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海外出張報告

ANS 1997年 軽水炉燃料性能に関する国際トピカルミーティング

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海外出張報告
ANS 1997年 軽水炉燃料性能に関する国際トピカルミーティング

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要　旨

米国オレゴン州ポートランドで開催された米国原子力学会（ANS）が主催する軽水炉燃料性能に関する国際トピカルミーティングへ1997年3月1日から8日までの８日間出張した。出張の目的は、動燃で得られたMOX燃料関連の研究成果を発表するとともに、動燃におけるMOX燃料の照射挙動評価技術の高度化に資するため同会議で発表される他の研究機関のMOX燃料に関する研究開発情報を収集することにあり、ほぼ当初の目的を達成できた。

本会議は、米国、フランス、日本等から延べ約270名（うち日本からは28名）の参加者のもと、軽水炉燃料の性能、特に高燃焼度化に主眼をおいて開催された。セッションの構成は、３つの基調講演、MOX燃料を含む７つの口頭発表セッションと２つのポスターセッションからなり、全部で91件（うち日本からは20件）の論文発表と活発な討議が行われた。

動燃からは、以下の３件を発表した。

1) 「ふげん」で集合体平均33.1GWd/tまで定常照射した36本タイプのMOX燃料についての照射および照射後試験結果を解析・評価し、"Behavior of MOX Fuel Irradiated in a Thermal Reactor"と題して口頭発表した。
2) 平成６～７年度にハルデン炉で実施した第１～４回ATR実証炉燃料の出力急昇試験結果を解析・評価し、"Power Ramp Tests of MOX Fuel Rods for ATR (IFA-591) "と題して口頭発表した。
3) 「ふげん」高燃焼度用54本タイプのMOX燃料の設計およびDuplex型MOX燃料の開発成果をまとめ、"Development of High Burn-Up MOX Fuel for ATR"と題してポスター発表した。

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目 次

1．目的

米国原子力学会（ANS）が主催する1997年 軽水炉燃料性能に関する国際トピカルミーティングへ出席し、動燃からはMOX燃料の設計および照射結果に係る２件の口頭発表と１件のポスター発表を行う。また、同会議で発表された他の研究機関のMOX燃料に関する研究開発情報を収集し、動燃における今後の研究開発に資する。

2．出張先

1997 International Topical Meeting on Light Water Reactor Fuel Performance, Portland, Oregon March 2-6, 1997

3．期間

平成9年3月1日～8日（8日間）

4．出張内容

4.1 会議の概要

米国原子力学会（ANS）が主催した本会議は、1997年3月2日から6日までの5日間、米国オレゴン州ポートランドにあるRed Lion Hotel Columbia River で開催された。

本会議が開催された背景には、電力会社が次世紀までに規制緩和による経済的なプレッシャーに直面していることがあげられる。電力会社は、安全運転、より高い設備利用率、劇的に減少した操業・メンテナンスコストのもとで果敢にこの問いに答えようとしている。操業コストの低減にともない、燃料コストの寄与はかつてないほど重要になっている。とはいえ、ここ１０年間、ウラン供給者間の競合とウランインベントリーの過剰により平均燃料コストは25%以上低下し、電力会社はこれを享受してきた。しかし、燃料コストが重要となった現在、ウラン価格の高騰に直面することになった。燃料コストを現状に維持するだけでも、電力会社は、材料物性の制限値に迫る燃料設計に頼らなければならなくなる。燃料製造メーカーにとってのチャレンジとは、電力会社、国立研究機関、大学と協力しつつ、先進的な設計の燃料を開発し、市場にもたらすことである。本会議は、燃料ユーザーと開発者が、先進的な情報、コンセプトおよび希望を語り合う場を提供するものである。

本会議は、米国、フランス、日本等から延べ約270名（うち日本からは28名）の参加者のもと、軽水炉燃料の性能、特に高燃焼度化に主眼をおいて開催された。セッションの構成は、３つの基調講演、①燃料性能Ⅰ、②MOX燃料、③燃料性能Ⅱ、④破損後の燃料挙動、⑤被覆管性能、⑥高燃焼度燃料、⑦反応度事故の７つの口頭発表セッションと２つのポスターセッションからなり、全部で91件（うち日本からは20件）の論文発表と活発な討議が行われた。プログラムおよび参加者リストをそれぞれ付録－1、－２に添付した。

4.2 動燃からの発表

動燃からは、以下の３件を発表し、質疑をまとめた。発表した論文およびOHPあるいはポスターを付録－３に添付した。

1) "Behavior of MOX Fuel Irradiated in a Thermal Reactor"

「ふげん」で集合体平均33.1GWd/tまで定常照射した36本タイプのMOX燃料についての照射および照射後試験結果を解析・評価し、口頭発表した。主な質疑は、以下の通り。

Q：仏国のMOX燃料に関する報告（出張者と同一セッションで２件前の報告）ではヘリウムの放出がほとんど見られないのに対し、ATR-MOX燃料では照射によるヘリウムの放出が多い理由は何か。

A：仏国の報告はPWRで照射したMOX燃料に関するものである。PWR燃料とATR燃料の製造仕様をヘリウム放出という観点から比較すると、２つの相違点が認められる。一つは燃料棒の線出力であり、もう一つはヘリウム封入圧である。

前者については、PWR燃料の線出力はATR燃料に比べ低いため、FPガスの放出率がATR（あるいはBWR）燃料に比べて小さい。（燃焼度30 GWd/t程度では、ATR燃料の場合は20％以下の放出率であるが、PWR燃料では数％程度である。）ヘリウムの放出はFPガスに比例していることから、PWR燃料のヘリウム放出はATR燃料に比べて小さくなるものと考えられる。

後者について、PWR燃料では製造時のヘリウム封入圧が高い（20気圧以上）ため、照射初期はペレット内部より外部の方がヘリウム分圧が高い状態となり、このためヘリウムはペレット中に固溶する（中性子や核分裂片の照射によりヘリウム原子がペレット中に打ち込まれる、とする説もある）。一方、ATR燃料では初期ヘリウム圧が３気圧程度と低いため、外部のヘリウムが内部に移行することは極めて少ない。

このように、PWR燃料ではヘリウムがペレットから放出されにくく、むしろ低燃焼度領域ではペレットの外側にあるヘリウム原子がペレット中に移行する。（このことはPWR-UO_2燃料で確認されているとのこと。）従って、PWR燃料では、燃料設計の違いから、ATR燃料のような顕著なヘリウムの放出は生じないものと考えられる。

2) "Power Ramp Tests of MOX Fuel Rods for ATR (IFA-591) "

平成６～７年度にハルデン炉で実施した第１～４回ATR実証炉燃料の出力急昇試験結果を解析・評価し、口頭発表した。主な質疑は、以下の通り。

Q：被覆管とペレットギャップは、300μmとのことであるが、出力急昇後のPCIによる直径変化は調べているか？

A：出力急昇後の直径データは持ち合わせていない。今後のPIEで確認する予定で

A：出力急昇後の直径データは持ち合わせていない。今後のPIEで確認する予定である。

Q：FEMAXI-ATRでの内圧計算例は良くあっている。計測値との差がどの程度か調べているか？

A：調べているが、現在手元にない。

C：オンライン計装データ、特にfast scanデータは、よく取れている。

A：ハルデン炉で計測したもので、動燃としても満足している。

Q：出力急昇時に破損するとするとPCMI破損か？

A：今回の試験では破損に至っていないが、破損するとするとPCMIによるものであると考えている。

Q：FEMAXI-IVが開発されているが、それとの関係はどうなっているのか？

A：今回開発したFEMAXI-ATRは、FEMAXI-IIIをベースにPuの物性値とMOXでのPIEデータを組み込んだもので、FEMAXI-IVは同じくFEMAXI-IIIをベースとしたものであるが別々に開発されたものである。

Q：UO_2に比べてMOXの方が、PCMIは厳しくないのではないか？

A：クリープの観点では、MOXの方がクリープ速度が高く、UO_2に比べてPCMIは厳しくないと考えられる。

3) "Development of High Burn-Up MOX Fuel for ATR"

「ふげん」高燃焼度用54本タイプのMOX燃料の設計およびDuplex型MOX燃料の開発成果をまとめ、ポスター発表した。主な質疑は、以下の通り。

（データベース関連）

Q：FPガスの放出率の計算には、Vitanzaモデルに加速項が加味されているが、図中のどこから効いているのか？

A：FPガス放出が各温度で２段階で上昇しており、高燃焼度側での上昇部分である。

C：FPガスの放出は、よく合っている。1%付近でばらつきが多いものだ。

C：ATRは炉型としては（横型にすれば）CANDUに似ている。MOXの照射データ、特に高燃焼度、あるいはMOX-Gdのデータは、軽水炉だけでなく、CANDUにとっても役に立つはずである。

Q：He放出率の図は、実測値か計算値か？

A：FEMAXI-ATRを用いた計算値である。

Q：He放出は、MAの崩壊によるものが支配的か？

A：そのとおり。核種としては^{242}Cmからものが支配的である。

Q：ATR実証炉はキャンセルされたと聞いたが？

A：設置主体の電源開発は、ATRをキャンセルしてフルMOX炉心のABWRを建設する。しかし、ここで得られたMOXのデータベースは、ATRのみならず軽水炉でのプルトニウム利用においても役立つものと考えている。

Q：FEMAXI-ATRの開発は何に反映されるのか？

A：ATR実証炉の開発は中止になったが、「ふげん」は今後10年間運転を継続することが決まっている。この中で、高燃焼度燃料のR&Dを計画しているが、この燃料を設計し、成立性を確認するにはFEMAXI-ATRの開発が不可欠である。
Q：熱伝導度と温度の関係を示す図は、動燃で測定したものか？
A：文献値である。

（Duplexタイプ燃料について）

Q：Duplexタイプの燃料は現行のBWRでも成立するのか？
A：現在、検討中である。
C：Gdのための専用ラインが要らないので画期的である。
Q：MOX-Gd燃料の照射はいつからか？
A：すでに試験燃料の製造は終わっており、1997年度に輸送し、照射を開始する予定である。
Q：ポスターの標記上、「Gd燃料が要らない」となっているがどういうことか？
A：均質固溶型のMOX-Gdでは、Gdが中性子毒であることからMOXラインとは別にMOX-Gdラインを建設する必要がある。しかし、Duplex型の場合は、中空のMOXを製造すればMOXラインが共有でき、別にMOX-Gdラインを建設する必要がない。Gd棒は、メーカーで製造後、燃料組立時にMOXの中空部に挿入すればよい。
C：DuplexタイプのMOX燃料のアイデアは、IAEAのTechnical Committee Meeting（1996年11月、東京）で発表されたものであり、よいアイデアである。

4.3 MOX燃料関連の発表

MOX燃料関連の発表は全部で７件あり、このうち３件は前節で紹介したように動燃からの報告であった。ここでは、残り４件の概要を紹介する。

(1) Recent Results from the Reactor MOX Fuel Performance in France and Improvement Program (Framatome, EdF, CEA)

Fragema社は1996年末までに約700体のMOX燃料を10機のフランスの900MWeプラントに供給してきた。EdFは、MOX燃料仕様プラントを1996年に10機、1997に15機、今世紀終わりまでに28機まで増やす予定である。このプラント数は、主にMELOX工場（1998年末までに120t/y）の製造能力に依存する。MOX燃料の装荷方法は、最初$UO_2$36体、MOX16体のOut-In-In 3サイクル使用であった。この時のMOX燃焼度は約38GWd/tであった。その後UO_2燃料を28体取替4サイクル使用としたがMOX燃料はそのままであった。1991年から2つのMOX装荷プラントでロードフォローを開始し、1995年からMOX燃料炉心でも制約を受けなくなった。その後、UO_2燃料がAFA2Gとして改良されると同様にMOX燃料にも適用された。さらにMOX燃料として、以下の２点が改良されている。
- 集合体内Pu分布の最適化；集合体内の出力分布を平坦化するために3種類のPu富

化度を使用しているが中央部の領域を増やし、外周部の割合を減らした。(外周領域は12本) この結果、ピーキング係数1.17が1.07になった。

- 燃料棒設計;MOX燃料棒の内圧は制限因子となっているが、この内圧の裕度を増すために燃料棒内空隙体積を増加させた。

この43-45GWd/t設計のAFA2G-MOX燃料は現在使用中である。今後、MOXプラントが増加してくるとUO_2燃料と同様な運用(燃焼度、使用サイクル等)とする必要がある。このために今後商業炉での照射MOX燃料PIEおよび試験炉での照射試験等が必要となる。

1987年以来、300体以上のMOX燃料が3サイクル照射を完了しており平均燃焼度37.5GWd/t、最大40GWd/tを達成している。試験目的により4サイクル照射、44.5 GWd/tを4体が達成している。1993年6月の Dampierre1号炉の第11サイクルでMOX燃料のリークが1体見つけられたがこれは異物破損と考えられる。このリーク燃料は再装荷された。

St.Laurent B1で3サイクル、約43GWd/tまで照射された最初のMOX燃料のPIEが実施された。また、St.Laurent B2で3サイクル、最後のサイクルでロードフォローを受けた燃料も試験も実施された。得られた被覆管外面腐食と寸法変化はウラン燃料と同等であることが確認された。しかしFPガス放出率はUO_2燃料より若干大きいことが分かった。この原因は、主に同様な燃焼度での出力がUO_2燃料より高いことによるものでMOX燃料での不均一さの影響は小さい。またロードフォロー時の挙動はベースロード時と同様であった。1/4装荷を実現するには高燃焼度域でのFPガス放出率のような照射挙動データの蓄積が必要である。

最近、4体のMOX燃料が Graveline4号で4サイクル照射、52GWd/t(Rod)達成した燃料棒のPIEにより出力が低い場合、燃焼に伴うFPガス放出の増加は見られなかった。また、MOXペレットの微細組織に関連してペレット端では、Puスポット部での気孔の集積、金属FPの析出が見られた。中心部に向かってこの気孔が合体した大きな気孔が見られる。中心部では気孔の集積により大きなボイドとその回りの金属FPが見られる。回りのUO_2マトリックスでは粒界にガスバブルの析出が見られ、FPガス放出が推定される。

MOX燃料ではCm242のα崩壊によりUO_2よりHe生成が多い。しかしパンクチャ試験による燃料棒内He量からHeの多くはペレット中に溶解していると推定される。

その他、MELOX工場でのMOX燃料(ADU、MIMAS法)の照射試験も実施している。

St.Laurent B1で、2, 3サイクル照射された燃料でランプ試験が実施され、最大480W/cmまで非破損であった。この耐PCI性能は、MOXペレットではクリープしやすいことによるものと考えられる。このことはランプ後にリッジが平坦化していること、さらに約350W/cmでペレットディッシュの埋まりが見られたことからも推定される。またペレット変形についてのDEFOMOX試験でもMOX燃料では被覆管変形が小さいことが確認された。

MOX燃料についての最初のRIA試験が St.Laurent B2で3サイクル照射された燃料(47GWd/t)で実施され、145cal/g・35msで非破損であった。

MOX燃料棒設計モデルでは、MOXでの物性および照射挙動を必要に応じて考慮している。例えばFPガス放出については、Puの非均一性を考慮し、データの蓄積により良い予測性を得ている。

今後、UO_2燃料と同じ4サイクル使用とするために4サイクル目を高い出力とする新たな照射試験を予定している。また、集合体内Pu富化度分布を最適化したことでPu富化度を5.3%から7.08%（Puトータル）まで引き上げる。

(2) High Burnup MOX Fuel and Fuel Rod Design Improvement (MHI, KEPCO)

PWRプラントでのMOX燃料利用ではPu富化度が約13wt%PuO_2まで、取出最高燃焼度45GWd/t（集合体）を想定しており、このPu添加と燃焼度であれば、多くの実績のあるウラン燃料と同等の特性と性能であると考えられる。ただし、Pu添加によるペレット熱伝導率の低下及びFPガス放出率の増加等の照射挙動への影響があるため、燃料棒設計においてこれらを適切に考慮している。このうち、FPガス放出挙動ではPu添加の影響とともにペレット中のPu均一性の影響を大きく受ける。すなわち、Puスポット部では発熱が大きくなるので出力（温度）、燃焼度ともに高くなり、ここからのFPガス放出が高くなる。現状の燃料棒設計では現時点での達成燃焼度が高い比較的均一性の悪いMOX燃料データまでを考慮しており、最近の製法による均一性のよい実用化燃料では、放出率が低くなると考えられ、これより新製法によるMOX燃料データを取得し、評価することで妥当な設計を取ることができると考えられ、これよりいくつかの照射試験が実施されている。

細径MOX燃料が Halden炉で照射されている。これは、BNFL製のSBR法で製造されたMOX燃料でPuスポットが小さく、均一性のよいペレットである。ペレット径を細くしているのは同じ照射時間、線出力で燃焼度の進行を加速するためである。すでに約45GWd/tまで達成している。ウラン燃料もリファレンスとして照射されており、これらのFINEコードによる解析評価からMOX燃料のガス放出率はウラン燃料に近く、約10%増程度であることが確認された。また、燃料中心温度測定からはMOXペレットに対して同コードが安全側に予測することが確認された。

また、BN社製MIMAS法MOX燃料が BR3炉で約57GWd/tまで照射された。燃料タイプは17×17型燃料でSBR法MOX燃料と比べるとPuスポットの径は大きいがその濃度は低いといった特徴がある。パンクチャ試験では約12%と高かったがこれは照射出力が30kW/mと高いことで説明される。同コードによる解析でもウラン燃料と同等の放出率を仮定して十分な予測性を有することが確認された。

その他、焼きしまり・スエリング挙動については、UO_2燃料と同等であり、Pu添加の影響のないことを確認した。

以上から最近の製造方法によるMOX燃料では、より均一なPu分布によりUO_2燃料の特性に近いことが確認され、さらなるデータの蓄積により、今後MOX燃料棒設計を見直し、より妥当な設計とし、特に内圧に関して裕度を確保できる見通しが得られた。

(3) Advanced Studies of the Behaviour of MOX Fuel (CEA, Framatome, EdF)

燃焼初期の熱的挙動およびFPガス放出挙動を見るためのGRIMOX2実験では、MIMAS法MOXペレットとUO_2ペレットを用いたセグメント燃料をインライン計測（中心温度、FPガス）しながら、燃焼度4.5GWd/tまで照射した。その結果、熱的挙動については、ペレット中心温度はMOXの方がUO_2に比べて6%高いことが分かった。これは主として熱伝導度が低いためである。FPガス放出挙動については、定常照射時は、MOXの方がFPガス放出率が高く、1500℃を超えると急激にガス放出が進むことが分かった。この場合の放出メカニズムとしては拡散が支配的であることが示された。また、出力過渡時は、生成したFPガスが直ちに放出され、定常時とは異なる放出メカニズムが示された。

炉内焼きしまり挙動をみるためのDENSIMOX実験では、ADU粉末とAUC粉末の2種類のMOX燃料を用いた。照射途中で中性子ラジオグラフィによるペレット寸法を測定した。炉内での最大焼きしまりは、使用したUO_2粉末により異なり、燃料の微細構造と関連しているとことが分かった。また、焼きしまり率は、炉外試験のものとほぼ一致した。

PCMI挙動をみるためのDEFORMOX実験では、MOXペレットとUO_2ペレットを用いて炉内での直径変化を測定した。出力ランプ時のペレット外径リッジ高さ変化を見ると、UO_2ペレットでは急激に増加した後、緩やかに低下するのに対して、MOXペレットでは急激な増加直後に、大きく低下している。これは、MOXのクリープによるもので、PCMIを緩和するという観点からは有利である。ただし、ペレット中間部でのリッジはMOXの方が大きくなっている。原因は調査中であるが、PIE結果から水平方向のペレット破砕によるものと推定している。

(4) High Burnup Modelling of UO_2 and MOX Fuel with METEOR/TRANSURANUS Version 1.5 (CEA)

PWR燃料用ロッド用の燃料挙動解析コードMETEOR/TRANSURANUS Version 1.5は、従来のUO_2燃料（燃焼度60GWd/t）に加えて、MOX燃料（45GWd/t）、$UGdO_2$燃料および出力過渡時の挙動を取り扱うことができるようモデルの追加・改良等を行った。

MOX燃料のモデリングに関連しては、MIMAS法で製造したMOX燃料を使用することからプルトニウムスポットがあり、これをスポット半径と濃度で規定してモデル化した。3サイクル照射のMOX燃料ペレットの測定値と比較し、コードが十分な精度を有することを検証している。

FPガス放出モデルについては、従来のモデルに、MOX燃料でのプルトニウムスポットの効果およびリム効果を加え、それぞれ測定値と計算値の比較により、コードが十分な精度を有していることを検証している。

仏国のR&Dにおいては、燃料中心温度計測、ピン内ガス圧測定、炉内外径測定等、炉内計測照射を数多く実施しているようであり、実験に基づく燃料挙動の解明といった点で、着実に成果を挙げているように感じられた。

また、動燃のMOX燃料の挙動を仏国データと比較すると、２つの点で明確な相違が認められた。

① FPガス放出挙動

動燃のMOX燃料は軽水炉のUO_2燃料と有意差のない放出率を示すが、仏国MIMAS燃料は高燃焼度領域（燃料要素平均燃焼度30 GWd/t以上）で放出率が上昇し、UO_2との差が明確になる。

MOX燃料のガス放出率が高燃焼度領域で上昇する主な原因について、仏国側は、MOX燃料は内部転換率がUO_2に比べ高いため照射末期の線出力が高く、このためMOX燃料の燃料温度が高くなり、これによりガス放出率が高くなる、としている。また、これ以外の原因としては、仏国が採用しているMIMAS法によるMOXペレットは開気孔率がUO_2燃料に比べて高いこと（約10倍）、及びMOX燃料の熱伝導度はUO_2に比べ幾分小さいこと、等が挙げられていた。

仏国のガス放出挙動が動燃の結果と異なる原因を検討すると、製造法の違いによるMOXペレットの性質の相違である可能性が高いと推測される。

仏国のMOXペレットはMIMAS法で製造される。同法はPu分布の均一性には優れるものの、幾分焼結性が劣ると推定され、そのため開気孔率が高いものと考えられる。一方、動燃のマイクロ波加熱直接脱硝粉（MH-MOX）は溶液の状態から直接粉末になるため、Pu分布の均一性が高いのみならず、焼結性にも優れている。開気孔率については、直接、MIMASペレットと比較してはいないものの、焼結性から推測するとMIMAS法より低い可能性が高く、従って、このため、動燃のMOX燃料のガス放出はUO_2燃料と有意な差がないものと推測される。今後、この点に関し調査・確認を進めていきたい。

但し、仏国の試験データによると、MOX燃料とUO_2燃料のガス放出率に明確な違いが生ずるのは約40 GWd/tからであるが、残念ながら動燃には40 GWd/tを越えるデータが十分でない。従って、動燃のMOX燃料がより高燃焼度ではUO_2燃料よりも大きなガス放出率を示す可能性があることを否定することはできない。よって、今後、40 GWd/tを越える燃焼度まで照射された高燃焼度燃料の照射後試験を行い、この点を確認する必要がある。

② He放出挙動

今回、出張者は比較的FPガス放出率の高い動燃のMOX燃料から有意量のヘリウムが検出されたこと、また、その放出量はFPガスの放出量に比例することを報告した。一方、仏国CEAは、彼らのPWR-MOX燃料からはヘリウムの放出は認められず、むしろ、照射後は製造時よりもヘリウムの量が減少している、と報告した。

この相違は、先にQ&Aの部分で述べたように、主として燃料の線出力および製造時のヘリウム封入圧に起因しているものと考えられる。即ち、ヘリウムの放出挙動にお

ける動燃のMOX燃料とCEAのMOX燃料との相違は、詰まるところPWR燃料であるかBWR燃料であるかの相違であると言え、このことは他の場で報告されているPWRあるいはBWRタイプMOX燃料の照射挙動とも一致している。

③ 過渡時UO_2燃料挙動

UO_2燃料の反応度投入型事象を模擬した最近のパルス照射試験において、燃料の高燃焼度化に伴う破損エネルギーの低下が認められてきている。また、同様の破損エネルギー低下がMOX燃料を用いたCABRI試験でも生じたとの報告があり、注目された。現在、動燃では日本原子力研究所との共同研究として、NSRRを用いたATR-MOX燃料のパルス照射試験を実施中である。この燃料は燃焼度がおよそ20 GWd/tと比較的低いものの、MOX燃料の破損エネルギーデータは極めて少ないため、貴重なデータとして期待される。

また、「ふげん」サイトにはより高い燃焼度のMOX燃料（燃焼度40 GWd/t以上）が照射を完了して保管されていることから、原研との共同研究を拡大しこの燃料を用いたパルス照射試験を行うことががができれば、水炉用MOX燃料の過渡時安全性研究により有益な試験データが供給できると考えられ、検討する価値があると思われる。

④ 高燃焼度UO_2燃料挙動

近年、高燃焼度UO_2燃料の特徴的な組織変化としていわゆる「リム組織」の発生が多くの研究者によって解析、報告されている。今回の会議では、リム組織が生じることによる問題点がより具体的に報告されていた。例えば、スエリングの増大、熱伝導度の低下、反応度投入事象下での破損限界エネルギーの低下、等について実験結果を基にリム組織の寄与について検討がなされていた。

また、このようなリム組織の研究においては、リム組織が極めて微細であることから、高分解能SEM（走査型電子顕微鏡）やTEM（透過型電子顕微鏡）、あるいは蛍光X線分析装置等、最新の電子機器が活用され、優れた成果を挙げていた。また、照射済み燃料の温度を自在にコントロールして過渡条件を模擬し、このような条件下でのガス放出挙動や組織変化挙動を研究している報告もあった。

これらの研究は水炉燃料のみならず高速炉燃料にも共通して重要であることから、動燃大洗工学センター燃料材料開発部の照射後試験施設にもこのような試験機器を導入していくことが必要であると感じた。

4.4 結論

本会議は、米国、フランス、日本等から延べ約270名（うち日本からは28名）の参加者のもと、軽水炉燃料の性能、特に高燃焼度化に主眼をおいて開催された。セッションの構成は、３つの基調講演、①燃料性能Ⅰ、②MOX燃料、③燃料性能Ⅱ、④破損後の燃料挙動、⑤被覆管性能、⑥高燃焼度燃料、⑦反応度事故の７つの口頭発表セッションと２つのポスターセッションからなり、全部で91件（うち日本からは20件）の論文発表と活発な討議が行われた。

動燃からは、以下の３件を発表した。

(1) "Behavior of MOX Fuel Irradiated in a Thermal Reactor"（口頭発表）
(2) "Power Ramp Tests of MOX Fuel Rods for ATR (IFA-591) "（口頭発表）
(3) "Development of High Burn-Up MOX Fuel for ATR"（ポスター発表）

本会議で発表された他の研究機関のMOX燃料に関する研究開発情報の収集としては、主としてフランスにおける商業炉および試験炉を用いたMOX燃料の照射および照射後試験結果ならびにコード開発の現状についてまとめた。

付録－1　プログラム

Listing of Presentations

1997 International Topical Meeting on LWR Fuel Performance

Portland, Oregon March 2-6, 1997

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Fragema & Framatome Cogema Fuels

Monday, March 3, 1997

08:30-10:15 Introductions and Keynote Presentations

Co-General Chairs, Jerry L. Ethridge (Pacific Northwest National Laboratory-USA) and Dave L. Larkin (Washington Public Power Supply System-USA)

Honorary Chair, David G. McAlees (Siemens Power Corporation-USA)

Technical Program Chair, Mitchel E. Cunningham (Pacific Northwest National Laboratory-USA)

Meeting the Challenge of Managing Nuclear Fuel in a Competitive Environment
Rosa L. Yang (EPRI)

Utility Perspective on Commercial Light-Water Reactor Fuel in Japan
Sadaaki Sasaki (TEPCO-Japan)

EDF Current Perspective on Commercial PWR Fuel
J.L. Provost (EdF-France)

10:15-10:45 Break

10:45-12:00 Fuel Performance I

Co-chairs: Richard Miller (Westinghouse-USA), Professor Tadatsune Okubo (Sophia University-Japan)

Westinghouse Fuel Performance in Today's Aggressive Plant Operating Environment
H. W. Wilson, H. F. Menke, H. Kunishi, R. S. Miller, L. R. Scherpereel (Westinghouse-USA)

Framatome and FCF Recent Operating Experience and Advanced Features to Increase Performance and Reliability
Gilles Ravier, Georges Masuy (Framatome- France), J. T. Willse (FCF-France)

LWR Fuel Performance and Material Development Activities for Extended Burnup in Russia
M.I. Solonin, Y.K. Bibilashvili, A.V. Medvedev, V.V. Novikov, O.A. Nikishov, A.V. Nikulina, N.B. Sokolov, F.F. Sokolov (A.A. Bochvar Research Insititute of Inorganic Materials-Russia), G.L. Lunin, V.N. Proselkov (Kurchatov Institute-Russia), A.K. Panjushkin (Mashinostroitelniy Zavod-Russia), O.B. Samoylov (OKBM-Russa), V.N. Golovanov, V.A. Ovchinikov, A.V. Smirnov, V.K. Shamardin (Research Institute of Atomic Reactores-Russia)

13:30-17:00 Mixed Oxide Fuels

Co-Chairs: Michio Ichikawa (JAERI-Japan), Xavier Thibault (EDF-France)

Recent Results from the In Reactor MOX Fuel Performance in France and Improvement Program
Patrick Blanpain (Framatome Nucl Fuel-France), Xavier Thibault (EdF Villeurbanne-France), Jean-Paul Pages (CEA/DRN/P-France)

High Burnup MOX Fuel and Fuel Rod Design Improvement
S. Doi (MHI Kobe-Japan), K. Yamate (Kansai Electric-Japan)

Behavior of MOX Fuel Irradiated in a Thermal Reactor
Takeshi Mitsugi, Naoya Kushida (PNC, Oarai-Japan), Keiichi Kikuchi (PNC-Japan)

15:00-15:30 Break

Analytical Studies of the Behaviour of MOX Fuel
Laurent Caillot, Gerard Delette, Jean Paul Piron, Clement Lemaignan (CEA, Grenoble-France), Alain Chotard (Framatome Nucl Fuel-France), Jean Philippe Berton (EdF Villeurbanne-France)

Power Ramp Tests of MOX Fuel Rods for ATR (IFA-591)
Soichiro Yano, Norihiko Onuki, Shusaku Kohno, Katsuichiro Kamimura (PNC-Japan)

17:30-19:30 Poster Session 1

Co-chairs: Barclay Andrews (Framatome Cogema Fuels-USA), Tony Turnbull (UK)

Application of a Semi-Empirical Rod Drop Model for Studying Rod Insertion Anomalies at South Texas Project and Ringhals Unit 4
Lars Bjornkvist (Vattenfall-Sweden), Ernie Kee (HL&P, USA)

Lessons Learned from Control Rods Irradiation Experience, Development of Advanced Absorbers and Their Refractory Properties Under Accident Conditions
Vladimir M. Chernyshov (Moscow Polimetal Plant-Russia), Vladimir M. Troyanov (IPPE-Russia)

The Summary of WWER-1000 Fuel Utilization in Ukraine
Anatoliy Afanasyev (USCNPU-Ukraine)

Fission Gas Release in ABB SVEA 10x10 BWR Fuel
David Schrire, Ingvar Matsson, Bjorn Grapengiesser (ABB Atom-Sweden)

Advanced BWR Channels
Adolfo Reparáz, Dave Barkhurst (Siemens), Stefan Linden, Hans Lippert (Siemens/KWU-B-Germany)

Development of High Burn-Up MOX Fuel for ATR
Shinichi Uematsu, Ichiro Kurita, Norihiko Onuki (PNC-Japan), Kouzo Kodaka, Kohji Terunuma (PNC, Oarai-Japan)

High Burnup Modelling of UO_2 and MOX Fuel with METEOR/TRANSURANUS Version 1.5
C. Struzik, M. Moyne (CEA/CEN Cadarache-France), J. P. Piron ([illegible] Grenoble-France)

Iodine Spiking Model for Pressurized Water Reactors
B. J. Lewis, F. C. Iglesias (RMC-Canada), A. K. Postma (Benton City Technol-USA), D. A. Steininger (EPRI-USA)

Vaporization of Low-Volatile Fission Products in Severe Light Water [illegible] Accidents
B. J. Lewis, W. T. Thompson, M. H. Kaye, B. J. Corse (RMC-Canada [illegible] C. Iglesias (Ontario Hydro-Canada), B. André, G. Ducros, M. [illegible] (CEA/CEN Grenoble- France), D. Maro (CEA-France)

Investigation of the Roles of Corrosion and Hydriding of Barrier [illegible] Fuel Pellet Oxidation in BWR Fuel Degradation
D. R. Olander, Wei-E. Wang, Yeon soo Kim, C. Y. Li, S. Lim (Univ. [illegible] California, Berkeley-USA), Suresh K. Yagnik (EPRI-USA)

Analysis of Primary Coolant Activity with Leaking Fuels in the [illegible] Water Reactor
Chan Bock Lee, Ig Sung Lim (KAERI-Korea)

Applications of Simple Rules of Fuel Failure in a Computer Model [illegible] for Nuclear Fuel Behaviour and Performance
Armando Carlos Marino (CNEA, Bariloche-Argentina), Eduardo J. Savino (CNEA-Argentina)

Fuel Performance at Nuclear Power Plant Bohunice
Ivan Smieško (SPUCO-Slovak Republic)

The Use of Three Cesium Isotopes for the Batch Identification of a Single Fuel Failure in a Pressurized Water Reactor
C. W. Sayles, R. Y. Chang (SCE-USA)

Failed Annular UO_2 Fuel in PWR Conditions: The EDITH 03 Experiment
Daniel Parrat (CEA, Grenoble-France), Yves Musante (Framatome Nucl Fuel-France), Alain Harrer (EdF Villeurbanne-France)

The Behavior and Management of Failed LWR Fuel Rods
Michael Kennard, Dion Sunderland, John Harbottle (Stoller, Pleasantville-USA)

