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

December 7, 2000

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

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

SUBJECT: SUMMARY - OCTOBER 12, 2000, MEETING WITH FRAMATOME COGEMA FUELS, DUKE COGEMA STONE & WEBSTER AND DUKE POWER TO BRIEF THE STAFF ON THE FUEL QUALIFICATION PLAN FOR THE DEPARTMENT OF ENERGY MATERIALS DISPOSITION PROGRAM

On October 12, 2000, representatives of Framatome Cogema Fuels (FCF), Duke Cogema Stone & Webster (DCS), Duke Power and the Oak Ridge National Laboratory (ORNL), met with members of the U.S. Nuclear Regulatory Commission (NRC) staff at NRC Headquarters in Rockville, Maryland. FCF requested this meeting to brief the staff on the Fuel Qualification Plan for uranium-plutonium mixed oxide fuel (MOX). The most recent NRC meeting on this subject was held on June 2, 1999. A list of attendees is provided in Enclosure 1. The handouts provided during the meeting are included as Enclosure 2.

MOX Program Overview

DCS stated that the objectives of the meeting were to discuss the MOX Fuel Qualification Plan, submitted by letter to the NRC on July 14, 2000, to discuss the supporting technology, to begin a dialog between the DCS team and the NRC staff, to identify DCS plans for submittals to the NRC, and to present the associated requested review schedules. The MOX Project Team includes Duke Energy, Cogema, Inc. and Stone & Webster as partners with major activities subcontracted to Duke Power and FCF.

Fuel Qualification Approach and European Experience

The objective of the MOX Fuel Qualification plan is to demonstrate safe and reliable operation of the fuel design based on (a) a proven fuel assembly design, (b) European experience and technology, and (c) lead assembly irradiation. The component design for the MOX fuel assemblies, up to the fuel rod, will be similar to the Mark-BW fuel assembly design which is currently in use in several US nuclear power plants. The dimensions of the MOX fuel rods are planned to be the same as for current Mark-BW fuel assemblies. The extensive European experience in the design, fabrication and operation of MOX fuel was summarized by noting that since 1987, over 1250 MOX fuel assemblies have been used in 19 nuclear power plants in France. The French irradiation experience was said to include use of MOX fuel in a power plant load-following mode, which is more demanding of the fuel than steady state operation, and MOX fuel reliability was said to be as good as UO2 fuel reliability. The Lead Assembly program would include the irradiation of lead assemblies for two fuel cycles, which would achieve a burnup of 40 Giga-Watt days per Metric Ton of heavy metal (GWd/MThm). This would be achieved in time to allow post irradiation examination (PIE) 12 months prior to irradiation of the first MOX production batch. The NRC staff discussed the desirability of having the Lead Assembly program include more lead test assemblies.

DCS's presentation also included a discussion of the principal differences in the Plutonium feed material based on whether it comes from a power reactor fuel cycle, as does the European material (reactor grade, RG), or whether it comes from the plutonium disposition program (weapons grade, WG). The principal differences are the impurities and the different isotopic concentrations in WG versus RG plutonium. The most significant impurity in WG is gallium, which will be reduced to the parts per billion level in the fuel pellets that go into the fuel assemblies. A significant difference in the isotopic concentrations is that WG plutonium has a higher concentration of fissile material (Pu-239 and Pu-241) than RG plutonium. This results in a corresponding need for a lower concentration of WG plutonium in the fuel pellet end-product than would be the case if RG plutonium were used. This will result in the reactivity of the fuel being comparable whether WG or RG plutonium is used.

Framatome also presented information on MOX fuel behavior up to high burnup levels (50 - 60 GWd/MThm). This range goes beyond the fuel assembly burnup levels said to be planned for the plutonium disposition program of 40 - 45 MWd/MThm. With regard to experience with RG MOX fuel, Framatone noted that (a) they have acquired a very large data base on fuel performance with good behavior demonstrated up to high burnup levels, (b) their fuel rod design code has the same prediction quality as the codes for UO2 fuel, and (c) they are continuing with a significant research and development effort in order to increase their fuel performance at very high burnup levels.

Fuel Assembly Design

DCS presented information on the fuel assembly design noting that there would be 463 kg of heavy metal per fuel assembly. DCS also described two fuel assembly enrichments (4.07 and 4.37 weight percent) with three fuel rod enrichments per assembly, ranging from 2.316% to 4.794 %.

Physics and Fuel Fabrication Process

Duke Power's presentation of the physics aspects of MOX fuel stated that at least 60% of the assemblies in all mixed cores will be standard uranium dioxide fuel assemblies. Information was also presented on plutonium mass and fissile plutonium versus burnup for standard UO2 fuel and RG and WG MOX fuel. The core physics analytical methodology was said to utilize codes which have been used to support more than 75 power plants worldwide. Revised methodology reports to reflect MOX are planned to be submitted in August 2001 for NRC staff review.

Framatome presented flow diagrams outlining the fuel fabrication process from the blending, using the MIMAS (Micronization Master blend) process, of the UO2 and PuO2 powders to the finished fuel pellets.

Lead Assembly Program

DCS indicated that irradiation of the lead assemblies is scheduled for October 2002 to March 2005; completion of fuel qualification and irradiation of the first production batch is scheduled for October 2007. DCS and the Department of Energy are currently evaluating the fabrication of the lead assemblies either in Europe or at the MOX Fuel Fabrication Facility with a projected decision date by January 1, 2001.

Licensing and Scheduling

Duke Power presented an overall licensing schedule showing projected review activities in the year 2001 on one license amendment application, three FCF methodology topical reports, and two Duke methodology topical reports. It was noted that further meetings will likely need to be scheduled to discuss the specific phases of the project as listed in the schedule.

Comments by Members of the Public

Following the meeting between the NRC staff and DCS, a representative of the Nuclear Control Institute offered comments on several points. The reactivity insertion accident (RIA) implications of the test data from the Cabri facility should be considered. European burnup experience with MOX fuels is not extensive beyond about 35 GWd/MT. European reactors are limited to about one third of a core being MOX assemblies whereas this proposal could involve up to 40% of a core loading with MOX assemblies. The NRC staff has been aware of these aspects and plans to consider them in its review.

Robert Martin
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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RidsNrrDssaSpsb (RBarrett)
RidsNrrDDeEmcb (JDavis)
SRosenberg (e-mail)
RidsNmssFcssFSpb (TJohnson, APersinko)
RidsRgn2MailCenter

PLUTONIUM DISPOSITION PROGRAM STATUS MEETING

Bob Martin	NRR/DLPM
Ralph Caruso	NRR/SRXB
Rich Emch	NRR/DLPM
Richard Correia	NRR/DLPM
Frank Rinaldi	NRR/DLPM
Laurence Losh	FCF
Geroge Meyer	FCF
Patrick Blanpain	Framatome
Skip Copp	Duke power
Anne Cottingham	Winston & Strawn
Frank McPhatter	FCF
Sidney Crawford	Consultant (self)
Muffet Chatterton	NRR/SRXB
Herbert Berkow	NRR/DLPM
Steven Nesbit	Duke Power
Jim Eller	Duke Power
Stephen Fisher	ORNL
Don Williams	ORNL
Don Spellman	ORNL
John Goshen	NRR/DLPM
Andrew Persinko	NRR/NMSS
Tim Johnson	NRC/NMSS
Chandu Patel	NRR/DLPM
S. Basu	NRC/RES
Ralph Meyer	NRC/RES
Phil Kasik	DOE
Patrick Rhoads	DOE
John Thompson	DOE
Edwin Lyman	NCI

AGENDA

Nuclear Regulatory Commission meeting with Duke Cogema Stone & Webster on DOE Materials Disposition Program

Fuel Qualification Plan

9:00 - 9:10 AM	MOX Program Overview *GA Meyer, DCS Fuel Qualification Manager*
9:10 - 9:20 AM	Fuel Qualification Approach *LL Losh, FCF Fuel Design and Licensing*
9:20 - 10:20 AM	European Experience *Patrick Blanpain, Framatome Fuel Division*
10:20-10:30AM	Break
10:30-10:45 AM	Fuel Assembly Design *LL Losh*
10:45-11:15 AM	Physics Aspects of Fuel Qualification *JL Eller, Duke Power*
11:15-11:30 AM	Fuel Fabrication Process *Patrick Blanpain*
11:30-11:45 AM	Lead Assembly Program *LL Losh*
11:45-12:00 noon	Schedule and NRC Actions Requested *GA. Copp, Duke Power*

Enclosure 2

DUKE COGEMA
STONE & WEBSTER

MOX Fuel Qualification

NRC/DCS Meeting
October 12, 2000

DUKE COGEMA
STONE & WEBSTER

Meeting Objectives

- Present the MOX Fuel Qualification Plan
- Review the technology supporting the plan
- Begin a dialog between the DCS Fuel Qualification team and the NRC staff
- Identify DCS plans for submittals to the NRC and requested review schdeules

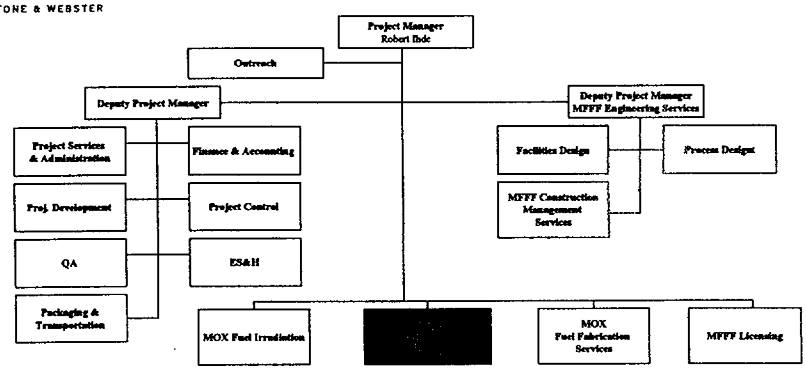
DUKE COGEMA
STONE & WEBSTER

MOX Project Team

- Duke COGEMA Stone & Webster (DCS)
  - Partners
    - Duke Energy
    - COGEMA, Inc
    - Stone & Webster
  - Major subcontractors
    - Duke Power
    - FCF
    - NFS

DUKE COGEMA
STONE & WEBSTER

DCS Organization



MOX Fuel Project Organization Structure

MOX Fuel Qualification

DUKE COGEMA
STONE & WEBSTER

- Objective - Demonstrate safe and reliable operation of the fuel design to be used for disposition of weapons grade plutonium
- Bases
  - Proven fuel design
  - European experience and technology
  - Lead assembly irradiation

FCF's Role in MOX Fuel Qualification

DUKE COGEMA
STONE & WEBSTER

- Lead Fuel Qualification and Design
  - Fuel Qualification Plan
  - Fuel Design and Licensing
  - Lead fuel fabrication team for lead assemblies
    - Cogema MOX fuel fabrication
    - FCF fuel assembly fabrication
    - FCF QA
  - Certify completion of qualification

DUKE COGEMA
STONE & WEBSTER

Fuel Qualification Plan

NRC/DCS Meeting
October 12, 2000

DUKE COGEMA
STONE & WEBSTER

Fuel Qualification Plan

- Document submitted to DOE and NRC
- Provides overall approach to fuel qualification
- Identifies technical approach to design of MOX fuel for the MD Program
- Provides schedules for NRC submittals
- Lists steps to be taken to complete qualification of MOX fuel for batch implementation

DUKE COGEMA
STONE & WEBSTER

Fuel Qualification Plan

- Terminology
  - Fuel Qualification Plan - The governing document that presents the strategy and process for qualifying the MOX fuel
  - Strategy - The overall approach to fuel qualification
  - Process - The steps to be taken to qualify MOX fuel

