NEUTRONIC AND THERMAL-HYDRAULICS CALCULATIONS OF U-Mo DISPERSION FUEL FOR UTILIZATION IN THE IEA-R1 REACTOR CORE

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ABSTRACT

The U-Mo dispersion fuels of Material Test Reactors (MTR) are analyzed in terms of their irradiation performance. The irradiation performance aspects are associated to the neutronic and thermal-hydraulics aspects in order to propose a new core configuration to the IEA-R1 research reactor of IPEN-CNEN/SP, using U-Mo dispersion fuels. Core configurations using U-10Mo-Al fuels with uranium densities ranging from 3 to 8 gU/cm^3 were analyzed with the computational programs CITATION and MTRCR-IEAR1. Core configurations for fuels with uranium densities ranging from 3 to 5 gU/cm^3 showed to be adequate to be used in IEA-R1 reactor and would present a stable in-reactor performance even at high burnup.

1. Introduction

The IEA-R1 reactor of IPEN-CNEN/SP in Brazil is a pool type research reactor cooled and moderated by demineralized water and having Beryllium and Graphite as reflectors. In 1997 the reactor received the operating licensing for 5 MW. In the last years, IPEN has developed the project and fabrication of U_3O_8-Al and U_3Si_2-Al dispersion fuels [1]. The U_3O_8-Al dispersion fuel is qualified up to a uranium density of 2.3gU/cm^3 and the U_3Si_2-Al dispersion fuel up to a uranium density of 3.0gU/cm^3. Nowadays the IEA-R1 reactor core is constituted of the fuels above, with low enrichment in U-235 (19.9% of U-235). These fuels follow rigorous technical specifications that were developed after a careful bibliography revision, comprising the world experience in the project, fabrication and fuel performance analysis of dispersion fuels. Analogous to the realized for the development, fabrication and qualification of U_3O_8-Al and U_3Si_2-Al fuels at IPEN-CNEN/SP, the aim of this paper [2] is the development of an extended bibliography revision on the irradiation performance of U-Mo alloy dispersed in an aluminum matrix (Al) and on hand of this revision, the attempt to establish a set of parameters that could help in the definition of the technical specifications for fabrication of this type of fuel and its posterior utilization in the IEA-R1 research reactor. A set of IEA-R1 core configurations using U-10Mo-Al fuel, with uranium densities ranging from 3.0 to 8.0gU/cm^3, was analyzed. Due the higher density of the analyzed U-10Mo-Al fuels compared to the U_3O_8-Al and U_3Si_2-Al qualified fuels at IPEN-CNEN/SP, it could be possible to reduce the number of fuel elements in the IEA-R1 reactor core, which generated the necessity to review the neutronic and thermal-hydraulics reactor core projects. The core neutronic calculation was developed with the computer program CITATION [3]. The thermal-hydraulics analysis was developed with the computer program MTRCR-IEA-R1 [4]. The MTRCR-IEA-R1 program permits the calculation of the fuel thermal and hydraulics parameters of the reactor core. The analysis has been made for a reactor operating power of 5 MW.

2. Bibliography revision

Since the eighties of the last century, countries that detain the nuclear technology have concentrated efforts in studying U-Mo dispersion fuels. This kind of fuel can have uranium densities up to 8gU/cm^3 and has been studied as a possible substitute fuel for U_3O_8-Al and U_3Si_2-Al fuels in research reactors with high power and high neutronic flux. The uranium density value of 8gU/cm^3 represents U-10Mo fuel particle loadings of about 50 vol.-% in the meat.

Early irradiation experiments with uranium alloys showed the promise of acceptable irradiation behavior, if these alloys could be maintained in their cubic γ-U crystal structure [5]. Many experiments have demonstrated that centrifugally atomized U-Mo powder can retain this gamma uranium phase during fuel element fabrication and irradiation and can be compatible with the aluminum matrix, becoming the prime candidate for dispersion fuels for research reactors. A set of irradiation tests has been conducted around the world for this alloy. Fourteen different fuel compositions, including twelve metallic alloys, have been irradiated as part of five separate experiments for high-density dispersion fuel development in the Advanced Test Reactor (ATR) at the Idaho National Engineering & Environmental Laboratory [6].

