NEUTRONIC PERFORMANCE OF THE U-MO FUEL TYPE IN THE REPLACEMENT RESEARCH REACTOR

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ABSTRACT

This paper describes the general neutronic characteristics of the Replacement Research Reactor (RRR) for the Australian Nuclear Science and Technology Organisation (ANSTO). An important neutronic characteristic of the RRR design is that it can safely handle two different types of LEU Fuel Assemblies (FA): the already internationally qualified U_3Si_2 fuel type with a Uranium density of 4.8 g/cm^3, and the under qualification U-Mo fuel type with higher densities. This paper presents the fuel assembly characteristics and makes emphasis on the neutronic behavior of the new fuel type for different Uranium densities and Molybdenum contents. It compares the neutronic characteristics of the new fuel assembly with respect to the standard U_3Si_2 fuel type. The comparison includes the cycle length, several safety variables and irradiation fluxes.

RRR GENERAL DESCRIPTION

The RRR Facility is a multi-purpose open-pool type reactor. The nominal fission power of the reactor is 20 MW. The core is located inside a chimney, surrounded by heavy water contained in the Reflector Vessel. The whole assembly is at the bottom of the Reactor Pool, which is full of de-mineralized light water acting as coolant and moderator and biological shielding.

The core is an array of sixteen plate-type Fuel Assemblies (FAs) and five absorber plates, which are called Control Plates (CP). The FAs are square shaped, within each FA there are twenty one fuel plates and it uses Cadmium wires as burnable poison. The coolant is light water, which flows upwards.

Reactor shut down can be achieved by two independent means, which are the insertion of five CPs into the core, or the partial drainage of the heavy water from the Reflector Vessel.

Two types of meat are considered: uranium silicide powder or uranium molybdenum powder, both dispersed in an aluminum matrix and with enrichment lower than 20%.

DESIGN CALCULATION CODES

The calculation of the RRR is done in several steps and using different validated codes. These steps and codes are summarized in a calculation line. The calculation line for RRR is divided in three different methodologies:

- Calculation using Macroscopic cross sections. This methodology is used for almost all the neutronic parameters. The equilibrium core burnup distribution is the most important calculated parameter.
- Calculation using Microscopic cross sections. This methodology is used for the calculation of the kinetic parameters and time dependent calculation.
- Montecarlo code. This calculation methodology is used for the verification of several neutronic parameters.

The first two methodologies are divided in 3 steps: Library generation, Cell calculation and Core calculation, and the last methodology is divided in 2 steps: Library generation and Montecarlo calculation. These methodologies and their interfaces are shown in Fig. 1. It shows several codes and a short description of the most relevant items is given.

