DEVELOPMENT OF INSPCT-S FOR INSPECTION OF SPENT FUEL POOL

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Goal

The goal of this work is to create an efficient and accurate tool for use by inspectors to predict the response of a neutron detector placed in a spent fuel pool (specifically, Atucha-I) with the aim of detecting nuclear safeguards violations.

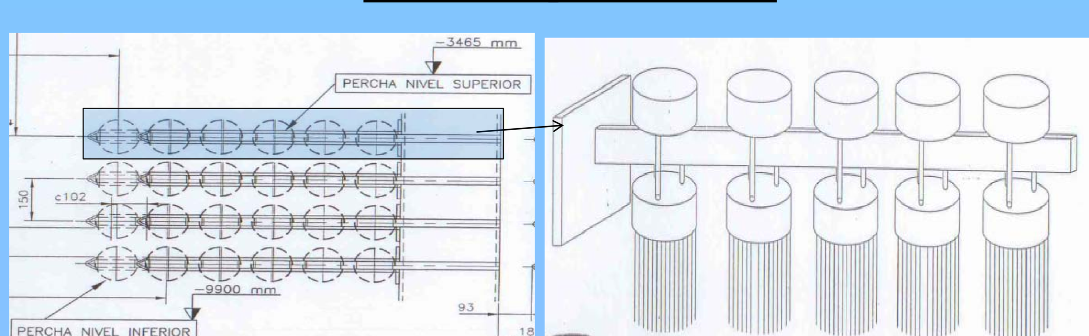
Background

Methods for spent fuel pool verification can be difficult, especially if the spent fuel assemblies cannot be easily isolated, e.g., the Atucha-I reactor in Argentina, due to interaction of detectors with many nearby assemblies. Additionally, the fuel is of low burnup and long cooling time, so traditional in-situ gamma measurement methods such as Cerenkov detectors fail due to a low gamma source strength.

Description of Atucha-I Reactor

- Located near Buenos Aires in Argentina
- 357 MWe
- Pressurized heavy water reactor
- 37 element circular fuel assemblies
- NU (Natural U) or SEU (Slightly Enriched U) (0.85 w%) fuel
- 5 – 13 GWd/MTU burnup
- Up to 40 year cooling time
- Monitored by IAEA for non-proliferation

Atucha-1 Spent fuel Pool



Top View Side view

Development of a hybrid Methodology

To estimate the response of a neutron detector, methods are needed for:

- Generation of neutron source in the pool
- Transport of neutron source to a detector

In order to develop an efficient and accurate methodology, we developed a hybrid methodology as follows:

Neutron source

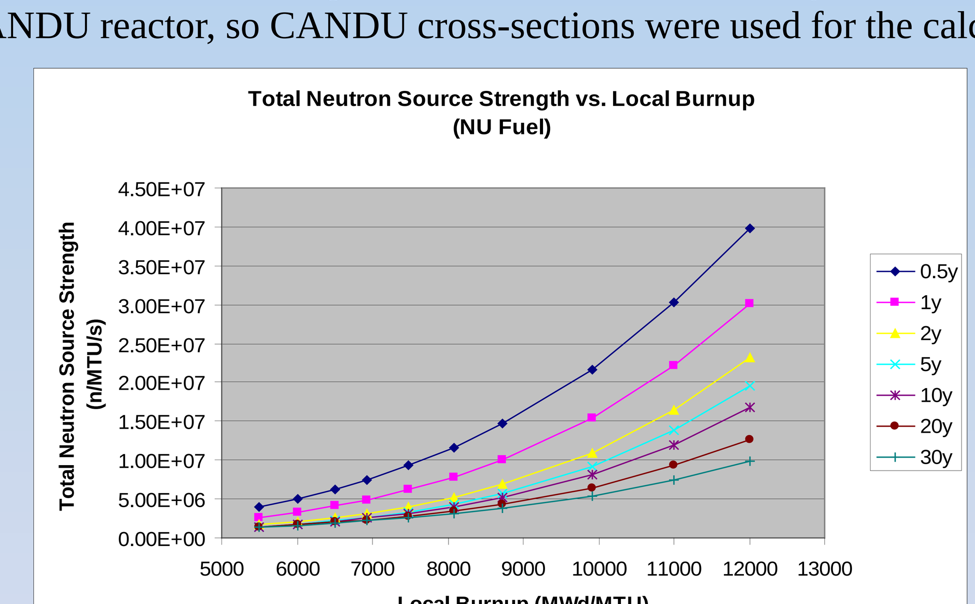
- A burnup/decay calculation provides the neutron source strength due to spontaneous fission and (α,n) reactions; referred to **Intrinsic Neutron Source**
- The fission matrix (FM) methodology using the Monte Carlo method to calculate the neutron source due to subcritical multiplication; referred to **Fission Neutron Source**

