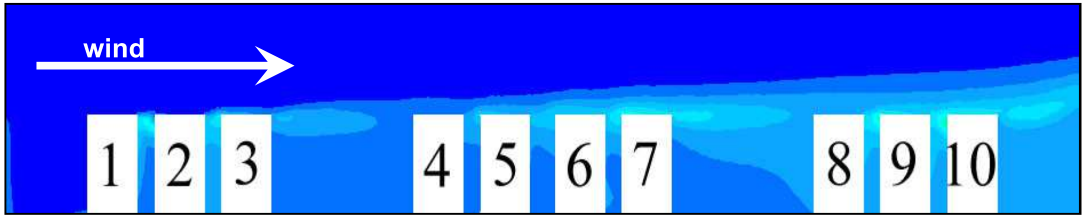
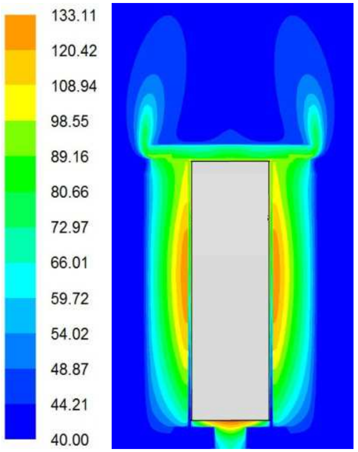
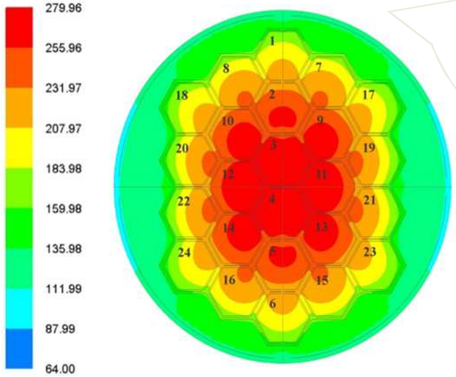
Session 7 - posters

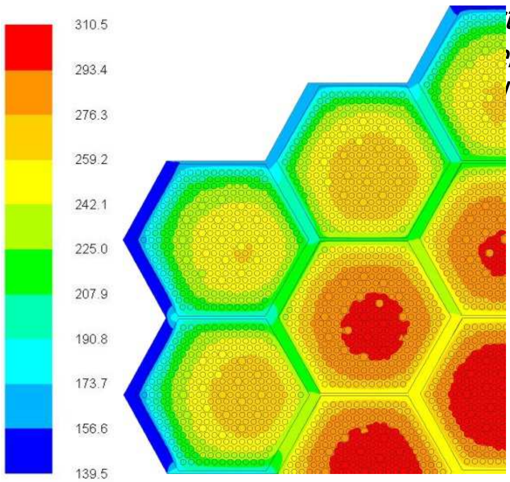
Direct heat transfer problems



• *the iterative approach for definition of thermal state of containers group with taking into account their mutual influences*

