DETAILED BURNUP CALCULATIONS OF IRR1 FUEL STOCKPILE BASED ON ITS COMPREHENSIVE IRRADIATION HISTORY

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ABSTRACT

The burnup of the Israeli Research Reactor 1 (IRR1) fuel elements was estimated, up until recently, within 15% uncertainty, which imposes some restrictions on fuel cycle management. Considering the high burnup values of IRR1 fuel, reducing the uncertainty of burnup values should enable longer utilization of IRR1 fuel using its current stockpile and without compromising its operational and safety limits. This study is part of a joint effort (together with Soreq Nuclear Research Center) which aims at reducing the uncertainty associated with IRR1 fuel burnup data to below 5%. A series of detailed three-dimensional full core burnup calculations of IRR1 fuel stockpile is performed based on a comprehensive and exhaustive description of the core elements and the irradiation history of the core. The data, which was collected and compiled by IRR1 personnel, details each core configuration that was assembled and irradiated since 1985, including its total burnup, as was recorded by the operators. These calculations are carried out in order to determine the amount of ^{235}U in IRR1 fuel elements.

1. Introduction

The Israeli Research Reactor 1 (IRR1) was built at Soreq Nuclear Research Center (SNRC) during the late ‘50s, reached first criticality in 1960, and is in operation now for about 55 years [1]. The current fuel stockpile of IRR1 consists of Highly Enriched Uranium (HEU) Material Test Reactor (MTR) metallic plate-type fuel elements (FEs), which are irradiated in the reactor core under highly irregular irradiation regime for the past 40 years. An irregular irradiation regime is typical of some research reactors, which often employ short and long irradiation and cooling periods, different core power levels, various types of experimental instrumentation, and numerous different core configurations of the fuel elements in the core. The reactor core was initially loaded with American fabricated fuel, but starting from 1975, French fabricated fuel (CERCA) started replacing depleted American fuel. Since 1985, the core is loaded solely with French fabricated fuel.

The knowledge of nuclear fuel's burnup data is an essential requirement for the enhancement of the safety, utilization, and performance of a nuclear reactor. Up until recently, the neutron flux distribution in the IRR1 core was calculated only approximately using GIL. GIL is a two-dimensional two-group neutron diffusion code (based on SCAR) used by IRR1 for the reactor routine operation for the past two decades. The two-group cross section library was produced by the LEOPARD code [2]. The code assumes core configuration with two sides reflected by graphite (north and south) and two sides reflected by water. Due to oversimplifications in the setup configuration and material balance, the uncertainty associated with the GIL calculations is approximated to be 10% [3]. GIL is coupled to a burnup module that evolves the total isotope content of each FE in time. In the past, there were attempts to experimentally measure the ^{235}U content of IRR1 FEs, based on gamma-ray spectrometry, the replacement of a fresh FE in the core with an irradiated one, and the measurement of the change in core reactivity using the stable period method [1,4-7].

Considering the high burnup values of IRR1 fuel (up to 70%) and that, fresh fuel supply is not foreseen in the near future, reducing the uncertainty of burnup values should enable longer and safer utilization of IRR1 fuel. This problem is definitely not unique to IRR1 and many other research reactors around the world experience similar problems.

A joint experimental and calculation effort was lunched by SNRC and Ben-Gurion University of the Negev (BGU), aiming at reducing the uncertainty associated with IRR1 fuel burnup data to below 5%. This effort was also part of the IAEA CRP T12029, on benchmarks of computational tools against experimental data on fuel burnup. In SNRC, an experimental campaign was conducted, in which the amount of ^{235}U in five different FEs was measured within 3% uncertainty using innovative Gamma-spectroscopy method. In addition, a three-dimensional Mote Carlo burnup coupling code MUTZAV (based on MCNP4b and DRAGON burnup module) was developed, validated, and used to calculate the burnup of the FEs and compare with the experimental result.
At Ben-Gurion University, a series of detailed three-dimensional full core deterministic burnup calculations of IRR1 fuel stockpile is performed based on a comprehensive and exhaustive description of the core elements and the irradiation history of the core. The data, which was collected and compiled by IRR1 personnel, details every core configuration that was assembled and irradiated since 1985, including its total burnup, as was recorded by the operators. These calculations are performed to determine the amount of ^{235}U in IRR1 FEs.

