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Economic Incentives and Recommended Development for Commercial Use of High Burnup Fuels in the Once-Through LWR Fuel Cycle

R. B. Stout
K. R. Merckx
J. S. Holm

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1.0 INTRODUCTION

The work summarized in this report was performed as a part of the Non-proliferation Alternatives Systems Assessment Program (NASAP) and constitutes an assessment of the economic incentives and development needs for commercial use of high burnup fuels in the once through light water reactor (LWR) fuel cycle.

The fuel burnup in current light water reactors was selected based on licensing considerations, reasonable assurance of acceptable fuel performance and on the assumption that the spent fuel would be reprocessed. Due to the current lack of reprocessing capability, utilities are operating LWR's on a once through fuel cycle and the lack of uranium and plutonium recycle have increased the uranium requirements by 30% to 35%. The decision on if and when commercial reprocessing plants will be licensed in the U.S. has yet to be made. In the interim, increasing the average discharge burnup of the fuel for LWR's operating on a once through fuel cycle will result in increased efficiency in the utilization of uranium and provide the potential for reduced electrical power production costs. However, prior to irradiating commercial quantities of fuel to high burnups, additional research, development and test data will be required.

This study quantifies the reduced uranium requirements and the economic incentives for increasing the burnup of current design LWR fuels from the

current range of 25 to 35 MWD/Kg to a range of 45 to 55 MWD/Kg. The changes in fuel management strategies which may be required to accomodate these high burnup fuels and longer fuel cycles are discussed.

The material behavior problems which may present obstacles to achieving high burnup or to license fuel are identified and discussed. These problems are presented in terms of integral fuel response and the informational needs for commercial and licensing acceptance. Research and development programs are outlined which are aimed at achieving a licensing position and commercial acceptance of high burnup fuels.

Attached as an appendix to this report is an abstract review from Exxon Nuclear Company's (ENC) reference file on published data pertaining to high burnup fuel performance.

2.0 SUMMARY

2.1 REDUCTION IN URANIUM REQUIREMENTS AND ECONOMIC INCENTIVES OF HIGH BURNUP FUELS

The reduction in uranium requirements, the fuel cycle cost and the total power production costs were calculated for increasing the discharge burnups in LWR's from the current 30 MWD/Kg up to the range of 45 to 55 MWD/Kg. These calculations were done for two, three and four-loop Westinghouse pressurized water reactors (PWR's) and for boiling water reactors (BWR's). The results of these calculations are discussed in Section 3.0, and the key conclusions are shown in Table 2.1. For constant cycle lengths, the uranium requirements can be reduced from 9% to 15% if batch average discharge exposures of 50 MWD/Kg can be achieved compared to current discharge exposures of 30 MWD/Kg.

To achieve acceptable power peaking factors, moderator temperature coefficients and cold shutdown margins, high burnup fuels will require burnable poisons. If the burnable poisons are in the form of rod control clusters some of the uranium savings will be lost due to the residual reactivity penalty at the end of cycle resulting from neutron absorption in the structural material and the residual poison. The alternative to the use of rod control clusters is gadolinium oxide (gadolinia) which is mixed in the fuel and burns out during the cycle and resulting in a minimal reactivity penalty at the end of cycle and therefore minimum uranium requirements. The uranium savings indicated in Table 2.1 were calculated using

Table 2.1 Savings Resulting from Increasing Batch Average Burnups
in LWR's from Current Burnups of Approximately
30 MWD/Kg to Burnups of 50 MWD/Kg

	<u>Fuel Cycle Mode</u>	
	<u>Constant Cycle Length Decreasing Batch Size</u>	<u>Increasing Cycle Length** Constant Batch Size</u>
Reduction in Uranium Requirements	9-15%	1-5%
Fuel Cycle Costs*	4-10% Reduction	1-4% Increase
Reduction in Total Electrical Power Production Costs	1-3%	1-3%

* Based on the economic parameters shown in Table 3.1.

** Increased cycle length is approximately from a 12 month to an
18 month cycle.

gadolinia. Gadolinia has been used extensively in BWR's and is currently being irradiated in PWR's. No major problems are foreseen in the use of gadolinia in high burnup fuels in LWR's. The use of gadolinia burnable poison in LWR's is discussed further in Section 3.0.

It should be emphasized that the results of this study do not include an optimization of the fuel design and fuel management for high burnup or for long cycles. Optimization of the fuel design to accomodate a particular range of burnup or cycle length and low neutron leakage fuel loading patterns should result in further reductions in uranium requirements.

It should be further emphasized that the majority of the economic analysis in this report assumed a constant fuel fabrication cost for both the current design fuel and fuel designed for high burnup. ENC estimates that the fuel fabrication cost on a per Kg basis will be from 25% to 50% higher for fuel designed to achieve 50 MWD/Kg than for current fuel designs. The fabrication cost of annular fuel is more expensive in direct proportion to the percentage reduction of the fuel pellet volume, even if the cost per pellet is identical to solid fuel pellets. A 10% reduction in pellet volume is typical for annular fuel. The remainder of the incremental cost for high burnup fuels would be attributed to the production of the annular pellet, adding lubricants, and possible thicker cladding requirements. In the case of spherepac fuels it is estimated that the fuel fabrication costs may be approximately the same as for current fuels.

The fuel fabrication costs represent only about 7.2% of the total fuel cycle costs. Increasing the fuel fabrication cost by 25% while keeping all other components of the fuel cycle fixed, would increase the total fuel cycle costs by less than 2%. From Table 2.1 the estimated savings in the fuel cycle cost of high burnup fuels are from 4% to 10%. The fuel fabrication costs for high burnup fuels could therefore increase from 50% to 125% above that of current design fuels before the increased fuel fabrication costs would negate the lower fuel cycle costs resulting from the high burnup.

2.2 MATERIAL CONSIDERATIONS AND RECOMMENDED PROGRAMS FOR COMMERCIAL ACCEPTANCE OF HIGH BURNUP FUELS

Section 4.0 describes the material considerations of achieving high burnup for commercial LWR fuels. In general no limiting conditions exist to prevent achieving batch average burnups between 45 to 55 MWD/Kg and peak pellet burnups in excess of 60 MWD/Kg. However, prior to achieving commercial acceptability, which includes safety and licensing considerations, the existing design bases for several phenomena need to be substantiated and the data bases pertaining to these phenomena need to be expanded. Of particular concern are the release of the inert gaseous fission products, the potential fuel cladding failure due to a stress corrosion mechanism and the possible changes in failure mechanisms at high burnup related to the effects of higher fission product inventories and their behavior during anticipated

transients. In some cases new programs may be required, in others the programs to answer the questions are already in place. Table 2.2 summarizes the major technology considerations for commercial utilization of high burnup in current LWR's. This table presents several fuel related phenomena, the concern of these phenomena at high burnup, and the reactor and fuel type to which it applies.

Section 5.0 discusses the programs to support the considerations discussed in Section 4.0. Table 2.3 summarizes the recommended programs to support commercial utilization of high burnup fuels in current LWR's. An estimate of the time scale for achieving solutions to each of the phenomena listed in Table 2.2 is presented in Table 2.4.

Table 2.2 Major Technology Requirements for Commercial Utilization
of High Burnup Fuel in Current Light Water Reactors

<u>Normal Operation</u>			
<u>Phenomenon</u>	<u>High Exposure Requirement</u>	<u>Reactor Type</u>	<u>Fuel Type</u>
Fission Gas Release	Verification of data base for burnup enhancement at limiting goal exposure and power level.	BWR, PWR	Pellets (Solid and Annular), Spherepac
	Data base for extended surface effects	BWR, PWR	Spherepac
Fuel-Cladding Interaction	Confirmation that ramping damage thresholds can be extrapolated to higher burnups.	BWR, PWR	All fuel types
	Quantification of the effective time period that conditioning remains effective.	BWR, PWR	Pellets, (Solid and Annular), Barrier, Spherepac
Failure statistics	Confirmation that present failure probability prediction methodology can be extended to higher burnups and the new fuel concepts.	BWR, PWR	Pellets (Solid and Annular), Barrier, Spherepac
Dimensional Stability			
Rod Bow	A data base which verifies that coolant gap closure remains limited.	PWR	17 x 17 Fuel Assembly Design
Cladding-Circumferential Strain	Expanded data base for gas release.	BWR, PWR	Pellets (Solid and Annular), Spherepac
Barrier Condition	Verification that fabrication technique has provided sufficient barrier throughout the intended fuel exposure lifetime effects.	BWR, PWR	Barrier

Table 2.2 (Continued)

<u>Phenomenon</u>	<u>High Exposure Requirement</u>	<u>Reactor Type</u>	<u>Fuel Type</u>
Corrosion Limitation	Corrosion data base to confirm limited nodular corrosion.	BWR	
	Corrosion data base assuring no accelerated corrosion.	PWR	
Fission Product Release from Failed Fuel	Confirmation that fission product contamination from failed fuel is not increased isgnificantly at higher burnup.	BWR, PWR	Pellets (Solid and Annular), Spherepac
Burnable Poison Rods	Confirmation that the behavior of fuel containing high concentrations of burnable poison is as predicted.	BWR, PWR	Rods containing High Gadolinia concentrations

Special Licensing Issues

<u>Phenomena</u>	<u>Possible High Exposure Requirement</u>	<u>Reactor Type</u>	<u>Fuel Type</u>
Anticipated Transients	Confirmation that volatile fission product accumulation does not lead to sudden fuel volume increase under power bursts.	BWR, PWR	Pellets - Annular
Chemical Stability of Barriers	Confirmation that potential interaction of metallic or graphite barriers with cladding, fuel or deposited fission products under loss of coolant accident (LOCA), power coolant mismatch (PCM), or reactivity insertion accident condition does not have an adverse effect.	BWR, PWR	Barrier

Table 2.2 (Continued)

<u>Phenomena</u>	<u>Possible High Exposure Requirement</u>	<u>Reactor Type</u>	<u>Fuel Type</u>
Fuel Relocation	Demonstration that fuel relocation is not significantly greater than that of solid pellet fuel for conditions causing cladding rupture.	BWR, PWR	Spherepac Annular

Table 2.3 Recommended Programs to Support Commercial Utilization
of High Burnup Fuel in Current LWR's

<u>Phenomenon</u>	<u>Program</u>
Fission Gas Release	Poolside gas release measurements from rods in lead assemblies and rodlet tests at higher powers for limiting power release rates.
Fuel-Cladding Interaction	Extended burnup rodlet test reactor tests to confirm failure and damage thresholds. (Develop technique to determine damage threshold.)
Failure Statistics	Partial core loadings with sipping measurements to increase data base.
Dimensional Stability	Rod spacing measurements and gap size measurements in lead PWR 17x17 assemblies. Post irradiation examination (PIE) of high exposure barrier fuel to determine barrier integrity.
Corrosion Limitation	Poolside measurement and inspection of lead assemblies for corrosion layer thickness.
Fission Product Release	Characterization of failures and measurement of release of fission products from ramp tests of rodlets in a test reactor.
Burnable Poison Rods	Poolside measurement of gas release and PIE of fuel chemistry.
Anticipated Transients	Power Burst Facility (PBF) tests on high burnup rodlets for confirmation of no unusual behavior. Direct electrical heating tests of high burnup fuel with PIE microstructural evaluation.
Chemical Stability of Barriers	Controlled heatup of chemically simulated fuel-barrier-cladding surfaces for confirmation of chemical stability and PBF verification of conditions with combined thermal-mechanical-chemical interactions.
Fuel Relocation	PBF tests for confirmation of no unusual behavior.

Table 2.4 Proposed Schedule for Commercial Utilization of High Burnup Fuels
in Current Light Water Reactors

Date	Extended Designs	Modified Designs*	New Designs**	
			Barriers and/or Annular	Spherepac and/or Doped
1980	Extended burnup reconstituted bundles	Lead assemblies (intermediate gadolinia concentrations)	Test rods and rodlets	Test rods and rodlets
1981	Rodlet PCI testing and gas release		Lead assemblies (high gadolinia concentrations)	
1982		Partial core loading	Fuel ramp test to estab- lish PCI capacity	Lead assemblies (high gadolinia concentra- tions) Fuel ramp test to estab- lish PCI capacity
1983	Extended burnup gas release base PCI burnup dependence		Partial core loading Transient confirmation tests	Transient testing
1984		Multicore loadings***		Gas release data base PCI burnup dependence
1985		Extended burnup gas release data base PCI burnup dependence		Partial core loading
1986		Gadolinia data base Verification of corrosion limitations	Extended burnup gas release data base	

* Modified Designs for: 33-36 MWD/KgU-BWF's; 38-42 MWD/KgU-PWR's

** New Designs (Major Modifications with less restriction on operation)
for 40-45 MWD/KgU-BWR's; 45-50 MWD/KgU-PWR's

*** Multicore Loadings where licensing data base is established and commercial
risk is acceptable assuming test programs yield favorable results

Table 2.4 Continued

Date	Extended Designs	Modified Designs*	New Designs**	
			Barriers and/or Annular	Spherepac and/or Doped
1987		Verification of failure statistics from multicore loadings	Gadolinia data base from multicore loadings***	Extended burnup gas release data base
1988			Verification of failure statistics from multicore	Gadolinia data base from multicore loadings
1989				
1990				Verification of failure statistics from multicore loadings***

- * Modified Designs for: 33-36 MWD/KgU-EWR's; 38-42 MWD/KgU-PWR's
- ** New Designs (Major Modifications with less restriction on operation) for 40-45 MWD/KgJ-BWR's; 45-50 MWD/KgU-PWR's
- *** Multicore Loadings where licensing data base is established and commercial risk is acceptable assuming test programs yield favorable results.

3.0 URANIUM REQUIREMENTS, ELECTRICAL POWER PRODUCTION ECONOMICS AND FUEL MANAGEMENT STRATEGIES FOR HIGH BURNUP FUELS

Currently pressurized water reactors are irradiating fuels to discharge burnups of about 30 MWD/Kg. Boiling water reactors are discharging fuels at slightly lower average burnups. These burnups were established partially due to fuel integrity limitations and partially due to economic considerations of a fuel cycle which assumed less expensive uranium and separative work, reprocessing of spent fuel and recycle of uranium and plutonium. Under these conditions the current discharge exposures of LWR fuels were near optimum.

Due to the current lack of reprocessing capability, utilities are actually operating LWR's on a once through fuel cycle. For a once through LWR fuel cycle, the uranium requirements per unit of energy can be significantly reduced if the discharge exposure of the fuel is increased. The maximum benefit in uranium reductions are achieved if the cycle length is held constant while increasing the burnup. If higher burnups are accompanied by increasing the cycle length from 12 to 18 month cycles then the increase in burnup needs to be larger to realize an equivalent reduction in uranium requirements. However, assuming a constant refueling outage time per cycle and the same operating capacity factor, increasing the cycle length results in a lower total power production cost due to an increase in the overall capacity factor.

The economics for pressurized water reactors and boiling water reactors are presented below. The economic analysis consists of the determination of three parameters; uranium requirements, fuel cycle cost, and total power production cost. The uranium requirements are defined as the weight of mined uranium to produce a unit of electricity, MTU/GWe-YR. The fuel cycle costs are composed of U_3O_8 purchase, conversion, separative work, fabrication, storage, and in-reactor inventory charges. Total power production costs consist of fuel cycle costs, plant capital costs, and maintenance and operating costs. Fuel cycle and total power production costs are in units of mills/kwhe. The values used in the economic analysis for the various components of the fuel cycle and power production costs are shown in Table 3.1. A summary of the characteristics of the plants considered in this report is presented in Table 3.2. Table 3.3 presents an estimate of the number and type of nuclear plants operating, under construction, or ordered as of August 1978. The majority of these reactors are expected to be operating by 1990.

The sensitivity of the economic results to alterations in various input parameters is discussed in Section 3.3. Three and four-loop pressurized water reactors are used to illustrate the sensitivity of the results.

3.1 PRESSURIZED WATER REACTORS

The detailed economic analysis presented below pertains to two, three, and four-loop Westinghouse pressurized water reactors. The two-loop reactor analyzed operates at 1,520 MWt with 121 assemblies in the core. The

fuel pins are arranged within the assembly in a 14x14 lattice with seventeen non-fuel locations. The three-loop reactor analyzed operates at 2,300 MWt with 157 assemblies in the core. The fuel pins are arranged within the assembly in a 15x15 lattice with 21 non-fuel locations. The four-loop reactor analyzed operates at 3,250 MWt with 193 assemblies in the core. The fuel assemblies for the four-loop plants and the three-loop plants are identical. The fuel design used in this study is the ENC standard reload design for each reactor type.

The enrichment versus discharge exposure relationships for pressurized water reactors are based on calculations using ENC standard methods. The microscopic and macroscopic cross sections as a function of exposure were calculated with XPOSE⁽⁸¹⁾. XPOSE is an Exxon Nuclear modified version of LEOPARD which is based on a modified MUFT-SUFOCATE model. The XPOSE code determines fast and thermal spectra using only geometry and temperature data as input. A burnup option is available which accounts for the variation of neutron spectra with fuel depletion. The burnup calculation is nonspatial.

The cross sections generated by XPOSE are then input to the reactor simulator code XTGPWR⁽⁸¹⁾. XTGPWR is a modified two-group coarse mesh reactor simulator code which can be run in either two or three dimensions. The calculations for this study were performed in two dimensions with automatic shuffling between cycles. The model is designed to accept regular or MND group cross sections and simulate the reactor for any steady state situation. MND cross sections were used in this study. The physical

conditions under which the cross sections were generated are input, and XTGPWR utilizes this information to store all cross sections on a common base (i.e., zero boron, zero xenon). The cross sections are recalled and modified to fit the reactor conditions and account for Doppler effects, time and power dependent iodine and xenon, boron, power and moderator density changes, burnable poison, and controls.

The results of the analyses are presented on a relative basis for each reactor type. The base case for each reactor type was assumed to be a twelve month cycle with a 90% capacity factor while operating and a six week refueling outage. This results in an overall capacity factor of 80%. The reference batch size was approximately one-third core. This base case has approximately a 30 MWD/Kg discharge exposure for the two and three-loop reactors and a 34 MWD/Kg discharge exposure for a four-loop case. The uranium requirements, fuel cycle costs and total power production costs as a function of batch size, cycle length and batch discharge exposure for each of these reactor types are shown in Figures 3.1 through 3.13 and Tables 3.1 through 3.12.

3.1.1 Uranium Requirements

The uranium requirements for two, three, and four-loop reactors are shown in Figures 3.1, 3.2, and 3.3. Uranium utilization as a function of batch size, discharge exposure, and cycle length is similar for all three reactor types. For constant cycle lengths the uranium requirements decrease with increasing burnup since the batch size is decreasing.

For increased cycle length at constant burnup the uranium requirements increase. If both the cycle length and burnup are increased, the uranium requirements decrease providing the burnup is increased sufficiently to compensate for the longer cycles. The variation in uranium requirements is greater as a function of batch size at constant cycle length than as a function of cycle length at constant batch size.

The uranium requirements behavior due to various changes in the reference case is summarized in Table 3.4. Decreasing the reload batch size from one-third to one-fifth core while maintaining constant cycle length results in an uranium requirement decrease of 15.2% for a two-loop reactor, 11.6% for a three-loop reactor and 8.7% for a four-loop reactor. The batch discharge exposure for the one-fifth core reload is 49 MWD/Kg for the two-loop reactors, 48 MWD/Kg for the three-loop reactors and 55 MWD/Kg for the four-loop reactors. Increasing the batch discharge exposure to 50 MWD/Kg, at a constant one-third core batch size, results in 4.7% decrease in uranium requirements for a two-loop reactor, a 1.9% decrease for a three-loop reactor and a .7% decrease for a four-loop reactor. For constant batch size, increasing the cycle length from twelve to eighteen months results in approximately a 50% increase in burnup and a decrease in uranium requirements of 4.4% for a two-loop reactor, 1.8% for a three-loop reactor and .3% for a four-loop reactor. Thus, it can be seen that there is a significantly greater decrease in uranium requirements by increasing the burnup through

reducing the batch size than for increasing the burnup through increased cycle length at constant batch size. Increasing the cycle length while maintaining a constant burnup results in an increase in uranium requirements. The amount of uranium necessary to produce 30 GWe-YR's of electricity is shown in Table 3.5 for the three reactor types. This table also demonstrates that larger improvements in uranium utilization are achieved when burnup is increased by reducing the batch size than by increasing the cycle length.

The differences in uranium requirements between the three reactor types are due in part to the different lattice structures. Additionally, the four-loop plants operate at power densities about 15% higher than the two and three-loop cores. It should also be noted that the base case discharge exposure for the four-loop plant is approximately 4 MWD/Kg greater than for the two and three-loop plants.

3.1.2 Fuel Cycle Costs

The fuel cycle costs for two, three and four-loop reactors are shown in Figures 3.4, 3.5, and 3.6. The fuel cycle costs are similar for the three reactor types. Fuel cycle costs increase with increasing cycle length and batch size but decrease with exposure at constant cycle length.

The variations in fuel cycle cost due to various changes in the base case are presented in Table 3.6. Reducing the reload batch size from one-third core to one-fifth core while increasing burnup and maintaining

constant cycle length results in a 10.3% reduction in fuel cycle costs for the two-loop reactor, a 7.2% reduction for the three-loop reactor and a 3.8% reduction for the four-loop core. Increasing the discharge exposure to 50 MWD/Kg from the reference case while maintaining a constant batch size (i.e., longer cycles) results in a 1.2% increase in fuel cycle costs for the two-loop core, a 3.5% increase for the three-loop core and a 2.7% increase for the four-loop core. Increasing the cycle length to eighteen months while increasing the burnup and keeping the batch size constant results in a .3% decrease in the fuel cycle cost in the two-loop reactor, a 2.3% increase in fuel cycle costs for the three-loop reactor, a 4.2% increase in fuel cycle costs for the four-loop reactor.

The components of the fuel cycle cost both increase and decrease with increasing batch discharge exposure. At constant batch size, as the batch average discharge exposure increases, the cost in mills/kwhe for separative work and in-reactor inventory increases. The cost decreases for U_3O_8 purchase, fabrication, conversion, and spent fuel storage. The total fuel cycle cost first decreases and then increases as a function of exposure at constant batch size with the trend toward increasing fuel cycle costs dominating over the exposure range of interest, 30 to 50 MWD/Kg. The separative work cost and in-reactor inventory charges are increasing faster with increasing exposure at constant batch size than the decrease in uranium requirements is reducing the other costs. The fuel cycle cost decreases as

a function of discharge exposure for constant cycle length and variable batch size. As batch size is reduced, increasing the exposure, the fuel cycle costs decrease significantly due to the large decreases in uranium requirements. The separative work cost and in-reactor inventory cost increase slightly as the exposure is increased by reducing the batch size but the decreased uranium requirements more than offset this effect.

3.1.3 Power Production Costs

The total power production costs are presented in Figures 3.7, 3.8, and 3.9 for the two, three and four-loop reactors. The total power production costs exhibit similar behavior for the two, three and four-loop reactors. Total power production costs decrease with increasing cycle length and decrease with decreasing batch size.

A summary of the effects on total power production cost due to changes in the reference case parameters is presented in Table 3.7. When the batch size is reduced from one-third core to one-fifth core and the cycle length is held constant, the total power production costs decrease by 3.3% for a two-loop reactor, 2.2% for a three-loop reactor and by 1.1% for a four-loop reactor. Increasing the batch discharge exposure to 50 MWD/Kg from the reference case while keeping the batch size constant results in a decrease in the total power production costs by 2.6% for a two-loop reactor, by 1.9% for a three-loop reactor, and by 1.2% for a four-loop reactor.

Increasing the cycle length to eighteen months while keeping the batch size constant decreases the total power production costs by 2.5% for a two-loop reactor, 1.8% for a three-loop reactor and 1.2% for a four-loop reactor.

Increasing the cycle length, and batch discharge exposure, at a constant batch size results in a reduction in total power production costs. The reduction in total power production costs occurs even though the fuel cycle costs are increasing at constant batch size. The reason for the decrease in total power production costs is that the capital charges are allocated against a larger yearly production of electricity as the cycle length is increased. With a constant outage time and constant operating capacity factor the overall capacity factor increases as the cycle length increases.

The advantages of increased batch discharge exposure, for a constant batch size, relies heavily on the assumption of equal outage time and equal capacity factor while operating for twelve and eighteen month cycles. If the outage time were to increase about 30% for the eighteen month cycle over the twelve month cycle, there would be little or no advantage to increased burnup from long cycles though the advantage from reduced batch size would remain.

3.2 BOILING WATER REACTORS

A typical boiling water reactor with an electrical output in the range 800 to 1,100 Mwe was used for the BWR economic analysis. A summary of the BWR plant characteristics is presented in Table 3.2. The reactor was

assumed to run at 85% capacity while operating and to have an eight week refueling outage. The BWR economic results in terms of uranium utilization, fuel cycle costs and total power production costs are presented on a relative basis as a function of cycle length and batch discharge exposure. The results are relative to a reference case consisting of an annual cycle, approximately one-fifth core reload, and a discharge exposure of 28 MWD/Kg.

The enrichment versus discharge exposure relationships for boiling water reactors are based on calculations performed with the simulator code XRA.⁽⁸²⁾ XRA is a derivative of the basic FLARE⁽⁸³⁾ code and calculates nodal power distributions for a three dimensional reactor core. The XRA calculation, based on one-group diffusion theory, is iterative in two modes: the first being source of power, while the second is the void iteration. Basically, the only inputs involved in the source solution are the infinite multiplication factor of the nodes and the corresponding migration areas.

The nodal infinite multiplication factor is input as a function of the following:

- Local moderator density
- Presence (or absence) of adjacent control rods
- Power dependent equilibrium xenon and Doppler reactivity
- Local fuel exposure
- Local burnable poison (moderator density and exposure dependent)

The migration area is input as a function of local moderator density.

The average nodal steam quality is based on inlet velocity, inlet enthalpy, and power integrated from the bottom of the core to the node of interest. The nodal void fraction is then calculated by a fit to a void-quality correlation curve.

The K_{∞} versus exposure relationships used in the simulator code are calculated with XFYRE.⁽⁸⁴⁾ The XFYRE code is intended to provide an accurate method of predicting local power distributions, reactivity and other important assembly neutronic characteristics as a function of burnup. The XFYRE computer program consists of two major modules. The first module calculates the epithermal and thermal microscopic cross sections of each isotope of each fuel rod type. The second major module of XFYRE consists of the assembly calculations to determine flux and power distributions and isotopic concentrations of each fuel pin. This module prepares the mesh and material zone intervals, performs iterative four-group flux solutions and makes the isotopic transmutations for each fuel rod at each time interval.

3.2.1 Uranium Requirements

The uranium requirements as a function of discharge exposure for a BWR reactor are shown in Figure 3.10 for a twelve month cycle, eighteen month cycle and quarter core reload. The uranium requirements decrease with discharge exposure at both constant cycle length and constant batch size. The larger rate of decrease in uranium requirements at constant cycle length is due to the decrease in batch size with increasing

discharge exposure. As in PWR's the uranium requirement is strongly influenced by the batch size. At a given burnup on eighteen month cycle always requires more uranium per unit energy than a twelve month cycle. For batch average discharge burnups of 28 MWD/Kg an eighteen month cycle requires 7.5% more uranium than a twelve month cycle. However, at near 50 MWD/Kg burnup an eighteen month cycle requires only about 4% to 5% more uranium than a twelve month cycle. These data indicate that high burnups are even more uranium efficient for longer cycle lengths.

The variations in uranium requirements for BWR's with changes in various parameters are summarized in Table 3.8. At a constant cycle length of twelve months, a reduction from one-fifth to one-sixth core reload results in a 6.7% decrease in uranium requirement. Increasing the discharge exposure from 28 to 50 MWD/Kg at a constant one-fourth core reload results in a 9.5% decrease in uranium requirements. Increasing the cycle length at a constant one-fifth core batch size from twelve to eighteen months yields a 9.1% decrease in uranium requirement.

3.2.2 Fuel Cycle Costs

Fuel cycle costs as a function of discharge exposure for a typical BWR are shown in Figure 3.11 for twelve month cycles, eighteen month cycles, and one-fourth core reloads. The fuel cycle costs decrease as a function of discharge exposure at constant cycle length. At a constant quarter core reload the fuel cycle costs first decrease then increase with

exposure, reaching a minimum at about 40 MWD/Kg. The fractional savings as a function of burnup are about the same for both twelve month and eighteen month cycles. The fuel cycle costs per unit of energy are larger for an eighteen month cycle than for a twelve month cycle at a given discharge burnup because of the higher initial enrichments and larger batch fractions required in an eighteen month cycle.

