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METHOD FOR ASSESSMENT OF RESIDUAL HEAT GENERATION WITHIN PREPARATION OF SPENT NUCLEAR FUEL SHIPMENT FROM RESEARCH REACTORS

In the presented paper is considered the approach to an assessment of the residual heat generation within preparation of spent nuclear fuel (SNF) shipment from research reactors. A possible approach to solving the problem for residual heat generation assessment at cyclic mode of facility operation is discussed, i.e. after several periods of operation and idle time, including at different levels of the reactor operational power. The approach is scrutinized in two possible mathematical models of calculations – weekly and annually modeling of the input data for calculations. In order to prove the correctness of the calculations based on an annual operating cycle in the mathematical modeling, calculations for both models under the same operating conditions were made and comparative analysis of the results was prepared. Although such approach ensures a correct assessment at variable operation parameters (power level, duration of operation and decay time of the fuel) for the purpose of the calculations is selected an operating mode with repeated parameter values in the separate operational cycles in order to facilitate the process of analysis. As conclusion it is noted that significantly simplified calculations by annual model give correct enough and conservative results on the basis of which may be provided the necessary information according the requirements of the country which is to accept the fuel for reprocessing. On the basis of these results the selection of a cask for the SNF shipment can be done. The presented approach is applied to evaluate the residual heat generation in the SNF during its shipment preparation from the nuclear research reactor IRT-2000, Sofia, Bulgaria, to the Russian Federation within RRRFR programme.

Keywords: research reactor spent nuclear fuel shipment, assessment of residual heat generation, cyclic operational mode.

1. Introduction

Important step in the management of the nuclear fuel cycle as a whole is the transportation of the nuclear material. This activity has its specificity, especially when transporting irradiated or SNF from research reactors. The process of SNF shipment preparation covers a number of activities, including:

review and analysis of the existing irradiated nuclear material and documentation compliance with the requirements of the country which is accepting the fuel for reprocessing;

review and analysis of the parameters of the casks suitable for transport of irradiated nuclear material;

review and analysis of the compatibility of the existing nuclear material with suitable casks.

An important parameter in the conduct of these three studies is the value of the residual heat generation in the spent/irradiated nuclear fuel. Presently in the publicly available scientific and technical literature could not be found a proven and validated method for calculating the residual heat generation in the SNF from research reactors which takes into account the specifics of their operating mode.

Most research reactors, especially those with a rated power of 2 MW or less, are operated in a cyclic mode. This means operation about 8 and staying about 16 hours per day during the five working days of the week and staying through the weekend. Furthermore – the reactor operation is not only in a cyclic mode and with different duration, but at a different power level and different reactor core configuration depending on the ongoing experiments. This can lead to cases where the operation of the separate fuel assembly has been interrupted for years, and then again continued. Typical specificity of research reactors is not only the specific mode of operation but also the characteristics of the fuel, which very often is in the category of highly enriched nuclear material and has a specific design as well. SNF shipment from research reactors with thermal power up to 2000 kW happens only two or three times, and sometimes only once for the facility lifetime, which allows for a significant duration of the decay time for the fission products in some of the fuel assemblies. These factors further complicate the assessment of residual heat generation at the moment of the shipment.

Developed and widely applied codes to calculate the numbers of parameters of the irradiated and/or

SNF from power reactors, including residual heat generation, are not applicable to research reactors without seriously complicating the calculations due to the cyclic operation on different power levels. To use these codes in practice, each operating cycle should be modeled with the relevant duration and power level as well as decay time duration of the fission products at the moment of the shipment. Assuming ten years of operation, 52 weeks in a year and five days per week this makes approximately 2600 calculations for operating cycles, most likely with different input parameters. This makes such calculations too cumbersome and unduly expensive.

All these factors necessitate the development of an adequate simplified numerical method for assessing the residual heat generation in the SNF at the moment of its shipment from the facility site, taking into account the specifics of the operation of nuclear research reactors.