2. The Israeli research reactor #1 (IRR1)

The Israeli research reactor #1 (IRR1) is a 5 MW swimming pool type reactor operated by the Israel Atomic Energy Commission (IAEC) since June 1960 (Fig 1) [1]. Pool nominal dimensions are 11.0×6.1×9.7 m^3, and it contains 400 m^3 of deionized water ("light water"). The reactor core is fueled by 24-30 highly enriched (93%) uranium (HEU) plate-type Material Test Reactor (MTR) Fuel Elements (Fes). Each FE contains 23 parallel fuel plates mounted between two lateral aluminum holders. Overall FE dimensions are 7.6×8.0×87.3 cm^3, with an active length of 60 cm (the fuel meat). Each fuel plate measures 0.051 cm thick, and it is enclosed in 0.038 cm thick aluminum cladding [1]. Each fresh fuel plate contains approximately 12.3 g of ^{235}U. The reactor is operated 2 days a week, 6-7 hours each day.
According to the fuel manufacturer (CERCA, France), cladding integrity is guaranteed up to rather high burnup levels. However, due to various operational considerations, e.g., fuel depletion and flux distribution in the core, IRR1 safety regulations permit maximal FE depletion of 80%, which is lower than the manufacturer's limit [8].

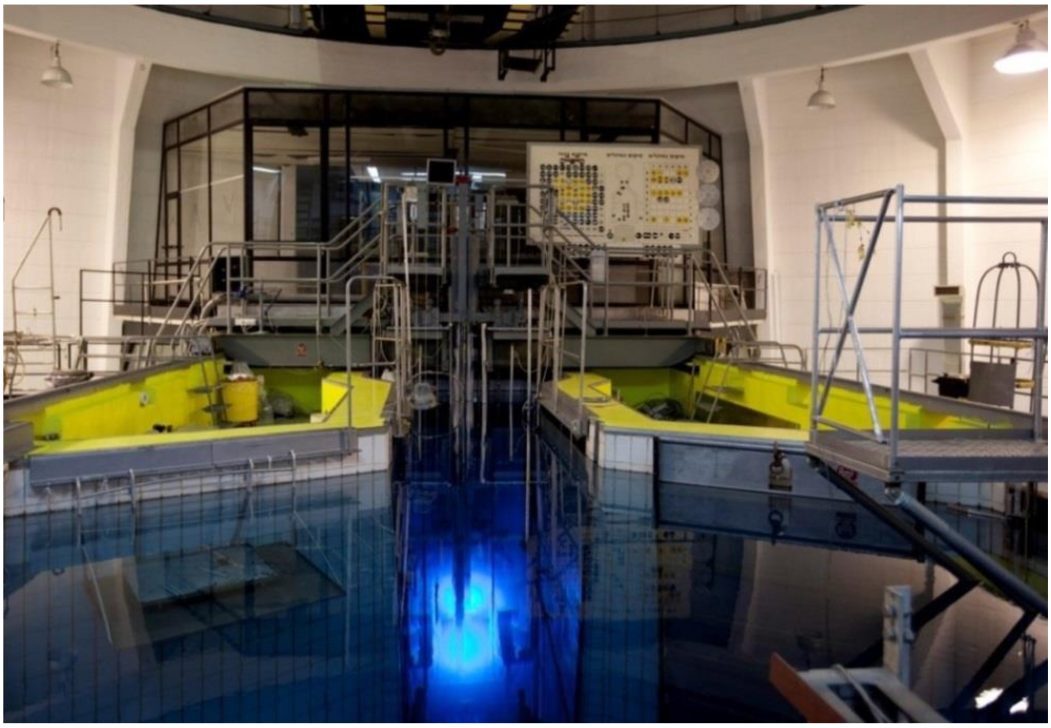

Fig 1. Front view of IRR1 [1].

General description of IRR1 facility	
Type of reactor	Open pool
Nominal power	5 MW
Maximal thermal neutron flux	
in core	1×10^{14} n cm^{-2} s^{-1}
in reflector	5×10^{13} n cm^{-2} s^{-1}
Fuel type	MTR flat parallel plates, UAl_x-Al dispersion
Enrichment	93% wt.
Coolant	Light water, downward flow
Moderator	Light water
Reflector	Graphite + light water
Heat flux	
Max.	35.4 W/cm^2
Average	11.6 W/cm^2
Nominal total flow rate	650 m^3/h
Flow rate through FEs	420-520 m^3/h (24 to 30 FEs)
Flow rate through single FE	17.5 m^3/h
Experimental instrumentation	6 radial beam tubes
	1 rabbit system
	2 tangential beam tubes
	1-3 irradiation positions

Tab 1: General description of IRR1 facility.