The variations in fuel cycle costs with changes in the input parameters are summarized in Table 3.8. A reduction in the batch size from one-fifth to one-sixth core results in a 4.4% reduction in fuel cycle costs. For a constant one-fourth core reload increasing the batch discharge exposure from 28 to 40 MWD/Kg results in a 2.3% reduction in fuel cycle costs while increasing the discharge exposure from 28 to 50 MWD/Kg results in a 9.0% reduction in fuel cycle costs. Increasing the cycle length from twelve to eighteen months at a constant one-fifth core batch size causes the discharge burnup to increase from 28 MWD/Kg to 45 MWD/Kg and results in a 2.8% reduction in fuel cycle costs.

3.2.3 Total Power Production Costs

The total power production costs versus batch discharge exposure are shown in Figure 3.12 for a twelve month cycle, an eighteen month cycle, and a one-fourth core reload. The total power production costs decrease as a function of batch discharge exposure at both constant cycle length and batch size.

A capital carrying charge and a maintenance and operating cost were added to the fuel cycle cost to obtain a total power production cost. The capital carrying charge represents about two-thirds of the total power production cost. Since capital carrying charges are applied regardless of whether the plant is producing electricity or not, it is advantageous from a cost per unit of electricity standpoint to produce the maximum amount of power per year. With the assumption of an eight week refueling outage per cycle and a 85% capacity factor while operating, an eighteen month cycle results in an overall capacity factor of 76.3% while a twelve month cycle results in a capacity factor of 72%. Under this assumption an eighteen month cycle will produce 5.9% more electricity per year than a twelve month cycle. When the increased energy production per year is incorporated into the total power production cost, eighteen month cycles result in a lower power production cost than twelve month cycles. Note that the margin of savings of eighteen month cycles over twelve month cycles increases with increasing burnup. This means higher burnups are even more desirable for eighteen month cycles than for annual cycles.

The variation in total power production as a function of batch size is highlighted in Table 3.8. At a constant twelve month cycle length, decreasing the batch size from one-fifth core to one-sixth core results in a 1.3% reduction in total power production costs. Increasing the discharge exposure at a constant one-fourth core reload from 28 to 50 MWD/Kg

results in a 3.4% reduction in total power production costs. At a constant one-fifth core reload increasing the cycle length from twelve months to eighteen months reduces the total power production costs by 4.2%.

3.3 SENSITIVITY OF THE ECONOMIC ANALYSIS

To add perspective to the implications of the results presented in Sections 3.1 and 3.2, calculations to indicate the sensitivity of the results to a few economic input parameters were performed. The sensitivity analysis was performed using three and four-loop Westinghouse reactors. The economic results presented in Sections 3.1 and 3.2 depend heavily upon the calculated relationship between enrichment and discharge exposure and upon the economic assumptions stated in Table 3.1. In this section the economic sensitivity is discussed in relationship to (1) the calculated enrichment versus discharge exposure; (2) the price of U_3O_8 ; (3) the cost of the nuclear plant; and (4) the fuel fabrication cost.

3.3.1 Uranium Enrichment Uncertainty

The variation of the total power production costs due to variations in the calculated enrichment versus discharge exposure relationship is presented in Table 3.9. A high and a low enrichment versus batch discharge exposure relationship was created by adding an additional uncertainty of 0.005 in enrichment for each MWD/Kg of exposure above 30 MWD/Kg. The following mathematical relationships were used:

$$\text{High Enrichment} = (\text{base enrichment} + .05) + (\text{exposure (MWD/Kg)} - 30) * .005$$

$$\text{Low Enrichment} = (\text{base enrichment} - .05) - (\text{exposure (MWD/Kg)} - 30) * .005$$

The enrichments utilized in the three-loop core analysis were adjusted through the above relationship and the economics were calculated for the new enrichments. The range of enrichments represented by the above equation represents a qualitative estimation of the uncertainty in the calculated enrichment versus discharge exposure.

It can be seen from Table 3.9 that decreasing the enrichment in the above manner increases the advantage of increased batch discharge exposure while increasing the enrichment does the reverse. Altering the calculated enrichment versus discharge exposure changes the uranium utilization and thus the fuel cycle and total power production cost. The effect of these enrichment alterations on uranium utilization is shown in Figure 3.13. The overall conclusions are not altered for any of these enrichments versus discharge exposure relationships. The total power production cost is still reduced by increasing the discharge exposure whether by reducing the batch size or increasing the cycle length. The increased capacity factor for longer cycle lengths still more than compensates for the increased fuel cycle costs.

Based on the above economic analysis there is a significant incentive to reduce the uncertainty in the enrichments required for high burnup fuels prior to actually committing to high burnup. Unfortunately almost all data that can be acquired from analysis of lead assemblies has a higher degree of uncertainty than that associated with the enrichment

requirement. Critical experiments on unirradiated fuels are of little help since the uncertainty of high burnup fuels is not the initial reactivity but the reactivity as a function of burnup. Data on reactivity as a function of burnup could be gathered by detailed isotopic analysis of a few highly burned lead assemblies. These data would help, but the uncertainty in isotopes versus burnup represents only a part of the total uncertainty associated with uranium enrichment requirements versus burnup. The major component of uncertainty in the calculation of the reactivity is due to uncertainty in the isotopic cross sections at high burnup and to the calculational methods. Analysis of data from the Multiple Cycle Plutonium Utilization Program may be of some help in this area.

This recommended solution to the uncertainty in uranium enrichment requirements versus burnup is full size reloads in actual operation reactors. The confirmation data will be the cycle length achievable as a function of enrichment and batch size. This method of solution will take a few more years to achieve results, but the results will be more conclusive than any tests which can be run. As additional confirmation and as a check against the high burnup isotopic calculations from current computer codes, a limited number of isotopic analysis should be performed on fuels which have achieved a burnup of 50 MWD/Kg or more.

3.3.2 Variation in Price of Uranium, Fuel Fabrication and Nuclear Power Plant Costs

The effect of variation in the price of U_3O_8 on the total power production costs is shown in Table 3.10. The price of U_3O_8 has a small effect on the reduction in total power cost which is due to high burnup achieved by a reduced batch size, but has a larger effect on the savings due to high burnups which were achieved by a longer cycle lengths. This is because of the increased uranium requirements for longer cycle lengths. For \$40/lb U_3O_8 , a one-fifth core reload is 1.1% lower in total power production costs than a one-third core reload and for \$60/lb U_3O_8 it is 1.3% lower. However, for twelve versus eighteen month cycles, the total power production costs are 1.6% lower for eighteen month cycles at \$20/lb U_3O_8 and 0.8% lower for eighteen month cycles at \$60/lb U_3O_8 .

The sensitivity of total power production cost to the plant cost is shown in Table 3.11. The reductions in total power costs which result from increased burnup decrease as a function of plant cost for high burnup achieved by reduced batch sizes but increase as a function of plant cost for high burnup achieved by increased cycle length. The table indicates that the advantages of eighteen month cycles depend significantly on the plant cost.

In Section 2.1 it was estimated that the fabrication costs for which are capable of a batch average discharge of 50 MWD/Kg may be from 25 to 50% higher than for fuels capable of current discharge exposures.

The figures and tables in this report were generated with the fuel fabrication cost remaining constant as a function of burnup. Using the economic parameters in Table 3.1, the fuel fabrication cost is 7.2% of the total fuel cycle costs. From Figure 3.6 and Table 3.6 the fuel cycle cost savings for a four-loop plant which are achieved by increasing the burnup from 34.2 MWD/Kg to 54.7 MWD/Kg through reducing the batch size from a one-third core to a one-fifth core, while at constant cycle length, is 3.8% of the fuel cycle cost. This is equivalent to 53% of the total fabrication cost. This means if the fabrication cost increased 53% to produce fuel capable of high burnup, the fuel cycle cost for the high burnup fuel would be equal to that for the current design fuel at the lower fabrication cost. These data indicate that the projected 25% increase in fabrication costs to produce fuels capable of high burnup would negate about 50% of the potential savings. A 50% increase would negate nearly all the potential savings. The economic data in Table 3.6 indicate the savings for two-loop and three-loop PWR's is larger and significantly higher percentage increases in the fabrication costs could be tolerated prior to reaching the breakeven point. The savings in the fuel cycle cost for high burnup fuels in BWR's is comparable to the four-loop plant (Table 3.8) which means the fabrication cost can increase comparably to reach the breakeven point.

3.4 BURNABLE POISON DESIGNS

In pressurized water reactors burnable poisons are used to control the power distribution and to control the moderator temperature coefficient.

The use of high burnup fuel, which implies high enrichment fuel, may require the use of burnable poisons for one or both of these purposes. There are presently three major burnable poison designs used in pressurized water reactors. These consist of: (1) rods of boron silicate glass placed in the guide tube channels or replacing fuel rods; (2) gadolinia mixed in the fuel pellets of selected fuel rods; and (3) rods containing boron carbide in aluminum oxide. The gadolinia design has distinct advantages over the other two designs. These are: first, no fuel or water displacement occurs with the use of gadolinia. Since PWR assemblies are under moderated this is important. Second, with gadolinia very little residual poison is left at the end of a cycle. There is a substantial poison effect at the end of cycle due to structural material and residual boron for both boron silicate glass and boron carbide in aluminum oxide. Third, the removal of the rods containing boron silicate glass or boron carbide in aluminum oxide prior to reinsertion of the assemblies for the second cycle necessitates additional handling of the fuel assemblies and thus costs additional manpower and increases the possibility of damage.

The achievement of high discharge exposures in PWR's by maintaining twelve month cycles and reducing the number of assemblies per batch may not require the use of burnable poisons if conventional out-in-in* loading

* Fuel on the outside of core first cycle, interior of core second and third cycles.

patterns are used. A fifth core reload at about 4.1 w/o will produce discharge exposures over 45 MWD/Kg in all types of PWR's. Traditional loading patterns may be found which maintain acceptable power peaking. For annual cycle lengths the beginning-of-cycle boron concentration is unchanged from current cycles and thus the moderator temperature coefficients are acceptable. However, if low leakage reload patterns such as in-out-in or in-in-out are used, then burnable poison will be required for all cycle lengths. These low leakage loading patterns would be more uranium efficient than the out-in-in loadings. All the PWR calculations in this study used the traditional out-in-in loading pattern. All BWR calculations used scatter loading patterns with highly burned fuel on the edge of the core.

The achievement of increased burnup through increased cycle length will necessitate the use of burnable poison. The primary reason for this requirement is that to achieve a long cycle additional reactivity must be added at the beginning of cycle. The addition of extra reactivity at the beginning of a cycle raises the boron concentration in a PWR and thus creates a more positive temperature coefficient. A general rule is that for each 1 MWD/Kg increase in cycle length, 100 gadolinia pins will be needed (1% reactivity). Since eighteen month cycle lengths are about 5 MWD/Kg longer in duration than twelve month cycles, an eighteen month cycle would require about 500 gadolinia pins to control the temperature coefficient.

The reactivity worth of the gadolinia at the beginning of the cycle is relatively independent of the concentration. If the gadolinia is required only to meet moderator temperature coefficient criteria, the concentration of gadolinia required will be low (i. e. 1-2%). If the gadolinia is required to meet power peaking limits, the concentration will need to be sufficient such that it will not burn out until after that point in the cycle where the power peaking is limiting. This may require higher concentrations if the power peaking is still limiting late in the cycle. Typically a 4 to 5% concentration of gadolinia will not burn out prior to 10 MWD/Kg burnup.

Eighteen month cycles for one-third core reloads result in discharge exposures in excess of 45 MWD/Kg. If the mechanical limitations limit the allowable discharge to 40 MWD/Kg, then the eighteen month cycles would require larger than one-third core reloads. Larger than one-third core reloads necessitate the loading of fresh assemblies on the interior of the core which will demand the use of burnable poison to control the power peaking in these assemblies.

3.5 OPERATIONAL RESTRICTIONS AND SURVEILLANCE REQUIREMENTS

Operational and fuel management restrictions are currently being placed on nuclear reactors because of fuel failure and safety or licensing considerations. Many of the restrictions are burnup dependent and would be more limiting at high burnups. However, several research and development programs are being conducted to reduce or eliminate these restrictions. If

successful, these programs would reduce the operational restrictions for high burnup fuel, or at least make the restrictions no more limiting than current fuels.

The current operational recommendations restrict the magnitude and the rates of power changes to avoid fuel failure due to pellet cladding interaction. Limitations on the increase in the power within a fuel are imposed mainly to avoid failure due to pellet cladding interaction and are based on the empirical results from power ramp tests in test reactors and correlations with observed failures. The shortcoming of the empirical approaches is that they are independent of fuel design, do not account for accumulative damage and do not consider the time span since the fuel was preconditioned. Thus, a purely statistical approach provides no near term benefits for improving fuel design or operational control. In addition the results are for a particular fuel design. New high burnup designs would require new empirical correlations.

As the correlation to evaluate failure probability incorporates more details of the fuel performance, more requirements are placed on the fuel management evaluation. Power ramp rates and the time since the last high power are important when considering power changes. Furthermore, when considering power ramp rates, the dynamics of the reactor control systems or methods of increasing power must be incorporated into the fuel management evaluations. This requirement becomes involved for boiling water reactors

where small reactivity adjustments are made by discrete control rod movements and where the power level is lowered before major rod adjustments and the power level is subsequently raised by coolant flow control.

The present criteria used in fuel management correlations for failure probability are based on the variables available from coarse mesh three dimensional core simulator programs which evaluate conditions within the fuel assembly. Fuel failure modeling codes are available for evaluating effects within one fuel rod, and they include many evaluations which are time consuming and require considerable computer core storage. The effective compromises between considering the variables available in the coarse mesh core monitoring programs and those necessary to model the essential characteristics of the rods within a bundle for a probability assessment of fuel failure have yet not be established.

If the fuel management techniques incorporate interactive prediction of failure probability, then increased expenditures for plant equipment and fuel inspection can be expected. Online computers will be required to perform such evaluations and additional bundle sipping and selective inspection of failed fuel will be required to establish the needed confidence levels for the failure algorithms. Hopefully the costs for plant modification and changes in operational procedures required to evaluate the fuel failure probability will be more than compensated for by improved fuel performance and increased operational flexibility compared to the presently recommended operational restrictions.

The implementation of interactive failure probability estimation is not a requirement for using high burnup fuel designs, but the additional economic incentives for using such systems with fuel neutronically designed to reach high exposures may be associated with the implementation of operation with high exposure fuel.

3.6 ECONOMIC RISKS OF HIGH BURNUP FUELS

The largest economic risk of high burnup fuels is from failure prior to achieving full burnup. The economic risks from fuel failure are a combination of two items: one; with no reprocessing capability the unused enriched uranium in the failed fuel is unavailable to produce energy: and two, failed fuel could cause reactor down time.

Fuels designed for high burnup require high enrichments. These fuels cost significantly more due to the increased requirements for U_3O_8 and separative work units of enrichment, and therefore the economic consequence of a failure are greater than failure of fuels designed for lower burnup. Data indicate that the failure rate of current fuel designs is exposure and power dependent. If the fuel failure rate is strongly dependent on exposure then new fuel designs will be required to reduce the risk of failure at high exposures.

To indicate the degree of risk of high exposure fuels Figure 3.14 compares fuel with a goal exposure of 40 MWD/Kg to fuel with a goal exposure of 30 MWD/Kg. To achieve equal uranium utilization, compared to fuel which

is designed for and achieves 30 MWD/Kg, the fuel designed for 40 MWD/Kg burnup must achieve at least 89% of 40 MWD/Kg or 35.6 MWD/Kg. To achieve equal fuel cycle costs high burnup fuel must achieve about 97% of goal exposure. Therefore, it is important that high burnup fuels have no greater probability of failure than fuels designed for current burnups. Changes in fabrication techniques or new fuel designs may be required to achieve this goal. Section 3.3 indicates that the fabrication costs for high burnup fuel can be increased substantially and still not negate the economic incentive for high burnup fuels. This is because fabrication represents less than ten percent of the current fuel cycle costs. As a result, there is considerable economic margin to implement new designs or innovations which may be necessary to achieve the goal of no increase in fuel failure at higher burnup.

Figure 3.1. Uranium Requirements as a Function of Cycle Length for a Westinghouse 2 Loop Plant.

Figure 3.2. Uranium Requirements as a Function of Cycle Length for a Westinghouse 3 Loop Plant.

Figure 3.3. Uranium Requirements as a Function of Cycle Length for a Westinghouse 4 Loop Plant.

Figure 3.4. Fuel Cycle Costs as a Function of Cycle Length for a Westinghouse 2 Loop Plant.

Figure 3.5. Fuel Cycle Cost as a Function of Cycle Length for a Westinghouse 3 Loop PWR.

Figure 3.6. Fuel Cycle Cost as a Function of Cycle Length for a Westinghouse 4 Loop Plant.

Figure 3.7. Total Power Production Cost as a Function of Cycle Length for a Westinghouse 2 Loop Plant.

Figure 3.8. Total Power Production Cost as a Function of Cycle Length for a Westinghouse 3 Loop PWR.

Figure 3.9. Total Power Production Cost as a Function of Cycle Length for a Westinghouse 4 Loop Plant.

Figure 3.10. Uranium Requirements as a Function of Discharge Exposure for a BWR.

Figure 3.11. Fuel Cycle Cost as a Function of Discharge Exposure for a BWR.

Figure 3.12. Total Power Production Costs as a Function of Discharge Exposure for a BWR.

Figure 3.13. Uranium Requirements for a High and Low Enrichment Uncertainty Estimation (3 Loop Westinghouse PWR).

Figure 3.14. Percent Savings in Fuel Cycle Costs and Uranium Utilization for High Exposure vs the Percent of Achieved Goal Exposure.

Table 3.1 Parameters for Economic Analysis

<u>Parameter</u>	<u>Cost*</u>
Capital Cost, \$/Kwe	800
Maintenance and Operating, mills/kwhe	2.0
U ₃ O ₈ , \$/lb	40
Separative Work, \$/Kg SWU	100
Conversion, \$/lb U	1.81
Fabrication \$/KgU	110**
Spent Fuel Disposal Cost Including Shipping, \$/KgU	135
Fuel Cycle Carrying Charge Rate	7.7
Capital Carrying Charge Rate	10.1
Pu Recycle	none
U Recycle	none

* In constant 1979 dollars

** For current LWR fuel

Table 3.2 Characteristics of Nuclear Plants Used In Analysis

Characteristic	2 Loop	3 Loop	4 Loop	BWR
Reactor Power Level MWt	1520	2300	3250	2381
No. Assemblies	121	157	193	548
Core Height (ft.)	12	12	12	12
Assembly Pitch (in.)	7.8	8.5	8.5	6
No. Fuel Pins Per Assembly	179	204	204	62
No. Guide Tubes & Instrument Tubes	17	21	21	0

Table 3.3 U.S. Reactors Which Were Operating, Under Construction
or on Order as of August, 1978

<u>Reactor Type</u>	<u>No. of Reactors</u>
Westinghouse 2 Loop	9
Westinghouse 3 Loop	21
Westinghouse 4 Loop	45
General Electric Boiling Water Reactors	60
Combustion Engineering	33
Babcock & Wilcox	29
Miscellaneous	3

Table 3.4 Uranium Requirements PWR's

<u>Change in Reference Case</u>	<u>Reactor Type</u>		
	2 Loop	3 Loop	4 Loop
<hr/>			
Reduce Batch Size From $\sim 1/3$ Core to $\sim 1/5$ Core (Constant Cycle Length)			
Discharge Exposure for $\sim 1/3$ Core (MWD/Kg)	29.6	29.8	34.2
Discharge Exposure for $\sim 1/5$ Core (MWD/Kg)	49.3	48.4	54.7
% Change in U Requirements	-15.2	-11.6	-8.7
<hr/>			
Increase Discharge Exposure Keeping Batch Size Constant ($\sim 1/3$ Core)			
Initial Discharge Exposure (MWD/Kg)	29.6	29.8	34.2
Final Discharge Exposure (MWD/Kg)	40	40	40
% Change in U Requirements	-3.8	-1.8	-.7
Final Discharge Exposure (MWD/Kg)	50	50	50
% Change in U Requirements	-4.7	-1.9	-.6
<hr/>			
Increase Cycle Length From 12 to 18 Months Keeping Batch Size Constant			
Discharge Exposure, 12 Month Cycle (MWD/Kg)	29.6	29.8	34.2
Discharge Exposure, 18 Month Cycle (MWD/Kg)	46.3	46.6	53.5
% Change in U Requirements	-4.4	-1.8	-.3
<hr/>			

Table 3.5 Uranium and Separative Work Requirements for
30 GWe-Years Electricity Production

Reactor	Batch Size	12 Month Cycles			18 Month Cycles		
		Burnup MWD/Kg	Uranium*	SWU**	Burnup MWD/Kg	Uranium*	SWU**
Westinghouse 2 Loop (121 Assemblies)	~1/3 Core	29.6	6328	4941	46.3	6046	5449
	~1/5 Core	49.3	5365	4752		--	
	% Change		-15.2	-4.0			
Westinghouse 3 Loop (157 Assemblies)	~1/3 Core	29.8	5324	4509	46.5	5820	5195
	~1/5 Core	48.4	5229	4567		--	
	% Change		-11.6	+1.3			
Westinghouse 4 Loop (193 Assemblies)	~1/3 Core	34.2	5736	4583	53.5	5734	5306
	~1/5 Core	54.7	-5253	4770		--	
	% Change		8.7	+4.1			
Boiling Water Reactor	~1/4 Core	21.2	6619	4494	33.4	6146	4746
	~1/6 Core	32.0	5632	4329		--	
	% Change		-14.9	-3.8			

* Uranium requirements in MTU/(30 GWe-years)

** Separative work units in MTSW/(30 GWe-years)

Table 3.6 Fuel Cycle Cost - PWR

<u>Change in Reference Case</u>	<u>Reactor Type</u>		
	2 Loop	3 Loop	4 Loop
Reduce Batch Size From $\sim 1/3$ Core to $\sim 1/5$ Core (Constant Cycle Length)			
Discharge Exposure for $\sim 1/3$ Core (MWD/Kg)	29.6	29.8	34.2
Discharge Exposure for $\sim 1/5$ Core (MWD/Kg)	49.3	48.4	54.7
% Change in Fuel Cycle Cost	-10.3	-7.2	-3.8
Increase Discharge Exposure While Keeping Batch Size Constant ($\sim 1/3$ Core)			
Initial Discharge Exposure (MWD/Kg)	29.6	29.8	34.2
Final Discharge Exposure (MWD/Kg)	40	40	40
% Change in Fuel Cycle Cost	-1.0	+5	+5
Final Discharge Exposure (MWD/Kg)	50	50	50
% Change in Fuel Cycle Cost	+1.2	+3.5	+2.7
Increase Cycle Length From 12 to 18 Months Keeping Batch Size Constant			
Discharge Exposure 12 Month Cycle (MWD/Kg)	29.6	29.8	34.2
Discharge Exposure 18 Month Cycle (MWD/Kg)	46.3	46.6	53.5
% Change in Fuel Cycle Cost	-.3	+2.3	+4.2

Table 3.7 Total Power Production Cost - PWR

<u>Change in Reference Case</u>	<u>Reactor Type</u>		
	2 Loop	3 Loop	4 Loop
Reduce Batch Size From $\sim 1/3$ Core To $\sim 1/5$ Core (Constant Cycle Length)			
Discharge Exposure for $\sim 1/3$ Core (MWD/Kg)	29.6	29.8	34.2
Discharge Exposure for $\sim 1/5$ Core (MWD/Kg)	49.3	48.4	54.7
% Change in Cost	-3.3	-2.2	-1.1
Increase Discharge Exposure While Keeping Batch Size Constant ($\sim 1/3$ Core)			
Initial Discharge Exposure (MWD/Kg)	29.6	29.8	34.2
Final Discharge Exposure (MWD/Kg)	40	40	40
% Change in Cost	-2.3	-1.6	-1.0
Final Discharge Exposure (MWD/Kg)	50	50	50
% Change in Cost	-2.6	-1.9	-1.2
Increase Cycle Length From 12 to 18 Months Keeping Batch Size Constant			
Discharge Exposure 12 Month Cycle (MWD/Kg)	29.6	29.8	34.2
Discharge Exposure 18 Month Cycle (MWD/Kg)	46.3	46.6	53.5
% Change in Cost	-2.5	-1.8	-1.2

Table 3.8 BWR Economics

<u>Parameter Change</u>	<u>% Change In Economic Parameter</u>		
	Uranium Requirements	Fuel Cycle Cost	Total Power Production Cost
Reduce Batch Size From $\sim 1/5$ Core (26.5 MWD/Kg Discharge) To $\sim 1/6$ Core (32.0 MWD/Kg Discharge) (Cycle Length Constant at 12 Months)	-6.7	-4.4	-1.3
Increase Batch Discharge Exposure From 28 To 40 MWD/Kg To 50 MWD/Kg (Constant Batch Size of $1/4$ Core)	-7.3 -9.5	-2.3 -.9	-2.9 -3.4
Increase Cycle Length From 12 Months (28 MWD/Kg Discharge Exposure) to 18 Months (45 MWD/Kg Discharge Exposure) (Constant Batch Size $\sim 1/5$ Core)	-9.1	-2.8	-4.2

Table 3.9 Effect of Enrichment Versus Discharge Exposure
Upon Total Power Production Cost, 3-Loop Core

<u>Change in Reference Case</u>	<u>% Change in Total Power Production Cost</u>		
	High* Enrichment	Base Enrichment	Low** Enrichment
Reduce batch size from ~1/3 Core (29.8 MWD/Kg Discharge) to ~1/5 Core (48.4 MWD/Kg Discharge) (Constant Cycle Length)	-1.7	-2.2	-2.7
Increase batch discharge exposure from 30 to 40 MWD/Kg: from 30 to 50 MWD/Kg: (Constant Batch Size)	-1.2 -1.3	-1.6 -1.9	-1.9 -2.3
Increase cycle length from 12 months (29.8 MWD/Kg) to 18 months (46.6 MWD/Kg Discharge) (Constant batch size and increased exposure)	-1.3	-1.8	-2.2

* High enrichment = (base enrichment + .05 wt%) +
(exposure (MWD/Kg) - 30) * .005 wt%

** Low enrichment = (base enrichment - .05) -
(exposure (MWD/Kg) - 30) * .005 wt%

Table 3.10 Effect of U_3O_8 Price Upon Total Power
Production Cost, 4-Loop Core

<u>Change in Reference Case</u>	<u>% Change in Total Power Production Cost</u>		
	$\$20/lb\ U_3O_8$	$\$40/lb\ U_3O_8$	$\$60/lb\ U_3O_8$
Reduce Batch Size From ~1/3 (34.2 MWD/Kg Discharge) ~1/5 Core (54.7 MWD/Kg Discharge) (Constant Cycle Length)	-1.0	-1.1	-1.3
Increase Batch Discharge Exposure From 34.2 MWD/Kg To 40 MWD/Kg: To 50 MWD/Kg: (Constant Batch Size)	-1.0 -1.5	-1.0 -1.2	-.8 -1.0
Increase Cycle Length From 12 Months (34.2 MWD/Kg Dis- charge) to 18 Months (53.5 MWD/Kg Discharge) (Constant Batch Size)	-1.6	-1.2	-.8

Table 3.11 Effect of Plant Cost on Total Power
Production Cost, 4-Loop Core

<u>Change in Reference Case</u>	<u>% Change in Total Power Production Cost</u>	
	<u>\$800/Kwhe*</u>	<u>\$1200/Kwhe*</u>
Reduce Batch Size From $\sim 1/3$ Core (34.2 MWD/Kg Discharge) to $\sim 1/5$ Core (54.7 MWD/Kg Discharge) (Constant Cycle Length)	-1.1	- .9
Increase Batch Discharge Exposure From 34.2 MWD/Kg to 40 MWD/Kg: to 50 MWD/Kg:	-1.0 -1.0	-1.0 -1.8
(Constant Batch Size)		
Increase Cycle Length From 12 Months (34.2 MWD/Kg Discharge) to 18 Months (53.5 MWD/Kg Discharge) (Constant Batch Size)	-1.2	-1.9

* In constant 1979 dollars.

