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Impact of extended burnup on the nuclear fuel cycle

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FOREWORD

Currently, burnup extension is one of the most advanced and intensively studied areas in fuel technology and management worldwide. The average design discharge burnups that are commercially available for BWRs and PWRs are in the range 35-45 MW·d/kg U and 40-50 MW·d/kg U, respectively. Economic incentives may exist for extending burnup even further, to at least 60 MW·d/kg U.

Burnup extension affects several important stages of the fuel cycle and concerns the whole nuclear fuel cycle industry. Increase of burnup reflects on the requirements for natural uranium, enrichment and fuel fabrication, reactor core configuration and its control, fuel performance and the back end of the fuel cycle, such as spent fuel handling, transportation, treatment and storage.

In view of the importance of high burnup for water reactor fuel utilization, the IAEA convened technical committee meetings (in 1981, 1984 and 1990) in the framework of the International Working Group on Water Reactor Fuel Performance and Technology which consists of representatives from 24 countries and three international organizations. Additionally, around 50% of the presentations at the IAEA Symposium on Improvements in Water Reactor Fuel Technology and Utilization in 1986 were related to subjects on high burnup.

The Water Reactor Fuel Extended Burnup Study (WREBUS) started in 1988 and was completed in 1991. It was the first internationally conducted study of its kind involving eleven IAEA Member States. The findings of the study have been published as IAEA Technical Reports Series No. 343 in 1992.

These proceedings present national approaches to extended burnup and experience in the subject, an evaluation of the state of the art and outline future trends in the field.

The IAEA wishes to thank all the participants of the Advisory Group Meeting for their valuable contribution and especially the Chairman, Mr. P. Lang, and the Session Chairmen, Messrs. P. Duflou, J. Griffiths and K. Hesketh. The IAEA officer responsible for the organization of the meeting and preparation of this document is Mr. G. Sukhanov, Nuclear Materials and Fuel Cycle Technology Section, Division of Nuclear Fuel Cycle and Waste Management.

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SUMMARY OF ADVISORY GROUP MEETING

1. INTRODUCTION

The Advisory Group Meeting was held in Vienna from 2 to 5 December 1991, to review, analyse, and discuss the effects of burnup extension in both light and heavy water reactors on all aspects of the fuel cycle. Twenty experts from thirteen countries participated in this meeting. There was a consensus that both economic and environmental benefits are driving forces toward the achievement of higher burnups and that the present trend of burnup extension may be expected to continue. The economic aspects for LWRs have been assessed and the results published in the IAEA Technical Reports Series No. 343 in 1992.

Batch average burnup levels were considered to be feasible goals, achievable within the next 10-20 years. For light water reactors burnup would be extended to 55-60 GW·d/t U for PWRs and WWER 1000 reactors and to 45-50 GW·d/t U for BWRs and WWER 440 reactors. For heavy water reactors burnup could be extended to 12-20 GW·d/t U. The extended burnup has been considered for the three main stages of the fuel cycle: the front end, in-reactor issues and the back end.

2. IMPACT OF EXTENDED BURNUP ON THE FRONT END OF THE FUEL CYCLE

The following potential effects of extended burnup on the front end of the fuel cycle have been identified and discussed:

- impact on resource utilization,
- impact of higher enrichments on front end facilities and operations,
- impact of necessary fuel design modifications on front end facilities,
- in the case of closed cycles, impact of extended burnup on the quality of plutonium and uranium recovered after reprocessing and on their subsequent utilization.

2.1. Impact of extended burnup on resource utilization

The extention of burnup leads to a significant decrease of specific uranium consumption (expressed in grams of uranium used per MW·h of generated power) and to a reduction of fuel assembly specific consumption (number of assemblies to be fabricated for a given energy produced). For LWRs, the magnitude of the decrease of uranium consumption depends on the way the burnup extension is carried out: it is more effective in the case of unchanged cycle length than when burnup and cycle lengths are simultaneously increased.

The effect on consumption of separate work units [SWUs] (number of SWUs used per unit of energy generated) is less clear (the change can be either or slight increase or decrease according to the burnup change) but estimates are that it would be small.

The issue of impact on resource utilization has been treated on a quantitative basis for LWRs in Technical Reports Series No. 343.

No complementary work has been found necessary on this subject in the short term.

2.2. Impact of higher enrichment on front end facilities

2.2.1. Light water reactors

The use of higher U-235 enrichment for use in LWRs has possible impacts upon safety for enrichment plants, fuel fabrication plants, transport containers and fresh fuel storage facilities.

This may require the expenditure of time and effort in documentation and minor modification in order to meet the licensing requirements, but essentially there were no generic problems envisaged. Most of the existing enrichment and fuel fabrication facilities are already licensed for enrichments up to 5% U-235 which is consistent with the framework of extended burnup under consideration. This conclusion remains valid bearing in mind that in defining the enrichment limit required, the necessary safety margin to allow for uncertainties needs to be considered and that when an average enrichment level is stated, some fuel management strategies may require enrichments above and below that average.

There may be generic issues related to adapting front end facilities to enrichments much greater than 5% U-235, however within the framework of the range of extended burnup considered by this group, such issues were not considered relevant.

2.2.2. Heavy water reactors

One option for achieving burnup extension for HWRs is the use of slightly enriched uranium (SEU) in place of natural uranium. In such a case, the current license limits for natural uranium fuel fabricators would need to be reviewed. In addition, typical storage and transport designs and procedures for fresh fuel have to be reviewed to verify criticality aspects.

An alternative way of obtaining SEU is to utilize reprocessed uranium from LWR fuels. This could imply modifications of fabrication facilities and special QC and dose control procedures in order to cope with the associated activity.

2.3. Impact of fuel design changes required for extended burnup on front end facilities

Extended burnup may require changes to fuel rod and assembly designs, and to fuel management strategies (see Section 3). The impact upon front end facilities arises from:

(a) The use of integral burnable absorbers for reactivity control;
(b) Changes in materials and design for fuel rods and assemblies.

Such new requirements could imply additions and modifications in current manufacturing processes and QC procedures. However, the participants did not identify any major problems associated with achieving these requirements. For some plants no changes are needed and for other plants changes have already been made.

2.4. Impact of extended burnup on the quality of plutonium and recycled uranium and on associated facilities

Extended burnup can change significantly the isotopic composition of the plutonium and uranium recovered after reprocessing, enhancing their radioactivity and increasing the content in neutron absorbing isotopes.

It is felt that such degradation of the quality of recycled uranium could require additional protective measures in the front end facilities where the material is treated before being reused in a reactor (enrichment facilities, manufacturing chains devoted to the use of REU, transport containers). However no need for significant additional modification of such facilities is anticipated for the envisaged burnup extension frame. The situation could become more critical in case of multiple recycling of the same material; however this is clearly not a short or medium-term issue.

In the case of recycling plutonium in MOX, the effect of the degradation of quality may give rise to significant dose uptake issues with possible implications on the plant design, unless remote operation and eventually maintenance are realized. This needs further consideration.

Attention should also be paid to possible impact of the presence of such active materials on decommissioning methods or cost of the relevant facilities.

Another issue is that the higher level of absorbing isotopes in the recycled materials leads to consequent even higher enrichments for LWRs, which in turn result in additional licensing problems.

3. IN-REACTOR ISSUES

3.1. Nuclear design

3.1.1. Impact on core characteristics

In all reactor types, extended burnups are achieved by higher initial fissile isotope concentrations. The increased thermal neutron absorption and the associated hardening of the neutron spectrum combine to influence the core physics characteristics in a number of ways. At the fundamental level main considerations are: firstly, fission products tend to decrease reactivity with increasing burnup, which may be offset by using burnable absorbers. One potential impact of that is to reduce shutdown margins. Secondly, the reactivity coefficients are modified. Thirdly, especially in the absence of burnable absorbers, the reactivity difference between new fuel and previously burned fuel is increased with extended burnups, potentially causing increased radial peaking factors.

The impact of extended burnup operation on the nuclear design characteristics are complicated by many factors. A detailed discussion of these factors is outside the scope of this summary, but an important point to note is that there is a tendency to modify the fuel management schemes and loading patterns at the same time as going to high burnups and this complicates the discussion. It is important to bear in mind this point when examining the impact of high burnups. The impact on loading patterns is discussed further in Section 3.1.3.

3.1.2. Impact on nuclear fuel design

The principal impact on the nuclear design of extended burnups and associated fuel management changes is the need to use higher enrichments. In LWRs this is associated with larger reactivity investments at the start of a fuel cycle which leads in turn to a requirement for more reactivity hold-down from control rods, soluble boron or burnable absorber. In PWRs and WWERs, the higher beginning of cycle hold-down requirements may result in the need to use burnable absorbers to ensure that the beginning of cycle moderator temperature coefficient remains within the design limit. BWRs may also require increased burnable absorber loadings, due to the limited reactivity hold-down that the control rods can achieve.

With respect to burnable absorbers the higher loadings needed with extended burnups gives an increased incentive to use integral fuel absorber designs. There are many integral fuel absorber designs available at present. In PWRs boron coatings and gadolinia are used extensively at present. In BWRs gadolinia is used exclusively. Both these types and future variants are suitable for use with extended discharge burnups. The use of integral fuel burnable absorbers impacts on the fuel management and fuel performance and is discussed further in Section 3.3.

There is a general tendency at present to increase the number of fuel rods in PWR and BWR assemblies and HWR bundles, a trend which exists independently of the trend towards extended burnups. This will increase thermal design margins, so that within-assembly peaking factor constraints may be relaxed. Thus the trend towards extended burnups will not necessarily result in any additional increase in the complexity of nuclear fuel designs.

3.1.3. Impact on fuel management strategies and loading patterns

The trend towards extended burnups tends to be accompanied by an increase in complexity of the fuel management strategies. Along with the trend towards extended burnups many LWR utilities are inclined to adopt low leakage loading patterns. These have several advantages, such as reduced vessel fluence and improved fuel utilization, but generally require more sophisticated core loading patterns. Thus, the radial peaking factors tend to be higher in low leakage cores, necessitating the use of burnable absorbers. The use of low leakage loading patterns in conjunction with extended burnup operation can thus lead to higher burnable absorber requirements. With respect to vessel fluence, extended burnup operation can be advantageous because the opportunity arises to use the highest burnup fuel assemblies to shield the reactor vessel.

3.1.4. Operational flexibility

An important consideration of adopting extended discharge burnup operation is that the operational flexibility available to the plant operators may be affected. Thus in LWRs the higher radial peaking factors associated with extended discharge burnup cycles, especially in conjunction with low leakage loading patterns, may reduce the margins available to thermal design limits and, depending on the specific details of a particular plant, may give rise to reduced operating flexibility.

Fuel management strategies will aim to keep within existing constraints, so that the impact of extending discharge burnups on operational flexibility is minimized.

3.1.5. Extension of codes to extended burnup conditions

Computer codes developed and validated for core calculations need to be extended and qualified for extended burnup conditions, to ensure that calculations done with such codes do not represent extrapolations but are normalized against data and therefore can be considered fully reliable for future extended burnup applications.

3.2. Fuel performance

3.2.1. Corrosion

The issue of clad (external) corrosion in high discharge burnup operation was identified as being the most important fuel performance issue for PWRs, particularly in plants with high coolant temperatures. Although clad corrosion is a potential concern for other reactor types, it was not seen to be a significant issue at present. Thus the different clad material

used in WWER reactors, based on an alloy of Zr and Nb shows exceptionally good corrosion performance. Similarly, in BWRs clad corrosion is not a major issue. The choice of clad material in relation to PWR clad corrosion is discussed in Section 3.3.2.

3.2.2. Fission gas accumulation and release

With extended burnups there is a greater accumulation of fission gases and more opportunity for diffusion processes to release the gas from the fuel pellets. However, this does not necessarily result in higher partial pressures from fission gas release as there is a trend in BWRs and HWRs towards increasing the number of fuel rods per assembly and therefore reducing the linear generation rate independent of the trend to extended discharge burnups. However, with a fixed number of rods per assembly, it is certainly the case that higher fission gas releases are obtained with extended burnups and this may make it necessary to introduce rod design changes, as discussed in Section 3.3.1. The increased fission gas release will give additional incentive to improve the fission gas release predictive capabilities of fuel performance codes.

An important point regarding the impact of extended burnups is that although the inventory of long lived fission products is increased, the inventory of short lived fission products is essentially unaffected, as they are at equilibrium. The impact of high discharge burnups on the short term radiological consequences of accidents, being dominated by the short lived volatile fission products, is therefore minimal.

At extended burnups there may be advantages in moving away from the present LWR design requirement that fuel rod internal pressures must not exceed the coolant pressure. It might prove feasible to replace this deterministic design criterion with a probabilistic argument and at high burnups there would be more incentive to implement such a change.

A more complete account of the effect of extended discharge burnups on fission gas release can be found in the reports of the IAEA meeting on fuel performance held at Studsvik and ANS/ENS LWR Fuel Performance Conference held at Avignon, April 1991.

3.2.3. Pellet clad interaction

While the trend towards extended burnups in most cases has no significant impact on the susceptibility of fuel clad to damage from pellet clad interaction, there is nevertheless an indirect impact resulting from the fuel management changes associated with high burnups. Loss of operational flexibility in connection with load follow operation in PWRs was identified as an area of particular concern in some countries.

With regard to BWRs and HWRs, however, the trend towards higher ratings is offset by the trend towards increasing the number of fuel rods per assembly or bundle making this consideration of less importance.

Possible materials changes to improve PCI performance are discussed in Section 3.3.2.

3.2.4. Fuel reliability

Available evidence points to there being no adverse impact of extended burnup operation on fuel reliability. Indeed all fuel is already designed and licensed for operation well below the levels at which life limiting phenomena occur. Moreover there is some evidence of there being a possible benefit. This can be understood from the reduced rate at which fresh feed assemblies are loaded in extended burnup operation, giving reduced risk of fuel defects related to manufacturing defects.

3.3. Mechanical design

3.3.1. Structural design impacts

As mentioned previously, there is a trend to increase the number of fuel rods in PWR, BWR and HWR assemblies. This decrease in the fuel linear power and temperature allows an increase in burnups. Other design changes can be also used e.g. to adopt a hollow pellet which decreases the fuel center temperature and therefore the FGR and the plenum volume.

3.3.2. Impact on materials

Clad external corrosion is a limiting factor for PWRs at the highest burnups and highest coolant temperatures. There is therefore a strong incentive to develop corrosion resistant cladding materials. Possible solutions currently under consideration or development are variants on the current Zircaloy alloys, duplex claddings in which a low corrosion alloy coats the outer surface of Zircaloy 4, thus obtaining a corrosion resistant material while retaining the bulk properties of Zircaloy 4.

With respect to pellet clad interaction and the underlying stress corrosion cracking mechanism, which is generally believed to promote crack growth, various materials properties developments are being pursued. The use of softer fuel pellets (such as niobia doped pellets), possibly in conjunction with hollow pellets is another approach being considered. The use of zirconium lined cladding is a well established means of enhancing pellet clad interaction margins in BWRs. Graphite coated clad is used in HWR bundles and zirconium liner clad is also being considered for HWR assemblies and bundles.

3.4. Thermalhydraulics

In relation to extended discharge burnups, there are no direct impacts on the thermalhydraulics. Assembly design changes introduced in connection with extended discharge burnup operation or otherwise, must be evaluated against thermal hydraulic compatibility considerations. Design changes in this category include the increased number of fuel rods per assembly or bundle being considered for BWRs and HWRs and possible changes in fuel to moderator ratio being considered for some BWRs for extended discharge burnup operation. The higher power peaking factors associated with extended discharge burnup operation tends to reduce margins to thermal limits and may provide an incentive to implement design changes to improve hydraulic mixing.

3.5 Mixed oxide fuels

The use of MOX fuels in thermal reactors alters the core characteristics in a way which is qualitatively similar to increasing the uranium enrichment. Thus MOX fuel of equivalent lifetime average reactivity has higher thermal neutron absorption than UO_2 fuel. This reduces the reactivity worth of control devices, modifies the reactivity coefficients and in LWRs requires Pu concentration zoning to control the within-assembly peaking in MOX assemblies at the interface with UO_2 assemblies. These considerations are well known and do not need to be elaborated here, since the use of MOX assemblies in LWRs is well established. The implementation of MOX recycle in HWRs is not yet established. The presence of MOX assemblies in LWR cores complicates the fuel management, but given that modern loading patterns are commonly very heterogenous, this is not a major consideration.

In most respects the issues related to extended discharge burnup operation are the same for MOX as for UO_2 fuels. There is evidence MOX fuels are less susceptible to PCI failures than UO_2 fuels, so that the considerations discussed under 3.2.3 may have a lower impact. The economics of MOX recycle are improved to a greater than for UO_2 with extended

discharge burnups, because the dominant cost component is the fabrication cost, which is essentially independent of burnup. Another factor for MOX is that the isotopic composition of recovered Pu will change depending on the discharge burnups of the fuel from which the Pu is recovered. This has a minor impact on the nuclear design of MOX cores.

4. IMPACT OF EXTENDED BURNUP ON THE BACK END OF THE FUEL CYCLE

4.1. General Issues

4.1.1. Criticality

The increase of initial fissile content of fuel for extended burnup will require an assessment of criticality implications. In most countries a credit for the reduction in reactivity of fuel which has attained its designed burnup is not allowed. This position may be relaxed with the introduction of instruments capable of reliable measurements of either burnup or reactivity. In the case of fuel containing burnable absorbers the regulatory authorities in some countries allow credit to be taken for these absorbers.

4.1.2. Heat output

The heat output from extended burnup fuel with the same cooling time is higher due to the increased fission product and actinide inventories. Typical values for this increase are given in the French contribution to the proceedings of this meeting.

4.1.3. Radioactivity

The radiation fields due to the fission products increase almost in proportion to the burnup, while those due to the actinides increase more rapidly. Typical values for the increase in radioactivity can be found in the French contribution to this meeting.

4.1.4. Accuracy of calculated actinide and fission product parameters

Extended burnup fuel provides a potential source of an excellent data base which could be used to validate code predictions for compositions, heat generation and radiation fields. Prior to the irradiation of this fuel between-code comparisons had indicated differences between the calculations of up to 30% for some nuclides.

4.2. Storage

Extended burnup results in a decrease in the rate of discharge of fuel assemblies. For a given storage capacity the time to use up that capacity is increased and the energy generated from the fuel stored in it is greater for extended burnup fuel. This provides somewhat longer cooling times without providing additional storage facilities.

While discharged extended burnup fuel is associated with higher radiation fields the water cover above the fuel for existing wet storage has been found to be adequate to reduce these fields.

While wet storage provides adequate cooling for the increased heat generation of extended burnup fuel longer pre-cooling may be needed before dry storage can be used. Dry storage facilities may, if not already designed for extended burnup fuel, also require additional shielding.

4.3. Transport

Transport casks for spent fuel are licensed to carry fuel up to a maximum heat load and dose rate which depend upon the burnup and decay time. The burnup limits are different between countries and range from 30 to 50 GW·d/t U. It may be necessary to re-license or redesign transport casks or alternatively to use longer cooling times and/or partial cask loadings.

4.4. Reprocessing

The impact of extended burnup fuel on reprocessing can be viewed from the plant owners or the customers viewpoint. To the customer less volume of waste is returned for a given amount of power generated. The specification of the waste for final disposal will stay within the guaranteed targets approved by the safety authorities. For the reprocessor, however, assuming constant throughput, extended burnup results in a slight increase in the waste from the reprocessing activities. This increase is due to the increased quantities of fission products, increased waste from solvent degradation and increased waste from sludge formation.

Existing reprocessing plants were designed for fuel up to a maximum burnup of 45 GW·d/t U and an enrichment of 4.5%. If, due to burn up extension, this maximum will be exceeded certain steps will need to be taken. Sufficient time would normally be available for these steps due to the approximately eight-year delay between the introduction of the extended burnup fuel into the reactor and its eventual arrival at the reprocessing plant.

These steps include reassessment of:

- criticality safety with particular emphasis on those areas applying fissile concentration controls;
- effluent discharges;
- the heat removal capability;
- the capability of monitoring equipment to measure the increased activity of plutonium;
- the effects of reduced fuel solubility;
- operational dose uptake;
- the effect of new burnable absorber designs.

4.5. Spent fuel conditioning and disposal In once-through cycle

Prior to fuel disposal the fuel assemblies are placed in a containing system. The impact on the increased heat generation on this system, the potential neutron dose and the shielding required for the increased fuel activity and on the disposal operation must be considered, if not avoidable by a number of other alternatives.

5. CONCLUSIONS

1. Exchange of information should be promoted on aspects such as licensing of extended burnup fuel, fuel design criteria and analysis of operational experience with extended burnup fuel.

2. Exchange of information on cross-section data for burnable absorbers is desired by some countries.

3. The programme to assess and benchmark computer codes for fuel thermal-mechanical performance at high burnups should be continued.

4. Measures to decrease uncertainties in predictions of isotopic composition and decay heat generation of spent fuel should be studied further.

5. The IAEA is addressing aspects of concern primarily to the LWR community. Development are also occurring in the HWR community which need to be addressed. Calculational benchmark meetingscould be set up aimed at fuel behaviour and discharge fuel properties at higher burnups and reactivity coefficients. Alternatively the HWR equivalent of WREBUS could be prepared.

PAPERS PRESENTED AT THE ADVISORY GROUP MEETING

DEVELOPMENTS IN SLIGHTLY ENRICHED URANIUM FOR POWER REACTOR FUEL IN ARGENTINA

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Abstract

Argentina has two Pressurised Heavy Water Reactors (PHWR) in operation at the present time: one is Atucha-1 and the second Embalse. Both are natural uranium fuelled.

The use of slightly enriched uranium (SEU) in PHWR is an effective way to increase the fuel discharge burnup. The main benefits from the use of SEU cores are the following:

- Fuel cycle costs decrease
- Uranium resources savings
- Spent fuel volume decreases.

Other advantages of core enrichment are described in the report.

In order to test the overall performance under irradiation of SEU fuel, a first series of twelve Atucha-1 FA prototypes (0.85% U-235) was fabricated by Argentine domestic supplier.

Work is carried out in Argentina in order to confirm the advantages and feasibility of PHWR fuel burnup extension programme.

1. INTRODUCTION

The fuel discharge burnup of Atucha-1 Nuclear Power Plant can be increased by using slightly enriched uranium (SEU) in reload fuel.

Main characteristics of the Atucha-1 PWHR are listed in Table 1.

The present Atucha-1 fuel data is summarized in Tables 2 and 3.

The transition from natural uranium core to SEU core would be obtained through two or three enrichment steps, in order to avoid changes to the thermalhydraulic design of the core. The first step will be the use of 0.85% U^{235}, followed by a second step of about 1.00%. One last step of about 1.20% is under consideration, as can be seen in Table 4.

TABLE 1. ATUCHA-1 REACTOR TECHNICAL DATA

REACTOR TYPE	**pressurized heavy water reactor (PHWR)**
THERMAL OUTPUT	**1179 MW**
GROSS ELECTRICAL OUTPUT	**367 MW**
COOLANT AND MODERATOR	**D_2O**
NUMBER OF FUEL CHANNELS	**253**
NUMBER OF FUEL ASSEMBLIES	**253**
REFUELLING AND FUEL SHUFFLING	**ON-POWER**

TABLE 2. ATUCHA-1 CORE TECHNICAL DATA

FUEL TYPE	**natural uranium dioxide**
TOTAL NATURAL URANIUM INVENTORY	**39.0 t**
QUANTITY OF FUEL ROD PER FUEL ASSEMBLY	**36**
QUANTITY OF SPACERS	
ZRY-4	**15**
INCONEL	**1 (bottom end)**

TABLE 3. ATUCHA-1 FUEL ROD TECHNICAL DATA

FUEL ROD OUTSIDE DIAMETER	**11.9 mm**
ACTIVE LENGTH OF FUEL ROD	**5300 mm**
CLADDING MATERIAL	**Zry-4**
AVERAGE FUEL ROD HEAT RATE	**232 W/cm**
MAXIMUM FOR STATIONARY CONDITION	**531 W/cm**
MAXIMUM FOR NON-STATIONARY CONDITION	**600 W/cm**
MAXIMUM DESIGN PEAK POWER	**690 W/cm**

TABLE 4. ENRICHMENT PLANNED STEPS OF HOMOGENEOUS SEU ATUCHA-1 CORE

. FIRST STEP	:	0.85 % U-235
. SECOND STEP	:	1.00 % U-235
. THIRD STEP	:	Not defined yet

TABLE 5. MAIN ADVANTAGES FROM USE OF SLIGHTLY ENRICHED URANIUM ON ATUCHA-1 CORE

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TABLE 8. ATUCHA-1 FUEL ASSEMBLY

	U-235 Enrichment			
	Natural	0.85 %	1.0 %	1.2 %
AVERAGE DISCHARGE BURNUP [MWd/kgU]	6	11.4	16	21
AVERAGE RESIDENCE TIME [efpd]	198	377	529	694
REFUELLING FREQUENCY [FA/efpd]	1.28	0.67	0.48	0.36
QUANTITY OF F.A./year [F_L = 85 %]	396	208	148	113
CONSUMPTION OF URANIUM [tU/year]	61	44	40	39

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- PURE ZIRCONIUM LAYER

TABLE 14. EMBALSE FUEL ASSEMBLY

	U-235 Enrichment			
	Natural	0.90 %	1.0 %	1.2 %
AVERAGE DISCHARGE BURNUP [MWd/kgU]	7.5	14.6	17.8	23.9
AVERAGE RESIDENCE TIME [efpd]	303	590	720	966
REFUELLING FREQUENCY [FA/efpd]	15.04	7.72	6.34	4.72
QUANTITY OF F.A./year [F_L = 85 %]	4665	2397	1966	1464

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- URANIUM RESOURCES SAVINGS
- SPENT FUEL VOLUME DECREASE

The main advantages from the use of slightly enriched uranium on Atucha-1 core, are:

- extension of fuel discharge burnup,
- savings in uranium reserves;
- reduction of the spent fuel volume and lower total fuel cycle cost.

These advantages are shown in Table 5.

Other consequences of core enrichment (see Table 6), are:

- extension of fuel residence time,
- reduction of fuel assemblies consumption and lower frecuency of on-power refuelling and fuel shuffling,
- reduction of fresh fuel stock and fuel transport;
- impact on spent fuel storage pool capacity.

In addition, during the transition to homogeneus SEU core there is also an increase in discharge burnup and residence time of the fuel with natural or lower enriched uranium. Reloading of spent natural fuel assemblies in order to extend their original low discharge burnup is also under analysis.

Table 8 compares the estimated average discharge burnups and other figures for different enrichments steps respect to our current experience with natural uranium core. For 0,85% U^{235}, the fuel consumption will be 52 percent respect to the present value.
The annual savings of uranium are above 28 percent considering a tail assay for enrichment of 0,20% U^{235} and a plant load factor of 85 percent.

The reduction of refuelling frecuency represents a significant advantage for the extension of the fuelling machine lifetime and manteinance operations.

2. FUEL DESIGN GUIDELINES

The increases in average fuel assemblies discharge burnup and residence time will require progressive changes of fuel rod and assembly designs.

The main design guidelines for SEU fuel assemblies are (Table 9):

- Maintain the fuel ability to operate reliably to extended burnups levels,
- Avoid the introduction of new power operation restrictions;
- Maintain the present margins of safe operation of the reactor.

3. FUEL PERFORMANCE REQUIREMENTS

Atucha-1 fuel assemblies with higher discharge burnups by using SEU will have to meet increased performance requirements due to:

- higher fuel burnup,
- longer residence time;
- higher fast neutron fluence (Øt).

as is shown in Table 10.

4. SEU LIFE LIMITING ASPECTS

For Atucha-1 SEU fuel rods, the main life limiting aspects are (Table 11):

-resistance to pellet-cladding interaction stress corrosion assisted (PCI-SCC) during fuel shuffling and core operational fluctuations,

-internal fuel rod pressure,

-fuel rod diametral changes with respect to fuel rod-spacer gap;

-waterside corrosion.

For fuel assemblies, the principal life limiting aspects are (Table 12):

- differential axial growing of fuel rod concerning to spacer-sliding pad interaction,
- stress relaxation of spring sliding shoes related with end of life (EOL) clamping forces between coolant channel and fuel assembly.

5. PCI REMEDIES

For higher enrichments and therefore higher burnups, is of special concern the potential of fuel failures by PCI-SCC. These failures may be caused by power ramps during on-power fuel shuffling, core load changes and core startups. To maintain satisfactory fuel performance without introducing restrictions to the present fuel managements and operation procedures, we are planning to use PCI remedies like (Table 13):

- graphite coating of cladding inner surface;
- pure Zirconium layer (as "barrier") bonded to cladding inner surface.

By using one of this remedies, we expect to prevent the decreasing of the PCI-SCC failure threshold at higher burnups.

6. PROTOTYPES IRRADIATION

To start the introduction of SEU fuel and for overall performance testing, a first serie of twelve Atucha-1 fuel assembly prototypes was fabricated by our domestic suppliers.
These fuel assemblies contain uranium enriched to 0.85% U^{235}.

Design and specifications changes were introduced by our engineering and design group. This prototypes will be irradiated in the near future at Atucha-1 Power Plant up to approximately 8,3 MWd/Kg.U.

7. SEU PROGRAMS FOR OTHER POWER REACTORS

Argentina has two NPP in operation. The second one is a CANDU-6 PWHR at Embalse.
The fuel for this reactor is also natural uranium.

Preliminary calculations were performed considering different levels of SEU for the Embalse reactor core.
The results are indicated in Table 14.

Reduction of fuel consumption and uranium resources savings are as important as it was stated for Atucha-1.

We have also the intention to analyze the extension of SEU programs to the Atucha-2 Power Plant which is under construction.

8. DOMESTIC RESOURCES

We intend to develope these fuel SEU programs by using our domestic capability on fuel design (Table 15).

The development of new manufacturing technology will be also carried out by our researchers and then transferred to our cladding and fuel manufacturing plants.

9. CONCLUSIONS

Analysis and calculations performed up to date suggest that the benefits of discharge burnup extension in PHWR are:

- Fuel cycle costs decrease.
  One preliminary estimation for Atucha-1 shows that with the first enrichment step (0,85%) it is possible to save approximately 10 million dollars per year. The fuel contribution to generation cost decreases about 30 percent.
- Uranium resources savings .
  For 0,85% enriched Atucha-1 core, the Uranium savings are about 28 percent per year.
- Spent fuel volume decreases.
  The amount of spent fuel reduces, for 0,85% enriched Atucha-1, to 52 percent per year.

These benefits are encouraging us to continue with these programs for both argentinean Nuclear Power Plants.

EXTENDED BURNUP IN CANDU

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Abstract

The use of enriched fuel, from either enriched natural uranium or uranium recovered from LWR discharged fuel, is an attractive option for the CANDU reactor system. The impact of the use of enriched uranium on resource utilisation and fuelling costs is discussed. Some technical problems result and their solution is described.

1. BACKGROUND

The CANDU* reactor is a heavy-water-moderated and -cooled, pressure-tube reactor (Figures 1 to 3). The pressure tubes are arranged horizontally on a square lattice pitch (380 channels in the CANDU 6 reactor). Each channel contains 12 natural-uranium UO_2 fuel bundles, 0.5 m long. Fuelling is on-line, and bi-directional, meaning that adjacent channels are fuelled in the opposite direction. When a channel is refuelled, a fuelling machine attaches to each end of the channel: fresh fuel is inserted in one end of the channel, and old fuel that has reached its discharge burnup is removed from the other end of the channel. In a CANDU 6 reactor, about 2 channels are refuelled per day using an 8-bundle shift. This method of fuelling gives the CANDU reactor the potential to accommodate a variety of different fuels.

On-line refuelling reduces the excess reactivity that needs to be controlled. Reactivity control (other than through refuelling) is primarily through adjuster rods, zone controller units, and shutoff rods. In a CANDU 6 reactor, there are 18 adjuster rods, arranged in three rows. The adjuster rods are vertically oriented between the fuel channels, and serve several purposes:

- The adjusters provide xenon override time. After a reactor shutdown, the level of xenon-135 increases for several hours. In a CANDU 6 reactor, the neutron absorption in the adjusters balances the absorption in the xenon which builds in after 30 minutes after a shutdown. If the adjuster rods are withdrawn after a trip, the reactor can be restarted within 30 minutes--otherwise the reactor poisons out.
- The adjusters provide shaping (flattening) of the flux and power distributions.
- Withdrawal of some of the adjuster rods provides reactivity shim in the event that the fuelling machines are unavailable.
- Withdrawal of some of the adjuster rods provides the reactivity needed to operate at less than full power (power set-backs or step-backs).

* CANDU: CANada Deuterium Uranium. Registered trademark.

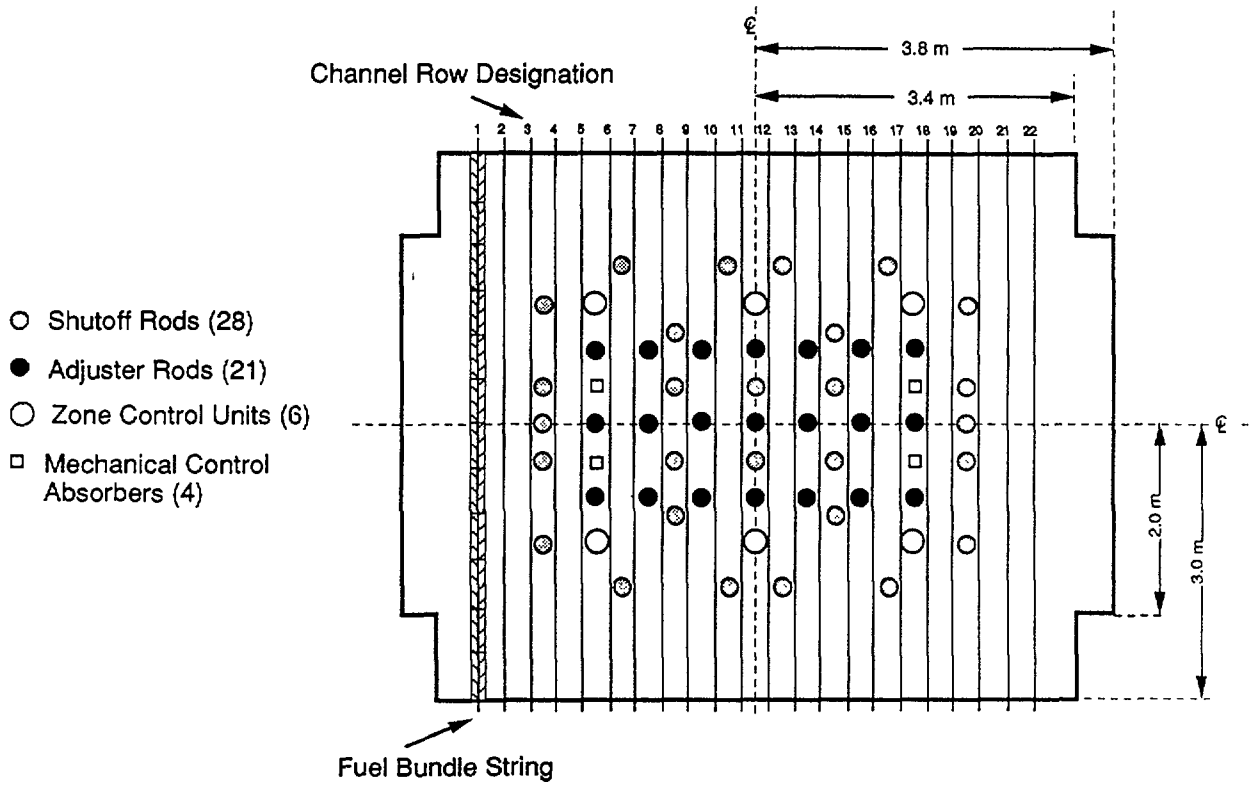

Figure 2:
Top View Of Core Showing Reactivity Device Locations

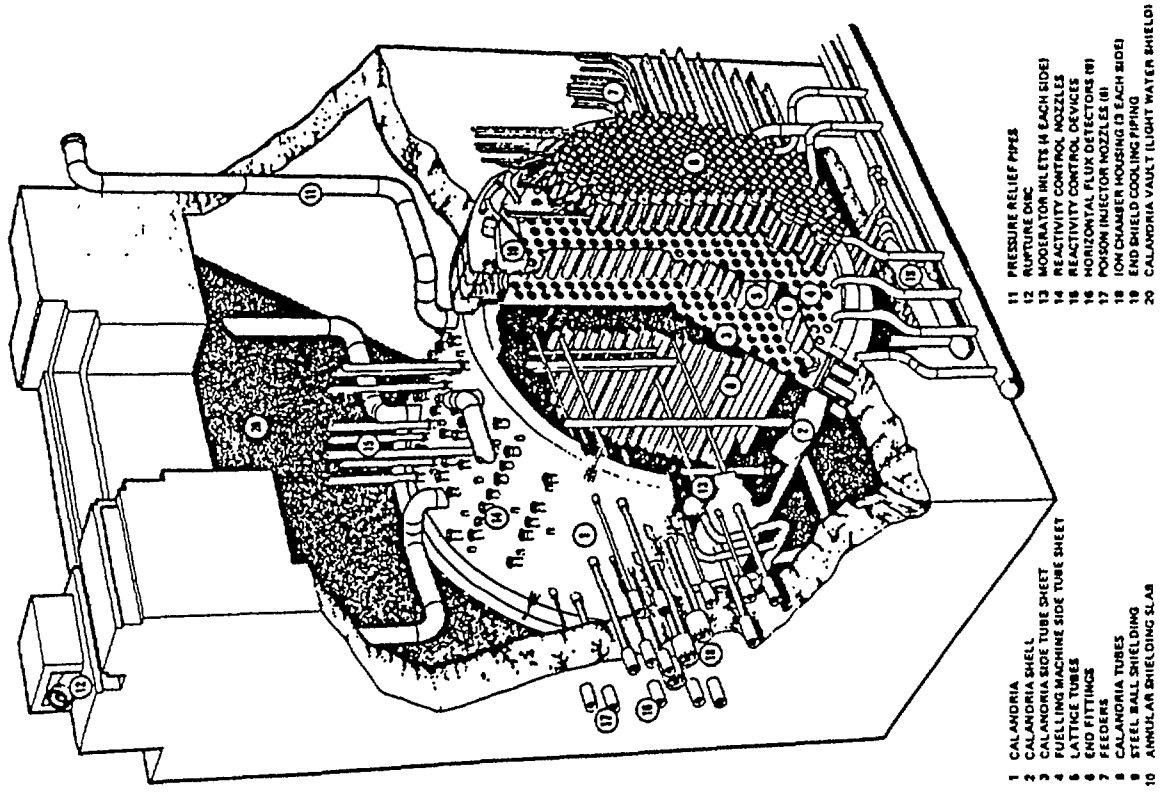

FIGURE 1: CANDU 6 Reactor Assembly

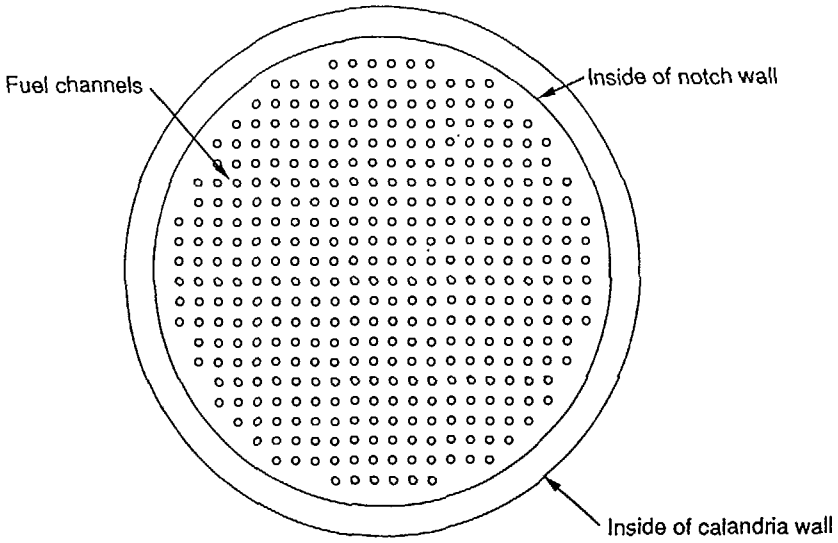

Figure 3: Face View Of Reactor

The zone-control system provides bulk- and spatial-reactivity control. This system consists of vertical tubes containing either two or three compartments. Each compartment contains light water (a neutron-absorber in CANDU), the level of which can be varied either uniformly in all the compartments (providing bulk-reactivity control), or individually (providing spatial control).

Rapid shutdown in CANDU is provided by two independent shutdown systems: shutoff rods, which are normally poised above the reactor core, and a liquid-poison injection system. The reactivity depth and speed of insertion of both systems ensures effective shutdown capability in the event of an accident.

2. IMPACT OF EXTENDED BURNUP ON CANDU URANIUM REQUIREMENTS AND FUEL-CYCLE COSTS

2.1 Slightly Enriched Uranium (SEU)

2.1.1 Natural-Uranium Requirements

While the natural-uranium-fuelled CANDU is the most neutron-efficient of all commercial reactors in operation today, enriching the uranium improves the neutron efficiency even further:[1-4] the increased natural uranium required to provide the source of fissile U-235 in the enriched fuel is more than compensated for by the increased energy extracted from the fuel. For example, while two kg of natural uranium are required to produce one kg of SEU with an enrichment of 1.2%, three times the energy is obtained from the fuel. Hence, the amount of natural uranium required to produce a unit of energy is reduced by a factor of 2/3, about 30%.

The savings in uranium resulting from the use of SEU in CANDU depends on the enrichment of both the product (the fuel), and the tails in the enrichment plant (Figure 4). The savings level out at an enrichment of about 1.2%, and increase with decreasing tails enrichment. An enrichment of 0.9% SEU results in annual uranium savings of between 20% and 30% relative to the use of natural uranium. With 1.2% SEU, the savings increase to between 30% and 40%. There is no resource incentive in going to higher enrichments, while the technical difficulties increase with higher enrichments. (Note that the exact savings depend to some extent on the burnup assumed for natural uranium, which depends on several factors, such as the size of the reactor and the designed xenon override capability.)

The advantage that the natural-uranium-fuelled CANDU enjoys over the light-water reactor (LWR) in uranium utilization is eroded with increasing burnups and other uranium-conserving fuel management techniques in the LWR. Moreover, a natural-uranium-fuelled CANDU could completely lose its traditional advantage in uranium utilization if new AVLIS technology makes lower tails enrichments economical. With SEU fuel, CANDU can maintain its advantage in uranium utilization, and can also benefit from decreases in enrichment prices that are expected to result from increased competition and new enrichment technologies (Figure 5).

While the use of SEU in domestic CANDU reactors could reduce their annual uranium requirements by about 30%, the effect of this reduction on the Canadian uranium mining industry would not be large: in 1988, 85% of Canada's output was available for export.[5] Moreover, the domestic use of SEU in CANDU would create a domestic market that could support a Canadian enrichment industry.[6] The export of enriched uranium, rather than natural uranium, would double the value of Canadian uranium exports.

