UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 17, 2001

LICENSEE: Duke Cogema Stone & Webster
Framatome Cogema Fuels
Duke Energy

FACILITY: Catawba Nuclear Plant, Units 1 and 2
McGuire Nuclear Plant, Units 1 and 2

SUBJECT: SUMMARY - DECEMBER 12, 2000, MEETING WITH OAK RIDGE NATIONAL LABORATORY (ORNL), FRAMATOME COGEMA FUELS, DUKE COGEMA STONE & WEBSTER, AND DUKE ENERGY, TO DISCUSS THE ORNL MOX FUEL RESEARCH AND DEVELOPMENT PROGRAM

On December 12, 2000, representatives of the ORNL, Duke Energy, Framatome Cogema Fuels (FCF), and Duke Cogema Stone & Webster (DCS), met with members of the U.S. Nuclear Regulatory Commission (NRC) staff at the American Museum of Science & Energy in Oak Ridge, Tennessee, to discuss the ORNL uranium-plutonium mixed oxide (MOX) fuel program. A list of attendees is provided in Enclosure 1. The information presented in the meeting was extensive, and the enclosed copy of the handouts provided during the meeting should be consulted for details since only brief summary comments are provided here. Throughout the presentations, similarities and differences between the ORNL test fuel and the mission fuel were noted.

The NRC staff opened the meeting by noting that interested members of the public are free to attend the meeting as observers but not to participate in the meeting pursuant to the Commission Policy Statement on "Staff Meetings Open to the Public; Final Policy Statement," 65 Federal Register 56964, 9/20/2000. A note to this effect was included in the staff's Notice of this meeting.

A summary of MOX fuel experience and its impact on the Advanced Test Reactor (ATR) test was presented. ORNL noted that this effort was originally focused on data collection but subsequent application of data to impurity issues (gallium) helped define the ATR test. The data were summarized in a report, ORNL/TM-13428, "Survey of Worldwide Light Water Reactor Experience with Mixed Uranium-Plutonium Oxide Fuel." This report was provided by the letter of Mr. Peter Hastings, DCS, to NRC, dated July 14, 2000.

A brief summary of the Fissile Materials Disposition (FMD) Program's gallium/cladding investigation was provided which included a discussion of the gallium/Zircaloy corrosion mechanisms and related topics. Two of the five conclusions from the FMD investigations were that large amounts of gallium react with clad as expected and trace amounts of gallium in fuel appear extremely unlikely to cause problems.

The measurement of achievable gallium separation from plutonium by means of the Purex solvent extraction method, as performed in the ORNL Radiochemical Engineering Development Center laboratory, was discussed. The resulting decontamination factors (DFs) were provided with the note that non-idealized effects, as well as improper process control, could result in lower DFs.

The weapons derived MOX fuel test program was discussed. This included sourcing the plutonium from dismantled weapons pit, fabrication of fuel pellets at Los Alamos, design of fuel pins and test assembly at ORNL, irradiation at the ATR at the Idaho National Engineering and Environmental Laboratory (INEEL), and periodic post-irradiation examinations (PIE) in the ORNL hot cells. The purpose of the tests was to investigate the behaviors of weapons-grade (WG) MOX fuel with and without treatment for removal of gallium. PIE is being carried out on capsules irradiated to various burnups. The PIE for intermediate withdrawal capsules (20.9 Gigawatt days/Metric ton (GWd/Mt)) has been completed and the PIE for 30 GWD/Mt has just begun. Further plans are to continue burnups for some capsules to beyond 30 GWd/Mt in 2001. ORNL's PIE results (through 30 GWd/Mt) were said to indicate excellent performance for WG derived MOX fuel with respect to densification and swelling, fission gas release, and cladding, and they showed no indication of gallium movement or any adverse effects of impurities.

Computer code support for safety analyses for MOX irradiation experiments in the INEEL ATR, including a list of nine codes, was also discussed. It was noted that ORNL is a member of the FRAPCON-3 users group and that this code has been modified at ORNL for use in predicting pellet densification and swelling and cladding dimensional changes.

A review of the ATR MOX fuel test PIE status for the 8, 21, and 30 GWd/Mt exposures was provided. Conclusions were that PIE is proceeding in a timely manner, observations were in accordance with predictions, no evidence of gallium migration or corrosion exists, and clad ductility testing is pending.

Fuel performance calculations in support of PIE were discussed. This included consideration of pellet cracking, fuel densification, and clad expansion and a description of MOX behavior by the fuel swelling models FRAPCON-3 and ESCORE.

ORNL presented information on its efforts to develop a well-suited test method for fuel cladding ductility tests. Ductility tests of irradiated clad from the ATR average power tests are planned after agreement on the test method (compressed plug method) is achieved. This cladding is unique because gallium was present in the fuel from the start. There will be no hydriding to mask any effect of gallium because the fuel pins are irradiated in an inert environment.

Reactor physics, criticality safety, and shielding analyses for MOX fuels were discussed. This included physics-related differences between reactor and weapons-grade plutonium and low enriched uranium, the use of burnable absorber rods in MOX assemblies, the ARIANE MOX destructive assay program, and joint FY 2001 study with FCF and physics models of the Catawba reactor with MOX fuel.

Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA) plutonium disposition reactor physics activities were discussed. The OECD/NEA provides a forum for cooperation among the 27-member countries. Several NEA-member countries have significant experience with MOX fuel. Activities to date have focused on benchmarking efforts in physics and fuel performance.

The OECD/NEA Task Force on Reactor-Based Plutonium Disposition (TFRPD) fuel performance benchmark activities was discussed. The Bureau of the OECD/NEA Nuclear

Science Committee established the TFRPD on December 15, 1998. It involves 50 participants from 29 organizations and 16 countries, with the objective that this task force could provide a forum and vehicle for international collaboration in the areas of weapons-derived MOX fuel performance and physics. It was stated that the highest priority activities should be experimental benchmarks. Also discussed was ORNL's participation in a benchmark comparison using the IFA-597 MOX experiment, offered by the Halden Reactor Project.

In conclusion, the NRC staff expressed its appreciation to the ORNL staff for the extensive efforts undertaken to support the meeting. The staff believes the meeting was beneficial, in that it provided an opportunity for the NRC staff, FCF and Duke Energy to discuss the programs undertaken by ORNL in support of the Fissile Materials Disposition Program in recent years. The staff also stated that, with respect to any prospective use of the ORNL data presented in the meeting in support of the mission fuel design, the relationship between the ORNL data and the actual mission fuel would need to be addressed in the licensing application to use the mission fuel. The staff would expect to address this subject, including the relationship between weapons-grade and reactor-grade MOX, in further detail during its review of the licensing application to use the mission fuel.

/RA/
Robert E. Martin, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413, 50-414, 50-369 and 50-370

Enclosures: 1. Attendance List
2. Handouts

cc w/enclosures: See next page

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Enclosures: 1. Attendance List
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cc w/enclosures: See next page

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NRC/ORNL MOX RESEARCH and DEVELOPMENT MEETING

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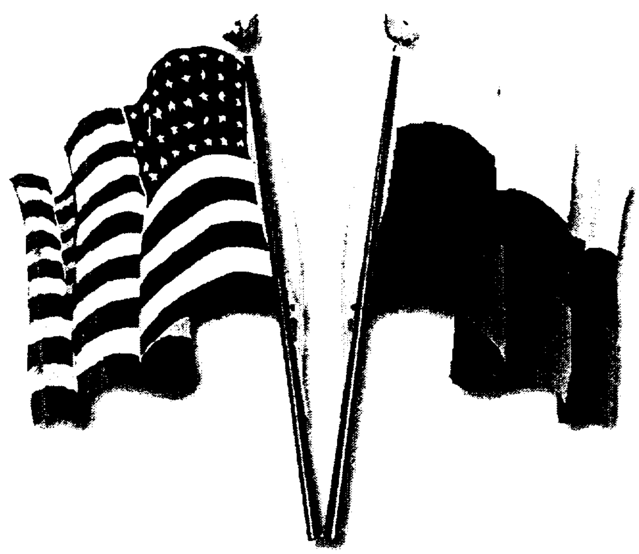
Don Spellman	ORNL
Ralph Caruso	NRC/NRR/SRXB
Margaret Chatterton	NRC/NRR/SRXB
Undine Shoop	NRC/NRR/SRXB
Harold Scott	NRC/RES
Sudhamy Basu	NRC/RES
Bob Martin	NRC/NRR/DLPM
George Meyer	Framatome Cogema Fuels (FCF)
Laurence Losh	FCF
Michael Bale	FCF
Glenn Copp	Duke Energy
Steven Nesbit	Duke Energy
Richard Clark	Duke Cogema Stone & Webster
Brian Cowell	ORNL
Jess Gehin	ORNL
Bill Hendrich	ORNL
Steve Hodge	ORNL
Scott Ludwig	ORNL
Claire Luttrell	ORNL
Gary Mays	ORNL
Robert Morris	ORNL
Mike Muhlheim	ORNL
Larry Ott	ORNL
Joe Pace	ORNL
Trent Primm	ORNL
Claud Pugh	ORNL
Theresa Stovall	ORNL
Ken Thoms	ORNL

Donald Williams	ORNL
Kent Williams	ORNL
Emory Collins	ORNL
Terry Yahr	ORNL
Patrick Rhoads	DOE
Jon Thompson	DOE
David Alberstein	Los Alamos National Laboratory
David Campbell	Consultant to ORNL
Delwin Mecham	INEEL/Bechtel BWXT Idaho, LLC
Edwin Lyman	NCI
Don Moniak	Blue Ridge Environmental Defense League

NRC/ORNL MOX Fuel Program Research and Development Meeting Oak Ridge, Tennessee December 12, 2000



Fissile Materials Disposition Program

Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37831
managed by
UT-BATTELLE, LLC
for the
U.S. DEPARTMENT OF ENERGY
under contract DE-AC05-00OR22725

AGENDA

ORNL MOX Fuel Program
Research and Development Meeting with NRC Staff
Oak Ridge, Tennessee

Tuesday, December 12, 2000

8:00 a.m.	Meeting Ground Rules	R. E. Martin (NRC)
8:10 a.m.	Welcome/Agenda Format	D. J. Spellman
	ORNL FMDP Overview	S. R. Greene
8:20 a.m.	Compilation of MOX Use History	B. S. Cowell
8:45 a.m.	Gallium/Clad Interaction Experiment	Dr. R. N. Morris
9:15 a.m.	Decontamination Factor Experiment	Dr. E. C. Collins
9:45 a.m.	In-Reactor MOX Fuel Test Program	Dr. S. A. Hodge
	— Background	
	— Test Objectives	
	— Irradiation History	
	— Ongoing Efforts and Future Plans	
	— Code Support for Safety Analyses	L. J. Ott
11:45 p.m.	Lunch	
12:30 p.m.	PIE of ATR MOX Fuel	Dr. R. N. Morris
2:00 p.m.	Fuel Performance Codes	L. J. Ott
2:30 p.m.	MOX Fuel Clad Ductility Testing	G. T. Yahr
3:00 p.m.	Reactor Physics Codes Analysis	
	— Reactor Physics Evaluations	R. T. Primm
	— OECD/NEA Benchmarking	Dr. J. C. Gehin/L. J. Ott
4:30 p.m.	Wrap-up	D. J. Spellman

ORNL MOX Fuel Program Research and Development

D. J. Spellman
Oak Ridge National Laboratory

Presented at

ORNL MOX Fuel Program
Research and Development Meeting
Oak Ridge, Tennessee
December 12, 2000

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Agenda Format

- Logistics
- Agenda Flow
  - What relevant information is available about past mixed-oxide (MOX) fuel experience?
  - How much of an issue is gallium?
  - How well can we get rid of it?
  - What will small gallium concentrations do to fuel performance and cladding ductility?
    - What testing have we done?
    - What have we discovered?
    - What codes analyses have been done?
    - How well do predictions match experimental results?
  - Discussions

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Summary of MOX Fuel Experience and Its Impact on ATR Test

B. S. Cowell
S. E. Fisher
R. T. Primm
Oak Ridge National Laboratory

Presented at

ORNL MOX Fuel Program
Research and Development Meeting
Oak Ridge, Tennessee
December 12, 2000

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U.S. MOX Fuel Experience Is Relevant to FMDP

- Effort was originally focused on data collection.
- Subsequent application of data to impurity issues helped define Advanced Test Reactor (ATR) test.
- Data were eventually summarized in ORNL report.
- Additional applications have followed.

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Early in FMDP, Questions About MOX Experience Arose Frequently

- Four "vendor reports" prepared under DOE Oakland contracts focused specifically on surplus plutonium
- One provided a good summary of applicable MOX history
- DOE requested a summary of U.S. and international experience to provide a convenient reference for answering frequent questions.
- ORNL attempted to update and supplement the GESMO summary, using its format as a guide.

During Database Development, Impurity Issues Rose to Prominence

- Of known impurities, gallium was viewed as most significant.
- Three-pronged approach was chosen:
  - Nonaqueous purification (TIGR) at LANL
  - Gallium-clad interaction studies
  - Irradiation test with typical impurities
- At the time, an irradiation test could not prejudice subsequent procurement, so tests were designed to be vendor-neutral.

Database Effort Strongly Influenced ATR Test Goals

- Determined that extensive experience in similar reactors exists, including some with high ^{239}Pu content
- Most useful experiment would tie the gallium-clad tests to irradiation conditions, while providing a test result with actual surplus plutonium.
- Test characteristics include
  - Vendor neutral test design due to procurement sensitivity,
  - Irradiation conditions as prototypic as possible,
  - Use of actual surplus plutonium, including dry purification,
  - Use of generic but representative MOX fuel fabrication process.
- ATR test will be described in more detail by Dr. Hodge.

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Collection and Organization of U.S. and International MOX Publications

- Relatively minor effort was needed to "harvest" the data, which leveraged residual value of significant past investments from DOE and its predecessors.
- Past U.S. MOX experience utilized high fissile (up to 91.4%) plutonium–similar isotopics to WG plutonium.
- A structure and format summarizing the overall experience was needed.
- Documentation was needed to summarize the literature and help educate all of the stakeholders on the history and use of MOX in reactors.

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Database Summarized in ORNL/TM-13428

- Describes U.S. and international operational experience
- Intended to provide roadmap to relevant data
- Includes substantial bibliography
- Organizes information in MS EXCEL table:
  - "Row" input is by rod, assembly, or batch at U.S. plant
  - "Column" describes ~ 50 characteristics such as
    - Reactor specifics
    - MOX assembly design and isotopics
    - Fuel fabrication technique
    - Maximum linear heat generation, average and peak burnup
    - Summary of destructive and nondestructive examinations
    - Noted performance features
    - Reference (source of published information)
- U.S. commercial and Saxton reactor irradiations were mapped in EXCEL table (PRTR, EBWR, and MTR not covered).
- CONCLUSION: While most U.S. fuel performance data are eclipsed by non-U.S. experience, the reactor physics information gleaned from the past is relevant to the FMDP.

U.S. MOX Fuel Irradiation History Summary

Reactor	Dates of Irradiation	Number of MOX Assemblies (rods)	Burnup (MWd/MT) Maximum Average Assembly (Peak MOX Pellet)	Examinations	Comments	FMDP Approach
Ginna (PWR)	1980-1985	4 (716)	39,800 (Data not found)	None	Assemblies intact (82% fissile plutonium)	**Prepared proposal for fuel exam**
Quad Cities-1 (BWR)	1975-1980s	5 (48)	39,900 (57,000)	Destructive and nondestructive exams—core physics oriented	Well-documented EPRI program (80 and 90% fissile plutonium)	**Neutronics benchmark analyzed**
Big Rock Point (BWR)	**1969-late** 1970s	53 (1248)	~20,000 est. (30,200)	Destructive and nondestructive examinations	Little documentation located	**Not analyzed**
San Onofre-1 (PWR)	1970-1972	4 (720)	19,000 (23,500)	Some destructive examinations	Documents for PIE have been found	**Neutronics benchmark analyzed**
Dresden-1 (BWR)	1968 - early 1970s	15 (103)	~19,000 (~25,000)	Data not found	Little documentation located	**Not analyzed**
Saxton (PWR research reactor)	1965-1972	9 (638)	Many reconstitutions (51,000)	Many fuel performance destructive examinations and physics tests	Relatively well documented; fuel performance data are abundant (91.4% fissile Pu)	**Neutronics analysis still under way**
Miscellaneous test reactors (EBWR, PRTR, MTR, ETR)	1960s/1970s	1000s rods	Data not found	Variety of destructive exams	Capsules/rods irradiated; little historical research	**Not analyzed**

*NOTE: All data in this table have not been confirmed; some values are estimated.

Database Supported FMDP Physics Studies

- Ginna MOX rods
  - No fuel examinations were conducted.
  - Proposed spent MOX parameter measurements for Ginna, but ARIANE data eclipsed this need.
- San Onofre-1 MOX rods
  - Performed isotopic analysis (ORNL/TM-1999/108, September 1999).
- Quad Cities—90%/80% fissile MOX rods
  - Performed power and isotopic analysis (ORNL/TM-13567, April 1998).
- Saxton: Cold criticals and irradiations, 91.4% fissile
  - Texas/TAMU analysis of physics data to be published in Spring 2001

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Collection and Publication of Database Has Benefited FMDP

- Helped demonstrate low technical risk of MOX fuel option for plutonium disposition
- Documented continuity of current application with previous U.S. MOX fuel experience
- Provided publicly available reactor physics data to Russia and the United States—a prelude to more extensive neutronics studies
- Provided springboard for dealing with WG plutonium/gallium issues

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A Brief Summary of the FMDP Gallium/Cladding Investigation

Dr. R. N. Morris
Oak Ridge National Laboratory

Presented at

ORNL MOX Fuel Program
Research and Development Meeting
Oak Ridge, Tennessee
December 12, 2000

Fuel Gallium Concentration History

- Originally, it was believed that MOX fuel fabricated from weapons material might contain high levels of gallium – 500+ ppm. Thus, corrosion due to gallium could be an issue, and a program was started in 1994 to investigate this situation.
- During the course of the program, it was discovered that almost all of the gallium leaves the fuel in the reducing atmosphere of the sintering furnace; fuel levels drop to ~5 ppm (problem for furnace internals).
- A decision was made in 1997 that the plutonium oxide would be polished to remove impurities and to control the powder characteristics. Thus, gallium concentrations will be below ~1 ppm.

Programs to Investigate Gallium Corrosion Were Established at ORNL and Texas A&M

- ORNL investigated corrosion mechanisms and observed gallium/Zircaloy corrosion under tensile conditions at temperature.
  - Tests were conducted by Dane Wilson.
- Texas A&M ion-implanted gallium into Zircaloy-4 to simulate events at greater than thermal energy.
  - Tests were conducted by Ron Hart.

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Task Needs Were Derived from NUREG-0800, 10 CFR 50, and Experience

- Design limit on gallium concentration
  - No fuel damage due to gallium for all irradiation conditions
  - Reasonable safety margins
- Understanding potential damage mechanisms
  - Aid to NRC reviewers, reactor owners, and fuel fabricators
- Radioactivity release during DBA bounded by current methods
  - Number of fuel failures not increased by gallium impurity

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Task Objectives for Out-of-Reactor Clad Testing

- Determine possible gallium-specific damage mechanisms
- Determine how damage mechanisms are affected by controllable system parameters
- Determine effects on clad mechanical properties under a range of parametric conditions

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Liquid Gallium Reacts with Zircaloy

- Liquid gallium reacts with Zircaloy at >300°C, which leads to the formation of intermetallic products that can deform the clad.

300°C　　350°C 200 h　　400°C

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Low Gallium Concentrations (ppm level) Refocus the Issue

- General corrosion is no longer an issue because of the relatively small amount of gallium.
  - Even gross migration will not concentrate enough gallium for GC.
  - No signs of migration in ATR experiments
- Liquid metal embrittlement (LME) was investigated.
  - This form of environmentally induced embrittlement can induce cracking or loss of ductility.
- Corrosion/mechanical tests were conducted.
  - Constant extension rate tests were performed.
  - Tensile testing was conducted.
  - Gallium and Ga_2O_3 were used at various temperatures.
  - Ce_2O_3 was used as fuel surrogate.

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Investigation Was Conducted in Two Phases

- Phase I, liquid gallium tests:
  - Corrosion – determine mechanisms and dependence on temperature and time
  - LME – constant extension rate tests to determine if Zircaloy is embrittled by liquid gallium
  - Corrosion/mechanical – determine the effect of corrosion on mechanical properties
- Phase II, gallium oxide tests (Ga_2O_3):
  - Corrosion – determine mechanisms and dependence on temperature, time, and concentration
  - Corrosion/mechanical– determine the effect of corrosion on mechanical properties
- Clads tested: Zircaloy-2, Zircaloy-4, and Zirlo

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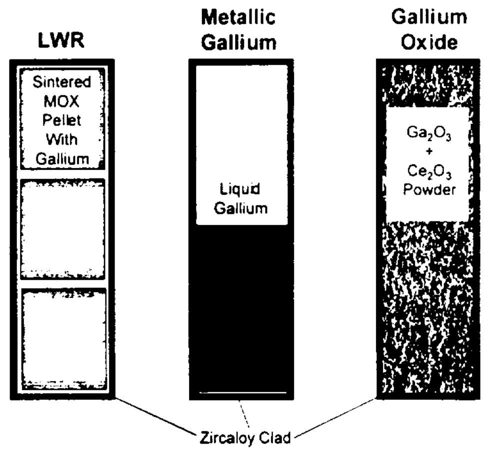
Phase I and II Testing Configurations



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None of the Investigated Mechanisms Lead to Credible Clad Damage at Low Gallium Concentrations

- Three mechanisms were investigated
  - Intermetallic compound formation (ICF)
  - Liquid metal embrittlement (LME)
  - Grain boundary corrosion (GBC)
- ICF was found to be temperature dependent with no distortion of the cladding up to 300°C
- ICF was mass-limited with no distortion of the cladding
  - Ten times more gallium than anticipated
  - Temperatures to 500°C

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None of the Investigated Mechanisms Lead to Credible Clad Damage at Low Gallium Concentrations (continued)

- The ORNL tests indicate no LME of Zircaloy by gallium.
- No indications of GBC were found.
- Metallographic, fractographic, dimensional, and microprobe posttest analyses were performed.

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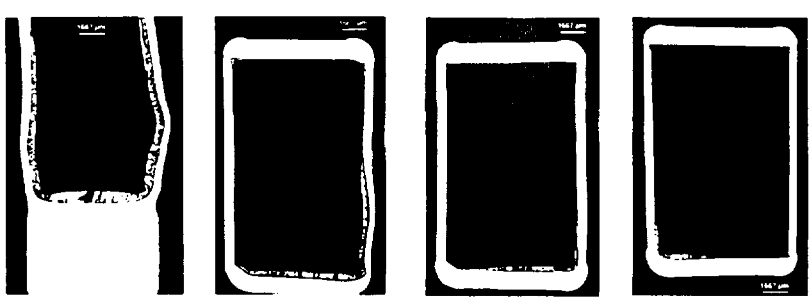
Distortion due to ICF from Gallium in Mission Fuel Is Unlikely

- No specimen distortion with ten times excess gallium (2 kg fuel at 2.5 ppm ⇒ 0.005 g Ga)



0.6 g gallium | 0.2 g gallium | 0.09 g gallium minor distortion | 0.05 g gallium no distortion

200 h at 500°C

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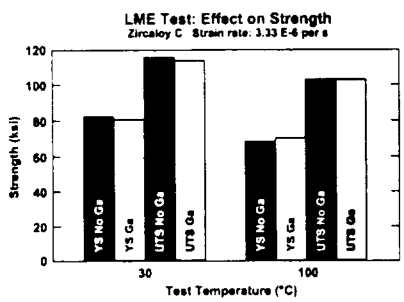
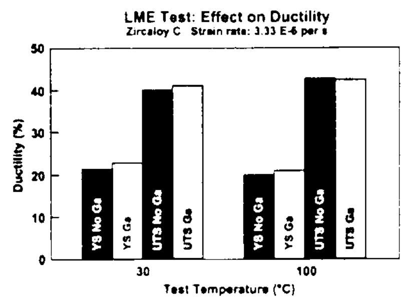
Constant Extension-Rate Tests Show No Evidence of LME Zircaloy by Gallium





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Phase II Tests Employed Ga_2O_3 in Ce_2O_3 Matrix

- Clad temperatures below 500°C result in negligible migration of gallium into the clad.
  - Gallium migration into the Zircaloy is negligible at 1% Ga_2O_3 concentration levels in the Ce_2O_3 matrix.
  - Gallium migration into the Zircaloy is negligible at 300°C and small at 500°C for 100% Ga_2O_3 (oxide layer formed).
- At 700°C reactivity was greater:
  - Formation of oxide layer and gallium-rich zone
  - Surface cracking of the Zircaloy
  - Distortion of the clad
  - Creep and heat treatment effects

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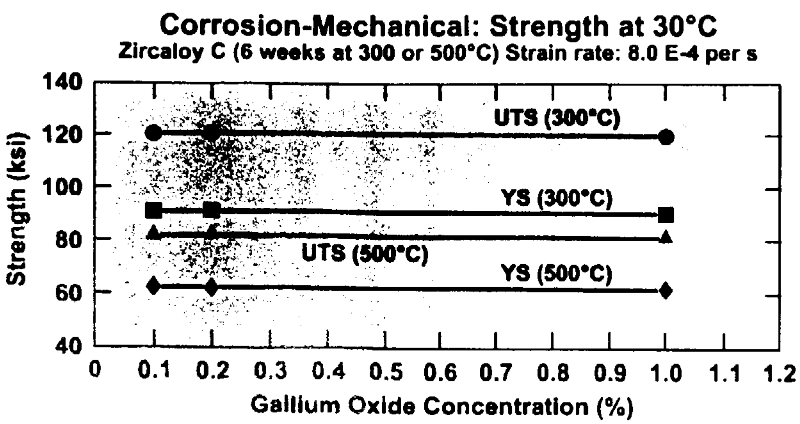
No Visual Differences Exist Between Ga_2O_3 Concentration Tests at 500°C

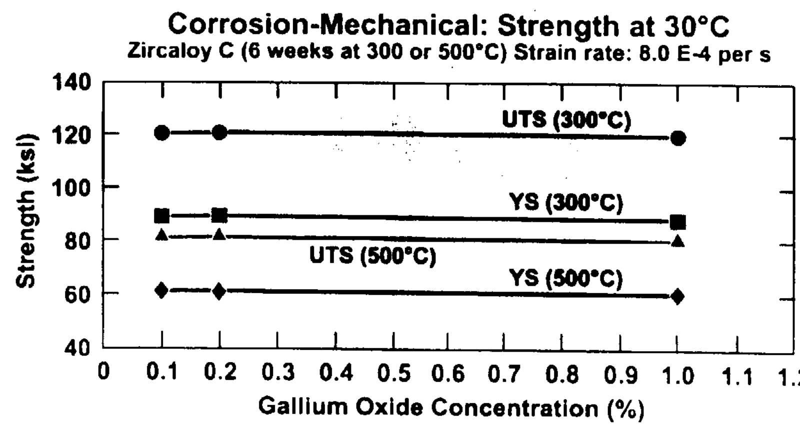
0.1% Ga_2O_3 0.2% Ga_2O_3 1% Ga_2O_3

Phase II Corrosion/Mechanical Tests

- After exposure at 300°C, there was no significant change in mechanical properties.
  - True for both mechanical tests at room temperature and at 300°C
  - All three clad materials
- After exposure at 500°C, there was some change in mechanical properties (room temperature).
  - Lower ultimate strength
  - Higher ductility
  - Effect found to be due to heat treatment at 500°C
    - No difference in gallium specimens when compared to unexposed controls
- No tests were conducted at 700°C.
  - Distortion (creep)

Phase II Corrosion/Mechanical Tests Indicate No Changes in Strength or Ductility





Conclusions

- Large amounts of gallium react with clad as expected.
- No evidence of liquid metal embrittlement or grain boundary attack exists.
- Only intermetallic compound formation was observed.
- No effect was observed on mechanical properties other than the loss of material.
- Trace amounts of gallium in fuel appear extremely unlikely to cause problems.

Results Are Documented

- ORNL material testing
  - D.F. Wilson et al., *Interactions of Zircaloy Cladding with Gallium: Final Report*, ORNL/TM-13684 (September 1998).
- Texas A&M beam work
  - M.K. West, *Gallium Interactions with Zircaloy*, ANRCP-1999-2 (January 1999).

