

# ZIRAT-8 SPECIAL TOPICS REPORT

## High Burnup Fuel Issues

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## FOREWORD

One of the major present challenges to nuclear energy lies in its competitiveness. To stay competitive the industry needs to reduce maintenance and fuel cycle costs, while enhancing safety features. Extended burnup is one of the methods applied to meet these objectives. There are a number of issues that need resolution to be able to successfully implement extended burnup. This special topic has been prepared within the ZIRAT-8 program and reviews the potential consequences of increased burnup on fuel.

Different section of this report has been prepared by the following authors: Ron Adamson, Friedrich Garzarolli, Peter Rudling, Al Strasser and Gunnar Wikmark. However, the whole report has been reviewed by all the authors as well as by Brian Cox. Thus, this report provides an unbiased and independent view of the current high burnup issues today by experts with vast experience in the fuel technology area.

Throughout the report, references are given to previous Special Topical Reports furnished in the ZIRAT-4 to –7 Programs, that are available to ZIRAT-customers, and they are:

- “CORROSION OF ZIRCONIUM ALLOYS” by Ron Adamson, Brian Cox, Friedrich Garzarolli, Peter Rudling, Al Strasser, Gunnar Wikmark, 2002.
- “DIMENSIONAL STABILITY OF ZIRCONIUM ALLOYS”, by Ron Adamson and, Peter Rudling, 2002.
- “WATER CHEMISTRY AND CRUD INFLUENCE ON CLADDING CORROSION”, by Gunnar Wikmark and, Brian Cox, 2001.
- “MECHANICAL PROPERTIES OF ZIRCONIUM ALLOYS”, by Ron Adamson and, Peter Rudling, 2001.
- “MANUFACTURING OF ZIRCONIUM ALLOY MATERIALS”, by Peter Rudling and, Ron Adamson, 2000.
- “HYDRIDING MECHANISMS AND IMPACT ON FUEL PERFORMANCE”, by Peter Rudling and, Brian Cox, 2000.

*Peter Rudling, Editor*

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## 1 INTRODUCTION (AL STRASSER)

### 1.1 DEFINITION

“Burnup” is defined as the total energy released per initial unit mass of fuel as a result of irradiation. Commonly used units are megawatt days per metric ton (MWd/MT) of fuel or the same in gigawatt days (GWd/MT). The term “fuel” can be ambiguous unless defined as initial uranium (U) metal, or  $\text{UO}_2$ , or their combination with plutonium (Pu) such as U+Pu or  $\text{UO}_2 + \text{PuO}_2$ . The most commonly used unit is initial U metal (or U+Pu for MOX) expressed as MWd/MTU.

The volume and location of the fuel need definition as well and can include average or peak values for a batch, an assembly, a rod, or a pellet. As examples, the economic analyses for extended burnup, as well as most vendors’ mechanical warranties are based on batch average burnups. However, the thermal-mechanical and nuclear limits are related to assembly and rod burnups.

In-reactor experiments, such as ramp tests, are evaluated relative to the peak burnup and peak power locations in the fuel rod and thermal mechanical analysis codes model the performance of the highest burnup and highest power locations within the rod to evaluate the most severe performance expected.

To carry this further to a microscopic level, the interpretation of the performance of the pellet rim regions that form at high burnups requires an estimate of the local burnup in the rim region. The same applies to inhomogeneities in MOX pellets formed by high Pu content particles in the MOX matrix that result in high local burnups and fission gas production on a microscopic scale.

Typical relationships of assembly, rod and pellet burnups to the batch average burnups are shown in Figure 1-1. *The differences in burnup definitions should be kept in mind as the reader proceeds through the Special Topic Report.*



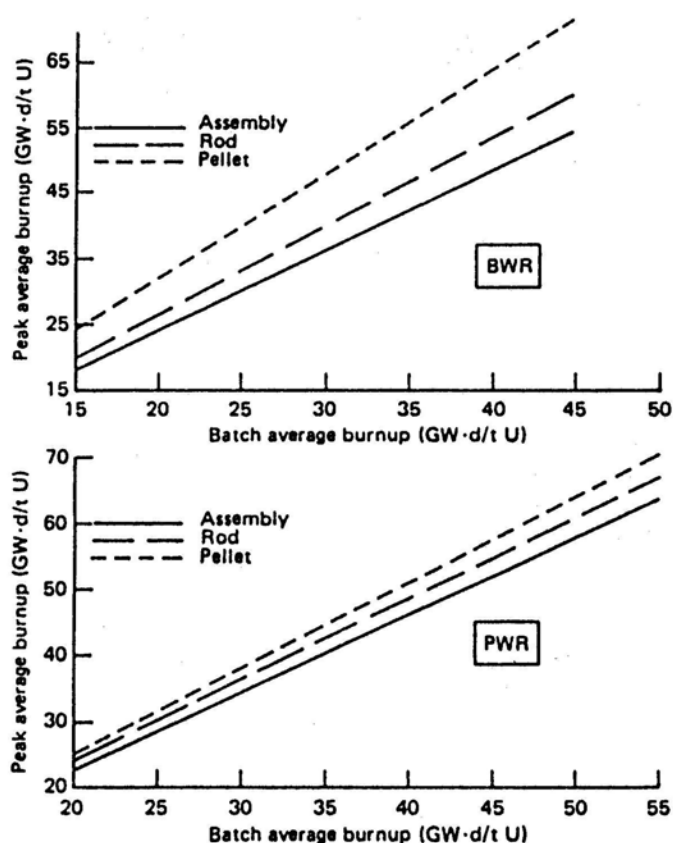


Figure 1-1: Relationship of Assembly, Rod and Pellet Burnup to Batch Average Burnup, Strasser, et al., 1986.

## 1.2 INCENTIVES

### 1.2.1 Overview

The incentives for operating fuel to high, or extended burnups are listed below. The list represents the incentives that existed in the early days of the nuclear industry, and while most of the incentives are still valid the value of and the emphasis on each one is slowly changing with time and with the modified nature of the fuel cycle as discussed subsequently.

1. Economics --- lower fuel cycle costs,
2. Capability for longer cycles --- increased capacity factors, decreased radiation doses,
3. Improved resource utilization --- decreased amount of uranium, Separative Work Units (SWU) and fuel assemblies per unit energy produced,
4. Increased margin to storage capacity limits,
5. Eventual decreased offsite shipping and storage costs.

Added incentives in European countries (and in the future in Japan) favored decreased number of fuel assemblies for reprocessing and refabrication due to,

1. High back end cost of reprocessing,
2. Reality of Pu recycle and its high fabrication cost.

The first major change in the fuel cycle in the US was the elimination of Pu recycling under the Carter Administration. This was a correct decision for the wrong reasons: recycling was stopped for fear of proliferation rather than its uneconomical nature without fast breeder reactors as part of the fuel cycle. Interestingly the federal restriction on reprocessing has been lifted since the Reagan administration but not implemented. The inertia of existing successful reprocessing plants and plutonium fuel fabrication plants in Europe has continued recycling in Europe; however, rising reprocessing costs have slowed this down there as well. Even the major client for mixed U+Pu oxide fuel (MOX), the French power industry, has decided to limit reprocessing to one recycle followed by permanent storage.

However, the fuel recycle economy was eliminated in the US and is fading away in Europe, on both continents, to a non-existent permanent spent fuel storage economy via an intermediate on site, or in some cases off-site, storage facility. *The inability to send fuel for reprocessing or to a permanent storage site has caused a spent fuel assembly log-jam in the spent fuel pools and eliminated the High Burnup Incentive #4, "Increased Margin to Storage Capacity".*

Longer fuel cycles, 18 to 24 months in duration, have been implemented in almost all the US plants in conjunction with increased fuel burnups for some time now. The implementation of long cycles is starting in Europe now as well.

The major economic gain of long cycles is due to the increased capacity factor, the result of decreased refueling times. Historically the US plants had relatively long refueling shutdowns and European plants relatively short ones, so that the economic gain due to long cycles was greater in the US. More recently, the US refueling operations have become more efficient and the economic gains due to long cycles are at a lower level as a result. Maintenance operations needed during the 24-month periods, also limit capacity factor gains. As 24 month cycles are approached the economics become plant specific --- economically justified in some plants and not at others --- so that the *economic gains due to longer cycles facilitated by extended burnups have been taken advantage of to a large degree by the current cycle lengths and burnup levels.*

The improved utilization of uranium (U) was a significant factor in the early days of very high U costs. The current plentiful supply at low price levels decreases the potential economic gains due to high burnup significantly. The cost of SWU for enrichment has also become more competitive since the DOE monopoly was broken. The decrease in the number of fuel assemblies required for the energy generated can be a significant saving, although the fabrication cost of the high burnup assembly designs balances this to some degree. Nevertheless, the highly competitive nature of the fuel fabrication business has kept the prices down.

The cost of shipping a decreased number of spent fuel elements will clearly save some money. The cost of intermediate storage and perhaps costs involved in final storage are a current area of uncertainty. *The significantly increased time required for high burnup fuel to decrease its decay heat in a spent fuel pool before it can be loaded into an intermediate dry storage cask and the unknown schedule for shipping the fuel from the dry cask to a permanent storage site prevents a reliable estimate for the capacity and cost required for the intermediate wet and dry storage facilities.* This is discussed in greater detail in Section 1.2.2 “Economics” and Section 6 “Spent Fuel Storage”.

## **1.2.2 Economics, Fuel Cycle Length, Resource Utilization,**

### *1.2.2.1 Overview*

The fuel cycle cost components for a permanent storage, back end cycle are:

- Natural uranium concentrate ( $U_3O_8$  equivalent)
- Conversion of the oxide to uranium hexafluoride ( $UF_6$ )
- Fabrication of fuel assemblies
- Operation of the plant to generate electricity
- Refueling of the reactor
- Wet and dry at-reactor storage of spent fuel
- Transportation to and storage at an intermediate site (optional)
- Transportation to a permanent storage site (if applicable)
- Carrying charges and present worth of money

Cost components for a reprocessing cycle are the same through wet storage and then replace the remaining steps above by:

- Transportation to the reprocessing plant
- Reprocessing of the fuel assemblies
- Transportation of the Pu to a fabrication plant and radioactive waste to a disposal site
- Radioactive waste disposal
- Carrying charges and present worth of money

After the last Pu recycle (currently only one is envisioned) the cost of permanent fuel assembly storage has to be added, probably starting with dry storage until a permanent site becomes available.

Early studies on the permanent storage cycle evaluated the effect of burnup and cycle length on economics. They concluded that the optimum burnup for 12 month cycles was between 40 and 50 GWD/MTU and for 18 month cycles the fuel cycle costs continued to decline up to 55 GWD/MTU, the highest burnup considered, Franks, et al., 1984 and Franks and Geller., 1986. The study was based on a Westinghouse 193 assembly plant. Since then, all US PWRs have gone to at least 18 month cycles to take advantage of its economics.

A study by EdF for their PWRs using input data from 1990 and based on reprocessing indicated a minimum fuel cycle cost between 50 and 60 GWD/MTU for 12 month cycles and for enrichments <5%, Demoulins, et al., 1993. Their results are shown in Figure 1-2.

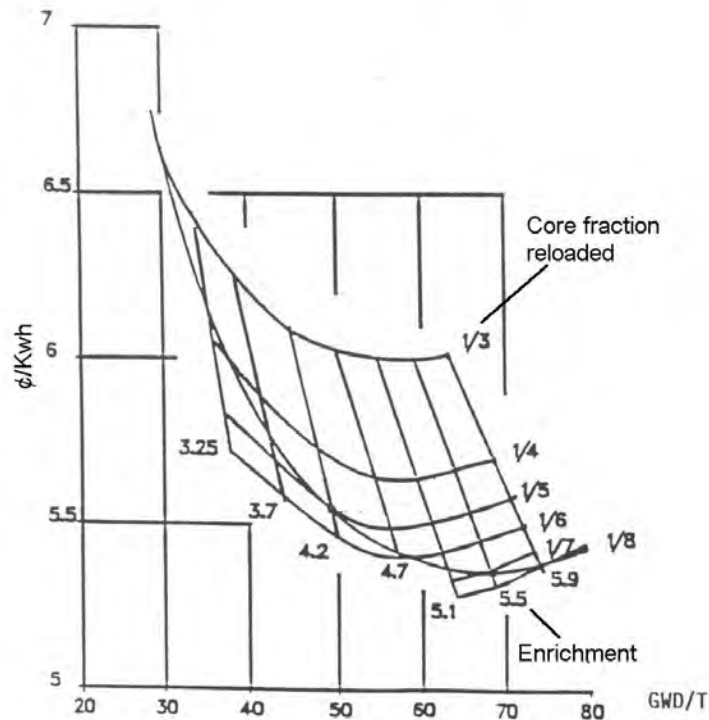


Figure 1-2: Fuel Cycle Cost Versus Burnup, Demoulins, et al., 1993.

A later study compared 12 and 24 month cycles and found that costs decreased to burnups as high as 60 GWD/MTU for 24 month PWR cycles, OECD, 1994. The study assumed that the quantity of fuel disposed was independent of burnup.

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A study to determine the optimum burnup for 18 month PWR cycles was made by Duke Power, EPRI TR-112571, 1999. A range of 60 to 70 GWD/MTU was concluded; however, enrichments >5% were required to achieve these levels.

A more recent study of 18 month cycle PWR fuel costs as a function of burnup indicates a relatively flat response with a broad range of minima between 55 and 70 GWD/MTU shown in Figure 1-3, Kazimi, et al., 2003. The work is based on a doctoral thesis at MIT and the input data used are not available yet. The decreasing storage costs with burnup appear to be unrealistic in view of the longer wet storage times required to meet dry storage thermal limits, a topic discussed subsequently.