DUKE COGEMA
STONE & WEBSTER

Fuel Qualification Plan

Overall Approach to Fuel Qualification

- Extensive European experience
  - Design, Fabrication, Operation
  - Qualification Programs
- Proven Fuel Assembly design
  - Change only where required for MOX
- Ensure validity of European database for weapons-grade plutonium
  - Impurities
  - Isotopics
- Confirm performance with Lead Assemblies
  - Irradiate LAs for 2 cycles, to > 40 GWd/MThm, prior to batch implementation
  - Complete 2 cycles irradiation and poolside PIE in 2006, 12 months prior to irradiation of first MOX production batch

DUKE COGEMA
STONE & WEBSTER

Fuel Qualification Plan

Overall Approach to Fuel Qualification

- Weapons-Grade Plutonium
  - Impurities - most notably Gallium
- Pellet Specification
  - Requirement added to European MOX specification
  - Incoming PuO_2 powder limited to 100 ppb gallium
  - Resulting pellet content < 5 ppb gallium

Fuel Qualification Plan

Overall Approach to Fuel Qualification

- Weapons-Grade Plutonium
  - Isotopics
    - Reactor-Grade
      - Pu-239 - 59%
      - Pu-241 - 11%
    - Weapons-Grade
      - Pu-239 - 94%
      - Pu-241 - 0.5%
- MIMIAS pellet fabrication process
  - Adjustment of Master Mix (UO_2/PuO_2) from 70/30 to 80/20
  - Maintains the same fissile content in plutonium rich particles
  - Maintains the same pellet microstructure

Fuel Qualification Plan European Experience (Oct. 12, 2000)

- MOX Fuel Irradiation Experience
- MOX Fuel Performance: Analytical and In-Reactor Data
- COPERNIC: MOX Physical Properties, Models and Validation

Patrick Blanpain

FRENCH IRRADIATION EXPERIENCE (1)

- Since 1987, more than 1250 MOX fuel assemblies delivered in nineteen 900 Mwe EDF units.

- Recycle rate : 30% MOX assemblies in core.

- "Hybrid" fuel management scheme :

  ➢ UO_2 assemblies (3.7% U5) irradiated 4 cycles (annual)
  ➢ MOX assemblies (equ. 3.25% U5) irradiated 3 cycles (annual)

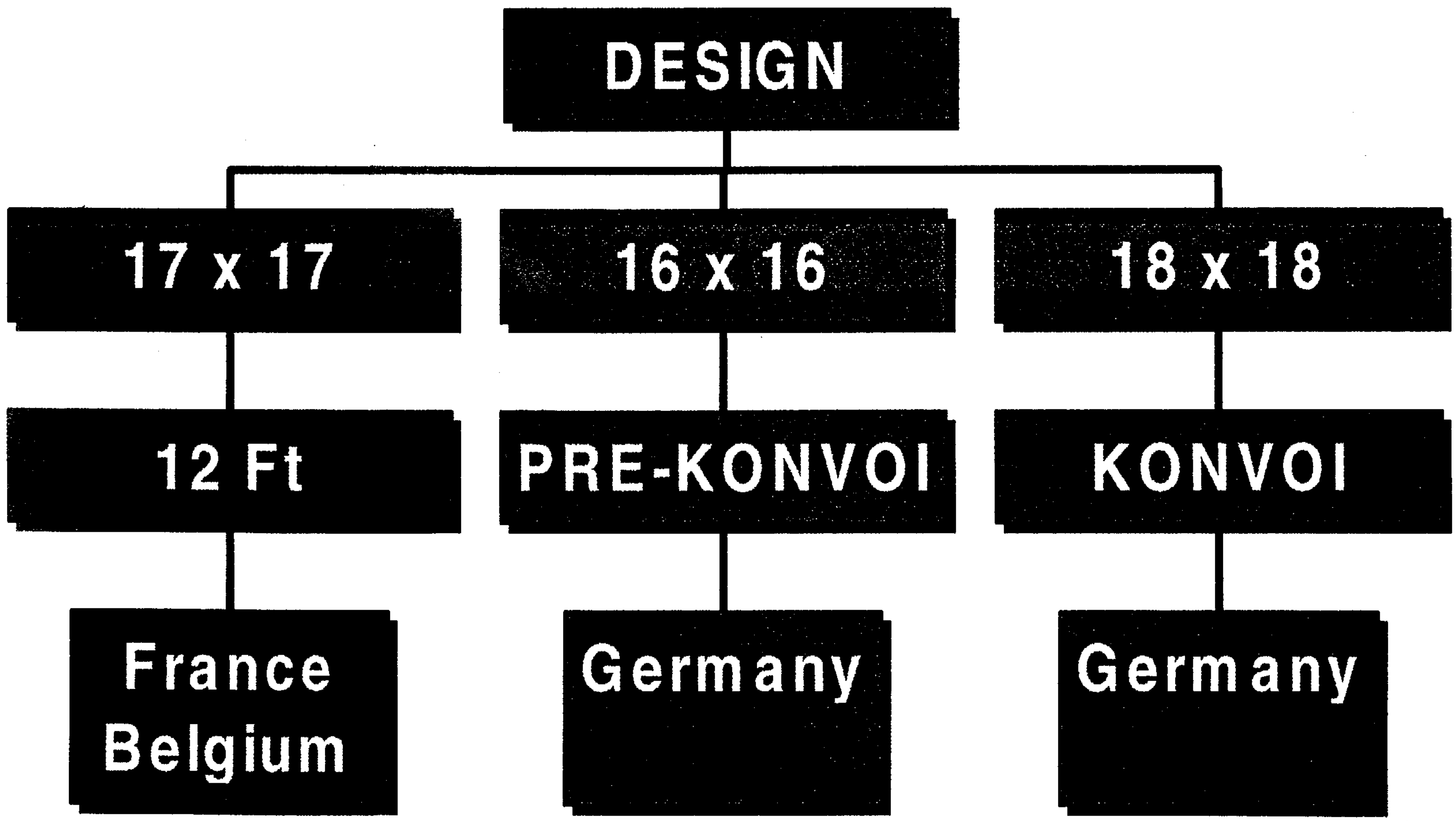
  - Pu/U + Pu up to 7.08% average assembly depending on the plutonium isotopic composition.

  - Average ass. Burnup : 43 GWd/tHM, Max : 47.6 GWd/tHM

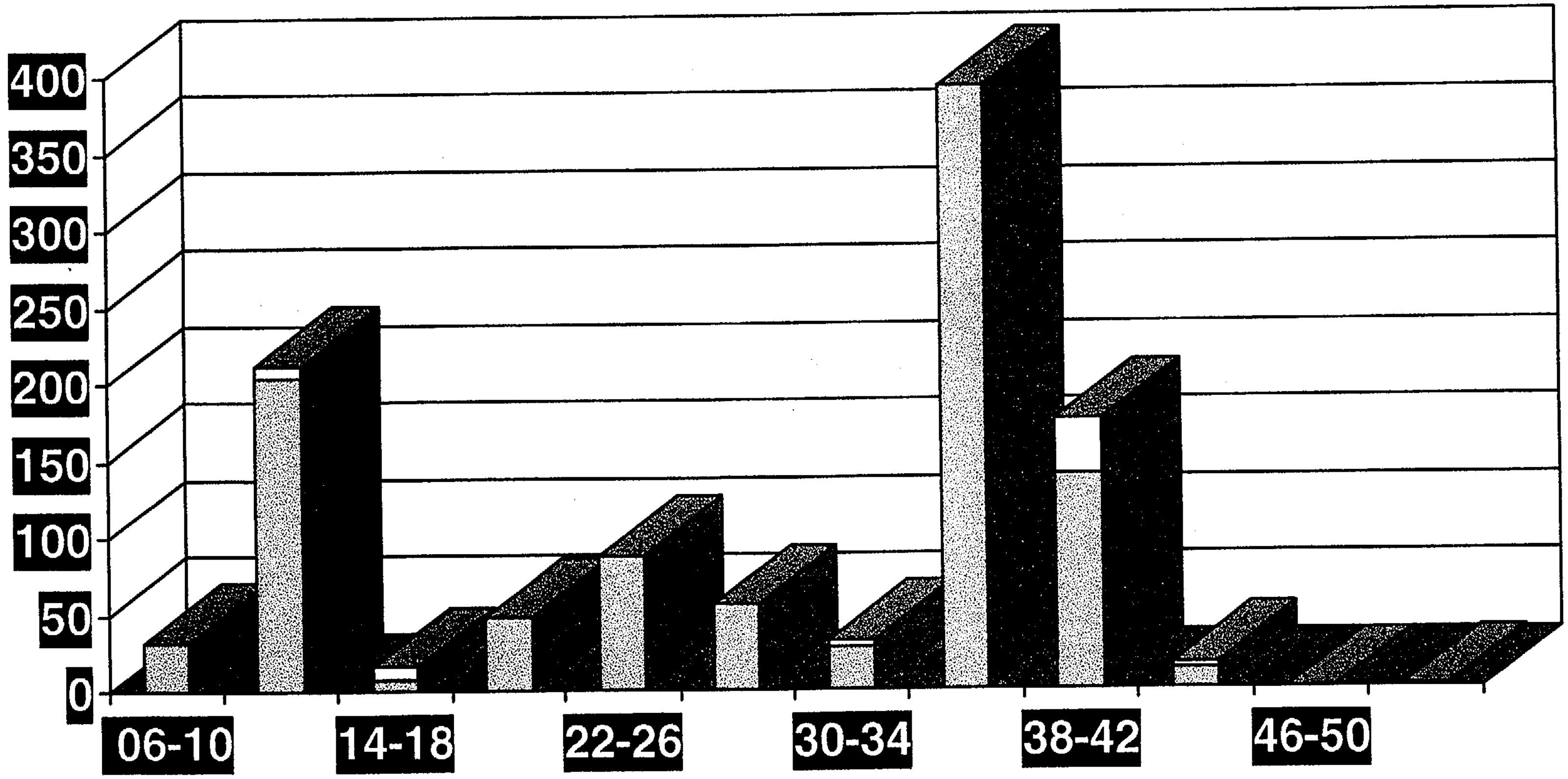
FRENCH IRRADIATION EXPERIENCE (2)

- At the end of 1999:
  - 560 assemblies have achieved 3 irradiation cycles
  - 6 assemblies have achieved 4 irradiation cycles (rod burnup of 53 GWd/tHM)
  - 2 assemblies have achieved 5 irradiation cycles (rod burnup of 61 GWd/tHM)
- MOX fuel can operate in load follow (daily and extended low power operation)
- MOX fuel reliability as good as UO_2 fuel

MOX FUEL ASSEMBLIES



FRAGEMA MOX EXPERIENCE end 99



MOX FUEL BEHAVIOR AT HIGH BURNUP (1)

- Experience feed-back (surveillance and analytical programs):
  - 60 commercial fuel rods examined in hot cells (BU up to 54 GWd/tHM - 4 cycles)
  - two assemlbies have completed a fifth irradiation cycle (BU =61 GWd/tHM), PIE underway
  - Power ramp testing and instrumented analytical irradiations have been or are being carried out (national & international programs)

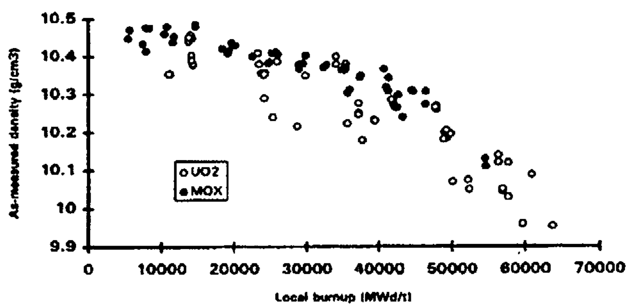
MOX FUEL BEHAVIOR AT HIGH BURNUP (2)