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Measurement of Achievable Plutonium Decontamination from Gallium by Means of Purex Solvent Extraction

Emory D. Collins
David O. Campbell
L. Kevin Felker
Oak Ridge National Laboratory

Presented at

ORNL MOX Fuel Program
Research and Development Meeting
Oak Ridge, Tennessee
December 12, 2000

Objectives and Test Conditions

- Objective was experimental measurement of gallium decontamination factor (DF) under idealized process conditions
- High DF was expected from expert opinions - problem was to be able to measure accurately the amount of gallium not separated from plutonium
- Initial concentration of gallium in WG-plutonium is ~1%, or 1E7 parts per billion (ppb)
- Radiotracing with gallium-72 (14 h half-life) produced by short irradiation in the High Flux Isotope Reactor (HFIR) rabbit facility shortly before conducting the solvent extraction tests was utilized
- Batch, single-stage extraction test was performed, followed by single-stage scrubbing test and finally, a single-stage plutonium strip test
- Tests were conducted in Radiochemical Engineering Development Center (REDC) laboratory, adjacent to HFIR. On-site REDC analytical chemistry lab enabled immediate analysis of test samples

The High Flux Isotope Reactor and Radiochemical Engineering Development Center at Oak Ridge National Laboratory

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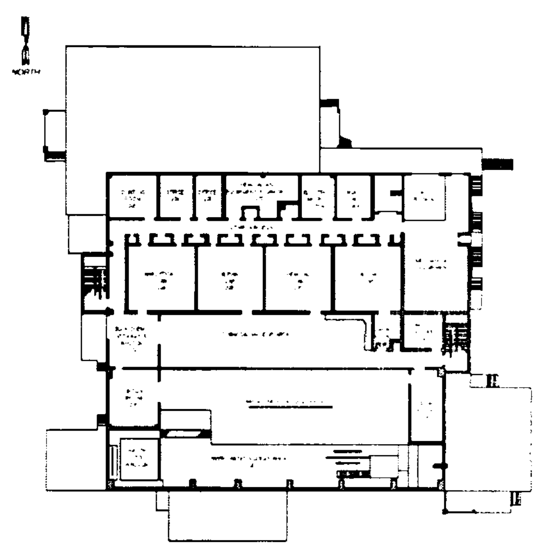
Layout of the REDC Building 7920 at the Oak Ridge National Laboratory



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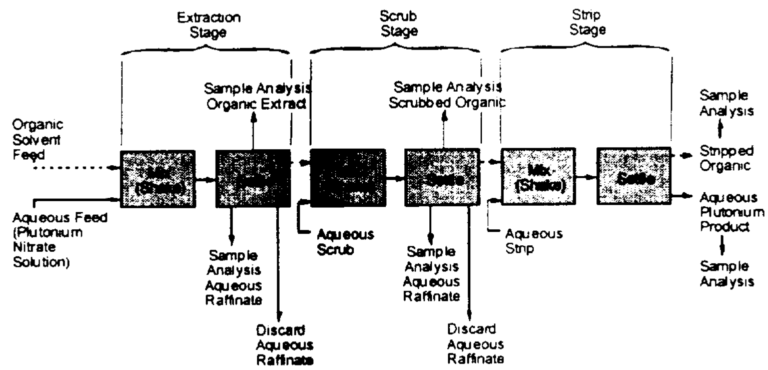
ORNL DWG 2000-346

Diagram of Single Stage Extraction, Scrub, and Strip Tests



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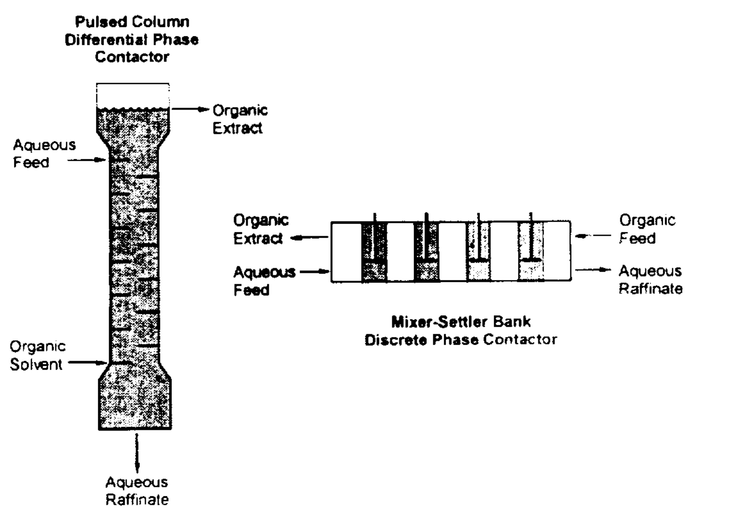
Continuous Multistage Countercurrent Liquid-Liquid Solvent Extraction



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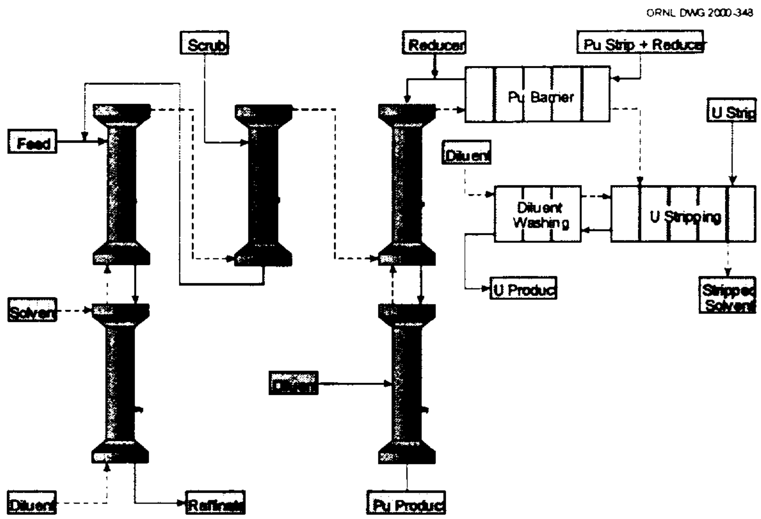
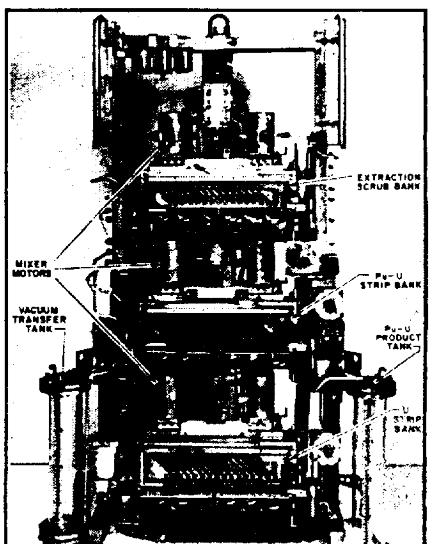


DCS Solvent Extraction Purification Flowsheet



SETF SOLVENT EXTRACTION RACK

Gallium Decontamination Test Results (extraction/scrub/strip)

Phase	Volume (mL)	HNO_3 (M)	Ga (Bq/mL)	K_d	5.15-MeV peak (Bq/mL)	Percentage of plutonium		
						Experimental	SEPHIS	
Extraction								
Aqueous feed	5.00	4.5	1.50×10^7 (±1.7%)		1.04×10^8	100	100	
Organic feed	5.00	0.03^b	0		0			
Organic extract	5.2^b	0.6^b	4.0(± 26%)	2.8×10^{-7}	9.3×10^7	92	93.3	3.6×10^6
Aqueous raffinate	4.8^b	2.9^b	*c*		1.1×10^7	10	6.7	
Organic feed	$5.0^{b,d}$	0.6^b	4.0(± 26%)		9.3×10^7	100	100	
Aqueous scrub	1.63	1.5	0		0			
Scrubbed organic	5.0^b	0.4^b	<2.3	<0.51	8.0×10^7	86	93.8	$>6.4 \times 10^6$
Aqueous raffinate	1.7^b	2.2^b	8.8		2.1×10^7	8	6.2	
Plutonium strip								
Organic feed	$4.8^{b,d}$	0.4^b	<2.3		8.0×10^7	100	100	
Aqueous strip	10.0	*e*	0		0			
Stripped organic	4.65^b	0.08^b	<0.4		1.75×10^6	2	1	
Aqueous plutonium product	10.1^b	0.4^b	<1.3		3.35×10^7	90	99	

[a] Gallium DF compared with feed, normalized per unit plutonium.

[b] Values based on SEPHIS calculation from input streams.

[c] Not measured.

[d] An estimated 0.2 mL of organic was lost to samples after the extraction step and after the scrub step.

[e] 0.1 M HNO_3-0.4 M HAN

Source: E.D. Collins, [illegible] Campbell, and L.K. Felker, "Measurement of Achievable Plutonium Decontamination from Gallium By Means of Purex Solvent Extraction," ORNL/TM-1999/312, January 2000.

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Gallium Removal Requirement

MOX $UO_2/PuO_2 \cong 20$

Zn in UO_2 (ppm)	Average Ga in MOX (ppb)
250	100
100	40
40	16
25	10
5	2

Proposed Specification for PuO_2

Ga in $PuO_2 \leq$ 100 ppb

Required Pu Decontamination Factor (DF) from Gallium $\cong 10^5$

Experimental Results and Conclusions Obtained from Gallium DF Tests

- Single-stage extraction gave gallium DF of 3.7E6
- DF after single-stage extraction and single-stage scrub was greater than 6E6
- DF after single-stage plutonium strip was greater than 5E6
- Concluded that, with multistage operation under idealized conditions, gallium DF would be greater than 5E6, reducing gallium concentration in plutonium product to less than 10 ppb
- Noted that the non-idealized effects of impurities, entrainment, crud formation, etc., as well as improper process control, could result in lower DFs

Weapons-Derived MOX Fuel Test Program in the Advanced Test Reactor

Fissile Materials Disposition Program
Average Power Test

Dr. S. A. Hodge
Oak Ridge National Laboratory

Presented at

ORNL MOX Fuel Program
Research and Development Meeting
Oak Ridge, Tennessee
December 12, 2000

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Mixed-Oxide (MOX) Fuel Irradiation Demonstration for the Department of Energy Fissile Materials Disposition Program (FMDP)

- Plutonium From Dismantled Weapons Pit
- Light Water Reactor Fuel Pellets Made at Los Alamos
- Fuel Pins and Test Assembly Designed at ORNL
- Assembled and Irradiated at the Advanced Test Reactor (ATR) at Idaho
  - Eleven Fuel Pins Irradiated
  - Six Inch Fuel Length – 15 Pellets Each
- Periodic Post-Irradiation Examinations (PIE) at ORNL Hot Cells (Building 3525)

Background: Weapons-Derived Plutonium Differs From Reactor-Grade Material in Isotopic Content and Level of Impurities (Additives For Weapons Purposes)

Purpose: Demonstrate Satisfactory Performance of MOX Fuel Fabricated From Weapons Components

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Presentation Outline

1. Average Power Test background
   - Components and configuration
   - Pellet characteristics and temperatures
2. Test objectives
   - Statement of original goals
   - Gallium concentrations
   - Significance of no clad hydriding
3. Irradiation history
   - Phases I, II, and III
   - PIE reports
4. Ongoing efforts and future plans
   - Equalization of burnups below 30 GWd/MT
   - Confirmatory PIE at 40 GWd/MT
   - Carry three capsules to 50 GWd/MT

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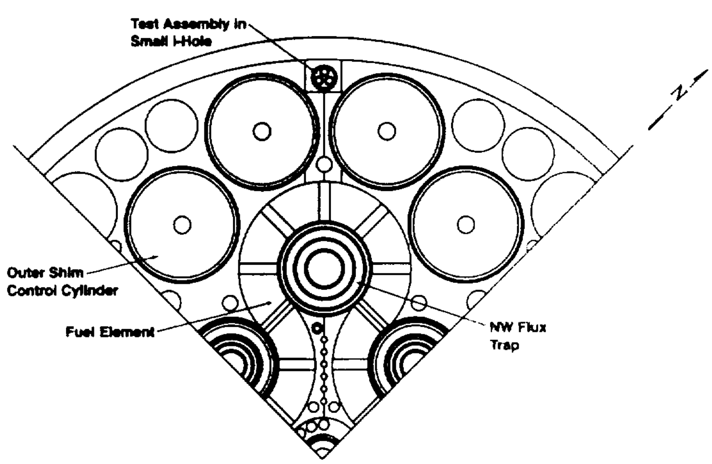
The Test Irradiations Occur in the Small I-Holes of the ATR Reflector



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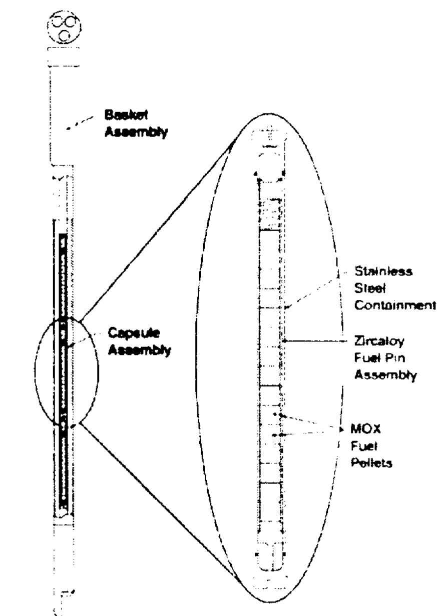
The Test Assembly Permits Simultaneous Irradiation of Nine Capsules



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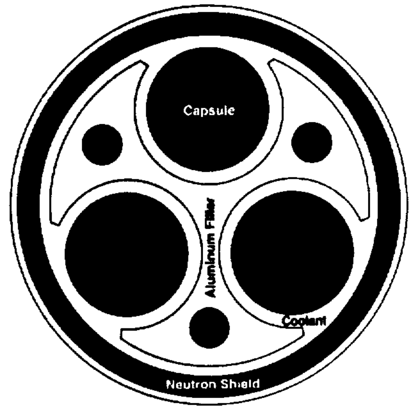
Three Capsule Assembly Columns Fit within the Basket Assembly; a Basket Shield Adjusts the Linear Heat Rates



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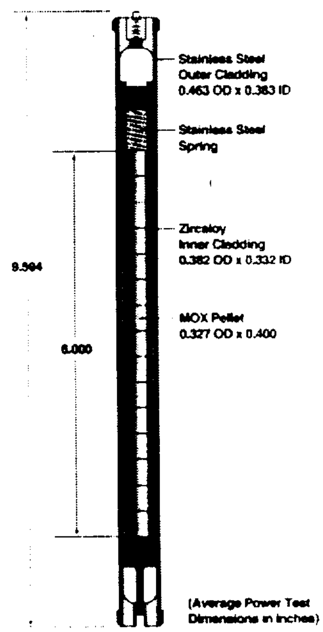
Each Capsule Assembly Contains One Zircaloy Fuel Pin with 15 Fuel Pellets



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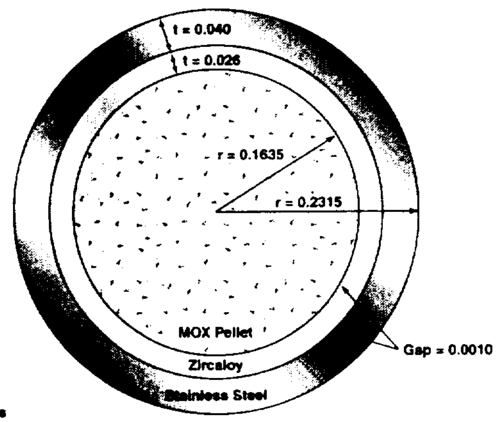
ORNL 2000-1644C EFG

Test Assembly Containment Is Provided by a Stainless Steel Capsule Surrounding Each Sealed Zircaloy Fuel Pin Assembly



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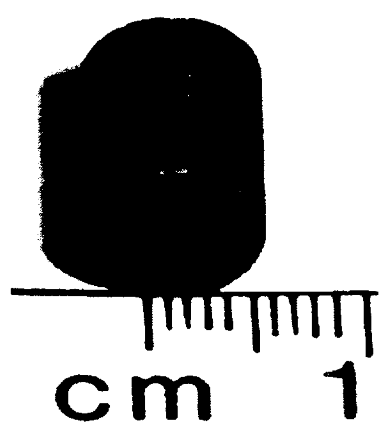
ORNL 2000-1645C EFG

MOX Pellet as Fabricated at LANL for the Average-Power Test



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The APT Includes MOX Pellets With and Without Treatment for Removal of Ga; Irradiation Was Initiated February 5, 1998

Fuel type[a]	Description[b]	Initial feed	Pu to PuO_2 conversion	PuO_2 purification	Intended burnup (GWd/MT)
1	5% WG Pu	1% Ga WG Pu	3-Step Hydox	None	8/20/30 40/50
2				Thermal	

[a]Each MOX fuel pin contains about 3.60 g Pu.
[b]UO_2 diluent is derived from ammonium diuranate (ADU).

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Part 2 Test Objectives

a. Statement of original goals
b. Gallium concentrations
c. Significance of no clad hydriding

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The Fuel Performance Demonstration Is Accomplished through Achievement of Several Goals

1. Utilize Pu derived from weapons components in a light-water reactor (LWR) environment
2. Contribute experience with fuel containing Ga to the MOX data base
3. Initiate irradiation of LWR WG MOX fuel in CY 1997
4. Exercise the infrastructure
   a. Convert Pu metal from weapons components to oxide
   b. Fabricate MOX pellets
   c. Transport fresh fuel
   d. Irradiate the fuel
   e. Transport irradiated fuel
   f. Perform PIE

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Fuel Performance Demonstration Goals (continued)

5. Fabricate MOX fuel that is as prototypic of current commercial RG MOX fuel as possible given limitations on equipment, schedule, and available feed materials

6. Trace the evolution and behavior of Ga and other impurities unique to the dry-processed surplus Pu throughout the fabrication process and the irradiation

7. Show that the test fuel residual Ga levels are acceptable even at temperatures in excess of those at peak fuel pins in commercial power reactors

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The Test Requirements Direct the Approach toward Attainment of the Established Goals

1. All test fuel is produced in the TA-55 facility at LANL.
2. Issues related to inclusion of burnable poisons in MOX fuel are not addressed.
3. Test fuels are fabricated to meet generic LWR MOX fuel pellet specifications developed by ORNL using process specifications developed by LANL.
4. The plutonium for the WG MOX test fuel is derived from one or more weapon components. The material pedigree is documented.
5. The uranium diluent procured for test fuels is characterized at LANL to verify the accompanying material certifications.
6. Test conditions reproduce LWR operating temperatures (clad and centerline) to the extent possible.

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Test Requirements (continued)

7. Fuel dimensions, cladding, fuel specifications, and burnups are selected in a manner that does not bias ongoing programmatic procurement activities.
8. The test fuels are removed from the reactor at selected points within a range of burnups.
9. Domestic facilities are used for fabrication, irradiation, and PIE.
10. The tests investigate the behaviors of WG MOX fuels with and without treatment for removal of gallium.

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ORNL 2000-1710C EFG

The Pellet Technical Specification Impurity Limits Are Representative of Commercial UO_2 Fuel

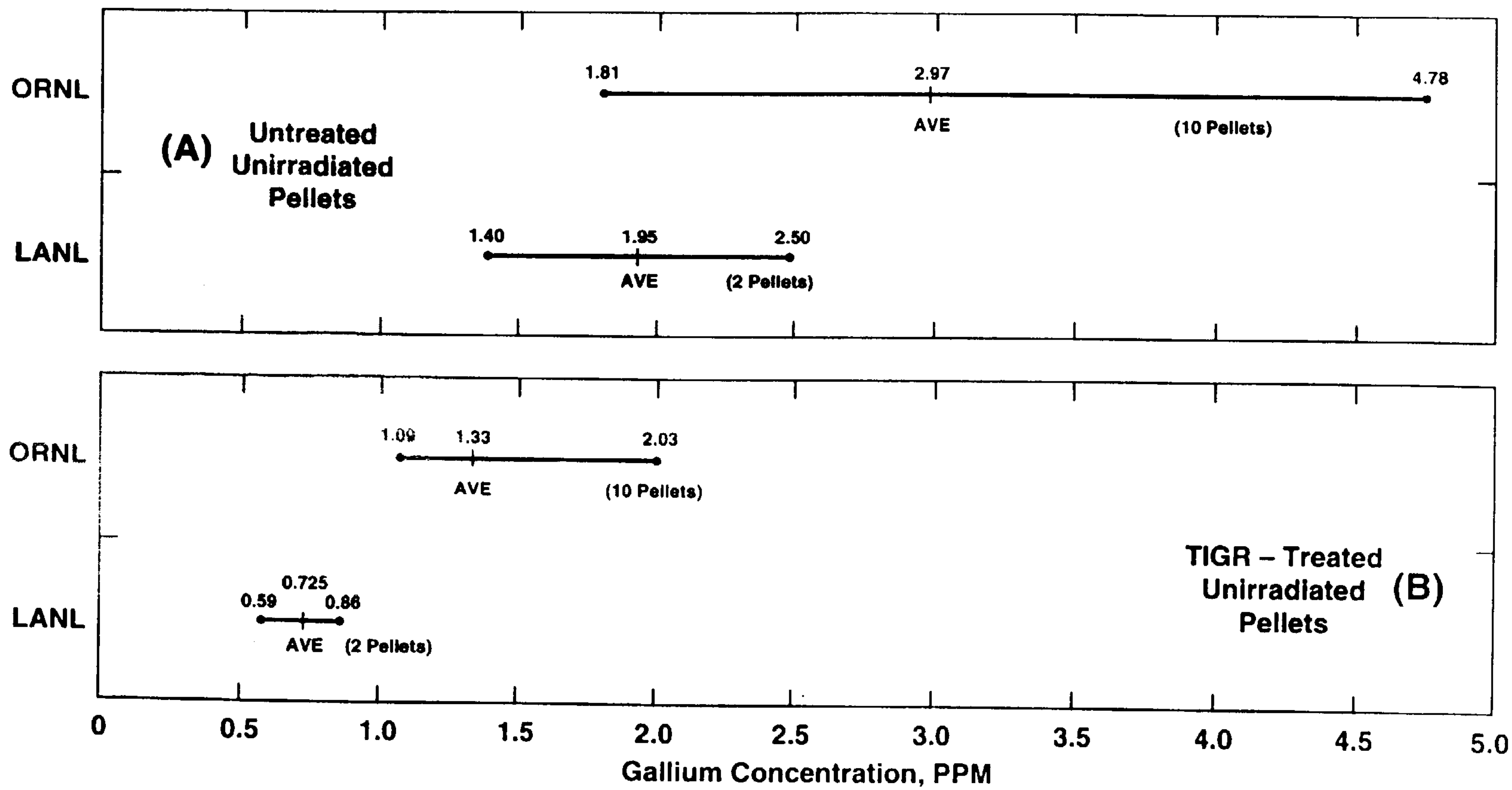
Element	Impurity limit (ppm)	Element	Impurity limit (ppm)
Aluminum	100	Iron	500
Boron	1	Lead	400
Cadmium	1	Magnesium	200
Calcium	250	Manganese	250
Carbon	250	Molybdenum	250
Chlorine	50	Nickel	250
Cobalt	250	Nitrogen	100
Dysprosium	1	Samarium	1
Europium	1	Silicon	250
Fluorine	50	Silver	25
Gadolinium	1	Tantalum	250
Gallium	As fabricated	Thorium	250
Hafnium	1	Tin	250
Hydrogen	1	Zinc	250
Indium	10	Total	2500

- Pellet density 95% of theoretical
- O/M ratio 1.995 – 2.010

Ga Concentrations Were Markedly Reduced during Pellet Preparation for the APT

Fuel	Ga concentration (ppm)	
	With thermal treatment	Without thermal treatment
Pu metal	10,000	10,000
PuO_2 powder	~8,800	~8,800
PuO_2 powder after treatment	~170	—
MOX	~8.5	~440
Sintered pellet (averages)		
LANL (2 pellets)	0.7	2.0
ORNL (10 pellets)	1.3	3.0

Unirradiated Fuel Batch Gallium Concentrations Are Recorded in MOX Average Power Test Fuel Pellet Initial Gallium Content ORNL/MD/LTR-182 Issued March 2000



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SAH-18

Recent Findings (NUREG/CR-6534) for the NRC Explain PWR Clad Cracking at High Fuel Burnup

- Failure ductilities observed at less than 1% strain criterion established by NRC Standard Review Plan
- Zirconium hydrides:
  - Produced by clad retention of 15% of waterside oxidation hydrogen
  - Form circumferentially while wall stress remains compressive
  - Precipitate radially when wall stress becomes tensile
  - Crack initiation sites at 300–400 ppm
  - Reduce ductility near zero
- PWR clad wall stress becomes tensile following hard pellet-clad contact after 40–45 GWd/MT
- Major limit to reaching higher burnups with current PWR designs

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Clad Exposure in the ATR MOX Test Is Representative of Commercial Experience. System Pressure Is Less, so the Clad Strains Outward.

Parameter	Units	BWR[a]	PWR[a]	ATR MOX
Discharge Burnup	GWd/MT	55	60	50
Exposure Time	days	1800	1500	1350–1500
Fast Neutron Fluence	cm^{-2}, E > 1 MeV	1E22	1E22	1.5E21
Clad Temperature	°C	280–320	290–400	220–370
Clad External Pressure	bar	70	158	1
Clad Hoop Stress	MPa	–60	–100	+3

[a]BWR and PWR values from Garzarolli, et. al ASTM STP 1295, 1996

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The ATR Tests Are Ideal for Revealing Any Effects of Gallium Since There Is No Masking by Hydride-Induced Clad Damage

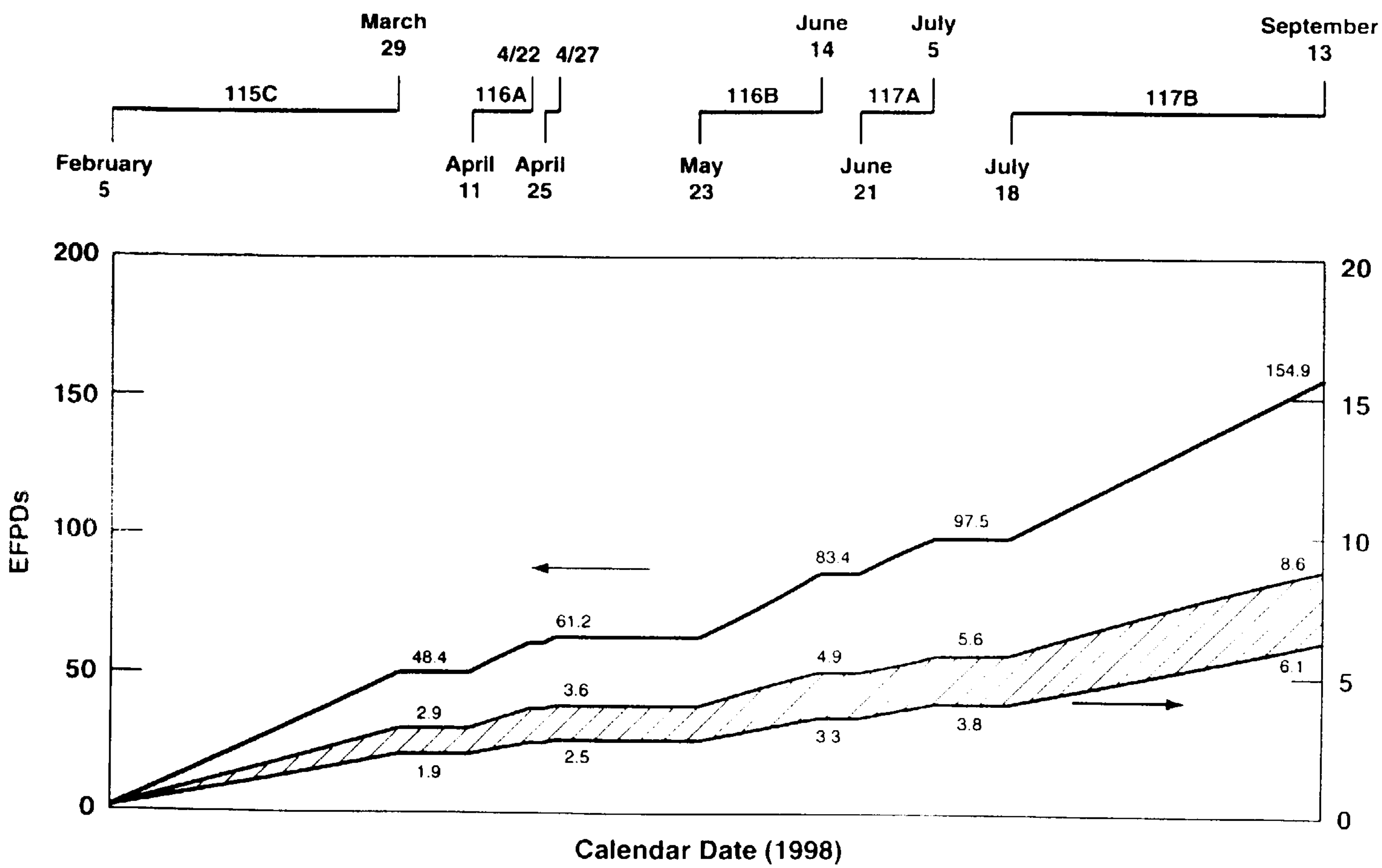
- Reduction of clad ductility to values near zero requires formation of crack initiation points
  - Hydrogen content in excess of solubility limit to form zirconium hydride
  - Tensile hoop stress so that hydrides form in radial direction
- Without hydrides, have only effects of fast flux
  - Similar to cold-working
  - Irradiated clad should withstand uniform strain of 3%–5%
- ATR tests
  - Have no hydrides
  - Prototypic integrated fast flux
  - Clad tensile stress
  - 0.6 to 4.8 ppm gallium in fuel

Part 3 Irradiation History

a. Phases I, II, and III

b. PIE reports

Phase I of the Average-Power Test Irradiation Was Completed on September 13, 1998, Leading to the First Withdrawal of Two Capsules for PIE



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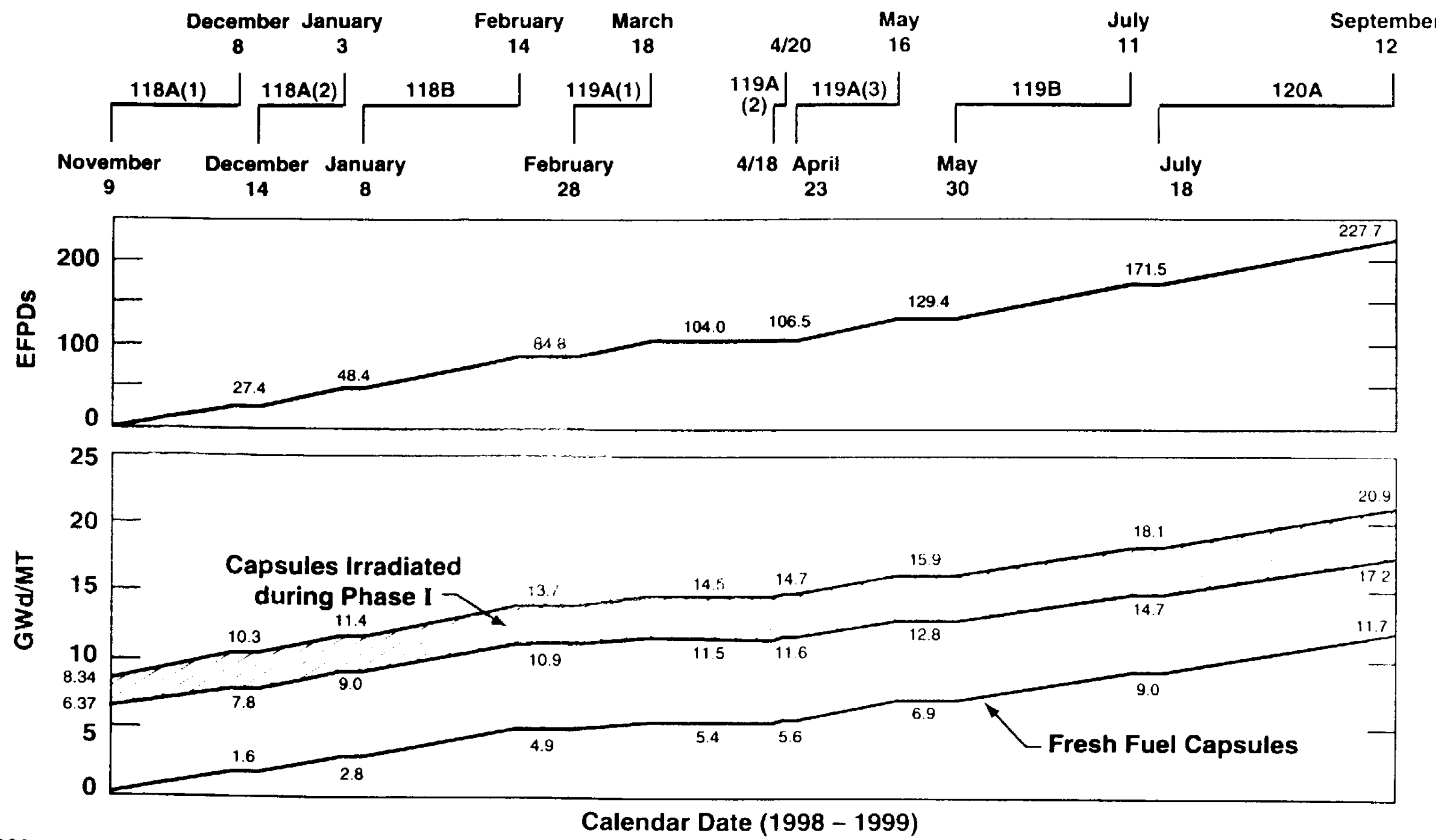
SAH-23

Early PIE: The Test MOX Fuel Behaved Normally so that Irradiation of Sister Capsules May Continue

Important Findings:

1. No significant difference between the performance of the TIGR-treated and the untreated MOX fuels.
2. Pellet cracking is evident, but considered normal in view of the thermal cycling experienced during the Phase I irradiation.
3. Gamma scans and burnup analyses are in accordance with the predictions of the MCNP code. The observed fuel swelling is as expected from the best-estimate CARTS code predictions.
4. Any transport of gallium from fuel to clad was limited to no more than about one fourth of that initially in the fuel.
5. This test fuel prepared with weapons-derived plutonium has behaved in accordance with European experience.

Lead Capsules 2 and 9 Attained 21 GWd/MT during the Second APT Irradiation Phase



July 26, 2000

PIE for Intermediate-Withdrawal Capsules (20.9 GWd/MT) Has Just Been Completed

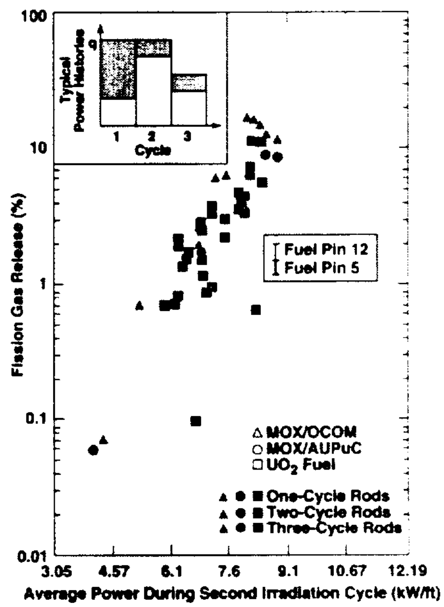
- Two capsules withdrawn mid-September 1999
- Fuel pins 5 (untreated) and 12 (TIGR)
- Destructive PIE began February 2000
  - First measurement of fission gas pressure
  - Smaller pellet-clad gap during irradiation
  - More likely that any effects of gallium would be observed
- Intermediate PIE Reports
  - "Quick Look" provided March 2000
  - Final report issued November 2000

Intermediate PIE: The Fuel Exhibits Normal Swelling, Densification, and Fission Gas Release. There Is Evidence of Outward Clad Creep.