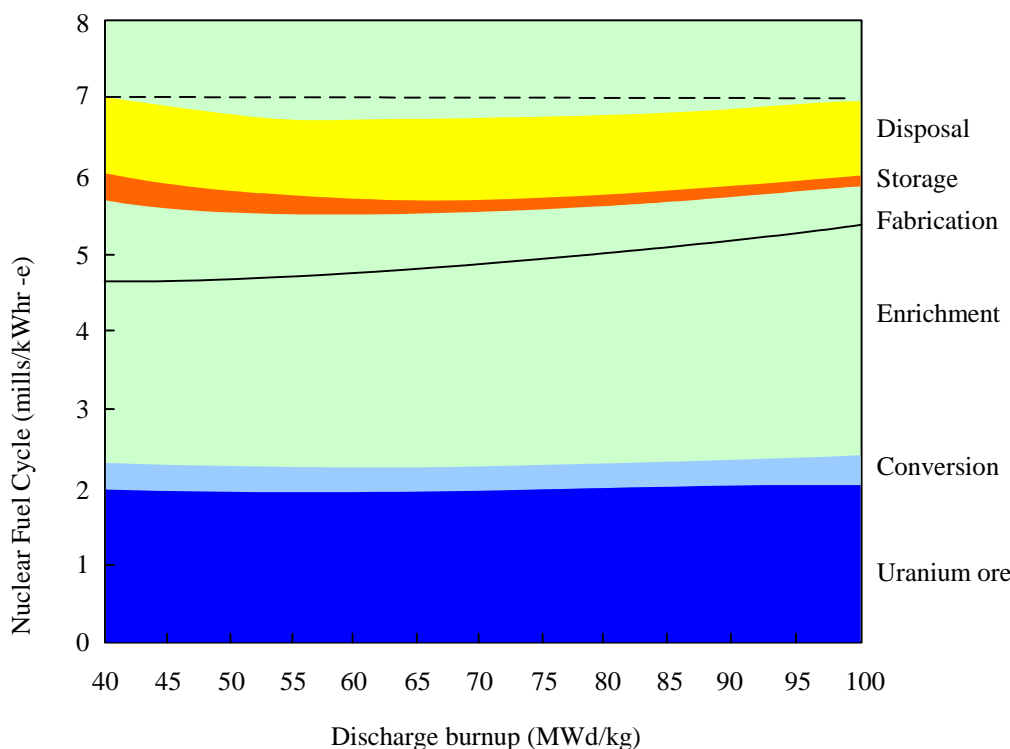


Figure 1-3: Fuel Cycle Economics vs. Burnup for 18 Month PWR Cycle, modified figure according to Kazimi, et al., 2003.

The most recent, detailed fuel cycle cost study, sponsored by the USDOE and EPRI, considers the effect of burnup below the 5% enrichment limit (Phase I study) as well as at levels above 5% enrichment (Phase II study) for both BWRs and PWRs, Secker, et al., 2001 and 2002). The major input assumptions, sensitivity studies and conclusions are summarized here. The reports provide much additional detail.

The cost and economic parameters used in the studies are given in Table 1-1 and the fuel cycle data that impact the economics in Table 1-2. The “low, medium, high” data represent the values used for sensitivity studies.

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1.2.2.2 BWRs with <5% Enrichment

Analyses were performed for a large GE NSSS with a 768 assembly core for 12, 18 and 24 month equilibrium cycle lengths and a small GE NSSS with a 368 assembly core with both GE 13 (9x9) and GE 14 (10x10) fuel. The maximum enrichment used was 4.9% and the standard reactivity margins were met. The nuclear design of the fuel is described in Section 1.4.2.

Examples of the batch average burnups as a function of core enrichments are shown in Table 1-3. The highest core enrichment of 4.3% is lower than its 4.9% maximum enrichment fuel rods, as a result of the lower enrichment gadolinia containing rods and the natural uranium blankets. The peak rod burnups were kept to the licensing limit of 62 GWD/MTU in all cases except for the highest burnup 18 and 24 month cycles where lead rods were permitted to exceed this limit. The difficulty in staying within the current licensing limit with lead rods was compromised, because of the current efforts --- hoped to be successful --- for raising the limit.

Table 1-1: Unit Prices and Economic Parameters, Secker, et al., 2001.

Component	Low	Nominal	High
Uranium (\$/kg U)	18.30	31.35	47.00
Conversion (\$/kg U)	3.50	5.00	7.00
Enrichment (\$/SWU)	80	105	130
Fabrication (\$/kg U) BWR	255	275	350
Fabrication (\$/kg U) PWR	185	210	275
Spent fuel storage (\$/Assembly) BWR	0	20,000	40,000
Spent fuel storage (\$/Assembly) PWR	0	50,000	100,000
Escalation rate for spent fuel storage (% per year)	2	3	5
Discount Rate (% per year)	7	9.5	12

Table 1-2: Fuel Cycle Data, Secker, et al., 2001.

Monetary base time year	2001
Lead time of uranium purchase	10 weeks
Lead time for conversion	8 weeks
Lead time for enrichment	4 weeks
Lead time for fabrication	2 weeks
Lag time for spent fuel storage	0 days
Loss factor for conversion	0.5 w/o
Loss factor for fabrication	0.0 w/o
Tail Assays ratio	0.3 w/o

Table 1-3: Batch Average Discharge Burnup Data as a Function of Enrichment for GE-14 Fuel – Large BWR, Secker, et al., 2001.

Enrichment (w/o)	Batch average discharge burnup (MWD/MTU)		
	12 Month	18 Month	24 Month
3.0	39407	37740	34517
3.1	40368	38905	35607
3.2	41638	40145	36668
3.3	42712	41129	37901
3.4	43844	42517	39106
3.5	44733	43811	40390
3.6	45658	44984	41633
3.7	46953	46222	42954
3.8	48324	47531	44218
3.9	49406	48679	45712
4.0	50926	50132	46981
4.1	51722	51674	48497
4.2	52963	53034	50113
4.3	54714	54468	51644

The fuel costs for a large BWR with GE-14 fuel as a function of burnup and cycle length are shown in Figure 1-4. The conclusions for all cases can be summarized as follows:

- Fuel costs decline at varying degrees with increased burnup for all cases,
- Fuel cost savings increase with increased cycle lengths and extended burnups; the 12 month cycle with the lowest savings is close to zero, assuming the shutdown time is the same for any length cycle.
- Fuel cost/kwhr increases with cycle length as one would expect,
- Fuel cost savings are significantly lower for a small BWR compared to a large one,
- Fuel cost savings as a function of burnup are very similar between GE-13 and –14.

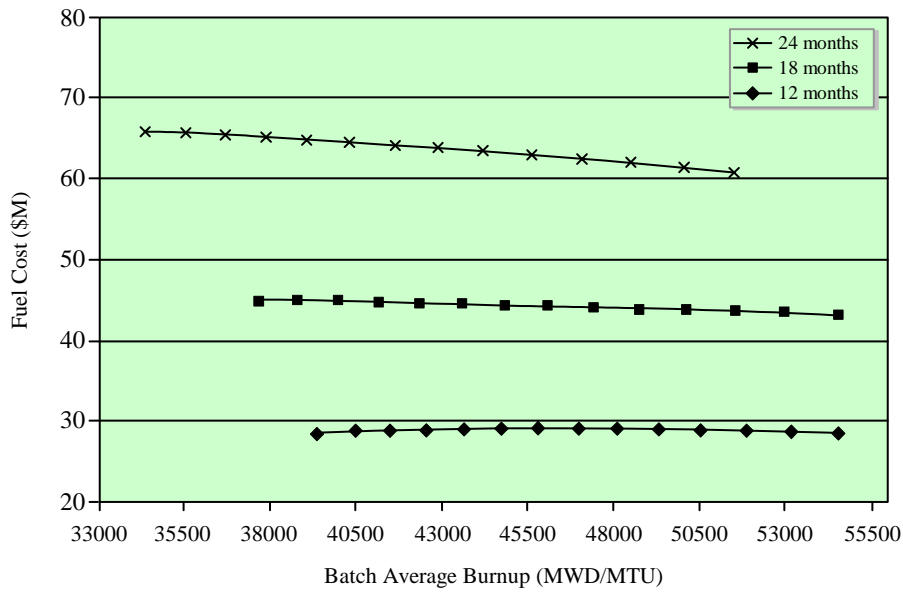


Figure 1-4: Large BWR GE-14 Fuel Cost for Different Cycle Lengths Nominal Parameters with 4.9% Max. Enrichment Rods, modified figure according to Secker, et al., 2001.

The results of the sensitivity studies of cost components as a function of burnup are given in Figure 1-5 for a large BWR with GE-14 and 18 month cycles. Variations in the fabrication and back-end costs have the largest effect on the fuel costs. The large savings for the “High Storage Cost” case do not appear to have considered the effect of the longer decay times required for high burnup fuel on storage cost (as in the case of the MIT cost analyses); consideration of this factor could affect the overall trend of the total fuel costs with burnup as discussed later.

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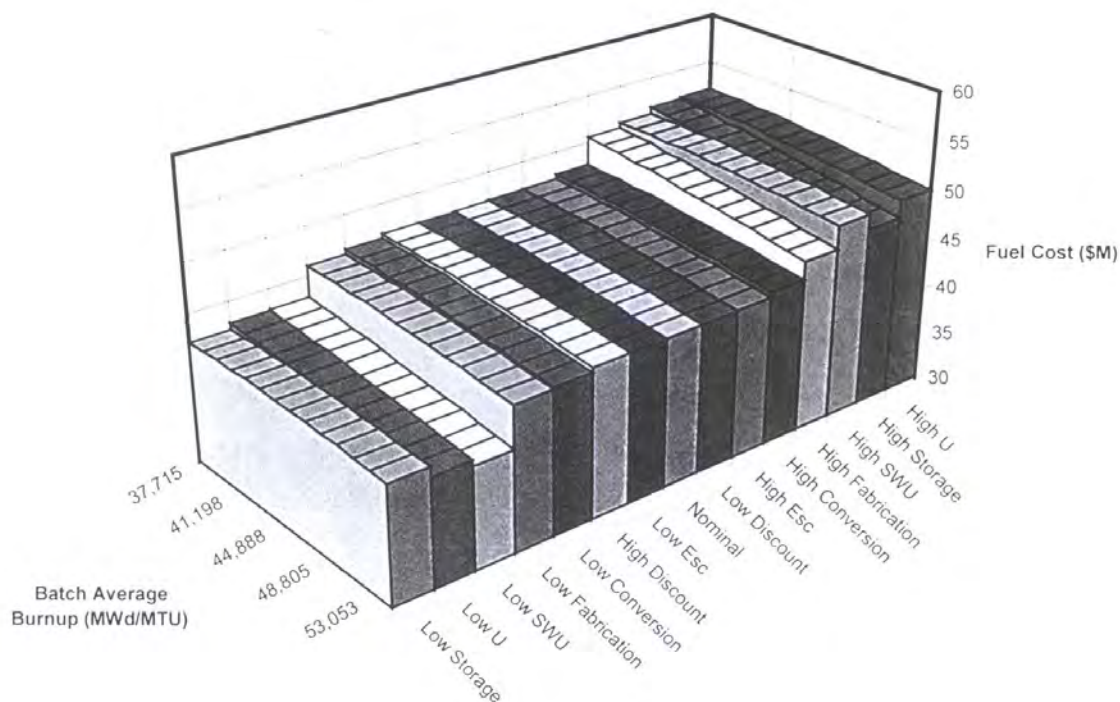


Figure 1-5: Comparison for Each Fuel Cycle Parameter Effect on Fuel Cycle Cost for Large BWR GE-14 with 18 Month Cycle and 4.9% Maximum Enrichment Rods, Secker, et al., 2001.

### 1.2.2.3 BWRs with >5% Enrichment

Analyses were performed by raising the maximum fuel rod enrichment to 5.95% and applying this to a large 764 assembly BWR fueled by Westinghouse SVEA-96 Optima2 (10x10) fuel. The effect of the enrichment increase on batch average burnup is shown in Table 1-4. The highest batch average burnup range of 63.2 – 65.2 GWD/MTU is equivalent to peak rod burnups of 75 – 77 GWD/MTU. The nuclear design of the fuel is described in Section 1.4.3 of this Report.

In order to determine whether the Phase I study (<5% enrichment) with the GE fuel designs can be compared to the Phase II study with the SVEA design, a SVEA reference design with 4.95% maximum enrichment rods was analyzed for a large BWR. The consistency between the SVEA and the GE-14 equilibrium core design results satisfied the authors that the results of the Phase II study can be compared to the results of Phase I.

The increased enrichment and reactivity case of Phase II, could not meet the shutdown margin and thermal margin limits of the Phase I Reference Case. In order to meet the same margins the fuel designs had to be modified and this was done by two different approaches. The “Optimized for Extended Enrichment” option was established by the combination of mechanical and nuclear design changes. The “Current Bundle Optimization” maintained the current mechanical design and was limited to nuclear design changes; this second option could not match the economic advantages of the first one. The specific design changes are discussed in Sections 1.4.2 and 1.4.3; the economic analysis results are given below.