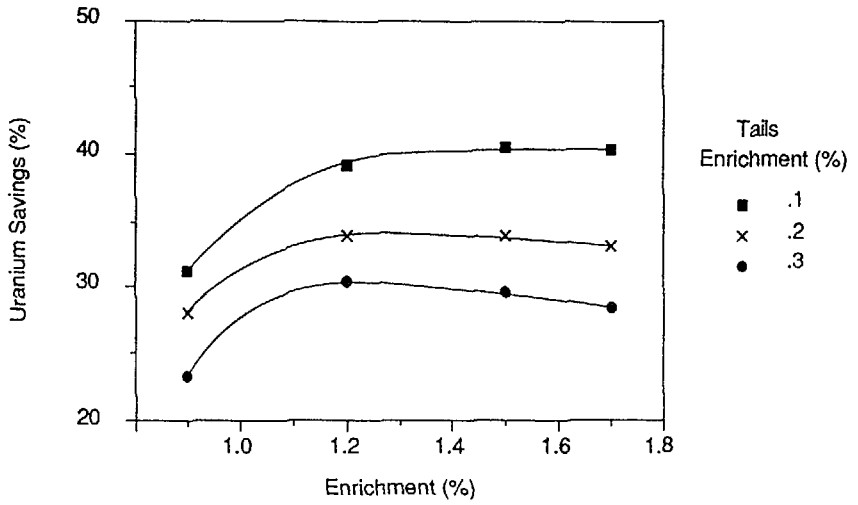

Figure 4:
Annual Uranium Savings With Seu In Candu

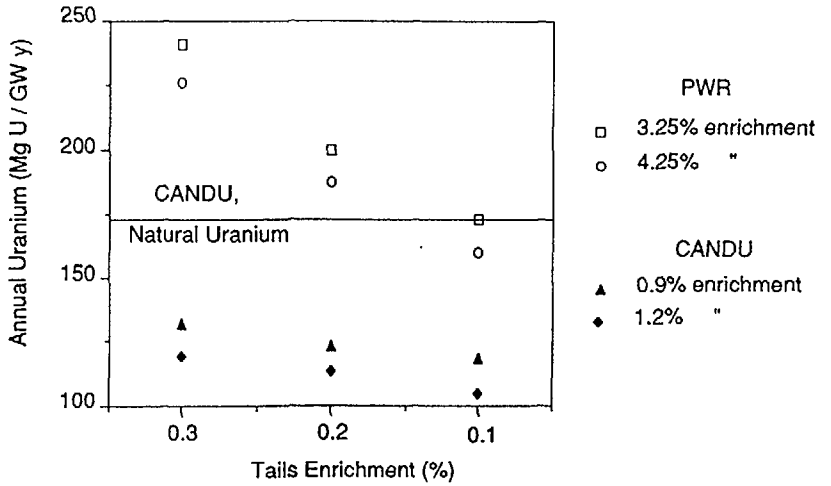

Figure 5:
Comparison of CANDU and PWR Uranium Requirements

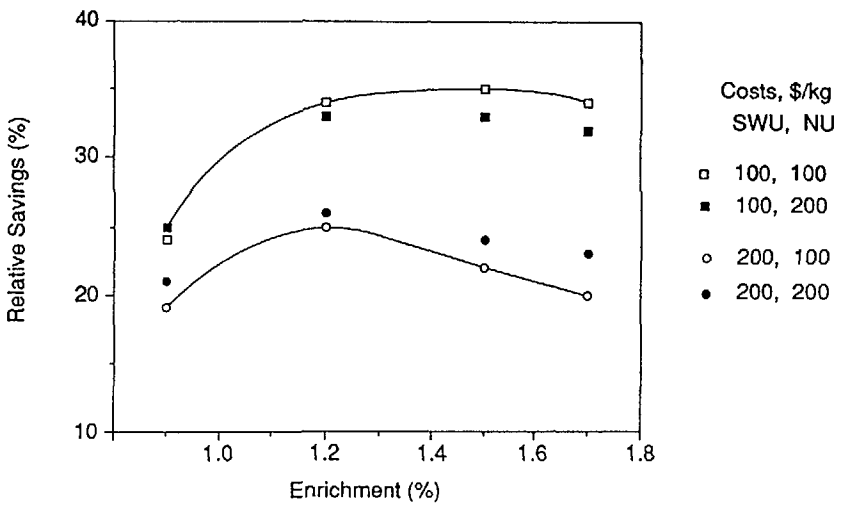

Figure 6:
Relative Fuel Cycle Cost Savings with SEU in CANDU

2.1.2 Fuelling Costs

In existing CANDU reactors, the only areas in which significant cost reductions can be made are in operation and maintenance, and in fuelling. SEU (and recovered uranium, RU, from spent LWR fuel) are the only fuel cycles in CANDU that are economic now, compared to natural uranium fuelling. Figure 6 shows the fuel-cycle cost savings (undiscounted) for SEU in CANDU, relative to natural-uranium fuelling, for representative component costs. Over a wide range of enrichment and natural-uranium costs, an enrichment of 1.2% is near-optimal, not only for uranium utilization, but also for fuel-cycle costs. While 0.9% SEU results in annual fuel-cycle cost savings of between 20% and 25%, these savings increase to between 25% and 35% with an enrichment of 1.2%. The corresponding absolute savings amount to over $100 million a year in a system of 15 GWe capacity.[3]

CANDU fuel-cycle costs are about a factor of two lower than for LWRs[7-9] -- an important advantage, since this more than offsets the added heavy-water cost component in the total unit energy cost. The use of SEU in CANDU will help maintain this advantage as LWR fuel-cycle costs are reduced through such improvements as higher burnup and smaller reload batches.

2.2 Recovered Uranium (RU)

2.2.1 Uranium Utilization

The use in CANDU of uranium recovered from spent LWR fuel during reprocessing is attractive from both the perspective of uranium conservation and fuelling costs. There is a strong resource incentive to recycle RU in CANDU, rather than re-enrich it for reuse in a pressurized-water reactor (PWR): twice the thermal energy can be extracted from the RU when recycled in CANDU rather than in an PWR.[9,10]

2.2.2 Fuelling Costs

Fuelling of CANDU with RU would be economical today. A major benefit stems from not having to re-enrich the RU before using it in CANDU. This results in significant simplifications and cost-savings. A recent study has shown the economic benefit of using RU directly in CANDU rather than re-enriching it in a PWR.[10] Figure 7 compares PWR and CANDU fuelling costs, for representative cost assumptions. Fuelling costs are shown for two values of RU: RU at the same cost as natural uranium (RU1), and RU at no cost (RU2). The fuelling cost advantage of CANDU over the PWR is extraordinary. CANDU fuelling costs with natural uranium are about a factor of 2 lower than PWR fuelling costs. With the cost of RU the same as natural uranium, PWR fuelling costs are reduced by only 2% (PWR, RU1 in Figure 7), whereas CANDU fuelling costs are nearly a factor of 3 lower than for the reference PWR (CANDU, RU1). With RU available at no cost, PWR fuelling costs decrease by 36% (PWR, RU2), whereas CANDU fuelling costs (CANDU, RU2) are a factor of 5.5 lower than for the reference PWR.

An upper value for RU can be derived by equating the cost of fabricating PWR fuel from RU to the cost of fabricating PWR fuel from enriched uranium from natural-uranium feed:

Cost of enriched uranium from natural-uranium feed ($/kg)

$= C_{U3O8} \cdot F_1 + C_{UF6} \cdot F_1 + C_{SWU} \cdot S_{NU} + C_{UO2}$

Cost of enriched uranium from RU ($/kg)

$= C_{RU} \cdot F_2 + (C_{UF6} + \Delta C_{UF6}) \cdot F_2 + (C_{SWU} + \Delta C_{SWU}) \cdot S_{RU} + (C_{UO2} + \Delta C_{UO2})$

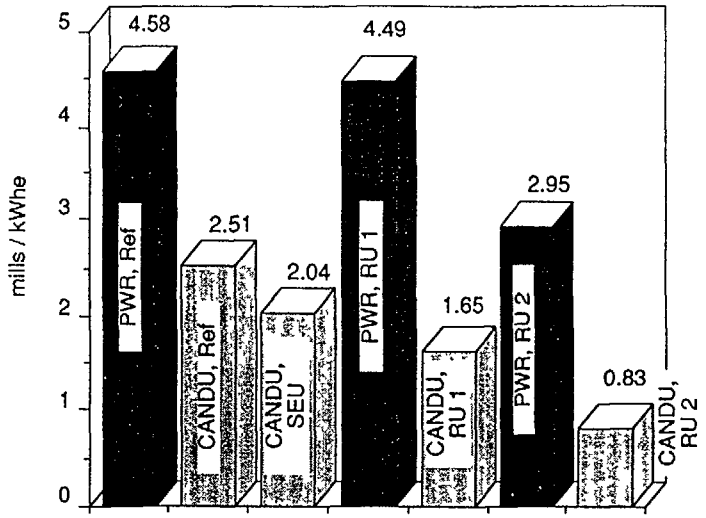

Figure 7:
Comparison of CANDU and PWR Fuelling Costs

This then yields the following value for the price of RU: C_{RU} =

$$1/F_2 \cdot [C_{U3O8} \cdot F_1 + C_{UF6} \cdot (F_1 - F_2) - \Delta C_{UF6} \cdot F_2 + C_{SWU} \cdot (S_{NU} - S_{RU}) - S_{RU} \cdot \Delta C_{SWU} - \Delta C_{UO2}]$$

The meaning of the symbols is given below, as well as illustrative values for the parameters (Reference 11 was used for several of these values; reference 9 states the enrichments, burnups and tails assumed).

C_{U3O8}:	price of natural uranium ($50/kg U)
C_{UF6}:	price of UF_6 conversion ($8/kg U)
C_{SWU}:	price of enrichment ($110/SWU)
C_{UO2}:	price of UO_2 fuel fabrication ($275/kg U)
F_1:	natural uranium requirement for 1 kg of enriched uranium (6.51)
S_{NU}:	separative work for 1 kg of enriched uranium from natural uranium feed (4.31 SWU)
ΔC_{SWU}:	premium for enrichment of RU ($10/SWU)
ΔC_{UF6}:	premium for UF_6 conversion using RU ($21/kg)
ΔC_{UO2}:	premium for UO_2 fabrication from RU (30/kg)
F_2:	recovered uranium requirement for 1 kg of enriched uranium (5.03)
S_{RU}:	separative work for 1 kg of enriched uranium from RU (3.95 SWU)

Using these values , a natural uranium price of $50/kg U results in a price of RU of $40 /kg; a natural uranium price of $80/kg results in a price of RU of $78/kg. Hence, in a PWR, recovered uranium has a value lower than that of natural uranium.

There are several reasons for the cost advantage of using RU in CANDU over re-enriching it and using it in a PWR.[12] The RU from spent PWR fuel contains about 0.4% U-236, a neutron poison whose neutronic effect is an order of magnitude greater in the harder PWR spectrum than in CANDU. Re-enriching the RU increases not only the U-235 concentration, but also the U-236 concentration. Consequently, the U-235 must be enriched to a higher level to compensate for the presence of the U-236. (For example, a U-236 concentration in the RU of 1.2% necessitates an enrichment of about 3.6% U-235, compared to 3.25% U-235 using natural-uranium feed for the same burnup.) This higher enrichment increases the requirements for both RU and SWU (enrichment), and the associated costs. There are also cost penalties associated with the conversion, re-enrichment, and fabrication of PWR fuel using RU, due to the radioactivity of the daughter products of U-232, the level of which also increases after re-enrichment.

3. TECHNICAL CHALLENGES ASSOCIATED WITH EXTENDED BURNUP

3.1 Fuel-Management Considerations

The use of enrichment in CANDU would increase the reactivity of the fresh fuel. In order to reduce the power ripple during refuelling, fewer bundles would be added. For example, with 0.9% SEU, which roughly doubles the fuel burnup from 7 MWd/kg with natural uranium to 14 MWd/kg, a 4-bundle shift fuelling scheme could be used; with 1.2% SEU, which triples the fuel burnup to about 21 MWd/kg, a 2-bundle shift fuelling scheme would be appropriate.

The optimal fuel-management strategy depends on such considerations as the fuel enrichment, and the number and location of adjuster rods. With 0.9% SEU (or RU), a regular 4-bundle shift fuelling scheme would result in good axial power profiles.

With 1.2% SEU, a regular 2-bundle shift fuelling scheme would result in acceptable power distributions either in a core without adjuster rods (such as the Bruce A nuclear generating station in Canada), or in the peripheral channels outside of the adjuster rod region (Figure 8).[13] In this situation, the power would peak at one end of the channel, and decrease along the length of the channel. This would be particularly attractive in those reactors in which fuelling is in the direction of coolant flow (such as the CANDU 6 and Pickering reactors). The power then would peak at the inlet end of the channel, where the coolant enthalpy is lowest, and this would improve the critical channel power (power at which the critical heat flux is first reached).

In reactors with adjuster rods, a regular 2-bundle shift fuelling scheme would result in an axial power distribution that is depressed in the vicinity of the adjuster rods (Figure 9).[13] This would increase the peak bundle power in the reactor, and result in power boosting during refuelling when the fuel is at relatively high burnups, which could lead to unacceptable fuel performance. Thus, with enrichments of 1.2% or greater, other fuel-management options must be employed. Two options that have been studied are axial shuffling and the checkerboard fuelling scheme.

3.1.1 Axial Shuffling

Axial shuffling provides the greatest flexibility in shaping the axial power distribution. Axial shuffling involves removing some or all of the bundles from the channel, rearranging the bundles in pairs, and reinserting some of the fuel-bundle pairs back into the channel in a different order, along with fresh fuel. Axial shuffling is the reference fuel-management strategy for the newest

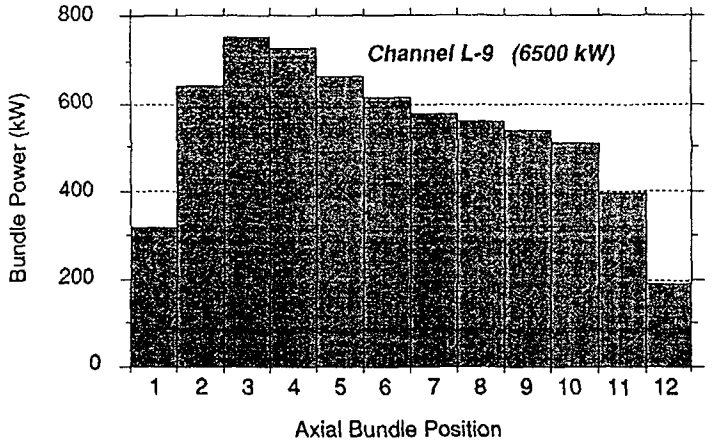

Figure 8:
Axial Power Profile For Regular 2-bundle Shift, Adjuster Rods Out

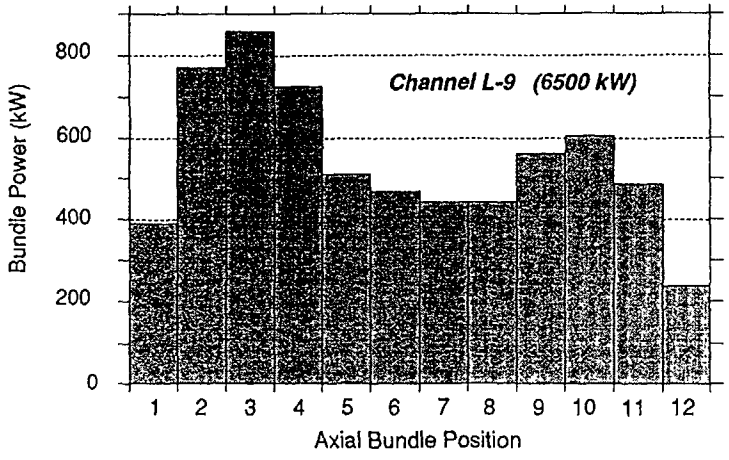

Figure 9:
Axial Power Profile For Regular 2-bundle Shift, Adjusters In Core

member of the CANDU family--the CANDU 3--since all fuelling is done from one end of the reactor (i.e., uni-directional fuelling). Preliminary studies indicate that axial shuffling is feasible in other CANDU reactors as well.

The ease of application of axial shuffling, and the number of bundles that can be removed from the channel, depend on several factors. For instance, the fuelling machine magazine capacity must be sufficient to hold all of the bundles discharged from the channel, the fresh bundles that are to be inserted into the channel, and other components, such as the channel closure and shield plug. The coolant flow must be sufficient to discharge all of the bundles.

One axial shuffling fuelling scheme has been identified that is attractive for 1.2% SEU.[14] This shuffling scheme is bi-directional, although in a reactor in which the fuelling is from one face of the reactor, such as in the CANDU 3, bi-directional fuelling can be simulated. If bundle positions along the channel are numbered from 1 to 12 starting at the "refuelling" end of the channel, then new bundles are first loaded into positions 1 and 2. Upon subsequent fuelling operations, the bundles in positions 1 and 2 are shifted as follows:

1 → 5 → 3 → 9 → 7 → 11 → discharged
2 → 6 → 4 → 10 → 8 → 12 → discharged

This axial shuffling scheme would be restricted to the central channels, in the vicinity of the adjuster rods. In the peripheral channels, a regular 2-bundle shift fuelling scheme would be employed.

This axial shuffling scheme results in low peak bundle powers. Figure 10 shows a typical distribution of bundle powers along a channel in the vicinity of the adjuster rods with axial shuffling. Moreover, the bundles see a declining power history with increasing burnup, which is desirable for good fuel performance at extended burnup (Figure 11).

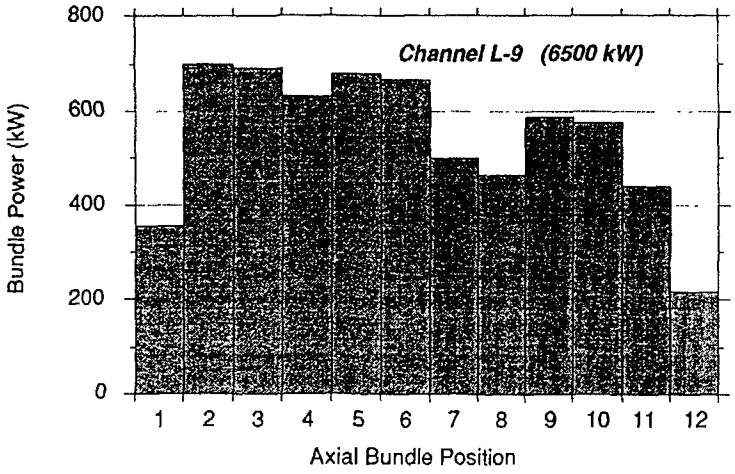

Figure 10:
Axial Power Profile For Axial Shuffling

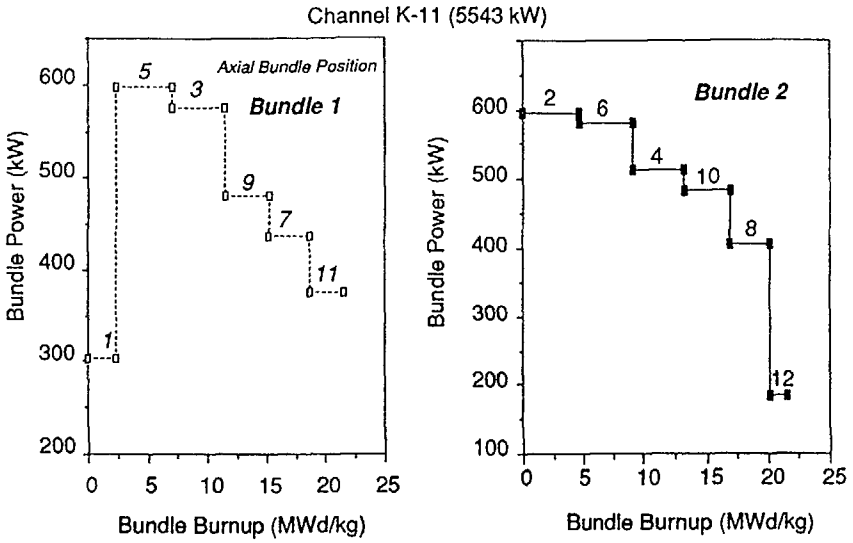

Figure 11:
Time-average Bundle Power Histories, Axial Shuffling

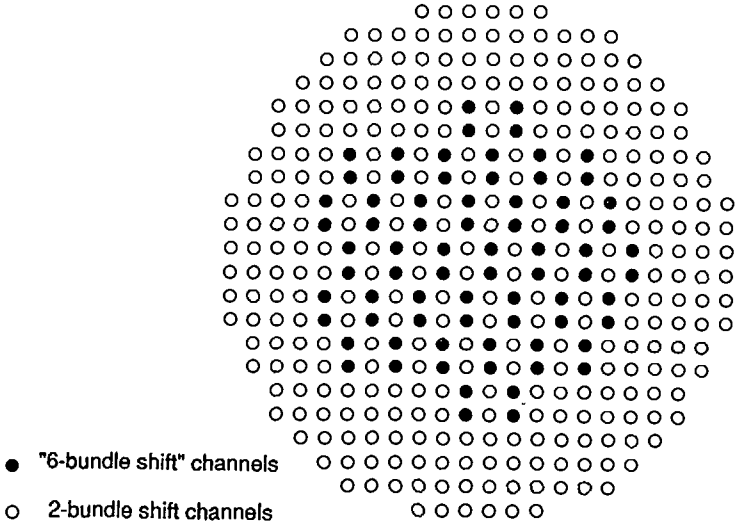

Figure 12:
Checkerboard Fuel Management Scheme

3.1.2 Checkerboard fuelling

The checkerboard fuel-management scheme is another strategy that could be used to accommodate 1.2% SEU in the vicinity of the adjuster rods.[15] In the checkerboard fuelling scheme, adjacent channel pairs are refuelled with different bundle shifts. Figure 12 is an example of a 6/2 checkerboard with 1.2% SEU. A 6-bundle shift fuelling scheme throughout the core would result in an axial power shape that is strongly peaked in the centre of the channel. A 2-bundle shift fuelling scheme peaks towards one end of the channel and is depressed in the centre of the channel if adjuster rods are nearby. The resultant axial bundle power distribution with the checkerboard is a blend of the two shapes, and is flatter axially (Figure 13). The 6-bundle shift channel is really a pseudo 6-bundle shift, comprising, for example, three quick 2-bundle shifts in succession, in order to reduce the power peaking that would result from a 6-bundle shift. The power histories resulting from checkerboard fuelling should be acceptable from the perspective of fuel performance.

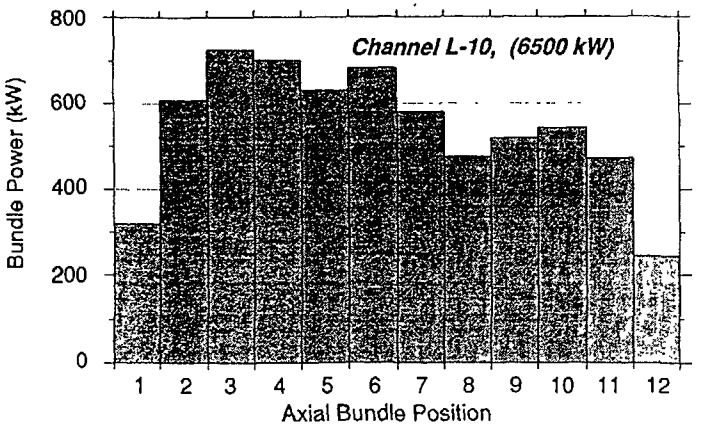

Figure 13:
Axial Power Profile For Checkerboard (Pseudo-6-Bundle-Shift Channel)

3.1.3 Future Reactors Optimized for the Use of SEU

In future reactors, many of the complications in fuel management resulting from the central location of the adjuster rods can be overcome by repositioning the adjusters and other reactivity devices. A configuration of reactivity devices has been identified that would be near-optimum for both 1.2% SEU and natural-uranium fuel.[16] This would allow the use of a regular 2-bundle shift with 1.2% SEU, and a regular 8-bundle shift fuelling scheme with natural uranium.

3.1.4 Strategies for Introducing SEU

The fuel-management considerations above focused on equilibrium fuelling. Another consideration is how the SEU is introduced in the reactor. It is unlikely that one would start a reactor with a fresh core of 1.2% SEU: the reactivity needing to be held down would be very large, which would result in a large loss of neutron economy, and could pose problems in safety and licensing. A more likely scenario would be to start with a fresh core of natural uranium, or with enrichment slightly above natural, and gradually introduce the higher enrichment. The transition from natural uranium fuel to 1.2% SEU has been studied for two cases: the checkerboard fuelling scheme with 1.2% SEU in a standard CANDU 6 reactor, and a regular 2-bundle shift fuelling scheme with 1.2% SEU in a core with repositioned reactivity devices.[17] Both strategies were workable.

3.1.5 Reactivity Control

The use of enrichment in CANDU will lower the reactivity worth of the adjuster rods (since the enriched fuel is blacker than natural uranium). Fortunately, the reactivity worth required to provide a given xenon override time is lowered correspondingly, so the same adjuster rods provide about the same xenon override time for both natural uranium and SEU.

The reactivity worth of the shutoff rods and zone controllers is also lowered, but is adequate. Software changes to the control system parameters (such as modifications to the controller gains) may be required.

3.2 Fuel Performance Aspects

3.2.1 CANDU Fuel Performance at Extended Burnup

With the optimum SEU enrichment of 1.2%, a bundle-average burnup of about 21 MWd/kg is attained, with bundle maxima of around 30 MWd/kg. This is three times the current burnup for natural uranium in CANDU. Potentially life-limiting factors that are important for extended burnup conditions include[2]:

- mechanical integrity,
- deuterium pickup and sheath corrosion,
- fission-gas release enhancement resulting in higher internal pressure,
- enhanced fission-product concentration, increasing susceptibility to stress-corrosion cracking (SCC), and
- dimensional stability (changes induced by solid and gaseous fission products).

The effect that is currently regarded as most important in potentially limiting fuel life at extended burnup is fuel chemistry. Several international programs are addressing fuel-chemistry effects (Halden, Studsvik, Riso, EPRI, BelgoNucleaire). The inventory of heavy metal, oxygen and fission products in UO_2 is modified at extended burnup. Some of the oxygen present is fixed by the fission products, and some of the resultant compounds are more stable than if the excess oxygen was fixed by UO_2-PuO_2. The rare earths, Nb, Zr, Ba and Sr, form stable oxides, but fission products like Cs, Mo, I, and Te do not. At high temperature, the latter compounds are less stable than the fuel phase. In a closed system, oxygen is transferred from these fission products towards the fuel phase; above some temperature the fuel phase then becomes hyper-stoichiometric. If a deviation from stoichiometry takes place in the dominant fuel phase, significant changes occur in thermal properties (thermal conductivity), diffusion properties or mechanical properties (creep rate). Such a deviation will significantly influence fuel-rod behaviour, characterized by higher temperatures and increased fission-product release.

Ontario Hydro has irradiated more than 600 000 bundles containing UO_2 in its CANDU reactors, with a very low bundle defect rate of about 0.1%. Over 3000 CANDU bundles have been irradiated to above-average burnups, with a few to a maximum of 30 MWd/kg. About 150 bundles have experienced burnups above 17 MWd/kg, mainly in Ontario Hydro reactors, but with relatively low powers, and declining power histories. The number of bundles drops sharply as the burnup increases. The experimental irradiations in research reactors associated with the initial development of CANDU fuel all involved enriched uranium. Some 66 bundles have been irradiated to high burnup (maximum 45 MWd/kg) at high power in experimental reactors, supplemented by data from the irradiation of 173 single elements.

The available data show that the current 37-element fuel design is capable of reliable operation to a burnup of 17 MWd/kg with a normal declining power history. Data also suggest that enhanced fission-gas release, producing potentially life-limiting effects, may begin to occur at burnups in excess of about 19 MWd/kg. This effect is attributed to changes in fuel chemistry discussed earlier.

Recently, post-irradiation examination (PIE) of several fuel bundles that reached extended burnups in the Bruce power reactor was performed.[18] No evidence of fuel-performance deterioration was observed up to 19 MWd/kg. However, two bundles having an average discharge burnup of 26 MWd/kg and having experienced a peak outer-element power of 52 kW/m had several defective elements. High gas release was measured (up to 28%). UO_2 ceramography revealed substantial grain growth, and tear-out of grains at the fuel periphery, associated with extensive porosity. The oxide thickness on the outer sheath surfaces was up to 7 μm. Outer-element diameters showed 2% increase. Existing fuel codes under-calculated grain growth, fission-gas release and diametral increase. Preliminary examinations of three other bundles having lower burnups (average 21 MWd/kg) but higher linear heat ratings (maximum of 58 kW/m) revealed failures associated with high fission gas release and excessive diametral strains.[18]

At extended burnups, there is now evidence that the fission-gas release can be higher than expected from low-burnup extrapolations, or from codes developed for low burnup. In LWR fuel, fission-gas releases larger than expected have been reported for both steady-power and transient conditions.

There appears to be a burnup-, and possibly linear-power-threshold at about 20 MWd/kg, above which enhanced fission-gas release (up to 28%), grain growth and diametral strain (up to 2%) may be observed, with defects due to stress-corrosion cracking.[18] The inference is that the extended-burnup effects are associated with a degradation in fuel conductivity and/or fuel-to-sheath heat transfer, resulting in acceleration of thermally activated processes. The primary defect cause in the extended burnup Bruce bundles mentioned above was stress-corrosion-cracking failure through the sheath at the braze heat-affected-zone of the inboard end-cap weld. The grain growth and fission-gas release in the two high-burnup bundles were typical of fuel with operating temperatures higher than expected from the power histories of the two bundles. However, these observations need to be confirmed by examination of a statistically significant sample.

A power-ramp test was recently performed on fuel from the Nuclear Power Demonstration (NPD) reactor as part of the AECL extended burnup program.[19] The fuel had a burnup of 35 MWd/kg at a declining power history, and the NRU reactor at Chalk River was used to ramp this fuel to about 36 kW/m. The NRU ramp proved sufficient to cause significant fission-gas release and diametral strain in the fuel elements, and although no defects occurred, incipient cracks were detected in the fuel sheaths. There is thus insufficient fuel performance data for the 37-element bundle to be assured of acceptable performance at extended burnup, particularly with load-following.

Technical improvements could be made to the internal design of the 37-element bundle to meet the performance challenges of extended burnup, such as improved sheath coatings, zirconium-barrier sheath, optimized pellet configuration and density, or graphite-discs.

In analyzing extended-burnup performance requirements, the assessment was that most long-term benefits for CANDU would be achieved with a new bundle design. A new bundle design with wide application in current and future CANDUs, for natural uranium or advanced fuel cycles, could be proven with the same development and demonstration effort required for a modified 37-element design.

3.2.2 The CANFLEX Bundle[2, 20]

The new CANFLEX bundle has 43 elements, and features two element sizes. CANFLEX is a logical extension of existing technology: progressive CANDU fuel designs have increased the number of elements in a CANDU bundle from 7 to 19 to 28 to 37, and now to 43 (Figure 14). The greater subdivision is also in line with technology trends in advanced LWR designs. Compared with the current design, CANFLEX will provide about a 20% reduction in peak linear heat ratings, which will enhance its burnup capability. The bundle is compatible with existing and future (for example, CANDU 3) fuel-handling systems. Some key points showing the wide application of CANFLEX are:

- in current and future reactors, lower element ratings at current bundle power,
- in future reactors, power uprating without exceeding current element ratings,
- achievement of extended burnup facilitated by option of lower element rating, and optimization of internal design,
- natural or enriched fuel,
- increased operating and safety margins, and
- power manoeuvring facilitated.

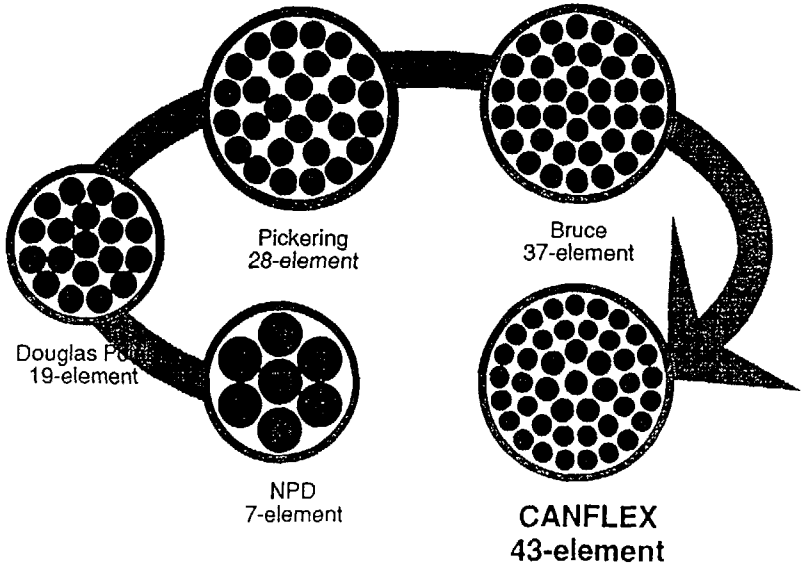

Figure 14: CANDU Bundle Evolution

The CANFLEX Development Program has been underway at the Chalk River Laboratories of AECL Research since 1986 February. It comprises performance testing, manufacturing, thermalhydraulics confirmation, reactor physics and endurance/handling tests. An important component of the CANFLEX product is the data base and modelling capability generated during development, to support large-scale manufacture and future licensing. The CANFLEX advanced fuel bundle has wide application in current and future CANDUs, and is the optimal carrier for advanced fuel cycles.

3.3 Licensing Issues

AECL has performed several safety studies with SEU fuel, both internally and for various clients, and these have not identified any safety problems. In terms of releases during potential accidents (such as loss-of-coolant), the higher absolute inventory of fission products (due to increased burnup) is balanced by a smaller fractional release (due to lower element ratings with the CANFLEX fuel bundle, and fuel-management schemes featuring a declining power history with burnup).

4. IMPACT OF EXTENDED BURNUP IN CANDU ON THE FRONT END OF THE FUEL CYCLE

4.1 SEU

CANDU reactors currently use natural uranium UO_2. The use of SEU would require additional steps in the production of CANDU fuel: conversion of U_3O_8 to UF_6, enrichment, conversion of UF_6 to UO_2, followed by fabrication of enriched UO_2 fuel. Conversion of U_3O_8 to UF_6 is provided by CAMECO in Canada. A facility would be required to convert the UF_6 product from the enrichment plant to UO_2. While this has been done on a small scale in the past in Canada, a new facility would likely be required to handle large volumes. While enrichment services are available commercially in the world, CAMECO is currently assessing the feasibility of the CRISLA (Chemical Reaction by Isotope Selective Laser Activation) enrichment process[21]. The availability of a domestic enricher in Canada would make the use of SEU even more attractive, and should be viewed as an industrial opportunity for Canada.

At the fuel fabrication plant, changes would be required to manufacture CANDU bundles from SEU (for example, to ensure criticality limits). Since enriched fuel is routinely fabricated by PWR fuel fabricators, this is not seen to be a problem.

The use of the CANFLEX bundle for SEU will also impact on fuel fabrication. There are two element sizes, compared to one in current CANDU bundle designs. However, the number of distinct element types (including number and orientation of bearing pads and spacer pads) may be similar to the 37-element bundle, depending on the final design. The fabrication costs for the 28-element and 37-element bundles are similar, so one would expect similar costs for the CANFLEX bundle in equilibrium, although there will be initial investment costs to tool-up for a new bundle design. Fuel cycle calculations for SEU conservatively assume a 20% cost penalty for CANFLEX fabrication.

While the use of extended burnup will reduce the quantity of CANDU fuel bundles required from fabricators in the long term, it is anticipated that this would be compensated by increased nuclear demand.

4.2 RU

The radiological implications of fabricating CANDU fuel from RU containing 0.8-0.9% U-235 have been examined.[22] An important conclusion was that the fabrication of highly purified RU presents no more of a hazard from internal or external exposures than does the fabrication of fresh LWR fuel. Although Cs-137 and Sr-90 are fission-product impurities in RU, no particular difficulties are foreseen in CANDU fuel fabrication, as long as their concentrations are low.

AECL is currently engaged in a cooperative program with COGEMA of France to assess the fabricability of CANDU fuel from RU. Of particular interest is whether Cs-137 would be driven off during sintering, and condense in the colder parts of the furnace, resulting in increasing fields over time. The experimental program will confirm that this is not a concern.

5. IMPACT OF EXTENDED BURNUP ON BACK END OF THE FUEL CYCLE

Increasing the burnup of CANDU fuel by a factor of two or three would reduce the discharge rate of irradiated fuel from the reactor by the same factor. This significant reduction in the volume of spent fuel produced could be a positive step in increasing public acceptance of nuclear power.

There are also economic incentives from the back-end of the fuel cycle to use SEU fuel in CANDU. Figure 15 shows that disposal costs (per unit energy) are reduced significantly with SEU. For the geological disposal concept developed by AECL, with spent natural-uranium fuel, the separation of the bore-holes in the disposal vault is limited by the physical properties of the rock. With SEU, the separation of the bore-holes is limited by the heat generated by the used fuel. Hence, there is an economic incentive (from the perspective of disposal) for longer cooling periods with SEU (such as 50 years). This has to be balanced with increased storage costs.

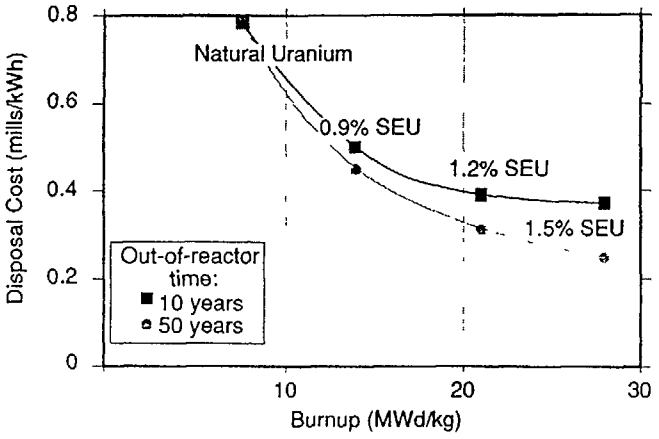

Figure 15:
Effect Of Burnup On Disposal Cost

6. CONCLUSIONS

There are resource and economic incentives for extending fuel burnup in CANDU through the use of SEU or RU. With SEU, natural uranium requirements are reduced by 20-40%, and fuelling costs by 20-35% relative to natural uranium fuelling, depending on the enrichment level and assumptions. The optimum enrichment is about 1.2%. The use of RU in CANDU is very attractive: double the energy can be extracted by burning the RU in CANDU compared to re-enrichment and use in a PWR. The potential savings in fuelling costs with RU are substantial.

The on-line method of fuelling CANDU provides both opportunities and challenges for accommodating enriched fuel. The opportunity is in the flexibility of using a variety of fuel management strategies; the challenge is in designing a fuel management scheme that ensures a declining power history with burnup. With 0.9% SEU (or RU), a simple 4-bundle shift fuelling scheme is acceptable. With 1.2% SEU, more imaginative fuel management schemes are available, such as axial shuffling or checkerboard.

At extended burnups, there is now evidence that fission-gas release can be higher than expected from low-burnup extrapolations. There also appears to be a burnup-, and possibly linear-power-threshold at about 20 MWd/kg, above which enhanced fission-gas release, grain growth and diametral strain may occur, leading to defects due to stress-corrosion cracking. The CANFLEX bundle is being developed to provide assurance of good fuel performance at extended burnup in CANDU.

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NECESSITY FOR AND FEASIBILITY OF BURNUP EXTENSION IN CHINA'S NUCLEAR INDUSTRY

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Abstract

The incentives to extend burnup are present in China, even the nuclear power development is at the starting stage. The studies for extending burnup is based on that the civilian nuclear power is the priority of the nuclear industry and that a fair-sized fuel industrial complex has been formed in China.

The Qinshan NPPs' reactor design and in-core fuel management are briefly presented. In the case of Qinshan NPP-1, the fuel cycle economic evaluation using the common economic input parameters are conducted. The study results show that extension burnup is very beneficial for reducing the costs of the fuel cycle and the utilization of nuclear resource.

A Graduate Research and Development Programme for extended burnup is taking place in China.

1. INTRODUCTION

The Nuclear Power has been playing important role in the woldwide energy electricity generation.

The safety and economy are the major issues for the development of nuclear power. One of the important areas is the effective utilization of nuclear resource and reducing nuclear power costs.

The burnup extension is an interesting study subject in the nuclear industry of China as around the world, even the nuclear power development is still at the starting stage. The study is based on that the civilian nuclear power is the priority of the nuclear industry and that a fair-sized fuel industrial complex has been formed in China.

The technology for PWR is adopted for the commercial nuclear power plants.

The Qinshan project is a Chinese designed and constructed PWR NPP [1] with 300 MWe capacity and a design burnup 24.0 MWd/KgU for early stage.

The project will contribute to gaining experience and mastering the technology for further development of nuclear power in China.

The Daya Bay project is an imported PWR NPP with 2x900 MWe capacity and a warranted design discharge burnup of 33 MWe/KgU. There are possibility and necessity of extension burnup for both NPPs, for the benefit of economy and technology.

During the process of reactor design and nuclear fuel Research and Development, some works on fuel performence, in-core fuel management as well as economic evaluation of fuel cycle cost have been already done.

A high burnup programme is in place in China.

2. QINSHAN NPP'S REACTOR DESIGN AND FUEL MANAGEMENT

2.1. Reactor Nuclear Design

The Qinshan NPP belongs to the small size PWR plant, but its features are nearly the same as the current PWR type plants in the world. [2]

The main characteristics of Qinshan Reactor are listed as follows:

Electrical output, MWe	300
Thermal output, MW	966
Core activity height, in	2.9
Equivalent core diameter, m	2.486
Number of fuel assemblies	121
Rod array	15x15
Fuel weight, UO2 (U), t	40.746 (35.917)
Water to fuel volume ratio	2.065
Average LHGR, W/cm	135
Maximum LHGR, W/cm	407
Fuel enrichment (first core), w/o	2.04; 2.67; 3.00
Fuel enrichment (refueling), w/o	3.0 - 3.4
Equilibrium cycle burnup, MWd/KgU	24 - 30

The low specific rating, therefore, the low LHGR is its obvious characteristic. The core design is a comprehensive result of many factors, since the core design involves trade-offs between the nuclear, thermal hydraulic and mechanical requirements. So the conservative design philosophy is adopted. The low LHGR design is not only beneficial to the safety and operational flexibility but also to fuel management, it offers plenty of room for various loading patterns, being able to accomplish multi-region loading and multi-batch refueling.

2.2. In-core Fuel Management

For the history reason, a rather low discharged burnup was designed. After gaining operation experiences, higher burnup should be achieved.

The nuclear fuel for the Qinshan NPP is designed and manufactured by Chinese own effort and a vast amount of Research and Development have been conducted [3]. The fuel in-pile testing has been completed [4].

The systematical analysis models and corresponding computer programs have been developed for fuel design, core design and in-core fuel management.

The works [5] which were already finished for the Qinshan's fuel management are:

- Refueling calculation for cycles 1 to 8 with burnup analysis, in the mode of Out-In-In

- Study for extending the burnup utilizing the moderator's temperature effects and fuel's Doppler effects at EOL

- Analysis of the effects of increasing the equilibrium refueling U-235 enrichment

- Improving burnable poisons using B.Silicate Glass instead of B.S.S.

- Fuel management analysis for Low Leakage(LL) and Very Low Leakage(VLL), both in the modes of In-Out-In and In-In-Out

- Low parasite design of the structural materials

- Optimization of in-core fuel management

3. ECONOMIC EVALUATION FOR FUEL CYCLE COSTS

There are some analysis models and codes for fuel cycle economic have been developed by various institutions [6] [7]. Among these models, the same principle - levelized discount cost model, is used.

The basic factors which impact on the costs of fuel cycle are:

- the basic price of material, processing and services
- economic parameters, and
- technical performance parameters.

According to the model and data recommended by IAEA, some evaluations for the fuel cycle costs are being conducted.

The analysis model used was checked by the benchmark, and the results consist with IEAE's result.

For example, in the case of Qinshan NPP-1, the evaluation results adopting common economic input parameters are as follows:

Z	Q (KgU)	T (Years)	Bu (MWd/KgU)	C5 (w/o)	Cost (Mills/Kwk)
121/40	11874	3.03	23.43	2.90	9.104
121/30	8906	4.03	31.24	3.30	7.640
121/20	5937	6.05	46.86	4.35	6.693
121/15	4453	8.07	62.48	5.40	6.325

where Z - ratios of core load fuel assemblies to reload fuel assemblies;

Q - batch U invetory, KgU;

T - fuel residence in-core time, years;

Bu - Discharge burnup for equilibrium cycle, MWd/KgU;

C5 - reload enrichment, w/o

In this study, the technical parameters are taken from Qinshan-1 design value as above, and the common econmic input parameters of WREBUS report is adopted.

For a given cycle energy generation (e.g. a 12M cycle and 300 MWe output), as the discharge burnup increases, the equilibrium batch size decreases, the reload enrichment increases and the average in-core residence time increases. The trends in fuel cycle cost then become a trade-off among the opposing trends of the various cost components.

The direct cost, generally, is decreased with the extended discharge burnup, provided the reload enrichment is not too high.

However, the indirect costs trend is usually opposite because the effect of the energy discount factor is unfavourable when the discharge burnup is increased.

In addition to sensitivity study for the basic price of various material and processing, the economical and technical analysis are necessary.

From the technology's point of view, the study should emphasize on the high burnup fuel performance and the feasiblity of the fuel management for such a high discharge burnup.

4. A HIGH BURNUP RESEARCH AND DEVELOPMENT PROGRAMME

The study results of other countries and China show that the extension burnup is very beneficial for reducing the fuel cycle's costs and the utilization of nuclear resouce.

According to the status and development requirement of nuclear industry in China, a Research and Developemnt programme on high burnup is proceeded.

A 3-phases Research and Development plan was suggested [8].

The objectives and developing stages are as follows:

Stage	Objective	Year
Phase-1	25 - 33 MWd/KgU	1996
Phase-2	40 - 45 MWd/KgU	- 2000
Phase-3	50 - 60 MWd/KgU	- 2010

The Research and Development on extension burnup consists of basic research, specific subjects and comprehensive studies.

The basic research includes:

- economic analysis of fuel cycle costs
- fuel performance data bank establishment
- computer packages development and improvement
- set up the standard and codes
- measurement, inspection techniques and facilities development

The specific subjects consist of:

- developing low parasite design
- developing Gd2O3-UO2 fuel
- waterside corrosion
- fuel in-pile testing

The comprehensive studies consist of:

- testing of fuel assembly performance
  - themal-hydraulic testing
  - mechanical testing
  - compatiblity testing
- lead fuel testing in reactor
- PIE
- advanced fuel management : optimatization of refueling and reactivity control
- coast-down operation

Among the subjects mentioned before, the most important ones are:

- advanced burnable poison Gd2O3 - UO2
- waterside corrosion
- advanced fuel management

Nevertheless, the most valuable issue of the programme is to accumulate practical experiences from the Qinshan and Daya Bay NPPs operations as to examine the fuel performance and to check the analysis models.

5. CONCLUSION

For development of civilian nuclear power programme, obviously, mastering the technology of extension burnup is not only necessary but also possible in China.

A gradual Research and Development programme for high burnup will be accomplished by our own efforts, and the international cooperation is also needed.

Since the international technology exchange and cooperation supported by IAEA is very successful and beneficial, the related activities should be continuely conducted.

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PRESENT STATUS OF THE NUCLEAR FUEL CYCLE IN THE ČSFR

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Abstract

The paper gives an overview of present fuel cycle activities in ČSFR and perspectives at its development. Discussed are results of preliminary evaluation of alternative fuel cycle options. Main attention is devoted to technical and economical problems of extended burnup, with special emphasis on VVER reactors. The most important part for further consideration is the backend of the nuclear fuel cycle, especially for the once-through alternative.

1. INTRODUCTION

The IAEA Water Reactor Extended Burnup Study has indicated strong economic incentives for burnup extension in LWR's. It has been shown that the main driving force is the backend cost, alike for whether once-through cycle and closed cycle.