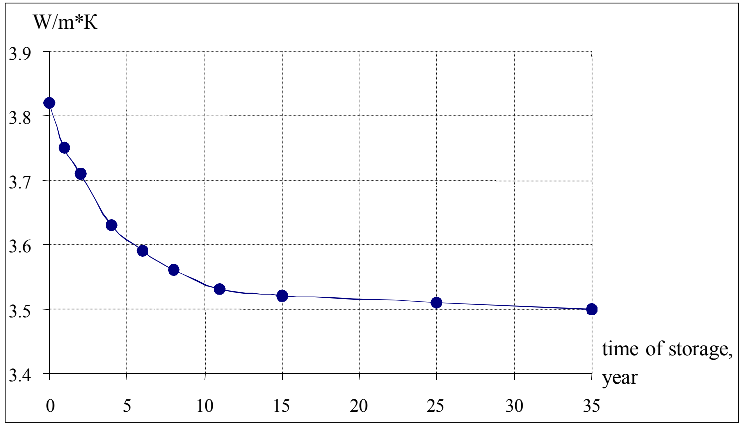




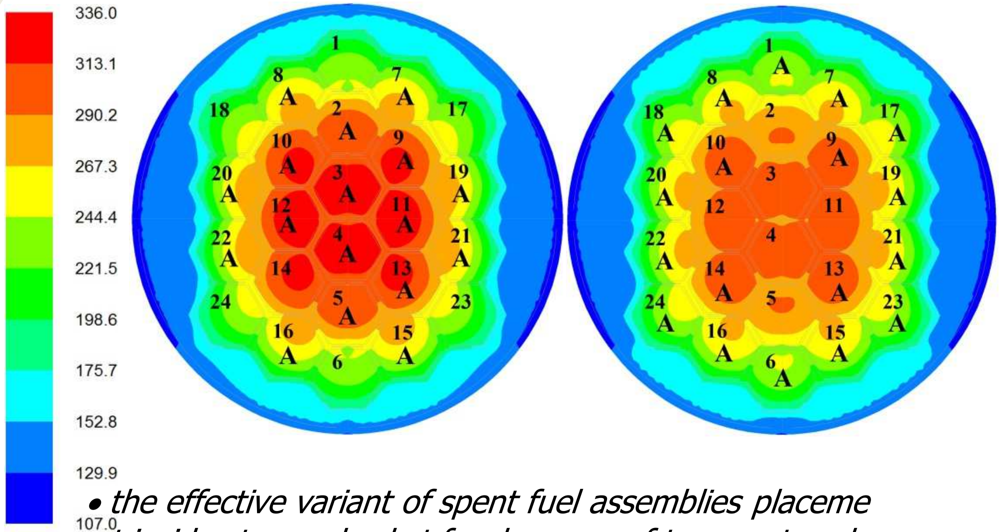


• *the iterative methodolog y for definition of thermal state of containers with sp ent nuclear fuel, spent fue l assemblies and fuel rods*

Inverse heat transfer problems



• *the equivalent heat transfer coefficients of storage ba sket for different time of storage*

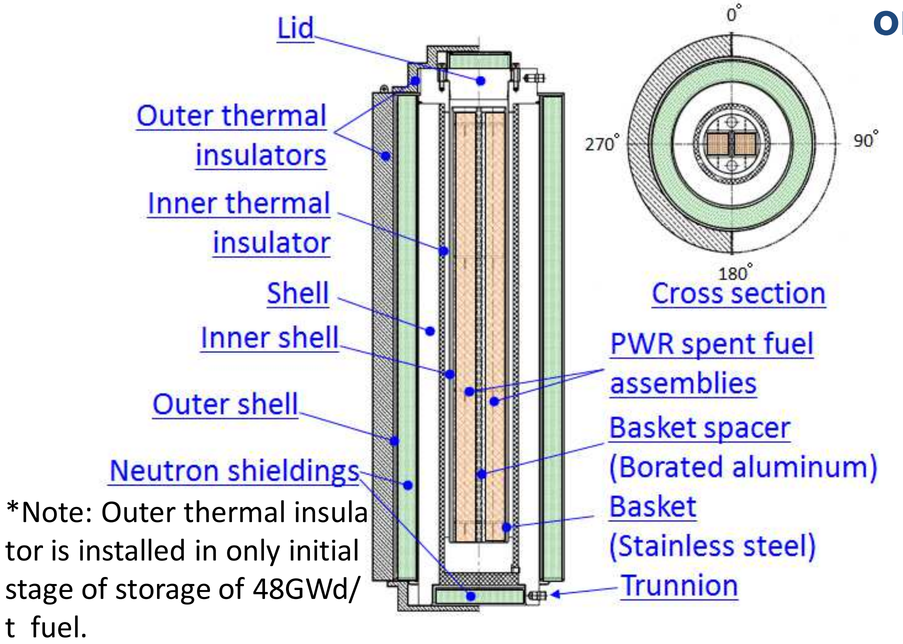


• *the effective variant of spent fuel assemblies placeme nt inside storage basket for decrease of temperature le vel*

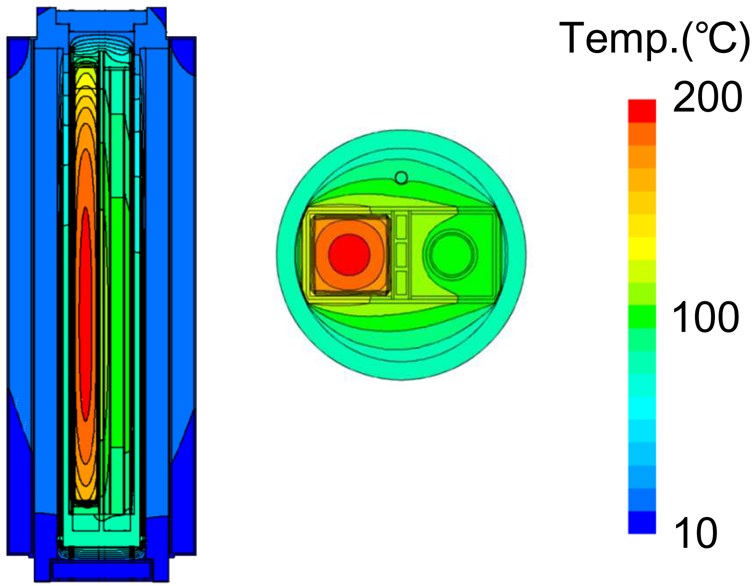
ID 45- Title : Preliminary Thermal Margin Comparison between A Simple SF Model and A Complex SF Model for a Dry Storage Cask.