Neutron transport & estimation of detector response

- Adjoint function methodology

Intrinsic Neutron Source

ORIGEN-ARP was used to perform the burnup calculations to determine the neutron source strength. The Atucha-I reactor is very similar to the CANDU reactor, so CANDU cross-sections were used for the calculation.



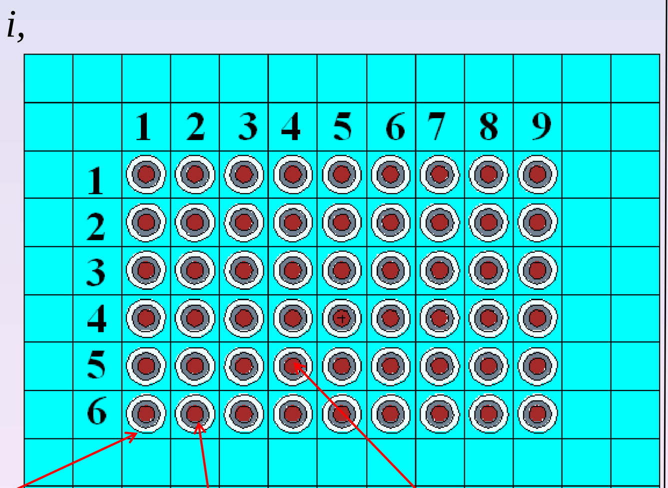
Subcritical Multiplication – Fission Neutron Source

The Fission Matrix (FM) method is expressed by:

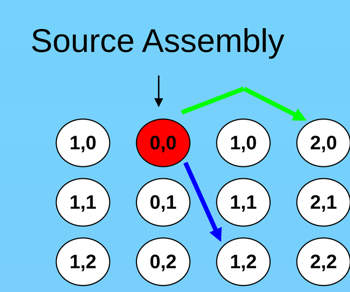
$$F_i = \sum_{i=1}^{N} a_{i,j}(F_j + S_j)$$

- $a_{i,j}$ is the number of fission neutrons produced directly in spatial cell i due to a neutron born in cell j
- F_i is the fission source in cell i
- S_i is the intrinsic source in cell i,
- N is the total number of cells

The FM coefficients are calculated by performing fix-source MCNP5 Monte Carlo calculations for different assemblies including corner, edge and interior using the following model.



a_{ij} **Coefficient** - It was noted that the coefficients varied little with the position of the source cell i *(i.e., corner, edge, or interior)*, just with the relative position of i to j. Below are the FM coefficients for an interior assembly:



Fission Matrix Coefficients			
	x-distance from source assembly		
y-distance	0	1	2
0	2.13E-01	4.98E-02	2.70E-03
1	4.56E-02	1.38E-02	1.22E-03
2	2.18E-03	1.11E-03	

Note - It is shown that the variation between a corner, edge and interior assembly is <1%. This finding reduces the necessary calculations to only one assembly location for different burnups and cooling times.

Calculation of Fission Neutron Source

After determination of the intrinsic source and the FM coefficients (a_{ij}), the above system of linear equations is solved iteratively to determine the fission source F_i.

Testing the FM method vs. a full MCNP Monte Carlo Calculation

Four assembly arrangements were tested, and the total multiplication factor (M) was calculated.

Assembly Arrangement	M (MCNP)	M (FM)	Difference	MCNP Uncertainty (1-σ)
2x6, uniform	1.7133	1.7104	-0. 29%	0.0008
9x6, uniform	1.9988	1.9966	-0. 22%	0.0007
9x6, non-uniform	2.0033	1.9968	-0.65%	0.0013
20x6, uniform	2.0513	2.0444	-0. 69%	0.0012

Computation time - the MCNP requires ~1hr, compared to < 1sec for the FM method.