The IAEA International Conference on the Safety of Nuclear Power [1] discussed the problems of final disposal of high level waste, and concluded that burnup extension is the one possibility how to reduce the spent fuel's amount. Also situation in long-term spent fuel storage could be improved with burnup extension.

The nuclear fuel cycle in ČSFR is now being evaluated from different reasons, both economical and strategic. Because no final decisions were yet reached, the paper gives only a review of present situation and the preliminary evaluation of some alternative.

The technical problems connected with burnup extension were already discussed within the WREBUS report and are included here for discussion purposes.

2. PRESENT SITUATION AND PERSPECTIVE

At present there are following NPPs in operation or under construction in the ČSFR [2]:

Power Plant				Start-up	Spent Fuel tU/a	Pool Capacity tU
NPP Bohunice	unit 1	VVER	440/230	04.79	13.97	38.34
NPP Bohunice	unit 2	VVER	440/230	06.80	12.53	38.34
NPP Bohunice	unit 3	VVER	440/213	11.84	13.97	38.34
NPP Bohunice	unit 4	VVER	440/213	09.85	13.97	38.34
NPP Dukovany	unit 1	VVER	440/213	05.85	13.97	38.34
NPP Dukovany	unit 2	VVER	440/213	03.86	13.97	38.34
NPP Dukovany	unit 3	VVER	440/213	12.86	13.97	38.34
NPP Dukovany	unit 4	VVER	440/213	07.87	13.97	38.34
NPP Mochovce	unit 1	VVER	440/213	12.93	13.97	71.9
NPP Mochovce	unit 2	VVER	440/213	09.94	13.97	71.9
NPP Mochovce	unit 3	VVER	440/213	09.95	13.97	71.9
NPP Mochovce	unit 4	VVER	440/213	04.96	13.97	71.9
NPP Temelín	unit 1	VVER	1000/320	05.94	23.36	273.05
NPP Temelín	unit 2	VVER	1000/320	11.96	23.36	273.05

All reactors under operation operate with 3 year cycles (some low-burnup fuel assemblies (FA) are in core for 4 years). The reactor and fuel design parameters are in TABLE 1 [2,3]. It is assumed that reactors under construction will start with three year cycles.

TABLE 1

Parameter	VVER 440	VVER 1000
Cladding outer diameter, mm	9.1	9.1
Cladding minimum thickness, mm	0.6	0.63
Fuel cladding diametral gap, mm	0.12-0.27	0.16-0.27
Gas plenum volume, cm^3	4	11
Fuel rod length, mm	2570	3840
Fuel column length, mm	2420	3510
Minimum fuel density, g/cm^3	10.4	10.4
Central hole diameter, mm	1.2	1.4/2.4
Fill gas	He	He
Fill gas pressure, MPa	0.1/0.5	2.0-2.5
Reactor thermal power, MW	1375	3000
Reactor electric power, MWe	440	1000
Coolant pressure, MPa	12.5	16.0
Coolant temperature - inlet, °C	268	290
Coolant temperature - outlet, °C	296	320
Mean linear heat rating, kW/m	12.7	16.7
Maximum linear heat rating, kW/m	33	49
Average burnup, MWd/t	30000	28000/40000
UO_2 inventory in fuel rod, kg	1.082	1.575
Number of fuel rods in assembly	126	312
Number of fuel assemblies in core	349	163
U inventory in assembly, kg	120.2	433.2
U inventory in core, t	41.9	70.6

Up to now the front-end operations for all VVER reactors (as for other former COMECOM countries) are provided by the USSR. This monopoly position is mainly due to the differences between VVER's FA and core and those of other PWRs. At present, some diversification in this respect is under consideration.

The original negotiated conditions of the transport of spent fuel to the USSR after 3 years storage were 10 years ago modified to prolonged storage (about 6 years) which forced the decision to construct Intermediate Spent Fuel Storage Facility (IMSFSF). Due to the lack of the spent fuel storage capacity before startup of the IMSFSF, a certain number of FAs was (after corresponding negotiations) transferred to the USSR. By the end of 80's the overall situation has drastically changed and long term storage or other alternative solutions are necessary. The present spent fuel storage capacity will be exhausted in 1994. Several options to solve this problem are under consideration, the most probable being construction of IMSFSF at Dukovany. The spent fuel storage capacities are following:

Facility	Capacity tU	Start-up
IMSFSF Bohunice	600	01.86
IMSFSF Dukovany	~ 2000	~ 01.94

As the delay in planned start-up of IMSFSF Dukovany could have unfavorable consequences, a preliminary evaluation of several alternatives of the whole nuclear fuel cycle (under the light of new economical and political conditions) was performed. Four alternatives were assessed [4] (Fig.1-6.):

- once-through fuel cycle with three year cycles,
- once-through fuel cycle with four year cycles,
- closed cycle with early reprocessing, and
- closed cycle with delayed reprocessing.

The preliminary conclusion are that extended burnup is the most direct way to reduce fuel cycle cost. The decision of choice between once-through and closed cycle is rather speculative and is influenced mainly by three important factors:

- relatively well defined and high cost of reprocessing and Pu and U recycling, therefore taken in the time they arise,
- undefined cost for spent fuel final disposal and for RA waste final disposal, therefore in some countries immediate political decision about financing this by a fee at spent fuel discharge from reactor core was considered, and
- not clearly seen time of final spent fuel or RA waste disposal.

Because of above mentioned factors and not easily chosen long term-value of discount rate, time schedule and financial problem issues make an actual problem.

Under above assumptions the ranking of alternatives is as follows·

1 Closed cycle with delayed reprocessing
2. Once-through fuel cycle with three year cycles
3. Closed cycle with early reprocessing

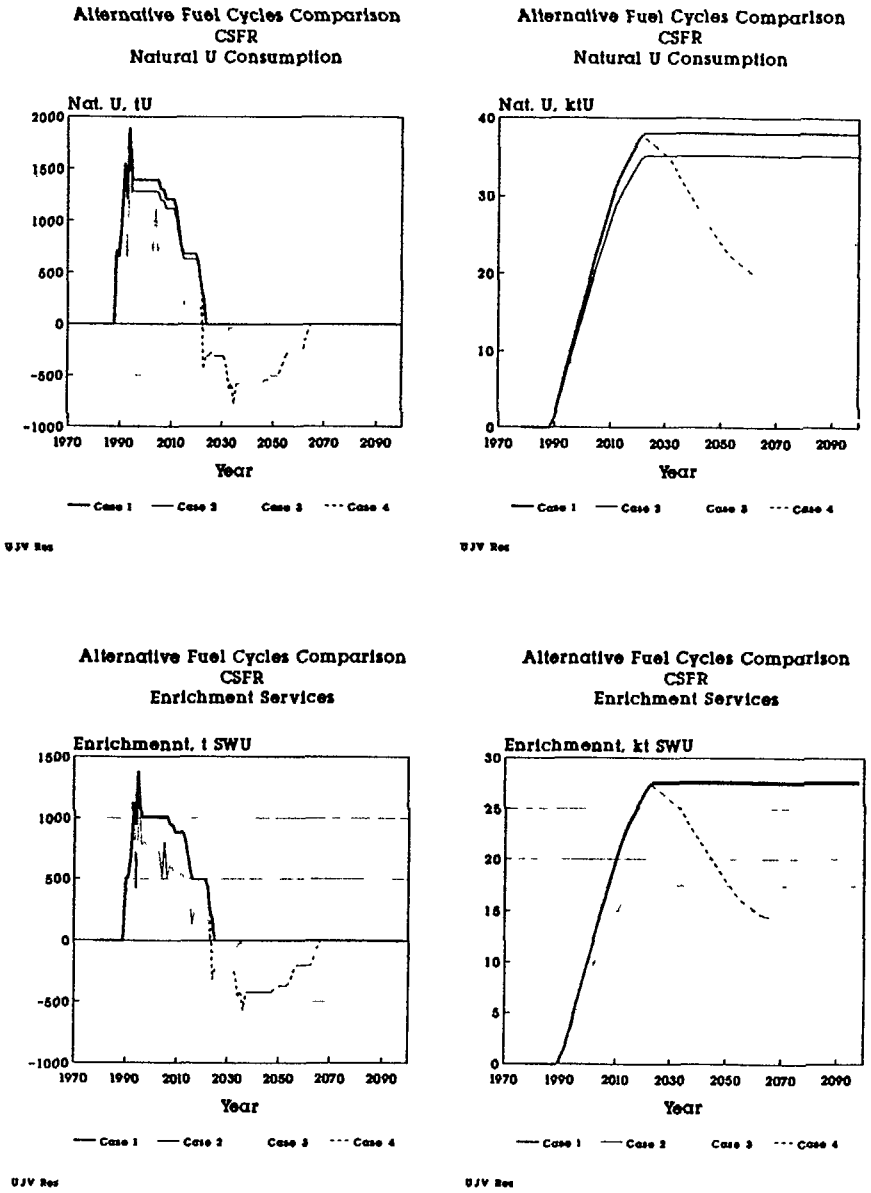
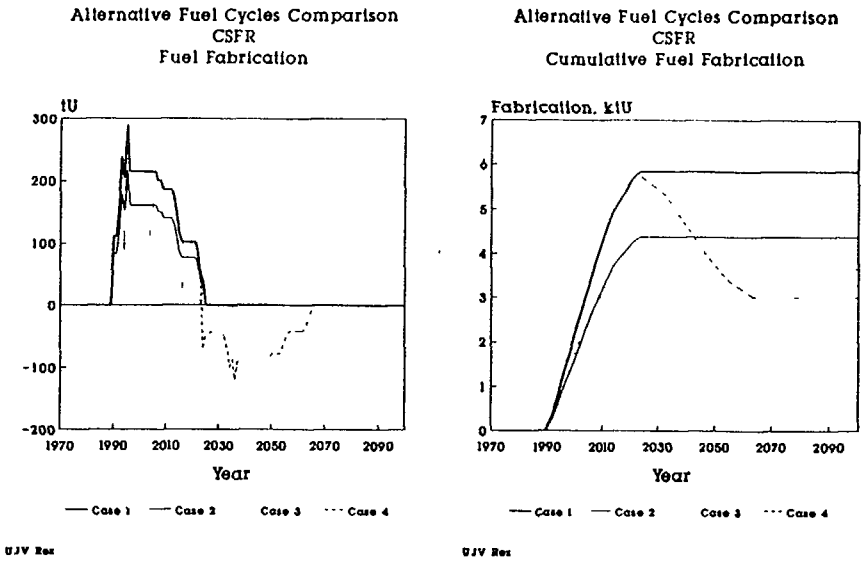


Fig. 1. Alternative Fuel Cycle Comparison for ČSFR



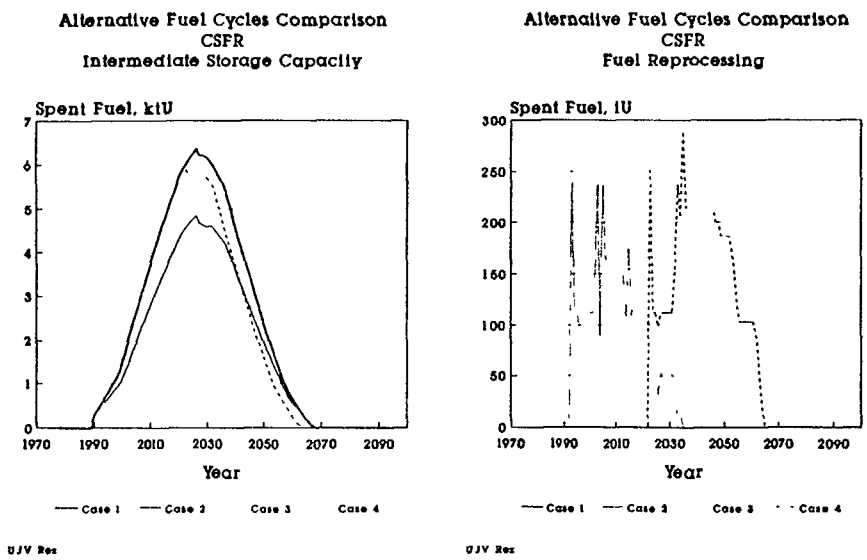
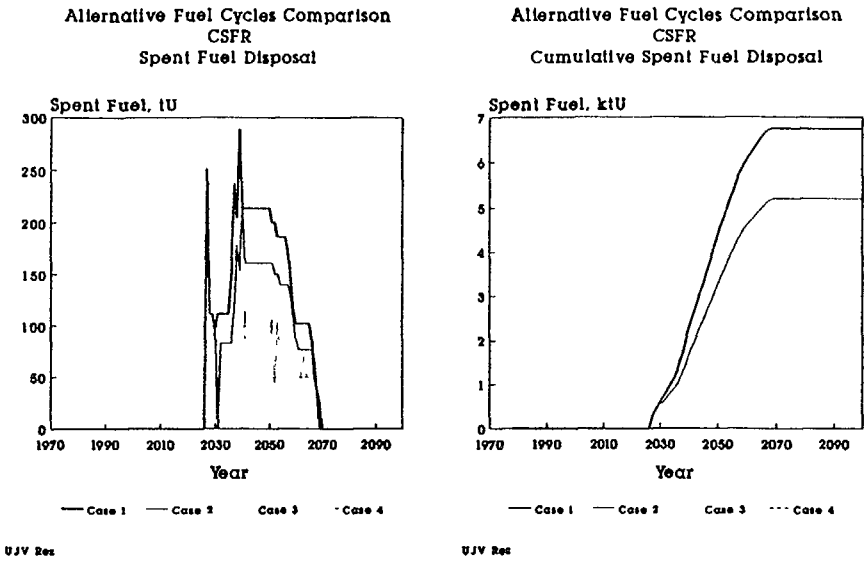


Fig. 2. Alternative Fuel Cycle Comparison for ČSFR



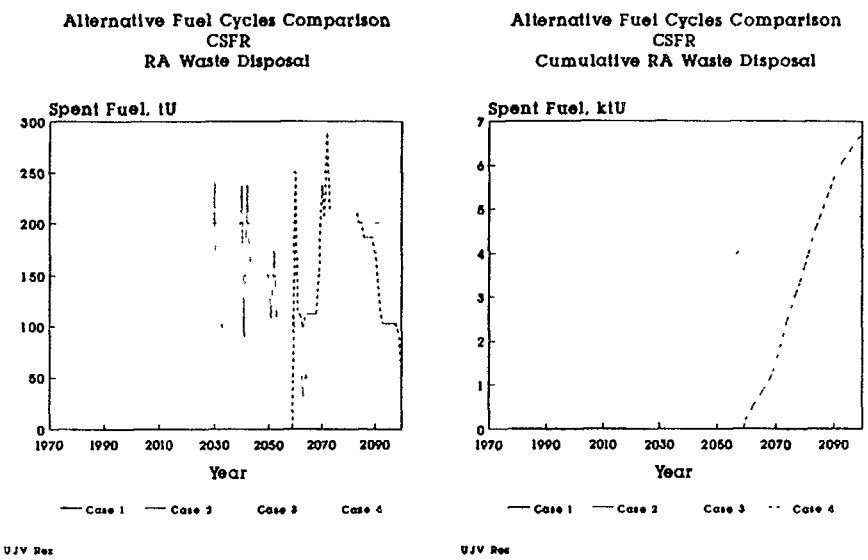
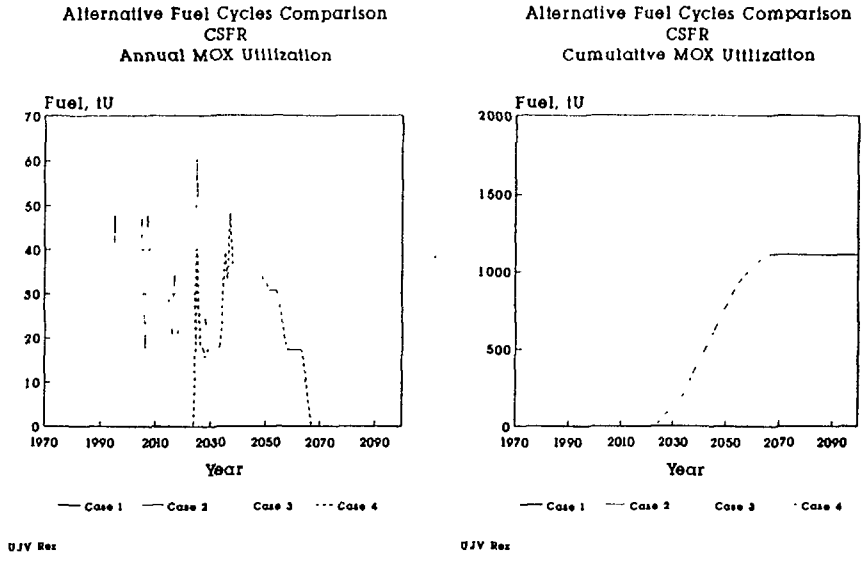


Fig. 3. Alternative Fuel Cycle Comparison for ČSFR



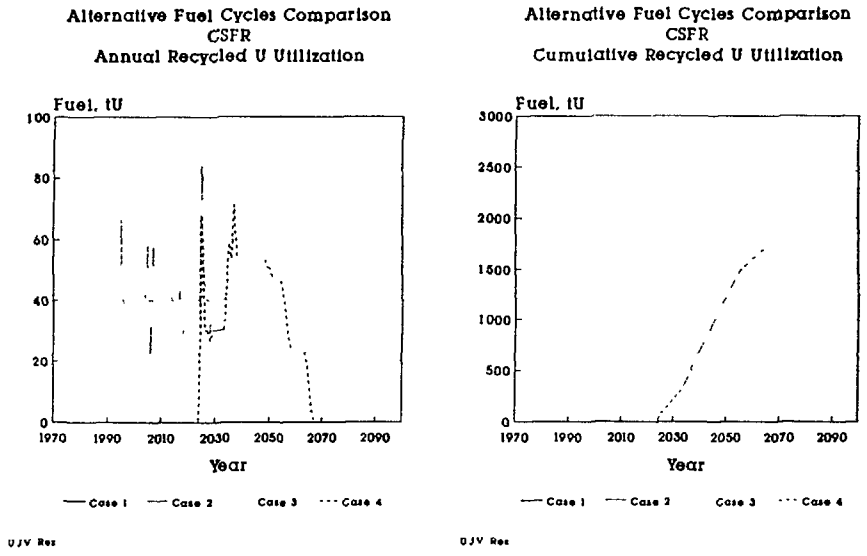
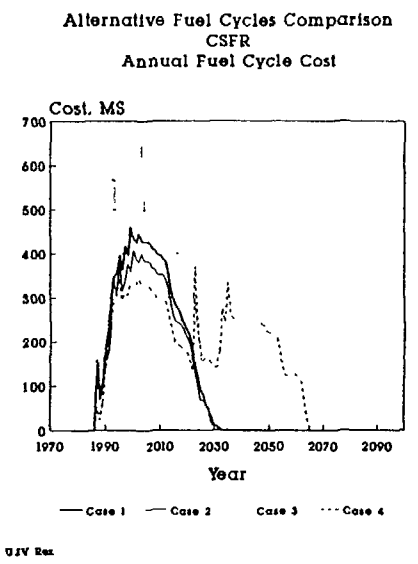
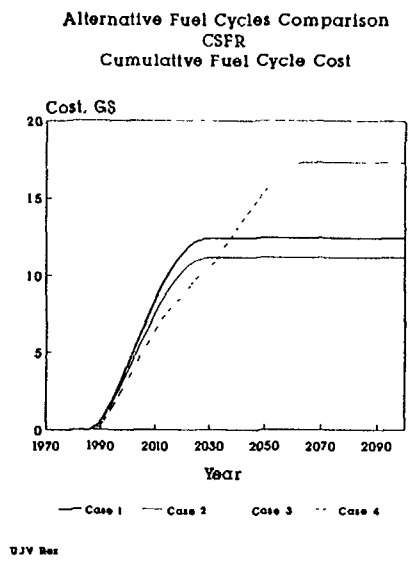
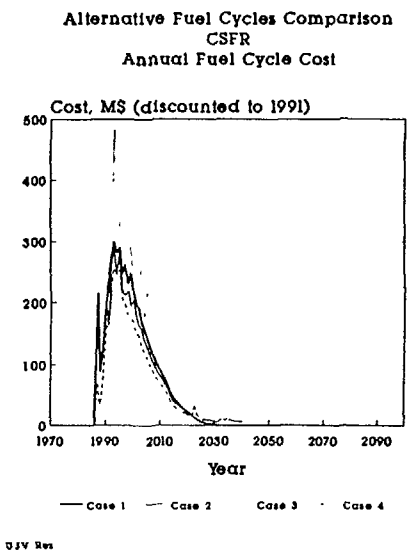


Fig. 4. Alternative Fuel Cycle Comparison for ČSFR







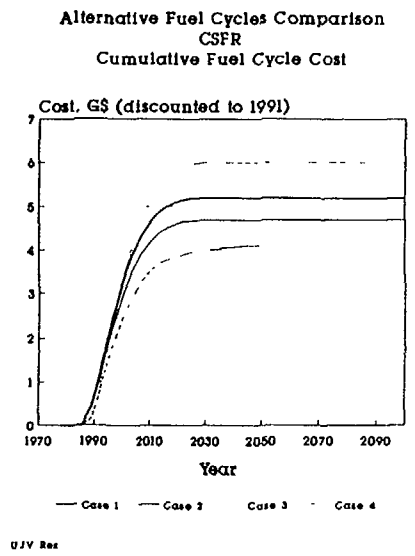


Fig. 5. Alternative Fuel Cycle Comparison for ČSFR

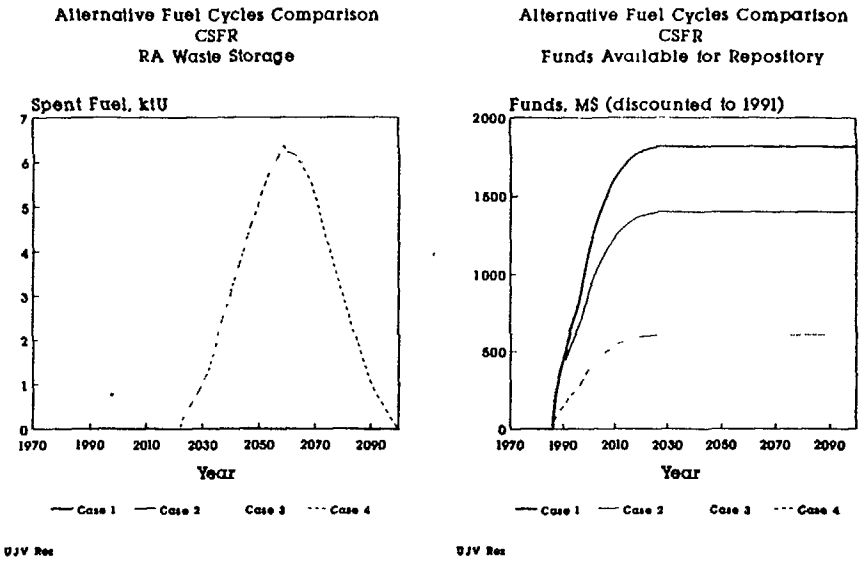


Fig. 6. Alternative Fuel Cycle Comparison for ČSFR

More detailed analyses are however necessary. More detailed discussion on the economical analysis is presented in chapter 4.

3. EXTENDED BURNUP IMPACT ON THE NUCLEAR FUEL CYCLE

The economical aspects of extended burnup in different LWR reactors, including both VVER 440 and VVER 1000 have been performed under the IAEA Water Reactor Extended Burn Up Study - WREBUS [5,6,7]. Included were also the main technical, licensing and environmental aspects. Therefore this paper deals only with main differences and requirements for more detailed analyses. The possibilities of such analyses depend however on the decisions on the backend options, therefore both once-through and closed cycles alternatives must be treated. The resolution between main fuel cycle options will be different in different states, the approach to such a decision should be however more general and a matter for discussion.

The detailed impacts could be divided into three parts:
- front-end,
- design and operation, and
- back-end.

3.1. FRONT END

Majority of the front-end activities are not covered in Czechoslovakia, nevertheless the economic impact of possible technical problems is also not without importance. The main technical problems could be:

- increase of required enrichment,
- fabrication of high enriched, high burnup fuel assemblies:
  - enrichment above licensed enrichment value,
  - application of burnable absorbers,
- fabrication of MOX fuel,
- enrichment of recycled U,
- fabrication of FAs with recycled U.

Specific problems of VVER reactors could be connected with new spacer grids.

Of interest is also fuel recycling in other advanced type reactors (e.g. CANDU)

3.2. DESIGN AND OPERATION

The main possible problems to be solved in connection with extended burnup were discussed within the WREBUS Study [3]:

- design of extended burnup FAs,
- testing of lead FAs,
- reactivity coefficients
- shutdown margin
- power distribution
- fuel design optimization
- cladding corrosion
  - hydrogen pick-up
  - dimensional problems
  - fission gas release
  - spacer grids
  - cladding collapse
- fuel design - MOX, recycled U
  - burnup impact
  - recycled U
- licensing problems

In distinction from PWR's, for VVER reactors the problem of cladding corrosion is not important, at least not for VVER-440 (because of better properties of Zr1Nb, lower temperatures and different water chemistry). Most important will be the fission gas release limits for both VVER-440 and VVER-1000. In addition, for VVER-1000 the reactivity coefficients, shutdown margin and cladding collapse are to be properly analyzed. In more details the VVER problems were discussed in [8,9].

3.3. BACKEND

Similar as for the design, also for the backend, the main possible problems to be solved in connection with extended burnup were discussed within the WREBUS Study [3]:

- spent fuel pool design
  - reactivity
  - heat removal
  - shielding
- intermediate storage fuel facility
  - optimum technology,
  - reactivity
  - heat removal
  - shielding
  - cladding integrity
- spent fuel transport
  - reactivity
  - heat removal
  - shielding
- final spent fuel disposal
- fuel reprocessing
  - burnable absorber impact
- RA waste storage
- final RA waste disposal

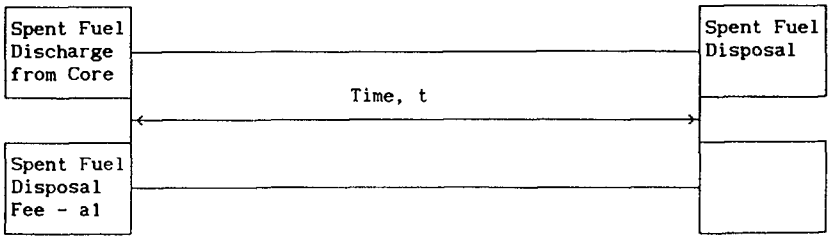
The most important is clearly the backend cost and its burnup dependence, and in case of once-through cycle the scaling factor is also of consequence.

4. BACKEND POLICY - MAIN PROBLEMS

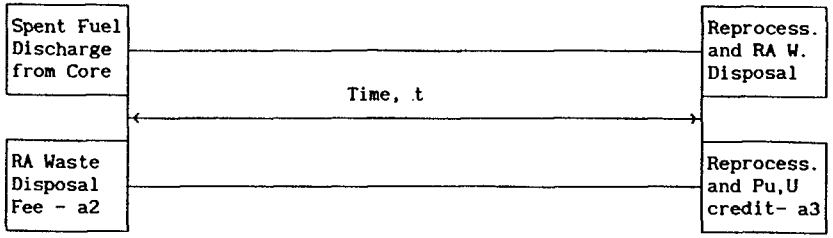
As was shown in the WREBUS Study, the most important factor to consider for burnup extension is the backend cost. However, backend is also the part with highest both technical and economical uncertainties. As it was already discussed in Chapter 2, the economical comparison of both once-through and closed cycle alternatives (because of different approach to payments: in the case of final disposal the payments are made in advance, in the case of reprocessing - such payments are in the time of fuel arrival to the reprocessing plant and uncertainties) is very difficult.

Let us assess the two alternatives:

Once-through cycle:



Closed cycle:



a1 [$/kgU] - Spent fuel disposal fee paid at the time of fuel discharge from core
a2 [$/kgU] - Ra waste disposal fee paid at the time of fuel discharge from core
a3 [$/kgU] - Reprocessing cost minus credit for Pu and U recycling (or plus penalty for their recycling)
t [years] - Time between fuel discharge and final solution (either final FA's disposal or RA waste final disposal)
i [%] - discount rate

Both cycles will be economically equal if:

$$a1.(1+i/100)^t = a2.(1+i/100)^t + a3 \qquad (1)$$

Solving this expression for t gives:

$$t = \log [(a3/a1)/(1-a2/a1)]/\log(1+i/100) \qquad (2)$$

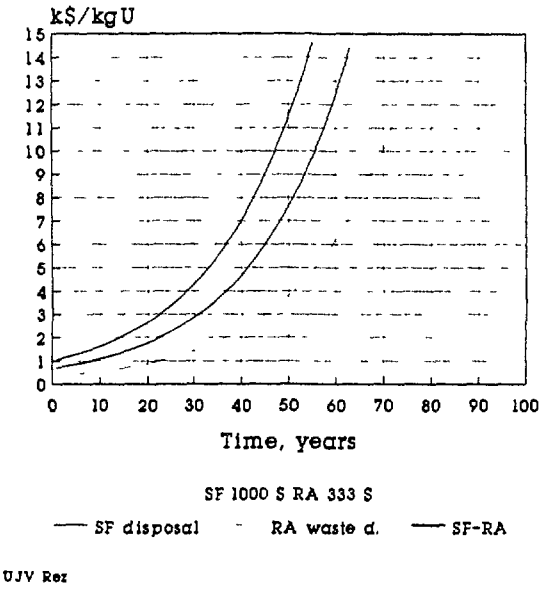
For the time < t the once through cycle is more attractive and for the time > t the closed cycle is more attractive.

The possible range for a2/a1 is 0.1-0.9 (the final disposal of RA waste from reprocessing must be cheaper than final disposal of spent fuel).
The possible range for a3/a1 is 1 - 10 (if the ratio is < 1, the fuel reprocessing will be always cheaper).

It is implicitly assumed, that for reprocessing either different fee is set, or the difference will be made available to utility in the time t in the corresponding amount $[(a1-a2).(1+i/100)^t]$.

Fig. 7 shows an example and Fig. 8 and 9 give the parametric evaluation of the equation (2).

This understanding is very important for both the utilities to make proper decision and for the state policy by proper setting the backend fee, for not to push the utility to solution which could be, from global point of view, less economical.



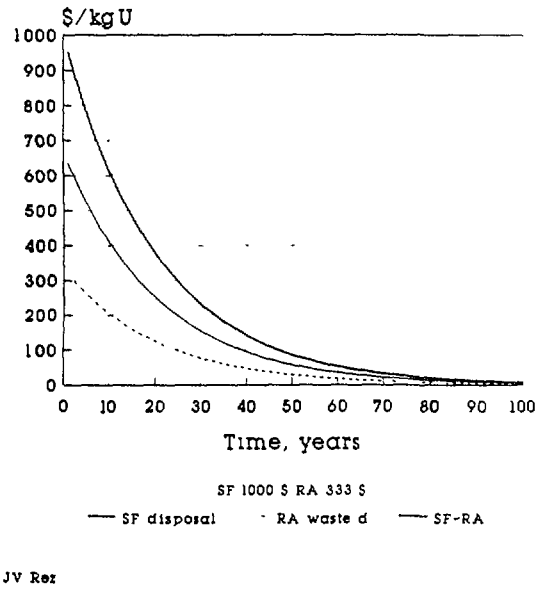
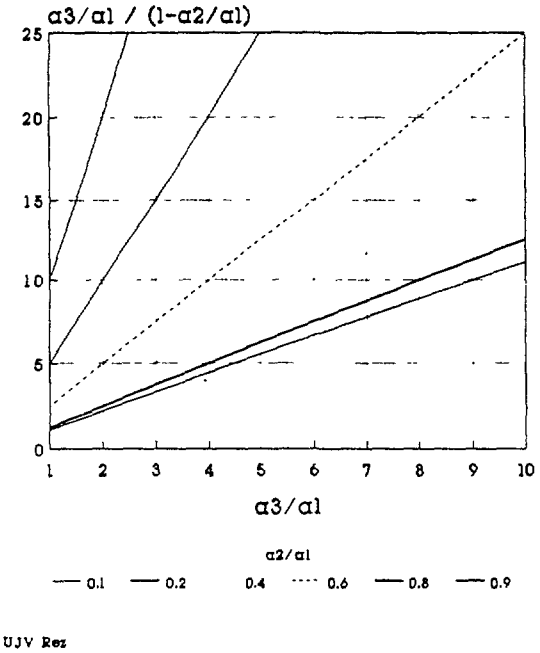


Fig. 7. Forward and Backward Discaunted Values of SF and RA Waste Disposal Cost



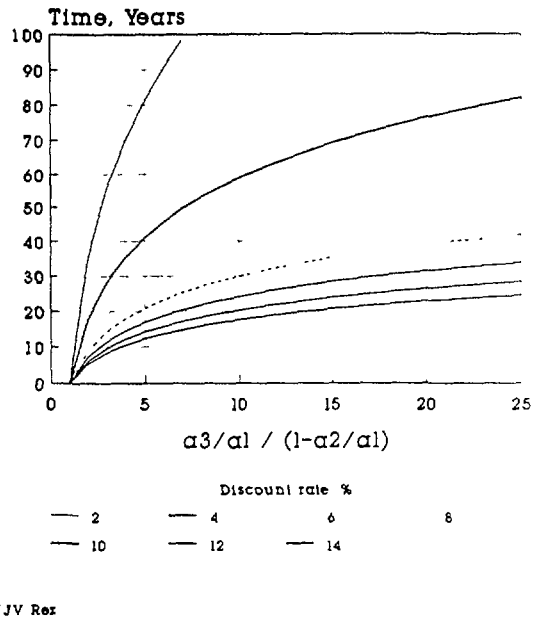


Fig. 8. Dependence of a3/a1/(1-a2/a1) on realtive backend costs (a3/a1 and a2/a1)

Fig. 9. Minimum Storage Time for Reprocessing to Be Economical

5 CONCLUSIONS

At present the average fuel burnup in Czechoslovak VVER reactors is 30 MWd/kgU with 3 year cycles There are incentives to implement 4 year cycle with burnup ~ 40 MWd/kgU Further burnup extension is of interest, but with the understanding of the main constrain of the lead FAs experimental verification

Slightly different situation will be with VVER-1000 These reactors are supposed to start with 3 year cycle, and corresponding burnup ~ 40 MWd/kgU and the perspective of extension to ~ 50 MWd/kgU with 4 year cycle, is already formulated

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NEAR-TERM PLANS TOWARDS INCREASED DISCHARGE BURNUPS IN FINNISH POWER REACTORS

Review of background aspects and current activities

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Abstract

Fuel cycles for the two reactor types used in Finland, 2 x VVER-440, 2 x ABB Atom BWR, are described. The current burn up limits and some of the technical aspects affecting the limits now and in the near future are reviewed. Examinations and research that the Finish power companies carry out to increase high burn up data base are listed.

1 Introduction

1.1 Nuclear Power in Finland

In 1990, about 35 % of the Finland's electricity production was covered by nuclear power. The generation is shared between two power companies Imatran Voima Oy (IVO), with its two 465 MWe modified VVER-440 PWR units (Loviisa 1, 1977; Loviisa 2, 1980), and Teollisuuden Voima Oy (TVO) operating two 735 MWe BWRs (TVO I, 1978; TVO II, 1980) built by ABB Atom AB. The accumulated experience currently amounts to some 50 reactor years in total. The operation histories of all these units have been excellent. The average load factors for the units have amounted to some 90 % for the recent years. The annual collective radiation doses on workers have been around 2 manSv for each of the two sites.

The Finnish reactors are run in base load mode. Large seasonal variations are typical of the capacity demand. Abundant hydro power and lowered demand cause spring and early summer to be naturally favourable times for refuelling.

A principle application to the Finnish Government for building a fifth nuclear unit of some 1000 MWe in capacity was placed by the power companies in May 1991. Offers for seven concepts from three suppliers are being currently assessed. Statement from the Government, followed by the decisive Parliamentary resolution may be expected in 1992.

2 The fuel cycles

2.1 TVO I and TVO II (twin ABB Atom BWRs)

2.1.1 Front end (TVO)

TVO procures the uranium and the various steps of the front end fuel services (uranium, conversion, enrichment, fabrication) separately with the fuel tailored to meet the needs. TVO's policy has been towards diversified supply, the fuel being now manufactured by ABB Atom AB in Sweden and by Siemens, Germany. The planning of fuel management is jointly carried out by the utility, the vendors and by the Technical Research Centre of Finland (VTT, independent national laboratory).

TVO I plant is now being run almost entirely with the Siemens 9x9-1 fuel, while in TVO II, the fuel type currently prevailing is ABB Atom's SVEA 64. In all types, axial natural uranium blankets are being utilized for improved axial power distribution. Burnable gadolinium absorbers are utilized in both units. Deviating from one-year refuelling interval is not considered appropriate in conditions that are expected to prevail in the near future. Low-leakage (in-out) loading strategies are followed. In reloading, about one third or one fourth of the core is replaced in Loviisa and TVO reactors, respectively.

2.1.2 Operation (TVO)

The average linear heat generation rates are 14.1 kW/m for TVO I (9x9 fuel) and 17.6 kW/m for TVO II; permitted local maximum is 41.5 kW/m. The currently utilized fuel rods are prepressurized. The cladding material is recrystallized Zircaloy-2.

Fuel performance has been good. Over the about 25 cumulated reactor years just a few failures have occured. None of the failures can be related to a burnup-dependent mechanism.

2.1.3 Back end (TVO)

After some cooling time the spent TVO fuel is stored in the interim storage at the plant site. The storage is planned to be able to accommodate the spent fuel from the whole life time of the two units. The facility is planned for storage time of 40 years, the design life being 60 years. For the final disposal, no decision has been made. The most probable solution is direct disposal in the Finnish granite bedrock. Several site candidates are being studied.

2.2.1 Front end (Loviisa)

The utility IVO procures the fuel for the two VVER-440 units solely from the original vendor, the latter being the only manufacturer of VVER fuel so far. The fuel and assembly characteristics are standard of the manufacturer; two or three fixed enrichments are offered. Partly based on the feedback from the Finnish experience and IVO's suggestions, however, the manufacturer has introduced several improvements to the fuel over longer term. The bulk of the fuel is with 3.6 % enrichment and used in three-year residence time mode. Some bundles, including the control rod followers, see only two years' life.

Detailed non-destructive and destructive examinations on fresh fuel from each delivered batch have been carried out in Finland by VTT by contract from IVO. This has been to substitute the limited possibilities to perform quality control audits at the fuel factory. There may not be need any more to routinely continue this practice.

Recently, there has been some interest by other companies in looking for options to start VVER fuel manufacturing by other companies. Siemens, for instance, has recently listed a VVER design fuel assembly with Zr-4 cladding.

2.2.2 Operation (Loviisa)

The average linear heat generation rate is 14.4 kW/m, permitted local maximum 32.5 kW/m. The fuel rods have been prepressurized to 6 bars helium. Coolant pressure is 123 bars. The cladding material is recrystallized Zr1%Nb. The pellets feature a central hole of nominally 1.4 mm in diameter. The dominating enrichment is 3.6 %. Fuel management planning is made by IVO.

No burnable absorbers are used in VVER's at the moment. The introduction of gadolinium for that purpose has been studied. It proves possible to be able to do this by the end of 1990's.

Fuel reliability has been good. The rod failure rate is estimated at $4 \cdot 10^{-5}$ over the accumulated 25 reactor years.

2.2.3 Back end (Loviisa)

According to the initial fuel delivery contract, all the spent fuel assemblies from the two Loviisa units have been and will be transported back to Soviet Union, the ownership and responsibility being rendered at the border. The transfer takes place after about five years' cooling in the spent fuel pool at the site. Typical lot then consists of the spent fuel corresponding one reaload from each of the units.

2.3 General economy

With great uncertainties in the future nuclear policy and reactor types and in future pricing of the more advanced fuel types, only comparative and highly speculative cost surveys are feasible. Only after several of the boundary conditions having become more fixed can any optimation of the future fuel cycles be accomplished on any realistic or absolute level. In general, the once-through fuel cycle is less sensitive to discharge burnup than that including high-cost reprocessing. Nevertheless, going higher in burnups is judged slightly but clearly favourable in Finland under conditions that one can see now.

3 Licensing issues

3.1 Current burnup limits

The national licensing authority, The Finnish Centre for Radiation and Nuclear Safety (STUK) places upper limits for the burnup individually on each fuel type utilized.

For the VVER fuel of the Loviisa plant the maximum rod average burnup is set at 48 MWd/kgU, with an additional requirement that no more than 12 assemblies may contain a rod averaging over 44 MWd/kgU.

For the TVO fuels, there applies an assembly average burnup limit of 36 MWd/kgU, additional restriction being that no more than 10 assemblies may exceed average burnup of 34 MWd/kgU.

During the 1980's, the allowed discharge burnups increased by 0.5 to 1.0 MWd/kgU/year. The fuel performance was quite good and no burnup related failures were found. STUK did, however, froze the burnup limits at the level reached in 1989, maintaining that no comprehensive assessment of the safety significance of increased burnup, particularly in relation to transient and accident conditions, exists.

In the near future, the Finnish power companies will evidently continue pursuing slightly higher burnups that are readily achieved technically with the current reactors and fuel design and management concepts. This might mean little change in the current rod maximum of 48 MWd/kgU for the Loviisa VVER fuel, provided the three-cycle mode will be followed, and maximum assembly burnups of some 40 MWd/kgU for the current TVO fuels.

Somewhat later on, there may be incentives to change over to four-cycle mode in VVER reactors. This would entail a profound re-evaluation of the fuel cycle economics, which has not been performed by IVO yet. TVO will consider more advanced fuel types. Introduction of those will bring the desirable burnup mark well over 40 MWd/kgU.

Possible introduction of a completely new reactor type in Finland would start another deep-going process of optimization and licensing. Issues of licensing play naturally an important role already in advance when the offered candidates are compared.

3.2 Technical fuel factors during operation

3.2.1 TVO fuels

One of the factors that contributed to the current burnup limits was the high fission gas release fractions found in some of the ABB Atom fuel rods. There were also indications that uncertainties in rod local power estimates were larger than anticipated. BWR fuels in general seem to be prone to fairly large scatter in fission gas release fractions. This may result from local power variations due to control blade movements, for instance, which are difficult to handle in calculations, and possibly from thermal feedback phenomena inside the rod due to relatively low pre-pressure levels typical of BWR fuels [1].

In addition, the effect of channel bowing on local power (via causing deviations in the water gap between the fuel channels and the resulting combined effects of altered thermal and fast neutron fluxes) have received attention as a possible high burnup factor. TVO has given up the earlier practice of reusing the channels.

TVO expects that the accumulating experience will show the scatter in release rates to be suppressed due to the adopted in-out loading strategies that curb large power variations later in life, and due to moving over to fuel designs with lower rod linear powers (from 8x8 to 9x9 and possibly to 10x10).Normal operation experience and extended irradiation programs suggest that cladding outside oxidation is probably not a major concern for the burnups foreseen.

Going to high burnups PCI needs to be looked at with any fuel type. Still, for example, it has been seen that essentially no ridging has formed on the claddings of ABB 8x8 rods with burnups up to 48 MWd/kgU. In destructive examinations, only slight bonding between the pellet and cladding was occasionally found. Fuel manufacturers claim improved PCI resistance in the future fuel types considered by TVO.

3.2.2 Loviisa VVER fuel

Over the life time of the Loviisa plants, the VVER fuel has gone through significant development. One of these has been the pellet manufacturing that, instead of the former extrusion method, is now accomplished by more standard pellet compressing technique allowing better command over the desired properties. Improvements that potentially have a direct influence on the high burnup performance include allowing more space in the assembly for rod axial growth, and going from 1 bar helium filling to 6 bars.

Also for VVER fuel the licensing has been cautious. One of the worries has been the fission gas release. There are some early results from irradiations in Soviet Union that show unacceptably high release fractions. The experience from Loviisa plant suggests gas release fractions of less than 1 %, as based on destructive PIE, or less than 2 % from poolside nondestructive determination (γ-scan, Kr85). Linear powers in Loviisa VVER-440 are comparatively low. Pellet design with a central hole further contributes to low fuel maximum temperatures.

Due to the high corrosion resistance of the cladding material (Zr1%Nb), comparatively low cladding temperatures and favourable water chemistry, the oxidation on the used Loviisa claddings is almost negligible. Oxidation is expected to be of no concern even at considerably higher burnups.

Destructive PIE has shown closed fuel-to-clad gap with occasional strong bonding. It is felt that even fairly deeply depleted rods may not be safe from PCI-inducing ramps especially in the neighbourhood of fresher assemblies after refuelling.

The fuel failures have been given great attention by IVO [2]. Leaking assemblies have normally been examined in the poolside stand. In a few of the cases, PCI failure is suggested or cannot at least be overruled. In some other cases, there is strong evidence that the faults seen are secondary failures. Recently, distorsions of the upper tie plate have been detected in several bundles in visual examinations. There may or may not be a correlation with fuel failures. These deformations seem to result from jamming of the upper end of the axially growing rods to the upper tie plate. The design needs corrective modification.

3.3 Licensing issues for front and back ends (IVO, TVO)

Within the non-revolutionary range of the foreseen burnup increases, the Finnish utilities see little technical problems regarding front and back ends of the fuel cycle. There are several safety factors which have to be taken care of but which have straightforward tehnical solutions, the issues being rather of economical character. An example of such is the enrichment limit for the fuel pools. Going somewhat higher of the current initial enrichments will need modifications in the storage racks and relicensing. The utilities also acknowledge that higher discharge burnups quite rapidly increase the neutron radiation in spent fuel, which needs to be taken into account in transport and storage. The interim storages are licensed for burnups high enough to accommodate the foreseen spent fuel batches.

