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ORIGEN-ARP Cross-Section Libraries for Magnox, Advanced Gas-Cooled, and VVER Reactor Designs

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Prepared by

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ABSTRACT

Cross-section libraries for the ORIGEN-ARP system were extended to include four non-U.S. reactor types: the Magnox reactor, the Advanced Gas-Cooled Reactor, the VVER-440, and the VVER-1000. Typical design and operational parameters for these four reactor types were determined by an examination of a variety of published information sources. Burnup simulation models of the reactors were then developed using the SAS2H sequence from the Oak Ridge National Laboratory SCALE code system. In turn, these models were used to prepare the burnup-dependent cross-section libraries suitable for use with ORIGEN-ARP. The reactor designs together with the development of the SAS2H models are described, and a small number of validation results using spent-fuel assay data are reported.

1. INTRODUCTION

ORIGEN-ARP cross-section libraries were developed for four non-U.S. reactor designs. The ORIGEN-ARP code system is described by Leal et al.[1] The cross-section libraries are burnup dependent and span the range of reasonable burnup values for these reactor types. The libraries were developed by using the SAS2H burnup sequence that is part of the SCALE system,[2] developed and maintained by Oak Ridge National Laboratory (ORNL).

Of the reactor designs studies, two were British designs—the Magnox Reactor and the Advanced Gas-Cooled Reactor (AGR)—and two were Russian designs—the VVER-440 and VVER-1000 reactors. In developing the cross-section libraries, the design parameters employed are intended to represent these various reactor types in a generic sense for the purpose of making predictions about spent-fuel characteristics and reactor performance. Therefore, for practical reasons, it was both necessary and appropriate to infer the design-parameter values from a combination of sources.

The ARP acronym in ORIGEN-ARP refers to the phrase Automatic Rapid Processing. This approach provides an alternative to the SAS2H sequence of the SCALE system. In the ARP approach, burnup-dependent libraries are generated for a given reactor design using SAS2H. These pre-generated libraries are suitable for the ORIGEN-S code. By interpolation among these libraries, point-depletion ORIGEN-S calculations can be carried out for that particular design of reactor without requiring the full complexity of a SAS2H simulation. Depending on requirements or on the reactor design being studied, the libraries allow for interpolation on three variables: burnup, enrichment, and water (moderator) density.

The SAS2H control module was originally developed for the SCALE code system to provide a method for generating radiation source terms for spent fuel to be used in a 1-D shielding analysis of a shipping cask. However, in addition to the determination of source terms, SAS2H is now often used to perform decay-heat and nuclide-concentration calculations for spent fuel.

The process used by SAS2H is illustrated schematically in Fig. 1. During this process, three SCALE modules, XSDRNPM, BONAMI, and NITAWL-II, are used. Details on these codes can be obtained in the SCALE documentation.[2] The calculation starts with input-specified data describing a fuel assembly as it is initially loaded into a reactor. The initial composition, average temperatures, geometry, and time-dependent specific power for the assembly are required. The SAS2H sequence performs 1-D neutron transport analysis of the reactor fuel assembly using XSDRNPM and a two-part procedure with two separate unit-cell-lattice models. The first model (Path A of Fig. 1) is a unit fuel-pin cell from which cell-weighted cross sections are obtained. The second model (Path B of Fig. 1) represents a larger unit cell (e.g., an assembly) within an infinite lattice. The larger unit cell zones can be structured for different assembly designs to account for assembly-specific attributes (water holes, burnable poison rods, etc.). The appropriate cell-weighted cross sections from Path A are passed on to Path B to be used for the homogenized fuel region that will be one of the zones in the Path-B model. Problem-dependent resonance self-shielding of cross sections is performed prior to each XSDRNPM calculation using the BONAMI and NITAWL-II codes. The neutron flux spectrum obtained from the second (assembly) unit-cell model is used to determine the appropriate nuclide cross sections for the burnup-dependent fuel composition. The cross sections derived from XSDRNPM calculations at each time step are used in an ORIGEN-S point-depletion computation that produces the burnup-dependent fuel compositions to be used in the next spectrum calculation. This sequence is repeated in user-specified burnup steps for a complete assembly operating history. The buildup and decay of nuclides in the fuel assembly are then computed by ORIGEN-S in a final pass based on the assembly's cooling time (i.e., the time period following the end of irradiation). Note that ORIGEN-S calculations have no spatial dependence. The neutron flux used to produce the ORIGEN-S cross sections is based on a radial average of an infinitely long uniform assembly.

These specifications (e.g., burnup, specific power, moderator temperature, etc.) can be representative of any axial location along the fuel assembly or can be chosen to represent an axial average of the fuel assembly.

If a given assembly design is simulated using SAS2H, and if further calculations are required for that particular assembly or for a very similar one, then one can simplify the task by saving the cross-section libraries from SAS2H and using them with ORIGEN-S to do follow-on calculations as needed. ORIGEN-ARP provides this particular capability. Libraries can be prepared for varying values of burnup, enrichment, and moderator density. Subsequently, interpolations can be performed among these libraries for any desired combination of these variables within the range of the original SAS2H simulations.

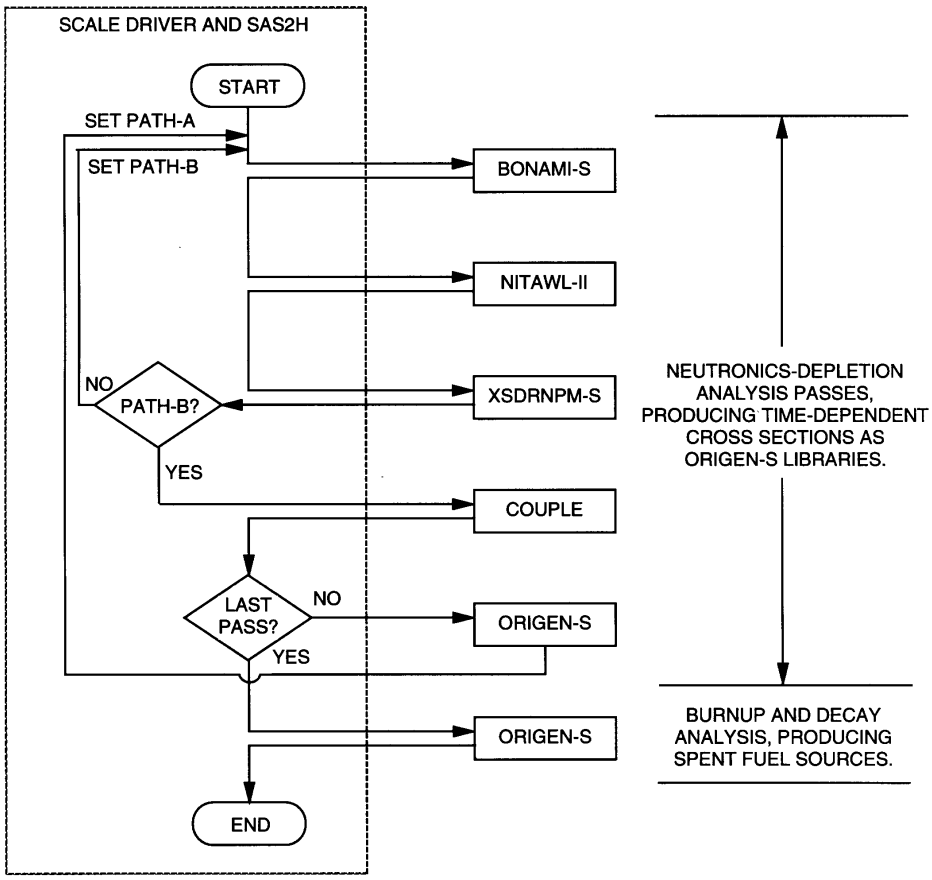

Fig. 1. Flow path invoked in SAS2H depletion and decay sequences.

2. REACTOR DESIGN DATA

This section describes the reactor design data used to develop cross-section libraries for the Magnox, AGR, VVER-440, and VVER-1000 representative assemblies. For each reactor, using the design data described, a SAS2H model was constructed to simulate burnup during typical reactor operation. The burnup-dependent cross-section libraries generated during the course of the SAS2H simulations were then used to construct the ORIGEN-ARP libraries.

2.1 MAGNOX REACTOR

A Magnox reactor takes its name from the material Magnox (an alloy of magnesium) used as cladding for the fuel rods. The fuel is natural uranium metal, and the Magnox clad has a low neutron absorption cross section. The fuel rods are placed in channels that are in a graphite-block structure with CO_2 coolant gas flowing in the channels.

Magnox reactor designs vary somewhat (see, for instance, Directory of Nuclear Reactors[3]). Depending on the size or rating of the reactor, the number of graphite blocks will vary, and later designs are, of course, larger and operate at higher power levels. As a primary example, we have used the Calder Hall Magnox reactor.[4] Calder Hall is one of the earlier designs, but it is generally typical of all Magnox reactors. The graphite moderating structure of the Calder Hall reactor is built of alternate layers of blocks and tiles. There are 112 square blocks in a horizontal plane forming a prism that is roughly circular and measures about 1077 cm across the corners. Along a diameter and parallel to a row of blocks are 12 blocks; each block measures about 81 cm on a side. This roughly circular structure is then inscribed in extra graphite blocks to give it a more nearly true circular shape.

Each graphite block contains 16 fuel-element channels in a 4 × 4 arrangement. The graphite blocks are in three distinct zones: an inner, middle, and outer zone containing 16, 36, and 60 blocks, respectively. The size of the cooling channels is different in the different zones; it is largest in the inner zone, where heat generation rates are greatest. A control rod channel is located at the center of each graphite block. The Calder Hall design is also reflected in the Bradwell and Berkeley reactor designs although these have larger cores.

Using Calder Hall design specifications, we constructed a SAS2H model of a Magnox reactor. The pin cell in SAS2H allows for fuel, gap, clad, and moderator. It is not a requirement that a gap be included. The fuel is uranium metal, the clad is Magnox, and the moderator is the graphite. Both clad and CO_2 coolant lie between the fuel and moderator.

There are a number of ways to construct a SAS2H model of a Magnox reactor. Specifically, we examined three possible approaches: One could consider the clad as the gap and consider the coolant as the clad (after all, Magnox has a negligible absorption cross section). Or one could consider the gas coolant to be the gap region, with the Magnox being the clad (This would reverse the true order of occurrence, and one would need to preserve the correct areas.) We adopted a third approach, however. We consider the fuel surrounded by Magnox, and we treat the Magnox as clad (i.e., there is no gap). In turn, the CO_2 coolant plus the graphite (using the appropriate volume fractions) are considered as moderator. The larger cell is then composed of a small central hole surrounded by a region of fuel, clad, and moderator. This larger cell was chosen to represent nine graphite blocks. Note, the central hole does not represent a control-rod hole. It is a 1-cm^2 hole (containing CO_2) that allows SAS2H to operate efficiently. SAS2H requires at least two zones for the larger cell and does not allow homogenized fuel material in the first zone. The choice of nine graphite blocks for the larger structure is arbitrary. It should, however, be mentioned that all three of these different modeling approaches were tested and were found to yield similar results.

It was pointed out that the fuel-rod channels have different radii depending on their location in the core. We used a value that is an average for the core based on the number of each type of fuel channel. The density used for the graphite was reduced to account for a control-rod hole at the center of each graphite block as well as the gaps between blocks that allow for Wigner expansion. It is possible that the expansion gaps increase the effective fuel-channel pitch. However, we used the design value of 20.32 cm. The Calder-Hall-like design parameters are listed in Table 1, as are those for the Wylfa reactor (see below).

The Calder Hall design is typical of the late fifties and early sixties. As stated, the design is also applicable to the Bradbury and Berkeley reactors. More typical later designs are the Oldbury, Sizewell, and Wylfa reactors.[3] The information that we have on the latter three reactor designs is from 1967. Besides the Calder-Hall-based model, we produced a second model that reflects the design of these later reactors. Together the two models should give a representation of most Magnox reactor designs. We use the designation Wylfa for this latter model although it can be considered representative of Wylfa, Oldbury, and Sizewell. The larger and smaller cell structures and the treatment of fuel, clad, and moderator all follow the Calder Hall example; however, the dimensions are slightly different. There is very little difference between the designs as regards the prediction of spent-fuel inventories.

