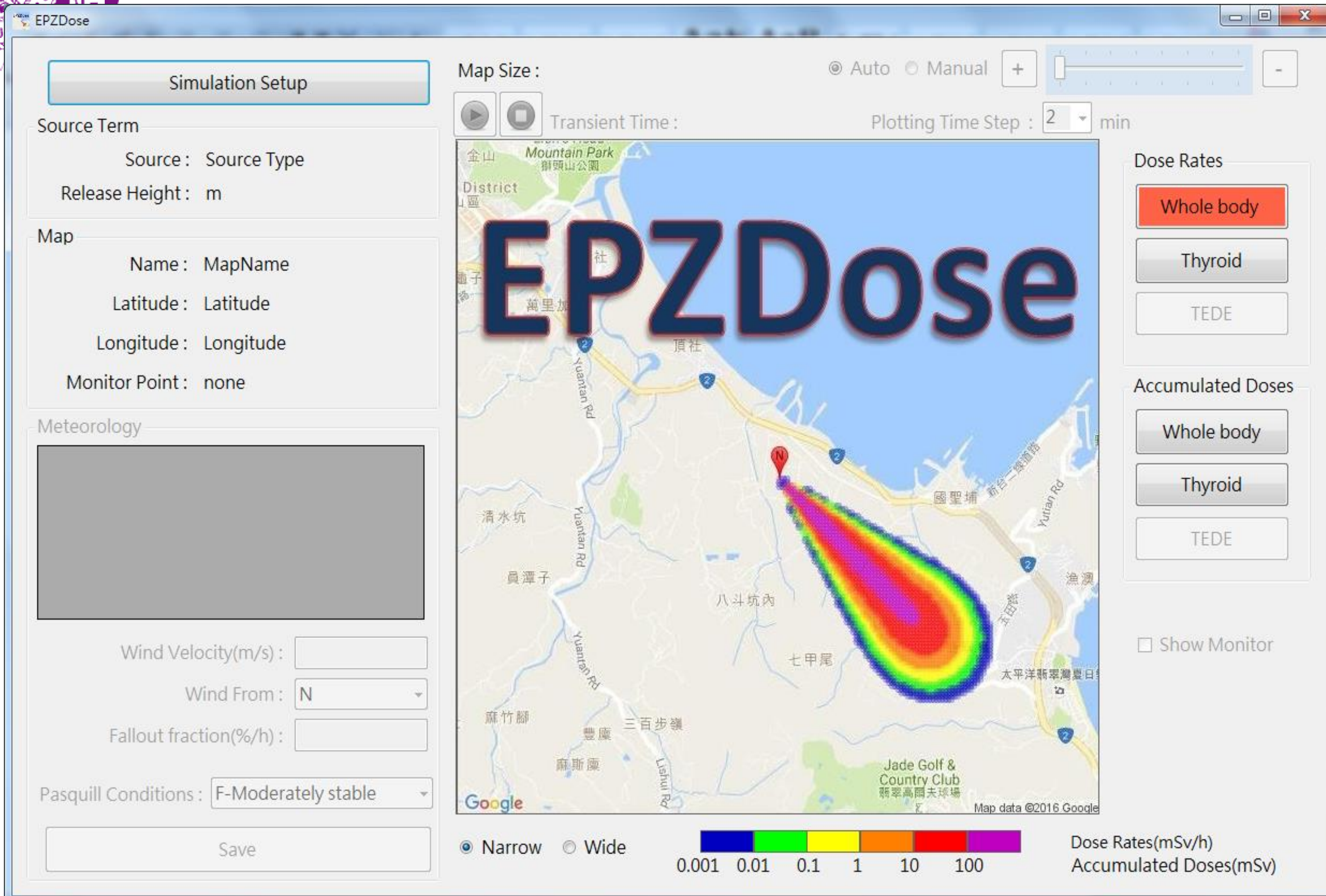
- C-14 gaseous and particulate contributions to dose and risk available
- Partially or fully submerged contaminated zone now treated
- Choice between ICRP60 or FGR12 for External dose factors added
- 64-bit and Vista computers now supported

RESRAD 6.4 (12/20/07)

- Added ICRP 72 age-dependent DCFs
- Improved data storage and retrieval, user specified directories.
- User specified ground DCF's now possible.
- C-14 inhalation dose and risk improved.