Adjoint Function Methodology

"Forward" transport equation is expressed by

$$H\psi = q \quad \text{in V}$$
$$\psi = 0 \quad \text{on } \Gamma \text{ for } \hat{n}\cdot\hat{\Omega} < 0$$

where

$$H = \hat{\Omega}\cdot\nabla + \sigma_t(\vec{r},E) - \int_0^\infty dE' \int_{4\pi} d\Omega' \sigma_s(\vec{r},E'\to E,\hat{\Omega}'\to\hat{\Omega})$$

"Adjoint" transport equation is expressed by

$$H^+\psi^+ = q^+ \quad \text{in V}$$
$$\psi^+ = 0 \quad \text{on } \Gamma \text{ for } \hat{n}\cdot\hat{\Omega} > 0$$

where

$$H^+ = -\hat{\Omega}\cdot\nabla + \sigma_t(\vec{r},E) - \int_0^\infty dE' \int_{4\pi} d\Omega' \sigma_s(\vec{r},E\to E',\hat{\Omega}\to\hat{\Omega}')$$

Calculation of Detector response - Using the "forward" flux

$$R = \langle \sigma_d \psi \rangle = \int_{V_d} dV \int_0^\infty dE \int_{4\pi} d\Omega\, \sigma_d(\vec{r},E)\psi(\vec{r},E,\hat{\Omega})$$

Derive the "commutation relation" between the "forward" and "adjoint" transport equations

$$\cancelto{0}{\langle \psi^+ H\psi \rangle - \langle \psi\, H^+\psi^+ \rangle} = \langle \psi^+ q \rangle - \langle \psi\, q^+ \rangle$$

Then, we obtain the following equality

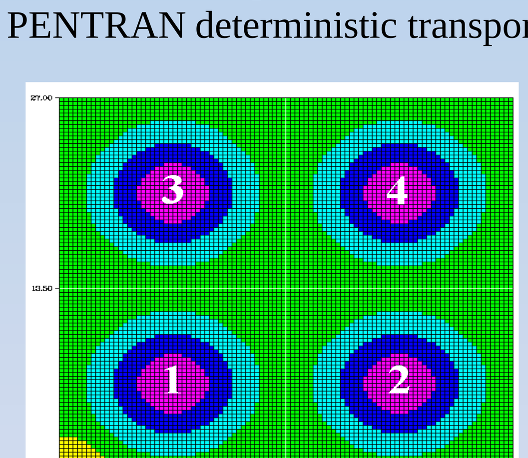
$$\langle \psi\, q^+ \rangle = \langle \psi^+ q \rangle$$

If we consider $q^+ = \sigma_d$, then

$$R = \langle \psi^+ q \rangle$$

Detector Importance and Field-of-View (FOV)

The detector importance functions, and thus the FOV, were calculated using the PENTRAN deterministic transport code for the following model.



Fission Chamber (94 w% U-235)

FOV formulation (Fractional Response)

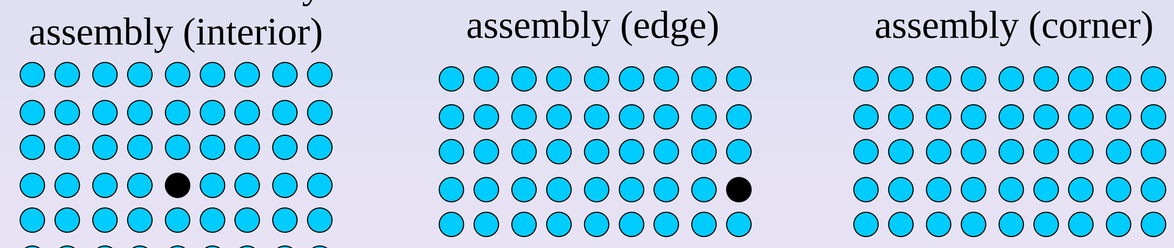
$$FR_i = \frac{\sum_g \psi^*_{ig} S_{ig} V_i}{\sum_j \sum_g \psi^*_{jg} S_{jg} V_j}$$

	Assembly Number			
Assembly	1	2	3	4
FOV	88.65%	5.31%	5.15%	0.90%

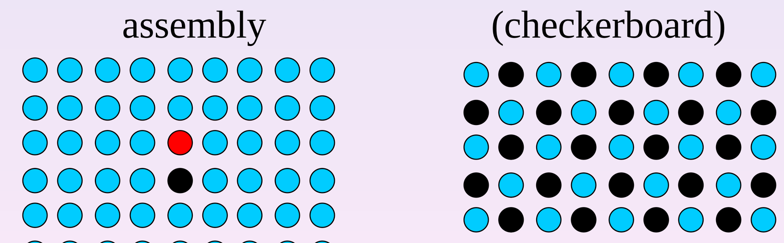
Sample Response Calculation

Using the methodology, we calculated the predicted response for numerous situations with missing fuel; below we show a few cases. We compare the response for these situations as compared to the base case of no missing fuel.