2. Mathematical modeling of the process

Obviously, at cyclic operational mode of the research reactors there is accumulation of fission products from the previous operation cycles. Precisely the fission products are the source of residual heat generation in a prolonged period after the reactor is stopped, but there is a decay time accumulation as well. The fission products accumulated from one operating cycle at a given moment of time directly depend on the power level, duration of the operation and the decay time at this moment. The proposed two simplified numerical methods are aimed at averaging the power levels of several cycles and their merging into one operating cycle after which starts accounting of the decay time duration.

2.1. Mathematical model 1 – weekly averaging

Most suitable for estimating of the residual heat generation N within weekly averaging would be the equation [1]

$$N = 10^{-2} \cdot 2.866 \cdot N_0 \cdot \left[t_s^{-0.2} - \left(t_s + T \right)^{-0.2} \right], \qquad (1)$$

where N – residual heat generation power, kW; N_0 – operating reactor power until shutdown, kW; t_s – decay time after reactor shutdown, min; T – duration of the reactor operation at power N_0, min.

To take into account the specificity of the cyclic operational mode with varying duration and power, as well as varying duration of the decay time of fission products, in the calculations is used the approach presented in [2], with some additional corrections in the mathematical modeling of the input data concerning duration and power level of the operation during the working week. The corrections are related to the source of the residual heat generation in the spent fuel in the long term – the long half-life radioactive fission products. This allows the operating cycles for the week to be merged into one. It is obvious that at such formulation of the task the fuel decay time between working days during the week is overlooked, but for the purpose of assessing the residual heat generation within the SNF shipment preparation they are irrelevant because they refer to fission products with short half-life. The duration of this united cycle is equal to the sum of the duration of the weekly cycles. After the duration of operation is defined, the other factor influencing the production of fission products – the power level, should be determined.

If every workweek the research reactor runs j times at different power N_j, kW, and for different duration T_j, min, (sometimes more than once per day) at the end of the week will be accumulated certain amount of produced energy Q_w, measured in kWmin. Then the energy produced weekly is

$$Q_w = N_1 \cdot T_1 + N_2 \cdot T_1 + \ldots + N_j \cdot T_j. \qquad (2)$$

When the amount of produced energy and duration of the work as a whole during the week are known the reactor operational power N_{wav} averaged over the week can be calculated

$$N_{wav} = \frac{Q_w}{T_w}, \qquad (3)$$

where T_W – duration of the reactor operation as a whole during the week, min

$$T_w = T_1 + T_2 + \ldots + T_j. \qquad (4)$$

Substituting the received values into Eq. (1), the residual heat generation power N_W as a result of this one week of operation is

$$N_w = 10^{-2} \cdot 2.866 \cdot N_{wav} \cdot \left[t_s^{-0.2} - \left(t_s + T_w \right)^{-0.2} \right], \qquad (5)$$

where t_s – decay time between reactor shutdown or fuel assembly removal from the reactor core at the end of the considered working week and the moment of the SNF shipment, min.

Assuming that the fuel decay time increases exactly with one week between two weekly cycles of work, for i number of weeks of operation at the moment of the SNF shipment according to [2] the accumulated residual heat generation is

$$N_{\Sigma i} = N_{w1} + N_{w2} + \ldots + N_{wi}, \qquad (6)$$

where

$$N_{wi} = 10^{-2} \cdot 2.866 \cdot N_{wavi} \cdot \left[t_{si}^{-0.2} - \left(t_{si} + T_{wi} \right)^{-0.2} \right]. \quad (7)$$

N_{Wi} – accumulated residual heat generation as a result from operational week i after decay time t_{si}, kW; N_{wavi} – average operational power level during week i, kW; T_{wi} – duration of the operation during week i, min; t_{si} – fuel decay time between the end of the operational cycle of the week i and the moment of the SNF shipment, min.