RESRAD 6.3 (8/25/05)

- Added ICRP-38 radionuclides
- Allow variable half-life cutoff
- DCF Editor is now common between RESRAD and RESRAD-BUILD


EPZDose
Simulation Setup
Source Term
Source : Source Type
Release Height : m
Map
Name : MapName
Latitude : Latitude
Longitude : Longitude
Monitor Point : none
Meteorology
Wind Velocity(m/s) :
Wind From : N
Fallout fraction(%/h) :
Pasquill Conditions : F-Moderately stable
Save
Map Size :
Auto
Manual
Transient Time :
Plotting Time Step : 2 min
EPZDose
Dose Rates
Whole body
Thyroid
TEDE
Accumulated Doses
Whole body
Thyroid
TEDE
Show Monitor
Narrow
Wide
0.001 0.01 0.1 1 10 100
Dose Rates(mSv/h)
Accumulated Doses(mSv)
Map data ©2016 Google

- Developments and Applications of EPZ Dose for Offsite Dose Evaluations in Nuclear Power Plant Accident Conditions, Int. Top. Mtg. on Nuclear Thermal-Hydraulics, Operation and Safety (NUTHOS-9), Kaoshiung, Taiwan, Sept. 9-13, 2012.
- EPZDose, An offsite dose evaluation code for radioactive material release accidents," PBNC, Vancouver, Canada, Aug. 24 – 28, 2014.
- Modeling Fukushima Dose Consequences with EPZDose, PBNC, Beijing, China, April 5-9, 2016.

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Nuclear Energy and Waste

Active

DECOVALEX

DOE-NE-NUCLEAR ENERGY

DEvelopment of COupled Models and their VALidation Against EXperiments in Nuclear Waste Isolation

The DECOVALEX Project is a unique international research collaboration for advancing the understanding and mathematical modeling of coupled thermo-hydro-mechanical (THM) and thermo-hydro-chemical (THC) processes in geological systems. DECOVALEX is an acronym for "Development of Coupled Models and their Validation against Experiments." Starting in 1992, the project has made important progress and played a key role in the development of numerical modeling of coupled processes in fractured rocks and

Energy Resources

PROGRAM DOMAIN

View Project Website

Project Contacts

Jens Birkholzer
Energy Geosciences
Division Director
Senior Scientist

Energy Geosciences Division Director, Jens Birkholzer was recently named the Chairman of the DECOVALEX Project, and LBNL is now serving as the coordinating organization for DECOVALEX. In addition, LBNL scientists have participated as modeling teams in various past DECOVALEX phases and several DECOVALEX projects. The current DECOVALEX Project phase (referred to as D-2019) started in Spring 2016 and will run through the end of 2019. Modeling tasks tackled in the current phase are:

- Task A: ENGINEER – Modeling advective gas flow through low permeability materials.
- Task B: Fault Slip Test – Modelling the induced slip of a fault in argillaceous rock.
- Task C: GREET – Hydro-mechanical-chemical-biological processes during groundwater recovery.
- Task D: INBEB – Hydro-mechanical interactions in bentonite engineered barriers.
- Task E: Upscaling Heater Tests – Upscaling of modelling and experimental results from small scale to one-to-one scale.
- Task F: FINITO – Fluid inclusion and movement in tight rock.
- Task G: EDZ Evolution – Reliability, feasibility and significance of Measurements of conductivity and transmissivity of the rock mass for the understanding of the evolution of a repository of spent nuclear fuel.