- Same behavior of MOX and UO_2 fuels concerning :
  - Fuel rod growth
  - Cladding diametral deformation
  - Cladding waterside corrosion
  - Pellet solid swelling
  - ZrO_2 internal layer

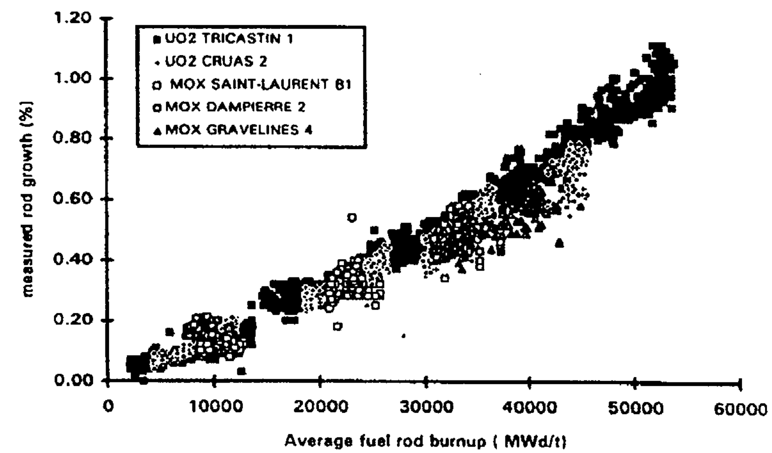
- Higher fission gas release than UO_2 fuel at equivalent burnups (higher heat rate during the last irradiation cycles)

- Better PCI behavior due to higher creep properties

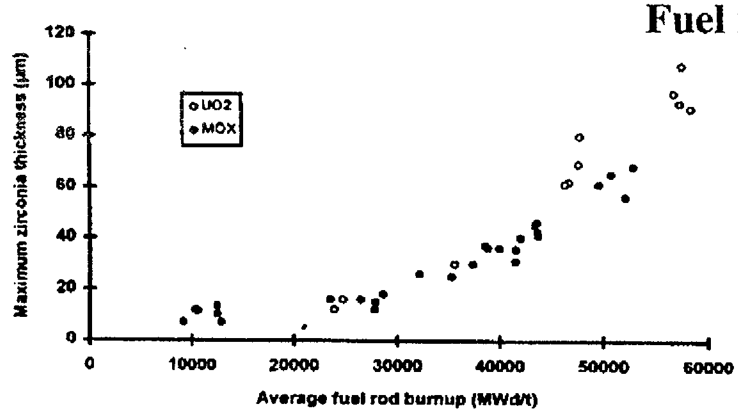
FUEL ROD GROWTH, CLADDING WATERSIDE CORROSION AND FUEL DENSITY AS FUNCTION OF BURNUP FOR MOX AND UO_2



Fuel density as function of burnup



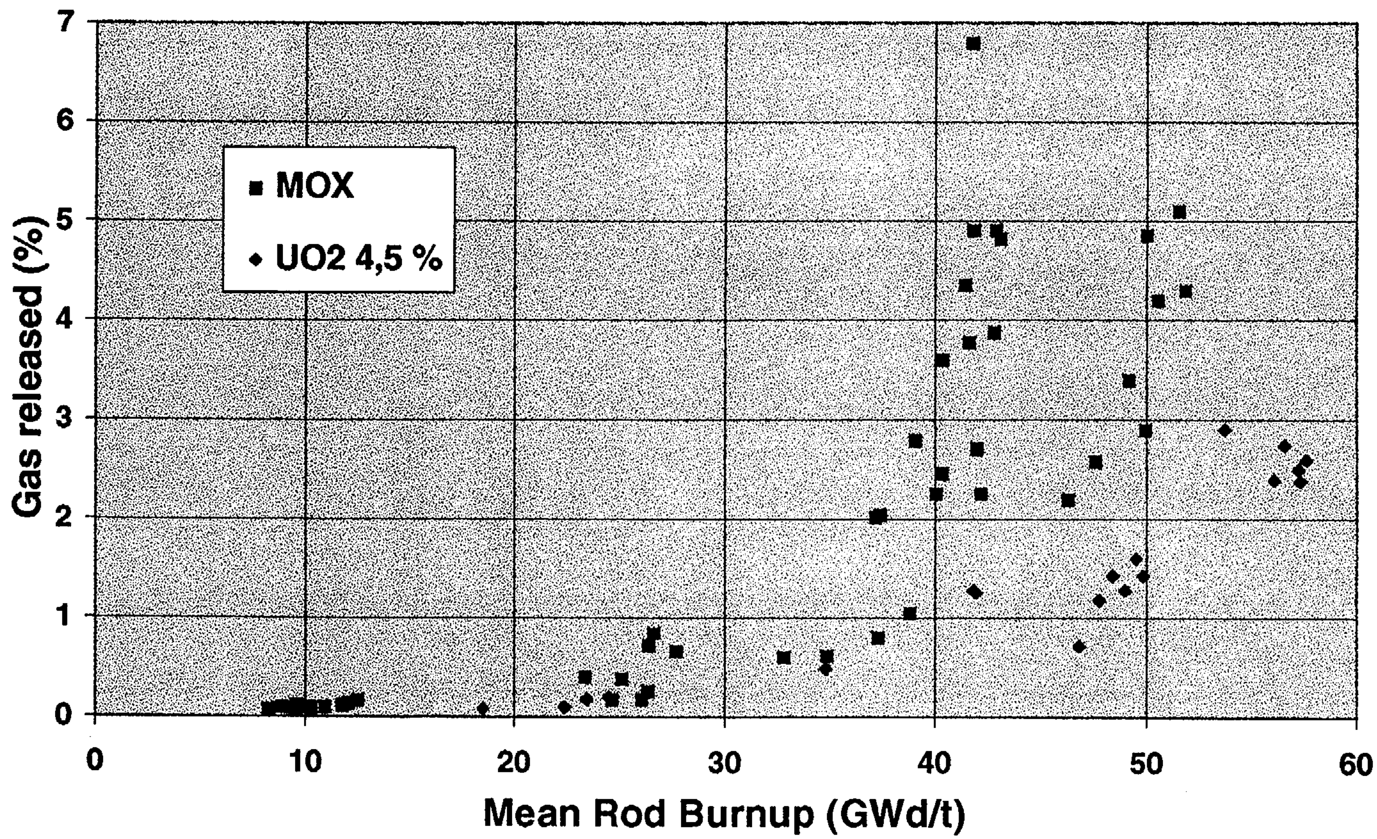
Fuel rod growth as function of burnup



Cladding waterside corrosion as function of burnup

FRAMATOME
NUCLEAR FUEL

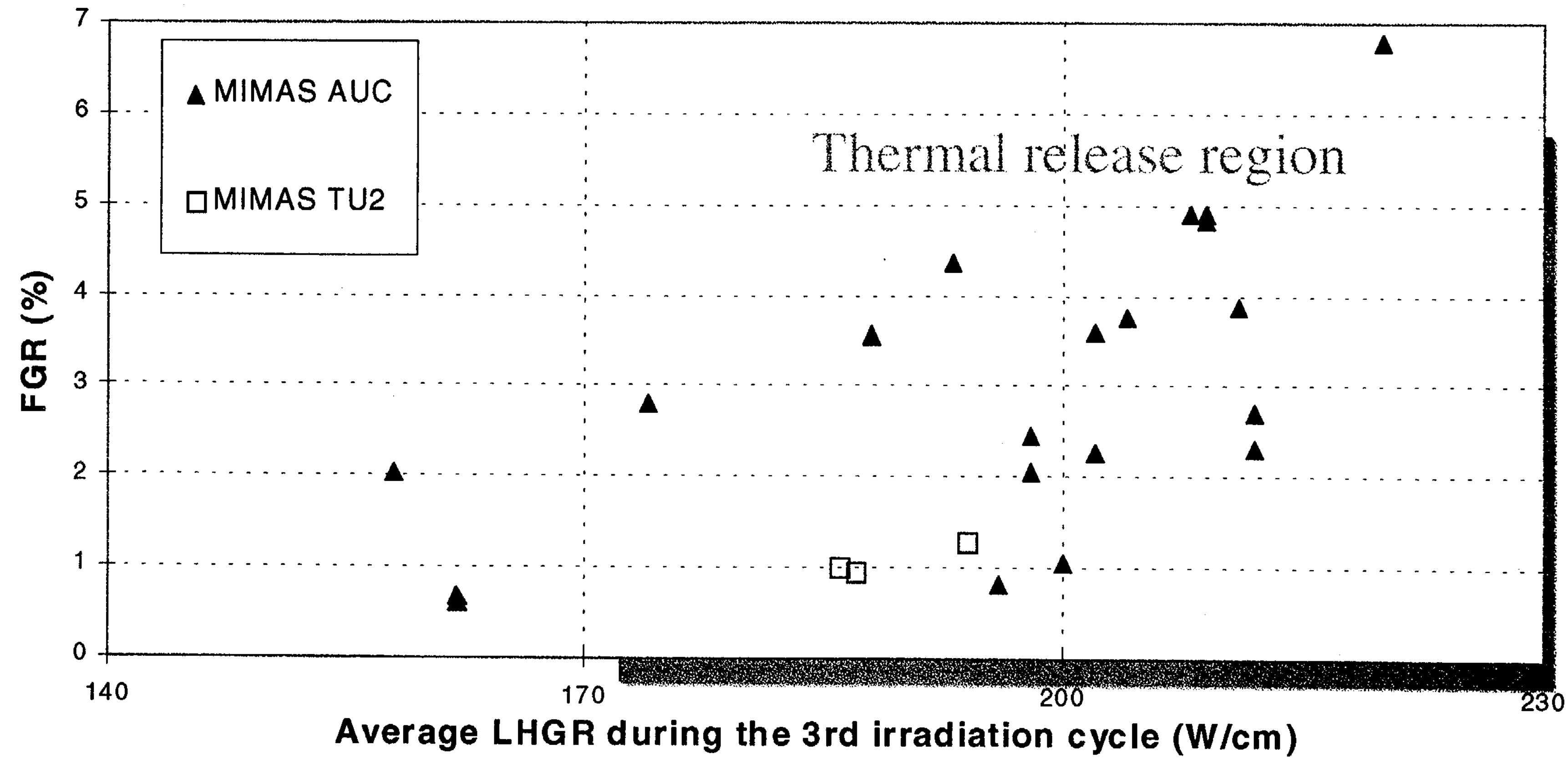
GAS RELEASE IN PWR MOX RODS AS COMPARED TO UO_2 RODS



MOX FUEL BEHAVIOR AT HIGH BURNUP (3)

- Fission Gas Release Behavior:
  - Neutronic properties: higher linear power density during the second and third cycles
  - Physical property: slightly lower thermal conductivity (centerline temperature +50°C at 200 W/cm)
  - Oxide microstructure: the presence of Pu rich particles (max 30% $PuO2$) due to MIMAS process has a small influence on the mechanism of FGR. Local high burnup zones lead to the formation of dense pore populations.
  - The current fuel rod design accomodates this higher FGR by an increase of the plenum volume and a lower Helium initial pressure (Fragema design in France and Belgium)