- Background
  - It is expected that amount of spent nuclear fuels(SNF) will exceed the capacity of temporary storage tanks in each nuclear power plant(NPP) in Korea after several years. Accordingly, the industry demand for commercialization of a storage cask has been increased.
  - According to the above demand, KINS(Korea Institute of Nuclear Safety) started to develop a thermal technical guideline including CFD analysis.
- Objective
  - In this study, the thermal margin comparison of calculation models is a preliminary review to establish a thermal analysis methodology using CFD code contained a complex SNF model.
- Results
  - Using the CE-type-16-by-16 SNF assembly, 4 CFD models (1) 1/4 assembly homogenized model, 1/8 assembly rod to rod model, one SNF whole homogenized model and one SNF whole rod to rod model) were designed for thermal analysis.
  - The result shows that the thermal margin of homogenized models is less then rod-to-rod model because of convective effect.

ID 46: Study on Temperature Estimation Method of PWR Spent Fuel Cladding in Dry Storage



Cross section of Test Container



Temperature contour calculated by FLUENT code (heat transfer test, electric heater 510W)

- The thermal analysis tool using ANSYS FLUENT code is developed with for estimating the stored spent fuel cladding temperature history during the storage period.
- Up to two spent fuels can be stored in the test container. In the initial stage of storage test (first 10 years), only a 42.8 GWd/t fuel will be stored. In the second stage (beyond 10years later), a 55GWd/t fuel will be added.
- The preliminary heat-transfer test using electric heaters was conducted. Validation of the analysis tool was made by comparing the analytical results with the measured temperature distributions of the test container during the heat-transfer test.
- The analysis results well agreed with the measured temperature distributions and it can be concluded that the fuel cladding temperature could be estimated from the surface temperature distribution of the test container. In the initial stage of storage (up to 10 years) of 42.8GWd/t fuel, the analysis tool is expected to estimate the cladding temperature with uncertainty of less than 10℃.

Huge non-spent potential of the “spent fuel”- the ways of its utilization

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Nuclear power is potentially the cheapest and waste-free source of electricity. How to realize this potential?

Actually the real potential of nuclear power is not realized. Up to now nuclear power still remains “in the shadow” of nuclear weapons. As a result, most of the NPPs use enriched uranium as a fuel (which is more expensive as compared with unenriched natural uranium). Having relatively small thermodynamic efficiency (as a consequence of security concerns), NPPs produce enormous thermal pollution, plus generate radioactive “nuclear wastes” (fortunately, in relatively small physical volumes with respect to traditional power production).
Cardinal solution for this problems would be creation of new generations of high-performance NPPs with **increased total efficiency** (which would mean attenuation of all shortages, inclusively reduction of spent fuel and other “wastes” quantities).
Taking into account this perspective, we consider opportune to propose also relatively short-term improvements (which could be realized basically in the limits of presently operating facilities and technologies).

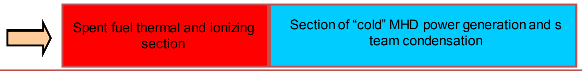
&

The resulting thermal pollution in nuclear power industry is inadmissibly large (about 70%); the output is the highly radiating “spent fuel”, which needs expensive storage; its reprocessing generates “nuclear wastes”

Solution for thermal pollution elimination using spent fuel

The problem of secondary (“waste”) energy, or harmful (and taxable!) thermal pollution, is common for all power plants, but it is more acute for NPPs. In [1] is presented a complex solution for this problem. “Waste” energy emissions are excluded – mainly due to application of thermal accumulators and large scale implementation of low-temperature food dehydration.
The possibilities of cost-effective non-thermal plasma generation using highly-radiating spent fuel offer additional unique conditions for ensuring microbiological security of the final products.
Additionally we propose to use highly-radiating spent fuel for intensification of heat and mass transfer processes at the NPPs, first of all for water vapor condensation. (By the way, the same technology actually could be used on each “traditional” power plant based on vapor cycle).
The idea is – to use ionizing and thermal potential of the spent fuel for condensation process improvement. The used water steam on its way to condenser is transformed into “cold” plasma of maximal density - due to ionizing radiation from the spent fuel, assisted by **α** and (or) **β** radionuclide emitters and (eventually, if necessary) -by nanosecond pulse generators. Moving “cold” plasma, being placed in constant magnetic field of induction **B**, transforms a part of its kinetic and thermal energy into electricity. In fact, we realize a magnetohydrodynamic generator, but in non-traditional form - with “cold” plasma. Such a procedure not only intensifies and simplifies condensation process, but also ensures additional electricity generation.
Partial water molecules dissociation - is also a positive result.
N.B. Because of the safety precautions for such applications should be used exclusively undamaged (checked through NDI etc.) spent fuel rods (bundles).