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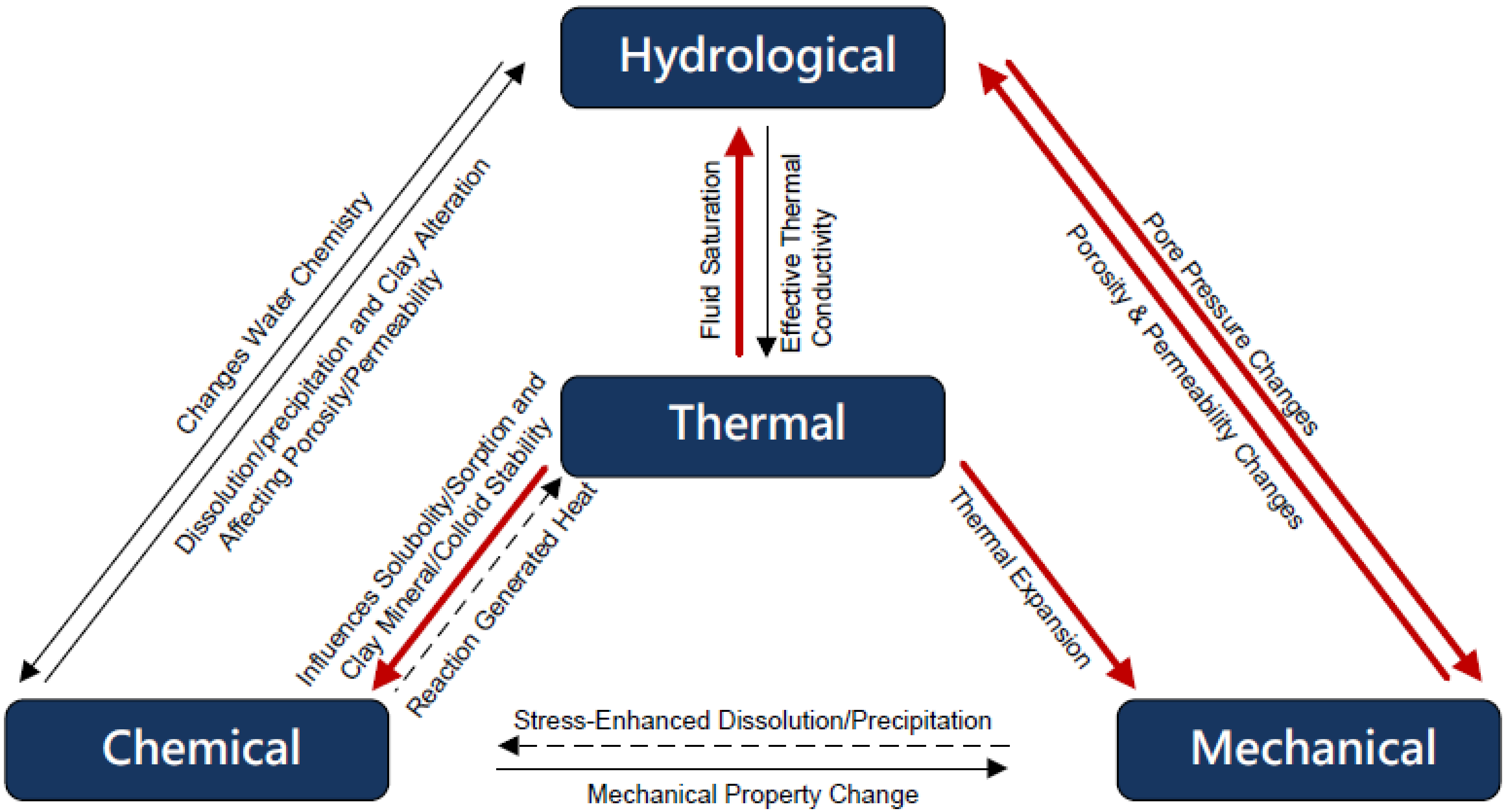

圖 3-1　熱傳、水力、力學與化學反應(THMC)之相互影響

- 將使用TOUGH2進行相關研究

結論

- 各項程式與分析技術皆可實際應用，相關之研究成果則可維護與提升國內核電廠之安全
- 將持續參與CAMP、CSARP 、 RAMP 、DECOVALEX國際合作計畫，來引進新程式或更新程式版本
- 將持續發展與精進相關程式之模式與分析技術
- 將積極投入除役與高放之相關研究與應用

動畫展示

Thanks