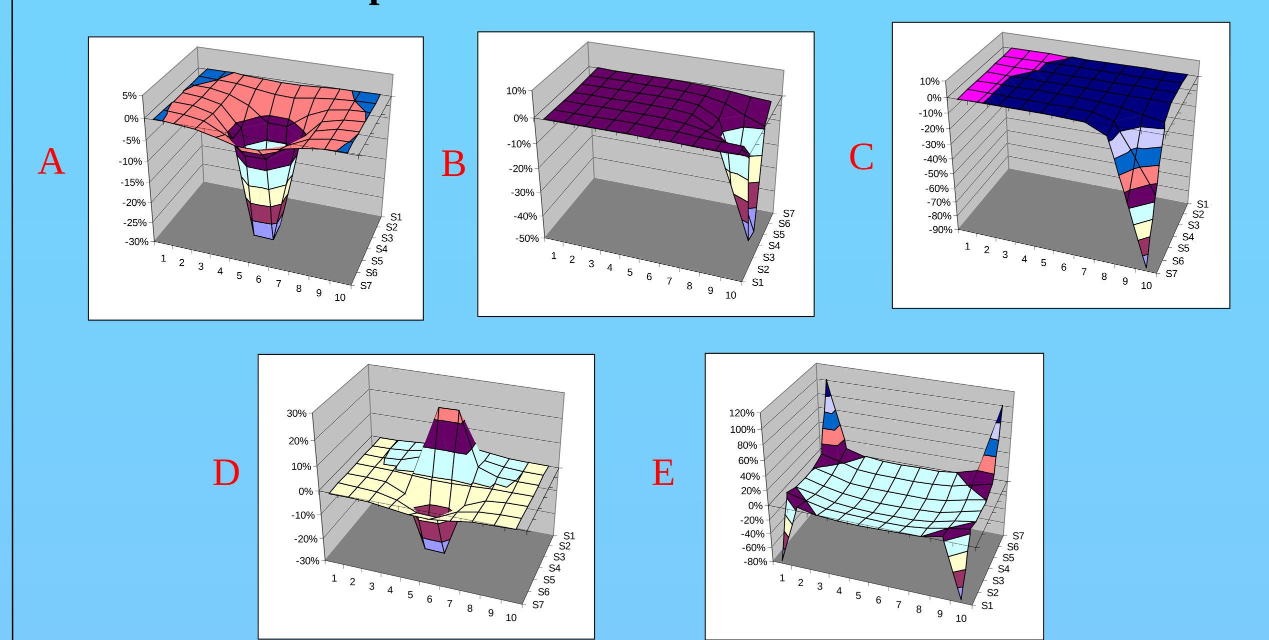
A = with a dummy assembly (interior) B= with a dummy assembly (edge) C = with a dummy assembly (corner)



D = with a dummy & a (7500 MWD/MT) assembly E= with dummy assemblies (checkerboard)