High-quality native products - as a result of secondary (“waste”) energy utilization from NPPs; the moderate low-temperature vacuum dehydration guaranties quality. Microbiological safety is ensured due to “cold” plasma [2]



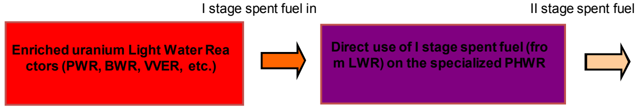
The scheme of improved steam condensation combined with “cold” MHD additional power generation

Integrated solutions for spent fuel utilization

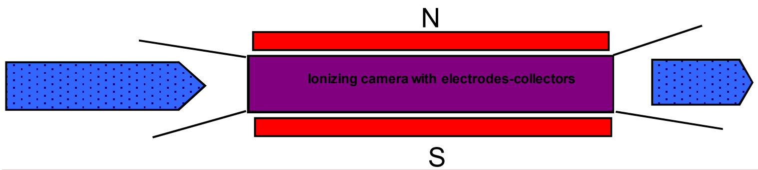
We anticipate that the progress in nuclear power will follow **2 directions: improvement of the presently running “nuclear thermal machine” (exothermic nuclear technologies), and direct transformation of radioactive heavy elements nuclear energy into electricity.**
As for the *first direction*, it could be realized through the increase of working temperature. We argue that implementation of microthermotechnical solution will be the most advantageous. (It means that instead of rods there will be used microwires, etc.). Such a solution favorites not only proper nuclear power generation process, but also subsequent spent fuel utilization and eventual reprocessing. At higher working temperatures - inevitably more intensive - ionization and decomposition (dissociation) of the cooling agent should be used. (We consider that again MHD mechanism could be enabled).
As for the *second direction*, it is entirely the domain of efficient application of the energetic potential of spent fuel and its reprocessing derivatives.
There is one more possibility to improve drastically the current situation in nuclear power domain, inclusively with spent fuel utilization.
We consider that it is rational to ensure the progressive growth of heavy water NPPs, which could re-use the entire quantity of the spent fuel coming from the “enriched uranium” NPPs. The necessary matching number (or installed power) of the natural uranium (“heavy water”) NPPs adapted for the re-use (consume) of this spent fuel could be smaller - because nuclear fuel “burning” in heavy water reactors is more intensive (faster) than in enriched uranium reactors. Such a secondary use of the “spent fuel” would permit substantial increase of the nuclear power generation - with relatively small capital investments and **without any growth of the spent fuel quantity**. In fact, spent fuel from the “enriched uranium” NPPs should be named “partially spent fuel”, or “first stage spent fuel”.

Extra NPPs technologies

We develop the next technologies for the spent fuel and its reprocessing derivatives utilization: cost-effective non-thermal plasma generation for large scale microbiological sterilization; “distributed” energy-active storage, inclusively geological one; geothermal energy and precious gas components extraction; wind (éolien) MHD power generation and other technologies.



The scheme of two-stage enriched uranium fuel utilization. For direct re-use of the I stage spent fuel coming from LWR it is necessary to implement specialized heavy water reactors (placed on the same site with the LWR, or apart). The quantity of spent fuel for the unity of power produced decreases substantially



The scheme of wind (éolien) MHD power generator. No moving parts. Charge separation occurs under the action of the Lorentz force **F** = Q [**v**×**B**]. The main ionizing agent – medium-lived “nuclear waste” strontium Sr^{90} (it can be assisted by nanosecond pulse generators). It is nearly “pure” beta emitter with E = (0,546 MeV + 2,28 MeV) and is one of the most abundant isotopes of uranium fission (about 4.5% of the total nuclear daughter products).

Conclusions:

Spent fuel and its derivatives contain huge energetic and technological potential. There is an urgent need of elaboration and implementation of the adequate technologies of its utilization. The respective terms “spent fuel”, “storage”, “disposal”, “nuclear wastes” should become archaic.

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[2] Boşneaga Iu. *Non-equilibrium plasma for sterilization in food industry.* (Poster report). **International Symposium on Food Safety and Quality: Applications of Nuclear and Related Techniques**. Vienna, Austria, 10 – 13 November 2014.

ID 82

Technologies for integrated safety management regarding gas generation in spent fuel wet or dry transport systems

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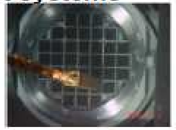


CONTEXT

Spent fuel transportation or storage request safety management regarding gas generation from water decomposition.

From storage pool at reactor, spent fuel is loaded in canister or casks and prior to transportation, a drying operation is carried out to ensure the achievement of minimal dryness criterion; then cavity is backfilled with inert gas. However, drying is not suitable in the case of leaking rods in a cask cavity. A solution is presented in another paper: "Safe Solutions For Transport and Dry Storage of defective Fuel Rods".

Safety analysis assess the radiolysis / H_2 risk by examining the influence of residual water in the cask.