2.1 Core configuration

The core elements are positioned on a grid plate, which is connected to a supporting structure. Since IRR1 is a research reactor, it is operated according to constantly changing experimental programs; hence, there is no equilibrium core configuration. Therefore, the number of fuel assemblies in the core varies between 24 and 30. The reactivity is controlled by four double Ag-Cd-In blades, driven by an electric motor located on the core bridge and operated from the control room. The main parameters of a typical core are shown in Table 2.

No. of standard fuel assemblies	20-26
No. of special (control) fuel assemblies	3-5
No. of special Beryllium control assemblies	0-1
Typical arrangement of fuel assemblies	5×5 or 5×6
Core lattice dimensions	77.1×81.0 mm^2
Core coolant and moderator material light water	Light water
Temperature of coolant and moderator Maximum	35°C at inlet

Tab 2: Main parameters of a typical IRR1 core configuration between 1985-2017.

2.2 Core elements

In order to calculate the neutron flux distribution in the core and the burnup distribution of the different FEs, the neutron transport equation (or its multigroup diffusion approximation) and the isotopic depletion equations (Bateman equations) need to be solved. More specifically, in order to use the multigroup diffusion approximation, as done in this study, multigroup homogenized cross sections must be calculated. This is done using unit cell Monte Carlo transport calculations.

The FEs pitch is 7.71 cm in the x-direction and 8.1 cm in the y-direction, and is constant throughout the core. Nonetheless, the core composition is highly heterogeneous both axially and radially. The unit cells are chosen such that they represent repetitive structures in the core and are modeled separately as two-dimensional infinite lattices. Neutron data libraries of 2-group homogenized cross sections are generated for all unit cells. In case fissionable materials are presents in the unit cell, the cross sections are generated for different burnup stages.

In order to better represent the core in the multigroup diffusion approximation, the axial heterogeneity is accounted for using axially different unit cells, i.e., two-dimensional slices of the same FE but in different axial sections. For example, a standard FA can be divided into two distinct unit cells, one containing the ^{235}U meat and another, which does not (above and below the active fuel region). The geometry of standard and special fuel elements are shown in Figures 2 and 3, respectively. The material balance is given in Table 3.

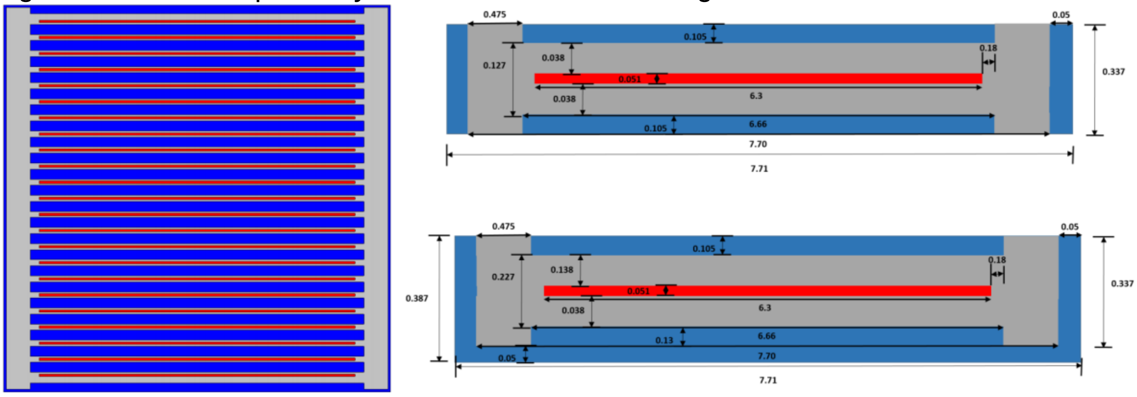

Fig 2. Geometry of an active zone of a standard fuel element. Blue, gray, and red represent water, Aluminum, and fuel meet, respectively. All dimensions are in cm.