4.0 FUEL PERFORMANCE AT HIGH BURNUP

The proposed peak pellet exposure limits of 60 MWD/Kg for light water reactor fuels are similar to those that have been obtained in prototypic LWR fuel^(46,72,73) and less than those obtained in fast reactor test irradiations^(13,15,26,27) where design exposure limits of 100 MWD/Kg are desired. No fuel material limiting conditions have been experienced in these reactor fuels which have been irradiated to high exposures. The retained fission products seem to reach saturation values, thus even the swelling of the fuels does not become unstable.⁽²⁶⁾ Growth and radiation hardening effects in the zircaloy cladding also seem to be saturated at higher fluences. Coolant induced exterior cladding corrosion should not be a limiting consideration, although in some claddings and coolant chemistries crud films, nodular corrosion, and possible higher corrosion rates⁽⁷²⁾ can occur.

Burnable poison rods have been used successfully in past reactor radiations. The predictability of individual fuel and cladding material properties (see Table 4.1) are satisfactory to perform design analyses and develop candidate fuel designs. Those factors associated with the fuel-cladding interaction and stress corrosion cracking are the least well defined and some licensing concerns have been encountered with gas release. The theory associated with these phenomena are not limiting the use of higher burnup fuel. Thus, this section describes the fuel element responses

that must be considered in reaching high exposure limits. None of these responses seem to create a barrier to reaching goal core average burnups of 45-50 MWD/Kg. Burnups of this magnitude have already been demonstrated with test irradiations, but because these irradiations were conducted as extended cycles of fuel initially designed for less burnup, the power levels reached during the high exposure were not prototypic of fuel designed for extended burnup in full core loads. Thus, the existing data base of prototypic LWR high exposure fuel is small and a statistically significant number of irradiation data are required to develop the needed confidence level for both a licensing data base and to commercially produce multiple numbers of full core loadings of higher burnup fuel.

4.1 FISSION PRODUCT CONSIDERATIONS

In this section an effort has been made to assess the effects of fission products on fuel performance. Release of inert fission gas is one of the most frequently discussed consideration because fission gases decrease the gap conductance and thermal conductivity of the fill gas, thereby decreasing gap conductance and increasing the stored energy of the fuel. The retained fission products and their location are important in determining the fuel behavior⁽²⁶⁾ especially under transient conditions. Release of the volatile products such as cesium and iodine must be considered since these products create a corrosive environment on the inner surface of the fuel cladding. If failure occurs, the release of fission products to the coolant causes a radiation problem that is significant in safety considerations.

4.1.1 Inert Fission Gas Release

The release of the inert fission gases has been recognized as an irradiation dependent concern in fuel element designs for many years. Thus, measurements of gas release have been made and programs are underway to enlarge the data base of gas release measurements (see Appendix B). The literature^(14,18,19,20,23,28) abounds with theoretical studies concerning gas release and bubble growth and migration. Data are available to support the existence of phenomena being modeled but the number of parameters in these models are so extensive that a more empirical approach is used for performance and licensing calculations. Because the goal exposures for breeder reactors are so much higher than those of light water reactors, measurements of gas release have been made at exposures which are beyond those proposed for light water reactors. The gas release measurements made for high exposure breeder reactor fuels have shown that a significant fraction of the inert fission gas generated is released, but the fuel in fast reactors operates at higher temperatures than that encountered in light water reactor fuel. The retention capability of the fuel material increases as the temperature is lowered.⁽²⁶⁾ The retention capacity at fuel temperatures encountered in the LWR fuels (below 1000°C) must be verified for a broad data base for exposures of up to 60 MWD/Kg.

The models for gas retention and gas release have been developed to a high level of sophistication.^(14,28) These models incorporate consideration of bubble formation, bubble resolution, bubble migration,

retention of bubbles, microstructural defects and bubble interlinkage along grain boundaries. Microstructural evaluations and techniques for such evaluations have been made to support these modeling concepts. The details of the fabrication process and their effect on such modeling approaches are not completely clarified so that the detailed design modifications to improve performance must be verified. Major fuel microstructural fabrication and parameters affecting gas release are known. The confirmation of these concepts to take advantage of these improvements must be supported with a substantial data base. Small grain size and pore size have been shown to lead to unstable fuel microstructures.⁽³¹⁾ Highly connected porosity increases gas release. What must be unequivocally verified is that for fuel with precharacterized stable fuel structures, the retention of inert gaseous fission products⁽²⁶⁾ in low temperature ranges ($T < 900^\circ\text{C}$) increases linearly with burnup up to 60 MWD/Kg. (i.e., the fraction of retained fission products stays constant with burnup). Thus high fission gas release rates should not be experienced in fuel material retained below this temperature for the goal fuel exposures covered by this study.

The earlier data that are publicly available on inert fission gas release were produced under nonprototypic conditions with fuels and irradiation histories that were not well characterized. Design and licensing considerations require that the variances between predictions and data be considered in establishing limiting conditions. If only prototypic

fuel is used, the variance increases because the data base is too small. If the total base is used without the fuel and irradiation conditions being properly characterized, the variance from empirical gas release predictions increases because of the fuel microstructural dependence of gas release. Therefore the data scatter with correlations caused by the lack of a sufficient characterization and a small prototypic fuel data base at high burnup creates a penalty for application to high burnup LWR fuel design.

Licensing calculations require that limiting conditions be considered and such conditions are not encountered where irradiation occurs under normal operating conditions. The high exposure fuel on which data are available normally operated at very low powers in the latter stages of its irradiation. These power levels are not those associated with the licensing limits for high exposure fuel. Thus, for the licensing evaluations of empirical gas release models, the data base must include data at the desired licensing limits of the high exposure fuel.

4.1.2 Retention of Fission Products

The retained fission products affect the fuel material behavior upon subsequent irradiation. When the fission products are not released, then the fuel pellet will undergo volume increase. At higher burnups and in the higher temperature regions of the fuel where fission bubble swelling occurs, the rate of release of the generated volatile fission products increases with burnup.⁽²⁶⁾ Fuel has a limited retention capability

of the gaseous and volatile fission products at temperatures greater than 900 C. Because of this the rate of fuel swelling in the high power fuel decreases at higher burnups and the rate of steady state deformation of the cladding in the limiting power fuel rods caused by fuel swelling decreases. The actual magnitude of the gaseous fission products retained can decrease in the high temperature nonrestructured and equiaxed structured regions because grain boundary bubbles interconnect and decrease the retention capacity.

One of the possible higher burnup transient fuel behaviors that has only partially been clarified is the role of volatile fission products such as cesium during temperature increases. The volatile cesium tends to be trapped in bubbles. If the bubbles are on the grain boundary and the cesium is in a volatile chemical state and the temperature increases to the vaporization range then the pressure in such bubbles increases rapidly.⁽²⁵⁾ Such pressure increases can cause grain boundary fractures or rapid bubble interlinking depending upon the rate of the temperature increase.⁽²⁴⁾ Also, as exposure increases the oxygen to metal ratio in uranium fuel material may decrease and should the fuel become hypostoichiometric then the chemical form of the retained cesium converts to a volatile cesium. If bubbles are interconnected then more fission products will be released and during temperature increases the fuel material should not encounter sudden volume increases.

Model development and data bases are being established for the consideration of retained fission products.^(24,28) Application of such concepts are important for assuring safe transient fuel behavior.

4.1.3 Pellet Cladding Interaction - Stress Corrosion Cracking

A significant body of information has been obtained to show that low ductility fuel cladding failures observed under power ramping conditions are related to the stress corrosion cracking^(7,68,69) of the zircaloy. This failure mode has been cited to be the one most difficult to avoid for high exposure fuel. Threshold mechanical conditions must be exceeded and the chemical environment must be severe enough for such stress corrosion cracking to occur. Simulation tests have concentrated on the iodine and stress levels necessary to cause such failures. Other constituents, such as cadmium and cesium, are also believed to be important in the stress corrosion process.^(25,69) The dynamics of the migration of cesium and its role in promoting pellet cladding interaction is not fully understood. Patches of reacted fission products have been observed on high burnup fuel claddings. Microcracking at inbedded particles in the cladding has also been observed. A limited number of observations have been made indicating that threshold failure levels do not continue to deteriorate at higher burnups⁽⁷²⁾.

Fuel elements are subjected to a number of power changes during burnup including possible cyclic load following. For multiple

loading conditions the growth of cracks in the cladding must be considered. The role of accumulative damage under cyclic load following is much less understood than the conditions encountered during single power ramps. Cladding creep down and fuel swelling decrease the power increments required to exceed damage thresholds and their influence should be considered in determining accumulative damage. Determination of damage thresholds as a function of exposure and exposure history would provide conservative criteria for the avoidance of damage accumulation.

Fuel performance must also be evaluated under anticipated transient and assumed accident conditions. Throughout the lifetime of the fuel, various types of anticipated transients such as turbine trips, rod movements, boron level dilution, etc. must be considered. Many of these transients result in power increases over short time periods which may not be sufficient time for fission products to migrate from the pellets to the cladding. Thus, not all potential fuel-cladding interactions under changing power conditions can be categorically classified as causing stress corrosion cracking, especially those associated with anticipated plant transient conditions. Tests on irradiated fuel under simulated transient conditions could help clarify the failure thresholds under these conditions.

4.1.4 Release of Fission Products After Cladding Failure

The safety consequences of fuel failure are determined by the amount of radioactive fission products that are released into the

coolant.⁽⁶⁵⁾ As the exposure of the fuel increases, its fission product content increases but the short lived products make the highest contribution to the activity. Thus, activity is power level dependent and may actually decrease for high burnup fuel. Cladding ductility saturates with exposure, thus, should a fuel failure occur at high burnup the exposed area of the fuel should not be any greater than if the failure had occurred at a lower burnup. Thus, activity release should not be enhanced by burnup, and the nature of cladding failures can be checked in those tests conducted to determine failure thresholds.

4.2 DIMENSIONAL STABILITY

Deformational mechanisms affect various fuel element components. Examples of deformational mechanisms are cladding collapse, fuel rod bow, spring relaxation and rod growth. These deformational mechanisms are usually fluence, burnup and time dependent and either become stable or become more limiting at high exposures. They may not result in fuel failure but they do influence fuel performance limits by increasing power peaking, perturbing the flow of the coolant, lowering margins to departure from nucleate boiling limits, creating conditions more favorable for fretting and altering the dynamic characteristics of the fuel element behavior. The avoidance of problems from such deformational mechanisms usually can be accomplished by careful consideration of the design details.

4.2.1 Cladding Collapse

If small grain, small pore sizes, and low density fuel material is used, fuel densification⁽³¹⁾ occurs during irradiation and gaps may be created in the fuel column. The radiation induced creep of the zircaloy cladding can then cause the initial ovality of the cladding to increase to such an extent that the cladding collapses between the gaps in the fuel column. Such behavior has been observed in pressurized water reactors and has resulted in the conservative design requirement of free standing cladding throughout the fuel exposure. The fabrication procedures for making stable fuel structures are much better understood at the present time than they were at the time that cladding collapse was observed. The use of preirradiation densification tests to check pellet fabrication procedures tests have eliminated large gaps being formed in the fuel columns of current fuel. Pressurization of the fill gas is also now used to assure that the cladding will be free standing. During exposure the fuel swells, the internal fill gas pressure increases, the cladding contacts the fuel material and power levels decrease so cladding temperatures decrease. Thus, if the cladding doesn't collapse within the first two cycles of irradiation then the subsequent collapse of the cladding is unlikely. This has been confirmed by gamma scanning and profilometry measurements on selected rods during cycle outages to assure that gaps do not exist in the fuel column and that large increases in ovality have not occurred. Thus, cladding collapse should not be a problem for high burnup fuel.

4.2.2 Cladding Circumferential Deformation

As the burnup of the fuel increases, more fission gas is released into the free volumes of the fuel rod, the cladding creeps down upon the fuel and the fuel material swells. These mechanisms increase the internal pressurization. If the fill gas pressure becomes sufficiently larger than the coolant pressure then the cladding will creep at a rate greater than the fuel swelling rate and the fuel-to-cladding gap will open. If this gap increases, the gap conductance will decrease causing the fuel temperature to increase which in turn causes more fission gas to release and the cladding deformational process becomes unstable. The major uncertainty in the fuel design is the confirmation that sufficient free volume remains and that the gas release is consistent with equations used to describe this behavior.

4.2.3 Bowing

Both fuel bundle bowing and rod bowing have been observed in pressurized water reactors. The bundle bowing is usually observed within the first few cycles and in those bundles that have been located on the core periphery. The mechanism of bundle bow is believed to be caused by the flux tilt which causes a differential growth of zircaloy guide tubes. At higher fluences, the growth tends to saturate and relative differences in guide tube growth decrease. Relocation of fuel bundles between cycles also tends to level out the fluence differences in the guide tubes. The problem

of bundle bowing can be reduced by proper selection of the zircaloy guide tube material. Thus, bundle bowing problems will not increase for high burnup exposure fuels.

Rod bowing and the resultant closure in fuel channels^(74,75) has been observed in pressurized water reactors. The initially observed rod bowing was caused by leaving insufficient axial space for the growth of the zircaloy cladding. The interference forces in the axial direction from tie plate restrictions on rod expansions caused rod bowing of such magnitude that fuel rods touched. Clearances between rods and tie plates were increased to allow for rod growth, but rod bowing was still observed in some fuel designs. Spacer springs can cause axial compressive forces when the fuel cladding elongates due to irradiation induced fuel cladding growth. These lateral forces increase the initial bow by a column buckling type of loading mechanism. Design changes can be made to reduce the effects of such rod bow problems. The rod bow is not an accelerating deformational mechanism since the axial restraint of the spacer springs reduces with exposure. The relative motion of the fuel rods also tends to decrease at higher fluences.

Since the rod bow problem is now recognized, design dependent solutions are available. Should rod bow become limiting in a particular bundle design, rod bow is decreased if the span-length between spacers is decreased. Hence, the use of additional spacers is always an available solution. Even if fuel rods do touch this does not mean that fuel failure

will occur. The major concern of excessive rod bow is one of flow perturbation and a decrease in margin to the departure of nucleate boiling limit. Confirmation that a particular bundle design does not encounter bowing limits is obtained by making rod-to-rod spacing measurements and such data are readily obtained by a demonstration irradiation program.

4.2.4 Spring Relaxation

Relaxation of various springs used in fuel element designs is a problem requiring consideration but should not be limiting to high exposure fuel. The bundle holddown springs, plenum springs, and spacer springs all have functional requirements that can be influenced by spring relaxation.

Holddown springs assure that the bundle will not move due to the forces caused by coolant flow. These springs also influence the dynamic properties of the fuel rod and bundle structural models and affects seismic LOCA analyses. Irradiation does not change the spring coefficients as much as it influences the restraining force on the bundle. The holddown springs of a reactor fuel element are out of the fueled region of the core. The effective neutron fluxes at these springs are several orders of magnitude less than those encountered in the other spring components; their relaxation is predictable and should not limit exposure.

Springs are placed in the plenum of the fuel rods to hold the pellets during shipping. They also ensure stack continuity during the densification period of initial fuel irradiation. Both of these functions

are initial life problems and the only effect of plenum spring relaxation will be its influence on initial restraint forces in the seismic-LCOA model. Again this will be a small effect because the spring coefficients are not changed; only the initial restraint forces are altered, and these effects are predictable. Thus, plenum spring relaxation is not viewed as a limitation to obtaining high burnups.

The spacer springs are used to laterally restrain the fuel rods. Such spacer springs are in the active fuel region and are exposed to high fluences consequently stress relaxation occurs. Stress relaxation is approximately a linear function of stress for fast flux induced relaxation; thus, the rate of relaxation decreases with higher burnups because the steady state stresses applied to the springs are reduced as the springs relax. The relaxation magnitudes, however, can be enough to allow rod liftoff under some cross flow conditions and possibly cause fretting problems in certain core locations. For licensing evaluations of the seismic LOCA event, calculated impact forces of fuel rods with the bundle tie plates will tend to increase as the spring restraints decrease thereby increasing the loadings on the cladding. Verification that present design predictions of spacer spring relaxation are adequate can be obtained from withdrawal force measurements and visual observations for fretting corrosion on lead assemblies during irradiations outages.

4.3 CORROSION LIMITATIONS*

Internal oxide corrosion of zircaloy-clad fuel rods generally is not a problem in light water reactors. Fuel-cladding chemical interactions are not widespread or severe except under extreme temperature conditions.⁽⁷⁰⁾ However, localized fission product attack and cracking of cladding may be quite rapid under stress, especially under extended burnup conditions when deformation of the cladding tends to become localized and cracking results from the large inventory of fission products that have accumulated. The discussion of such stress-corrosion cracking was treated in Section 4.1.

External corrosion of cladding must be accounted for in the design of extended burnup fuel rods. The primary considerations with regard to external corrosion are uniform corrosion (crud buildup and wall thinning), localized corrosion (nodular) and hydrogen pickup. The large reactor-to-reactor system variability has a profound influence on the external corrosion of fuel rods. This variation can overshadow the effect of extended burnup operation. However, as will be discussed later, certain reactor systems may exhibit accelerated corrosion that will limit extended burnup exposures. A nationwide or international comparative study would be needed to determine the reasons for the extremes in LWR corrosion behavior seen between various reactors before accurate predictions and control of corrosion could be achieved.

* Reader should refer to data in WAPD-TM-1412, DOE R&D report, "Corrosion and Hydriding Performance Evaluation of Three Zircaloy-2 Clad Fuel Assemblies after Continuous Exposure in PWR Cores 1 and 2 at Shippingport, Pa", January, 1980.

While crud buildup has been linked to early reactor fuel failures, proper water chemistry control procedures have reduced the effect to acceptable levels. No serious impact on extended burnup operation should occur in those reactor systems with controlled crud buildup under existing coolant chemistry conditions.

Uniform corrosion has been measured in a variety of out-of-pile and in-reactor tests. For BWR fuel assemblies designed to achieve burnups of about 45 MWD/Kg in six years, the wall thinning is calculated to be about 16 μm (0.6 mils) using recommended design correlations.⁽¹⁾ For PWR fuel assemblies designed to achieve about 45 MWD/Kg in about four years, the wall thinning is about 18 μm (0.7 mils). Such effects can be incorporated into extended burnup designs by increasing cladding thickness allowances.

PWR's generally do not exhibit localized cladding corrosion as do BWR's. However, certain PWR's exhibit uniform cladding corrosion far in excess of the values stated above. There is some evidence that water chemistry conditions, such as high oxygen content, accelerate cladding corrosion.⁽⁵⁴⁾ While some PWR oxide thicknesses have been found to be only about 10 μm (0.4 mils) at 38 MWD/Kg, other reactors have had oxide thicknesses of about 45 μm (1.8 mils).⁽⁵⁴⁾ The projected relationship between oxide thickness and burnup indicates unacceptable corrosion at extended burnup in some reactors. Investigations have shown that thick oxide films undergo internal cracking and separations. It is believed that this results in poor thermal conductance

in the film, higher cladding oxide-metal interface temperatures and accelerated corrosion.⁽⁵⁴⁾ More work is needed to characterize the coolant and cladding conditions which lead to accelerated uniform generalized corrosion.

Localized corrosion phenomena, such as nodular corrosion, potentially have a more damaging effect on cladding integrity than uniform corrosion. Nodular corrosion attack has been observed predominately in BWR's⁽⁵⁵⁾ where free oxygen is generated in the core and also has been found in high temperature steam corrosion of zircalloys out-of-pile.⁽⁵⁶⁾ Estimates are that nodules under extended burnup conditions may penetrate to as deep as 250 μm (10 mils).⁽⁵⁷⁾ Such cladding penetrations are potential sites of cladding failure. Some reactor fuel failures have occurred in assemblies displaying significant amounts of nodular corrosion; however, no direct causal relationship between the nodules and rod failures has been established.⁽⁵⁸⁾

Nodular corrosion appears to depend on the heat treatment, surface preparation, and composition of the cladding. It also is dependent on the oxygen content of the coolant. Nodular corrosion is not well understood but may be controllable by a combination of cladding design and water chemistry control. A systematic study is needed to evaluate the important parameters required to assure the integrity of fuel rods which undergo increasing exposure times in-reactor.

Part of the hydrogen generated during cladding corrosion is absorbed into the cladding during operation. Hydrogen pickup may be estimated for normal LWR extended burnup operations using published design correlations.⁽¹⁾ The hydrogen pickup during pretransition zircaloy corrosion is related to the extent of oxide buildup. For BWR assemblies designed for about 45 MWD/Kg burnup in six years the hydrogen pickup in the cladding is calculated to be about 55 ppm above the as-fabricated values using the relationships recommended in Reference 1. For PWR assemblies designed for about 45 MWD/Kg burnup in four years the hydrogen pickup in the cladding is calculated to be about 180 ppm above the as-fabricated value. Such levels do not constitute a design problem since the cladding will not be severely embrittled. In fact, it has been claimed that cladding hydrogen levels on the order of 100 ppm are beneficial with respect to stress corrosion cracking,⁽⁵²⁾ but this concept has not received general acceptance.

4.4 BURNABLE POISON FUEL

Burnable poison will be required in high burnup fuel. If the cycle length is increased from the current 12 months, increased reactivity is required at the beginning of the cycle. For these longer cycles in PWR's, burnable poisons will be required to reduce the boron concentration at the beginning of cycle (BOC) to a level such that the moderator temperature coefficient is negative. In BWR's the burnable poisons will be required to meet cold shutdown requirements. If the discharge burnup is

increased through the reduction in the number of assemblies loaded each cycle while maintaining a constant cycle length, then the reactivity of each assembly loaded will be high at beginning of life. Under these conditions burnable poisons will be required to maintain acceptable power peaking limits.

If burnable poisons are used to control shutdown margins or moderator temperature coefficients then a large number of burnable poison rods may be required. The required number will be proportional to the cycle length. The concentration of these rods will not need to be high since the excess core reactivity will burn down after a few MWD/Kg into the cycle. If on the other hand burnable poisons are used for power shaping in a few high reactivity fuel assemblies, the required number of pins is lower but higher concentrations will be required to reduce the power peaking throughout most of the first cycle the fuel is in the core.

Candidate poisons for such applications are boron, cadmium, samarium, gadolinium, dysprosium and erbium.⁽³³⁾ Boron carbide rods have been used as separate elements within fuel designs and with proper moisture controls have been demonstrated to be an acceptable poison control method. Samarium has been avoided because of radioactivity problems during fabrication and cadmium because of high vapor pressure at high temperatures. The rare earths gadolinium, dysprosium and erbium behave similarly. Gadolinium in the form of its oxide Gd_2O_3 , has been the burnable poison most frequently

added directly to urania fuel. The advantage of adding the poison directly to the fuel is that when the burnable poison has burned out, there is little reactivity penalty due to the structural materials which are present when the burnable poisons are placed in separate rods. For this reason gadolinia burnable poisons are being increasingly considered for high burnup fuels in both BWR's and PWR's.

4.4.1 Gadolinia Bearing Oxide Fuels

Gadolinia has a solid solubility up to 60 mole % in urania and is physically compatible far beyond its expected range of application. To date fuels with gadolinia contents of up to 4% have been fabricated and irradiated.^(76, 80) Some mixing problems can be encountered during fabrication therefore the microhomogeneity of the fuel must be checked after fabrication.⁽³³⁾ Normal fuel densities can be obtained but procedures for sintering must be slightly altered to obtain desired fuel microstructures. For higher goal exposures at constant cycle lengths, higher compositions of gadolinia will be desirable. Proposed fuel compositions with >5% gadolinia will require lead assembly verification.

The physical property of greatest importance to the fuel behavior which is altered by additions of gadolinia is the thermal conductivity.⁽³²⁾ Gadolinia reduces thermal conductivity and increases fuel temperatures. As the result of urania-gadolinia solid solution formation during irradiation, oxygen is made available to oxidize the zircaloy clad.

The oxygen potential of the irradiated urania fuel is also reduced to a stiochiometric value and can increase the volatility of the cesium fission product. The addition of gadolinia burnable poison to fuel rods alters the power histories of these rods. The gadolinia bearing fuel rods initially have lower linear heat generation rates (LHGR's) which later in the fuel life increase. The lower initial power conditions prevent the fuel from being preconditioned by its initial irradiation. Depending on the U_{235} enrichment of the gadolinia bearing rods and the particular irradiation history of bundles containing these rods, linear heat generation rates high enough to surpass threshold interaction levels could be encountered for rods that were not preconditioned. This power history is not associated with the goal exposure but with design variations that could arise in meeting the desirable nuclear characteristics for long exposure fuel. Because of the varying power generation histories of the gadolinia bearing fuel rods, their behavior cannot be modeled empirically with the current data available from the urania rods. Since there are less gadolinia bearing rods than urania rods in a bundle, the relative data base for gadolinia burnable poison is much smaller and should be expanded especially if the high burnup designs rely on gadolinia concentrations greater than are presently being used.

4.4.2 Boron Carbide Dispersions

Boron carbide dispersed in alumina ($B_4C-Al_2O_3$) is a burnable poison candidate material. (38,39,40) Present LWR experience with zircaloy clad rods of $B_4C-Al_2O_3$ pellets have been satisfactory once the

manufacturing difficulties with the high moisture levels were avoided. Keeping such rods in reactor for long exposures should not cause difficulty. The helium fission gas generation stops once the boron conversion to lithium ($B^{11} \rightarrow Li^7$) is completed. Alumina swelling, degeneration of thermal conductivity^(41,42,43) and further release of the previously created helium are the main design concerns for the high burnup rods. Since power generation in the burnable poison rods is mainly from gamma heating at high burnups, temperatures do not become high and further residence of the poison fuel rods in the reactor should not be a problem. Thus, boron carbide-alumina dispersions provide an alternate burnable poison candidate to the gadolinia bearing fuel rods, but due to the end of cycle reactivity penalty of the structural material and the residual $B_4C-Al_2O_3$ the neutron economy of boron carbide control rods is not as favorable as the gadolinia bearing fuel element designs. This reactivity penalty of boron carbide must be compensated for by additional enrichment and therefore is not as uranium efficient a burnable poison as is gadolinium.

4.5 DATA BASE FOR HIGH EXPOSURE FUELS

To answer many of the questions posed about the effects of high burnup on fuel performance an adequate data base is required. Such a data base should contain information on the design parameters of the fuel, characterization of the fuel and cladding materials, irradiation histories and post irradiation observations. Verification of performance evaluation

procedures is required to obtain licensing approval. Design and licensing calculations are required for conditions that are more restrictive than actually encountered in the operating reactors. Verification of the calculational procedures to predict behavior under a number of conditions is required to establish confidence in these procedures and to establish the safety margins in the fuel systems as they will be irradiated under the proposed operational limitations.

Fuel performance data on rods which have undergone post irradiation examination are a main source of information for the verification of irradiation dependent behavior models such as gas release, fuel swelling, redistribution of fission products, rod bowing, corrosion, etc. A verification file for the FRAP-S⁽²⁾ Code has been established and is an example of a fuel performance data base. EPRI is currently developing a detailed data base⁽⁶³⁾ of light water reactor fuels manufactured by the various vendors. In the fast reactor area a Breeder Reactor Fuel Information Center has been established to provide data on many of the test irradiations that have been conducted for this program. Guidelines for the contents of approximate data accumulation are being established by users of such data bases in performing the required verifications of performance and licensing evaluation procedures. Thus, before commercial application of fuels containing major design modifications can be implemented, or operational limits can be extended, it would be desirable to establish the data bases for assuring the evaluation of the safety margins of these fuels operating to such limits.

4.5.1 The Gas Release Data Base

From the inception of oxide fuel utilization the release of the inert gaseous fission products from fuel materials has been recognized as a behavior that must be modeled. A significant data base for prototypic light water reactor fuel exists for fuels irradiated up to 30 MWD/Kg, but much of these data are not based on design limiting power levels at high exposure. The existing data base in the public domain usually has a description of the fuel material, design characteristics of the fuel rod, maximum exposure and the mean and maximum linear heat generation rate. The operational history of the fuel is usually the area where the description is the weakest. Gas release data is available mostly from measurements made of fuel rods which have been exposed to a variable power history. Model evaluation with such a data base requires adjustment of parameters of the performance code. The FRAP-S verification file⁽²⁾ is probably the largest publicly available source for this type of light water reactor fuel data. In this file are data from high burnup rods irradiated in the Saxton reactor^(4,5,6,10) at high powers near the end of life. These data are not prototypic for uranium fuel since the fuel material was mechanically blended $\text{PuO}_2\text{-UO}_2$. In such rods the burnup distribution on a micro-scale is heterogeneous with very high burnups being reached in local regions of the fuel. Most of the data collected for light water reactor rods were from designs used a decade ago. In many of these designs, the fuel micro-structure was such that the fuel underwent high densification.

