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THE KFA/NUCLEBRÁS R+D PROGRAM ON TH–UTILIZATION IN PWR

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TH-UTILIZATION IN PWR

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INTRODUCTION

The utilization of thoria fuel in thermal reactors, due to the
high η value of U-233, brings about better resource utilization
compared with conventional uranium fuel. This particularly holds
for advanced reactor types specifically designed for thoria
application, but in light water reactors (LWRs) also, a substantial
advantage is anticipated for a closed fuel cycle. The closing of
the fuel cycle is a prerequisite for any long-term use of thorium
in thermal reactors. The quickest and easiest way to demonstrate
the feasibility of the complete fuel cycle is to prove this on
first generation commercial reactors.

To utilize the available know-how and research capacities, a joint
cooperative program was initiated between KFA and NUCLEBRÁS, with
the participation of KWU and NUKEM on the German side. The
objectives of the program, which has run since mid-1979, are as
follows:

1. to analyze and evaluate thorium utilization in pressurized
   water reactors (PWRs)

2. to design the PWR fuel element and core for the thorium fuel
   cycle

3. to manufacture, test, and qualify Th/U and Th/Pu fuel elements
   under operating conditions

4. to study the closing of thorium fuel cycle by reprocessing of
   spent thorium-containing PWR fuel elements.

The project utilizes as much as possible existing technologies
and available know-how for core design, fuel element fabrication,
and reprocessing. Therefore, the existing techniques and equipment
for the fuel cycle in high-temperature reactors (HTRs), i.e.,
manufacturing and reprocessing of fuel, as well as for the fuel cycle

in LWRs, i.e., fabrication and head-end treatment, will be applied. Advanced methods and processes (pellet fabrication from fuel kernels) will render an essential potential for employment. In the long term, the program could be extended to incorporate advanced methods comprising all steps of the fuel cycle, such as the fuel design, manufacture, and disposal.

STATUS OF WORK

The first activity comprised a review of the accessible literature about thorium utilization in LWRs (Ref.1). The evaluation has shown that a good base of knowledge is available for fuel characterization methods and, to a lesser extent, for the general behaviour of Th/U mixed-oxide fuel under irradiation. No statements can be made with regard to the properties depending directly on the manufacturing process, such as fission gas release, pellet-cladding interaction (PCI), accident behaviour, etc.

Strategy and Nuclear Core Design Calculations. Investigations on the thorium fuel cycles in PWRs have been performed with the same theoretical methods as are used for the U/Pu fuel. An outstanding task in the initial program phase was the modification and validation of standard PWR-related codes for calculation of thoria-based fuel.

Testing was done by analysis of Brookhaven National Laboratory exponential experiments.[2,3] When comparing the calculated and measured ratio of epicadmium to subcadmium capture in Th-232, we see that the results of the Th-FASER code agree within the incertainty limits, especially in the region of moderator-to-fuel ratio of main interest. The calculation of k_{eff} with measured buckling showed an underprediction of 1 to 2%. We follow the opinion of Ref.3 that this is caused mainly by experimental data and by zero-dimensional calculations. Further improvements require explicit calculation of resonance interferences. Work in this direction should be done in parallel with nuclear design work using Th-FASER. The present version of Th-FASER has been used to generate two-group libraries for MEDIUM core calculations.

The three-batch Th/HEU once-through and self-generating recycle (segregated and mixed) reload schemes for a KWU 1300-MW(e) standard PWR have been investigated. Calculations for a four-batch core have also been carried out. The overall core calculations have confirmed that no modifications need to be done for thorium fuel when incorporated in a standard PWR fuel element. For the Th/Pu cycle, only the once-through reload scheme in a three-batch full Th/Pu core has been investigated up to now. Tables I and II summarize the information about the fuel cycles. First results show that a full Th/Pu core can be realized with standard PWR fuel elements. Preliminary calculations related to the safety analysis of both Th/HEU and Th/Pu, based in reactivity balance calculations, have shown sufficient control margin.

TABLE I

Characteristics of Once-Through Fuel Cycles

		Th/HEU-Fuel 3 batch core	Th/Pu-Fuel 3 batch core
Enrichment of first core	w/o U-235 or w/o Pu_{fiss}	2.9 (69 FAs)[a] 3.6 (68 FAs) 4.2 (64 FAs)	2.8 (65 FAs) 3.6 (64 FAs) 4.4 (64 FAs)
Enrichment of reload batches	w/o U-234 or w/o Pu_{fiss}	4.2 (64 FAs)	4.4 (64 FAs)
Boron concentration First core (BOC) Equilibrium core (BOC)	ppm	1461 1177	2754 1374
Cycle length First core Equilibrium core	fpd	330 290	317 267

[a]FA = fuel assembly.