Material	Isotope	Density [atoms/cm-b]	Material	Isotope	Density [atoms/cm-b]
Clad (Al)	^{6}Li	1.01638×10^{-8}	Light water	^{1}H	6.66324×10^{-2}
	^{7}Li	1.65525×10^{-7}		^{16}O	3.33162×10^{-2}
	^{10}B	3.99888×10^{-10}	Fuel meat	^{235}U	1.63006×10^{-3}
	^{11}B	8.87024×10^{-8}		^{238}U	1.21150×10^{-4}
	^{27}Al	2.71029×10^{-1}	Absorber	Ag	4.16103×10^{-2}
	^{106}Cd	1.43837×10^{-10}		In	2.85641×10^{-3}
	^{108}Cd	1.00509×10^{-10}		Cd	7.00091×10^{-3}
	^{110}Cd	1.38265×10^{-9}	Nickel	Ni	9.14127×10^{-2}
	^{111}Cd	1.40645×10^{-9}	Beryllium	^{9}Be	1.20674×10^{-1}
	^{112}Cd	2.62553×10^{-10}		^{16}O	1.38529×10^{-3}
	^{113}Cd	1.32004×10^{-9}	Graphite	^{12}C	8.03067×10^{-2}
	^{114}Cd	3.07589×10^{-9}	B_4C	^{10}B	2.16828×10^{-2}
	^{116}Cd	7.89623×10^{-10}		^{12}C	5.42070×10^{-3}

Tab 3: Material balance of IRR1 core elements.

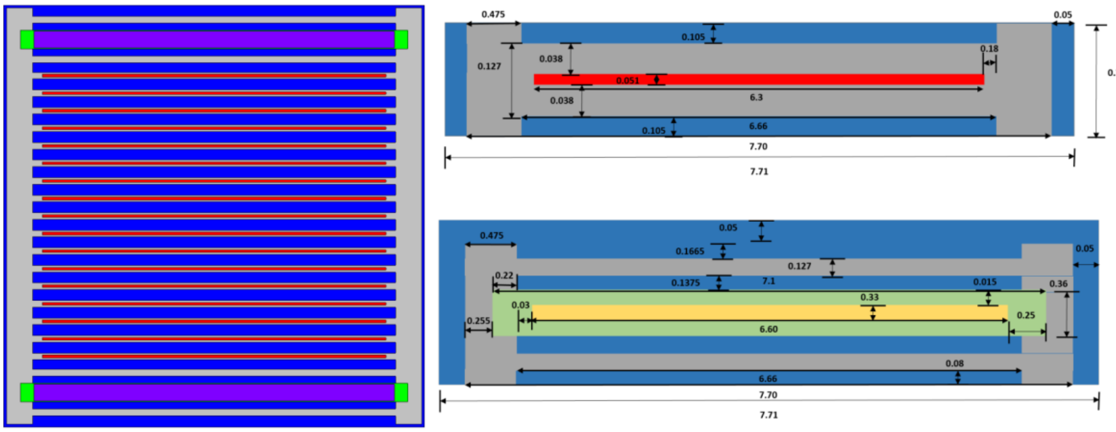

Fig 3. The geometry of an active zone of a rodded special fuel element. Blue, gray, and red represent water, Aluminum, and fuel meet, respectively. Pale green and pale yellow (on the right) and bright green and purple (on the left) represent the nickel coating and the absorber meat, respectively. All dimensions are in cm.

The beryllium block in the core functions as a neutron reflector and moderator by means of isotropic elastic scattering. Beside the beryllium block, which is located at the center of the FE, the element contains two control rods. The location of the control rods in the element is similar to their location in the special FE. The beryllium block is shaped as a corner-less rectangular, as shown in Figure 4.

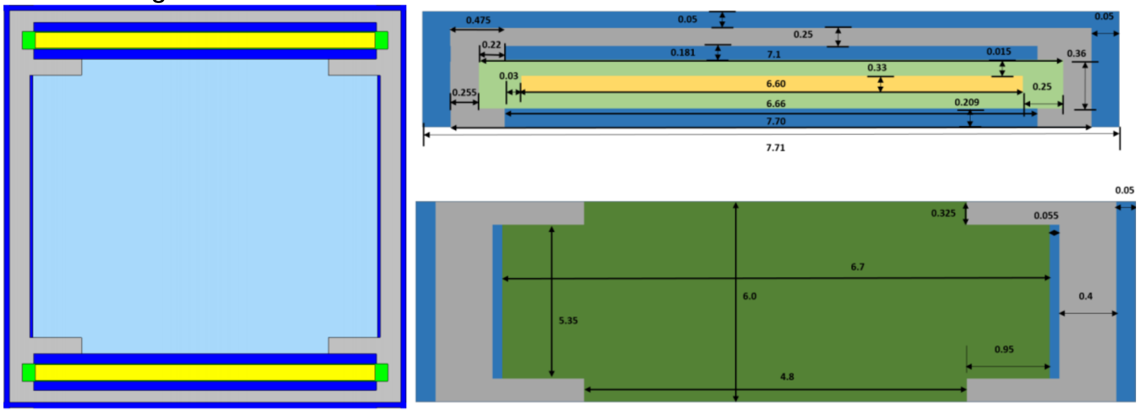

Fig 4. Geometry of a rodded beryllium element. Light blue, yellow, and light green represent beryllium, absorber, and Nickel, respectively. All dimensions are in cm.