The very high burnup gas release data are mostly LMFBR fuels. Some of these data are on uranium fuel, but LMFBR fuel operates at higher temperatures than LWR fuel and hence should not be used to evaluate empirical release models for light water reactor fuels.

The limits of gas retention, especially in the low temperature ranges of operation typically below 1,000°C should be determined.⁽²⁶⁾ Verification of retention capacity at low temperatures should be supported from a large data base if it is to be applied to remove existing overly conservative recommended empirical correlations⁽²¹⁾ based on fuel irradiated under nonprototypic conditions. Such retention is dependent upon microstructural features such as grain size and open porosity. Detailed fuel characterization data are needed along with the gas retention measurements to make these evaluations applicable for model evaluation.

Because of the importance of determining the high burnup effects on gas release, in many of the programs which have extended burnup limits on prototypic LWR fuels the gas release in high burnup rods has been or will be measured. These data will provide empirical values for prototypic fuel at high exposures but will not completely satisfy the data requirements requested by the NRC.* This is because the bundles subjected to extended burnup in normal loading patterns will operate at power levels lower than the power generation rate proposed for the licensing limits. In several

* Personal conversation with Ralph Meyers of the NRC Staff

programs which are in the advanced planning stages such as the DOE/BNW/ Industry-High Burnup Effects Program and the Riso Fission Gas Project, there are provisions to put high burnup rods in specially designed bundles or in a test reactor to expose these rods to higher power levels ("bumped") at the end of their exposure. These tests will produce gas release data at power generation rates which are typical of those used for the licensing limits. Development of a gas release data base with both normal and licensing limit power levels at end of life is required to establish a firm licensing position for high burnup fuels.

4.5.2 Fission Product Release From Failed Fuel

Special study programs⁽⁶⁶⁾ have measured the fission product release rates from heated rods which had intentionally defective cladding. Methods for interpreting such data in terms of operational failures and accounting for the effects of coolant entering the fuel have not been developed. Operational data is continually obtained by the utilities on the plant coolant activity. Analyses can be made for various isotopes of iodine and the relative isotopic iodine levels can be used to give an indication of the delay time between the generation of the fission products and the time at which measurements were taken. Measuring coolant activity is not sufficient in itself unless a correlation can be made in terms of fuel failures. Data on fuel failures of commercial reactors is usually obtained from sipping measurements made at the end of fuel cycles.

Even when bundle sipping measurements are made and failed fuel is detected dismantling of the bundle is seldom done. Dismantling of the bundles is required to determine which rod and at what location the failure occurred. This detail of data is required along with classification of failure type to assure correlation between failure models and release models. Although references to release rates exist,⁽⁷⁷⁾ no data bank of activities and failure observations which is available to the public was disclosed by this review.

4.5.3 Failure Data For Pellet Cladding Interaction

A number of test programs have been conducted to determine the conditions for failure when fuel is ramped from a lower to higher power level. Such ramp data is usually collected in test reactors.⁽⁶⁰⁾ These data have provided a valuable source for model confirmation. The data are usually collected on low exposure short fuel rods designed to determine the failure threshold limit from pellet cladding interaction. Systematic variations of design variables have been made in some controlled ramp experiments. These test rods usually undergo detailed post irradiation examination and provide a well characterized data source. Currently sets of rodlets from fuel which has been irradiated to high burnup in commercial reactors are being exposed to power ramps in test reactors. Data from such tests should provide a close correspondence to prototypic conditions and provide failure thresholds for pellet cladding interaction.

The specialized conditions encountered in power ramp tests are not necessarily representative of the low probability failures encountered under the varying operational conditions that are encountered in commercial fuel. Many observations of failures have been made in commercial reactors, but the intensity of such observations have usually been highest when a fabrication or coolant chemistry problem was encountered. The failures observed in such cases usually have not been a result of pellet cladding interaction. In the early periods of commercial operation of reactors a number of hydriding fuel failures were observed. Some of these failures were caused by unstable fuel microstructures. Other failures were encountered because of high crud levels due to copper containing tubing (Monel) used in the heat exchangers. These types of problems have been remedied by design changes to either the fuel or to the reactor system.

Generally failures which were observed at intermediate burnups have been attributed to pellet cladding interactions, especially in those reactors where the rate of power changes were not controlled by operational guidelines. Sipping measurements of fuel bundles have been made and failures as indicated by high activity levels have been attributed to pellet cladding interaction. Probabilistic failure models have been evaluated with these data. Most bundles identified as failures by sipping have not been disassembled so that the indicated failure data could not be directly connected with a particular failure mode. In most of the incidences where detailed inspections have been made, the results have been

proprietary and are not generally available for failure model evaluation. Even when rods have been subjected to post irradiation examination to identify the failure mode, coolant may have entered the rod prior to its inspection and localized hydriding due to coolant entry was observed even though hydriding may not have been the cause of the initial failure. Thus, data pertaining to identification of failure mode and location are difficult to accumulate. Consequently statistical models for failure by a given failure mode are generally evaluated with indirect failure data.

For accumulated failure model evaluation, detailed rod and ramping histories are required. Ramping histories are difficult to obtain even when core following techniques are used to regenerate fuel histories but programs such as the EPRI PSMS program are presently in existence to obtain such detailed data. Major power ramps have also been assessed for some core loadings which do not have interactive core fuel management by looking at reactor operational logs. For these reactor cores a data base for PCI failure statistics could be constructed but the exposures for these fuels will be less than those for the proposed extended burnup limits.

4.5.4 Failures Induced by Transient Reactor Operation

The type of failures normally associated with pellet cladding interaction and power ramping are not prototypic of those that would be encountered under Class 2 and 3 of the more severe anticipated reactor transients that are analyzed for licensing. In many of the design

type accident situations short bursts of power or short overheating of the cladding is encountered. Such transients arise from events such as accidental rod movement (rod withdrawal) and perturbations in flow conditions. The Power Burst Facility has been constructed to duplicate such conditions.⁽²⁰⁾ Some of the rods tested in this facility have been irradiated but not to the proposed burnup levels associated with extended burnup. The initial rods tested in this facility were tested to very severe power conditions beyond those that are expected to be found in any of the transient accident conditions. These test conditions were proposed to establish failure thresholds but they are not prototypic of the types of failures which would be encountered under the anticipated reactor conditions and may have activated different failure mechanisms than would be encountered under more realistic transients. Thus the data base for predicting failures during transient conditions is not available. Tests are planned for the Power Burst Facility⁽⁶²⁾ which are typical of the limiting transients which are believed possible in reactor operation. These tests should include high burnup fuel rods for those cases where the effects of the transient are expected to be more severe at high burnup.

4.6 DESIGN MODIFICATIONS

A number of different fill gas, fuel and cladding design modifications have been proposed for minimizing LWR fuel-cladding interaction (FCI) and maximizing achievable burnup. Most commercial nuclear fuel

vendors currently have potential extended burnup design modifications under irradiation. These include helium pressurization of BWR fuel rods,^(44,45) increased helium pressurization of PWR⁽⁴⁶⁾ fuel rods, chamfered pellets,⁽⁴⁷⁾ annular pellets,⁽⁴⁵⁾ niobia-doped pellets,⁽⁴⁸⁾ particle fuels (vipac and spherepac),^(45,49) thicker cladding⁽⁴⁷⁾ and cladding internally coated with pure metals^(50,51) or graphite^(45,49,52).

Pressurization of BWR fuel rods or higher pressurization of PWR fuel rods has a multifold effect on fuel clad interaction (FCI) and extended burnup. The increased pressure reduces the rate of cladding creepdown due to internal coolant pressure, delays the onset of fuel-cladding mechanical contact and increases the helium fill gas content so that released fission gas has less influence on the gap conductance. Prepressurization extends the periods during which intermittent high power fuel conditioning is effective and reduces the temperature increases of the fuel due to degradation of the gap conductance by release of inert fission gas. Prepressurization may also result in higher allowable ramp rates and peak powers during normal maneuvering transients and higher burnup before reaching steady-state cladding strain limits (as predicted by rod performance code calculations). The improvement of fuel-cladding gap conductance and effective thermal conductivity of the fuel by pressurization, especially late in life when a significant amount of fission product gases have accumulated, causes the fuel to operate at lower temperatures and this, in turn, causes

less fission gas and corrosive volatile fission product release. The reduced amount of fission products released as well as the increased amount of fill gas dilution of fission gases reduce the propensity for fission product stress corrosion cracking of the zircaloy cladding.

A conservative upper limit for fill gas pressurization can be set by the requirement for a negative differential pressure between the inside and outside of the cladding throughout the operation of the fuel. The internal pressure should remain low enough to maintain a conservative assurance of fuel-cladding gap having a positive tendency to close by pressure loadings. The licensing evaluations for the loss of coolant accident require that cladding deformations be calculated and these deformations are influenced by internal rod pressure. The fission gas release rates at extended burnup are necessary to perform these analyses.

Chamfered pellets reduce clad ridging due to thermal distortion at pellet corners. The reduction of cladding stress and strain in such regions should reduce the propensity for stress corrosion.

Annular pellets operate at lower temperatures than solid pellets for a given linear heat generation rate. Thus annular pellets are subject to less thermal expansion and fission gas release. Annular pellets also are thought to exhibit more flexibility under compression. Cladding stresses are not expected to build up to levels as high as those encountered in solid pellet fuel rods.

Recent information from BNFL⁽⁴⁸⁾ indicates that adding small amounts of niobia to urania produces pellets that have a large grain size. Fission gas release may be reduced by a factor of two to four at intermediate burnups⁽⁴⁸⁾ in these fuels.

Particle fuels (vipac and spherepac) have been irradiated in a variety of domestic and foreign thermal and fast reactor fuel development programs. Particle fuel rods exhibit less ridging and cladding strain than comparable pellet fuel rods,⁽⁷⁸⁾ and this behavior suggests an extended burnup capability. Particle fuel rods also display improved behavior during power ramp tests.⁽⁷⁹⁾ However, fission gas release in particle fuels is predicted to be higher than pellet fuel due to the large surface area of the particle fuel. The relative performance capability of particle and improved pellet fuel rod designs at extended burnups in light water reactors remains to be determined. More definitive information on fission gas release in both particle and pellet fuels is required at extended burnups.

Increased cladding thickness has been identified as an important parameter in delaying creepdown, extending preconditioning duration and useable fuel life and in minimizing localized clad strain such as clad ridging at pellet interfaces.⁽⁵³⁾ Increased cladding thickness has been proposed for irradiation at two domestic reactors to target burnups of 40-50 MWD/Kg.⁽⁴⁷⁾

High ductility cladding is being examined which gives irradiated ductilities comparable to that of material in the as-fabricated condition.⁽⁴⁷⁾ Presumably, less cladding embrittlement would result in less susceptibility to late-in-life stress corrosion failures.

Stress corrosion cracking generally is recognized to occur much more readily in alloys such as zircaloy than in pure metals. One solution for FCI prevention is to metallurgically bond a pure metal layer to the cladding internal surface. Currently a copper plating and a zirconium coextrusion layer are being evaluated in power ramp tests.⁽⁵⁰⁾ Extension of testing to extended burnup is planned.

Graphite and siloxane coatings applied to the inner surfaces of cladding tubes have produced improvement in FCI performance of fuel rods.⁽⁵²⁾ It is not well understood, however, whether the graphite and siloxane primarily act to lubricate the fuel-cladding interface or as a barrier to protect the cladding from fission products. There is some evidence of the latter effect.⁽⁵²⁾

Table 4.1 Predictability of Material Property As A
 Function of Extended Burnup

<u>Fuel Material</u>	
<u>Property</u>	<u>Predictability</u>
Elastic Moduli	Only small porosity effect dependent on burnup.
Inert Fission Gas Release	Interconnection of bubbles to form open porosity and diffusion of gas to open porosity dependent on burnup. Phenomena defined with no unstable behavior expected but effect of burnup dependence not adequately defined.
Thermal Conductivity	Mainly dependent on porosity and internal crack structure with burnup dependent effects of cracking poorly defined. No new phenomena expected at extended burnup but all other fuel properties highly dependent on this property.
Swelling	Accumulation of solid fission products and porosity by bubble formation. Phenomena defined and no unstable behavior expected.
Volatile Fission Product Release	Fuel chemistry and porosity structure dependent, both of which change with burnup. Most difficult phenomena to model burnup effects.
Creep Properties	Porosity effect and diffusion effects from changing chemistry, but not as significant as fissioning rate effects.
Fracture Strength	Changes with interconnected porosity, but evaluations for fuel behavior not highly dependent on this property.
Heat Capacity	No exposure dependence.
Densification	Low exposure phenomena.

Table 4.1 (Continued)

<u>Fuel Material</u>	
<u>Property</u>	<u>Predictability</u>
Melting Point	Chemical effect which is only slightly burnup dependent.
Thermal Expansion	No expected exposure dependence.
Restructuring	Mainly temperature sensitive and occurs at lower burnups when temperatures are higher.
Fuel Emissivity	No exposure dependence.
Diffusion Coefficients	Sensitive to chemical state thus burnup dependent but these coefficients only required for advanced theoretical models.
<u>Cladding Material</u>	
Stress Corrosion Susceptibility	Structure, surface condition and chemical environment dependent. Most difficult to model as far as exposure dependence. Chemical environment is highly dependent on fuel behavior.
Ductility	Reaches saturation values already encountered in existing fuel designs.
Creep Rates	Fast flux dependent more than fluence dependent, thus not exposure sensitive.
Anisotropic Growth	Fluence dependence with no expected unstable behavior.
Thermal Conductivity	Little dependence on exposure.
Thermal Expansion	Little dependence on exposure.

Table 4.1 (Continued)

<u>Cladding Material</u>	
<u>Property</u>	<u>Predictability</u>
Heat Capacity	Little dependence on exposure.
Elastic Moduli	Little dependence on exposure.
Waterside Corrosion	Surface condition and coolant chemistry dependent. Accelerated rates have been observed at higher exposures when thickness of films increased local cladding temperatures.

5.0 RESEARCH AND DEVELOPMENT REQUIREMENTS

To obtain bundle average burnups in the range of 45 MWD/Kg, peak pellet burnups in the range of 60 MWD/Kg will be reached. Such burnups have been obtained in test irradiations and in a few bundles that have been retained in the reactor for additional cycles beyond normal discharge. Thus, the exposure goals necessary for higher core average burnup have been achieved. A number of new fuel concepts are available^(47,50) which hold promise that the higher goal exposures can be achieved without restrictive reactor operational guidelines.

What is currently required is to develop an experience base with sufficient accuracy for verification of these concepts and the additional data that would be required for licensing. The normal procedures of introducing lead bundles followed by partial core loadings is an appropriate way to develop a statistically significant data base if appropriate measurements and tests are made with fuel from these programs. These initial irradiations should be under conditions where the fuel is well characterized and extensive poolside and post irradiation examinations observations are made. These observations would be put into a central data base file system, such that devised by EPRI and is currently being considered for aid by DOE. Included in the high burnup bundles should be removable rodlets for subsequent testing in a test reactor to determine ramping limits and to develop a gas release data base for limiting power operation levels at the higher goal

exposures. Such data should produce the necessary data base for performance model verification, failure threshold determination and for the licensing calculations associated with stored energy and end of life fuel internal pressurization.

Developing operational experience in itself is not sufficient for developing the licensing data base required for anticipated transient behavior. Special considerations are required to develop these bases which will require additional post irradiation testing of the type run in the Power Burst Facility. A summary of data needs and programs required for providing these needs is presented in Tables 2.2 and 2.3.

5.1 FISSION GAS RELEASE

The necessary data pertaining to the quantity of fission gas released under normal operational conditions can be collected from lead assemblies of each fuel design. Presently planned programs by EPRI and DOE for extended burnup demonstrations can provide the necessary data if appropriate post irradiation gas release measurements are made. Cooperative utility and fuel vendor supported programs are being conducted where gas release measurements are being made at the higher exposures. The data bases developed from these test programs will be incomplete if attention is not directed toward gas release at limiting or licensing power levels. Removable rodlets with extended periods of irradiation at licensing power levels or specially reconstituted bundles can be used to obtain such specialized

data. Such experiments would be most valuable if comparisons can be made between gas release measurements of rods after normal operation and after being driven to a higher power.

Special consideration of several effects which are proposed in the more fundamental approach to gas release modeling should be verified. Gas retention capacity measurements at temperatures below 1000°C and in the burnup range of 30 to 60 MWD/Kg should be made, also gas retention dependence on grain size in high burnup fuel should be evaluated. Present restrictions in modeling gas release may be too conservative for all these operational and modified fuel microstructural conditions. The more empirical gas release correlations do not account for irradiation induced microstructural changes to the fuel and the effects of exposure in the temperature range of 900-1200°C can effect the high exposure release of fission gas. If a period of higher power operation occurs during intermediate burnups then the fuel is likely to develop connected grain boundary porosity. Extending the microstructural data base for fuel operating in this temperature range and obtaining gas release data from higher burnup fuel which has been exposed to irradiation conditions that cause restructuring is needed to clarify these effects. In addition, the fabrication characteristics of fuel and the subsequent effect on gas release should be confirmed in regards to their effectiveness. Additions of niobia to the urania fuel have been proposed⁽⁴⁸⁾ which increases the grain size and is reported to reduce fission gas release.

Confirmation of these concepts to alter release thresholds and gas retention capability should be verified. Other fuel concepts such as Spherepac,⁽⁷⁾ should be thoroughly evaluated for gas release in a manner similar to that employed for pellet fuel. Relative release rates for unique fuel forms must be established before they can be used in commercial core loadings.

Gas release during anticipated transient and accident heat up conditions are required for some safety evaluations where the kinetics of release are important. For fundamental data on these conditions, the post irradiation type of experiments being conducted by direct electrical heating⁽²⁴⁾ should provide adequate data for safety modeling. Attention should be directed at intermediate temperature levels especially when higher burnups are reached so that data are obtained in the actual temperature ranges which are predicted to occur. The microstructural changes in grain boundary bubbles caused by cesium collected on the grain boundaries bubbles⁽²⁶⁾ and the chemical state of the cesium as influenced by the oxygen to metal ratio must be considered in establishing the time dependence effects in such accident simulation conditions.

5.2 FUEL-CLADDING INTERACTION, STRESS CORROSION CRACKING AND POWER LIMITATIONS

A significant effort is presently being directed in understanding the role of stress corrosion cracking^(25,52,64,65,68,69) during fuel cladding interactions. With such significant efforts presently being extended

it is mainly necessary to obtain the appropriate test data to incorporate into present interpretational procedures. The fuel-cladding interaction phenomena are not unique to high exposure fuel, but fuel failures by this mode are the most likely to cause limitations on reactor operational conditions, especially as the proportion of a reactor core which contains high burnup fuel becomes significant. The test irradiations being conducted to higher exposure levels should include removable rodlets which can be subsequently loaded into test reactors and ramped to establish empirical limits on power ramping capability of the fuel.

As exposure limits increase the possibility of damage accumulation becomes a significant factor. The accumulative damage determination is not adequately covered by the single power ramp tests. It would be helpful for all levels of fuel exposure if measurement techniques could be developed to determine damage thresholds at the intermediate power levels and how damage⁽³⁴⁾ accumulates or grows. Since new fuel forms containing internally coated cladding and design modifications are being incorporated to reach higher goal exposures, damage thresholds should be significantly extended. Experimental determination of a damage threshold and the extent of damage poses one of the most difficult test measurement problems. If the extent of damage could be measured then quantitative experimental data would be available to establish operational guidelines and to evaluate the additional capacity of such fuel to undergo further exposure. Quantitative evaluation

of potential fuel performance under prototypic irradiation conditions could be established at much greater confidence levels than are presently possible from fuel failure data from demonstration programs containing a small number of fuel assemblies.

5.3 OTHER CONSIDERATIONS

The economic optimization of reactor operation requires adequate models for the low probability failure modes and their dependence on the method of reactor operation. Evaluation of such models requires significant quantities of data on the fuel type being used. Partial core loadings with sipping, irradiation history and fuel characterization and checking of failure modes, if any occur, should provide the needed data.

Dimensional stability of fuel should not be a problem for extended burnup fuels, but confirmation data should be obtained from lead assembly tests. Rod-to-rod spacing and profilometry should provide sufficient information along with the proposed gas release data described in Section 5.1. Special designs such as barrier fuel should confirm that the barriers remain effective throughout the intended exposure goal.

No new fission product release problems from extended burnup fuels are expected, but the safety and plant operational implications are important. Irradiation tests which ramp the fuels to failure should be used to expand the existing data base and confirm that extended burnup does not influence the geometry of the cladding failure or the fission product release through the failures.

Corrosion problems are plant dependent and indications have been obtained from some measurements that corrosion rates could increase in the range of burnup being considered. Measurement of corrosion layers are planned in the lead assembly tests for various reactors. These data are needed to provide the required verification that corrosion is not limiting.

Higher goal discharge exposure make the use of higher gadolinia content in the fuel necessary to obtain acceptable power peaking. Existing technology and measurements made on rods of lower gadolinia content and on unirradiated fuel rods with higher gadolinia content do not indicate any problems with the use of gadolinia. However, applicability of these data under prototypic irradiation conditions and for gadolinia contents greater than 5% should be confirmed. The data base for rods of higher gadolinia content should be developed for the entire range of fuel exposure over which it is expected to operate as must the data base for any new fuel composition (niobia doped fuel would require development of its data base).

The licensing concerns for high exposure fuel are not expected to create greatly increased problems except that gas release could rapidly increase with burnup. This concern was discussed in Section 5.1. The anticipated transient and possible failure modes are not new problems for higher burnups, but in certain cases, especially under rapid power increases, the increased volatile fission product content and its increased diffusion to the grain boundary might enhance failure probability. Programs are

being conducted in this area especially with the Power Burst Facility and the direct electric heating simulation tests. These programs should include high burnup fuel materials.

Special fuel designs such as the barrier concepts, annular fuel and spherepac fuel can introduce new concerns. These concerns are not necessarily high burnup problems but arise because these fuels are being proposed for high burnup application. Introduction of new chemical forms at the fuel-cladding surface need to be checked for chemical compatibility for all transient and accident conditions where high cladding temperatures are reached. New fuel forms such as annular fuel and spherepac fuel must be checked under accident conditions for new fuel relocation mechanisms. Since accumulated fission products influence microcracking of the fuel, the relocation mechanisms should be checked on fuel near its intended goal exposure. Rodlets of irradiated fuel tested in the Power Burst Facility under limiting power conditions should provide data that confirms whether or not similar test data on pellet fuel remains applicable to the new fuel forms.

5.4 TEST DATA

The extension of fuel burnup levels does not create any new measurement problems for fuel. Since the time and expense of obtaining the high goal exposures is greater, effort should be directed toward maximizing the information output from the various fuel irradiations. The fulfillment

of the data needs recommended in this report emphasize a lead assembly and rodlet type of program. From such a program, the primary data comes from poolside and post irradiation examinations.

5.4.1 Poolside Examination

Poolside examinations can be performed rapidly and at intermediate levels of exposure. Thus, poolside techniques should be used to obtain the bulk of the high exposure data base. If failures do occur in bundles it is extremely valuable to determine which rods fail, where the failures occurred and to classify the type of failure. Such data are required for statistical failure evaluation.

The gas release for the fuel rods can be determined by poolside measurement techniques and a significant number of measurements must be made to develop a statistically significant data base. Both rotational and longitudinal profilometry measurements should also be made. The rotational measurements are made to obtain mean cladding deformation, to determine when fuel contact occurs and to evaluate high exposure fuel swelling. The changes in ovality are also best evaluated with rotational profilometry measurements. The longitudinal traces are required to determine when ridging occurs. Rod length changes should be made to check growth models and to assure that sufficient spacing has been provided in the bundle design. Rod-to-rod spacing measurements are required to assure no rod bowing limitations are being encountered at extended burnups. The

gamma scanning measurements are made to determine localized fission product relocation, gapping in fuel columns and fuel column length changes.

Visual examinations and special examinations to determine corrosion layer thicknesses should be made to assure that the corrosion rates are not rapidly accelerating. All of the preceeding measurements are currently included in existing programs and continuation of making access available to a large base of such data is recommended to be extended into the high burnup range.

More work is needed to develop a reliable nondestructive technique that can be correlated with the accumulative damage occurring to the fuel cladding. The current techniques have not proven adequate to do such evaluations.

5.4.2 Post Irradiation Examination

The evaluation of gas retention at lower temperatures and fission product redistribution should be done with data generated by post irradiation examinations. Efforts should be made during the microstructural evaluations to determine the threshold for interconnected grain porosity. Fission product distribution in the fuel should be determined to aid in developing the models which can predict the chemical fission product state and the distribution of fission products within the fuels. These models should be useful for fission product release and estimating the chemical environment of the cladding. Observations of interaction products on the

cladding inner surface and location of where the damage is initiated will be required to determine damage threshold criteria. Criteria for detecting such thresholds by nondestructive testing techniques also need to be established.

Specialized tests should be conducted to determine stress corrosion cracking thresholds that simulate both the mechanical and chemical conditions. If one test technique could obtain general acceptance then comparative analyses could be made for the various fuel and cladding concepts. The direct electrical heating experiments should be included as part of post irradiation examinations for the high burnup fuel specimens⁽²⁴⁾ to determine the specialized gas release mechanisms resulting from transients. Particular attention should be focused on those fuels which have retained a high inventory of retained fission products to assure that this inventory does not pose a potential for enhanced failure under accident conditions.

5.4.3 Operational Data Requirements

As the procedures for interpreting data more accurately reflect the dependence on radiation histories, the need for an adequate description of the test irradiation conditions increases. For pseudo steady state model verification, the effective operational history characterization requires reporting of rod coolant flow conditions and power level on a periodic bases along with the axial power distribution for each reported power level. To evaluate behavior under power ramping conditions,

the change in axial power distribution and the ramping rate for this change needs to be described. Any overpower of the rod occurring due to xenon transient effects should also be reported.

Records of the operational history of the fuel rod are quite lengthy and become quite expensive to generate and store. Computer compatible filing techniques are required for efficient storage and recording of these data. Proposed procedures for filing such information have been included as part of the FRAP-S verification file, the Fuel Performance Improvement Program, the data system for the EPRI fuel irradiation programs and the Breeder Reactor Fuel Systems Information Center. These efforts should provide the guidance for establishing working procedures for new test programs.

5.5 PROPOSED SCHEDULE

The proposed schedule for irradiating commercial quantities of high burnup fuels in LWR's is shown in Table 2.4. The recommended research needs for extending present exposures limits of LWR fuel is built upon demonstration of existing concepts. It assumes that the licensing authorities will allow partial core loadings once the basic data bases are established with test irradiations. A lead assembly program is employed to obtain any indications of unexpected behavior.

The proposed schedule is based on favorable results coming from the test programs, active and continuous support of the demonstration programs sufficient economic incentive to receive utility acceptance, a

sufficient market for the vendors to invest their funds and a cooperative environment by the licensing authorities so that any special issues will receive prompt review. The proposed program is a demonstration program where every proposed concept is based on experimental data indicating that improved performance can be expected and no technical delays will be encountered to alter the concept. The initial phases of this program are already in place, thus no program delays are considered for starting the program. Since to obtain high exposure on fuels takes from five to seven years under prototypic LWR conditions, data generation steps must be run in parallel and decisions must be made with partial results from the previous stages of the demonstration. Actual commercial acceptance was assumed to take place while the data base was developing and most of these data would be generated by poolside examination. Reconstitution must be part of the program with ramping and transient tests on new concepts being conducted before goal exposures are obtained.

The schedule is developed for three phases of data accumulation. The preliminary data will be accumulated with fuel and test rods from two sources. The first is from fuel which is presently undergoing irradiation and will be subjected to extended irradiation by retaining the fuel in the reactor for additional cycles. The second is from special test reactor irradiations. From these rods can be established gas release data for

intermediate power level generation (25-36 kw/m) and failure thresholds for high burnup fuel rods which are subjected to power ramping. These data can then be used to design fuel for higher burnups.

Design modifications to existing LWR fuel are believed to be sufficient to obtain extensions of existing exposure limits (33-36 MWD/Kg for BWR's and 38-42 MWD/Kg for PWR's). No new licensing issues and concerns over unexpected performance should delay lead assembly and partial core loadings of such fuel on a limited demonstration program. Even before such fuel reaches its goal exposure, the expanded data base generated from current design fuel reaching higher exposures, ramping test results of fuel with these design modifications, and poolside measurements should provide sufficient justification for multiple core loadings of such fuel. If the economic incentives are sufficient, fuel performance under test reactor conditions is favorable, and it is demonstrated that no problems are encountered up to present exposure limits, the assumption has been made that acceptance of high burnup fuels can be obtained before actual verification of the failure statistics are available.