High Pressure Steam Corrosion and Hydriding of Zircaloy-4
Yong-soo Kim (Hanyang Univ-Korea), Duck-kee Min (KAERI-Korea), Sun-ki Kim (Hanyang Univ-Korea), Seung-gy Ro (KAERI-Korea), Young-kil Ok (Hanyang Univ-Korea), Moon-ghu Park (KEPRI- Korea)

Irradiation Behavior of Zr Alloys for Ultra High Burnup Fuel
Yoshinori Etoh, Sachio Shimada (Nippon NFD-Japan), Ronald B. Adamson (GE, Pleasanton-USA), Takayoshi Yasuda (Hitachi-Japan), Toshiaki Kogai (Toshiba-Japan), Yoshiaki Ishii (TEPCO- Japan)

Modelling of the Mechanical Behaviour of Zircaloy-4 Cladding Tubes from Unirradiated State to High Burn-Up
I. Schaffler-Le Pichon, Ph. Geyer (EdF/DER/MTC-France), P. Delobelle (U.A. CNRS-France), P. Bouffioux (EdF/DER/EMA-France)

Anisotropy Viscoplastic Behavior of Zircaloy 4 Cladding Tubes Under Simulated Load Follow Conditions
Pol Bouffioux (Edf/DER/EMA-France), Philippe Geyer (EdF/DER-France)

Effects of Water Radiolysis on Zircaloy Corrosion
Yoshitaka Nishino (Hitachi, PIS-Japan), Takayoshi Yasuda (Hitachi Works-Japan), Eishi Ibe, Masao Endo (Hitachi, PIS-Japan)

Development of Ultra-High Burnup Fuel for BWR
A. Fukazawa, Y. Shirai, H. Harada (TEPCO-Japan), T. Furuya, N. Itagaki, Y. Yuasa, Y. Mozumi (NFI-Japan)

Corrosion Behavior of Zircaloy 4 Cladding Material: Evaluation of the Hydriding Effect
Martine Blat (EdF/DER-France), Josseline Bourgoin (EdF/GDL/SCMI-France)

Tuesday, March 4, 1997

08:00-12:00 Fuel Performance II

Co-chairs: Rosa Yang (EPRI), Enrico Sartori (OECD/NEA)

Recent GE BWR Fuel Experience
G. A. Potts (GE, Wilmington)

Siemens Fuel Performance Overview
Keith N. Woods (Siemens-USA), Wolfgang Klinger (Siemens/KWU BTE-Germany)

Improved BWR and PWR Fuel Designs and Operating Experience at ABB
L. V. Corsetti, Z. E. Karoutas, H. R. Freeburn (ABB CE-USA), L. Hallstadius, S. Helmersson (ABB Atom-Sweden)

BWR Fuel Performance and Recent R&D Activities in Japan
Keizo Ogata (Nippon NFD-Japan), Akio Fukazawa (TEPCO-Japan), Kenichi Ito (Hitachi-Japan), Takao Koyama (Nippon NFD-Japan), Kazuhiro Takei (TEPCO-Japan), Toshio Matsumoto (Toshiba RD&E-Japan)

10:00-10:30 Break

Irradiation Characteristics of BWR Step II Lead Use Assemblies
H. Hayashi, M. Kitamura (NPEC-Japan), K. Ito, T. Kubo (Hitachi-Japan), T. Nomata, T. Kogai (Toshiba-Japan), Y. Wakashima, H. Sakurai (Nippon NFD-Japan)

Design and Performance of Mitsubishi PWR Fuel for Increased Reliability
S. Abeta, R. Fukuda, K. Kita (MHI-Japan)

Burnup Extension of Japanese PWR Fuels
K. Yamate (Kansai Electric-Japan), A. Oe, M. Hayashi, T. Okamoto (NFI-Japan), H. Anada, S. Hagi (Sumitomo-Japan)

13:30-17:00 Post Defect Fuel Behavior

Co-chairs: Larry Noble (GE-USA), Vladimir Onoufriev (IAEA)

Assessment of BWR Fuel Degradation by Post-Irradiation Examinations and Modeling in the Defect Code
S. K. Yagnik, O. Ozer, B. C. Cheng, R. L. Yang (EPRI-USA), R. O. Montgomery, Y. R. Rashid (Anatech-USA), J. H. Davies, E. V. Hoshi, R. B. Adamson (GE, Pleasanton-USA)

Fe-Enhanced Zr Liner Cladding
A. Seibold, R. Manzel (Siemens/KWU-Germany), K. N. Woods (Siemens-USA), N. Itagaki (NFI-Japan)

Studies of the Secondary Hydriding Process in Fresh Test Fuel with Simulated Primary Defects
Christian Gräslund, Gunnar Lysell (Studsvik-Sweden), Keizo Ogata (Nippon NFD-Japan), Toru Takeda (NFI-Japan)

Post-Irradiation Examination of Failed KKK - Barrier Fuel Rods
A. Hüttmann, M. Ketteler, J. Skusa (HEW-Germany), H. Heckermann (RWE Energie-Germany), G. Rudholzer (Bayernwerk-Germany), R. Manzel (Siemens/KWU-Germany)

15:00-15:30 Break

Recent ABB BWR Failure Experience
Lembit Sihver, Lars Hallstadius, Gunnar Wikmark (ABB Atom-Sweden)

SADDAM: An On Line Computer Code to Assess in Operation Defective Fuel Characteristics and Primary Circuit Contamination
C. Leuthrot, J. B. Genin (CEA/DRN/SECA-France), P. Ridoux, A. Harrer (EdF Villeurbanne-France)

An Evaluation of the Potential for PCI in BWR Barrier Fuel Failures
Dion J. Sunderland, Michael W. Kennard, John E. Harbottle (Stoller, Pleasantville-USA)

Wednesday, March 5, 1997

08:00-12:00 Cladding Performance

Co-chairs: Keith Woods (SPC-USA), Roland Traccucci (Framatome Nucl Fuel-France)

Fuel Performance and Water Chemistry Variables in LWRS
B. C. Cheng (EPRI), J. M. Brown (Union Electric, St. Louis-USA), K. G. Turnage (Southern Nuclear-USA), E. A. Armstrong (ComEd, Downers Grove-USA), M. Hudson (NUSCO, Waterford-USA)

Distinctive Crud Pattern and Failed Fuel Pins
David Mitchell (FCF)

In-Reactor Fuel Cladding Corrosion Performance at Higher Burnups and Higher Coolant Temperatures
G. P. Sabol, R. J. Comstock, G. Schoenberger, H. Kunishi, D. L. Nuhfer (Westinghouse-USA)

Update on the Development of Advanced Zirconium Alloys for PWR Fuel Rod Claddings
Jean Paul Mardon (Framatome Nucl Fuel-France), Garry Garner (FCF-France), Pierre Beslu (CEA Cadarache-France), Daniel Charquet (Cezus-France), Jean Senevat (Zircotube-France)

9:45-10:15 Break

Performance of Standard and Advanced Fuel Rod Cladding for High Burnup Applications in PWRs
S. R. Pati (ABB CE-USA), P. Jourdain (ABB Atom-Sweden), G. P. Smith, A. M. Garde (ABB CE-USA), L. Hallstadius (ABB Atom-Sweden)

Behavior of Zircaloy-4 and Zirconium Liner Zircaloy-4 Cladding at High Burnup
L. F. Van Swam (Siemens-USA), A. A. Strasser (Aquarius Svc-USA), J. D. Cook (RG&E-USA), J. M. Burger (ESEERCO-USA)

Irradiation Test on High Performance Fuel
H. Ikehata (Nucl Development-Japan), K. Yamate (Kansai Electric-Japan), S. Abeta (MHI Yokohama-Japan), T. Okubo (Sophia Univ-Japan), T. Takahashi (MHI Kobe-Japan), H. Uchida, I. Komine, Y. Inoue (NPECO-Japan)

Effects of SPP Dissolution on Mechanical Properties of Zircaloy-2
S. T. Mahmood, K. W. Edsinger, D. M. Farkas, R. B. Adamson (GE, Pleasanton-USA)

13:30-17:00 High Burnup Fuel Performance

Co-chairs: Satya Pati (ABB CE-USA), Motoyasu Kinoshita (CRIEPI-Japan)

BWR and PWR Fuel Performance at High Burnup
L. F. Van Swam, G. M. Bain, W. C. Dey (Siemens), D. D. Davis (CP&L-USA), H. Heckermann (RWE Energie-Germany)

Fission Gas Release and Pellet Structure at Extended Burnup
R. Manzel (Siemens/KWU-Germany), M. Coquerelle (CEC-Germany)

High Burnup BWR Fuel Pellet Performance
S. Vaidyanathan, R. D. Reager, R. W. Warner, C. Martinez (GE, Pleasanton-USA), Y. Shirai (TEPCO-Japan), Y. Iwano (Nippon NFD-Japan)

Effect of Irradiation-Induced Microstructural Evolution on High Burnup Fuel Behavior
K. Une, K. Nogita, S. Kashibe, T. Toyonaga, M. Amaya (Nippon NFD-Japan)

15:00-15:30 Break

Performance of Improved UO_2 Pellets at High Burnup
M. Hirai, T. Hosokawa, R. Yuda, K. Une, S. Kashibe, K. Nogita (Nippon NFD-Japan), Y. Shirai, H. Harada (TEPCO-Japan), T. Kogai (Toshiba-Japan), T. Kubo (Hitachi-Japan), J. H. Davies (GE-Pleasanton-USA)

Thermal Diffusivity Measurement of High Burnup UO_2 Pellet
Jinichi Nakamura, Tsuneo Kodaira, Masaaki Uchida, Takeshi Yamahara, Hiroshi Uetsuka, Akira Kikuchi (JAERI-Japan)

Assessment of UO_2 Conductivity Degradation Based on In-Pile Temperature Data
W. Wisenack (OECD-Norway)

17:30-19:30 Poster Session 2

Co-chairs: Carl Beyer (PNNL-USA), Eric Kolstad (HRP-Norway)

The Compilation of a Public Domain Database on Nuclear Fuel Performance for the Purpose of Code Development and Validation
P. M. Chantoin (IAEA-Austria), E. Sartori (OECD-France), J. A. Turnbull (Independent Consultant)

Thermodynamics of the High Burnup UO_2 Structure Formation and Fission Gas Release
Serguei E. Lemehov (Kurchatov Inst-Russia)

High Burnup Rim Project: Progress of Irradiation and Preliminary Analysis
M. Kinoshita, S. Kitajima, T. Kameyama, T. Matsumura (CRIEPI-Japan), E. Kolstad (OECD-Norway), Hj. Matzke (CEC-Germany)

A New Integral Fuel Burnable Absorber Design for LWR's
Üner Çolak (Hacettepe Univ-Turkey), H. Hasan Durmazuçar (Cumhuriyet Univ-Turkey), Gungor Gündüz (OrtaDogu Teknik Univ-Turkey)

Thermal Conductivity Measurements of High Burnup UO_2 Pellet and a Benchmark Calculation of Fuel Center Temperature
Koichi Ohira, Noboru Itagaki (NFI-Japan)

Advanced Fuel Development for Burnup Extension
T. Takahashi (MHI Kobe-Japan), K. Yamate (Kansai Electric-Japan)

Improvement of FRAPCON-2 Code Based on FUMEX Exercises
Zhang Yingcaho, Zhang Shishun, Sun Songging, Chen Peng (CIAE-China)

The IAEA CRP FUMEX Influence on the Fuel Rod Performance Modeling Quality in the Czech Republic
Radek Svoboda (NRI-Czech Republic)

Change of Thermodynamic Properties of UO_2 Fuel Doped with Magnesium and Other Metals
Takeo Fujino, Nobuaki Sato (Tohoku Univ-Japan), Kousaku Fukuda (JAERI-Japan)

SIERRA: A Fuel Performance Code to Predict the Mechanical Behavior of Fuel Rods up to High Burnup
M.R. Billaux, S.H. Shann, L.V. Van Swam (Siemens), F. Sontheimer (Siemens/KWU-Germany)

Experimental Study of Fission Gas Release at the Dodewaard BWR by Comparison of Krypton-85 and Puncturing Measurements with ESCORE Results
J.A.A. Wouters, Wim J.M. Siegers (KEMA-Netherlands), Wim R. van Engen (N.V. GKN-Netherlands)

Using Radially Inhomogeneously Enriched UO_2 Fuel to Reduce the Rim Effect
K. Bakker, H. Hein, R.J.M. Konings (ECN-Netherlands)

Fuel Performance Analysis by Code "FAIR" and Development of Neural Network for Fuel Pin Analysis
P. Swami Prasad, B.K. Dutta, H.S. Kushwaha

An Assessment of Recent High Burnup Modifications to NRC Fuel Performance Code FRAPCON-3
Donald D. Lanning, Carl E. Beyer (PNNL-USA), Gary A. Berna, Kurt Davis (INEL-USA)

Capabilties and Validation of the ASFAD Performance Code for WWER-440 High Burnup Fuel Rods
Serguei E. Lemehov (Kurchatov Inst-Russia)

The Prediction Method of Fuel Cladding-Coolant Heat Transfer During a Reactivity-Initiated Accident
T.N. Dinh (RIT-Sweden)

Analysis of Fuel Behavior During Rod Ejection Accident in Korea Standard PWR
Chan Bock Lee, Chung Chan Lee, Oh Hwan Kim, Jin Gon Chung, Chong Chul Lee (KAERI-Korea)

On the Influence of an Embrittled Rim on the Ductility of Zircaloy Cladding
Todd M. Link, Donald A. Koss, Arthur T. Motta (Penn State-USA)

Examination of the Spent VVER Fuel Behavior Under Accident Conditions Using Electrically Heated Installations
V. Smirnov, et al

Analyzing the Rod Drop Accident in a BWR with High Burnup Fuel
David J. Diamond, Lev Neymotin (BNL-USA)

Post-Test Examinations of High Burnup PWR Fuels Submitted to RIA Transients in CABRI Facility
Didier Lespaiux, Jean Noirot, Patrick Menut (CEA Cadarache-France)

The SCANAIR Code for the Description of PWR Fuel Rod Behavior Under RIA: Validation on Experiments and Extrapolation to Reactor Conditions
J. Papin, H. Rigat, F. Lamare, B. Cazalis (CEA Cadarache-France)

Cladding Metallurgy and Fracture Behavior During Reactivity-Initiated Accidents at High Burnup
H.M. Chung and T.F. Kassner

Thursday, March 6, 1997

08:00-12:30 Reactivity Initiated Accidents

Co-chairs: Ralph Meyer (NRC-USA), Franz Schmitz (IPSN)

NSRR/RIA Experiments with High Burnup PWR Fuels
Toyoshi Fuketa, (JAERI-Japan), Yukihide Mori (MHI Kobe-Japan), Hideo Sasajima, Takehiko Nakamura (JAERI-Japan), Yoshihiro Tsuchiuchi (NFI-Japan), Kiyomi Ishijima (JAERI-Japan)

Hydride Morphology and Hydrogen Embrittlement of Zircaloy Fuel Cladding Used in NSRR/HBO Experiment
Fumihisa Nagase, Hiroshi Uetsuka (JAERI-Japan)

The Main Outcomes from the Interpretation of the CABRI REP-Na Experiments for RIA Study
J.M. Frizonnet, J.P. Breton, H. Rigat, J. Papin (IPNS-France)

RIA Related Analytical Studies and Separate Effect Tests
F. Lemoine, M. Balourdet (IPNS-France)

10:00-10:30 Break

Investigation of the Behavior of VVER Fuel Under RIA Conditions
Vladimir Asmolov, Larissa Yegorova (Kurchatov Inst-Russia)

Review and Analysis of RIA-Simulation Experiments on Intermediate and High Burnup Test Rods
R.O. Montgomery, Y.R. Rashid (ANATECH-USA), O. Ozer, R.L. Yang (EPRI-USA)

Fuel Failure Risk Assessment Under Rod Ejection Accident in PWRs Using the RIA Simulation Tests Database—The French Utility Position
S. Stelletta, N. Waeckel (EdF/SEPTEN Villeurbanne-France)

A Regulatory Assessment of Test Data for Reactivity Accidents
Ralph O. Meyer (NRC-USA), Richard K. McCardell (INEL-USA), Harold H. Scott (NRC-USA)

付録－2　参加者リスト

1997 INTERNATIONAL TOPICAL MEETING
ON LIGHT WATER REACTOR FUEL PERFORMANCE
March 2-6, 1997

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付録－3　発表資料

BEHAVIOR OF MOX FUEL IRRADIATED IN A THERMAL REACTOR

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ABSTRACT

As part of the first stage of high burnup uranium-plutonium mixed oxide (MOX) fuel development, series of irradiation tests were conducted on the 36-rod fuel assemblies. Four assemblies were irradiated up to an assembly burnup of 33.1 GWd/t in two thermal reactors. The results of post-irradiation examinations (PIEs) were analyzed to investigate irradiation behavior of MOX fuel.

The investigation showed that there was no distinguishable difference in irradiation behavior between MOX and UO_2 fuel except for helium gas release and plutonium heterogeneity observed in a MOX fuel pellet. It also verified the design consideration of the ATR 36-rod fuel assembly.

I. INTRODUCTION

More than 600 MOX fuel assemblies which corresponds to more than 16,000 MOX fuel rods were fabricated and irradiated by Power Reactor and Nuclear Fuel Development Corporation (PNC), since the operation start of the 165 MWe prototype advanced thermal reactor (ATR) Fugen. To date, there observed no indication of fuel failure by pool-side inspection or PIE, which indicates excellent reliability of PNC MOX fuel in a thermal reactor.

Recently, our efforts for research and development have been focused on mainly improving fuel cycle cost. Two stages of MOX fuel development aiming burnup extension have been set up for this purpose.

The 36-rod fuel assembly with a design burnup of 38 GWd/t which consists of 36 fuel rods compared to 28 fuel rods in original design of the Fugen driver fuel assembly, was developed for the first stage. To verify design procedure of the 36-rod fuel assembly and to accumulate irradiation experience of high burnup MOX fuel, series of irradiation tests were performed; irradiation test of three fuel assemblies designated as E03, E04, and E05 in the Fugen core, and that of one fuel assembly designated as Type-E in the Steam Generating Heavy Water Reactor (SGHWR) of Atomic Energy Authority (AEA) U. K.. The results of irradiation tests and verification of design procedure for the Type-E fuel assembly was reported[1] already.

For the second stage of MOX fuel burnup extension, a 54-rod fuel assembly with a design burnup of 55 GWd/t is under development. Current status of high burnup MOX fuel development in PNC will be presented elsewhere in this meeting[2].

In this paper, the results of irradiation tests of E04 and E05 fuel assemblies are presented, focussing on irradiation behavior of a PNC MOX fuel at high burnup.

II. IRRADIATION TEST

A. Test Assemblies

The schematic drawing of the 36-rod fuel assembly is shown in Fig. 1. Its design specification is listed in Table 1, compared with the Fugen MOX driver fuel assembly. It has 36 fuel rods to be distributed in three cylindrical rings (six rods in inner-ring, 12 in intermediate-ring, and 18 in outer-ring). Six of 12 fuel rods in intermediate-ring are tie-rods which interconnect the upper and lower tie plates through the fuel bundle. Fuel rods are positioned by 12 spacers made of Inconel with axial locations fixed by one spacer supporting rod.

Basically, the design of the 36-rod fuel assembly follows its predecessor Fugen driver fuel. One of apparent differences is cladding outer diameter which is decreased from 16.46 mm of the Fugen driver fuel to 14.5 mm to accommodate 36 MOX fuel rods within a bundle with the same diameter. Another difference is Pu content which is increased to incorporate higher burnup. Note that specification for Pu content is higher in inner and middle-ring fuel rods than that in outer-ring fuel rods to balance linear heat rate among three rings. In addition, the Pu content of ATR MOX fuels is slightly lower than that of light water reactors (LWRs). This is because capture of thermal neutrons by heavy water in case of the ATR is significantly lower than that by light water in case of LWRs.

MOX fuel pellets loaded into E03, E04, and E05 fuel assemblies were fabricated from master blended MOX powder prepared by the microwave heating (MH) process[3]. In the MH process, the master blend is obtained through direct co-conversion of U-Pu mixed nitrate solution by microwave heating. Then, the master blend is diluted and mixed with UO_2 powder by ball milling to obtain specific Pu content. Due to direct co-conversion and subsequent ball milling, MH pellets have excellent homogeneity of Pu distribution.

The test assemblies were fabricated at the Plutonium Fuel Fabrication Facility of Tokai works of PNC, where the Fugen driver fuel is also fabricated.

B. Irradiation

The test assembly E03, E04, and E05 were loaded into center core region of Fugen and irradiated side by side. This implies that these three assemblies have almost same linear heat rate history, except the fact that E04 was irradiated for 1203 days and obtained an assembly burnup of 25.1 GWd/t, while E03 and E05 were irradiated for 1697 days and reached to 33.1 GWd/t. The history of rod average linear heat rate for E05 is shown in Fig. 2. The linear heat rate reached maximum of 30.2 kW/m at 120 day, then decreased gradually. Relatively large increase of approximately 3 kW/m due to change of control rod pattern was observed at 1320 day.

E04 and E05 were selected to be examined from three test assemblies irradiated at the Fugen, and their PIE were performed at the Reactor Fuel Examination Facility of Japan Atomic Energy Research Institute and at hot cell facilities in Oarai Engineering Center of PNC.

III. RESULTS AND DISCUSSION

A. Fission Gas Release Behavior

Measured fission gas release rate of the MOX fuel rods from E04 and E05 are shown in Fig. 3 as a function of rod average burnup, compared with those of UO_2 fuels irradiated in a Japanese BWR[5]. The fission gas release rate is also plotted in Fig. 4 as a function of the experienced maximum linear heat rate after 10 GWd/t. Fission gas release rates of E04 and E05 fuel rods increase with burnup and linear heat rate, and are scattered between almost 0 and less than 20 %. They are in good agreement with the Fugen MOX driver fuel and also within the data spread of the UO_2 fuels. It is apparent that fission gas release behavior of the MOX fuel is quite similar to that of UO_2 fuel.

It is well known that homogeneity of MOX fuel affects its fission gas release rate at high burnup. Simple mechanical blending may cause non-uniform Pu distribution in the fuel matrix, consequently resulted in high fission gas release. However, there observed no distinguishable difference between PNC MOX fuel and the UO_2 fuels in this work. Mishima et. al. investigated fission gas release behavior of MOX pellets including MH pellets and mechanically blended pellets, using instrumented irradiation rigs up to a rod average burnup of 28 GWd/tMOX[6]. They reported that fission gas release rate of MH pellets was lower than that of the mechanically blended pellets. Also reported was that its release behavior was similar to that of UO_2 fuels, and there was no significant difference of fission gas release rate between MOX and UO_2 fuels.

The pellets loaded into E04 and E05 were MH pellets, and as described later, its excellent homogeneity was confirmed by PIE. Therefore, it is considered that these experimental results suggest that fission gas release behavior of PNC MOX fuel is comparable to that of UO_2 fuels in a burnup range up to 35 GWd/t.

B. Helium Release Behavior

In addition to Xe and Kr, significant amount of helium was detected by puncture tests. In Fig. 5, amount of helium measured is plotted as a function of rod average burnup, compared with those of UO_2 fuel irradiated in a Japanese BWR[5]. Amount of helium released is apparently larger in MOX rods than in a UO_2 rods. It is reported that primary sources of He generation in a UO_2 fuel rod are ternary fission and alpha decay of ^{242}Cm[7]. In MOX fuel, ^{242}Cm generation during irradiation is much faster than UO_2 due to its high Pu content, which results in larger generation of He. Fig. 6 depicts relationship between amount of helium and fission gas (Xe and Kr) released. The amount of helium released is almost proportional to that of fission gas released. These are consistent with the results reported earlier[7, 8], and implies that release behavior of helium is similar to that of fission gas with respect to its dependence on burnup and linear heat rate.

From fuel design point of view, behavior of helium release is important to estimate internal gas pressure of a fuel rod. As shown in Fig.5, amount of helium released increases with burnup. However, the comparison of internal pressure between measured and calculated by

ATFUEL code which was used in thermo-mechanical calculation for the test fuel rods, confirmed that the ATFUEL predicts internal pressure of the MOX fuel rod conservatively up to a rod average burnup of 35 GWd/t.

For designing higher burnup MOX fuel rod, a new evaluation code which is equipped with more mechanistic model for prediction of rod internal pressure, was developed in PNC[2].

C. Pellet Microstructure

Fig. 7 shows typical microstructure of a PNC MOX fuel pellet of E04 irradiated up to a local burnup of 30.8 GWd/t. Grain growth and fission gas bubble precipitation on grain boundary were observed at pellet center region reflecting the fact that the fission gas release rate of this fuel rod was 15.6 %.

Other microstructural characteristics of E04 fuel assembly are occurrence of narrow porous band observed in pellet peripheral region of an outer-ring fuel rod and porous structure of Pu agglomerates.

Fig. 8 shows scanning electron micrograph of pellet peripheral region, comparing with Xe and Nd profiles measured by EPMA. A large number of pores are accumulated in peripheral region. Thickness of this porous band extended to approximately 50 micron from the pellet surface. However, this band didn't cover all over pellet peripheral region, but existed only discontinuously. In this band, Nd intensity by EPMA increases steeply toward pellet surface, indicating burnup onset due to rim effect. On the other hand, Xe intensity doesn't follow the Nd intensity profile, rather decreases slightly. Recently, microstructure change with these characteristics are observed in pellet peripheral region of high burnup UO_2 fuel. Many authors reported that the microstructure change started at an average cross-sectioned burnup of around 40 GWd/t or at a local burnup of 70 to 80 GWd/t. The average burnup of the fuel pellet shown in Fig. 8 is 28.4 GWd/t. This burnup is lower than those reported for UO_2 fuel. Moreover, this is even lower than middle-ring fuel shown in Fig. 7 where the porous band was not observed.

Since neutron is moderated by heavy water which surround fuel assemblies in the ATR core design, moderated neutron goes into a fuel assembly from outer ring. Therefore, a local burnup is highest at outer surface of fuel assembly, i.e. pellet rim of an outer-ring fuel rod which faces to moderator. From Nd profile measured, a local burnup of porous band is deduced to be approximately two times larger than the pellet average burnup, i.e. around 60 GWd/t, which is well agreed with threshold burnup of transition zone reported by Lassman et. al.[9] where porous band starts to occur. In case of middle ring fuel rods, onset of burnup at pellet peripheral region is not so steep that the local burnup at pellet edge doesn't reach the threshold in spite of higher average burnup.

It is easily supposed that the significant development of this porous microstructure may affect physical properties such as thermal conductivity and/or swelling. However, no experimental result associated with the porous band, such as significant development of grain growth or fuel to cladding gap closure, was obtained through the PIEs.

At the pellet periphery, Pu agglomerates were observed. Fig. 9 shows scanning electron micrograph of a Pu agglomerate together with intensity profile of Xe, Pu and Nd measured by EPMA. From the micrograph, it is clearly seen that the Pu agglomerate contains large population of fine pores, which cause dark appearance of the agglomerates in ceramograph. The Pu intensity profile showed that the maximum diameter of Pu agglomerates was 50 micron approximately. Quantitative measurement by EPMA revealed that maximum Pu concentration in the agglomerates was around 14 %. It also showed that profile of Nd concentration almost traced that of Pu, while increase of Xe concentration in Pu agglomerates was not obvious. Since a local burnup of Pu agglomerates was estimated to exceed above-mentioned threshold burnup of 70 to 80 GWd/t for porous structure from Pu concentration measured, it is considered that the porous microstructure of Pu agglomerates resulted from its high burnup.

Plutonium agglomerates have such characteristics, however, there observed no measurable effect of Pu agglomerates on irradiation behavior of MOX pellets, partly because the MOX fuel pellets loaded into the ATR 36-rod fuel assemblies were highly homogenized by MH process.

D. Cladding Inner Surface Oxidation

In a MOX fuel rod, it is anticipated that thickness of oxide layer on cladding inner surface become larger than that in a UO_2 rod at high burnup, because Pu atoms generate more noble metal atoms by fission than U atoms does and oxygen potential of MOX fuel is higher than that of UO_2. However, as shown in Fig. 7, oxidation of cladding inner surface in a MOX fuel rod of E04 was benign. The maximum oxidation thickness was around 7 micron in high linear heat rate region, which is consistent with the literature[10]. Relationship between location of corrosion and presence of Pu agglomerate where higher oxygen potential can be expected due to high Pu concentration was vague also.

These result suggests that oxidation of cladding inner surface of a MOX fuel rod is small and may not be affected by presence of Pu agglomerates in the burnup range up to 35 GWd/t.

IV. CONCLUSION

Through the series of irradiation tests up to an assembly burnup of 33.1 GWd/t, a lot of useful experiences have been accumulated on the irradiation behavior of a MOX fuel. The analysis of the PIE results showed excellent perfomance of the high burnup MOX fuel assembly. Comparison of irradiation behavior between MOX and UO_2 fuel showed that there was no distinguishable difference in irradiation behavior between MOX and UO_2 fuel except for helium release and Pu heterogeneity.

MOX fuel releases helium more than UO_2 fuel does, since ^{242}Cm is generated faster in MOX fuel. However, it is confirmed that the internal pressure in a MOX fuel rod is predicted by the ATFUEL code conservatively up to a rod average burnup of 35 GWd/t. Though plutonium agglomerates with porous structure were observed, there detected no measurable effect on irradiation behavior of MOX fuel.

ACKNOWLEDGEMENTS

The authors would like to thank T. Kobayashi of Electric Power Development Co., Ltd. for useful discussion and the program administration, and many hot lab. staff members at JAERI and PNC for their PIE works.