Important Findings:

1. No significant difference exists between the performance of the TIGR-treated and the untreated MOX fuels.
2. Gamma scans and burnup analyses are in accordance with MCNP code predictions. Observed fuel swelling is as expected from CARTS code predictions.
3. The gas release fraction (implied from pressure and ^{85}Kr activity measurements) is within expectations based on the European MOX experience.
4. Pellet densification is prototypic of commercial MOX fuel.
5. Clad expansion is about 0.3%.
6. No evidence of gallium migration to the clad.
7. This test fuel prepared with weapons-derived plutonium is behaving as expected.

Intermediate-Withdrawal Fuel Pins Exhibit Gas Release Fraction in Accordance with Linear Heat Generation Rate Experience



Fission gas release as reported by Goll et al. (1993)

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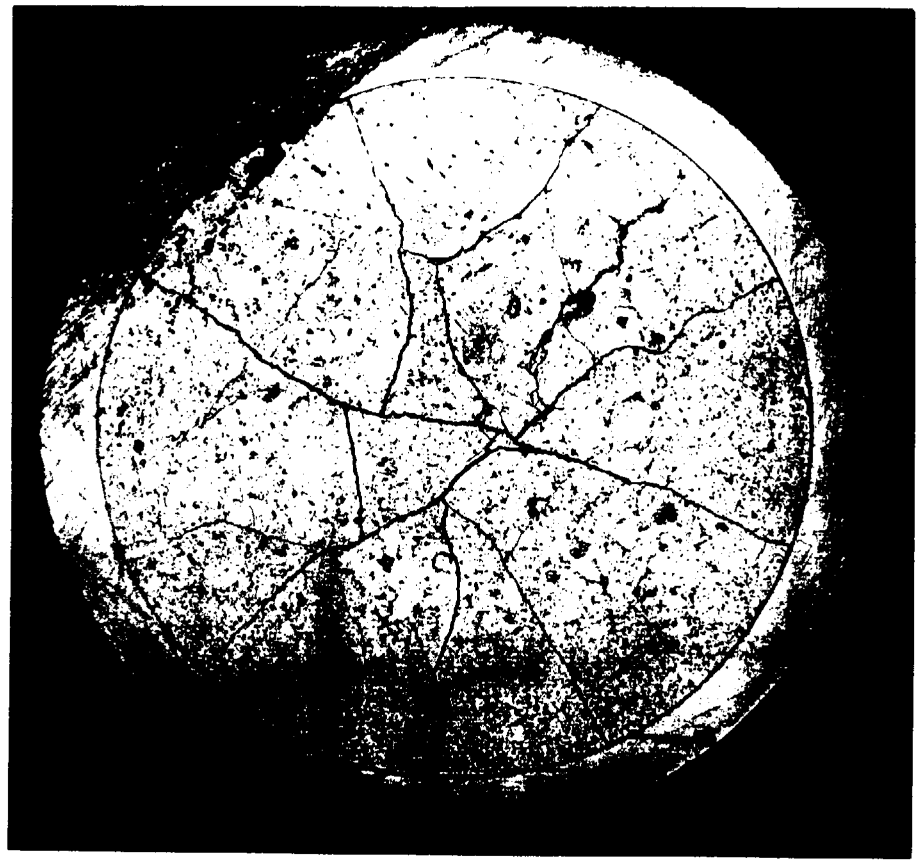
Fuel Section from Intermediate – Withdrawal (21 GWd/MT Burnup)



MXR82612 Sample #6147 400 μm

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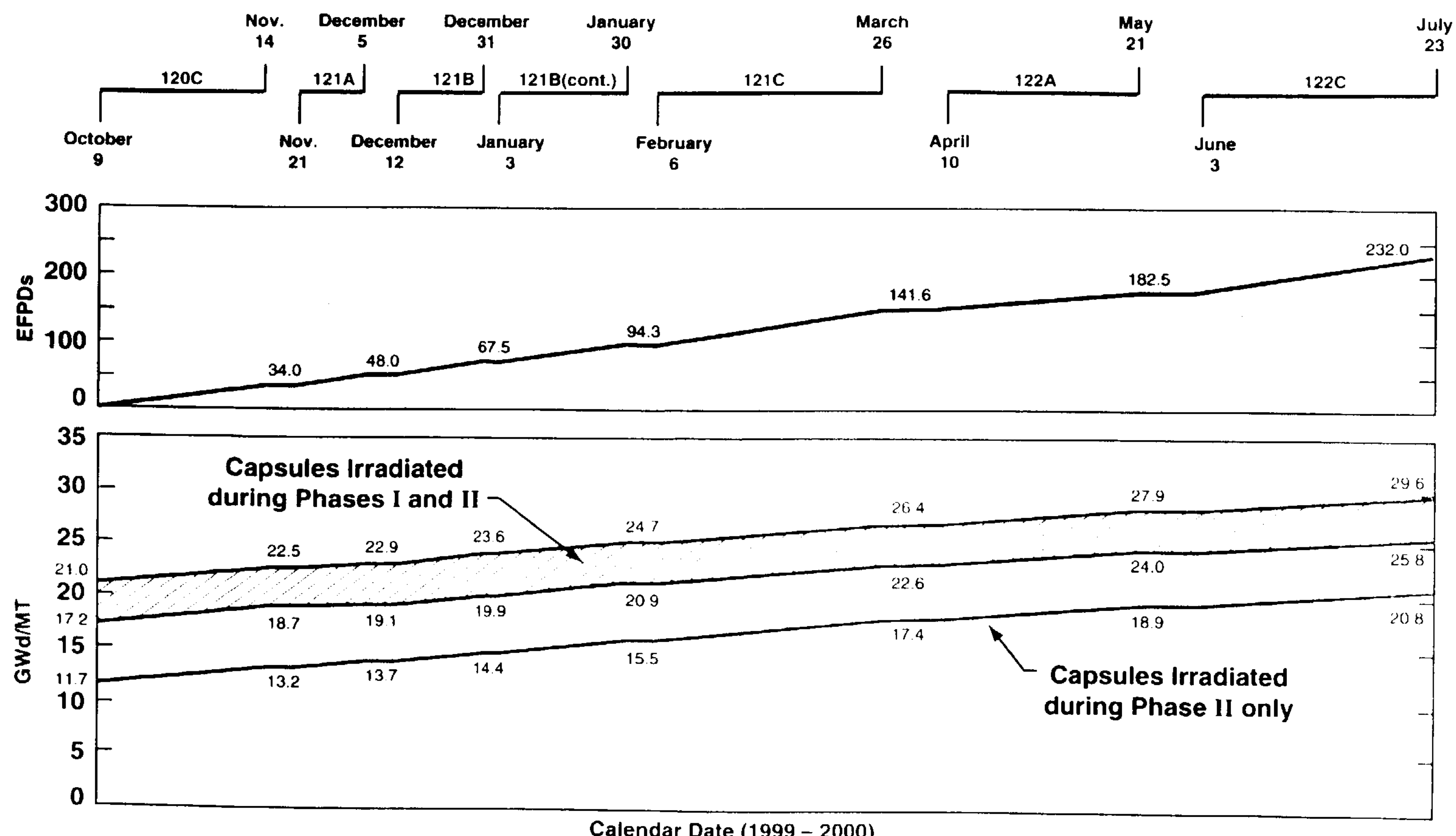
SAH-29

APT Irradiation Phase III Part 1 Brought Lead Capsules 3 and 10 to 30 GWd/MT



July 27, 2000

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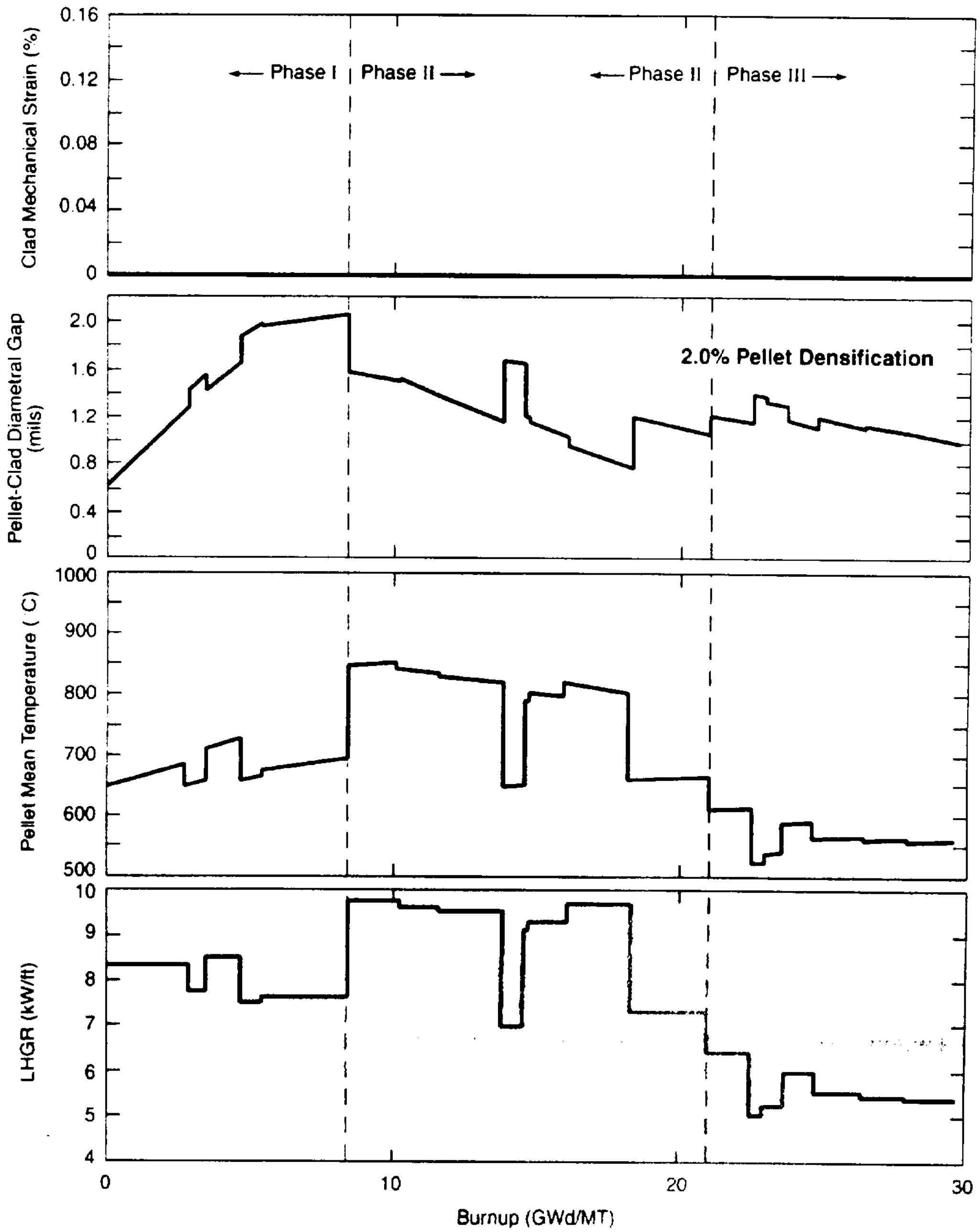
SAH-30

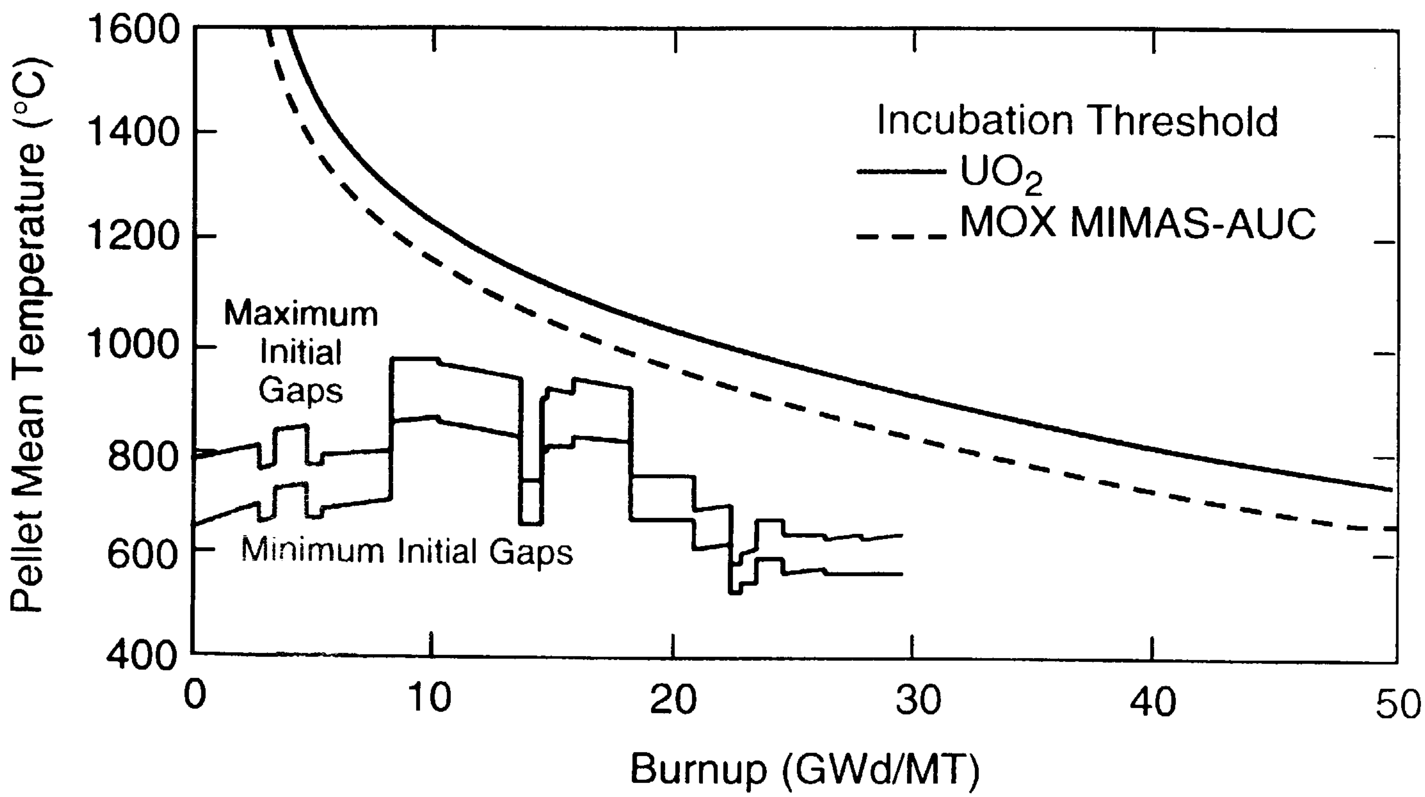
Capsules 3 and 10 Best Estimate: Pellet-Clad Gap Closure Has Not Yet Occurred



PIE for Capsules Withdrawn at 30 GWd/MT Has Just Begun

- Irradiation completed July 23, 2000
- Fuel pins 6 (untreated) and 13 (TIGR)
- Nondestructive PIE steps completed
  - Capsule gamma scans
  - External dimensions
  - Surface temperatures
- I-131 activity requirement (5 mCi) met November 18
  - Capsule and fuel pin pressure measurements
  - Fuel pin external dimensions
- Quick Look report
  - Information required before exceeding 30 GWd/MT
  - Issue February 2001

Calculations Indicate that Fuel Temperatures Have Not Exceeded the Incubation Threshold for Accelerated Fission Gas Release.



Source: ***Fuel Qualification Plan – European Experience***, **DCS Presentation to NRC, October 12, 2000.**

Part 4 Ongoing Efforts and Future Plans

a. Equalization of burnups below 30 GWd/MT

b. Confirmatory PIE at 40 GWd/MT

c. Carry three capsules to 50 GWd/MT

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ORNL 2000-1[illegible]C EFG

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The Average-Power Test Examines the Effects of the Gallium Impurity

- Will virtually eliminate gallium via wet polishing
- Target is 10 ppb, but practical considerations may limit guarantee to < 1 ppm
- Successful completion of this MOX irradiation test will demonstrate that 1-5 ppm in fuel is acceptable

☞ Desirable to carry test irradiation beyond highest burnups (~ 44 GWd/MT) expected by FMDP

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ORNL 2000-1[illegible]1C EFG

UT-BATTELLE

Irradiation of Some Capsules to Levels Beyond 30 GWd/MT Will Begin in January 2001

- Extension of burnup beyond 30 GWd/MT
  - Take five capsules to higher burnups
  - Two withdrawn for PIE at 40 GWd/MT
  - Three to reach 50 GWd/MT
- Requires
  - Additional Safety Analyses
  - Design review and approval by SORC
  - PIE results at 30 GWd/MT
  - Confirmatory PIE at 40 GWd/MT

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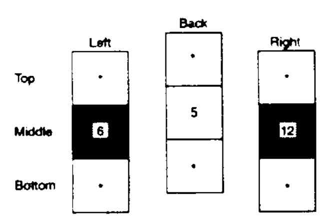
Now Increasing Lag Capsule Burnups While 30 GWd/MT PIE Is Underway

- Initial burnup range:
  - Capsules 4 and 13 at 28.9 GWd/MT
  - Capsule 5 at 25.8
  - Capsules 6 and 12 at 20.8
- Three lowest-burnup capsules placed at test assembly midplane
  - Continue irradiation toward 30 GWd/MT
  - Permitted by existing approved test plan
- Irradiate for three ATR cycles (4 months)
  - 114 EFPDs
  - Capsules 6 and 12 in front positions
  - Capsule 5 in rear



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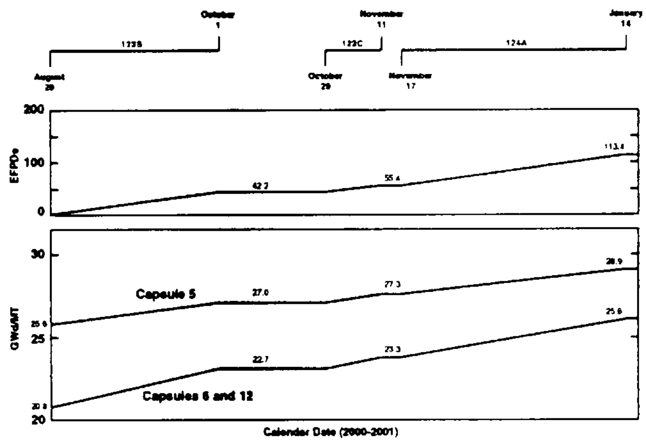
Lag Capsules 5, 6, and 12 Approach 30 GWd/MT during APT Irradiation Phase III Part 2



November 21, 2000

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ORNL 2000-1884C EFG

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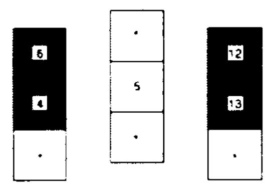
Capsule Loading Configuration and Withdrawal Schedule for Phase IV

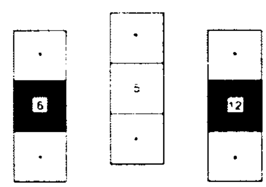
Phase IV Part 1

- To 40 GWd/MT
- Begin January 2001
- Move to southwest I-hole in December
- End April 2002
- Remove capsules 4 and 13 for PIE



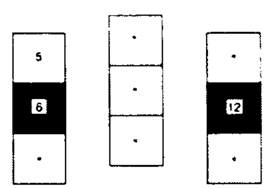
Phase IV Part 2

- Burnup equalization below 40 GWd/MT
- End August 2002



Phase IV Part 3

- Begin September 2002
- Reach 45 GWd/MT in April 2003
- Reach 50 GWd/MT in December 2003



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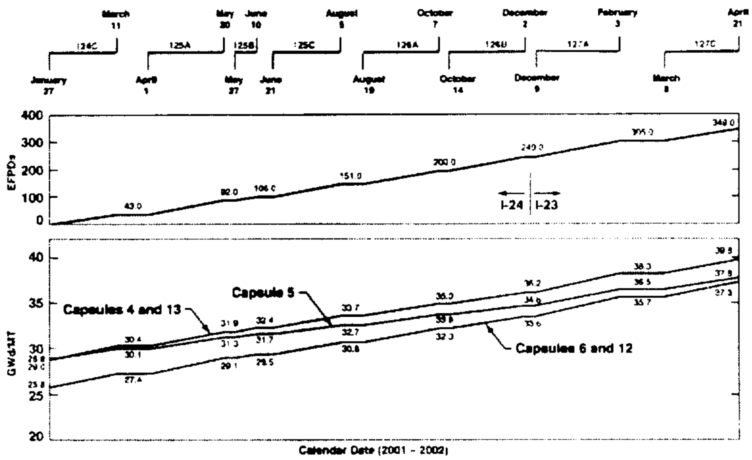
APT Irradiation Phase IV Part 1 Will Carry Lead Capsules 4 and 13 to 40 GWd/MT



November 10, 2000

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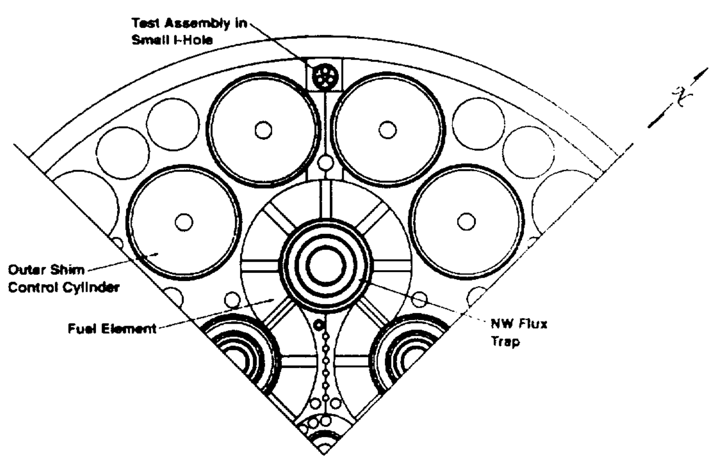
ORNL 2000-1838C EFG

At 36 GWd/MT the Test Assembly Will Be Moved from the Northwest to the Southwest I-Hole of the ATR Reflector



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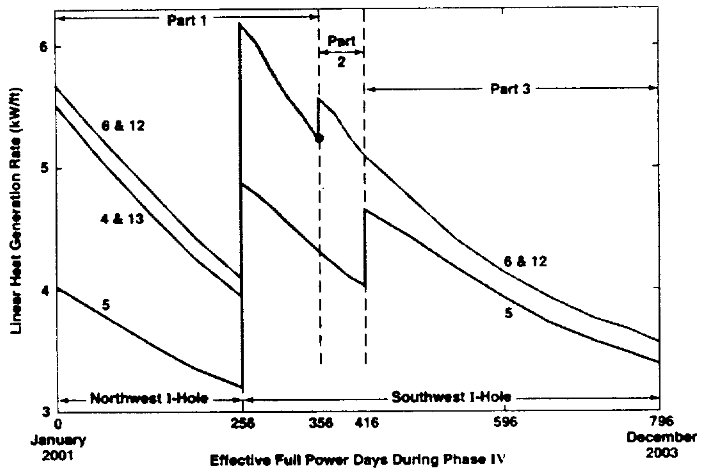
ORNL 2000-1490C EFG

Moving to the Southwest I-Hole Reduces the Time Required to Reach 50 GWd/MT



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ORNL 2000-1868C EFG

The Confirmatory PIE at 40 GWd/MT Is Essential to the Application of the Safety Analyses

- Hot cell predictions based on as-run experience
  - MCNP Code
  - CARTS
- Confirmation is obtained by non-destructive examination of the fuel pin (capsule must be opened)
  - Gamma scan to confirm length of pellet stack
  - Measure fuel pin gas pressure and Kr85 inventory
  - Obtain diameter profile over fuel pin length

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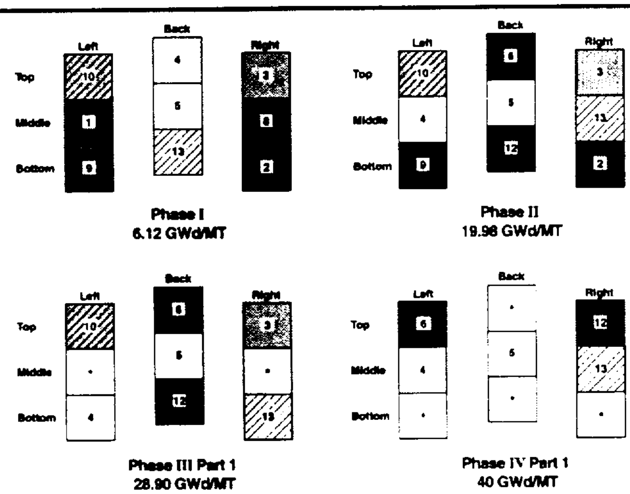
ORNL 2000-1867C EFG

Test Assembly Locations for Capsules 4 and 13 (40 GWd/MT Withdrawal)



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ORNL 2000-1666C EFG

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The Phase IV Irradiation Schedule Includes Provision for Confirmatory PIE at 40 GWd/MT

2001

January	Begin Phase IV at 28.9 GWd/MT
December	Reach 36 GWd/MT; move test assembly to southwest I-hole

2002

April	Reach 40 GWd/MT; remove capsules 4 and 13 for PIE
April – August	Irradiate capsules 5, 6 and 12 to approach 40 GWd/MT
September	Move capsule 5 to front with 6 and 12, and continue irradiation

2003

April	Reach 45 GWd/MT
December	Reach 50 GWd/MT

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ORNL 2000-1858C EFG

Linear Heat Generation Rates (LHGRs) in the Average-Power Test Exceed the U.S. PWR Average

U.S. PWRs:

- 5.2 – 6.7 kW/ft
- Peak axial power in an average PWR rod: 6.4 – 8.4 kW/ft

ATR tests

- As-run kW/ft for capsules withdrawn at 30 GWd/MT
  - 8.0 Phase I
  - 8.8 Phase II
  - 5.7 Phase III (Part 1)
- Many more thermal cycles than normal commercial experience

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ORNL 2000-1670C EFG

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PIE Results (Through 30 GWd/MT) Indicate Excellent Performance for Weapons-Derived MOX Fuel

- Densification and swelling
- Fission gas release
- Clad
- No indication of gallium movement or any adverse effects of impurities

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The Test Apparatus and Associated Safety Analyses Are Placed in Standby for a Possible Second Pellet Irradiation

Item	Pending
• Fuel pins and hafnium oxide end pellets at LANL	None – Bottom fuel pin end caps have been welded.
• Capsules and basket assemblies (3) at INEEL	None
• Safety documentation and Experiment Safety Assurance Package for High-Power Test (16 kW/ft)	Review and approval by SORC

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ORNL 2000-1713C EFG

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Average-Power Test Documents

Fissile Materials Disposition Program Light Water Reactor Mixed Oxide Fuel Irradiation Test Project Plan (ORNL/MD/LTR-78, Rev. 2)

A. REQUIREMENTS AND SPECIFICATIONS

1. *Design, Functional, and Operational Requirements for the Advanced Test Reactor Mixed Oxide Fuel Irradiation Experiment* (ORNL/MD/LTR-76). Author: Ken Thoms. Revision 1 issued September 30, 1997.
2. *Technical Specification: Mixed-Oxide Pellets for the Light-Water Reactor Irradiation Demonstration Test* (ORNL/MD/LTR-75). Author: Brian Cowell. Revision 0 issued June 1997.
3. *Purchase Order: Mixed-Oxide Pellets and Fuel Pin Assemblies* (ORNL/MD/LTR-77). Author: Brian Cowell. Revision 0 issued August 28, 1997.
4. *Purchase Order: Mixed Oxide Capsule Assemblies* (ORNL/MD/LTR-90). Author: Brian Cowell. Revision 0 issued August 12, 1997.
5. *Design, Functional, and Operational Requirements for Phase IV of the Average-Power Mixed-Oxide Irradiation Test* (ORNL/MD/LTR-187). Author: Ken Thoms. Revision 1 issued July 31, 2000.

B. PROCEDURES AND QUALITY CONTROL

1. *Fabrication, Inspection, and Test Plan for ATR MOX Fuel Pellets* (LANL Document NMT9-AP-QA-007-R00). Author: Ken Chidester. Revision 0 issued October 28, 1997.
2. *Fabrication, Inspection, and Test Plan for MOX Fuel Pin Preparation (FITP)* (LANL Document NMT9-AP-QA-008-R00). Author: Marty Bowidowicz. FITP Revision 0 issued November 14, 1997; Weld Qualification Plan issued November 21, 1997.
3. *Fabrication, Inspection, and Test Plan for the Advanced Test Reactor (ATR) Mixed-Oxide (MOX) Fuel Irradiation Project* (INEEL/EXT-97-01066). Author: Gregg W. Wachs. Revision 0 issued November 5, 1997.
4. *Experiment Safety Assurance Package for Mixed Oxide Fuel Irradiation in an Average Power Position (I-24) in the Advanced Test Reactor* (INEEL/EXT-98-00099). Authors: S. T. Khericha, R. C. Pederson, R. C. Howard, and John Ryskamp. Issued November 2, 1999.

Average-Power Test Documents (continued)

C. DESIGN AND SAFETY ANALYSES

1. *Thermal/Hydraulic Calculations for the LWR MOX Irradiation Test Assembly at 12 kW/ft* (ORNL/MD/LTR-85). Author: Larry Ott. Revision 0 issued October 1, 1997.
2. *Effects of Fission Gas Release and Pellet Swelling Within the LWR Mixed Oxide Irradiation Test Assembly* (ORNL/MD/LTR-83). Author: Steve Hodge. Revision 1 issued November 11, 1997.
3. *Design Calculations in Support of the Advanced Test Reactor Mixed Oxide (ATR-MOX) Fuel Irradiation Experiment* (ORNL/MD/LTR-92). Authors: Kin Luk and Jim Corum. Revision 0 issued November 6, 1997. Addendum 1 for fuel pin end caps issued January 13, 1998.
4. *Capsule Loading and Operation Schedule* (ORNL/MD/LTR-91). Author: Steve Hodge and Brian Cowell. Revision 2 issued February 17, 2000.
5. *Flow Test of the MOX Test Basket Assembly* (ORNL/MD/LTR-118). Author: Larry Ott. Revision 1 issued February 4, 1998.
6. *Flow Test of the Model-2 MOX Test Basket Assembly* (ORNL/MD/LTR-149). Author: Larry Ott. Revision 0 issued August 19, 1998.
7. *Fission Gas Release and Pellet Swelling Within the Capsule Assembly During Phase IV of the Average-Power Test (ORNL/MD/LTR-184) Author: Steve Hodge. Revision 0 issued July 21, 2000.*
8. *Thermal/Hydraulic Calculations for Phase IV of the LWR MOX Irradiation Average-Power Test* (ORNL/MD/LTR-191). Author: Larry Ott. Revision 0 issued July 26, 2000.
9. *Design-Calculations for Phase IV of the Advanced Test Reactor Average-Power Mixed Oxide Fuel Irradiation Experiment.* (ORNL/MD/LTR-192) Authors: Claire Luttrell and Terry Yahr. Revision 0 issued August 2000.
10. *Overview of Safety Analyses for MOX Irradiation Phase IV Extended Burnup* (ORNL/MD/LTR-194). Author: Steve Hodge. Revision 0 issued June 14, 2000.

D. TRANSPORTATION

1. *Fresh Test Fuel Shipment Plan for the LWR MOX Fuel Irradiation Test Project* (ORNL/MD/LTR-87). Authors: Leonard Dickerson and Mimi Welch. Revision 0 issued September 17, 1997.
2. *Irradiated Test Fuel Shipping Plan for the LWR MOX Fuel Irradiation Test Project* (ORNL/MD/LTR-101). Author: Scott Ludwig. Status: Revision 0 issued October 16, 1998.