Additional elements in the core include graphite blocks (marked as “G”) and various irradiation positions, as shown in Figure 5. The irradiation positions include a “GS” position (a graphite block with a water-filled 3.6 cm diameter channel), a “D” position (a 3.5 cm inner-diameter aluminum channel of width of 0.75 cm), a “DC” position (similar to “D” with an inner Cd ring of width 0.1 cm), and a “TBC” position (a B_4C ring coated with thin aluminum layer). All Irradiation positions have the same aluminum frame identical to that of the graphite block.
The oval stainless-steel regulation rod is marked as “R” and has an inner diameter of 5.215 cm in the X-direction, an inner diameter of 1.37 cm in the Y-direction, and is made of 0.25 cm thick stainless steel alloy.

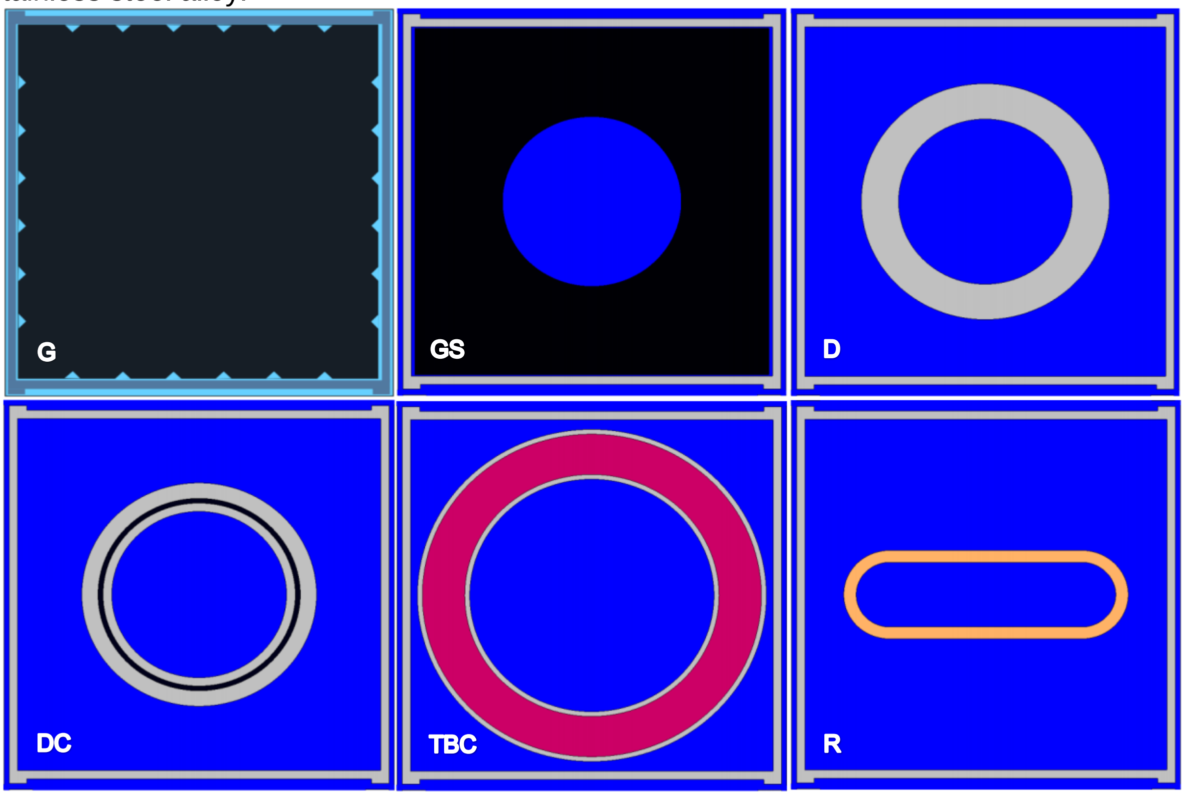

Fig 5. Additional elements in the core include graphite blocks, various irradiation positions, and a regulation rod.

2.3 IRR1 cores since 1985

Since 1985, IRR1 core is composed solely of French (CERCA) supplied fuel. In order to calculate the burnup of IRR1 fuel stockpile, the full core burnup calculations are based on the following data given by SNRC:

• Loading pattern for each core. The data of each core configuration was used to set up each core. In order to model the pool, the core was surrounded by a layer of water of varying width between 15 and 30 cm.