The cost parameters and fuel cycle data were the same as those used in the Phase I study, shown in Table 1-1 and Table 1-2. Similar sensitivity studies were made as well. The economic analyses assume no change in cost due to enrichment, manufacturing, transportation, licensing, wet or dry storage and disposal when fuel >5% enrichment is used. Consequently, these analyses are a good start, but are incomplete for a realistic prediction of economic trends.

The fuel costs for a large BWR as a function of batch average burnup operating on 24 month cycles with two types Westinghouse SVA-96 Optima2 designs are shown on Figure 1-6.

Table 1-4: Batch Average Discharge Burnup Data as a Function of Enrichment for SVEA-96 Optima 2 Fuel – Large BWR, Secker, et al., 2002.

Maximum Nominal Enrichment (w/o U <sup>235</sup> )	Maximum Actual Enrichment (w/o U <sup>235</sup> )	Assembly average enrichment (including blankets) (w/o U <sup>235</sup> )	Enrichment Average over Non-blanket Enriched Zone (w/o U <sup>235</sup> )	Batch Average Discharge Burnup (MWD/MTU)	
				Optimized for Extended Enrichment	Current Bundle Optimization
5.0	4.95	4.25	4.56	52400	52400
5.5	5.45	4.67	5.02	59400	58400
6.0	5.95	5.10	5.48	65200	63200

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ZIRAT-8 Special Topic on High Burnup Fuel Issues

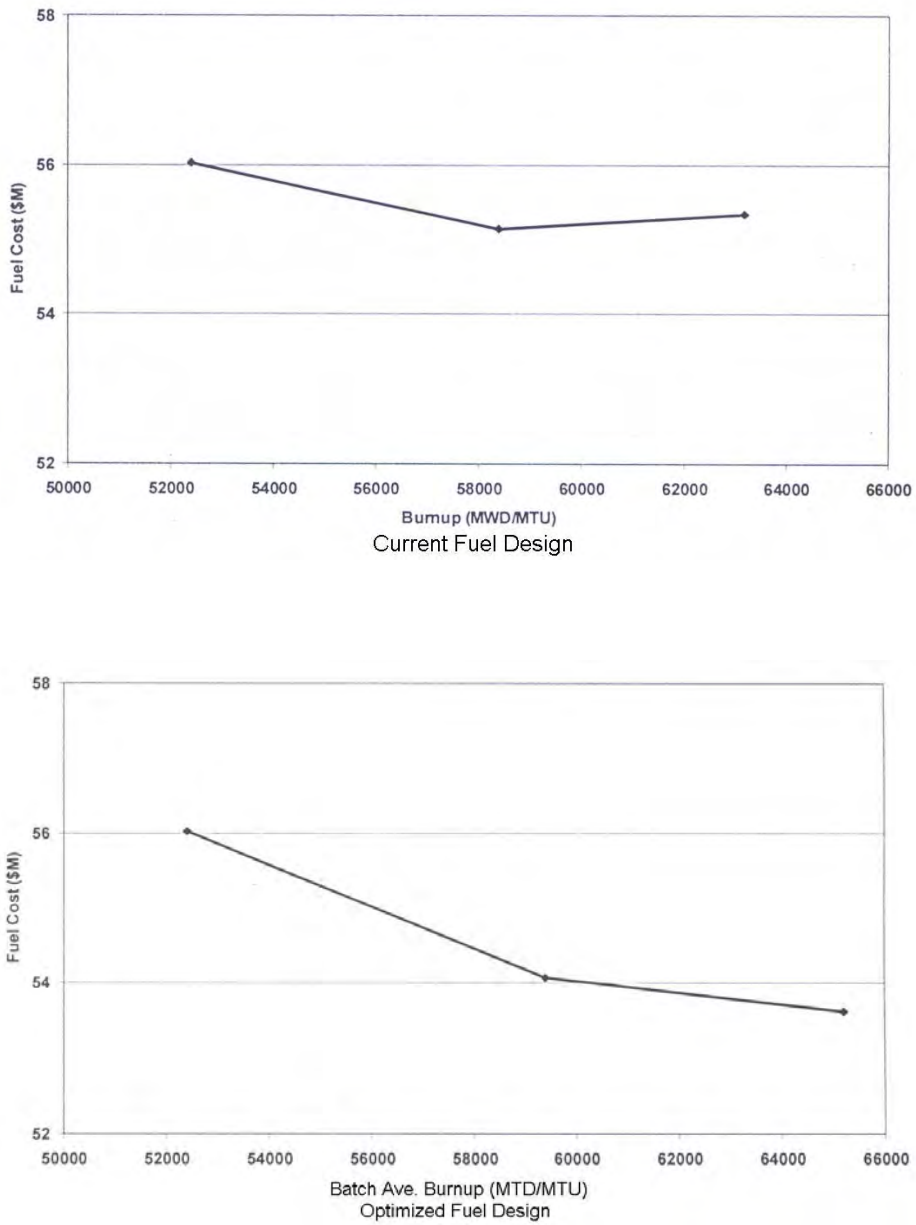


Figure 1-6: Fuel Cost for 24 Month Cycle with Westinghouse SVEA-96 Optima2 for Extended Enrichment Fuel Design, Secker, et al., 2001.

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For the “Optimized” design:

- Fuel costs decline with increased burnup leveling off at the higher values; the fresh fuel savings are more than the increase in uranium, conversion, and enrichment cost components due to a lower required fresh feed amount,
- Levelized fuel costs (not shown) followed the same trend,
- Mechanical design changes provided greater batch size savings than the “Current” design, permitting smaller fresh fuel batches to be used to reach the cycle energy requirements and resulting in lower fuel costs for the “Optimized” design,

For the “Current” design:

- Fuel costs decreased by increasing batch average burnup from 52.4 to 58.4 GWD/MTU, but then increased by further increasing burnup to 63.2 GWD/MTU. In the latter case the uranium cost components at the high enrichment levels outweighed the corresponding energy benefit, or stated differently the fresh fuel savings are less than the increase in uranium cost components required for the high burnup level.
- Levelized fuel costs (not shown) followed the same trend.

The results of the sensitivity analyses for the cost components as a function of burnup are shown for both the “optimized” and the “current” design on Figure 1-7. The comments made for the analyses at <5% enrichment apply to these as well. The reversal in economic trends can be observed for the “current” design, particularly for the high-end values of the uranium and fabrication cost components.

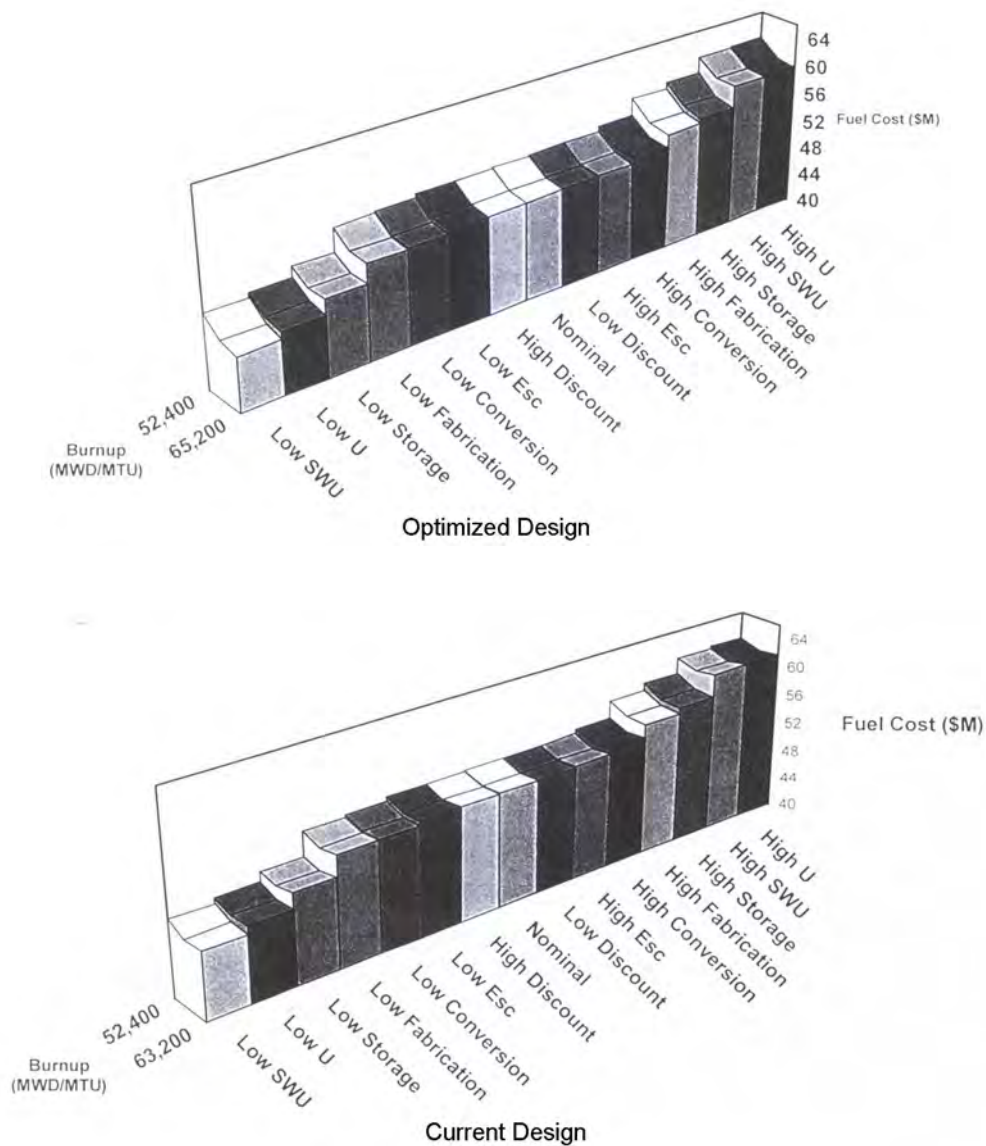


Figure 1-7: Comparison of Each Fuel Cycle Parameter Effect on Fuel Cycle Cost for Large BWR Westinghouse SVEA-96 Optima2 in a 24 Month C, Secker, et al., 2002.

A comparison of all the BWR cost analyses for 24 month cycles, below and above 5% enrichment, is shown on Figure 1-8.

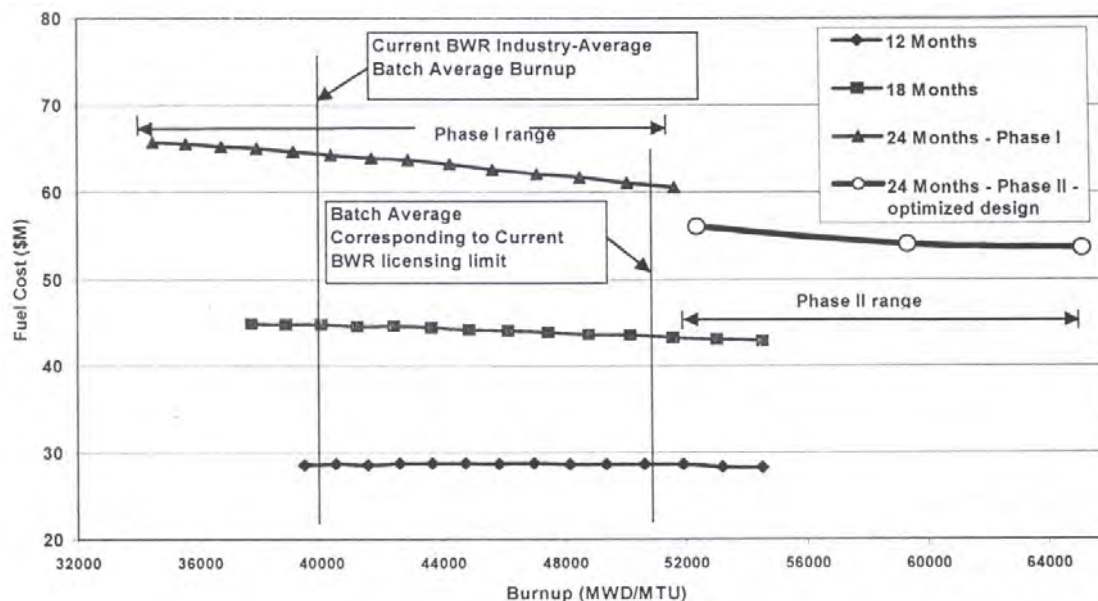


Figure 1-8: Fuel Cost as a Function of Discharge Burnup for 24 Month Cycle BWR, Secker, et al., 2002.

#### 1.2.2.4 PWRs with <5% Enrichment

Analyses were performed for a large 4 loop, 193 assembly core Westinghouse NSSS for 12, 18 and 24 month equilibrium cycles with both Westinghouse Performance+ and Robust Fuel Assembly (RFA) designs. The RFA design differs from the Performance+ by a larger diameter fuel rod and lower H/U ratio that reduces the number of fresh assemblies to be loaded within the same max. rod enrichment limit.

As in the BWR designs, the maximum enrichment rod was 4.9% and the additional cycle design limits are listed in Table 1-5. Very short, 15 day, refueling outages were assumed with a 98% operating capacity factor. These are unrealistic since the average outage times in 2001 were 34 days and the capacity factors an average of 90.7%. The optimistic assumptions will tend to make long cycles less advantageous. The nuclear and mechanical designs are discussed in Section 1.4.4.

A comparison of the core characteristics with the two fuel types and the burnups they can achieve in 18 month cycles are shown as an example in Table 1-6.

Table 1-5: PWR Cycle Design Constraints, Secker, et al., 2001.