An evaluation/mitigation of hydrogen generation in transportation packages is necessary to ensure that a flammable mixture will not be formed and to verify that casks do not accumulate an unsafe concentration of hydrogen (lower flammable limit (LFL) for H_2 < 4% vol. at room temperature in air).

ISSUES

Currently time constraints are requested to avoid H_2 accumulation.

In France, transport of leaking rods is limited since 2009 and investigations are done since to evaluate hydrogen risk.

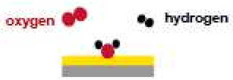
New process : H_2 measurements will be carried out before transport and extrapolation of H_2 concentration in the cavity will be used to fix the maximal transport duration.

Mitigation of the hydrogen risk in transportation casks

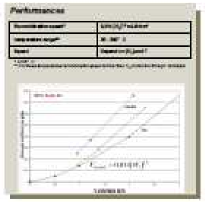
Technology for wet transportation of spent fuel : catalytic recombining system

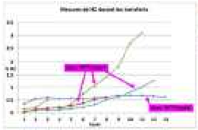
Metallic recombiners are catalysts of the reaction hydrogen and oxygen which are present in the system.

Water is produced, which may be decomposed by radiolysis but at a slower rate than the recombination. Therefore the amount of Hydrogen and other species reach an equilibrium.



These recombiners are efficient in a wide range of temperature (35-200°C) and pressure (> 1 bar).





At nuclear site: comparison with/without recombiners

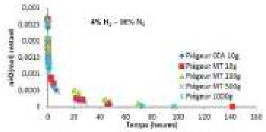
Technologies for dry transportation/storage : hydrogen getters

Currently : Efficient drying procedure, inert gas used in casks/canisters

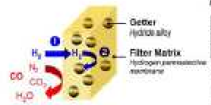
NEW: use Getters. Hydrogen is absorbed by these materials and chemically bound irreversibly in the crystalline structure.

1- Oxide getters studied by CEA

Composition :	Metal Oxides Ag_2O/MnO_2
Type :	☐ Recombiner ☑ Getter → no O_2 necessary
Reversibility of chemical reaction:	NO
Capacity :	0,138 [illegible]
Absorption kinetics :	V = 486 [illegible]
Temperatures :	< 400° C
Poisons :	Poison: HCl, resistance to CO and CH4
Irradiation influence:	Chemical properties maintained : Gamma irradiation until 4Mgy
Industrialisation :	Done



2- Composite getters Zr_2Fe



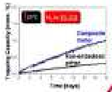
❶ Hydrogen diffusion through the matrix

❷ Hydrogen trapping by the getter

*Performances properties**

Sorption capacities	1 ml H_2 per cm³
Temperature range	-25 to 200° C
Speed	Depending on T and P**

In the situation of gas mixtures, the composite getter has shown high trapping performances compared to simple hydride (non-embedded getter).



SUMMARY & ADVANTAGES

Different technologies need to be selected for each situation of transport or storage

Safety Management :

- Flammability risk eliminated by a simple and reliable technology
- Time constraints suppressed

Licensing : Qualified products for spent fuel storage conditions

Operationnal advantage : Maintenance and dose exposure reduced.