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Table 1 Summary of design specifications

Items	Fugen MOX Driver Fuel	ATR 36-rod fuel (E03, E04, E05)
1. Fuel Pellet		
Material	PuO_2-UO_2	PuO_2-UO_2
Outer Diameter (mm)	14.40	12.40
Height (mm)	18	13
Density (%T.D.)	95	←
Shape	Solid with Dish & Chamfer	←
Pu Fissile Enrich (wt%)	0.4~1.62	0.98~2.45
2. Fuel Rod		
Cladding Material	Zircaloy-2	←
Outer Diameter (mm)	16.46	14.50
Inner Diameter (mm)	14.70	12.70
Fuel Stack Length (mm)	3700	3647
Filling Gas	He	←
& Pressure (MPa)	0.1	0.3
3. Fuel Assembly		
Length (mm)	4388	4398
Bundle O. D. (mm)	111.6	←
Rod Number	28	36
Inner	4	6
Intermediate	8	12
Outer	16	18
Spacer Number	12	12

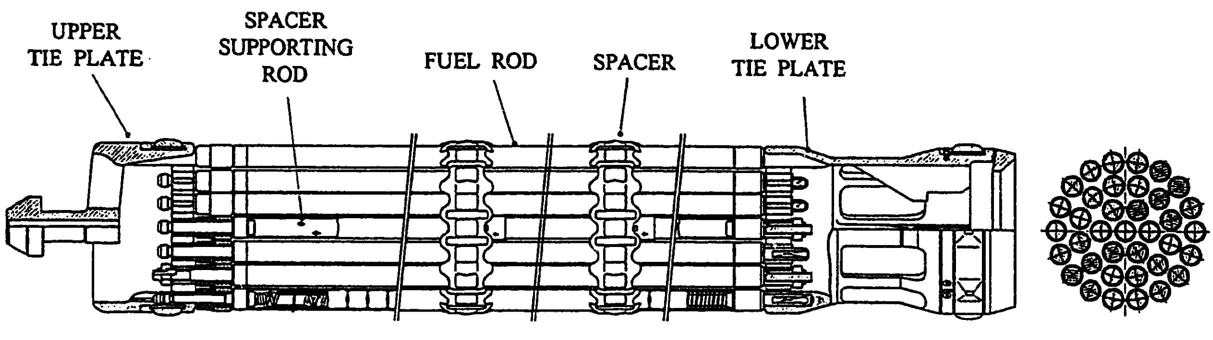


Fig. 1 Schematic drawing of an ATR 36-rod fuel assembly

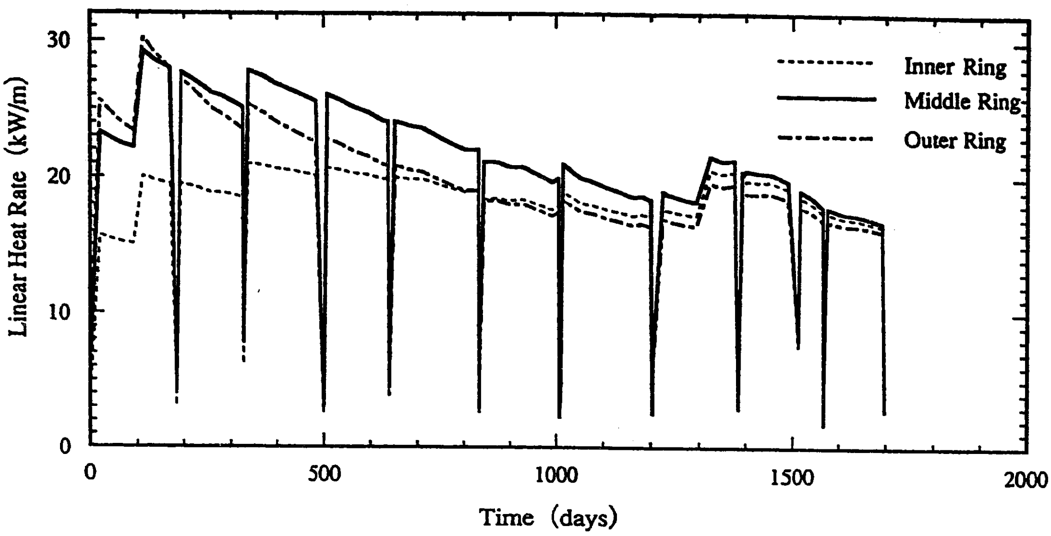


Fig. 2 Irradiation history of Fugen E05 fuel assembly

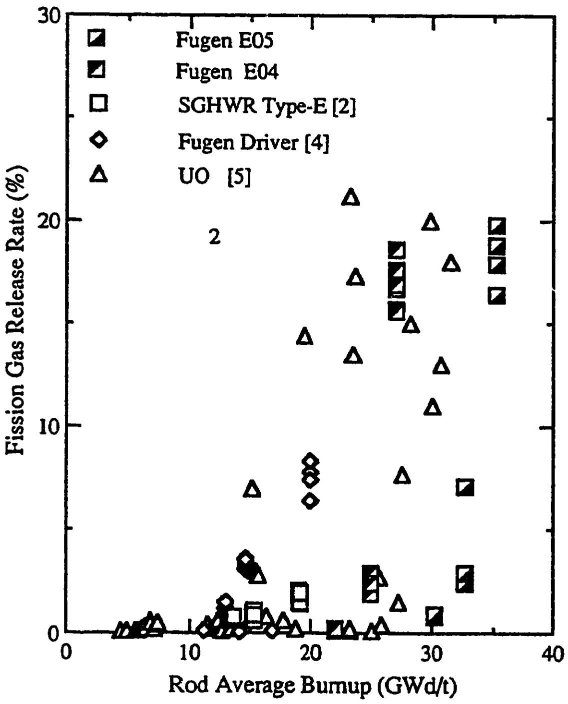


Fig. 3 Fission gas release rate as a function of burnup

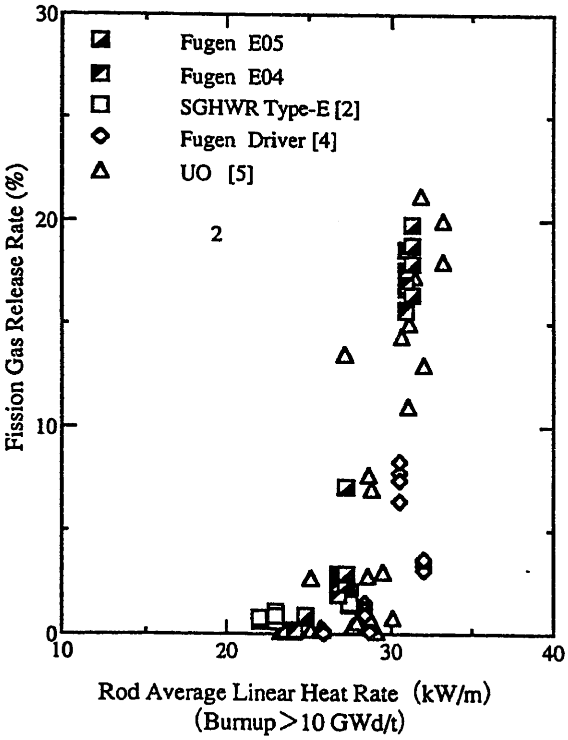


Fig. 4 Fission gas release as a function of linear heat rate

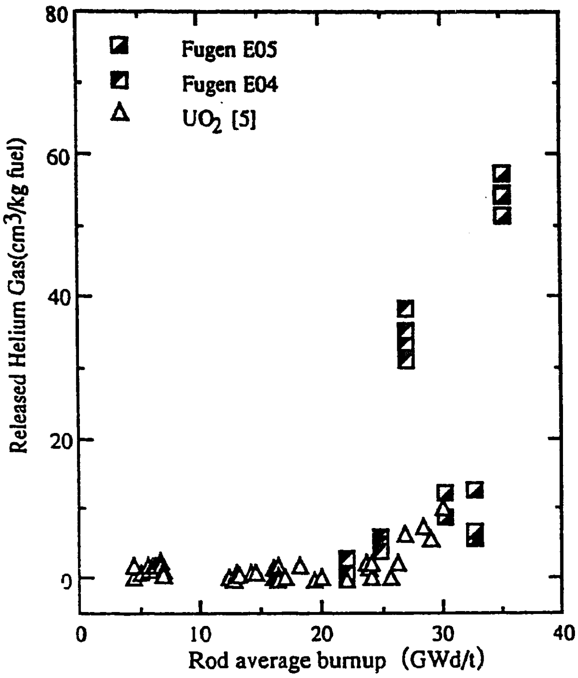


Fig. 5 He gas release as a function of rod average burnup

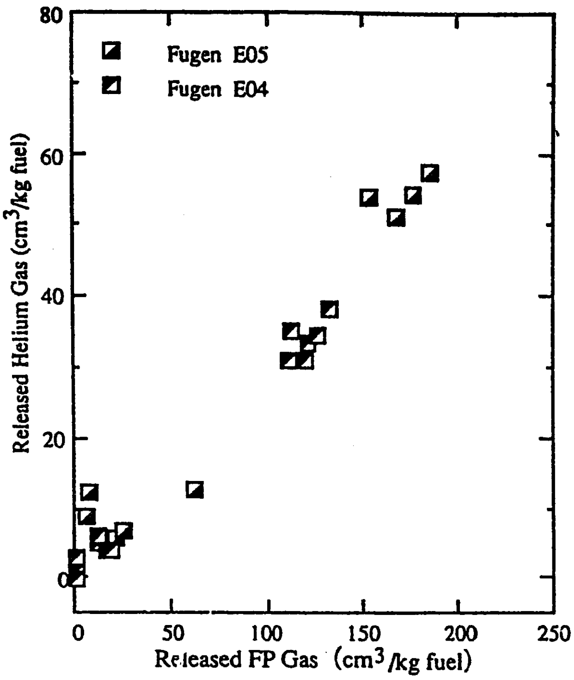


Fig. 6 Comparison between He gas and FP gas release

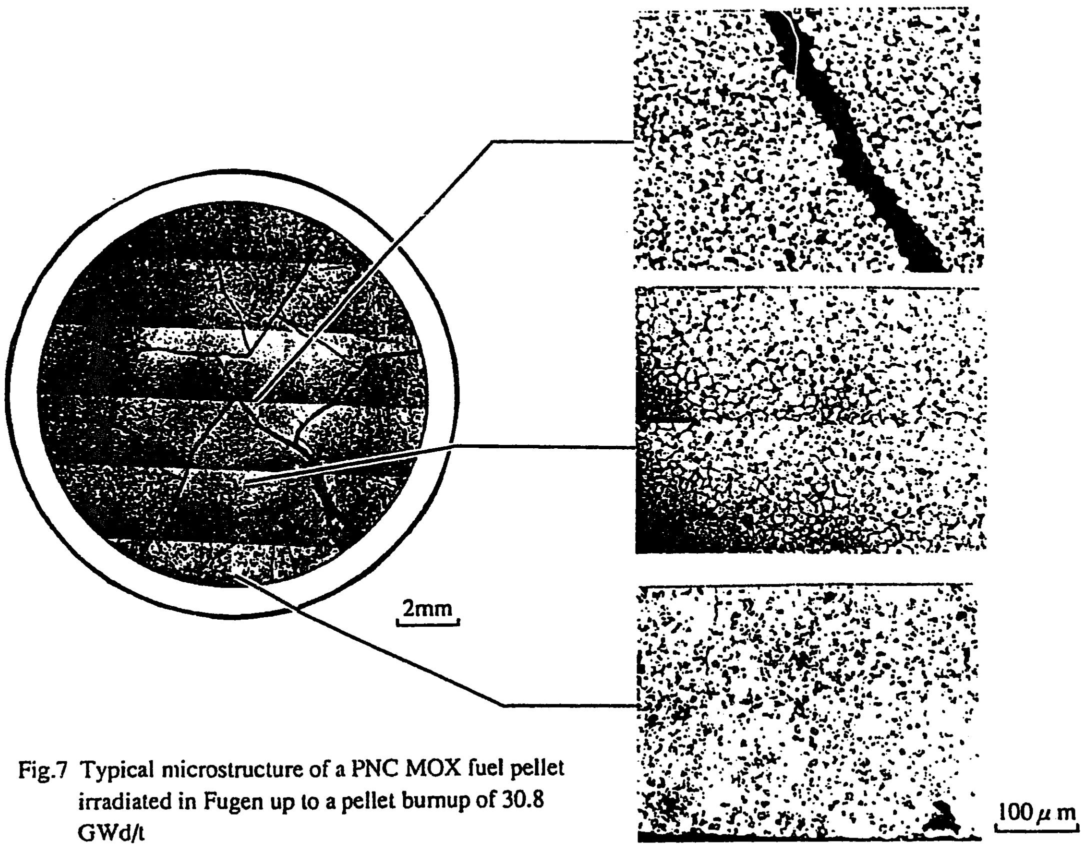
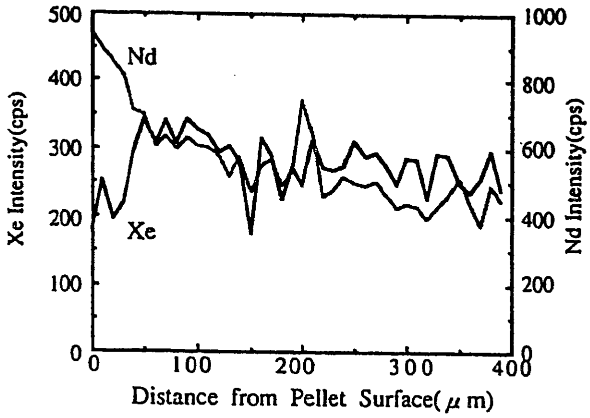


Fig.7 Typical microstructure of a PNC MOX fuel pellet irradiated in Fugen up to a pellet burnup of 30.8 GWd/t



Xe and Nd Profiles by EPMA

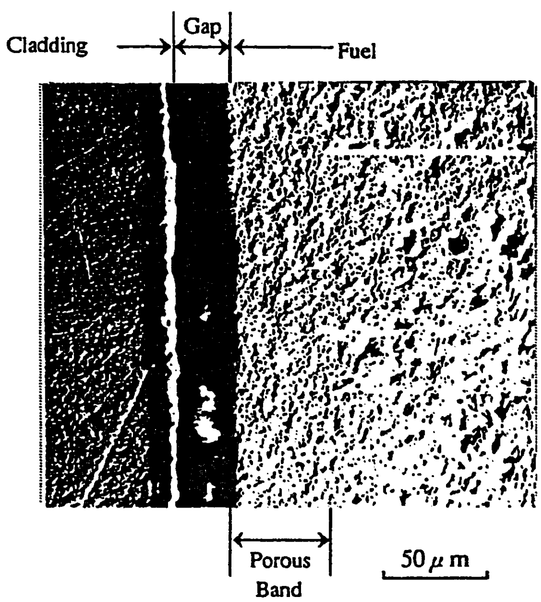
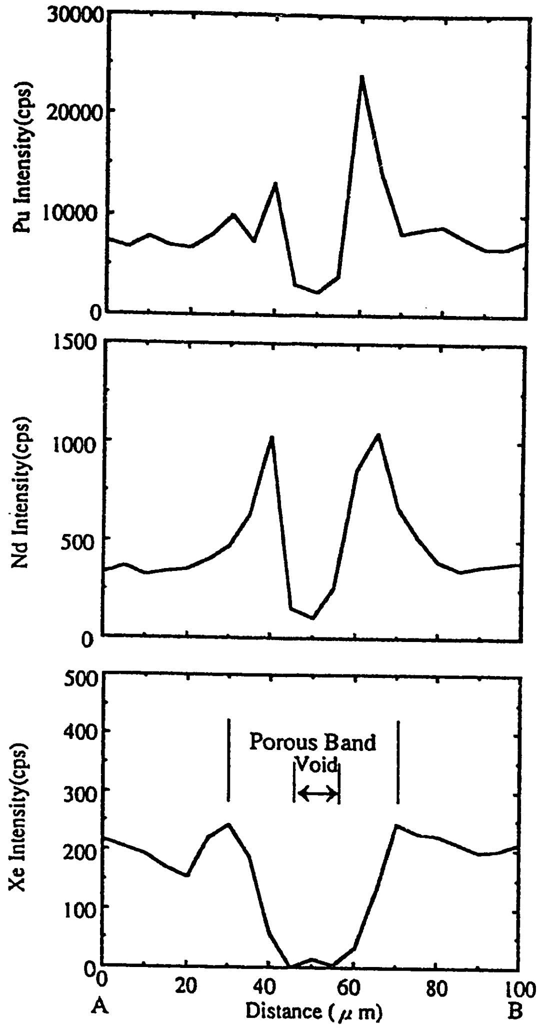


Fig.8 Scanning electron micrograph of pellet peripheral region, comparing with Xe and Nd profiles measured by EPMA



Pu, Nd and Xe Profiles by EPMA

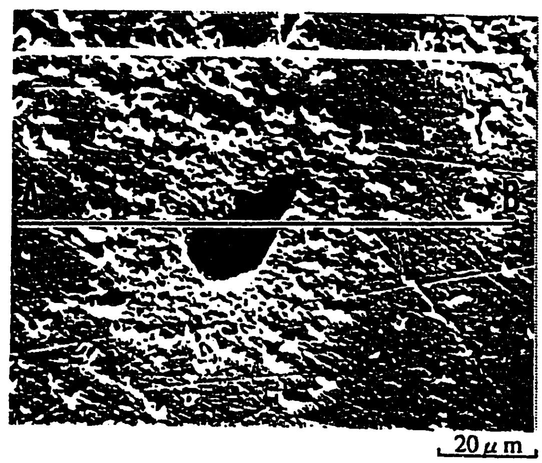


Fig.9 Fission products distribution in a Pu agglomerate

BEHAVIOR OF MOX FUEL IRRADIATED IN A THERMAL REACTOR

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Fuels & Materials Division

Oarai Engineering Center

Power Reactor & Nuclear Fuel Development Corporation

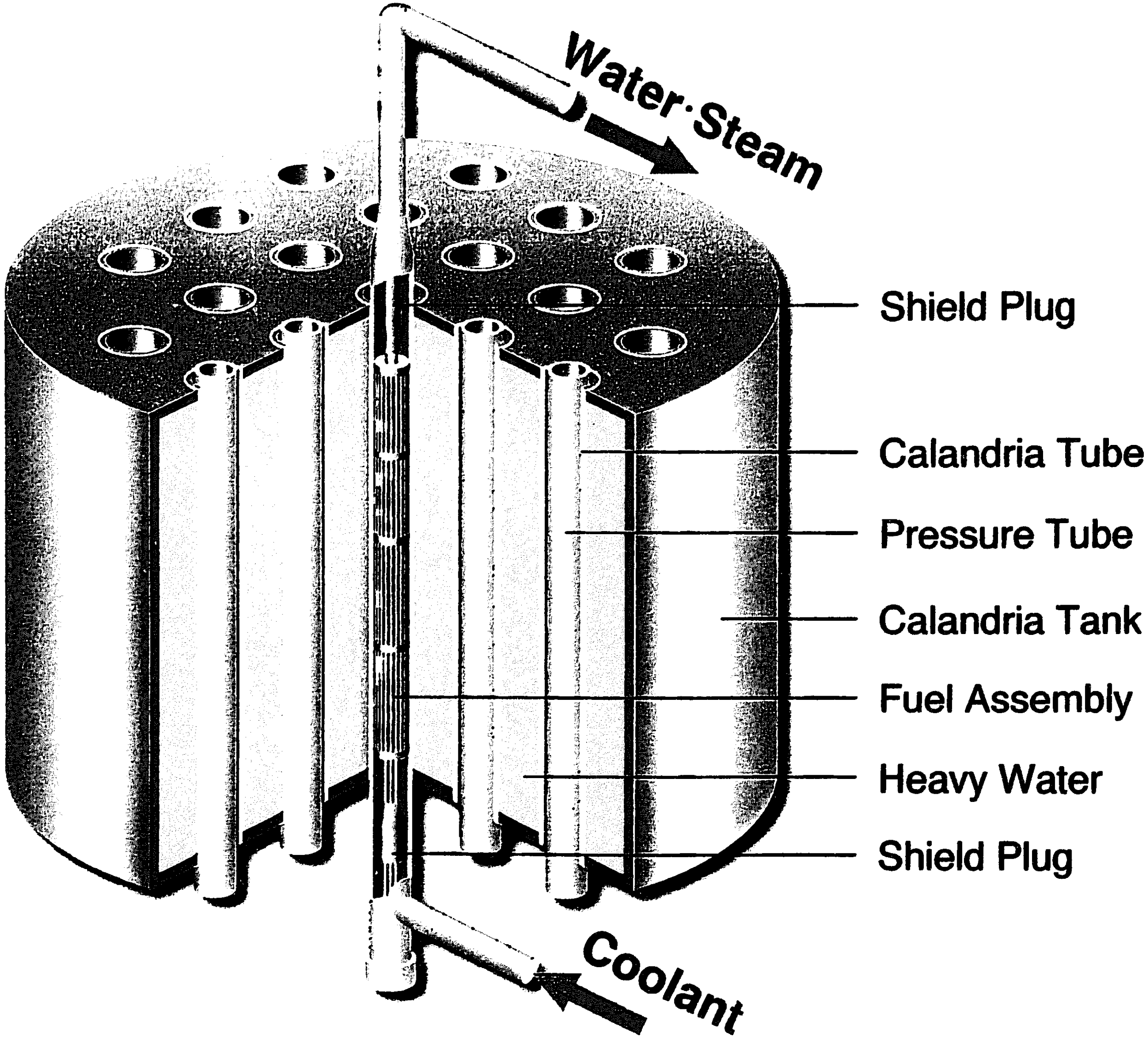
Introduction

- Basic policy : Plutonium recycle
  (Reprocessing, Fast Breeder Reactor, and Thermal Reactor)

- It is important
  - to consolidate the wide range of technological system related to plutonium recycle, proceeding to commercial use of FBRs
  - to consume plutonium comes from LWRs in operation.
  - to improve fuel cycle economy.

- PNC irradiated more than 16,000 MOX fuel rods in ATR Fugen since 1978, and has been developing high performance MOX fuel for water reactors.

Schematic View of Fugen Core



Development of High Burnup MOX Fuel for ATR

Original design

28-rod Type Fuel Assembly for Fugen Driver
- Maximum Assembly BU : 20 GWd/t
- Maximum Linear Heat Rate : 57.4 W/m

↓

1st Step

36-rod Type Fuel Assembly
- Maximum Assembly BU : 35 GWd/t
- Maximum Linear Heat Rate : 49.2 W/m
  - SGHWR : Type-E
  - Fugen : E03, E04, E05

↓

2nd Step

54-rod Type Fuel Assembly for Fugen Burnup Extension
- Maximum Assembly BU : 55 GWd/t
- Maximum Linear Heat Rate : 40 W/m
  - Fugen : Irradiation tests

Objectives

- To irradiate series of MOX fuel assemblies to high burnup.
- To obtain useful information by extensive post-irradiation examination.
- To evaluate irradiation behavior on high burnup MOX fuel by comparing with UO_2 fuel.
- To verify and to improve design procedure for high burnup MOX fuel.

Characteristics Interested in MOX fuel

Pu Addition →

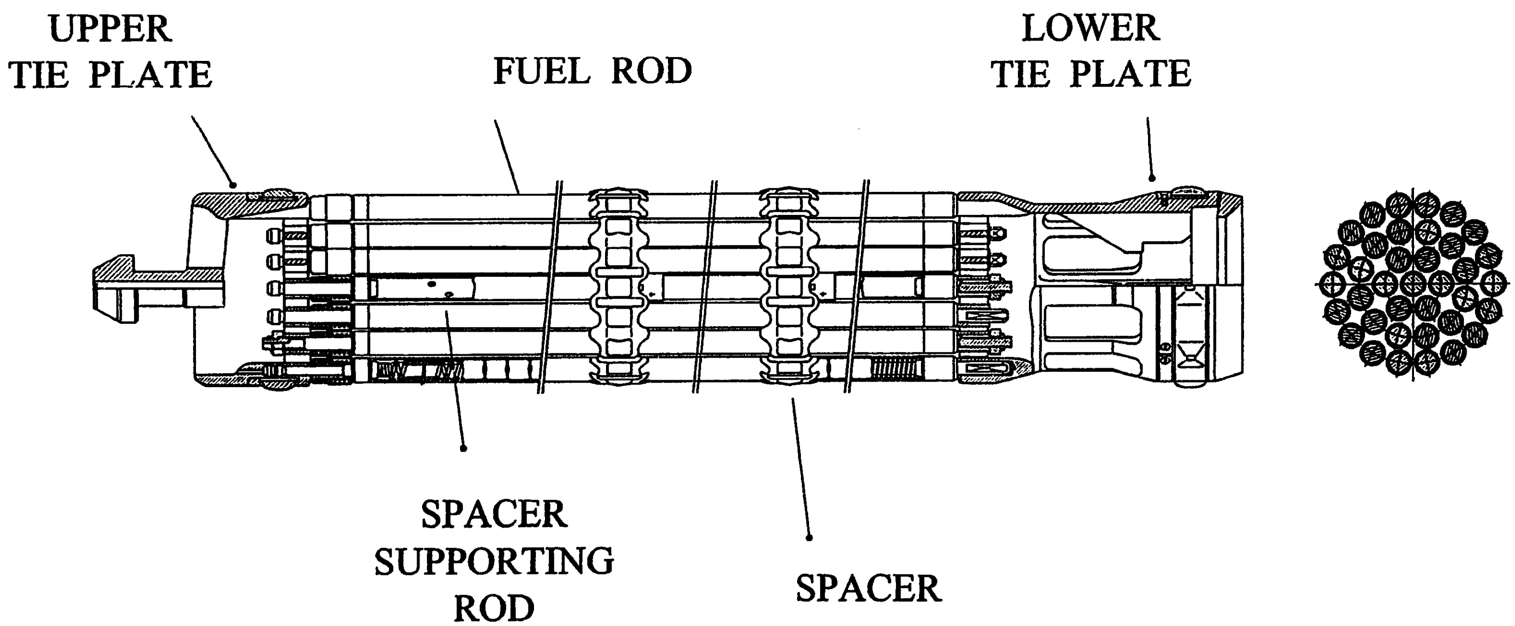
Fabrication Process →

Physical Properties

- Thermal Conductivity
- Melting Temperature
- <u>Oxygen Potential (Corrosion)</u>

Irradiation Behavior

- <u>FP Gas Release</u>
- <u>He Gas Release</u>
- Density Change
- <u>Microstrucutre Change</u>
- <u>Pu distribution</u>

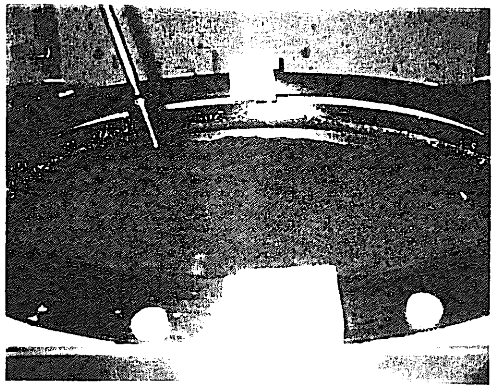


Schematic drawing of an ATR 36-rod fuel assembly

Summary of design specifications

Items	Fugen MOX Driver Fuel	ATR 36-rod fuel (E03, E04, E05)	*BWR (Typical 8x8)*
1. Fuel Pellet			
Outer Diameter (mm)	14.40	12.40	*10.6*
Density (%T.D.)	95	←	*95*
Pu Fissile Enrich (wt%)	0.4～1.62	0.98～2.45	-
2. Fuel Rod			
Cladding Material	Zircaloy-2	←	
Outer Diameter (mm)	16.46	14.50	*12.5*
Fuel Stack Length (mm)	3700	3647	*3660*
Filling Gas	He	←	←
& Pressure (MPa)	0.1	0.3	*0.1*
3. Fuel Assembly			
Length (mm)	4388	4398	*4470*
Bundle O. D. (mm)	111.6	←	
Rod Number	28	36	*63*
Inner	4	6	-
Intermediate	8	12	-
Outer	16	18	-

Pellet Fabrication Process



U, Pu Mixed Nitrate Solution

Co-conversion

Calcination

Reduction

$(U, Pu)O_2$ Powder

Microwave Heating Process

UO_2 Powder / $(U, Pu)O_2$ Powder

UO_2 Powder / PuO_2 Powder

Blending

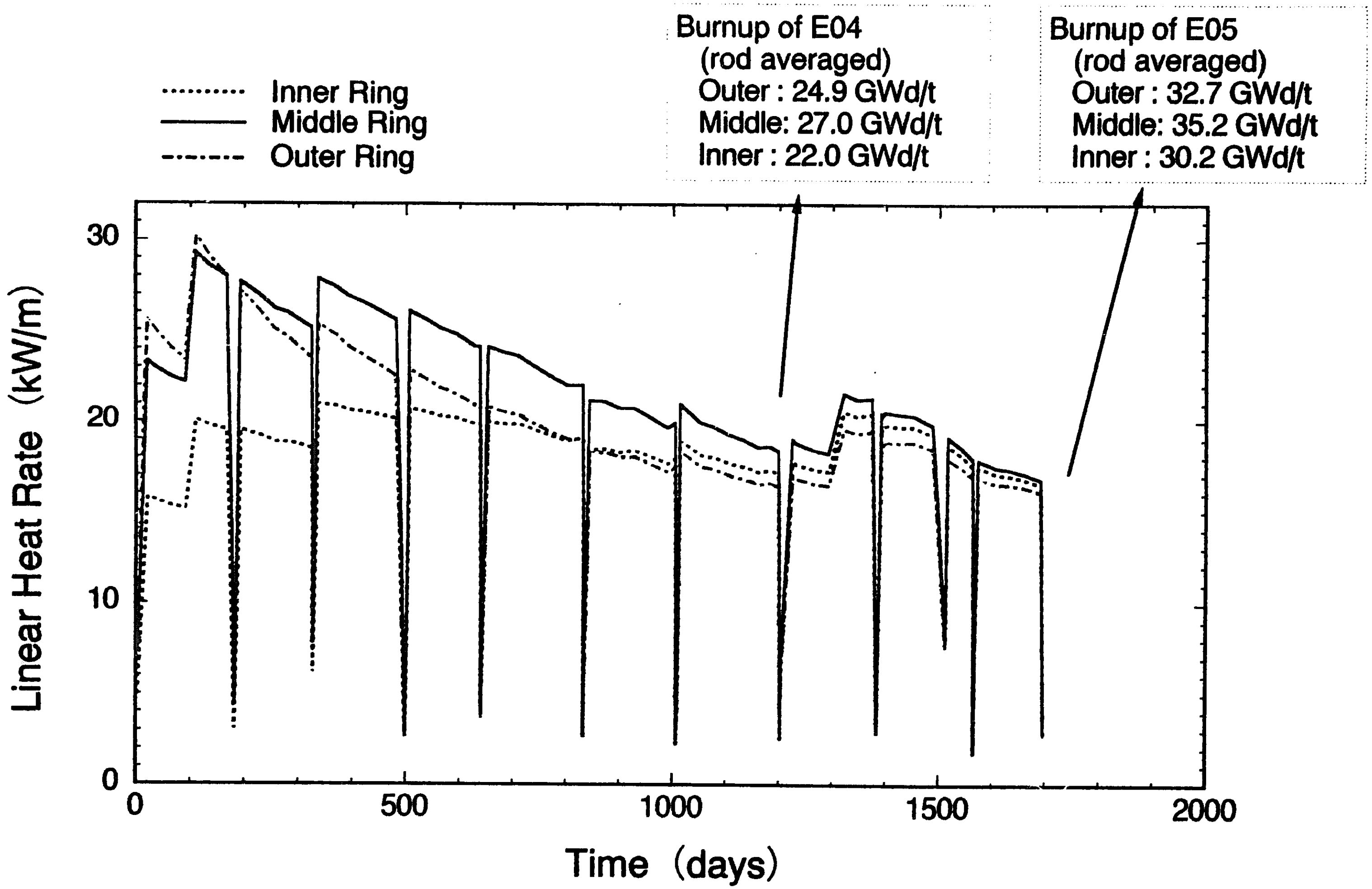
Ball Milling

Pressing

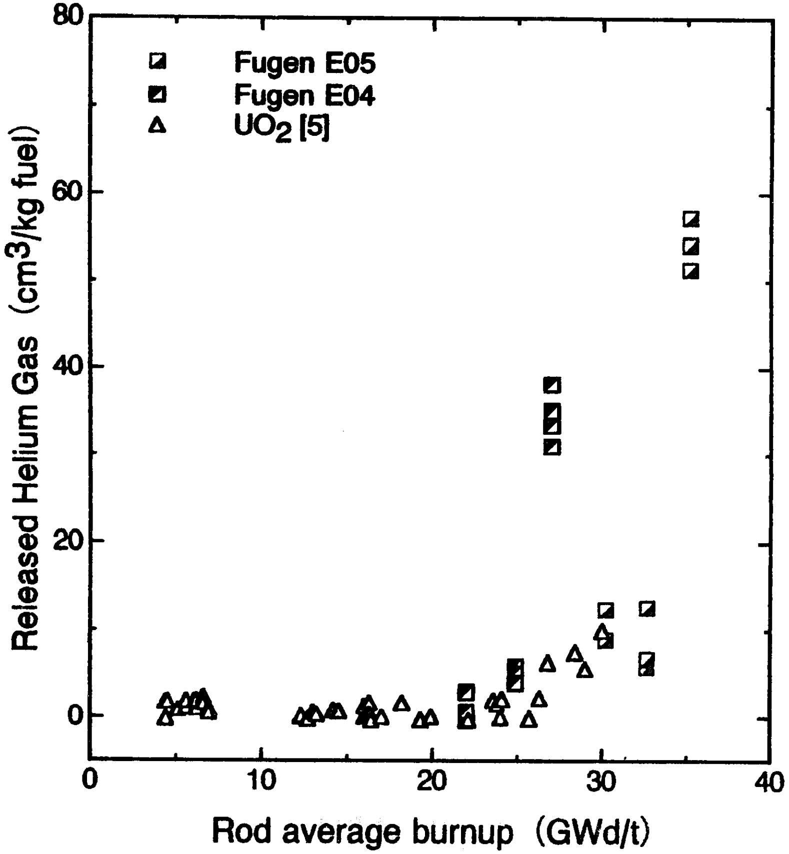
Presintering

Sintering

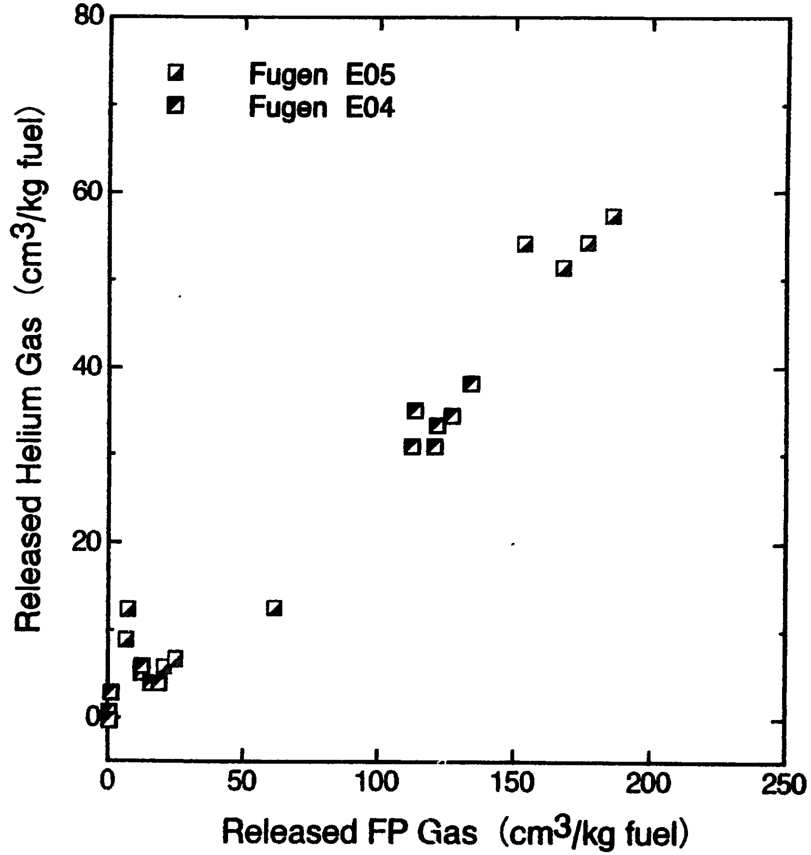
MH Pellet | MB Pellet



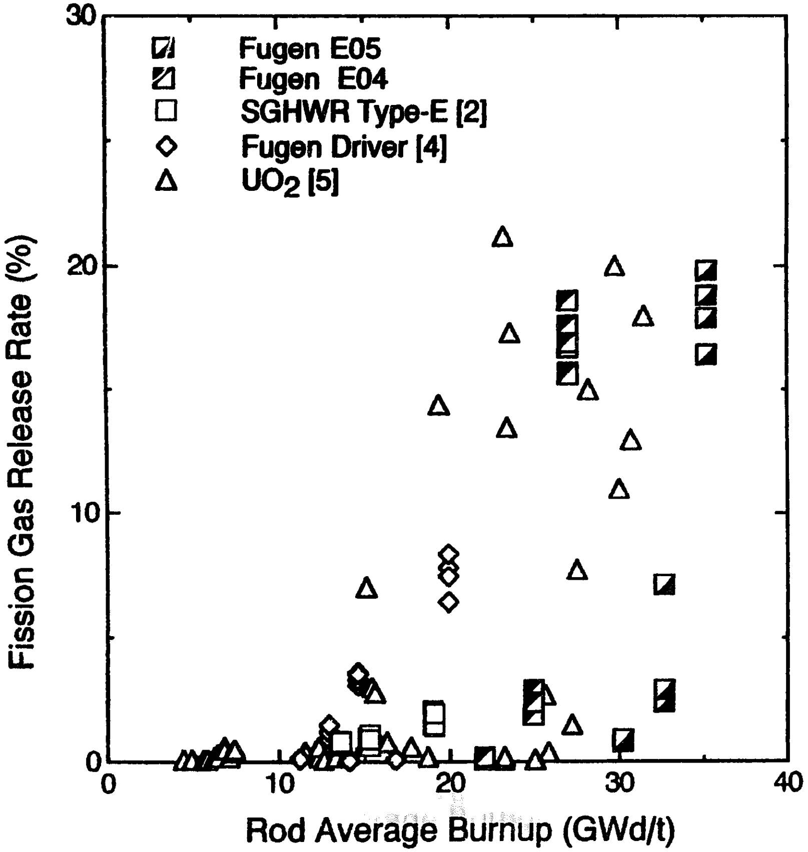
Irradiation history of Fugen E05 fuel assembly