To obtain high exposures without restrictive operational guidelines, significant design modifications to the fuel may be necessary. No untested concepts are considered for this phase, but those presently in the testing phases are put into a demonstration program. The barrier and annular fuel concepts are not considered major departures from existing

technology. Annular fuel designs have already been demonstrated and some barrier fuel has been irradiated on commercial scale to low burnups. Enough new fabrication techniques are involved with these concepts and the gadolinia concentrations for high exposure fuel should be high enough that their demonstration and acceptance will require developing a data base that is more than a slight extension of the extended burnup base from existing fuel. The gas release base should for the most part be applicable but the expected improvements for pellet-cladding interaction failures should be demonstrated. A data base for gadolinia bearing fuel which contains higher concentrations of gadolinia (>5%) should be developed. This data base will provide justification for the performance of fuels which contain higher concentrations of gadolinia. Because of the present concern over transient fuel behavior, new licensing concerns in this area are likely. Higher exposure should not be the concern, but modifications in fuel form and the fuel-cladding interface will require some verification that no new transient behavior mechanisms are operable. Such testing if planned and scheduled at the start of the demonstration program should not delay acceptance of these concepts and should not require demonstration on fuel that have reached goal exposures.

The spherepac fuel and doped fuel are presently undergoing irradiation and to date have demonstrated good performance. These modifications are felt to be greater than the barrier and annular fuel concepts

since they are significant modifications to the fuel material. Their data bases will require significant development since an extensive data base for performance at lower exposures does not currently exist for these fuels as exists for conventional fuels. The schedule for these fuels has been extended to permit development of these data bases before making the commitments.

Multicore loadings should require at least data from the lead assembly irradiations and demonstration that fabrication of fuel in production quantities does not lead to production problems. Since the fuel forms are new, licensing issues may require special consideration which can cause delays. The extended schedule should allow enough time for the data base to be developed to settle such issues.

5.6 ESTIMATED COST

The research and engineering costs to develop a high burnup fuel for use in current high power density PWR's or BWR's has been estimated by ENC to be between forty-two and fifty-four million dollars. This includes the cost of programs which have already been completed, programs which are currently in progress or programs which are in the process of being implemented. The components of this estimate are shown in Table 5.1 and include the costs to design, license, test, irradiate and conduct post irradiation tests on high burnup fuels. The costs include a lead assembly program and one batch of reload fuel in either a current high power density PWR or BWR.

Sufficient money to develop lead assembly programs for both annular and spere-pac fuels are included in the cost estimates. For the full reload batch only the incremental costs for one high burnup fuel design are included. The incremental costs for the first reload fuel include costs which are expected to occur one time and not costs which will be incurred each cycle as a result of the fuel being designed for high burnup.

As an example, any increased fabrication cost for the first full reload of high burnup fuels would be part of the cost of the fuel and those costs would be part of the utility purchase price for the fuel. Increased costs for initial licensing of a reload batch of high burnup fuel are included in the estimate.

Table 5.1 Lead Assembly Program on Two Advanced Fuel Design Concepts
 Annular and Sphere-Pac

Fuel Performance Improvement Program (FPIP)	\$17 *
Gadolinia Performance Study	3
Incremental Fabrication and Fuel** Characterization Costs (Leads + Reload Batch)	3- 4
Fabrication of Segmented Rods	1- 2
Fuel Management and Data Collection Costs During Irradiation	1- 2
Poolside Measurements and Data Collection Costs (4-5 Poolside Campaigns for Leads) (Includes the destructive PIE on sacrificial rods in mid-irradiation for leads.)	3- 5
PIE on Discharge Rods to Benchmark Fuel Performance Codes, etc.	2- 3
<hr/>	
TOTAL IRRADIATION PROGRAM	\$30-36
<hr/>	
<u>PRIMARY RAMP PROGRAM</u>	
Shipping and Waste Disposal Costs	\$ 1- 2
Ramping 30 Sphere-Pac	3- 4
30 Annular	3- 4
Post-Irradiation Examination of 20 Rods	3- 5
<hr/>	
TOTAL PRIMARY RAMPING COSTS	\$10-15
<hr/>	
Incremental Licensing Costs to Develop Revised Fuel Performance Models and Safety Review and Analysis	\$ 2- 3
<hr/>	
TOTAL	\$42-54*
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* Millions of 1979 Dollars

** Includes poison rods for extended burnup with a few segmented poison rods for ramping at high burnup and testing poison rod performance.

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ANS TOPICAL MEETING ON LWR FUEL PERFORMANCE, PORTLAND,
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ANS TOPICAL MEETING ON LWR FUEL PERFORMANCE, PORTLAND, APR 30-MAY 3, 1979.

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Simulation of Reactor Cores for Fuel Management and Cycle
Analysis of Boiling Water Reactors", XN-CC-25, Revision 1,
Exxon Nuclear Company (January, 1976).
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"Reactor Physics Quantatively Report, April, May, June 1969,
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Dimensional Diffusion Theory Code for the Microscopic
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APPENDIX A REFERENCE FILE

6
 1K 5COMPCODE URANIA UPU02 DENSITY RESTRUCTUR
 1A REYMANN,G.A.
 1A MAGRMAN,D.L.
 1A LETSUN,S.B.
 1T MATPRO=VERSION 10--A HANDBOOK OF MATERIALS PROPERTIES FOR USE IN
 1T THE ANALYSIS OF LIGHT WATER REACTOR FUEL ROD BEHAVIOR.
 1R TREE=NUREG-1180,FEB1978.
 1C A COMPILATION OF FUEL, ZIRCALOY CLADDING, AND FILL AND FISSION GAS
 1C PROPERTIES. THESE PROPERTIES ARE THOSE REQUIRED TO RUN THE FRAP
 1C STEADY STATE AND TRANSIENT FUEL ROD PERFORMANCE CODES. MATPRO
 1C DOCUMENTATION INCLUDES A CALCULATIONAL MODEL FOR THE PROPERTY,
 1C AN ASSESSMENT OF THE DATA BASE, REFERENCES AND A LISTING OF THE
 1C CALCULATIONAL SUBROUTINE. FUEL PROPERTIES INCLUDE HEATCAPACITY,
 1C THERMAL CONDUCTIVITY, EMISSIVITY, THERMAL EXPANSION, ELASTIC
 1C MODULUS, POISSON'S RATIO, CREEP RATE, FRACTURE STRENGTH, SWELLING,
 1C DENSIFICATION, RESTRUCTURING, FISSION GAS RELEASE, CESIUM AND
 1C IODINE RELEASE, MELTING POINT, HEAT OF FUSION. SOME PROPERTIES
 1C CORRELATIONS ARE STATISTICAL FITS AND WHERE APPLICABLE PHYSICAL
 1C MODELS ARE USED.
 1Z KRM STB
 11K 3PERFILE COMPCODE FILESYS
 11A COLEMAN,U.R.
 11A SCOFIELD,H.R.
 11A HAMPTON,N.L.
 11A LAATS,E.T.
 11T FUEL BEHAVIOR VERIFICATION DATA REPORT - REPORT NUMBER 1-
 11T DESCRIPTION OF STEADY STATE FUEL ROD MEASUREMENT STORAGE.
 11R TFR=TR-153,DEC1976.
 11C REPORT DESCRIBES DATA FORMATS AND CONTENTS OF A FILE COLLECTED AS
 11C COMPATIBLE WITH THE REQUIREMENTS OF THE FRAP-S CODE. HISTORIES OF
 11C IRRADIATION CONDITION CAN BE DESCRIBED. ALSO DATA CARD TYPES ARE
 11C INCLUDED FOR BOTH POOLSIDE AND POST IRRADIATION EXAMINATION DATA
 11C ACTUAL DATA AND THEIR REFERENCES ARE NOT IN THE REPORT BUT THE
 11C FILE HAS BEEN OBTAINED AND INTERPRETED. AN EXTENSIVE COMPILATION
 11C OF DATA HAS BEEN COLLECTED.
 11Z KRM
 13K 5SOLGEL TEMPMEAS IRRADIATEDPERFILE VIPAC
 13A FITTS,R.W.
 13A MILLER,F.L.
 13T A COMPARISON OF SPHERE-PAC AND PELLET (U+PU)O2 FUEL PINS IN LOW-
 13T BURNUP INSTRUMENTED IRRADIATION TESTS.
 13R NUCL TECH, VOL 21, PP26-38, JAN 1974.
 13C SPHERE-PAC FUEL IS CLAD FILLED WITH 420-590 UM DIA 95 PERCENT DENSE
 13C SOL-GEL SPHERES, INFILTRATED WITH 44 UM DIA SPHERE. THIS YIELDS
 13C .73 LARGE AND .27 SMALL MICROSPHERES. FINAL DENSITY .82 TD.
 13C BACKFILL WITH HELIUM. ONE MOLY SHEATHED THERMOCOUPLE FAILED. USED
 13C W-.26RE SAEATHED THEREAFTER BUT ATTACK OF THESE WAS OBSERVED.
 13C COMPLETE RESTRUCTURING ABOVE 1650C (COLUMNAR GRAINS).
 13C THERMAL COUPLE THROUGH FUEL CENTER, 304SS CLAD, NAK CAPSULE.
 13C SHOULD BE A TRACEABLE DATA SOURCE. SETS OF THERMAL COUPLE DATA.

13C 12 PER PIN AND FIVE PINS. TEMPERATURES TO 2000C, LHGR TO 10KW/FT.
13C CALCULATED GAP CONDUCTANCE 1.93 W/(CM**2*C)
13C CENTER TEMP. VS. LHGR CURVES SHOWN FOR PELLET + SOLGEL FUEL PINS.
13C SOLGEL PINS OPER. AT LOWER CENTER TEMPS. THAN H2 PCT. TO PELLET PINS
13C BY 11.7 PCT.

13Z KRM PGS

21K 6PERFILE VIPAC GASRELEASEIRRADIATEDPELLET UPU02

21X 2 22 23

21A SMALLEY, W.R.

21T SAXTON CORE II FUEL PERFORMANCE EVALUATION PART I - MATERIALS.

21R WCAP-3385-56 PART I, SEP 1971.

21C PROGRAM FOR PLUTONIUM-ENRICHED FUEL WITH BOTH PELLETIZED AND VIPAC

21C FUEL. ASSEMBLIES ARE 9X9 WITH 6.6W/O PU ENRICHED FUEL WITH SEVEN

21C PELLETIZED FUEL AND TWO WITH VIPAC WITH MOST OF FUEL ZR-4 CLAD.

21C ACTUAL CORE II LIFE WAS 9.19 (MWD/KGM). ROD PITCH = 0.580 (IN).

21C PELLETS HAD .94TD AND MECHANICALLY VIPAC .87TD DENSITY. CENTRAL

21C ASSEMBLY IS A SPECIAL 3X3 ASSEMBLY WHICH WAS EXAMED AFTER

21C AVG. BURNUP=4.5 (MWD/KGM) AND PEAK BURNUP=6.1 (MWD/KGM) AND FOUR RODS

21C REMOVED. CORE II OPERATION ENDED OCT. 18, 1968 WITH CORE AVG.

21C BURNUP=10.94 (MWD/KGM), PU AVG. BURNUP=17.4 (MWD/KGM) AND PEAK PU

21C PELLET BURNUP=28.0 (MWD/KGM). TWO OF FOUR RODS WITH EARLY REMOVAL

21C WERE VIPAC AND HAD A FRACTIONAL GAS RELEASE OF .108 VS .021 FOR

21C THE PELLET RODS. INDICATIONS VIPAC HAD 25PPM HIGHER MOISTURE

21C CONTENT THAN PELLET FUEL WHICH INCREASE CLADDING CONTENT 15PPM.

21C OVERPOWER TEST ASSEMBLY HAD GREATER PITCH AND WAS IN THE 3X3

21C CENTRAL LOCATION. RODS A AND R INITIALLY IN 9X9 BUNDLE THEN

21C INSERTED IN 3X3 ASSEMBLY. A PEAK POWER OF 18.7 KW/FT WAS REACHED.

21C OTHER STANDARD RODS CH AND WERE ALSO EXAMED TO PROVIDE A BASES FOR

21C COMPARISON. REMAINDER OF RODS EXAMED WERE FROM REGULAR BUNDLES AT

21C END OF CORE II LIFE. THREE RODS FROM CENTER ASSEMBLY D3, TWO FROM

21C C3, ONE FROM E2, AND THREE VIPAC FROM D4. PROFILOMETRY WAS DONE

21C ON IRRADIATED RODS BUT NO PREIRRADIATION MEASUREMENTS WERE MADE.

21C GAS RELEASE MEASUREMENTS WERE MADE ON MOST OF THE RODS. IN GAS

21C RELEASE ROD D HAD VERY HIGH NITROGEN MAKING ITS MEASUREMENTS

21C SUSPECT. FILL GAS WAS .9HE AND .1AR WITH VIPAC FUEL RELEASING

21C NITROGEN BECAUSE OF ATMOSPHERE USED IN PREPARING UO2.

21C OXIDATION THICKNESSES ON CLADDING INNER SURFACE WERE OF SIMILAR

21C THICKNESSES FOR BOTH PELLET AND VIPAC FUEL (.1 TO .5 MILS) WITH

21C SOME PARTICLES BONDED IN VIPAC FUEL. VIPAC FUEL WAS MORE

21C HYPERSTOICHIOMETRIC THAN PELLET FUEL.

21C FUEL MICROSTRUCTURES OF PELLET SAMPLES SHOWED CLUSTERS OF GAS

21C BUBBLES AROUND ORIGINAL PUO2 PARTICLES. LARGE AMOUNTS OF EQUIAXED

21C GRAIN GROWTH BUT NO COLUMNAR GRAIN GROWTH OR CENTRAL VOID.

21C HIGHEST BURNUP VIPAC ROD (K4) INPILE SINTERED TO .8 OF RADIUS.

21C CLADDING WAS COLD TUBE REDUCED AND ANNEALED AT 1025F FOR 2HRS.

21C BOTH TENSILE AND TUBE BURST TESTS WERE MADE OF IRRADIATED CLADDING

21C WITH MAJORITY OF TESTS AT 675 F.

21Z KRM

22K 3PERFILE VIPAC DESNRPT

22X 2 21 23

22A BASSLER, E.A.

22A FISCHER, D.C.

22A GEORGES, N.J.

22A MC CABE, E.A.
22T SAXTON PLUTONIUM PROGRAM - MECHANICAL, THERMAL AND HYDRAULIC
22T DESIGN OF SAXTON PARTIAL PLUTONIUM CORE.
22R EURAEC-1491, WCAP-3385-52, DEC 1965.
22C NINE MAIN ASSEMBLES WITH 638 RODS WITH BOTH PELLET AND VIPAC FUEL.
22C 148 VIPAC RODS WITH 10 HAVING SST CLAD THE REST ZIRC. 490 RODS
22C WITH PELLETIZED FUEL WITH 20 HAVING SST CLAD AND THE REST ZIRC.
22C REGULAR DESIGN IS A PITCH OF .580 INCH. ONE CORNER OF BUNDLE HAS
22C ONE PARTIAL ROW OF ROD REMOVED FOR CONTROL BLADE - 4 RODS IN THESE
22C TWO ROWS OR 9 RODS REMOVED. ALSO TWO LATTICE LOCATIONS LEFT VACANT
22C FOR FLUX WIRE THIMBLE, SOURCE RODS OR REMOVABLE FUEL RODS. ALSO
22C ASSEMBLY WITH REMOVABLE CENTER NINE RODS AND EIGHT RODS AND FLUX
22C WIRE THIMBLE IS PUT IN THIS SUBASSEMBLY. PITCH IN THIS ASSEMBLY
22C IS .538 INCHES. UPPER PLENUM LENGTH IS .856 INCHES WITH A PLENUM
22C SPRING. PELLET LENGTH IS .366 INCHES AND PUO2 ENRICHMENT.
22Z KRM
23K 4PERFILE DESNRPT VIPAC THERMOCOND
23X 2 21 22
23A BIANCHERIA, A.
23A STANUTZ, R.N.
23A ALLIO, R.J.
23A HOUSTON, M.D.
23A RAY, W.E.
23A ROSE, R.G.
23T SAXTON PLUTONIUM PROGRAM - MATERIALS DESIGN AND FABRICATION OF THE
23T SAXTON PARTIAL PLUTONIUM CORE.
23R EURAEC-1492, WCAP-3385-53, DEC 1965.
23C CLADDINGS WERE TYPE 304 SS AND ZIRCALOY-4. FUEL IS NATURAL UO2
23C ENRICHED TO 6.6 W/O PUO2. PU CONTAINED 8.6 W/O PU-240. PELLETS
23C WERE .94TD AND VIPAC WAS NUPAC .87TD USING BNWL HIGH ENERGY
23C IMPACTION PROCESS FEED. PELLET FEED MATERIAL WAS MECHANICALLY
23C MIXED. REVIEW OF VIPAC EXPERIENCE IN PRTR WITH EARLY FAILURES DUE
23C FUEL CONTAMINATION BY FLORIDE, MOISTURE AND OIL. SAVANNAH RIVER
23C EXPERIENCE REVIEWED WITH CHLORIDE, MOISTURE, POOR WELDS AND SPACER
23C DESIGN CAUSE FOR EARLY FAILURES. CORRECTIVE MEASURES EFFECTIVE IN
23C BOTH PROGRAMS. WASHOUT AND WATER LOGGING CAN OCCUR WHEN VIPAC TOO
23C LOW DENSITY BUT NO PROBLEMS ENCOUNTERED IN LATER GENERATION
23C DESIGNS. THERMAL CONDUCTIVITY CURVE GIVEN FOR .87TD VIPAC.
23C DRAWING IN REPORT WERE INTERCHANGED BETWEEN PELLETS FOR ZR AND SS
23C CLADDINGS. FUEL WAS PREPARED FROM BLENDS OF MINUS 165 MESH PUO2
23C AND MINUS 65 MESH UO2. MOISTURE CONTENTS OF 30 PPM WERE SPECIFIED
23C FOR PELLETS. IN THE VIPAC THE MOISTURE WAS SPECIFIED AS 100 PPM.
23C PELLET GREEN DENSITIES WERE 4.5-4.8 GM/CC, 0.5 O/D STEROTEX LUR.
23C AND SINTERED AT 1700C FOR FOUR HOURS. VIPAC HAD FOUR SIZE RANGES
23C WITH FRACTIONS .65 -6+10MESH, .125 -20+35MESH, .125 -35+65MESH,
23C .10 -200MESH. FEED PARTICLE DENSITY WAS .986TD. ROD IDS AND
23C ASSEMBLY LOADINGS GIVEN IN REPORT. DISCUSSION OF FABRICATIONS
23C AND SPECIFICATIONS GIVEN.
23Z KRM
24K 780L GEL GAS RELEASE IRRADIATED TEMP MEAS PIEXAM VIPAC UPUO2
24K 811 PELLET RESTRUCTURURANIA THERMOCOND
24A LOTT, A.L.
24A PETERSON, S.

24T FAST BREEDER REACTOR OXIDE FUELS DEVELOPMENT - FINAL REPORT.
24R ORNL-4901, NOV 1973.
24C RESULTS REVIEWED FOR A 5 YEAR PROGRAM TO DEVELOP A SOL-GEL MIXED
24C OXIDE FUEL FOR SPHERE-PAC LOADING. MICROSPHERE FORM FROM SOL-GEL
24C BY DEBOLVING CONTROLLED AMOUNTS IN DROPS. SUSPENSION IN ORGANIC
24C LIQUID WHERE WATER IN DROPS REMOVED BY GELATION. SEPARATION FROM
24C ORGANIC LIQUID, DRYING AND FIRING TO OBTAIN HIGH DENSITY SPHERES.
24C PROBLEMS ARE CLUSTERING DURING GELATION AND CRACKING DURING DRYING
24C WHEN LARGER SPHERES ARE FORMED. > 0.2 MM DIA. SPHERES HAD > .99
24C TD. MAXIMUM SINGLE SPHERE PACKING DENSITY IS .635. IF SECOND
24C SIZE SPHERE USED DENSITY INCREASES AS D1/D2 INCREASES EXCEPT
24C SEGREGATION OCCURS. USING D1/D2 OF 4 AND D2/D3 OF 10 AND .88TD CAN
24C BE OBTAINED. RATE OF FILL DECREASED AS ROD LENGTH INCREASES. TIME
24C FOR TWO PHASE INFILTRATION TOO LONG SO BLENDING FIRST THEN
24C INFILTRATION WITH FINE. SORBED GAS RELEASE GREATER THAN .05CM³/GM
24C WAS A PROBLEM ESPECIALLY FOR MIXED OXIDES. HEATING TO 850C NEEDED
24C TO MEET LIMIT. REPORT DESCRIBE H₂O MOISTURE AND SORBED GAS
24C CHARACTERIZATION. TECHNIQUES TESTED ON MIXED OXIDE.
24C STRUCTURE OF COARSE MICROSPHERES .420-.597 MM .954 TO .991 TD WITH
24C POROSITY AT GRAIN BOUNDARIES. GRAIN VARIED .0017-.0032 MM BUT
24C SIZES AS LARGE AS .040 MM HAVE BEEN OBSERVED. FINE MICROSPHERES
24C LESS THAN .044 MM WITH .02 HOLLOW OTHERWISE .995 TD. GRAIN SIZE
24C .001 TO .020 MM.
24C THE ORNL IRRADIATIONS INCLUDED SPHERE-PAC, SOL-GEL PELLETS,
24C COPRECIPITATED PELLETS AND VIPAC. FUEL COMPOSITION, ROD DIAMETER
24C AND CLADDING WERE PROTOTYPIC FOR LMFBR. DATA NOT LWR PROTOTYPE BUT
24C GENERAL BEHAVIOR OF SOL-GEL SPHERE-PAC SHOULD DESCRIBE PHENOMENA.
24C 86 RODS IRRADIATED IN PROGRAM.
24C FIRST TEST SERIES IN ETR WHERE OF 40 RODS 6 WERE OF ENRICHED UO₂
24C AND 17 (U,Pu)O₂ SPHERE-PAC. ACTIVE FUEL LENGTH 3.1 (IN)
24C ROD O.D. = 0.252 AND 0.230 (IN) WITH .84 TD AND COARSE BED .3-.4 MM
24C AND FINES .LT. .044 MM. BURNUPS TO OVER 100 MW/KGU COVERING WIDE
24C RANGE LHGR 8.4-50 KW/FT. GAS RELEASE RATES IN GENERAL > .0.3
24C AND AT HIGH BURNUPS > .0.60 WITH LHGR 8.8-11.2 KW/FT. THE UO₂
24C ROD WENT TO EXPOSURES OF 40-60 MW/MTU AT HIGH LHGR'S 16.4-21.6
24C (KW/FT) AND SPHERE-PAC HAD LOW DENSITY. CENTRAL VOID AND MAJOR
24C RESTRUCTURING, NON PROTOTYPIC LWR CONDITIONS. HIGH BURNUP PINS HAD
24C DECREASING LHGR AND ACTUAL HISTORIES CORRELATED WITH FISSION GAS
24C RELEASE. INITIALLY FUELS STARTED AT 12.5-14.7 KW/FT AND CENTRAL
24C VOID AND SINTERING OCCURED.
24C THERMAL PERFORMANCE TESTS CONDUCTED ON THREE CAPSULES WITH
24C THERMAL COUPLE. LAST TWO TESTS CONDUCTED SO TMAX=1500C AT 12KW/FT
24C AND TMAX=2000C AT 16 KW/FT. LOW DENSITY PELLET FUEL REACHED 2000C
24C AT LOWER LHGR THAN SPHERE-PAC ROD. BOTH FUELS RESTRUCTURED WITH
24C COLUMNAR GRAIN GROWTH ABOVE 1650C. (U,Pu)O₂ FUEL WITH .82 TD HAD
24C HIGHER GAP CONDUCTANCE THAN PELLET FUEL. = 1.93 TO 0.73 (W/CM**2)
24C BY BACK CALCULATION. FUEL CLADDING CHEMICAL INTERACTION IN PELLET
24C FUEL AND NONE IN SPHERE-PAC. DESCRIBED IS TECHNIQUE FOR ANALYSIS
24C OF THERMOCOUPLE DATA. MORE DATA SCATTER IN PELLET DATA THAN
24C SPHERE-PAC. WITH POWER CYCLING IN BOTH PELLET AND SPHERE-PAC
24C FUEL CENTER LINE TEMPERATURES INCREASED 250C AT 8-13 KW/FT.
24C TOOK PELLET 3HR AND SPHERE-PAC 1HR TO RECOVER.
24C TWO INSTRUMENTED CAPSULES, 43-120 AND 43-121, WERE IRRADIATED IN

24C ETR. FOUR PINS PER CAPSULE WITH 1 IN FUEL LENGTH AND 0.25 O.D.
 24C WERE INSTRUMENTED WITH THERMOCOUPLES. TWO SOL-GEL-DERIVED PELLET
 24C AND TWO SPHERE-PAC PINS ARE IN EACH CAPSULE. PEAK LHGR=17.2 KW/FT
 24C WITH 0.103 FIMA EXPOSURE OR 100 MW/D/KGU. SOME POWERS NEAR PEAK
 24C DID OCCUR NEAR END OF EXPOSURE. CENTER HOLES AND LARGE REGIONS OF
 24C COLUMNAR GRAIN STRUCTURE FORMED. GAS PEASES .75-.90 OBSERVED WITH
 24C PELLETS HAVING GREATEST RELEASE.
 24C EBR-II TEST IN TWO SERIES WITH SERIES I COMPOSED OF 5 PINS OF
 24C SPHERE-PAC IN 13.5 IN FUEL COLUMN (U.6PU.2)01.98 FUEL AND .25IN OD
 24C CLADDING. SMEAR DENSITIES OF .82TD. LHGR=14 KW/FT. BURNUP=.06FIMA
 24C AND GAS RELEASE FRACTIONS.8U. MAJOR FUEL RESTRUCTURING AND CENTER
 24C HOLE FORMED. COLUMNAR GRAIN STRUCTURE TO OUTER RING OF SPERE-PAC.
 24C COLUMNAR GRAINS HAD .96TD AND FORMED ABOVE 1700C. SERIES II HAD
 24C 37 PIN ASSEMBLY WITH PELLET. SPHERE-PAC AND VIPAC (U.6P.2)02 FUEL.
 24C VIPAC AND SPHERE-PAC HAD .8U. .83 AND .85 TD.
 24C CONCLUSIONS FROM VARIOUS TESTS ARE THAT --
 24C SPHERE-PAC HAS MUCH HIGHER GAP CONDUCTANCE THAN PELLETS. SAME
 24C THERMAL CONDUCTIVITY FOR SAME DENSITY AND RESTRUCTURES ABOVE 1650C
 24C WITH ADDITION HIGH EXPOSURE SINTERING. COLUMN LENGTH CHANGES UP
 24C TO .042 FOR .84TD FUEL OBSERVED WITH CHANGES PROPORTIONAL TO LHGR.
 24C PU REDISTRIBUTION OCCURS WITH PU CONCENTRATING AT HIGHER TEMP.
 24C MOLYBDENUM, CESIUM AND TELLURIUM ACCUMULATED NEAR CLADDING.
 24C CHEMICAL INTERACTION WITH CLADDING WERE LESS WITH SPHERE-PAC THAN
 24C PELLETS.
 24C SOLGEL GAP CONDUCT. WAS 1.93W/CM*2=C BY TWO METHODS. IN ORR
 24C INSTRUMENTED TESTS, MEASURED W/TCIS THE TCENTER. TCAD OUTER.
 24C LHGR. OF 82PCT.TD UPU02. THE LOWER TCENTER OF SPHEREPAC COMPARED
 24C TO PELLET HAD TO BE DUE TO HIGHER GAP CONDUCT. THESE WERE SHORT
 24C TERM IRRADIATIONS, 1200-2200 HR.
 24C IN SERIES I EBR-II TESTS USED RESTRUCTURE OF UNINSTRUMENTED PINS.
 24C RADIUS OF COLUMNAR REGION (1700C) INDICATED HI GAP CONDUCT. IN PIN
 24C OPERATED TO 57GWD/MTM AT HI TEMP. (81-82PCT.TD). INDICATE GAP
 24C CONDUCT. OF SPHEREPAC STAYS HI EVEN W/RESTRUCTURING. NO FUEL
 24C CLADDING CHEMICAL INTERACTION SEEN WHICH WOULD INFLUENCE GAP CONDUCT.
 24C ORNL REPORTS DO NOT GIVE VOL. AVG. FUEL TEMPS. BUT COULD MODEL
 24C FUEL AND CALCULATE THEM IN ORDER TO DERIVE FISSION GAS RELEASE
 24C CORRELATIONS.
 24Z KRM PGS
 27K 1FILE8Y8
 27A GREENE,N.M.
 27A FORSBERG,V.M.
 27A RAIFORD,G.B.
 27A ARWOOD,J.W.
 27A SIMPSON,D.B.
 27A FLANAGAN,G.F.
 27T SACRD= A DATA BASE FOR FAST REACTOR SAFETY COMPUTER CODES.
 27T SUMMARY DOCUMENT.
 27R ORNL=5477/V1,SEP1978.
 27C SACRD IS SAFETY ANALYSIS COMPUTERIZED REACTOR DATA. CONTAINS
 27C TABLES, VALUES, AND CORRELATIONS FOR A NUMBER OF PROPERTIES.
 27C THE CONTENTS OF THE SACRD INFORMATION ARE HANDLED BY THE JOSHUA
 27C SYSTEM = DP-1380. FITS OF DATA ARE PUT IN AS FORTRAN SUBPROGRAMS.
 27Z KRM