Average-Power Test Documents (continued)

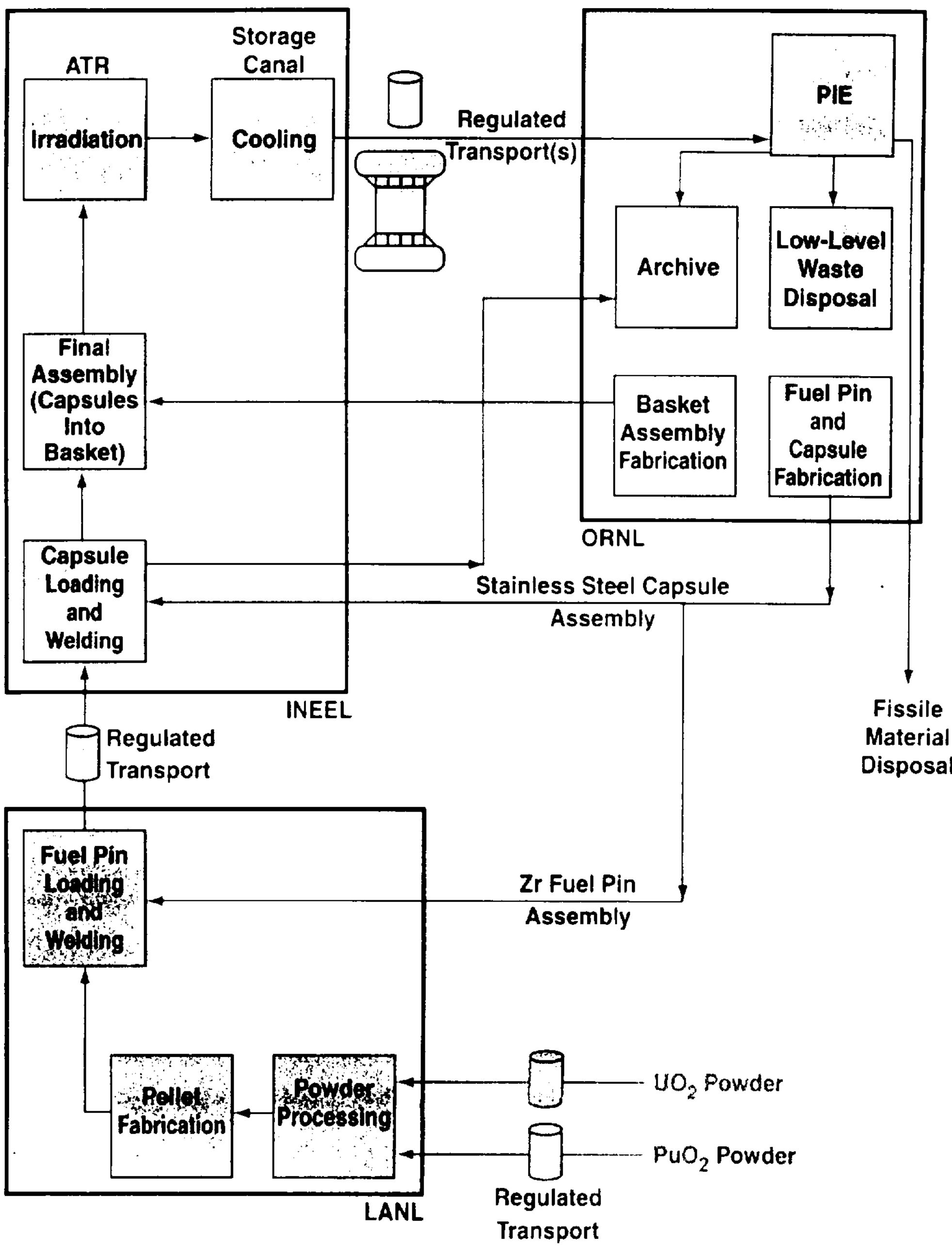
E. PIE

1. *MOX Capsule Post-Irradiation Examination Vol 1: Test Plan for Low Burnup Fuel* (ORNL/MD/LTR-93). Author: Bob Morris. Revision 0 issued August 20, 1997.
2. *PIE Plan Volume II* (ORNL/MD/LTR-93). Author: Bob Morris. Revision 0 issued December 11, 1997.
3. *MOX Average Power Early PIE; 8 GWd/MT Quick Look* (ORNL/MD/LTR-163). Author: Bob Morris. Revision 1 issued February 23, 1999.
4. *MOX Average Power Early PIE: 8 GWd/MT Final Report* (ORNL/MD/LTR-172). Author: Bob Morris, C.A. Baldwin, et al. Revision 0 issued November 18, 1999.
5. *MOX Fission Gas Pressure Measuring Apparatus* (ORNL/MD/ LTR-176). Author: Bob Morris, C.A. Baldwin. Revision 0 issued January 31, 2000.
6. *MOX Average Power Test Fuel Pellet Intial Gallium Content* (ORNL/MD/LTR-182). Author: Bob Morris, Joe Giaquinto, and Steve Hodge. Revision 0 issued March 7, 2000.
7. *MOX Average Power Intermediate PIE: 21 GWd/MT Quick Look* (ORNL/MD/ LTR-185). Author: Bob Morris, C.A. Baldwin, S. A. Hodge, C. M. Malone, and N. H. Packan. Revision 0 issued March 21, 2000.
8. *Post-Irradiation Examination Plan For ATR MOX Capsules Withdrawn at 30 GWd/MT and Higher* (ORNL/MD/LTR-195) Author: Bob Morris. Revision 0 issued September 18, 2000.
9. *MOX Average Power Intermediate PIE: 21 GWd/MT Final Report* (ORNL/MD/LTR-199). Author: Bob Morris. Revision 0 issued November 10, 2000.
10. *Implications of the PIE Results for the Intermediate-Withdrawal (21 GWd/MT) MOX Capsules* (ORNL/MD/LTR-203). Authors: Steve Hodge and Larry Ott. Revision 0 issued December 7, 2000.

F. CLAD DUCTILITY TESTING

1. *A Simple Method for Measuring Ductility of Irradiated Fuel Clad—Design of Apparatus and Proof of Principle* (ORNL/MD/LTR-201). Author: W. R. Hendrich, G. T. Yahr. Now available in draft.

The MOX Fuel Test Irradiation Is a Cooperative Endeavor of ORNL, INEEL, and LANL



Successful Accomplishment of the MOX Fuel Irradiation Test Project Involves Several Fields of Expertise

ORNL	Project Planning and Analysis	Brian Cowell
	Test Assembly Design and Fabrication	Ken Thoms, Dennis Heatherly
	Thermal Hydraulics	Larry Ott
	Structural Analysis	Claire Luttrell
	PIE	Bob Morris
	Clad Integrity Tests	Terry Yahr, Bill Hendrich
	Neutronics Advisor	Joe Pace
	Transportation	Scott Ludwig
INEEL	MOX Project Manager	Bob Pedersen
	Neutronics Calculations and Irradiation Scheduling	Gray Chang, Bill Terry
	Reactor and Canal Operations	Rob Howard
	Experiments Thermalhydraulics	Dick Ambrosek
	ATR Reactor Safety	Soli Khericha, Terry Tomberlin
	Project Advisors	John Ryskamp, Del Mecham
LANL	Project Lead	Dave Alberstein
	Fuel Fabrication	Ken Chidester, Tim George, Tom Blair, Marty Bowidowicz

Code Support for MOX Irradiation Safety Analysis

L. J. Ott
Oak Ridge National Laboratory

Presented at

ORNL MOX Fuel Program
Research and Development Meeting
Oak Ridge, Tennessee
December 12, 2000

Outline of Presentation

- Primary function: safety analyses for MOX irradiation experiments in the ATR
  - Comparison of ATR operating conditions with commercial experience
  - Compliance with the experiment "design, functional and operational requirements . . ."
    - Fuel pin and capsule design
    - Code requirements
    - Mechanical design requirements
    - Design conditions
    - ATR technical requirements
    - ATR operational requirements
  - Satisfy the ATR Safety and Operations Review Committee (SORC)
- Fuel performance calculations for the 21 GWd/MT MOX capsules
  - CARTS
  - FRAPCON-3

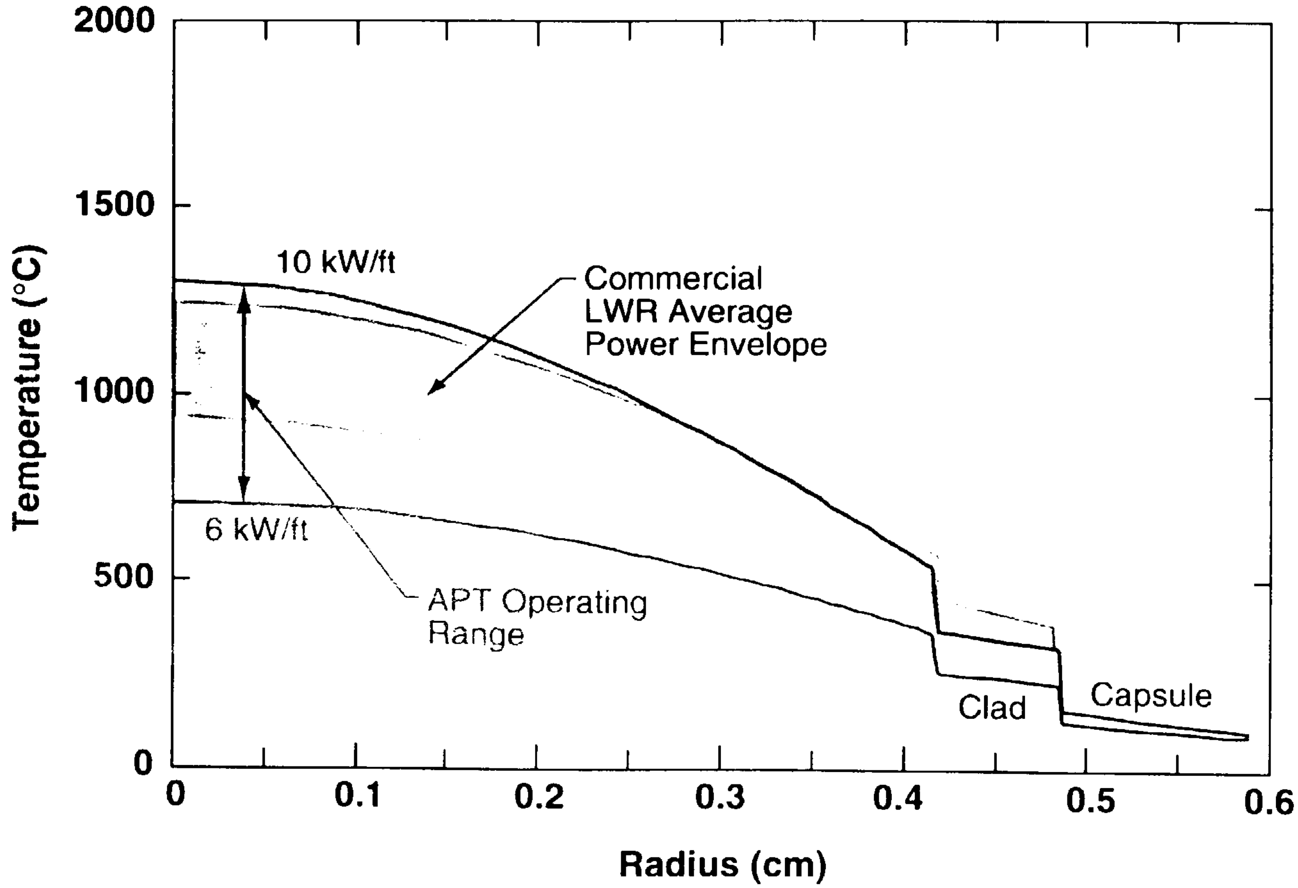
Primary Codes Employed

ABAQUS	commercial finite element structural analysis program
CARTS	experiment-specific Capsule Assembly Response-Thermal Swelling code (developed for this project at ORNL)
ESCORE	EPRI-sponsored fuel performance code
FFFAP	steady state fluids code developed at ORNL
FLUENT	commercial computational fluid dynamics (CFD) code
FRAPCON-3	USNRC-developed fuel performance code
HEATING	a general structural thermal analysis code developed at ORNL
MATPRO	USNRC-developed material properties correlations and computer subroutines
MCNP	Monte Carlo Neutron Photon neutronics code employed by INEEL

ATR MOX Experiments Are Designed to Replicate Commercial LWR Fuel Conditions during Normal Operation

- Source of information is the annual fuel performance report (NUREG/CR-3950) prepared by PNNL for the NRC
  - For 1991 (Vol. 9, published in 1994)
  - Covers **all** U.S. NSSS suppliers and fuel fabricators
- APT simulates the axial peak power position in an "average" fuel pin.
- HPT is proposed to simulate the core peak power position.

The MOX Fuel Temperatures in the APT Blanket the Operating Ranges for Commercial LWR Fuel of Similar Dimensions



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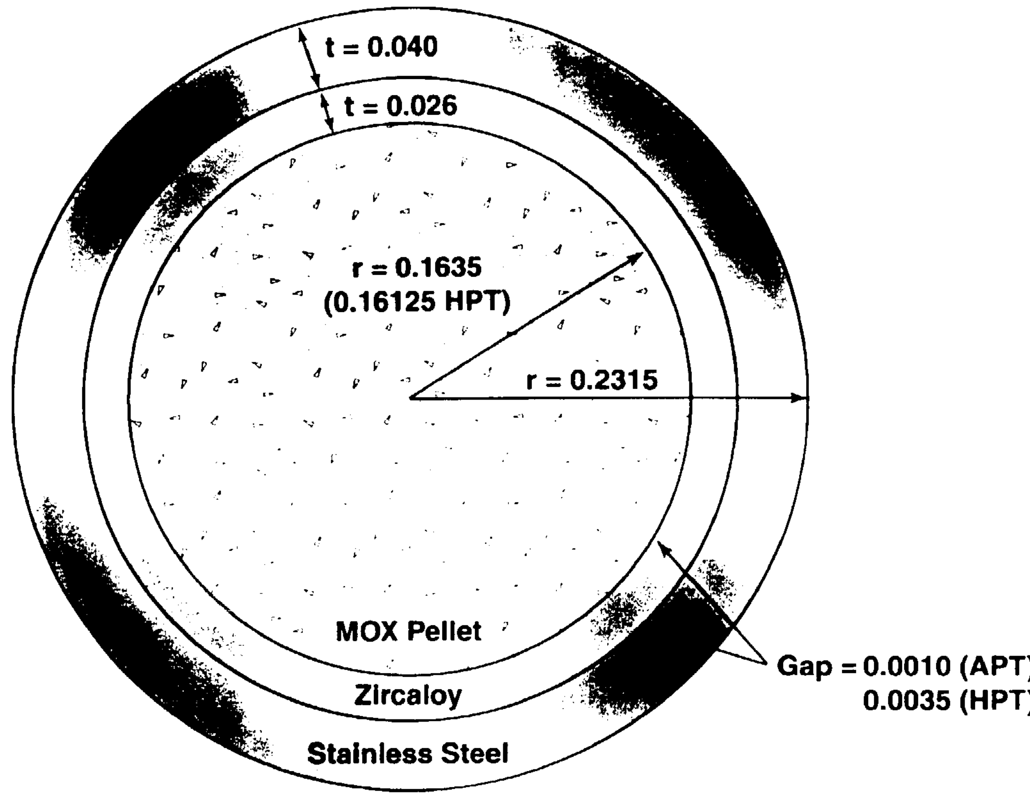
ORNL 2000-1826C EFG

UT-BATTELLE

LJO-5

Test Assembly Containment* Is Provided by a Stainless Steel Capsule Surrounding Each Sealed Zircaloy Fuel Pin Assembly

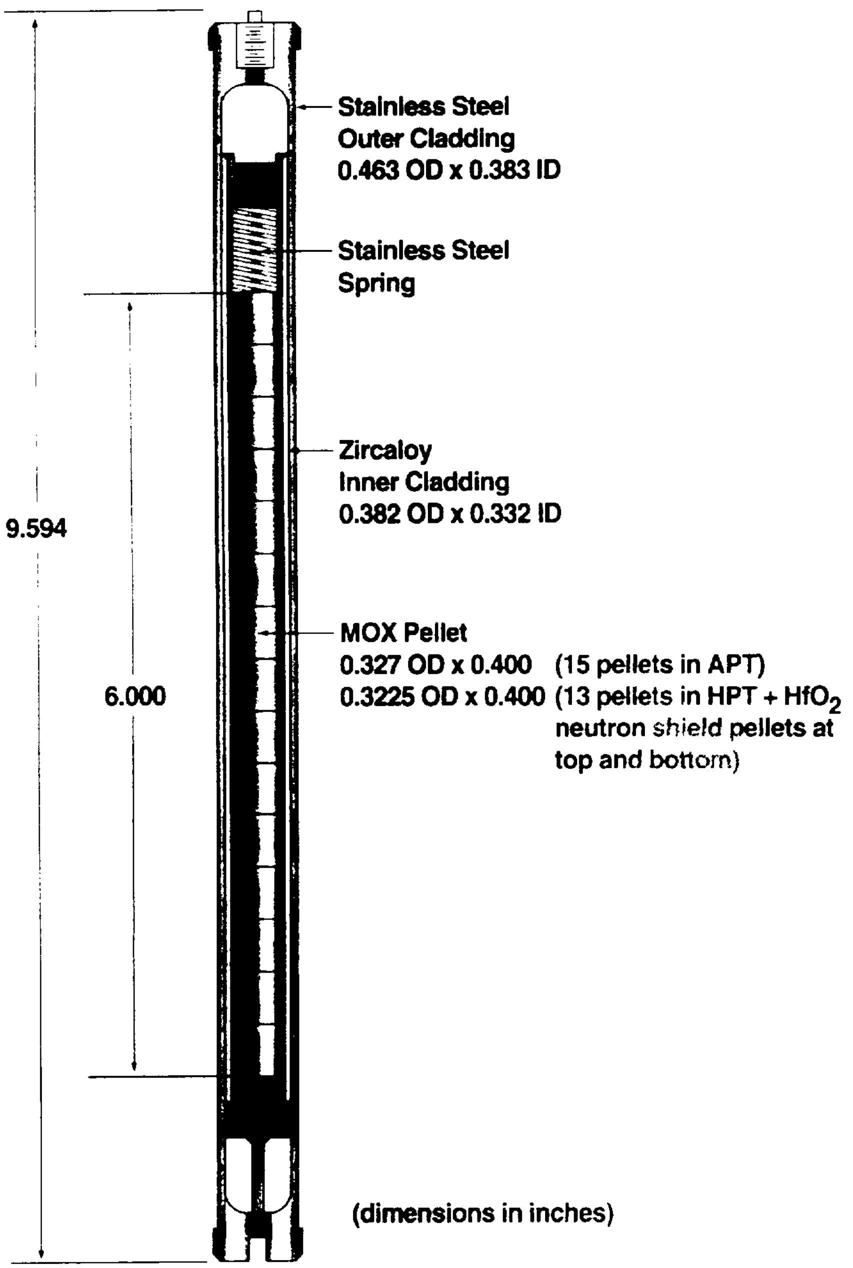
Average Power Test (APT) Capsule



Dimensions: Inches

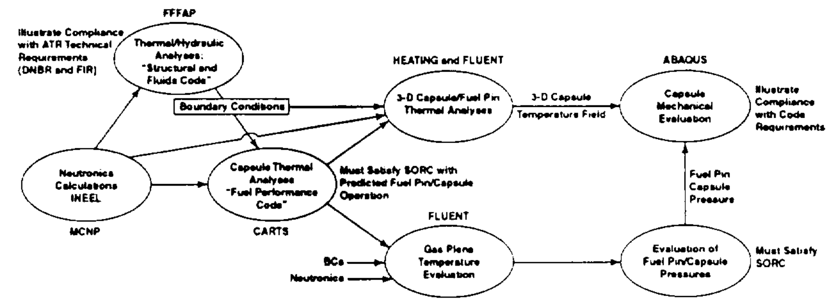
*ATR requirement, must meet the intent of ASME Section III, Class 1 standards

Each Capsule Assembly Contains One Zircaloy Fuel Pin with 15 Fuel Pellets



Three Thermal/Hydraulic Safety Analyses Have Been Performed for the ATR MOX Irradiation Experiments

- APT (≤30-GWd/MT burnup): ORNL/MD/LTR-85, 1997
- HPT (≤50-GWd/MT burnup): ORNL/MD/LTR-138, 1999
- APT extension (≤50-GWd/MT burnup): ORNL/MD/LTR-191, 2000



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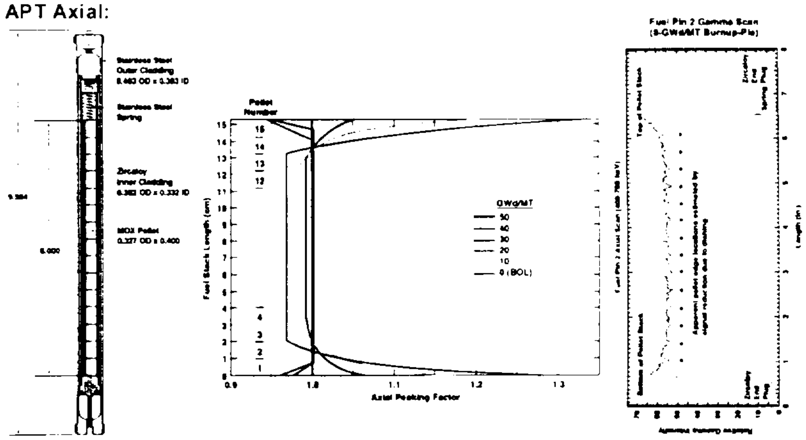
ORNL 2000-1829C EFG

Thermal Evaluations Required Three-Dimensional Power Generation Models* within Fuel Pin/Capsule Components

APT Axial:



*MCNP Calculations at INEEL by G. Chang

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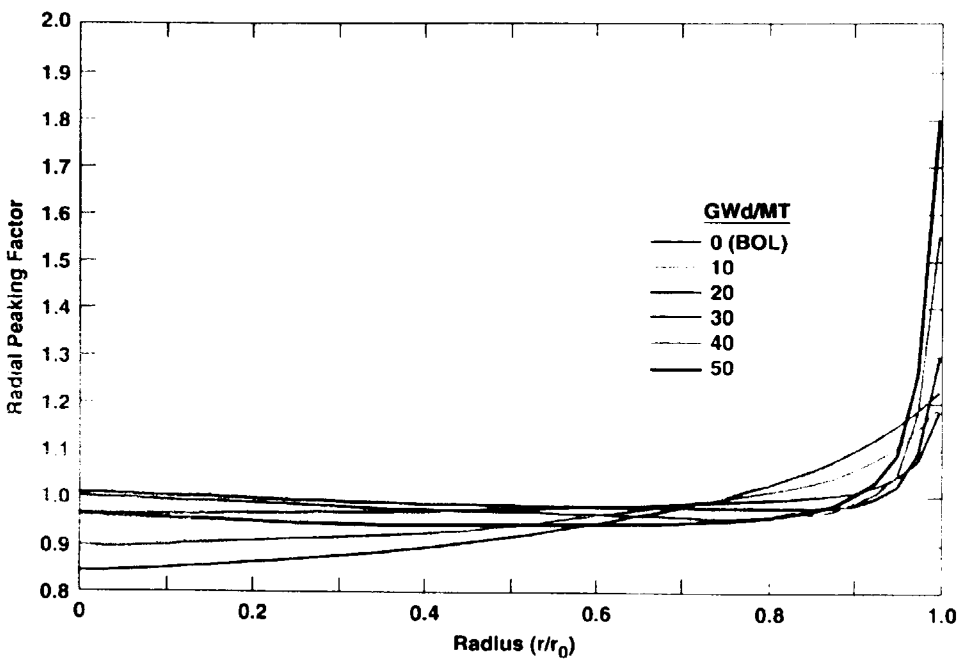
ORNL 2000-1830C EFG

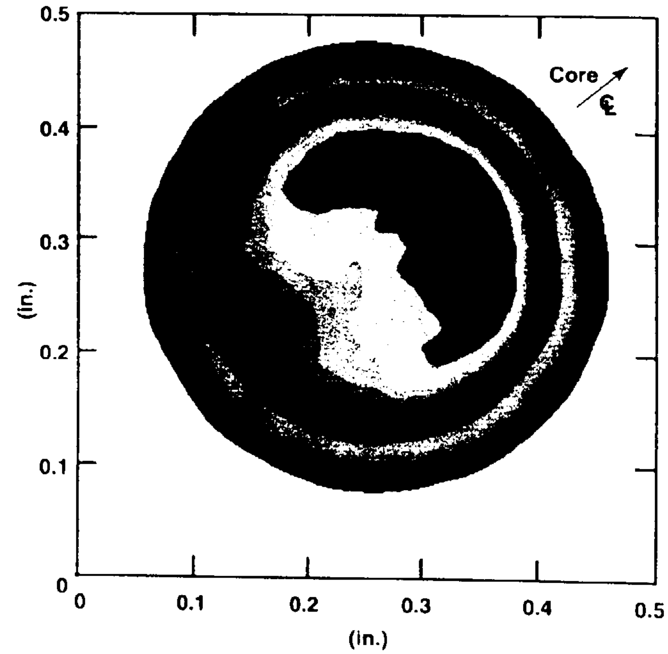
Thermal Evaluations Required Three-Dimensional Power Generation Models* within Fuel Pin/Capsule Components (continued)

APT Radial:



APT Azimuthal:

Relative Gamma Intensity
(Fuel Pin 2: 8-GWd/MT Burnup, PIE)
Smoothed Pellet Mount 6141 (450 to 800 keV)



- PIE supports Chang's azimuthal dependence

*MCNP Calculations at INEEL by G. Chang

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ORNL 2000-1831C EFG

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LJO-10

The Fuel Pin/Capsule Thermal and Mechanical Calculations Are the Focal Point of the Thermal Safety Analyses

- Must model fuel, (Zircaloy) clad, and stainless steel capsule wall
- Provides input to the 3-D thermal analyses and gas plena temperature analyses
- Addresses SORC concerns regarding uncertainties:
  - Dimensions
  - Material properties
  - Boundary conditions
    (i.e., surface convective heat transfer and spatial power generation)
  - Fission gas release
  - Models
    - Gap conductance
    - Fuel densification
    - Fuel swelling

In 1997, no code was available to address the capsule wall and the uncertainty analyses required by the SORC.

ORNL Developed the Experiment-Specific Capsule Assembly Response–Thermal and Swelling (CARTS) Code to Simulate the MOX Capsule Assembly

Capabilities:

- Determines the coupled thermal and mechanical behavior of the capsule assembly as burnup advances
- Uncertainty analyses:
  - Component dimensions
  - Component physical properties (i.e., thermal conductivity, thermal expansion)
  - Fission gas release
- Model options:
  - Fuel densification:
    - ESCORE
    - FRAPCON-3
  - Fuel swelling:
    - ESCORE
    - FRAPCON-3
  - Fuel thermal conductivity:
    - MATPRO
    - FRAPCON-3
  - Gap thermal conductance:
    - MATPRO
    - ESCORE
    - NUREG/CR-0330

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CARTS Code

- Simplified one-dimensional fuels and containment behavior code
- Writeup reviewed by INEEL, LANL, ORNL, and MPRA
- Code and models reviewed extensively by Stoller Nuclear Fuel Corp. (Dion Sunderland)
- Benchmarked against industry ESCORE BWR calculations
- Comparisons with FRAPCON-3 for Phase II irradiation

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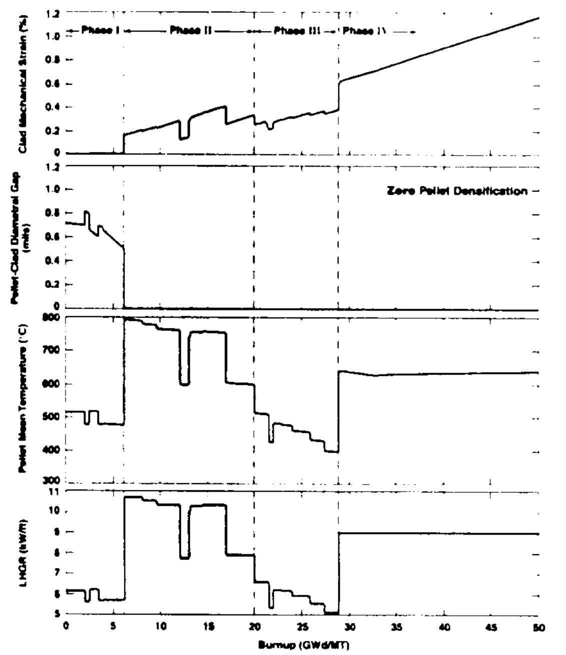
CARTS Prediction for 9 kW/ft during Phase IV Based on Irradiation History for Capsules 4 and 13



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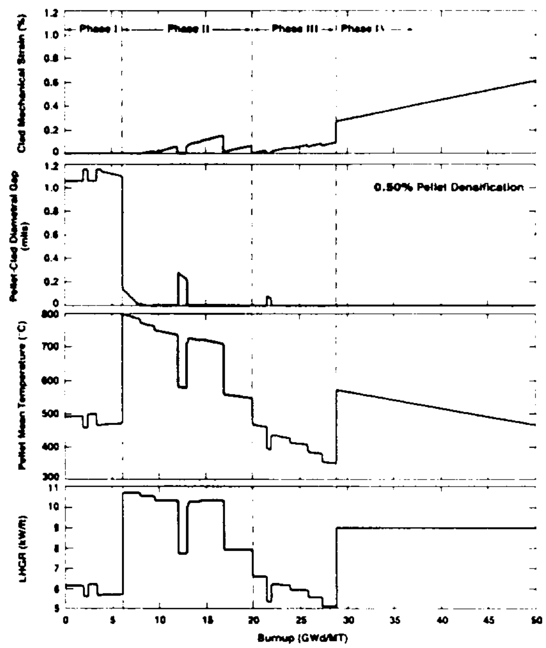
CARTS Prediction for 9 kW/ft during Phase IV with Pellet Densification



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Primary SORC Concerns for the Proposed HPT and the APT Extension

- Proposed HPT: centerline fuel melting
  - At LHGRs of 16 kW/ft; "worst case" CARTS simulations yielded centerline temperatures more than 50°C below melting
- APT extension: significant mechanical strain of stainless steel wall due to fuel swelling
  - At 9 kW/ft and 0.5% fuel densification, CARTS simulations predict **no** mechanical strain in stainless steel wall
  - At 9 kW/ft and 0.0% fuel densification, CARTS simulations predict a maximum of 0.38% mechanical strain in capsule wall
    - Acceptable per capsule mechanical evaluation with ABAQUS (Luttrell and Yahr)

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Phase II (21-GWd/MT Burnup) Intermediate-Withdrawal Capsule Simulations with the CARTS and FRAPCON-3 Codes

- FRAPCON-3 should not be used to model MOX fuel (per PNNL Web site)
- FRAPCON-3 simulation uses 7% LEU as a surrogate for the 5% MOX (per Chang's HPT analyses)
- Good agreement with CARTS until FRAPCON-3 starts predicting excessive fission gas release (contrary to PIE measurements)