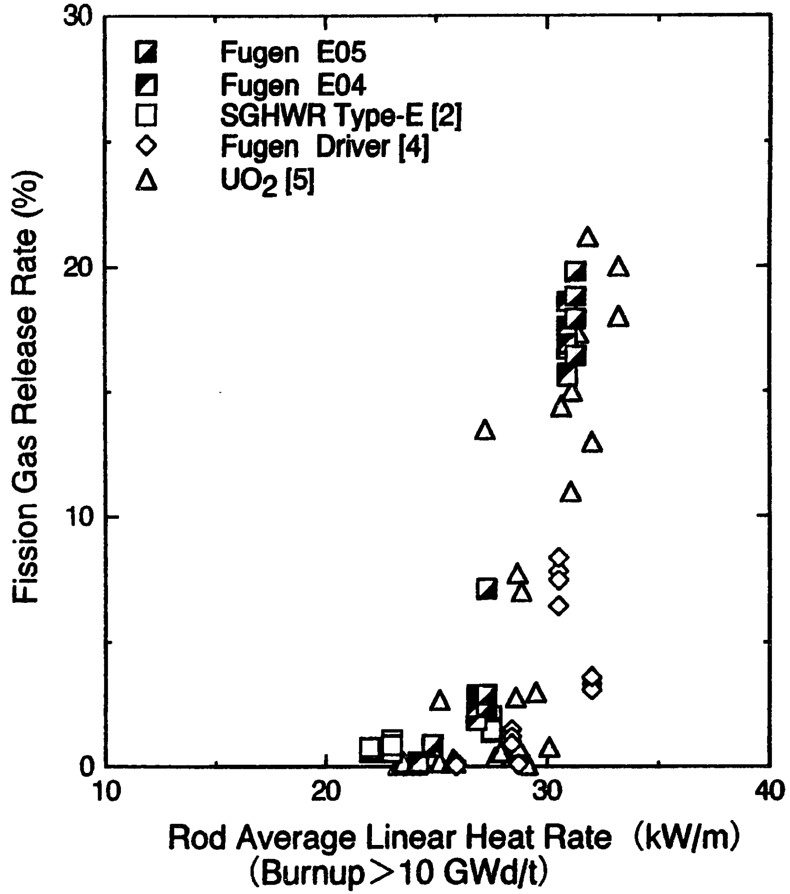
He gas release as a function of rod average burnup



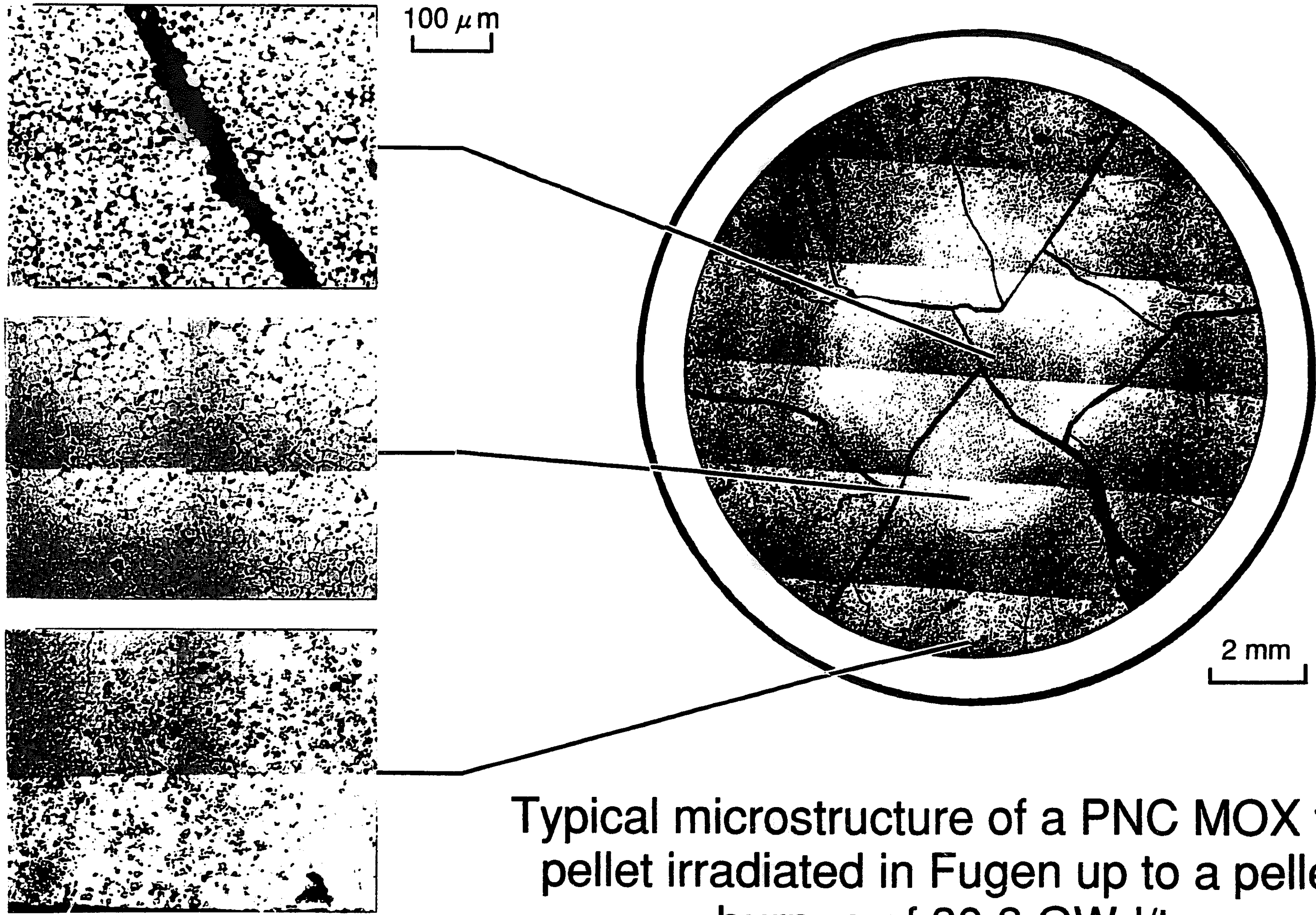
Comparison between He and FP gas release



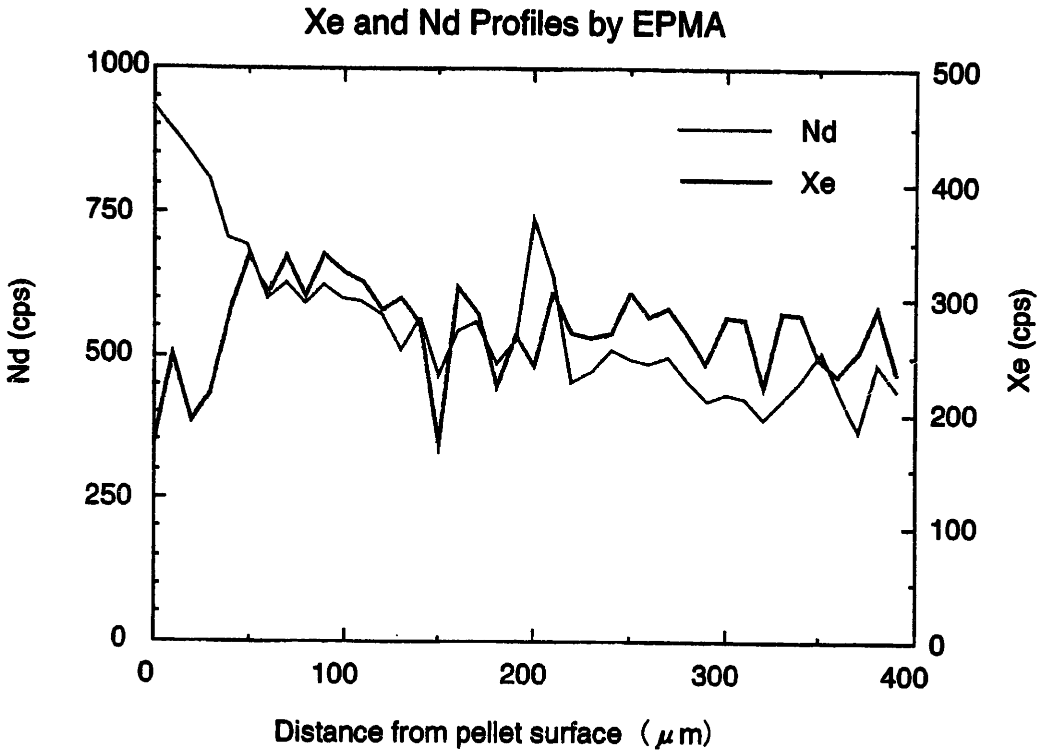
Fission gas release rate as a function of burnup

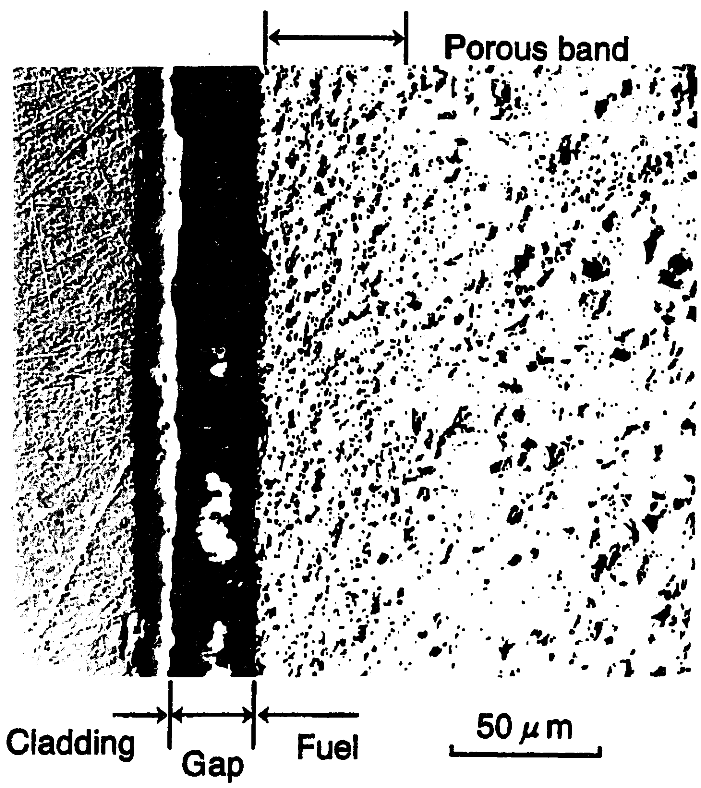


Fission gas release as a function of L. H. R.

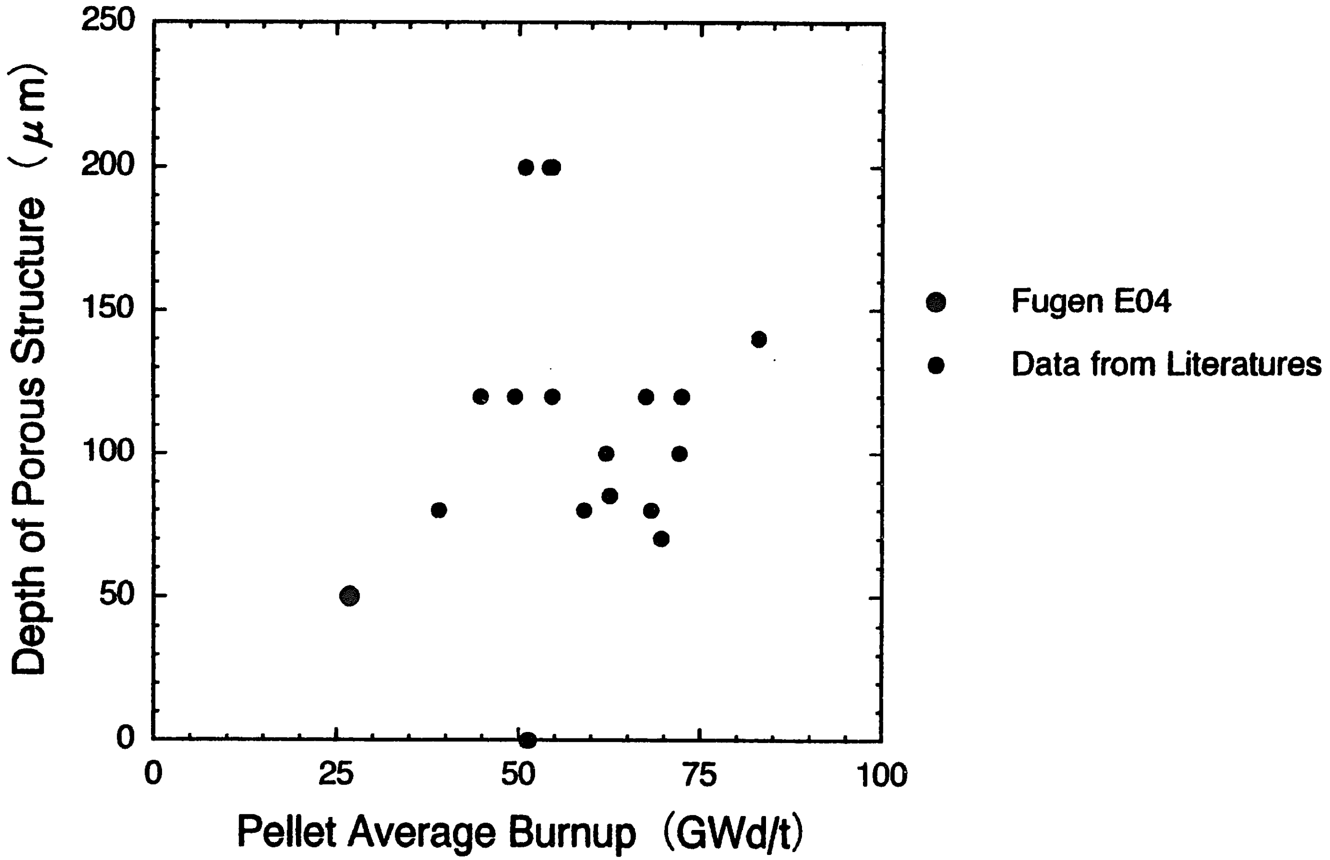


Typical microstructure of a PNC MOX fuel pellet irradiated in Fugen up to a pellet burnup of 30.8 GWd/t



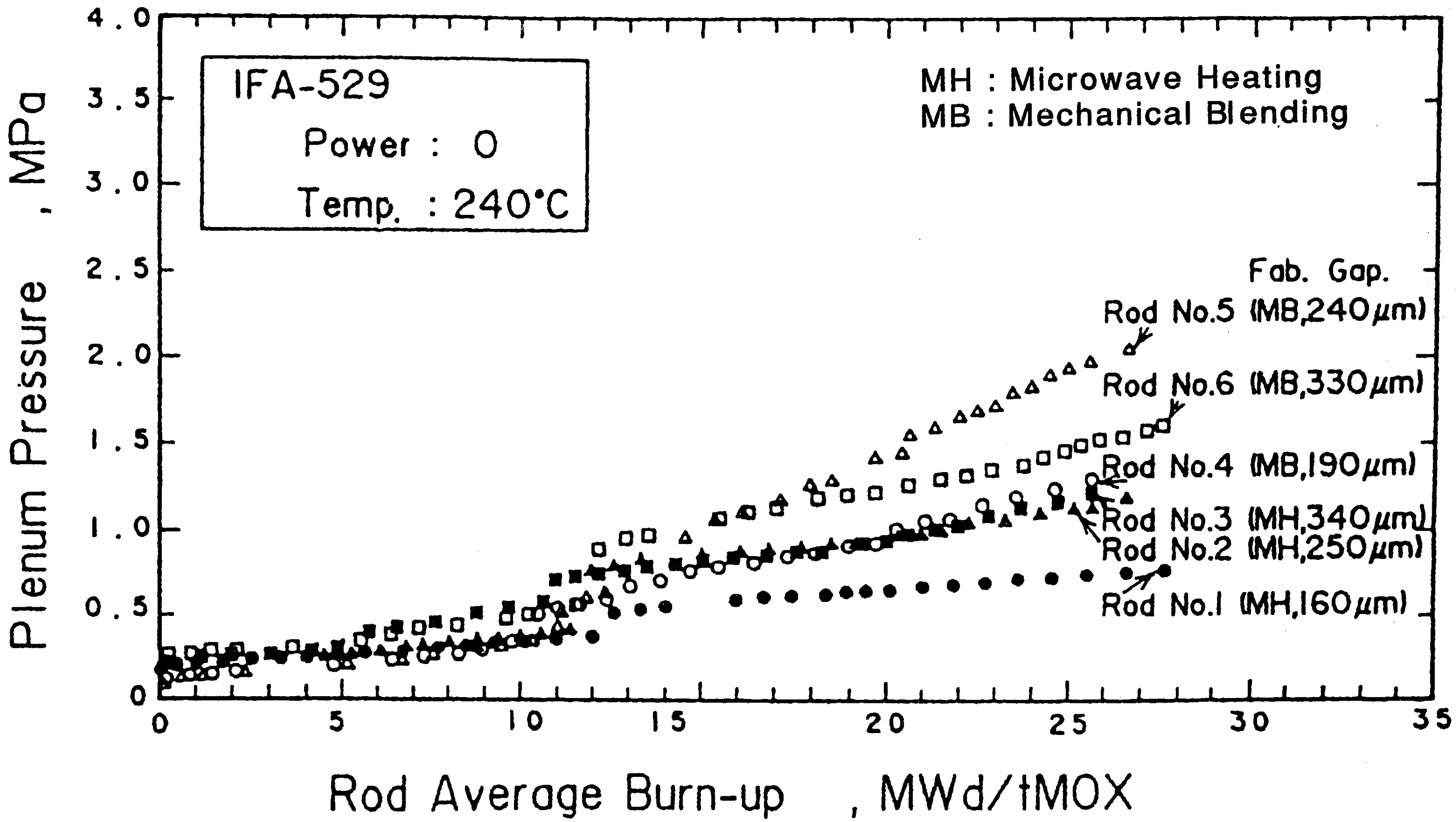


Scanning electron micrograph of pellet peripheral region, comparing with Xe and Nd profiles measured by EPMA

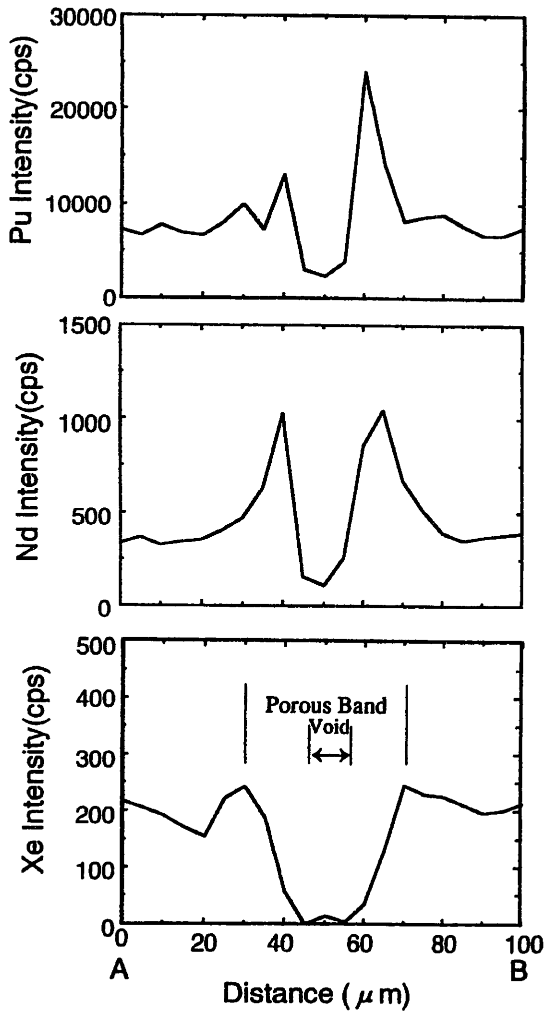


Depth of Porous structure as a function of pellet average burnup

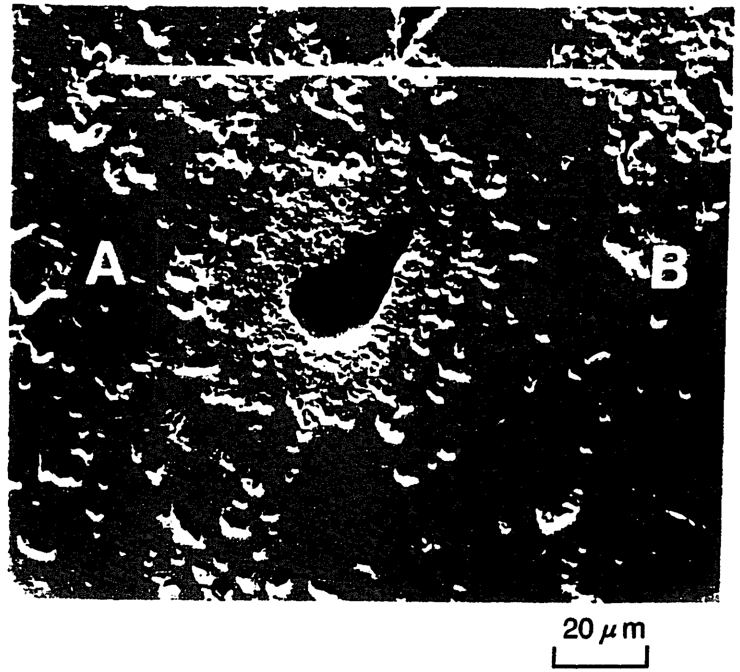
Reference : T. Mishima, et.al., IAEA Technical Committee Mtg. on Recycling of Plutonium and Uranium in Water Reactor Fuels, Cadarache, France, Nov. 13-16, 1989



In-reactor measurement of plenum pressure in HBWR



Pu, Nd and Xe Profiles by EPMA



Secondary electron image

Fission products distribution in a Pu agglomerate

Conclusion

- Irradiation behavior of PNC MOX fuel was investigated through series of irradiation tests up to an assembly burnup of 33.1 GWd/t. Evaluation of data accumulated showed that there was no distinguishable difference in irradiation behavior between MOX and UO_2 fuel except for helium release and Pu heterogeneity.

- MOX fuel releases helium more than UO_2 fuel does, since ^{242}Cm is generated faster in MOX fuel. However, it is confirmed that the internal pressure in a MOX fuel rod is predicted by the ATFUEL code conservatively up to a rod average burnup of 35 GWd/t.

- Though plutonium agglomerates with porous structure were observed, there detected no measurable effect on irradiation behavior of MOX fuel.

POWER RAMP TESTS OF MOX FUEL RODS FOR ATR (IFA-591)

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ABSTRACT

Plutonium-uranium mixed oxide (MOX) fuel rods of instrumented rig IFA-591 experiment were ramped in HBWR to study the Advanced Thermal Reactor (ATR) MOX fuel behavior during transient operation and to determine a failure threshold of the MOX fuel. Eleven segments were base-irradiated in ATR "Fugen" up to 18.4GWd/tM. Zirconium liner claddings were adopted for four segments of them. The segments were disassembled, non-destructive post irradiation examinations (PIEs) were performed before power ramp tests. All segments have instrumentations for in-pile measurements of cladding elongation or plenum pressure.

The following results were obtained. All segments were heated up to the maximum linear heat rating of 58.3~68.4kW/m without failure. Relaxations of cladding caused by axial deformation at high temperature and thermal feed back phenomena were observed. There are no differences between pellet-cladding mechanical interaction (PCMI) behaviors of two type rods of Zry-2 and Zr liner claddings. It seems that fission gas release are caused through three steps as a rapid, a continuous and an additional release. Fuel rod behavior analysis code 'FEMAXI-ATR' is validated under transient conditions.

I. INTRODUCTION

The ATR is a heavy water moderated and light water cooled reactor developed by Power Reactor and Nuclear Fuel Development Corporation (PNC) in Japan. A main feature of ATR is its flexibility on fuel utilization. A large number of MOX fuel assemblies have been, and are being irradiated in the ATR prototype plant "Fugen". PNC has been developing MOX fuels for thermal reactors.

The objectives of the power ramp tests in the instrumented rig IFA-591 are to study the ATR MOX fuel behavior during transient operations and to determine a failure threshold of the MOX fuel. An assembly with eleven segment rods for the power ramp tests was irradiated in Fugen up to 18.4 GWd/tM. Zirconium liner claddings were adopted for four segments of them. The segments were disassembled, non-destructive PIEs were performed before the power ramp tests.[1~4] The ramp conditions as a ramp speed, a hold time, etc. were determined more severely than considerable design conditions at transient cases. Ramp modes were (A) multi step test for six segments and (B) single step test for five segments. Each segment has an instrumentation for in-pile measurement of cladding elongation (EC) or plenum pressure (PF). In this paper, PCMI and fission gas release behaviors during the power ramp tests are discussed with the EC and PF data. The experiment of IFA-591 was carried out as a part of joint research program between PNC and Japan Atomic Energy Research Institute (JAERI) with the participation in the OECD Halden Reactor Project.

II. EXPERIMENT

Figure 1 shows the program time schedule.

A. Fuel Rods

The assembly equipped with eleven segment rods for the power ramp tests was fabricated by PNC Japan. These rods contained MOX pellets enriched to 3.71wt% plutonium fissile content, having a density of 95% of TD. Design parameters of the rods are listed in table 1. The rods to be discussed are from IFA-591-1 (rod No.1) to -11 (rod No.11). Rods of IFA-591-2,-5,-8,-9 consisted of Zr liner claddings. Positions of the segment rods in the assembly are shown in figure 2. The segment rod features are shown in figure 3.

B. Irradiation

The assembly was loaded in the core of Fugen, operated by PNC, in 1987. It was irradiated under normal ATR conditions. Achieved burnups of the segment rods are listed in table 2.

C. Inspection before Power Ramp Tests

After the base irradiation at Fugen, non-destructive PIEs for the segment rods were performed to confirm to be sound at JAERI and Kjeller Laboratory.

D. Power Ramp Tests

The power ramp tests were performed in IFA-591, which was equipped with a ^{3}He local power controller (the ^{3}He method) and hydraulic cylinders for moving the rods between the upper (low flux) position and the lower (high flux) position (the drop method). The rig of IFA-591 has three rods at each loading which were ramped in order. It was set in the special loop kept at ATR coolant conditions, that is, 286℃ and 7.2 MPa during the power ramp tests.

The eleven rods were power ramp tested in the Halden Reactor. Two typical power ramp test sequences are shown in figure 4. Ramp sequence A is a multi step ramp by the ^{3}He method for four rods of Zry-2 and two rods of Zr liner cladding, and Ramp sequence B is a single step ramp by the drop method for three rods of Zry-2 and two rods of Zr liner cladding.

III. RESULTS AND DISCUSSION

A. Inspection Result

It was observed 1) that there were no remarkable defects and deformations of claddings by visual examination, dimensional measurement and eddy current testing, 2) that axial burnup distributions of each fuel were almost constant without peak by gamma scanning tests , 3) that stacks of MOX pellets were sound with no fragments by neutron radiography. So it was confirmed that all segment fuel rods were to be suitable for the power ramp tests.

B. Result of Power Ramp Tests

Burnups for the power ramp tested rods ranged from 14.8 to 22.2 GWd/tM and ramp terminal powers ranged from 58.3 to 68.4 kW/m. The power ramp test results are shown in figure 5. All eleven rods were sound during power ramp tests.

C. Safety Evaluation

All segments were heated up to the ramp terminal power of 58.3~68.4kW/m without failure. It is confirmed that the current design of MOX fuel is conservative for the transient to a maximum burnup of 22.2GWd/tM.

To compare with results of the latest BWR UO_2 fuel,[5] the ramp sequence A' and B' for the BWR UO_2 fuel are shown in figure 6, and its results are shown in figure 7 overlapped the results of ATR MOX fuel. It was observed that the ATR MOX fuels were sound at higher terminal power. In figure 7, the threshold for the ATR MOX fuel was higher than the BWR UO_2 fuel ranged from 15.5 to 22.2 GWd/tM, because the threshold line for the BWR UO_2 fuel could be drawn at about 50kW/m between sound and failed marks and the ATR MOX fuels were sound over the level.

D. Pellet-Clad Mechanical Interaction Behavior

Figure 8 shows the EC result of IFA-591-3 (Zry-2, 21.2 GWd/tM) during the power ramp test. The ramp terminal power reached to 62.2kW/m of seven steps. It was observed a rapid elongation right after the ramp and then a relaxation. Figure 9a shows the rapid elongation in detail at fourth step. Figure 9b shows the thermal feedback phenomena in detail at third step. After the rapid elongation, the rod was slowly elongated to 80 μ m for 150 seconds. This power level was in agreement with a rapid fission gas release. During a power decrease after the power ramp test, it was observed that a elongation rate of cladding grew lower at 55kW/m. Its power level was in agreement with an additional fission gas release. Figure 10 shows the additional elongation by the thermal feedback phenomena was observed during a power decrease after the ramp sequence B for IFA-591-7. Concerning of shrinkage of the cladding with temperature decrease, the additional elongation is assumed about 100 μ m caused by PCMI relaxation. The time for the additional elongation is about one hour and is longer than a case of the third step of the power ramp, because of lower temperature. Pellet-cladding mechanical interaction behaviors of two type rods of Zry-2 and Zr liner claddings were almost the same.

Figure 11 shows a relaxation ratio (relaxation /elongation) at each power ramp for IFA-591-1 to -3. The relaxation ratio was increased with linear heat rating, more than 80% was relaxed over 55kW/m. Relaxations were caused by axial creep deformation of pellet and slip between the cladding and pellet at high temperature. Relaxation rate didn't depend on burnups and kind of claddings.

Figure 12 shows a relationship between EC and coolant temperature. During the power decrease after the power ramp test, it was observed that the elongation rate of cladding below 55kW/m was almost linear and parallel to it during the power increase before the power ramp test. The difference between them caused a plastic deformation of cladding as of 53 μ m (0.01%) by PCMI. The other rods' deformations were small as 0.01%.

E. Behavior of FP Gas Release

Fission gas release rates estimated by the PF instrument during the power ramp tests were 42.5, 41.1, 37.2% for each IFA-591-4,-5,-6. The rate during the base irradiation were estimated 0.03 to 0.19% by puncturing test for other fuels in the same assembly. Total fission gas release rate was less than 45%.

Figure 13 shows the PF of IFA-591-6 (Zry-2, 22.2GWd/tM) during the power ramp tests. The ramp terminal power reached to 66.8kW/m of eight steps. There were three parts:

1. A rapid release. At the first step, the PF had no change. At the second step, the PF increased rapidly. It seems that the pellet temperature was over an onset of the gas release from grain boundaries and pores of crystal.

2. A continuous release. After the third step, the PF continuously increased although the power increased stairly. It seems that the fission gas released from inside of grains caused by diffusion and change of microstructure.

3. An additional release. It was also observed that the additional fission gas release caused by PCMI relaxation at 55kW/m during power down. The additional fission gas release rate is 20~30% of total fission gas release. It is saturated for about one hour.

Figure 14 shows that the onset powers of the rapid release are 40kW/m for IFA-591-4 (16.3GWd/tM) and 35kW/m for IFA-591-6 (22.2 GWd/tM). Higher burnup rod has lower threshold power of the rapid release in this power range.

Figure 15 shows that the FP gas release rates before the power ramp tests are the same or lower than the other test results.[5-7] The rates were estimated 0.03 to 0.19% by puncture test of the other fuel rods of the same assembly. The rates after the power ramp tests are overlapped in the same figure. These data are on a trend of the other points.

There are no differences between behaviors of the fission gas release of two type rods of Zry-2 and Zr liner claddings.

F. Validations of Fuel Rod Behavior Analysis Code 'FEMAXI-ATR'

The FEMAXI-ATR code has been developed for analysis of fuel rod irradiation behavior.[8-9] Material property models and irradiation behavior models of MOX fuel are considered in this code. And, under the transient condition, that is power ramp, there are some transients behavior models. The model that makes roughness of pellet surface smooth by pellet and cladding contact pressure, and the model that rod internal gas flows to rod axial direction by difference of FP gas concentration and difference of FP gas pressure,-and other transient models are considered.

Validations of the code under the transient condition were carried out using the ramp test results. Figure 16 shows an example of validation result for rod internal gas pressure. The trend of calculated values at each ramp step resembles to that of measured ones as follows. 1) Increasing of internal gas pressures during increasing of powers were small except second step. 2) Increasing of internal gas pressures during fixed powers and power down were large. In the cases of behaviors of rod elongation, measured values and code calculated values show good agreement, too.

IV. SUMMARY

The following results were obtained by the power ramp tests;

(1) All eleven segments were heated up to the maximum terminal power of 58.3~68.4kW/m without failure. It was confirmed that the current design of MOX fuel was conservative for the transient to a maximum burnup of 22.2GWd/tM.

(2) Relaxations caused by axial deformation at high temperature were observed. Thermal feed back phenomena were observed at multi and single step modes. Plastic deformation of cladding by PCMI was small as 0.01%. There were no differences between PCMI behaviors of two type rods of Zry-2 and Zr liner claddings.