4 Actions taken to qualify higher fuel burnups

Both of the Finnish power companies carry out extensive programmes to collect data on the performance of their fuel types. The Finnish licensing practice does not favour large-scale prototypic irradiations in Finnish power reactors. To substitute this, the utilities go in for a number of actions some of which are listed in the following.

4.1 Poolside examinations

Both of the Finnish utilities have a versatile poolside fuel inspection facility which have been used very actively for extensive routine examination on all of the fuel types, as well as in more detailed surveys like those mentioned in the sections below [3]. The facilities feature a number of standard equipments. In addition, more sophisticated examinations have been occasionally ordered from outside contractors. Examinations performed poolside in Finland include:

- visual inspection of the assemblies and rod
- assembly distorsions
- rod length measurements
- clad profilometry
- oxide thickness measurement, fuel-to-clad gap measurement with "squeezing" method
- γ-scanning including fission gas release determination by the Kr85 activity method.

4.2 Extended irradiations of single assemblies and detailed PIE examinations of selected rods in hot cells

TVO has carried out an exteded irradiation of several ABB Atom 8x8 fuel assemblies. The irradiation lasted five and six cycles and the rod maximum burnup of 48.5 MWd/kgU (max. pellet 55.5 MWd/kgU) was reached. Nine rods from two of these bundles were included, after poolside examina-

tion, in the international Battelle High Burnup Effects Program for detailed PIE examinations at the Windscale Laboratories, England [4]. Valuable reference was also gained to calibrate the coming poolside examinations. For example, an excellent match was arrived at between the fission gas release fractions measured by Kr85 method and those from puncturing. Some of the conclusions are referred to in section 3.2.1 above. No clearly restricting high burnup phenomena were found. More PIE examinations (EPMA, burnup determinations) are being carried out at Windscale under commission from TVO.

4.3 Poolside and hotcell examinations of Loviisa VVER fuel

In 1985-91 16 assemblies and five absorber assemblies have been examined at the poolside inspection stand. During annual refuellings, the lengths of 47 irradiated assemblies and their rods (up to burnups of 49 MWd/kgU) have been measured [3]. Four rods from three special high burnup assemblies were examined destructively in Studsvik laboratories. More recently two special three- and four-cycle assemblies have been unloaded. These assemblies and some of their rods have been precharacterized in detail. The poolside examinations of these assemblies and rods will be performed in 1992. Rod burnups range up to 48 MWd/kgU.

4.4 Test reactor irradiations of VVER fuel

Under co-operation between the Kurtchatov Institute of Moscow and IVO, an experimental program is carried out that produces highly qualified behaviour data from closely simulated VVER-440 and VVER-1000 conditions [5]. Up to now three 18 rod bundles (of 1m active length) instrumented with fuel thermocouples and elongation sensors have been irradiated in the MR test reactor in Moscow. The fourth one, featuring also rod internal pressure sensors is going into the reactor; fifth one is under construction. Among the rods in the bundles, design parameters like fuel density, fuel-to-clad gap and fill gas are varied. The irradiations are followed by hot cell examinations.

In the second phase of the program, a facility will be installed and applied that allow reinstrumentation of high burnup rods with fresh thermocouples. This phase is starting in 1992.

Plans also exist of a third phase where ramp and cycling behaviour of VVER fuel would be studied.

4.5 International research programmes

The Finnish organizations have participated in most of the internationally sponsored fuel research programmes. Usually the participation has been actualized under co-ordination of the national laboratory VTT. Examples of on-going or recently completed efforts are the OECD Halden Reactor Project, the many Studksvik fuel projects, the Battelle High Burnup Effects Program and the Third Risø Fission Gas Project. For past few years, it has obviously become more difficult to launch such programmes. This reflects the groving demand of more focused fuel-type specific experiments.

5 The role of fuel performance codes

Computerized fuel behaviour models are essential in understanding the many phenomena and interactions that govern the nuclear fuel behaviour. Actually, the most efficient use of the experimental results is to condense them to the form of a model parameter or a submodel in a code. Successful modelling is a sign of advanced understanding.

The high burnups of the nuclear fuel place great challenges to fuel modellers. Even currently, one can spot several weak points in our descriptions of the integral behaviour.

Fission gas release is one of the phenomena that remains difficult to predict. Occasionally, even the fuel vendors have had large scatter in simulating the behaviour of their own fuels. The problem lies not only in gas release models as such. The actual temperatures and temperature distributions of high burnup fuel may not be known accurately enough. Further, it is very important to use methods in power calculations that are sufficiently sophisticated.

More mechanical data should be acquired before detailed modelling of PCI or clad creep-out (with the possible lift-off effect) is actualized.

Still, there is strong need of physically based fuel performance codes, though it looks like that certain calibration against each fuel type in question and detailed fuel-type specific materials data will be needed.

In Finland, the larger fuel performance codes are run by the national laboratory VTT. In the framework of a national research programme, new codes are being acquired and developed to extend the analytical capabilities into appearing new ranges of fuel design and operating data.

6 Conclusions

In Finland the performance of the four reactor units and their fuels has been very good. However, extremes as

regards reactor power densities and linear powers or discharge burnups have been avoided. The licensing policy adopted by the Finnish authority STUK has been cautious.

Extending the discharge burnups is judged clearly, yet maybe slightly, favourable in economical and environmental terms under the advocated once-through fuel cycle concept.

The Finnish utilities pursue slightly increased discharge burnups for the nearest future for their fuel types now in use.

The utilities and carry out remarkable fuel examination campaigns of their own. Introduction of new fuel or reactor types will bring incentives to further extend the burnup limits.

International co-operation is important for Finland in experimental fuel research.

Computer modelling of the high burnup phenomena should keep up with the recent experimental findings. The effect of burnup on transient and accident behaviour and related modelling should not be forgotten either.

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COMPATIBILITY OF EXTENDED BURNUP WITH FRENCH FUEL CYCLE INSTALLATIONS

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Abstract

Based on the fact that about 75 % of national electricity is supplied by nuclear power plants, economy of nuclear energy is a very important concern in France. At this level of penetration and taking into account the experience gained through the 15 past years during which about 50 plants have been constructed and operated, the only effective way for reducing the cost of nuclear energy appears to be burnup extension. This program is currently in progress and involves all the actors of the fuel cycle, from enrichment phase to reprocessing activities. The burnup extension program carried out in France is reviewed.

1 - PREAMBLE

In all countries concerned by nuclear energy, burnup extension is one of the main concerns of the plant owner. In fact, such extension has a significant impact on the cost of electricity produced.

For reference, figure 1 attached to the present paper shows the variation of fuel cycle cost in relation with burnup value. This figure was established by EDF about 3 years ago on the basis of the economic conditions prevailing at that time, and may suffer some minor modifications linked with variations affecting the constituting elementary costs.

However, the general shape of this figure is still basically correct and shows that, for a given refueling policy, fuel cycle cost reaches a minimum value for burnup ranging from 50 to 60 GWD/T. This can be achieved provided that fuel enrichment is raised to at least 4.5 %.

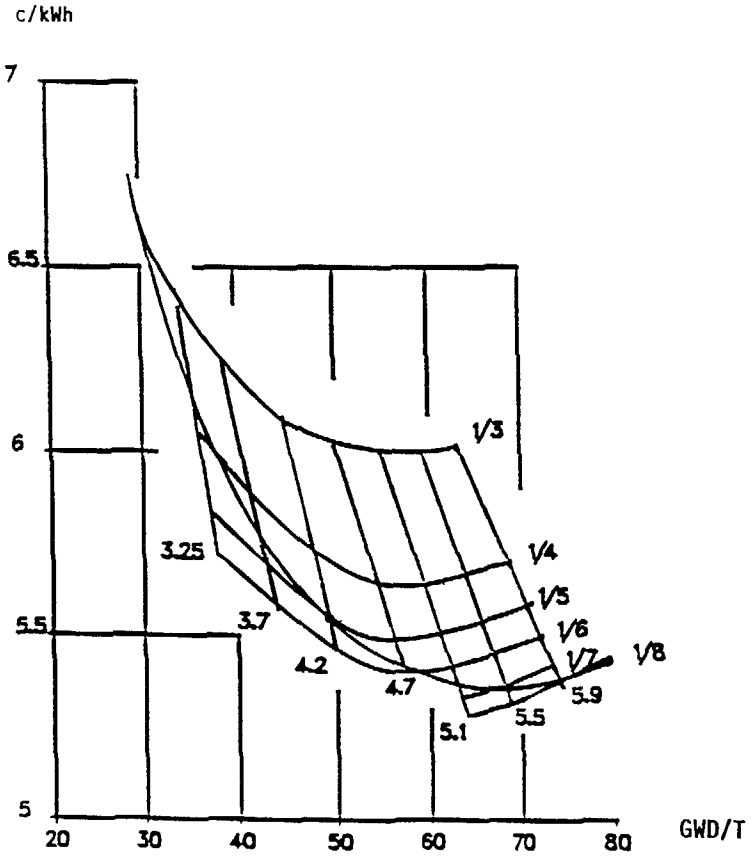

FIG. 1. Fuel cycle cost versus burnup.

In this respect, EDF have started since some years to progressively increase the fuel enrichment of their domestic plants ; this policy is illustrated by the following milestones :

Step	Period	Enrichment %	B.U * GWD/T
1	1977	3.25 (900 MWe)	33
	1984	3.10 (1300 MWe)	33
2	1987	3.70 (900 MWe)	42
	1994	3.60 (1300 MWe)	42
3	> 2000	5.00	60

(* Typical average unloading B.U. value)

It is worth noting the followings :

- At the first step, EDF policy consisted in loading fuel elements enriched up to 3.25 %. This enabled to operate on a 1/3 core yearly refueling basis.

- More recently, enrichment has been raised to 3,70 % on 900 MWe plants, enabling the burnup to be increased up to 42 GWD/T and to operate on a 1/4 core yearly refueling basis.

- EDF target is to raise fuel enrichment up to 5 % so as to reach a 60 GWD/T burnup. This challenge needs a lot of problems to be overcome.

Experience gained during the 2nd step will be helpfull for that task ; it is the experience gained by EDF as operator and COGEMA as reprocessing firm in the framework of this first enrichment extension we are going to discuss here after.

It covers the following technical aspects :

- enrichment and fuel manufacturing
- transportation of fresh fuel elements
- storage of fresh fuel elements
- core loading map and reactivity control
- storage of spent fuel elements
- evacuation and transportation of spent fuel elements
- reprocessing

fully covered by the French industry.

2 - ENRICHMENT AND FUEL MANUFACTURING

Up to now, uranium enrichment and fuel elements manufacturing for national needs have been mostly carried out by French firms. Yearly national needs, including first core loading and refueling, are summarized in the following table (year 1991) :

- Natural uranium needs : 8 000 T/year
- Enrichment capacity : 6 M SWU/year
- Manufacturing : 2560 fuel assemblies/year

With respect to enrichment, which is carried out by EURODIF in the Georges Besse Plant at Pierrelatte, enrichment rate up to 5 % is allowed by French Authorities regulations for public needs. This limit is consistent with EDF forecast needs up to the step 3 (5 %) of the program.

It is worth noting that small quantities of uranium enriched at 4.5 % have already been produced by this diffusion plant for EDF needs.

As regards manufacturing of fuel elements, this activity is carried out in France in accordance with regulations issued by the French Government. These regulations specify the maximum enrichment allowed for the uranium to be used by the manufacturer.

This enrichment is fixed at 5 % for the ROMANS factory (decree dated 2 nd march 1978) and the PIERRELATTE factory (decree dated 7 th sept. 1982).

This limit match with EDF goal for step 3 as defined here above.

3 - TRANSPORTATION OF FRESH FUEL ELEMENTS

Fresh fuel elements transportation is regulated by the French Authorities in conformance with the domestic and international regulations applicable.

Fresh fuel containers shall be approved by the Ministry responsible on a case by case basis.

One of the main problems raised by enrichment extension in that field is that of criticality ; for that purpose, fresh fuel containers shall be equipped with absorbing materials (copper, borated steel).

At the time being, 3 types of authorized containers are used for EDF needs as specified in the following table :

Brand	Maximum enrichment %	with Absorbing material	Fuel type
FRAMATOME RCC3	3.9	Cu	900
FBFC RCC	3.8	Cu	900
FBFC RCC4	4.5	B	1300

The capacity of these containers is 2 elements.

These containers fulfill EDF needs for several years ; further developments will have to be carried out for facing the problems linked with the higher enrichment forecast.

4 - STORAGE OF FRESH FUEL ELEMENTS

As regards the storage on site of fresh fuel elements, the only problem which was identified was that of criticality.

Radiation shielding did not raise any problem.

First of all, it must be kept in mind that EDF policy is to store fresh fuel elements in the plant fuel building pool, so that the problem is to determine if there is any risk of criticality (keff < 0.95) for underwater storage. In this respect, it has been determined that this condition is fulfilled by fresh fuel elements enriched up to 4.5 % when stored in stainless steel racks in non borated water if the cell pitch exceeds 380 mm.
For 900 MWe plants, and for the first 1300 MWe series (P4), this pitch is 410 mm, so that no problem will occur in this field in the near future.

For the next series (P'4 and N4), the pitch is reduced to 280 mm (compacted spent fuel storage). These compacted racks meet the criticality requirement for up to 4.5 % enriched fuel by incorporating neutron absorbant ; this was taken into account at the design stage of the series concerned so that extending fuel enrichment will not require any re-racking operation until the step 3 of the program.

5 - CORE LOADING MAP AND REACTIVITY CONTROL

900 MWe plants were initialy loaded with 3.25 % enriched fuel and operated on a 1/3 core yearly refueling basis. Core loading map was according to figure 2 ("Out-In-In scheme").

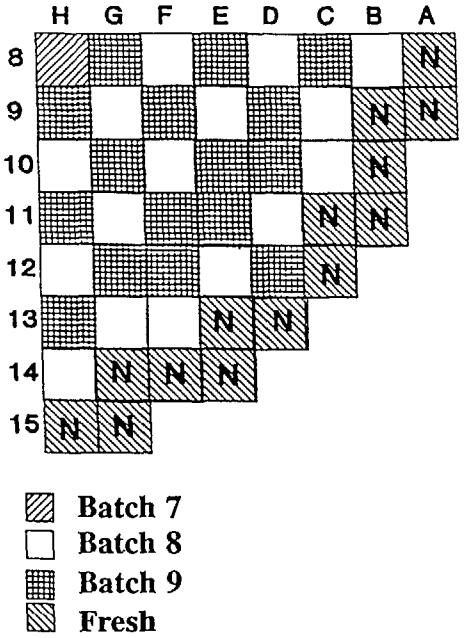

FIG. 2. 900 MWe core loading map: 1/3 core 3.25% U5.

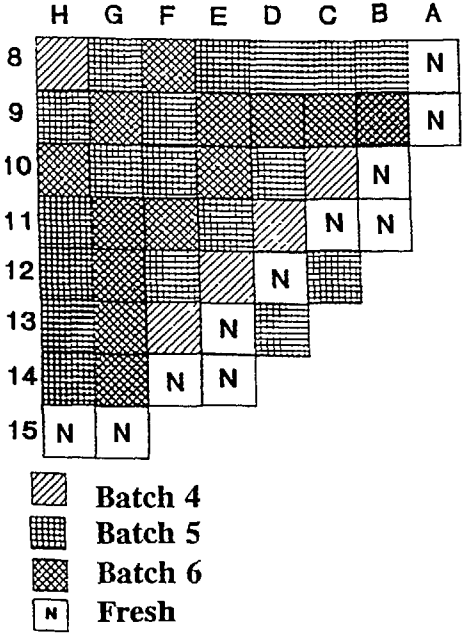

FIG. 3. 900 MWe hybrid core loading map: 1/4 core 3.7% U5.

When enrichment was raised to 3.7 % in the year 1987 on three 900 MWe plants (Blayais 1, Gravelines 1 and Chinon B1), the same fuel management led on one hand to change the core load map to a "hybrid" one in order to reduce the core hot point factor in some situations and to limit the neutron fluence on the reactor vessel, and on the other hand, to use burnable poison (gadolinium) in order to control the reactivity.

Actually, EDF policy is to operate all the reactors loaded with 3.7 % enriched fuel on a 1/4 core basis ; under this condition, burnable poison is no longer mandatory. In fact, it has been determined that reactivity control only requires in these circumstances absorbing Pyrex rods at first core loading. As regards the core loading map, an "hybrid" scheme has been set-up (see figure 3).

Investigations carried out in the framework of the 2nd step project have demonstrated that fuel enrichment extension up to 3.7 % does not induce significant changes in the core management scheme. Higher enrichment and fuel management like 1/3 core would require burnable poison to control the reactivity ; experience gained in this field demonstrates that gadolinium burnable poison is a reliable technology for future enrichment extension.

Otherwise, initial boron concentration and control rod worth shall be checked every time enrichment is modified or fuel management is changed. As regards boron concentration, it is mainly connected with cycle duration ; in this respect 1/3 Core - 3.25 % and 1/4 Core - 3.7 % schemes lead approximately to the same boron concentration requirement. On the other hand, it has been determined that higher enrichment would possibly require enriched boron control rods for reactivity control at the end of the fuel cycle.

6 - STORAGE OF SPENT FUEL ELEMENTS

Spent fuel storage on site raises 4 kinds of questions which must be answered when extending the burnup value :

- radiation dose in the plant fuel building
- criticality
- residual heat removal
- operational lag storage.

6.1 - Radiation dose on the service floor of the fuel building is to be kept below the "green area" upper limit value (25 μ Sv/h) under any operating condition.

In fact, it was computed that the water depth within all the pools concerned by spent fuel handling and/or storage was large enough for accomodating high activity fuel elements. For reference, the dose criteria quoted above requires a minimum water thickness of 255 mm shall be maintained between the water surface and the head of a fuel element enriched at 4.5 % and irradiated at 45 GWD/T. Actually, the water thickness in EDF plants is in no case smaller than 3 m.
However, special measures shall be taken as regards spent fuel elements stored in the viscinity of the watertight doors of the spent fuel pool : operators have been instructed not to handle and not to store spent fuel within a short distance from the doors. Forbidden storage cells concerned may be loaded with fresh fuel assemblies.

6.2 - Taking into account the measures taken against the risk of fresh fuel criticality (see section 4), spent fuel storage does not raise any problem in this field.

6.3 - Figure 4 appended provides a comparison as regards residual heat between two significant cases (3.25 % - 33 GWD/T and 4.5 % - 45 GWD/T). The power developped within 4 days following reactor shutdown is only given for information and is not to be considered when dimensioning the residual heat removal system (PTR) of the plant.

In the worst situation (i.e. 4 days after reactor unloading), there is only a small difference - 4.7 % - between the residual powers relieved by the both types of assemblies considered. This variation can be accomodated by the PTR system without any difficulty.

6.4 - Due to the requirements presented by the high activity and the heat load of spent fuel discharged from the reactor (see section 7 hereafter), an operational lag storage is required.

On the basis of a 2 years decay period, as discussed here after, and taking into account a fuel management operating on a 1/3 or 1/4 core yearly refueling basis, the operational lag storage capacity should be about one core for each plant.

This condition is fulfilled by all EDF plants : for reference, CPO desactivation pool capacity enables to store 2 cores ; this capacity has been extended to 7/3 core for CP1 and P4 series and to 3 1/4 core for P'4 and N4 series.

7 - EVACUATION AND TRANSPORTATION OF SPENT FUEL ELEMENTS

Evacuation of spent fuel assemblies from the plant desactivation pool is submitted to conditions linked to the activity of the assemblies, their residual heat and their reactivity.

Burnup extension, and Correlative high enrichement have obvious effects on these parameters.

As regards the residual heat, though the difference between a 3.25 % enriched fuel assembly irradiated at 33 GWD/T and a 4.5 % enriched fuel assembly irradiated at 45 GWD/T is quite small during the first days following reactor shutdown, it becomes significant after 6 months (see figure 4 appended).

COOLING TIME	RESIDUAL HEAT (W/T)		Δ P (%)
	3,25 % 33 000 MWD/T	4,5 % 45 000 MWD/T	
1 h	5,68 E5	5,82 E5	2,4
1 d	2,41 E5	2,50 E5	4,16
4 d	1,49 E5	1,56 E5	4,7
15 d	8,10 E4	8,79 E4	8,5
1 m	5,94 E4	6,55 E4	10,2
6 m	2,11 E4	2,51 E4	25,8
1 y	1,20 E4	1,51 E4	25,8
1,5 y	8,63 E3	1,07 E4	28
2 y	6,23 E3	8,30 E3	33,2
3 y	3,87 E3	5,38 E3	39
5 y	2,11 E3	3,14 E3	48,8

FIG. 4. Spent fuel residual heat.

On the other hand, the initial enrichment is to be taken into account as regards the risk of criticality during transportation and at the reprocessing plant (reprocessing a damaged assembly, weak irradiation of the upper part of the assemblies). Similarly, a minimum burnup value is required for the assemblies to be evacuated so as to reduce their reactivity.

Consequently, spent fuel evacuation is subjected in France to two kinds of requirements :

A - Spent fuel cask requirements

- Radiation dose at the surface of the spent fuel cask shall be less than 2 m Sv/h, and 100 μ Sv/h at a distance of 1 m.
- Thermal power of assemblies enclosed shall be lower than a specified value.
- Initial enrichment shall be lower than a specified value.

Spent fuel casks used in France for EDF needs are listed hereafter, together with their specific characteristics.

NPP Type	Brand	Max. Power (kw)	Max. BU (GWD/T)	Max. Enrich. (%)
900	TN 12/1	120	42	3.75
900	TN 12/2	93	42 45*	3.75
900	LK 100 B	85	42	3.75
1300	TN 13/2	109	42 45*	3.75

* Over-shielded type

This table shows that no spent fuel cask is at the time being qualified for transportation of over 3.75 % enriched fuel.

B - Reprocessing plant requirements

The following data are those applicable to La Hague reprocessing plant.

- Installations were designed for a full cask thermal power of 85 Kw.
- As well, the initial design was made on the base of an initial enrichment of 3.5 % and a burnup value of 33000 MWD/T.

At the time being, higher initial enrichment together with higher burnup are accepted, provided that :

- it is checked that the spent fuel assemblies have actually been irradiated during one cycle, at least,
- the spent fuel assemblies have stayed in the desactivation pool for a minimum period of time, as set forth here after :

BU GWD/T	35	40	43	45
900 (days)	270	310	420	550
1300 (days)	320	370	470	580

These requirements have two major consequences :

- As regards the decay time required, a limitation of burnup extension will rapidly be created by the plant desactivation pool capacity. External lag storage facilities may be necessary in that case.
- With respect to the burnup checking requirement, French firms have already developed a system enabling to monitor spent fuel assemblies in reactor pool or in reprocessing plants with N.D.A methods. The prototype device called PYTHON enables by passive and active methods (neutron and gamma counting) to measure the 3 main physical parameters of a fuel assembly : burnup, cooling time and effective multiplying factor. A first industrial system is today in operation in France.

8 - REPROCESSING

This activity is carried out by COGEMA in their factory at La Hague.

For the purpose of this report, high burn-up fuel is defined by comparison to the characteristics of the nominal UP3 fuel (i.e.

TABLE 1

HBU FUELS - COMPARED CHARACTERISTICS
ESTIMATED VALUES

Data refers to 1 metric ton of initial uranium (MTU)

DATA	NOMINAL UP3	HBU	XCOEF.	OBSERVATIONS
Burn-up Cooling time U235 initial	33,000 MWd/MTU 3 years 3.5 %	47,000 MWd/MTU 5 years 3.7 to 4.5 %	1.4 1.66	
Fission products : Including solubles insolubles gazeous FP activity	35 kg 26 3 6 7.4 x 10^5 Ci	49.5 kg 6.5 x 10^5 Ci	1.4	
Activation products . hull . end fittings	12 to 14,000 Ci	+ 20 % + 20 %	1.2	
Actinides : . Uranium 232 . Uranium 235 . Uranium 236 Total Plutonium including : . Pu 238 . Np . Am . Cm Total activity	 0.090 ppm/U235 0.9 % residual U 0.4 à 0.5% res.U 8.9 kg 2 % total Pu 483 g 257 g 29 g 6,736 Ci	 0.150 to 0.220 ppm U235 0.8 % residual U 0.6 % residual U 11 kg Lim. 3.5 % total Pu 700 g 600 g 80 g 13,000 Ci	 1.25 1.75 1.50 2.75 2.50 2.00	 depends on initial enrichment idem - penalty for U estimates
Zr fines	1kg	1 kg	1.00	

33,000 MWd/MTHM, 3.5 % U235 initial enrichment, 3 years cooling before reprocessing) as follows :

- a typical value of 47,000 MWd/MTHM,
- an U235 initial enrichment between 3.7 % and 4.5 %,
- 5 years of cooling time before reprocessing.

The main characteristics of theses two reference fuels are compared and estimated in Table 1 while Table 2 compares the values of the nominal UP3 fuel with those of a high burn-up fuel after only 4 years of cooling time. Major differences can be observed as there are 40 % more fission products, 25 % more plutonium, an increase of 100 % in the alpha content as well as an increase by a factor of 3.5 of the neutron emission (due to Cm 244). A summary listing of the impact of these new fuel characteristics on the reprocessing plant is as follows :

- increase of the U235 initial enrichment : criticality impact from the storage pool NPH to the R1 buffer tank unit,

TABLE 2

COMPARED CHARACTERISTICS OF NOMINAL UP3 FUEL AND HBU FUEL

DATA	NOMINAL UP3 FUEL	HIGH BURN-UP FUEL
Burn-up U235 Cooling Time	33,000 MWd/MTU 3.5 % 3 years	45,000 MWd/MTU 3.7 % 4 years
ACTINIDES		
. mass U	955.4 kg	940.8 (0.985)
. mass Pu	9.74 kg	11.37 (1.17)
. U + Pu	963.14 kg	952.17 (0.987)
. beta, gamma	1.22 10^5 Ci	1.5 10^5 (1.23)
. alpha	7.057 10^3 Ci	1.47 10^4 (2.1)
. heat release	237.3 W	500.1 (2.1)
. neutron emission Pu	3.058 10^8 n/s	1.068 10^9 (3.5)
. alpha activity	3868 Ci	6608 (1.7)
. heat release Pu	126.3 W	216.7 (1.7)
. neutron emission Pu	7.194 10^6 n/s	1.162 10^7 (1.6)
. % Pu 238	1.8 %	2.88 % (1.6)
. % Pu 240	22.7 %	24.6 % (1.08)
. % Pu 241	12.2 %	12.8 % (1.05)
. alpha activity/kg Pu	397.2 Ci/kg Pu	581.1 (1.46)
. thermal power/kg Pu	12.97 W/kg Pu	19.06 (1.47)
. Cm 244	24.03 g	82.46 (3.43)
ACTIVATION PRODUCTS		
. total mass	316 kg	316 (1)
. activity	6193 Ci	5735 (0.93)
. Heat release	37.22 W	38.23 (1.03)
. Co 60	2261 Ci	2366 (1.05)
FISSION PRODUCTS		
. mass FP	33.82 kg	46.07 (1.36)
. activity	761500 Ci	747400 (0.98)
. heat release	2792 W	2692 (0.96)
. Ru/Rh 106	6.4 10^4 Ci	4.614 10^4 (0.72)
. Kr 85	7.9 10^3 Ci	9.37 103 (1.19)

- increase of the total mass of the fission products : impact on the FP concentration and vitrification units,

- increase of the total mass of plutonium : impact on the throughput capacity of R4 plutonium unit and BST1, criticality impact due to the increase in Pu concentrations,

- increase in the Pu alpha activity : compared to the fission products activity, this leads to a decrease by a factor of 1.7 of the ratio beta activity / alpha activity with a possible impact on the definition of the monitoring equipment,

- increase in Curium 244 and neutron emission : impact on the threshold values for the atmospheric release estimates due to the importance of Curium in their determination. Impact on the neutron emission of the fission product solutions, due to Cm 244, thus on the biological shieldings of R1, R2, R7, SPF.,

- increase of the Pu specific thermal power : impact on the temperature of Pu solutions in R4 and of PuO_2 in BST1 and on the equilibrium temperature of the FS47 transport containers,

- increase of the Pu neutron emission : impact on the R4, BST1 biological shielding.

Basically, two ways are possible to reprocess high burn-up fuels through the reprocessing plant : 1) Through the on line reprocessing of batches of segregated high burn-up fuel which implies some new licensing issues, 2) Through dilution of the high burn-up fuel at the earliest possible stages of the process.

Dilution of a high burn-up fuel in the nominal fuel must be done early in the process i.e. just after the dissolution step : the two dissolution lines of T1 could work simultaneously (one line for the nominal fuel, one line for the high burn-up fuel). Furthermore, a buffer tank could be installed between T1 an T2 to receive and account the high burn-up fuel and to keep the high burn-up solutions in stand-by until they can be fed in the main flow without throughput capacity reduction of the plant.

The successive UP3A facilities are listed below along with the main consequences from the on line reprocessing of batches of high burn-up fuel :

NPH-TO Pools

- The criticality safety aspect of spent fuel pool storage with the storage baskets requires to verify the burn-up as soon as the cask is unloaded and before the fuel is placed in the storage basket.

T1 - SHEARING : Dissolution-clarification

- The main problem is located at the clarification unit which has to process 30 % more fines, thus needing more frequent cycles. This limitation should constitute one of the bottlenecks all along the plant process.

T2 - High activity solvent extraction and waste concentration

- Depending on the limiting value which will be determined for the Pu/U ratio, HBU fuel can or cannot be processed on line in this facility. This is a limiting point from a licensing standpoint.

- Apart from this, one other important incidence can be evaluated concerning the U IV quantities needed for Pu extraction, which could be also a limitation for the overall capacity.

- FP concentration will have to process 40 % more fission products (by increasing concentrations and/or volumes of solutions to handle).

T3 - U and Pu purification

- Although not directly a process limitation, the evolution of the isotopic composition of U has to be considered :

  . increase of irradiation due to U232 increase (need of complementary shielding),

  . the penalty due to U236 (neutron absorber) significantly increased.

- Within the Pu purification line, a significant increase of the irradiation level is caused by the increase of Pu238, which requires more shielding. This irradiation is a cause of faster damaging of the solvent, reducing its efficiency.

T4 - PuO_2 conversion and packing

- 25 % more Pu to be reprocessed means 25 % more final product to pack, handle and store.

T7 - Vitrification

- Products to be vitrified include FP (+ 40 %), clarification fines (+ 30 %), alkaline solutions (rinsing of FP evaporators, alkaline wastes, basic effluents) with an increased volume by some 25 to 50 %. These quantities to be processed require that three T7 lines work simultaneously.

Total Reprocessing waste volume

For the same mass of heavy metal reprocessed, a high burn-up fuel would lead to 2 % more waste than for the reference fuel irradiated to 33 Gwd / MTHM. However, the comparison of the ratio of waste volumes by the energy produced by the fuel in a typical 900 MWe reactor producing 233 Gwh / MTU, shows that for the LWR reference fuel, there are 30.5 l of waste packaged per Gwh while for a high burn-up fuel there are 22.9 l of waste package per Gwh. Therefore for an equal quantity of energy produced, the high burn-up fuel leads to 23 % less waste volume.

Beta and gammas activities for hulls and ends activities are increasing by 20 % and 100 % but still remain below the guaranteed parameters. Supplementary shieldings would be necessary for the package handling.

Technological wastes : non alpha as well as alpha wastes still remain below the guaranteed parameters for each category.

EXTENSION OF BURNUP IN INDIAN PHWRs

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Abstract

The Indian nuclear power programme is based upon PHWRs in the first phase in which plutonium will be produced. In the second phase, this plutonium would be used in the fast breeders to enhance fissile materials i.e. plutonium and also generate U^{233} by irradiating thorium. In the third stage, Th-U^{233}reactors will be used for producing power.

Presently many PHWRs of medium size are either operating or are under construction. An effort is being made to extend the burn up of these PHWRs in case there is a delay in commercial operation of the fast breeder reactors. Because of the fast reactor programme, a fuel reprocessing facility is available. This assures availability of plutonium. Hence, recycling plutonium becomes a natural choice for extending burn up of PHWRs.

It was decided that the burn up extension programme should be implemented in the present reactors without any change in the basic hardware of the reactor. A number of lattices with MOX enrichment in natural uranium or depleted uranium were analysed and two of these were chosen for detailed studies. It has been established that a lattice with 0'4 wt per cent MOX in natural uranium in the central seven rods of the nineteen rod cluster and natural uranium in the outer twelve rods would increase the average exit burn up of the reactor from the present value of 7 000 MWd/te to 10 300 MWD/te.

The transition from the present natural uranium reactor to MOX reactor would be feasible without any power derating at any stage. The control and safety characteristics would almost remain unchanged with the MOX fuel.

1. INTRODUCION

Indian nuclear power programme is based upon Pressurised Heavy Water Reactors (PHWRs) using natural uranium fuel in the first phase. The plutonium produced in these reactors would be used to fuel the fast breeder reactors in the second phase. These fast breeder reactors would also be used to irradiate thorium and to enhance the fissile material in the form of

either plutonium or U^{233}. In the third stage, Th-U^{233} reactors would be the source of nuclear power production. The basis for the three stage programme is the moderate uranium resources and abundant thorium resources in the country.

A design has been standardised for a small size PHWR in the 200-230 MWe range and five reactors of this type are operating, one has been commissioned recently and one more is poised for commisioning soon. Five reactors of this catagory are in various stages of construction. The design of a 500 MWe PHWR is also finalised and the work on a number of such reactors would start soon.

India has two (Boiling Water Reactors(BWRs) of 200 MWe each of vintage type but does not have a plan to extend the present 20,000 MWD/Te burn up of these reactors. In 1982, a feasibility study of using plutonium enriched fuel in these BWRs was carried out. In future, if the need arises, these efforts could be revived. An effort to extend the burn up in PHWRs presently is being made and has to be viewed in the context of possible delay in establishing commercial fast breeder reactors of the second phase.

As a sequel to the decision to have a fast reactor programme, the reprocessing facility for the fuel discharged from PHWRs is available. This ensures availability of plutonium for recycling. Hence recycling of plutonium seems to be the natural choice for extending the burn up in PHWRs. In the context of the present three stage programme, recycling of plutonium in PHWRs would be only a short term measure mainly to improve resource utilisation and an important consideration in this respect is that plutonium should be available for the fast reactor programme when required and the quality of the second stage plutonium should not be seriously impaired.

The problems arising while extending burn up of LWRs and PHWRs are similar as far as management of power distribution is concerned.

The paper gives a brief description of the PHWR, discusses the problems associated with extension of burn up and gives results of some studies made for medium sized Indian PHWRs.

2. NATURAL URANIUM FUELLED PHWRs

The reactor consists primarily of a horizontal cylindrical vessel called calandria which is penetrated by a large number of fuel channels arranged in a square lattice. The fuel channels consist of a calandria tube and an inner pressure tube through which the fuel bundles are moved by the on-power fuelling machine. The pressurized hot heavy water coolant flows through the pressure tube. The cool moderator is contained within the low pressure calandria. The control and shut-off rod devices are vertical and contained within the low pressure moderator. Table-1 gives various details for both 220 and 500 MWe units.

The fuelling scheme is on-power bidirectional multibundle shift scheme. For the natural uranium fuelled reactor, 8 bundle shift has been

TABLE-1

Design data for Indian PHWRs

1.	Electric power MWe	235	500
2.	Thermal power to coolant MWth	756	1730
3.	Calandria vessel diameter cms	600	780
4.	Number of fuel channels	306	392
5.	Lattice pitch cms	22.86	28.6
6.	number of bundles per channel	12	13
7.	Maximum channel power MWth	3.2	5.5
8.	Number of adjuster rods	6	21
9.	Number of shim/control rods	2	4
10.	Zone control compartments	-	14
11.	Number of shut of rods	14	28
12.	Number of vertical liquid poison tubes(SSS)	12	-
13.	Number of horizontal poison injection tubes	-	6

chosen. The core is divided into two or three radial zones to achieve radial power flattening in equilibrium condition and in each zone a discharge burn up is so chosen that the designed total power is obtained and the limits on the maximum channel powers and the maximum bundle power are satisfied.

The LWRs have the problem of controlling the excess reactivity that is built in to compensate for reactivity depletion when the reactor operates at power. The PHWR, using on power refuelling, has only a small amount of excess reactivity present in the reactor at any time in the equilibrium condition. The only time the problem of controlling excess reactivity is faced in a PHWR is when the reactor has initial fuel loading of fresh fuel. This excess reactivity is controlled with poison in the moderator which is removed gradually. The radial power flattening is achieved by making a judicious choice of the number and locations of either fuel bundles with lower U^{235} content or thorium bundles. On power fuelling is also used to maintain proper power shape.

Positions of various devices are chosen based on the neutron flux distribution for the fuelling scheme such that the devices give desired reactivity worths.

3. CHARACTIRISTICS OF PLUTONIUM

A reactor operating with fuel which has U-238 in it, contains varying amounts of plutonium isotopes at different irradiations. These plutonium isotopes contribute to the reactivity of the reactor and its energy production by way of in situ burn up of plutonium, differently according to the fuel irradiation. Fig.1 gives the percentage of energy production by U-235 and plutonium for the case of natural uranium fuel with nineteen rod cluster used in 220 MWe PHWRs. It can be seen that by the time the fuel is

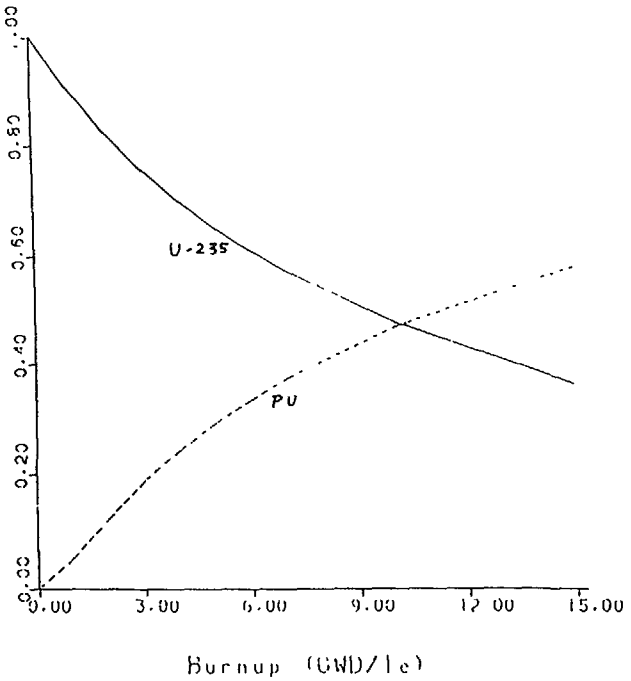

FIG. 1. Contribution to energy from U-235 and plutonium vs burnup.

discharged at 7000 MWD/Te burn up, the plutonium contribution to energy is about 40 percent. Further burn up extension of the fuel and possible in situ burn up of plutonium may not be possible in natural uranium fuelled PHWRs due to limitation of reactivity. When plutonium is introduced in the reactor in the form of mixed oxide of uranium and plutonium (MOX), with a view to decrease requirement of uranium, some problems may be encountered because of the particular characteristics of plutonium fuel.

3.1 General Characteristics and their Effects

It is known that the fission cross section of both Pu-239 and Pu-241 are higher than for U-235; so are the capture cross sections. At low

TABLE-2

Microscopic Fission and Capture Cross Sections(barns) at 2200 m/s neutron speed for Uranium and Plutonium Isotopes

Isotope	Fission	Capture	Beta
U-235	579.0	100.0	0.0065
U-238	-	3.0	0.0173
Pu-239	741.0	267.0	0.0020
Pu-240	-	290.0	-
Pu-241	1009.0	368.0	0.0049
Pu-242	-	19.0	-

energies in the vicinity of 1.0 ev, there are a number of fission resonances of Pu-239 and Pu-241 and absorption resonances for Pu-239, Pu-240, and Pu-241. Prominent are the fission resonance at 0.3 ev for Pu-239 and absorption resonance at 1.0 ev for Pu-240. Added to this is the fact that the isotopic composition of plutonium undergoes continuous change. These facts result in a highly complex contribution of plutonium to reactivity and power production in the soft spectrum of PHWRs.

The following general effects can be observed in LWRs/PHWRs while using fuel enriched with plutonium.

3.1.1 Reactivity Worth of Absorber Elements

The spectrum in lattices with plutonium is hardened and because of higher thermal absorption cross section of plutonium isotopes and occurence of absorption resonances, the other absorbing elements close to the plutonium bearing fuel have to compete with it; also in the vicinity of fuel with plutonium, the flux would be depressed. This would result in the reactivity worth of the absorber elements being lower than in the case of natural uranium fuel.

3.1.2 Power Peaking

There is a significant difference in the nuclear properties of a lattice with natural uranium and one with plutonium enrichment. Fig.2 shows variation of thermal group fission cross section as a function of burn up for natural uranium and two types of plutonium bearing lattices of nineteen rod PHWR clusters which we call MOX-7 and MOX-7/12, to be described later. One can observe a sharper decrease in the cross section in the MOX fuel than in the natural uranium fuel for the same change in burn up. Because of this reason, existence in close proximity with each other of MOX fuel bundles

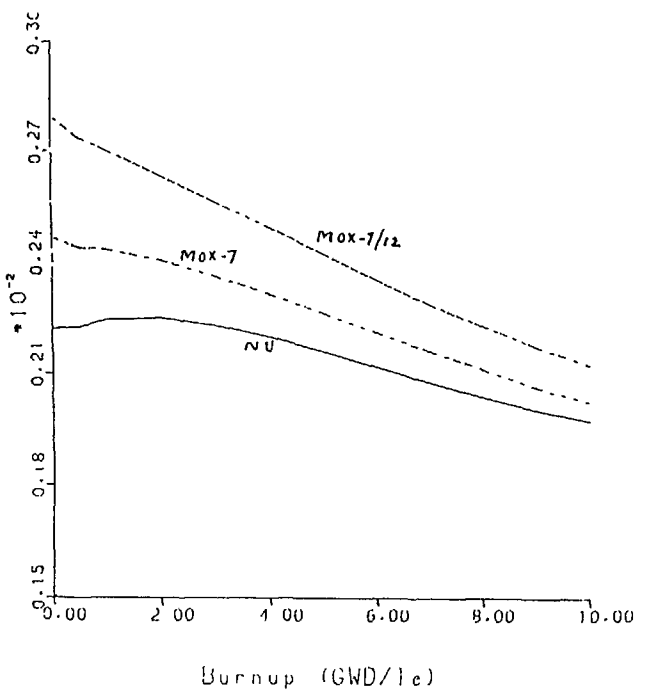

FIG 2 Thermal fission cross-section vs burnup for MOX and natural uranium

with significant difference in irradiation or of natural uranium bundles and MOX bundles would lead to power peaking .

3.1.3 Reactivity Coefficients

The hardened spectrum and higher resonance absorption in plutonium isotopes result in reduction of reactivity coefficients of plutonium bearing fuel in comparison with natural uranium fuel. The magnitude of the fuel and moderator temperature coefficients as well as the coolant void coefficient would be reduced. The latter has a bearing on the safety of the reactor and can be suitably used.

3.1.4 Control

The fraction of delayed neutrons from Pu-239 is smaller than for U-235 and that for Pu-241 lies between the two. The effective delayed neutron fraction will depend upon the percentage of U-235, U-238,Pu-239 and Pu-241 in the fuel as well as the contribution of U-238 fissions. However it would be smaller than for the case where all fissions are from U-235. Such a situation exists to some extent in a PHWR in burn up equilibrium condition and the control system acts fast enough to take account of the smaller value of the delayed neutron fraction is always possible.

4. OPTIONS FOR BURN UP EXTENSION IN INDIAN PHWRS

Since the burn up extension programme is treated as interim, it would be carried out, to start with, in the present medium sized reactors for which the design is already standardised. Hence plutonium recycling would involve minimum deviation in the basic design of the fuel cluster, the reactor hardware and the control systems etc. In such an approach, small development efforts would be possible but major design changes would be ruled out. The alternative approach is to optimise the reactor design for the new fuel which may involve redesigning of the lattice, reassessment of control requirements and redesigning of the control system etc. This approach may involve an intensive development programme. Presently, we are not taking this approach.

In the approach chosen, since the fuel cluster and the coolant channel design is not to change significantly, the limits on the power that a fuel rod in the cluster and each of the coolant channels can produce is finalised. Also the locations and the number of the control elements and the fuel handling system are finalised. The burn up extension would have to be planned within these contraints.

The limits on the fuel rod power in the cluster(typically given by integral k-detheta), and channel powers would mean that the new fuel configuration is to be so optimised that the ensuing local and global power distributions are very close to the earlier one, if not better. The power distribution in the cluster can be optimised to be within the limit of integral k-detheta even with MOX fuel by resorting to differential enrichment in various rods in the bundle.

A fuel cluster for extended burn up would have to withstand high burn up, high power ramps and preferably should have higher operating margins; there should be provision to collect fission gases. These characteristics could be achieved by having sufficient clad thickness, graphite coating of fuel pellets and higher number of fuel rods in the cluster. For the burn up extension that is envisaged right now, it is felt that the present collapsible cladding may not pose any problems. A number of fuel bundles with graphite coating of pellets and a cluster of 22 instead of usual 19 rods are being irradiated in NAPP-1 to get experience.

The limit on the power in each channel according to the coolant flow in that channel, fixed by orificing, may still be violated unless the plutonium enrichment is small but in that case the burn up would not increase significantly. In case high burn up increase is to be achieved, the global power distribution in the reactor operating with plutonium fuel should be close to what is available with natural uranium fuel. This would also ensure that the reactivity worths of the various control devices are not reduced. Such a power distribution would be possible only by proper fuel management strategies because the peaks in the power distribution would depend upon whether fuel bundles with nuclear parameters differing largely reside in the neighbourhood of each other.