In such an enunciation of the task each operating week can be individual as duration and power level of operation, and hence the power of the residual heat generation resulting from this week of operation is individual. This allows with sufficient accuracy to be taken into account the specific mode of operation of the research reactor which proves the correctness of the mathematical model of computation. Furthermore, when considering ten years of operation such an approach reduces the number of calculated operating cycles from 2600 to 520.

2.2. Mathematical model 2 – yearly averaging

Despite the satisfactory accuracy of the model concerning the operational mode of research reactors it is nonetheless too clumpy when considering the prolonged periods of operation and fuel decay time. On the account of the low operational parameters of the fuel used in research reactors compared with the ones of power reactors, the operational cycle of the former can reach and even exceed ten years. Similarly the fuel assemblies decay time is accumulated during idle time in the reactor core with the time spent in the repository for irradiated/spent fuel. This total decay time can also reach ten or more years.

On grounds of the foregoing could be considered a model where the approach is analogous to the already presented, but this time the duration and the average power of operation are examined and merged into one operating cycle for a period of 52 weeks, which is approximately equivalent to one year. In this case more suitable for estimating of the residual heat generation would be the equation, where duration of the reactor operation and decay time after reactor shutdown are in days [3]

$$N = 10^{-3} \cdot 6.5 \cdot N_0 \cdot \left[t_s^{-0.2} - \left(t_s + T \right)^{-0.2} \right], \quad (8)$$

where N – residual heat generation power, kW; N_0 – operating reactor power until shutdown, kW; t_s – decay time after reactor shutdown, days; T – duration of the reactor operation at power N_0, days.

Then Eqs. (2) - (7) acquire the following form.

The amount of produced energy Q_y, measured in kWdays during the considered operational year is

$$Q_y = Q_{w1} + Q_{w2} + \ldots + Q_{w52}, \quad (9)$$

where Q_{wi} is calculated by Eq. (2) but T_j is measured in days.

Operational power level N_{yav} averaged over the year

$$N_{yav} = \frac{Q_y}{T_y}, \quad (10)$$

where T_y – duration of the reactor operation as a whole during the considered operational year as the sum of weekly duration of operation, days

$$T_y = T_{w1} + T_{w2} + \ldots + T_{w52}. \quad (11)$$

Substituting of the received values into Eq. (8), the residual heat generation N_y as a result of the considered operational year is

$$N_y = 10^{-3} \cdot 6.5 \cdot N_{yav} \cdot \left[t_{si}^{-0.2} - \left(t_s + T_y \right)^{-0.2} \right], \quad (12)$$

where t_s – decay time between the end of the considered operational year and the moment of the SNF shipment, days.

Assuming that the fuel decay time increases exactly with one year between two annual cycles of work for i number of years of operation at the moment of the SNF shipment according to [2], the accumulated residual heat generation is

$$N_{\Sigma i} = N_{y1} + N_{y2} + \ldots + N_{yi}, \quad (13)$$

where

$$N_{yi} = 10^{-3} \cdot 6.5 \cdot N_{yavi} \cdot \left[t_{si}^{-0.2} - \left(t_{si} + T_{yi} \right)^{-0.2} \right]. \quad (14)$$

N_{yi} – accumulated residual heat generation as a result from operational year i after decay time t_{si}, kW; N_{yavi} – average operational power level during operational year i, kW; T_{yi} – duration of the operation during operational year i, days; t_{si} – fuel decay time between the end of the operational cycle of the year i and the moment of the SNF shipment, days.

For ten years of operation such an approach reduces the number of considered operating cycles from 520 (Mathematical model 1) to 10. Despite of the chosen model of computation the following records must be available in order to ensure the necessary input information:

history of the operating power of the reactor core as a whole;

history of the operating power of the fuel assembly as a part of the core.