• Year of the specific configuration. The data of all core configurations was given in chronological order and indicated the year each configuration was irradiated.

• The energy produced (MWh). Given that the reactor power is 5 MW (3 MW starting of 2016), the temporal duration (in hours or days) can be deduced from each core's MWh.

• Initial fuel repository depletion per FE. This data is essential for obtaining absolute final burnup values of each FE in the IRR1 stockpile. This data is accompanied by potential significant uncertainties. The French fabricated fuel was gradually put into service between 1975-1985 along with the old American fabricated burnt fuel. Hence, in order to obtain an accurate estimation of the fuel initial burnup values, one has to obtain the data on core configurations and irradiation periods starting with the first core in 1960. This is beyond the scope of this study; thus, some assumptions were made in order to estimate the initial burnup values of the French fabricated fuel stockpile.

3. Methodology

Unit cell calculations and few-group cross section generation are performed for each element in the reactor core using the Monte Carlo code Serpent. Three-dimensional full core burnup calculations are subsequently performed sequentially on all 164 different core configurations irradiated in IRR1 since 1985 using the multi-group nodal diffusion code DYN3D.

3.1 Transport unit cell calculations

Serpent is a continuous energy Monte Carlo neutron transport code with burnup capabilities developed at VTT research center in Finland [9,10]. This code allows for modeling of complicated three-dimensional geometries and was originally developed as an alternative to deterministic lattice codes for the generation of homogenized multigroup constants for reactor analyses using nodal codes. The current version of Serpent contains libraries based on JEFF-2.2, JEFF-3.1.1, ENDF/B-VI.8, and ENDF/B-VII evaluated data files. In this work, the ENDF/B-VII evaluated data files are used.

Serpent also has a built-in subroutine for fuel depletion that is based on the CRAM method [11]. All nuclides and meta-stable states data contained in the decay libraries are available for Serpent calculations, where the total number of different nuclides produced from fission, transmutation, and decay reactions is in the order of 1500. The atom densities of all included nuclides with decay data are tracked in the burnup calculation, and the number of nuclides with cross sections typically ranges from 200 to 300.

In this study, Serpent is used to generate few group cross sections as a function of fuel burnup.

3.2 Full core three-dimensional nodal diffusion and burnup calculations

In this study, the code DYN3D is used for full core calculations. The code DYN3D [12,13] is a three dimensional coupled neutron kinetics and thermal-hydraulics code, developed at Helmholtz-Zentrum Dresden-Russendorf (HZDR) for dynamic and depletion calculations in light water reactors with rectangular or hexagonal lattice geometry. The multi-group neutron diffusion equation is solved by nodal methods coupled to a thermal-hydraulic model (FLOCAL). Cross section libraries generated by different lattice codes for different reactor types are linked with DYN3D. Although DYN3D was developed for analyzing power reactors and was intensively validated and verified against power reactors' benchmarks, in this study DYN3D is used for studying research reactors, such as the one considered in this work [14].

3.3 Few-group cross sections

The few-group cross section calculations of core elements containing fissile material are usually straightforward, e.g., standard FE. These are calculated using reflective boundary conditions and can be and the generation of cross section can be branched on the burnup. The energy spectrum is usually correct making the cross section collapse for energy groups a good approximation. Other core elements, either FE with strong absorbers or elements without fissile material, need a different treatment for generating a correct neutron energy spectrum in the unit cell. In these cases, a mini-core structure is employed (see Fig 6), with standard (fresh) FE surrounding the element, providing source neutrons in the appropriate spectrum.