Detector Response Deviations for Proliferation Cases A-E



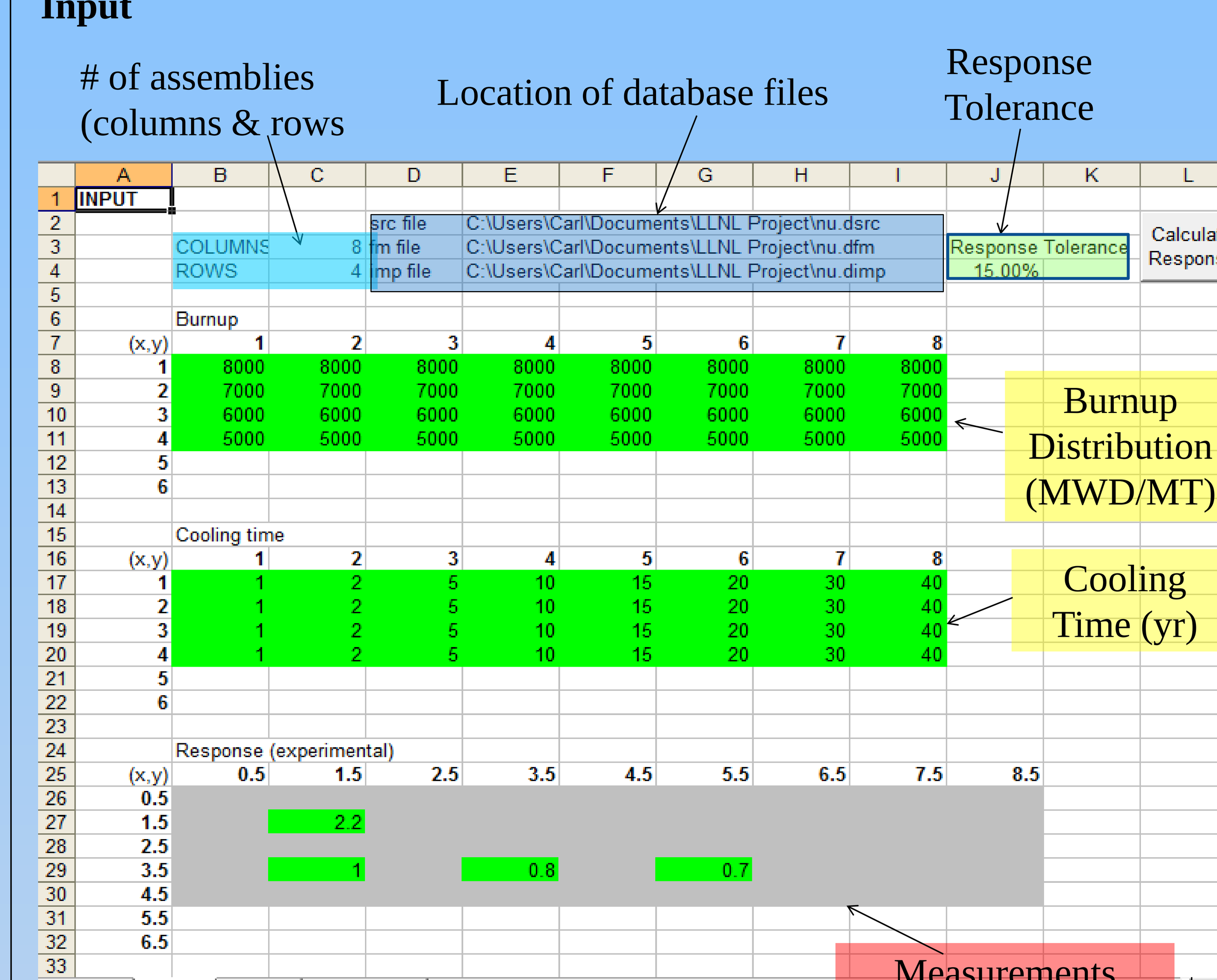
INSPCT-S SOFTWARE

The INSPCT-S (Inspection of Nuclear Spent fuel-Pool Calculation Tool ver. Spreadsheet) software is developed based on the above methodology and a set of databases. It uses an Excel 2003 spreadsheet, and Visual Basic for Applications (VBA) code within Excel to perform input and output processing, while the brunt of the calculations are performed using a dynamic-link library (DLL) created using FORTRAN-95. The spreadsheet nature of the program makes the software user-friendly and allows for easy input, output, and visualization. Below, we provide detailed information for using INSPCT-S.

