Aspects of Spent MOX-Fuel Transports

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1. INTRODUCTION

With the decision to reprocess spent fuel the recycling of Plutonium (Pu) into thermal reactors was concluded in the Federal Republic of Germany (FRG), too. A demonstration programme was started in 1968 in the small PWR and BWR reactors and continued up to the end of the seventies. The large-scale technical feasibility and the economic use of reprocessed Plutonium has thereby been demonstrated.

The Plutonium recycling programme recommenced in 1981 and has now reached a point where about 50 % of the German reactors are using or are intending to insert Mixed-Oxide (MOX) fuel elements to their cores on a regular basis to replace up to 30 % of the normal reload of UO2-fuel.

Up to now natural uranium has been used as base material for the MOX-fuel elements. In the future reprocessed uranium may be used for MOX-fuel element fabrication. If this latter process is utilized it may be a recommendation that U-236 is extracted during reprocessing to avoid a devaluation of the recovered U-235.

NOTE: Figures and tables are available from the authors.

As NTL plays a major role in transporting spent fuel to reprocessing or storage plants it is of essential interest that NTL studies carefully the MOX-fuel element parameters.

Starting with the manufacturing process and following the irradiation history NTL is in a position to perform reliable calculations and predictions for the safe handling and transportation of spent UO2- and MOX-fuel from the reactors to its eventual destination.

2. MOX-FUEL ELEMENT DESIGN

2.1 General aspects

The MOX-fuel elements have to be designed to be compatible with the UO2-fuel elements in the core such as thermohydraulic, thermal and mechanical design characteristics.The mechanical design parameters (geometry and structural materials) of MOX-fuel elements are identical to UO2-fuel elements with the exception of the fissile material content of the fuel rods.

The main differences between MOX- and UO2-fuel are the following:

- the fabrication process of the fissile material for MOX-fuel is different due to the presence of Plutonium
- the basic physical properties of the pellets (thermal expansion, conductivity, creep) are different
- the power distribution is changed across the fuel section due to the different neutronic properties
- the fuel chemical behavior is modified (different oxygen potential of PuO2)
- the fuel microstructure (porosity distribution, grain size) varies to a certain extent.

2.2 Impact on safety

For the reactor operator this question is mainly related to the impact of safety during operation and storage of the MOX-fuel elements. It concerns:

- the relevant parameters of the core physics during reactor operation
- the fuel storage pond in respect to criticality aspects
- reactor accident and transient analysis
- activity inventory and release
- decay heat and heat transfer
- radiological and health physics aspects during handling and storage.

MOX-and UO2-fuel elements are only compatible to a certain extent in relation to their nuclear properties as the initial composition of the fissile material is different resulting in differences in the neutronic behavior and activity inventory after irradiation which needs to be considered for shielding , decay heat and dose rate evaluations.

The above mentioned facts are of major importance to NTL for determing the right type of flask for the fuel to be loaded.

It can be summarized that the studies about the impacts of MOX-fuel on safety during operation, transport and reprocessing have proved that this type of fuel can be treated very similar to UO2-fuel taking into account the differences.

2.3 Fabrication of MOX-fuel elements

This section briefly describes the present situation in Germany. For the MOX-fuel of the first generation PuO_2-powder from the oxalate process was mechanically mixed with normal UO_2-powder. During reactor performance the fuel rods containing the MOX-fuel pellets showed no differences to the UO_2-fuel elements but during the reprocessing of these MOX-fuel elements residual Plutonium was found in the pure nitric acid thus demonstrating insolubility of the Plutonium.

To meet the solubility requirements of the reprocessing plants the AUPuC-Process and the OKOM-Process were developed as shown in Figure 1. The solubility of the MOX-fuel is greater than 99 % as specified.

Plutonium in form of nitrate is available from the German Reprocessing Plant (WAK) at Karlsruhe.

Presently more than 80 % of the Plutonium supply is received from COGEMA at La Hague. This Plutonium is supplied in form of PuO_2-powder.

Compared to UO_2-fuel the requirements for the MOX-fuel fabrication are considerable higher, for example up to 3 different enrichment zones per element, uniformity of the isotopic composition of the fissile material, homogenization of blendings etc. are required.

To meet these requirements additional controls are performed to verify:

- Plutonium isotopic composition by mass-spectrometry
- Americium (Am) content by gamma-spectrography

- Plutonium dispersion by alpha-autoradiography
- Rod gamma-scanning.