	Westinghouse PWR
Enrichment (w/o U <sup>235</sup> )	≤ 4.95
Maximum Rod Relative Power, $F\Delta_H$	1.70
Maximum Local Relative Power, $F_Q(z)$	2.60
Shutdown Margin (% $\Delta\rho$ )	> 1.30
Maximum Allowed MTC (pcm/°F)	5.0
Target MTC (pcm/°F)	< 3.0
Target BOC Maximum Boron Concentration (ppm)	1750

The fuel cycle costs for a large PWR with Westinghouse Performance+ and RFA fuel as a function of burnup and cycle length are shown in Figure 1-9. The conclusions for all cases can be summarized as follows:

- Fuel costs decline at varying degrees with increased burnups for all cases,
- The 24 month cycles do not meet the current NRC burnup limits with Performance + fuel (just barely with RFA fuel) and the economics are so poor compared to the shorter cycles that operators of these large Westinghouse plants are not considering 24 month cycles according to the reference. However, a few low power density plants that can design for a ½ core reload are running on 24-month cycles.
- Within the limited data, the cost savings for the RFA fuel are similar to the Performance+ fuel.

The results of the sensitivity studies of the economic parameters for 18 month cycles are shown on Figure 1-10. The trends are similar to the BWR trends.

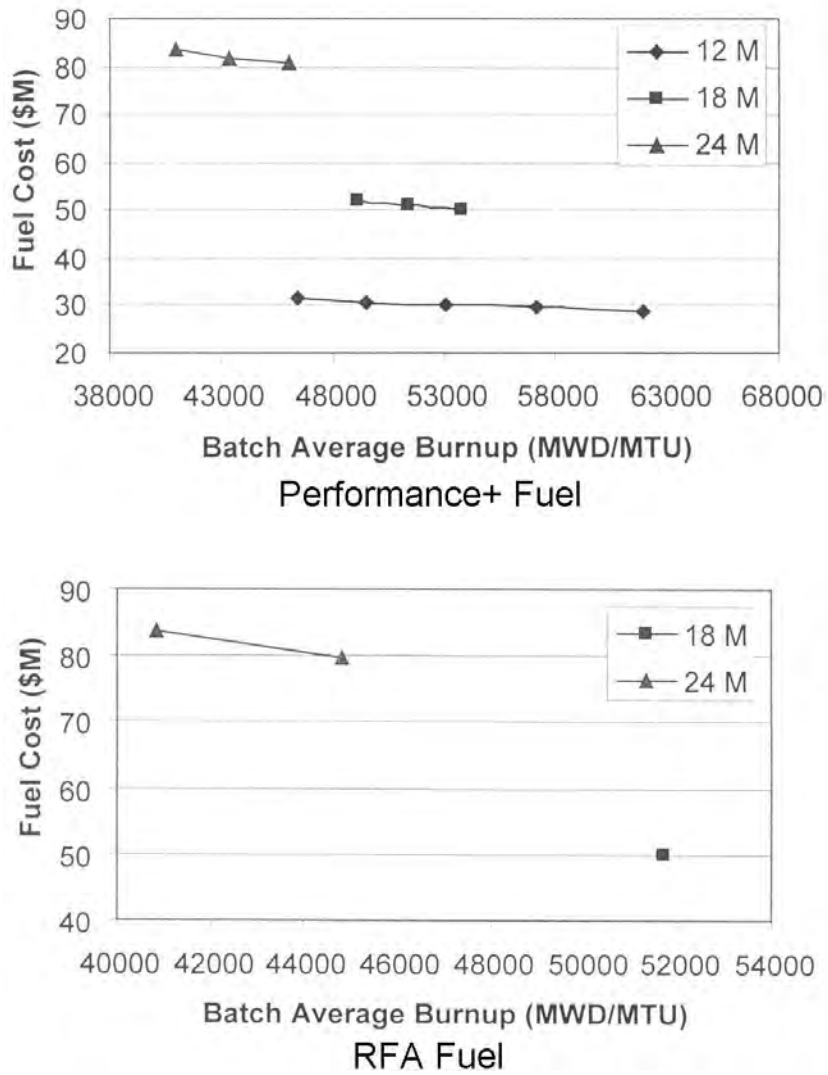


Figure 1-9: PWR Fuel Cost for Different Cycle Lengths at Nominal Parameters, Secker, et al., 2001.



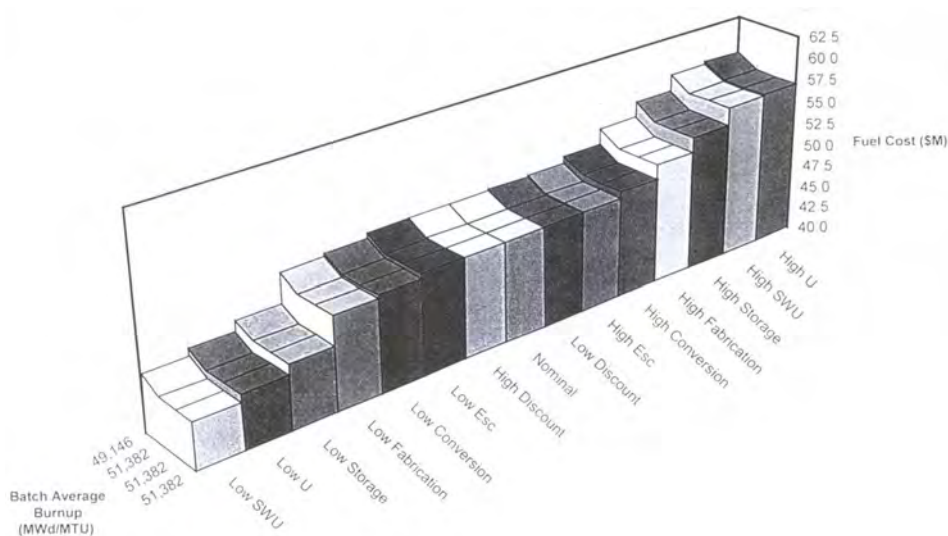


Figure 1-10: Comparison of Each Fuel Cycle Parameter Effect on Fuel Cycle Cost for PWR Performance+ Fuel with 18 Month Cycle.

#### 1.2.2.5 PWRs with >5% Enrichment

The analyses for the high enrichment cycles were continued using the same 4 loop Westinghouse NSSS with Performance + and RFA fuel designs. The batch burnup discharge goals were in the 65-70 GWD/MTU range and the peak rod burnup goal was 75 GWD/MTU. The cycle design limits were the same as in Table 1-5, except that no limit was placed on the enrichment. The same optimistic refueling time of 15 days and capacity factor of 98% were used. The design is described in Section 1.4.5.

A comparison of the core characteristics with the two fuel types and the burnups they can achieve in 18 month cycles are shown as an example in Table 1-6 and, Table 1-7.

The fuel cycle costs for a large Westinghouse PWR with Westinghouse Performance+ and RFA fuel as a function of burnup and cycle length are shown in Figure 1-11. The conclusions can be summarized as follows:

- Fuel costs decline with increased burnup for all cases,
- Levelized fuel costs (not shown) also decline with increased burnup,
- Fuel cost savings with increased burnup are very similar between Performance + and RFA fuel.

The results of the sensitivity studies show that the expected cost trends observed in the other cases. High cost savings with high storage costs do not take into account the potential increased storage costs due to increased cool down times required for the high burnup fuels, Figure 1-12.

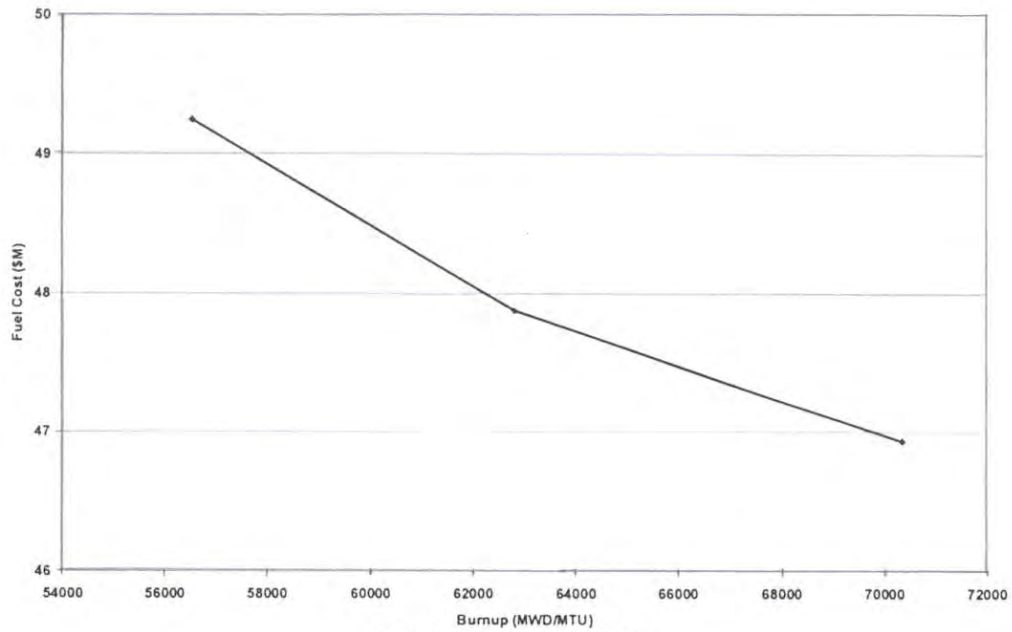
Table 1-6: 18 Month Cycle PWR Fuel Requirements and Core Characteristics, Secker, et al., 2001.

	92 Feed	88 Feed	84 Feed	80 Feed
Fuel Type	Performance+	Performance+	Performance+	RFA
Cycle Length (EFPD)	521.3	521.3	521.3	521.6
Number of Feeds Enr1	64	60	8	48
Enrichment 1 (w/o)	4.545	4.695	4.595	4.620
Number of Feeds Enr2	28	28	76	32
Enrichment 2 (w/o)	4.940	4.950	4.950	4.950
8" Annular Blanket Enrichment (w/o)	3.200	3.200	3.200	3.200
Number of IFBA	9664	8768	8832	6144
Number of WABA	0	0	0	0
Maximum HFP Boron Concentration (ppm)	1462	1505	1492	1753
Maximum HZP Boron Concentration (ppm)	1957	2004	2003	2414
Maximum HZP MTC (pcm/°F)	+4.4	+4.7	+4.1	+1.3
Maximum $F_{\Delta H}$	1.51	1.49	1.48	1.50
Maximum $F_Q(Z)$	2.04	1.98	1.92	1.97
Center Assembly Lead Rod Burnup (MWD/MTU)	64763	65584	63994	69054
Other Assembly Lead Rod Burnup (MWD/MTU)	57314	57379	58989	60219
Batch Average Discharge Burnup (MWD/MTU)	49146	51382	53826	51709

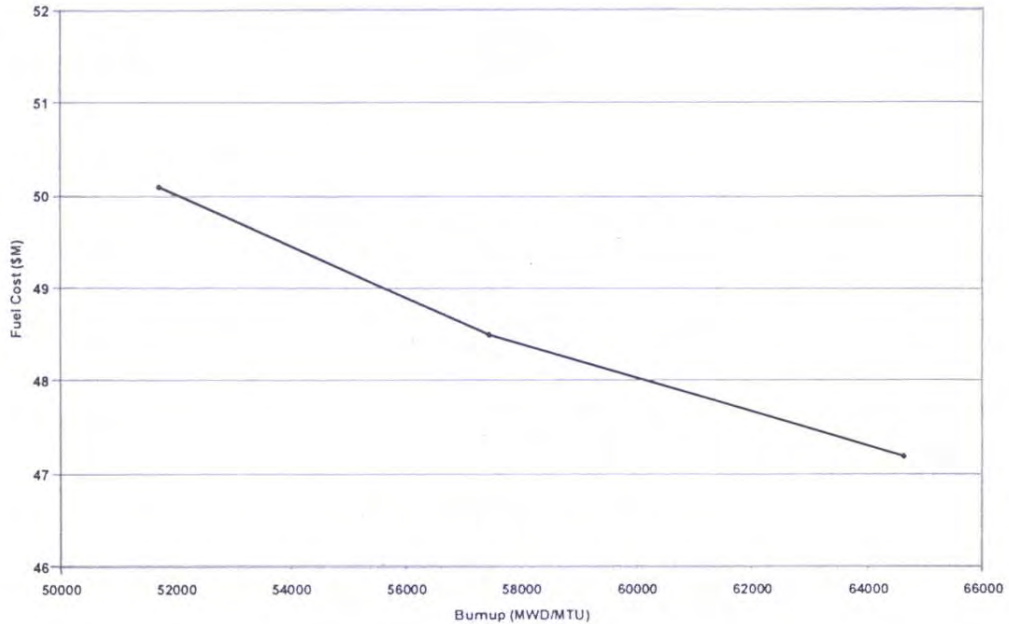
Table 1-7: 18 Month Cycle PWR Fuel Requirements and Core Characteristics With Performance+ and RFA Fuels, Secker, et al., 2002

Fuel Type	80 Feed Performance+	72 Feed Performance+	64 Feed Performance+	80 Feed RFA	72 Feed RFA	64 Feed RFA
Cycle Length (EFPD)	521.6	521.6	521.4	521.6	521.6	521.7
Number of Feeds Enr1	48	24	64	48	24	64
Enrichment 1 (w/o)	4.940	5.177	6.145	4.580	4.800	5.665
Number of Feeds Enr2	32	48	-	32	48	-
Enrichment 2 (w/o)	5.340	5.717	-	4.950	5.315	-
8" Annular Blanket Enrichment (w/o)	3.200	3.200	3.200	3.200	3.200	3.200
Feed Region Average Enrichment, Including Blankets (w/o)	4.936	5.336	5.891	4.593	4.972	5.447
Number of IFBA	7744	7104	7248	7744	7104	5504
Maximum HFP Boron Concentration (ppm)	1587	1718	1704	1543	1696	1900
Maximum HZP Boron Concentration (ppm)	2157	2316	2336	2207	2396	2629
Maximum HZP MTC (pcm/°F)	+3.7	+3.6	+2.2	-0.1	-0.1	+0.9
Maximum F <sub>AH</sub>	1.57	1.54	1.57	1.53	1.55	1.57
Maximum F <sub>Q(z)</sub>	2.08	2.05	2.04	1.99	2.02	2.09
Center Assembly Lead Rod Burnup (MWD/MTU)	76395	73455	83837	68689	68339	76896
Other Assembly Lead Rod Burnup (MWD/MTU)	65758	78756	82267	60994	73078	75285
Batch Average Discharge Burnup (MWD/MTU)	56544	62827	70349	51709	57456	64643

A comparison of all the PWR cost analyses for 18 month cycles, below and above 5% enrichment levels, is shown in Figure 1-13.