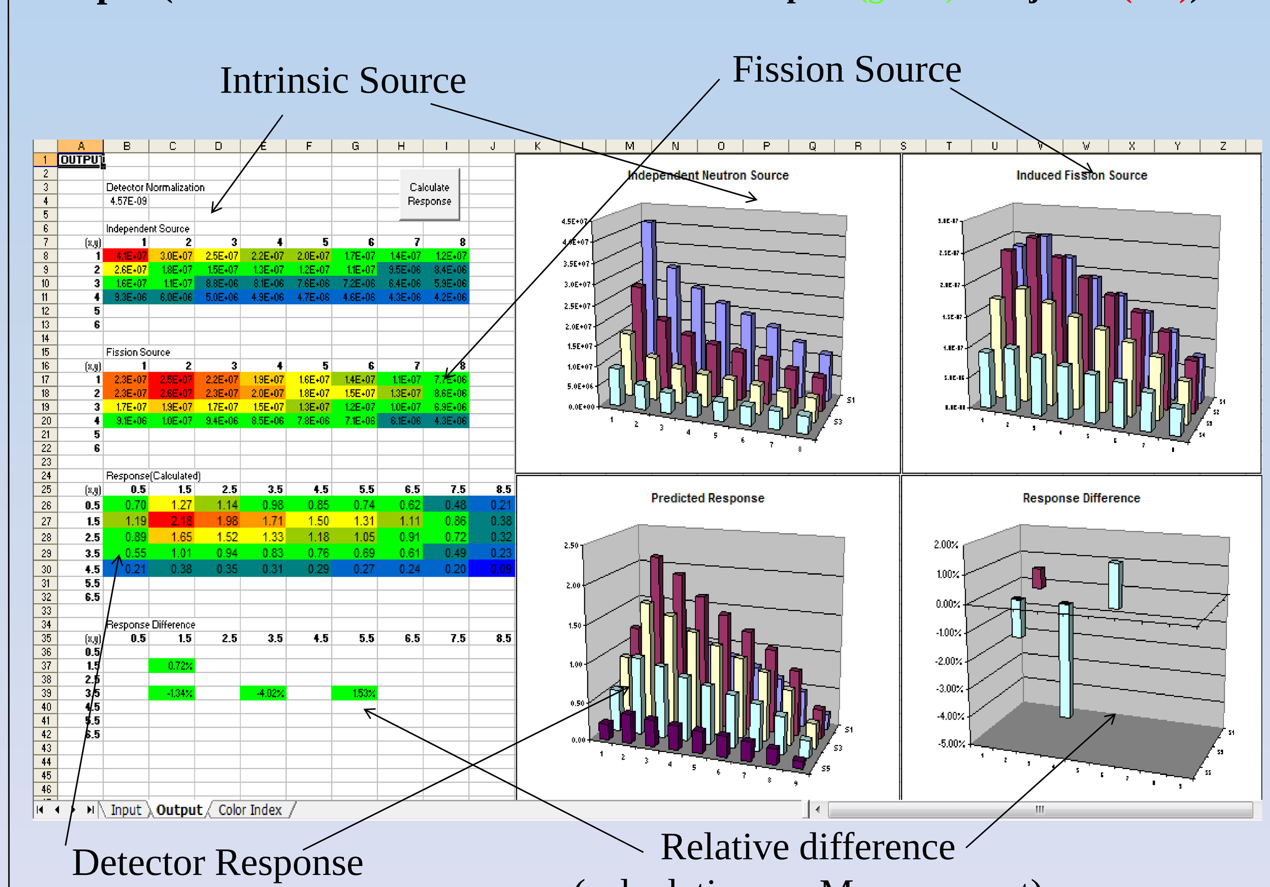
Databases

There are six(6) database files that cover the possible spent fuel parameters that exist at the Atucha-I reactor. This means that for each fuel type (Natural Uranium and Slightly Enriched Uranium) there are three database files with data on intrinsic neutron source strength, fission matrix coefficients and detector importances. For NU, the file names are "nu.*" while for SEU, they are "se.*"

Input



Output (colors in the relative difference refer to accepted (green) & rejected (red))



Summary, Conclusions & Future Work

This work develops a spreadsheet-based software for inspection of spent fuel pool, referred to INSPCT-S. This software is based on a new hybrid Fission Matrix (FM) Monte Carlo and adjoint function deterministic methodology. Databases for intrinsic source, Fission Matrix coefficients, and adjoint function distribution as a function of burnup and cooling time for Natural Uranium (NU) and slightly Enriched Uranium (SEU) fuels are prepared for the Atucha-I spent fuel pool. INSPCT-S uses Excel 2003 spreadsheet and Visual Basic for Applications (VBA) code. The spreadsheet provides a simple interface that allows for very easy input, operation, and output without detailed knowledge of the process. The computation time is < 1 sec, and therefore in conjunction with experiments, offers an excellent tool for inspection of spent fuel pool for Nuclear Safeguards. INSPCT-S will be used for inspection of the Atucha-1 spent fuel pool. Also, the INSPCT methodology will be tested for power reactors.

*Most of the work was performed while Prof .Haghighat and PhD Candidate William Walters were at the University of Florida; work was funded by LLNL