The following examples (Fig. 2 to 5) show typical MOX-fuel element design schemes for PWR reactors.

It has been proven that unirradiated MOX-fuel rods loose about 1 % of their reactivity value per year of storage as Pu-241 is decaying into Am-241.

3. MOX-FUEL BEHAVIOR DURING IRRADIATION

The following parameters during irradiation are of particular interest for the transport of spent MOX-fuel:
- Activity inventory
- Number of fuel cycles (final burn-up) and out of core storage time
- Decay heat and time of storage prior to transport
- Physical behavior during irradiation (integrity).

3.1 Activity inventory

Calculations for comparing the activity inventory of UO2- and MOX-fuel (having approximately the same irradiation history) show that the maximum activity inventory of the MOX-fuel after irradiation (at the reactor shut-down) is smaller than for UO2-fuel.

The alpha-activity of the actinides resulting mainly from the decay of Am-241 and the resultant Curium (Cm-242) is considerable higher (about a factor of 3) compared to UO2-fuel and it is increasing with the build-up of Am-241, Am-243, Cm-242 and Cm-244.

Because of the build-up of Cm-242 and Cm-244 the neutron emission is also increased significantly.

The activity of the fission products such as the isotopes of Krypton (Kr) and Strontium (Sr) is lower compared to UO2-fuel, the activity of Iodine (I-131) and of the Caesium (Cs) isotopes is similar.

In total the activity inventory of the fission products in spent MOX-fuel is smaller compared to UO2-fuel.

3.2 Number of fuel cycles (final burn-up) and out of core storage time

The final burn-up of the MOX-fuel is of particular interest for the transporter as this value is the basic information for calculating the activity inventory of the fuel elements. Under normal circumstances the average burn-up per flask loading is taken into account which has now reached a figure of about 40 GWd/t for MOX-fuel and this does not seem to be an ultimate figure as many reactors have started using the fuel during 4 and even 5 irradiation cycles.

The increasing burn-up will necessitate longer cooling times for the spent fuel elements at the reactor site to be in compliance with the limits of the transport flasks in respect to decay heat and dose rate.

For reactors having large capacity storage ponds (compact racks) there is no real problem to keep the fuel elements longer at the site. But for older reactors with a small storage capacity there may be difficulties.

3.3 Decay heat and time of storage prior to transport

The decay heat curve of the MOX-fuel does not follow closely the decay heat curve of the UO2-fuel.

Even during longer storage time (over many years) the decay heat of MOX-fuel elements is only slightly decreased because of the build-up of Cm-244.

That means in practice longer cooling time may not always solve the problem in respect to decay heat.

3.4 Physical behavior during irradiation (integrity)

It can be stated that there are no remarkable differences during irradiation in respect to the physical behavior of MOX-fuel compared to UO2-fuel. Up to now only sound MOX-fuel was transported by NTL to the reprocessing plants. The criteria applied to verify the integrity of the MOX-fuel elements prior to transport are basically the same as for UO2-fuel elements.

4. TRANSPORT MANAGEMENT OF SPENT MOX-FUEL

4.1 General considerations

The following aspects have to be carefully evaluated before any transport can be carried out:

Do the fuel elements to be transported meet the criticality and soundness requirements, the limits in respect to decay heat and activity?

Can the permitted dose rate limits be met in respect to the type of flask to be used?

The regulatory limits as stipulated by the international IAEA-regulations for the transport of dangerous goods (IAEA Safety Series No. 6, etc.) form the basis for the flask safety analysis report and corresponding supplements.

NTL has over nearly two decades accumulated a wealth of experience and a considerable amount of practical and substantiated analytical data of spent fuel transports for small flasks (40 t) up to big flasks (115 t) carrying up to 6 t of irradiated fuel.

NTL has established a sophisticated data bank and is applying the FAKIR computer code, which has been developed in the recent years. NTL is in a position to predict the expected decay heat, activity and dose rate for each individual transport (wet and dry type flasks) with good accuracy.

For MOX-fuel element transports a modified FAKIR-computer code is under development and will be available very soon.

It is the future intention of NTL to modify the FAKIR-computer code in such a way that it can be applied to UO2-and MOX-fuel elements in one computer run.

To substantiate the FAKIR development and to provide assurances that all transport requirements are met, NTL has its own computer system which uses internationally recognized calculational methods and data sets.