Performance+ Fuel



RFA Fuel

Figure 1-11: Fuel Cost for 18 Month PWR Cycle with >5% Enrichment Fuel, Secker, et al., 2002.

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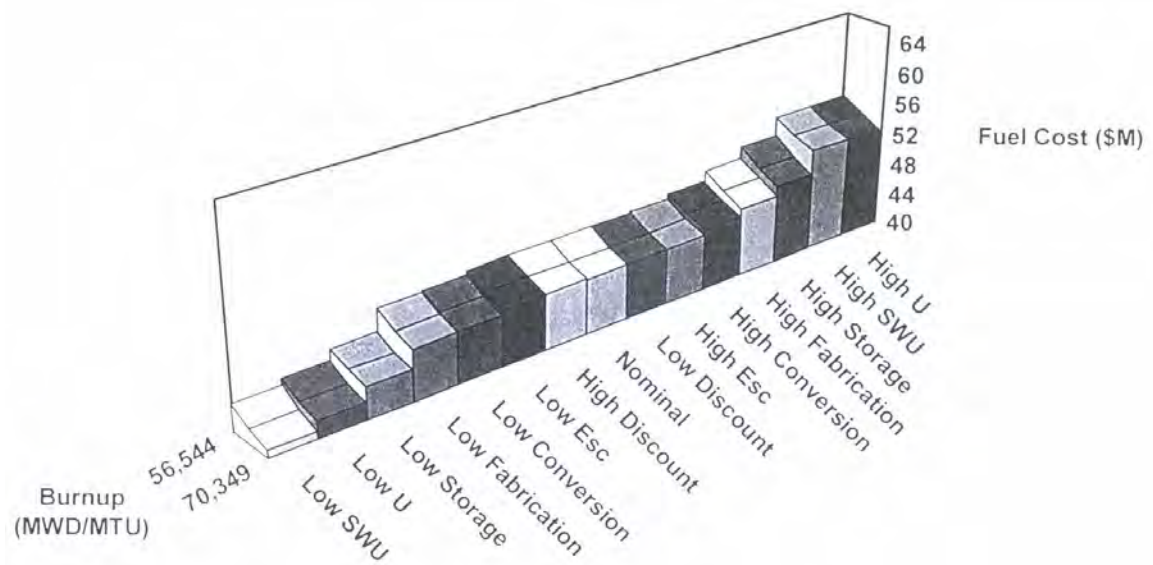


Figure 1-12: Comparison of Each Fuel Cycle Parameter Effect on Fuel Cycle Cost for PWR Performance+ Fuel with 18 Month Cycle, Secker, et al., 2002.

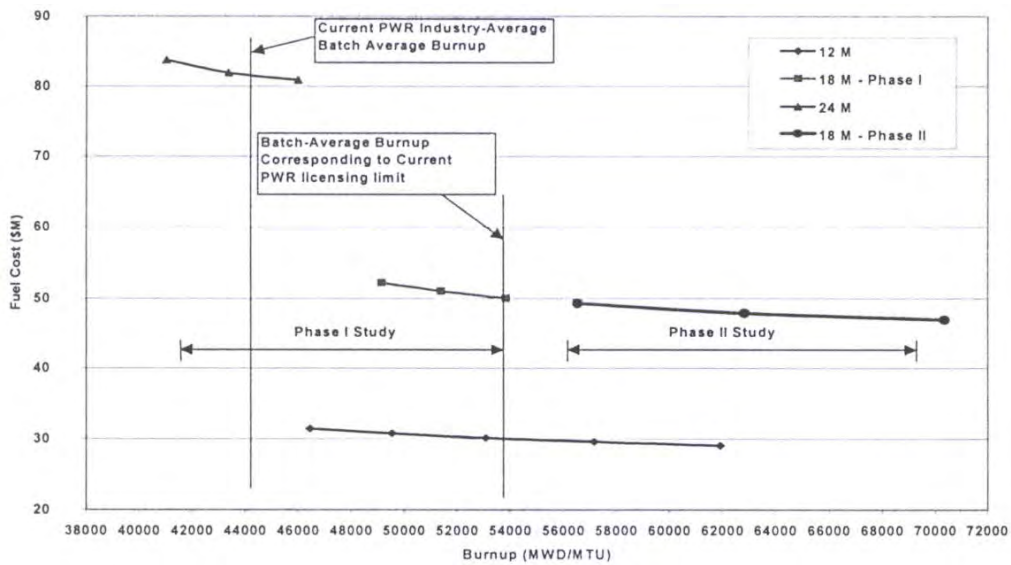


Figure 1-13: Fuel Cost as a Function of Discharge Burnup for an 18 Month Cycle PWR, Secker, et al., 2002.

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#### 1.2.2.6 *Comments on the Economic Analyses*

The results of fuel cycle cost estimates are very sensitive to input cost parameters, financing methods and even tax laws applicable in different geographical areas. The data cannot be evaluated or compared to one's own situation without complete knowledge of the analytical process. Even then, the need to do a plant specific analysis will more than likely be needed. The discussion and references above are useful as a guide for methodology and trends; however, the actual numbers should not be used without confirmation for one's own plant specific case.

The economic analysis results show that fuel cycle costs decrease up to batch average burnups of 70 GWD/MTU with fuel assembly and core designs that are optimized for extended burnups. Application of current designs with increased enrichments to extended burnup service is likely to increase costs since the margins to design and licensing limits are eroded by the high reactivity non-optimized fuel. Absolute cost savings can be large and up to several million \$/year; however, the relatively flat nature of the cost vs. burnup curve can change from a downward to an upward slope with realistic changes of input parameters.

The most recent, detailed economic analyses of the value of increased burnup are by the DOE/EPRI reports summarized above. They developed good optimized fuel and core designs that form a good basis for the economic analyses. However, the economic analyses are incomplete and their appropriate modification may change both absolute fuel costs and cost trends with burnup. The report itself has cautionary notes in this regard, Secker, et al., 2002:

- The following potential contributors to fuel costs were not considered: “allowance for funds used during construction (AFUDC), accelerated depreciation, investment tax credits, property taxes, fuel leasing vs. ownership, insurance costs”,
- For >5% enriched fuel: “The economic analysis assumes no change to enrichment, manufacturing, transportation, licensing, or storage/disposal costs when fuel in excess of 5.0 w/o is used”. “Taken together the costs associated with new fuel designs and the increased manufacturing costs may significantly erode the benefits of using fuel with enrichments greater than 5 w/o”.

In addition to the above, two other important factors were not considered:

- Carrying charges, or the cost of money (for fuel they increase with cycle length),
- Increase in cost of spent-fuel storage,

Both of these factors could reverse trends even at <5% enrichment.

## 2 IMPACT OF IRRADIATION ON FUEL ASSEMBLY MATERIAL MICROSTRUCTURE - BASICS

### 2.1 ZIRCONIUM ALLOYS (RON ADAMSON)

In previous ZIRAT special topical reports, the influence of microstructure of zirconium alloy performance has been described – mechanical properties (Adamson and Rudling, 2001), dimensional stability, Adamson and Rudling, 2002, corrosion, Adamson, et al., 2002 and manufacturing, Rudling and Adamson, 2000. Perhaps second in importance only to the actual elemental composition of the alloys, microstructure is the key to most fabrication and in-reactor requirements. The term “microstructure” as used here means the metallurgical details of the alloys which can be manipulated by thermomechanical processing and by choice of ingot chemistry, and then measured with the goal of obtaining specific alloy behavior. These include:

- grain size and shape
- dislocation density
- alloy phase presence and distribution
- second phase precipitate (particle) size, distribution and composition
- alloy matrix solute composition
- texture

For a given alloy composition, these microstructure components are fixed during fabrication by the specific cold working, hot working, quenching and annealing (thermo mechanical) regimens used, as described in the Manufacturing Special Topical Report and in countless open literature references.

As an example a typical BWR Zircaloy-2 fuel rod would have the following metallurgical microstructure characteristics:

- equiaxed grains with an average diameter (size) in the 4-8 $\mu$ m range
- a very low dislocation density due to a final stage recrystallization anneal
- a completely alpha phase matrix
- second phase precipitates (SPPs) of crystalline form uniformly distributed in the matrix
- SPPs of type Zr (Fr,Cr)<sub>2</sub> with an average size in the range .06-.10 $\mu$ m, deliberately set by thermo mechanical treatments for corrosion performance
- SPPs of type Zr<sub>2</sub> (Fe, Ni) with an average size near .15 $\mu$ m, again chosen for corrosion performance.
- a matrix “solute” content of about 13,000 ppm Sn and 1400 ppm O, deliberately added for strength considerations

- A matrix “solute” content of on the order of 350 ppm Fe+Cr+Ni, as a result of the chosen thermo mechanical treatments. (Since this number is controversial, see Kruger, et al., 1992).
- A crystallographic texture resulting from the chosen thermo mechanical treatments characterized by a basal pole distribution of 10% in the axial direction, 30% in the circumferential direction and 60% in the radial direction.

For a PWR cold worked/stress relieved (SRA) Zircaloy-4 fuel rod, there would be some differences: the grain size could not be determined but the grain shape would be highly elongated in the axial direction, the dislocation density would be very high due to the SRA thermo mechanical state, the SPPs would be only the  $Zr(Fe,Cr)_2$  type due to the alloy composition, the chosen SPP size would be on the order of  $0.2\mu m$  to optimize PWR uniform corrosion resistance, and the Fe+Cr “solute” content would be lower than for the BWR rod.

For a PWR recrystallized Zr1Nb rod, the main differences in microstructure from Zircaloy result from the absence of Sn, Fe, Cr and Ni in the alloy: the SPPs are mainly  $\beta Nb$ , the matrix solute content is about 60,000 ppm Nb (0.6 w/o Nb).

And for a cold-worked/stress relieved Zr1Nb1Sn0.1Fe rod the main differences from Zircaloy are that the SPPs are  $Zr(Nb,Fe)_2$  and  $\beta Nb$ , and the matrix solute content is about 60,000 ppm Nb).

The pertinent questions for this Special Topic are “does the microstructure evolve or change during reactor service” and “what significant differences are there between microstructure at moderate and high burnups”. The answer to the first question is a resounding “yes!” and to the second is “not many”, as described below.

During in-reactor service, the microstructure does indeed evolve, due primarily to the effects of neutron irradiation on the atom-scale arrangement of the elements in each zirconium alloy. For the fuel burnup of 50 MWd/kgU each atom in the alloy lattice has been displaced, at least momentarily, from its normal lattice site about 20 times. Subsequent effects include enhanced diffusion rates, creation of small dislocations in the form of loops, dissolution of SPPs, changes in the solubility of “solute” elements, and changes in the composition of the alloy matrix. For straightforward descriptions of such processes, see the ZIRAT reports, Adamson and Rudling, 2001 and Adamson and Rudling, 2002, a broad review paper on neutron irradiation effects, Adamson, 2000, or a review of Zr-Nb microstructures, Rudling, 2000. Some details are given here on effects for moderate burnup (~50 MWd/kgU) and high burnup (75-100 MWd/kgU).



### 2.1.1 Grain Size

Table 2-1 gives some trends for grain size. For Zircaloy, there does not appear to be any strong effect of irradiation. If anything, some small grain growth could occur at high burnups, as reported by Yang, 1988. For RXA Zr1Nb a small grain growth (4→8 μm) is reported for irradiation at 350°C to low fluences (or burnup), Shishov, et al, 1996. For the Zr2.5Nb alloy, the E125 alloy irradiated at 350°C undergoes complete recrystallization at 350°C but, similar to Canadian CW Zr2.5Nb, none at 305°C.

Table 2-1: Grain Size

Material	Irradiation Effect		
	As-Fabricated Range, μm	Moderate Burnup	High Burnup
Zircaloy, RX	4-8	no change	no change to minor grain growth
Zircaloy, CWSR	-	no change	no change
Zr1Nb	3-7	moderate grain growth	?
Zr2.5Nb		300°C, no change	no change
extrude, CW	-	350°C, full RX	
PRX	-	305°C, no RX	
Zr Sn Nb Fe			
635 – RX	3-5	350°C, grain growth (8)	
635 – PRX	~3	315°C RX	
Zirlo – SR	-	310°C, minor RX	?