(3) It seems that fission gas release are caused through three steps as (A) a rapid release from grain boundaries and pores of crystal, (B) a continuous release from inside of grains caused by diffusion and change of microstructure, and (C) an additional release caused by PCMI relaxation during power down.

(4) Fuel rod behavior analysis code 'FEMAXI-ATR' is validated under the transient condition.

ACKNOWLEDGMENTS

Authors gratefully appreciate staff members of the Halden Reactor Project for their endeavor to fuel power ramp tests, staff members of JAERI and Kjeller Laboratory for their contribution to PIEs and also Messrs. M.Kato, K.Kikuchi, T.Iijima of PNC for their contribution to co-evaluate ramp results.

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Table 1 Design parameters of the segment fuel rod

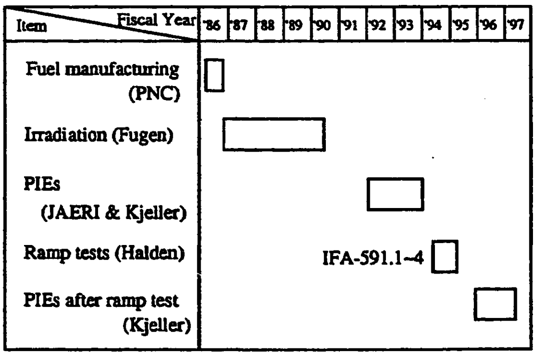
Item	IFA-591 fuel
1.Fuel Pellet	
Type	Sintered MOX
Form	Solid with dish and chamfer
Outer Diameter	12.4mm
Height	13.0mm
Density	95%TD
Pu Enrichment ($^{239}Pu+^{241}Pu$)/(Pu+U)	3.71%
2.Fuel Cladding	
Material	Zry-2 / Zry-2 with Zr liner
Outer Diameter	14.5mm
Wall Thickness	≥0.82mm
Zr Liner Thickness	0.075mm
3.Fuel Rod	
Rod Length	520mm
Pellet Stack Length	365mm
He Pressure	0.3MPa

Table 2 Achieved burnups of segment rods

Segment Rod No.	Burnup GWd/tM	Max. Linear Heat Rating kW/m	Ramp Test		Loading No.
			Ramp Sequence	Terminal Power kW/m	
IFA-591-1	15.5	18.1	Multi	65.3	IFA-591.1
-2*	15.5	18.1	Multi	66.3	IFA-591.1
-3	21.2	23.7	Multi	63.2	IFA-591.1
-4	16.3	18.1	Multi	66.4	IFA-591.2
-5*	16.3	18.1	Multi	68.4	IFA-591.2
-6	22.2	23.7	Multi	66.8	IFA-591.2
-7	16.3	18.1	Single	65.5	IFA-591.4
-8*	14.8	18.1	Single	68.0	IFA-591.4
-9*	16.6	18.1	Single	65.2	IFA-591.3
-10	21.2	23.7	Single	59.4	IFA-591.3
-11	22.2	23.7	Single	58.3	IFA-591.3

* : Zr liner cladding

Fig.1 Program time schedule



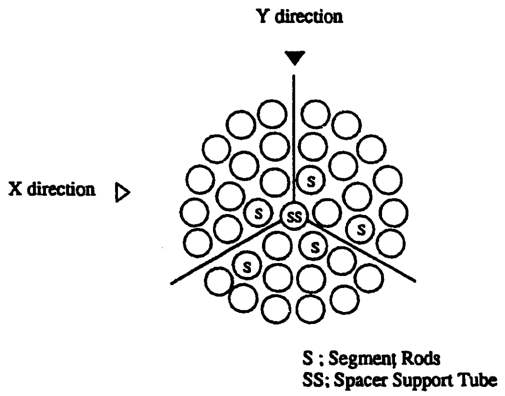


Fig.2 Positions of the segment rods in the assembly

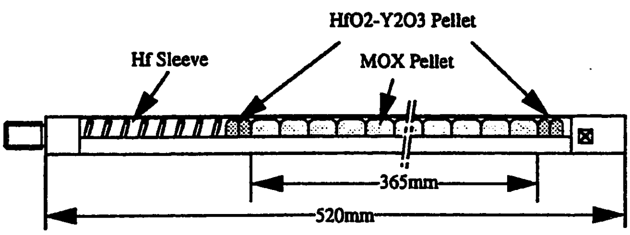


Fig.3 Schematic of the segment rod

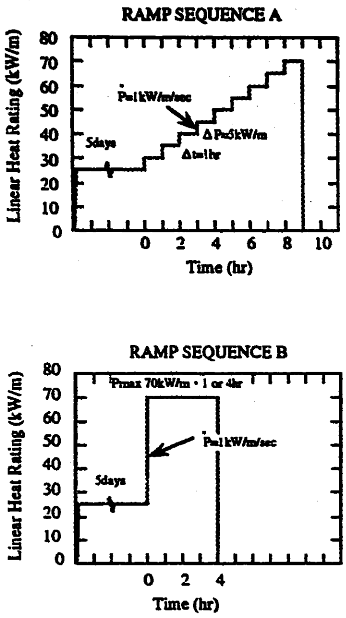


Fig. 4 Ramp Sequence

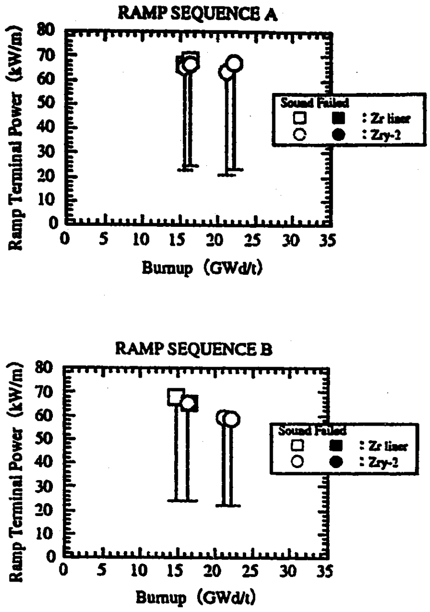


Fig. 5 Ramp Test Results

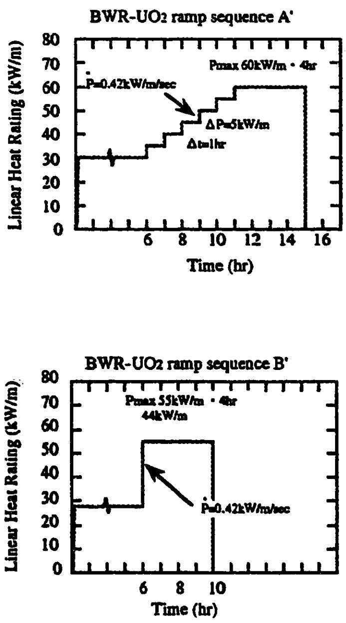


Fig.6 Ramp Sequence for BWR UO2 fuel 6)

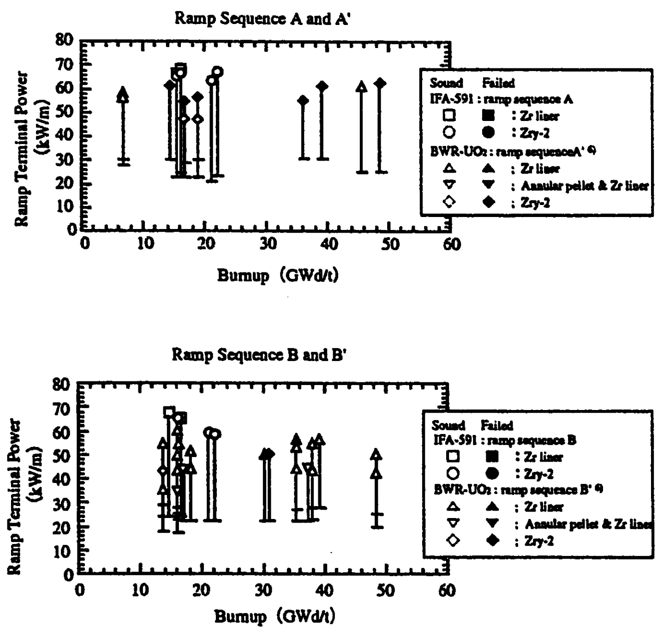


Fig.7 Ramp Test Results for ATR MOX (IFA-591) compared with BWR UO2 fuel 6)

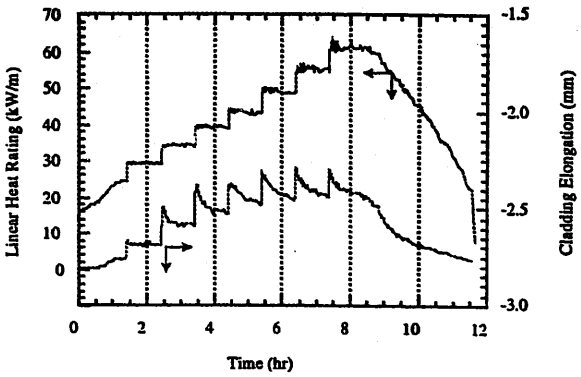


Fig.8 Linear heat rating and cladding elongation of IFA-591-3 as a function of time

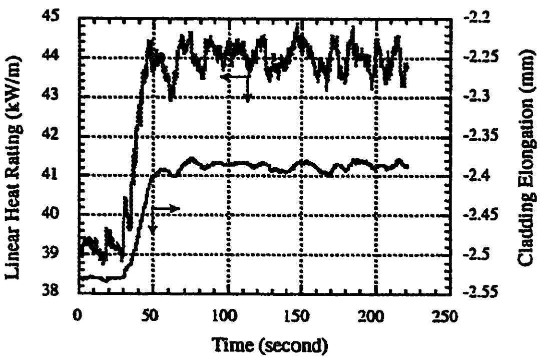


Fig. 9a Linear heat rating and cladding elongation of IFA-591-3 step4 as a function of time

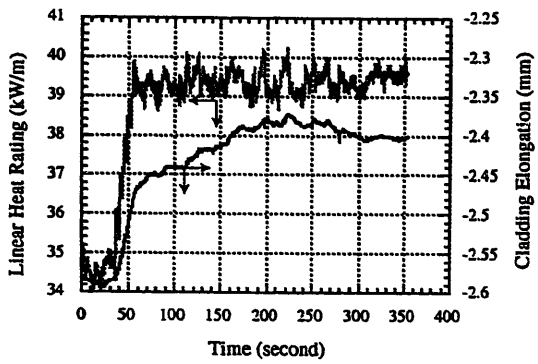


Fig. 9b Linear heat rating and cladding elongation of IFA-591-3 step3 as a function of time

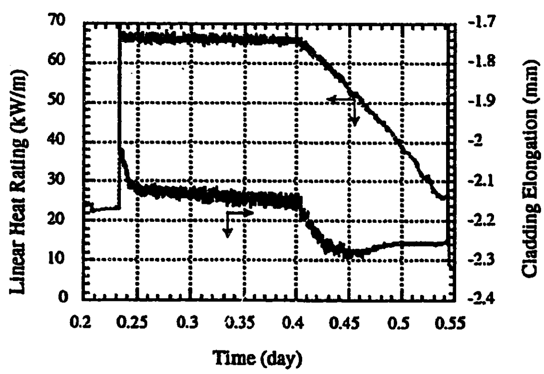


Fig.10 Linear heat rating and cladding elongation of IFA-591-7 as a function of time

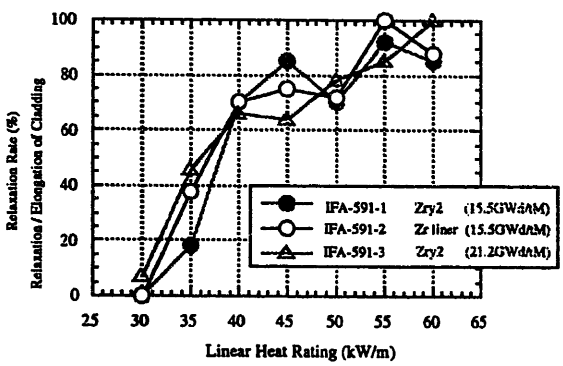


Fig.11 Relaxation of cladding as a function of linear heat rating

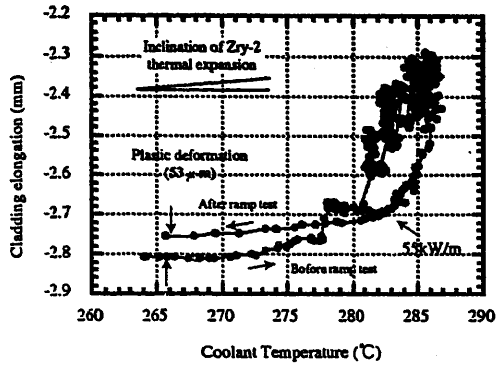


Fig.12 Cladding elongation of IFA-591-3 as a function of coolant temperature

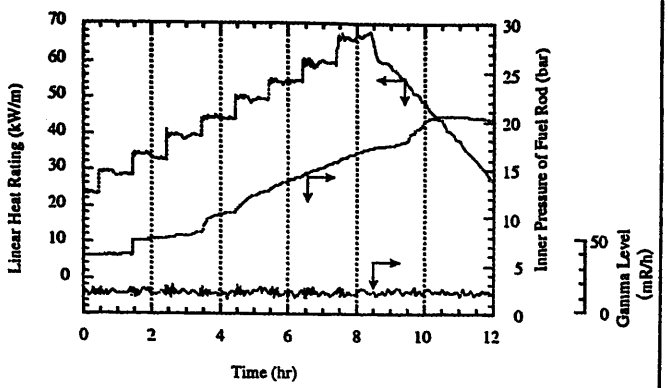


Fig.13 Linear heat rating and pressure of IFA-591-6 as a function of time

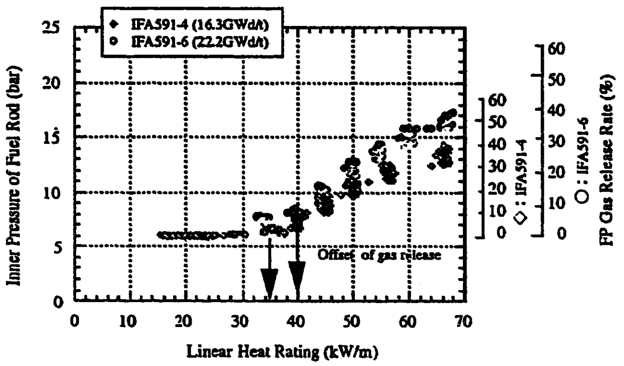


Fig.14 FP gas release of IFA-591-4 and -6 as a function of linear heat raing

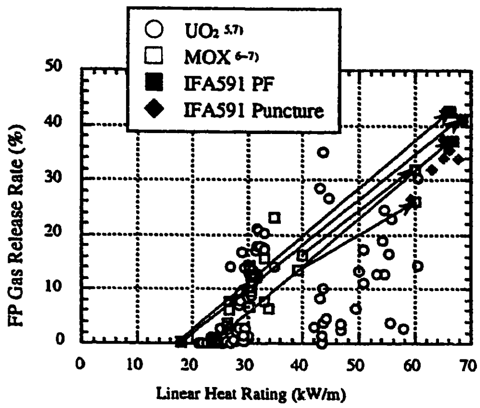


Fig.15 FP gas release rate as a function of linear heat rating experienced above 10GWd/t

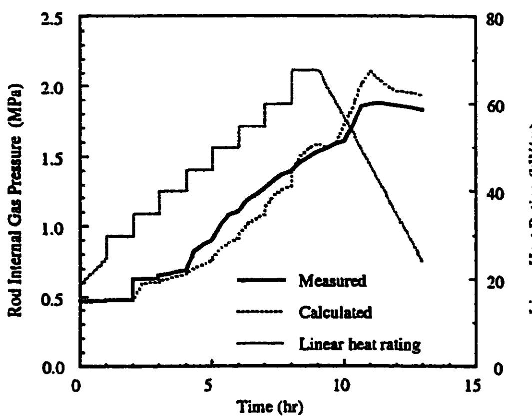


Fig.16 Validation result for rod internal gas pressure of IFA-591-6

Power Ramp Tests of MOX Fuel Rods for ATR (IFA-591)

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Tokai Works, Japan

Objectives

To investigate the ATR MOX fuel behavior during transient operations.

To determine a failure threshold of the MOX fuel.

To develop an irradiation behavior analysis code for MOX fuel .

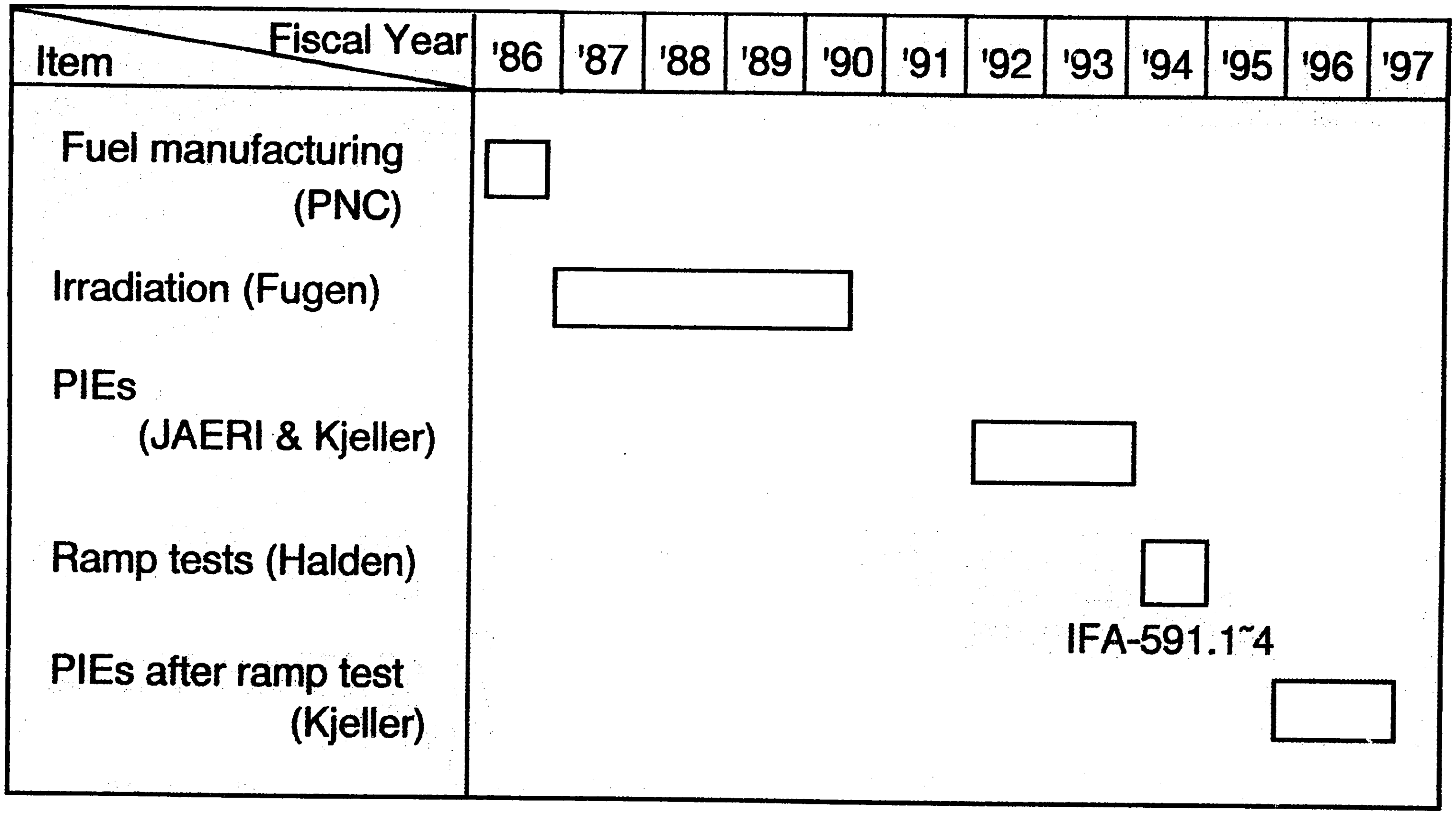


Fig. Program time schedule

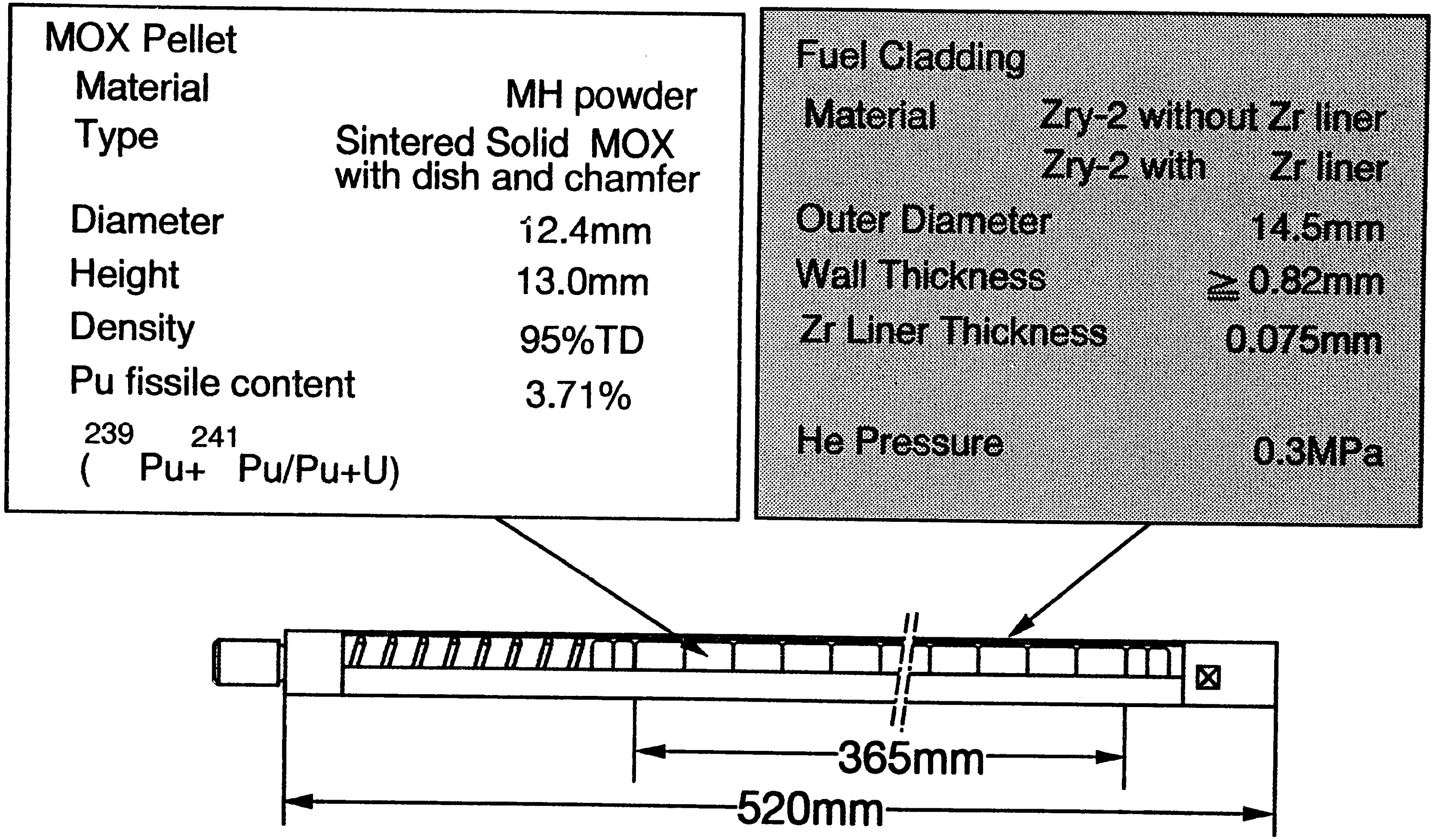


Fig. Schematic and design parameters of the segment rod

Table Achieved burnups of segment rods

Segment Rod No.	Burnup GWd/tM	Max. Linear Heat Rate kW/m	Ramp Test		Loading No.	Instr.
			Ramp Sequence	Terminal Power kW/m		
IFA-591-1	15.5	18.1	Multi	65.3	IFA-591.1	EC
-2*	15.5	18.1	Multi	66.3	IFA-591.1	EC
-3	21.2	23.7	Multi	63.2	IFA-591.1	EC
-4	16.3	18.1	Multi	66.4	IFA-591.2	PF
-5*	16.3	18.1	Multi	68.4	IFA-591.2	PF
-6	22.2	23.7	Multi	66.8	IFA-591.2	PF
-7	16.3	18.1	Single	65.5	IFA-591.4	EC
-8*	14.8	18.1	Single	68.0	IFA-591.4	EC
-9*	16.6	18.1	Single	65.2	IFA-591.3	EC
-10	21.2	23.7	Single	59.4	IFA-591.3	EC
-11	22.2	23.7	Single	58.3	IFA-591.3	EC

*: Zr liner cladding

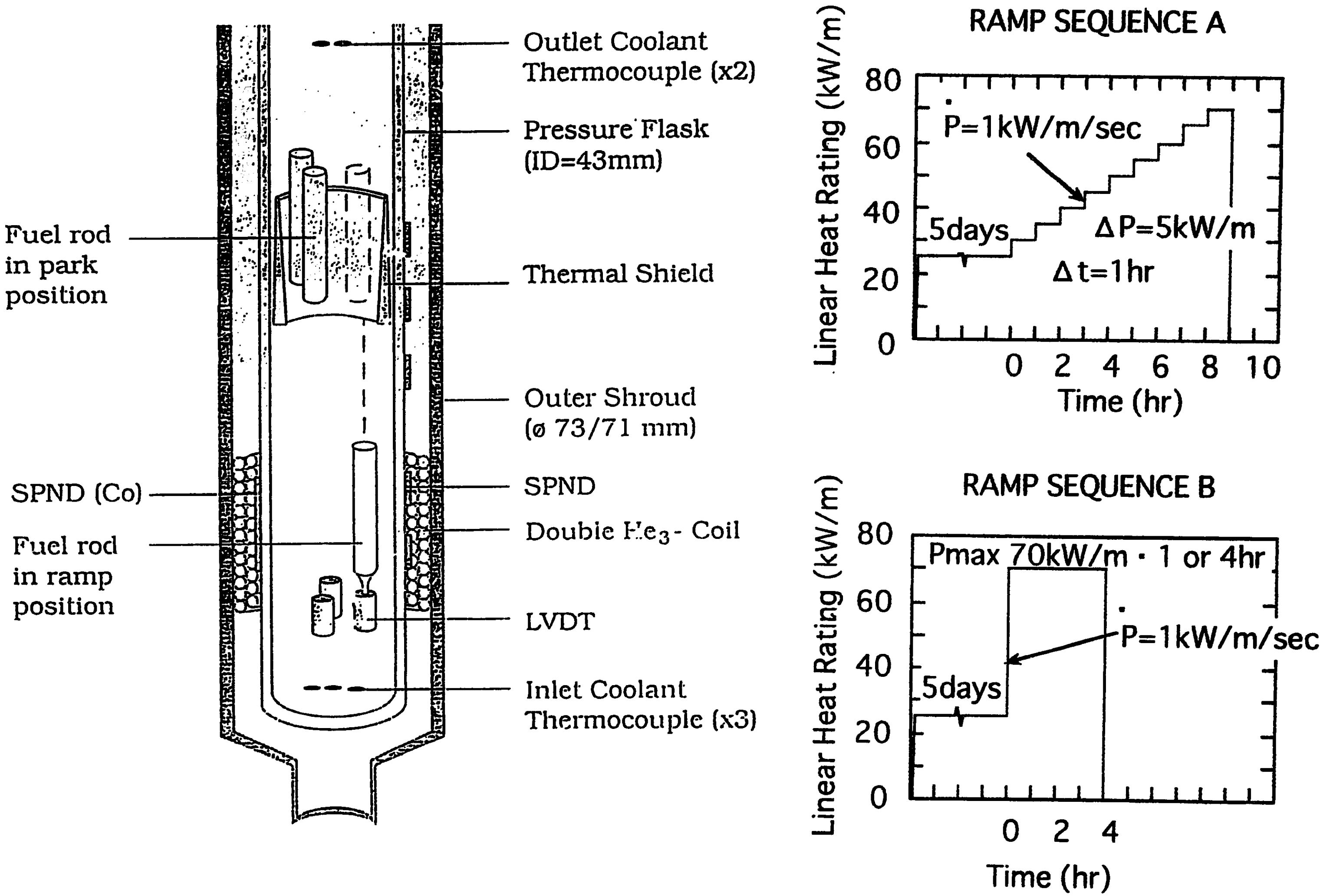


Fig.3 Principal Lay-out of the Rig and Ramp Sequence

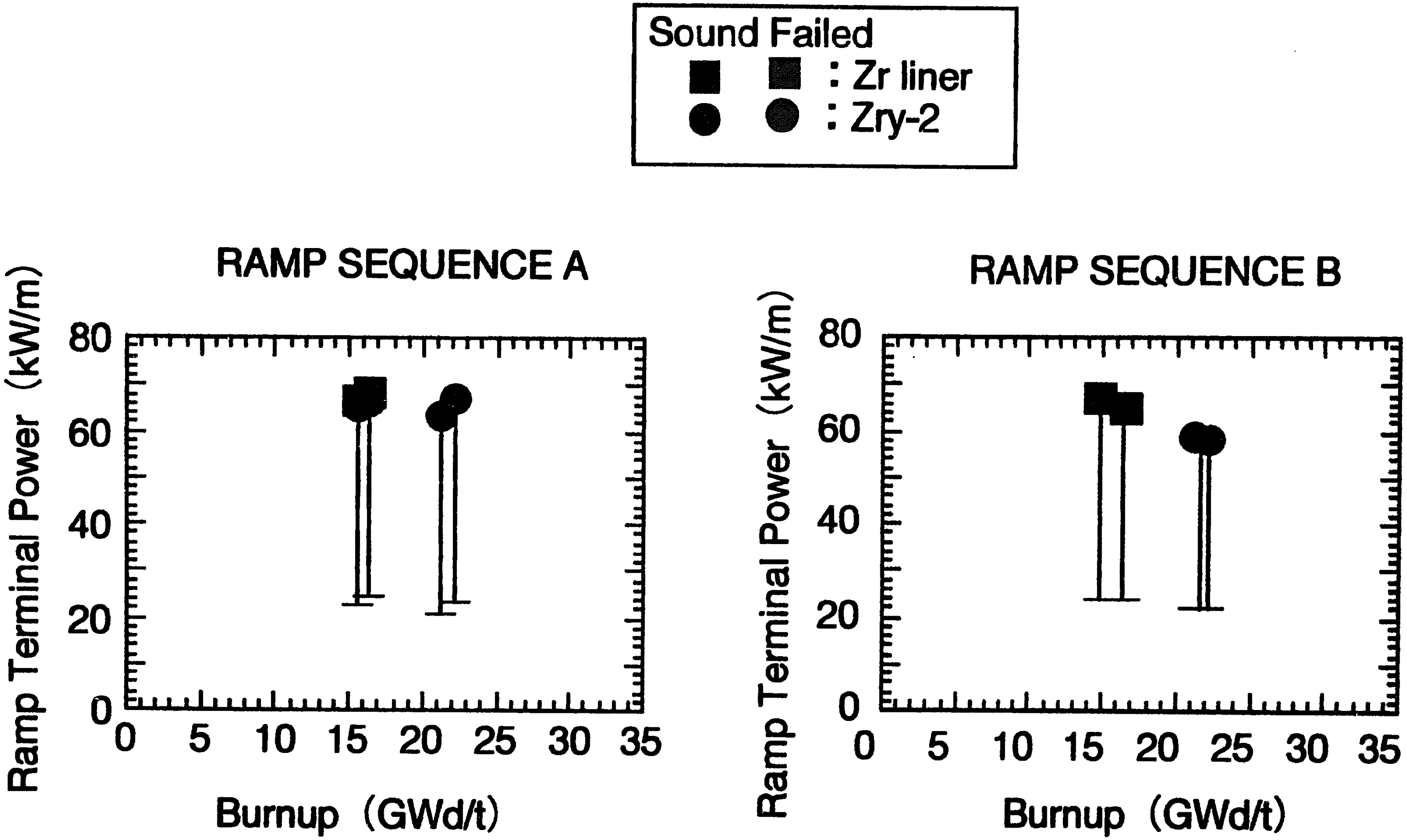


Fig. Ramp Test Results

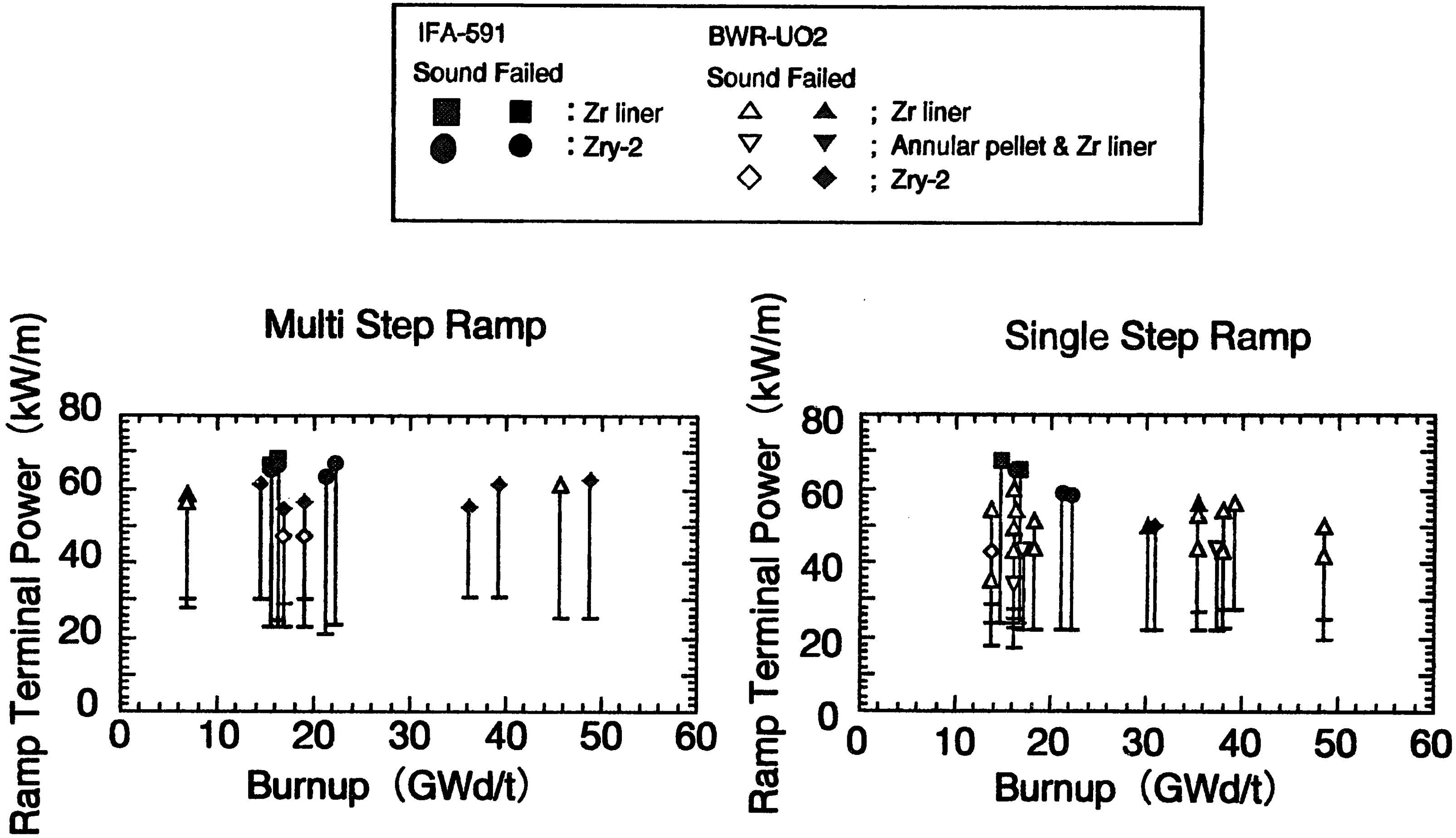


Fig. Ramp Test Results for BWR UO2 fuel compared with ATR MOX (IFA-591)