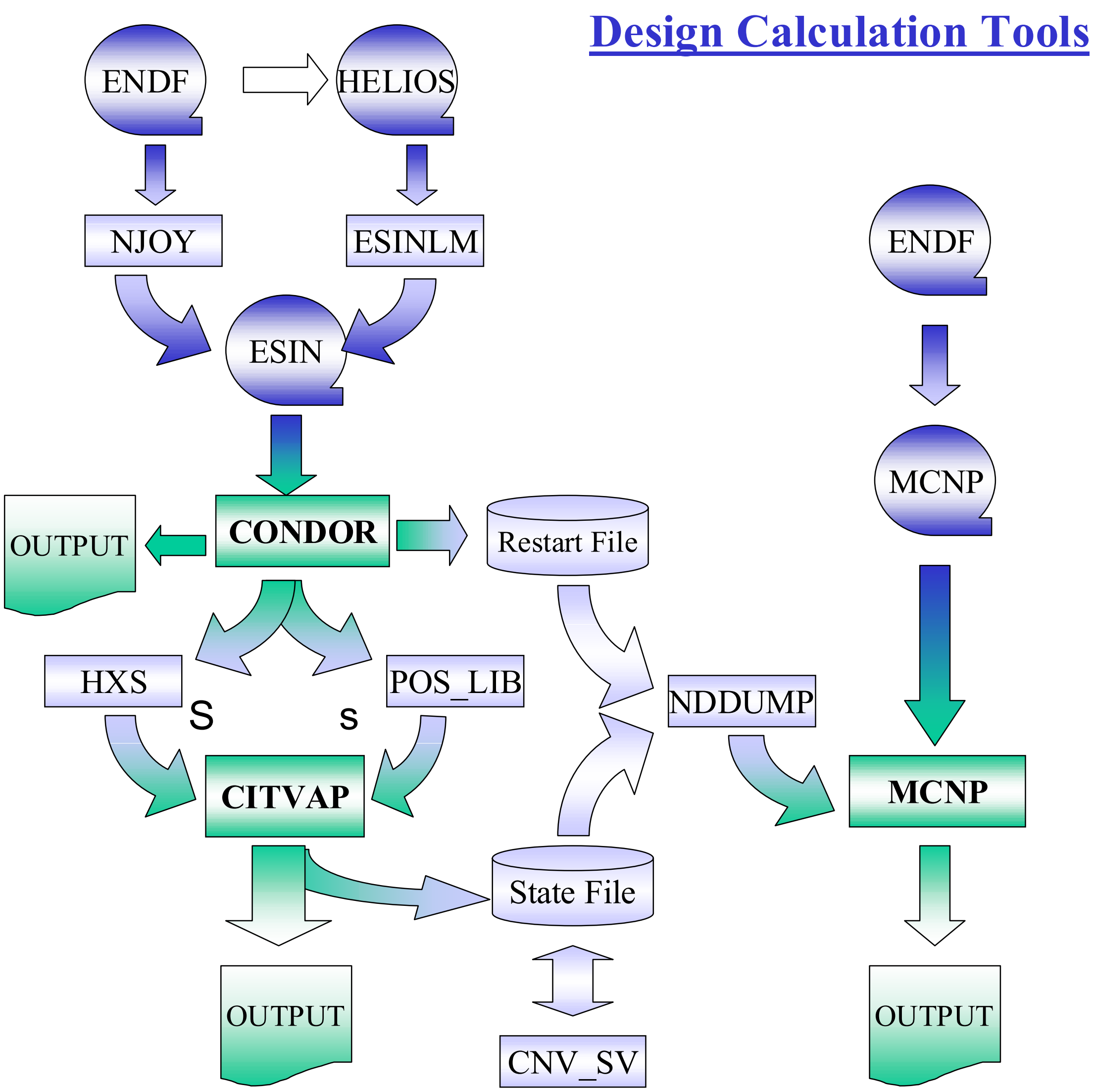

Figure 1. Calculation Line Scheme

Nuclear Data Library. Two different primary data are used to generate ESIN type libraries.

WIMS-ESIN[1]. This results from the 69 groups WIMS library[2], which has good thermal detail as well as resonant parameters. A new set of isotopes was added from the ENDF/B-VI: Ir and Te, using NJOY system.

HELIOS-ESIN. Primary data of the HELIOS[3] are from the ENDF/B-VI library. The library has three different group structures: 190, 89 and 34 groups.

CONDOR. The CONDOR Code[4] for neutron calculations is used to calculate fuel cells, fuel-rod clusters, as well as fuel plates with slab geometry or 2D geometry. Flux distribution within the region to be calculated is obtained through the collision probability method or the Heterogeneous Response Method in a multi-group scheme with various types of boundary conditions.

HXS. The HXS program[5] (Cross section handler) represents a major utility. It handles macroscopic cross-sections (identified by a name) in library form.

CITVAP. The CITVAP reactor calculation code[6] is a new version of the CITATION-II code[7], developed by INVAP's Nuclear Engineering Division. The code was developed to improve CITATION-II performance. The code solves 1, 2 or 3-dimensional multi-group diffusion equations in rectangular or cylindrical geometry. Spatial discretization can also be achieved with triangular or hexagonal meshes. Nuclear data can be provided as microscopic or macroscopic cross section libraries.

MCNP Montecarlo Code[8]. This well-known Montecarlo transport code for neutron and gamma calculations uses ENDF/B-VI cross Sections[9] in any order and performs 3-D calculations. It is used to verify some neutron parameters through an independent calculation method.

UTILITIES[10]. A short description of the main utilities is given:
POS_LIB program uses the CITVAP "state file" with a given burnup distribution and CONDOR output to generate microscopic CITVAP cross section. It also generates the Burnup dependent numerical densities to be used in the core calculation.
NDDUMP uses the CITVAP "state file" and CONDOR "restart file" to generate burnup dependent materials to be used in MCNP code.
CNV_SV is an administrative program to handle CITVAP "state file".