The mode of operation and management of the nuclear fuel cycle at research reactors has its specificity which is quite different from that of power reactors. The frequent changes of the core configuration required by the nature of the ongoing experiments leads to a considerably larger volume of records. Depending on the involved rules for records keeping by the Quality Assurance System for management of the nuclear fuel cycle, the produced energy could be constantly accounted for the operational power of the reactor core as well as the separate fuel assemblies and could be averaged regularly. It would be felicitous that the relevant records are entered in the file of each fuel assembly. This would facilitate both current nuclear material accountancy and upcoming preparation for SNF shipment.

In order to prove the correctness of the calculations based on an annual operating cycle (Mathematical model 2) both mathematical models were prepared under the same operating conditions and comparative analysis of the obtained results was made.

3. Calculations and results

Although such approach ensures a correct assessment at variable operation parameters (power level, duration of operation and decay time of the fuel) for the purpose of the calculations is selected an operating mode with repeated parameter values in the separate operational cycles in order to facilitate the process of analysis. The calculations have been made with the following initial conditions and assumptions:

ten-year period of operation and twenty years decay time of the fuel;

the idle time during the weekends is accounted in the calculations by Mathematical model 1;

an operating cycle of 12 hours of work and 12 hours of downtime, five days per week is adopted;

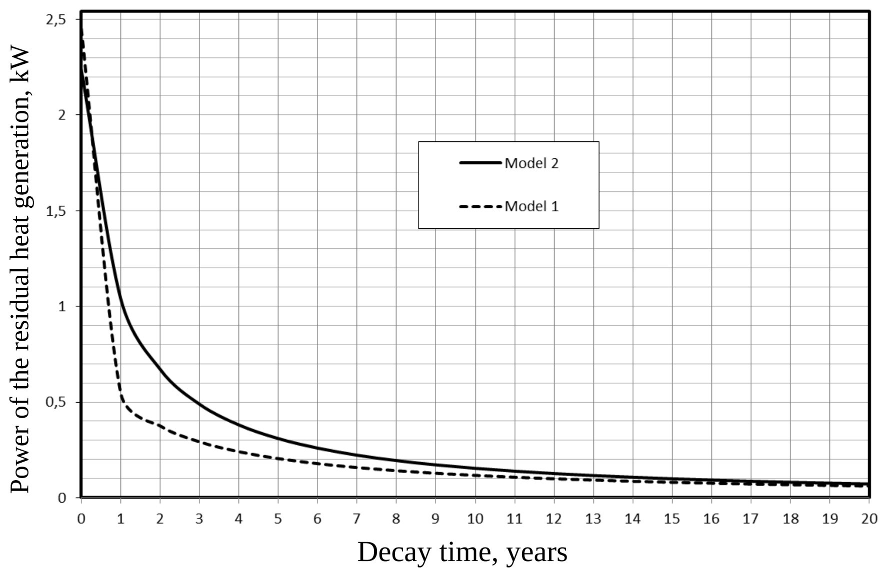
the reactor is operated at a power level of 2000 kW;

transmutation of the fission products as a result of irradiation in the reactor core during operation is ignored;

the calculations results are for the reactor core as a whole;

the contribution of a separate fuel assembly is assessed subsequently on the basis of the design and operational documentation and the assembly’s operational file which is a part of the nuclear material accounting records.

The calculation results for the amendment of the residual heat generation with the increase of the decay time obtained with both models are presented graphically in Figure.



Calculation results for the amendment of the residual heat generation with the increase of the decay time obtained with both models.

Deviations of the calculation results obtained with Mathematical model 2 from those with Mathematical model 1 are presented numerically in Table 1.

4. Discussion

The significant decrease in the number of the considered operating cycles makes it possible to calculate relatively easy the parameters of SNF with the available computer codes for power reactors. However, preliminary preparation of input data for calculations must be carried out according to the presented above methodology.