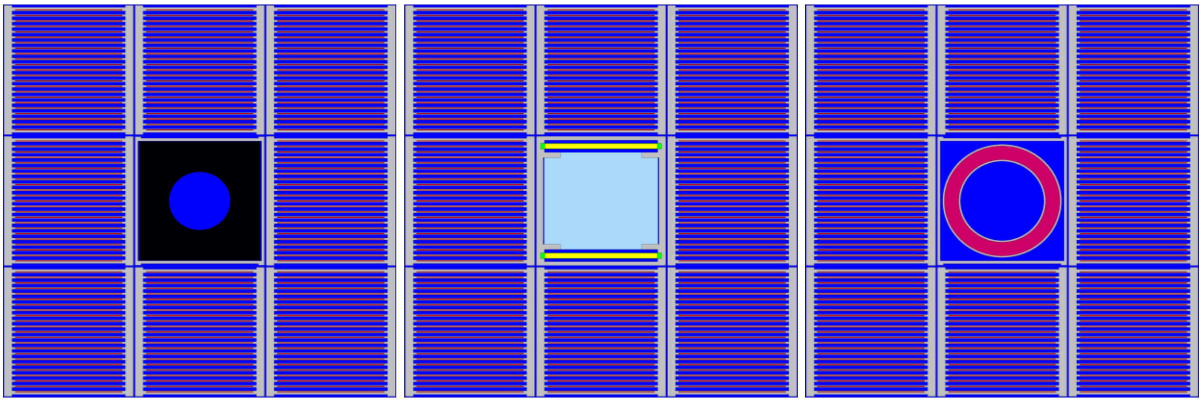

Fig 6. Examples for mini-core structures for the generation of few-group cross sections.

3.4 Validation

In order to validate the few-group cross section generation procedure, a comparison between the infinite multiplication factor k_∞ of a unit cell is made between the reference results from Serpent Monte Carlo transport calculations and the results from DYN3D nodal diffusion calculations. The validation is done by comparing the values of k_∞ as a function of burnup. The results are shown in Fig 7 for standard FE and for special FE with control blades. The results for special FE without control blades are very similar to those of standard FE.

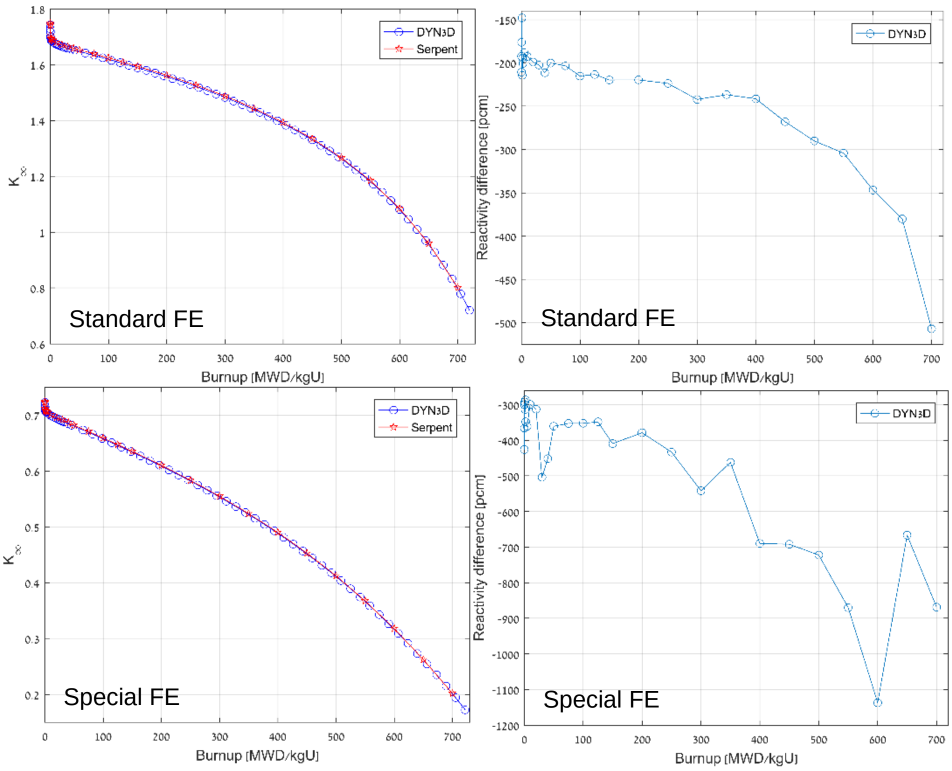

Fig 7. Comparison of k_∞ values between Serpent and DYN3D as a function of burnup.

4. Results

The burnup calculation results are summarized in Table 4 for all standard FE (1-45) and in Table 5 for all special FE (1C-5C).