Table 1. Design parameters for two representative Magnox reactors

Parameter	Calder Hall	Wylfa
Fuel density	17.98 g/cc	17.98 g/cc
Fuel radius	1.461 cm	1.38 cm
Clad radius	2.04 cm	1.61 cm
Coolant channel radius	4.849 cm	4.905 cm
Coolant temperature	650 K	650 K
Fuel material	Natural U metal	Natural U metal
Control rod hole radius	8.25 cm	8.25 cm
Clad material	Mg (1% Al, 0.05% Be)	Mg (1% Al, 0.05% Be)
Clad density	1.65 g/cc	1.65 g/cc
Moderator density	1.628 g/cc	1.628 g/cc
Moderator temperature	650 K	650 K
Fuel pin pitch	20.32 cm	19.7 cm
Coolant material	CO_2	CO_2
Coolant pressure	10 –27 bar	10 –27 bar
Typical power rating	1.5 –2.0 MW(th)/t	2.7 MW(th)/t
Typical fuel burnup	4500 MW(th)d/t	4500 MW(th)d/t
Maximum fuel burnup	9000 MW(th)d/t	9000 MW(th)d/t

Note that the fuel is natural uranium metal. The Magnox clad material consists of conventional cylindrical cladding with Magnox cooling fins on the outside. We chose not to represent the fins. Because Magnox is neutronically inert, it was not considered important to do so. The graphite density is an effective density. The graphite moderator included a small amount of boron as an impurity. Also note that we do not include any control rods in our model.

A variety of temperatures for fuel, clad, coolant, and graphite can be encountered in various descriptions of Magnox reactors. It is likely that temperatures for the Wylfa reactor should be greater than for Calder Hall. The temperatures that we chose to use were 800 K, 700 K, and 650 K for fuel, clad, and graphite, respectively. These temperatures that we have chosen are in

the high range of what we have seen reported. It might also be acceptable to reduce each one by up to 100 K. However, we feel that choosing the average temperature of the graphite moderator is not ideal. The effective moderator temperature must account more heavily for the region closest to the fuel. A report by Turner et al.[5] contains useful discussion on temperature and other issues in graphite-moderator reactors.

2.1.1 Magnox Library Details

The typical burnup for a Magnox reactor is shown in Table 1 as 4500 MWd/t. (Note that the power rating is higher for the Wylfa reactor.) It was felt that the 4500 MWd/t value was probably typical of early Magnox operating history. In fact, during validation studies to be discussed later, we encountered experimental samples where the burnup extended to 9000 MWd/t. This latter value was chosen as the maximum burnup, and the Magnox libraries were allowed to span a range of burnup values up to 9000 MWd/t.

Magnox reactors burn natural uranium. Therefore, there is no requirement for the libraries to cover variations in enrichment. Neither is there a need to address variations in moderator density. (Typically, ORIGEN-ARP libraries span a range of enrichment values and sometimes, such as for BWR cases, they also span a range of moderator densities.) For a Magnox reactor, the libraries that we developed consist of a set of 21 libraries covering a range of 0 to 9000 MWd/t in 20 steps of 450 MWd/t. (The first library contains the fresh-fuel cross sections and is followed by 20 burnup-dependent libraries.) A power level of 1.875 MW/t was used, and each of the 20 cycles was of 240 days duration. The cumulative and mid-cycle burnup values are shown in Table 2.

With a burnup of 9000 MWd/t, a power level of 1.875 MW/t seems low. (It implies a burnup time of 4800 days.) However, our original intention was for a burnup of 4500 MWd/t, and a power level of 1.875 MW/t was consistent with this. The libraries were extended to 9000 MWd/t when the validation data were obtained, and this would perhaps have allowed for a higher power level. However, the maximum power level reported for the validation studies was slightly less than twice our value of 1.875 MW/t, and for some of these exposures it was around 1.5 MW/t.

Table 2. Magnox reactor burnup values

Library number	Cumulative burnup (MWd/t)	Mid-cycle burnup (MWd/t)
1	0	0
2	450	225
3	900	675
4	1350	1125
5	1800	1575
6	2250	2025
7	2700	2475
8	3150	2925
9	3600	3375
10	4050	3825
11	4500	4275
12	4950	4725
13	5400	5175
14	5850	5625
15	6300	6075
16	6750	6525
17	7200	6975
18	7650	7425
19	8100	7875
20	8550	8325
21	9000	8775

2.2 ADVANCED GAS-COOLED REACTOR (AGR)

The AGR is also a gas-cooled and graphite-moderated reactor. It was developed as a follow-on design to the Magnox reactor. In the latter, natural uranium fuel dictated the use of the very low absorption Magnox cladding material. Magnox, however, cannot tolerate the high coolant gas temperatures required in the more efficient advanced design, and stainless steel cladding was used instead. But this necessitated enriching the ^{235}U in the fuel. Furthermore, uranium metal would undergo phase changes at the higher temperatures, and this led to the use of uranium-oxide fuel. Because of the lower thermal conductivity of the oxide, the larger-diameter metal fuel rods were replaced by clusters of smaller-diameter UO_2 fuel pins so as to avoid overheating the fuel. Incidentally, this latter design change causes the fuel-enrichment requirement to increase further. With smaller-diameter fuel rods, there is less self-shielding of the ^{238}U resonances, with a consequently lower resonance escape probability. For more information on AGR design see, for instance, Directory of Nuclear Reactors,[3] Kershaw and Durston,[6] Wilson,[7] or World Nuclear Industry Handbook.[8]

The AGR contains a large graphite moderating structure (see Fig. 2). Within this structure are cooling channels containing clusters of 36 fuel pins each. The coolant material is CO_2 gas. The fuel pins are composed of UO_2 encased in stainless steel cladding, and the fuel pellets contain a central axial hole to allow the accumulation of fission gases. The 36 fuel pins that are in

each cooling channel are arranged in three rings of 6, 12, and 18 pins in such a way that there is effectively a triangular pitch to the arrangement. A stainless steel tie bar is located at the center of the cluster. A graphite sleeve surrounds the coolant channel. This sleeve is surrounded by an annular cooling channel, which is, in turn, surrounded by more graphite that is cylindrical in shape. (Some older units have a double graphite sleeve that includes another annular cooling channel.) This cylindrical structure is repeated in a rectangular geometry. The spaces that are on the diagonals between the fuel bundle cylinders also contain graphite in the form of square blocks. There are channels between the graphite blocks and cylinders that contain CO_2. At the points where a cylinder abuts one of these blocks or another cylinder, there are keys, or cogs, that allow the graphite components to lock, thus giving stability to the overall structure. One design, for instance, has 332 fuel channels in the core in an arrangement that forms a 16-sided prism. At the periphery, this structure is anchored to the containment vessel.

In using SAS2H to analyze the AGR, the fuel-pin cell was composed of the UO_2 fuel surrounded by stainless steel clad, which was in turn surrounded by CO_2, considered to be moderator. (However, the CO_2 coolant has little moderating effect.). The fuel material was assumed to fill the entire volume within the clad. Thus, no internal hole was specified and the effective fuel density was adjusted accordingly. An effective pitch was calculated for the fuel pins that are on a triangular mesh. This pitch determines the effective fuel-pin cell size. For the larger (Path B) cell, we specify the central tie bar, surrounded by an annulus of CO_2. This is surrounded by homogenized fuel-pin-cell material that accounts for 36 fuel pin cells (from Path A) and another annulus of CO_2 immediately to the inside of the graphite sleeve. There follows the graphite sleeve; the annular cooling channel; and, finally, the surrounding graphite plus CO_2. The model therefore consists of a fuel bundle (36 fuel pins), the associated coolant, and the surrounding structure of graphite and CO_2. Table 3 lists operating parameter values for a typical AGR.

As mentioned, the 36 fuel pins are in three concentric rings. Table 3 lists the radii of these rings. A fuel-pin cell consists of the fuel pin and some surrounding coolant, and the size of the average cell follows from the average pitch. Thus, the total amount of coolant associated with the 36 fuel pins determines the area of the homogenized fuel-pin-cell material in Path B. When one calculates this latter area, there remains an annulus of coolant between it and the central tie bar as well as another annulus of coolant immediately inside the graphite sleeve. Conceptually we associate each fuel-pin ring with a zone of coolant, and the radii of the edges of these three zones (four values) are given in Table 3. These four values are such that the homogenized fuel-pin-cell material has the correct area and the three fuel-pin rings sit centrally in their respective radial zones.

The AGR fuel consists of UO_2 with various enrichment values. With enriched uranium fuel, there will invariably be some ^{234}U and ^{236}U present. In specifying enriched uranium fuel for U.S. reactors, there are prescriptions for the amounts of ^{234}U and ^{236}U that should be present. However, these prescriptions depend on the characteristics of the U.S. fuel cycle and it was felt that they might not apply to fuel enriched elsewhere. Therefore, only ^{235}U and ^{238}U were specified in defining the AGR fuel composition. Although this might have a small effect on the prediction of ^{234}U and ^{236}U concentrations during burnup, it would not affect the cross-section libraries being generated.

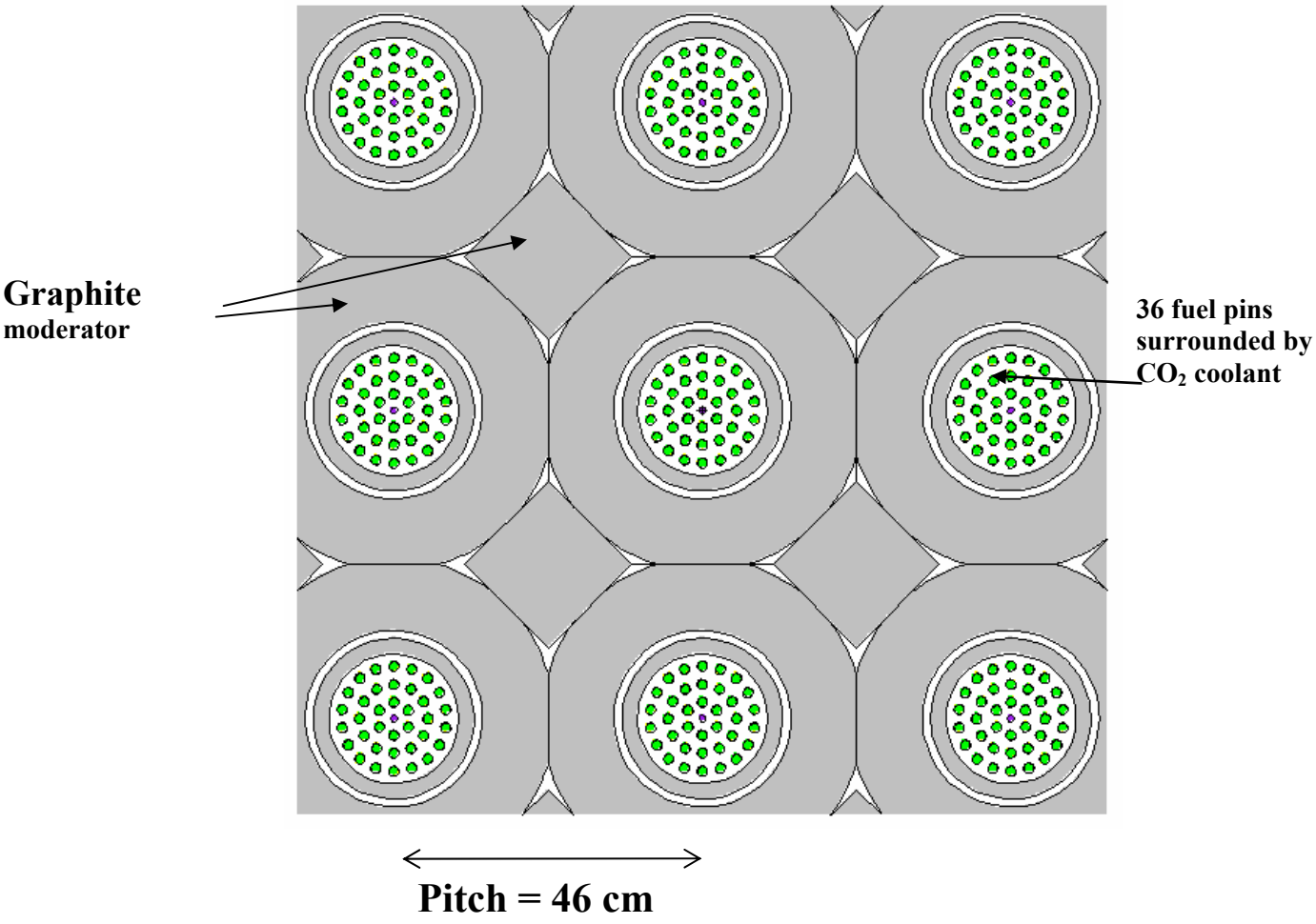

Fig. 2. Typical Advanced Gas-Cooled Reactor lattice. Each fuel bundle consists of 36 fuel pins in CO_2 coolant surrounded by graphite moderator.