PRESENT DESIGN

General aspect. A compact core with 16 FA has been designed to fulfill all the neutronic design criteria. The Fig. 2 shows a scheme of the core layout with the FA and the control rods and the layout of the irradiation facilities. Their position was optimized maximizing the margins to fulfill the flux requirement and the flux perturbation between them.

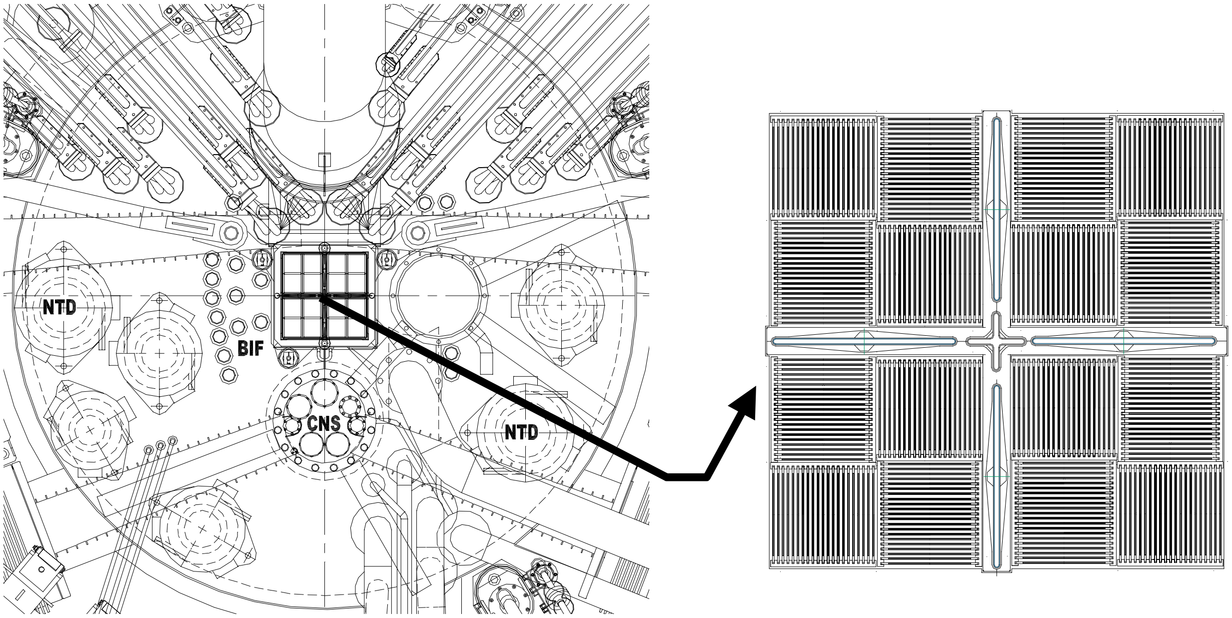

Figure. 2. Upper View of the Reflector Tank, main Irradiation Facilities and Core layout.

Fuel Assembly characteristics. This paper presents a parametric study on the FA type: U_3Si_2 and U-x%Mo fuel with two different characteristics. Figure 3 shows a quarter of the FA with the Cd wires as burnable poisons, the same geometry is used for both type of FA.and the FA main characteristics are summarized in the next table:

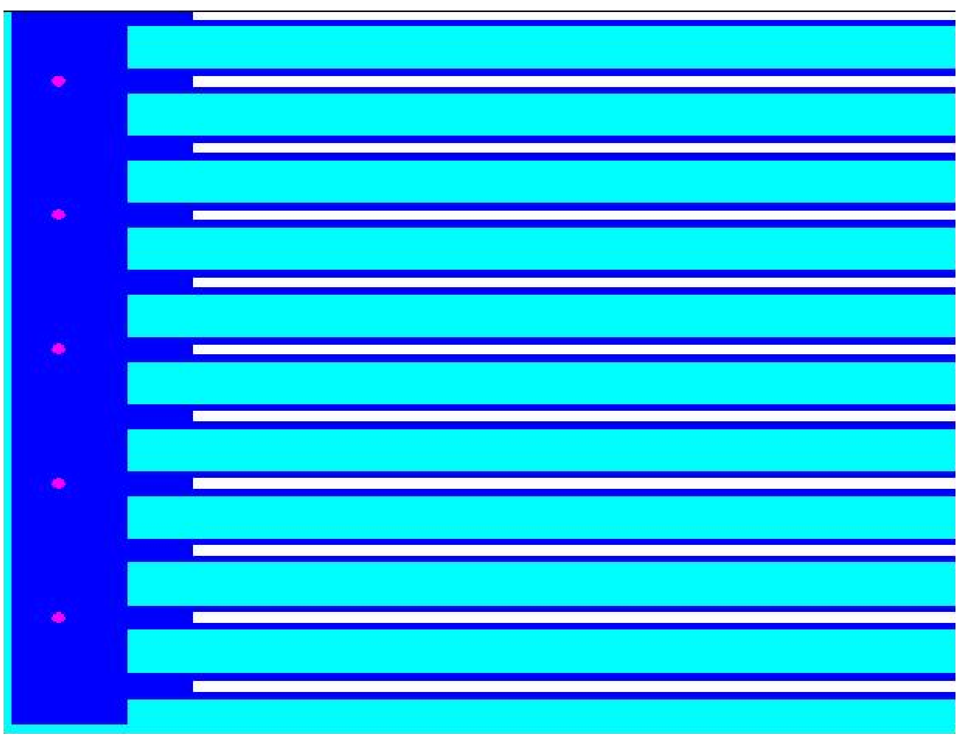

Figure 3. A quarter of FA with Cd as Burnable Poison.

Table 1: FA Main Characteristics

FA type	Label	U-Density [g/cm^3]	Mass U_{235} [g]
U_3Si_2-Al	Si480	4.8	~ 485
U6%Mo-Al	Mo600	6.0	~ 607
U9%Mo-Al	Mo700	7.0	~ 708
U6%Mo-Al	Mo70x	7.0	~ 708
U7%Mo-Al	Mo800	8.0	~ 809

Core Characteristics. The table 2 shows the core reactivities in different conditions, and the verification of some design criteria like shutdown margins. The cycle length was adjusted to have similar end of cycle reactivities.

Table 2: Core parameters

Variable	Unit	Limit	Si480	Mo600	Mo700	Mo70x	Mo800
Cycle length	days	>26	30.0	27.8	33.4	34.5	40.5
Number of FA per cycle	-	-	3	2	2	2	2
FA annual consumption.(*)	# FA	-	34.2	24.5	20.6	20.0	17.2
Hot with Xe BOC reactivity	pcm	-	4230	3610	3710	3820	3890
Hot with Xe EOC reactivity	pcm	>1200	1510	1420	1430	1450	1400
Cold without Xe BOC reactivity	pcm	-	8210	7690	7840	7930	7940
Cold without Xe EOC reactivity	pcm	-	5540	5570	5600	5630	5640
Power Peaking Factor	-	< 3	2.14	2.38	2.51	2.53	2.65
Cold No Xe BOC Shut Down Margin	pcm	> 3000	10760	10160	9230	8980	8350
Cold No Xe BOC SM (single failure)	pcm	> 1000	6130	5790	5110	5180	5170
Second Shutdown System	pcm	> 1000	6370	6000	5910	6300	5740

(*) Two days for refueling and maintenance per cycle is considered.

Flux on irradiation facilities. This subsection gives neutronic fluxes per irradiation facility type. The reference values presented are for the BOC state of the U_3Si_2 FA type, and the difference with the U-x%Mo fuel.

Bulk Production Irradiation Facilities. These facilities have 17 irradiation tubes with 5 targets each. Table 3 shows the average thermal flux per type (very high, high and medium flux) for all the targets.

Table 3. Bulk production irradiation facilities

Facility	Number of tubes and targets	Si480 (ref flux)	Mo600	Mo700	Mo70x	Mo800
		(n cm^{-2} sec^{-1})	(%)	(%)	(%)	(%)
Very High Flux	2 / 10	2.4+14	0.5%	-0.7%	0.4%	-2.1%
High Flux	3 / 15	1.5+14	1.4%	0.4%	0.7%	-0.4%
Medium Flux	12 / 60	8.9+13	-1.0%	-2.1%	-1.4%	-2.6%

<u>Pneumatic Conveyor Facilities</u>. These facilities have 19 irradiation rigs with a different number of targets per facility (it ranges from 1 to 5 targets per rig). Table 4 shows the thermal flux (except FF: fast flux facility) for each level of flux requirement.