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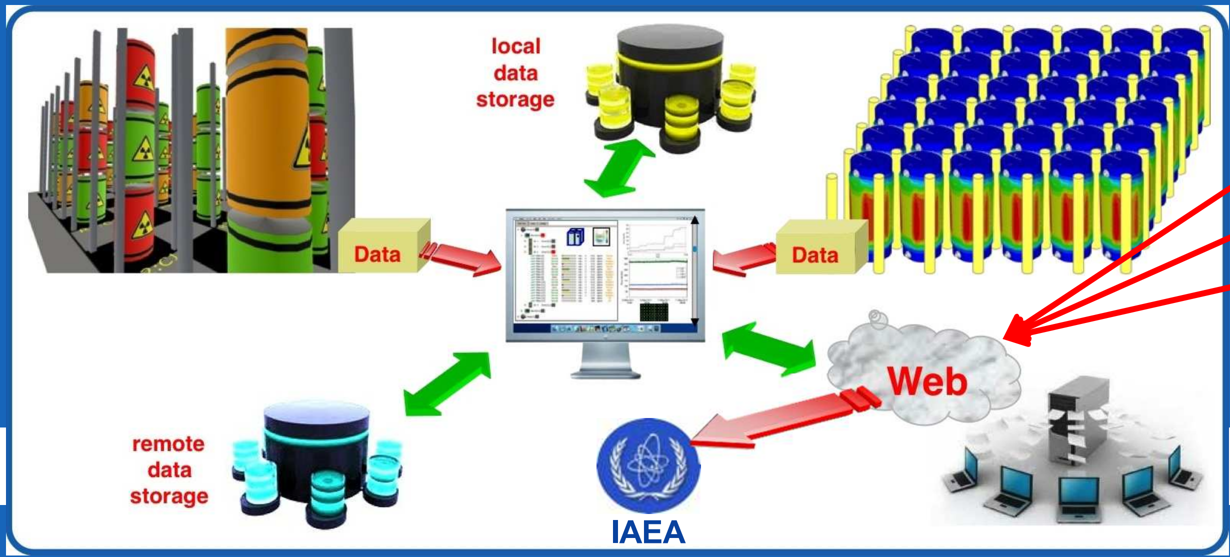
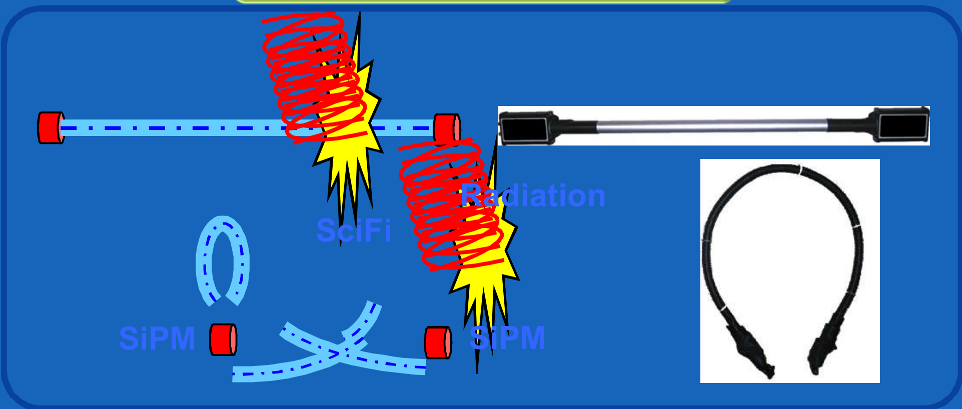
Low cost gamma and neutron radiation sen sors for real-time cask monitoring

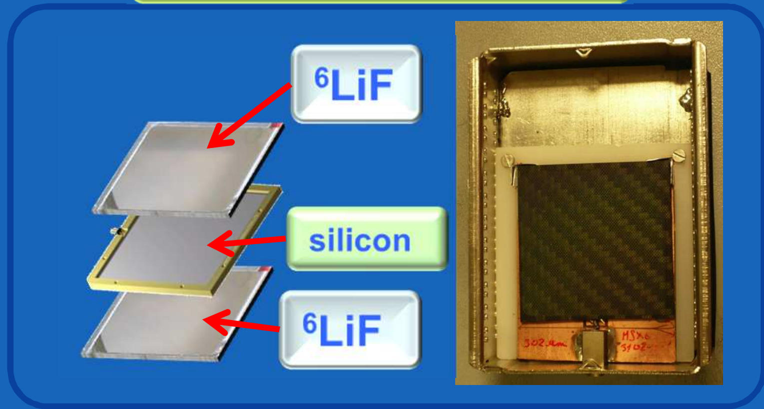


- *On-line display and data check*
- *Counting rate channel by channel*
- *Programmable alarm levels*
- *Remote data availability*

gamma detectors



neutron detectors



A Novel Approach for Monitoring Highly Active Wastes

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IAEA-CN-226-125 P

Summary

- We demonstrate the feasibility of a **3D gamma imaging** device for fuel debris location in the Primary Containment Vessel (PCV) and Reactor Pressure Vessel (RPV) in the Fukushima Daiichi reactor.
- We have constructed a gamma camera for monitoring Highly Active (HA) environments with radiation doses up to **10,000Gy/hr**.
- The results of this feasibility study will be used to extend the N-Visage imaging capabilities.

Gamma Imaging is Ideal for Detecting and Locating Radioactive Sources

- Fuel is the most dominant source of radiation in the PCV and it can be detected through gamma imaging.
- The N-visage gamma imaging scanner and software have already been very successful regarding Medium Activity (MA) monitoring.

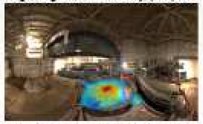

Fig. 1. Using the N-visage gamma camera to locate a MA source at Fukushima Daiichi

In some instances, such as locating fuel debris at Fukushima-Daiichi, radiation fields can be extremely intense and therefore irreversibly damaging to human resources and sensitive equipment. Although the current version of N-visage is reliable for scales up to approximately 1Gy/hr, there is a necessity for radiation-hard gamma camera that can function in the region of 10,000Gy/hr.

We have constructed a lightweight and compact camera, aimed to be used in the context of the N-visage method, with the following characteristics:

- Images at exposure rates of 10,000Gy/hr.
- Rapidly imaging.
- Robust to electrical noise.

The N-visage Imaging method

The N-visage approach replaces the standard cylindrical collimator shape with a slot. This allows a higher signal in the detector and a lower collimator mass.