In this strategy of burn up extension, we would be closing the uranium cycle. We would also try to gain experience to prepare for the Th-U^{233} reactors of the third stage with minimum of design changes now in the present reactors and without losing the present reactor performance characteristics. A cycle using thorium, Once Through Thorium (OTT), in which irradiation of pure thorium bundles would be segragated from irradiation of MOX fuel bundles, is being considered. In this cycle, benefit of in situ burn up of U^{233} produced would be obtained and experience with irradiation of thorium would also be obtained, whereas the option of closing the thorium cycle need not be exercised immediately.

The next section gives our experience in establishing feasibility of a scheme to extend burn up by recycling MOX in the medium sized reactor with minimum deviation from reactor parameters using natural uranium fuel.

5. A SCHEME FOR EXTENDING BURN UP FOR MEDIUM SIZED PHWRS

It was decided that the integral k-detheta of 40 watts/cm as in the case of the medium sized PHWR using natural uranium fuel would be the limit on the maximum heat producing rod in the fuel cluster. The flux distribution in the cluster would decide the maximum power that a bundle was capable of producing. Investigations have been carried out with 19, 22 and 23 rod clusters with plutonium enrichments either in natural uranium or depleted uranium oxide.

TABLE - 3

Lattice Calculations to Analyse Various Options

FUEL	Central 7 Rods	Outer 12 Rods	k-infinity	Max.Bndl. Power KW.	REMARKS
N.U.	N.U.	N.U.	1.1154	420	-
MOX-7/12	0.4 MOX	0.2 MOX	1.2373	441	-
0.4 P.C.MOX-7	0.4 MOX	N.U.	1.1829	384	-
0.6 P.C.MOX-7	0.6 MOX	N.U.	1.2074	362	-
MOX-7IN D.U.	0.9 MOX	N.U.	1.1972	372	LOW BURN UP
MOX-7/12 D.U.	0.9 MOX	0.7 MOX	1.2473	435	LOW BURN UP

After analysing different cases, as shown in Table 3, the following two fuel lattices were studied in detail to start with. (a) MOX-7 with central seven rods having 0.4 wt.percent plutonium oxide in natural uranium oxide and the outer twelve rods of natural uranium oxide and (b) MOX-7/12 having central seven rods with 0.4 wt.percent and outer twelve rods 0.2 wt.percent plutonium oxide in natural uranium oxide.

There were two problems in the scheme viz. design of a reactor in equilibrium with MOX fuel and the transition of the present natural uranium reactor to the MOX reactor. While optimising, for both the cases, the constraints of total power, limits on bundle and channel powers, limitations on the fuel handling system and the control system were satisfied.

5.1 Equilibrium Reactor

For the MOX-7 reactor, a central region with all natural uranium fuel and an outer region with MOX-7 fuel were devised while retaining eight bundle shift. This was essential to avoid peaking of power in the central core that otherwise would have occured. The average exit burn up of the reactor would go up from 7000 MWD/Te to 10300 MWD/Te and the refuelling requirement would come down from an average of 7.4 fuel bundles to five fuel bundles for each full power day.

The reactor with MOX-7/12 could give exit burn up of 13500 MWD/Te without violation of any contraints if a combination of two and four bundle refuelling scheme is resorted to. In case the present eight bundle scheme is continued, the maximum channel power limit would be exceeded. With this value of the exit burn up, the requirement of refuelling would come down, on an average, to 3.8 fuel bundles per day. However, the following development effort would be necessary.

(1) For an average exit burn up of 13500 MWD/Te the maximum burn up the fuel would go up to about 20000 MWD/Te Fuel development to withstand this high value would be necessary.

(2) Plenum for fission gas collection would be necessary.

(3) In the case of a combination of two and four bundle refuelling scheme, the fuel bundles would have to withstand power ramps. Graphite lubrication of the fuel may solve this problem.

(4) Though the number of bundles to be refuelled would be less than four per day, on an average, two channels may have to be visited every day. The fuel handling system should be geared to take this load efficiently.

5.2 Transition to MOX Reactor

The studies carried out show that it is possible to change over from the present natural uranium equilibrium reactor to a MOX reactor without any derating of the reactor power. This however would require introducing natural uranium fuel bundles along with the MOX bundles to avoid axial power peaking in the first phase of transition for the first option of MOX-7 fuel. The transition in the case of MOX-7/12 would require introduction of some depleted fuel bundles in addition to the natural uranium bundles.

In order to avoid introducing natural/depleted uranium bundles as well as for avoiding the central natural uranium fuelled region, higher than 19 rods cluster could be helpful.

The transition to a MOX reactor could also be carried out by loading the reactor with all MOX fuel bundles at the time of replacing the present pressure tubes. This situation would require controlling excess reactivity which would be higher than for the fresh natural uranium fuel when the reactor operation starts. Also the core radial flattening would have to be optimised with depleted or thorium bundles. But with the experience of controlling such reactivity and achieving radial flattening in all the earlier natural uranium fuelled reactors, this should not pose a problem.

5.3 Safety and Control Characteristics

It was found that the reactivity worth of the control system, called adjuster rod system, would decrease only by about four percent for the MOX-7 reactor and by eight percent for the MOX-7/12 reactor.

TABLE - 4

Burnup (GWD/T)	Void Reactivity at Different Burn ups		
	Void Rectivity(mk)		
	Natural Uranium	MOX-7	MOX-7/12
0.0(NoXe)	8.47	7.94	6.57
0.0(with Xe	8.94	8.26	6.78
4.0	5.95	6.52	6.44
6.0	5.61	6.16	6.22
8.0	-	5.77	5.93
10.0	4.94	5.38	5.62

The reactivity addition when all the coolant gets voided is almost the same for both the reactors as can be seen from Table 4.

It was also found that the minimum value of beta effective of both the reactors would not be significantly different from the value of beta effective of the natural uranium reactor at equilibrium.

6. CONCLUSIONS

From the studies carried out on the medium sized Indian PHWRs, extension of burn up in this size reactors is feasible with plutonium obtained from the natural uranium fuelled PHWRs. In the scheme that could be started with minimum of development effort, saving of natural uranium fuel of about 40 percent is possible. The transition to such a reactor is possible without power derating. The control and safety characterisics would remain almost unchanged.

BURNUP EXTENSION PLAN OF BWR FUEL AND ITS IMPACT ON THE FUEL CYCLE IN JAPAN

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Abstract

In Japan burn up extension of BWR fuel is being pursued in a stepwise way to 45 GWd/t (batch average). Within this burnup range, no major technical issue and no major licensing program will be expected. In case of burnup extension over 45 GWd/t some modifications will be necessary throughout the nuclear fuel cycle to provent problems on criticality, shielding, etc.

0. BURNUP EXTENSION PLAN OF BWR FUEL IN JAPAN

0.1 Introduction

In Japan number of nuclear power plants have been steadily operating in these years and playing quite an important role in electricity supply with fairly high capacity factors.

Nuclear power generation has several advantages compared with other power generation methods and one of the prominent features should be that it has lower power generating cost(total power generating cost including construction cost, maintenance cost, fuel cost and etc). Extension of operating cycle length will further improve the economical advantages of nuclear power plants through higher capacity factor, in this case it is favourable to use the fuel which can be usable in the high burnup regions. In addition to this, extended burnup results in lower fuel cycle cost even if the cycle length is kept constant. Burnup extension also leads to the decrese of number of spent fuel, which is desirable to mitigate the burden on backend of fuel cycle. So it is desirable to introduce high burnup fuel in commercial use as soon as possible. In those background, developement of high burnup fuel has been pursued.

0.2 Target Burnup

Extended burnup has significant impact on the economical aspects of nuclear fuel and earlier introduction is desired as mentioned already. But as the target for high burnup goes higher and higher, the task for research and development would become the more difficult. In this connection it appears to be a more effective and steady way to apply the technologies already established one after another and put forword to the direction to high burnup in a stepwise manner. Our approach on the development of high burnup fuel is shown in figure0-1.

Current database on the high burnup experiences in the foreign commercial reactors demonstrates that no particular failures associated with the increased burnup are seen in this burnup range(52GWD/t). We have been pursued our development

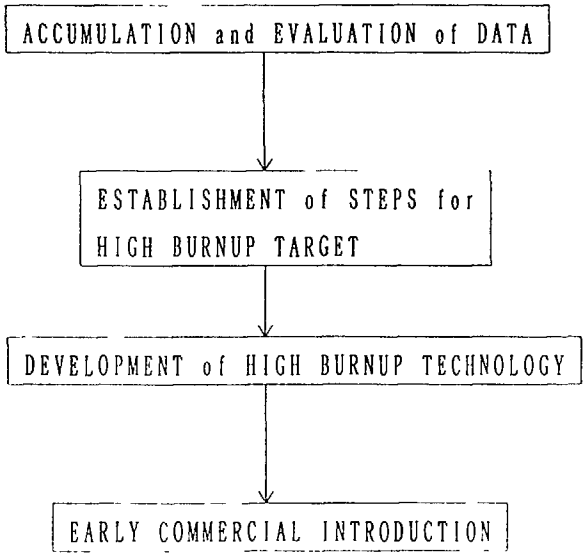


Fig. 0-1. Steps for High Burn-up

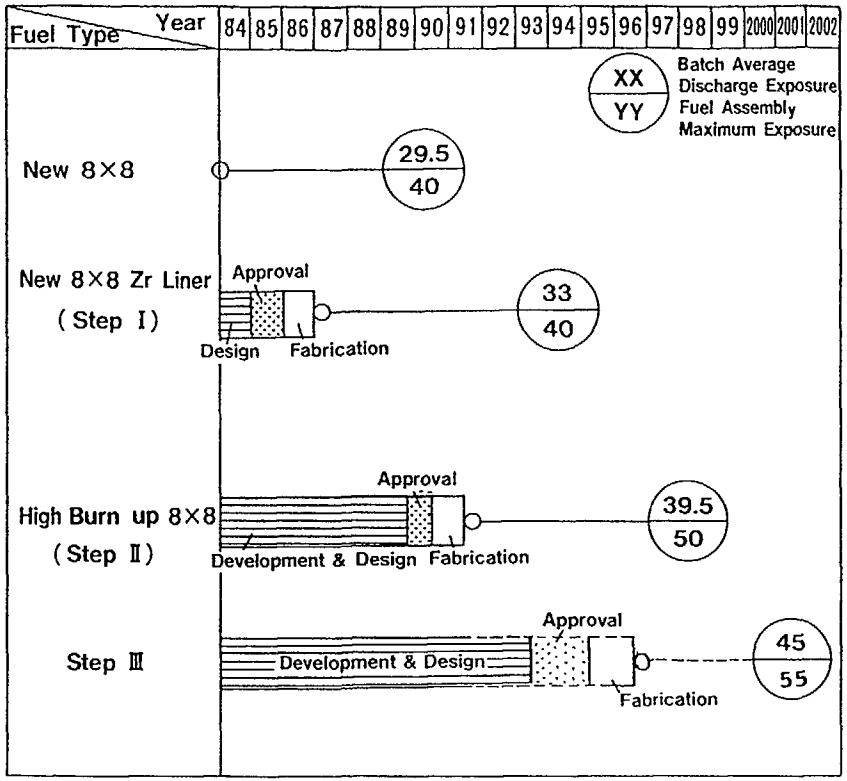


Fig. 0-2 BWR Burnup Extension Plan

effort in the following steps. Target of high burnup fuel introduction and their schedule are shown in fig 0-2.

Step I	Average discharge burnup	approx.33GWd/t
	Maximum assembly burnup	40GWd/t
Step II	Average discharge burnup	approx.39GWd/t
	Maximum assembly burnup	50GWd/t
Step III	Average discharge burnup	approx.45GWd/t
	Maximum assembly burnup	55GWd/t

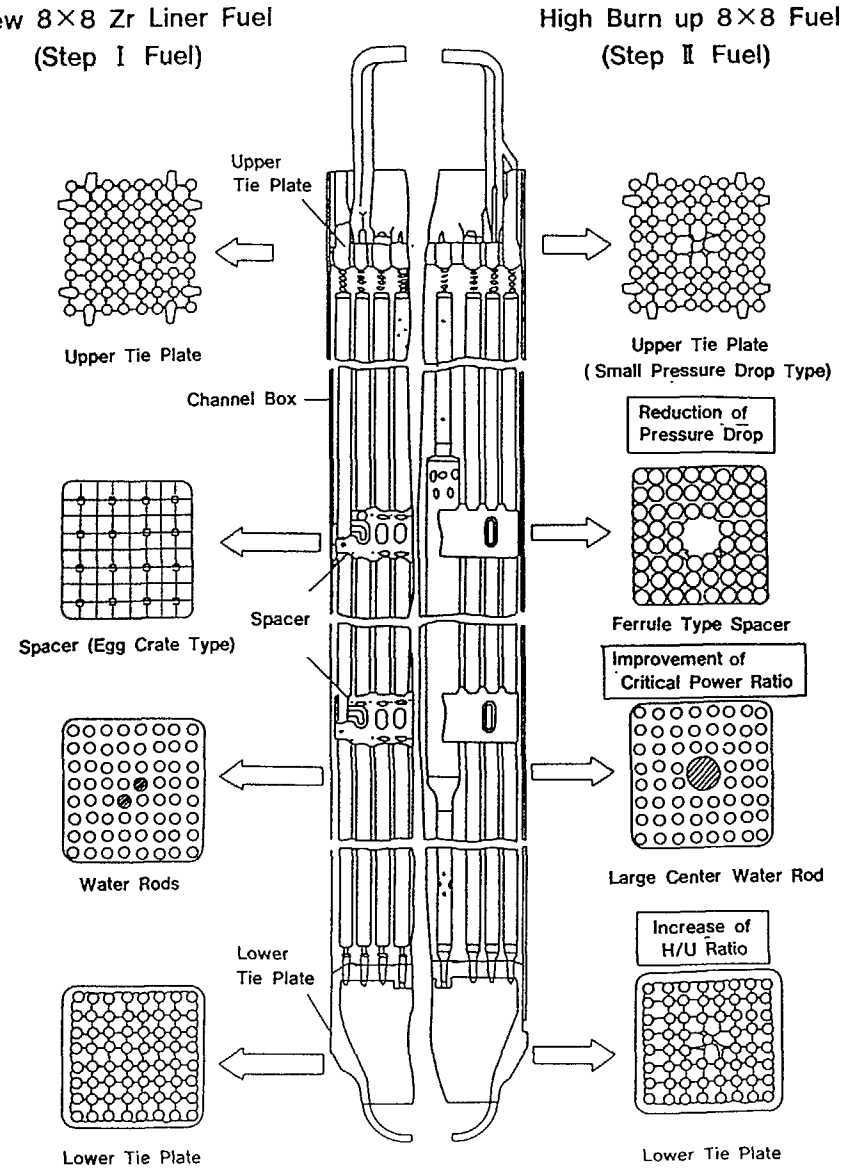


Fig. 0-3 Structure of StepⅡ Fuel

0.3 Fuel Design

(1)StepⅠ Fuel

StepⅠ fuel has been used since 1986 in Japan. This fuel utilizes the Zirconium barrier fuel, and enough reliability can beconfirmed through current data. (Though over 8000 fuel assemblies have been introduced in Japanese BWR core, there is no fuel failure yet.)

(2)StepⅡ fuel

StepⅡ Fuel is the next stage fuel whose batch average discharge exposure is about 39 GWd/t. Figure 0-3 shows a comparison between StepI fuel and StepII fuel. In StepII fuel, ferrule type spacer is introduced. Using the ferrule type spacer, Critical Power Ratio can be improved about 15%.

Another structual characteristic of StepII fuel is large center water rod which occupies the four fuel rods region in the bundle.

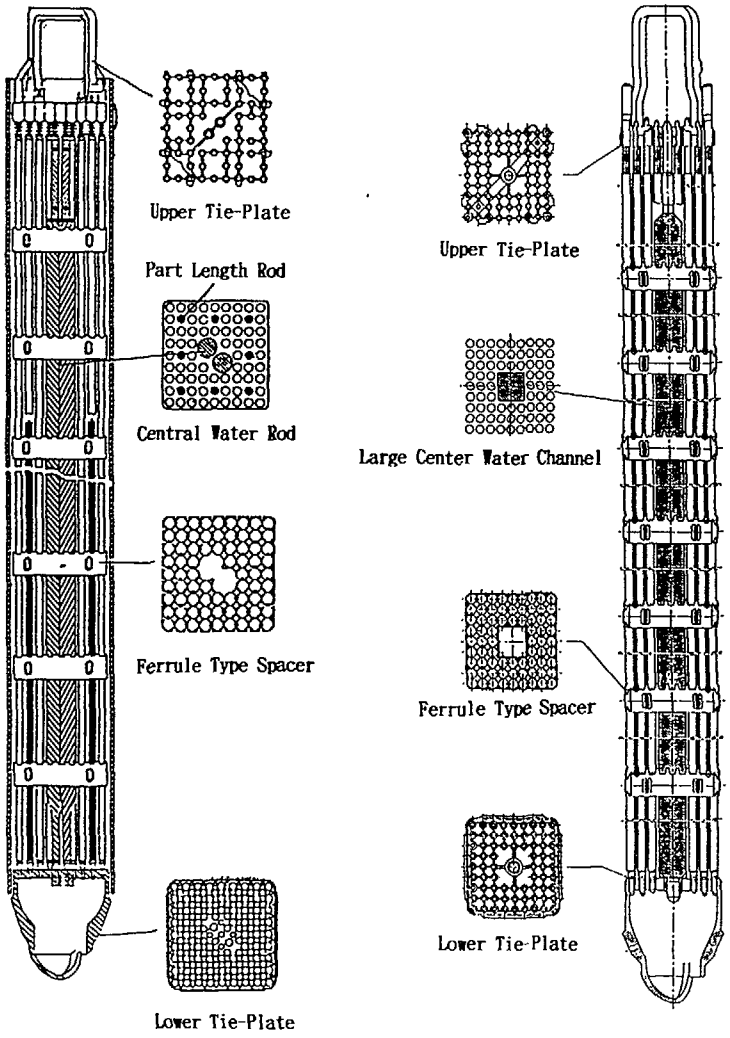
Because average bundle enrichment of StepⅡ Fuel(3.4%) is greater than that of StepⅠ Fuel, neutron spectrum hardening causes a change for the worse of void coefficient and shut down margin. Therefore, it is necessary to soften neutron spectrum by increase of H/U ratio. To increase H/U ratio, large center water rod is introduced.

(3)StepⅢ Fuel

Japanese BWR utilities have been investigated two types of fuel design as stepⅢ Fuel (figure 0-4).

The direction of improvements towards burnup extension of StepⅢ fuel are basically same. In nuclear design terms, it is necessary to increase H/U ratio.

In StepⅢ fuel, bundle average enrichment will be increased to about 4.0%. So we must use bigger water rod in fuel bundle. In StepⅢ A-type fuel, two central water rods will be introduced into the seven-fuel rods region. And in StepⅢ B-type fuel, large water channel will be introduced into the nine-fuel rods region.



Type A Type B

Fig. 0-4 Structure of StepⅢ Fuel

Most remarkable design change of StepⅢ fuel is introduction of 9 x 9 lattice. 9 x 9 lattice arrangement bring certain merits. One is 20% decrease in average linear heat generation rate and another one is a greater flexibility in fuel assembly design. For example, flexibility of Gd fuel rods arrangemnet will increase, and we can also simplify fuel rod enrichment sprit design.

1.EFFECT OF BURNUP EXTENSION ON NUCLEAR FUEL CYCLE STRATEGY

1.1 Overview

Japan's nuclear power generation has been established technologically, and compared very favorably in performance with that of the other countries which have nuclear power plants. In the nuclear fuel cycle, we are now able to fabricate all the nuclear fuel we need, but as for such critical processes as enrichment and reprocessing we mostly depends on overseas facilities.

To establish the entire nuclear fuel cycle in Japan, two companies are planning to construct three key nuclear fuel cycle facilities at Rokkasyo-mura in Aomori Prefecture, Nortern Part of Japan. These facilities are a spent fuel reprocessing plant, a uranium enrichment plant and a low-level radioactive waste storage facility.

1.2 Effect On Front End Strategy

(1)consumption of natural uranium

To ensure the steadily supply of uranium resources, its sources should be diversified, the ratio of development to import should be increased, and prospective resources (interests) should be acquired whereever possible.

Figure 1-1 shows relative comparison of natural uranium consumption among high burnup fuels.

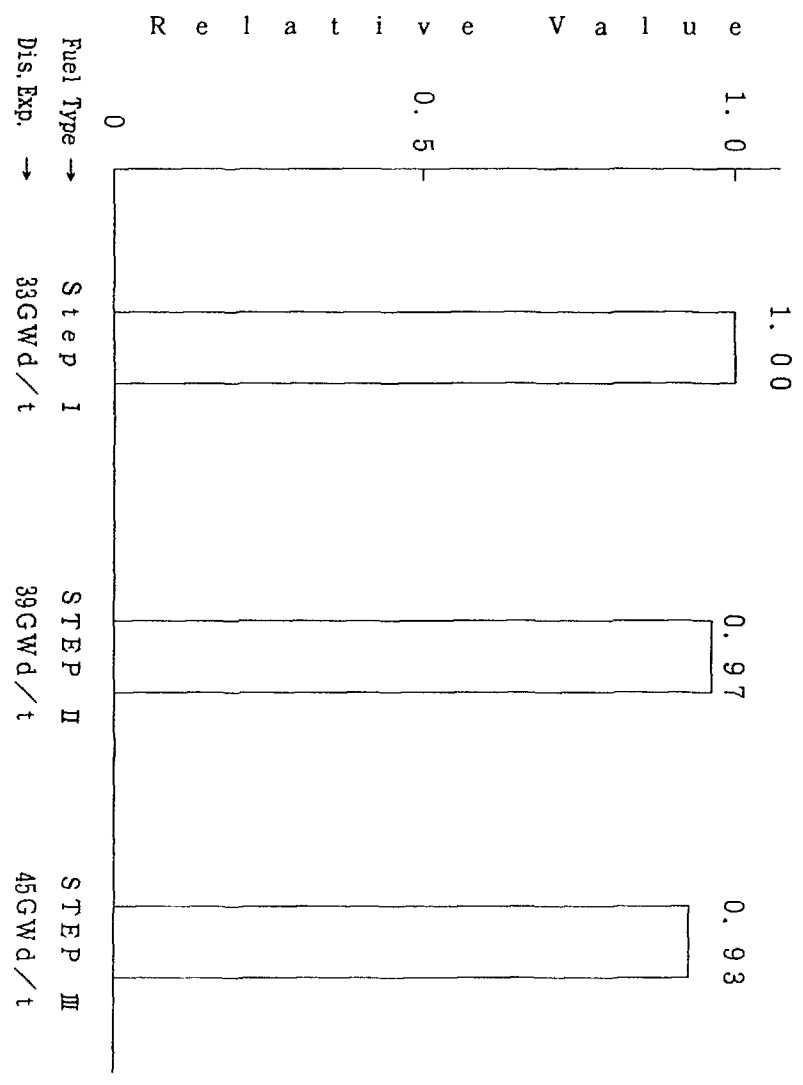


Fig. 1-1. Relative Comparison of Uranium Requirement

Though higher burnup fuel has a tendency of decreasing an amount of natural uranium required, the defferences are not so large. The reason of this decrease results from decreasing batch size, because these calculation is based on the same operational period, high burnup fuel which has a small batch size is burned effectively. In the case that operational period is extended in propotion to extended burnup, an amount of natural uranium required become almost constant.

Since the effect of burnup extension on consumption of natural uranium is not so large, Japan's future strategy will not be affected. Japanese electric power companies have secured some amount of U_3O_8 by long-term contract with overseas mining companies. This assured quantity of uranium will meet the domestic demand for the time being. However, given the depressed state of exploration and development for new resources worldwide,along with the inevitably long lead time for uranium development,the global supply-demand situation of uranium may get tight.
So, we should be encouraged to ensure long-term, stable supply of natural uranium.

(2)Enrichment

In Japan, uranium enrichment services has been consigned to United States and France. On the other hand, it is expected that smooth and steady operation of domestic enrichment facility will be promoted by utilizing the already established centrifugal separation method. A gradual expansion of domestic enrichment facilities is expected.

Figure 1-2 shows relative comparison of SWU requirement among high burnup fuels. There is no difference among three type of fuels.

(3)Fuel Fabrication

Figure 1-3 shows relative comparison of fuel fabrication services among three type of fuels. As the amount of fuel

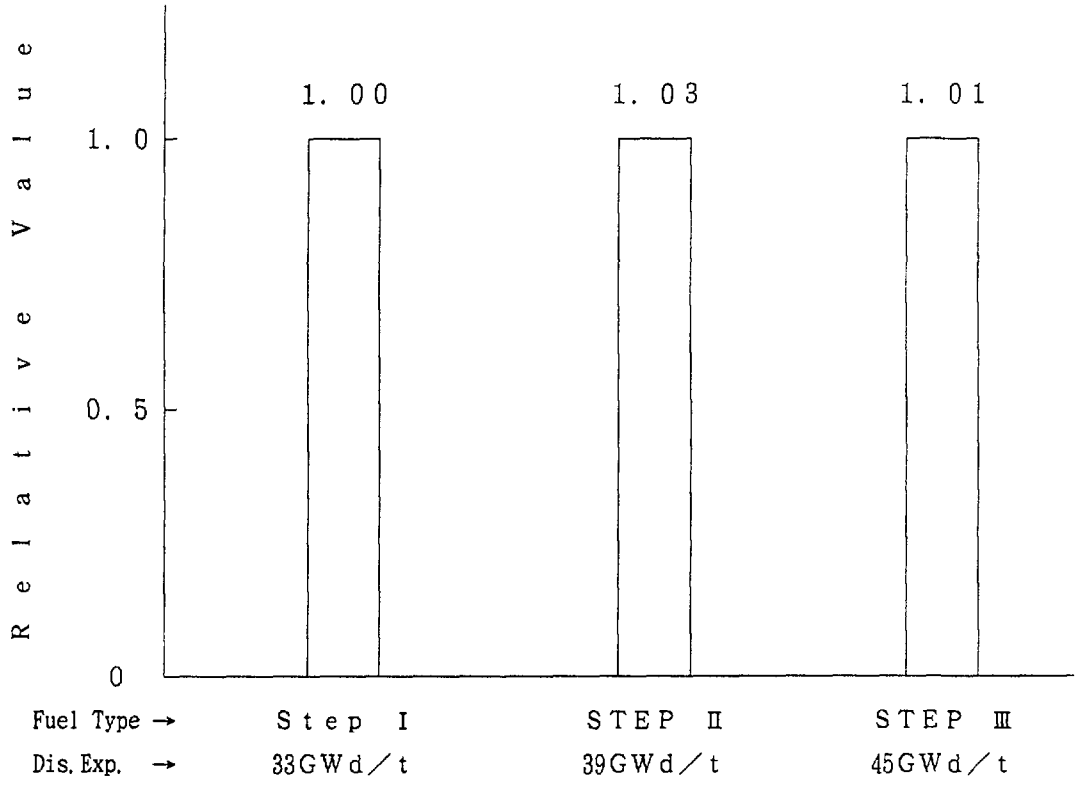


Fig. 1-2. Relative Comparison of SWU Requirement

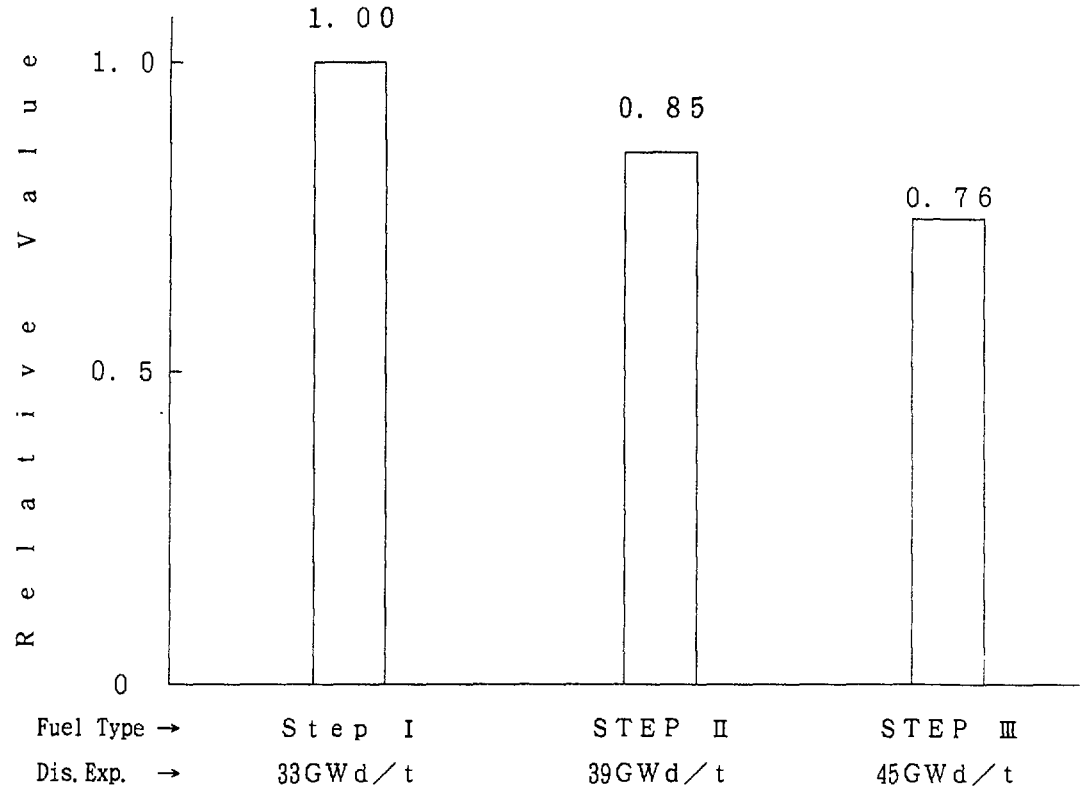


Fig. 1-3. Relative Comparison of Fablication

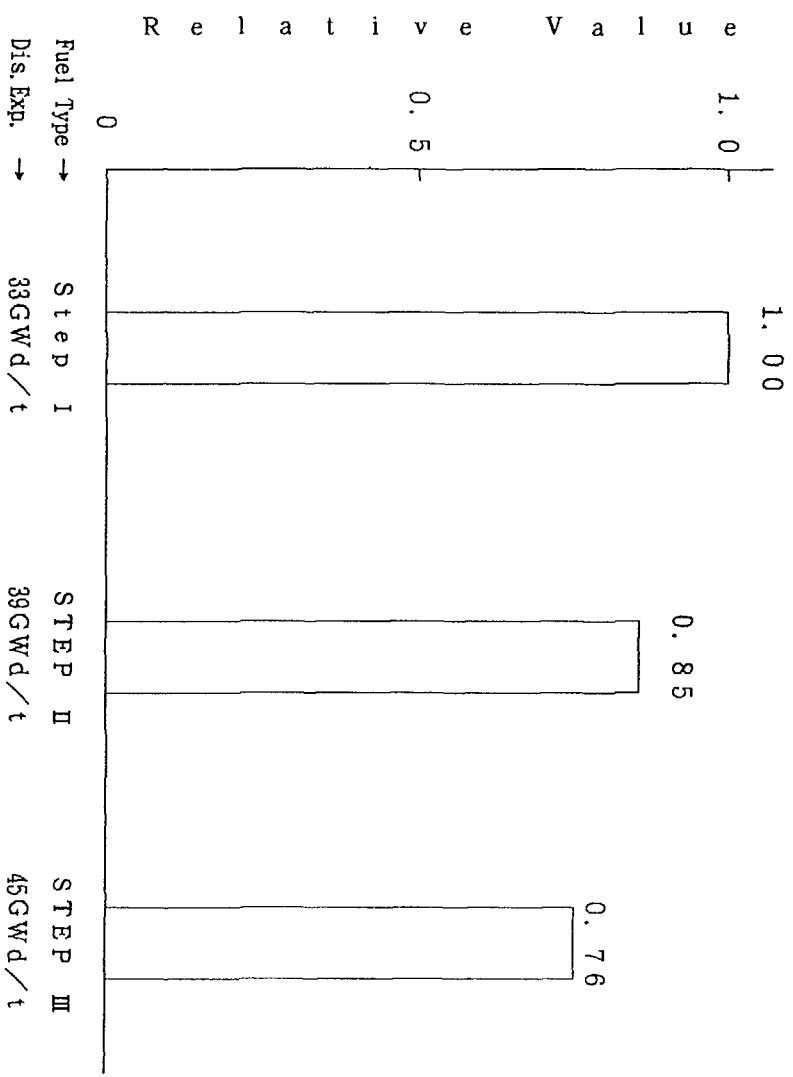


Fig. 1-4. Relative Comparison of Spent Fuel

fabrication service per fuel bundle is almost same, burnup extension contributes to decreasing fuel fabrication services for one cycle operation (usually 12 month in Japan).

Current domestic fuel fabrication capacity is enough to satisfy domestic demand. Moreover, the lead time required to build fuel fabrication facilities is relatively short, it would be easy to increase the capacity in a short period for expanded power generation.

1.3 Effect On Back End Strategy

(1)Reprocessing

In the case that Japanese utilities use high burnup fuel for 1 year's operation, fuel batch size decrease can be expected. Figure 1-4 shows relative comparison of the amount of spent fuel after 1 year cycle operation. For example, spent fuel generation will decrease about 25% by using StepⅢ fuel, when it is compared with using StepⅠ fuel.

Although Japanese utilities have made reprocessing contracts with BNFL and COGEMA in these years, spent fuel is expected to be reprocessed domestically in future. First commercial reprocessing plant(capacity:800t/y) will start operation around middle of 1990'.

Before reprocessing, spent fuel will be stored properly. Requirements is going to be studied and implemented so that the spent fuel can be stored in dry and in high density.

2.1 Increasing Reactivity In The Core And Associated Problems With Fuel Management And Reactivity Control

To extend fuel discharge burnup, it is generally required to increase enrichment. There is a tendency that simply increasing enrichment cannot gives us most of the merit, because of spectral hardening which is the consequence of increased thermal neutron absorption. Increased enrichment also leads to increase of reactivity defference and power mismatching between fuel bundles.

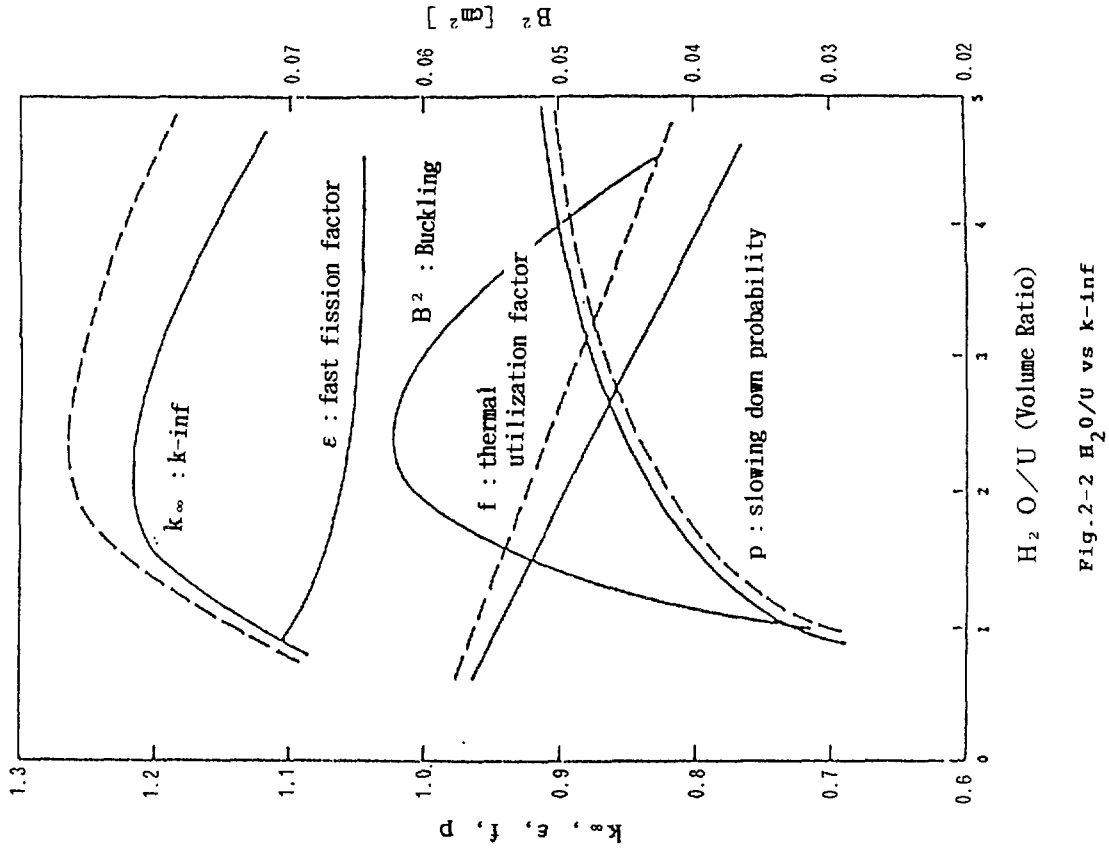


Fig.2-2 H_2O/U vs k-inf

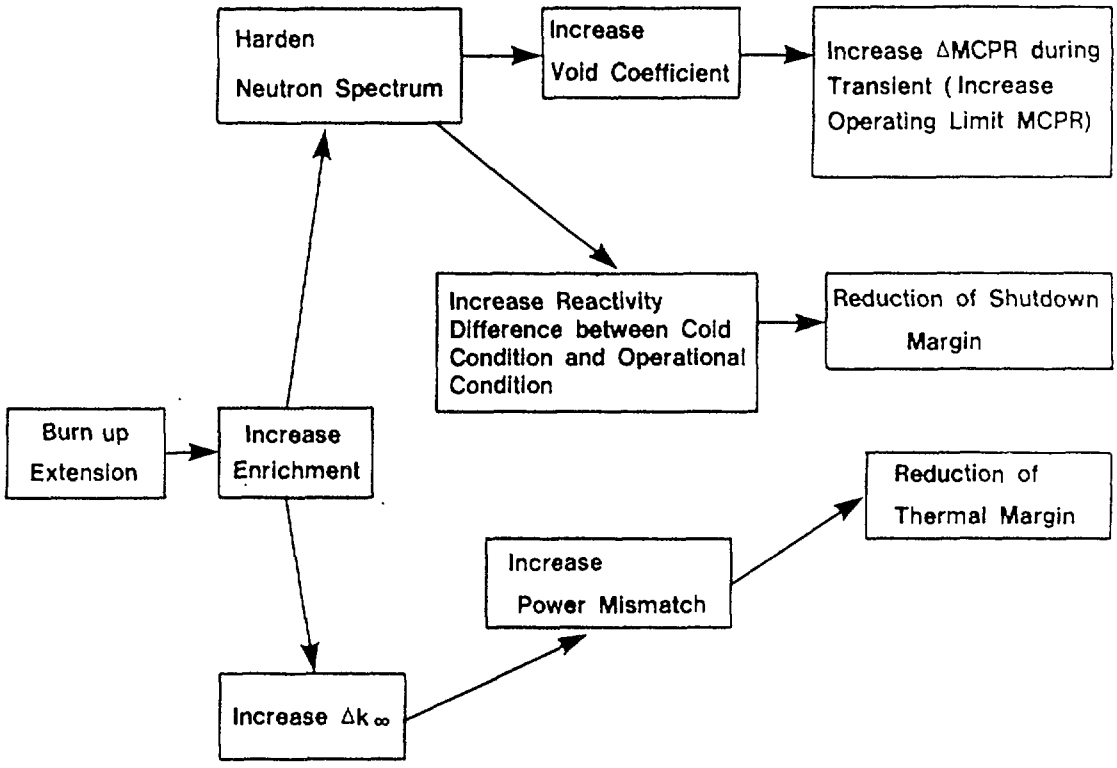


Fig.2-1 Burn up Extension Characterisity (Nuclear Aspects)

The main effects of the burnup extension in the nuclear property are all derived from these two characteristics. As typicaly illustrated in Figure2-1, spectral hardening and increased power mismatching leads to three effects listed below.

-Increased absolute value of the void reactivity coefficient
-Increased cold-hot reactivity swing
-Increased power peaking

As the consequence of these effects, two changes of the core property listed below occur.

-Decreased thermal margin
-Decreased cold shut down margin

In the following sentence, the countermeasures of these changes are described.

(1) Optimization of water to uranium ratio (Optimization of lattice design with increased enrichment).

First we should consider how the discharge burnup will change as the increased enrichment. With these effects, we must consider the change of the initial K-infinity and the change of K-infinity with burnup.

Figure2-2 shows the change of the main parameters which consist of K-infinity, using water to uranium ratio as a parameter.