TABLE II

Fuel Composition of Burnup-Equivalent Once-Through and First Recycling Generation Fuel Assemblies (FAs)

Th-MEU, Once-through Fuel	Loading w/o	Unloading w/o
Th 232	95.48	96.34
U 233	-	1.46
U 234	-	0.17
U 235	4.20	1.15
U 236	-	0.55
Np 237	-	0.00
U 238	0.32	0.28

Reload batch n_FA = 44, 4.2 w/o U235 (MEU)
Burn up: 35.5 MWd/kg_HM

Th-U, Segregated Recycling Fuel	Loading w/o	Unloading w/o
94.82	96.28
1.89	1.69
0.23	0.33
1.85	0.83
0.79	0.81
	0.00
0.42	0.34

Reload batch: 44 FA of 3.75 w/o U_fiss,rec. + 20 of 4.2 w/o U235 (MEU)
Burn up: 33.9 MWd/kg_HM

Th-U, Mixed Recycling Fuel	Loading w/o	Unloading w/o
95.05	95.98
1.24	1.68
0.15	0.33
2.68	0.83
0.52	0.81
-	0.00
0.38	0.34

Reload batch n_FA = 64, 3.905 w/o U_nis
Burn up: 34.0 MWd/kg_HM

Th/Pu-fuel cycle

Th/Pu fuel, Pu extracted from U-FA

	Loading w/o	Unloading w/o
Th 232	93.93	95.62
U 233 + Pa 233	-	1.32
U 234	-	0.30
U 235	-	0.02
U 236	-	-
U 238	-	-
Pu 239	3.65	0.64
Pu 240	1.41	1.12
Pu 241	0.75	0.74
Pu 242	0.25	0.44

Burn up: 34.6 MWd/kg_HM

Th-Pu/U_rec fuel without adding Pu

make up extracted from U-FA

	Loading w/o	Unloading w/o
93.51	95.11
1.99	2.18
0.16	0.36
0.07	0.08
-	-
-	-
0.90	0.10
1.66	0.75
1.09	0.60
0.67	0.04

Burn up: 33.8 MWd/kg_HM

Fuel fabrication. For the fabrication of $(Th,U)O_2$ fuel pellets, a combination of two fully developed fabrication routes is used: manufacturing of $(Th,U)O_2$ microspheres by the method of external gelation, and pellet fabrication by the method of direct pelletizing.

The ex-gel conversion has been developed and is now used for the production of HTR fuel.[4] The direct pelletizing as an economically and technically advanced process is utilized in the manufacturing of uranium-based LWR fuel.[5] Both processes have a high degree of standardization and show a broad basis of production experience. The combination of both is used for Th-based LWR fuel production according to the flowsheet in Fig. 1.
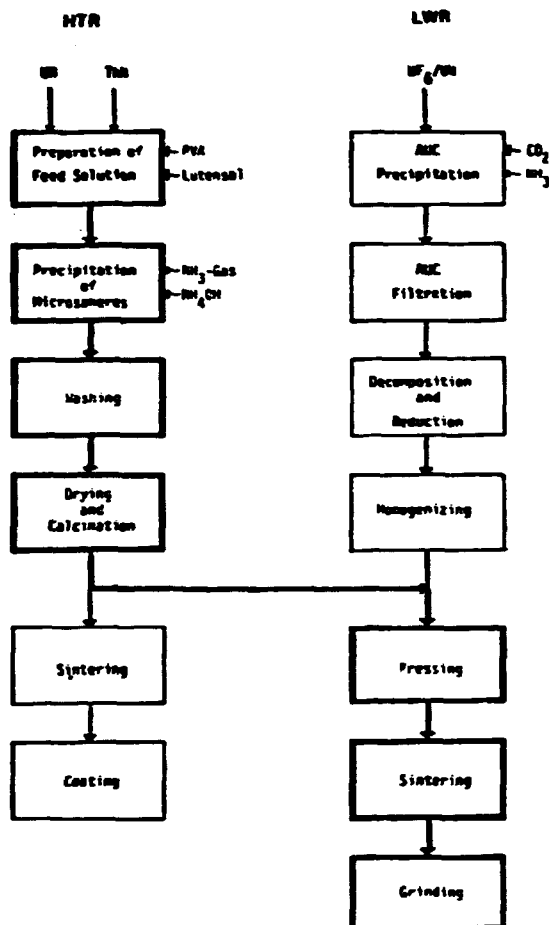
Fig. 1. Combination of available technologies for the manu-
facturing of mixed-oxide fuel.