For the Russian E635 alloy (Zr1.2Sn1Nb.35Fe) significant changes occur early. By moderate burnup, complete in-reactor recrystallization of initially partially recrystallized material occurs at temperatures between 310-350°C, Shishov, et al, 1996 and Nikulina, et al., 1996. For the Zirlo alloy (Zr1Nb1Sn.1Fe), originally in the SR condition, only minor recrystallization at 310°C was reported, Nystrand and Bergqvist, 1997.

In summary, the grain structure of Zircaloy and RXA Zr1Nb appears to be stable under irradiation, with minor grain growth possible at high burnup. For E635 complete recrystallization occurs at moderate burnup, so no additional high burnup effect is expected. Minor recrystallization of SR Zirlo has been reported, with effects at high burnup unknown. Recrystallization has the effect of lowering the dislocation density, with an accompanying reduction of strength and increase in creep strength. In essence, the performance differences normally expected between a CW or SRA and RXA material would be greatly diminished by recrystallization.

### 3 FUEL PERFORMANCE ISSUES DURING NORMAL OPERATION AND ANTICIPATED OPERATIONAL OCCURRENCES (PETER RUDLING)

The thermo-mechanical properties of the *LWR* fuel assembly is crucial for its satisfactory performance in-reactor. Standard Review Plan, SRP, section 4.2, lists different thermo-mechanical failure modes of the fuel assembly and its components, that either had occurred up to its last revision in 1981 or were thought to be potential failure modes. The *SRP*, section 4.2, also lists the design criterion to each failure mode to ensure that the fuel assembly behaviour is satisfactory. These design criteria are set to ensure that:

- the fuel assembly will not *fail* during normal operation (class I) and anticipated operational occurrences (class II). *Failing* in this sense has a broader meaning, namely that the fuel rod may not be breached and that the dimensional changes of the assembly during irradiation must be limited. The latter requirement is to ensure that control rods can be inserted and that the fuel can be handled during shutdown. Also the BWR fuel outer channel cross section must not have increased to such an extent that it is impossible to pass it through the upper core grid during reloading.
- the fuel remains coolable during an accident (class III and IV). Class IV design basis accident are *LOCA*, *RIA* and earthquake. During *LOCA* and *RIA*, it is important that the fuel cladding may not fail in a brittle fashion during the *reflooding*<sup>1</sup> phase during *LOCA* and due to *PCMI*<sup>2</sup> during the *RIA* transient. During class III and IV situations limited fuel failures are however accepted. Another criterion that must be fulfilled in these situations is that it should be possible to insert the control rods.

Table 3-1 lists the primary failure causes that have occurred in commercial reactor during class I and II operation. Table 3-2 provides corresponding information for hypothetical failures.

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<sup>1</sup> This is the last phase during a *LOCA* situation when the core is reflooded with water that cools the fuel cladding surface imposing very large thermal stresses that may fracture the fuel cladding.

<sup>2</sup> Pellet Cladding Mechanical Interaction, i.e., interaction without the influence of fission products such as iodine (that would instead result in PCI (Pellet Cladding Interaction)).

Table 3-1: Primary failure causes for LWR fuel that has occurred during normal operation and anticipated operational occurrences.

Primary Failure Cause	Reactor type most prone to the failure type	Short Description
Excessive Corrosion and hydriding	PWR and BWR	An accelerated corrosion process results in cladding perforation. This corrosion acceleration can be generated by e.g. PWR or BWR CRUD deposition (CILC <sup>3</sup> ), enhanced spacer shadow corrosion, ESSC, <sup>4</sup> (in BWRs), dry-out due to excessive fuel rod bowing.
Manufacturing defects	PWR and BWR	Non-through-wall cracks in the fuel cladding developed during the cladding manufacturing process that may propagate through the whole cladding thickness during a power ramp causing fuel failure. Defects in bottom and/or top end plug welds. Primary hydriding due to moisture in fuel pellets and or contamination of clad inner surface by moisture or organics. Too large gap between the fuel rod and the spacer grid supports (poor spacer grid manufacturing process) leading to excessive vibrations in the PWR fuel leading to fretting failures. Chipped pellets may result in PCI failures both in liner and non-liner fuel
PCI	BWR	Pellet Cladding Interaction-an iodine assisted stress corrosion cracking phenomenon that may result in fuel failures during rapid power increases in a fuel rod. There are three components that must occur simultaneously to induce PCI and they are: 1) large enough tensile stresses- induced by the power ramp, 2) access to freshly released iodine-occurs during the power ramp, provided that the fuel pellet temperature becomes high enough and 3) a sensitised material – Zircaloy is normally sensitive enough for iodine stress corrosion cracking even in unirradiated state.
Cladding collapse	PWR	This failure mechanism occurred due to pellet densification. This failure mode has today been eliminated by fuel design changes and improved manufacturing control.
Fretting	Debris – BWR and PWR Grid/rod –PWR Baffle jetting- PWR	This failure mode has occurred due to: Debris fretting in BWR and PWR Grid-rod fretting - Excessive vibrations in the PWR fuel rod causing fuel failures. This situation may e.g. occur due to different pressure drops in adjacent fuel assemblies causing cross-flow. Baffle jetting failures - Related to unexpectedly high coolant cross-flows close to baffle joints
Excessive dimensional changes	BWR and PWR	Excessive fuel assembly bowing resulted in dry-out and corrosion failures in a BWR. Similar excessive fuel assembly bowing in PWRs have resulted in difficulties to insert control rods during scram and a necessity to downrate reactor power to maintain thermal margins (dnb and LOCA). Excessive dimensional changes of fuel components have also resulted in handling difficulties during outage.

<sup>3</sup> Crud Induced Localised Corrosion – an accelerated form of corrosion that have historically resulted in large number of failures in BWRs. Three parameters are involved in this corrosion phenomenon, namely: 1) Large Cu coolant concentrations- as a result of e.g. aluminium brass condenser tubes, 2) Low initial fuel rod surface heat flux – occurs in Gd rods and 3) Fuel cladding that shows large initial corrosion rates- occurs in cladding with low resistance towards nodular corrosion.

<sup>4</sup> This corrosion phenomenon resulted recently in a few failed rods in a BWR. The mechanism is not clear but seems to be related to galvanic corrosion. This corrosion type may occur on the fuel cladding in contact or adjacent to a dissimilar material such as Inconel. Thus, this accelerated type of corrosion occurred on the fuel cladding material at spacer locations (the spacer springs in alloy BWR fuel vendors fuel are made of Inconel). Water chemistry seems also play a role if the fuel cladding material microstructure is such that the corrosion performance is poor. Specifically coolant chemistry with low Fe/(Ni+Zn) ratio seems to be aggressive (provided that the cladding material shows poor corrosion performance. A fuel cladding material with good corrosion resistance does not result in ESSC even in aggressive water chemistry.

Table 3-2: Hypothetical failure causes in the *SRP* section 4.2 (that has not occurred in LWRs).

Primary Failure Cause	Reactor type most prone to the failure type	Short Description
Fuel centerline melting	PWR and BWR	According to <i>SRP</i> Section 4.2, it has also been traditional practice to assume that failure will occur if fuel centre melting takes place. The melted fuel may either through its volume expansion of 10% cause PCMI failures or the fuel melt may actually cause the cladding to locally melt and result in fuel clad penetration.
Rod overpressure	PWR and BWR	If the rod internal pressure becomes larger than the reactor system pressure, the fuel cladding may start to creep outwards. If the fuel cladding outward creep rate becomes larger than the fuel swelling rate (due to fission product production during irradiation), the pellet-cladding gap may increase. This phenomenon is named <i>liftoff</i> . Since this gap constitute a significant barrier towards the heat flux, an increased gap may result in an increase in fuel pellet temperature. This higher temperature will in turn increase the fission product release rate thus increasing the fuel rod overpressure even more leading to an even higher outward cladding creep rate. Thus, a thermal feedback effect may result that could quickly lead to fuel failure.
Fatigue	PWR and BWR	Fatigue stresses may be induced in the fuel assembly components due to, e.g., the turbulent coolant flow. According to the <i>SRP</i> , section 4.2, the cumulative number of strain fatigue cycles on the structural components should be significantly less than the design fatigue lifetime, which is based upon the data by O'Donnell and Langer, 1964, and includes a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles.

## 4 WATER CHEMISTRY(GUNNAR WIKMARK)

The development to higher burn-up in itself has no obvious direct impact on or effect from the water chemistry, as discussed. Nevertheless, the trend to longer cycles, higher local and average core power densities, higher operating temperature (in PWRs), etc. will have an impact on the water chemistry and crud behaviour. Furthermore, the measures taken to enable the higher burn-up will also result in changes in the requirements of the “tailored” water chemistry that is becoming increasingly popular in all LWRs. This means that the requirements on the amount, and sometimes forms and techniques applied for water chemistry modification will be affected by the strive for higher burn-up. An example of such impact discussed in the following is the boron shime for PWRs.

It is also evident that a batch fuel burn-up increase by a factor of two or more that we have already experienced for the last period of 20 years is correlating to a significant increase in the numbers of fuel failures, as was reviewed and discussed recently, see Wikmark and Cox, 2001. This development could be resulting from two independent trends but, as is discussed below, this is probably not the case. Instead, we will find that the increased demands and loads emerging from the high burn-up will lead to a situation where the water chemistry must be monitored and controlled much more carefully when higher burn-ups are targeted. Unfortunately, we do not have a consistent and independent data to assess the real impact of the previous development. One reason is the efforts made in improving many chemistry parameters along with the increase in fuel burn-up.

It should also be pointed out that higher batch burn-ups are in general still being a goal and not reality. This means that for most of the effects discussed below, there is no experience gained yet. The reason for this is that testing a few rods or even a few bundles to a significantly higher burn-up will not have any global effect on the water chemistry, since the relative change from the few rods or bundles on the water chemistry are negligible globally in the plant. More effect could be expected from the impact of the water chemistry on the few rods or bundles operating at higher temperature, heat flux levels, etc. The lack of reports on such impact does probably indicate absence of focus on these issues in high burn-up programs rather than the absence of any water chemistry and crud effects. Unfortunately, this also implies that very little results have been reported on such aspects.

### 4.1 *NECESSARY CHANGES IN WATER CHEMISTRY DUE TO HIGH BURN-UP*

#### 4.1.1 **BWRs**

The boiling conditions in the BWRs requires basically that the coolant is as pure as possible in order to avoid significant deposits on the heat transfer surfaces, i.e. cladding. Since no additions are required to operate the BWR from point of nuclear technology, there are no changes necessary to adapt the effects of higher burn-up. It will, however, affect the global and local chemistry due to changes in crud deposition as discussed below.

#### 4.1.2 PWRs

The PWR operation relies on a balance between the reactivity in the core and absorber in the water, in the form of added boron as borate. Higher burn-up implies in reality a higher core reactivity. During initial operation, this can be adjusted by addition of thermal neutron absorbers, “poisons” (gadolinia, erbia, or  $ZrB_2$  are currently used), to the fuel rod. With time, the higher reactivity must be controlled by addition of more B-10, the boron isotope being effective as thermal neutron absorber. This increase could be obtained by increasing the amount of boron (with 19.9% B-10) or to use boron enriched in B-10. The former solution does necessarily lead to a higher boron concentration, but also a higher lithium concentration, in order to retain a higher pH, as illustrated in Figure 4-1. The lithium concentration would have to be even higher if a co-ordinated, rather than the today common modified, chemistry aiming for pH 7.2 is used. Such a pH 7.2 fully co-ordinated chemistry for a plant operating for a 24-month cycle has yet not been performed for any PWR to date (2003), and we are hence lacking experience of such operation. The possible negative impact on fuel cladding corrosion from an increase in lithium concentration has been recently reviewed and discussed, Wikmark and Cox, 2001, Wikmark, 2000.

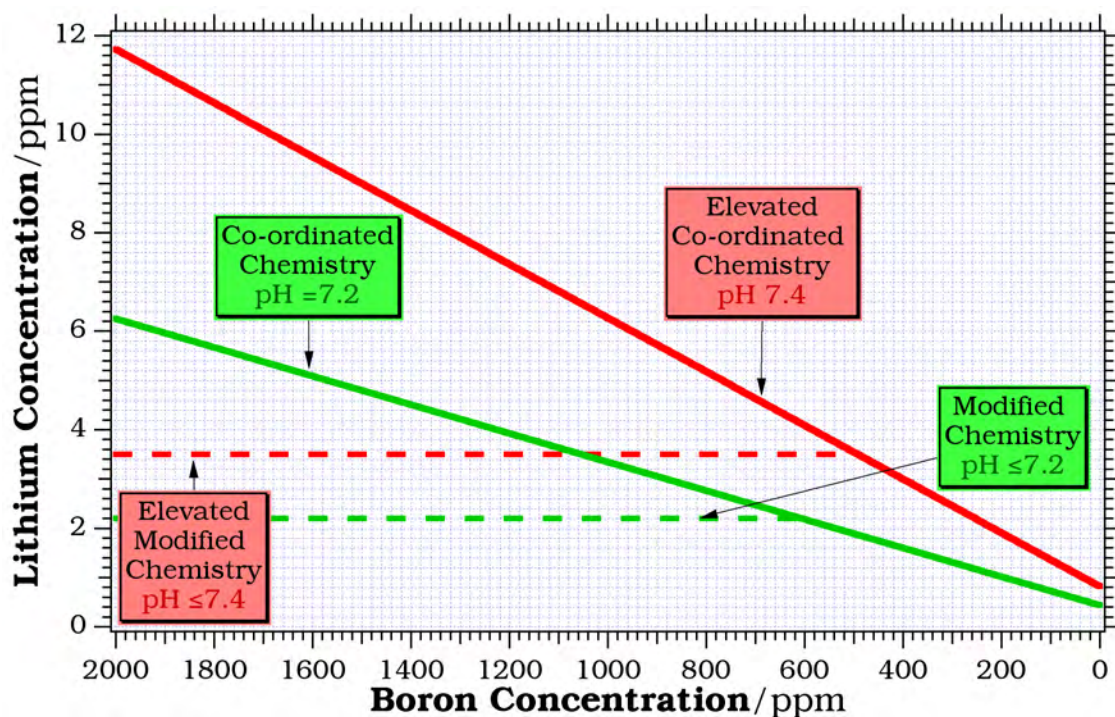


Figure 4-1: Approximate relation between boron and lithium concentrations in 300°C for various PWR water chemistry regimes.