Table 3. Advanced Gas-Cooled Reactor design parameters

Fuel density	10.76 g/cc
Effective fuel density	8.7 g/cc
Fuel radius (in/out)	0.3175/ 0.7258 cm
Fuel material	UO_2 (2.7 – 3.4% ^{235}U)
Fuel temperature	900 K
Clad outer radius	0.7655 cm
Clad density	7.98 g/cc
Clad material	53% Fe, 20% Cr, 25% Ni, 1% Nb, 0.08% C
Clad temperature	805 K
Fuel pin average pitch	2.886 cm
Tie bar density	8.02 g/cc
Tie bar material	31% Fe, 45% Ni, 17.5 % Cr, 3.8% Mo, 1.3% Al, 1.3% Ti, 0.08 % C
Tie bar radius	0.5061 cm
Pin ring radii	2.46/5.08/7.8 cm
Pin-zone radii in coolant	1.1557, 3.7719 6.4389, 9.1567 cm
Coolant channel radius	9.607 cm
Sleeve outer radius	11.91 cm
Coolant annulus radius	13.5 cm
Coolant pressure	30 – 40 bar
Coolant temperature	737 K
Bulk moderator effective radius	25.953 cm
Moderator material	Graphite
Moderator density	1.65 g/cc
Bulk moderator temperature	703 K
Assembly pitch	46 cm
Power rating	13 MW(th)
Mean fuel burnup	27,000 MW(th)d/t

We have no comparison tests with which to judge the accuracy of the ORIGEN-ARP libraries for the AGR case. Later in this report we describe validation checks performed on other ORIGEN-ARP libraries; however, when this work was being conducted, no experimental data were available to us on AGRs. Unfortunately, we are also not as confident of our ability to simulate AGR burnup conditions with SAS2H as we are for the case of the other three reactors discussed here. The reason for our concern is that the fuel-pin cell contains essentially no moderating material. The moderator in the AGR is the graphite that is in the larger cell. Therefore, the pin-cell cross sections are not likely to be accurate. We include the AGR libraries here. It is our hope to be able to perform validation studies at a later time.

2.2.1 AGR Library Details

Table 3 lists the mean fuel burnup for an AGR as 27,000 MWd/t. The ORIGEN-ARP libraries were developed to cover the burnup range up to 30,000 MWd/t.

ARGs operate with a variety of enrichment values. Table 2 indicates the enrichment range as most typically being 2.7 to 3.4%. The libraries developed in this work span an enrichment range of 1.5 to 4.5% in 0.5% increments. Specifically, the AGR ORIGEN-ARP libraries consist of seven sets of items. These seven sets are for enrichment levels of 1.5, 2.0, 2.5, 3.0, 3.5, 4.0,

and 4.5%. Each set contains 21 burnup-dependent libraries beginning with a fresh-fuel library followed by 20 libraries for burnup values that increase by 1500 MWd/t up to 30,000 MWd/t. A power level of 12.5 MW/t was used, and each of the 20 burnup steps is of 120 days duration. The burnup values are shown in Table 4.

Table 4. Advanced Gas-Cooled Reactor burnup values

Library number	Cumulative burnup (MWd/t)	Mid-cycle burnup (MWd/t)
1	0	0
2	1,500	750
3	3,000	2,250
4	4,500	3,750
5	6,000	5,250
6	7,500	6,750
7	9,000	8,250
8	10,500	9,750
9	12,000	11,250
10	13,500	12,750
11	15,000	14,250
12	16,500	15,750
13	18,000	17,250
14	19,500	18,750
15	21,000	20,250
16	22,500	21,750
17	24,000	23,250
18	25,500	24,750
19	27,000	26,250
20	28,500	27,750
21	30,000	29,250

2.3 THE VVER-440 REACTOR

VVER type reactors are pressurized-water reactors of Russian design. In a VVER, the core, the assemblies, and the lattice elements within an assembly are hexagonal in shape. A VVER-440 assembly consists of 126 fuel pins and a central lattice element containing an instrument hole. There are 349 assemblies in the VVER-440 core. The distance across an assembly between flat surfaces is sometimes called the "wrench" distance and, if there is no water film between assemblies, this is also the assembly pitch. Similarly for the lattice elements within an assembly, this is the lattice-element pitch. Half of this distance is the height of an equilateral triangle, of which there are six within the hexagonal element. Thus, the area of each of the 127 lattice elements in the assembly follows from the lattice-element pitch.

We define a fuel pin cell as the fuel, surrounded by clad, and surrounded by the water moderator in the lattice element. The elements form a triangular grid. The fuel in a VVER-440 is UO_2. The fuel pellets have a central axial hole, and there is a gap between pellet and clad. In the pin-cell model, the fuel was smeared over all the volume within the clad and the density was

adjusted to reflect this. The larger Path B model contained, in order, the water in the central instrument hole, the central instrument-hole tube, the remaining water outside the tube in the central lattice element, the homogenized fuel material equal to 126 fuel-pin cells, and an outer film of water to account for the water between the last ring of hexagonal cells and the flat outer boundary of the assembly. A VVER-440 assembly is shown in Fig. 3, and the structure of a single lattice element can be seen in Fig. 4. Table 5 lists typical design parameters for a VVER-440.

Table 5. Typical design parameters for a VVER-440

Fuel density	10.1 g/cc
Effective fuel density	9.224 g/cc
Fuel material	UO_2
Enrichment	1.6 - 4.4 % ^{235}U
Fuel inner radius	0.075 cm
Fuel outer radius	0.3783 cm
Clad inner radius	0.388 cm
Clad outer radius	0.455 cm
Clad and central tube material	Zr (1% Nb)
Clad density	6.45 g/cc
Central tube radius (in/out)	0.44/0.515
Lattice element pitch	1.22 cm
Moderator density	0.73 g/cc
Specific power rating	32 MW(th)d/t
Fuel temperature	980 K
Clad temperature	630 K
Moderator temperature	575 K
Mean fuel burnup	28,600 MW(th)/t

Various estimates of density were encountered for the VVER-440 UO_2 fuel. From previous work involving a specific VVER-1000 case, we had a value of 10.13 g/cc for that particular reactor's fuel density. Based on this, we used a value of 10.1 g/cc for the VVER-440. As was the case for the AGR, the fuel was defined as consisting of ^{235}U and ^{238}U only.

2.3.1 VVER-440 Library Details

The mean fuel burnup for a VVER-440 assembly is shown as 28,600 MWd/t (Table 5). However, this value probably represents data on past burnup experiences. The current trend is towards significantly higher burnup for light-water-reactor fuel. Therefore, with this in mind, and in line with our experiences involving PWR fuel, we chose to produce burnup-dependent libraries up to 60,000 MWd/t. A range of enrichment values from 1.5 to 5.0% was chosen.

The VVER-440 ORIGEN-ARP libraries consist of a group of libraries for each of seven enrichment values: 1.5, 2.0, 2.5, 3.0, 3.5, 4.0, and 5.0%. For each of these enrichment values, there are 21 burnup-dependent libraries. Library number 1 is for fresh fuel and is followed by 20 libraries at burnup increments of 3,000 MWd/t up to a total of 60,000 MWd/t. In generating these libraries with SAS2H, a power level of 40 MW/t was used with 20 cycles each of 75 days duration. Table 6 shows the burnup values for the libraries.

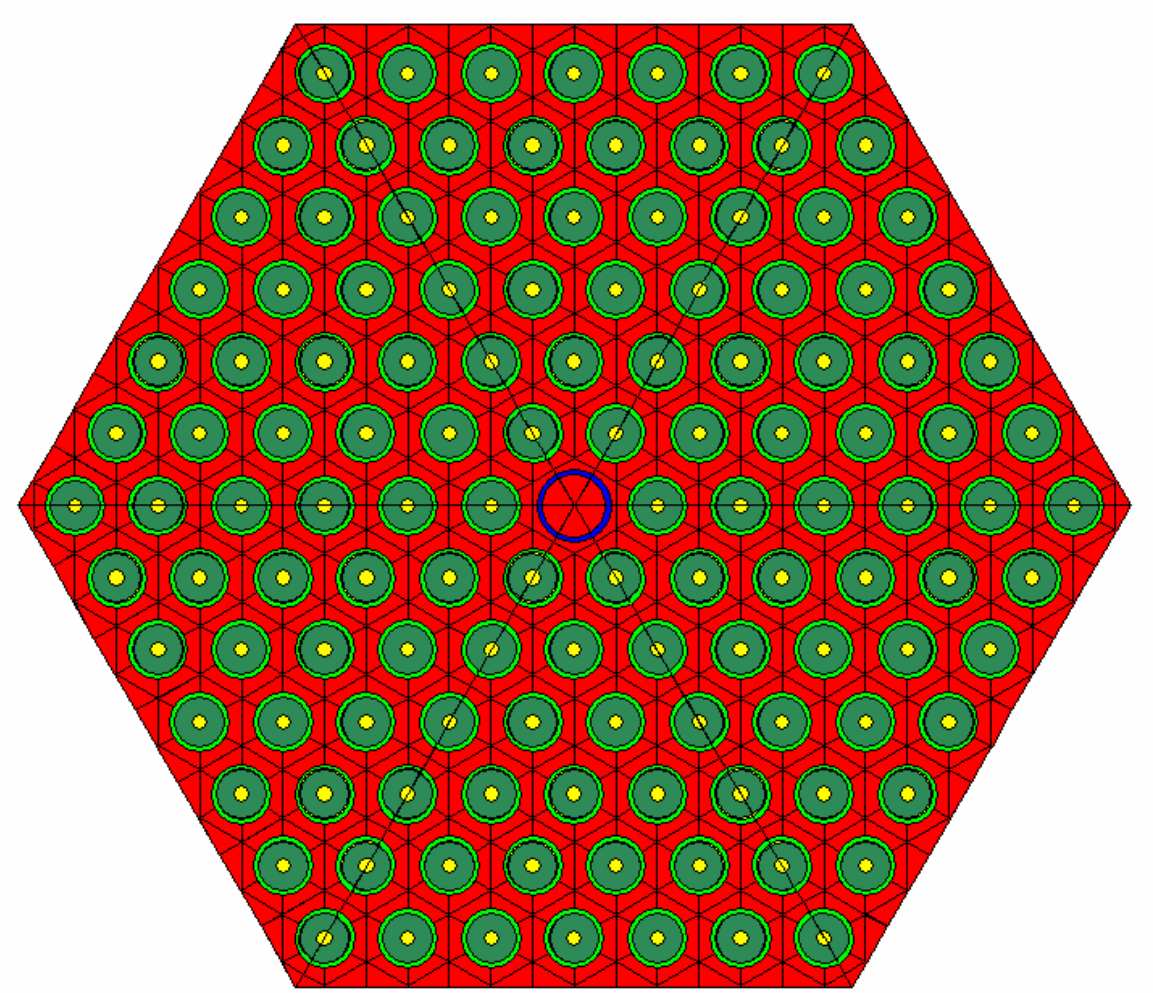

Fig. 3. VVER-440 assembly showing a central water hole and 126 fuel pins.