This figure also shows the change of the parameters with increased enrichment:the thermal neutron utilization factor f increases with increased burnup. The slowing down probability p slightly decreases, because of spectral hardening. As the total effect of these parameter changes K-infinity changes as illustrated dotted line in this figure, thepeak value point changes to right side (to higher water to uranium ratio).This means that in the condition of using same water to uranium ratio, it is difficult to obtain the maximum gain byincreased enrichment

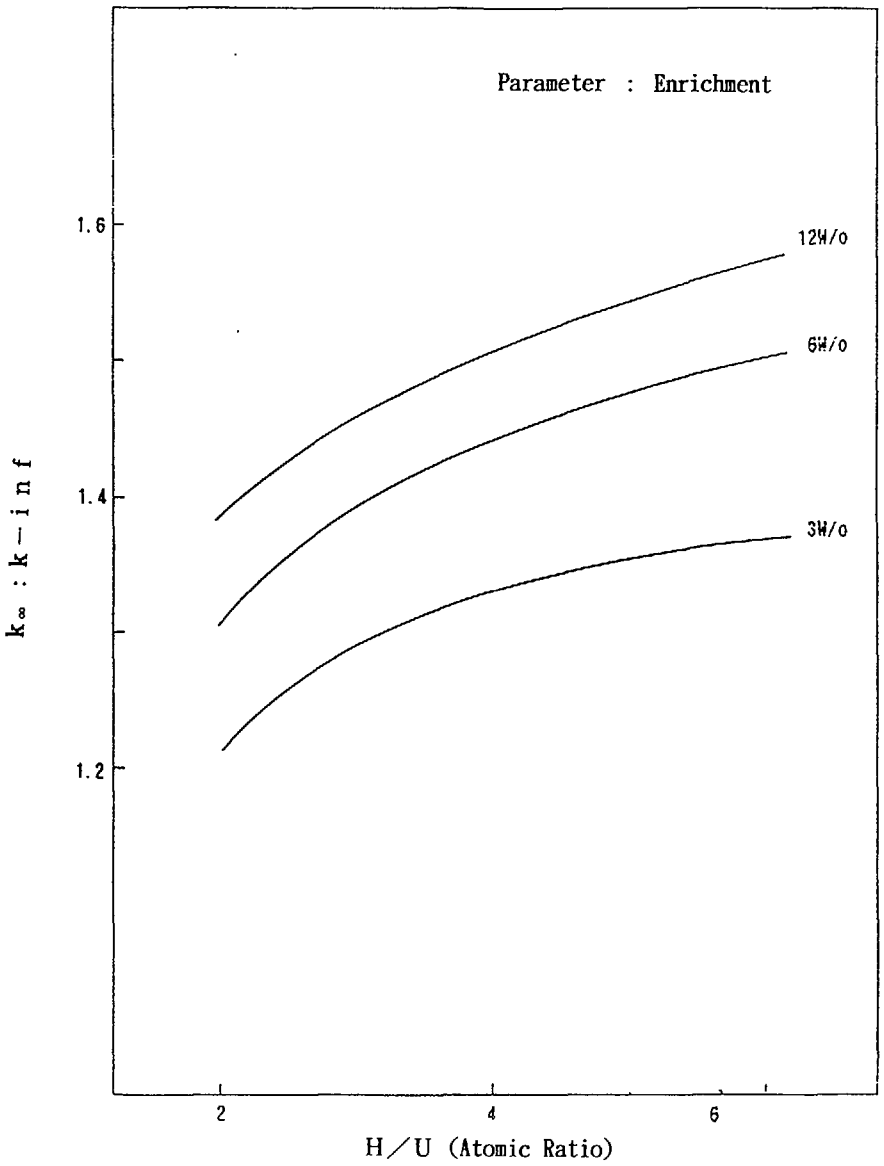


Fig.2-3 H/U vs k-inf

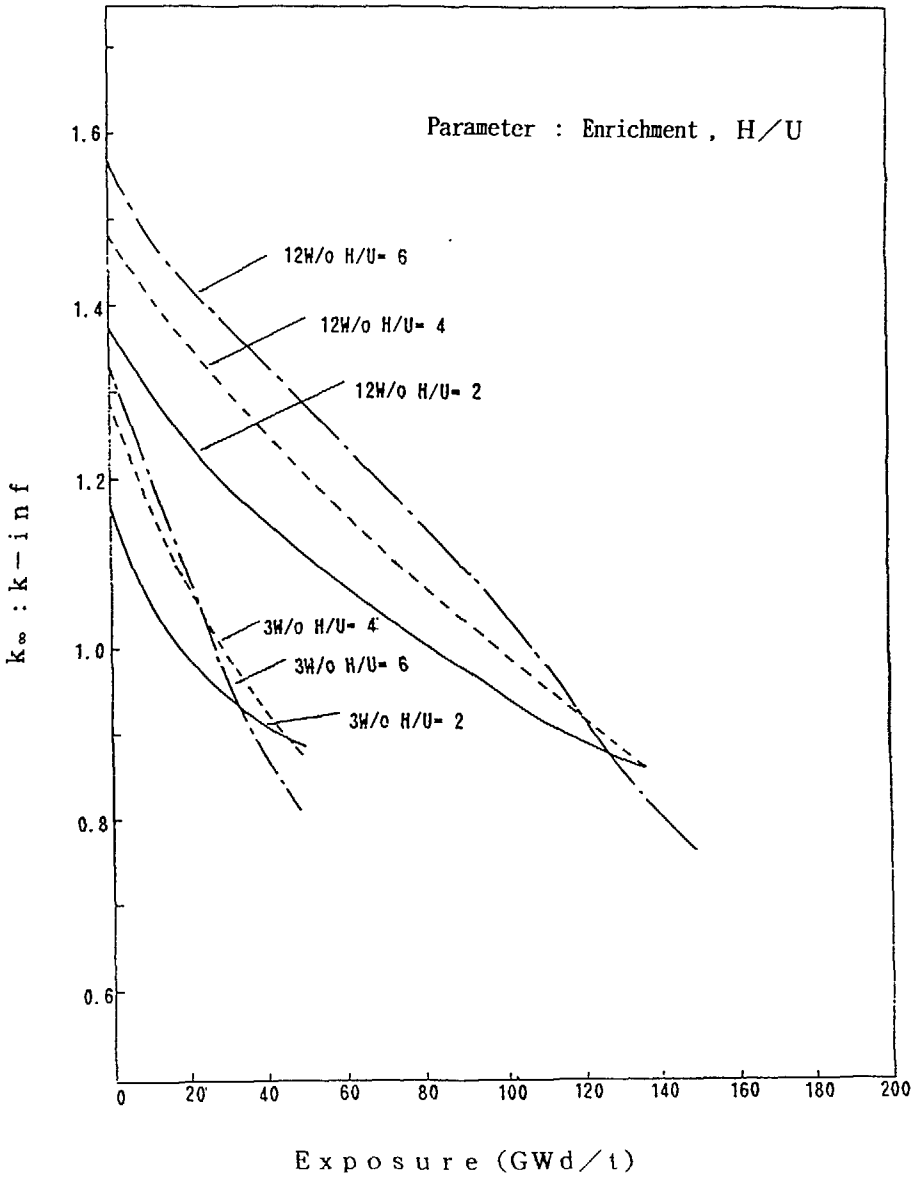


Fig.2-4 K-inf vs Exposure

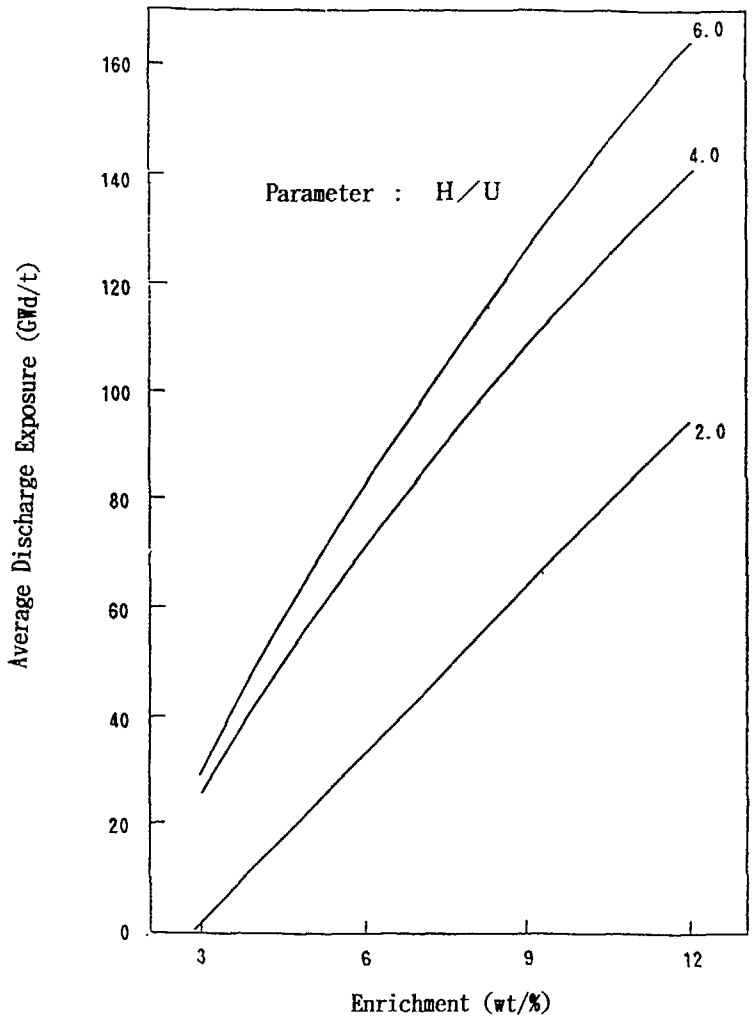


Fig.2-5 Average Discharge Exposure vs Enrichment

because the point will be far from peak point and K-infinity increase will be very small.

For example, Figure2-3 shows the relation between K-infinity and enrichment as a parameter of hydrogen to uranium number density ratio.

But spectral hardening causes not only bad effect, but also recovery effect of K-infinity in the high burnup region based on the increased Pu generation and higher conversion ratio.

To determine which effect is dominant, the burnup calaulation was made (See Figure2-4). Based on this result, the average discharge burnup was caluculated by linear reactivity model (See Figure2-5). As shown in this Figure, to achieve higher average burnup, it is favorable to have higher H/U ratio in spite of recovery effect of K-infinity. So it is very important to adapt optimized fuel lattice design in the view point of H/U ratio.

As mentioned above, higher H/U ratio is essential in the view point of reactivity. It is also effective as a countermeasure of increased peaking and decreased cold shut down margin.

(2) Power peaking factor

To design BWR core, it is convinient to divide power distribution into some factors. For this purpose three power peaking factors are defined as described below,

Radial peaking factor: peak bundle power to average bundle power ratio

Axial peaking factor : axially maximum planer power to average power ratio

Local peaking factor : maximum fuel rod power to average fuel rod power ratio in the bundle cross section

As the consequence of burnup extension, radial peaking factor tends to be increased because of the increased reactivity

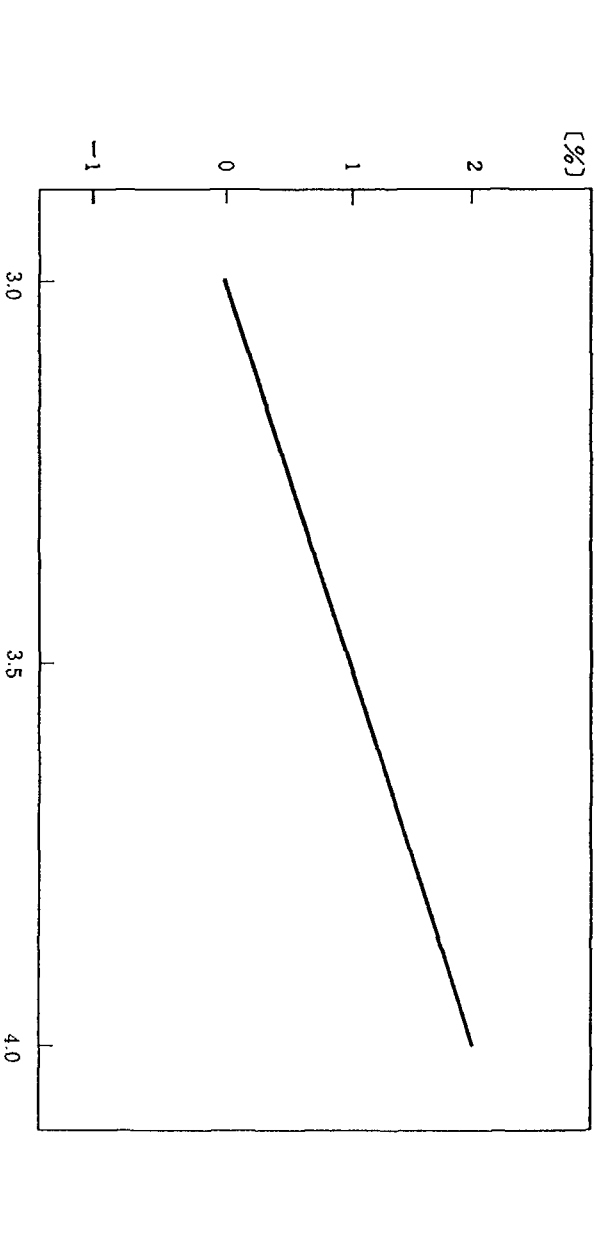


Fig.2-6 Average Enrichment vs Local Peaking Increase

defference between fuel bundles. For example, according to increased average discharge burnup, from 30GWd/t to 45GWd/t, the radial peaking factor is increased 3% in batch average(the same batch size and same fuel lattice configuration are assumed).

Local peaking factor tend to be increased, because of thermal neutron defference between peripheral region and central region in the fuel bundle and increased number of the gadorinia rods to suppress large excess reactivity(see Figure2-6).

As the consequence of these effect, the margin to thermal limit is decreased. So it is necessary to adequately control axial peaking factor and use adequate fuel lattice configuration to suppress large local peaking factor: as for BWR the techniqe to control axial peaking factor has been established by means of axially zoned reactivity core, and we can use this technique to generate enough thermal margin. As for the local peaking factor, we can suppress it by locating water rod in the central region of the bundle.

(3) Improvement of void reactivity coefficent

The void coefficent is defined as described below.

$$\frac{1}{Keff} \cdot \frac{d\ Keff}{d\ V} \qquad (V: \text{in channel void fraction})$$

The void reactivity coefficent is an important factor to dominate transient event and accident at BWR. Because the absolute value of coefficent is more than ten times larger than the other reactivity coefficent which is common to other type of reactors, doppler coefficent, moderator temparature coefficient and etc. In case of increased enrichment without modification of fuel lattice configuration, the sensitivity of K-infinity on water to uranium ratio, the absolute value of void coefficient, tends to be increased.

The increased absolute value of void coefficent tends to affect some results of transient events and leads to some difficulties such as decreased thermal margins

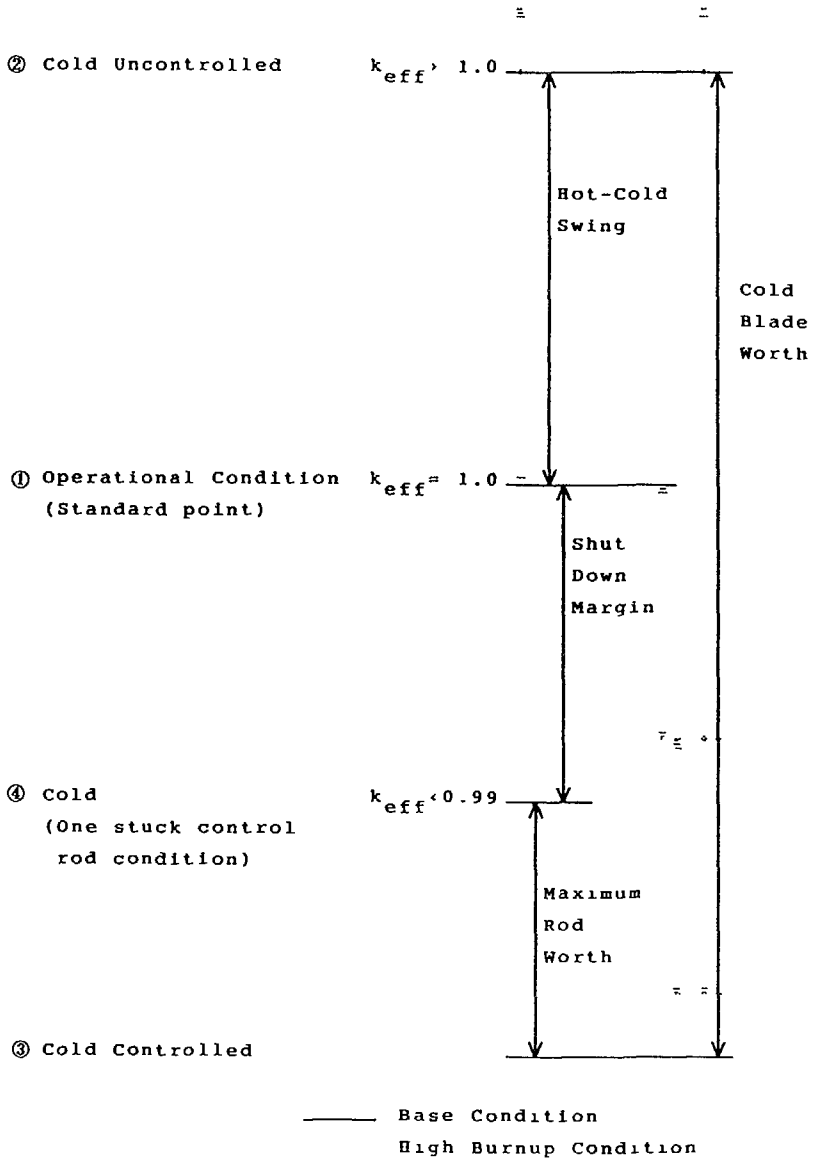


Fig 2-7 Degradation of cold shut down margin by high burnup

So it is also desirable to modify fuel lattice configuration to achieve adequate water to uranium ratio according to increased enrichment from the stand point of void reactivity coefficent.

(4) Improvement of cold shut down margin

Figure2-7 conceptually shows the relations of reactivities related to cold shut down margin. As for BWR the reactivity of the core becomes maximum value on the cold condition(② in Figure2-7), because on the operational condition(① in Figure2-7), there is a negative reactivity feedback. This reactivity defference between cold condition and hot operational condition is called cold to hot reactivity swing. If all control rods are inserted in the core, the core reactivity decreased by worth of control rods. When the control rod which has maximum worth is withdrawed at this condition, the Keff of the core becomes the value at ④ . The reactivity defference between ① and ④ becomes cold shut down margin.

In case of increased burnup without modification of fuel bundle configuration spectral hardening leads to increased cold to hot swing. Thus cold shut down margin is decreased shown as a dotted line in figure2-7.

To avoid this effect, the increase of water to uranium ratio with modification of the fuel lattice is considered as a countermeasure.

2.2 Optimization Of Fuel Assemblies And Fuel Element Design To Overcome Technical Limits

(1) Burnup extension within 45 GWd/t

These designs described below will be introduced.

— Increase of fuel rod plenum volume

This leads to reduction of the internal pressure. As the fuel burnup is increased, the production of FP gas is also increased. This increases the fuel rod pressure is so high, fuel cladding is lifted off. This is a bad effect on fuel mechanical reliability.

— Inclease of the He initial pressure and optimization of pellet-cladding diametral gap

These lead to improved thermal transmission characteristics. FP gas increase press and bad effect on thermal performance of fuel rod. The accumlation of FP gas reduce the thermal transmission rate, so the pellet temperature gets higher. As the burnup is incleased, He initial pressure of the fuel should be increased.

— Optimization of cladding thickness

To extend burnup with sufficient design margin, the optimization of cladding thickness is desirable.

— Ferrule type spacer

Ferrule type spacer produces better CPR performance and maintains thermal margins.

— High-Flow upper tie plate

Reduce two-phase bundle pressure drop and maintains thermal-hydraulic stability margins.

(2) Burnup extension beyond 45 GWd/t

R&D program was initiated to develop new fuel materials in order to explore the possibility for burnup levels up to about 70 GWd/t batch average.

Improved pellet materials and zirconium alloys are now being irradiated in the Halden reactor and a high power density commercial reactor in USA. A part of the specimens has been retrieved and examined in GE and NFD hot labratories. From 1990 the Halden BWR corrosion loop experiment, in which EPRI is partially participating, started to investigate the effects of

reactor coolant water chemistry on zirconium alloys and to accelerate the selection of the improved zirconium alloys.

After selecting new improved materials, the necessity of safety-related experiments such as LOCA and RIA simulations will be considered and the irradiation of Lead Use Assembly incorporating improved materials will be planned in detail.

These R&D activities are also useful for the burnup extension level less than 70 GWd/t and backfitable.

2.3 Licensing Problems

As for licensing, the items described below should be considered.

-The maximum value of average discharge burnup is described in E/P submitals. In case of burnup extension, Ministry of International Trade and Industry,MITI,and Atomic Safety Committee review the submitals even if it is a small number. The design methods to be used in this review were already reviewed by theses organizations and summerized in the report.

-As for the fuel fabrication facilities, the applicable enrichment range of the guide line is limited to less than 5%, in the case that more than 5% enrichment uranium is necessary, this guide line would be revised.

-The length of operational cycle is limited by law to 13 month, the burnup extension doesn't directly lead to extended cycle operation.

3. Impact on front end of nuclear fuel cycle

3.1 Burnup Extension Within 45GWd/t

The maximum value of enrichment depends on the assumed operational cycle length, but even if longer cycle, such as 18 month, is assumed, the maximum value is less than 5%. In this case, the impacts on front end become as described below.

(1)Fuel fabrication and reconversion

From the stand point of criticality control, the maximum batch quantity must be decreased according to the enrichment. But the modifications of the current facilities will not be needed, so the impact is small.

(2)Uranium and fresh fuel transportation

Current containers can be basically applicable for the transportation, though the maximum capacity should be limited in some cases.

3.2 Burnup extension beyond 45GWd/t

The maximum value of enrichment may exceed 5%. In this case, the impacts on front end should be investigated as described below.

(1)Fuel fabrication and reconversion

The applicable enrichment range of the guideline for safety analysis of fuel fabrication facilities in Japan is limited to less than 5%. In case that more than 5% enrichment uranium is needed, the revise of this guideline is needed.

The changes of criticality control in the facilities will be needed. They can be done by means of the decrease of the maximum batch quantity and modifications of the facilities, though the efficiency will be decreased. Figure 3-1 shows the facilities which supposed to be affected.

From the stand point of exposure control, the increase of exposure dose will be relatively smaller, so the major modification of the facilities will not be needed.

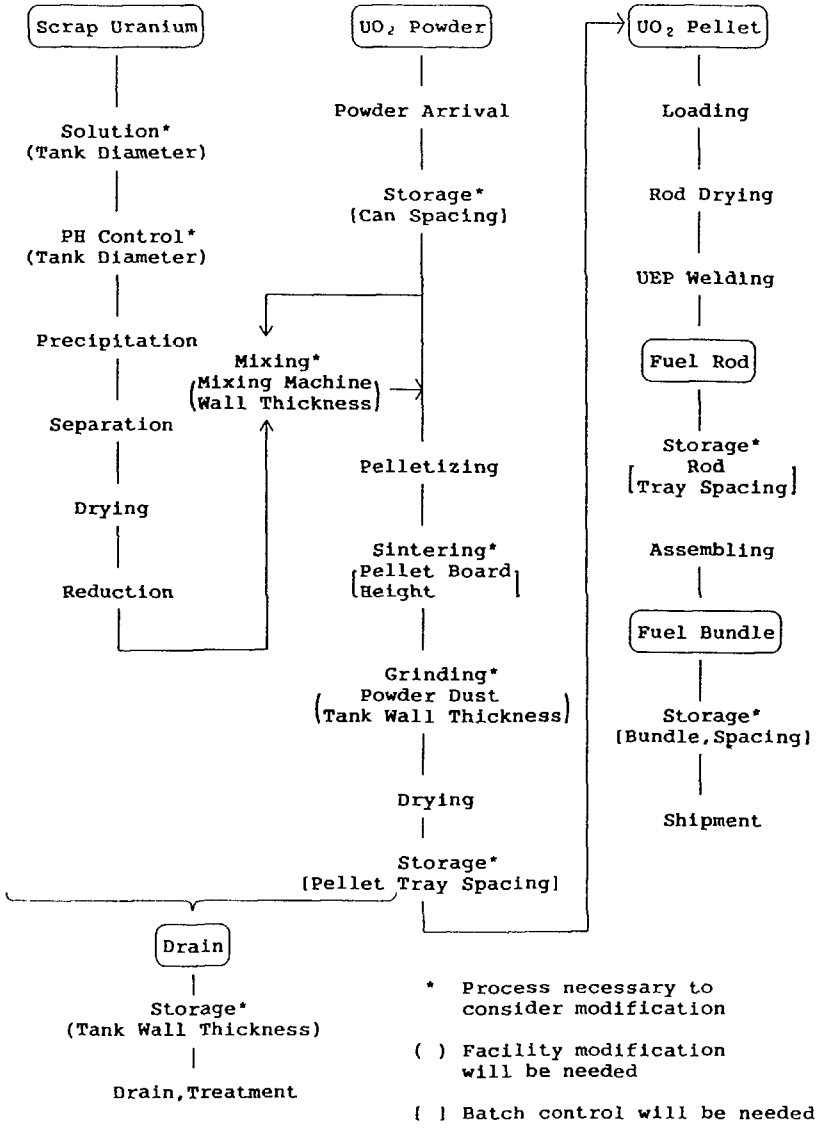


Fig.3-1-1 Fuel fablication process and facilities necessary to consider modification

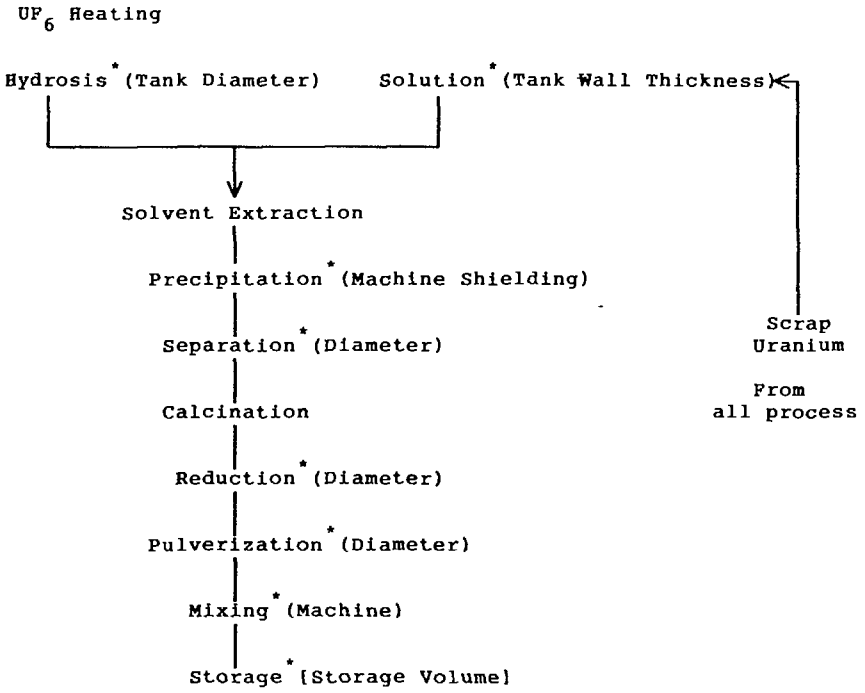


Fig3-1-2 Conversion Process and Facilities necessary to consider modification.

As for ventilation, drain and solid waste control, the impact of increased enrichment also will be small, so the major modification of the facilities will not be needed.

(2)Uranium and fresh fuel transportation

According to IAEA Safety Standards published in 1985, the package which contains a few Kg of uranium with over 5% enrichment will be categorized into B-Type. So the current package for fresh fuel transportation will not be applicable.

The capacity of the current package for UO_2 powder will be limited to too small value to transport efficiently.

As for UF_6 transportation, current cylinder will not be applicable by the regulation of DOE,ORO 651 Rev6. Therefore cylinder will be needed.

4. Impact On Back End

According to burnup extension, the increase of accumulated activities, decayheat generation and number of neutron generation ocurr. The calculated examples of these effects are shown in figure 4-1∼3 as a function of cooling time. The calculated example of FP generation is shown in figure 4-4.

4.1 Burnup Extension Within 45 GWd/t

(1)Spent fuel transportation

As for the criticality, the current casks will be applicable with modification of basket in the cask. As for the heat removal,they will be applicable with extended cooling time, almost 6 month. From the view point of shielding, as for the gamma-ray they will be applicable with extended cooling time. As for the neutron, it will be required to optimize the number of assemblies for a cask and shielding structure.

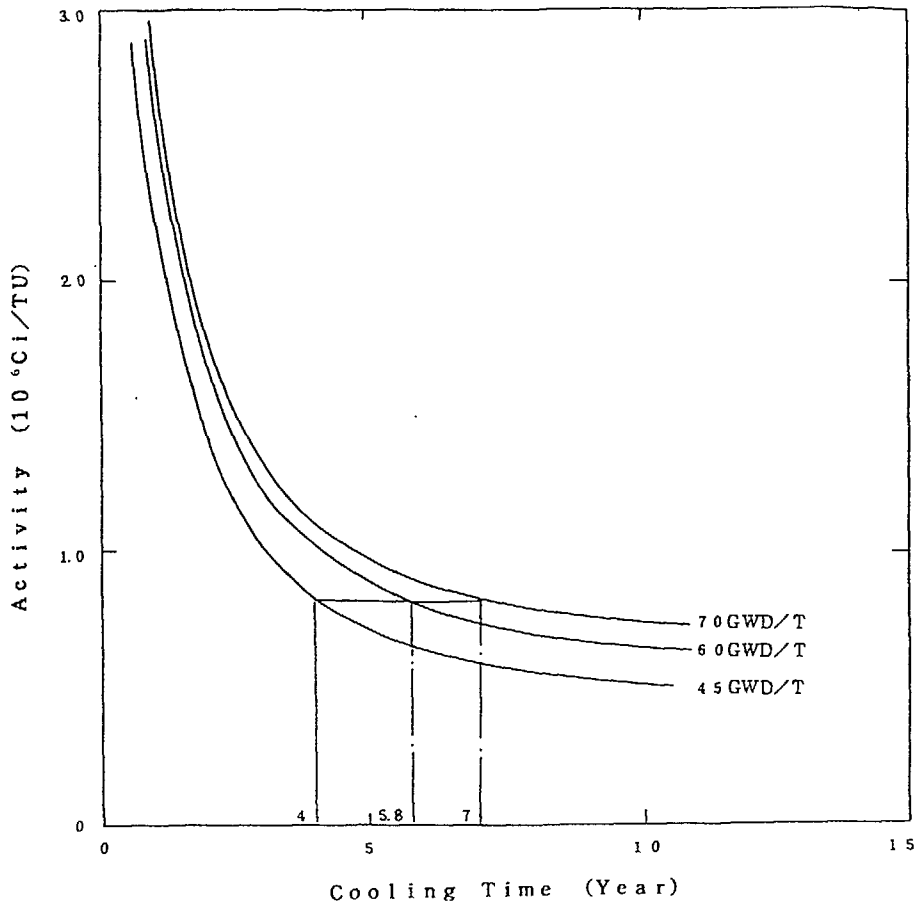


Fig. 4-1 Activity vs Cooling Time

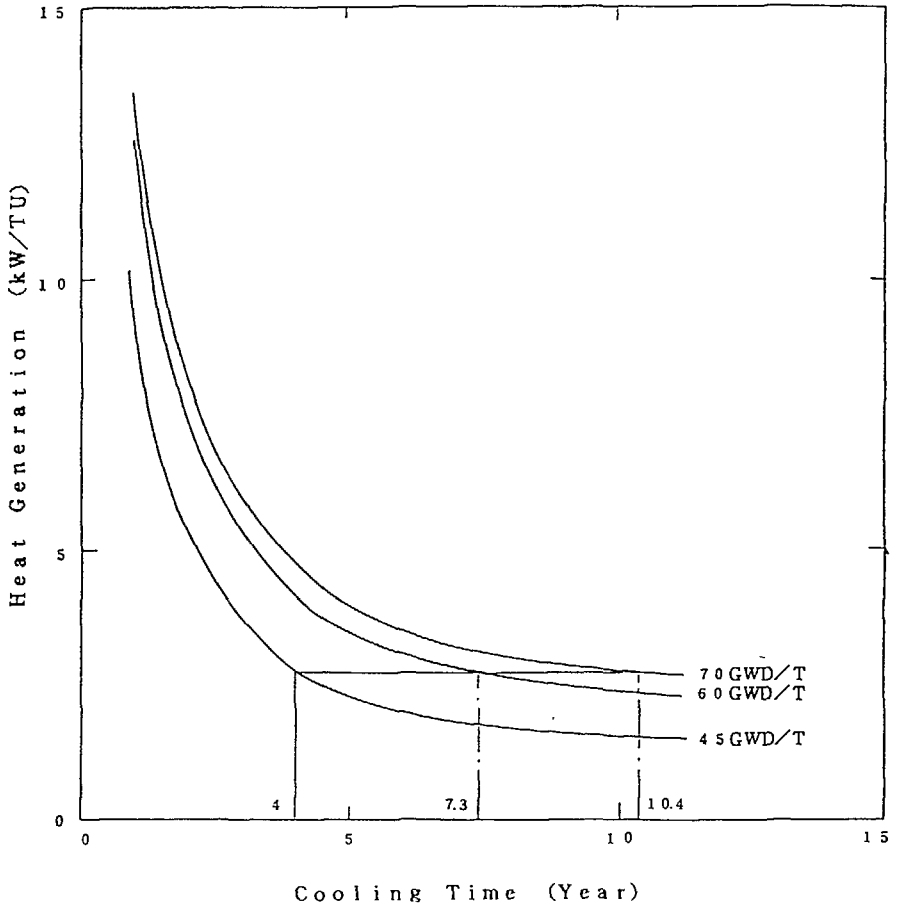


Fig. 4-2 Heat Generation vs Cooling Time

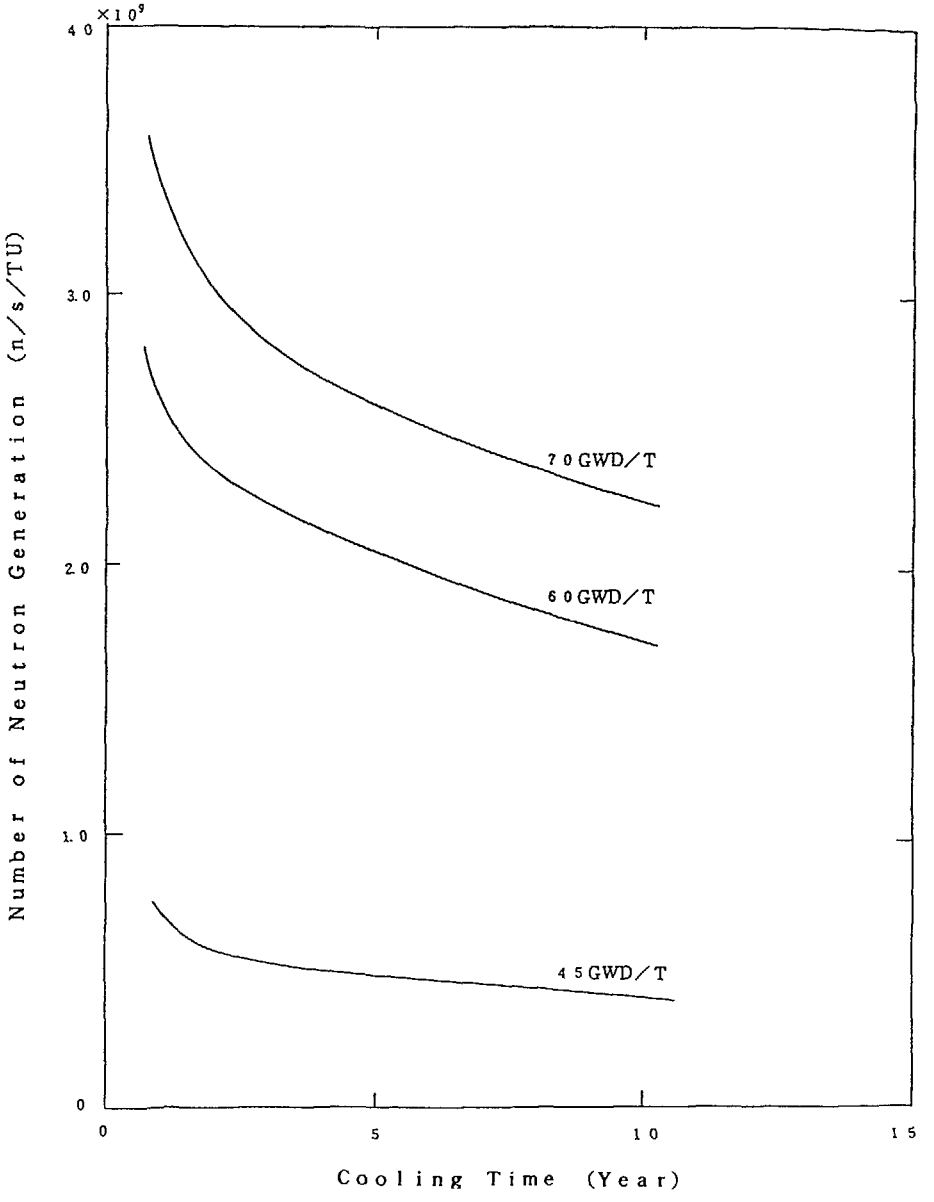


Fig. 4-3 Number of Neutron Generation vs Cooling Time

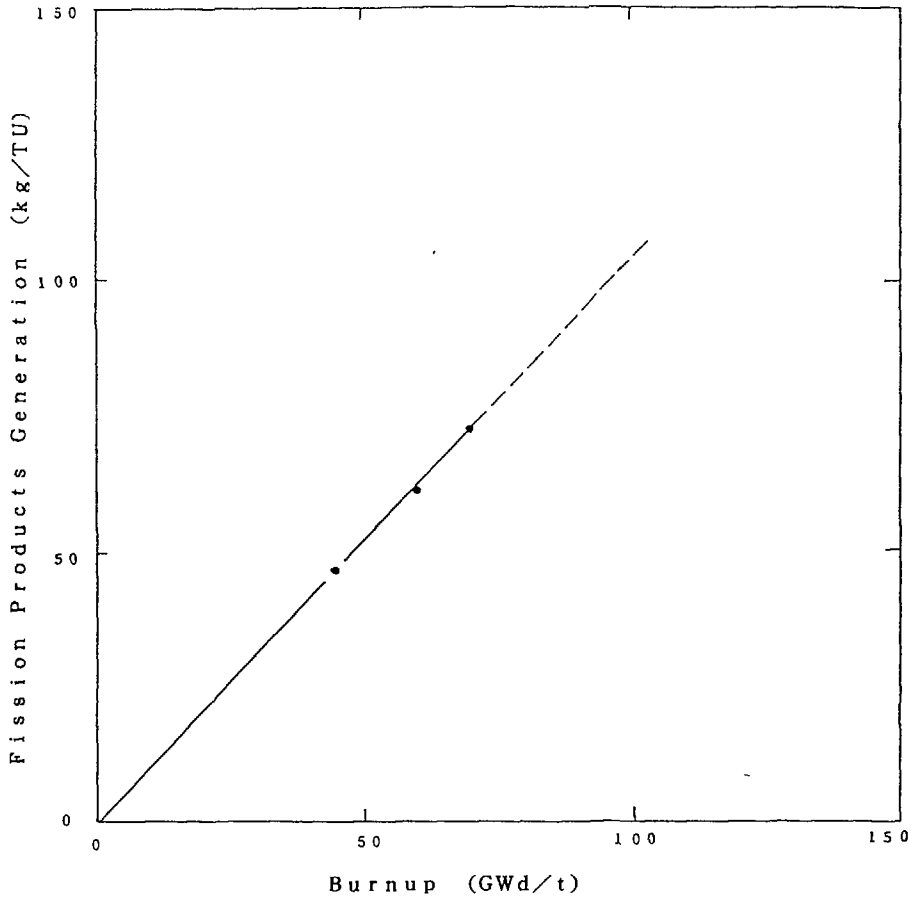


Fig. 4-4 Fission Products Generation vs Burnup

(2)Reprocessing facilities

The JNFS reprocessing facility is designed to be able to reprocess burnup within 45 GWd/t. Table 4-1 shows the basic specification of the fuel which JNFS can accept.

4.2 Burnup Extention Beyond 45 GWd/t

(1)Spent fuel transportation

From the view point of criticality, heat removal and shielding, it is supposed that the major modifications of cask design and longer cooling time are needed.

Table 4-1 Specification of JNFS Acceptable Spent Fuel and Extended Burnup Fuel

	JNFS Acceptable Spent Fuel	Extended Burnup Fuel
Bundle Average Initial Enrichment before Irradiation (w/o)	5	< 4.5
Maximum Allowable Bundle Average Enrichment of Spent Fuel (w/o)	3.5	—
Average Burnup par Day (GWd/t)	45	< 45
Bundle Maximum Burnup (GWd/t)	55	55(60)*
Cooling Time before Shearing (Year)	4	—

*( ) Design Burnup

(2)Reprocessing facilities

Figure 4-5 shows the conceptual illustration of reprocessing process flow and anticipated problems with extended burnup. As shown in this figure, these problems described below should be considered in connection with extended burnup.

-The thickness of shielding wall must be increased in proportion to the increase of neutron generation.

-The increase of filtering ability capacity of the facilities will be needed to cope with the increase of nuclides which need release control to the environment.

-Prevention of blockage of liquid flow due to sludge deposition by increase of clearing capacity will be needed.

-Shortening of extraction time or the increase of frequencies of solvent regeneration will be needed, because of activity increase in the solvent resulting from FP.

-The increase of disposal capacity of radioactive waste disposal facilities and storage capacity of solidified high level waste will be needed as the consequence of increased FP generation.

While, the items listed below should be considered in connection with modifications of fuel design : additive contained pellet, new cladding materials and new bundle configurations.

-Impact of pellet material on solubility and insoluble residue.

-Possibility of increased cutting powder of cladding material and deformation of cutting face

-Applicability of cutting machine to bundle configuration.

Fig. 4-5 Flowsheet for Spent Fuel Reprocessing

show predicted problems for high burnup spent fuel reprocessing

HIGH BURNUPS FOR WATER REACTOR FUELS - A UK PERSPECTIVE

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Abstract

The present position with respect to the construction of nuclear power plants in the UK and prospects for the future is summarised. The paper reviews the effect that adopting high discharge burnup cycles has on the fuel manufacturing process, the core design, the performance of the fuel and on the reprocessing of discharged fuel. It is concluded that there are no fundamental difficulties which would preclude high burnup cycles in the UK's PWRs, although a number of detailed design and licensing issues would need to be addressed. Some suggestions are made regarding the content of the IAEA report on the impact of high burnup on the nuclear fuel cycle.

1. INTRODUCTION

The electricity supply industry in the UK has just undergone a major upheaval following the Government's implementation of its plans to remove both the supply and distribution sectors from state ownership Formerly, the Central Electricity Generating Board (CEGB) was responsible for the supply and distribution in England and Wales, while the South of Scotland Electricity Board (SSEB) and the North Scotland Hydro Board were responsible for the supply and distribution in Scotland Following privatisation CEGB was split up into three major suppliers, National Power, Powergen and Nuclear Electric and a separate company responsible for distribution The SSEB was split into two companies, Scottish Power and Scottish Nuclear, while the North Scotland Hydro Board was renamed Scottish Hydro Apart from Nuclear Electric and Scottish Nuclear, all these new companies were privatised The recent legislative changes have also made it possible for independent generating companies to be formed to provide competition to the main generators and already a number of privately funded generating stations are being built

Such major changes to the electricity industry have naturally had a major impact on the UK nuclear industry The Government s original plans were to privatise the nuclear components of CEGB and SSEB along with the fossil fuel components, but these plans were changed at a late stage , the final decision being to separate off the nuclear components into separate companies, Nuclear Electric and Scottish Nuclear, and to retain them under government ownership Prior to the time at which the decision was made to retain the nuclear generating stations under government control, the Government s intention was to construct four Pressurised Water Reactors (PWRs) of a Westinghouse design three in England and one in Wales by about the end of the century At the same time as it decided not to privatise the nuclear generation, the Government also decided to postpone a decision on whether to proceed with three of the planned PWRs until after a major review of the nuclear industry to be conducted in 1994 However, the Government confirmed that the first PWR in the series, Sizewell 'B', which was already under construction would proceed to completion

Thus we have the current position in which only a single nuclear power station is under construction in the UK, at Sizewell 'B' and plans for other stations are on hold pending a major review Following the review, it will be decided whether Nuclear Electric should proceed with duplicates of Sizewell 'B , opt for alternative PWR designs or not proceed with any new nuclear plants An encouraging recent development is that British Nuclear Fuels is actively exploring the possibility of constructing one or two PWRs at its Chapelcross and Sellafield sites The first phase of this study has recently been completed The study has indicated that it would be technically feasible to build at least one nuclear power plant of up to 1500 MW(e) at Chapelcross If the project were to go ahead, construction would be on land owned by British Nuclear Fuels adjacent to the existing Magnox station

It is a safe assumption that any future programme of thermal nuclear power plant construction in the medium term in the UK will be based on PWR designs, which explains the UK's interest in Light Water Reactors (LWRs) The Goverment's decision to suspend the construction of the follow-on plants to Sizewell 'B' was a result of many factors The principal reason was due to difficulties in raising private capital to fund nuclear power plants, given the radical changes to the industry This in turn was cited to be due to uncertainties over whether a secure market could be found for the output of the nuclear stations within the new electricity supply structure, but another important factor was cited to be the economics of nuclear generation

Partly because of the restructuring of the industry, the cost of nuclear generation has recently become the principal factor which has figured in the public debate over the future of nuclear power in the UK The future of the UK nuclear industry will therefore depend to an important extent on being able to demonstrate it to be economically competitive Although the bulk of the cost of a nuclear plant over its lifetime is associated with recovering the construction costs, the fuel cycle costs make a significant contribution to the total (≈ 10%) One means of improving the fuel cycle economics is to increase fuel discharge burnups Another is to utilise Mixed Oxide (MOX) and recycled uranium fuels The UK is actively pursuing all these options Higher discharge burnups can therefore contribute to the overall case for a continued nuclear programme in the UK

This presentation reviews the current position with regard to the PWR fuel cycle in the UK and identifies the technical issues concerning high burnups which are considered most significant The scope of the review covers the impact on the front end of the fuel cycle, the in-reactor fuel management and performance, as well as the impact on fuel reprocessing

2 CURRENT STATUS OF PWR FUEL CYCLE IN UK

Construction of Sizewell B is currently in progress and on schedule for completion by 1994 The plant is based on the Westinghouse SNUPPS design[1], and is similar to the

Wolf Creek and Callaway plants in the USA British Nuclear Fuels are supplying the initial charge fuel and have been responsible for producing the nuclear design report and carrying out the associated safety related calculations for the first operating cycle The initial core loading pattern is virtually identical with those used in the first cycles of Wolf Creek and Callaway, with enrichments of 2 1, 2 6 and 3 1 w/o for the three fuel regions Since the first reload is not due until 1996, or thereabouts, the form of the reload fuel management has not yet been finalised by Nuclear Electric Nevertheless, comprehensive calculations of the performance and safety characteristics of nominal reload fuel cycles have been carried out to satisfy the safety authorities that the core characteristics in reload cycles will be acceptable

These nominal reload studies, carried out by British Nuclear Fuels, have been based on a very conservative 12 month, three batch fuel cycle, which can be achieved with a feed enrichment of 3 1 w/o The discharge burnup depends on the load factor assumed and is in the region of 33 GWd/t Up to the present time, relatively little priority has been given to establishing more advanced fuel cycles , the main priority has been to construct the plant to time and to cost and to obtain good operating performance This conservative approach is likely to be continued in the first reload cycles

In the longer term, however, it is likely that the priorities will change and that there will be more emphasis on more economic fuel cycles, in both Sizewell 'B' and later PWRs, as has already been considered for BNFL's proposed PWRs British Nuclear Fuels have already reviewed extensively high burnup PWR fuel cycles and have concluded that a four batch 12 month cycle would give substantial benefits in terms of fuel cycle costs[2], as discussed in the next section

3. URANIUM FEED, ENRICHMENT AND OVERALL FUEL CYCLE COSTS OF 4 BATCH 12 MONTH CYCLES

A four batch 12 month fuel cycle would be well suited to the annual load demand cycle, which has the peak demand in the winter months and the minimum demand in the summer BNFL's study[2] has shown that a four batch 12 month fuel cycle in a 1100 MW(e) plant with a discharge burnup of about 44 GWd/t would reduce the feed uranium and conversion requirements by about 3% relative to a 3 batch 12 month cycle. With the feed enrichment increased from 3 1 w/o to 4 4 w/o, the enrichment costs for the four batch cycle would, however, be some 4% higher, so that the combined uranium feed, conversion and enrichment costs of the two fuel cycles would be about equal On the assumption that the fabrication and spent fuel management costs per unit weight of fuel are approximately the same for the two fuel cycles, the high burnup fuel cycle shows an advantage of approximately 13% in fuel cycle costs

There is thus a substantial benefit to be gained from increasing the equilibrium discharge burnup of the UK PWRs to $\approx$ 45 GWd/t Provided that the safety characteristics of high burnup cycles prove to be satisfactory when evaluated against the UKs requirements, this will be a major incentive for the adoption of such a fuel management scheme, or a similar one in the UK PWRs Preliminary studies for higher burnup fuel cycles confirm that there are continued fuel cycle cost savings at even higher discharge burnups but recognising the need for well planned gradual steps higher discharge burnup cycles are not likely to be implemented in UK PWRs immediately

An issue which has not yet been addressed in the UK is the economic impact of utilising MOX assemblies in high discharge burnup cycles Since the manufacturing cost of MOX assemblies is essentially independent of the initial fissile content, MOX fuel can potentially give economic savings when utilised in high discharge burnup cycles such as the above four batch 12 month fuel management scheme

4. IMPACT OF HIGH BURNUP FUEL CYCLES ON THE FRONT-END OF THE FUEL CYCLE

4.1 Impact on Fuel Fabrication

There are implications of high burnup fuel cycles on two aspects of PWR fuel fabrication

An important consideration is that of the criticality clearance for the fabrication plant British Nuclear Fuel's oxide fuel fabrication plant is cleared for the manufacture of up to 5 w/o enrichment, a figure which is typical of modern oxide fuel fabrication facilities This will be adequate to cope with fuel intended for burnups of up to at least 45 GWd/t, even allowing for split enrichment feeds With discharge burnups significantly higher than 45 GWd/t, careful attention to the criticality issue would have to be given If a requirement arises for fuels enriched beyond 5 w/o, an application would have to be made to the licensing authority This would involve at the very least re-calculation of the potential for criticality and may also require revised procedures to be implemented, in addition to significant cost penalties in plant equipment At present, it is not clear whether there are any practical considerations which will limit the highest enrichment levels manageable and what level of enrichment and discharge burnup such a practical limit would correspond to.