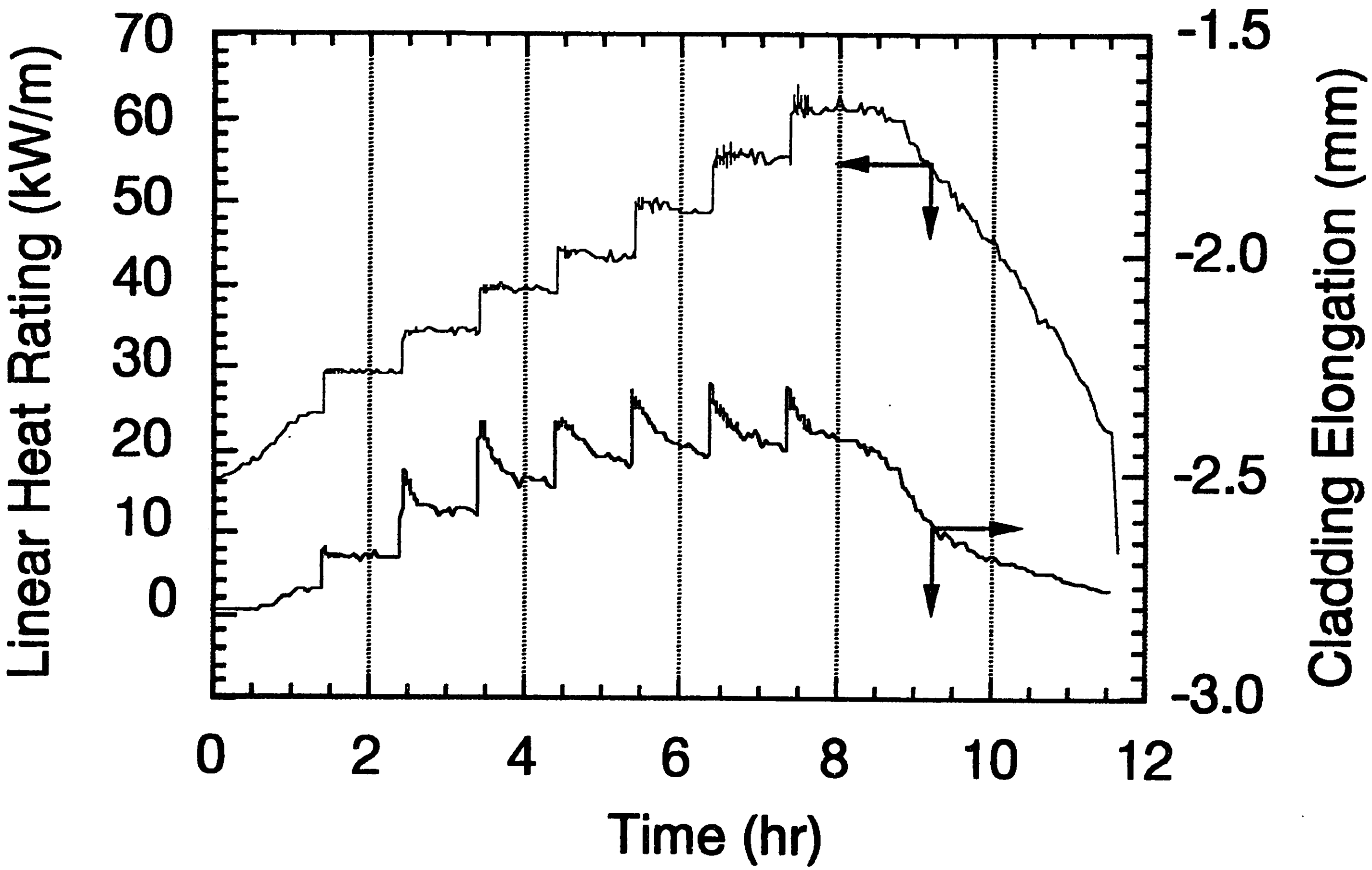


Fig. Linear heat rating and cladding elongation of IFA-591-3 as a function of time

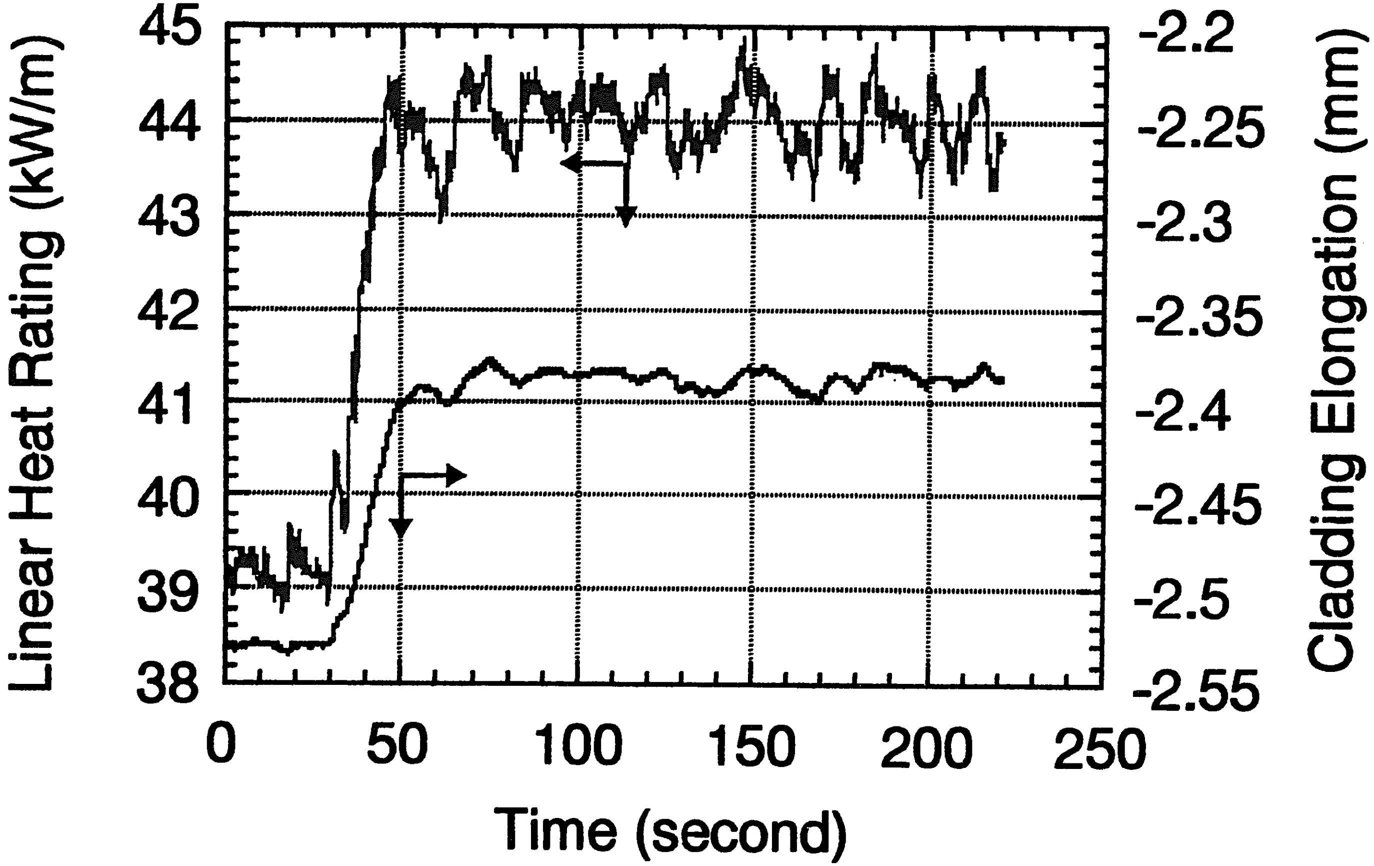


Fig. Linear heat rating and cladding elongation of IFA-591-3 step4 as a function of time

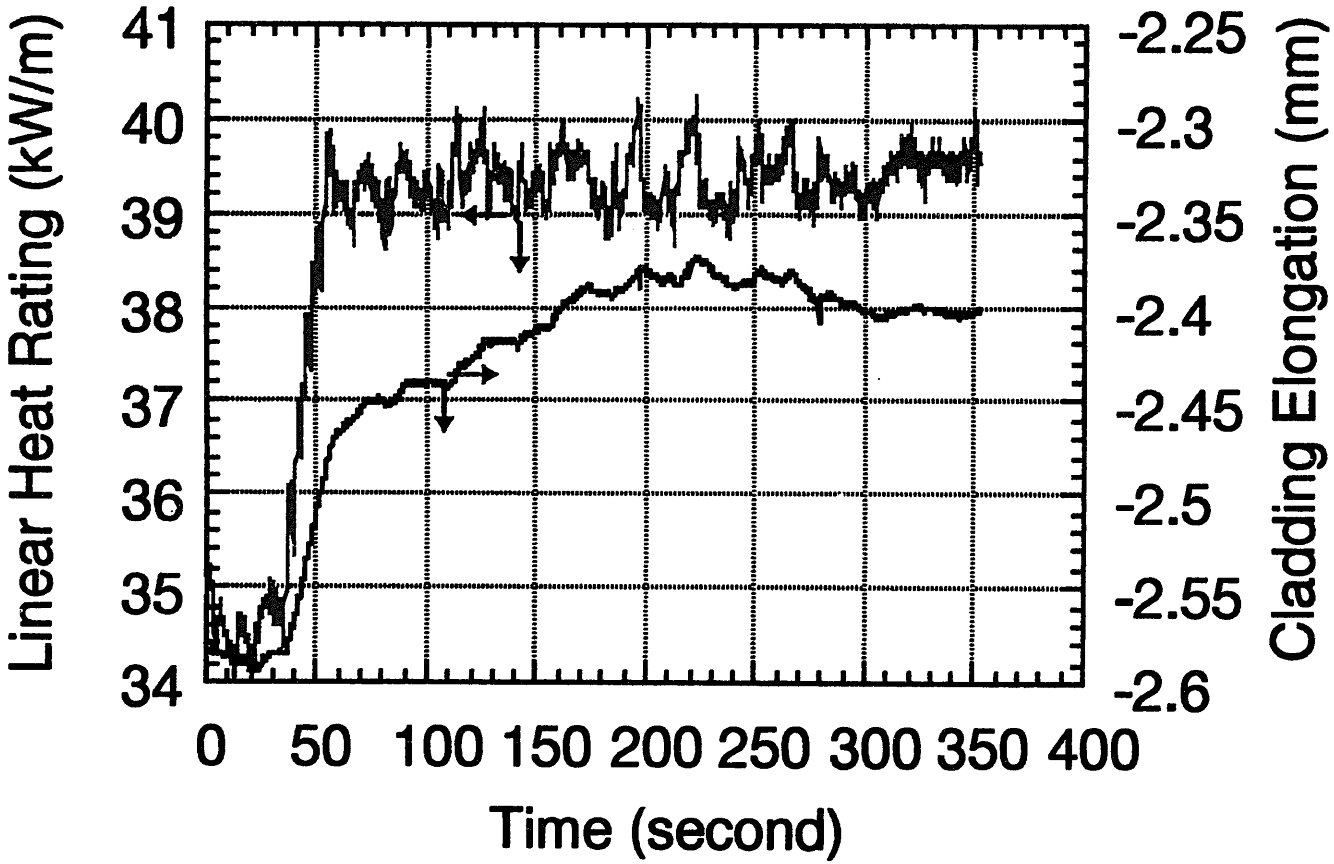


Fig. Linear heat rating and cladding elongation of IFA-591-3 step3 as a function of time

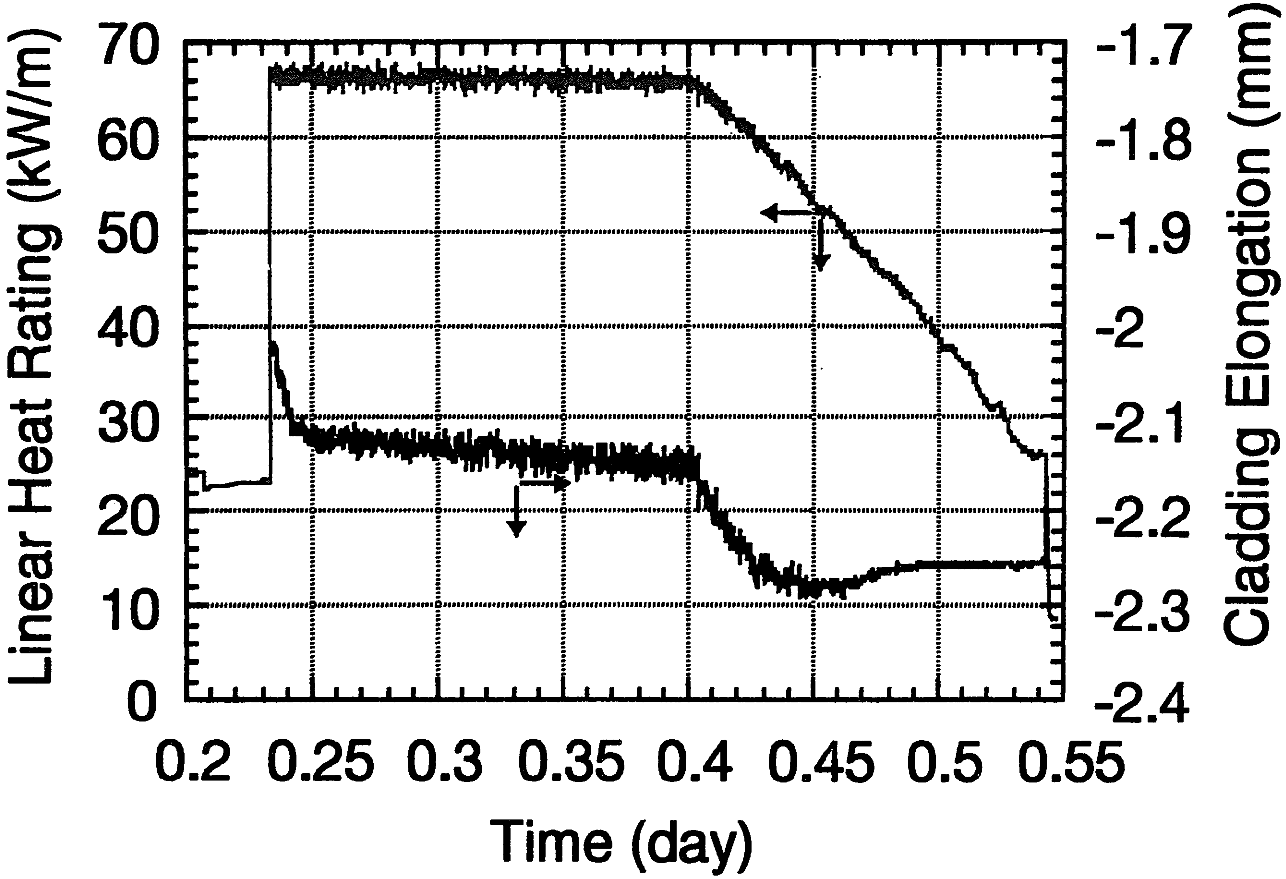


Fig. Linear heat rating and cladding elongation of IFA-591-7 as a function of time

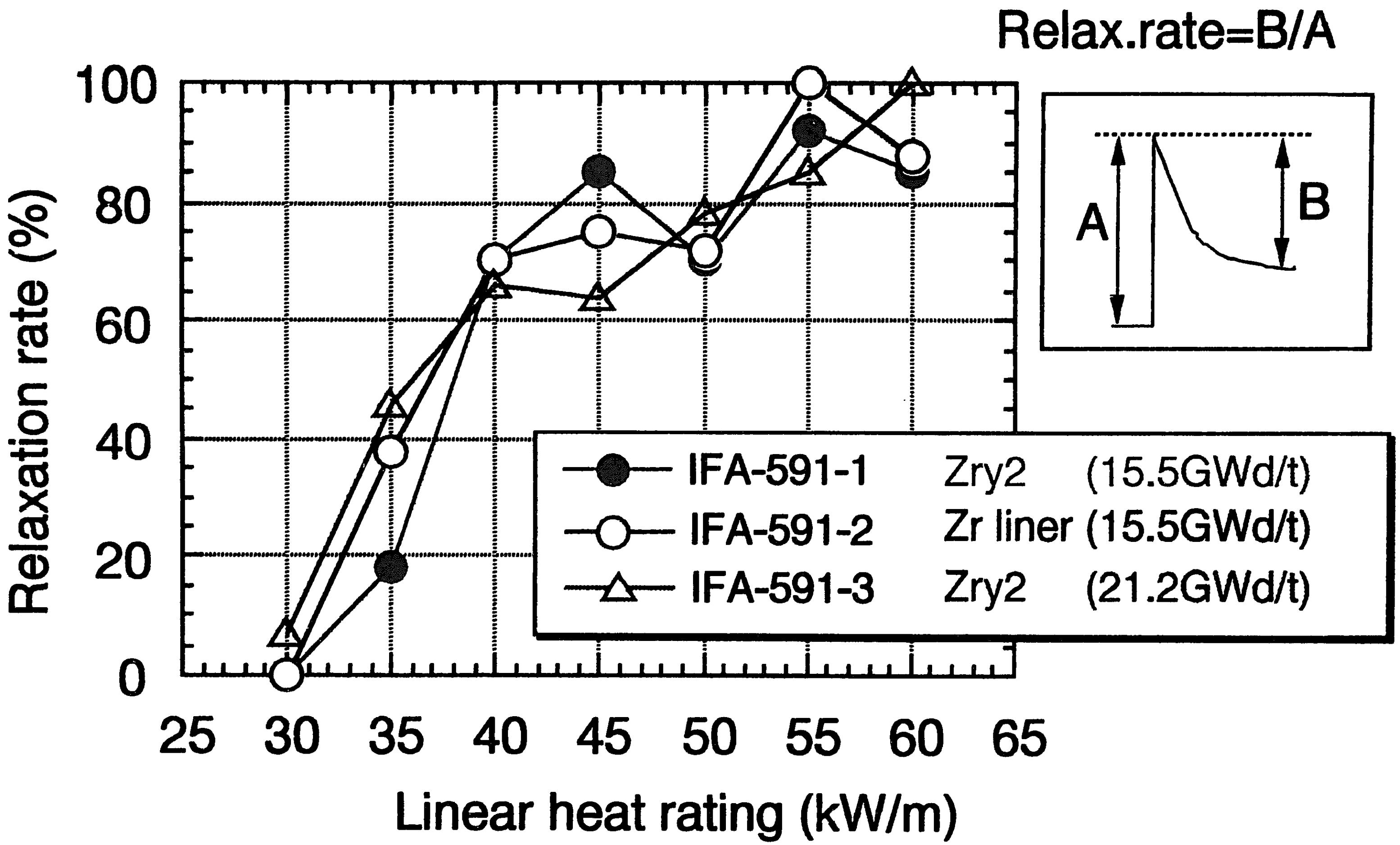


Fig. Relaxation of cladding as a function of linear heat rating

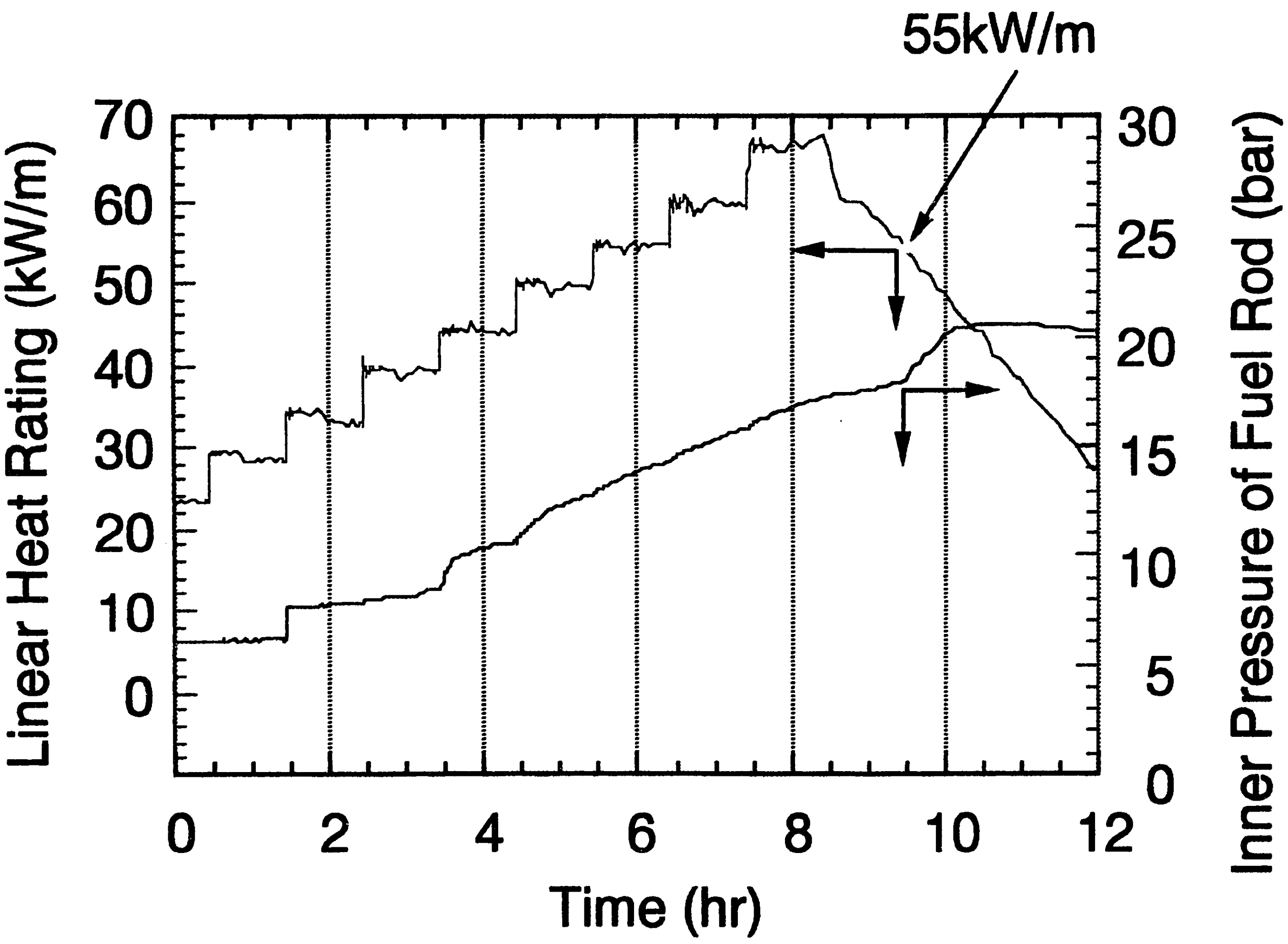


Fig. Linear heat rating and inner pressure of IFA-591-6 as a function of time

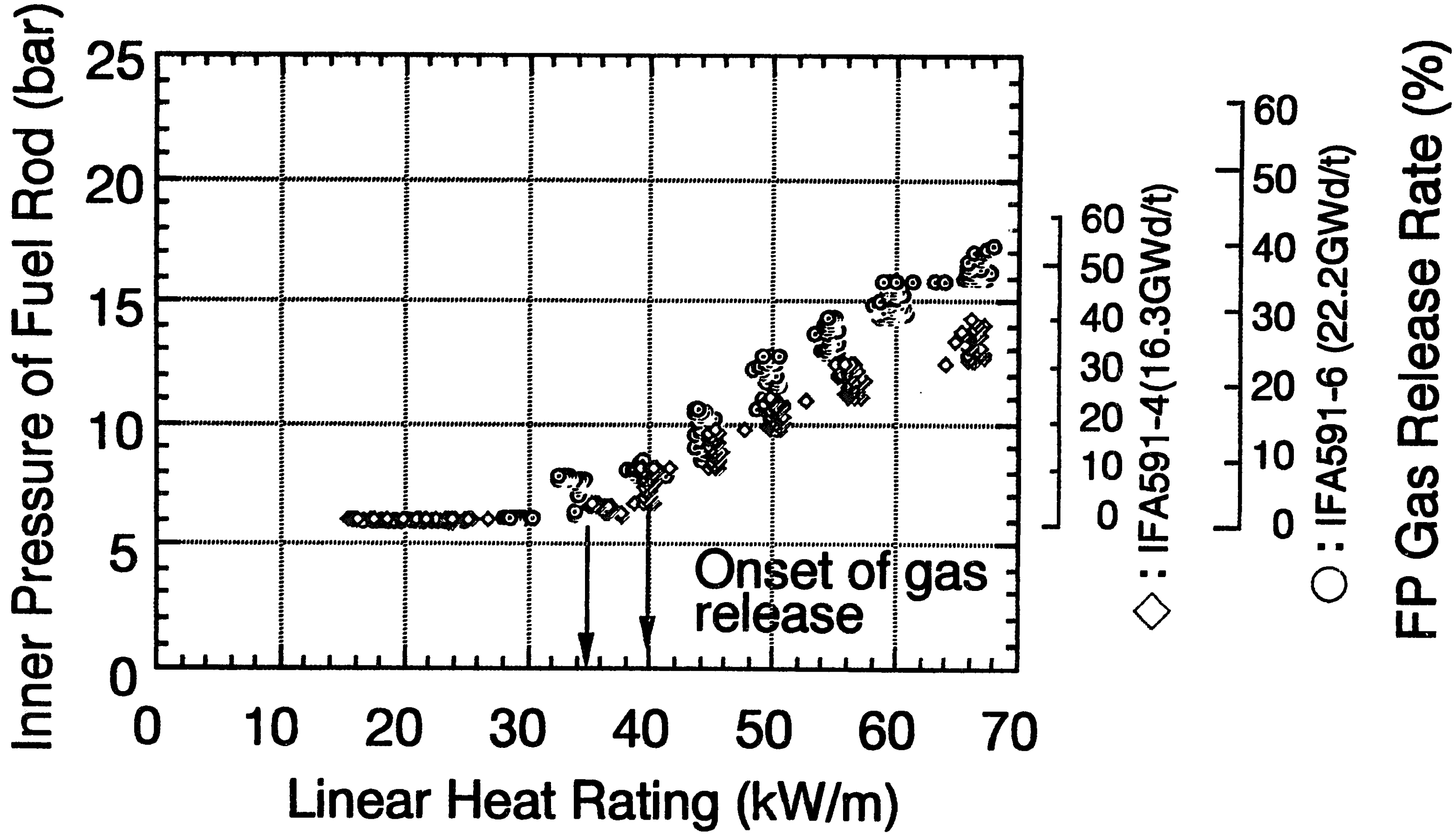


Fig. FP gas release of IFA-591-4 and -6 as a function of linear heat rating

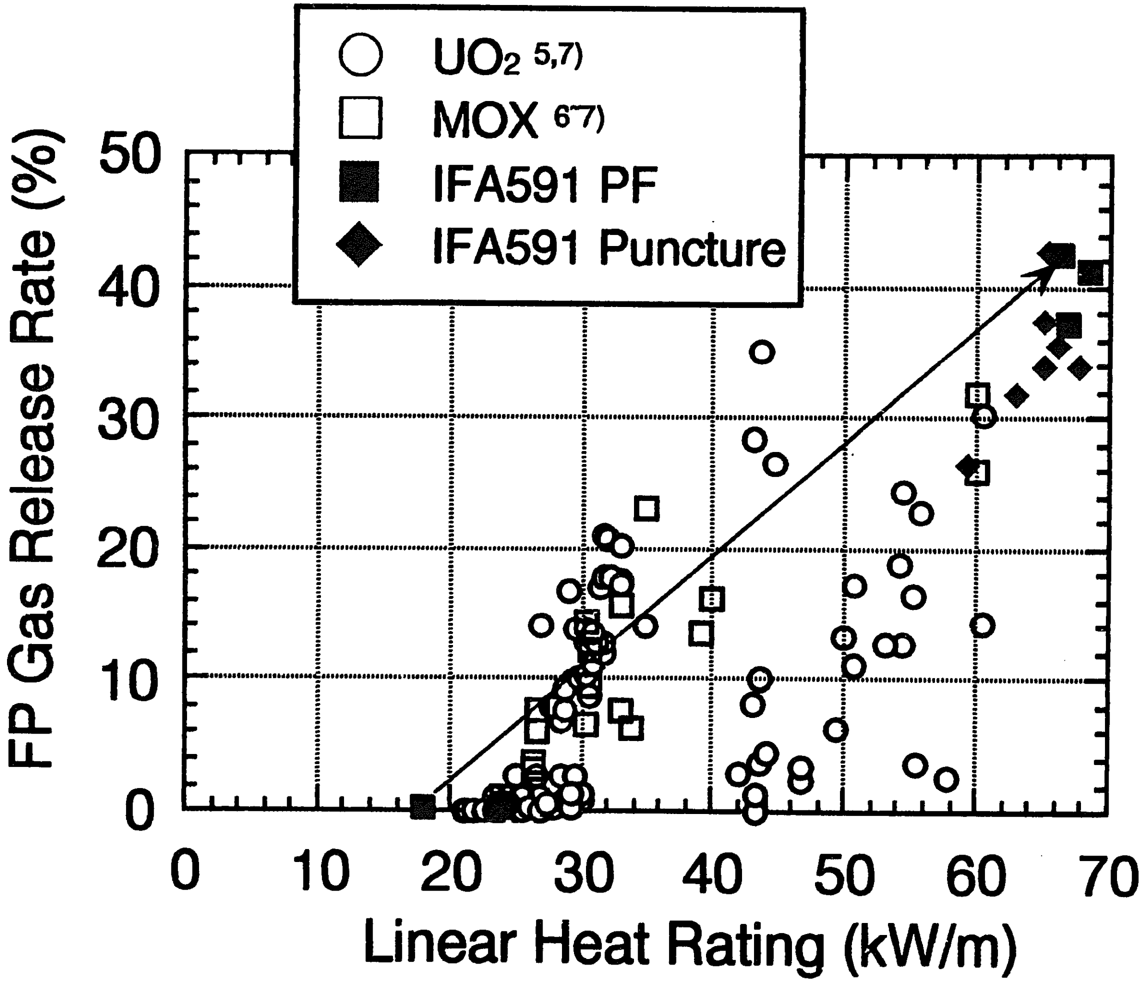


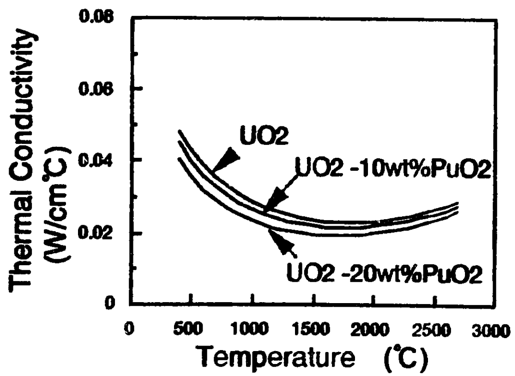
Fig. FP gas release rate as a function of linear heat rating experienced above 10GWd/t

Development of FEMAXI-ATR Code

UO2 rod irradiation behavior analysis code
→FEMAXI-III

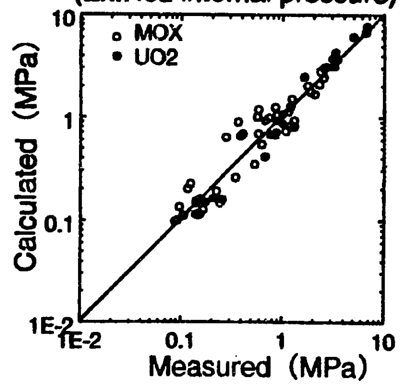
MOX rod irradiation behavior analysis code
→FEMAXI-ATR (Steady State)