The reactor core is generally configured of about 20 or more fuel assemblies in order to operate at power level of 2000 kW. If a configuration of 20

Table 1. **Deviations of the calculation results obtained with Mathematical model 2 from those with Mathematical model 1**

Decay time, years	Deviations of Model 2 from Model 1, %	Deviations of Model 2 from Model 1, kW	Decay time, years	Deviations of Model 2 from Model 1, %	Deviations of Model 2 from Model 1, kW
1	47,4	0,493	11	22,6	0,032
2	44,0	0,296	12	21,4	0,027
3	40,2	0,196	13	20,3	0,024
4	36,8	0,140	14	19,4	0,021
5	33,8	0,105	15	18,5	0,018
6	31,3	0,081	16	17,7	0,016
7	29,1	0,064	17	16,9	0,015
8	27,2	0,053	18	16,2	0,013
9	25,5	0,044	19	15,6	0,012
10	23,9	0,037	20	14,9	0,011

Table 2. **Average residual heat generation (RHG) amendment of a fuel assembly with the increase of the decay time calculated by Mathematical model 2**

Years	1	2	3	4	5	6	7	8	9	10
RHG, W	51.97	33.58	24.44	19.03	15.48	12.98	11.14	9.72	8.61	7.70
Years	11	12	13	14	15	16	17	18	19	20
RHG, W	5.39	6.34	5.81	5.36	4.97	4.63	4.33	4.06	3.83	3.61

fuel assemblies with a conditional even distribution of power is adopted, the average power of the residual heat generation (RHG) amendment of a fuel assembly calculated by Mathematical model 2 with the increase of the decay time is presented in Table 2.

Table 3 presents the characteristics of SNF shipment casks TUK-19, ŠKODA VPVR/M and NAC-LWT concerning the presented study.

Comparing the data in Table 2 and Table 3 shows that after three years of decay time SNF can be shipped in any of the presented casks.

Table 3. **Characteristics of SNF shipment casks TUK-19, ŠKODA VPVR/M and NAC-LWT**

Cask type	Capacity in number of fuel assemblies	Admissible heating rate in the cask, W	Average RHG of a fuel assembly, W
TUK-19	4	112	28
ŠKODA VPVR/M	36	900	25
NAC-LWT	28 MTR (HEU)	1260	45
	42 MTR (LEU)	1000	24

As can be seen on graphic presentation (see Figure) of the results for the initial year of the fuel decay time, there is an increase of the power of the residual heat generation from the calculation by Mathematical model 1. This is due to the significantly shorter decay time of the fission products generated in the fuel as a result of the last operating cycles of the reactor. Just when the decay time reaches one year the calculation results from Mathematical model 2 are higher than those from Mathematical model 1. The percentage increase seems significant, but the difference in the absolute values for the power of the residual heat generation is acceptable according to the information presented in Tables 2 and 3. Furthermore, the increased value ensures conservatism of the obtained results and gives some safety margin in the selection of the cask for SNF shipment.

5. Conclusions

The significantly simplified calculations by Mathematical model 2 give correct enough and conservative results on the basis of which may be provided the necessary information according the requirements of the country which is to accept the fuel for reprocessing. On the basis of these results the selection of a cask for the SNF shipment can be done.

The presented approach is applied to evaluate the residual heat generation in the SNF during its shipment preparation from the nuclear research reactor IRT-2000, Sofia, Bulgaria, to the Russian Federation within RRRFR programme.