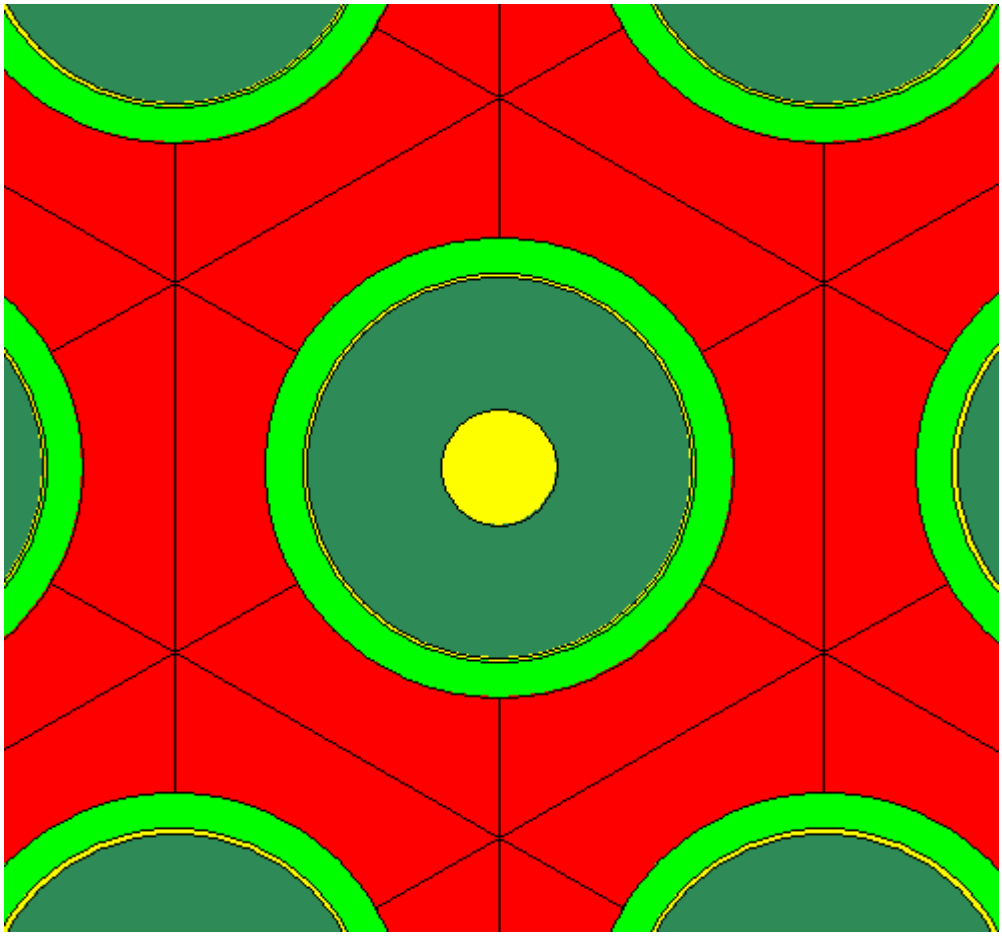

Fig. 4. Typical VVER lattice element. The element is hexagonal in shape. Within the lattice element, there are a fuel rod (with a central hole), a gap, clad, and surrounding moderator water.

Table 6. VVER-440 burnup values

Library number	Cumulative burnup (MWd/t)	Mid-cycle burnup (MWd/t)
1	0	0
2	3,000	1,500
3	6,000	4,500
4	9,000	7,500
5	12,000	10,500
6	15,000	13,500
7	18,000	16,500
8	21,000	19,500
9	24,000	22,500
10	27,000	25,500
11	30,000	28,500
12	33,000	31,500
13	36,000	34,500
14	39,000	37,500
15	42,000	40,500
16	45,000	43,500
17	48,000	46,500
18	51,000	49,500
19	54,000	52,500
20	57,000	55,500
21	60,000	58,500

2.4 THE VVER-1000 REACTOR

The VVER-1000 reactor is a larger version of the VVER-400 with different pin-cell dimensions and the addition of burnable poison rods (see Fig. 5). These larger assemblies contain 331 lattice elements. There are 163 fuel assemblies in a VVER-1000 core. In a typical assembly there is a central instrument tube, 18 control or burnable-poison elements, and 312 fuel elements. The arrangement of the control (burnable-poison) elements is such that they naturally fall into two rings, an inner ring of 6 and an outer ring of 12. A fuel pin cell consists of UO_2 fuel surrounded by clad and the water moderator within the fuel cell lattice element. The control elements are composed of burnable-absorber material containing boron. The detailed composition of the material is given in Table 7. The burnable-absorber material is surrounded by clad, a guide-thimble (also clad material), and water.

We describe a specific case of a VVER-1000 in the validation section below,[9] and for this case, there are burnable absorber rods in the assembly for the duration of just the first cycle (approximately 15,000 MWd/t). This was therefore assumed to be a reasonable arrangement for a

VVER-1000 and the libraries were developed with burnable absorber in place for the first 15,000 MWd/t of burnup.

There are 18 burnable absorber rods in a VVER-1000 assembly. The SAS2H models simulated one burnable absorber rod and the associated section of the assembly that surrounds it (one-eighteenth of an assembly). Therefore, the larger cell was constructed to reflect, in order, one absorber rod, the absorber-associated cladding, the water in the absorber pin cell, one-eighteenth of the fuel inventory, and one-eighteenth of the central water hole. Beyond 15,000 MWD/t burnup, the absorber material was replaced with water. Design parameters for a VVER-1000 are summarized in Table 7.

Table 7. Design parameters for a VVER-1000

Fuel density	10.13 g/cc
Fuel material	UO_2
Enrichment	1.6 – 4.4 % ^{235}U
Radius of fuel (in/out)	0.115/0.3775 cm
Fuel Clad radius (in/out)	0.386/0.4582
Clad and central tube material	Zr (1% Nb, 0.03% Hf)
Clad density	6.45 g/cc
Central tube radius (in/out)	0.45/0.515 cm
Burnable absorber material	Al:91.75%; B:1.25%; Cr:3%; Ni:2%; Zr:2%
Burnable absorber radius	0.35 cm
Burnable absorber density	2.9 g/cc
BA cladding and tube material	Zr (1% Nb, 0.03% Hf)
BA cladding and tube density	6.45 g/cc
BA clad radius (in/out)	0.35/0.41 cm
BA tube radius (in/out)	0.55/0.63 cm
Lattice element pitch	1.275 cm
Moderator density	0.69 g/cc
Typical power level	45 MW(th)/t
Fuel temperature	966 K
Clad temperature	630 K
Moderator temperature	578 K
Typical burnup	45,000 MW(th)d/t

We note here an important point regarding the pin-cell model: The fuel rods contain a central cylindrical fission-gas plenum, thus giving rise to an inner and outer radius being quoted for the fuel. There is also a gap between fuel and clad. It is often acceptable to smear the fuel over the total volume inside the clad and thus to ignore the gap (and to adjust the fuel density accordingly). One could do this here and ignore both the plenum and the gap. However, in a validation test to be mentioned later, we seemed to improve the performance of the simulation if we retained the correct density for the fuel and combined the plenum and the gap, thus considering the combination to be effectively the gap. We do not know how general this result might be, and we have not tested it for the other reactor designs where plenum and gap volumes might be important.

2.4.1 VVER-1000 Library Details

The burnup and enrichment ranges chosen for the VVER-1000 libraries were the same as those for the VVER-440 libraries. Similarly, the incremental values for burnup and enrichment were the same. Thus, the arrangement of the libraries is exactly the same as in the VVER-440 case. There are seven enrichment values: 1.5, 2.0, 2.5, 3.0, 3.5, 4.0, and 5.0%. There are 21 burnup-dependent libraries for each enrichment value, with the total burnup being 60,000 MWd/t.

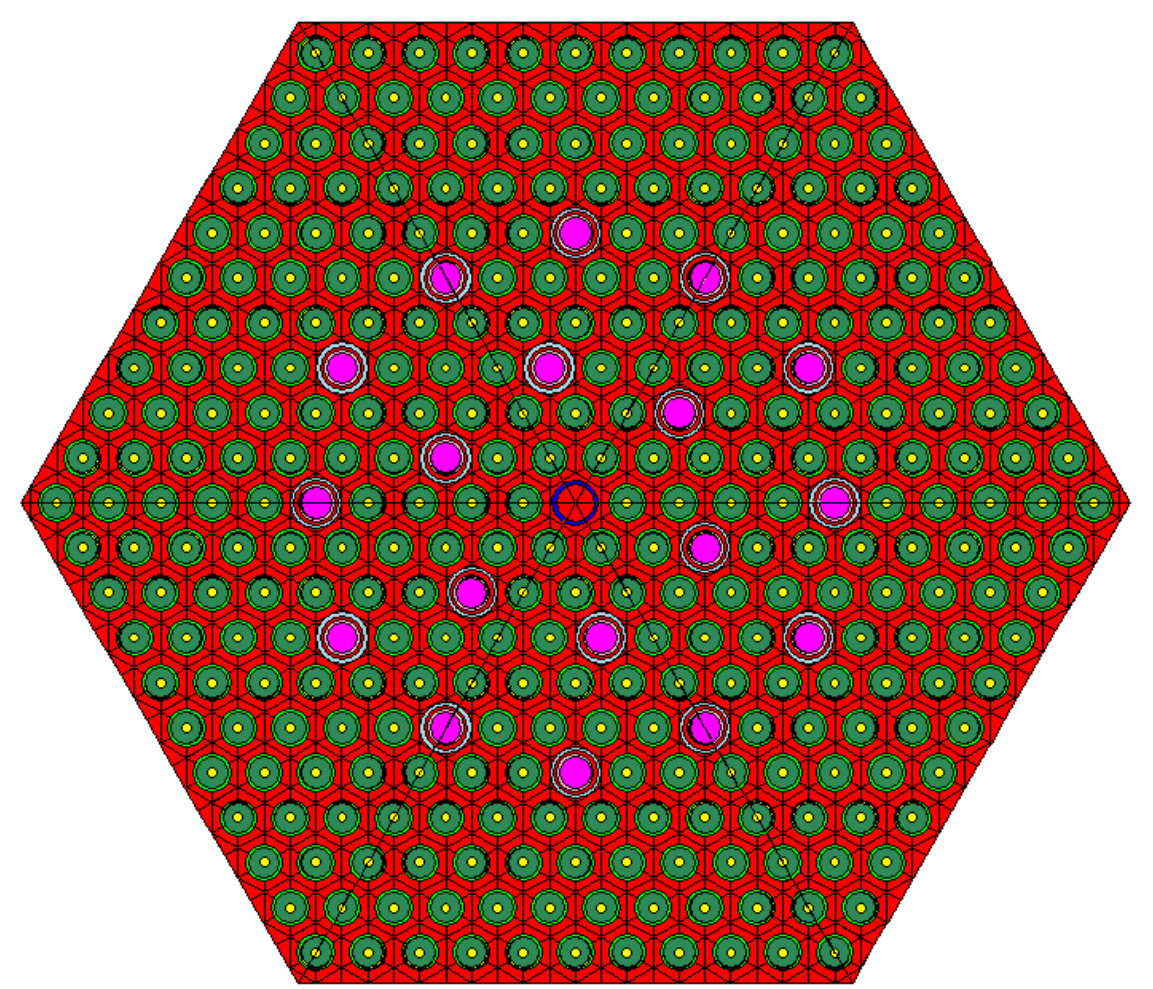

Fig. 5. VVER-1000 assembly containing a central water/instrument hole, 312 fuel pins, and 18 control-rod, or burnable-poison, elements. The control-rod elements are arranged in 2 zones, one of 6 and one of 12 elements.

3. LIBRARY VALIDATION

Where possible, validation checks were carried out for the models described above. The most definitive type of validation check for a reactor burnup code is a comparison of nuclide concentration predictions against nuclide concentration measurements on spent fuel. A small number of spent-fuel assay results were identified, and we present comparisons with these measurements below. These validation checks are not exhaustive. However, they do indicate the appropriateness of the cross-section libraries for their intended purpose, which is to describe the composition and decay properties of spent fuel for these four reactor types under typical conditions.

3.1 MAGNOX RESULTS

Experimental measurements on samples exposed in the Hunterston-A Magnox reactor have been reported by Nair.[10] We report here on comparisons between simulations and measurements of nuclide concentrations for four of these samples. The burnup for the four samples spanned a range of values. Table 8 lists burnup values for the samples as estimated by the reactor operator, by mass spectrometry, and as estimated from the ^{148}Nd concentrations. For samples B, C, and D, the burnup values are quite high for what would appear to be normal Magnox operating conditions. Therefore, the results for Sample A should be considered important as to their relevance to normal operation with Samples B, C, and D supplying further validation for extended operating conditions.

In the comparisons being discussed, burnup simulations for Samples A, B, and C were carried out using ORIGEN-ARP and the Magnox libraries. However, for Sample D the estimated burnup was beyond the range of the libraries and, therefore, for this one sample, the simulation was performed with the SAS2H model used to generate the libraries.