● Material property Models and Irradiation behavior Models of MOX fuel



● Verification using PIE data

(Ex.Rod internal pressure)



Development of FEMAXI-ATR Code

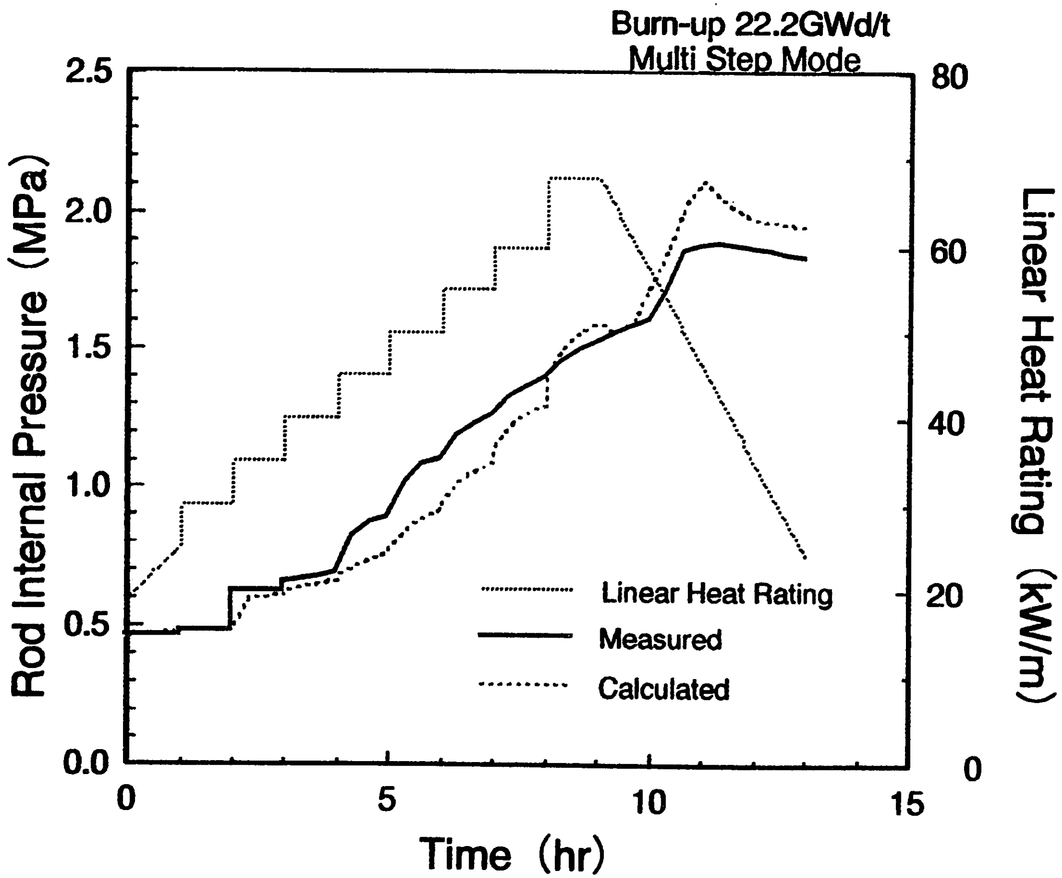
MOX rod irradiation behavior analysis code
→FEMAXI-ATR (Transient)

● *Measurement of Rod Behavior under the Transient Condition*
☆Rod Internal Pressure, Rod elongation

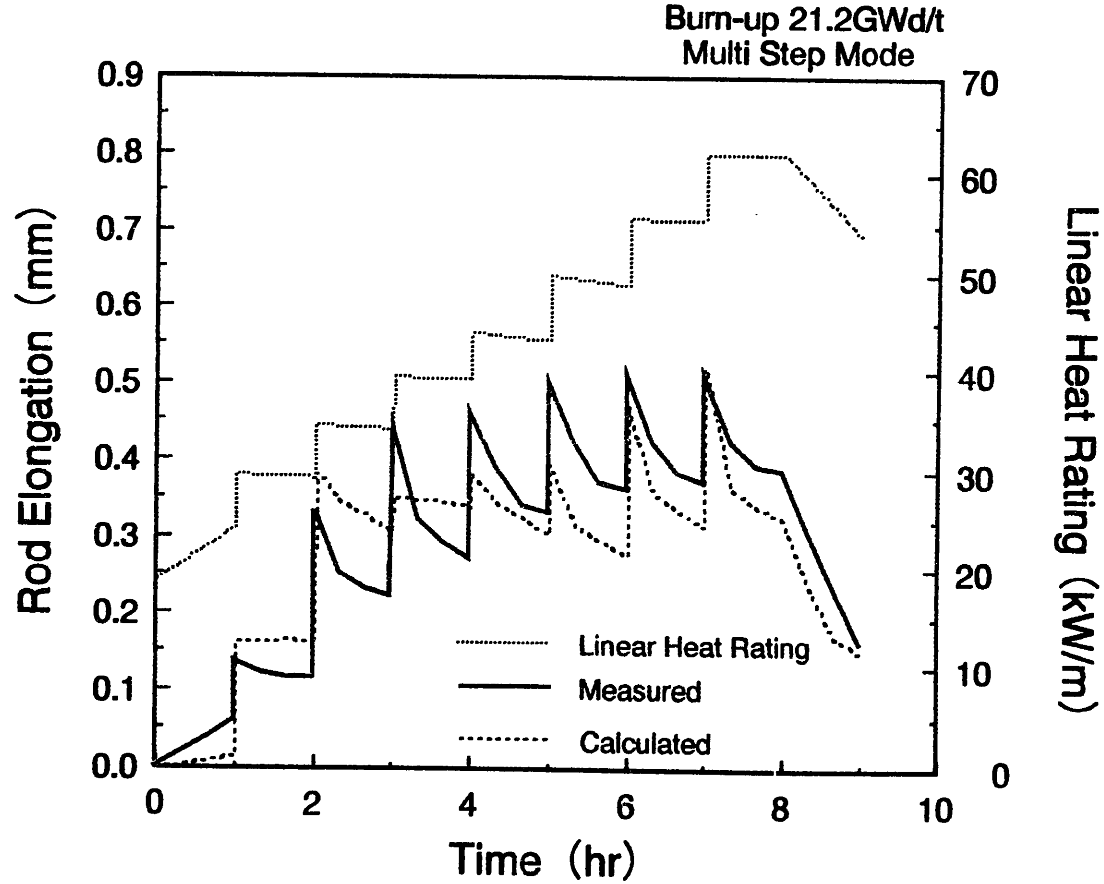
● *Development of irradiation behavior models*
☆Pellet surface smooth by pellet and cladding contact pressure
☆Rod internal gas flows to rod axial direction by gas concentration , etc.

● *Verification*
☆in-pile measurment data

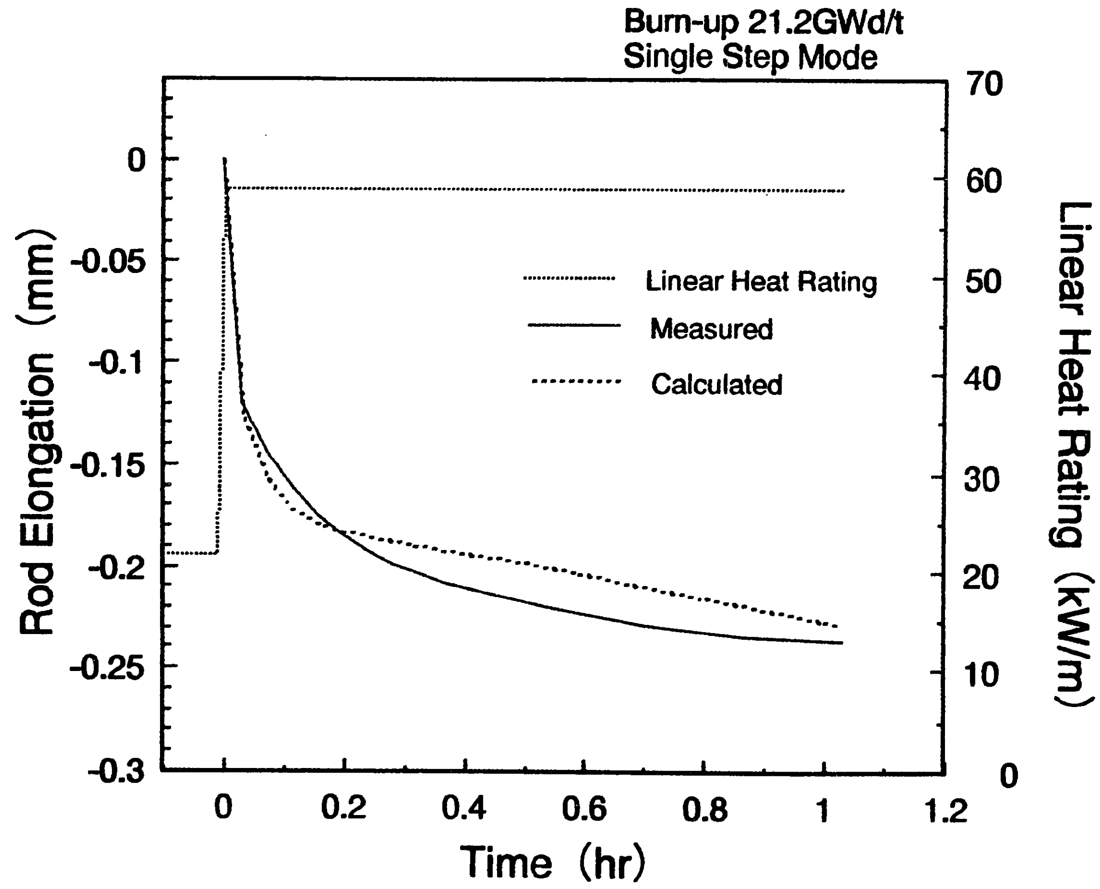
Verification Result of FEMAXI-ATR (Rod Internal Pressure)



Verification Result of FEMAXI-ATR (Rod Elongation)



Verification Result of FEMAXI-ATR (Rod Elongation)



Conclusions

1. The failure threshold is over 68.4kW/m
   Conservative design

2. PCMI behavior
   Axial relaxations
   Thermal feed back phenomena

3. FP gas release
   Three steps (rapid, continuous, additional)
   Total FP gas release rate $\leqq$55%

4. FEMAXI-ATR
   Validation under the transient condition

Development of High Burn-up MOX Fuel for ATR

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ABSTRACT

PNC is operating a prototype of the ATR, called Fugen(165MWe), since 1979. About six hundred plutonium-uranium mixed oxide fuel (MOX fuel) assemblies have been irradiated in Fugen without a failure. PNC is developing a high burn-up MOX fuel for the ATR (AH MOX fuel, maximum assembly burn-up : 55GWd/t) from the economical viewpoint. It is expected to contribute to the development of the MOX fuels for thermal reactors. The statistic design estimation method for the AH MOX fuel including the FEMAXI-ATR code was established. Out-of-pile tests have been performed to obtain thermal hydraulic and mechanical characteristics for the AH MOX fuel. PNC is also developing the MOX-Gd fuel rod for the AH MOX fuel.

I. PURPOSE OF ATR FUEL DEVELOPMENT

The Power Reactor and Nuclear Fuel Development Corporation (PNC) is operating a prototype of the advanced thermal reactor (ATR), called Fugen(165MWe), without a fuel failure, which is the heavy-water-moderated, boiling-light-water-cooled, pressure-tube-type thermal reactor [1]. About six hundred plutonium-uranium mixed oxide fuel (MOX fuel) assemblies have been loaded in Fugen since 1979. The ATR has the advantage of the flexibility in the fuel utilization because fast neutrons are slowed down in the heavy water moderator region(see Fig.1). In the case of plutonium utilization in the ATR, plutonium isotopic composition slightly affects on the nuclear characteristics of

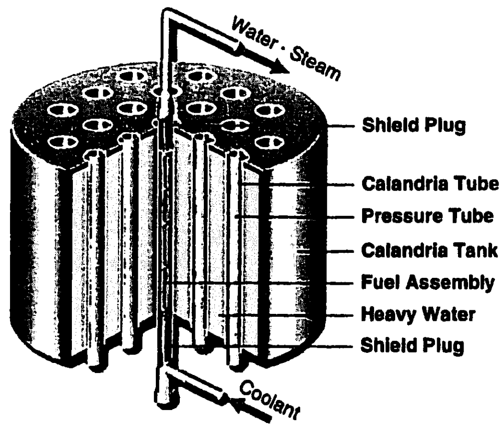

Fig.1 Conception of ATR Core Structure

the ATR [2]. PNC has developed the 28-fuel-rod-type MOX fuel assembly(28-MOX fuel, maximum assembly burn-up: 20GWd/t) and the 36-fuel-rod-type MOX assembly(36-MOX fuel, maximum assembly burn-up: 38GWd/t)for the ATR (see Fig.2). Six hundred twenty seven 28-MOX fuels have been loaded in Fugen as the driver fuel assembly(at November 1996). Eleven 36-MOX fuels also have been loaded in Fugen as the lead fuel assembly, and the integrity of the 36-MOX fuel has been confirmed up to a burn-up of 33GWd/t [3].

PNC is developing a high burn-up MOX fuel assembly for the ATR (AH MOX fuel) from the economical viewpoint. It is expected to contribute to the development of MOX fuels for thermal reactors. A maximum assembly burn-up is 55GWd/t, which is the same as those of the high burn-up UO_2 fuel assemblies for LWRs in Japan. PNC has a plan of the irradiation experiment with a few AH MOX fuels in Fugen(see Fig.3).

II. OUTLINE OF AH MOX FUEL

The structure and principal specifications of the AH MOX fuel are shown in Fig. 4 and Table 1. To fit into the pressure tube of the reactor, the fuel assembly is of a cylindrical configuration in which fuel pins are arranged in three concentric rings. Each pressure tube houses one assembly. The fuel assembly is composed of 54 fuel pins(24 fuel rods in the outer-ring, 18 fuel rods in the intermediate-ring, 12 fuel rods in the inner-ring respectively) of 10.8mm in outer diameter, a spacer supporting rod, upper and lower tie-plates, and 12 spacers. The upper and lower tie-plates and spacers maintain the fuel pins in their desired positions. The fuel assembly is about 4.5m long and 110mm in diameter. The fuel rod has almost the same structure as that of the BWR fuel rod. The average linear heat generation rate will be decreased because the outer diameter of the fuel rod is decreased and the number of fuel rods is increased in comparison with the 28-MOX fuel (16.46mm in outer diameter) loaded in Fugen. Six UO_2 fuel rods with Gd_2O_3 in 18 fuel rods in the intermediate-ring restrain the excess reactivity in the beginning of life to decrease the power mismatch between the fresh fuel and the

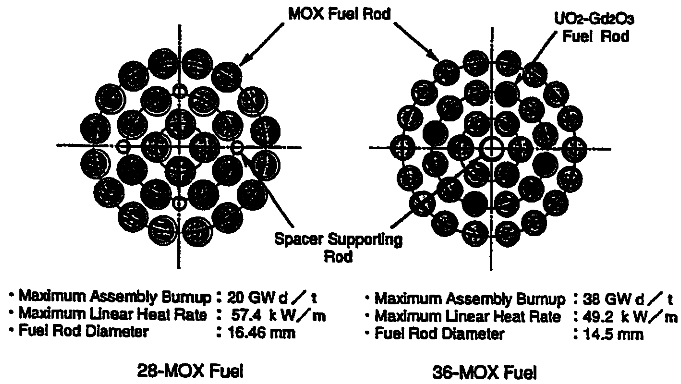

Fig.2 Cross Section of ATR Fuel Assembly

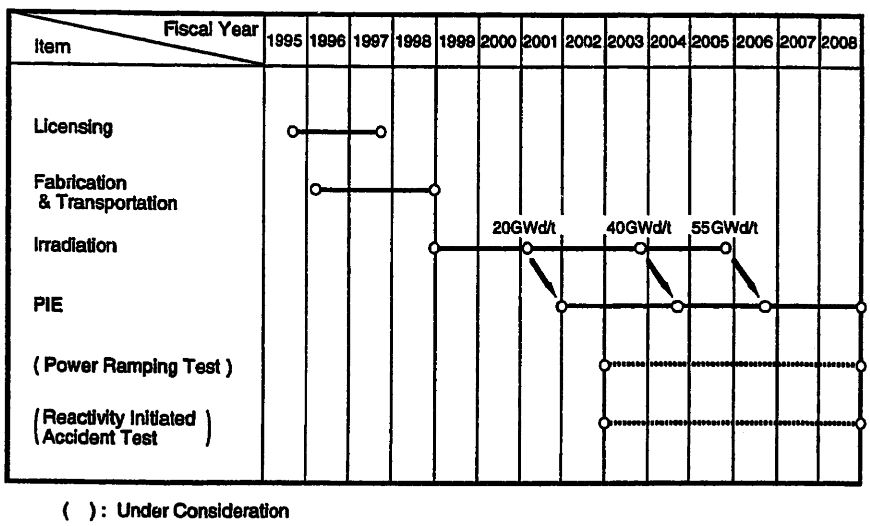

Fig.3 Schedule of AH MOX Fuel Irradiation Test

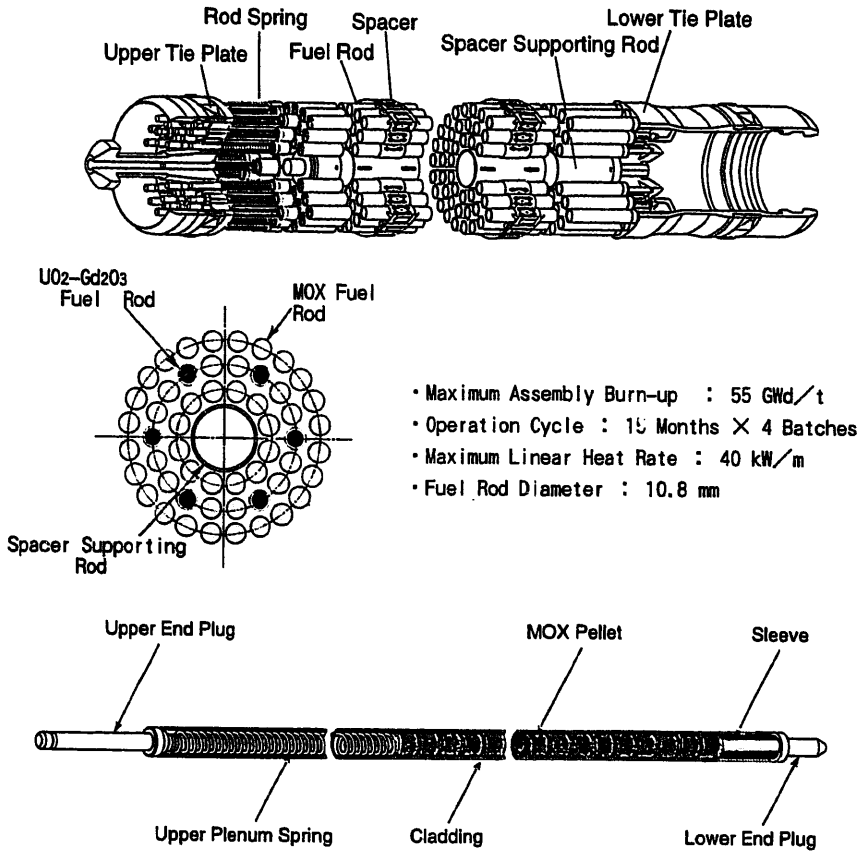

Fig. 4 AH MOX Fuel Assembly

remaining fuel. The spacer supporting rod is located in the center of the fuel assembly and is filled with non-void water, contributing to relaxation of the radial power distribution in the fuel assembly.

III. ISSUES IN DEVELOPING AH MOX FUEL

Extending burn-up causes increase of the internal pressure of the fuel rod at the end of life due to increase of fission gas release. Increase of the initial helium pressure in the fuel rod improves the thermal behavior, restraining increase of the internal pressure. The assembly structure of the AH MOX fuel was designed from the aspect of solutions to increase of the fuel rod elongation and the fuel pressure drop. A conventional deterministic design evaluation method for the ATR MOX fuel rod has been used for the designs of the 28-MOX fuel rod and the 36-MOX fuel rods so far, having sufficient reliability. However, it is considered that the result of the design evaluation is severe in the case of the AH MOX fuel because extending burn-up brings increases of internal pressure and cladding corrosion thickness, and the deterministic design evaluation method has an excessive safety margin. Therefore, it was decided that a statistical design evaluation method with considerations of distributions of fuel dimensions in production and operation conditions was developed and adopted for the AH MOX fuel rod. On the other hand, the statistical design evaluation method was already introduced in the case of designing for step II fuel rod(maximum assembly burn-up:50GWd/t)for BWRs in Japan from the view point of rationality. It is necessary for having benefit of this method to predict thermal and mechanical performances of the high burn-up MOX fuel rod accurately during the irradiation period. A MOX fuel rod performance evaluation code "FEMAXI-ATR" was developed as a best fit code(see Fig.5). It is necessary to predict rationally the gap widths between fuel rods of the AH MOX fuel during irradiation because the quantity of fuel rod bowing increases due to increase of the irradiation period. A gap-width between fuel rods evaluation code "DYNAGAP" was developed, which introduced the statistic calculation method.

Table 1 Fuel Specification

Item	54-Fuel-rod Assembly
1. Fuel Pellet	
Material	PuO_2-UO_2 UO_2-Gd_2O_3
Outer Diameter(mm)	9.10
Height(mm)	9.5
Density(%T.D.)	95.0
Shape	Solid with Dish & Chamfer
Pu Fissile Enrichment(wt%)	3.6~4.8
2. Fuel Rod	
Cladding Material	Zircaloy-2 with Zr-Liner
Outer Diameter(mm)	10.80
Inner Diameter(mm)	9.30
Diametral Gapsize(μm)	200
Fuel Stack Length(mm)	3,600
Filling Gas	He
Filling Gas Pressure(MPa)	0.5
3. Fuel Assembly	
Length(mm)	4,381
Bundle Outer Diameter(mm)	111.6
Rod Number	54
Inner-Ring Rod	12
Intermediate-Ring Free Rod	12
Intermediate-Ring Tie Rod	6
Outer-Ring Rod	24
Spacer Number	12

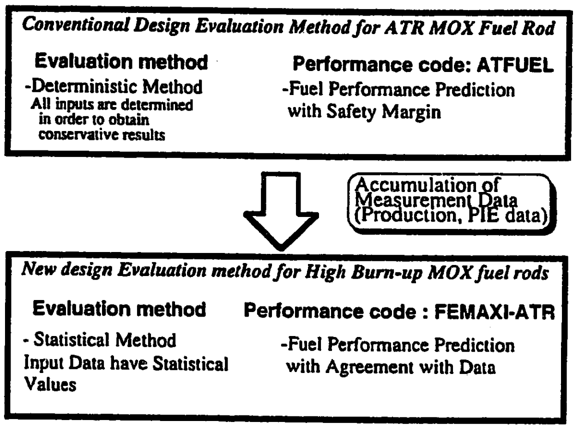

Fig.5 Comparison of Design Evaluation Methods for ATR Fuel Rod

IV. CODE DEVELOPMENTS FOR AH MOX FUEL

A. DEVELOPMENT OF "FEMAXI-ATR"

The FEMAXI-ATR code is based on "FEMAXI-III" [4] that has enough experiences in evaluating performances on the UO_2 fuel rod. It has fuel material property models and irradiation performance models specific to the MOX fuel as shown in Fig.6~9. Principal introduced models of fuel material properties are melting point [5], thermal conductivity [6,7], thermal expansion factor [8], Young's module [8], creep rate [8]. Principal introduced models of irradiation performances are FP gas release [9], xenon to krypton ratio, production of helium which are based on the results of the irradiation tests in Fugen[3], the SGHWR[11] in UKAEA,and the HBWR in the Halden reator project. In the model of production of helium, the production quantity due to decay of Cm242 corresponding to the initial plutonium density is considered, which is calculated by the ORIGEN code. A correlation [10] between helium gas release and FP gas release is also considered. Verifications of the FEMAXI-ATR code were performed by comparing calculated values with the results of the MOX fuel irradiation tests in Fugen [3], the SGHWR [11], the HBWR[12], and the Dodewaard reactor [13]. Code verifications were also done by using UO_2 fuel data [14,15,16]. The data bases are shown in Table 2. Verification items were fuel centerline temperature, fuel rod internal pressure, fission gas release, helium gas release, and fuel rod outer diameter change. Calculated values and measured ones had a good coincidence at a burn-up of up to about 60GWd/t as shown in Fig. 10~14. Thus, it was verified that the FEMAXI-ATR code could predict accurately thermal and mechanical performances of the ATR MOX fuel rod during the irradiation period even at high burn-up. Uncertainty of the code predictions to measured values is considered as a safety margin in the statistical design evaluation method.

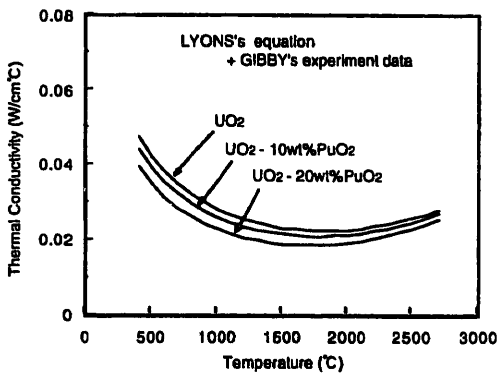

Fig.6 Model of FEMAXI-ATR (Material Property)
(Fuel Pellet Thermal Conductivity)

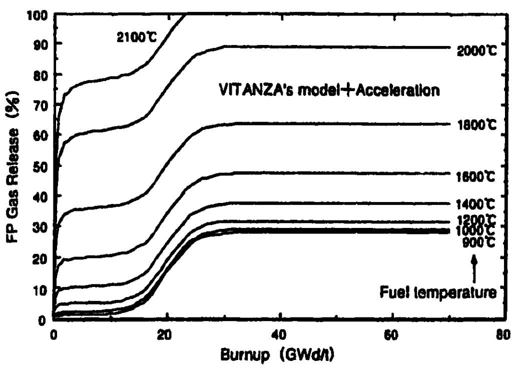

Fig.7 Model of FEMAXI-ATR (Material Property)
(FP Gas Release vs. Burnup and Temperature)

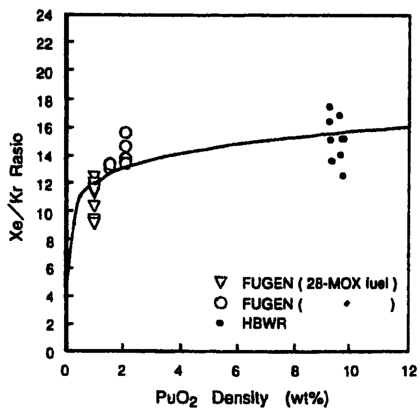

Fig.8 Model of FEMAXI-ATR(Irradiation Behavior)
(Xe / Kr Ratio vs. PuO2 Density)

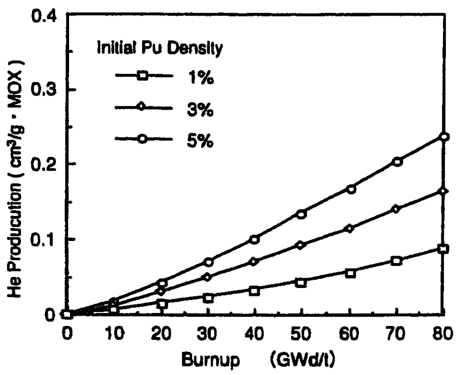

Fig.9 Model of FEMAXI-ATR (Irradiation Behavior)
(He Producution vs. Burnup)

B. DEVELOPMENT OF "DYNAGAP"

The DYNAGAP code was developed to predict in-core behaviors of gap widths between fuel rods. It deals with all fuel rods as an assembly model statistically. Fuel rod bowing is calculated by using a elastic rod model with considerations of initial displacement of rod bowing, axial force caused by differences of rod elongation between each ring, circumferential distribution of thermal and irradiation

Table2 Data Bases for Verification of "FEMAXI-ATR"

Item	Number of Fuel Rod	Specification: Pellet-Cladding Gap (mm)	Pellet Density & Pu Content	Initial He Gas Pressure (MPa)	Rod Average Burn-up (GWd/t)	Maximum LHR (kW/m)	Data Base
Pellet Maximum Temperature	5	0.18 ~ 0.31	93~95 (%T.D.) 4.5~7.6 (wt%)	0.1 ~ 0.3	≦45	≦41	• Halden (MOX)
FP Gas Release Rate	80	0.09 ~ 0.33	92~96 (%T.D.) 0.7~9.6 (wt%)	0.1 ~ 2.1	≦62	≦55	• Halden (MOX) • SGHWR (UO_2, MOX) • FUGEN (MOX) • Japanese BWR (UO_2) • NFIR (UO_2) • BR-3 (UO_2) • Dodewaard (UO_2, MOX) • Inter Ramp (UO_2)
Rod Inner Pressure	72	0.09~ 0.33	92~96 (%T.D.) 0.7~9.6 (wt%)	0.1 ~ 2.1	≦62	≦55	• Halden (MOX) • SGHWR (UO_2, MOX) • FUGEN (MOX) • Japanese BWR (UO_2) • NFIR (UO_2) • BR-3 (UO_2) • Dodewaard (UO_2, MOX) • Inter Ramp (UO_2)
He Gas Volume	56	0.12~ 0.33	92~96 (%T.D.) 0.7~9.6 (wt%)	0.1 ~ 2.1	≦62	≦55	• Halden (MOX) • SGHWR (UO_2, MOX) • FUGEN (MOX) • Dodewaard (UO_2, MOX)
Cladding Diameter Change	42	0.09~ 0.30	92~96 (%T.D.) 0.7~3.3 (wt%)	0.1 ~ 2.1	≦60	≦55	• SGHWR (UO_2, MOX) • FUGEN (MOX) • NFIR (UO_2) • BR-3 (UO_2) • Inter Ramp (UO_2)

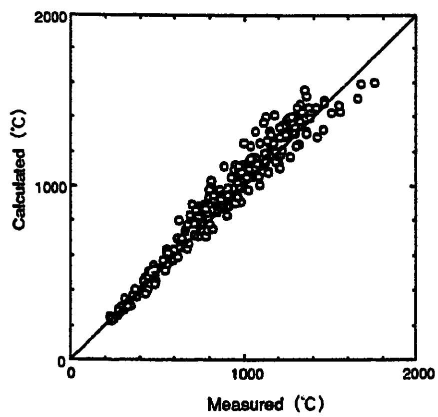

Fig.10 Verification Results of FEMAXI-ATR
(Fuel Centerline Temperature)

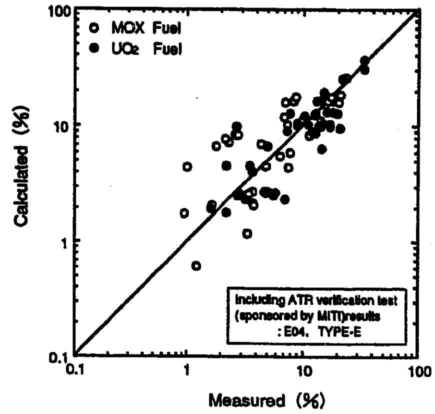

Fig.11 Verification Results of FEMAXI-ATR
(Fission Gas Release)

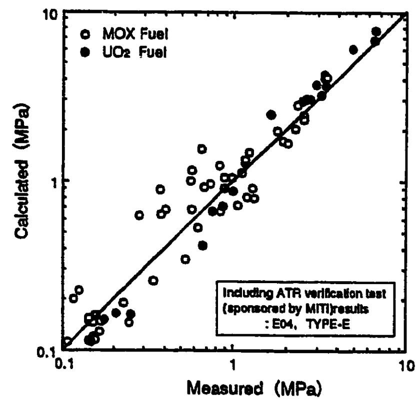

Fig.12 Verification Results of FEMAXI-ATR
(Fuel Rod Internal Pressure)

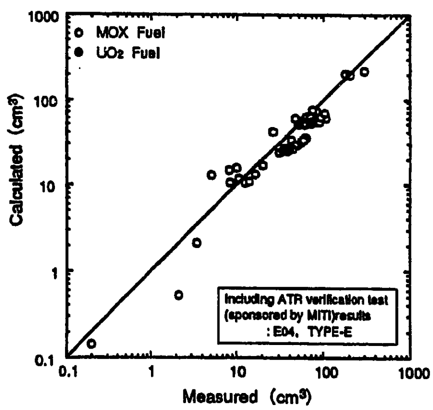

Fig.13 Verification Results of FEMAXI-ATR
(He Gas Release)

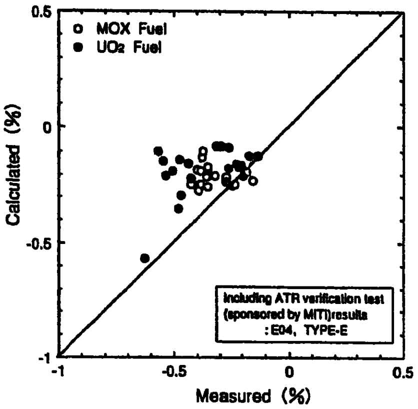

Fig.14 Verification Results of FEMAXI-ATR
(Fuel Rod Outer Diameter Change)

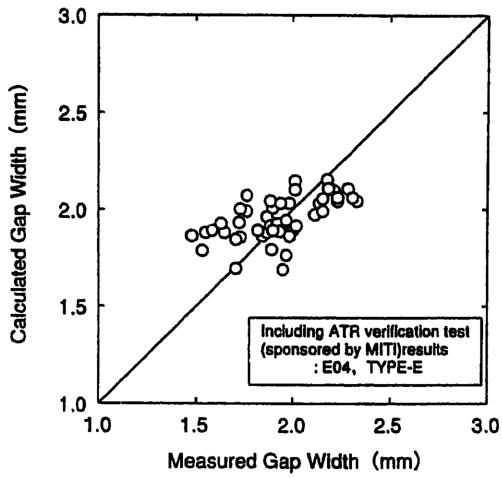

Fig.16 Verification Results of " DYNAGAP "

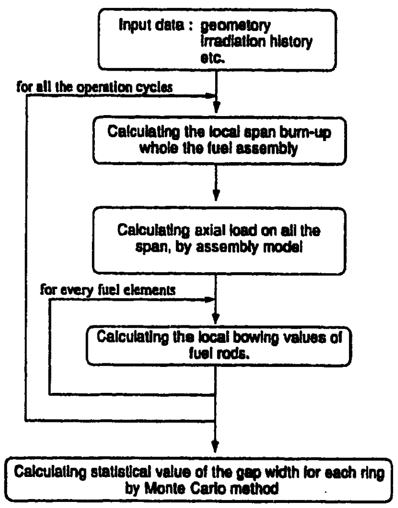

Fig.15 Flow Chart of "DYNAGAP"

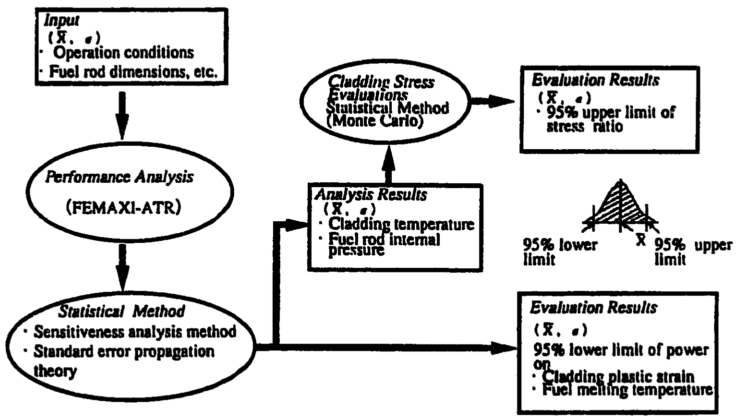

Fig.17 Flow Diagram of Fuel Design Evaluation
(Statistical Evaluation Method)

effects on fuel rods, stress relaxation by creep and relaxation of binding force of spacer. Thereafter, the distribution of the gap width value between fuel rods for each ring is obtained by applying Monte Carlo method to calculated bowing values because the direction of bowing for each fuel rod is at random. The flow chart of this code is shown in Fig. 15. Code verifications were done by comparing calculated values with measured ones which were based on the results of the post irradiation tests for the 36-MOX fuels(E04, TYPE-E) irradiated in Fugen[3] and SGHWR[11]. Calculated values and measured ones have a good coincidence as shown in Fig. 16. The code was modified for using in design estimation so that calculation values could cover all measured ones.