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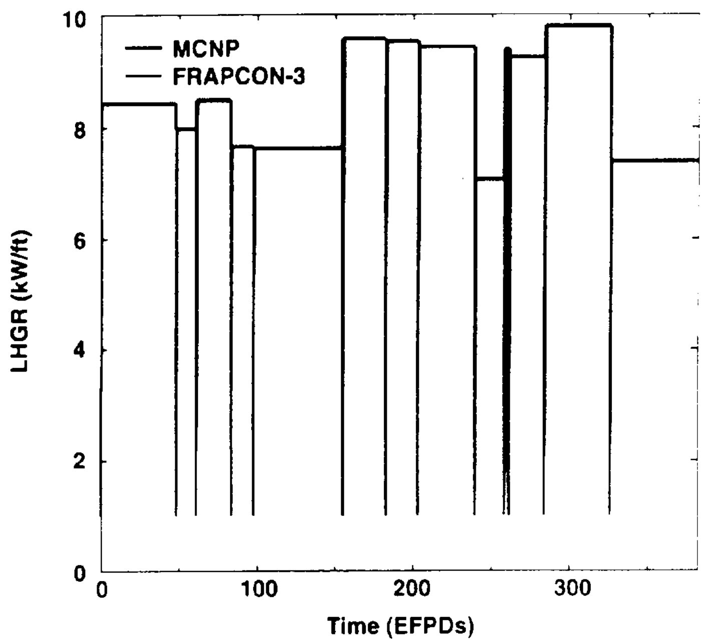
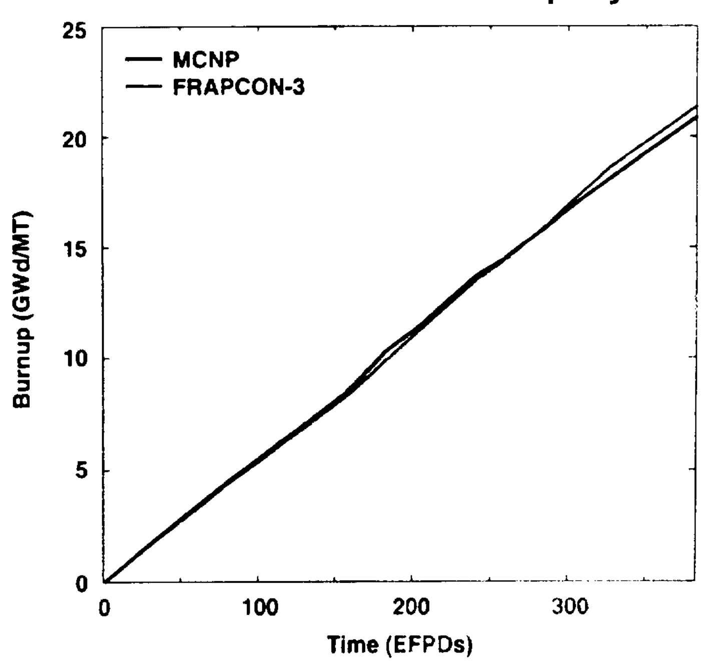
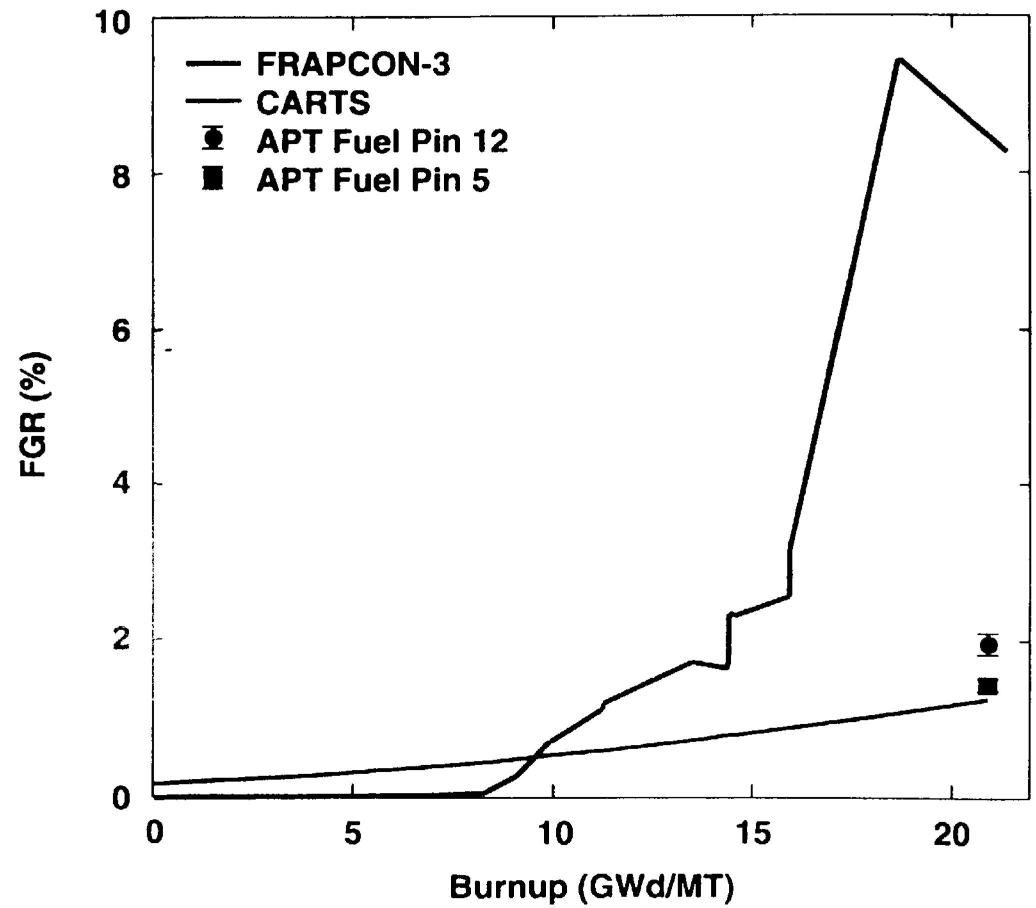
FRAPCON-3 Input LHGRs Match the MCNP Calculated Power for Intermediate-Withdrawal Capsules



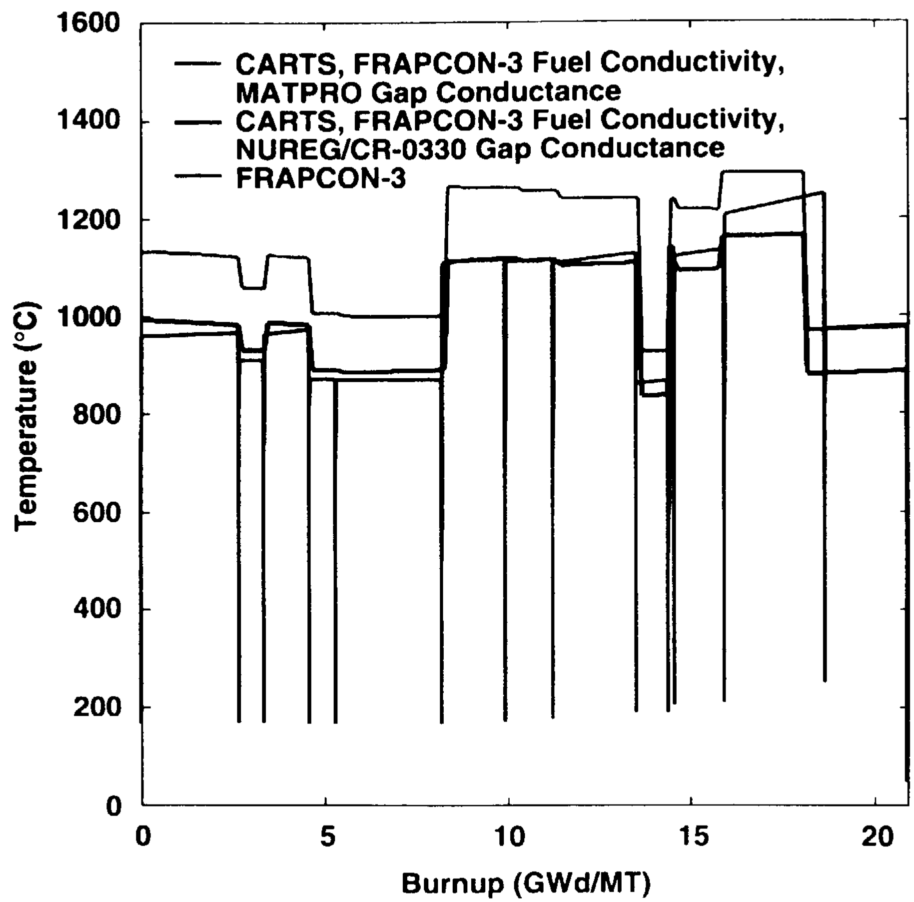
Good Comparison between FRAPCON-3 and MCNP Calculated Burnups Indicates Accurate Estimate of LHGRs and Burnup by MCNP



When Applied to Phase II of the APT, FRAPCON-3 Predicts Fission Gas Release in Excess of that Measured



FRAPCON-3 and CARTS (Using FRAPCON-3 Models) Calculated Fuel Centerline Temperatures Agree Closely until FRAPCON-3 Calculates High Fission Gas Release from Fuel



ORNL Is a Member of the FRAPCON-3 User's Group

- FRAPCON-3 v1.3 is being applied (at ORNL) to the APT MOX fuel pins post-PIE analyses.
- Results and observations have been discussed at User Group Meetings.
- The intermediate PIE report (ORNL/MD/LTR-199) has been mailed to D. Lanning at PNNL.

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FRAPCON-3 Is Being Modified at ORNL to Analyze the OECD/NEA MOX Benchmarks

- To include MOX thermal conductivity correlations
  - Halden
  - Duriez (with Lucuta degradation factors)
- To allow input of plutonium isotopics
  - Required for the TUBRNP subcode (calculates the radial power distribution and burnup)

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A Brief Review of ATR MOX Fuel Test PIE Status

Dr. R. N. Morris
Oak Ridge National Laboratory

Presented at

ORNL MOX Fuel Program
Research and Development Meeting
Oak Ridge, Tennessee
December 12, 2000

Goals of the PIE

- Monitor the course of the irradiation
  - Routine ATR safety issues
  - Collect data as a function of burnup
- Examine the fuel pin and capsule for material interactions
  - Irradiation of weapons-grade material
    - No significant concern
  - Gallium impurity
    - Much lower than originally thought (a few parts-per-million)
    - May have been present at very low levels in LWR fuel systems for 40 years

8-GWd/MT PIE Examination

- Fuel and clad performance
  - No signs of pellet-clad mechanical interaction
  - No signs of corrosion
    - No indications of gallium migration
  - No difference between TIGR and non-TIGR treated
  - Fuel behavior as expected
    - Slight densification (<0.5%)
- Only clad ductility testing remains

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8-GWd/MT PIE Examination (continued)

- Nominal gallium in fuel system materials
  - Unexpected trace gallium in unirradiated stainless steel and clad
  - Generation from zinc impurities
- Clad ductility testing deferred
  - More complex issue than expected
  - Test analysis and design under way (subject of separate presentation)

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ORNL 2000-1758C EFG

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8-GWd/MT PIE Examination (continued)

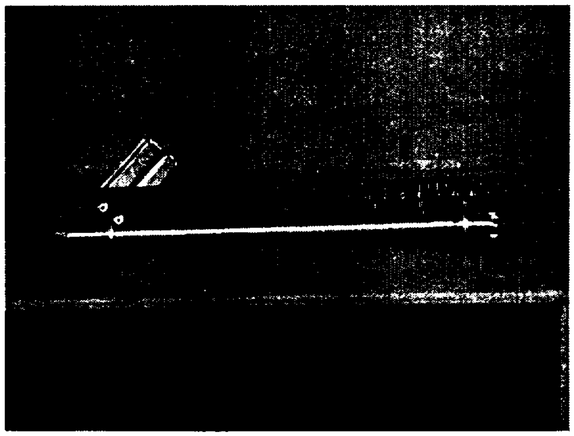
- Documentation:
  - *MOX Average Power Early PIE: 8 GWd/MT Quick Look*, ORNL/MD/LTR-163, Rev. 1 (February 1999)
  - *MOX Average Power Early PIE: 8 GWd/MT Final Report*, ORNL/MD/LTR-172 (November 1999)

8-GWd/MT PIE Exam Involved Capsules 1 and 8

- Capsule 1
  - Contained fuel pin 2
  - Untreated fuel
- Capsule 8
  - Contained fuel pin 11
  - TIGR-treated fuel
- Both capsules were otherwise identical and irradiated under identical conditions
  - EOI September 13, 1998

Capsule 1 in Hot Cell

- Excellent condition



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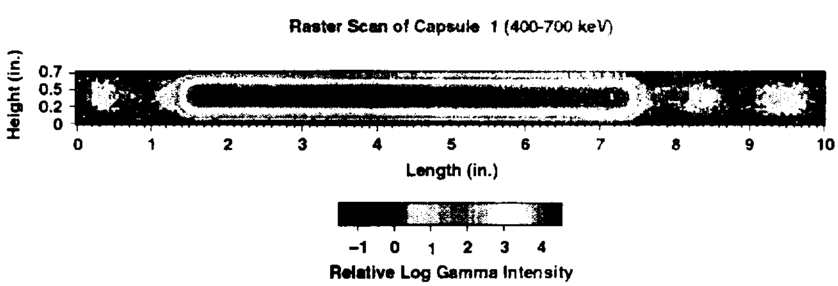
Capsule 1 Gamma Scan

- No internal abnormalities



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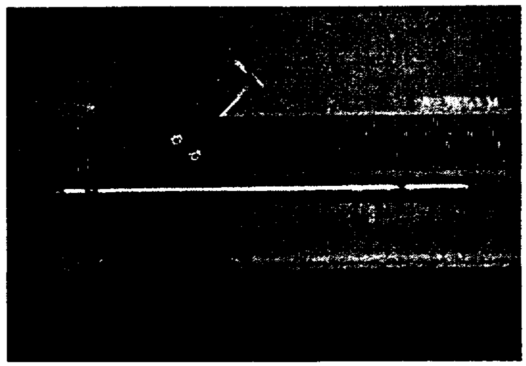
Fuel Pin 2

- Easily removed from capsule 1
- Excellent condition



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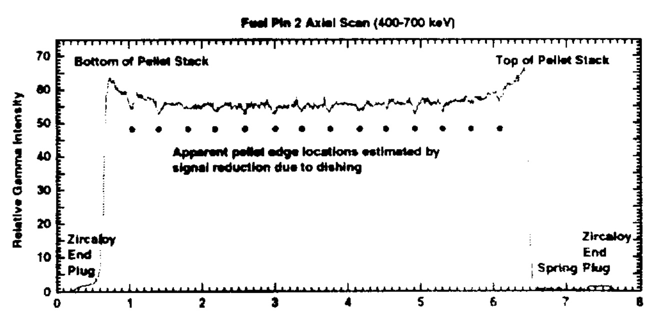
Fuel Pin 2

- Gamma scan reveals intact pellet stack



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Fuel Pin 2 Pellets

- Pellets easily removed from pin
- No signs of clad interaction

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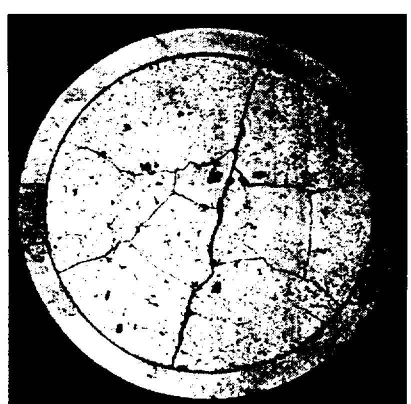
Fuel Pin 2 Metallography

- Expected behavior
  - Normal cracking
  - Visible pellet-clad gap
  - No visible restructuring



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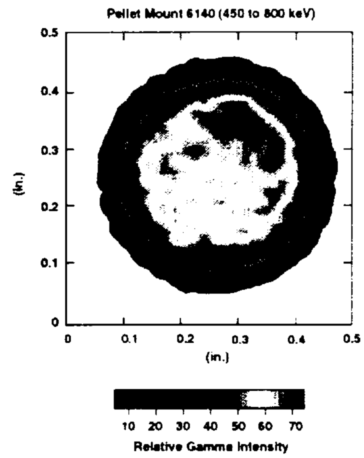
Gamma Scan of Fuel Pin 2 Cross Section

- Qualitative picture of predicted asymmetric burnup
- Resolution is actually quite coarse (6 to 8 points along radius), so this information is only qualitative



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ORNL 2000-1787C EFG

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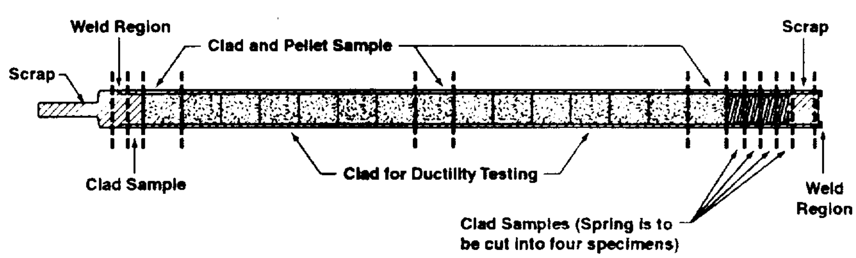
Fuel Pin 11 Was Sectioned for Radiochemistry

Fuel Pin Cutting Guide for Gallium Tracking



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Radiochemistry Results

- Good burnup agreement (fuel pin 2, pellet 2)
  - 8-GWd/MT vs 8.6-GWd/MT calculation
- No gallium movement observed within uncertainties
  - Some difficulties with analysis methods
    - Unexpected nominal trace gallium levels in fuel system materials (tenths of a part-per-million)
    - Pin 11 had lower initial inventory
  - Measured clad levels approximately 1 ppm
    - Clad inventory bounded at one-fourth pellet inventory

Gallium Levels in Fuel System Materials Were Investigated

- Gallium in unirradiated Zircaloy-4 tube and bar stock was 0 to 0.35 ppm (5 samples).
- Gallium in unirradiated stainless steel stock was 23 to 34 ppm (7 samples).
- ^{68}Zn transmutes into ^{69}Ga.
  - Zinc is a listed fuel impurity.
  - Roughly 1 ppm of zinc gives 0.8 ppb of gallium at the end of fuel life.
- Depleted uranium powder was not analyzed yet.
- Gallium appears to have been present in the LWR fuel system at the sub-ppm to ppm level for years.

Archive Pellet Gallium Levels Were Measured to Improve Statistics

- Ten pellets from both batch A and B were analyzed for gallium content.
- Batch A average was 2.97 ppm (untreated).
  - Varied from 1.81 to 4.78 ppm
  - Standard deviation of 1.01
- Batch B average was 1.33 ppm (treated).
  - Varied from 1.09 to 2.03
  - Standard deviation of 0.28

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8-GWd/MT Summary

- Capsule and fuel pin are performing well.
- PIE has verified Code predictions.
  - No significant fuel restructuring
  - Asymmetric burnup
  - Slight densification (examined in detail at 21 GWd/MT)
  - Burnup level within uncertainties

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8-GWd/MT Summary (continued)

- No gallium effects were observed.
  - No gross transport
    - Limited by measurement (since improved)
    - Bounded at one-fourth pellet inventory
  - "Nominal sources" are better understood:
    - Trace impurity ~ 0.01- to 0.1-ppm level
    - Zinc transmutation at part-per-billion level
- Clad samples archived

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21-GWd/MT PIE Exam Involved Capsules 2 and 9

- Capsule 2
  - Contained fuel pin 5
  - Untreated fuel
- Capsule 9
  - Contained fuel pin 12
  - TIGR-treated fuel
- Both capsules were otherwise identical and irradiated under identical conditions
  - EOI September 12, 1999

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21-GWd/MT PIE Examination: Only Clad Ductility Testing Remains

- Fuel and clad performance
  - No signs of pellet and clad mechanical interaction
  - No signs of gallium migration or corrosion
  - Densification and swelling examined
- Fission gas pressures within expected limits
- No difference between TIGR and non-TIGR
- Plutonium-rich agglomerates noted
- No abnormal behavior noted

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ORNL 2000-1775C EFG

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21-GWd/MT PIE Examination: Only Clad Ductility Testing Remains (continued)

- Documentation:
  - *MOX Average Power Intermediate PIE: 21 GWd/MT Quick Look*, ORNL/MD/LTR-185 (March 2000)
  - *MOX Average Power Intermediate PIE: 21 GWd/MT Final Report*, ORNL/MD/LTR-199 (November 2000)

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Capsule 2 in Hot Cell

- Excellent condition

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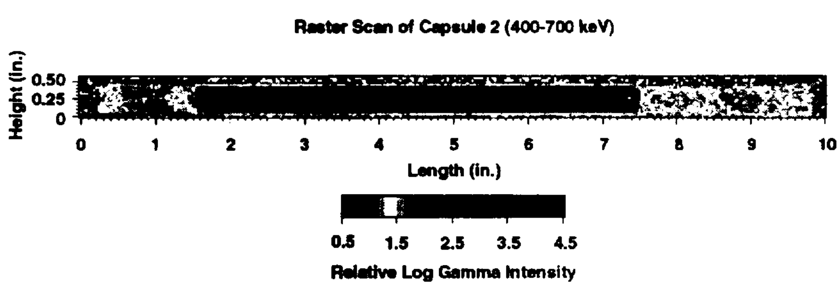
Capsule 2 Gamma Scan

- No internal abnormalities



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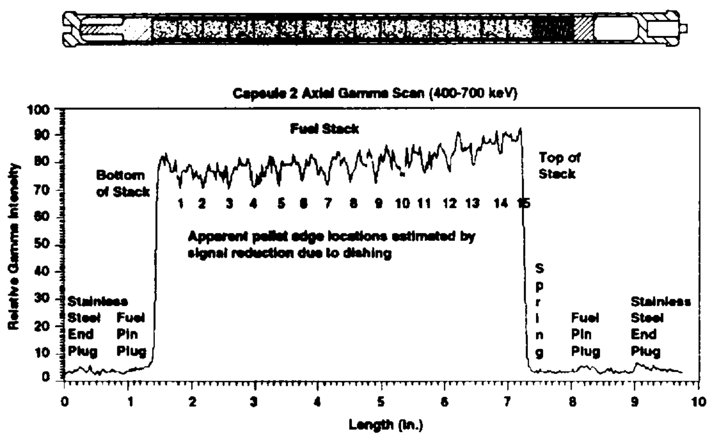
Capsule 2 Fuel Pin 5

- Gamma scan reveals intact pellet stack



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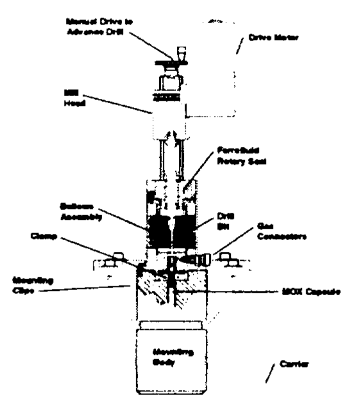
Fission Gas Pressures in Capsules 2 and 9 Were Measured

- Apparatus and calibration were documented in *MOX Fission Gas Pressure Measuring Apparatus*, ORNL/MD/LTR-17 (January 2000).
- Performance was tested by a series of blind tests.



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Estimation of Fission Gas Release

- The fuel pins were found to be leak tight
- Pressures/^{85}Kr inventory
  - Fuel pin 5 23.8 psia 4.38 mCi
  - Fuel pin 12 27.5 psia 5.86 mCi
- Estimated release fraction
  - 1.5 to 2.0% based on ^{85}Kr
  - Slightly higher than predicted (~1.2%)

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Fuel Pin 5

- Easily removed
- Excellent condition

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Fuel Pin Metrology

- Dimensional inspections
  - Clad diameter and length
    - Pellet swelling
    - Clad creep
- Both fuel pins (5 and 12) revealed no abnormalities
  - Very small (0.2–0.3%) clad creep/irradiation growth
  - No pellet stack encroachment into gas plenum
  - Analysis of data with CARTS revealed no pellet-clad contact during irradiation
    - Fuel creep/swelling is similar to that of commercial fuel

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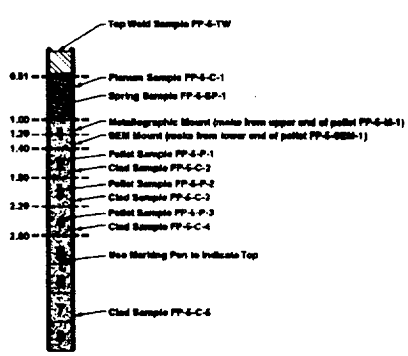
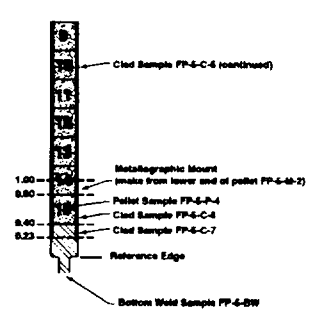
Both Capsules Were Sectioned

Detailed Cutting Guide for
Top Half of Fuel Pin 5



Measurements are referenced to the top of fuel pin

Detailed Cutting Guide for
Bottom Half of Fuel Pin 5



Measurements are referenced to the bottom edge of fuel pin

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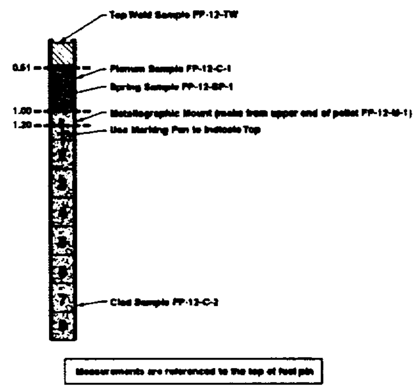
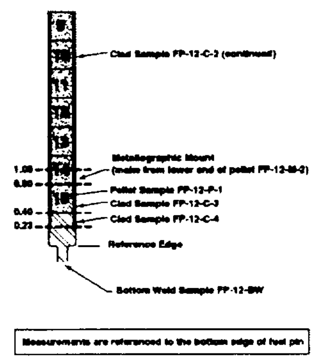
Both Capsules Were Sectioned (continued)

Detailed Cutting Guide for Top Half of Fuel Pin 12



Detailed Cutting Guide for Bottom Half of Fuel Pin 12



Fuel Pin Specimens

- Metallography
  - Estimate pellet-clad gap
  - Fuel restructuring—temperature indication
  - Indications of plutonium-rich agglomerates
- Radiochemistry
  - Gallium tracking for pellet and clad
  - Burnup
- SEM
  - First look at agglomerate/matrix plutonium distribution

Fuel Pin Specimens (continued)

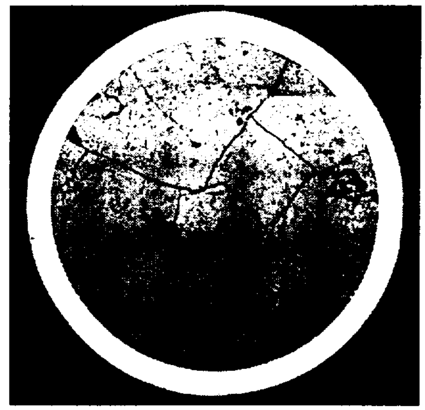
- Clad ductility testing
  - Test design/apparatus under design
  - Subject of subsequent presentation
  - Archive material from 8 GWd/MT
- Metallographic mount gamma scanning
  - Examine burnup profile

Two Metallographic Mounts from Each Fuel Pin Were Examined

- No differences were noted between the non-TIGR fuel (capsule 2, fuel pin 5) and the TIGR-treated fuel (capsule 9, fuel pin 12).
- Specimens came from both the top and bottom of the fuel pins.
- Numerous plutonium-rich agglomerates were observed.
  - Up to 500 μm in size
  - Uneven distribution
- Pellet-clad gap was observed.

Metallographic Mounts Showed Expected Behavior

- Normal cracking
- Pellet-clad gap
- MOX agglomerates
- Pin 12, M-1



MX R82516 6143

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ORNL 2000-1788C EFG

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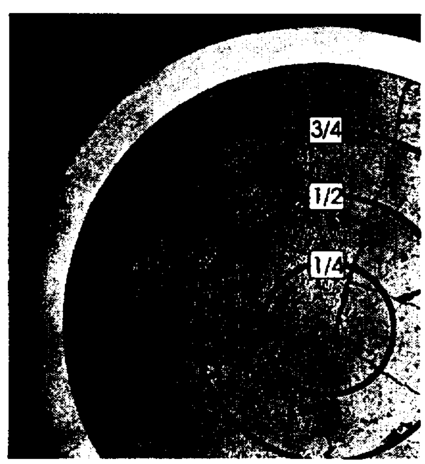
Agglomerates Are Visible in the Cooler Regions of the Pellet

- Graduated zoom of pin 12, M-2



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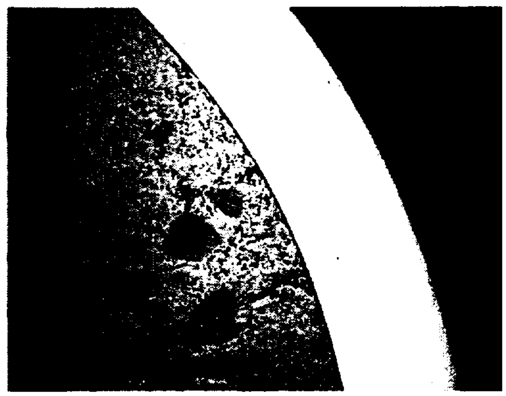
Agglomerates Are Unevenly Distributed which Is Typical in MOX Fuel

- In general the agglomerates in this test are unusually large and uneven in size
- Fuel pin 12, M-1



MXR82497 6143 200µm As Polished

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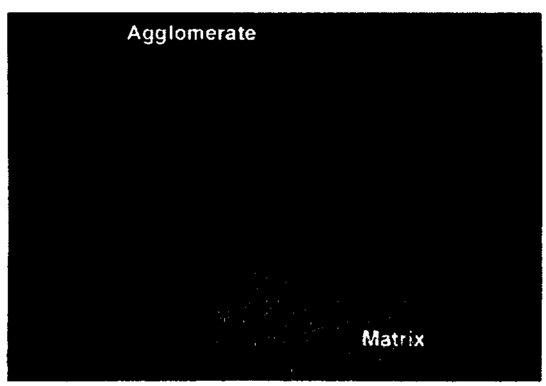
ORNL 2000-1781C EFG

SEM Analysis Shows Large Plutonium Difference between Agglomerate and Matrix

- Ratio of plutonium signals is about 7.
- Agglomerate is near the midradius point.