FISSION GAS RELEASE OF 3-CYCLE MOX FUEL RODS



ON GOING ANALYTICAL PROGRAM

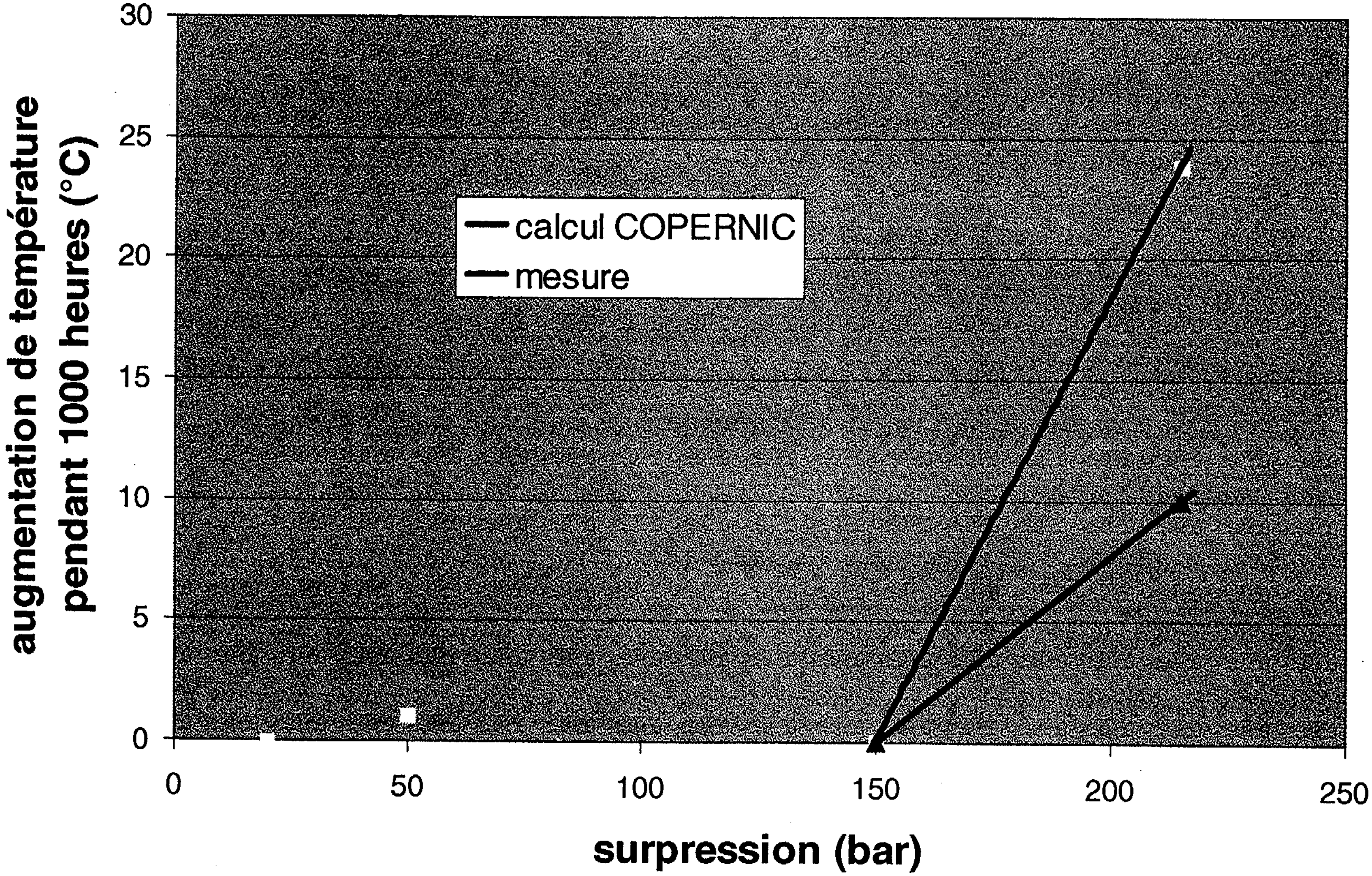
- Focused on the role of the microstructure on the Fission Product distribution, migration and release; in pile densification, mechanical properties
- Better understanding of the very high burnup effects
- In-pile and hot cell experiments: national and international programs (Halden, BN,....)
- To improve modelling

HIGH BURNUP ANALYTICAL EXPERIMENT
IFA 610.2 (HRP)

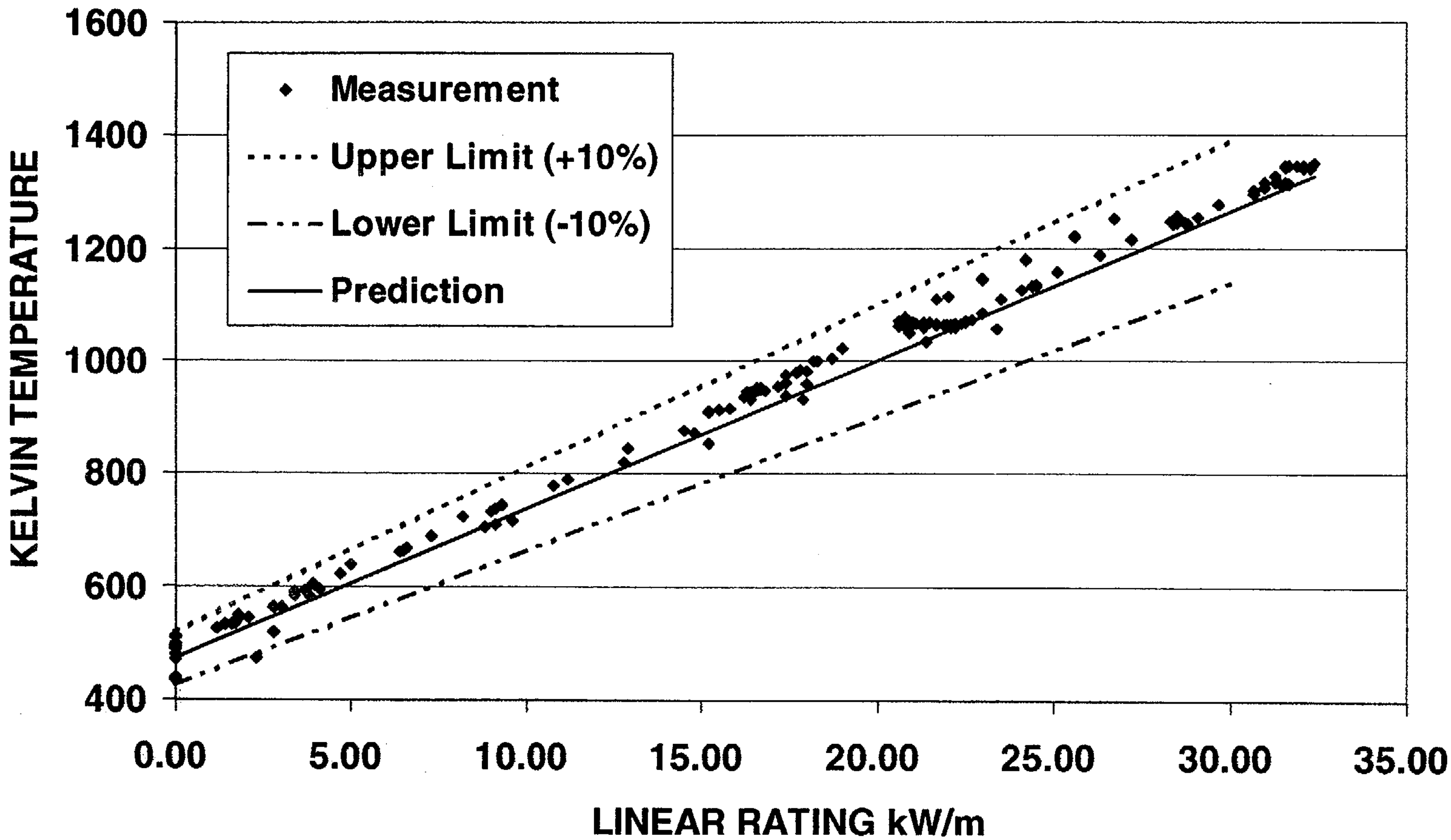
- 4 cycles MOX rodlet (55 GWd/t).
- Rod overpressure/cladding lift-off test.
- Instrumented with a thermocouple, a clad extensometer and a gas line at each end for internal pressurisation.
- Pressurized with argon, increasing overpressure levels

➡ slight temperature increase at 215 bar overpressure.

IFA-610.2 : LIFT-OFF THRESHOLD



FIGARO experiment (MOX)

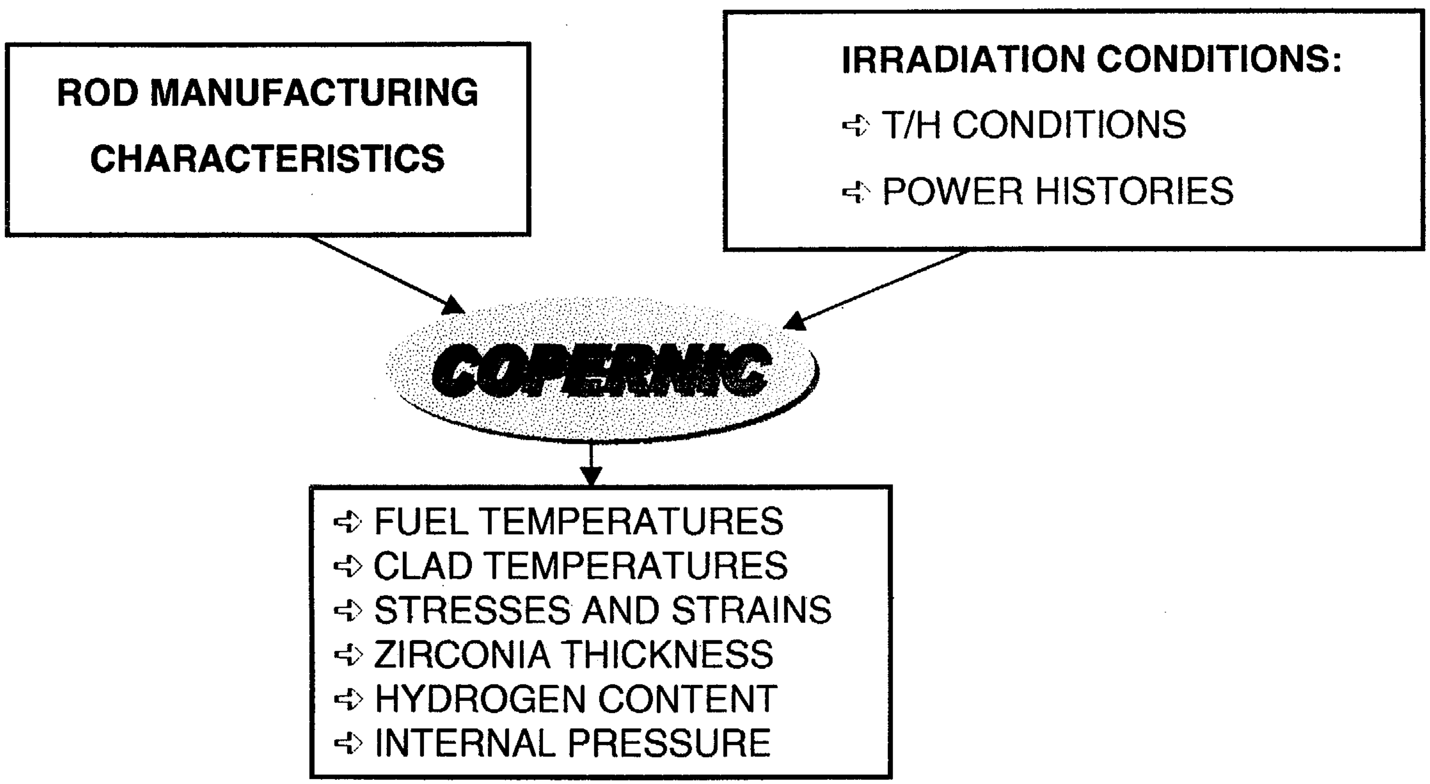


FRAMATOME NUCLEAR FUEL

COPERNIC: FUEL ROD DESIGN COMPUTER CODE

. SIMULATES BEHAVIOR OF FUEL ROD DURING IRRADIATION

. EVALUATES FUEL ROD THERMAL MECHANICAL PERFORMANCES



COPERNIC: VALIDITY RANGE

FUELS:
- UO_2, MOX, UO_2-Gd_2O_3

CLADDINGS:
- ZIRCALOY-4 (STRESS-RELIEVED AND RECRYSTALLIZED)
- ALLOY 5 ADVANCED CLADDING (M5)
- STRESS-RELIEVED DUPLEX 2 (D2)