V. STATISTICAL DESIGN EVALUATION

The flow diagram of the statistical design evaluation method is shown in Fig.17. In this method, input values and evaluation values are dealt with statistically. Evaluation values are obtained

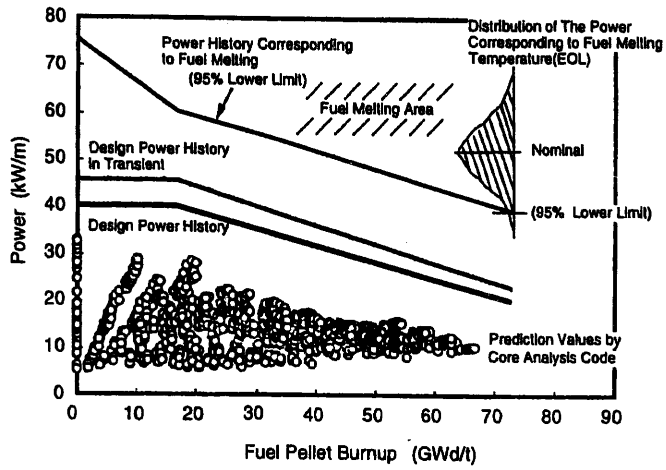

Fig.18 Evaluation Result on Fuel Melting

by applying the sensitiveness analysis method and the standard error propagation theory to the results of analyses by the FEMAXI-ATR code. The statistical evaluation values of the cladding stresses are obtained by the Monte Carlo method in the evaluation of the cladding stress. Thereafter, the 95% value to the upper limit in the distribution of the cladding stress is compared with the allowable stress which is based on the shear strain energy hypothesis. This statistical design evaluation method was applied to the evaluation of the AH MOX fuel rod. It was verified analytically that the integrity of the AH MOX fuel rod was maintained over the whole irradiation period. Fig.18 shows the evaluation result on melting of fuel. It is clear that the 95% value to the lower limit in the distribution of the power corresponding to melting of fuel is sufficiently higher than the design power history in transient.

VI. CHF TEST, HYDRAULIC TEST, ENDURANCE TEST, AND VIBRATION TEST

Four kinds of out-of-pile tests have been performed to obtain thermal hydraulic and mechanical characteristics for the AH MOX fuel. A critical heat flux(CHF) test has been performed at the 14 MW Heat-Transfer-Loop(HTL). The HTL test section is shown in Fig.19. In the experiment, one of the local peakings of the cluster was at the beginning of the life where was the severest in terms of thermal hydraulics. Fig.20 shows the relationship between CHF and steam quality for the whole data of

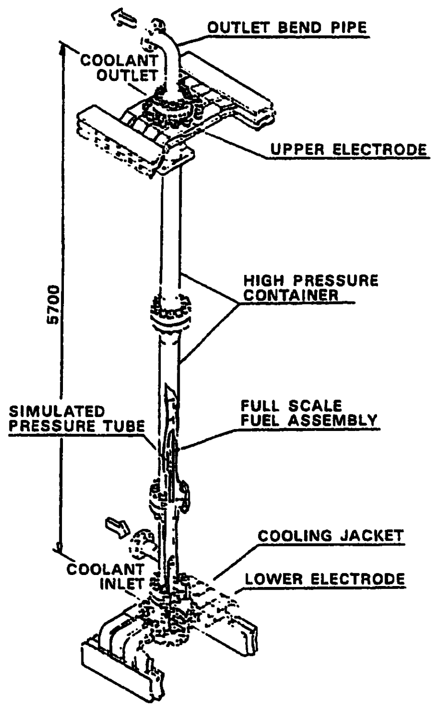

Fig.19 HTL Test Section

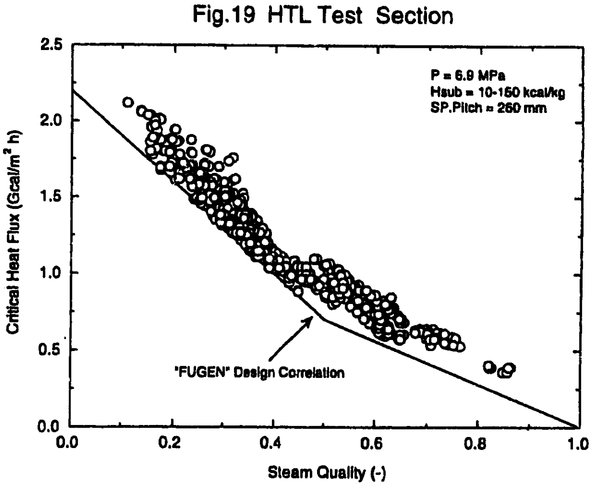

Fig.20 Comparison between All of Experimental Data and "FUGEN" Design Correlation

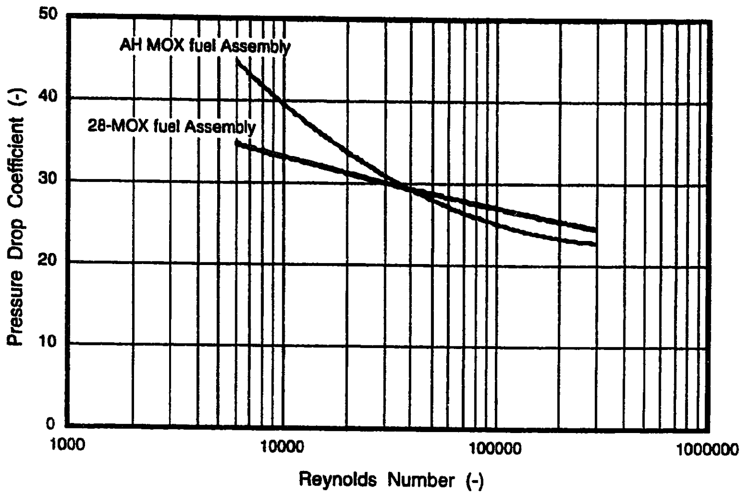

Fig.21 Pressure Drop Coefficient of Fuel Assembly

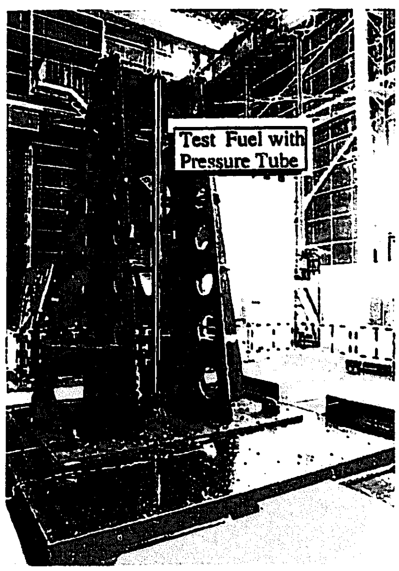

Fig.22 Vibration Test Equipment

the AH MOX fuel. It has been confirmed that the measured CHFs were higher than the design correlation line for the 28-MOX fuel. The tests of hydraulics and mechanical endurance have been also carried out using a first mockup AH MOX fuel and the Component-Test-Loop(CTL), in order to obtain pressure drop coefficients and two-phase multiplier coefficients of components constructing the AH MOX fuel assembly, the fretting characteristics of the fuel claddings at the contact points with the spacers and the integrity of the fuel components. It has been confirmed that each correlation of pressure drop coefficients(see Fig.21) and two-phase multiplier coefficients was almost the same as that of the 28-MOX fuel at normal reactor operating conditions. It was predicted that the integrity of the AH MOX fuel was retained at the maximum assembly burn-up of up to 55GWd/t from the results of the endurance test. In addition, the vibration tests for the AH MOX fuel have been performed to obtain the vibration characteristics, and to demonstrate the integrity of the AH MOX fuel under earthquake conditions because the skeleton structure with one spacer supporting rod was different from that of the 28-MOX fuel(see Fig.22). The test fuel had the same specifications as those of the AH MOX fuel except imitation pellets for weight, and acceleration detectors set in fuel rods and the spacer supporting rod. It has been confirmed that the vibration

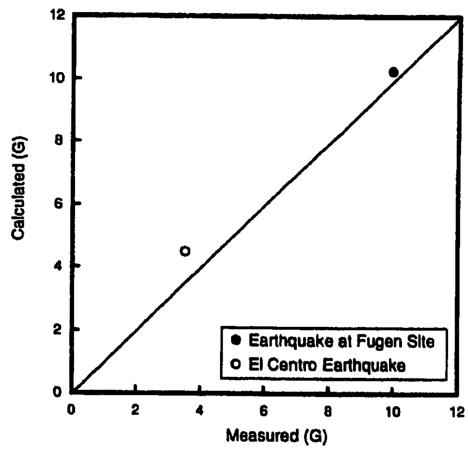

Fig.23 Verification Results of Aseismic Analysys Code " CATFISH " (Maximum Accelaration)

characteristics of the AH MOX fuel was almost the same as that of the 28-MOX fuel. It is considered that the binding forces by 12 spacers for the fuel rods and the spacer supporting rod are enough to make the fuel assembly vibrate as a single body. The vibration test simulating design-base earthquakes has been curried out. It has been confirmed that the integrity of the AH MOX fuel was retained under the design-base seismic conditions. A verification of an aseismic analysis code "CATFISH" was done by comparing calculated values with measured ones. It was confirmed that calculated values and measured ones had a good coincidence as shown in Fig. 23.

VII. DEVELOPMENT OF MOX-GD FUEL

The AH MOX fuel has six UO_2 fuel rods with Gd_2O_3 as mentioned above. However, the utilization of the MOX fuel rod with Gd_2O_3 can make the reactor core composed by MOX fuels completely. A duplex-pellet-type MOX fuel rod with Gd_2O_3 (DU MOX-Gd) is being developed by PNC which means that Gd_2O_3 rods (or Gd_2O_3-ZrO_2 rods) are inserted in holes of annular MOX pellets(see Fig.24) [17]. The characteristics in the utilization of this fuel rod are , 1)the manufacturing processes for the DU MOX-Gd fuel pellets are not required, 2) the effect of gadolinium as burnable poison is milder in comparison with the case of using MOX fuel rod added Gd_2O_3 homogeneously because the Gd_2O_3 rod is located at the center of the fuel pellet so that the number of thermal neutrons absorbed by gadolinium is smaller than that in the case of the MOX fuel rod added Gd_2O_3 homogeneously. Furthermore, there is a possibility to adopt a ATR fuel assembly with one kind of plutonium fissile content if the DU MOX-Gd fuel rods are utilized in the fuel assembly more positively. This advanced fuel technology can be applied to the MOX fuels for LWRs.

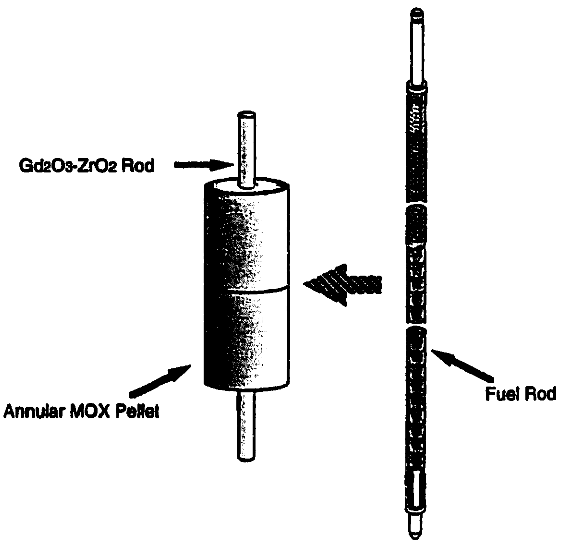

Fig.24 Duplex-Type MOX-Gd Fuel

VIII. CONCLUSIONS

1) PNC is developing the AH MOX fuel(maximum assembly burn-up:55GWd/t) which is expected to contribute to the development of the MOX fuels for thermal reactors.
2) PNC has the plan of the irradiation experiment with a few AH MOX fuels in Fugen.
3) The statistic design estimation method for the AH MOX fuel including the FEMAXI-ATR code was established.
4) It was verified by analysis and out-of-pile tests that the integrity of the AH MOX fuel was maintained over the whole irradiation period and under the design-base seismic conditions.
5) It has been confirmed that the thermal hydraulic and vibration characteristics were almost the same as those of the 28-MOX fuel.
6) PNC is developing the DU MOX-Gd fuel rod for the AH MOX fuel as the advanced MOX fuel technology.

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Development of High Burn-up MOX Fuel for ATR

Power Reactor & Nuclear Fuel Development Corporation
S.Uematsu, I. Kurita, N.Onuki, K.Kodaka, K. Terunuma

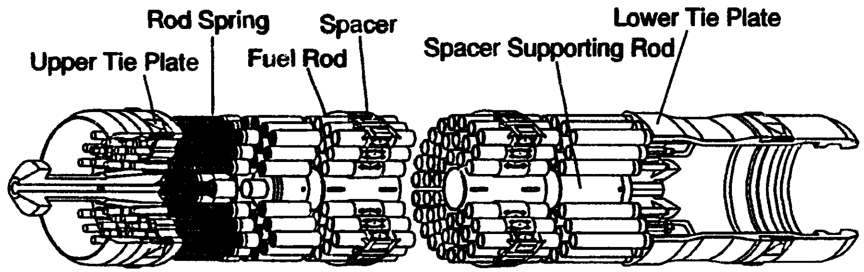
Development of ATR Fuel

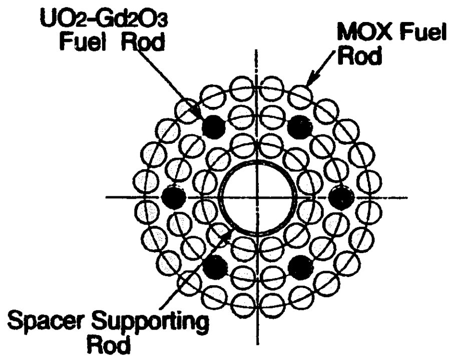
- The Power Reactor and Nuclear Fuel Development Corporation (PNC) is operating a prototype of theadvanced thermal reactor (ATR), called Fugen(165MWe), without a fuel failure,

- **About six hundred plutonium-uranium mixed oxide fuel (MOX fuel) assemblies have been loaded in Fugen since 1979.**

- **PNC is developing a high burn-up MOX fuel assembly for the ATR (AH MOX Fuel : assembly burn-up 55GWd/t) from the economical viewpoint to contribute to the development of MOX fuels for thermal reactors.**

AH MOX Fuel Assembly





- Maximum Assembly Burn-up : 55 GWd/t
- Operation Cycle : 15 Months × 4 Batches
- Maximum Linear Heat Rate : 40 kW/m
- Fuel Rod Diameter : 10.8 mm

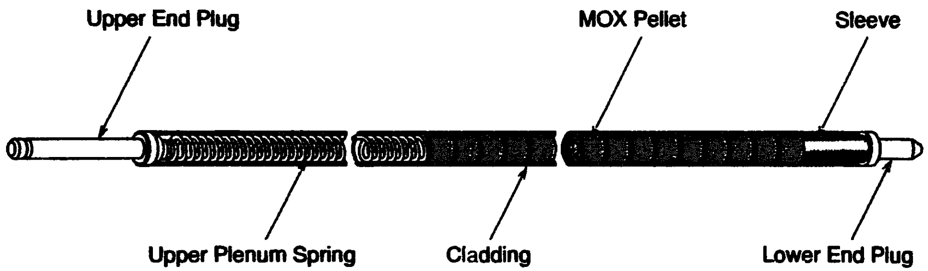


Fig. AH MOX Fuel Assembly

Table Fuel Specification

Item	54-Fuel-rod Assembly
1. Fuel Pellet	
Material	PuO_2-UO_2 UO_2-Gd_2O_3
Outer Diameter(mm)	9.10
Height(mm)	9.5
Density(%T.D.)	95.0
Shape	Solid with Dish & Chamfer
Pu Fissile Enrichment(wt%)	3.6~4.8
2. Fuel Rod	
Cladding Material	Zircaloy-2 with Zr-Liner
Outer Diameter(mm)	10.80
Inner Diameter(mm)	9.30
Diametral Gapsize(μm)	200
Fuel Stack Length(mm)	3,600
Filling Gas	He
Filling Gas Pressure(MPa)	0.5
3. Fuel Assembly	
Length(mm)	4,381
Bundle Outer Diameter(mm)	111.6
Rod Number	54
Inner-Ring Rod	12
Intermediate-Ring Free Rod	12
Intermediate-Ring Tie Rod	6
Outer-Ring Rod	24
Spacer Number	12

Design Evaluation Methods

Conventional Design Evaluation Method for ATR MOX Fuel Rod

Evaluation method

-Deterministic Method
All inputs are determined in order to obtain conservative results

Performance code: ATFUEL

-Fuel Performance Prediction with Safety Margin

Accumulation of Measurement Data (Production, PIE data)

New design Evaluation method for High Burn-up MOX fuel rods

Evaluation method

- Statistical Method
Input Data have Statistical Values

Performance code : FEMAXI-ATR

-Fuel Performance Prediction with Agreement with Data

FEMAXI-ATR code

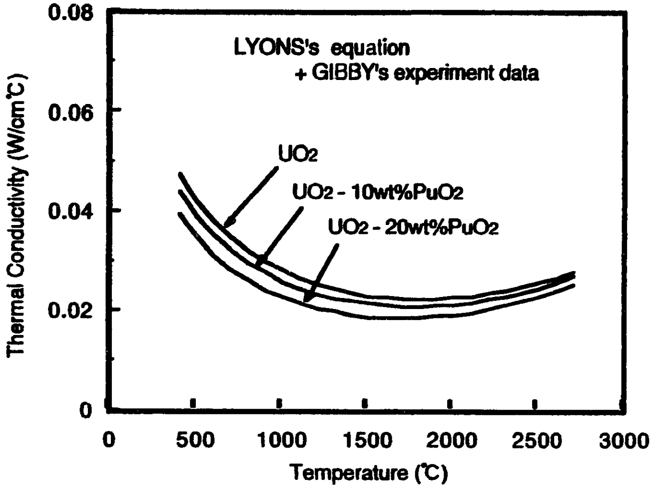


Fig. Fuel Pellet Thermal Conductivity

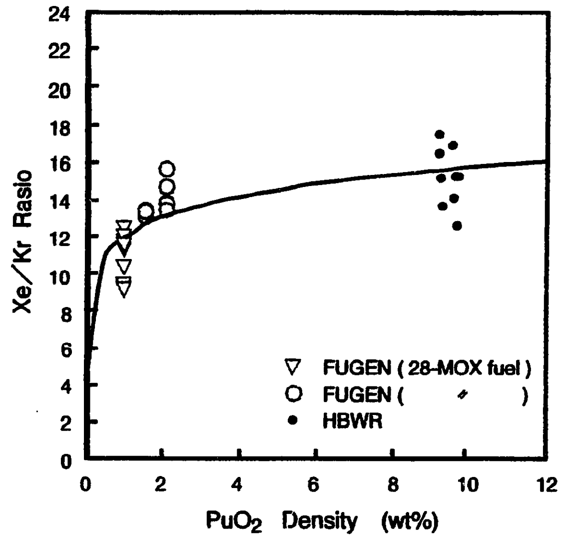


Fig. Xe / Kr Ratio vs. PuO2 Density

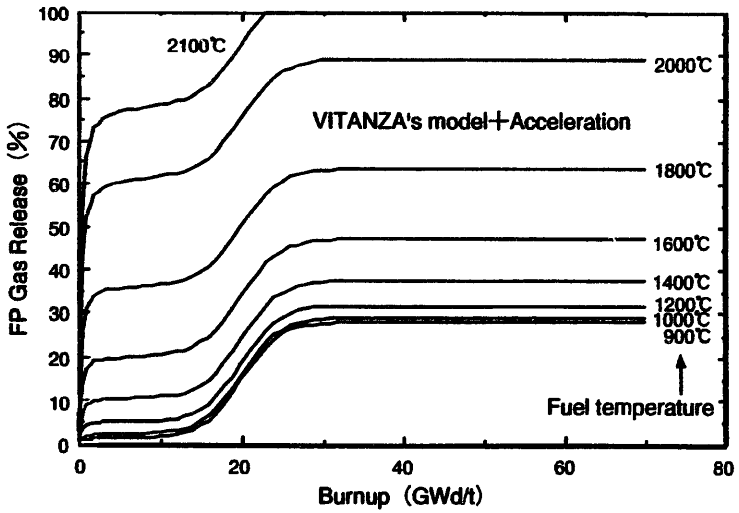


Fig. FP Gas Release vs. Burnup and Temperature

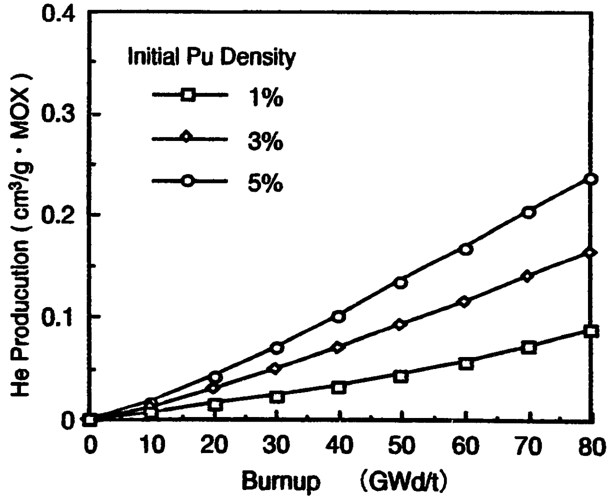


Fig. He Producution vs. Burnup

Verification of FEMAXI-ATR

Item	Number of Fuel Rod	Specification					Data Base
		Pellet-Cladding Gap (mm)	Pellet Density & Pu Content	Initial He Gas Pressure (MPa)	Rod Average Burn-up (GWd/t)	Maximum LHR (kW/m)	
Pellet Maximum Temperature	5	0.18 ~ 0.31	93~95 (%T.D.) 4.5~7.6 (wt%)	0.1 ~ 0.3	≦45	≦41	· Halden (MOX)
FP Gas Release Rate	80	0.09 ~ 0.33	92~96 (%T.D.) 0.7~9.6 (wt%)	0.1 ~ 2.1	≦62	≦55	· Halden (MOX) · SGHWR (UO_2, MOX) · FUGEN (MOX) · Japanease BWR (UO_2) · NFIR (UO_2) · BR-3 (UO_2) · Dodewaard (UO_2, MOX) · Inter Ramp (UO_2)
Rod Inner Pressure	72	0.09~ 0.33	92~96 (%T.D.) 0.7~9.6 (wt%)	0.1 ~ 2.1	≦62	≦55	· Halden (MOX) · SGHWR (UO_2, MOX) · FUGEN (MOX) · Japanease BWR (UO_2) · NFIR (UO_2) · BR-3 (UO_2) · Dodewaard (UO_2, MOX) · Inter Ramp (UO_2)
He Gas Valume	56	0.12~ 0.33	92~96 (%T.D.) 0.7~9.6 (wt%)	0.1 ~ 2.1	≦62	≦55	· Halden (MOX) · SGHWR (UO_2, MOX) · FUGEN (MOX) · Dodewaard (UO_2, MOX)
Cladding Diameter Change	42	0.09~ 0.30	92~96 (%T.D.) 0.7~3.3 (wt%)	0.1 ~ 2.1	≦60	≦55	· SGHWR (UO_2, MOX) · FUGEN (MOX) · NFIR (UO_2) · BR-3 (UO_2) · Inter Ramp (UO_2)

Verification of FEMAXI-ATR

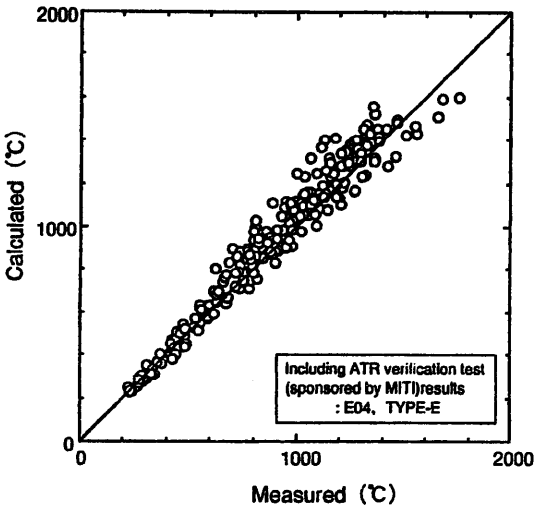


Fig. Fuel Centerline Temperature

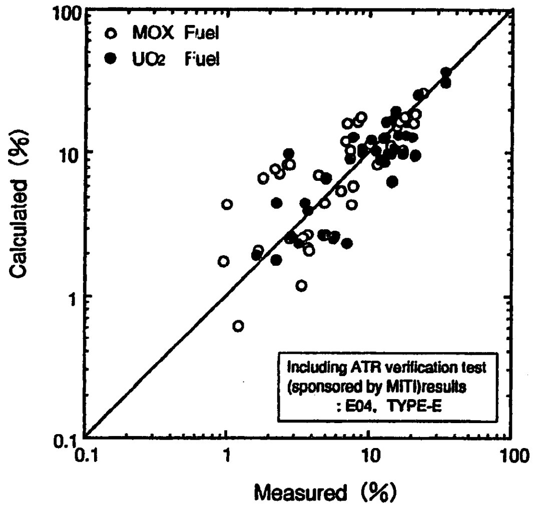


Fig. Fission Gas Release

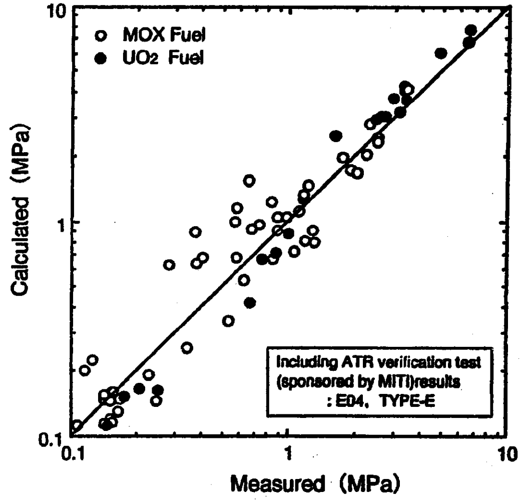


Fig. Fuel Rod Internal Pressure

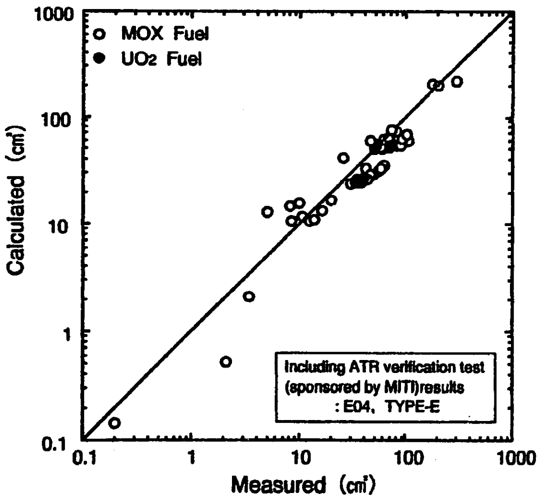


Fig. He Gas Release

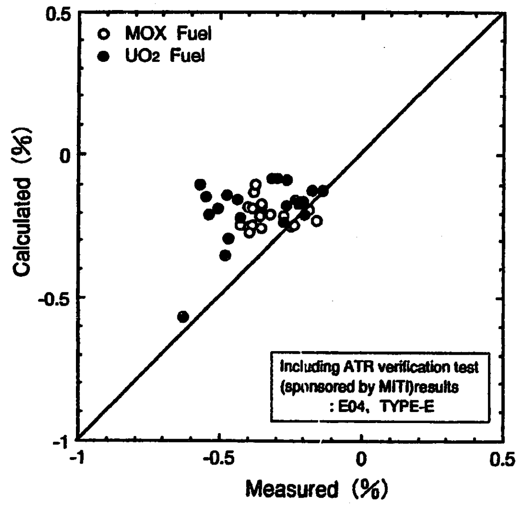
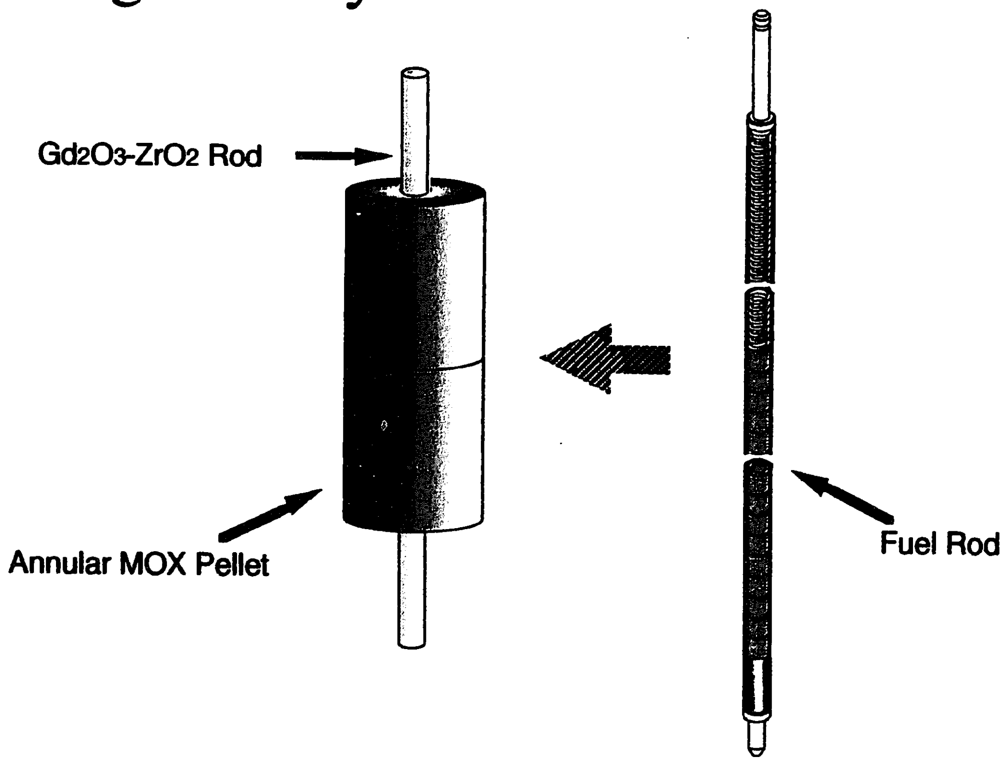


Fig. Fuel Rod Outer Diameter Change

Development of Duplex-Type MOX-Gd fuel

1)The manufacturing processes for the Duplex-Type MOX -Gd fuel pellets are not required.

2)The effect of gadolinium as burnable poison is milder in comparison with the case of using MOX fuel rod added Gd_2O_3 homogeneously.



Conclusions

1)PNC has the plan of the irradiation experiment with a few AH MOX fuels in Fugen.

2)The statistic design estimation method for the AH MOX fuel including the FEMAXI-ATR code was established.

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4)PNC is developing the DU MOX-Gd fuel rod for the AH MOX fuel as the advanced MOX fuel technology.