Plutonium - Green Uranium - Red Plutonium Agglomerate in Mount 6147

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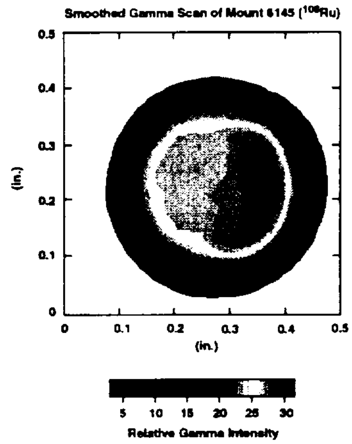
Gamma Scan of Fuel Pin 5 Cross Section

- Qualitative picture of predicted asymmetric burnup
- Resolution is actually quite coarse (6 to 8 points along radius), so this information is only qualitative



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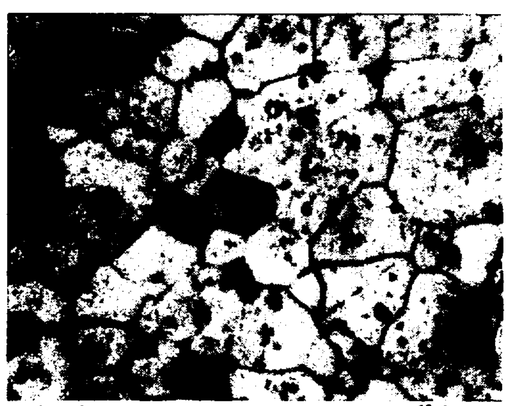
Grain Boundaries Similar to Preirradiation

- Etched mount from fuel pin 5, M-1



MXR82579 6146 Near Edge 10μm 2.5mL H_2SO_4 + 48mL H_2O_2

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ORNL 2000-1784C EFG

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Gallium Analysis Indicates No Significant Migration

- Average of the clad specimens
  - Pin 5 0.32 ppm (5 samples)
  - Pin 12 0.43 ppm (3 samples)
- Nominal clad gallium impurity level in the tenths of a part-per-million range
- Average of the pellets
  - Pin 5 2.2 ppm (2 samples)
  - Pin 12 1.3 ppm (1 sample)
- Gallium levels of pellets within expected range

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Gallium Migration Bounding Estimate Improved

- Conservative bounding estimates
  - Assumes no initial gallium in clad
  - No credit for fuel adhering to clad
- 8 GWd/MT
  - TIGR
    - Upper bound of 27% mass transfer to clad
  - Non-TIGR
    - No data
- 21 GWd/MT
  - TIGR
    - Upper bound of 14% mass transfer to clad
  - Non-TIGR
    - Upper bound of 9% mass transfer to clad

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Burnup Follows Predictions

- Three pellets were analyzed for burnup using the ^{148}Nd method:

Specimen	Pellet Location	Calculated Burnup (GWd/MT)	Radiochemistry Burnup (GWd/MT)
FP-5-P-1	2	22.3	23.3
FP-5-P-4	15	21.2	22.0
FP-12-P-1	15	21.2	22.5

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21-GWd/MT PIE Summary

- Capsule and fuel pin metrology as expected
  - Slight clad creep/growth
- Fission gas measured and within predicted limits
- No evidence of gallium migration or corrosion
- Metallography as expected
  - MOX agglomerates more typical of early MOX fuel than present commercial product
  - Grain boundaries as expected
  - No significant fuel restructuring
  - Fuel behavior used to improve CARTS fuel model
- No abnormal behavior observed
- Clad samples archived

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30-GWd/MT PIE Examination Under Way

- Capsule 3
  - Contains fuel pin 6
  - Untreated fuel
- Capsule 10
  - Contains fuel pin 13
  - TIGR-treated fuel
- Both capsules were otherwise identical and irradiated under identical conditions
  - EOI July 23, 2000

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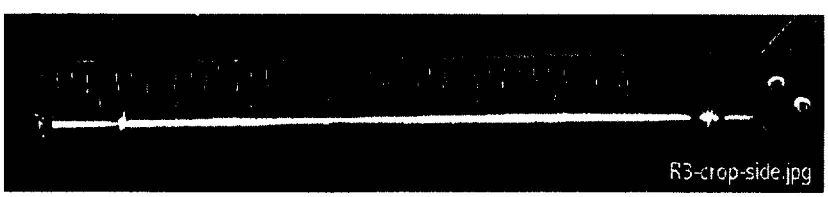
Capsule 3 in Hot Cell

- Excellent condition



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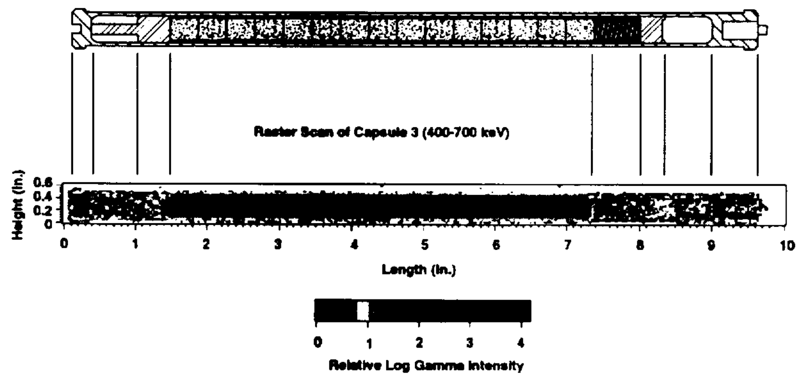
Capsule 3 Gamma Scan

- No internal abnormalities
  - The nominal fuel pin schematic has been shifted within the capsule and fuel stack and the spring scaled to model the scan.



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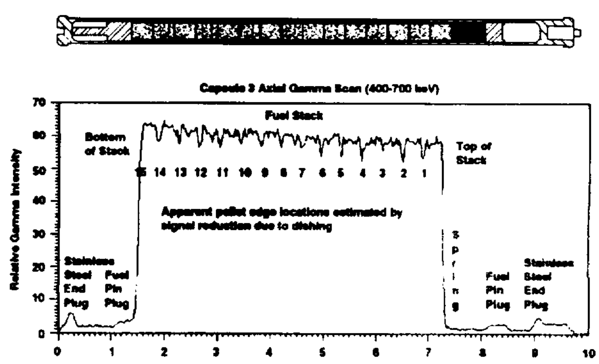
Capsule 3 Fuel Pin 6

- Gamma scan reveals intact pellet stack.
  - The nominal fuel pin schematic has been shifted within the capsule and the fuel stack and the spring scaled to model the scan.



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MOX PIE Overview

- PIE is moving along in a timely manner.
  - Capsule fission gas measurement capability was demonstrated.
  - SEM provided plutonium distribution data.
- Observations were in accordance with predictions.
  - No abnormal fuel swelling
  - Large plutonium-rich agglomerates
  - Fuel behavior is within the international database
- No evidence of gallium migration or corrosion exists.
  - Small amount of gallium in MOX mostly eliminated during sintering.
  - Trace gallium may have been present all along.
- Clad ductility testing is pending.
  - Samples being archived

Fuel Performance Calculations in Support of PIE

L. J. Ott
Oak Ridge National Laboratory

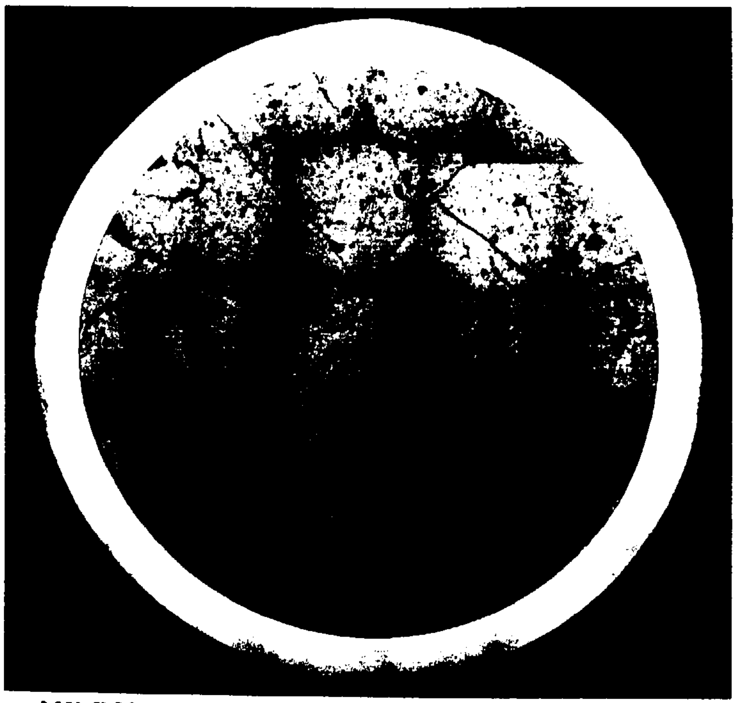
Presented at

ORNL MOX Fuel Program
Research and Development Meeting
Oak Ridge, Tennessee
December 12, 2000

Outline of Effort to Support PIE MOX Fuel Observations

- Predictions of capsule conditions before opening:
  - Fuel and structural dimensions
  - Fission gas pressure
- Interpretation of PIE metrology and metallographic mounts
- Assessment of APT fuel densification and swelling

Seven Metallographic Mounts Have Been Prepared from the Early (~8- GWd/MT) and Intermediate (~21-GWd/MT) Withdrawal Capsules



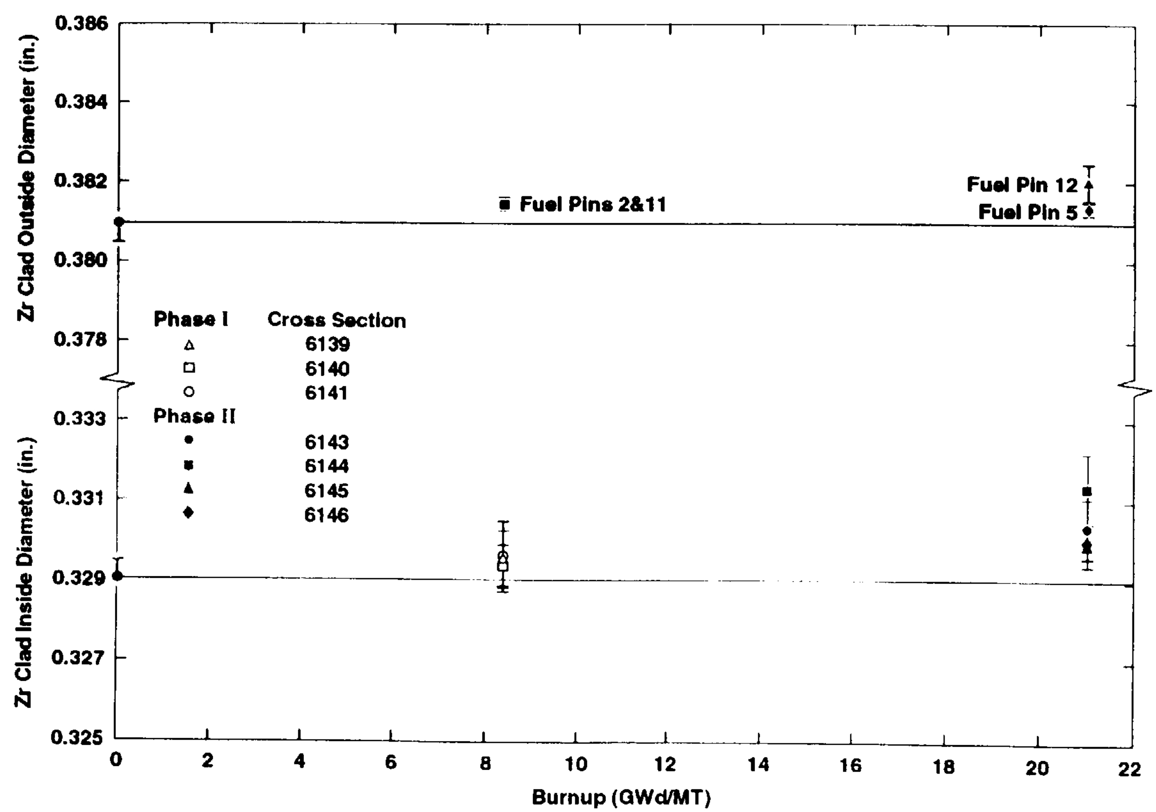
MX R82516 6143

Fuel Pin Metallographic Mount Identification

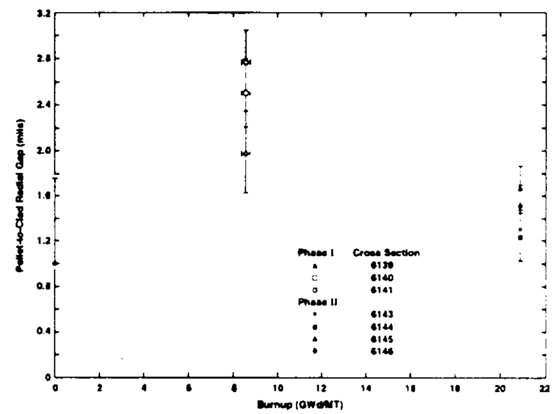
APT Irradiation Phase	Mount ID No.	Capsule	Fuel Pin	Fuel Type	Axial Location (Pellet No.)
I	6139	1	2	Non-TIGR	5
I	6140	1	2	Non-TIGR	5/6 Interface
I	6141	1	2	Non-TIGR	6
II	6143	9	12	TIGR	1
II	6144	9	12	TIGR	14
II	6145	2	5	Non-TIGR	1
II	6146	2	5	Non-TIGR	14

- Precise measurement of fuel pin outer diameter followed by close examination of metallographic mounts leads to:
  - Zircaloy clad wall thickness and inner diameter
  - Free area within the fuel pin (gaps and cracks)
  - Fuel pellet "outer diameter"

Cold Clad Inner and Outer Diameters as Measured in APT MOX Fuel Pins Reveal an Increase in Diameter with Irradiation



Cold Radial Pellet-to-Clad Gaps Reflect Prototypic Fuel Densification and Swelling



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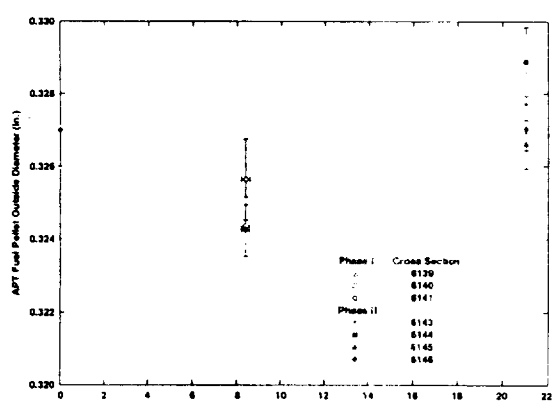
Cold Pellet Outer Diameter Also Illustrates the Fuel Densification and Swelling



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The APT MOX Fuel Densification Can Be Assessed via FRAPCON-3 and ESCORE Fuel Performance Models

- CARTS simulations utilizing
  - FRAPCON-3 fuel densification and swelling models
  - ESCORE fuel densification and swelling models
- Fuel densification and swelling models are explicitly **assumed** applicable:
  - The only degree of freedom is the fuel densification assumed.
  - The "best estimate" densification is within the expected European data range of 1–2%.

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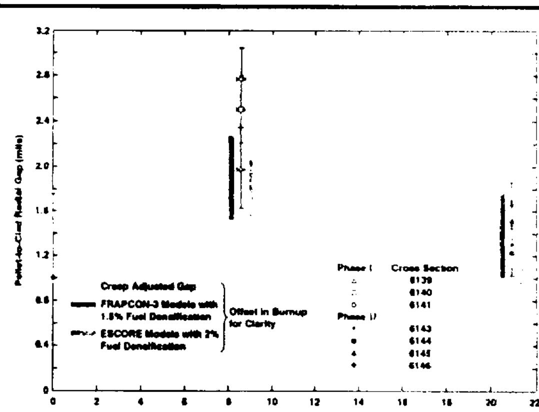
The Cold Radial Pellet-to-Clad Gap Behavior in the APT MOX Fuel Pins is Best Described with the ESCORE Models by Assuming a Densification of 2%. With the FRAPCON-3 Models, a Value of 1.5% Is Appropriate.



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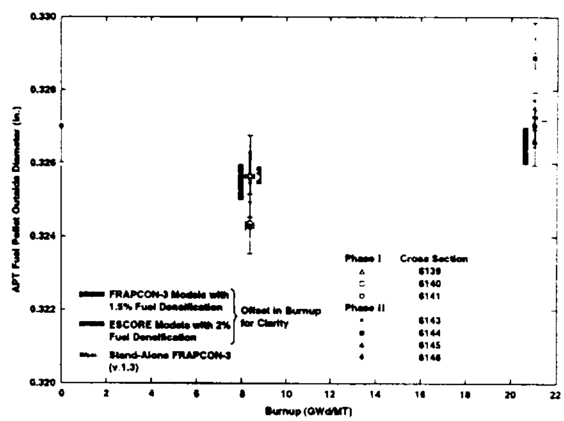
The APT MOX Cold Pellet Outer Diameter Trace Is also Adequately Predicted with 2% Densification for the ESCORE Models and 1.5% Densification for the FRAPCON-3 Models



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Conclusions

1. No significant difference exists dimensionally between the TIGR-treated and the untreated MOX fuels.
2. Pellet cracking is evident, but considered normal in view of the thermal cycling and LHGRs experienced during the irradiation.
3. Fuel densification (1-2%) is prototypic of commercial MOX fuel.
4. Clad expansion at 21 GWd/MT is ~0.3%.
5. FRAPCON-3 or ESCORE fuel swelling models adequately describe the APT MOX behavior (given the appropriate fuel densification).

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Ductility Test for MOX Fuel Clad

Terry Yahr
Bill Hendrich
Claire Luttrell
Oak Ridge National Laboratory

Presented at

ORNL MOX Fuel Program
Research and Development Meeting
Oak Ridge, Tennessee
December 12, 2000

Why Are We Developing a Clad Ductility Test ?

- A potential concern is that gallium in MOX fuel may reduce the clad ductility.
- Irradiation alone reduces clad ductility to only 3–5%.
- Fuel clad from Light Water Reactor Mixed-Oxide Fuel Irradiation Test has no hydrides, so any effect of gallium can be observed
- Zircaloy clad is anisotropic, and fuel swelling produces hoop strain in clad.
- Available tests are not well suited for measuring fuel clad hoop strains <5% accurately.

Outline of Presentation

- Previous methods of testing fuel clad
  - Axial tension
  - Tube burst
  - Expanding mandrel
  - Ring-stretch test
  - Hourglass ring-stretch test
- Compressed-plug test
- Summary

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Zircaloy-4 Cladding Has Lower Ductility in the Circumferential Direction than in the Axial Direction

- Axial tensile properties

– UTS	120,000 psi	111,000 psi
– YS	83,000 psi	83,000 psi
– Elongation	22%	20%

- Circumferential tensile properties (burst test)

– Hoop stress	140,000 psi	140,000 psi
– Elongation	15%	12%

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Closed-End Burst Test Is Required by ASTM B811, but It Has Disadvantages

- Minimum clear length is ten times OD.
  - 10 x 0.381 = 3.81 in.
  - Fuel pins are 7.203 in. long
- Strain is determined from posttest measurement of circumference to an accuracy of ±0.005 in.
  - Strain in 0.381-in. OD specimen is accurate to ±0.4%
- Concerned about high pressure and fluid release in hot-cell

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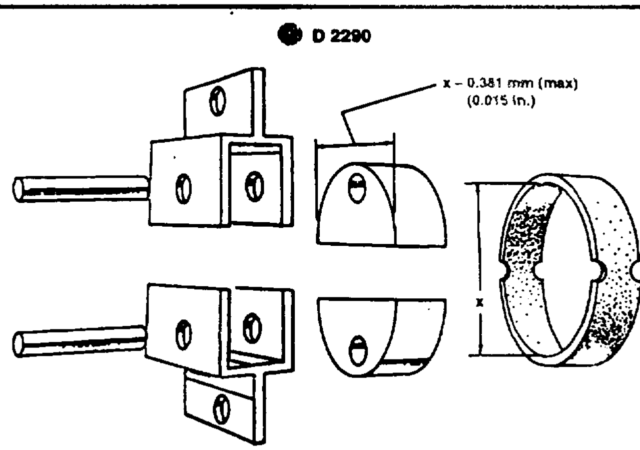
Ring-Stretch Specimen Has Been Widely Used



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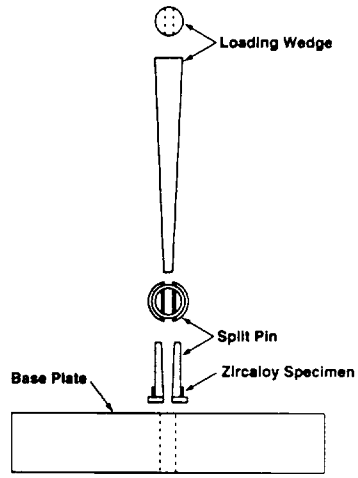
Wedge-Loaded Split-Pin Applies Stress in Circumferential Direction to Short Length of Cladding



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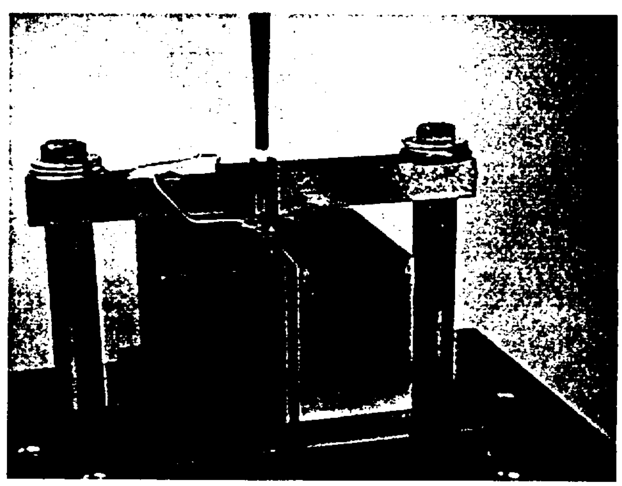
Specimen Mounted in Test Fixture with Pin Ready to Be Inserted



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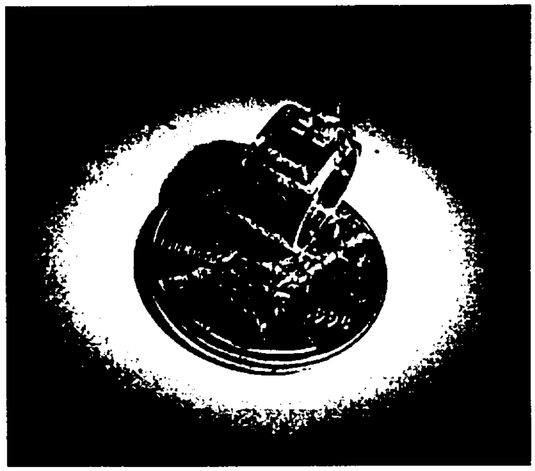
ORNL 2000-1911C EFG

Specimen Is 0.250 in. Long with an I.D. of 0.329 in. and a Wall Thickness of 0.026 in. The Strain Gage Has a Gage Length of 0.032 in.



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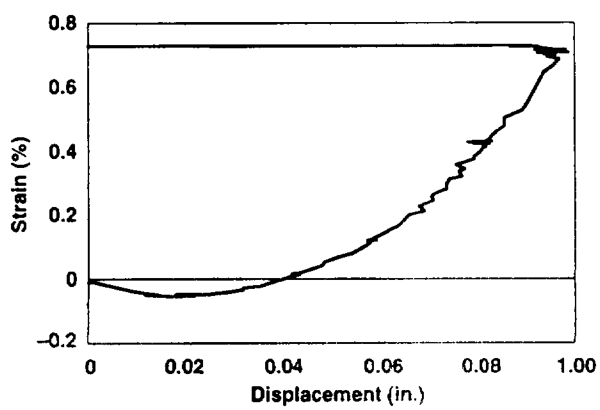
UT-BATTELLE

Strain in the Wedge-Loaded Ring-Tensile specimen Is Initially Negative



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ORNL 2000-1984C EFG

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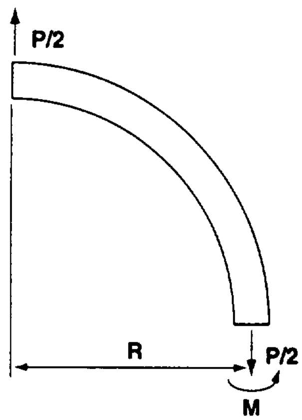
Bending Predominates in Ring-Stretch Test When Insert Is Smaller Than Ring



- Elastic stress
  - 0.381 OD
  - 0.329 ID
  - 0.300 long
- $S_i = P/2A + RPt/4I$
- $S_i = 64P + 2625P = +2689\ P$
- $S_o = P/2A - RPt/4I$
- $S_o = 64P - 2625P = -2562\ P$

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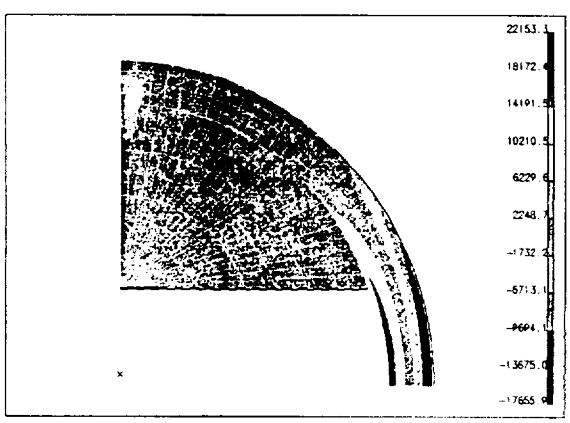
Loading with an Undersized Circular-Segment Produces Tension on the Inside Surface



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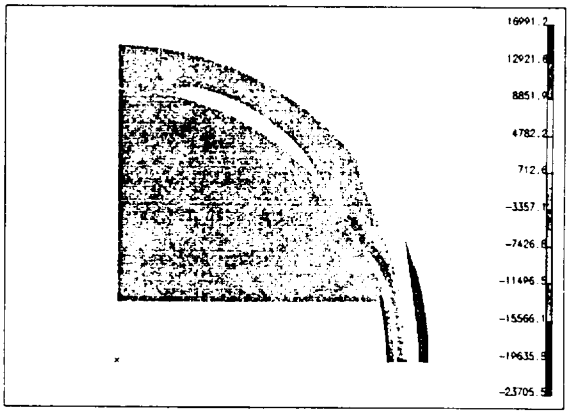
Loading with an Oversized Circular-Segment Produces Tension on the Outside Surface



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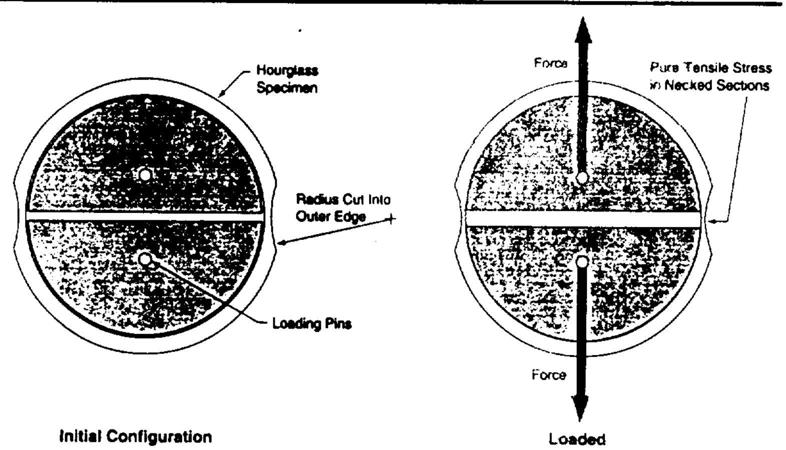
Machined Ring Will Produce Uniform Stress across Clad Cross-Section and Constitutes an Acceptable Test for Evaluating the Residual Ductility of the Clad Ring Samples Since We Are Looking for Gallium Effect



Clad Ductility Hourglass Test Setup

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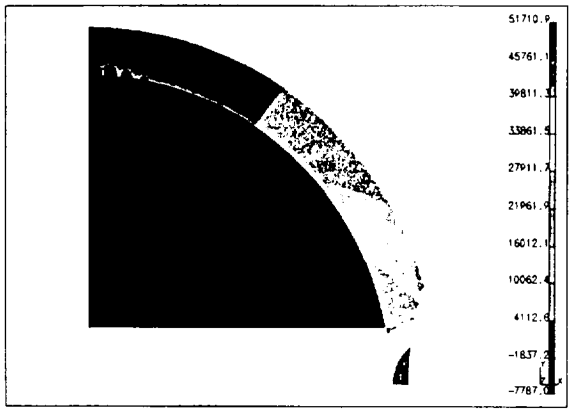
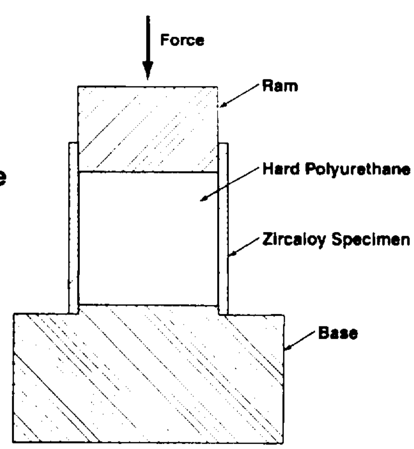
Finite-Element Analysis Showed that the Through-Wall Bending Is Reduced in the Hourglass Ring-Stretch Specimen

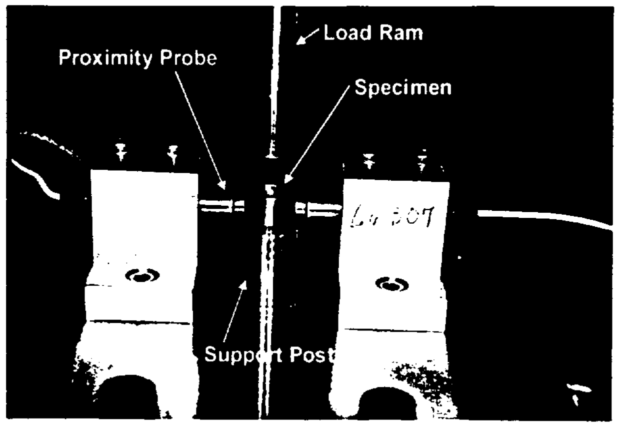


Compressing a Polyurethane Plug Fitted Inside a Short Piece of Cladding Forces It to Expand in a Manner Similar to Swelling Fuel

- Circumferential strain is the change in diameter divided by the initial diameter



Load Ram Compresses a Polyurethane Plug that Is Inserted in the Specimen



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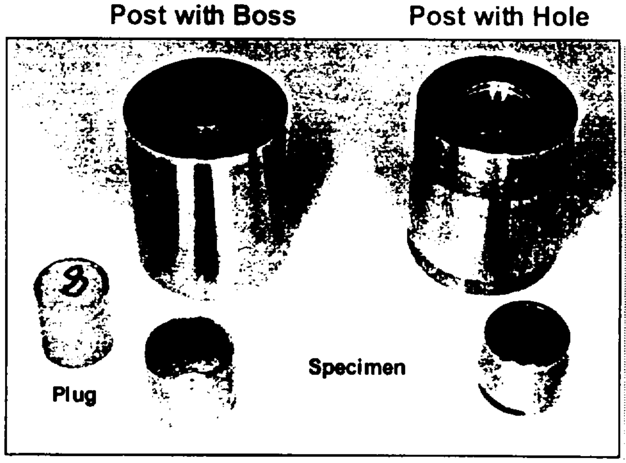
The Lower Post Supports and Accurately Positions the Specimen

Post with Boss Post with Hole



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ORNL 2000-1915C EFG

Tests on Unirradiated Zircaloy-4 Cladding Demonstrated that the Compressed Plug Approach Can Induce >10% Strain (0.020-in. boss)

Test No.	Specimen No.	Plug Hardness (Shore)	Peak Compressive Load (lb)	Peak Hoop Strain (%)	Residual Hoop Strain (%)
1	1	A-95	1700	3.4	2.6
2	2	D-75	2000	3.1	2.7
3	3	A-90	1600	5.6	4.8
7	254	A-95	3000	12.5	11.5
9	255	A-95	3000	9.2	8.2
4	351	A-95	2750	7.3	6.3
5	352	A-95	2700	5.4	4.4
6	353	A-95	2900	10.7	9.6
8	354	A-95	2750	7.0	6.0

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Inserting the Plug into a 0.020-in. Deep Hole Was also Successful with Virgin Zircaloy-4

Test No.	Specimen No.	Plug Hardness (Shore)	Peak Compressive Load (lb)	Peak Hoop Strain (%)	Residual Hoop Strain (%)
12	257	D-75	3800	8.8	7.7
13	302	A-95	2700	5.4	4.6
14	303	A-95	2700	5.4	4.6

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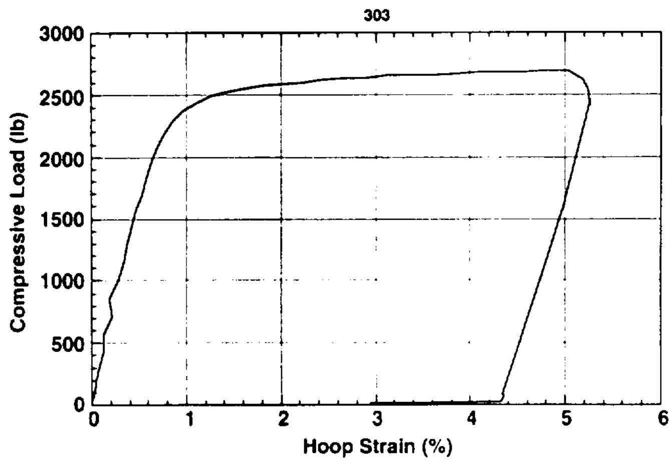
Load vs Strain Curve for Unirradiated Zircaloy-4 Specimen Is Similar to a Stress-Strain Curve



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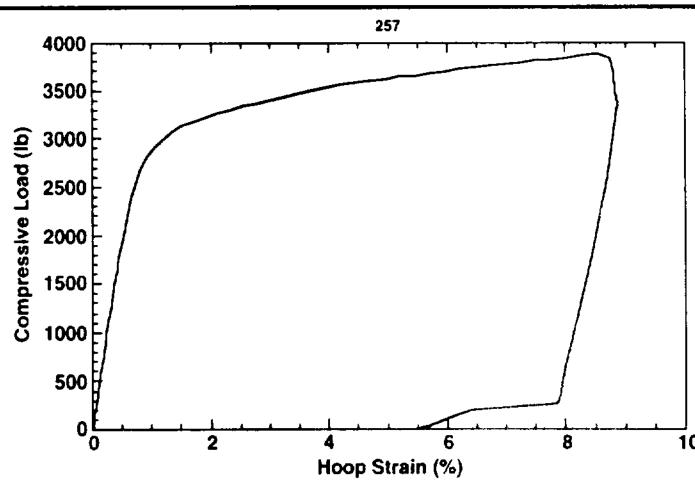
Some Unirradiated Zircaloy-4 Specimens Were Taken to High Strains, but None Have Fractured



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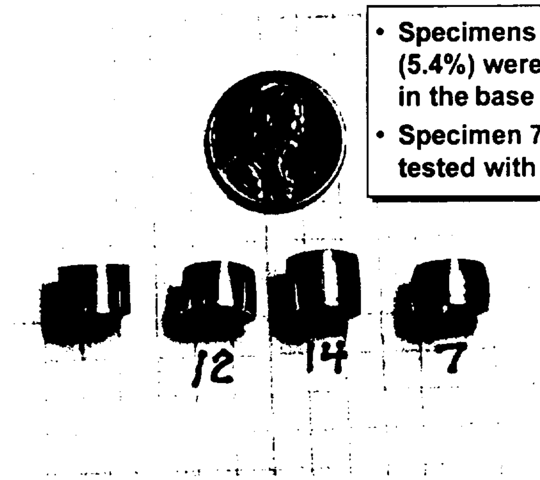
Barreling Can Be Controlled to Some Extent



- Specimens 12 (8.8%) & 14 (5.4%) were tested with a hole in the base
- Specimen 7 (12.5%) was tested with a boss on the base

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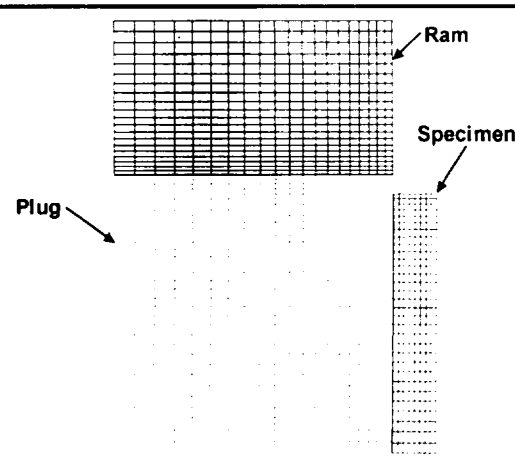
Finite-Element Model of Compressed-Plug Clad Ductility Test Included 1,320 8-Mode Axisymmetric Elements



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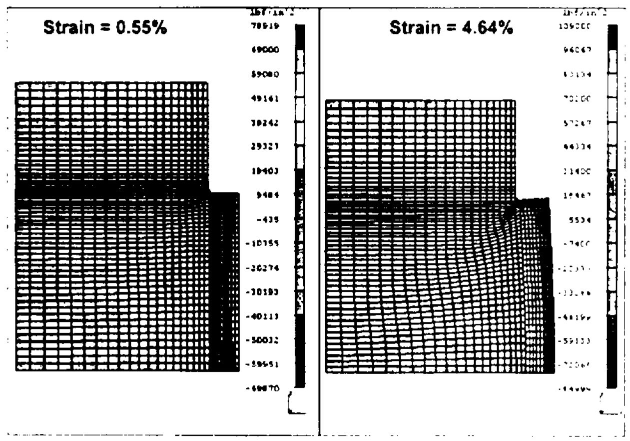
Hoop Stresses in the Specimen Were Determined by FEA.