ENRICHMENTS AND CONTENTS IN WEIGHT PER CENT:
- UO_2 : UP TO 9 % U_{235}
- MOX: UP TO 11 % PU
- UO_2 -GD_2O_3: UP TO 10 % GD_2O_3

INITIAL DENSITY:
- GREATER THAN 92.5 % TD

MAXIMUM ROD POWER:
- UP TO 80 kW/m

ROD AVERAGE BURNUP:
- UO_2: 0-67 GWd/tMM
- MOX: 0-53 GWd/tMM
- UO_2-Gd_2O_3: 0-55 GWd/tMM

COPERNIC : EXPERIMENTAL DATABASE

<table>
<tr><th>THERMAL</th><th>FGR</th><th>MECHANICAL</th></tr>
<tr><td>HALDEN Project
IFA 562.16→102 GWd/tM
IFA 515.2→64 GWd/tM

- Thermal model</td><td>HBEP
NFIR
HBC Program
TRANSRAMP IV program

- Steady-state FGR</td><td>Power ramps (OSIRIS, STUDSVIK)
TRANSRAMP IV
RECOR

- Relocation</td></tr>
<tr><td>EXTRAFORT (62 GWd/t)

- Thermal model</td><td rowspan="4">GONCOR
HATAC
REGATE
HBEP
TRIBULATIONS
Power ramps
(OVER-RAMP, STUDSVIK, OSIRIS)

- Transient FGR</td><td>FRAMBOISE

- Densification</td></tr>
<tr><td rowspan="3">HBC ramps melt
CONTACT 2B
BOSS 09
US/NRC Database
GRIMOX 01-02
GDGRIF 1-2

- Thermal model</td><td>NFIR
GONCOR
Power ramps (OSIRIS, STUDSVIK)

- Gaseous swelling</td></tr>
<tr><td>ZS campaigns
CEA-SRMA tests

- Low-stress creep</td></tr>
<tr><td>High-stress creep / relaxation tests
Power ramps (OSIRIS, STUDSVIK)
TRANSRAMP IV

- High-stress creep</td></tr>
<tr><td colspan="3">PWR/BR3/CAP/ZORITA/PRIMO/GAIN rods</td></tr>
</table>

New experiments are integrated : FIGARO, IFA610.2, BR3, ...

THERMAL MODEL: Qualification base

. UO_2 FUEL

- 5 FRA RODS + 14 US/NRC RODS
- POWER TO MELT EXPERIMENT (HBC4)
- HALDEN PROJECT (2 IFA 562.2 RODS)
- EXTRAFORT (62000 MWd/tU)

- **BURNUP: 102GWd/tM**
- DIAMETRAL GAP: 381 µm
- GAS COMPOSITION: He, He-Xe, Xe (1-32 BAR)
- MAX.LHGR: 80 kW/m

. MOX FUEL

- GRIMOX 1: 700 MWd/tM
- GRIMOX 2: 4500 MWd/tM

. GADOLINIUM OXIDE FUEL

- GDGRIF 1: 2000 MWd/tM
- GDGRIF 2: 7000 MWd/tM
- Power to melt (HBC5): 16700 MWd/tM
- **IFA 515.10-2: 64 GWd/tM**

HIGH-BURNUP FUEL TEMPERATURE

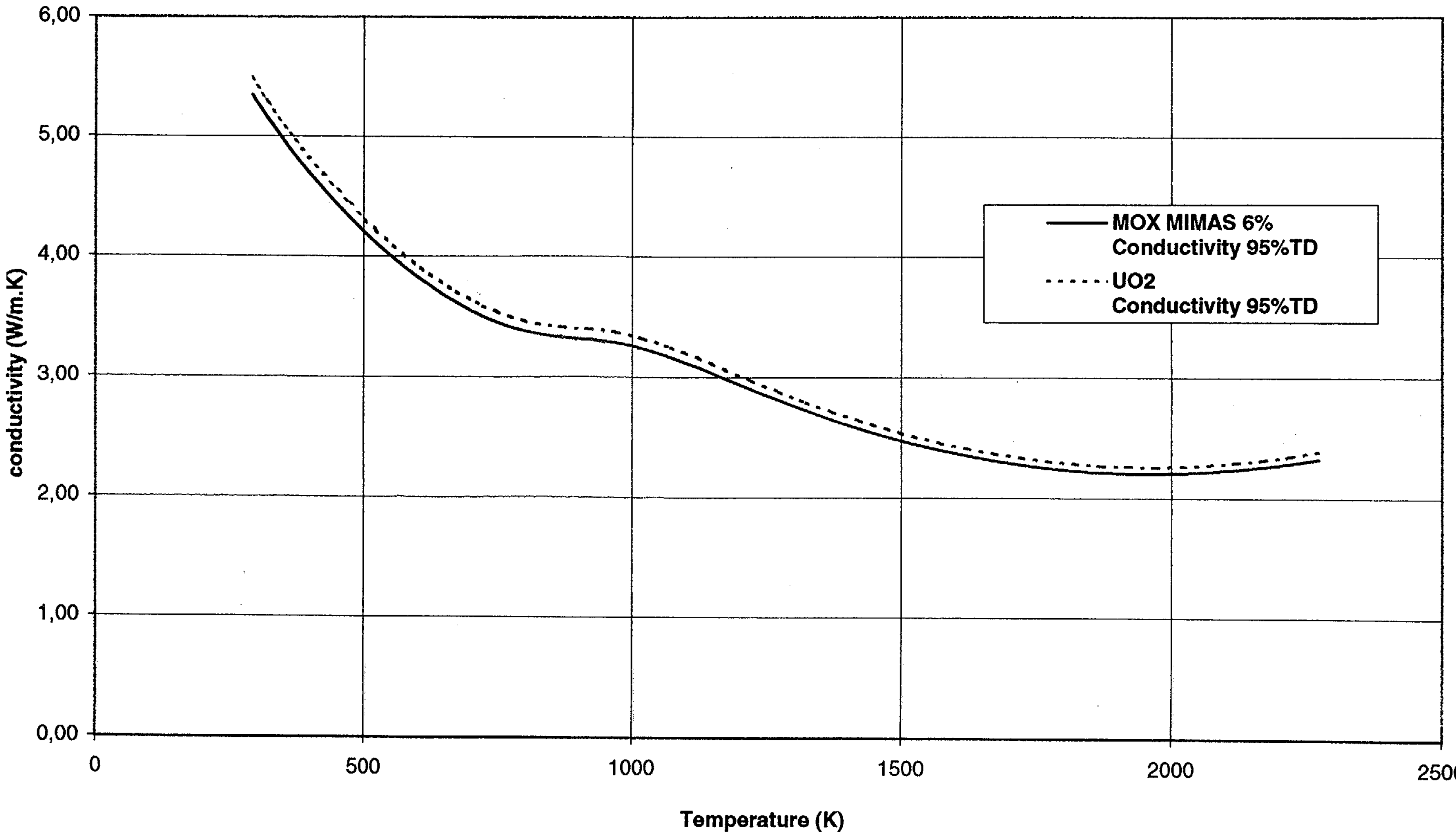
- ❑ Thermal model benchmark
  - > 2000 centerline temperature data
- ❑ Recent fuel centerline measurements
  - EXTRAFORT experiment: French program up to 62 GWd/tM
  - HALDEN experiment IFA 562.2-16: > 100 GWd/tM
  - FIGARO experiment: MOX, ~47 GWd/tM
  - HALDEN IFA 515.10-2: UO_2-Gd_2O_3, > 60 GWd/tM
- ❑ Updated thermal model in COPERNIC
  - Fuel thermal conductivity
  - Thermal gap
  - Radial Power Profile and RIM specific to HALDEN

THERMAL MODEL: MIXED OXIDES

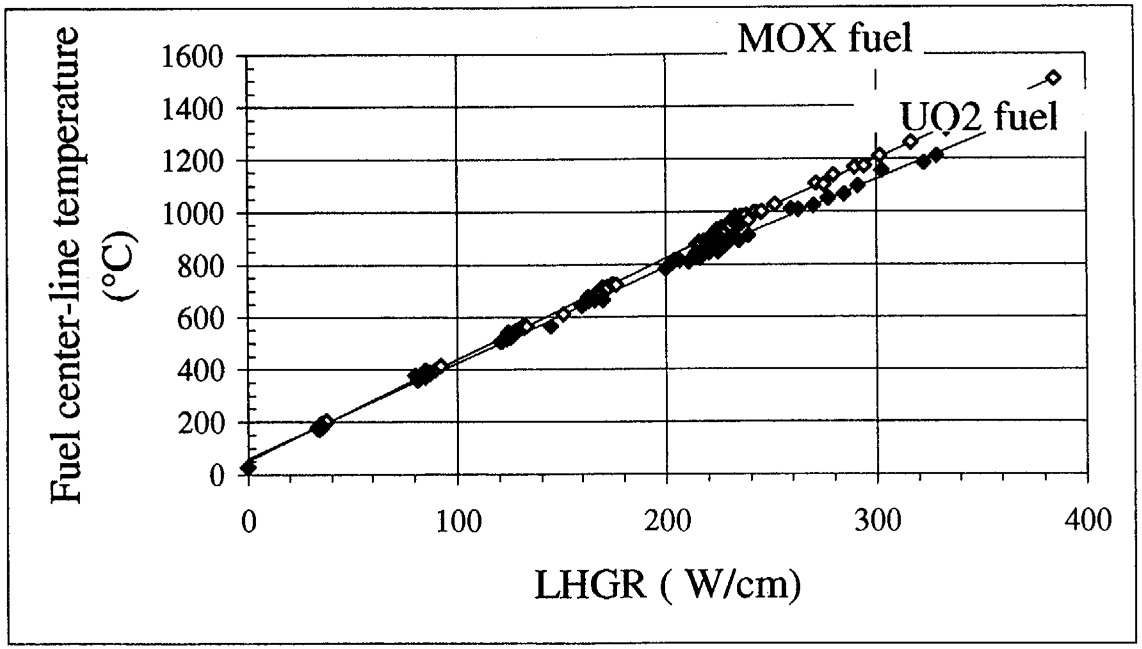
- **Experimental database:**
- MOX:
  - GRIMOX (low burnup, 23 kW/m, 7% Pu)
  - IFA 610.2 (55 GWd/tM)
  - FIGARO (47 GWd/tM, 33 kW/m, 6% Pu)
- UO_2-Gd_2O_3
  - GDGRIF (low burnup, 31 kW/m, 8% Gd_2O_3)
  - IFA 515.10-2 (64 GWd/tM, 8% Gd_2O_3)

- **Models**
- $\lambda_{MOX}/\lambda_{UO2} = F(Pu)$
- $\lambda_{Gd2O3}/\lambda_{UO2} = G(T, Gd_2O_3)$