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МЕТОДИКА ОЦІНКИ ПОТУЖНОСТІ ЗАЛИШКОВОГО ТЕПЛОВИДІЛЕННЯ В РАМКАХ ПІДГОТОВКИ ДО ВИВЕЗЕННЯ ВІДПРАЦЬОВАНОГО ЯДЕРНОГО ПАЛИВА З ДОСЛІДНИЦЬКИХ РЕАКТОРІВ

Розглядається метод оцінки потужності залишкового тепловиділення при підготовці до вивезення відпрацьованого ядерного палива (ВЯП) з дослідницьких реакторів. Обговорюється можливий підхід до вирішення проблеми оцінки потужності залишкового тепловиділення в умовах циклічного режиму роботи установки, тобто після декількох періодів експлуатації і часу простою, у тому числі на різних рівнях робочої потужності реактора. Цей підхід ретельно вивчається у двох можливих математичних моделях розрахунків – щотижневому і щорічному моделюванні вхідних даних для розрахунків. Щоб довести правильність розрахунків, заснованих на річному робочому циклі в математичному моделюванні, були зроблені розрахунки для обох моделей у тих же умовах експлуатації і був підготовлений порівняльний аналіз результатів. Хоча такий підхід забезпечує правильну оцінку експлуатаційних параметрів циклічного режиму роботи установки (рівень потужності, тривалість робочого циклу і час витримки після зупинки), з метою полегшення процесу розрахунків було обрано робочий режим із повторюваними значеннями параметрів в окремих робочих циклах. Відзначається, що значно спрощені розрахунки за річною моделлю дають досить правильні і консервативні результати, на основі яких може бути надана необхідна інформація згідно з вимогами країни, яка повинна прийняти паливо для переробки. На основі цих результатів також можна зробити вибір типу контейнера для транспортування ВЯП. Представлений підхід був використаний для оцінки потужності залишкового тепловиділення у ВЯП дослідницького реактора ІРТ-2000 (Софія, Болгарія) при його підготовці до вивезення в Російську Федерацію в рамках програми RRRFR.

Ключові слова: вивезення відпрацьованого ядерного палива з дослідницьких реакторів, оцінка потужності залишкового тепловиділення, циклічний режим експлуатації.

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МЕТОДИКА ОЦЕНКИ МОЩНОСТИ ОСТАТОЧНОГО ТЕПЛОВЫДЕЛЕНИЯ В РАМКАХ ПОДГОТОВКИ К ВЫВОЗУ ОТРАБОТАВШЕГО ЯДЕРНОГО ТОПЛИВА ИЗ ИССЛЕДОВАТЕЛЬСКИХ РЕАКТОРОВ

Рассматривается метод оценки мощности остаточного тепловыделения при подготовке к вывозу отработавшего ядерного топлива (ОЯТ) из исследовательских реакторов. Обсуждается возможный подход к решению проблемы оценки мощности остаточного тепловыделения в условиях циклического режима работы установки, т.е. после нескольких периодов эксплуатации и времени простоя, в том числе на разных уровнях рабочей мощности реактора. Этот подход тщательно изучается в двух возможных математических моделях

расчетов – еженедельном и ежегодном моделировании входных данных для расчетов. Чтобы доказать правильность расчетов, основанных на годовом рабочем цикле в математическом моделировании, были сделаны расчеты для обеих моделей в тех же условиях эксплуатации и был подготовлен сравнительный анализ результатов. Хотя такой подход обеспечивает правильную оценку эксплуатационных параметров циклического режима работы установки (уровень мощности, продолжительность рабочего цикла и время выдержки после останова), в целях облегчения процесса расчетов выбран рабочий режим с повторяющимися значениями параметров в отдельных рабочих циклах. Отмечается, что значительно упрощенные расчеты по годовой модели дают достаточно правильные и консервативные результаты, на основе которых может быть предоставлена необходимая информация в соответствии с требованиями страны, которая должна принять топливо для переработки. На основе этих результатов также можно сделать выбор типа контейнера для транспортировки ОЯТ. Представленный подход был использован для оценки мощности остаточного тепловыделения в ОЯТ исследовательского реактора ИРТ-2000 (София, Болгария) при его подготовке к вывозу в Российскую Федерацию в рамках программы RRRFR.

Ключевые слова: вывоз отработавшего ядерного топлива из исследовательских реакторов, оценка мощности остаточного тепловыделения, циклический режим эксплуатации.

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