In reality, the required increase in boron concentration from higher burn-up is not that very pronounced unless also the cycle length will be increased. Today, a 24-month cycle requires some 1600 – 1900 ppm boron at beginning of cycle (BOC) with current fuel batch burn-ups. It will possibly be very difficult to stay below 2000 ppm B with 24-month cycles and a batch burn-up close to  $60 \text{ MWd} \cdot (\text{kgU})^{-1}$  unless enriched boron or large amounts of burnable absorbers are used. It is important to keep in mind that the maximum BOC Boron content is usually limited to about 1700 ppm by the core design since a higher B content would result in a positive temperature coefficient for reactivity.

Enriched boron has been employed in German PWRs, yet not for extended cycles at high burn-up. Instead, the two reasons have been to retain sufficient boron solution on site, without risking boron over-saturation or expansion of boron solution storage tanks, Staudt, et al., 2002 or to cope with the enhanced reactivity of MOX fuel at shut-down, Wikmark, 2000. Experience of technical requirements, costs, and safety issues for use of enriched boron is hence already available.

## 4.2 *EFFECTS ON THE WATER CHEMISTRY FROM HIGH BURN-UP FUEL*

### 4.2.1 **Radiolysis**

A strive for higher burn-up generally results in a higher local reactivity and power, and in a higher average core power density. The gamma and fast neutron dose rates will hence increase. The radiolysis product formation on dose rate is, however, not very sensitive to an increase in dose rate, and only marginal radiolysis product concentration increase in the core would be expected from this. We do hence not anticipate a significantly higher radiolysis product formation (in-core hydrogen peroxide, oxygen, and radical formation, as well as BWR steam oxygen and hydrogen concentrations) in BWRs or PWRs by increasing the batch burn-up by, for instance, 20% from today's power density levels.

By far more important are the core loading changes that could result from an enhanced burn-up. The HWC efficiency, i.e. oxidant suppression, in BWRs is hence extremely sensitive to the dose rate in the down-comer where recombination reactions between the oxidants and hydrogen peroxide occur, Lundgren, et al., 2003. A low leakage core loading will result in a very inefficient recombination and hence a sluggish ECP response to hydrogen injection. On the other hand, having a higher remaining power level, and hence higher gamma and fast neutron emission, from the peripheral bundles, as a result of higher burn-ups, could significantly increase the HWC efficiency in many plants.

### 4.2.2 **Other in-core chemistry changes**

As discussed in Section 4.2.1, the integral effect on water chemistry from an increased burn-up is relatively marginal and is not anticipated to produce a significant change in BWRs or PWRs. It should, however, be noted that higher power output could lead to more hydrogen peroxide and higher temperature of the BWR condensate. This would result in higher sulphate release from the Condensate Clean-up system (CCU), Wikmark, et al., 2002.

## 5 FUEL PERFORMANCE ISSUES DURING DESIGN BASIS ACCIDENTS (LOCA, RIA) (PETER RUDLING)

### 5.1 LOCA

#### 5.1.1 Introduction

The loss-of-coolant accident (LOCA) is a major postulated accident considered in licensing. In the safety analysis for a postulated LOCA, it is generally estimated that the peak clad temperature would reach a value in the range of 900 to 1400K, and the fuel cladding would be exposed to high temperature steam for several minutes until emergency core cooling system, ECCS, quenches the fuel bundle. The LOCA sequence can be divided into three phases, Figure 5-1:

- Ballooning and burst of the cladding occur since the rod internal pressure becomes much higher than the system pressure of the reactor pressure vessel and strength of the fuel cladding decreases as the temperature increases.
- The cladding is oxidized by steam and it becomes brittle when severely oxidized.
- The embrittled cladding may rupture by thermal shock caused by rapid cooling during the reflooding stage or due to post-LOCA events.

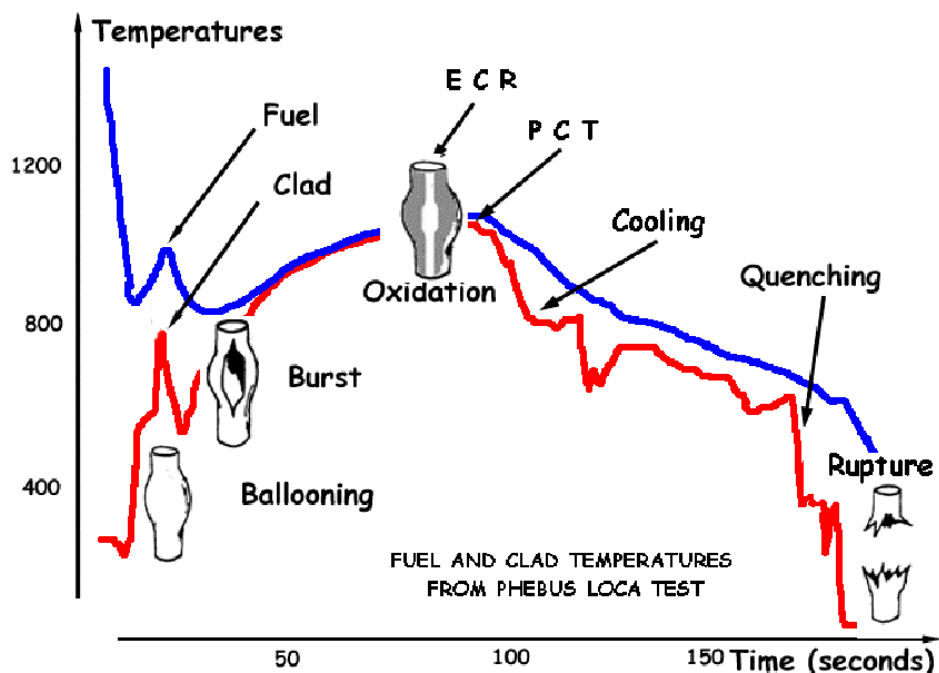


Figure 5-1: PWR LOCA cycle, Maillat, et al., 2001.



For LOCA analysis, it is generally assumed that a certain amount of fuel rods fail and release fission products, but that emergency core cooling systems (ECCS) operate in such a way that fuel rod fragmentation is avoided, thus preserving a coolable geometry, and moreover provide long term core cooling. The ECCS criteria in most countries are:

- Peak Cladding Temperature, PCT, < 1204<sup>15</sup>°C (or 2200°F)
- Equivalent Cladding Reacted, ECR<sup>16</sup>, < 17<sup>17</sup>%
- Hydrogen gas produced < 1%<sup>18</sup>.
- Fuel must have coolable geometry<sup>19</sup>.
- Core temperature maintained at low value for extended time<sup>20</sup>.

The first two criteria are addressing clad embrittlement.

The Equivalent Cladding Reacted, ECR and Peak Clad Temperature, PCT, criteria were formalized in 1973 and based on retention of clad ductility at 275°F (135°C, the saturation temperature during reflood) in slow ring compression tests of 2-sided oxidized non-ballooned unirradiated Zircaloy-2 and -4 cladding samples (in these samples the hydrogen content was low). The selection of the 17% ECR value was specific to the use of the conservative Baker-Just clad oxidation correlation. *It is not clear today what test should be used to ensure that the fuel retains “coolable geometry” during quenching and post-LOCA event. In Russia and in Japan, the fuel rod LOCA behaviour is assessed in quenching tests (in Russia without clad restraint and in Japan with clad restraint). The criteria is that the fuel rod must not rupture during the quenching process. In the rest of the countries, the assessment is done by the ring compression test mentioned above. The ring compression test is more conservative (perhaps over-conservative) compared to the quenching tests. Thus, a material may pass the quench test but may not pass the ring compression test.*

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<sup>15</sup> In Japan in the corresponding temperature is 2000°C

<sup>16</sup> The ECR is defined as the total thickness of cladding that would be converted to stoichiometric ZrO<sub>2</sub> from all the oxygen that are contained in the fuel cladding as ZrO<sub>2</sub>, and oxygen in solid solution in the remaining clad metal phase. Subsequently, in the NRC Information Notice 98-29 (August 3,1998), the NRC stated that total oxidation, as mentioned in 10 CFR 50.46 (acceptance criteria for LOCA analysis), includes both preaccident oxidation and oxidation occurring during a LOCA.

<sup>17</sup> This value is 15% in Japan and 18% in Russia but the limits are based upon LOCA quenching tests.

<sup>18</sup> Total amount of hydrogen shall not exceed 1% of the hypothetical amount generated by the reaction of all the zirconium in the fuel cladding with the water/steam.

<sup>19</sup> The coolable geometry may be lost by either fuel clad ballooning causing coolant channel blockage or fuel cladding fragmentation due to clad embrittlement.

<sup>20</sup> After any operation of the ECCS, the core temperature shall be maintained at an acceptably low value and decay heat removed for the extended period of time required by long-lived radioactivity.

During the late 70's – early 80's, slow ring-compression tests of ballooned and bursted samples showed that the 1973 criteria failed to ensure retention of ductility at 135°C in narrow local regions near the burst opening<sup>21</sup>, where H content exceeds about 700 ppm, see e.g. Figure 5-3. This phenomenon was not known in 1973. However, the 1973 criteria still ensured resistance to 0.3 J impact tests, and survival after fully constrained quench tests for low-burnup Zircaloy<sup>22</sup>. The implications of the results, are such that for high-burnup fuel cladding tubes with a H uptake, prior to the LOCA event, exceeding about 700 ppm:

- the 17% ECR criterion may fail to ensure retention of ductility at 135°C and,
- the clad will not survive a fully constrained quench test without rupturing while it may survive an unconstrained quench test.

Also, the 1204°C peak cladding temperature (PCT) limit was selected on the basis of slow-ring compression tests that were performed at 25-150°C. Samples oxidized at 1315°C are far more brittle than samples oxidized at 1204°C in spite of comparable level of total oxidation. This is because oxygen solid-solution hardening of the prior-beta phase is excessive at oxygen concentrations >0.7wt%. Consideration of potential for runaway oxidation (*due to that the oxidation process of the Zirconium material becomes to exothermic to be cooled by water*) was a secondary factor in selecting the 1204°C limit.

### 5.1.2 LOCA fuel performance

*The LOCA criteria used today are based upon tests of non-irradiated Zry-2 and -4 fuel cladding. There are no data on e.g. irradiated fuel rods. Large programs are ongoing to test claddings of various types (e.g. Zr-4, Zr-2, ZIRLO and M5) at high burnup levels, see section 5.3. Also, tests will be performed on fuel rods to e.g. assess the impact of TFGR on LOCA fuel performance.* In the following, relevant data obtained up to this date, to predict fuel cladding performance at high burnups are presented. It will be shown that the data suggests that:

- Oxide thickness prior to the LOCA event does not significantly impact fuel LOCA performance
- Hydrides in the fuel cladding may increase the LOCA fuel cladding oxidation rate somewhat (a few percent at the most) but this is taken care of in the very conservative Baker-Just model
- Increased hydrogen cladding content decreases cladding ductility and at a hydrogen concentration exceeding about 700-800 wtppm in the fuel cladding the ductility at 135°C is small.

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<sup>21</sup> It appears that the stagnant conditions of water/steam in this location will significantly increase the hydrogen pickup during LOCA clad oxidation.

<sup>22</sup> Following these results, Japan modified the basis of its ECR criterion to ensure survival after fully constrained quench tests by specifying a maximum of ECR 15%.

## **6 FUEL PERFORMANCE ISSUES DURING INTERMEDIATE WET AND DRY STORAGE (AL STRASSER)**

### *6.1 WET STORAGE*

Fuel to be stored in the spent fuel pool does not have any licensing burnup limits per se. All the fuel assemblies that are discharged from the reactor core need to be stored in the fuel pool. The fuel pool itself has to meet certain design criteria and some of these may be impacted by increasing burnup levels. As these design limits are approached, the fuel pool may have to be modified to meet them.

Two design criteria can potentially limit storage of high burnup assemblies and/or decrease the storage capacity of the pool: criticality considerations and the pool system's cooling capacity.

The increased average enrichment of fresh high burnup fuel assemblies may require either larger assembly to assembly spacing or increased absorbers in the pool racks. Assemblies that have reached their goal high burnup should be able to obtain burnup credit for the fission product poisons in the fuel and this would extend the margin to the reactivity limits. Partially burned, high enrichment assemblies will need to be assessed on a case by case basis. The requirement to be able to unload the core at any burnup level may necessitate reservation of a section of the pool for accommodating the high enrichment assemblies. Since most pools already have sections assigned for various reactivity level fuels, this should only be a further step in the evolution of that methodology. *Planning for any required pool modifications and related licensing actions should occur as soon as higher enrichments are being considered.*

Increased burnup increases the inventory of Pu isotopes that are the source of the decay heat, therefore the assemblies with high burnup will generate more heat than the low burnup assemblies and this heat has to be dissipated by the pool cooling system. As the batch average burnups increase with time, the fraction of high burnup, high decay heat assemblies will increase in the pool and the cooling system may have to be upgraded to keep the water temperature at the specified level. Depending on the Tech. Spec. for the plant, the time to boiling in the event of a partial loss of coolant may have to be considered as well. The projected number of fuel assemblies with >45 GWD/MT burnup expected to be discharged as a function of time are shown in Figure 6-1 for BWRs and Figure 6-2 for PWRs and indicate the ever increasing fraction of high burnup assemblies.