Loi COPERNIC V2.4 - BU = 0

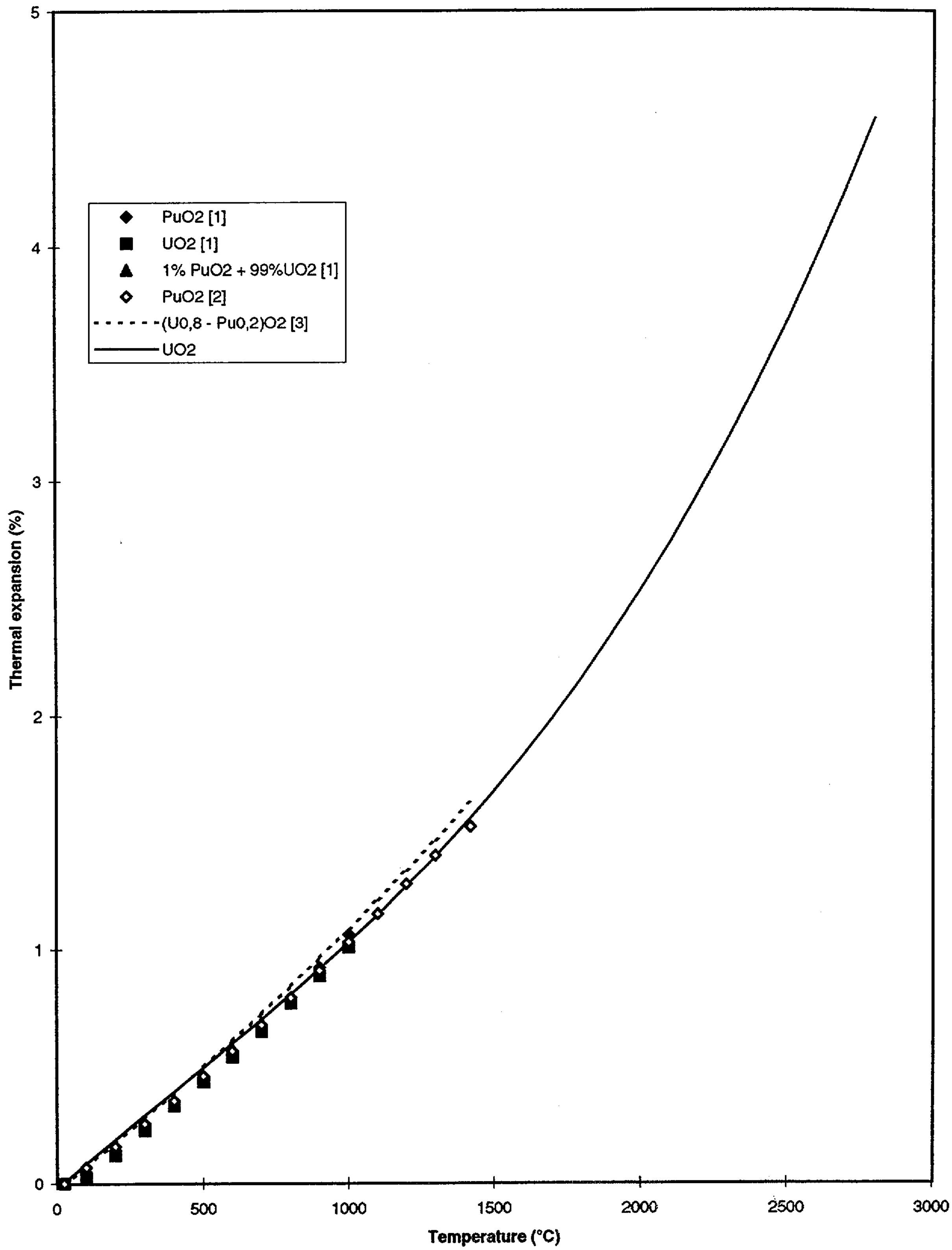


GRIMOX 2 EXPERIMENTAL PROGRAMME



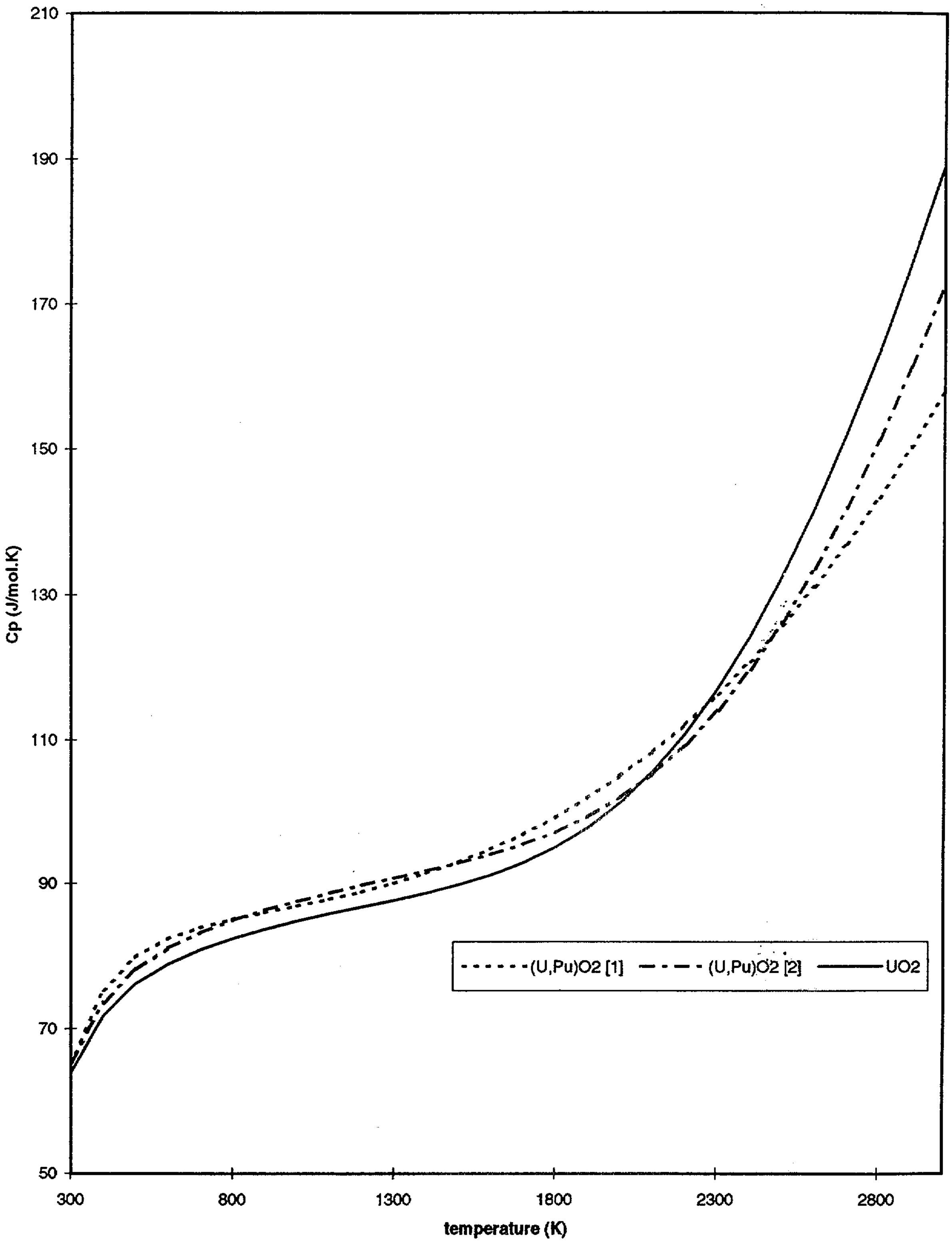
Relationship between center-line temperature and power for MOX and UO_2 fuels

Thermal expansion of UO_2, PuO_2 and $(U,Pu)O_2$



[1] Brett N.H., and Russel L.E., "The thermal expansion of PuO2 and some others actinide oxides between room temperature and 1000°C", in "Plutonium 1960", E.Grison & al Edts, Proc. 2nd Inter.conf. Plutonium metallurgy, Grenoble France,19-22 April 1960, p 397-410.
[2] Tokar M. et al., "Linear thermal expansion of plutonium dioxide", Nuclear technology, Vol. 17, p 147-152 Feb. 1973
[3] Lorenzelli R. et al., "Dilatation tehrmique d'oxydes mixtes (U,Pu)O2-x en fonction de l'écart à la stoechiométrie" JNM 68

Specific heat comparison UO_2 - $(U,PuO)_2$



[1] Baker R.D. "Quaterly Progres Report on the advanced Fuels Program - April 1 to June 30,1972", LA-5067-PR, October 1972, p. 31 and 33.
[2] Gibby R.L. et al., "Analytical expressions for enthalpy and heat capacity for uranium - plutonium oxide", HEDL-TME- 73-60, June 1973