28K 1FILESYS
 28X 1 27
 28A GREENE,N.M.
 28A FORSBERG,V.M.
 28A RAIFORD,G.B.
 28A ARWOOD,J.W.
 28A FLANAGAN,G.F.
 28T SACRD= A DATA BASE FOR FAST REACTOR SAFETY COMPUTER CODES.
 28T VERSION 1 CONTENTS ARE GLOSSARY OF TERMINOLOGY.
 28R ORNL-5477/V2,SEP1978.
 28C A DESCRIPTION OF THE FILES CURRENT CONTENTS IS GIVEN ALONG WITH
 28C MORE INFORMATION ON THE FILE STRUCTURE. CATALOGING IS BY FIVE
 28C QUALIFIERS= FILE NAME, PROPERTY, MATERIAL, VERSION NAME, DATA TYPE
 28C UNITS ARE CHANGED IN DIFFERENT FILES AND TABLES OF SYMBOLS ARE
 28C INCLUDED IN THIS REPORT.
 28Z KRM
 31K 5GABRELEASEPIEXAM PERFILE UPUO2 PELLET
 31X 3 21 22 23
 31A SMALLEY,M.R.
 31T EVALUATION OF SAXTON CORE III FUEL MATERIALS PERFORMANCE
 31R WCAP-3385-57,JUL1974.
 31C THE 250 PUO2=UO2 RODS FROM THE SAXTON CORE II WERE RECONSTITUTED
 31C INTO SEVEN LOOSE LATTICE ASSEMBLIES AND IRRADIATED IN CORE III TO
 31C 51(MWD/KGU). GAMMA SCANNING, PROFILOMETRY, LENGTH MEASUREMENT,
 31C ANALYSIS OF FILL GAS, METALLOGRAPHY, CLAD HYDROGEN, AND MECHANICAL
 31C TESTING WERE PERFORMED.
 31C ROD RR WAS REMOVED EARLY AND REACHED A PEAK BURNUP OF 35(MWD/KGM).
 31C SPIRAL TRACE PROFILOMETRY INDICATING A 0.5 TO 1.4(MIL) DIAMETER
 31C DECREASE AS COMPARED TO ENDS. NO PRE-IRRADIATION GIVEN.
 31C BY ISOTROPIC MEASUREMENTS A PEAK BURNUP OF 33.2 MWD/KGU AND AN
 31C AVERAGE BURNUP OF 25.1 MWD/KGU WERE DETERMINED. FUEL TO CLAD
 31C BONDING WAS OBSERVED. PHOTOMICROGRAPHS INDICATED RESTRUCTURING
 31C UP TO 60 PERCENT OF RADIUS OF PELLET.
 31C ROD REMOVED AT MID-LIFE SHUT DOWN
 31C THE SEVEN RODS L2,B0,NI,RO,F8,GL,MQ WERE REMOVE FROM CENTER 3X3
 31C FOUR FROM 503-17-2, AND TWO FROM 503-17-9 ASSEMBLIES. ROD AVERAGE
 31C BURNUPS RANGED FROM 25.5 MWD/KGU TO 33.7 MWD/KGU. PEAK BURNUPS
 31C WERE FROM 1.2 TO 1.3 HIGHER AT 12 AND 18 INCHES FROM ROD BOTTOMS.
 31C ROD MQ HAD FAILURE INDICATIONS SO NO GAS RELEASE DATA. FILE HAS
 31C ONLY ONE STRAIN MEASUREMENT BUT ROTATIONAL PROFILOMETRY WAS DONE.
 31C MORE PELLET TO CLAD BONDING NOTED THAN IN RODS AT END OF CORE II.
 31C FISSION GAS BUBBLES WERE ASSOCIATED WITH PUO2 PARTICLES IN REGION
 31C OUTSIDE GRAIN GROWTH REGION. HOMOGENIZATION OCCURRED IN HIGH
 31C TEMPERATURE REGIONS. TENSILE AND BURST TESTS WERE PERFORMED ON
 31C THE CLADDING. CLADDING RETAINED SIGNIFICANT DUCTILITY.
 31C END OF LIFE RODS
 31C RODS IM,PF,BE,BK,GM,JX,LS,FI WERE SELECTED FOR EOL EXAMINATION
 31C 37 DEFECTED RODS WERE FOUND IN LOOSE LATTICE CORE BUT NONE OF THE
 31C ABOVE. ROD JX WAS FAILED. LOWER POWER RODS SHOWN SIGNIFICANTLY
 31C LOWER RELEASES THAN HIGHER POWER RODS. COMPARATIVE EARLY HISTORY
 31C AND THEN LOWER POWER REDUCES RELEASE FRACTION WHEN ROD FI AND LS
 31C COMPARED WITH RODS CH,TE,TI,TP,TT OF CORE II. RELEASE FROM ROD
 31C GM IS QUESTIONABLE. FAILURES ARE PROPOSED TO BE BY OXIDATION OF

31C OTHER SURFACE POSSIBLY WHERE CRUD LAYERS FORMED. ROD JX AND BE
 31C BOTH HAD HEAVY OXIDES. FUEL CRACKING SEEMED LESS AT HIGHER BURNUP
 31C WITH LESS CRACK VOLUME AND RESIDUAL FUEL-CLAD GAPS. COLUMNAR
 31C GRAIN GROWTH IN RODS BE, BK, JX AND SOME IN IM AND PF. LARGE PORES
 31C ASSOCIATED WITH PUO₂ PARTICLES. BOTH EQUIAXED AND COLUMNAR GRAIN
 31C GROWTH REGIONS SHOWED REDISTRIBUTION OF SMALL BUBBLES TO GRAIN
 31C BOUNDARIES. LARGE GAS BUBBLES IMMEDIATELY OUTSIDE EQUIAXED ZONE.
 31C TENSILE AND BURST TESTING OF CLADDING PERFORMED. A PERIOD OF TWO
 31C MONTHS LOWER POWER AT END OF LIFE CAUSED HIGHER TENSILE AND BURST
 31C STRENGTHS. DUCTILITY LOWERED BY STILL .GT. 2 PERCENT.

31Z KRM
 54K SFISPROD THUO2 PIEXAN IRRADIATEDRELOCATIONRESTRUCTUR

54A BERMAN, R.M.

54T FISSION PRODUCT DISTRIBUTION IN OXIDE FUELS

54T LWR DEVELOPMENT PROGRAM

54R WAPD-TM-1236, DEC 1976.

54C DATA IS PRESENTED ON THE RELOCATION OF FISSION PRODUCTS AFTER
 54C IRRADIATION TO $14.59E+20$ AND $20.83E+20$ (FIS/CM**3).

54C SPECIMEN WERE 20 W/O UO₂. TRAVERSES WERE MADE FOR XENON,

54C ZIRCONIUM, STRONTIUM, CERIUM, BARIUM, R. THENIUM, CESIUM,

54C NEODYMIUM, MOLYBDENUM, AND THORIUM. COUNT DATA IS IN TABLES.

54Z KRM

56K 3UPUO2 PLUTONIA CHEMCUMPA

56A DAVIES, J.H.

56A EWART, F.T.

56A TAYLOR, R.G.

56T THE EFFECT OF BURNUP ON THE O/M RATIO OF IRRADIATED (U,PU)O₂

56T AND PUO₂

56R AERE-R6264, DEC 1969.

56C THIS REPORT ABOUT THE CHANGING CHEMICAL NATURE OF (U,PU)O₂ FUEL

56C SYSTEM WITH IRRADIATION. THEORETICAL CONSIDERATIONS ON CHEMICAL

56C ACTIVITY OF GENERATED FISSION PRODUCTS PREDICTS OXYGEN EXCESS

56C WILL BE GREATER IN MIXED OXIDES THAN IN UO₂. FISSION PRODUCT

56C YIELDS AND RELATIVE CHEMICAL STABILITY WERE PROGRAM TO CALCULATE

56C OXYGEN POTENTIAL OR PARTIAL MOLAR FREE ENERGY--

56C $DEL G(O_2) = RT \ln P(O_2)$

56C O/M RATIO OF IRRADIATED FUEL DIFFICULT TO DETERMINE BECAUSE FUEL

56C CONTAINS THE FISSION PRODUCTS, BUT CO/CO₂ GAS EQUILIBRATION METHOD

56C SELECTED AS MOST APPROPRIATE. SPECIAL IMBEDDED (U,7,PU,3)O₂ AND

56C PUO₂ SAMPLES IRRADIATED (NOT ACTUAL PELLETS). BURNUPS FROM

56C 4.3 TO 16.8 (ATOM PERCENT) ACHIEVED AND EXCESS O/M MEASURED.

56C DATA IS GIVEN O/M INCREASED WITH BURNUP. DISCUSSION IS INCLUDED

56C ON REDISTRIBUTION THAT SHOULD OCCUR IN ACTUAL FUEL BECAUSE OF

56C DIFFUSION.

56Z KRM

64K 5UPUO2 SWELLING THERMOCONDGASRELEASESREVIEWART

64A CRAIG, C.N.

64A APPLEBE, W.K.

64A PERRY, K.J.

64A DUNCAN, R.N.

64A BAILEY, W.E.

64A SPALARTS, C.N.

64T STEADY STATE PERFORMANCE OF PUO₂-UO₂ FAST REACTOR FUEL

04R PROCEEDINGS OF THE CONFERENCE ON FAST REACTOR FUEL ELEMENT
04R TECHNOLOGY, 13-15 APRIL, 1971, NEW ORLEANS, P555-578.
04C PLUTONIUM MIGRATION WAS OBSERVED IN FAST REACTOR FUEL PINS WITH
04C O/M OF 1.94 AND 1.97 BUT NOT IN 2.00 FUELS WITH LMGR.GT.8KW/FT
04C AND BURNUPS.GT.50(MWD/KGM) PLOTS OF (U.8.PU91)O2 VERSUS TEMPERATURE
04C AND FOR VARIOUS STOICHIOMETRY ARE GIVEN. CONDUCTIVITY DECREASED AS
04C O/M DECREASED ABOVE 80 (MWD/KGM) NEARLY ALL GENERATED INERT
04C FISSION GAS IS RELEASED. SUMMARY DATA FOR HIGH IRRADIATION OF
04C LMFBR FUEL PINS GIVEN.
04Z KRM
05K 46SWELLING GASRELEASETHEORY URANIA
05X 1 422
05A NICHOLS,F.A.
05A WARNER,H.R.
05T SWELLING AND GAS RELEASE MODELS FOR OXIDE FUEL RODS
05T LWR DEVELOPMENT PROGRAM
05R PROCEEDINGS OF THE CONFERENCE ON FAST REACTOR FUEL ELEMENT
05R TECHNOLOGY, 13-15 APRIL 1971, NEW ORLEANS, P.237-289.
05C THEORETICAL PRESENTATION OF INERT FISSION GAS BEHAVING AND THE
05C ACCUMULATION IN PORES. THE MIGRATION OF PORES IS ALSO TREATED AND
05C HOW THE BUBL PROGRAM HANDLES THESE PHENOMENA IS ALSO COVERED.
05C PAPER IS NOT A DATA SOURCE BUT A DEFENCE OF THE BUBL PREDICTIVE
05C CAPABILITY
05C PAPER DISCUSSES CHANGES TO INITIAL BUBL CODE THAT ARE IMPORTANT
05C FOR LOW TEMPERATURES ESPECIALLY IMMOBILIZATION OF SMALL BUBBLES
05C DISCUSSION OF IMMOBILIZATION VERSUS RESOLUTION IS PRESENTED. PAPER
05C POINTS OUT THAT A LARGE AMOUNT OF FISSION GAS CAN BE KEPT IN
05C SOLUTION UP TO 1000C. GRAIN BOUNDARIES ALSO ATTRACT BUBBLES.
05C BUBL WAS CHANGED TO HAVE PINNED SMALL BUBBLES WHICH GIVES A HIGHER
05C DENSITY OF SMALL BUBBLES. ANOTHER CHANGE WAS TO NOT IMMEDIATELY
05C PUT BUBBLES IN THE IN-TRANSIT CATEGORY ONCE CRITICAL SIZE IS
05C REACHED.
05Z KRM
05K 2GASRELEASEUPUD2
05A SPALARIS,C.N.
05A SANDUSKY,D.W.
05A ZEBROSKI,E.L.
05T EFFECT OF TEMPERATURE AND BURNUP ON FISSION-GAS RELEASE IN
05T MIXED OXIDE FUEL
05R INTERNATIONAL SYMPOSIUM OF CERAMIC NUCLEAR FUELS
05R KRUGER,O.L., KAZNOFF, A.L. (ED.)/ ACS, MAY 1969.
05C LOW SWELLING OCCURS IN FAST REACTOR FUELS WHICH MAY BE RELATED TO
05C HIGH GAS RELEASE.
05D DELV/V = .002 TO .004 DEP 1.E+20FIS/CM**3
05P A SUMMARY PERFORMANCE FILE OF DATA INCLUDED WITH BURNUPS FROM 1 TO
05P 330 GWD/MTU. THERMAL IRRADIATIONS BUT OF FAST REACTOR FUELS
05P FISSION RELEASE FRACTION GIVEN.
05C HIGH GAS RELEASES OBSERVED AND INCREASING WITH BURNUP.
05C VERY SMALL DEPENDENCE WAS NOTED ON POWER BUT RELEASE LINEARLY
05C DEPENDENT ON DENSITY. POSITION OF COLUMNAR GRAIN BOUNDARY SHIFTED
05C WITH BURNUP. A LOT OF UNCHARACTERIZED DATA GIVEN IN THIS REPORT.
05C DATA IS NOT REPRESENTATIVE OF LWR CONDITIONS.
05Z KRM

93K 4GASRELEASEUPU02 IRRADIATEDPIEXAM
93A STAHL,D.
93A PATRICIAN,T.J.
93T FISSION GAS BEHAVIOR DURING A MILD OVER-POWER TRANSIENT
93R ANL-8069, FEB 1974.
93C TWO RODS PNL-17-10 AND -24 WERE IRRADIATED IN EBR-1 AT LHGR=10KW/FT
93C TO 3.2 AT=PERCENT BURNUP. -10 WAS EXAMINED AFTER IRRADIATION AND -24
93C WAS SUBJECTED TO OVERPOWER TRANSIENT IN TREAT. FUEL WAS TO=9325.
93C (U.75,PU.25)01.958 WITH .493-CM-DIA. PELLETS. DOCUMENT DESCRIBES
93C THE DETAILS OF THE OBSERVED MICROSTRUCTURE OF THE TWO RODS.
93C IN PNL-17-10, PORES IN NONRESTRUCTURED ZONE APPROX. 650(A). IN
93C RESTRUCTURED EQUIAXED REGION FRACTURES ARE INTERGRANULAR, BUBBLES
93C ARE LARGER AND INCREASE IN SIZE WITH TEMPERATURE. IF INTRAGRANULAR
93C FRACTURE THEN BUBBLES SAME SIZE AS IN COOLER REGION. ON BOUNDARY
93C BUBBLE COALESCENCE OBSERVABLE. IN TRANSIENT TESTED PNL-17-24 MORE
93C INTERGRANULAR BUBBLES IN NONRESTRUCTURED REGION. IN RESTRUCTURED
93C REGION A NEW POPULATION OF SMALLER BUBBLES OBSERVED ON GRAIN
93C BOUNDARY. QUANTITATIVE MEASUREMENTS INDICATED NEW SMALLER BUBBLES
93C NUCLEATED BUT LARGER APPROX. SAME SIZE. GAS ATOM CONTENT IN BUBBLES
93C CALCULATED FROM MICROSTRUCTURE COMPARED WITH THAT RELEASED BY
93C LASER SAMPLING AND FOUND TO BE SIMILAR. DATA ONLY USEFUL FOR
93C MICROSTRUCTURAL PHENOMENOLOGICAL MODEL EVALUATION.
93Z KRM
138K 3GASRELEASEUPU02 PERFIL
138A ZIMMERMANN,H.
138T FISSION GAS BEHAVIOR IN OXIDE FUEL ELEMENTS OF FAST BREEDER
138T REACTORS
138R NUCLEAR TECHNOLOGY, VOL. 28, P. 127-133, JAN 19769
138P FUEL IRRADIATIONS DESCRIBED WERE IRRADIATED IN FR2-KARLSRUHE,
138P BR2-MOL, OFR-DOUNREAY, RAPSODIE-CADARACHE. REPORTS DESCRIBING
138P IRRADIATION ARE REFERENCED. FOR FAST REACTOR FUEL, ENOUGH HISTORY
138P REFERENCED TO BE THE SOURCE FOR PERFORMANCE FILE ENTRIES.
138C FISSION GAS MEASURED IN THREE STEPS- RELEASED, IN PORES-BUBBLES
138C AND IN MATRIX. GAS IN PORE AS AMOUNT RELEASED BY GRINDING.
138C THEN CHEMICALLY DISLOVED FOR IN MATRIX. ALL DATA IS PLOTTED IN
138C GRAPHICAL FORM AND VERSUS MEAN PARAMETERS SUCH AS AVERAGE TEMP.
138C BEFORE DISCHARGE, BURNUP, INITIAL PELLET DENSITY.
138C SHARP INCREASE OF RELEASE IN .02 TO .07 ATOM FRACTION BURNUP.
138C RELEASE FOR LOWER BURNUP DECREASES FOR INCREASING DENSITY OR
138C INCREASING O/M RATIO. BREAK AWAY RELEASE BELIEVED TO OCCUR AT
138C LOWER BURNUPS IN (U,PU)02 FUEL MECHANICALLY MIXED THAN IN UO2.
138C SINCE BURNUP HIGHER IN PU CONTAINING GRAINS. ABOVE 1700C-COLUMNAR
138C GRAINS-VERY LITTLE RETAINED GAS. QUALITATIVE DESCRIPTION OF
138C RELEASE GIVEN AND A REFERENCE SOURCE FOR POSSIBLE DETAILED DATA.
138Z KRM
249K 4GASRELEASESWELLING THEORY URANIA
249A POEPPPEL,R.B.
249T AN ADVANCED GAS RELEASE AND SWELLING ROUTINE
249H PROC CONF FAST REACTOR FUEL ELEMENT TECHNOLOGY, 331 (1971).
249C PRECIPITATION RATE DIFFUSION CONTROLLED. GROWTH BY COALESCENCE
249C RESOLUTION CAN OCCUR IN IRRADIATION ENVIRONMENT.
249C COALESCENCE BECAUSE OF RANDOM MOTION AND MAINLY BIASED
249C MIGRATION DUE TO THERMAL GRADIENT DRIVING FORCE.

249C TWO MODEL TYPES-PREScribed NUMBER AND PREScRIBED SIZE.
249C BUBL-1 IS COMBINATION WITH FIXED NUMBER OF SITES UNTIL RELEASE
249C AND THEN FIXED SIZE WHILE MIGRATION.
249C DISTRIBUTION FUNCTION APPROACH CAN BE LINKED TO OBSERVATIONS.
249C PAPER DESCRIBES THE GRASS CODE.
249C THREE TYPES OF BUBBLE DISTRIBUTIONS
249C 1-BUBBLES NOT PINNED TO STRUCTURAL DEFECTS
249C 2-BUBBLES PINNED TO DISLOCATIONS
249C 3-BUBBLES PINNED TO GRAIN BOUNDARIES
249C CRITERIA ARE GIVEN WHEN SIZE IS LARGE ENOUGH TO PULL FREE.
249C FREE BUBBLE MIGRATION VELOCITY IS THE LARGEST OF
249C 1-SURFACE DIFFUSION
249C 2-VOLUME DIFFUSION
249C 3-EVAPORATION CONDENSATION
249C DISTRIBUTION BASED ON DISCRETE LOGARITHMIC SCALE OF NUMBER OF
249C GAS ATOMS WITH SIZE DETERMINED WITH GAS EQ. PRESSURE IN
249C EQUILIBRIUM WITH SURFACE TENSION AND HYDROSTATIC PRESSURE.
249Z KRM
258K 2GASRELEASETHEORY
258A RONCHI, G.
258A MATZKE, H.J.
258T CALCULATIONS ON THE IN-PILE BEHAVIOR OF FISSION GAS IN OXIDE FUELS.
258R JOUR. OF NUCLEAR MATERIALS, VOL.45 PP15-28,(1972).
258C GAS PRODUCED = DISSOLVED + INTRAGRANULAR BUBBLES + GRAIN BOUND.
258C CONSTANT VOLUME (SAME NUMBER OF NUCLEI) DIFFUSION EQUATION
258C SOLVED FOR DISSOLVED GASES GOING TO INTRAGRANULAR BUBBLES.
258C RESOLUTION RATE NEEDED FOR LOW TEMPERATURE EFFECTS.
258C TRANSPORT OF BUBBLES TO THE BOUNDARIES BY MIGRATION NOT
258C CONSIDERED ONLY ATOMIC DIFFUSION. (RESTRICTS T.GT.1000C USE.)
258C GAS REACHING THE GRAIN BOUNDARIES IS THAT REACHING THE PLENUM.
258C AT LOWER TEMPERATURES, <LT.1000C, GAS GENERATED ONLY PARTIALLY
258C REACHES BUBBLES BECAUSE OF THE RESOLUTION EFFECT.
258Z KRM
264K 2GASRELEASEEXTRADATA
264A TURNER, S.E.
264A BEYER, C.E.
264A DAVIS, J.R.
264A HAIRE, M.J.
264A KLOTZ, R.J.
264A LORENZ, R.A.
264A MEYER, R.O.
264A MILLER, J.V.
264A NOBLE, L.D.
264A NOTLEY, M.J.F.
264A RIN, C.S.
264A RITZMAN, R.L.
264T STATUS REPORT AND 5.4 FUEL PLENUM GAS ACTIVITY (N21A) = (FISSION
264T PRODUCT RELEASE FROM UO2 FUEL).
264R APR1977.
264C STUDY INCLUDES --REVIEW AVAILABLE DATA, SURVEY OF APPLIED MODELS,
264C FISSION PRODUCTS. MODEL SELECTED BY GROUP IS THE B00TH DIFFUSION
264C AND DEVELOPMENT OF ANALYTICAL MODEL FOR RELEASE OF VOLATILE
264C FOR RELEASE FROM A SPHERE. KRYPTON AND XENON MAY BE ASSUMED TO

264C HAVE SIMILAR DIFFUSION COEFFICIENTS BUT THIS IS NOT TRUE FOR
 264C IODINE. USE OF BOOTH MODEL RECOGNIZED AS OVER SIMPLIFICATION AND
 264C DIFFUSION COEFFICIENT EMPIRICALLY DERIVED CAN NOT BE GIVEN ATOMIC
 264C INTERPRETATION. RECOIL RELEASE WILL HAVE TO BE CONSIDERED.
 264C DATA BASE SELECTED FOR CORRELATION IS THAT USED BY MEYER-HANN.
 264C THIS IS A RESTRICTED CAPSULE DATA BASE. IN DATA REDUCTION BURNUP
 264C BASED ON 200 MEV/FIS, HEAT GENERATION IN FUEL 182 MEV/FIS, AND
 264C BASED ON 200 MEV/FIS, HEAT GENERATION IN FUEL 182 MEV/FIS, AND
 264C 26.9 CC/MWD FOR XE AND 4.1 CC/MWD FOR KR PRODUCTION. LYONIS
 264C THERMAL CONDUCTIVITY AND MAXWELL-EUKEN POROSITY CORRECTION.
 264C MATHEMATICAL SERIES SOLUTIONS FOR SPHERICAL SYMMETRY WITH CONSTANT
 264C AND TIME DEPENDENT PARAMETERS WITH EXTENSION TO TIME DEPENDENT
 264C POWER HISTORY ARE GIVEN. THE CONSTANT POWER-COEFFICIENT RELATION
 264C WAS FIT TO THE DATA BASE USING A GRID SEARCH TECHNIQUE. FITTED
 264C PARAMETERS WERE==
 264C $DO/A^{**2}=.00221$ (SEC**2)=PRE-EXPONENTIAL DIF.COEF. TERM
 264C $Q=49.7$ (KCAL/MOLE) ACTIVATION ENERGY
 264C SAXTON DATA WAS USED TO CHECK VALIDITY FOR HIGH BURNUP. VARYING
 264C POWER HISTORY FORM WAS USED FOR THIS CHECK. WITH ABOVE COEF. DATA
 264C WAS UNDERPREDICTED. DISCREPANCY ATTRIBUTED TO BURNUP EFFECT
 264C WHICH WAS APPROXIMATED AS==
 264C $DBO = DO*F$,AND, $F=100*(BU/30000)$ WITH BU (MWD/MTM)
 264C SINCE SAXTON RODS ARE MIXED OXIDE BLENDED FUEL, THE POSSIBLE
 264C EFFECTS OF THIS WERE DISCUSSED. PARTICLE SIZE OF PUO_2 LESS THAN
 264C 44 MICRONS. LOCALIZED CONCENTRATION OF FISSION PRODUCTS IN
 264C PARTICLE .GT. 20 MICRONS COULD BE SIGNIFICANT FACTOR.
 264Z KRM
 265K 1GASRELEASE
 265A MEYER,R.O.
 265A BEYER,C.E.
 265A VOGLEWEDE,J.C.
 265T FISSION GAS RELEASE FROM FUEL AT HIGH BURNUP.
 265R NUREG-0418(DRAFT),DEC1977.
 265C EVIDENCE OF INCREASED RELEASE ABOVE 30MWD/KGU. SOURCES OF DATA
 265C SUPPORTING INCREASED RELEASE ARE SAXTON MIXED OXIDE DATA, ZORIYA
 265C WESTINGHOUSE DATA, AND LMFBR MIXED-OXIDE DATA FROM EBR-II. NRC
 265C HAS REQUIRED USE OF THEIR RECOMMENDED BURNUP ENHANCED RELEASE
 265C CORRELATION UNTIL AN APPROVED MODEL IS OBTAINED. THE BURNUP
 265C ENHANCED RELEASE FRACTION, FENH, IN TERMS OF PREVIOUS FRACTION, F==
 265C $FENH(BU,T) = F(T) + (1.-F(T))*Y$ FOR $BU,GR.20$ (MWD/KGU)
 265C $Y = (1.-EXP(-.0436*(BU-20.)))/(1.+(.665/F(T))$
 265C $*EXP(-.1107*(BU-20.))$
 265C ANOTHER RESTRICTION IS THAT RELEASES LESS THAN $F=0.01$ ARE NOT
 265C ACCEPTABLE. IN APPENDICES ARE LMFBR AND KWU GAS RELEASE DATA BUT
 265C DETAILS OF HISTORY AND FUEL DESIGN ARE INADEQUATE.
 265Z KRM
 266K 2COMPCODE GASRELEASE
 266A ROBERTS,E.
 266A BALFOUR,M.G.
 266A HOPKINS,G.W.
 266A SHALLEY,M.R.
 266A DEQUIDT,E.
 266T FUEL MODELING AND PERFORMANCE OF HIGH BURNUP FUEL RODS.