The ex-gel fabrication leads to $(Th,U)O_2$ microspheres, some properties of which are very advantageous for the direct pelletizing, as was shown in laboratory-scale investigations:

1. The two oxides are in a coprecipitated state; that means that there is no disturbing influence of the formation of solid solution during sintering, as is observed sometimes in mechanically mixed oxides.

2. Form, size, and morphology of the microspheres guarantee an easy handling (excellent flowability, for example).

3. The sinterability of the microspheres is very good.

4. The dust generation during processing is low.

To reach the required pellet quality, the adaptation of some of
the microsphere's properties and, to a lower extent, of the
pelletizing parameters also is necessary. This relates mainly
to the adjustment of a suitable pellet microstructure, which
requires a lowering of the generally high mechanical strength
of the microspheres. By optimizing the heat treatment after
washing, a sufficiently low mechanical strength was obtained.
In addition, an adaptation of the precipitation conditions has
been performed to assure a low content of residual organic
material in the press-fit microspheres; beyond that, a special
low-temperature treatment before drying has been developed for
the same reason.

With respect to the pressing of microspheres, no essential changes
of the highly standardized LWR pressing process were found to be
necessary. A good lubrication of the die wells has to be provided
for. The use of lubrication additives admixed to the kernels has
a beneficial effect on the green and sintered density, but is not
adequate for a good microstructure. Higher pressing forces and,
consequently, higher green densities than in UO_2 production have
been found to be suitable for sufficient sintered density and
good microstructure. Because of the high sinterability of the
kernels, reduced sintering temperature of - 1200 to 1400°C are
sufficient. As a sintering atmosphere, reducing, inert, and
even oxidizing gases, e.g., air, can be used. This is due to the
excellent phase stability of $(Th,U)O_2$ up to 50 mol% UO_2 and at
high temperature. These investigations resulted in the laboratory
scale fabrication of $(Th,U)O_2$ pellets that fulfill all LWR-
specific requirements.

Fuel Performance Prediction and Qualification. For the prediction
of the irradiation behaviour, a consistent data set for $(Th,U)O_2$
has been developed on the basis of a literature review. Most of
the data are available, or can be calculated from the values of
pure ThO_2 and UO_2. Physical and thermal properties of $(Th,5\%U)O_2$
solid solutions were developed for adapting them to the CARO
computer code. Models have also been developed for ThO_2 and UO_2
elastic moduli. The dimensional behaviour of $(Th,5\%U)O_2$ fuel has
been calculated on the basis of the densification model described

in Ref.6. The results are as follows:

1. The swelling rate of $(Th,U)O_2$, is ~ 50% higher than that of pure UO_2.

2. The prediction of densification behaviour is not yet very reliable, but there is an indication of a smaller densification rate in $(Th,U)O_2$ than in pure UO_2.

These results confirm the former qualitative statement that the pore structure of the ex-kernel should be optimized. CARO code calculations on $(Th,U)O_2$ fuel show the following:

1. The maximum temperature of $(Th,5\%U)O_2$ PWR fuel is expected to be more distant from the melting point $(3250^{\circ}C)$ than that of UO_2 fuel $(2800^{\circ}C)$.

2. $(Th,5\%U)O_2$ fuel rods have less stored energy than UO_2 rods under corresponding conditions.

An in-pile fuel qualification will be performed by an irradiation test of six $(Th,U)O_2$ fuel rods, fabricated under pilot plant conditions, in FRJ-2 in Jülich. Start of irradiation is envisaged for the spring of 1983.

Reprocessing Studies. A literature search of all aspects of the back end of the thorium-uranium fuel cycle has been completed. The progressing evaluation of the literature indicates lack of knowledge in some areas, particularly in the head end. Cold laboratory tests in dissolution and solvent extraction are proceeding.

The dissolution experiments with cold $(Th,U)O_2$ pellets from an early development stage have shown a high dissolution rate of this fuel in Thorex solution. The dissolution of Zircaloy cladding tubes by Thorex solution has been identified as a main specific problem of reprocessing $(Th,U)O_2$ LWR fuel, since the expected zirconium concentration in the feed solution is about three times higher than of the UO_2 fuel.

CONCLUSIONS

The cooperative Brazilian/German thorium utilization in PWRs program creates a technological basis for industrial application of thorium fuel cycles in PWRs. By choosing an advanced fuel fabrication route that contains several features favorable for refabrication purposes, a spinoff for other cycles as well can be expected.

By 1983, a statement toward technological feasibility of Th/U ex-kernel fuel will be available. After that, an investigation of more detailed aspects connected with the reactor-related fuel performance and investigation of technological feasibility of Th/Pu fuel,as well as hot reprocessing experiments, will follow. The final goal is a fuel element test in a currently running PWR.

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