THERMAL MODEL: Conclusions

. STATE-OF-THE-ART MODELS

- LOCAL EFFECTS OF BURNUP
- STEADY-STATE & TRANSIENT REGIMES

. SUB-MODELS QUALIFIED SEPARATELY

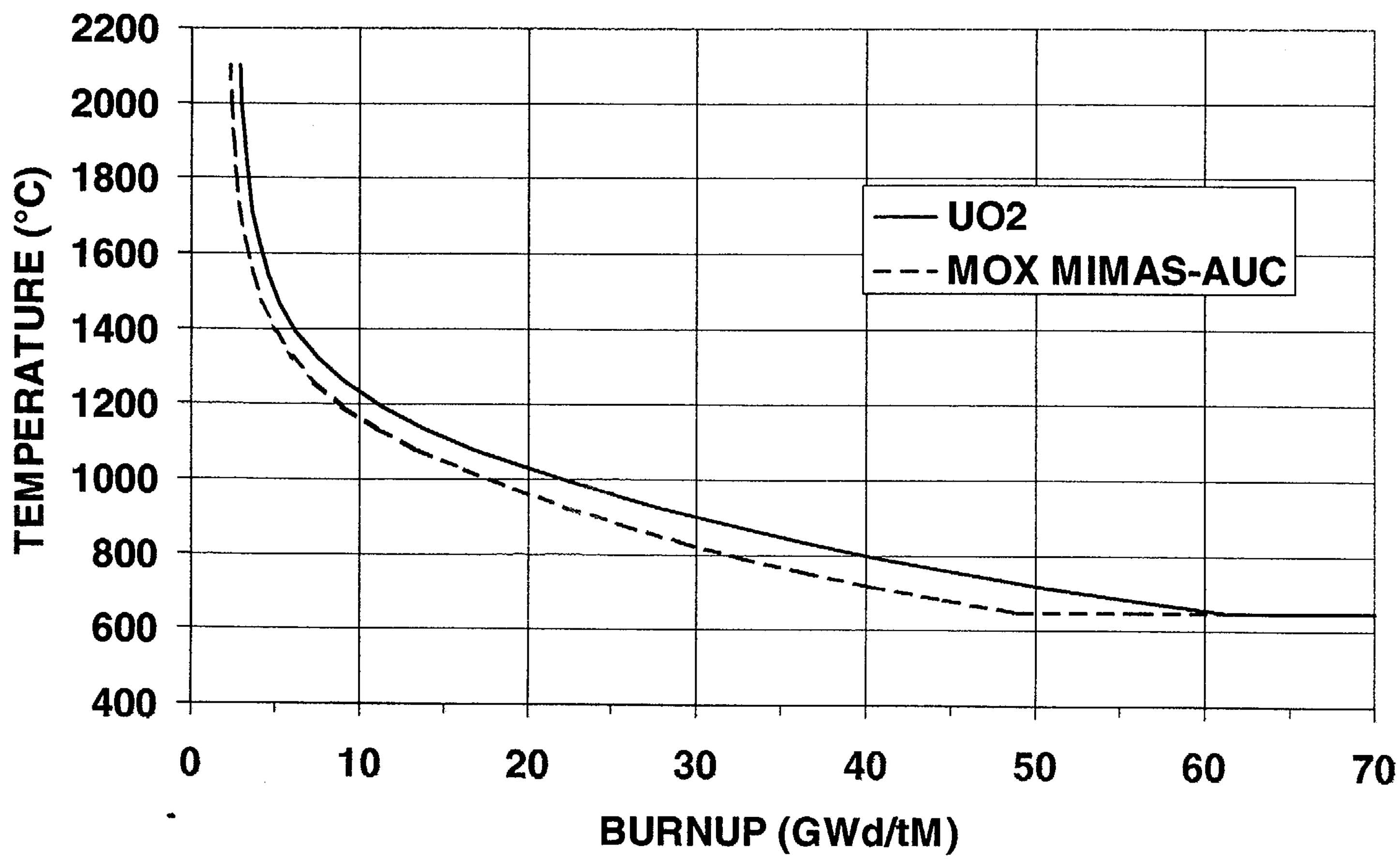
. EXPERIMENTAL DATABASE

- SPECIFIC EXPERIMENTS & INTERNATIONAL PROGRAMS
- EXTENDED TO VERY HIGH BURNUPS (102 GWd/tM) & HIGH LHGRs

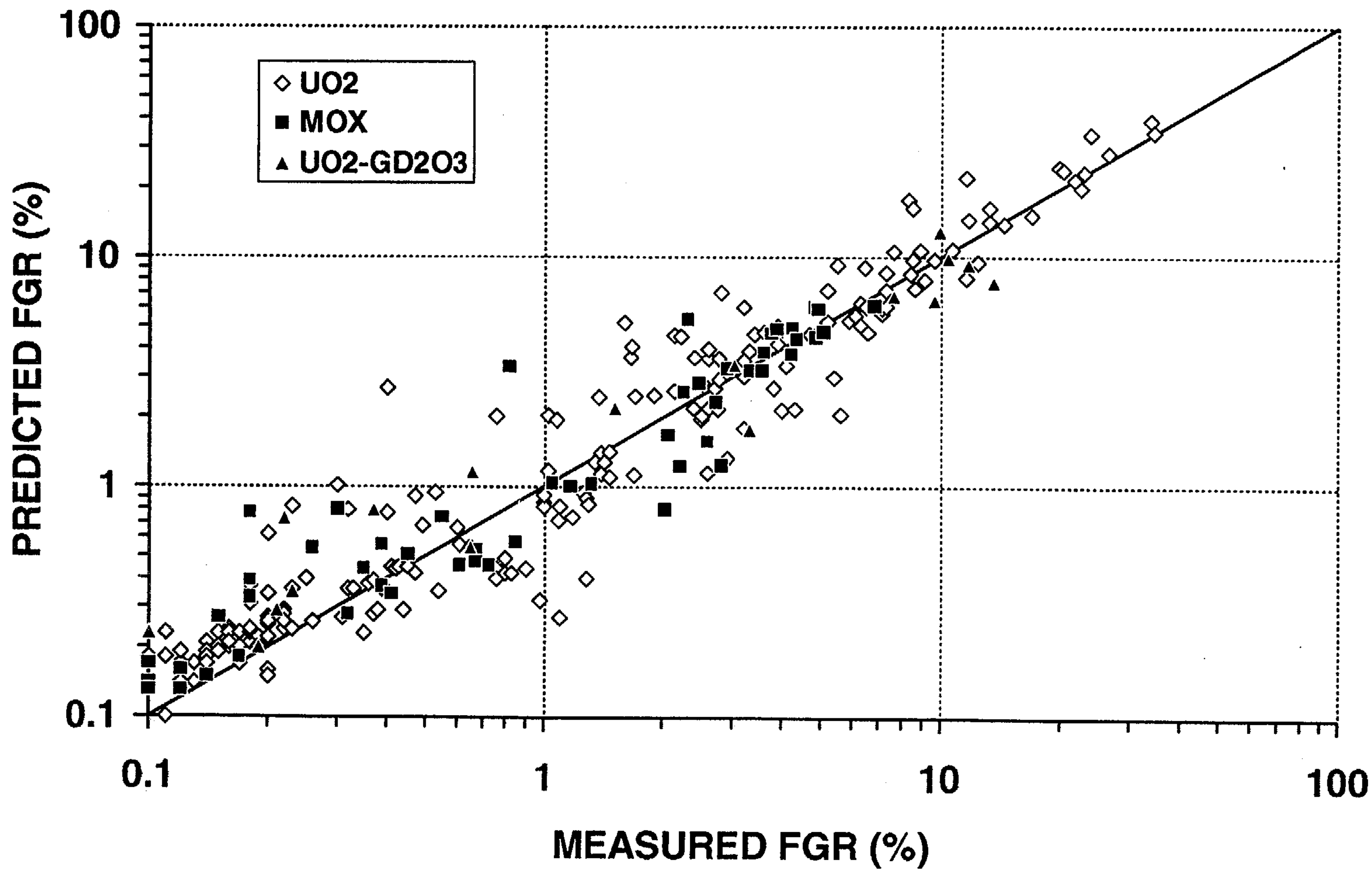
FISSION GAS RELEASE (FGR)

- Steady-state + Transient databases
  - UO_2, UO_2-Gd_2O_3, MOX
- Steady-state > 280 fuel rods up to 67 GWd/tM
  - UO_2 steady-state: calibrated with more than 200 rods
  - Rods equipped with M5-alloy cladding up to 63 GWd/tM
- Transient > 50 fuel rods up to 62 GWd/tM
- Measurements soon available:
  - UO_2 fuels irradiated 6 cycles in a French PWR
  - MOX fuels irradiated 5 cycles
  - Re-irradiation in the HALDEN reactor of a MOX fuel (4 cycles in a PWR - transient FGR)