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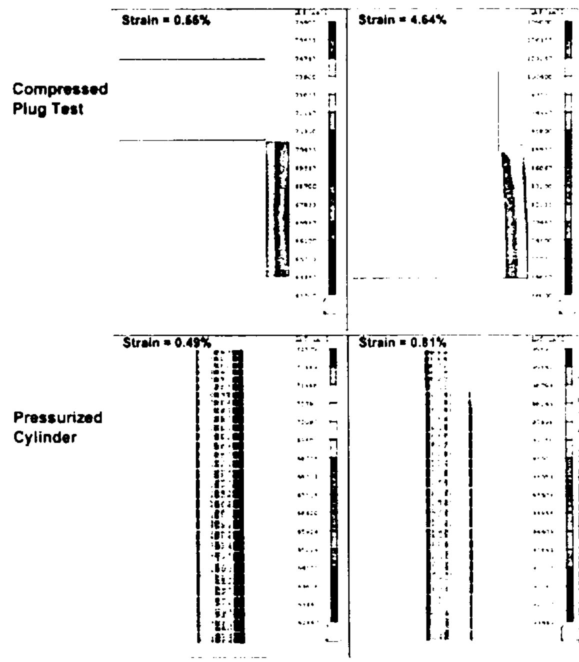
Stress Distributions in Compressed-Plug Test and Pressurized Cylinder Are Similar



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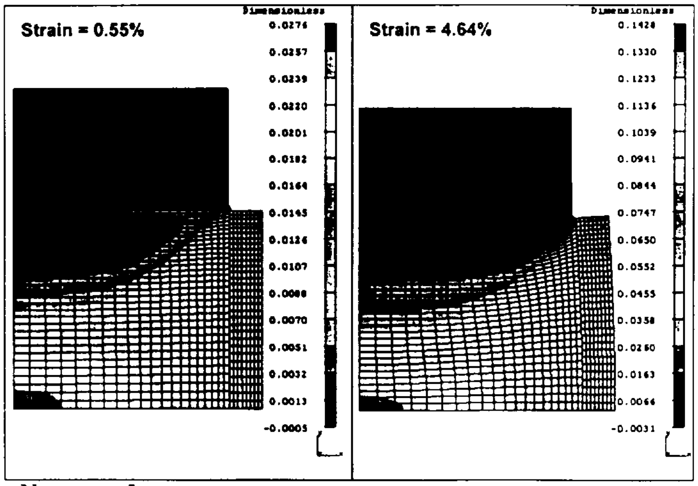
ORNL 2000 1829C EFG

The Maximum Strain Is at the Middle of the Compressed-Plug Specimen



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UT-BATTELLE
ORNL 2000-1830C EFG

Tool Steel Specimens (UTS = 300 ksi) Were Tested to Ensure that Irradiated Specimens Can Be Tested and to See if Fracture Could Be Detected

Test No.	Specimen No.	Plug Hardness (Shore)	Peak Compressive Load (lb)	Hoop Fracture Strain (%)
18	307	D-75	8100	4.4
19	308	D-75	8750	3.4
20	309	A-95	7900	5.8
22	311	D-90	11310	3.9
23	3012	A-95	6800	4.8

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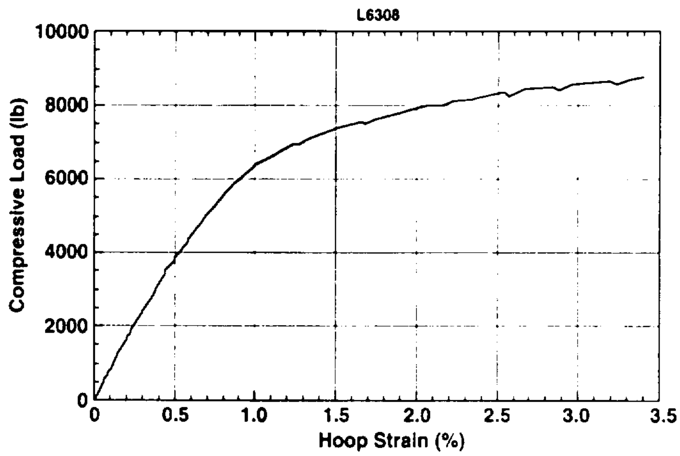
Tool Steel Specimens Fractured at Higher Loads than Were Applied to the Zircaloy-4 Specimens



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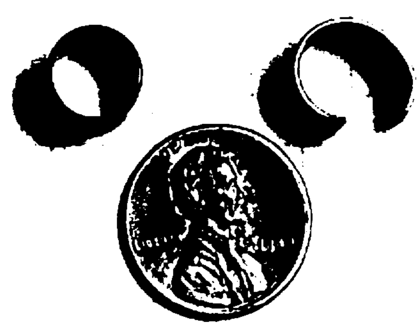
Tool Steel Specimens Deform When They Fracture



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Test Is Being Developed to Measure the Ductility of Irradiated MOX Fuel Clad

- Irradiation reduces the ductility of Zircaloy fuel clad.
- It is important to measure ductility in a circumferential direction because Zircaloy is anisotropic.
- A ring of the clad is loaded by applying an axial compressive load to a plug of polyurethane fitted inside the specimen.
- Change in the specimen diameter is measured while the load is being applied.
- Circumferential strain is change in diameter divided by diameter.
- Unirradiated Zircaloy-4 has been strained to >10%.
- Tool steel specimens have fractured.

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Bench Testing Has Demonstrated Proof-of-Principle for Compressed Plug Concept

- Concept is applied to unirradiated Zircaloy-4 and tool steel.
- Strains are imposed in similar manner as by swelling fuel.
- Specimen preparation is simple.
- Small specimens make good use of limited irradiated clad.
- Results are straightforward to interpret.

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Ductility Tests on Irradiated Clad from ATR Average Power Test Will Be Done after All Interested Project Participants Concur on the Test Method

- Clad is unique because gallium was present in fuel from the start.
- A spectrum of samples is being accumulated, with burnups of 8, 21, 30, 40, and 50 GWd/MT.
- There will be no hydriding to mask any effect of gallium because the fuel pins are irradiated in an inert environment.
- Compressed Plug method has several advantages over other methods, including ring-stretch test.
  - Loading is prototypic.
  - Strain can be measured accurately.
  - Simple specimen preparation
  - It may be possible to measure stress as well.

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Further Development of the Compressed Plug Test Is Planned to Ensure Success in Hot-Cell

- Investigate critical dimensions (analytical and experimental)
  - Minimum ram diameter
  - Avoid extrusion of plug
  - Control amount of barreling
- Investigate use of lubricant
- Strain gage specimen for comparison with strain determined from change in diameter
- Develop method for cutting specimens in the hot-cell
- Publish "proof-of-principle" report
- Design and assemble apparatus to test irradiated material from in-reactor MOX fuel tests

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Potential Work Beyond Current Project

- Develop way to determine stress-strain curve
- Develop apparatus for testing at elevated temperature
- Gain acceptance as ASTM Test Procedure

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Reactor Physics, Criticality Safety, and Shielding Analyses for MOX Fuels

R. T. Primm, III
Oak Ridge National Laboratory

Presented at

ORNL MOX Fuel Program
Research and Development Meeting
Oak Ridge, Tennessee
December 12, 2000

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Content of Presentation

- Physics-related differences between reactor- and weapons-grade plutonium and LEU
- Use of burnable absorber pins in MOX assemblies
- ARIANE MOX destructive assay program and joint FY 2001 study with FCF
- Physics models of Catawba reactor with MOX fuel

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Topic: Physics-Related Differences — Validation Studies

- Extensive validation with all types of MOX critical experiments
- SCALE, HELIOS (assembly) , NESTLE (core), and MCNP
- k_{eff} and power distribution C/E values independent of ^{240}Pu content: range from 0.985 to 1.0 for MOX; 0.99 for LEU (ENDF/B-VI)
- Discrepancies between measured and calculated pin powers at MOX/LEU interfaces (<10%)
- Within a MOX region, measured-to-calculated pin power ratios are the same as for LEU (agree to about 1%)
- Multiplication factors for MOX increase as pitch increases

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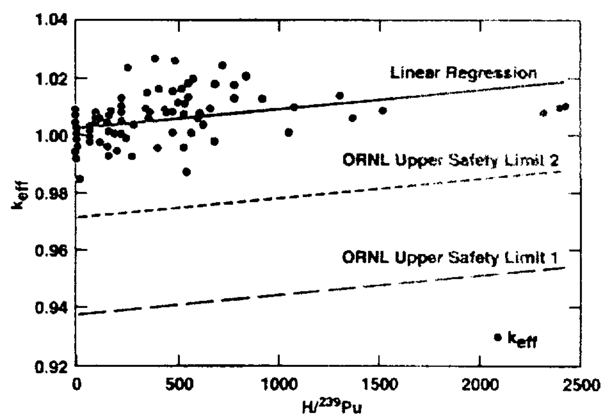
Results of Benchmark Calculations Show Trend with Moderation



101 Calculations with 238-Group ENDF/B-V: k_{eff} vs $H/^{239}Pu$

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Data Sources for Reactor Physics Benchmarks

Publicly Available Critical Experiments

- SAXTON
- ESADA
- Battelle - PNNL
- KRITZ
- VENUS

Publicly Available Reactor Irradiation Data

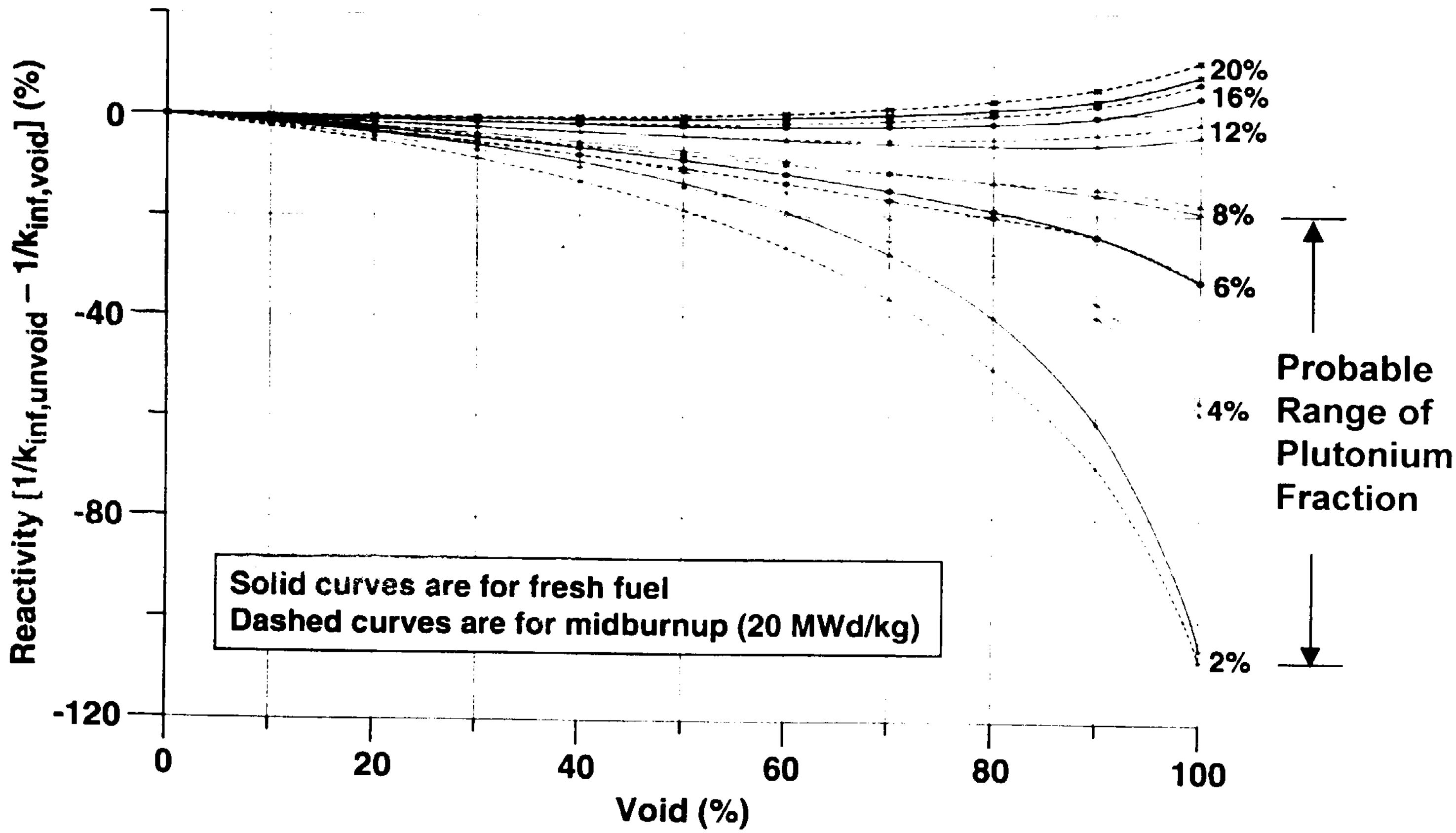
- Quad Cities
- San Onofre
- Saxton
- ARIANE
  - Beznau
  - Gosgen
  - Dodewaard

Proprietary Critical Experiments and Reactor

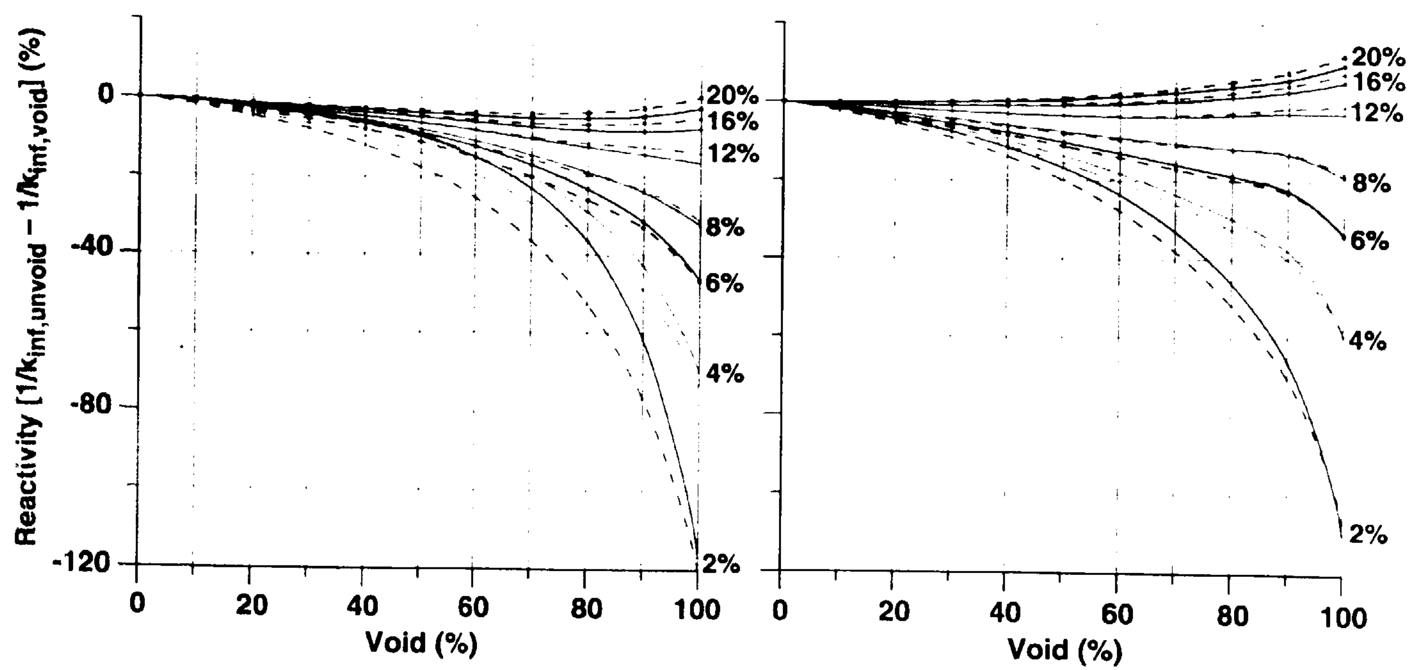
- KRITZ
- VENUS
- EPICURE
- ERASME
- Oldest 17 x 17 French data available
- Oldest French MOX data
- Others

Physics-Related Differences — Void Coefficient

- Confirmed that value is negative for WG plutonium



Void Coefficient for Weapons-Grade MOX Is Bounded by Reactor-Grade MOX and LEU*



Solid curves are for fresh fuel;
Dashed curves are for midburnup (20 MWd/kg)

*See PHYSOR 2000 paper by Ellis.

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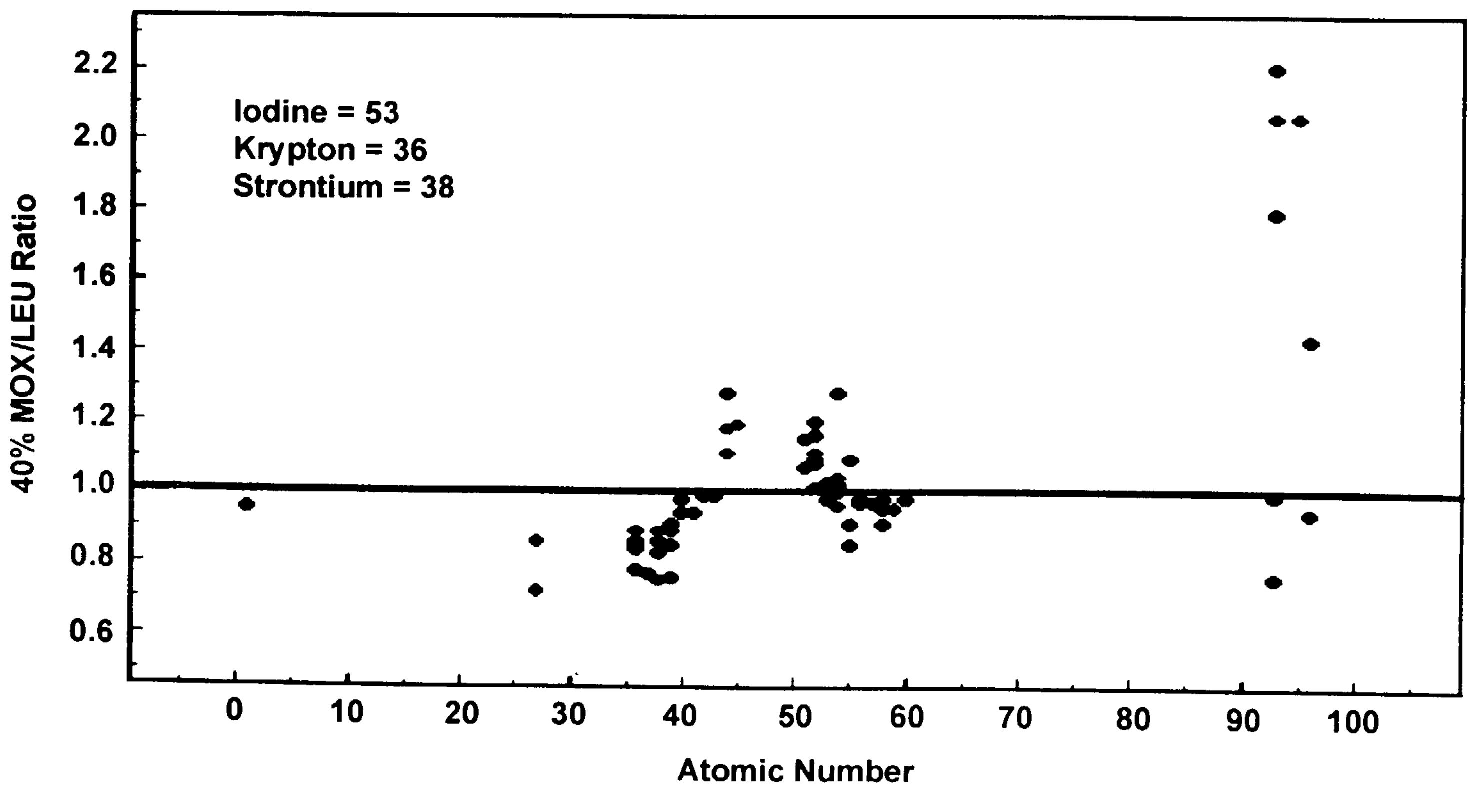
Strontium and Krypton Inventories for Partial MOX Cores Are Lower than Those for LEU (ARIANE validation)



Topic: Use of Burnable Poison Rods in MOX Assemblies

- A computational benchmark for a PWR 17 x 17 MOX assembly was sponsored by the American Nuclear Society; available at http://www.engr.utk.edu/org/ans/benchmark/ansmoxbm.html
- CASMO-4 (assembly code used by Duke) was used by some participants
- Removal of poison pins from MOX assembly after one cycle of irradiation
- Infinite MOX lattice

At BOL, Power Peaks at Center of Assembly (upper left corner)

0 MWd/kg (BOC1)

	1.106	1.037		1.016	1.014		1.003	0.974
1.106	1.052	1.022	1.020	1.005	1.003	1.011	0.992	0.973
1.037	1.022	1.009	1.019	1.007	1.005	1.012	0.994	0.973
	1.020	1.019		1.022	1.020		1.004	0.973
1.016	1.005	1.007	1.022	1.019	1.034	1.027	0.993	0.972
1.014	1.003	1.005	1.020	1.034		1.011	0.977	0.967
	1.011	1.012		1.027	1.011	0.985	0.970	0.963
1.003	0.992	0.994	1.004	0.993	0.977	0.970	0.962	0.963
0.974	0.973	0.973	0.973	0.972	0.967	0.963	0.963	0.960

After One Cycle (poison rods still in), Peak Is Less but Still at Center of Assembly

15 MWd/kg with BP Rods (EOC1)

	1.064	1.041		1.031	1.027		1.007	0.964
1.064	1.032	1.024	1.034	1.015	1.012	1.022	0.992	0.962
1.041	1.024	1.016	1.035	1.018	1.014	1.023	0.994	0.963
	1.034	1.035		1.040	1.038		1.009	0.963
1.031	1.015	1.018	1.040	1.038	1.056	1.042	0.991	0.957
1.027	1.012	1.014	1.038	1.056		1.019	0.966	0.949
	1.022	1.023		1.042	1.019	0.980	0.952	0.941
1.007	0.992	0.994	1.009	0.991	0.966	0.952	0.939	0.936
0.964	0.962	0.963	0.963	0.957	0.949	0.941	0.936	0.932

After Rods Pulled, Hot Spot Shifts to Pins Close to Water Hole and Increases (as for LEU fuel)

Local power density at "new" hot spot increases by 4.4% after poison rods are pulled.

15 MWd/kg, BP Rods Pulled (BOC2)

	1.041	1.053		1.053	1.049		1.016	0.942
1.041	1.014	1.025	1.056	1.025	1.021	1.042	0.989	0.939
1.053	1.025	1.021	1.059	1.029	1.026	1.044	0.991	0.939
	1.056	1.059		1.071	1.069		1.020	0.940
1.053	1.025	1.029	1.071	1.067	1.099	1.076	0.986	0.930
1.049	1.021	1.026	1.069	1.099		1.037	0.945	0.916
	1.042	1.044		1.076	1.037	0.970	0.921	0.903
1.016	0.989	0.991	1.020	0.986	0.945	0.921	0.901	0.895
0.942	0.939	0.939	0.940	0.930	0.916	0.903	0.895	0.889

For an Obsolete Assembly Design, When Poison Rods Are Pulled, the Hot Spot Moves to Adjacent-Water-Hole Location, and Power Density Increases (larger increase of 7.5%)

15 MWd/kg (EOC1) with BP Rods

	1.094	1.043		1.029	1.034		0.912	0.922
1.094	1.059	1.031	1.030	1.020	1.024	1.043	1.054	0.919
1.043	1.031	1.025	1.029	1.021	1.026	1.044	1.055	0.919
	1.030	1.029		1.034	1.041		0.914	0.921
1.029	1.020	1.021	1.034	1.039	1.052	1.057	1.056	0.919
1.034	1.024	1.026	1.041	1.052		1.047	1.046	0.922
	1.043	1.044		1.057	1.047	1.045	1.064	0.943
0.912	1.054	1.055	0.914	1.056	1.046	1.064	0.944	0.788
0.922	0.919	0.919	0.921	0.919	0.922	0.943	0.788	0.814

15 MWd/kg (EOC1) with BP Rods Pulled

	1.049	1.065		1.069	1.071		0.928	0.883
1.049	1.024	1.031	1.071	1.035	1.037	1.079	1.047	0.879
1.065	1.031	1.034	1.073	1.039	1.042	1.083	1.049	0.879
	1.071	1.073		1.089	1.095		0.933	0.882
1.069	1.035	1.039	1.089	1.093	1.130	1.119	1.045	0.873
1.071	1.037	1.042	1.095	1.130		1.079	1.003	0.865
	1.079	1.083		1.119	1.079	1.024	1.002	0.877
0.928	1.047	1.049	0.933	1.045	1.003	1.002	0.878	0.729
0.883	0.879	0.879	0.882	0.873	0.865	0.877	0.729	0.751

OECD/NEA Plutonium Disposition Reactor Physics Activities

Dr. Jess C. Gehin
Oak Ridge National Laboratory

Presented at

ORNL MOX Fuel Program
Research and Development Meeting
Oak Ridge, Tennessee
December 12, 2000

Overview

- Role of the OECD Nuclear Energy Agency (NEA) in plutonium disposition
- Task force on reactor-based plutonium disposition
- Benchmarking efforts
  - VENUS-2
  - KRITZ-2
  - VVER-1000 assemblies with uranium/gadolinium pins
- Future plans

Issues with MOX Assemblies in Control Rod Locations

- No current experience in French reactors but perhaps there will be some soon
- DP reports that locations are not the "control rod ejection limiting" locations
- ORNL would expect that not only would most reactive rod be examined but some discussion would take place of "worst" conditions to exist in a MOX assembly for rod ejection, even though the worst case in MOX does not correspond to maximum reactivity worth overall; the uncertainties in the MOX calculation may be larger than those for LEU (quantifying the margin of uncertainty for MOX cores is a proposed area of work for Russians for VVERs)

Other Comments from the Literature

- Following quote from "MOX Fuel Utilization in Belgian NPPs," March 1997 (FRAMATOME, three-loop 900-MW(e), 17 x 17)
- Discrepancies in MOX detector response prediction vs measurement were seen. "Because no explanation was found for these unusual deviations on MOX detector responses, it was decided, as a short term measure, to determine a bias to the MOX fission chamber responses calculated with that (CASMO-3) specific methodology."
- "In the cases of rod misalignment and rod drop, peaking factors are higher for mixed cores, but margins subsist [sic] regarding the criteria verification."