Table 8. Sample burnup estimated for the Hunterston-2 Magnox reactor

Sample	Operator (MWd/t)	Mass spec. (MWd/t)	^{148}Nd (MWd/t)
A	3900 ± 400	3200 ± 100	3323
B	7000 ± 700	6500 ± 200	6714
C	7000 ± 700	7000 ± 200	7000
D	9400 ± 900	9000 ± 300	9154

Calculated-to-experimental ratios for the four samples A, B, C, and D are shown in Figs. 6, 7, 8, and 9 for some uranium, plutonium, and neodymium isotopes. The nuclides chosen for display have either very long or infinite lifetimes because measurement dates were not quoted in the report by Nair. The comparisons with the measurements published by Nair show that our models are well representative of Magnox reactor burnup conditions.

A second, and more indirect, check on our modeling results is also worth mentioning (it involves comparisons with other calculations): In a discussion of North Korean nuclear issues, Albright and O'Neill[11] refer to estimates of plutonium production in Magnox reactors.[5] The burnup values were quite small. On page 116 of their book, Albright and O'Neill discuss the likely inventory of plutonium that might have been extracted from a graphite-moderated reactor.

Two calculations are mentioned by Albright and O'Neill. These refer to the case of the full core (50 metric tons) having been unloaded and to the case of just half of the core being unloaded. If the full core were unloaded, the average burnup is assumed to be about 200 MWd/t and the 50 t is estimated to yield about 9.5 kg of total plutonium, with about 1.5% being ^{240}Pu. The SAS2H model used to develop the Magnox ORIGEN-ARP libraries predicts 8.84 kg of plutonium with 1.6% of this being ^{240}Pu.

Alternatively, if half the core were unloaded, Albright and O'Neill assume that the average burnup would be 300 MWd/t and they quote an estimate of 7 kg of plutonium containing 2.3 to 2.4% ^{240}Pu. Using our models for the half-core situation, we estimate 6.9 kg of plutonium containing 2.4% ^{240}Pu.

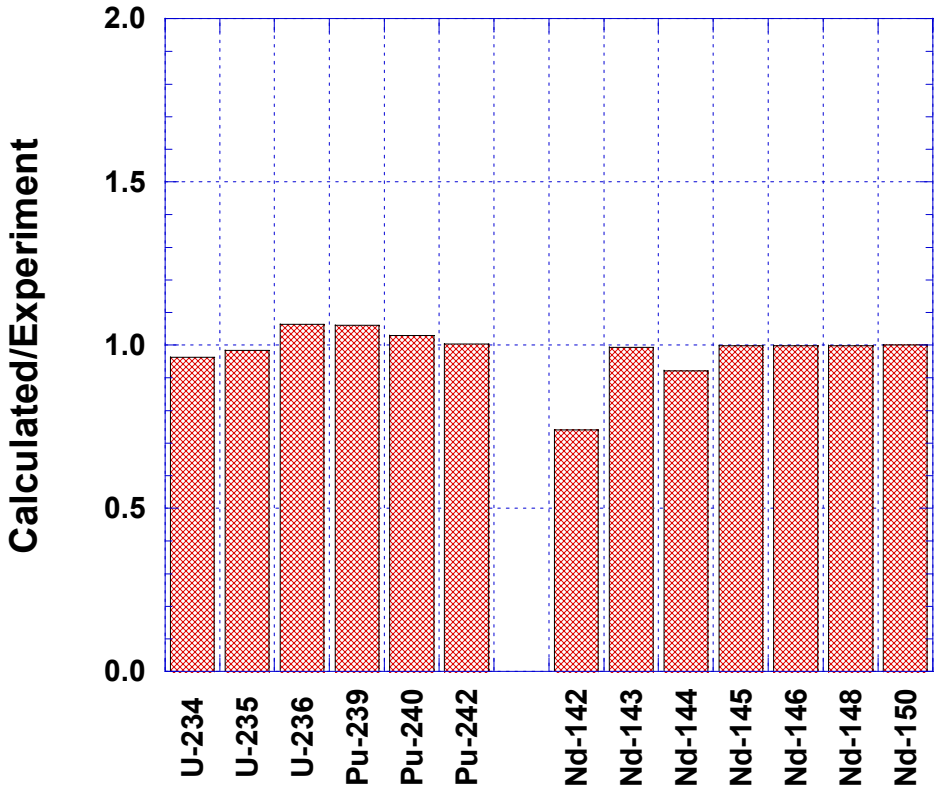

Fig. 6. Calculated-to-experimental ratios for Sample A exposed in the Hunterston-2 Magnox reactor. The sample burnup was 3323 MWd/t. The calculated values were obtained using ORIGEN-ARP and the Magnox libraries.

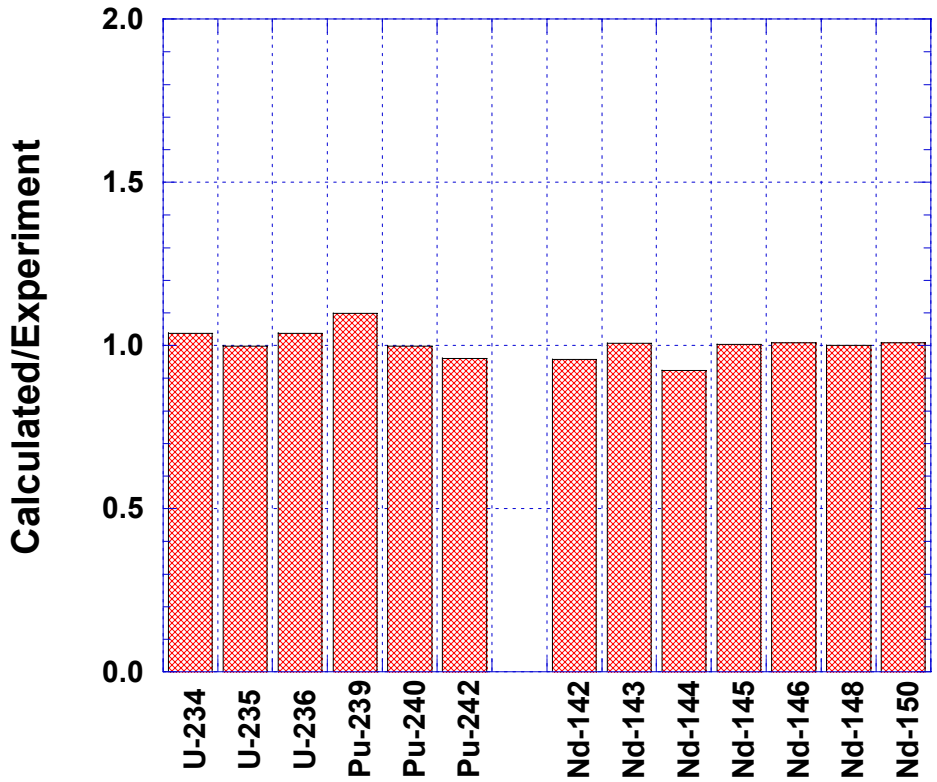

Fig. 7. Calculated-to-experimental ratios for Sample B exposed in the Hunterston-2 Magnox reactor. The sample burnup was 6714 MWd/t. The calculated values were obtained using ORIGEN-ARP and the Magnox libraries.

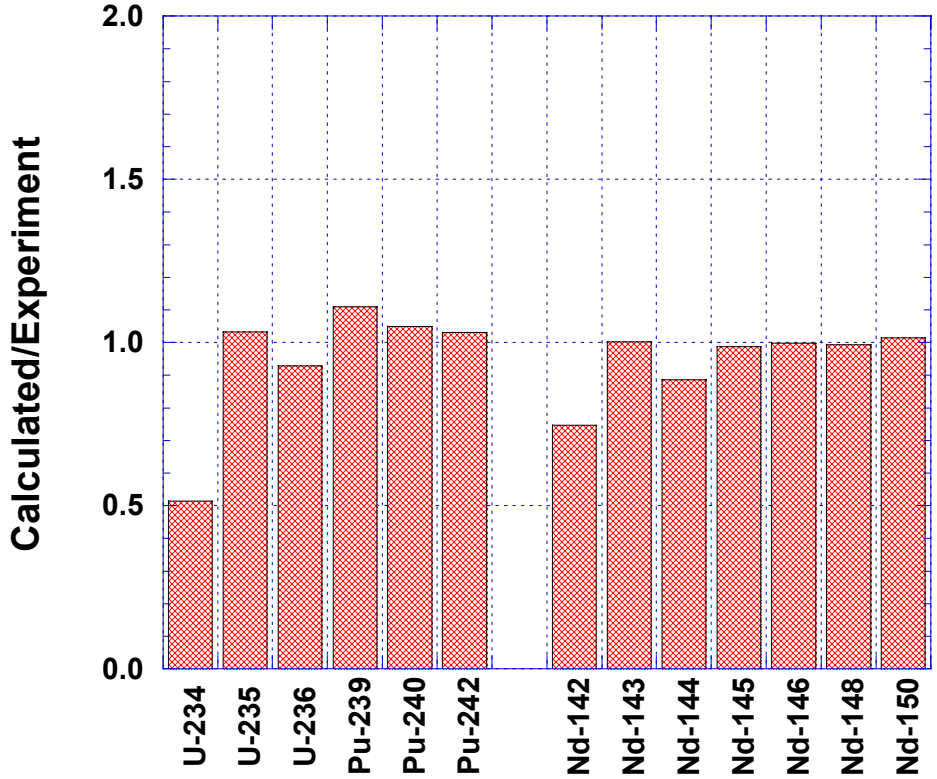

Fig. 8. Calculated-to-experimental ratios for Sample C exposed in the Hunterston-2 Magnox reactor. The sample burnup was 7000 MWd/t. The calculated values were obtained using ORIGEN-ARP and the Magnox libraries.

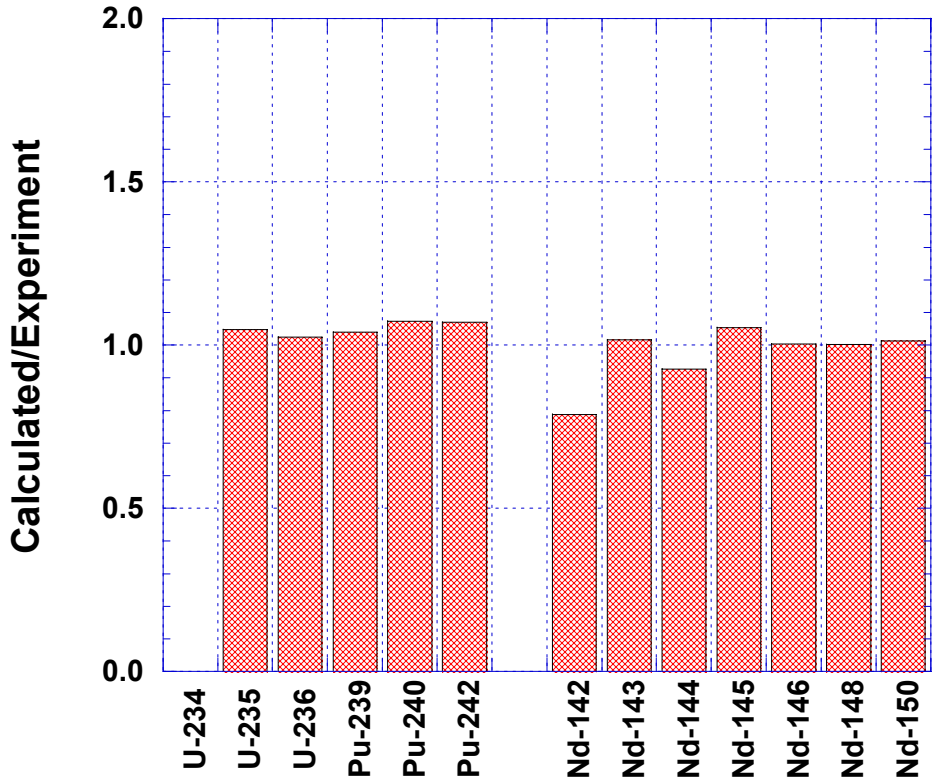

Fig. 9. Calculated-to-experimental ratios for Sample D exposed in the Hunterston-2 Magnox reactor. The sample burnup was 9154 MWd/t. Because the burnup is outside the range of the ORIGEN-ARP libraries, these calculations were performed using the SAS2H model for a Magnox reactor.