266R ANS WATER REACTOR FUEL PERFORMANCE ST. CHARLES, ILL., MAY 1977.
266C JUSTIFICATION OF WESTINGHOUSE PAD CODE WITH SOME ZORITA DATA FOR
266C BURNUPS TO 65 MWG/KGU. PAD ANALYZES AXIAL SEGMENTED ROD FOR
266C PERIODS OF CONSTANT POWER WHICH ARE CHANGEABLE. HAS USUAL MODELS
266C FOR FUEL BEHAVIOR BUT REPORT DOES NOT ANALYTICALLY DESCRIBE ANY OF
266C THE MODELS. FRACTIONS OF GAS RELEASE COULD BE OBTAINED FROM PLOTS
266C BUT NO HISTORIES OF DESIGN DETAILS GIVEN. COMMERCIAL LHGR OF FROM
266C 4.0 TO 9.6 KW/FT WERE EXPERIENCED. FISSION GAS RELEASE VERY
266C SENSITIVE TO BOTH BURNUP AND POWER. ROD DIAMETERS DECREASED AND
266C RIDGES UP TO 1 MIL WERE FORMED. 50 ZORITA RODS PUNCTURED WITH
266C 32 RODS IRRADIATED TO 28,000 EFPH. NO WAY TO GET MODELING DATA
266C FROM THE PAPER BUT RELEASES UP TO .27 OBSERVED IN ZORITA RODS.
266Z KRM
268K 3GASRELEASETHEORY REVIEWART
268A DIAS, J.W.C.
268A MERCKX, K.R.
268T THEORIES OF FISSION GAS BEHAVIOUR
268R ANNAL NUCL.ENERGY, VOL 3, PP41-53, 1976.
268C THIS REVIEW COVERS THE THEORETICAL MODELS FOR GAS ATOM SOLUBILITY
268C DIFFUSIVITY, BUBBLE NUCLEATION, BUBBLE GROWTH BY DIFFUSION AND BY
268C SURFACE VOLUME, AND VAPOR DIFFUSION IS COVERED IN THE BUBBLE
268C TRANSPORT MODELLING. RETARDING FORCES ON BUBBLES DUE TO STRUCTURAL
268C DEFECTS ARE ALSO DISCUSSED.
268Z KRM
291K 4SWELLING URANIA GASRELEASEFAILURE
291A BANDYOPADHYAY, G.
291T RESPONSE OF OXIDE FUEL TO SIMULATED THERMAL TRANSIENTS
291R NUCLEAR TECHNOLOGY, 41, P. 349-358, DEC. 1978.
291C DIRECT ELECTRIC HEATING OF IRRADIATED PELLETS IS USED TO SIMULATE
291C TRANSIENT ACCIDENT CONDITIONS. TEST STACKS ARE 25(MM). ENOUGH
291C ENERGY IS OFTEN ADDED TO CAUSE FUEL MELTING. IN IRRADIATED FUEL
291C TESTS, GROSS SWELLING OCCURS IN SLOWER ENERGY DEPOSITION RATE
291C TESTS. HIGH ENERGY DEPOSITION RATES CAUSE GRAIN BOUNDARY SEPARATION.
291C DATA FOR SUPPORT OF SAFETY STUDIES ON REACTIVITY INSERTION TYPE OF
291C ACCIDENT MODELS AND POSSIBLE FUEL DISPERSION.
291Z KRM
296K 3URANIA GASRELEASEFISPRON
296A CUBICCIOTTI, D.
296T STATE-OF-THE-ART MODEL FOR RELEASE OF VOLATILE FISSION PRODUCTS
296T FROM UO2 FUEL
296R EPRI PROJECT RP355-11, (SRI PROJECT 7453) FINAL REPORT, DEC. 1978.
296C RESULTS OF A SIX MONTH STUDY BY SRI FOR A FISSION PRODUCT RELEASE
296C MODEL BACKGROUND PRESENTED. RELEASE CREATES CHEMICAL ENVIRONMENT
296C FOR SCC. CORRELATION BETWEEN FISSION GAS (KR+XE) AND VOLATILE
296C FISSION PRODUCTS (CS, I, TE, CD, ETC.) EXISTS BUT MAJOR DIFFERENCES
296C ALSO OCCUR. A REVIEW OF FISSION GAS RELEASE MODELS IS INCLUDED
296C THAT RELATES GAS RELEASE TO VOLATILE PRODUCTS (INCOMPLETE REVIEW).
296C MODEL PROPOSED COVERS (CS, I, TE, CD, RB, BR, SE, AG, SN, BA, PO).
296C THESE ARE NOT RELEASED AT LOW POWERS. RELEASED AFTER OR DURING
296C RAMPS PRODUCTS MIGRATE TOWARD PERIPHERY OF FUEL AT MODERATE
296C POWERS. EFFECTS OF CESIUM ON PRESSURE IN BUBBLES IN 1300 TO 1700K
296C RANGE COULD BE MUCH GREATER THAN XE AND KR. DEPOSITED CS ON GRAIN
296C BOUNDARIES CAN BE CAUSE OF TEARING DURING RAMPS. SLOW TEMPERATURE

296C INCREASES ALLOW BUBBLE GROWTH AND CONDENSATION IN FUEL AND MAYBE
296C MAJOR FACTOR IN RATE EFFECTS ALSO, PERIOD FROM LAST RISE CAN BE
296C SIGNIFICANT BECAUSE IT INFLUENCES BUILDUP OF VFP. AFTER MODERATE
296C BURNUPS ESTIMATED THRESHOLDS WERE 1000-1100C FOR MIGRATION,
296C 1400C FOR RELEASE AND RAMP RATES GREATER THAN .09 W/M*SEC FOR
296C PULSE RELEASE
296C COMMENTS ARE INCLUDED ON A NUMBER OF REFERENCES.
296Z KRM
299K TURANIA UPII02 SWELLING GASRELEASEFISPROD EXTRADATA CREEP
299K 8 8DENSIFY
299X 1 138
299A ZIMMERMAN, H.
299T UNTERSUCHUNGEN ZUM SCHWELLEN UND SPALTGASVERHALTEN IN OXIDISCHEM
299T KERNBRENNSTOFF UNTER NEUTRONENBESTRAHLUNG
299R KFK 2467, MAY 1977.
299C UO2 CAPSULES WERE IRRADIATED BETWEEN 1300K AND 2000K IN FREE AND
299C RESTRAINED CONDITIONS AND RELEASE, RETAINED IN LARGE BUBBLES AND
299C WERE ALSO MEASURED, THEY HAD HIGHER TEMPERATURE GRADIENTS.
299C AT 1250K, .014 GAS ATOM PER HEAVY METAL ATOM, WHILE AT 1900K
299C 0.0025 GA/HMA IN UO2 AND 0.0002 GA/HMA IN (U,PU)O2 W/HT TEMP.
299C GRADIENTS WERE THE LIMITING RETAINED GAS CONCENTRATIONS.
299C SWELLING DUE TO RETAINED FISSION GAS REACHED LIMITING VALUES
299C OF .2 DELTA V/V AT 2000K AND .05 AT 1250K. HIGHER SWELLING DOES NOT
299C OCCUR BECAUSE OF THE RELEASE OF FISSION GAS. THE SOLID FISSION
299C PRODUCT DECREASED FROM .0065 TO .003 (DV/V) PERCENT BURNUP AS THE
299C TEMPERATURE INCREASED BECAUSE OF THE RELEASE OF VOLATILE FISSION
299C PRODUCTS
299C THIS REPORT CONTAINS A REVIEW OF THE LITERATURE ON SWELLING
299C BUBBLE FORMATION AND MIGRATION AND NONGASEADOUS FISSION PRODUCT
299C DISTRIBUTION, REDISTRIBUTION OF THE SOLID FISSION PRODUCTS AS WELL
299C AS GAS RELEASE INFLUENCES SWELLING. AN EXTENSIVE REVIEW OF THE
299C EXPERIMENTAL SWELLING LITERATURE IS GIVEN. A GROSS CORRELATION OF
299C UO2 AND (U,PU)O2 IN THE LOWER TEMPERATURE REGIONS.
299C THESE SPECIMENS REACHED A MAXIMUM OF ONE PERCENT BURNUP.
299C THE GAS RELEASE APPROACH FOR THIS REPORT STRESSES THE RETENTION
299C CAPABILITIES OF THE FUEL. A TEMPERATURE DEPENDENT BREAKAWAY
299C RELEASE MECHANISM OCCURS.
299D BREAKAWAY BURNUP = 6.0 (PERCENT, (T.LT.1320K)
299D = 4.0 (T.GT.1320).OR.(T.LT.1770K)
299D = 2.0 (T.GT.1770K).OR.(T.LT.2020K)
299D REF. FROST, INT. SYM. CER. NUCL. FUELS, MAY 1969.
299C TWO TYPES OF CAPSULES WERE USED TO OBTAIN FREE SWELLING AND
299C RESTRAINED SWELLING RESULTS. FREE SWELLING USED THIN DISKS,
299C AND THE RESTRAINED USED AN AXIAL PISTON AND THICK WALLED CYLINDER
299C TABLE 3 PAGE 36 HAS 41 VOLUME CHANGE AND 31 RESIDUAL GAS DATA.
299C THE NON PRESSURIZED CAPSULES SHOWED A LARGE TEMPERATURE DEPENDENCE
299C FOR SWELLING BUT THIS DID NOT CONTINUE TO HIGH BURNUPS MUCH LESS
299C TEMPERATURE AND NEGLIGIBLE PRESSURE DEPENDENCE WAS SHOWN BY THE
299C PRESSURIZED CAPSULES.
299C FOR THE UNRESTRAINED SAMPLE, THE GAS RELEASE WAS CALCULATED AS THE
299C DIFFERENCE BETWEEN GENERATED AND RETAINED, BECAUSE OF THE SMALL
299C SAMPLE SIZE. FOR THE CALCULATIONS, .31 GAS ATOM/FISSION WAS ASSUMED.
299C THIS ACCOUNTS FOR THE THERMAL NEUTRON TRANSFORMATION OF XE-135 TO

299C STABLE XE-136. THE XE/KR RATIO OF 7.4 FOR U-235 FISSIONS WAS ALSO
 299C FOUND TO AGREE WITH MEASUREMENTS.
 299C THE RESULTS FROM A NUMBER OF (U,Pu)O₂ IRRADIATIONS ARE SUMMARIZED
 299C AND REFERENCE REPORTS ARE CITED. TABLE 4 PAGE 39. THE FUEL
 299C COMPOSITIONS WERE TYPICAL OF FAST REACTOR FUEL. IN THESE SAMPLES,
 299C .4 TO .7 FRACTION WAS IN THE COLUMNAR REGION.
 299C THE INREACTOR DENSIFICATION PROCESS IS FORMULATED FOR BOTH
 299C THERMAL AND IRRADIATION DEPENDENCE. VACANCY GENERATION BY FISSION
 299C FRAGMENTS AND DISLOCATIONS AND GRAIN BOUNDARIES AS SINKS ARE
 299C CONSIDERED.
 299C THE RELATIONS FOR GAS RELEASE ARE DEVELOPED AS A TWO STAGE PROCESS.
 299C FIRST STAGE IS DIFFUSION TO THE GRAIN BOUNDARY AND THIS RELEASE
 299C FRACTION IS MULTIPLIED BY A STRUCTURE DEPENDENT AND GAS CONTENT
 299C FACTOR WHICH ALSO HAS A DIFFUSION COEFFICIENT. THESE RELATIONS
 299C ARE FIT TO THE MEASURED RELEASE VALUES. A CHANGING POWER CASE IS
 299C ALSO DISCUSSED WITH RESPECT TO BREAK AWAY RELEASE AND ACCUMULATION
 299C DURING THE VARIOUS POWER REGIMES. THE RELEASE OF CESIUM IS ALSO
 299C COMPARED TO THE INERT GAS RELEASE ON FOUND TO BE SIMILAR BUT
 299C DELAYED IN BURNUP DEPENDENCE. SINCE CE133 AND CE135 ARE DAUGHTER
 299C PRODUCTS OF XE, THEY COULD BE RELEASED AS XE. RELEASE OF CESIUM
 299C AFFECTS SOLID FISSION PRODUCT SWELLING BECAUSE IT CONTRIBUTES
 299C 60 PCT. THEORETICALLY .0066(DV/V) PERCENT BU COMES FROM ALL PRODUCTS
 299C 0.7 OF THIS IS FROM VOLATILE PRODUCTS WHICH 0.8 IS CESIUM. THE
 299C RESULTANT SWELLING IS
 299D $DV/V = (0.0019 + 0.0047(1-FCS)) * BU$ -- FOR UO₂
 299D BU = BURNUP (PERCENT)
 299D $DV/V =$ FRACTIONAL VOLUME CHANGE
 299D FCS = FRACTION OF CESIUM RELEASED
 299C SWELLING CONTRIBUTIONS OF THE INERT FISSION GASES COMES FROM THREE
 299C SOURCES - 1) GRAIN BOUNDARY BUBBLE, 2) BUBBLE IN THE GRAINS AND
 299C 3) GAS IN THE LATTICE. THE LARGEST CONTRIBUTION COMES FROM
 299C GRAIN BOUNDARY BUBBLE. SMALL GRAIN SIZES HAVE MORE GRAIN BOUNDARY
 299C AREA, HENCE MORE SWELLING. AT HIGHER TEMPERATURES GRAIN GROWTH
 299C OCCURS AND DECREASES THIS CONTRIBUTION. RELATIONS ARE DEVELOPED
 299C FOR THE SWELLING CONTRIBUTION. RELATIONS ARE DEVELOPED
 299C FOR THE SWELLING CONTRIBUTION FROM EACH SOURCE.
 299C QUOTED LIMITS FOR ABILITY FOR ACCUMULATE GAS WERE DETERMINED
 299C WITH ISOTHERMAL TESTS. THE TEMPERATURE GRADIENTS IN FUEL RODS
 299C REDUCE THIS ACCUMULATION CAPABILITY IN THE HIGH TEMPERATURE
 299C REGIONS.
 299C THIS PAPER CONTAINS A GREAT DEAL OF INFORMATION AND IS
 299C DEVELOPED IN ENOUGH DETAIL TO BE USEFUL. IT HAS 162 REFERENCES AND
 299C COVERS BOTH THE GERMAN, ENGLISH, AND AMERICAN LITERATURE. IT IS
 299C IN GERMAN.
 299Z KRM
 312K SUPUO2 PIEXAM SWELLING GASRELEASEVIPAC
 312A PERRY, K.J.
 312A BAILEY, W.E.
 312T HIGH BURNUP PERFORMANCE OF POWDER AND PELLET MIXED-OXIDE FUEL
 312R GEAP-10385, AEC RESEARCH AND DEVELOPMENT REPORT, JULY 1971.
 312C TWO-LMFBR-TYPE FUEL RODS WERE IRRADIATED IN A THERMAL FLUX TO
 312C OBTAIN PERFORMANCE DATA FOR COMPARING VIPAC AND PELLET FUELS UNDER
 312C TYPICAL ADVANCED LMFBR OPERATING CONDITIONS. THERMOCOUPLES USED TO

312C MONITOR TEMP. BOTH FUEL TYPES CLAD IN SS347, FUEL SHEAR DENSITY OF
312C 84 PCT. TD, LINEAR POWER 23.5KW/FT, CLAD TEMP 1300 TO 1400 F,
312C BURNUP 149,000MWD/T
312C PIE RESULTS SHOWED MAX DIAMETER INCREASE OF 1.2 PCT. FOR ROD WITH
312C PELLET FUEL AND 0.7 PCT. FOR VIPAC FUEL ROD.
312C FISSION GAS RELEASE OF 91 PCT. CALCULATED FOR PELLET ROD AND
312C 78 PCT. FOR VIPAC ROD. FUEL COLUMN LENGTH INCREASES OF 8 PCT. AND 3 PCT.
312C FOR PELLET, POWDER FUEL WERE MEASURED
312C FUEL SWELLING RATES MEASURED AT .28 DELTA V/VPER 10*2
312C FISSIONS/CC(7 PCT./100,000MWD/T).
312C CENTER FUEL MELT OCCURRED IN BOTH RODS (MP REDUCED 50-100C,
312C GT.100,000MWD/MTM).
312C FUEL MICROSTRUCTURES SHOWED THERMAL PERFORMANCE OF TWO FUELS
312C SIMILAR AT HIGH BURNUPS.
312C 2ND PHASE ZONE 0.001 IN THICK DUE TO CLADDING REACTION WITH
312C VIPAC (FE,PD,MO,BA,CB)
312C DIFFUSION ZONE = 0.003-.004 IN. IN CLADDING, WALL THICKNESS REDUCED
312C UP TO .003 IN.
312C 2ND PHASE ZONE=0.003-.004 IN. IN PELLET ROD, WALL REDUCED UP TO .005 IN.
312Z CJC PGS
406K STHEORY URANIA COMPCODE SWELLING GASRELEASE
406A REST:J.
406T GRASS-88T = A COMPREHENSIVE MECHANISTIC MODEL FOR THE PREDICTION
406T OF FISSION-GAS BEHAVIOR IN UO2-BA9E FUELS DURING STEADY-STATE AND
406T TRANSIENT CONDITIONS
406R NUREG/CR-0202, ANL-78-53, JUNE 1978.
406C THE GRASS-88T COMPUTER CODE IS A MECHANISTIC CODE ACCOUNTING FOR
406C DIFFUSION, BUBBLE MIGRATION, GRAIN BOUNDARY COALESCENCE, CHANEL
406C FORMATION AND GRAIN BOUNDARY SEPARATION. BUBBLE SIZE DISTRIBUTION
406C RELATIONS ARE TRACKED THROUGH THESE PHENOMENA, THUS THE CODE IS
406C VERY DETAILED AND IS TOO INVOLVED FOR A MODEL IN A PERFORMANCE CODE.
406C THIS CODE IS AND HAS EVOLVED FROM AN EXTENDED PROGRAM AT ANL WHERE
406C MICROSTRUCTURAL OBSERVATIONS AND SPECIALIZED TESTS HAVE BEEN
406C CONDUCTED TO VERIFY THE MODEL. THE GRASS CODE IS BUILT AROUND THE
406C BUBBLE DISTRIBUTION MODELS.
406C FOR EACH ANNULAR RING A DISCRETE NUMBER OF SIZE CELLS ARE FORMED.
406C A SET NUMBER OF GAS ATOMS IS ASSIGNED TO EACH CELL TYPE. NUMBER
406C OF BUBBLES ENTERING AND LEAVING EACH CELL TYPE IS TRACKED THRU THE
406C PHYSICAL MECHANISMS PREVIOUSLY DESCRIBED. WHEN BUBBLE SIZE IS
406C REQUIRED AS IN CALCULATING SWELLING THE GAS LAWS AND SURFACE
406C TENSION ARE USED TO CALCULATE SIZE. BESIDES SIZE, THE CELL TYPES
406C COVER LATTICE, DISLOCATION AND BOUNDARY LOCATIONS. THIS REPORT
406C ONLY OUTLINES THE CALCULATIONAL PROCEDURES.
406C A COMPREHENSIVE SENSITIVITY STUDY IS INCLUDED SHOWING BURNUP,
406C TEMPERATURE, GRAIN SIZE AND TEMPERATURE GRADIENT SENSITIVITY
406C ONLY TWO REPRESENTATIVE LWR FUEL IRRADIATIONS WERE INCLUDED IN
406C STEADY-STATE VERIFICATION AND HIGHER TEMPERATURE BEHAVIOR ALSO
406C CHECKED. MUCH WORK ON SIMULATED TRANSIENTS -DEH EXPERIMENTS- DONE
406C FOR TRANSIENT ANALYSIS VERIFICATION, GRAIN BOUNDARY BUBBLE LINKAGE
406C AND CRACKING IMPERTANT IN TRANSIENT TESTS.
406Z KRM
422K 48WELLING GASRELEASECOMPCODE URANIA
422X 1 421

422A WARNER, H.R.
 422T BUBL-1 - A STATISTICAL FUEL SWELLING AND FISSION GAS RELEASE MODEL
 422R WAPD-TM-942, SEP. 1970.
 422C THE THEORY OF DISLOCATION AND GRAIN BOUNDARY TRAPPING, VELOCITY
 422C IN A TEMPERATURE GRADIENT IS INCLUDED. THE COMPUTATIONAL CONCEPTS
 422C FOR THE COMPUTER SIMULATION FOR THE UNIT CELL STATISTICS IS GIVEN
 422C NUCLEATED BUBBLES, WHOSE RATE OF NUCLEATION IS DETERMINED BY
 422C FISSIONING RATE, BOMBARD DISLOCATIONS. (RE-SOLUTION IS NEGLECTED)
 422C POSITION ON DISLOCATION LINE, WHOSE LENGTH IS DETERMINED BY
 422C DISLOCATION DENSITY, IS DETERMINED BY RANDOM NUMBER GENERATOR.
 422C IF OVERLAP OCCURS, COALESCENCE IS ASSUMED. GAS LAW AND SURFACE
 422C TENSION USED FOR BUBBLE SIZE. BUBBLE MOTION DEPENDENT ON SIZE
 422C ALONG DISLOCATION ALLOWED. HANGUP AT NODES ALSO OCCURS. PULL OFF
 422C OCCURS WHEN CRITICAL SIZE IS REACHED AND BUBBLE PUT IN TRANSIENT
 422C DISTANCE TO GRAIN BOUNDARY DEPENDENT ON GRAIN SIZE IS SEQUENTIALLY
 422C SELECTED AND CALCULATED VELOCITY GIVES TIME OF GRAIN BOUNDARY
 422C ARRIVAL. BUBBLES RANDOMLY STRIKE GRAIN BOUNDARY AND COALESCENCE
 422C CHECKED BY FOR OVERLAP. WHEN CRITICAL SIZE REACHED, GRAIN BOUNDARY
 422C RELEASE OCCURS AND TRANSIENT TIME TO RELEASE AT CRACK OR FUEL
 422C CENTER. PARAMETERS FOR URANIA CALCULATIONS ARE GIVEN. VERIFICATION
 422C CALCULATIONS MADE FOR DATA SOURCES OF 1) STORHOK AND CHUBB
 422C 2) DANIEL. SENSITIVITY FOR TEMPERATURE GRADIENT, PRESSURE,
 422C DISLOCATION DENSITY, DISLOCATION NODES, GRAIN SIZE, RELEASE
 422C DISTANCE, FISSION RATE STUDIED WITH CODE. AT LOW TEMPERATURES
 422C DISLOCATION DENSITY SIGNIFICANT. BETWEEN 1100 TO 1600C PREDICTIONS
 422C DEPENDENT ON MOST PARAMETERS. NOT A DATA SOURCE BUT THIS CODE WAS
 422C ONE OF THE FIRST HAVING MANY OF PHENOMENA OF BUBBLES MODELLED.
 422Z KRM
 426K 4GASRELEASESWELLING URANIA DENSIFY
 426A BELLAMY, R.G.
 426A RICH, J.B.
 426T GRAIN-BOUNDARY GAS RELEASE AND SWELLING IN HIGH BURNUP URANIUM
 426T DIOXIDE.
 426R JOURNAL OF NUCLEAR MATERIAL, 33, P.64-76, 1969.
 426C A PRIMARY REFERENCE ON LOW TEMPERATURE GAS RELEASE, SWELLING AND
 426C IN REACTOR DENSIFICATION. SOURCE REPORT OF OBSERVATIONS OF
 426C DENSIFICATION AT LOW TEMPERATURES LOW TEMPERATURE BUT HIGH BURNUP
 426C GAS RELEASE IS TREATED BY A KNOCK-OUT MODEL. DATA PRESENTED ON
 426C LOW TEMPERATURE HIGH DENSITY URANIA FUEL IRRADIATIONS.
 426Z KRM
 429K 3DENSIFY CREEP URANIA
 429A REGULATORY STAFF
 429T TECHNICAL REPORT ON DENSIFICATION OF LIGHT WATER REACTOR FUELS.
 429R USAEC, NOV. 14, 1972.
 429C PRESENCE OF SOME POROSITY IS DESIRABLE TO COUNTER TENDENCY TO SWELL AS
 429C FISSION PRODUCTS ACCUMULATE IN THE MATRIX.
 429C DESCRIPTION OF 1) FUEL BEHAVIOR IN DIFFERENT REACTORS, 2) DENSIFICATION,
 429C 3) ASSOCIATED PHENOMENA AND 4) STAFF CONCLUSIONS
 429C ANNIHILATION OF SMALL PORES HAS OCCURRED AS A RESULT OF LOW
 429C TEMP. NEUTRON IRRADIATION FOR RELATIVELY SHORT PERIODS OF TIME.
 429C PORE MORPHOLOGY COULD HAVE A PROFOUND EFFECT ON IRRADIATION
 429C INDUCED DENSIFICATION
 429C TWO PROPOSED MECHANISMS - 1) RESOLUTION-FISSION FRAGMENTS

429C AID IN TRAPPING LATTICE VACANCIES WHICH MIGRATE AWAY FROM
429C AND VACANCIES; THE ATOMS HAVE A HIGHER MOBILITY, MIGRATE TO
429C SINKS, LEAVING AN EXCESS OF VACANCIES AND RESULTING
429C IN AN INCREASE IN THE FUEL ATOM DIFFUSION RATES
429C SINTERING IS DRIVEN BY PORE SURFACE ENERGY RATE IS
429C ENHANCED BY FISSION GENERATED EXCESS VACANCIES
429C URANIUM MOVES BY A VOLUME DIFFUSION MECHANISM.
429C VACANCY SOURCE IS PROVIDED BY FISSION SPIKES.
429C VACANCY SINK IS PROVIDED BY GRAIN BOUNDARIES, FREE
429C SURFACES AND DISLOCATIONS.
429C STAFF ESTIMATE OF IN-REACTOR URANIUM DIFFUSION
429C COEFFICIENT IS $2.5 \times 10^{-17} \text{ CM}^2/\text{SEC}$ AT A FISSION RATE
429C OF $1.8 \times 10^{12} \text{ F/CC-SEC}$.
429C THE IN-REACTOR DIFFUSION RATES ARE EQUIVALENT TO THERMALLY
429C ACTIVATED DIFFUSION RATES AT 1400C
429C IT IS BELIEVED THAT DIFFERENCES IN VENDOR PELLET MANUFACTURING
429C PROCESSES ARE NOT SUFFICIENT TO ALLOW DIFFERENTIATING
429C BETWEEN FUELS FOR THE PURPOSES OF SAFETY ANALYSIS
429C TREND FOR INCREASED FUEL COLUMN SHRINKAGE WITH DECREASED
429C INITIAL SINTERED DENSITY.
429C RELATIVE AXIAL SHRINKAGE IS GREATER THAN RELATIVE RADIAL SHRINKAGE.
429C NO RELATIONSHIP BETWEEN FUEL COLUMN LENGTH CHANGE AND
429C FUEL ROD PREPRESSURIZATION
429C DENSIFICATION EFFECTS 1) DECREASE IN PELLET LENGTH CAUSES
429C LINEAR HEAT GENERATION RATE TO INCREASE PROPORTIONALLY
429C 2) AXIAL GAPS RESULT IN INCREASED LOCAL NEUTRON FLUX
429C AND GENERATION OF LOCAL POWER SPIKE, 3) INCREASE
429C IN PELLET-CLADDING CLEARANCE GAP CAUSES DECREASE IN
429C THERMAL CONDUCTANCE AND DEGRADATION OF HEAT TRANSFER
429C CAPABILITIES.
429C TOTAL EFFECT CAN CAUSE FUEL ROD TO CONTAIN MORE
429C STORED ENERGY, INCREASE LINEAR HEAT GENERATION RATE,
429C DECREASE HEAT TRANSFER CAPABILITY AND CREATE
429C POTENTIAL FOR LOCAL POWER SPIKE.
429Z KRM STB

600K 3GADOLINIA THERMEXP RESTRUCTUR
600A WADA,T.
600A NORO,K.
600A TUBUKUI,K.
600T BEHAVIOUR OF UO₂-GD₂O₃ FUEL.
600R PROC.INTERNAL CONF.ON NUCLEAR FUEL PERFORMANCE,B8NE,NO.63,OCT1973.
600C SAMPLES PRESSED AT 3-5000 KG/CM², SINTERED AT 1600C FOR 2 HRS;
600C TD=.94 TO .96. INCOMPLETE FORMATION OF SOLID SOLUTION AND
600C SPECIMENS MECHANICALLY BLENDED. NO EFFECT OF GADOLINIA ON THERMAL
600C EXPANSION. MIGRATION OF GADOLINIA NOT OBSERVED. GADOLINIA
600C INHIBITED FORMATION OF COLUMNAR GRAINS. NO DETAILS OF EXPERIMENTS
600C GIVEN FOR SOURCES OF SITE REPORTS.
600Z KRM
601K 2GADOLINIA MICROSTR
601A LITTLECHILD,J.E.
601A BUTLER,G.G.
601A LESTER,G.W.