The use of high burnup fuel assemblies may also have implications for the manufacturing plant in that there may be a need to provide integral fuel burnable absorbers, such as Gd_2O_3 rods, in which case a separate manufacturing facility for the integral poison rods may be necessary This will certainly increase the unit costs of fuel fabrication, but probably not to such an extent as to invalidate the fuel cycle cost benefits of high burnup cycles

4.2 Impact on Fuel Management

The increased initial enrichments needed for high burnup fuel cycles have implications for the fuel management Increasing the initial enrichment introduces a larger reactivity differential between fresh and previously irradiated fuel, which tends to cause the radial power peaking factors to rise Depending on the radial peaking factor limits applicable to a particular plant, and on the fuel management scheme (ie whether out/in or low leakage), this may necessitate the use of burnable absorbers for radial flux/power peaking control

Since the reload fuel management scheme to be used in the UK PWRs has not yet been decided it is too soon to say whether burnable poisons will be needed with high burnup cycles If this proves to be the case gadolinia burnable poison may well be favoured as it allows separate control over the initial reactivity hold-down and the burn-out rate

Table 1

Impact of High Burnup Fuel Cycles on Core Characteristics

Parameter	Effect of High Burnup Cycle	Implications
Radial peaking factor $F_{\Delta H}$	Increases	Reduced margin to $F_{\Delta H}$ limits
Axial peaking factor	No significant effect	
Overall peaking F_Q	Increases	Reduced margin to F_Q limits
Moderator temperature coefficient	More negative	Reduced shutdown margin Increased feedback in heat-up faults Increased reactivity insertion in cooldown faults
Doppler and boron coefficients	No significant effect	
Control rod worths	Reduced	Reduced trip reactivity worths Reduced shutdown margins
Xenon reactivity worths	Reduced	Improved operating flexibility

Gadolinia poisons do require careful attention to ensure that within-assembly peaking and the effects of reduced thermal conductivity do not become limiting, but these effects can be minimised by careful selection of the UO_2 enrichment in the gadolinia rods With modern nuclear design codes, questions regarding the prediction of gadolinia behaviour should not be a major consideration

The higher enrichments used in high burnup fuel cycles also have a significant impact on the reactivity coefficients and control rod reactivity worths Increasing the enrichment tends to make the moderator temperature coefficients more negative With respect to beginning of cycle moderator coefficients this is a beneficial effect as the requirement to have a non-positive moderator coefficient at operating conditions can be met more easily, with fewer (if any) burnable poisons required for moderator coefficient control The more negative moderator temperature coefficients may have a detrimental effect on cooldown faults as the potential reactivity insertion is increased On the other hand, reactivity feedback in heat-up faults is increased, which is a benefit and tends to make the overall impact on safety case calculations roughly neutral Other reactivity coefficients, such as Doppler and boron coefficients, are affected also, but the impact on the core management is less significant than that of the moderator coefficient

The reduced control rod worths in high burnup cycles arises because of the increased thermal absorption in the high enrichment fuel Combined with the effect of the more negative moderator temperature coefficients the reduced control rod worths tend to reduce the shutdown margins This needs careful attention in the fuel management to ensure that the shutdown margins are not reduced to unacceptable levels

A comprehensive assessment of the impact on fuel management of high burnup cycles in the UK PWRs has not yet been carried out Preliminary studies reported in Ref 2 for a 4 batch 12 month cycle, however, have identified that the radial peaking factor aspect of fuel management is most likely to be the limiting consideration

Table 1 summarises the impact of high burnup fuel cycles on the principal safety related core parameters, identifying against each parameter how the parameter changes as the discharge burnup increases

4.3 Impact on Fuel Rod Design

For discharge burnups up to 45 GWd/t, which is the highest burnup level which British Nuclear Fuels envisage to be likely in the forseeable future in the UK PWRs, no new fuel performance issues are expected to arise and the fuel performance safety assessments are not expected to be affected provided that the existing core design limits are satisfied At discharge burnups extending up to 60 GWd/t, one fuel performance issue has been identified to be limiting, that of fuel cladding corrosion

The increased dwell time of the fuel in high burnup fuel cycles, in combination with the higher ratings to which the fuel will be subjected at high burnups, will increase the potential for clad corrosion This was identified as an issue by a recent IAEA study on fuel performance at high burnups[3] Several PWR fuel vendors are currently developing improved cladding materials, such as modified zirconium alloys and duplex cladding (in which the Zircaloy clad is coated with a corrosion resistant layer) British Nuclear Fuels are monitoring developments in this area closely, and may choose to adopt such an advanced cladding if a clear need for it is shown to exist

Other secondary fuel performance issues, which arise from the more severe rating histories that fuel is subjected to in high burnup cycles, rather than from the performance of the fuel itself, are those of fission gas release and pellet clad interaction

Although British Nuclear Fuels' Integrated Dry Route (IDR) fuel shows excellent fission product retention properties, the longer exposure time of the fuel in conjunction with increased ratings at high burnups increases the potential for fission gas release British Nuclear Fuels, in common with other fuel vendors, are assessing this potential problem and methods of reducing fission gas release

Finally, there is the issue of pellet clad interaction There is a clear need to demonstrate that the failure threshold for this mechanism does not become significantly more

restrictive at high burnups and that in combination with higher local ratings in high burnup fuel cycles, does not result in excessive numbers of fuel failures in postulated reactor faults

5. IMPACT OF HIGH BURNUP FUEL CYCLES ON THE BACK-END OF THE FUEL CYCLE

It is premature as yet to decide on whether or not fuel from the UK PWRs will be reprocessed Nevertheless, the option to reprocess will be available in the UK in British Nuclear Fuels Thermal Oxide Reprocessing Plant (THORP) THORP is currently being licensed for operation with LWR fuels of up to 40 GWd/t discharge burnup Following from a preliminary assessment of the safety implications of reprocessing high burnup fuel (up to 60 GWd/t) in THORP, it has been concluded that with suitable changes in operating procedures, a satisfactory safety case can be made The following discussion highlights the most important areas in the reprocessing plant which are affected by high burnup fuels

Mechanical handling and feed pond operations at the mechanical head end are unaffected, as are the processing rates and cycle times for the mechanical operations in the basket handling cave Dose uptakes, shielding, cooling and criticality considerations in these areas are, however, affected and would need to be assessed

Dissolution times in the chemical head end plant are unaffected, though there is an increase in the amount of insoluble fission products in high burnup fuels, which will affect the amount of feed liquor volume processed per batch in the centrifuges This will not significantly affect the on-line cycle time, however, because the off-line time for removal of insolubles is relatively short compared to the on-line process time Preliminary studies indicate that a satisfactory safety case for the chemical head end plant could be made with high burnup fuels

Constraints related to maximum Pu capacities would lead to a reduction in the maximum throughput in the chemical separation plant, due to the higher Pu content of high burnup fuels A further consideration is a constraint related to maximum activity level of feed liquor These considerations may make it necessary to perform blending operations to ensure that the liquor derived from the high burnup material is diluted

Overall, the processing of high burnup fuel will give rise to no new effluent streams, and the arisings of solid, liquid and gaseous effluents will be similar to those of medium burnup fuels although the volumes and concentrations may differ somewhat

Thus there is no reason why high burnup fuels should not be reprocessed in THORP with suitable modifications to operating procedures BNFL is currently preparing a full safety justification for reprocessing high burnup fuels in THORP and no additional input in this area is seen to be necessary at present

6. RECOMMENDATIONS FOR THE IAEA HIGH BURNUP STUDY

This presentation has summarised the current status of high burnup LWR fuel cycle work in the UK Following from the points made in this report, the following recommendations are made regarding the contents of the IAEA report on the impact of high burnups on LWR fuel cycles

1 The IAEA report should attempt to determine whether there is any practical limit on enrichment due to criticality considerations in manufacturing plants If so, the report should identify approximately at what burnup levels any such limitation would be effective The report should identify any special measures which may be necessary to prevent criticality in manufacturing plants over and above those already required at present burnup levels

2 The IAEA report should include a survey of the effect of high burnup cycles on the core neutronics characteristics and identify the factors which are most likely to be limiting from the perspective of fuel management and operational flexibility An assessment of the economic impact of the use of MOX fuels in high discharge burnup cycles would be valuable The report should include a concise discussion of the relative merits of short (≈ 12 month) and long (≈ 18 to 24 month) fuel cycles, as an aid to utility managers in selecting fuel management schemes

3 The IAEA report should incorporate a review of the fuel performance implications of high burnup cycles based on the conclusions of the IAEA study on fuel performance at high burnups reported in Ref 3

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EFFECTS OF EXTENDED BURNUP ON THE NUCLEAR FUEL CYCLE

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Abstract

Burnup affects to varying degrees the front end of the fuel cycle, the in-reactor part of the fuel cycle, and its back end. These effects are reviewed and discussed in this paper from the US perspective of PWRs and BWRs operating with storage and disposal of spent fuel. The front end effects include reduced requirements for uranium and fuel fabrication and licensing of the front-end operation for the higher enrichments needed for higher burnups. The principal in-reactor issues are (1) the fuel design and performance impacts of the longer in-core residence times on corrosion and hydriding, fission gas release, dimensional and structural changes, pellet-cladd interaction, and ultimately the resulting effect on fuel reliability, and (2) the effects on fuel management of greater potential differences in reactivity between adjacent assemblies and the potential for reduced control-rod worth and changes in some reactivity coefficients. The principal effects on the back end are the reduced volume of spent fuel to be handled and the higher decay heat generation of this spent fuel. Finally, there is a need to be able to take credit for burnup for licensing spent fuel storage and shipping.

1. Introduction

The discussion of the effects of the extension of burnup of light water reactor fuel on the nuclear fuel cycle presented here is divided into three parts: effects on the front end of the cycle, in-reactor effects, and effects on the back end of the cycle. The back end discussion applies to the once-through fuel cycle, as this is by far the more economic option and the option which is being selected in most countries employing water reactors. A large part of this discussion is taken from relevant parts of the WREBUS report being published by the IAEA (1). These sections of WREBUS were originally written by the present author; hence, no useful purpose would be served by attempting to write a new document on the same subjects. In incorporating these sections from WREBUS, editing, summarizing, and supplementing with additional information has been done, as necessary to adapt these sections to the organization and purposes of this paper.

Burnup may be extended in different ways, as follows: (1) at constant cycle length with accompanying reduction in the fraction of the core reloaded, (2) while extending cycle length proportional to burnup and keeping the fraction of the core reloaded unchanged, or (3) while simultaneously changing cycle length and reload fraction. As a first approximation, most European utilities extend burnup at constant cycle length while most U.S. utilities extend both burnup and cycle length with the expectation that in the longer term (for "equilibrium cycles") reload fractions will not change much. It is important to recognize that some of the effects of extending burnups depend on which of these approaches is used. When such differences occur, the conditions to which effects apply need to be clearly stated.

2. Front-End Effects

The front-end effects can be divided into two parts: effects on front-end resource requirements and the effects of the higher enrichments needed for higher burnups on fuel fabrication plants and fresh fuel storage facilities.

2.1 Effect on Resource Utilization

In Reference (2), three measures of resource utilization were discussed, each of which is defined in terms of the amount of energy produced per unit of the resource consumed. Historically, the most important and most widely used measure has been uranium utilization. Other significant measures of fuel utilization are the separative work unit (SWU) utilization and the fuel assembly utilization. The SWU utilization characterizes the amount of separative work needed per unit of energy produced. The fuel assembly utilization is a measure of both fuel fabrication and, more importantly, spent fuel generation per unit of energy produced. For a given fuel design, it also is a measure of the amount of metal (e.g. Zircaloy) needed for the fuel fabrication process. Because of difficulties and high costs associated with the back-end of the fuel cycle in many countries, the fuel assembly utilization as a measure of spent fuel generation has gained strongly in importance and emphasis in recent years. A fourth utilization may be similarly defined for conversion; since all uranium is converted to hexafluoride prior to enrichment, the conversion utilization varies in exactly the same manner as the uranium utilization.

2.1.1 Effect on Uranium Utilization

The effect of extending burnup on the specific uranium consumption is shown for several reactor/cycle-length combinations in Fig. 3.4.7.1. of the WREBUS study (1). The specific uranium consumption is expressed in grams of natural uranium used per MW-hour of electric energy generated and may be used as an indicator of uranium utilization. The results of Fig. 3.4.7.1. show that specific uranium consumption goes down (i.e., that uranium utilization improves) as burnup increases, almost without exception.

Comparison of 12- and 18-month cycles for the same reactor at the same burnup shows that in the longer cycles, uranium utilization is degraded by typically 10-15 percent. A similar comparison of 18- and 24-month cycles for the U.S. BWR shows that this further lengthening of the cycle adversely affects uranium utilization by between 6 and 9 percent.

The less uranium-efficient performance of longer cycles arises from the poorer neutron economy inherent in the design of longer cycles, which results from the need to load more excess reactivity at beginning of cycle and hold it down by control absorbers. Burnup extension typically has the potential to save up to about 15 percent in specific uranium consumption when implemented with

unchanged cycle length. For reactors operating at the historical burnups in the neighborhood of 30 MWd/kgU, approximately half of this uranium saving can be realized if burnup and cycle length are simultaneously increased, to the higher values of burnup shown in the figure and from 12 to 18 months.

2.1.2 Effect on SWU Utilization

The effect of increasing burnup on the consumption of SWUs per unit of energy generated is generally small, and the direction of the change can be either increasing or decreasing. In most studies which cover a wide enough range (3), the specific SWU consumption at first decreases, reaches a minimum at intermediate levels of burnup extension, and then increases again as burnup is extended to the highest values. The greatest difference found between an extreme value and the minimum value is only a few percent. Because of the relatively small magnitude of most of these changes, the lack of uniform and consistent trends in their direction and magnitude, and the relatively lesser interest in SWU utilization generally, it is concluded that the effect of extending burnup on SWU utilization is relatively small and unimportant.

2.1.3 Effect on Fuel Assembly Utilization

For a fixed fuel assembly design containing a fixed weight of enriched uranium, burnup itself is a direct measure of fuel assembly utilization, as it represents the energy produced per unit of enriched uranium loaded. To calculate a specific fuel assembly consumption analogous to the specific uranium and SWU consumption discussed in the preceding sections, burnup would need to be multiplied by the weight of enriched uranium per assembly and by the thermal efficiency of the plant, in order to obtain electrical energy per assembly, and the reciprocal of this product would represent assemblies per unit of electric energy. This calculation is hardly worth doing, since burnup is the only variable and therefore is a direct measure of fuel assembly utilization.

The specific assembly consumption is proportional to the reciprocal of the burnup and therefore always decreases as burnup increases. For small to moderate changes, a given percentage increase in burnup gives virtually the same percentage increase in fuel assembly utilization. This explains the significance of burnup extension in reducing the amount of spent fuel which must be disposed of and in reducing assembly-specific costs, both fuel fabrication and back-end.

2.2 Effects of Higher Enrichments

For some fuel suppliers, processing higher enrichments has required or may require amendment of their fuel fabrication license. While this may require time and documentation, no difficulties are foreseen for the moderate increases in enrichment presently contemplated. The current license limits on enrichment for the five U.S. fuel fabricators are:

ABB-Combustion Engineering	5.0%
Babcock & Wilcox	4.1%, being raised to 5.1%
General Electric	6.0%
Siemens Nuclear Power	5.0%
Westinghouse	5.0%

Fresh fuel storage has been studied for PWR fuel of up to five percent enrichment, and the results show that no criticality concerns exists for typical storage rack designs.

3. In-Reactor Effects of Extended Burnup

3.1 Fuel Performance Considerations

The principal fuel design and performance technical issues in extending burnup are fission-gas release and internal fuel-rod pressure, cladding corrosion and hydriding, fuel dimensional and structural changes (rod growth, assembly growth, rod bowing, clad creepdown or expansion, grid-spring relaxation, etc.), and fuel integrity; i.e., resistance to pellet-clad interaction (PCI) and other failure mechanisms. Most of these issues can be resolved through straightforward design changes such as providing more space in the assembly for fuel-rod growth, more space in the fuel rod for fission-gas release, or lowering the prepressurization pressure. Of potentially greater concern is waterside corrosion; this is the most likely fuel life limiting phenomenon for present zircaloy cladding metallurgy, for high enough burnup extension and/or high enough coolant temperature.

Advanced zirconium-based alloys with better corrosion resistance, in order to facilitate burnups higher than achievable with present alloys, are now becoming commercially available. Fuel failure through PCI is not a significant concern at extended burnup, both because advanced designs that enhance PCI margins are available and because at higher burnups the local reactivity is likely to be insufficient to produce power ramps that result in risk of PCI. The PCI concern is more validly directed toward the performance at lower burnups of higher enrichment fuel designed to achieve extended burnup; this performance must still be maintained free of PCI failures through PCI-margin enhancing design improvements, lower linear heat rates, or control of fuel local power and power increases.

A traditional concern when extending burnups has been the potential effect on fuel reliability. There exists an instinctive feeling that operating the fuel longer is imposing a more severe duty on it, which should lead to increased fuel failures. This is, in fact, not borne out by experience. Reliability of fuel (and of many other products) may be best understood by considering the traditional bathtub-shaped curve of failures vs. usage or time in service characterizing the occurrence of failures of any component or system. The initial, sharply decreasing part of this curve represents the preponderance of manufacturing and quality assurance related defects early in life. This is followed in the intermediate range by a low, horizontal line representing the effect of a small number of random and often external causes of failure. Finally, late in life, wear-out effects predominate and the failure curve begins to rise sharply. With regard to nuclear fuel, its technology must be well understood and its design conservative and well-tested, if it is to be licensed on a substantial scale. This means that its operation is always restricted to well below the point at which wear-out effects begin to take place. Hence, reliability has not in practice been found to decrease with increasing burnup. In fact, for fuel with high rates of initial failures, reliability may well improve as burnup is increased, since one of the effects of burnup increase is a reduction in the amount of new fuel loaded each year.

To establish utility confidence and regulatory acceptability for extended burnup fuel design and performance, test assembly irradiations to the desired maximum assembly burnups contemplated are helpful prior to full implementation of extended burnups by utilities. This includes taking existing designs to higher burnups and testing new, advanced designs through the process of design, fabrication, test-assembly licensing, irradiation to high levels, and post-irradiation examination. Many such projects have been successfully accomplished for the extended burnup levels now considered acceptable (about 50 MWd/kgU for PWRs, 45 MWd/kgU for BWRs), but new projects would facilitate extension of burnups beyond these levels.

Along the same lines, data on the several aspects of fuel performance at burnups higher than the established range of present analytical methods and computer codes may need to be tested against these methods and codes. After any necessary modifications, this will establish the validity of the methods over the newly extended burnup range of interest.

Burnup extension may require some dimensional or materials changes to the fuel designs, and such changes, as well as the costs of extending warranties to higher burnups, recovering applicable research/development/testing/engineering costs, and in some cases increases in the amortization of capital costs over smaller volumes of fuel fabrication may be expected to increase unit fabrication costs. Estimates of these cost increases have been made and the conclusion seems to be that such increases have been and are expected to be quite moderate, no more than about 10-15 percent.

3.2 Fuel Management Issues

Extending burnup requires increasing the enrichment level of fresh fuel and results in lower enrichments and reactivities of discharged fuel. Physical consequences of this include:

- larger potential differences in reactivity between neighboring assemblies in the core, resulting in greater difficulty in maintaining power distributions within prescribed limits, and

- hardening of the neutron spectrum, which reduces control-rod worth and shutdown margin and may adversely affect some reactivity coefficients.

Because of these changes, the constraints in working out acceptable core management patterns for higher levels of burnup become more severe; this may also affect some safety analyses and core maneuverability. Experienced fuel management personnel have, however, generally been able to develop acceptable fuel loading patterns for substantial increases in burnup.

Burnable absorbers are not needed for burnup extension by itself. When longer cycles and/or low leakage fuel management are used in conjunction with burnup extension, burnable absorbers become necessary for acceptable fuel cycle designs; specifically, to hold down excess reactivity at beginning of cycle, to control power distributions more precisely, to reduce end-of-cycle absorber residuals, and to maintain reactivity coefficients within desired ranges. The use of improved integral burnable absorbers may be particularly important for substantial cycle length extension of PWRs, as there may be no acceptable alternative ways to control the large excess reactivity needed at the beginning of cycles and because reducing end-of-cycle absorber residuals may be important. Low leakage fuel management becomes increasingly important as burnup (and reload fuel enrichment) are increased, since placing higher enrichment reload fuel at the periphery of the core can unacceptably increase neutron leakage and produce excessive fast neutron fluence on the reactor vessel. Another fuel management concern is the design of the transition from lower burnups and out-in fuel management to extended burnup with low leakage fuel management, either with the same number of core regions if cycles are simultaneously lengthened, or with increased number of regions if cycle length is unchanged. Such transitions can be difficult to design if it is desired to achieve the new conditions rapidly.

3.3 Licensing Aspects

A number of safety, licensing, and environmental assessments of burnup extension have been conducted in the United States, and as a result of these efforts fuel with progressively increasing levels of burnup has been licensed over the last few years. The overall conclusions of the various assessments are:

- No qualitatively different phenomena are expected at higher burnups;
- No substantial safety issues exist;
- The environmental effects of burnup extension are mildly positive; and
- Substantial licensing-oriented work may be needed whenever burnup is extended to new, higher levels--to obtain data to support licensing and to show that these data either justify the extended burnup fuel operating within previously established licensing limits, or to provide technical justification for any required extension of these limits.

Most of this previous work has applied to burnup extension up to 50 MWd/kg discharge batch average for PWRs and 45 MWd/kgU for BWRs, but there is no reason to believe that the principal conclusions would be substantially different for further, moderate extension of burnup, although, of course, this remains to be proven.

4. Back-End Effects

The most important effect of extending burnup on spent fuel storage, shipment and disposal, arises from the lesser quantities of spent fuel generated at higher burnups. As indicated in Section 2.1.3, for moderate percentage increases in burnup, the amount of spent fuel is reduced by the same percentage. Thus, there is less spent fuel to be stored, shipped, and disposed of for higher burnups. Existing storage capacities last longer before becoming full, fewer spent fuel shipments need to be made, and any given repository tonnage capacity will accommodate spent fuel which has produced proportionately more energy. The incremental or variable costs of these operations are approximately proportional to the amount of spent fuel; hence, burnup extension yields substantial savings in the costs of the various back-end operations.

Spent fuel of higher burnups has higher decay heat generation rates than spent fuel of lower burnups, when compared at equal cooling times after discharge from the core. This is of importance because some of the proposed repository designs are heat-load limited. Spent fuel shipment and disposal in the U.S. is expected to take place on a complex schedule reflecting the relative availability of spent fuel storage space in various utilities' spent fuel

pools. As a first approximation, this will probably mean that for a considerable time after shipments from plant sites begin, these shipments will be made only from plants with virtually full pools. By extending burnup, a utility decreases its annual rate of discharge of spent fuel and thereby extends its period before pool fill, proportionately to its extension of burnup. A more valid and practical basis for comparison of decay heat generation at different burnups for spent fuel shipping and disposal is therefore at cooling times proportional to burnup. When compared in this way, spent fuel at extended burnup still exhibits higher decay heat rates, but by much less than when compared at the same cooling time. Several alternatives for taking care of this remaining extra heat generation (in those cases where it may actually be limiting) exist; these range from blending with some of the old and cold fuel at the site to simply providing longer cooling times for the highest burnup assemblies.

In addition to these effects of quantity of spent fuel and its decay heat generation rate, spent fuel storage and shipping can present criticality concerns under the overconservative licensing assumption that all fuel be considered fresh; i.e., without credit for reduced reactivity resulting from burnup. Such credit may be taken if measurements verifying the burnup are made, or it sufficiently stringent administrative controls are provided to satisfy the licensing authorities that credit for burnup should be allowed. In the United States, a burnup meter has been developed and successfully testd for verifying burnup of spent fuel assemblies. The use of a burnup meter with spent fuel of higher initial enrichment should resolve the criticality issue for spent fuel storage (and, if needed, for spent fuel shipment.) For spent fuel shipment, the limiting factors to be considered for each fuel design and shipping cask combination are heat rejection capability, shielding, fission-gas release, and criticality. The results can be very complex since so many combinations exist and since the same consideration is not always limiting. At the risk of oversimplification, it may be stated that for U.S. cask designs almost all fuel assemblies to burnups of 55 MWd/kgU (PWR) and 50 MWd/kgU (BWR) can be shipped in almost all of the existing casks, if one or more of the following are acceptable in some instances: 1) credit for burnup is taken in the criticality analyses, 2) somewhat longer cooling times are acceptable prior to shipping, 3) partial cask loadings are used in lieu of longer cooling times when these are unacceptable, and 4) minor design modifications are made to some of the casks. Some relicensing of the casks for the higher burnup fuel may be needed. For burnups beyond those cited, it should be expected that more of these measures must be taken if existing casks are used. However, the very high burnup assemblies under consideration for the future will not actually be discharged from reactors and need to be shipped for many years, and in that time frame casks of new design anticipating the much higher burnup requirements should become available, obviating the need for excessive cooling times or partial cask loadings.

5. Conclusions

Burnup extension reduces uranium consumption--but more so at lower burnups than at the highest values. Although cycle-length extension alone increases uranium consumption, simultaneous extension of burnup and cycle length can achieve about half of the uranium saving possible through equivalent burnup extension at constant cycle length. Other front-end effects are similarly favorable or neutral.

No major technical problems or issues are known which would prevent moderate extension of burnup for the majority of existing plants. For some PWRs operating at the highest coolant outlet temperatures and using zircaloy cladding, corrosion may limit the permissible degree of burnup extension.

No substantial safety or environmental issues are known to exist for burnup extension, although substantial work may be needed to support licensing to higher burnups. The net environmental effects of burnup extension are mildly positive.

The major benefits of burnup extension arise from the reduced number of fuel assemblies needed to generate a given amount of energy. This reduces the amount of fuel to be fabricated and the amount of spent fuel to be stored, shipped, and disposal of.

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IMPROVEMENT OF VVER FUEL BURNUP AND ITS EFFECT ON THE FUEL CYCLE

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Abstract

The paper discusses the effects of burnup improvement on the VVER fuel cycle. The experience gained in operating the cores of VVER-440 and VVER-1000 reactors with the focus on safety, reliability, and economy issues raised in transferring to an enhanced burnup. The ways to optimize VVER cores, fuel assemblies, and fuel elements with the aim to improve the fuel burnup and to improve the reactor safety are analyzed. In the paper, the secondary are the effects of burnup enhancement on a final stage of nuclear fuel cycle.

I. SOVIET NUCLEAR POWER IN 1987-1991

At present the NP fraction in electric energy production in the USSR is about 12%. 47 NPP units of total installed capacity of 36412 MW(e) were under operation in this country as of January 1, 1991, the NP structure being as follows: 16 units of VVER (16000MW(e)), 8 units of VVER-440 (3412MW(e)), 14 units of RBMK-1000 (14000MW(e)), and 2 units of RBMK-1500 (3000MW(e)). The capacity factor (CF) averaged over this period was 60-70%.

The Chernobyl accident gave birth to a wide opposition to NP development in the USSR. This slowed down the rate of new NPP commissioning. As a result, the program on the Soviet NP development has been revised. A draft of a anew Nuclear Power Program concentrates on the further improvement of safety and performance of operating NPPs and on the development of improved-safety VVER-1000 units incorporating passive safety systems [1]. It is supposed that the activities along these lines will be coordinated, as before, by the Ministry of Nuclear Power and Industry and supervised by the USSR State Committee on Supervision of safe operation in industry and nuclear power.

2. EXPERIENCE IN OPERATING VVER CORES WITH ENHANCED FUEL BURNUP

2.1. VVER-440

Early VVER-440 loadings were intended for a 3-year fuel lifetime at an average enrichment of 3.25% with out-in fuel transfer scheme. The refuelling-averaged fuel burnup of 29MW.kg/day was achieved with the cycle 292 eff.days (7000hr) long.

The experiments on switching VVER-440 to a 4-year lifetime of 3.6%-enriched fuel performed in early eighties at Kola NPP proved to be a success as well as subsequent experiments with 4-year fuel operation at Units 3 and 4 of Novo-Voronezh NPP and Units 1 and 2 of Kola NPP where fuel was also enriched to 3.6% [2].

The increase of fuel enrichment up to 4.4% has allowed not only the increase in the fuel burnup but also the increase in the cycle duration up to ~ 350 eff.days (~8000 hr) with a 4-year lifetime. 78 fuel assemblies with 4.4% - enriched fuel (1/4 fraction of the whole core) were for the first time loaded to core periphery of Unit 3 of Kola NPP.

In 1990 a 4-year lifetime of the core with enhanced-enrichment fuel was completed. Operating assemblies contained 4.4% -enriched fuel, while fuel sections of control and protection assemblies contained 3.6%-enriched fuel. The duration of fuel cycles were 246, 336, 300, and 370 eff.days. The average and maximum burnup of unloaded fuel assemblies was 42MW day/kg and 48MW day/kg, respectively.

The experience gained served the basis for switching VVER-440 reactors to a 4-year fuel cycle involving 4.4%-enriched fuel. In order to validate the future transferring to 5-year fuel cycle, 12 fuel assemblies of Kola NPP Unit 3 remained under operation for the fifth year of service. This experiment validated by preliminary thermal-physics and neutronics calculations proved to be a success.

2 2 VVER-1000

The early VVER-1000 loads were intended for a 2-year fuel cycle at an average fuel burnup of 29MW day/kg and enrichment of 3.6%. At present VVER-1000 reactors are being switched to 3-year fuel cycle at an average burnup of more than 40MW day/kg and average enrichment of loaded fuel of 4 23%.

Novo-Voronezh NPP Unit 5 and Kalinin NPP Unit 1 have already reached a steady-state condition with three refuellings during the core life; the burnup averaged over one fuelling has reached the value of 43-45MW day/kg.

The implementation of a 3-year fuel cycle at Zaporozhye NPP Unit 5 was started with the first fuelling. The other operating VVER-1000 units are under transferring from 2- to 3-year core lifetime. Detailed information concerning this problem is given in [3].

In order to validate 3-year lifetime, fuel element and assembly operational reliability was both predicted and measured in experiments and their parameters were optimized. This will be considered below. At present the whole complex of calculations and experiments is in progress to validate a 4-year fuel life for VVER-1000 reactors.

During scheduled preventive maintenance essentially each VVER-1000 unit was tested for fuel clad integrity. Over a period from 1982 to 1990 4190 assemblies were tested and only 2 assemblies were unloaded before the scheduled time. Not a single assembly failed after three years of operation.

3. EFFECT OF IMPROVED BURNUP ON DEMAND FOR NATURAL URANIUM AND ENRICHMENT LEVEL

The experience gained and estimation performed show that in an open fuel cycle (without recycling of reprocessed uranium and plutonium) the optimal level of fuel burnup in existing VVER reactors is about 50 MW day/kg, in contrast to early design burnups of 29-32 MW day/kg The present-day values will help to reduce the demand for natural uranium by about 20% [4].

Thus, the switching of VVER-440 reactors to a 4-year fuel lifetime where enrichment of loaded fuel was 4 4% in operating fuel assemblies and 3.6% in control and protection assemblies allowed the fuel burnup to be improved up to 42 MW day/kg and the consumption of natural uranium to be reduced by ~12% [4].

The switching of VVER-1000 reactors to a 3-year, 4 4%-enriched fuel lifetime allowed the fuel burnup to be improved up to 43-45MW day/kg and the consumption of natural uranium to be reduced by ~ 15% [4].

The two abovementioned switching have already been accomplished. Below presented are some predictions [4]·

- the reduction of deleterious neutron absorption in VVER cores by replacing steel in assembly parts with zirconium and the reduction of hafnium content in zirconium will allow an improvement of burnup by 2-3% and 8-10% for VVER-440 and VVER-1000, respectively, or the reduction of loaded fuel enrichment by ~ 0.4% with respect to U-235. This may provide the reduction in natural uranium consumption by 3-7% and 8-12% for VVER-440 and VVER-1000, respectively;

- the implementation of refuellings with reduced radial neutron leakage provides the additional improvement of fuel burnup by 5-6% and similar reduction of natural uranium consumption;

- the application of integrated burnable neutron absorbers in VVER-1000 fuel load will allow the core arrangement with reduced radial neutron leakage In this case, the fuel burnup may increase b y ~ 6-9% [5] and the natural uranium consumption may reduce by a similar value

Calculations on VVER fuel cycle optimization in terms of a fixed cycle duration (12, 18 months for VVER-440 and 12 months for VVER-1000) were performed within the framework of the IAEA REBUS program in 1989-1991

The dependence of a fuel enrichment required for achieving a particular fuel burnup (Fig.1) was determined basing on physics calculations of core refuellings with out-in fuel transport Physics calculations were experimentally verified up to fuel burnup of 45MW day/kg. The change in a VVER-1000 curve character (Fig 1) is due to a specific core arrangement. With the increasing fuel burnup and associated increase in fuel enrichment, a core with a partly reduced radial neutron leakage gets formed due to the placement of partly outburned assemblies to particular positions in the core periphery even if in general the fuel transportation scheme is out-in

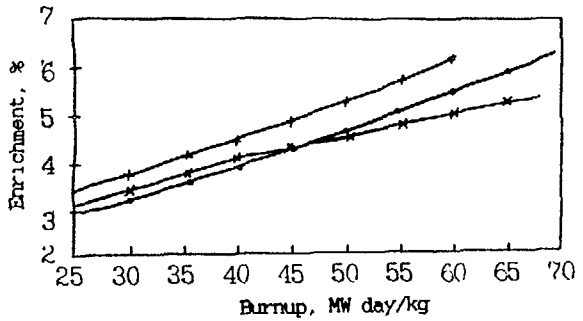

Fig 1 Enrichment vs fuel burnup [6]
—•— VVER-440, 12 months
—I— VVER-440, 18 months
—x— VVER-1000, 12 months

In this case the core optimization is performed to achieve a uniform power density rather than the best fuel utilization [6]. These two approaches differ most essentially in the burnup range of up to ~ 40MW day/kg.

In the REBUS program, the fuel burnup effect on the consumption was predicted in the units of enrichment work ($WU) For VVER-440 the value of $WU/KW.hr falls with the increasing burnup to medium values followed with a slight rise, the maximum difference being not more than 7% [7] For VVER-1000 the value of $WU/KW hr. reduces monotonously with the increasing burnup, the difference achieves 21 5% when the burnup increases from 33 to ~65MWday/kg [7]

The data on the VVER-440 and VVER-1000 fuel burnup effect on natural uranium consumption estimated within the framework of the REBUS program is shown in Fig 2 The reduction in the natural uranium consumption estimated by REBUS is rather close to earlier data predicted by Soviet specialists for particular cases of VVER switching from 3 - to 4-year fuel lifetime (11 5 and 12%, respectively) and VVER-1000 switching from 2 - to 3-year fuel lifetime (~14 and ~15%, respectively)

VVER fuel burnup effect on fuel cycle economy is not considered at the present meeting Detailed information on this issue may be found in [7]

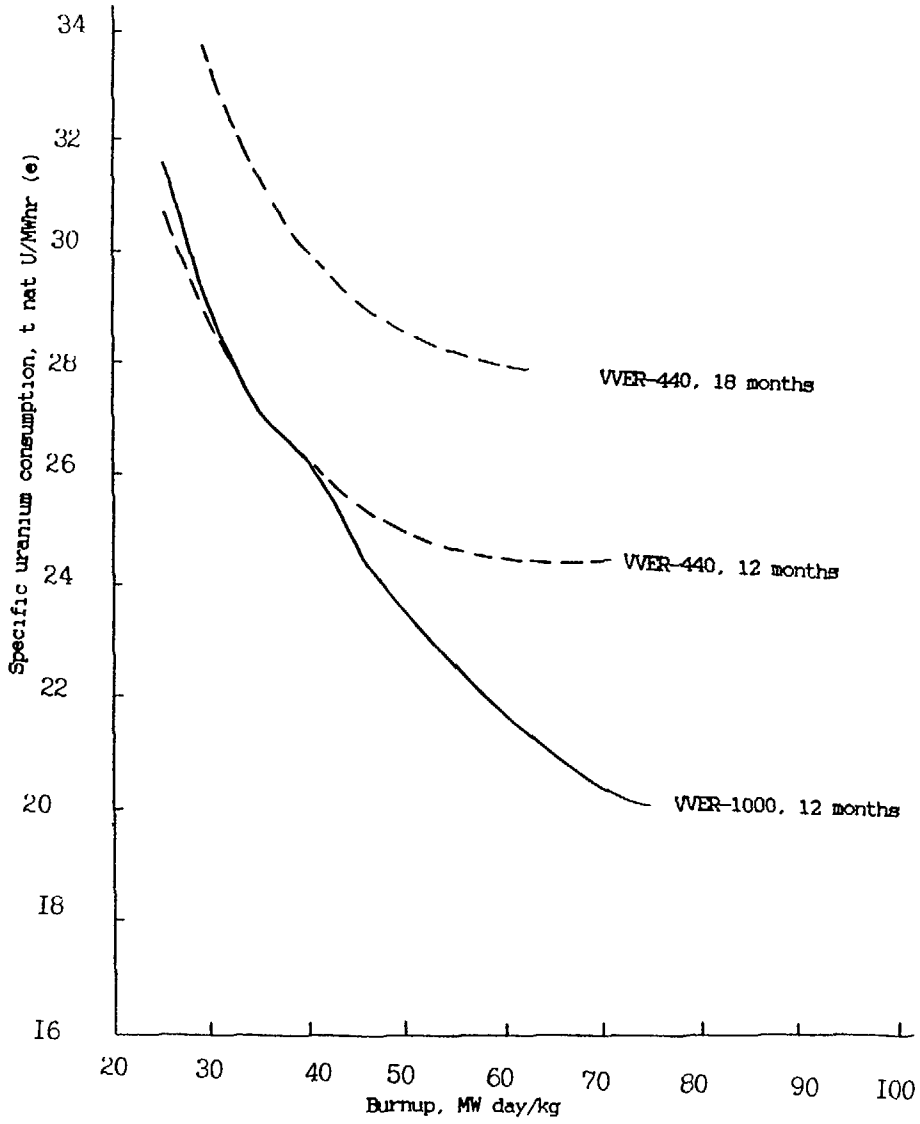

Fig 2 Specific natural uranium consumption vs fuel burnup for VVER [7]

4. FUEL HANDLING IN THE CORE AND MONITORING OF VVER REACTOR

REACTIVITY BEHAVIOUR IN THE PROCESS OF FUEL BURNUP INCREASING

4 1 VVER-440

In the beginning of a fuel cycle the VVER-440 and VVER-1000 core possesses a maximum reactivity margin intended for bringing the reactor to an operating condition. Fuel burnup during 7000 hr (292 eff.days) at a nominal power is 1375 MW(th) and 3000MW(th), respectively.

The VVER reactivity margin is compensated with a boric acid dissolved in a primary coolant

Fast (power and part of temperature) effects of reactivity in emergency and transient conditions are compensated with mechanical members of a control and protection system

The use of enhanced-enrichment (up to 4.4%) fuel in VVER poses additional requirements to providing nuclear safety.

The fifth fuel load to the Kola NPP Unit 3 reactor included 78 assemblies enriched to 4 4% Maximum reactivity margin was 17.6%. To load was experimental, therefore the concentration of boric acid in coolant under shutdown conditions was increased up to 16g/kg. In this case subcriticality of a completely loaded reactor in cold condition with withdrawn mechanical members of the control and protection system was 8.3% (according to safety requirements subcriticality should be not less than 2%). It should also be noted that with an enhanced-enrichment fuel in a reactor, a differential efficiency of boric acid in its absolute value reduces by more than 25% in comparison with its value for a design regime of VVER-440 refuelling This effect favours safety in the case of accidental dissolving of boron-containing coolant with pure water, but it aggravates the reactor maneuvering because of the reduction in the rate of reactivity release by water-exchange when it is necessary to compensate non-stationary xenon poisoning.

Further calculations showed that the shutdown concentration of boric acid at the level of 13 27g/kg provides a sufficient subcriticality Taking into account experimental and calculational errors a shutdown concentration of 14g/kg [2] can be recommended for fuel loadings having the cycle 350-380 eff days long

In VVER reactors, the value and sign of coefficient of reactivity in coolant temperature ($\alpha_T = \partial\rho/\partial T_c - 1$) depend on critical concentration of boric acid in coolant, startup position of a control group, core load composition, degree of assembly burnup, and on a very temperature Core characteristic structural materials are such that for VVER-440 loads of rated energy capacity (292 eff days, fuel enrichment of 2 4% and 3 6%) at any combination of startup parameters at the beginning of a fuel cycle α_T is negative over an operating temperature range from 200 to 285°C. With the increasing energy capacity of the VVER-440 load (320 eff.days) a necessity arises to compensate an excessive reactivity margin by increasing the boric acid concentration. This may cause the alteration of an sign in the operating temperature range and a higher startup temperature may become necessary. For loads of Kola NPP Unit 3 a conservative startup temperature of 260°C was chosen at which α_T at

TABLE I

Power and temperature coefficients of reactivity at energy level of VVER-440 power

	Experiment		Calculation	
I. Transient fuel cycle				
Power, %/MW	-1.00	10^{-3}	-0.92	10^{-3}
Temperature, %/grad	-1.85	10^{-2}	-1.77	10^{-2}
2. Transient fuel cycle				
Power, %/MW	-1.44	10^{-3}	-1.27	10^{-3}
Temperature, %/grad	-2.42	10^{-2}	-2.63	10^{-2}
3. Transient fuel cycle				
Power , %/MW	-1.26	10^{-3}	-1.21	10^{-3}
Temperature, %/grad	-2.42	10^{-2}	-2.85	10^{-2}
4 Stationary fuel cycle				
Power, %/MW	-1 0	10^{-3}	1 19	10^{-3}
Temperature , %/grad	-2 80	10^{-2}	2 88	10^{-2}

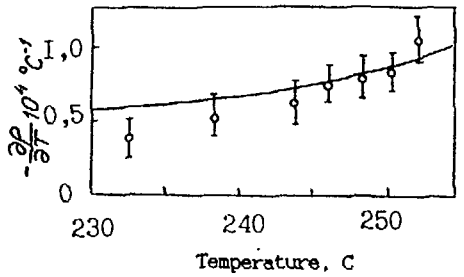

Fig.3 Water and fuel coefficient of reactivity vs temperature with boron acid concentration of 8.7 g/kg
—— calculation
o experiment

any position of a control group over a control range is ensured negative [2].

The values of reactivity coefficients for transient and stationary (all fuel is enriched to 4.4%) loads are shown in Table 1. The temperature dependence of α_T is shown in Fig.3.

4.2. VVER-1000

In the VVER-1000 cores, the total reactivity margin is compensated by two systems: mechanical members of a control and protection system (61 members) in the form of bunches of thin absorbing components made of boron-containing material (18 AC in a bunch) and boric acid dissolved in coolant.

With a 3-year fuel lifetime (fuel enrichment is 4.4%) in VVER-1000 reactors, to improve safety a part of reactivity margin is compensated by burnable absorber. Rods with burnable absorber are used in fresh assemblies during the first year of operation.

In order to improve reactor safety, considered is an introduction of burnable absorber into the fuel (the use of uranium-gadolinium fuel or pellets with deposited boron-containing material).