The irradiation performance data obtained from these tests had led the US-RERTR program to narrow its focus toward the U-Mo binary alloy system as its primary candidate for use in a high-density dispersion fuel. In the experiments RERTR-1 and RERTR-2, the tested fuel plates were fabricated with fuel particles loadings of only 25 to 30 vol.-% in the meat, giving meat-averaged densities of ~ 4 gU/cm^3. The particular focus of these experiments was to observe the phenomena of fuel-matrix interaction and the fuel particle swelling under irradiation. The fuel plate powers, and consequently temperatures, were maintained low. Fuel plates fabricated with the U-4Mo alloy showed poor behavior. The U-Mo alloys fabricated with at least 6wt.% performed well up to 70% burnup. The RERTR-3 experiment was designed to test experimental fuel plates under irradiation conditions considered aggressive for research reactor fuels. Forty-seven miniature fuel plates were fabricated and irradiated to a nominal U-235 burnup level of 40%. Based on the results of the RERTR-1 and -2 experiments, the RERTR-3 experiment focused principally on the U-Mo binary alloy fuels with 6Mo-10Mo wt.%. In this experiment, the test fuels were fabricated with fuel particle loadings of over 50 vol.-% in the meat, giving meat-averaged uranium densities of up to 8.5 gU/cm^3. PIE of these fuel plates showed generally acceptable fuel performance. The fuel swelling was relatively low, with no tendency toward breakway behavior from microscopy. However, at the elevated fuel temperatures of this experiment significant fuel-matrix interaction was observed. In fact, fuel-matrix interaction was so extensive that no matrix Al remained in the hot central portion of the fuel meat in some fuel plates. Nonetheless, acceptable fuel plate performance was achieved even in cases where all of the matrix Al phase was consumed.

The experiments RERTR-4 and RERTR-5 were designed to test larger fuel plates irradiated to nominal U-235 burnup levels of 50 and 80%. These experiments continue to focus on the U-Mo binary alloys with 6Mo-10Mo wt%. The test results of the RERTR-4 [7] experiments indicated that the formation of the aluminide interaction phase appeared to be the only aspect of fuel behavior that is significantly affected by temperature. The irradiation behavior of the U-Mo fuel alloy itself was deemed athermal over the temperature range tested. The most important observations of the RERTR-4 experiment were: stable, apparently athermal swelling of the U-Mo alloy particles with the presence of small uniformly distributed fission gas bubbles. No evidence of unstable, break-way, swelling, characteristic of other high-density fuels, has been found. The formation of a U-Mo/Al interaction phase will be significant, consuming practically all matrix aluminum at higher temperatures (150°C). The fission induced swelling rate of this compound is, however, low and very stable. The interaction phase occupies a larger volume than its U-Mo and Al constituents and therefore, contributes to swelling.

This contribution, however, is limited by the amount of Al matrix available. The main effect of the interaction product formation is the reduction of thermal conductivity of the meat [8], which should be carefully assessed for a particular fuel design. As the interaction proceeds, a low-conductivity reaction-product phase builds-up, with the corresponding depletion of high-conductivity Al matrix phase. This leads to a substantial degradation of fuel meat thermal conductivity with time, and fuel centerline temperatures can increase with burnup even plate power decreases. The U-10Mo-Al dispersion fuel has been the most studied and has presented excellent performance under irradiation to nominal U-235 burnup level of 80% and with uranium densities ranging from 3 to 9 gU/cm^3.

Therefore, based on the above results, this type of U-Mo dispersion was chosen to be analyzed for posterior utilization in the IEA-R1 reactor core.