ID	^{235}U content [g]		$\Delta^{235}U$ [g]	BU [MWD/kgU]		ΔBU [MWD/kgU]
	1985	2017		1985	2017	
1	153.2	115.1	38.1	353.5	462.3	108.8
2	147.8	119.4	28.4	368.8	450.0	81.2
3	182.8	125.9	56.9	270.3	431.2	160.9
4	194.8	128.7	66.1	236.9	423.3	186.5
5	218.0	127.2	90.8	172.4	427.4	255.0
6	217.6	122.1	95.5	173.5	442.1	268.6
7	208.0	120.7	87.3	200.2	446.1	245.9
8	210.5	117.8	92.7	193.2	454.5	261.3
9	203.1	124.1	79.0	213.8	436.3	222.5
10	207.4	121.5	85.9	201.9	443.9	242.1
11	238.4	123.2	115.2	116.9	439.0	322.1
12	238.9	120.8	118.1	115.6	446.0	330.4
13	240.3	118.4	121.9	111.8	452.8	341.0
14	256.7	140.5	116.2	67.6	389.7	322.0
15	256.1	124.6	131.5	69.2	434.9	365.6
16	268.8	127.2	141.6	35.3	427.6	392.3
17	279.9	134.0	145.9	5.8	408.0	402.2
18	279.0	127.1	151.9	8.2	427.8	419.6
19	282.1	153.9	128.2	0.1	351.6	351.5
20	281.6	131.5	150.1	1.2	415.2	414.0
21	281.4	280.9	0.5	1.8	3.1	1.3
22	280.9	146.1	134.7	3.2	373.5	370.3
23	280.4	159.6	120.8	4.5	335.4	330.9
24	279.4	166.8	112.6	7.0	315.1	308.1
25	279.7	161.5	118.2	6.3	330.0	323.7
26	280.6	176.0	104.6	4.0	289.4	285.4
27	280.3	170.4	109.9	4.7	304.9	300.2
28	279.9	185.1	94.7	5.9	263.8	257.9
29	280.2	190.6	89.5	5.1	248.5	243.3
30	281.8	181.9	99.8	0.9	272.7	271.8
31	282.8	206.6	76.1	0	204.0	204.0
32	280.9	230.4	50.4	3.3	138.4	135.1
33	281.1	218.9	62.2	2.8	170.0	167.2
34	282.1	234.0	48.1	0	128.7	128.7
35	281.3	235.5	45.8	2.2	124.8	122.7
36	282.2	258.7	23.4	0	62.2	62.2
37	282.6	266.7	15.9	0	40.7	40.7
38	282.3	267.1	15.2	0	39.8	39.8
39	281.7	274.7	7.0	0.9	19.6	18.7
40	282.5	282.5	0	0	0	0
41	281.3	281.3	0	0	0	0
42	281.3	281.3	0	0	0	0
43	280.7	280.7	0	0	0	0
44	283.2	283.2	0	0	0	0
45	252.6	242.3	10.3	78.7	106.5	27.9
Total	**11568.2**	**8186.5**	**3381.0**	**3047.5**	**12380.8**	**9337.5**

Tab 4: Burnup calculation results for all standard FE.

ID	^{235}U content [g]		$\Delta^{235}U$ [g]	BU [MWD/kgU]		ΔBU [MWD/kgU]
	1985	**2017**		**1985**	**2017**	
1C	120.4	69.6	50.8	321.1	516.2	195.1
2C	131.4	81.5	49.9	279.9	469.8	189.9
3C	158.3	68.0	90.3	180.0	522.7	342.7
4C	164.5	70.9	93.6	157.2	511.4	354.2
5C	187.7	78.6	109.1	73.7	481.0	407.3
Total	**762.3**	**368.6**	**393.7**	**1011.9**	**2501.1**	**1489.2**

Tab 5: Burnup calculation results for all special FE.

5. Summary & conclusions

A series of detailed three-dimensional full core burnup calculations of IRR1 fuel stockpile are performed based on a comprehensive and exhaustive description of the core elements and the irradiation history of the core. The data include all core configurations assembled and irradiated since 1985, including its total burnup. These calculations are carried out in order to determine the amount of ^{235}U in IRR1 fuel elements. Unit cell calculations and few-group cross section generation are performed for each element in the reactor core using the Monte Carlo code Serpent. Three-dimensional full core burnup calculations are subsequently performed sequentially on all 164 different core configurations irradiated in IRR1 since 1985 using the multi-group nodal diffusion code DYN3D.

The results shown here do not include the impurities in the aluminum, graphite, and other structural materials. The burnup calculations shown here are performed using the correct power level of 5 MW, which implies an incorrect number of effective full power days. Similar calculations, carried out using the correct number of operation days with an adapted (lower) power level, gave similar results and validated this approximation. The burnup calculations were performed with the control blades inserted to height 75-80%.

The results indicate a total burnup of 10822.5 MWD/kgU between 1985-2017. This result translates into a total energy production of 3161.5 MWD and the total depletion of 3775 g^{235}U, which indicate an average value of 1.20 g^{235}U/MWD.

6. Acknowledgments

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