This use of burnable absorber helps to get more negative reactivity coefficient with respect to coolant temperature at the beginning of fuel lifetime [4,5], and also to increase the number of control and protection members in order to improve the efficiency of

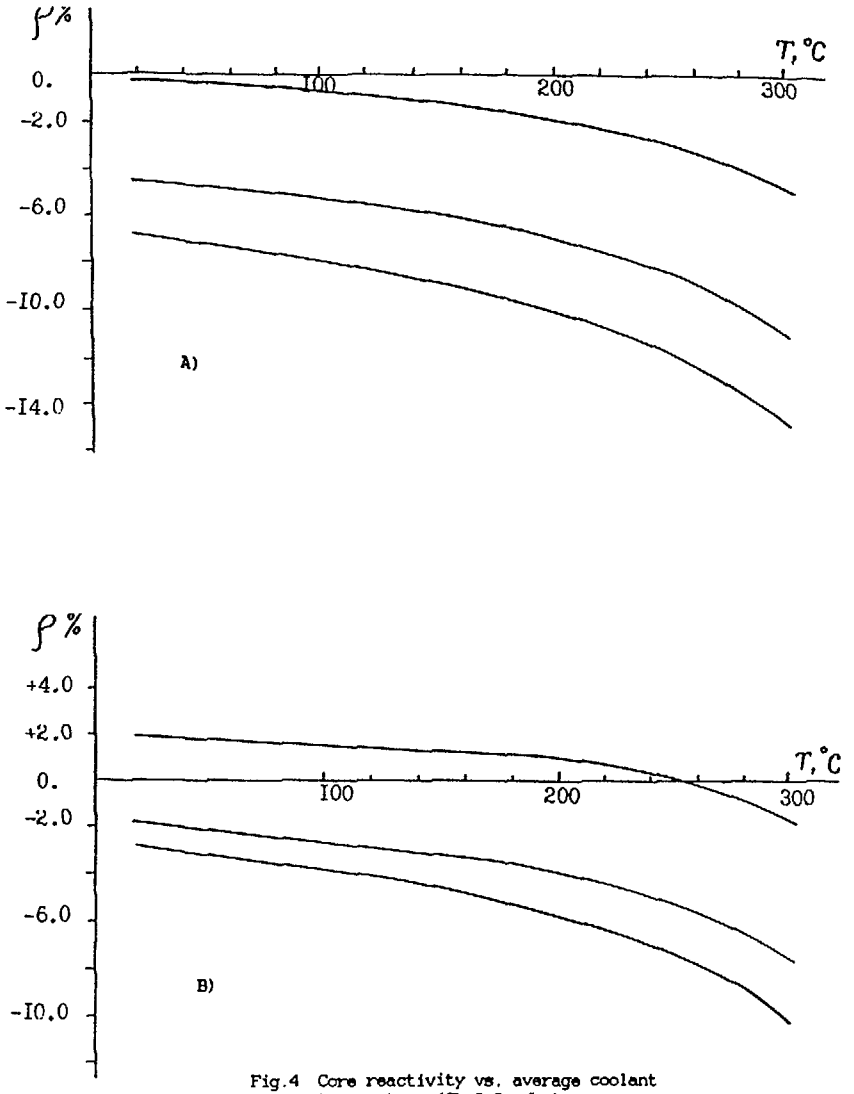

Fig.4 Core reactivity vs. average coolant temperature (Cb=0.3 g/kg)
a) all members of control and protection systems are inserted
b) the highest-worth group is "stuck" in upper position

61.97.121 - number of members of control and protection system

reactor emergency protection (e g with 97 members of control and protection system a repeated criticality at the end of a burnup cycle is excluded even if the whole highest-worth rod group is stuck in the upper position (Fig 4)

5 OPTIMIZATION OF FUEL ELEMENT AND ASSEMBLY DESIGN AND MATERIALS IN VVER REACTORS WHEN TRANSFERRING TO ENHANCED FUEL BURNUP

5 1 VVER-440

The fuel element and assembly designs were improved to provide reliable operation of fuel elements and assemblies in switching from 3- to 4-year fuel cycle

Here we mention some of these improvements·

- pressure of helium filling the fuel-clad space in fuel elements was increased from 0.1 to 0 5 MPa. In combination of this factor and a central hole of 1.2-2.0 mm dia in the fuel pellet provided the reduction of fuel temperature and hence of gaseous fusion product release [8],

- to provide a free travel of fuel column and to reduce the formation of grits, the fuel pellets with facets were introduced:

- to compensate the radiation-induced growth of fuel elements, the space between the fuel element upper ends and the protective grid of the assembly head was increased up to 25 mm

A number of technological improvements was also introduced with the aim to improve the quality of fuel pellets and fuel elements the tolerances on the moisture and fluorine content in fuel pellets, fuel column length and mass, fuel clad diameter etc were made closer

Comprehensive post-reactor studies on VVER-440 assemblies operated in 3- and 4-year fuel cycle conditions indicated good state of fuel elements and other assembly components Below we give some results obtained with an assembly operated in NVNPP Unit 3 during 5-year cycles and achieved the average burnup of 49 MW day/kg [8]

- fuel clads preserve the plasticity margin, mechanical properties do not differ from those of fuel elements of assemblies operated for three one-year cycles,

- oxidation and hydration is small, oxide film thickness is less than 3 μm,

- fuel has not undergone any essential structural change, column-like grains are not observed;

- size changes of fuel elements are unimportant,

- internal gas pressure at the end of exposure does not excess the coolant pressure (gas release is insignificant);

- studies have not found any signs indicative of impossibility of further operation

5.2. VVER-1000

In switching to a 3-year VVER-1000 fuel cycle with burnup higher than 40MW day/kg and enrichment of 4 4%, with the aim to improve the operational reliability of fuel elements the diameter of fuel pelleted central holes was increased from 1.4 to 2 4 mm. This helped to reduce the fuel temperature and hence the gaseous product release, and also to increase a free space. At a pressure of helium inside a clad of 2 0-2 5 MPa, a sufficient stability of fuel clad is provided by the end of the burnout cycle [9].

Besides, a number of steps was taken to improve the fuel and clad quality. In particular, the moisture and fluorine content in pellets was reduced, the production of grits was reduced (pellets with facets we applied), and admissible maximal depth of clad defects was reduced down to 50 mcm etc [9]

Analysis of commercial fuel element and clad parameters showed reasonable conformity with technical requirements and good stability [10]

It is envisaged to continue optimization and improvement of fuel element and assembly design and performance in the nearest future with the aim of providing high operational reliability at a more extended burnup The following improvements are expected to be accomplished

- fuel element elongation up to 100 mm without changing fuel assembly overall size This will increase a free space in a gas collector or a fuel column height,

- introduction of Zr to spacers,

- application of integrated burnable neutron poisons (uranium-gadolinium fuel or boron-compound deposition on pellet surfaces)

Technology, safety, and economy issues relative to uranium-gadolinium-fuelled VVER-1000 reactors are also covered in this country by the IAEA BAF program of coordinated studies. Some results obtained within the framework of this program are presented in [5].

Post-reactor studies on VVER-1000 fuel elements, exposed in the MR reactor under relatively severe conditions (up to a burnup of 80 MW day/kg) showed that the change in fuel-element geometric size is insignificant, the oxidefilm thickness on the fuel and coolant side is small (<5 μm) and the fuel clad preserves a considerable plasticity margin [11].

Post-reactor studies on 2- and 3-year operated VVER-1000 fuel assemblies confirmed the validity of particular selected fuel element and assembly design and parameters.

Post-reactors data agree reasonably well with preliminary fuel element calculations performed by the RETR and PIN 0.4 codes.

6.VALIDATION (LICENSING) OF AN EXTENDED VVER-FUEL BURNUP

Early fuel loads in VVER-1000 and -440 were experimentally and theoretically validated up to burnup of the order of 29-32 MW.day/kg. Any further step to extending burnup required additional validation of VVER-440 and -1000 switching to a 4- and 3-year fuel lifetime, respectively, with a burnup averaged over one fuelling of up to 40-42 and 40-45 MW.day/kg, respectively. Appropriate technical safety specifications were formulated and approved by State

These specifications included also the following basic fuel-related documents· design specifications, loop experimental results, calculations on fuel-element and assembly operational reliability by the tested PETR and PIN 0.4 codes, neutronics calculations determining the condition of fuel operation (BIPR-7, PERMAK etc.), analysis on operational experience with fuel and cores.

The possibility is investigated to extend the VVER-fuel burnup to 50 MW.day kg (5- and 4-year fuel cycles for VVER-440and 1000, respectively, Zr introduction to spacers, refuelling with reduced radial neutron leakage) Burnup extension due to the above improvements will require additional validation and appropriate technical specifications on Safety.

7. EFFECT OF EXTENDED VVER-FUEL BURNUP ON THE INITIAL STAGE OF A NUCLEAR FUEL CYCLE

The application of enhanced-enrichment fuel assemblies for VVER-440 fuelling poses the additional requirement to nuclear safety in handling fresh fuels. To validate nuclear safety it is necessary to provide such conditions of fresh fuel storage at which the subcriticality of not-less than 0.05 at any accident could be ensured [12]

The pitch of a fresh, 4 4%-enriched fuel assembly lattice which ensures nuclear safety was calculated by the change of an effective multiplication factor in infinite lattices placed in water of different density with different content of boric acid [13]

The racks for storing fresh fuel assemblies should be designed so that assemblies be spaced in a triangular lattice at a pitch of 225 mm, which is sufficient to ensure safety.

Similar means and conditions for storing fresh fuels are applied to VVER-1000 reactors.

8 EFFECT OF AN EXTENDED VVER FUEL BURNUP ON A FINAL STAGE OF NUCLEAR FUEL CYCLE

The effects of an extended fuel burnup on transportation and long-term storage of spent VVER nuclear fuels are considered in detail within the framework of the IAEA BEFAST" program of coordinated studies, while the issues of VVER-440 spent nuclear fuel reprocessing (VVER-1000 spent fuel will be reprocessed by the RT-2 plant now under construction) were delivered last fall to the Meeting of IAEA Advisory Committee on Spent nuclear fuel handing In this section, the effects of fuel enrichment in U-235, assembly capacity, burnup and operation conditions on decay heat of spent nuclear fuels under long-term storage conditions will be exemplified by VVER-440

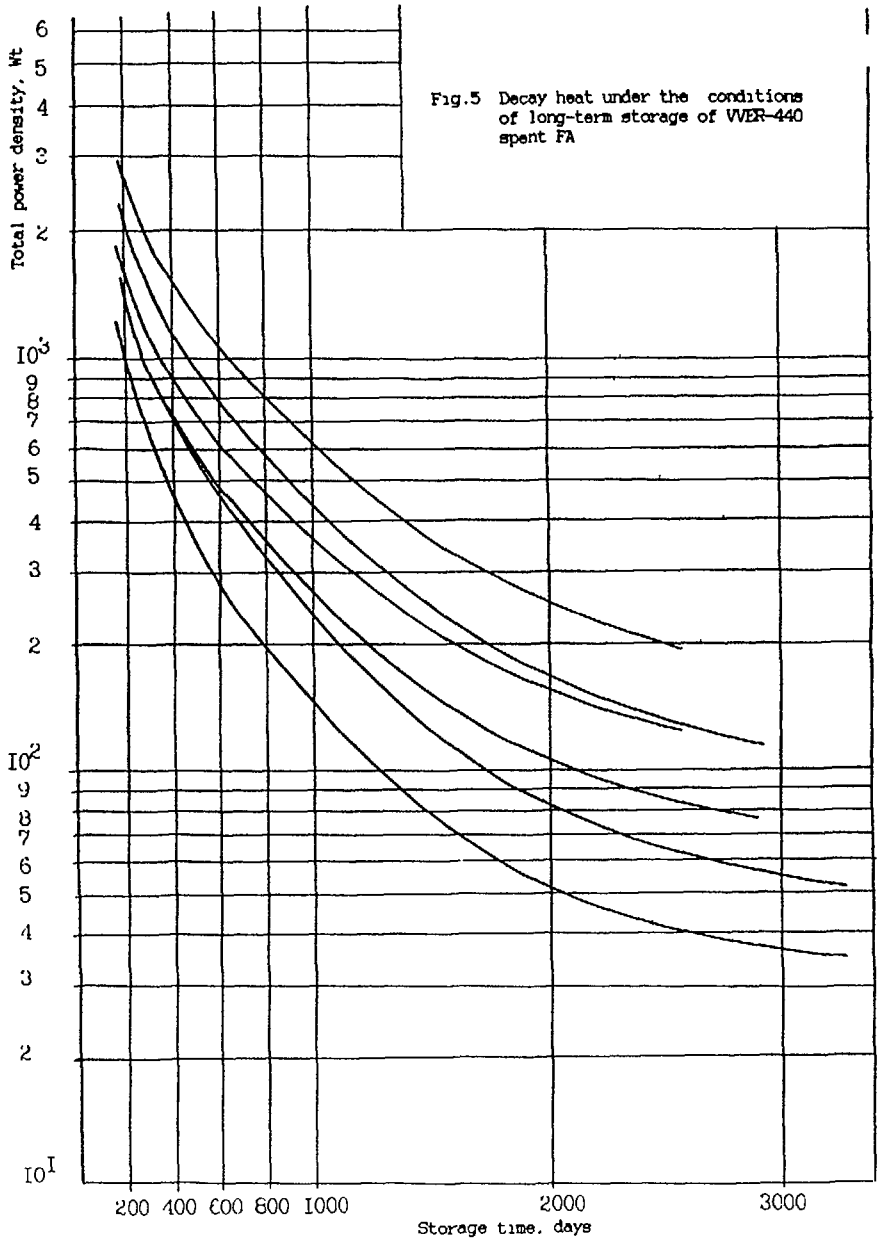

Fig.5 Decay heat under the conditions of long-term storage of VVER-440 spent FA

TABLE 2

Values of enrichment, operational power of a fuel assembly, and number of exposure cycles 294 eff.days long (taking into account the duration of one fuelling - 30 days), VVER-440

(for Fig.5)

Number of curve in Fig.5	Enrichment, %	Operational power of fuel assembly, MW	Number of fuel cycles
1	1.6	5.95 (max)	1
2	1.6	3.94 (av.)	1
3	2.4	5.95 (max)	2
4	2.4	3.94 (av.)	2
5	3.6	5.94 (max.)	3
6	3.6	3.94 (av.)	3

Fig.5 shows the dependences of VVER-440 spent fuel decay heat under the conditions of long-term storage. Table 2 gives data on operation conditions, initial fuel enrichment and burnup.

Calculations on decay heat were performed using into account only the products of U-235 and PU-239 fission (β - and γ- radiations were included). The contribution of actinides to heat release is neglected.

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BURNUP AS A SAFETY PARAMETER FOR HANDLING TRANSPORT PACKAGES AND STORAGE FACILITIES FOR SPENT NUCLEAR FUEL

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Abstract

Four types of casks used in the Soviet Union for spent fuel transportation, namely TK-6, TK-10, TK-11 and TK-13 are considered in this paper in connection with issues arising from burnup increase. It has been shown that nuclear safety (undercriticality) and permissible radioactive content lost from a cask are not limiting factors in burnup increase sense. Residual heat dissipation and biological shielding of the casks are the issues relevant to burnup increase and discussed in the paper. Some attention is given to ensuring safety of highly burnt spent fuel discharged from VVER and RBMK type reactors.

1. Introduction

Specific development of nuclear industry in most countries of the world demands:

- increase in initial fuel enrichment for operating and projected NPP;
- increase in fuel burnup during NPP operation;
- increase in SNF storage capacity;
- justification of dry method for SNF storage.

In overcoming the first two problems it is necessary to estimate whether the number of existing casks is capable of transporting such fuel with enhanced characteristics from NPP to a reprocessing site If a cask does not provide the safety level required, it is necessary to consider transportation methodology being suitably changed.

Safe transportation of NPP spent fuel is generally provided with the following characteristics of a cask:

- efficient and reliable radiation shielding;
- secured sub-criticality of packages;
- ability for fuel assembly decay heat removal;
- permissible losses of volatile fission products.

General approach, to complete the cask loading with spent fuel in case of fuel parameter values surpassing those assumed in cask designing, is as follows:

Determine which of the above four characteristics represents a limiting factor. Then either select the most suitable load for a cask(reducing the number of fuel assemblies, increasing their cooling time) or change transportation pattern(replacing the cooler in a cask or using neutron-shielded casks for transportation).

2. Cask Description

There are four types of casks used frequently for NPP spent fuel transportation in the Soviet Union, they are TK-6, TK-10, TK-11 and TK-13.

The TK-6 cask is designed for transporting by rail thirty PWR assemblies from WWER-440-type(as well as BK-50-type) reactor plants. It is a thick-walled forged flask weighing more than 78 tons. The cask walls, bottom and lid protect against ionizing radiation, no special neutron shielding being provided. Fuel decay heat is removed by natural convection and radiation from the cask outer surface with many ribs. The inner cavity of the cask may be filled with either gas or water.

The TK-10 is intended to transport by rail six PWR assemblies from WWER-1000(or WWER-500 and ACT-500)-type reactor. It is manufactured as a thick-wall forged steel flask, on the outer side of which there is a liquid neutron shielding of ethylene glycol (67%) mixed with water. Residual decay heat is removed by natural convection and radiation from the cask smooth outer surface.

Table 1

Spent Fuel Package Characteristics

Cask identification	TK-6	TK-10	TK-11		TK-13	
Fuel						
Type of fuel	PWR (WWR-440)	PWR (WWR-1000)	BWR (RBMK-1000)	LMFBR (BN-600)	PWR (WWR-1000)	BWR (ACT-500)
Initial enrichment, wt% U-235	3.6	4.4	2	33	4.4	2
Number of assemblies	30	6	51	35	12	12
Average burnup,GWD/MTU	20/28	42	20	80	42	15
Cooling time,years	3	3	3	3	3	3
Maximum decay heat,kW	8/12	13	10.3	10.7	20	20
Loaded fuel,tU/pack	3.6	2.6	5.9	1.0	5.2	4.8
Cavity						
Diameter,cm	147.5	100	148.5		132	
Length,cm	348	503	379		495.5	
Coolant in a cask	gas/water	water	gas	gas	gas	gas
Pressure in a cask,atm	1 66/2.5	2.5	2	2	1.66	1.66
Shielding:						
Gamma shield material	steel	steel	steel		steel	
Thikness of shielding, cm	36	38	36		36	
Gamma dose rate 2m from side surface, mrem/h	4/3	2	4	3	5.6	1.6
Neutron shield material	-	eth. glycol	-	-	ethylene glycol	
Thickness of shielding,cm	-	12	-	-	12.7	
Neutron dose rate 2m from side surface, mrem/h	4/2.6	0.4	3	1	1	0.02
Overall characteristics						
Outside diamerter,cm	219.5	200	219.5		229.5	
Length of cask,cm	410.5	613	445.5		600.0	
Mode of surface	steel fins	smooth surface	steel fins		smooth surface	
Cask shape	vertical cylinder	horiz. cylind.	vertical cylinder		horizontal cylinder	
Transport means	r.car	r.car	railway car		railway car	
Empty weight,t	76.5	84	86.5	86.5	106	106

The TK-11 cask may be used for rail transportation of fifty -one BWR assemblies from RBMK-1000-reactor or thirty-five LMFBR assemblies from BN-type reactor. It is also useful for transporting fuel assemblies from WWER-440 and RBMK-1500-type reactors. This cask is similar to the TK-6 one by design and safety parameters.

The TK-13 cask is design for railway transporting twelve PWR assemblies from WWER-1000 reactor or twelve BWR assemblies from the ACT-500-reactor. The cask basic construction is a forged steel flask with external neutron shielding of ethylene glycol(67%)-and-water mixture. Decay heat is dissipated from a smooth outer surface of the cask by natural convection and radiation.

Some technical characteristics of the casks are given in Table 1.

3. Casks Design Potentialities

According to the 'State Bodies' requirements for supervision over safe transportation, package criticality questions should be decided upon without burnup being taken into consideration (i.e. for fresh fuel). Therefore, nuclear safety provision is not a limiting factor for the most of containers designed for maximum fuel enrichment.

The casks have a sufficiently reliable leak-proof system eliminating volatile radioactive isotopes losses (both under normal transport conditions or in emergency situation),that surpass permissible values in case of loading the cask with fuel promising the maximum burnup. Thus, permissible radioactive content lost from a cask is not a limiting factor either

With burnup rising above the average values shown in the Table 1 neutron source of spent fuel drastically increases. Furthermore, dose rate outside a cask is likely to exceed the regulatory limits adopted. In this case, transportation may be done in a water-filled cask providing for additional neutron

Table 2

Radiation Doses Outside Shielding of Casks with Fuel Loaded with Maximum Design Burnup

Cask identification	TK-6	TK-10	TK-11		TK-13	
Type of fuel (UO2)	PWR (WWR-440)	PWR (WWR-1000)	BWR (RBMK-1000)	LMFBR (BN-600)	PWR (WWR-1000)	BWR (ACT-500)
Enrichment,wt%U-235	3.6	4.4	2.0	33.0	4.4	2.0
Cooling time,years	3	3	3	3	3	3
Coolant for transport	gas/water	water	gas	gas	gas	gas
Maximum burnup,GWd/tU	20/40	50	20	150	50	17
γ-quantum dose rate at 2m distance	2/1	0.34	1	2.5	3.1	1.6
Neutron dose rate at 2m distance	8/7.5	0.05	4	0.3	4.3	0.02
Heat output for cask, kW	8/12	13	10 3	10.7	20	20

shielding, otherwise the number of assemblies to be transported may be reduced.

Biological shielding of the casks being considered is designed to considerable allowance accounting for a cask to be loaded with fuel of the maximum burnup which is possible with reactors of this type. Table 2 shows ionizing radiation dose rates 2 m away from side surface of a cask fully loaded with fuel of maximum burnup, cooled for the design time-period

The temperature of fuel assembly elements cladding in the transport should not exceed 350 C. This puts the requirements for the system of fuel assembly decay heat removal, and it is just this condition that is often a limiting factor in determining the cask loading Thus, not all the loadings shown in Table 2 are feasible, for the removal system does not provide values of temperature adopted in the regulations for

packages. For example, with the TK-13 cask fully loaded and WWER-1000-type fuel being cooled during up to 3 years, its burnup is not to exceed 42 GWd/tU. An elevated pressure of coolant in the cask cavity which is expedient not with standing the worse conditions for a cask leak-tightness secured, is accounted for an effort to advance in FA decay heat removal.

4. Opportunities Taken from the Modified Transport Methodology

In accordance with the advanced program for safe operation of RBMK-reactor power plants, maximum fuel enrichment was increased from 2.0% to 2.4% with possible resultant growth in fuel burnup to about 2.8 GWd/tU. Transportation of such fuel in the TK-11 casks necessitates changes in the transport procedure:

1. For securing nuclear safety and pressure permissible in the cask cavity under emergency conditions, transportation should be made in gas-filled casks only;

2. Meanwhile, radiation shielding becomes the principal limiting factor. Thus, fuel with initial enrichment of 20% and burnup of 22GWd/tU can be transported only after being cooled for not less than 8 years. As to fuel enriched initially to 2.4% and burnup of 28.5 GWd/tU, dose rate outside the shielding exceeds standard values, even though the fuel is cooled for 15 years. In this case the number of fuel assemblies loaded into TK-11 casks is to be reduced; for example, it is possible, without breaking the safety standards, to load 15 assemblies being cooled for 8 years instead of the planned 51 assemblies loaded in a cask. For more extended burnup fuel a new cask should be developed or a reactor storage built.

To improve the economy, some nuclear power plants with WWER -440 reactors intend to use fuel with increased 4.4% enrichment. In this case, as seen from the experience of Kolskaya NPP, the fuel burnup can reach 50 GWd/tU. Such fuel can be transported in TK-6, only in the water-filled cask, and its full loading is not permissible according to both nuclear safety and radiation protection requirements. The problem can be solved by replacing the outer row of fuel assemblies with their imitators which will serve as an additional shield and supersed excess water. The cask loading, in this case, reduces to 18 fuel assemblies.

In some instances, transportation of fuel, cooled for less than the design period, is required. Such conditions can arise when the schedule for fuel dispatch to the reprocessing plant is changed or when the NPP is decommissioned ahead of schedule as the Armenian NPP, for example.

As for TK-6, the system for optimizing loading of cask baskets is developed. The system is based on sorting out the fuel assemblies to be loaded in a cask, with burnup below 40 GWd/tU, or reducing their number for loading. In winter-time transportation, an increase in total release up to 15 kW instead of 12 kW per a cask being fully loaded, proves to be possible.

Similar recommendations for transportation of WWER-type fuel with a relatively shorter cooling period as compared to a designed one, are developed to fit the TK-10 and TK-13 casks. According to the experimental data, the cask shield is well within the safety margins, and the limiting factor is a cask design capacity for decay heat to be dissipated. For example, assemblies initial fuel enrichment below 4.4%, which are used under reactor start-up conditions without extended burnups, can be transported when fuel cooling time is shorter than designed. Furthermore, radiation safety requirements are satisfied if decay heat from all the assemblies in a TK-10 cask does not exceed 13 kW and that in a TK-13 cask-20 kW.

5. Increasing Waste Fuel Storage Capacity

This problem has become more acute in connection with a decision taken to abandon the reprocessing of RBMK-type spent

fuel and, consequently, its long-term storage in quantities exceeding capacities available At present, interim storage densification is suggested to be doubled. The principal limiting factor for a design and densification rate to be chosen is a nuclear safety requirement. The problem can be solved with using solid neutron absorbers, such as baskets made of steels with boron, gadolinium etc. admixed This way is rather widely used, though involving considerable material and financial costs. So, in recent time the possibility of solving the problem with burnup accounted for, is discussed.

Design studies conducted at the Physics-Energetics Institute, Obninsk, show that even though the 1986 decision on increasing initial fuel enrichment up to 2.4% being considered, burnup taken as a nuclear safety parameter allows to overcome the problem of densified RBMK-type fuel storage without using solid neutron absorbers. Consideration was given to various FA arrangement pitches adopted in storage projects and planned for their being desified: 250x160 mm, 230x110 mm, 250x80 mm, 125x110 mm, 115x110 mm. The analysis shows that with any fuel arrangement pitches planned and in emergency situatios, a multiplication factor for assembly infinite lattice does not exceed the value of 0.95, on condition that fuel is stored in canisters and its burnup goes over 10 GWd/tU (that corresponds to average burnup for the first-year fuel campaign). If the fuel is stored without canisters, the burnup should be not less than 21 GWd/tU.

In its turn, using burnup as a safety parameter requires for solving thje following problems:

1) nuclear safety provision for storage of fuel assemblies with burhup being less than a designed one,

2) elimination of personnel errors when burnup being estimated,

3) reliable evaluation of burnt-up fuel isotopic content,

4) reliable estimation of burnup distributed along a fuel assembly,

5) carrying out base experiments for verifying a criticality analysis program and neutron-physical constants' library available with regard to fission products and actinides being present in burnt-up fuel.

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IMPACT ON THE BACK END OF THE NUCLEAR FUEL CYCLE

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Abstract

Extending burnup affects the characteristics of spent fuel to a larger extent than a simple linear extrapolation. Increased alpha (and consequent decay heat generation) and neutron activities are the predominant features impacting on spent fuel management, both in the open and closed cycle options. Reprocessed U has properties only marginally affected by the higher burnup. To the contrary, the properties of Pu are strongly modified and higher Pu contents are required. However, the time scale for the MOX fuel industry to cope with this challenging future is stretched over a long enough period for the MOX manufacturing plants to be adapted in due time and for the adequate data base to be implemented.

1. Introduction.

The impact of extending fuel burnup should be considered in the perspective of both the open (i.e. final disposal of spent fuel as final waste) and the closed (i.e. reprocessing and recycling of the fissile material) cycle, since each alternative is equally used (worldwide) as the reference back end option (fig.1), based on (1). As the risks associated with spent fuel become predominantly influenced by the actinides, rather than by the fission products, after 10 to 30 years storage time (fig.2), it is important to assess how much extended burnup affects radioactivity to be managed in the fuel cycle. This paper takes, as an example, a 50 % increase of discharge burnup of PWR fuel, from 32÷33 GWd/tU to 48-50 GWd/tU.

Reference policy of the concerned countries.	Installed capacity		
	World	of which USA	France
Open cycle	52 %	3I %	-
Closed cycle	45 %	-	17 %
Indefinite interim storage	17 %	-	-

FIG. 1. Back end options, based on [1] and expressed as percentages of worldwide installed nuclear capacity, as of December 1991 (the percentages do not add up to 100% since several countries have selected indefinite interim storage for part of their spent fuel to provide a delay for selecting the most appropriate back end option).

2. Spent fuel activities.

For such 50 % increase in burnup, the radioactivity of fission products (fig.3) in the spent fuel (beta and associated gamma activity) is increased by 36 % per kg U(fig.4) and hence decreased by 9 % per kWh generated. To the contrary, the radioactivity of actinides (fig.4) in the spent fuel (alpha and associated gamma and neutron activities and heat generation) is increased by 250 % per kg U and hence by 140 % per kWh generated. Proper management of the actinides becomes therefore an important issue. For instance, it is unclear whether and to what extent the high alpha and neutron activity would deteriorate the cladding during very long term storage of high burnup spent fuel.

Reprocessing and recycling of the U and the Pu as fuel takes care of the largest part of the actinides stream. The increasing quantities of Cm and Am in the spent fuel justify the developments being contemplated to implement specific management schemes for those products (isolation and transmutation or fission).

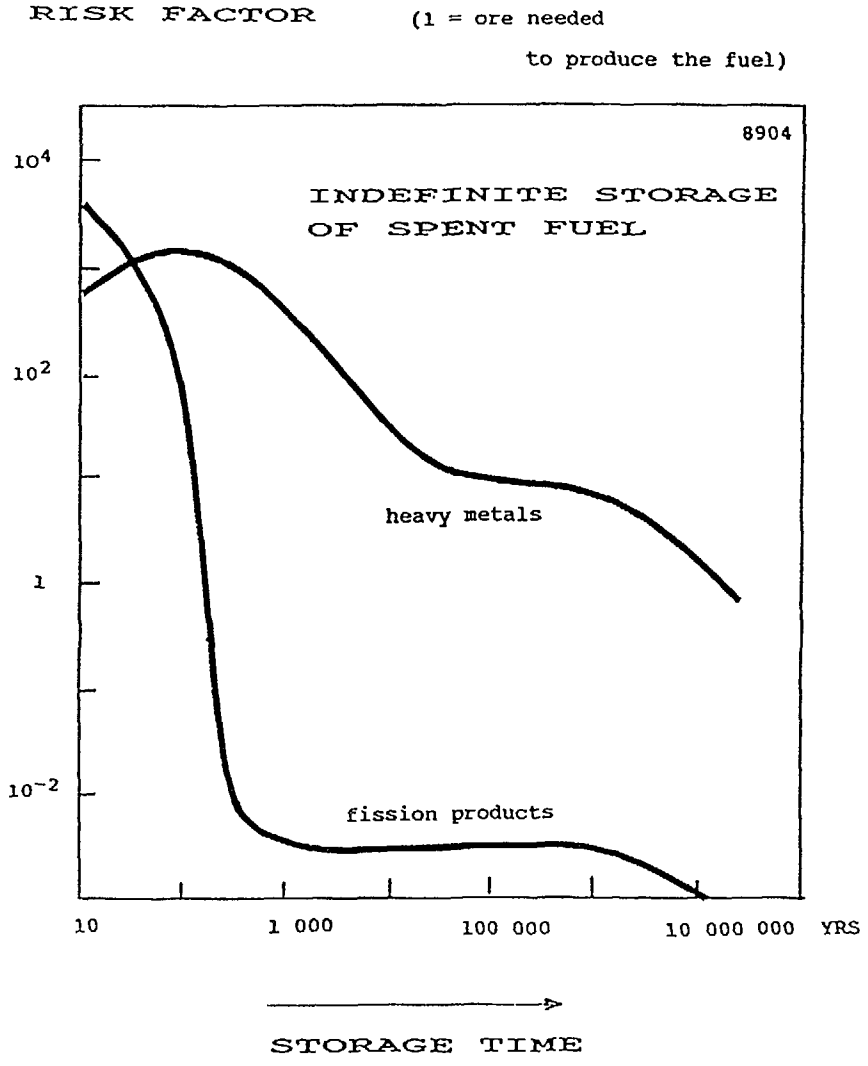

FIG. 2. Evolution of the two predominant components of the risk.

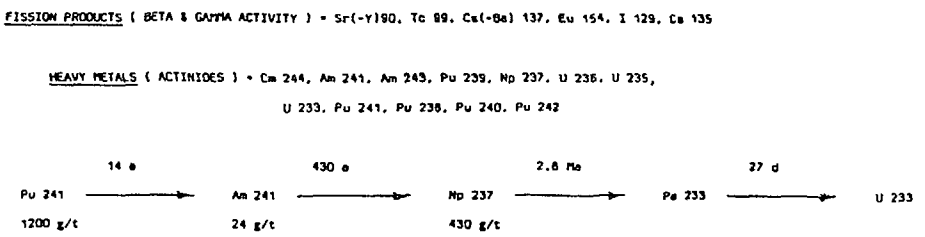

FIG. 3. The most important isotopes with long half-lives.

	PERIOD	g/tHM		Ci/tHM	
		33 GWd/t	50 GWd/t	33 GWd/t	50 GWd/t
90 Sr-Y	28 a +	480	590	140 000	170 000
99 Tc	210 ka	900	1200	15	21
135 Cs	2.9 Ma	1100	1700	1	1.5
137 Cs-Ba	30 a +	1100	1700	100 000	150 000
154 Eu	16 a	4.3	8	5 800	11 000
FPs	10 a	3600	5200	250 000	340 000
237 Np	2.8 Ma			0.3	0.4
Pu (α)	7-380 ka	9000	11000	3 800	8 000
241 Am	450 a			800	3 000
244 Cm	18 a			1 900	12 000
ACTINIDES	10 a			6 500	23 000

FIG. 4. Inventory of long life fission products and actinides.

3. Spent fuel storage and final disposal.

For a 36 % increased discharge burnup, the residual heat of spent fuel is increased by 26 % after a cooling time of 1 yr and by 49 % after 5 yrs (2). Since the decay heat is one limiting feature for spent fuel transportation, the storage period in the spent fuel pools at the reactor site might have to be increased considerably. For instance, taking the decay heat generation versus cooling time per assembly from the presentation (3), nl for an increase of discharge burnup of 67 %, the decay period must be extended from 5 yrs to 14 yrs if the admissible heat load is 800 W per fuel assembly, or from 4 yrs to 8 yrs if it becomes 1000 W per fuel assembly. Consequently, in this example, the at-reactor-site storage capacity expressed in number of years of reactor operation could be reduced by respectively 60 % or 17 %. It might become limiting in power plants having low to moderate spent fuel capacity.

The large increase of residual heat per kg U (250 % in the example of fig.4) becomes also a limiting feature for direct disposal of spent fuel at waste sites where both package weights and heat generation are key issues (e.g. clay formations). It could ultimately lead to the necessity of separating the Pu and actinides from the bulk of the weight of spent fuel.

4. Spent fuel storage and reprocessing.

Based on a past publication (4) and on a more recent one (2) and taking as a unit, the spent fuel corresponding to a same quantity of electricity generated (fig.5), extended burnup affects negatively spent fuel transport and the solvent extraction step at the reprocessing plant. All the other operations are drawing benefits from burnup extension ; a.o. the high level vitrified waste quantity decreases per kWh generated.

5. Reprocessed uranium.

Rep U originating from extended burnup fuel (fig.6) has a 20 % higher reactivity worth than Rep U from standard burnup fuel (i.e. its value as a fuel is not affected),

Operation	Attribute	(4)	(2)
Transport & storage	number of FAs	- 31 %	- 30 %
	BOL enrichment	+ 28 %	+ 8 to 29 %
	n/sec after 3 yrs	+120 %	+ 140 %
Shearing & dissolution	number of FAs	- 31 %	- 30 %
	Ru-Rh 106 after 3 yrs	- 16 %	- 50 %
Solvent extraction	Pu	- 16 %	- 16 %
	Cm	+110 %	+ 140 %
	activity	+ 40 %	+ 19 %
Pu conversion	heat generation	- 6 %	+ 3 %
HWL vitrification	glass	- 10 %	- 5 %

FIG. 5. Quantitative variation, per kW·h generated, of the attributes influencing reprocessing, when discharge burnup of the spent fuel increases from 33 to 47 GW·d/t U.

Burn-ups Cooling before reprocessing	34000 MWD/tU (INIT. U-235 : 3.25 %) 5 years		45000 MWD/tU (INIT. U-235 : 4.2 %) 5 years	
	Composition	Activity (Bq/gU)	Composition	Activity (Bq/gU)
U-232 (α)	0.82 ppb	660	2.49 ppb	1989
U-234 (α)	0.0128 %	29180	0.0191 %	43554
U-235 (α)	0.75 %	590	0.928 %	734
U-236 (α)	0.47 %	10670	0.583 %	10561
U-237 (β,γ)	$5.5\ 10^{-9}$ %	165800	$4.2\ 10^{-9}$ %	126826
U-238 (α)	98.77 %	12250	98.47 %	12213
Total α	100 %	53350	100 %	69051
Total β		165800		126826
U-232/U-235	0.109 ppm		0.27 ppm	

FIG. 6. Reprocessed uranium composition and activity.

GWd/tU	33				43	45			53
Pu age	EOL	0	2 yrs.	6 yrs.	EOL	0	2 yrs.	6 yrs.	EOL
Pu 236	$12X10^{-6}$	nd	$3.4X10^{-6}$	nd	$15X10^{-6}$	nd	$4.8X10^{-6}$	$1.9X10^{-6}$	$20X10^{-6}$
Pu 238	1.3	1.8	1.8	1.7	2.0	2.8	2.8	2.7	2.7
Pu 239	56.5	58.2	58.2	58.2	52.6	55.3	55.3	55.3	50.4
Pu 240	23.2	22.7	22.7	22.7	24.1	23.1	23.1	23.1	24.2
Pu 241	13.9	12.2	11.1	9.2	14.7	12.6	11.5	9.5	15.2
Pu 242	4.7	5.0	5.0	5.0	6.2	6.3	6.3	6.3	7
Am 241	0.35	-	1.1	3.0	0.44	-	1.1	3.1	0.51
Reference	(5)	(7)	(7)	(7)	(5)	(7)	(7)	(8)	(5)

FIG. 9 Pu isotopic composition (percentage by weight) as a function of burnup of the original spent fuel (GW-d/t U) and of Pu age (storage time after reprocessing). EOL is the final discharge from the reactor core, i.e. some years before reprocessing.

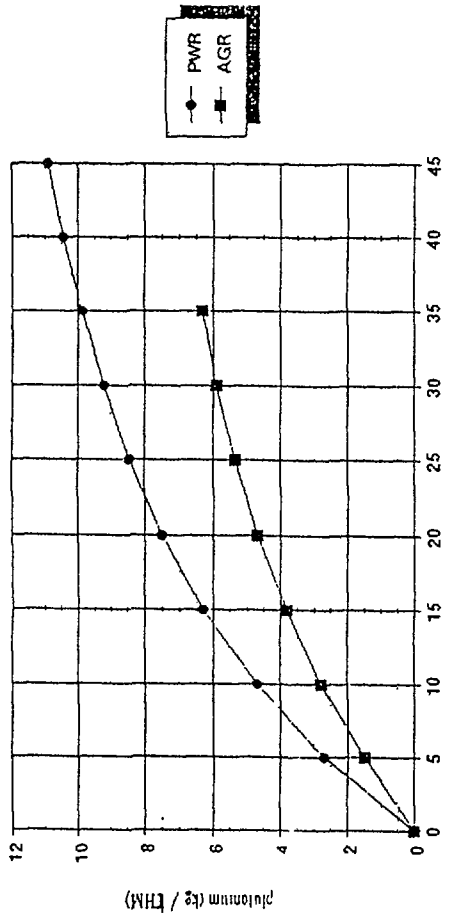

FIG. 7. Specific mass of plutonium as a function of fuel burn-up.

ALPHA : Pu 238 + Pu 239 + Pu 240 + Am 241

GAMMA : Pu 236 decay products + Pu 238 + Am 241

NEUTRONS : alpha - n reaction + sponaneous fission

(Pu 238 + Pu 240 + Pu 242 + Am 241)

FIG. 8. Pu radioactivity

ACTIVITY	FACTOR
ALPHA	1.8
HEAT GENERATION	1.8
GAMMA < 200 keV	1.5
" 0.2 - 1 MeV	1.1
" > 1 MeV	1.6
NEUTRON	1:5

FIG. 10. Increase of Pu radioactivity when spent U fuel burnup is increased from 33 to 48 GW·d/t U.

	Year	Plutonium content (approx.)	Equivalent U-235 content (approx.)	Average reload burnup (MWd/tHM)	In-reactor life (years)
	1990	5 %	3.25 %	36 000	3
	1995	7.5 %	3.7 %	42 000	4
>	2000	10 %	4.5-5 %	55-60 000	5

FIG. 11. Expected improvements in MOX fuel performance [9].

	U FUEL	MOX FUEL	Rep U FUEL
1980's	3.2 % U5 32 GWd/tHM	COMMERCIAL Pu ex 30 GWd/tU 5 % Put 33 GWd/tHM	SMALL SCALE DEMO
1990's	4.1 % U5 45 GWd/tHM	Pu ex 30 GWd/tU 7 % Put 45 GWd/tHM	LARGE SCALE DEMO U ex 30 GWd/tU 5 % U5 45 GWd/tHM
2000's	5 % U5 60 GWd/tHM	Pu ex 45 GWd/tU 8 % Put 60 GWd/tHM	COMMERCIAL U ex 30 GWd/tU 5.5 % U5 60 GWd/tHM
2010's	SAME	Pu ex 60 GWd/tU + 30 & 45 GWd/tHM 10 % Put	U ex 45 GWd/tU 6 % U5
2020's	SAME	Pu ex 60 GWd/tU + 45 & 60 GWd/tHM 12 % Put	U ex 60 GWd/tU 7 % U5

FIG. 12. Extended burnup implementation time scales.

based on data presented earlier (5). On the basis of more recent data (4), the reactivity worth is decreased by 20 %. The radioactivity increase (fig.6) is mainly due to the hard gammas emitted by daughter products of U 232, which form non-volatile fluorides. As a result, the inconvenience is essentially limited to the enrichment plant and to the hexafluoride transport containers, in which those non-volatile decay products are being deposited. They must be decontaminated more frequently thereby generating additional process waste.

6. Plutonium.

The net quantity of Pu produced decreases, as part of the generated Pu is burned in situ (fig.7). The radioactivity of Pu (fig.8) increases however sharply (fig.9), due to the higher proportion of alpha and neutron emitting isotopes (fig.10) and of hard gamma activity of the daughter products of Pu 236. The challenge to the MOX fabrication plant will not only result from this increased specific radioactivity, but also from the higher Pu contents required for extended burnup MOX fuel (fig.11).

These higher Pu contents and the resulting neutron spectrum hardenig decrease the effect of the capture resonances of U 238. It results in the void coefficient becoming less negative and even ultimately positive , a situation which requires an extension of the experimental data base (10).

The actual perspective to be faced is however much more progressive than what might be deduced from equilibrium cycle values (e.g. fig.11) : Pu to be incorporated in MOX fuel of any target discharge burnup will, for many decades, be Pu from the previous generation of discharge burnups (fig.12). This will provide timely opportunity for the industry to collect an adequate data base and to implement appropriate technical solutions to this challenging evolution of feed Pu and MOX fuel to be fabricated, on the grounds of lessons progressively learned from commercial operation of the fabrication and of power plants with MOX fuels.

Finally, it is not unlikely that the nuclear industry will have to utilize, in MOX fuel, Pu arising from the dismantling of nuclear warheads as the most efficient way to demilitarize that Pu. Mixing it to Pu arising from high burnup spent fuel will result in completely annihilating the increased radioactivity of the latter.

DIFFERENCES	APs	FPs	ACTINIDES
> 30 %	Zr 93	Mo 96 Xe 128 Sm 147 Eu 156	Pu 239
> 20 %		Cs 135	Cm 244

FIG. 13. Predictive capability of activation products, fission products and actinides content of fuel at high burnup (examples).

7. Final remark.

Altogether, the nuclear industry and R & D institutes are taking the necessary steps or preparing the appropriate initiatives to cope in due time with the back-end consequences of extending discharge burnups.

It is, however, observed (fig.13) that large differences exist in predicting the quantity of each isotope in high burnup spent fuel. This might mislead the developpers in preparing for the future. Opportunity should be taken of available high burnup U and MOX fuels to build, at once, an experimental data base, on which predictive codes can be benchmarked for the future burnup ranges of interest.

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based on data presented earlier (5). On the basis of more recent data (4), the reactivity worth is decreased by 20 %. The radioactivity increase (fig.6) is mainly due to the hard gammas emitted by daughter products of U 232, which form non-volatile fluorides. As a result, the inconvenience is essentially limited to the enrichment plant and to the hexafluoride transport containers, in which those non-volatile decay products are being deposited. They must be decontaminated more frequently thereby generating additional process waste.

6. Plutonium.

The net quantity of Pu produced decreases, as part of the generated Pu is burned in situ (fig.7). The radioactivity of Pu (fig.8) increases however sharply (fig.9), due to the higher proportion of alpha and neutron emitting isotopes (fig.10) and of hard gamma activity of the daughter products of Pu 236. The challenge to the MOX fabrication plant will not only result from this increased specific radioactivity, but also from the higher Pu contents required for extended burnup MOX fuel (fig.11).

These higher Pu contents and the resulting neutron spectrum hardenig decrease the effect of the capture resonances of U 238. It results in the void coefficient becoming less negative and even ultimately positive , a situation which requires an extension of the experimental data base (10).

The actual perspective to be faced is however much more progressive than what might be deduced from equilibrium cycle values (e.g. fig.11) ; Pu to be incorporated in MOX fuel of any target discharge burnup will, for many decades, be Pu from the previous generation of discharge burnups (fig.12). This will provide timely opportunity for the industry to collect an adequate data base and to implement appropriate technical solutions to this challenging evolution of feed Pu and MOX fuel to be fabricated, on the grounds of lessons progressively learned from commercial operation of the fabrication and of power plants with MOX fuels.

Finally, it is not unlikely that the nuclear industry will have to utilize, in MOX fuel, Pu arising from the dismantling of nuclear warheads as the most efficient way to demilitarize that Pu. Mixing it to Pu arising from high burnup spent fuel will result in completely annihilating the increased radioactivity of the latter.

DIFFERENCES	APs	FPs	ACTINIDES
> 30 %	Zr 93	Mo 96 Xe 128 Sm 147 Eu 156	Pu 239
> 20 %		Cs 135	Cm 244

FIG. 13. Predictive capability of activation products, fission products and actinides content of fuel at high burnup (examples).

7. Final remark.

Altogether, the nuclear industry and R & D institutes are taking the necessary steps or preparing the appropriate initiatives to cope in due time with the back-end consequences of extending discharge burnups.

It is, however, observed (fig.13) that large differences exist in predicting the quantity of each isotope in high burnup spent fuel. This might mislead the developpers in preparing for the future. Opportunity should be taken of available high burnup U and MOX fuels to build, at once, an experimental data base, on which predictive codes can be benchmarked for the future burnup ranges of interest.

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