3. Definition of a new core for the IEA-R1 Research Reactor using U-Mo dispersion fuel

For the definition of a new IEA-R1 reactor core, neutronic and thermal-hydraulics calculations were developed for the U-10Mo-Al fuels with densities ranging from 3 to 8gU/cm^3. The uranium density value of 8gU/cm^3 was chosen because it represents U-Mo fuel particle loadings of about 50 vol.-% in the meat, value normally considered as a limit to maintain the mechanical integrity of the fuel plate under irradiation. The uranium density value of 3gU/cm^3 was chosen as the minimum value utilized, because it is the maximal meat uranium density qualified for the U_3Si_2-Al fuel fabricated at IPEN-CNEN/SP. Densities smaller than 3 gU/cm^3 are not of interest. Nowadays the IEA-R1 reactor core has a typical configuration with 24 elements, being 20 standard elements of U_3O_8-Al and U_3Si_2-Al dispersion fuels, 4 control elements, and one Beryllium irradiation element in the central position of the core. In the neutronic calculation the computer program CITATION was utilized for the three-dimensional core calculation and for burnup calculation. The radial and axial power density curves were utilized as input data for the thermal and hydraulics core analyses. The neutronic calculation results showed that the analyzed 3x3 core configuration (9 elements), with 4 standard fuel elements, 4 control elements, and 1 Beryllium irradiation element in the central position of the core and with uranium densities ranging from 6 to 8gU/cm^3 was very reactive and technically inadequate for the IEA-R1 reactor core at 5 MW. The analyses were concentrated in reactor core configurations using 8 standard fuel elements, 4 control elements and 1 Beryllium irradiation element in the central position and with uranium densities ranging from 3 to 5gU/cm^3. The beginning of life neutronic calculation showed that the cores with uranium densities of 4 and 5gU/cm^3 presented high reactivity excess (1.1367 and 1.1604, respectively). In those cases, for uranium densities of 4 and 5 gU/cm^3, a new core configuration was defined having only 11 elements (6 standard fuel elements, 4 control elements and 1 Beryllium irradiation element in the central position of the core).

In the year 2000 was finalized at IPEN-CNEN/SP through the commercial program Engineering Equation Solver (EES) a new thermal-hydraulics model, the model MTCR-IEA-R1 [3]. Using this computer model it has been possible to realize the steady-state thermal and hydraulics core analyses of research reactors with MTR fuels. The following parameters are calculated along the fuel element channels: fuel meat central temperature (T_c), cladding temperature (T_r), coolant temperature (T_f), the Onset of Nucleate Boiling (ONB) temperature (T_{onb}), the critical heat flux (Departure of Nucleate Boiling-DNB), flow instability and the thermal-hydraulics safety margins MDNBR e FIR. The thermal-hydraulics safety margins MDNBR and FIR are calculated as the relation between, respectively, the critical heat flux and the heat flux for flow instability and the local heat flux in the fuel plate. Furthermore, the MTCR-IEA-R1 model also utilizes in its calculation, the involved uncertainties in the thermal hydraulics calculation as, for instance, fuel fabrication uncertainties, error in the power density distribution calculation, in the coolant flow distribution in the core, reactor power control deviation, in the coolant flow measures, and in the safety margins for the heat transfer coefficients. The calculated thermal and hydraulics core parameters are compared with the design limits established for MTR fuels: a) cladding temperature < 95°C; 2) safety margin for the onset of nucleate boiling higher than 1.3, or the ONB temperature higher than coolant temperature; 3) safety margin for flow instability higher than 2.0; and 4) safety margin for critical heat flux higher than 2.0.