Topic: ARIANE — a Destructive Assay of MOX and LEU BWR and PWR Pins; Final DOE-Funded ORNL Domestic Physics Study

- Program managed by Belgonucleaire
- Fuel pins were extracted from assemblies
- Segments, 3 cm in length, were cut
- Samples sent to three analytical chemistry labs
- Analyses performed for the following:

Actinides

^{237}Np

^{232}U, all uranium isotopes above 233

Plutonium isotopes from 238 to 244

Americium and Curium isotopes with half-lives > 1 year

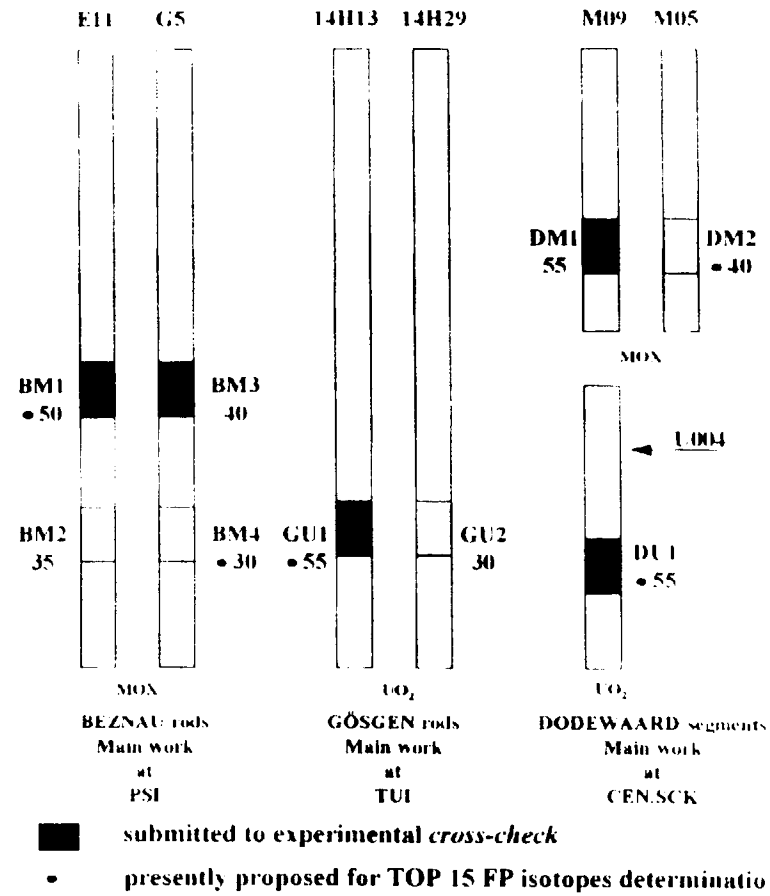
Fission Products

^{90}Sr, ^{95}Mo, ^{99}Tc, ^{101}Ru, ^{106}Ru, ^{103}Rh,
^{109}Ag, ^{125}So, ^{129}I,
^{133}Cs, ^{134}Cs, ^{135}Cs, ^{137}Cs,
^{142}Nd, ^{143}Nd, ^{144}Nd, ^{145}Nd, ^{146}Nd, ^{148}Nd, ^{150}Nd
^{144}Ce, ^{147}Pm,
^{147}Sm, ^{148}Sm, ^{149}Sm, ^{150}Sm, ^{151}Sm, ^{152}Sm, ^{154}Sm
^{151}Eu, ^{153}Eu, ^{154}Eu, ^{155}Eu,
^{155}Gd

The Final Data Report for ARIANE Is Expected in Mid-December 2000.

Schematic of locations of samples analyzed in ARIANE (burnup in GWd/MT)

Additional PWR and BWR samples (not shown) were added after the start of the program.



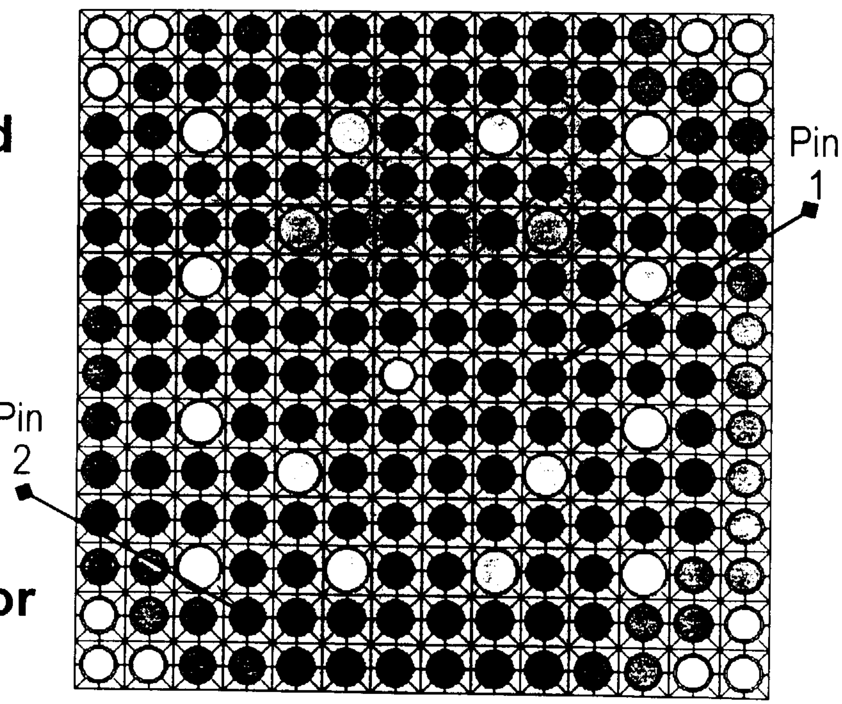
Average C/E Ratios for ARIANE and Other LEU/PWR Data (preliminary, nuclides with C/E < 0.9 or > 1.1)

Nuclide	*ARIANE MOX*	*All ARIANE*	*PWR-LEU Samples*[a] *(number of samples)*	*Consistent Discrepancy*
^{90}Sr	0.79	0.79	1.06(9)	
^{134}Cs	0.86	0.88	0.78(16)	X
^{135}Cs	1.12	1.10	1.06(9)	X
^{148}Sm	0.89	0.89	0.84(3)	X
^{149}Sm	1.12	1.10	0.66(3)	
^{151}Sm	1.30	1.29	1.32(3)	X
^{152}Sm	1.16	1.19	1.22(3)	X
^{155}Eu	0.58	0.61	0.74(3)	X
^{241}Am	1.24	1.22	0.88(9)	
^{242m}Am	1.25	1.29	0.89(6)	
^{246}Cm	0.80	0.81	—	

[a] Calculated with SCALE Version 4.2, SAS2H, 44-group ENDF/B-V; calculations performed by M. D. DeHart, ORNL (e-mail dehartmd@ornl.gov)

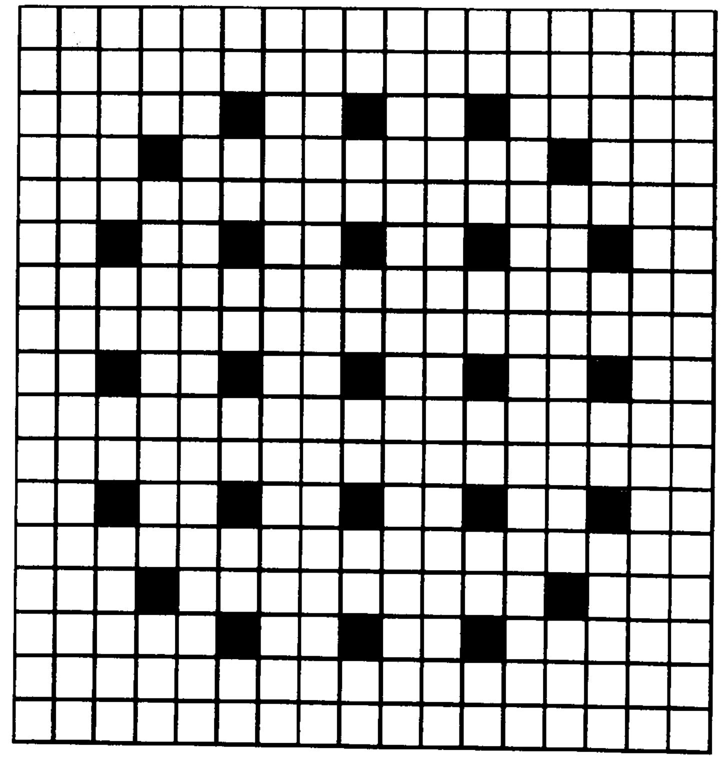
Distribution of ARIANE Data Is Restricted for Two Years Following the End of the Program

- Data useful for reactor physics code validation (integral check on end-of-life inventories; good indicator of proper spectral calculation) and environmental source-term calculations with SCALE
- Data can and will be transmitted to NRC
- ARIANE-like problem created for joint ORNL/DP-FCF study



Topic: ORNL Is Performing a Single Calculation of the "Equilibrium MOX" Catawba Core

- Four-group cross-section libraries (color sets) created by using HELIOS program with ENDF/B-VI library
- Reactor core model created with NESTLE
- Documentation of source of input data contained in ORNL/TM-1999/255



MOX assembly model

An "Equilibrium MOX" Core Configuration Is Being Modeled by ORNL Staff

	4.17 20 @3.5 0	4.24	4.40 20 @3.0 0	4.45	4.17 20 @4.0 0	4.17 128 IFBA 0	
4.17 20 @3.5 0		4.17 24 @3.5 0		4.40 24 @3.5 0		4.37 16 @2.0 0	
4.24	4.17 24 @3.5 0		4.07 24 @4.0 0			4.40 128 IFBA 0	
4.40 24 @3.0 0		4.07 24 @4.0 0			4.17 128 IFBA 0	4.37 0	
4.45 1	4.40 24 @3.5 0		4.45 1	4.24 1	4.37 20 @2.0 0	4.24 1	
4.17 24 @4.0 0			4.17 128 IFBA 0	4.37 20 @2.0 0	4.37 0		
4.17 104 IFBA 0	4.37 16 @2.0 0	4.40 128 IFBA 0	4.37 0	4.24			

Fuel enrichment (^{235}U or Pu)
BPR at ^{10}B enrichment or IFBAs
Fuel cycles irradiated to date

MOX Feed

Once-burned MOX

LEU Feed

Once-burned LEU

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OECD/NEA Plutonium Disposition Reactor Physics Activities

Dr. Jess C. Gehin
Oak Ridge National Laboratory

Presented at

ORNL MOX Fuel Program
Research and Development Meeting
Oak Ridge, Tennessee
December 12, 2000

Overview

- Role of the OECD Nuclear Energy Agency (NEA) in plutonium disposition
- Task force on reactor-based plutonium disposition
- Benchmarking efforts
  - VENUS-2
  - KRITZ-2
  - VVER-1000 assemblies with uranium/gadolinium pins
- Future plans

Role of the OECD/NEA in Plutonium Disposition

- The OECD/NEA provides a forum for cooperation among member countries.
- NEA Membership consists of 27 countries (not Russia).
- Most importantly, several NEA member countries have significant experience with MOX fuel.
- The role of OECD/NEA is to provide a forum for an international exchange of information on MOX fuel for the plutonium disposition mission.

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Task Force on Reactor-Based Plutonium Disposition

- Formed at the request of the United States and the Russian Federation to directly address the issue of plutonium disposition.
- Experts from several countries participate in two meetings annually and in benchmarking activities.
- Work to date has focused on benchmarking efforts in physics and fuel performance.
- OECD/NEA has obtained the release of previously proprietary critical experiment data (VENUS-2, KRITZ-2).
- Additional calculational benchmarks provide a good comparison of methods and data.

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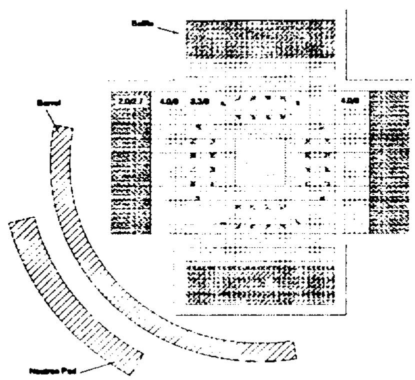
VENUS-2 Experiment

- Experimental data released to OECD/NEA by SCK/CEN (Belgium).
- Blind benchmark performed by 12 participants.
- ORNL analyzed the experiment with HELIOS and ENDF/B-VI data.
- Other participants used a wide variety of methods and data.
- Specifications are being developed for a 3-D VENUS-2 benchmark exercise.
- OECD/NEA report is being prepared.
- ORNL/TM documents ORNL modeling.

VENUS-2 MOX Core



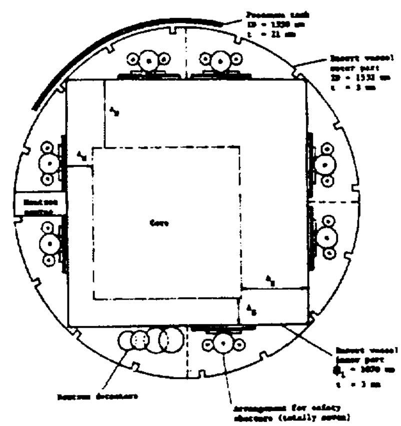
VENUS-2: Multiplication Factor Results

Multiplication Factor Results

Institution	Code	k_{eff}	Comments
NEA	DORT	0.99452	44G
KAERI	HELIOS-1.5	0.99817	35G
SCK	DORT	0.99233	44G
PSI	BOXER	1.00378	21G
US-TRKOV	GNOMER	0.99450 0.99977	JEF-2.2 (4G) ENDF/B-VI (4G)
NEA+KAERI	MCNP4B	1.00213	
JAERI	MVP	NO CALCULATION	
ORNL	HELIOS-1.4	1.00150 0.99907 0.99870	ENDF/B-VI (34G) ENDF/B-VI (89G) ENDF/B-VI (190G)
KI	MCU-B	0.99650	
KFKI	MCNP4B	1.00050	
GRS	MCNP4B	1.00430	
US-JERAJ	MCNP4B	0.99570	

KRITZ-2 Experiment

- Experimental data were released to OECD/NEA under TFRPD.
- Experiments were performed at temperatures up to 245°C.
- Critical water level and fission rate distributions were measured.
- ORNL prepared detailed specifications for this benchmark.
- ORNL analyzed experiment with HELIOS.
- Other participants are calculating experiment and will submit results in January.



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KRITZ-2: Critical Configurations

Core	Rod type	Number of rods	Temp. (°C)	Boron conc. (ppm)	$B_z^2 \times 10^4$ (cm^{-2})	HK (mm)
KRITZ 2:1	U	44 x 44	19.7	217.9	14.75	652.8
			248.5	26.2	6.25	1055.2
KRITZ 2:13	U	40 x 40	22.1	451.9	8.01	961.7
			243.0	280.1	5.98	1109.6
KRITZ 2:19	Pu	25 x 24	21.1	4.8	16.37	665.6
			235.9	5.2	7.70	1000.1

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KRITZ-2: Multiplication Factor Results

KRITZ Core/ Temperature	MCU	CASMO	APOLLO/ JEF2 (B_z)	WIMS6/ JEF2, 172 groups	WIMS6/ JEF2, 69 groups	APOLLO/ JEF2 (B_r)	HELIOS -1.5 35g
2:1 20 °C	0.9963	1.00050	0.99928	0.9997	1.0003	0.99843	1.00003
2:1 245 °C	0.9933	0.99830	0.99889	0.9986	0.9995	0.99801	0.99714
2:13 20 °C	0.9972	1.00074	1.00127	1.0002	0.9995	1.00239	1.00115
2:13 245 °C	0.9968	1.00154	1.00142	0.9988	0.9996	1.00013	0.99952
2:19 20 °C	0.9975			1.0005	1.0014		1.00133
2:19 245 °C	0.9975			0.9979	0.9981		1.00026

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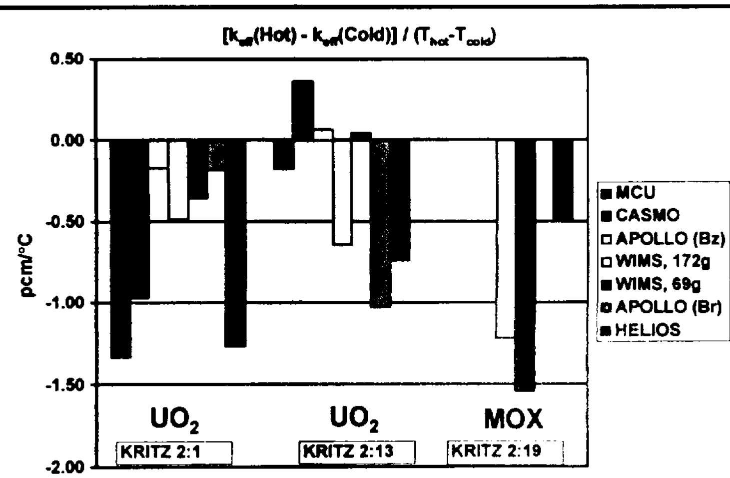
KRITZ-2: Temperature Effect Results



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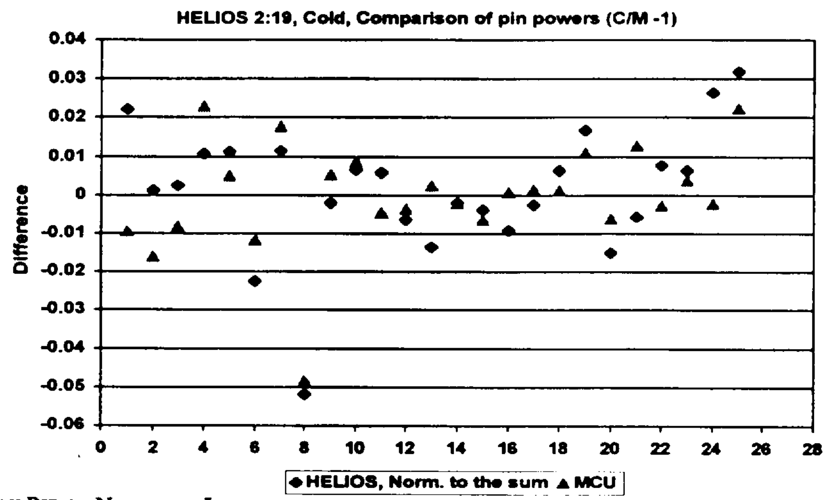
KRITZ-2: MOX Case Relative Pin Fission Rates (cold)



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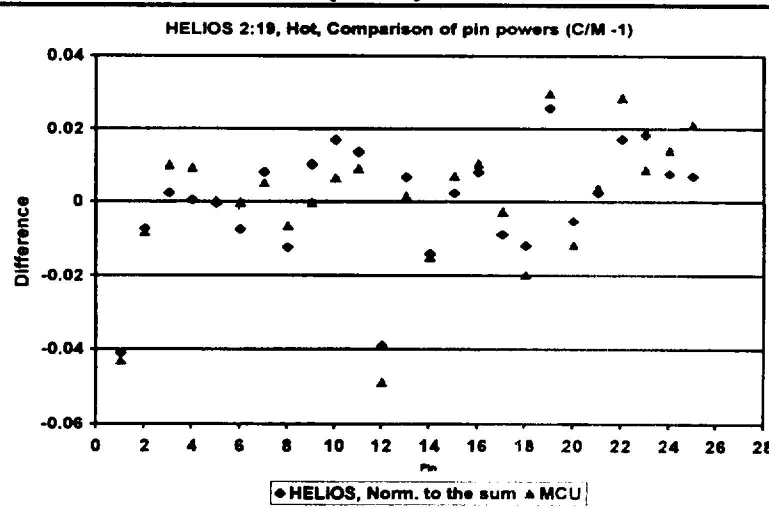
KRITZ-2: MOX Case Relative Pin Fission Rates (hot)



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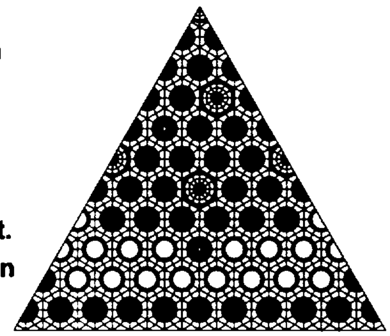
VVER-1000 Assembly Benchmark

- Calculational benchmark proposed by Kurchatov Institute (KI).
- Consists of isolated UOX and MOX assemblies with uranium/gadolinium pins with burnup.
- Calculations performed by five participants.
- ORNL analyzed with HELIOS.
- Results indicate excellent agreement.
- Provides good verification of Russian methods.
- Report being prepared by KI.
- ORNL/TM documenting ORNL analysis.

1/6 Assembly Model



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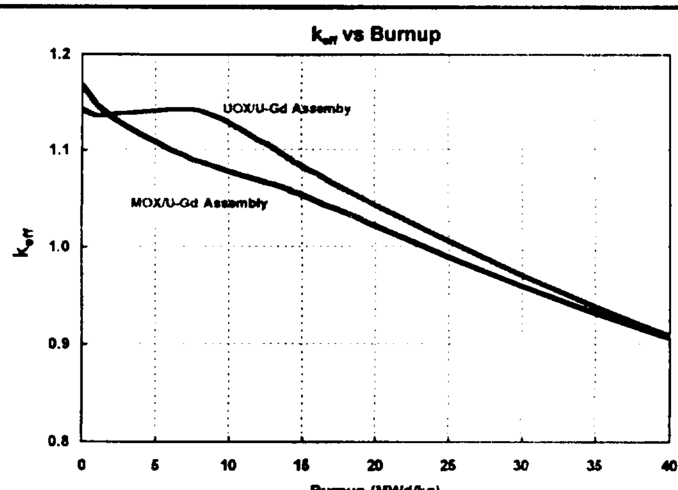
VVER-1000 Assembly Benchmark: k_{eff} Behavior vs Burnup



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VVER-1000 Assembly Benchmark Results

- Multiplication factors within ±0.5% for both UOX and MOX.
- Pin-by-pin fission rate distribution with 2.5% for UOX and 3% for MOX.
- Isotopic compositions show good agreement.

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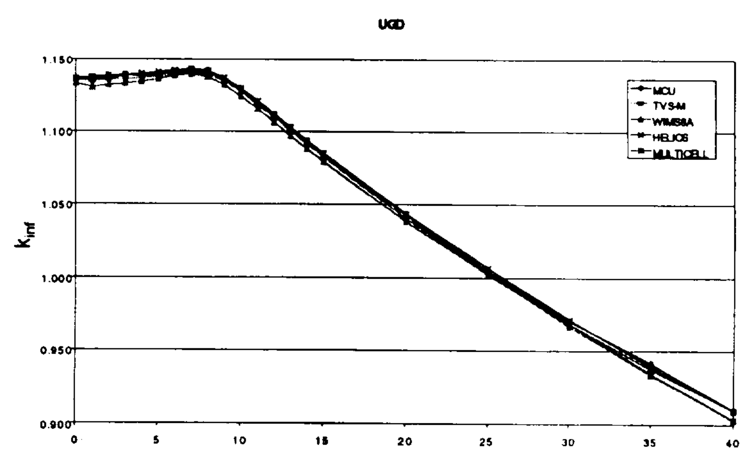
VVER-1000 Assembly Benchmark: UGD Multiplication Factor Comparison



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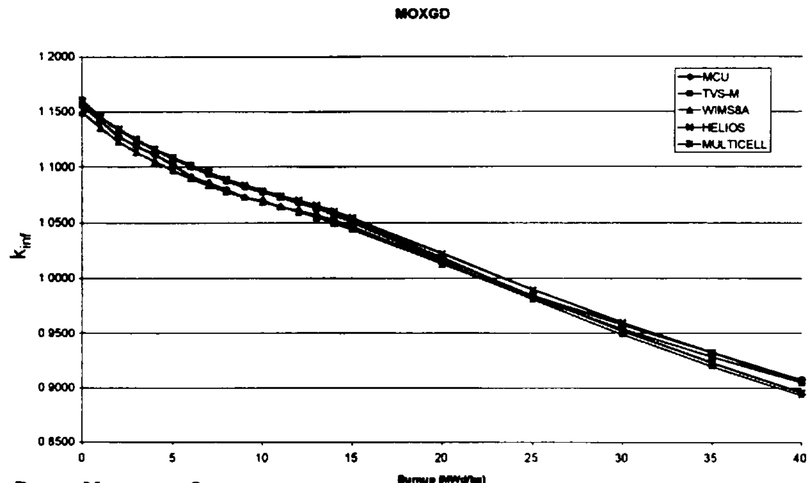
VVER-1000 Assembly Benchmark: MOXGD Multiplication Factor Comparison



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Future Benchmark Plans

- OECD/NEA is working hard to facilitate the release of more experimental data in both physics and fuel performance areas.
- Extension of VENUS-2 benchmark to 3-D.
- Benchmark regarding delayed neutron data and parameters for WG MOX.
- Benchmarks based on reactor startup data are being investigated.
- Possible release of recent experiments by SCK/CEN at VENUS facility with WG MOX.

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OECD/NEA Task Force on Reactor-Based Plutonium Disposition (TFRPD) Fuel Performance Benchmark Activities

L. J. Ott
Oak Ridge National Laboratory

Presented at

ORNL MOX Fuel Program
Research and Development Meeting
Oak Ridge, Tennessee
December 12, 2000

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Background

- The TFRPD was first proposed at the OECD/NEA's Workshop on the Physics and Fuel Performance of Reactor-Based Plutonium Disposition in Paris in September 1998.
  - 50 participants from 29 organizations and 16 countries
  - Strong consensus that this task force could provide a forum and vehicle for international collaboration in the areas of weapons-derived MOX fuel performance and physics
  - Specific recommendations:
    - The collection and publication of relevant materials and experimental databases
    - The execution of computational benchmarking and validation exercises
- The Bureau of the OECD/NEA Nuclear Science Committee established the TFRPD on December 15, 1998:
  - Meetings to be held in conjunction with the NEA Working Party on the Physics of Plutonium Recycling and Innovative Fuel Cycles (WPPR) meetings
  - Strongly encouraged appropriate participation of "all players in Russia"
  - Highest priority activities should be experimental benchmarks

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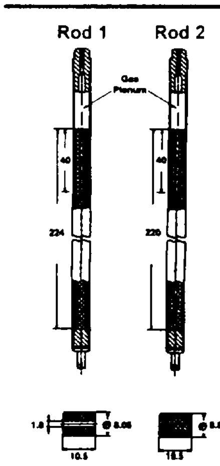
First MOX Fuel Performance Benchmark Exercise

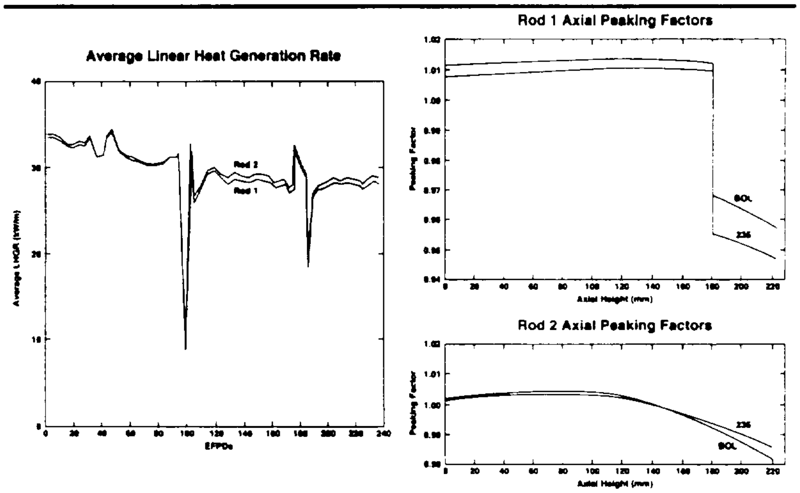
- During the second TFRPD meeting (November 1999), W. Wiesenack (Halden Reactor Project) offered the IFA-597 MOX experiment, covering solid and hollow pellets, for a benchmark comparison against experiment, if the U.S. NRC and the R.F. Kurchatov Institute agreed.
- ORNL received the preliminary benchmark specification for comment at the end of April 2000.
- Reviewer comments and preliminary analyses on the benchmark exercise were presented at the third meeting of the task force in June. IFA-597 was adopted as the first fuel performance benchmark.
- "Blind" calculations were performed; results will be sent to HRP by end of December 2000.
- HRP review and evaluation of exercise is to be presented at fourth meeting of task force (Jan. 31 – Feb. 2, 2001)

HRP IFA-597



- Two MOX fuel pins
  - Rod 1: solid pellets except for 4 top pellets
  - Rod 2: all annular pellets
  - 7.4% plutonium oxide
- Both rods instrumented with thermocouples (~40 mm below top)
- Rod pressure measured in both
- Rods irradiated for 235 EFPDs in Halden HBWR reactor
  - Approximate burnup of 14 GWd/MT
- Boiling assumed for the entire length of the fuel rod (per HRP)

HRP IFA-597 Irradiation History through 235 EFPDs



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ORNL 2000-1892C EFG

The IFA-597 Analyses at ORNL Employ the Locally Modified Version of FRAPCON-3 v1.3

- Corrects inconsistencies in code usage of the variable COMP (% plutonia in fuel)
  - Originally prompted PNNL notice that v1.3 should not be used to model MOX
- Includes MOX thermal conductivity correlations
  - Halden (HWR-589)
  - Duriez with Lucuta degradation factors (ORNL/TM-2000/351)
- Allows input of plutonium isotopics
  - Required by subcode that calculates the radial power distribution and burnup
- Uses current MASSIH fission gas release model (Vol. 1 of NUREG/CR-6534)
  - May require MOX data to adjust the temperature and burnup dependence of the diffusion constant and the resolution rate

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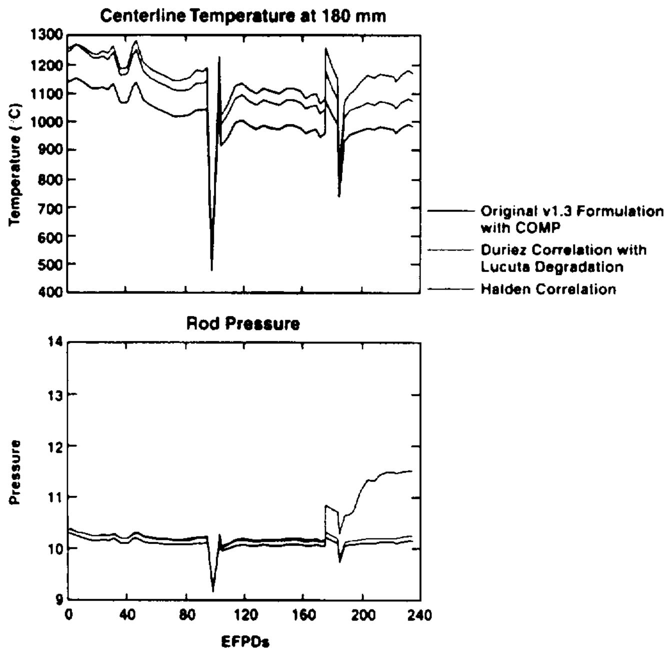
Comparisons of FRAPCON-3 Simulation Results for Rod 2



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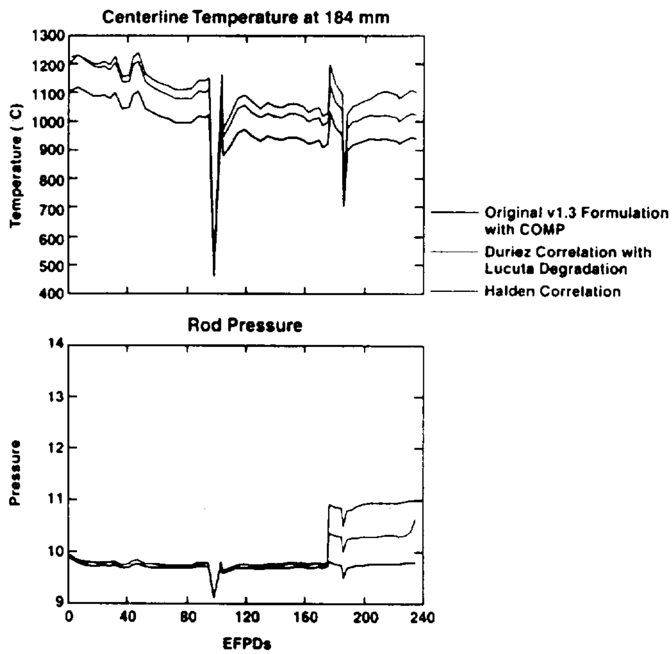
Comparisons of FRAPCON-3 Simulation Results for Rod 1



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Conclusions

- FRAPCON-3 v1.3 (ORNL version) simulations of IFA-597 have been completed; the results will be sent to HRP for presentation at the fourth meeting of the TFRPD on January 31, 2001
- FRAPCON-3, with the Halden MOX thermal conductivity, closely replicated the MOX fuel rod(s) thermal response
  - **Caution**: IFA-597 was included in the database from which the Halden correlation was developed
- ORNL modifications to FRAPCON-3 v1.3 will be forwarded to PNNL
  - ORNL studies will be presented at the FRAPCON-3 User's Group Meetings
- MOX thermal conductivity is 5–10% less than that of LEU
  - May require modification of the MASSIH fission gas release model parameters (via review of additional existing experimental MOX fuel pin response data)

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ORNL 2000-1898C EFG

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