601T THE PRODUCTION OF BURNABLE POISON OXIDE FUEL.
601R PROC. INT. CONF. ON NUCLEAR FUEL PERFORMANCE, 88NE, NO. 65, OCT 1973.
601C GADOLINIA HAS SOLUBILITY TO 40WT PERCENT GD2O3. GD GOES INTO
601C U4+ LATTICE SITE AND CONVERTS U4+ TO U6+. GD2O3 TENDS TO REDUCE
601C SINTERABILITY AND GRAIN SIZE AS WELL AS O/M RATIO.
601Z KRM
602K 1 FAILURE
602A GELHAUS, F.E.
602T THE CLADDING FLAW GROWTH INDEX- A METHOD FOR COREWIDE FUEL ROD
602T RELIABILITY PREDICTION.
602R EPRI NP-324-SR, FEB 1977.
602C AN ACCUMULATIVE INDEX BASED ON EXPOSURE AND POWER INCREASE EVENTS
602C IS PROPOSED TO ASSESS FAILURE PROBABILITY. CALCULATION FOR LOCAL
602C ACCUMULATION IS RELATED TO NODAL NEUTRONICS-HYDRAULIC CORE
602C SIMULATION CALCULATIONS. CLADDING FLAW GROWTH INDEX = CFGI
602C REQUIRES THRESHOLD BE SURPASSED (OXIDE FILM BE REACHED) THEN
602C ENVIRONMENT FOR FLAW EXTENSION (SCC CONDITIONS) OCCUR. GENERAL
602C CONCEPT PRESENTED.
602Z KRM
603K 2 FAILURE PROFILE
603A BORRESEN, S.
603A POMEROY, D.L.
603A ROLSTAD, E.
603A SAUAR, T.O.
603T NUCLEAR FUEL PERFORMANCE EVALUATION.
603R EPRI NP-409, JUN 1977.
603C REPORT DESCRIBES EPRI PROJECT RP509 TO DEVELOP COMPUTERIZED LOGIC
603C SYSTEM FOR PREDICTING FUEL RELIABILITY. POSHO MODEL USEFUL BUT
603C REQUIRES FURTHER IMPROVEMENT. SYSTEM PREDICTIONS CHECKED AGAINST
603C FAILURE MEASUREMENTS IN QUAD CITIES UNIT TWO -BWR- AND MAINE
603C YANKEE -PWR. FAILURE DATA AND OPERATIONAL DATA OF POWER SHOCKS
603C OBTAINED IN DETAIL FOR THESE REACTORS. STATISTICS PREDICTIONS
603C REASONABLE EXCEPT AN ORDER OF MAGNITUDE TO LOW FOR MAINE YANKEE
603C TYPE B FUEL. POSHO MODEL BASED ON STRESSES PRODUCED DURING POWER
603C CHANGES WITHOUT CONSIDERING DETAILS OF SCC. MEASURE OF SHOCK IS
603C GENERATION RATE MINUS THRESHOLD LEVEL (G-Q0) WHERE THRESHOLD IS
603C HISTORY DEPENDENT. RATE OF POWER RISE AFFECTS G-Q0 SINCE Q0 CAN
603C INCREASE DURING RAMP. BUNDLE DIVIDED INTO ROD GROUPS AND AXIAL
603C POWER DISTRIBUTIONS ACCOUNTED FOR BY CONSIDERING SEGMENTS OF THE
603C BUNDLES. AN ACTIVITY RELEASE ALGORITHM WAS ALSO DEVELOPED.
603C POWER LEVEL AND PROBABILITY OF FAILURE USED IN ALGORITHM.
603P DETAILED DISCUSSION OF CONDITIONS AT QUAD CITIES UNIT TWO GIVEN
603P THAT LEAD TO 94 FAILURES.
603C PREDICTIONS DID NOT SEEM TO ACCOUNT FOR ACCUMULATIVE EFFECTS.
603C USE OF PREDICTIONS FOR LOCATIONS FOR SIPPING SEEMS JUSTIFIED.
603C -LOCATION OF SAFE ASSEMBLIES WITH LESS THAN 0.05 FAILURE PROB.-
603P DETAILED DISCUSSION IS GIVEN OF MAINE YANKEE. 41 FAILURES OF
603P TYPE B FUEL WERE DETECTED FOR CORE 1 AND 7 FOR CORE 1A.
603P 1 AND 3 FOR TYPE A -- 1 AND 2 FOR TYPE C FOR CORES 1 AND 1A.
603P FUEL FABRICATION DETAILS BETWEEN TYPES NOT GIVEN.
603C REPORT GIVES GOOD IDEA OF DETAILS REQUIRED FOR LWR FUEL FAILURE
603C ANALYSIS BASED ON PCI.
603Z KRM

604K 2FAILURE LUBRICATED
604A HARDY,D.G.
604A BAIN,A.S.
604A MEADOWCROFT,R.R.
604T PERFORMANCE OF CANDU DEVELOPMENT FUEL IN THE NRU REACTOR LOOPS.
604R ANS WATER REACTOR FUEL PERFORMANCE MEETING,MAY1944,ST.CHARLES,ILL.
604C CANDU FUEL FAILURE PROBABILITY IS RELATED TO MAXIMUM POWER, POWER
604C INCREASE, FUEL BURNUP AND DWELL TIME AT POWER. THRESHOLD CRITERIA
604C ARE GIVEN FOR NON-CANLUB FUEL. DATA ON CANLUB FUEL IS SHOWN TO
604C SUPPORT THE IMPROVEMENTS OBTAINED WITH THE CANLUB FUEL. THE
604C GRAPHITE COATING ACTS AS LUBRICANT, ABSORBER AND BARRIER TO
604C IMPROVE PERFORMANCE. TREATMENT OF DATA INDICATES CONCEPTS OF
604C FUELOGRAM CONCEPTS.
604Z KRM
605K 1FAILURE
605A PANKASKIE,P.J.
605A HEASLER,P.G.
605A MOHR,C.L.
605T PROFIT--A PRELIMINARY MODEL FOR CALCULATING THE PROBABILITY OF
605T FUEL FAILURE IN A TRANSIENT.
605R BNW,PERSONNEL COMMUNICATION,NOV1978.
605C WORK IS A REFITTING OF FUELOGRAM APPROACH FOR FAILURE PROBABILITY
605C BASED ON POWER INCREMENT, INITIAL POWER AND BURNUP. LOW STRAIN
605C ENERGY ABSORPTION GIVEN AS PHYSICAL CAUSE FOR FAILURES AND A
605C TIME TO FAILURE FUNCTION PROPOSED. CORRELATION HAS NO PARAMETERS
605C FOR DESIGN OF FUEL NOR AN ACCUMULATIVE DAMAGE CRITERION.
605Z KRM
606K 3B4CDISPER SWELLING GASRELEASE
606A HEAVER,S.C.
606T IRRADIATION BEHAVIOR OF AL2O3-B4C FOR USE IN LIGHT WATER REACTORS.
606R USNI-73-34,OCT1973. (U.S.NUCLEAR,INC.)
606C KAPL IRRADIATED SPECIMENS OF .65 TO .85 TD SHOWED LITTLE SWELLING
606C AND 0.01 FRACTIONAL HELIUM RELEASE. BMI WITH GREATER THAN 0.9 TO
606C SHOWED 0.03 DELV/V. AL2O3 IRRADIATED TO 10**22 HAD 0.01 DELV/V
606C WITH SMALL GRAIN SIZE AND MORE SWELLING WITH LARGER GRAIN SIZE.
606C REPORT CONTAINS SUMMARY DATA AND REFERENCE SOURCE FOR SEVERAL
606C IRRADIATION PROGRAMS OF BOTH B4C-AL2O3 AND ALUMINA. IN 150-600C
606C AND LARGE GRAIN SIZE, THE SWELLING OF ALUMINA THE GREATEST.
606Z KRM
607K 3B4CDISPER SWELLING GASRELEASE
607A BURIAN,R.J.
607A FROMM,E.O.
607A GATES,J.E.
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607C HELIUM RELEASE AND 0.013 TO 0.024 DELV/V FOR B4C-AL2O3 PELLETS.
607C PELLET CRACKING AND CHIPPING DURING UNLOADING OBSERVED.
607C METALOGRAPHY SHOWED FINE CRACKING MAIN MICROSTRUCTURAL CHANGE.

607C CRACKED PELLETS WERE SEVERELY CORRUDED IN CORROSION TESTING.
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607C SHOWED 1000F REQUIRED BEFORE SIGNIFICANT HELIUM RELEASE.
607Z KRM
608K 3B4C DISPER SWELLING GAS RELEASE
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609C FLUENCES .GT. $2.8E+21$ (N/CM**2). VOLUME INCREASES PLOTTED VS.
609C FAST-FLUENCE (.GT. 0.1MEV) WITH IRRADIATION TEMPERATURE GIVEN.
609C FRACTURING FUNCTION OF IMPURITY CONTENT.
609Z KRM
610K 3ALUMINA THERMOCONDUCTIVITY SPECIFIC ELASTIC MOD
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611A WILKS, R.B.
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611Z KRM
612K 1PRESSURIZE
612A BAILEY, W.E.
612A ARMIJO, J.B.
612A JACOBSON, J.
612A PROBSTLE, R.A.
612T BWR FUEL PERFORMANCE
612R ANS TOPICAL MEETING ON LWR FUEL PERFORMANCE, PORTLAND, AP 30-MY3, 1979.
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629A BURCHILL, W.E.
629A HOLDERNESS, J.H.
629A JAGELER, E.F.
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629A MASON, J.H.
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629C AND ABILITY OF FRAP-T3 TO MEET REQUIREMENTS. THIS INPUT INTO TASK
629C TO DEVELOP EPRI TRANSIENT CODE. NRC GUIDES ARE REGULATION REVIEW
629C GIVES TABLES OF ACCIDENTS AND REQUIREMENTS FOR EVALUATION. FOR
629C MODERATE FREQUENCY EVENTS BOTH CORE AVERAGE AND LIMITING CHANNEL
629C THERMAL, MECHANICAL, CHEMICAL AND ROD PERFORATION SHOULD BE
629C EVALUATED. INFREQUENT AND LOCA OUTPUT REQUIREMENTS SPECIFIED.
629C ACCIDENT CATEGORIES ARE REVIEWED BY TYPE SPECIFYING CODES USED BY
629C VENDORS AND REQUIRED EVALUATIONS WITH JUDGEMENTS OF WHICH REQUIRE
629C FUEL PERFORMANCE EVALUATIONS SINCE REPORT IS FOCUSED ON SPECIFYING
629C REQUIREMENTS FOR SUCH A CODE. ABILITY OF FRAP-T3 TO MEET THESE
629C REQUIREMENTS IS REVIEWED.
629Z KRM
630K 2TRANSIENT LICENSING
630A CHARYULU, M.K.
630A CROUCHER, D.W.
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630T WITH AND WITHOUT SCRAM.
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630C SHORT PERIODS AS MOST SEVERE CONDITIONS. CLADDING COLLAPSE AND
630C STRESS CORROSION CRACKING ARE MOST PROBABLE DAMAGE MECHANISM.
630C FOUR TESTS PLANNED TO INVESTIGATE PHENOMENA. REVIEW OF ACCIDENT
630C GIVEN. TABLES, POWERS REACHED, DURATIONS AND PLOTS OF EXPECTED
630C CONDITIONS GIVEN FOR BOTH BWR'S AND PWR'S. IN BWR'S A HIPI EVENT
630C FOR AN ATWS (ANTICIPATED TRANSIENT WITHOUT SCRAM) MAY REACH 7.75
630C TIMES OPERATING POWER. FRAP-T4 WAS USED TO ANALYZE MAY OF THESE
630C EVENTS. REPORT INCLUDES RECOMMENDATIONS FOR INSTRUMENTATION AND
630C POST TEST EXAMINATIONS.
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105K 2FILESYS COMPCODE
105T LIST OF DATA VARIABLES FOR THE EPRI FUEL PERFORMANCE DATA BASE.
105R SAI REPORT FOR EPRI PROJECT RP971, 1977.
105C DRAFT REPORT FOR COMMENT AND REVIEW. SIX DATA CATEGORIES--
105C DATA SOURCE IDENTIFICATION
105C PRE-IRRADIATION CHARACTERIZATION OF FUEL ROD
105C PRE-IRRADIATION CHARACTERIZATION OF ASSEMBLY

105C IRRADIATION HISTORY
 105C END-OF-CYCLE NDE
 105C END-OF-LIFE DESTRUCTIVE EXAMINATION
 105C EXTENSIVE LIST OF DATA WITH PRIORITIES AND AVAILABILITY ARE TO BE
 105C ASSIGNED. LISTS ARE EXTENSIVE AND FORM OF DATA NOT DESCRIBED SO
 105C SYSTEM SEEMS FAR FROM BEING WORKABLE.
 25Z
 26K 2ZIRCALOY SCC
 26A JONES,R.L.
 26A SMITH,E.
 26A MILLER,A.K.
 26T OBSERVATIONS ON DAMAGE ACCUMULATION DURING IODINE-INDUCED STRESS
 26T CORROSION CRACKING OF ZIRCALOY CLADDING.
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 26C LINEAR DAMAGE ACCUMULATION BASED ON FRACTIONAL TIME UNDER CHANGING
 26C CONDITIONS WAS PROPOSED. TESTS UNDER CHANGING CONDITIONS WERE RUN
 26C TEST DATA REPORTED.
 108Z
 109K 2FISPROD DIFCOEF
 109X 1 110
 109A RITZMAN,R.L.
 109T CESIUM AND TELLURIUM RELEASE FROM UO2.
 109R SUMMARY OF ANS-5.4 MEETING, NOV.8,1978.
 109C LITERATURE DATA DEVELOPED FOR RELATIVE CESIUM/XENON AND
 109C TELLURIUM/XENON DIFFUSIVITY RATIOS TO GET D1 FOR THE ANS-5.4
 109C DIFFUSION RELEASE MODEL. THE RECOMMENDED DIFFUSION COEFFICIENTS ARE
 109C $D_{IX} = .00212 * \exp(-49700./RT)$
 109C $D_{II} = .00126 * \exp(-40800./RT)$
 109C $D_{IC} = .000168 * \exp(-37600./RT)$
 109C $D_{IT} = 2.43 * \exp(-62200./RT)$
 109C $T = (K), R = ((CAL/MOL)/(K)), D1=D0/A**2*(SEC**=-1)$
 109C WIDE SCATTER OF DATA WAS NOTED.
 110Z
 111K 1FISPROD
 111A LORENZ,R.A.
 111A COLLINS,J.L.
 111A MANNING,S.R.
 111T FISSION PRODUCT RELEASE FROM SIMULATED LWR FUEL.
 111R NUREG/CR-0274,ORNL/NUREG/TM-154,OCT1978
 111C SIMULATED CONDITIONS FOR SFTA -SPENT FUEL TRANSPORTATION ACCIDENT-
 111C AND LOCA CONDITIONS. IMPLANTED COATINGS ON FUEL SURFACE USED FOR
 111C SIMULATIONS TO REPRESENT MATERIAL RELEASED TO PELLET-CLADDING GAP
 111C REGION. TESTS ARE TO PROVIDE DATA TO FISSION PRODUCT TRANSPORT
 111C ANALYSIS PROGRAM AT BCL. FOR NON MELT CONDITIONS, PRODUCTS IN THE
 111C GAP ARE THE SIGNIFICANT CONTRIBUTORS. TABLES OF RELEASES GIVEN
 111C FOR GAP MATERIAL WITH TELLURIUM BEING A SMALL FRACTION OF A
 111C PERCENT AT 700C AND BELOW. IODINE AND CESIUM WERE SIGNIFICANT.
 111C TESTS WERE IN AIR AND STEAM AND FOR LESS THAN A DAY IN DURATION
 111C TEMPERATURES BETWEEN 500 TO 1300C. PRODUCTS MIGRATE WITHIN THE
 111C FUEL RODS AND RELOCATES INITIAL DISTRIBUTIONS. NO IDEAS GIVEN ON
 111C HOW TO MODEL THE PHENOMENA.
 111Z
 112K 2ZIRCALOY SCC

112A PEELS, M.
112A STEHLE, M.
112A STEINBERG, E.
112T OUT-OF-PILE TESTING OF IODINE STRESS CORROSION CRACKING IN
112T ZIRCALOY TUBING IN RELATION TO THE PCI-PHENOMENON.
112R FOURTH INTERNATIONAL CONFERENCE ON ZIRCONIUM IN THE NUCLEAR
112R INDUSTRY, 1978.
112C IF IODINE ARE THE TEST SPECIMENS. RESULTS ARE REPORTED AS
112C RELATIVE UNIFORM ELONGATION WITH STRESS, IODINE CONCENTRATION,
112C AND STRAIN RATE AS INDEPENDENT PARAMETERS. TIME OF FAILURE AND
112C UNIFORM ELONGATION WERE ASSUMED TO BE PROPORTIONAL. AT 400C,
112C $1E-6$ (GM/CM**2) WAS IODINE THRESHOLD LEVEL. FOR IODINE
112C CONCENTRATIONS ABOVE THRESHOLD, MINIMUM DUCTILITY WAS AT A STRAIN
112C RATE OF .002 (1/HR). MAXIMUM CRACK DENSITY OCCURS WITH BASAL
112C POLES AT 50 TO 70 DEGREES TO THE RADIAL DIRECTION. SPLIT RING
112C TESTS USED TO DETERMINE CRACK NUCLEATION AND CRACK GROWTH TIMES.
112C THUS METALLURGICAL CONDITION, STRAIN RATE, AND IODINE LEVEL MUST
112C ALL BE CONSIDERED IN DETERMINING TIME TO FAILURE.
117Z
118K 3ZIRCALOY SCC FAILURE
118A MILLER, A.K.
118A CHALLENGER, K.D.
118A SMITH, E.
118A RANJAN, G.V.
118A CIPOLLA, R.C.
118T ZIRCALOY CLADDING DEFORMATION AND FRACTURE ANALYSIS.
118R EPRI NP-856, AUG 1978.
118C IODINE CONCENTRATION, TEXTURE, TIME AND IRRADIATION EMBRITTLEMENT
118C ARE THE SECONDARY FACTORS AFFECTING THE STRESS AND TEMPERATURE
118C DEPENDENT STRESS CORROSION CRACKING PROCESS. STRESS INTENSITY
118C K_{ISCC} IS LOW (3 KSI*(IN)**.5), THIS INITIATION IS CRITICAL STEP.
118C IODINE CONCENTRATION IS ASSUMED PROPORTIONAL TO TOTAL ROD BURNUP.
118C FISSION GASES RELEASED AND INVERSELY PROPORTIONAL TO AREA OF
118C CLADDING EXPOSED TO IODINE. THIS AREA IS AROUND ENDS OF PELLETS=
118C $12(GM/CM**2) = 2.E+6*((FRACTIONAL\ BURNUP)*(FRACTION\ FG\ RELEASE))$
118C DAMAGE FUNCTION APPROACH ASSUMED WITH LINEAR ACCUMULATION OF
118C DAMAGE. FLUENCE SHOULD BE .GT. $3.E+20$ (N/CM**2).
118C FAILURE REQUIRES AT LEAST THE THRESHOLD STRESS FOR TRANSGRANULAR
118C CRACK FORMATION TO BE REACHED. FAILURE INVOLVES CLEAVAGE OF GRAINS
118C FAVORABLE ORIENTED AND PLASTIC FLUTING OF OUTERS. THRESHOLD STRESS
118C COULD BE AS LOW AS .3 TO .5 OF IRRADIATED YIELD STRENGTH. TEXTURE
118C IMPORTANT BECAUSE IT AFFECTS NUMBER OF GRAINS ORIENTED FOR THE
118C CLEAVAGE FAILURES. THRESHOLD STRESS IS REQUIRED TO INITIATE A
118C TRANSGRANULAR CRACK FROM A FLAW. CIRCUMFERENTIAL STRESS IS ASSUMED
118C TO BE DOMINANT STRESS. A CRACK PROPAGATION MODEL IS DEVELOPED FOR
118C THE CLEAVAGE-PLASTIC DEFORMING MATERIAL. THEORETICAL AND TEST
118C RESULTS INDICATE THAT IRRADIATED ZIRCALOY MUST BE USED TO EVALUATE
118C BEHAVIOR BECAUSE OF IRRADIATION HARDENING AFFECTING THE PLASTIC
118C DEFORMATION PROCESS (DISLOCATION CHANNELING). EXPERIMENTAL RESULTS
118C REFERENCED (YAGEE, ETC. EPRI RP1027) THAT YIELDED A THRESHOLD STRESS
118C OF 200 (MN/M**2) OR 30 (KSI). TABLE OF TEST RESULTS GIVEN ON
118C P 3-44, 3-45. THE YAGEE RESULTS ARE THOSE FROM ANL.
118C MODELING OF CRACKED PELLETS IS DONE TO DEVELOP STRAIN

118C CONCENTRATIONS. FOR THE HARD PELLET MODEL, SIX CRACKS AND TYPICAL
118C CLADDING THICKNESS. STRESS C.F. #1.2.
118Z
119K 2ZIRCALOY SCC
119A HOLUB, F.
119A KOHLI, R.
119T THE EFFECT OF SIMULATED FISSION PRODUCTS ON THE MECHANICAL
119T PROPERTIES OF ZIRCALOY-2.
119R EUROPEAN NUCLEAR CONFERENCE, FORATOM VII, HAMBURG, MAY 1979.
119C ZIRCALOY TUBES EXPOSED TO MU, TE, SE, I, BR, CO, IN, SB. TESTS AT 350,
119C 500, 700C AFTER 50, 250, 500 AND 1000 HR ANNEALS. ONLY 700C TESTS
119C SHOWED SIGNIFICANT DIFFERENCES FROM REFERENCE ZR-2. MAJOR DUCTILITY
119C DROPS (ORDER OF SIGNIFICANCE) AT 700C WERE FROM MO (.002),
119C BR (.07), I (.13), TE (.16), CO (.17).
119Z KRM
631K 2LICENSING TRANSIENT
631A MEHNER, A.S.
631A HOBBS, R.R.
631A SEIFFERT, S.L.
631A MACDONALD, P.E.
631A MCCARDELL, R.K.
631T PERFORMANCE OF UNIRRADIATED AND IRRADIATED FUEL RODS TESTED UNDER
631T FILM BOILING CONDITIONS.
631R ANS TOPICAL MEETING ON LWR FUEL PERFORMANCE, PORTLAND,
631R APR 30-MAY 3, 1979.
631C REVIEW OF TESTS OF POWER-COOLING-MISMATCH IN PBF. TABLE OF TESTS
631C AND A FEW QUOTED RESULTS.
631Z KRM
632K 6ANNULAR DOPING URANIA GASRELEASE CREEP THERMOCOND
632X 1 616
632A RAVEN, L.F.A.
632T GAS RETENTIVE ANNULAR FUEL PELLETS PALLIATIVE OR PANACEA FOR THE
632T L.W.R.
632R A.C.S. MEETING, CINCINNATI, MAY 1979.
632C DESIGN STARTED AS AN EXTENSION OF THE AGR PROGRAM WITH CONTROLLED
632C POROSITY FUEL DEVELOPED AND IRRADIATED IN THE SGMWRIS. TO GO TO
632C HIGH BURNUPS A LARGE GRAIN SIZE FUEL WAS COMBINED WITH ANNULAR AND
632C CONTROLLED POROSITY. NB2O5/NIOBIA DOPED FUEL INCREASED GRAIN SIZE
632C AND CREEP RATE. EPRI SUPPORTED TEST WITH 200 RODS TO BEGIN IN
632C 1980. MEASURED FISSION GAS RELEASE AT 8.5 AND 28.3 MWD/KGD WERE
632C LESS THAN HALF FOR DOPED COMPARED TO NONDOPED RODS.
632Z KRM
633K 4IRRADIATED URANIA PELLET CORROSION
633A GARZAROLLI, F.
633A MANZEL, R.
633A STEHLE, H.
633T KWU FUEL PERFORMANCE WITH EMPHASIS ON HIGH BURNUP.
633R ANS TOPICAL MEETING ON LWR FUEL PERFORMANCE, PORTLAND, APR 30-MAY 3, 1979.
633C SOME EXPOSURES OF 45 MWD/KGD REACHED AND ACCELERATED CORROSION
633C OBSERVED. NO SEVERE PROBLEMS NOTED. KWU FUEL REPORTED ON.
633Z KRM
634K 4IRRADIATED URANIA PELLET COMPCODE
634A BALFOUR, M.G.

634A DEQUIDY, E.
634A HOPKINS, G. W.
634A ROBERTS, E.
634A SMALLEY, W. R.
633T FUEL MODELING AND PERFORMANCE OF HIGH BURNUP FUEL RODS
633R ANS TOPICAL MEETING WATER REACTOR FUEL PERFORMANCE, ST. CHARLES, ILL.
633R MAY 9-11, 1977, PP 133-146.
634C DESCRIPTION OF W.E. PAD CODE AND DATA BASE FOR CODE VERIFICATION.
634C ZORITA IRRADIATION TO 65 MWD/KGU AND SAXTON -U, PU02, TO 51 MWD/KGU
634C DESCRIBED. BEHAVIOR OF RODS PREDICTED AND SATISFACTORY.
634Z KRM
635K 1ROOBOW
635A CAYE, T. E.
635A GUTHRIDGE, J. R.
635A REAVIS, J. R.
635A ROD BOW OBSERVATIONS.
635R WCAP-5172, OCT 1973, PRESENTATION CREEP-RELATED, FUEL-CLAD MECHANICAL
635R INTERACTION MEETING, BETHESDA, MD., PRELIMINARY INFORMATION.
635C REVIEW OF WESTINGHOUSE EXPERIENCE OF ROD BOW IN PWR'S. TWO TYPES
635C OF BOW PROBLEMS REPORTED. ONE WITH INSUFFICIENT AXIAL SPACE AND
635C THE OTHER A GENERIC PROBLEM.
635Z KRM
636K 1ROOBOW
636A CAYE, T. E.
636A CERNI, S.
636A REAVIS, J. R.
636T RECENT ROD BOW OBSERVATIONS.
636R PRESENTATION ERDA FUEL PERFORMANCE SEMINAR, JAN 1976, GERMANTOWN, MD.
636C WESTINGHOUSE EXPERIENCE AND BOW CORRELATIONS FOR 15x15 BUNDLES
636C PRESENTED.
636Z KRM
637K 1GADOLINIA
637A POTTS, G. A.
637T URANIA GADOLINIA NUCLEAR FUEL PHYSICAL AND IRRADIATION
637T CHARACTERISTICS AND MATERIAL PROPERTIES.
637R NEDO-20943, JAN 1977.
637C THIS WAS AN NON-PROPRIETARY VERSION OF G.E. REPORT WITH MOST OF
637C ACTUAL DATA MISSING BUT IT DID INDICATE GENERAL EFFECTS OF
637C GADOLINIA ADDITIONS.
637Z KRM
638K 2FAILURE FISPROD
638A AXTELL, C. E.
638A HOLLOWAY, J. H.
638A OSBORNE, R. N.
638T POWER REACTOR HIGH PERFORMANCE UO₂ PROGRAM ANALYSIS OF FISSION
638T PRODUCT RELEASE RELATED TO CENTERMELT FUEL FAILURES.
638R GEAP-10039, JUN 1969.
638C TWO RODS OPERATED AT 27KW/FT FAILED GAVE OFF 23,000 MICRO-CI/SEC
638C DURING POWER ADJUSTMENTS. RELEASE MEASUREMENTS AT AIR EJECTOR OF
638C BIG ROCK POINT. CHEMISTRY OF OFFGAS DETERMINED. SHOWED FUEL HAD
638C LESS THAN 5 MWD/KGU. REPORT HAS ONLY CHEMICAL DATA AND DOES NOT
638C CHARACTERIZE FAILURES.
638Z KRM

639K 3 VIPAC PELLET PROFIL
639A HOFFMAN, J.P.
639A LYONS, M.F.
639A WEIDENBAUM, B.
639T DIMENSIONAL CHANGES IN HIGH PERFORMANCE PELLET AND POWDER FUEL
639T RODS - PROFILOMETRY STUDIES.
639R GEAP-5100-8, 1966.
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640K 2VIPAC PELLET
640A BAGGER, C.
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640T PERFORMANCE EVALUATION OF UO₂-ZR FUEL IN POWER RAMP TESTS.
640R INT. CONF. ON NUCLEAR POWER AND ITS FUEL CYCLE, SALZBERG, AUSTRIA, 1977.
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