The simplest physical realization of the imaging system consists of a spherical collimator with a hollow center containing a radiation detector. A slot is cut half way through the sphere to form the aperture.

Fig. 2. Example slot-collimator scan pattern. Squares indicate the presence of intermediate poses that have not been drawn.

Compared to a tubular collimator design, this configuration has a small maximum dimension and it measures the radiation incident from every angle, i.e. it is measuring the full radiation fluence at each point and can hence be used to analyse the full radiation dose.

The distribution of radioactive sources is obtained as follows:

- The N-visage scanner is used to take dose-readings at specific locations and generate a model containing the positions of all the potential sources and shielding is generated.
- Each potential source location is assumed to be affecting the magnitude of each dose-reading.
- The linear inverse problem **d = A*S** is solved.

  d: The collection of dose readings.
  A: The systems matrix encoding the distances and shielding between the locations of the dose readings and the potential sources.
  S: The radioactive-source configuration provided by the solution.

Results

The performances of a Silicon Diode and a Scintillator gamma detector were tested in a high-dose radiological environment created by a Co60 source.

- The results from the Silicon Diode detector are shown in the figure below. The data show that the Silicon Diode exhibits a linear behaviour from 20 Gy/hr up to 21000 Gy/hr and it is thus capable of functioning in radiation hard environments such as the PCV and RPV at Fukushima.

Fig. 3. Results from Silicon Diode detector at 0Gy(Blue), 0.5MGy(Black), and 1MGy total dose.

- The results from the scintillating crystal detector are shown in the next figure. The detector exhibits an almost perfect linear response to dose-rate from approximately 300 up to 21,000 Gy/hr. The radiation hard scintillating crystal is therefore also appropriate for gamma imaging in extreme environments.

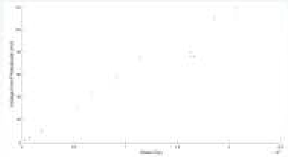

Figure. 4. Results from the detector. Blue 'x' are values from a first 'run' with increasing dose field. Green 'x' are from a second 'run' with a decreasing dose field. Red 'x' show consistent aberrations throughout the data-set.

- The relative speed of response to a fluctuating dose field, which is indicative to a detector's ability to adjust to a change of dose-rate, was measured for both detectors. This was achieved by inserting the detectors within a rotating simple slit collimator and monitoring the readout scheme at different angles. The collimator was spun at three different speeds; 1 Hz(fast), 0.59 Hz(medium) and 0.24 HZ (slow.) . The silicon diode detector could not be evaluated due to the slow response of the readout.

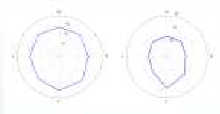

Figure. 5. Polar plots of collimator response for (Left) Silicon Diode Detector (nA) and (Right) Scintillator detector (mV). F – Front (w.r.t. source), B – Back, L – Left and R – Right.

Figure. 6. Response of the Scintillator read-out scheme with a moving collimator at a medium speed.

Conclusions

- HA monitoring through gamma imaging is feasible by modifying the existing N-Visage technology.
- We can provide pragmatic pictorial representations of HA radioactive debris at the Fukushima establishments in the near future.
- Our results are expected to have significant applications regarding the surveillance, monitoring and decommissioning of nuclear power stations.

Acknowledgments

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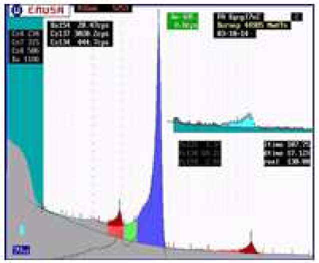
ID 125

ID 152 - GAMMA NEUTRON DYNAMIC SCANNER

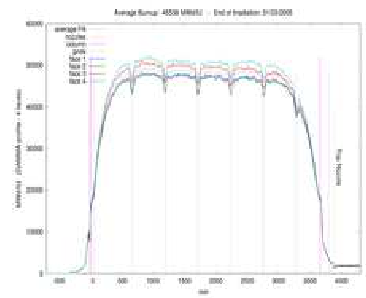
- Dynamic scanner for radiometric inspection of irradiated fuel assemblies based on gamma and neutron measurements. **Capabilities:**
  - Gamma and Neutron burnup profile
  - Burnup evaluation: Gamma spectrometry & Neutron counting analysis.