Table 4. Pneumatic conveyor facilities

Facility	Number of rigs and targets	Si480 (flux)	Mo600	Mo700	Mo70x	Mo800
		(n cm^{-2} sec^{-1})	(%)	(%)	(%)	(%)
LV-1	'1 / 3	3.6+12(*)	~-1.5%	~-3%	~-3%	~-3%
LV-2	'2 / 6	8.7+12(*)	~-1.5%	~-3%	~-3%	~-3%
LV-3	'4 / 12	1.6+13(*)	~-1.5%	~-3%	~-3%	~-3%
LV-4	'2 / 6	3.2+13(*)	~-1.5%	~-3%	~-3%	~-3%
LV-5	'2 / 6	5.2+13(*)	-1.5%	-3.0%	-2.5%	-3.4%
LV-6	'2 / 6	7.2+13(*)	-1.4%	-3.9%	-2.5%	-3.4%
LV-7	'2 / 10	1.2+14(*)	-3.2%	-3.0%	-4.3%	-5.3%
NAA	'1 / 1	2.5+13(*)	~-1.5%	~-3%	~-3%	~-3%
DNAA	'1 / 1	5.9+12(*)	~-1.5%	~-3%	~-3%	~-3%
FF	'2 / 6	7.6+12(#)	0.3%	-0.3%	1.1%	3.0%

(*) Thermal Flux. En < 0.6 eV. (#) Fast Flux. En > 1 MeV.

<u>Large Volume Irradiation Facilities</u>. These facilities have 6 irradiation rigs and they are dedicated to the neutron transmutation doping. Table 5 shows the average thermal flux for all the facilities.

<u>Secondary Sources</u>. A cold source and a thermal source are located near the core. These facilities have 2 neutron beams each and several neutron guides. The following table shows the source average flux (cold or thermal).

Table 5. NTD and Secondary Sources

Facility		Si480 (flux)	Mo600	Mo700	Mo70x	Mo800
		(n cm^{-2} sec^{-1})	(%)	(%)	(%)	(%)
NTD	6 Targets	8.9+12(*)	-1.5%	-3.0%	-2.5%	-3.4%
Cld Source	2 Beams	7.5+13(#)	-2.9%	-4.6%	-2.5%	-5.2%
Th. Source	2 Beams	1.6+14(&)	-2.9%	-6.2%	-5.2%	-7.5%

(*) Thermal Flux. En < 0.6 eV. (#) Cold Flux. En < 0.01 eV. (&) Thermal Flux. En < 0.1 eV.

CONCLUSIONS

- The RRR has a safe and high performance core design, which fulfills all the irradiation fluxes and the operational requirements.
- U_3Si_2 and U-Mo FA types can be safely handled, fulfilling the cycle length operational requirement and irradiation flux performance.
- A very important reduction on the fuel consumption can be obtained without loosing flux performance.

REFERENCES

[1] JOSE RELLOSO, **Improvements of the WIMS library from ENDF/B-IV**. DIN/GN/001-96

[2] ASKEW, FAYERS AND KEMSHELL, **A general description of the lattice code WIMS,** UKAEA, 1967.

[3] A A FERRI, B FREDIN, F D GIUST, A JONSSON, T SKARDHAMAR Y R J J STAMM'LER. **Helios Methods, Chapter IX: The library.**. Studsvik Scandpower, December 1997.

[4] EDUARDO VILLARINO, **CONDOR Calculation Package**, Physor 2002, "International Conference on the New Frontiers of Nuclear Technology : Reactor Physics, Safety and High-Performance Computing."

[5] EDUARDO VILLARINO, **MTR_PC v2.6 User Manual: HXS 4.1,** July 1995.

[6] EDUARDO VILLARINO, **CITVAP v3.1 Improved version of CITATION II**. MTR_PC v2.6 system: User manual. July 1995.

[7] FOWLER, T. B.; VONDI, **D. R., Nuclear Core Analysis Code: CITATION**, ORNL-TM-2496, July 1969.

[8] BRIESMEISTER, J. F., ED., **MCNP - A General Monte Carlo N-Particle Transport Code, Version 4C,** LA13709-M, Los Alamos National Laboratory (April 2000).

[9] **RSICC Data Library Collection DLC-200 MCNPDATA**. Oak Ridge National Laboratory.

[10] EDUARDO VILLARINO, "**CONDOR-CITVAP-MCNP Calculation Line Description"** XIII ENFIR "National Meeting of Reactors Physics and Termo Hydraulics" Brazil"

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