3.2 VVER-440 RESULTS

As part of a cooperative effort with staff at the State Scientific and Technical Center on Nuclear and Radiation Safety in Kiev, Ukraine, ORNL had access to some VVER-440 spent-fuel assay data. These data were preliminary and somewhat incomplete. Only total burnup values were available, with no information on power levels or cycle durations. Furthermore, there is no information on the location of the analyzed samples in their respective assemblies. With these shortcomings in mind, we carried out simulations for a sample with an initial enrichment of 3.6% and for which concentrations were reported over a good range of burnup values. Figures 10, 11, 12, and 13 show concentration values versus burnup for ^{235}U, ^{238}U, ^{239}Pu, and ^{244}Cm, respectively. The plots show the reported measured values and the values determined from an ORIGEN-ARP simulation (where the ORIGEN-ARP libraries were developed using the SAS2H model).

The comparisons for the VVER-440 show reasonable agreement between simulation and measurement. We do, however, see overprediction of ^{239}Pu. Perhaps a better knowledge of the power-history details might improve the simulations in this regard.

3.3 VVER-1000 RESULTS

As part of the Fissile Materials Disposition Program (FMDP) at ORNL, a number of VVER-1000 irradiation cases involving low-enriched uranium (LEU) fuel were analyzed.[9] The simulations in these analyses employed the collision-probability code HELIOS.[12] From among the cases studied, the one involving Sample 15 from the Balakova-2 reactor was chosen for verification purposes here. The ^{235}U enrichment for Sample 15 was 4.4 %. It was located well within the reactor assembly and therefore not subject to boundary effects.

In Fig. 14 we show comparisons between SAS2H calculations and experimental measurements for the case of the Sample 15 irradiation. Comparisons are shown for uranium, plutonium, americium, curium, and neodymium isotopes. Figure 15 shows the equivalent comparisons where ORIGEN-ARP (with SAS2H-generated libraries) was used for the burnup simulation. The values being compared in Figs. 14 and 15 are at discharge. Presumably the measurements were carried out at some time following discharge and, for the short-lived species, concentration values were estimated for the time of discharge. As one might expect, the comparisons shown in Figs. 14 and 15 are essentially the same because the SAS2H model for the Fig. 14 results is the one that generated the libraries used for the Fig. 15 results.

Using the combined concentrations of ^{145}Nd and ^{146}Nd, the burnup for this sample was determined to be 44,900 MWd/t using the HELIOS code[9] and it was determined to be 44,400 MWd/t in the investigations reported here. This amounts to about a 1% difference, which is well within the uncertainty that one normally associates with sample-burnup estimates.

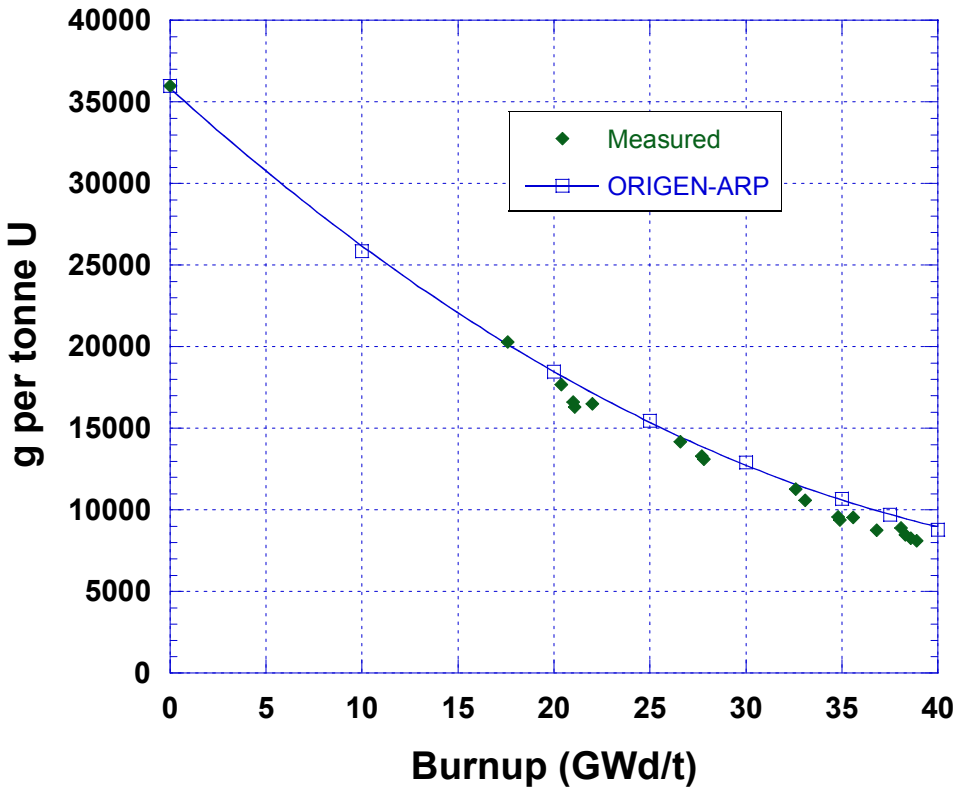

Fig. 10. Uranium-235 concentration versus burnup for a Ukrainian VVER-440. The figure shows measured values and values predicted with ORIGEN-ARP.

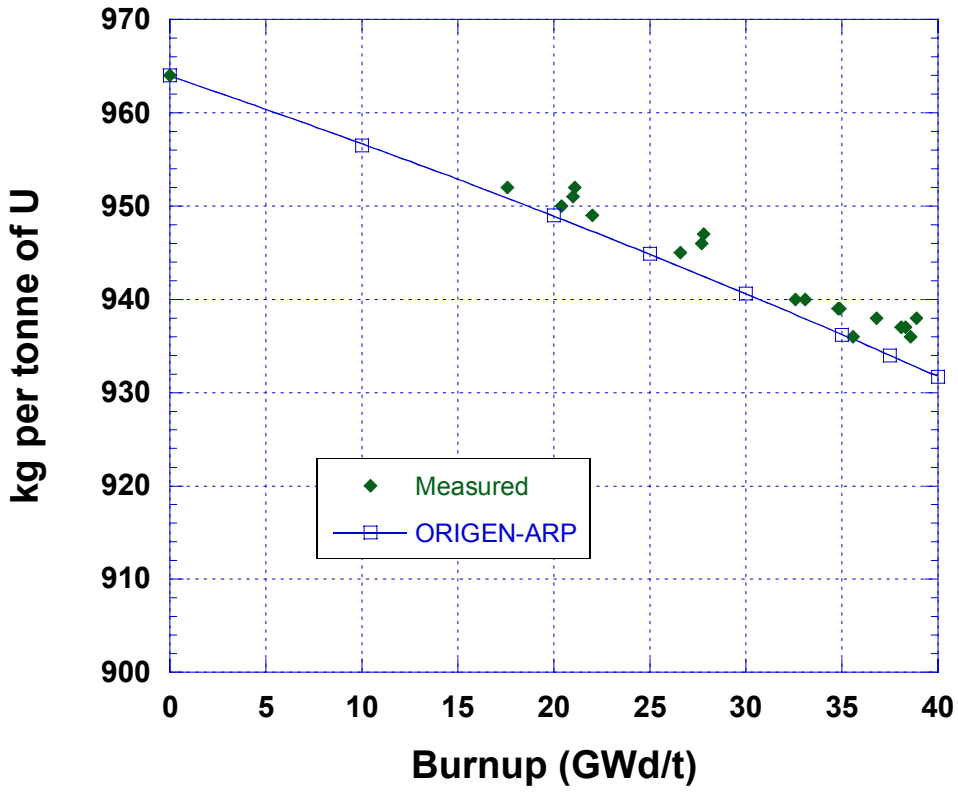

Fig. 11. Uranium-238 concentration versus burnup for a Ukrainian VVER-440. The figure shows measured values and values predicted with ORIGEN-ARP.

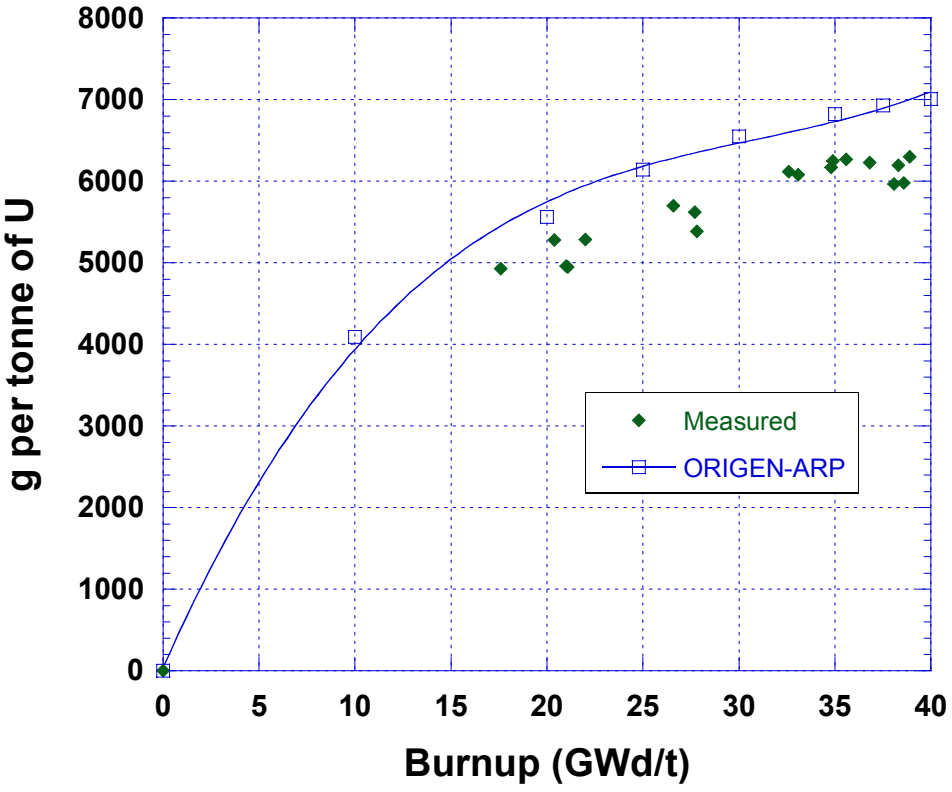

Fig. 12. Plutonium-239 concentration versus burnup for a Ukrainian VVER-440. The figure shows measured values and values predicted with ORIGEN-ARP.

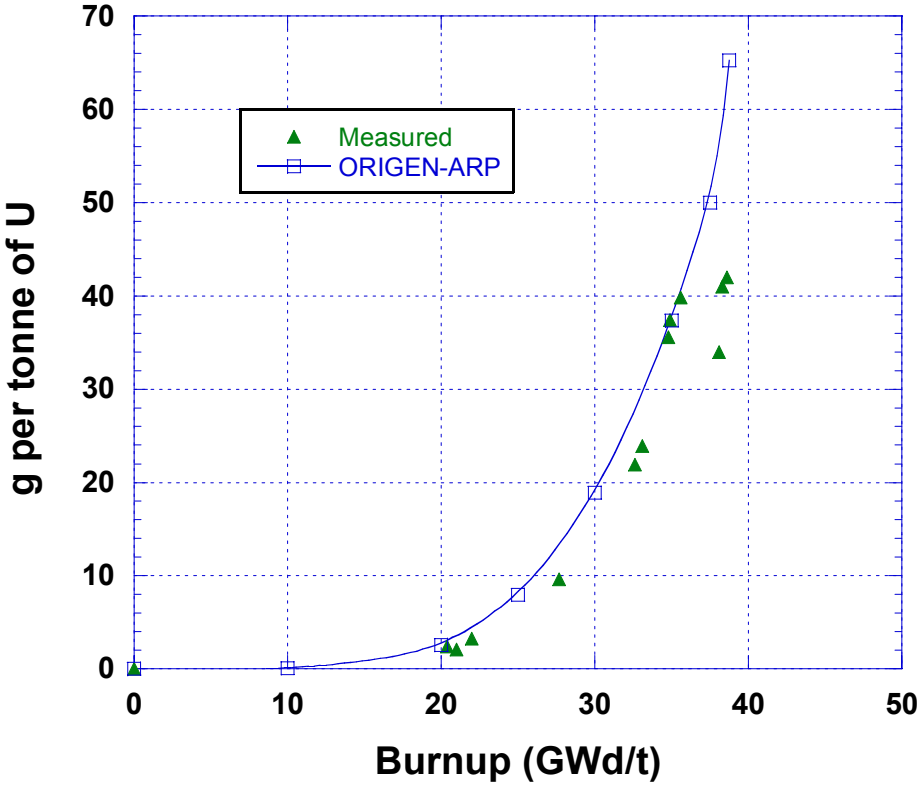

Fig. 13. Curium-244 concentration versus burnup for a Ukrainian VVER-440. The figure shows measured values and values predicted with ORIGEN-ARP.