All aspects of the cooling system need to be reviewed including the heat exchanger, pumps, piping sizes, clean-up system capacity, and the hydraulics of the pool itself. *A comparison of the heat removal limits of the system with the expected increase of total decay heat in the pool should be made in order to predict the date when the acceptable margins to the heat removal limit will be reached. This will permit timely modification of the cooling system.*

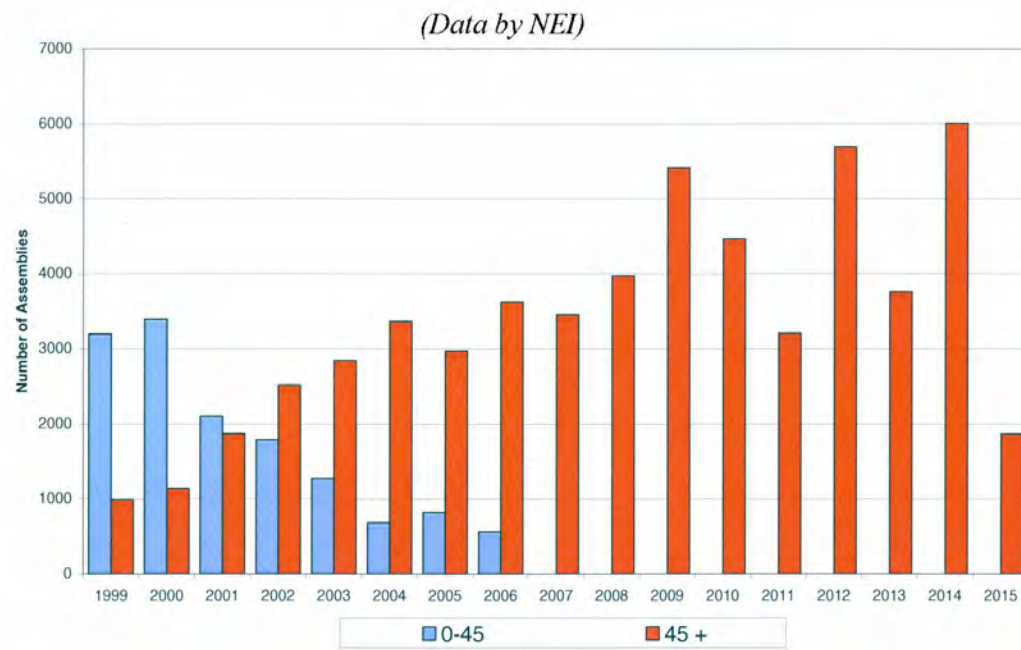


Figure 6-1: Projected BWR Burnup Distribution Through 2015, Gruss, 2002.

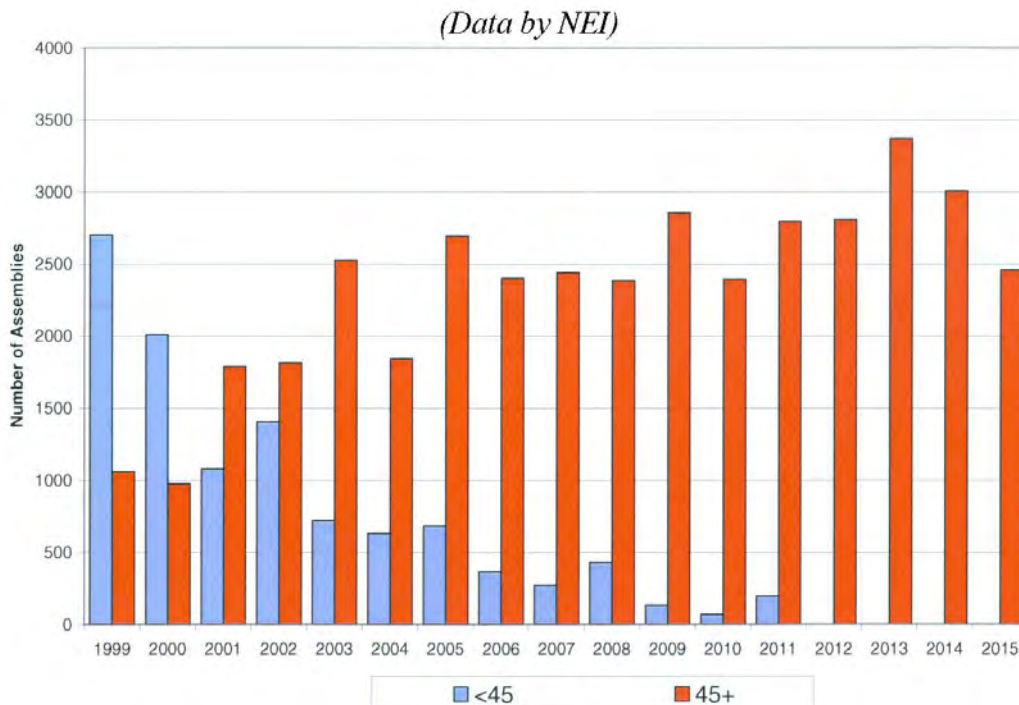


Figure 6-2: Projected PWR Burnup Distribution Through 2015, Gruss, 2002.

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## 6.2 DRY STORAGE

The burnup limits for dry storage vary from country to country as do the criteria they are based on.

The parameters that affect fuel performance during dry storage are summarized in the Introduction, Section 1.8.2. Dry cask storage containers dissipate the fuel decay heat by natural convection of the cask helium (He) atmosphere and conduction through the cask container walls; there are no moving parts or forced cooling in this system. As a result, the cladding can reach temperatures of several hundred degrees C. The pressure differential across the cladding can be significant since the fuel rod internal gas pressure is made up of:

- He gas pre-pressurization during fabrication,
- Fission gases,
- He from transmutation of B in burnable absorbers,
- Alpha decay of the Pu isotopes in the fuel during storage,

and their pressure is further raised by the fuel decay heat. The pressure outside the cladding in the cask is only slightly above atmospheric. Creep deformation of the cladding will occur at a relatively constant rate early in life, when the internal gas pressure, the cladding stresses and the cladding temperature are at their highest. As the decay heat decreases with time, the gas pressure and the cladding temperature both decrease. In addition, the internal free volume of the fuel rod increases as the cladding creeps outward, decreasing the gas pressure and cladding stresses further. All three of these factors eventually reduce the creep rate to a negligible value.

Creep-rupture is the most likely cladding failure mode during dry storage and there is general consensus on this mechanism. The parameters that determine the potential for creep rupture are the cladding stress level, the cladding temperature and the rate of decay heat decrease and all three of these parameters are burnup dependent.

***The sources of the internal pressure*** are all burnup dependent, except for the as – fabricated pre- pressurization level. *Fission gas pressure* used in the licensing process is based on the licensed fission gas release model of each specific vendor and will vary as a function of burnup and temperatures reached during the fuel's operation in the reactor. Fission gas production, Xe and Kr, will increase as a function of burnup, but the fraction released will depend on other factors. The high burnup rims that form on pellets may be particularly sensitive to gas release with increasing temperatures. Additional data on fission gas release are discussed in Section 2.2.5 of this Report.

## 7 DISCUSSION AND SUMMARY

The first and second part of this section provides the incentives for high burnups and a summary of the different potential high burnup fuel failure mechanisms, respectively. Subsequently, the potential burnup limitations of the different commercial alloys are discussed based upon the potential high burnup failure mechanisms. Finally, a section discussing the impact of intermediate storage on fuel burnup limitations are provided.

### 7.1 INCENTIVES FOR GOING TO HIGH BURNUPS

The list below represents the incentives that existed in the early days of the nuclear industry for operating fuel to high burnups. Most of the incentives are still valid however, the value of and the emphasis on each one is slowly changing with time. The incentives are:

- Economics --- lower fuel cycle costs,
- Capability for longer cycles --- increased capacity factors, decreased radiation doses. The economic gains due to longer cycles facilitated by extended burnups have been taken advantage of to a large degree by the current cycle lengths and burnup levels.
- Improved resource utilization --- decreased amount of uranium, Separative Work Units (SWU) and fuel assemblies per unit energy produced,
- Increased margin to storage capacity limits. However, the inability to send fuel for reprocessing or to a permanent storage site has caused a spent fuel assembly log-jam in the spent fuel pools and effectively eliminated this high burnup incentive,
- Eventual decreased offsite shipping and storage costs. However, the significantly increased time required for high burnup fuel to decrease its decay heat in a spent fuel pool before it can be loaded into an intermediate dry storage cask and the unknown schedule for shipping the fuel from the dry cask to a permanent storage site prevents a reliable estimate for the capacity and cost required for the intermediate wet and dry storage facilities.

Added incentives in European countries (and in the future in Japan) favored decreased number of fuel assemblies for reprocessing and refabrication due to,

- High back end cost of reprocessing,
- Reality of Pu recycle and its high fabrication cost.

The economic incentives for extending burnup levels will most likely disappear at batch average burnups in the range of 60 to 70 GWD/MT. Economic analyses that represent all costs at conditions prevailing today have not been published and those tend to flatten out at these burnups. Unaccounted costs and uncertainties in the back-end costs will if anything increase fuel costs at burnup levels above these. Since the increase or decrease of fuel costs at these burnup levels are very sensitive to the input, the break-even point for extended burnup requires a plant specific analysis.

In the opinion of this author, based on this and other factors discussed, extension of burnup to levels that require >5% enrichment are highly unlikely.

The reduction in margins to nuclear, thermal and safety analysis limits poses challenges to fuel management methods in order to maintain the desirable as well as the licensing margins. Modified fuel designs and fuel management methods have succeeded to meet the design and licensing limits with 4.95% as well as 5.95% enriched fuel. The major modifications have been the increased amount of burnable absorbers to hold down the increased reactivity, more sophisticated reactivity zoning and nuclear calculations for their accommodation. Other modifications include optimization of the H/U ratios, improved spacer designs and optimization of fuel management methods. Detailed studies of designs >5% enrichment may reach nuclear or thermal limits that will be difficult to maintain by design modifications.

## 7.2 *POTENTIAL FAILURE MECHANISM AT HIGH BURNUPS*

From the previous discussions in this report, it appears that increased burnup may reduce the margins towards the following failure modes during class I and II operation:

- Corrosion
- PCI/PCMI
- Dimensional changes

and the corresponding failure modes during LOCA and RIA:

- “brittle” fuel rods failure during LOCA resulting in “non-coolable fuel geometry”
- fuel dispersal during RIA

The effect of increasing burnup on the above failure modes are discussed in the sequel.

### 7.2.1 **Corrosion**

Increased burnup will increase the degree of zirconium alloy material corrosion since higher burnup also means in general longer residence time in the reactor. This holds true for all zirconium alloy material fuel components, Figure 7-1. However, the development of new PWR alloys have significantly reduced corrosion rates compared to that of Zry-4 and thus increased the margins towards corrosion failures.

To reach higher burnups, the enrichment of the fuel must increase and therefore, the fuel rod power over its lifetime will increase. This situation tends to increase the fuel clad temperature that for PWRs will increase the corrosion rate (since the corrosion rate is much less dependent on temperature in BWRs, similar effect will not be seen in BWRs).

Also, higher enrichment fuels in PWRs will require an increase in the LiOH coolant concentration. This since reactivity control in PWRs is to a large extent controlled by the boron concentration in the coolant, and increased fuel reactivity will require an increase in the boron coolant concentration that in turn will require an increase in the LiOH coolant concentration to maintain the optimum pH. The tendency for increased fuel rod power with increased burnup may result in increased tendency for subcooled boiling in the hottest channels and may together with the increased LiOH coolant content tend to deteriorate the protectiveness of the zirconium oxide layer with accelerated corrosion as a result.

Both in BWRs and PWRs there is a tendency for SPP dissolution with increasing fast fluence (that corresponds to increased burnup). The larger SPPs in Zircalloys for PWR application results in a slower dissolution rate compared to that of Zircalloys for BWRs. When the SPPs have dissolved there is a larger risk that corrosion rate accelerates provided that the coolant chemistry is “aggressive” (what “aggressive” means we do not yet know).

Both in PWRs and BWRs it appears that hydrides at the metal/oxide interface may accelerate corrosion rate, thus with increased burnup, corrosion produced hydrogen absorbed in the zirconium alloy material will increase. This hydrogen may eventually precipitate out as hydrides and as such may accelerate the corrosion rate. Again, the development of new corrosion resistance PWR alloys will also reduce hydrogen pickup (that is the product of corrosion rate and hydrogen pickup fraction) reducing the tendency for hydride driven corrosion acceleration late in life.

There are also some other important water chemistry changes that are not driven by the objective to reach higher burnups but for reasons such as reducing cracking tendencies (NMCA with HWC, HWC, and Zn-injection), and to limit radiation dose (Zn-injection, increased pH in PWRs). In most cases these water chemistry changes results in a more aggressive corrosion environment that may limit fuel burnup.



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