BURNUP CONTROL BEFORE DRY STORAGE

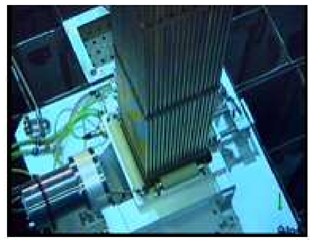
MONITORING OF FUEL BURNUP BEHAVIOUR AT REACTOR

BURNUP AXIAL IRREGULARITIES DETECTION



FEATURES

- Computational control of axial position\geometry control is guaranteed.
- Equipment qualified: complete burnup characterization for FA > 3,5 years Cooling Time
- Burnup scanner uncertainty <1.2% (2sigma)-4 faces inspection
- < 2% (2sigma) - 2 faces inspection
- Valid for PWR and BWR fuel assemblies
- Installation on the spent fuel pool racks



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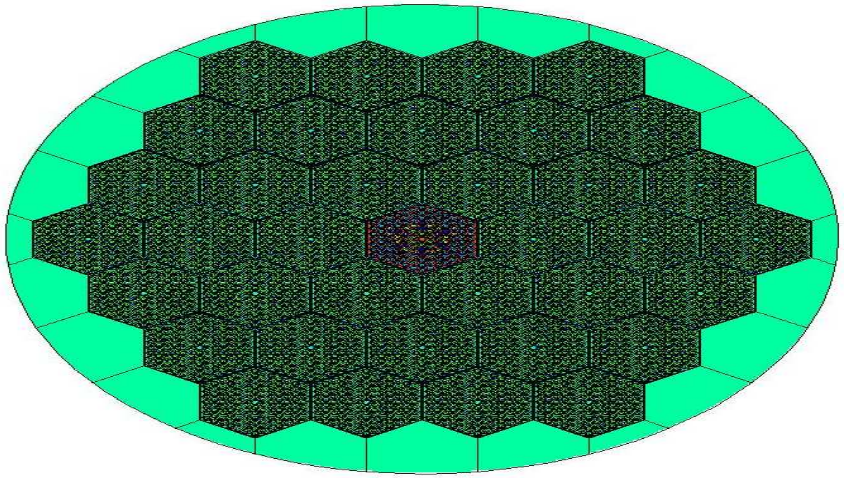
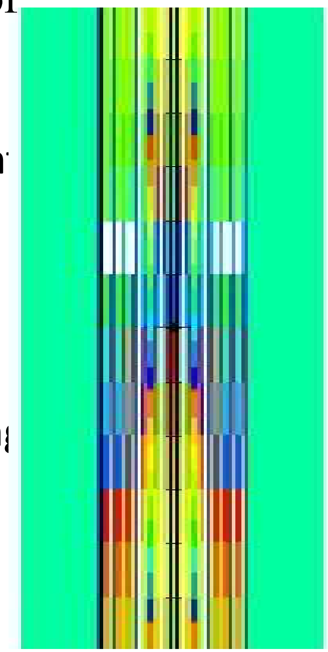
To assess applicability of MCNP6 for WWER-440 fuel type burnup credit analysis its depletion model was verified against WWER-440 spent fuel assembly chemical assay data [1].

Model was composed of 37 WWER-440 fuel assemblies to maintain criticality of depleting system and provide appropriate neutron spectra for peripheral fuel rods of depleting fuel assembly (central one).



Axial nodalization of fuel assembly was chosen to be identical to the experimen and available power history:

- 28 axial nodes for fuel rod
- axial gradient of moderator density was modeled by 12 nodes
- boron history was modeled accordin to [1]
- MCNP6 doesn't allow to model fuel temperature history



Conclusions

- good agreement for the most of measured isotopes - ^{95}Mo, ^{99}Tc, ^{101}Ru, ^{133}Cs, ^{143,145}Nd, ^{147,150,152}Sm, ^{235,236}U, ^{238,239,240,242}Pu, ^{241,243}Am
- worse agreement - ^{109}Ag, ^{155}Gd , ^{151,153}Eu, ^{149,151}Sm, ^{234}U, ^{237}Np

[1] L. J. Jardine: Radiochemical Assays of Irradiated VVER-440 Fuel for Use in Spent Fuel Burnup Credit Activities, Lawrence Livermore National Laboratory, UCRL-TR-212202

ID 71-Title : Development of the Licensing Procedure and Regulatory Framework for the Spent Fuel Storage Cask in Korea

- Background
  - There is no independent licensing procedure on the spent fuel storage cask in nuclear safety act because it is considered as the main safety equipment in the interim storage facility in Korea.
- Objective
  - The aim of this study is to develop the licensing procedure for the storage cask in nuclear safety act and to develop the revision draft of nuclear safety act on the interim storage facility additionally.
- Results
  - Independent licensing procedure for the spent fuel storage cask in nuclear safety act was developed and it was composed of the design approval of storage cask, administrative applying procedure, technical criteria, manufacture inspection, manufacture inspection criteria, periodical inspection, and periodic inspection criteria, etc.
  - In order to develop the revision draft of nuclear safety act on the interim storage facility, the licensing procedure of interim spent fuel storage facility was separated from the current ‘disposal facility, etc.’ in accordance with the article 63 of nuclear safety act independently.