4.2 Transport performance

Up to now NTL has totally performed more than 900 transports of spent fuel from the FRG where about 20 shipments were loaded with MOX-fuel elements.

9 transports contained only MOX-fuel elements and the other transports were carried out as mixed transports of UO2-and MOX-fuel elements together. The following figures (Fig. 6 to 8) show the mode and number of transports performed by NTL from Germany (domestic and abroad).

To meet the requirements, in particular the dose rate limits, NTL has established selective loading pattern for individual shipments as it is shown on Figure 9.

Depending upon the fuel parameters to be transported NTL calculates the optimized loading pattern in respect to decay heat, activity and dose rate.

In special cases where there are no sufficiently long cooled UO2-fuel elements available for loading with MOX -fuel elements or only MOX -fuel elements have to be transported the provision of additional shielding is taken into account.

There are two possibilities for realization:

- Fitting additional shielding material to the transport vehicle

- Using a jabroc cover fitted to the flask (see Fig. 10).

It is the standard practice now to transport the MOX-fuel elements together with the UO2-fuel elements. The MOX-fuel elements are preferably positioned into the centre compartments of the fuel frame of the flask.

The UO2-fuel elements around the MOX-fuel elements lead to an effect of additional gamma shielding.

5. EXPERIENCE WITH TRANSPORTS OF SPENT MOX-FUEL

As the number of transports with spent MOX-fuel elements is small compared to the number of UO2-fuel element transports, NTL is collecting all information about the MOX-fuel history and transport performance.

The majority of transports with MOX-fuel elements have been performed with wet type of flasks but this is changing now as transports of MOX-fuel elements from the big 1300 MW-reactors are commencing in big dry type of flasks.

Fig. 11 and 12 show dose rate surveys of transports performed with only MOX-fuel or mixed with UO2-fuel elements. It is interesting to note that the predicted dose rate values were in good compliance with the measured values. Furthermore the dose rate values for comparable UO2-fuel transports are listed. The results show clearly that the neutron emission of MOX-fuel is considerable higher than the one for UO2-fuel. This is at least valid for the complete MOX-loading case. The factor is depending upon the burn-up and cooling time of the fuel elements.

6. FURTHER DEVELOPMENTS AND TRENDS

To meet the requirements given by the regulations and the needs of the clients transports of fuel with high burn-up and in some cases after a short cooling time will have to be performed. There are even several possibilities that can be taken into consideration:

- Design of new type of flasks for MOX-fuel elements.

- Design of new fuel frames for existing type of flasks. NTL is at present designing a new fuel frame to meet this demand.

  As a consequence the payload of the flasks may be decreased. The possibilities of arranging the fuel element compartments in a typical NTL- flask for PWR-fuel elements are demonstrated in Figure 13. The preliminary calculation results are predicting that version b is the better configuration.

- Mixing of MOX-fuel elements with longer cooled UO2-fuel elements using standard flask types.
  This solution may not always be possible in future as the situation at some reactors may require transports after short cooling periods (6 to 9 months).

The present trend in Germany in respect to the use of Plutonium from reprocessing is clearly directed to use it as MOX-fuel (up to 1/3 of a reactor reload) in both reactor types (PWR and BWR).

As the commercial reprocessing plants are now accepting MOX-fuel elements the number of transports containing MOX-fuel will increase steadily in the future. Therefore it is of great importance that all parties in the back-end of the fuel cycle are well informed in advance about the fuel parameters for correct preparation of these shipments. NTL has always considered safety and security as the major aspects for the performance of nuclear transports.

7. CONCLUSION

As recycling of Plutonium can now be considered as a standard technique, the number of transports with spent MOX-fuel is increasing from year to year.

Although in general the MOX-fuel elements are identical to the mechanical design of the UO2-fuel elements, the fissile content of the MOX-fuel elements is different. Therefore the content of fission products (gamma emission) and actinides (alpha and neutron emission) varies considerably depending upon the burn-up and the cooling time.

These together with the radiological considerations need to be taken into account when preparing transport campaigns.

NTL is establishing methods and using internationally recognized computer codes to ensure that shipments containing MOX-fuel elements meet the regulatory requirements.

At present the existing fleet of flasks is able to cope with the identified number of MOX-fuel elements.

The future trend in MOX-fuel element irradiation and continued acceptance of MOX-fuel elements will determine whether new type of flasks will be necessary to meet the increasing amount of transports to the reprocessing or storage plants.

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