As mentioned in the bibliography revision (item 2), due to the fuel particle-aluminum matrix interaction, the fuel meat thermal conductivity reduces during irradiation. For studying the thermal behavior of the U-10Mo-Al fuel with the program MTRCR-IEA-R1, it was necessary to provide the meat fuel thermal conductivity as input data. Two values of thermal conductivity for the U-10Mo-Al meat were utilized: 70Wm°C and 13W/m°C. The value 70W/m°C was utilized because it represents the meat thermal conductivity value for U-Mo fuel particle loadings of about 50 vol.-% in the meat [8]. For fuel particle loadings smaller than 50 vol.-%, the meat thermal conductivity is higher, bringing smaller fuel temperatures when compared with the temperatures calculated for the meat thermal conductivity value of 70Wm°C. When the aluminum in the meat is completely consumed, with 100% of fuel particle-matrix interaction products, the meat thermal conductivity reaches values around 13W/m°C [8]. The value 70W/m°C would be more representative for reactor cores where the fuel temperatures would present values under 100°C, or for cores in the beginning of life. The meat uranium density value of 13W/m°C would be more representative for cores with higher temperatures, where the fuel particle-matrix interaction is so extensive that no matrix Al remained in the hot central portion of the fuel meat. In the thermal-hydraulics analysis the computer program MTRCR-IEA-R1 was utilized with the radial and axial power distribution curves provided for the computer program CITATION. The input data for the thermal–hydraulics simulations were obtained from reference [2]. The U-Mo fuel-plate geometric dimensions used in the simulations were the same of those of U_3Si_2-Al fabricated at IPEN-CNEN/SP. The core with thirteen elements and uranium density of 3gU/cm^3 in the fuel-plate was simulated first with the MTRCR-IEA-R1 (Simulation 1) for a meat thermal conductivity of 13W/m°C, without uncertainties treatment involved (nominal condition) and with uncertainties treatment involved (Simulation 2). Afterwards the same core was simulated for a meat thermal conductivity of 70W/m°C, in the nominal condition (Simulation 3) and with uncertainties treatment involved (Simulation 4). The results are presented, respectively, in the Table 1. The same sequence was utilized for the core simulations with ten elements and uranium densities, of respectively, 4gU/cm^3 and 5gU/cm^3 (Simulations 5 to 8 and 9 to 12, Table 1).

Simulation	T_f (°C)	T_r (°C)	T_c (°C)	T_{onb} (°C)	MDNBR	Flow Instability (FIR)
01	47,73	69,05	93,37	120,5	7,73	24,06
02	52,95	86,58	127,3	122,3	4,26	14,37
03	47,73	69,05	74,86	120,5	7,73	24,06
04	52,95	86,58	96,36	122,3	4,26	14,37
05	47,79	69,57	98,81	121,1	7,13	29,54
06	53,05	87,43	136,3	123	3,92	17,64
07	47,79	69,57	76,6	121,1	7,13	29,54
08	53,05	87,43	99,19	123	3,92	17,64
09	47,86	73,68	104,3	121,3	6,41	29,53
10	53,16	93,82	145	123,2	3,63	17,64
11	47,86	73,68	81,04	121,3	6,41	29,53
12	53,16	93,82	106,1	123,2	3,52	17,64

Tab 1: Simulations with the computer program MTRCR-IEA-R1 for reactor cores with 12 and 10 elements, meat uranium densities of 3, 4 e 5gU/cm^3 e thermal conductivities of 13W/m°C and 70W/m°C.

4. Main results and conclusion

The simulation results described in Table 1 show that no design limit is achieved for the analyzed cores. The calculated cladding temperatures are under the value of 95°C, reaching for the reactor cores with uranium density fuels of 5gU/cm^3 (simulations 10 and 12) the maximal value of 93.82°C. This was expected because these analyzed cores had the highest meat uranium density in the fuel plates, which also achieved the highest calculated peak factor (F_q = 2.1850) in the axial power distribution calculation of the fuel element hot channel, when compared to the peak factors (F_q = 2.076 and F_q = 2.118, respectively) for the core configurations with meat uranium densities of 3 and 4gU/cm^3 in the fuel plates. The temperatures in the simulations, without uncertainties treatment, are well below those obtained with uncertainties treatment. From Table 1 it can be seen that the coolant temperature (T_f) for all simulations are below the ONB temperature, indicating one-phase flow in the simulated cores. The margins for critical heat flux (MDNBR) and flow instability (FIR) are well above the value 2.0, admitted as design limit. The maximal fuel meat central temperature was 145°C. At this temperature the fuel particle-matrix interaction would be completed, reaching the meat thermal conductivity value of 13W/cm°C. The interaction phase U-10Mo-Al would consume practically all of the matrix Al phase. However, as seen in the item 2, for U-10Mo-Al dispersions the fuel behavior would be stable even at higher burnup.

These simulation results were utilized to propose the meat uranium density of $5gU/cm^3$ for the mini-plates of U-Mo dispersion fuel to be fabricated at IPEN-CNEN/SP and tested in the IEA-R1 reactor. With this uranium density in the fuel meat, the number of fuel elements used in the IEA-R1 reactor would be reduced, bringing economic advantages and also reducing the number of spent fuel elements to be stored in the reactor pool.

5. References

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