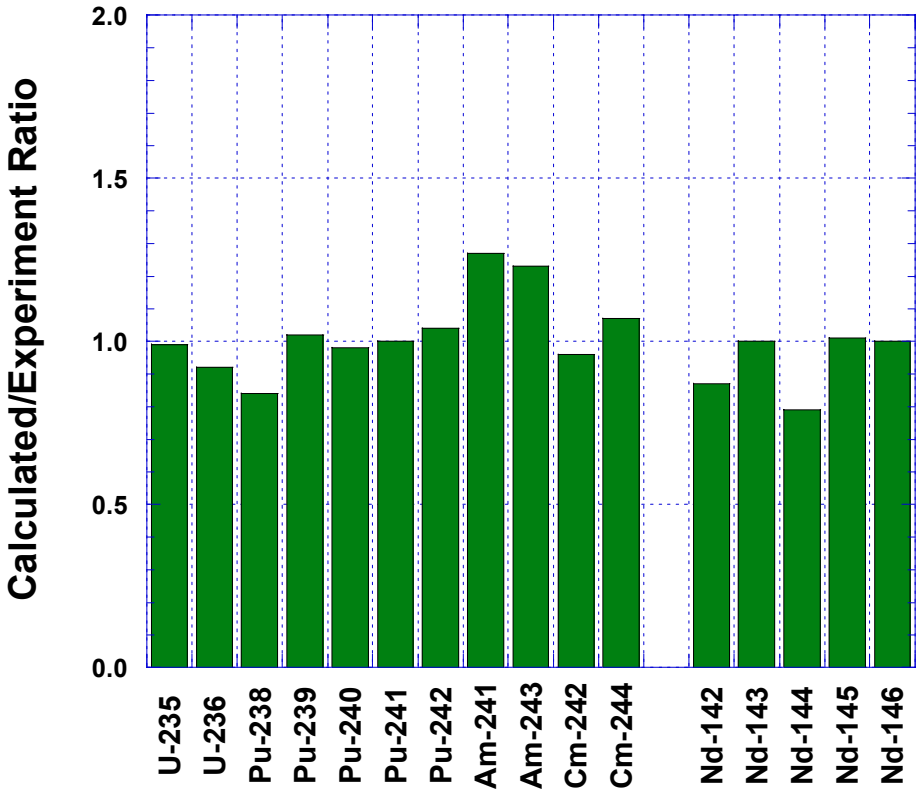

Fig. 14. Comparisons between calculations with SAS2H and measurements for a UO_2 sample exposed in the Balakova-2 VVER-1000 (burnup = 44,400 MWd/t). These results were obtained with the SAS2H model that was used to generate the VVER-1000 ORIGEN-ARP libraries.

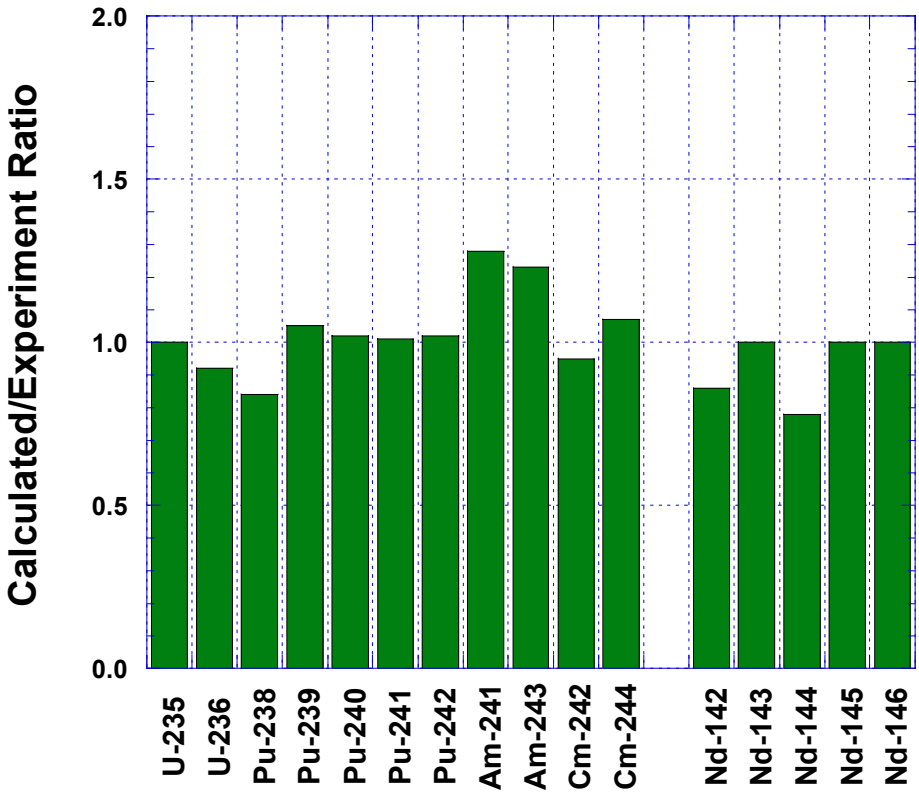

Fig. 15. Comparisons between ORIGEN-ARP calculations and experimental values for a UO_2 sample exposed in the Balakova-2 VVER-1000 (burnup = 44,400 MWd/t).

4. SUMMARY

This report describes the generation of ORIGEN-ARP cross-section libraries for four reactors: the Magnox reactor, which is gas cooled and uses natural uranium metal fuel; the Advanced Gas-Cooled Reactor, which uses enriched uranium in oxide form; the VVER-440, which is a Russian pressurized-water reactor (PWR); and the VVER-1000, a larger Russian PWR. These libraries were created using SAS2H models to simulate burnup in the reactors. The resulting burnup-dependent cross-section libraries are suitable for the ORIGEN-ARP burnup simulation system.

Reactor operating parameters were obtained from various published sources and the models and cross-section libraries are intended to apply to each of the four reactor designs in a generic sense (i.e., they represent a typical reactor of that type).

For the Magnox, VVER-440, and VVER-1000 systems, we were able to locate data giving experimentally measured concentrations for important nuclides in spent fuel. These were compared to values calculated using the models we developed. The comparisons were reasonably good and provide validation for the models. For the VVER systems, there is more favorable agreement between calculation and experiment for the VVER-1000 than there is for the VVER 440. But we emphasize that the VVER-440 irradiation conditions were not available in as much detail as was the case for the VVER-1000. No experimental data were located for the AGR case, however.

5. REFERENCES

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APPENDIX A

Examples of four SAS2H input streams are listed in this Appendix. They refer respectively to a Magnox reactor, an Advanced Gas-Cooled Reactor, a VVER-440, and a VVER-1000.

SAS2H Input for Magnox reactor.

```
=sas2h    parm=skipshipdata
 MAGNOX reactor sas2h input
44groupndf5 latticecell
' ------------------------------------------------------------------
'      Using Calder Hall as a model.
' ------------------------------------------------------------------
'    Natural uranium-metal fuel rods.
'
u      1 den=17.98  1 800  end
' ------------------------------------------------------------------
'    some extra nuclides needed from library
zr-94  1 0 1-20     800 end
tc-99  1 0 1-20     800 end
ru-106 1 0 1-20     800 end
rh-103 1 0 1-20     800 end
rh-105 1 0 1-20     800 end
xe-131 1 0 1-20     800 end
cs-133 1 0 1-20     800 end
cs-134 1 0 1-20     800 end
ce-144 1 0 1-20     800 end
pr-143 1 0 1-20     800 end
nd-143 1 0 1-20     800 end
nd-145 1 0 1-20     800 end
nd-147 1 0 1-20     800 end
pm-147 1 0 1-20     800 end
sm-149 1 0 1-20     800 end
sm-151 1 0 1-20     800 end
sm-152 1 0 1-20     800 end
eu-153 1 0 1-20     800 end
eu-154 1 0 1-20     800 end
eu-155 1 0 1-20     800 end
u-232  1 0 1-20     800 end
u-233  1 0 1-20     800 end
u-237  1 0 1-20     800 end
pu-236 1 0 1-20     800 end
pu-237 1 0 1-20     800 end
pu-243 1 0 1-20     800 end
pu-244 1 0 1-20     800 end
cm-245 1 0 1-20     800 end
cm-246 1 0 1-20     800 end
cm-247 1 0 1-20     800 end
cm-248 1 0 1-20     800 end
bk-249 1 0 1-20     800 end
cf-249 1 0 1-20     800 end
cf-250 1 0 1-20     800 end
cf-251 1 0 1-20     800 end
cf-252 1 0 1-20     800 end

' ------------------------------------------------------------------
'        clad is magnox.
'
arbm_magnox  1.65 3 0 0 0 12000 98.95 13000 1.0 4000 0.05
             2 1.0  700 end
```

```
'
' ----------------------------------------------------------------
'        moderator is co2 and graphite smeared together.
'
arbm-co2   0.0530 2 0 1 0 6012 1 8016 2 3 0.15206 650  end
'
c      3 den=1.628 0.84794 650 end
'
'     this is boron to account for impurities in the graphite.
'
arbm-bormod 1.628 1 1 0 0 5000 100 3 5.51e-7 650 end
' ----------------------------------------------------------------
'    mixtures for larger-unit-cell
'
arbm-cool  0.0578 2 0 1 0 6012 1 8016 2 4 1.00 650  end
'
end comp
' ----------------------------------------------------------------
'    fuel pin cell geometry
'
'    the pin cell consists of a fuel rod surrounded by clad (no gap),
'    and surrounded by moderator.
'
'    the coolant is smeared together with the graphite and the
'    mixture is treated as the moderating material.
'
squarepitch 20.32 2.922  1 3 4.08 2 end
'
' ----------------------------------------------------------------
'     assembly and cycle parameters
'
'     assembly contains 1 tonne of heavy metal.
'
'     assembly is 9 graphite blocks (144 fuel pins).
'     larger cell should be just 500-type material, but,
'     a 1 sq cm hole of coolant is added for SAS2H to work.
'
npin/assm=144 fuelngth=57.622 ncycles=21 nlib/cyc=1 lightel= 0
printlevel=4 inplevel=2 numzones=2 end
'
4 0.5642 500 138.568
'
'   we assume cycles of 120 days at 1.875 MW(th)d/t
'   total burnup is 4500 MWd/t.
'
 power=1.875    burn=1.0e-15 down= 0  end
 power=1.875    burn=120.0   down= 0  end
 power=1.875    burn=120.0   down= 0  end
 power=1.875    burn=120.0   down= 0  end
 power=1.875    burn=120.0   down= 0  end
 power=1.875    burn=120.0   down= 0  end
 power=1.875    burn=120.0   down= 0  end
 power=1.875    burn=120.0   down= 0  end
 power=1.875    burn=120.0   down= 0  end
 power=1.875    burn=120.0   down= 0  end
 power=1.875    burn=120.0   down= 0  end
 power=1.875    burn=120.0   down= 0  end
```

```
 power=1.875    burn=120.0   down= 0  end
 power=1.875    burn=120.0   down= 0  end
 power=1.875    burn=120.0   down= 0  end
 power=1.875    burn=120.0   down= 0  end
 power=1.875    burn=120.0   down= 0  end
 power=1.875    burn=120.0   down= 0  end
 power=1.875    burn=120.0   down= 0  end
 power=1.875    burn=120.0   down= 0  end
 power=1.875    burn=120.0   down= 0  end
' ------------------------------------------------------------
end
```

SAS2H Input for Advanced Gas-Cooled Reactor

```
=sas2h    parm=skipshipdata
 Advanced Gas-Cooled Reactor (AGR) sas2h input
44groupndf5 latticecell
' ----------------------------------------------------------------
'     UO2 fuel rods.  Effective density - fuel smeared over central
hole.
'     Fuel enrichment is 3.5%.
'
uo2    1 den=8.70  1 900  92235 3.50  92238 96.50 end
'      --------------------
'    some extra nuclides needed from library
zr-94  1 0 1-20     900 end
tc-99  1 0 1-20     900 end
ru-106 1 0 1-20     900 end
rh-103 1 0 1-20     900 end
rh-105 1 0 1-20     900 end
xe-131 1 0 1-20     900 end
cs-133 1 0 1-20     900 end
cs-134 1 0 1-20     900 end
ce-144 1 0 1-20     900 end
pr-143 1 0 1-20     900 end
nd-143 1 0 1-20     900 end
nd-145 1 0 1-20     900 end
nd-147 1 0 1-20     900 end
pm-147 1 0 1-20     900 end
sm-149 1 0 1-20     900 end
sm-151 1 0 1-20     900 end
sm-152 1 0 1-20     900 end
eu-153 1 0 1-20     900 end
eu-154 1 0 1-20     900 end
eu-155 1 0 1-20     900 end
u-232  1 0 1-20     900 end
u-233  1 0 1-20     900 end
u-237  1 0 1-20     900 end
np-238 1 0 1-20     900 end
pu-236 1 0 1-20     900 end
pu-237 1 0 1-20     900 end
pu-243 1 0 1-20     900 end
pu-244 1 0 1-20     900 end
am-242 1 0 1-20     900 end
cm-245 1 0 1-20     900 end
cm-246 1 0 1-20     900 end
cm-247 1 0 1-20     900 end
cm-248 1 0 1-20     900 end
bk-249 1 0 1-20     900 end
cf-249 1 0 1-20     900 end
cf-250 1 0 1-20     900 end
cf-251 1 0 1-20     900 end
cf-252 1 0 1-20     900 end
cf-253 1 0 1-20     900 end
'
' ---------------------------------------------------------------
'
'          Clad material.
```

```
'
arbm-NbSS  7.98 5 0 0 0 24000 20.0 28000 25.0 41000 1.0 26000
           53.92 6012 0.08 2 1.0 805  end
'
'          Fuel-channel CO2
'
arbm-channel-co2   0.0308 2 0 1 0 6012 1 8016 2 3 1.0 737  end
' ----------------------------------------------------------------
'          materials for larger-unit-cell
'
arbm-tie-bar 8.02 7 0 0 0 28000 45.0 24000 17.5 42000 3.8
         13000 1.31 22000 1.31 6000 0.08 26000 31.0 4 1.0 737 end
'
'
'    Graphite sleeve and CO2 around the sleeve.
'
arbm-sleeve-gr        1.65 1 0 0 0 6012 100.0 5 1.0     703 end
arbm-sleeve-co2 0.0323 2 0 0 0 6012 27.303 8016 72.697
                                           6 1.0     703 end
'
'    Outer graphite blocks and associated CO2.
'
arbm-outer-gr         1.65 1 0 0 0 6012 100.0 7 0.94586 703 end
arbm-outer-co2  0.0323 2 0 0 0 6012 27.303 8016 72.697
                                           7 0.05414 703 end
'
end comp
' ----------------------------------------------------------------
'    fuel pin cell geometry
'
triangpitch 2.886 1.4516 1 3 1.531 2 end
'
' ----------------------------------------------------------------
'     assembly and cycle parameters
'
'     assembly contains one metric ton of heavy metal.
'
npin/assm=36 fuelngth=2189.00 ncycles=21 nlib/cyc=1 lightel= 0
printlevel=4 inplevel=2 numzones=7 end
'
4 0.5061 3 1.1557 500 9.1651 3 9.607 5 11.9063 6 13.5 7 25.9527
'
 power=12.50 burn=1.0e-15 down=0.0 end
 power=12.50 burn=120     down=0.0 end
 power=12.50 burn=120     down=0.0 end
 power=12.50 burn=120     down=0.0 end
 power=12.50 burn=120     down=0.0 end
 power=12.50 burn=120     down=0.0 end
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 power=12.50 burn=120     down=0.0 end
 power=12.50 burn=120     down=0.0 end
 power=12.50 burn=120     down=0.0 end
```

```
 power=12.50 burn=120     down=0.0 end
 power=12.50 burn=120     down=0.0 end
 power=12.50 burn=120     down=0.0 end
 power=12.50 burn=120     down=0.0 end
 power=12.50 burn=120     down=0.0 end
 power=12.50 burn=120     down=0.0 end
' ------------------------------------------------------------------
end
```

SAS2H Input for a VVER-440.

```
=sas2h    parm=skipshipdata
  VVER-440 sas2h input
44groupndf5 latticecell
' ------------------------------------------------------------------
'  UO2 fuel rods.  Fuel smeared to include central hole and gap.
'             Fuel enrichment is 3.5%.
'
uo2    1 den=9.224  1 980  92235 3.50  92238 96.50 end
'      --------------------
'    some extra nuclides needed from library
zr-94  1 0 1-20     980 end
tc-99  1 0 1-20     980 end
ru-106 1 0 1-20     980 end
rh-103 1 0 1-20     980 end
rh-105 1 0 1-20     980 end
xe-131 1 0 1-20     980 end
cs-133 1 0 1-20     980 end
cs-134 1 0 1-20     980 end
ce-144 1 0 1-20     980 end
pr-143 1 0 1-20     980 end
nd-143 1 0 1-20     980 end
nd-145 1 0 1-20     980 end
nd-147 1 0 1-20     980 end
pm-147 1 0 1-20     980 end
sm-149 1 0 1-20     980 end
sm-151 1 0 1-20     980 end
sm-152 1 0 1-20     980 end
eu-153 1 0 1-20     980 end
eu-154 1 0 1-20     980 end
eu-155 1 0 1-20     980 end
u-232  1 0 1-20     980 end
u-233  1 0 1-20     980 end
u-237  1 0 1-20     980 end
np-238 1 0 1-20     980 end
pu-236 1 0 1-20     980 end
pu-237 1 0 1-20     980 end
pu-243 1 0 1-20     980 end
pu-244 1 0 1-20     980 end
am-242 1 0 1-20     980 end
cm-245 1 0 1-20     980 end
cm-246 1 0 1-20     980 end
cm-247 1 0 1-20     980 end
cm-248 1 0 1-20     980 end
bk-249 1 0 1-20     980 end
cf-249 1 0 1-20     980 end
cf-250 1 0 1-20     980 end
cf-251 1 0 1-20     980 end
cf-252 1 0 1-20     980 end
cf-253 1 0 1-20     980 end
'
' ------------------------------------------------------------------
'
'          Clad material.
'
```

```
arbm-clad  6.45 2 0 0 0 40000 99.0 41093 1.0 2 1.0 630 end
'
' ----------------------------------------------------------------------
---------
'
'          Moderator is H2O (600 ppm boron)
'
h2o        3 0.7300 575 end
arbm-bormod 0.730 1 1 0 0 5000 100 3 600.0-6 575 end
'
'          Shroud material.
'
arbm-shroud  6.45 2 0 0 0 40000 97.5 41093 2.5 4 1.0 575 end
'
end comp
'
' -----------------------------------------------------------------
'    fuel pin cell geometry
'
triangpitch 1.22 0.776 1 3 0.91 2 end
'
' -----------------------------------------------------------------
'     assembly and cycle parameters
'
'     assembly contains one metric ton of heavy metal.
'
npin/assm=126 fuelngth=2063.88 ncycles=21  nlib/cyc=1 lightel= 0
printlevel=4 inplevel=2 numzones=6 end
'
3 0.44 2 0.515 3 0.64054 500 7.21858 3 7.40577 4 7.56018
'
 power=40.00 burn=1.0e-15 down= 0  end
 power=40.00 burn=75.0    down= 0  end
 power=40.00 burn=75.0    down= 0  end
 power=40.00 burn=75.0    down= 0  end
 power=40.00 burn=75.0    down= 0  end
 power=40.00 burn=75.0    down= 0  end
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 power=40.00 burn=75.0    down= 0  end
 power=40.00 burn=75.0    down= 0  end
 power=40.00 burn=75.0    down= 0  end
 power=40.00 burn=75.0    down= 0  end
' ----------------------------------------------------------------
end
```

SAS2H Input for a VVER-1000

```
=sas2h    parm=skipshipdata
VV10004060     VVER-1000 sas2h input
44groupndf5 latticecell
' ----------------------------------------------------------------
' This is a model centered around ONE BA rod (1/18 of assembly)
' ----------------------------------------------------------------
'         UO2 fuel rods.  Fuel true density.
'         Central hole area will be added to gap.
'         Fuel enrichment is 4.0%.
'
uo2    1 den=10.13  1 966  92235 4.00  92238 96.00 end
'      --------------------
'    some extra nuclides needed from library
zr-94  1 0 1-20     966 end
tc-99  1 0 1-20     966 end
ru-106 1 0 1-20     966 end
rh-103 1 0 1-20     966 end
rh-105 1 0 1-20     966 end
xe-131 1 0 1-20     966 end
cs-133 1 0 1-20     966 end
cs-134 1 0 1-20     966 end
ce-144 1 0 1-20     966 end
pr-143 1 0 1-20     966 end
nd-143 1 0 1-20     966 end
nd-145 1 0 1-20     966 end
nd-147 1 0 1-20     966 end
pm-147 1 0 1-20     966 end
sm-149 1 0 1-20     966 end
sm-151 1 0 1-20     966 end
sm-152 1 0 1-20     966 end
eu-153 1 0 1-20     966 end
eu-154 1 0 1-20     966 end
eu-155 1 0 1-20     966 end
u-232  1 0 1-20     966 end
u-233  1 0 1-20     966 end
u-237  1 0 1-20     966 end
pu-236 1 0 1-20     966 end
pu-237 1 0 1-20     966 end
pu-243 1 0 1-20     966 end
pu-244 1 0 1-20     966 end
cm-245 1 0 1-20     966 end
cm-246 1 0 1-20     966 end
cm-247 1 0 1-20     966 end
cm-248 1 0 1-20     966 end
bk-249 1 0 1-20     966 end
cf-249 1 0 1-20     966 end
cf-250 1 0 1-20     966 end
cf-251 1 0 1-20     966 end
cf-252 1 0 1-20     966 end
'
' ----------------------------------------------------------------
'
'          Clad material.
'
```

```
arbm-clad  6.45 3 0 0 0 40000 98.97 41093 1.0 72000 0.03 2 1.0 630 end
'
'          Moderator is H2O (734 ppm boron)
'
h2o        3 0.6900                            578 end
arbm-bormod 0.6900 1 1 0 0 5000 100 3 734.0-6 578 end
'
' ---------------------------------------------------------------
'
'    Mixtures for larger unit cell.
'
'    Burnable Absorber.
'
arbm_ba     2.9 7 1 0 0 5010 0.2258 5011 1.0144 13027 91.7511
            26000 0.1919 28000 1.9127 24000 2.9903 40000 1.9138
            4 1.0 578 end
'
'    clad for absorber (composition is same as fuel clad).
'
arbm-cclad  6.45 3 0 0 0 40000 98.97 41093 1.0 72000 0.03 5 1.0 578 end
'
end comp
' ---------------------------------------------------------------
'    fuel pin cell geometry
'
'       use a gap and include plenum in gap area
'       fuel radius is effective radius when plenum is added to gap.
'
triangpitch 1.275 0.71911 1 3 0.9164 2 0.772 0 end
'
' ---------------------------------------------------------------
'     assembly and cycle parameters
'
npin/assm=312 fuelngth=883.972 ncycles=21 nlib/cyc=1 lightel= 0
printlevel=4 inplevel=2 numzones=5 mxrepeats=0 end
'
'  modeling 1/18 of assembly (around one BA rod).
'     (assembly contains one metric ton)
'
4 0.35 5 0.51235 3 0.66943 500 2.8663 3 2.91385
'
'    Following lines may be commented out for checking purposes.
'
 4 0.35 5 0.51235 3 0.66943 500 2.8663 3 2.91385
 4 0.35 5 0.51235 3 0.66943 500 2.8663 3 2.91385
 4 0.35 5 0.51235 3 0.66943 500 2.8663 3 2.91385
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 3 0.35 5 0.51235 3 0.66943 500 2.8663 3 2.91385
 3 0.35 5 0.51235 3 0.66943 500 2.8663 3 2.91385
 3 0.35 5 0.51235 3 0.66943 500 2.8663 3 2.91385
 3 0.35 5 0.51235 3 0.66943 500 2.8663 3 2.91385
 3 0.35 5 0.51235 3 0.66943 500 2.8663 3 2.91385
 3 0.35 5 0.51235 3 0.66943 500 2.8663 3 2.91385
'
'
 power=1.0e-9 burn=1      down=0.0 end
'
'    Following lines may be commented out for checking purposes.
'
 power=40.00 burn=75      down=0.0 end
 power=40.00 burn=75      down=0.0 end
 power=40.00 burn=75      down=0.0 end
 power=40.00 burn=75      down=0.0 end
 power=40.00 burn=75      down=0.0 end
```

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