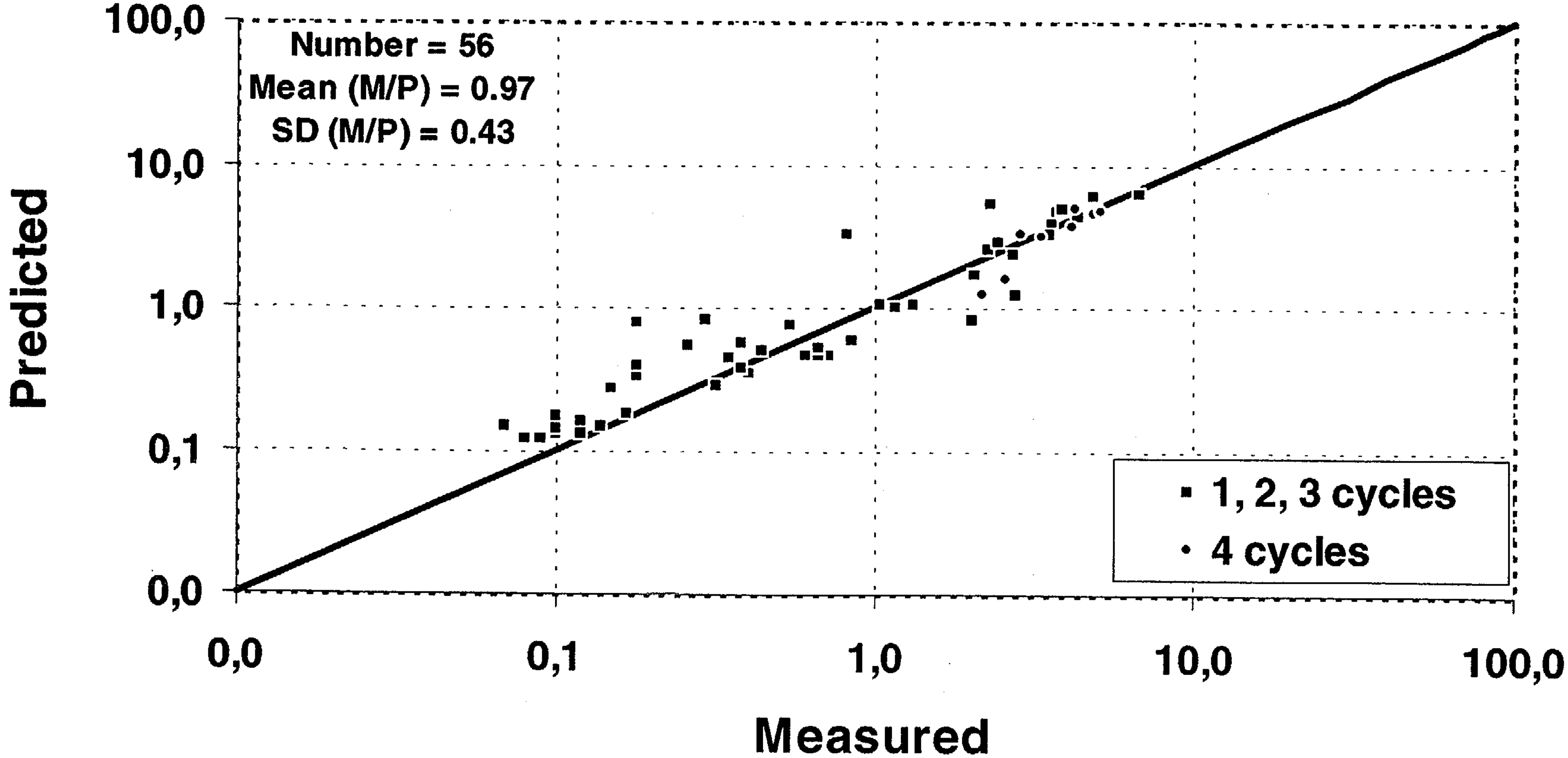
FGR MODEL: INCUBATION THRESHOLD



UO2 STEADY-STATE FGR

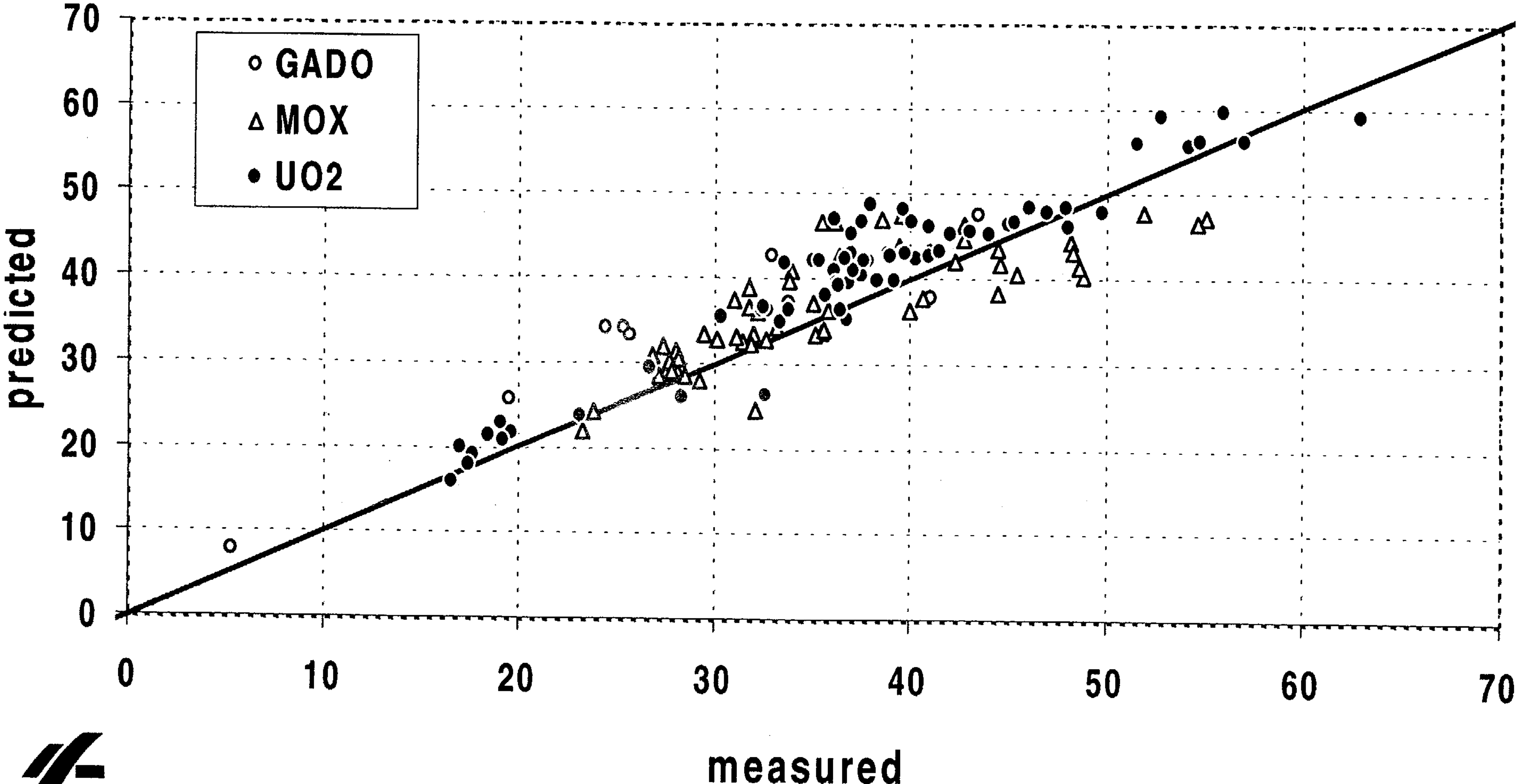


FGR MOX : COPERNIC V2.2



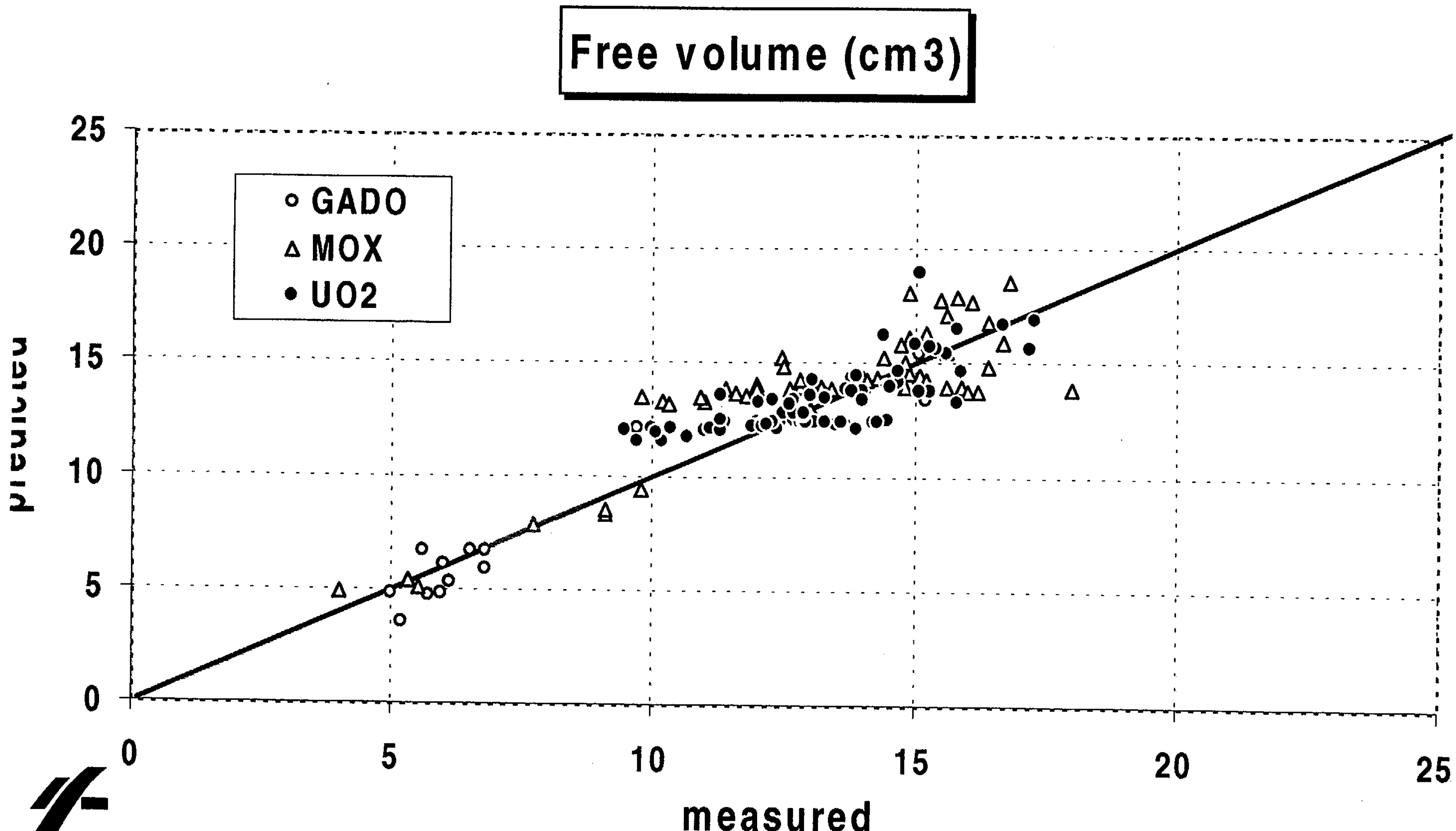
GLOBAL VALIDATION

Internal pressure (bar)



FRAMATOME NUCLEAR FUEL

GLOBAL VALIDATION



FRAMATOME NUCLEAR FUEL

Design and Sales Division

CONCLUSIONS

- Acquisition of a very large data base (surveillance and analytical programmes)
- Very good behaviour (reliability, performance) up to rod burnup of 60 GWd/tM
- Fuel rod design code: same prediction quality as UO_2 fuel
- Continuation of a significant R&D effort in order to increase the fuel performances (very high burnup)

DUKE COGEMA
STONE & WEBSTER

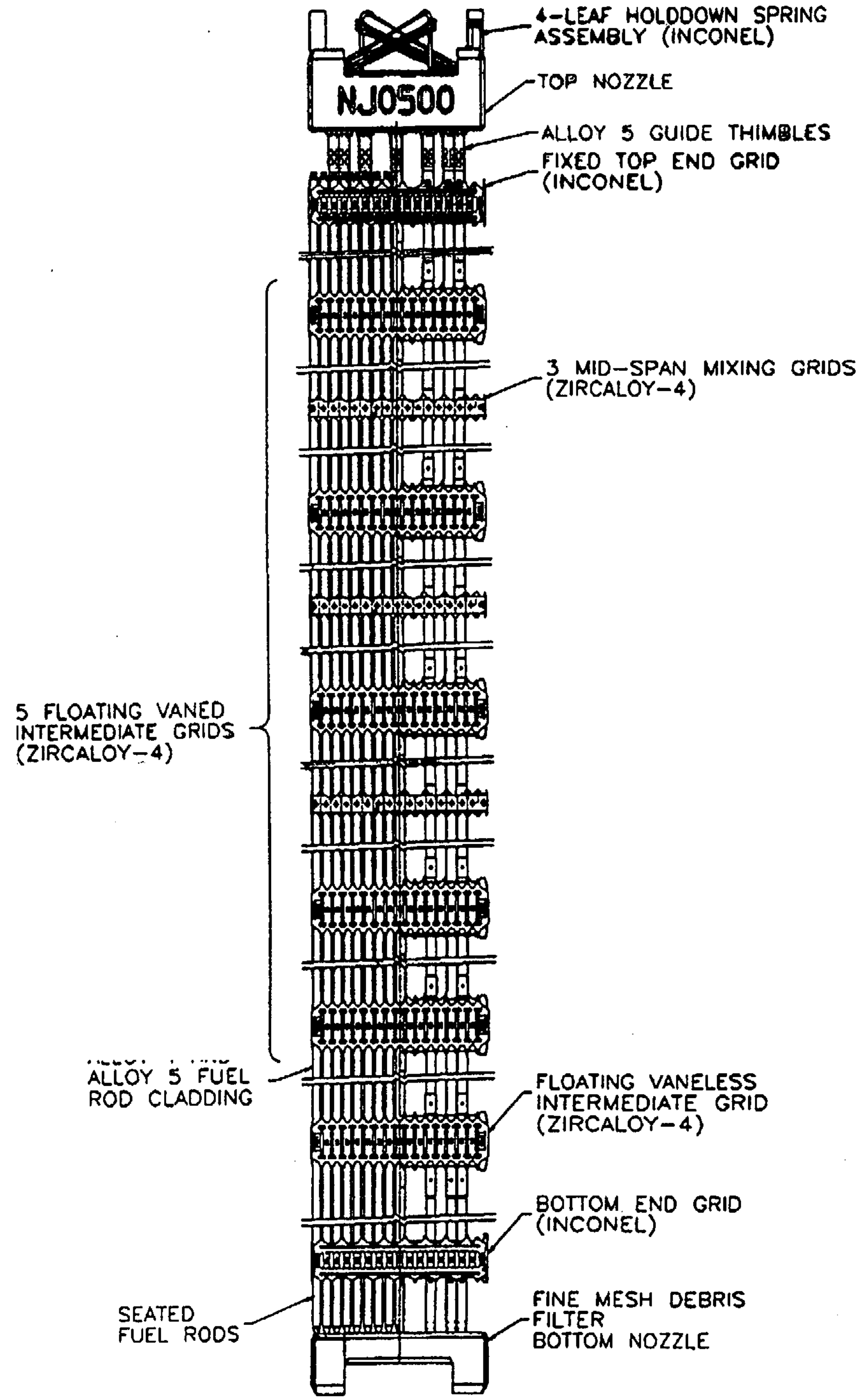
Fuel Assembly Design

NRC/DCS Meeting

October 12, 2000

Advanced Mark-BW Fuel Assembly Design

June 2, 1999



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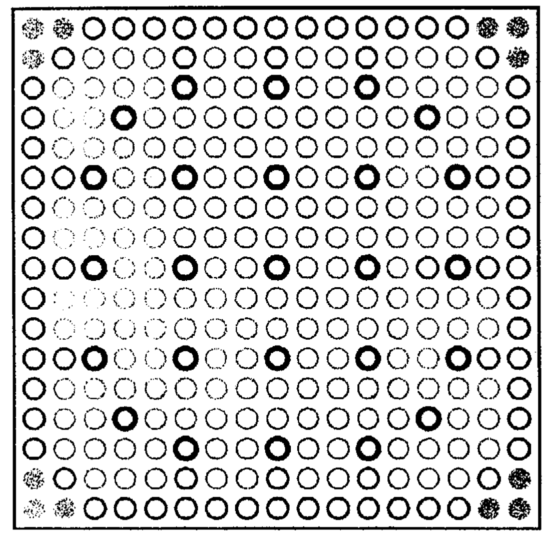
Fuel Qualification Plan

Fuel Assembly Design

- Fuel rod design
  - Same dimensions as UO_2 rod
    - 144 inch stack height
    - 22.5 mil cladding wall
    - 6.5 mil diametral gap
    - 463 kg hm fuel assembly loading
  - MOX pellet
    - Specification based on Framatome specification
    - Additional requirement on gallium (applied to PuO_2 powder)

Fuel Assembly Cross Section

- Use of zoning minimizes power peaking
- Three enrichments per assembly
- Representative of final batch design



	4.07 w/o Design	4.37 w/o Design	No. of Rods
High w/o Pu-Total	4.364 w/o	4.794 w/o	184
Medium w/o Pu-Total	3.583 w/o	3.583 w/o	68
Low w/o Pu-Total	2.316 w/o	2.316 w/o	12
Guide Tube			
Instrument Tube			

MOX Fuel Project

Steady State Core Physics Methodology

Jim Eller

October 2000

Duke Power SM
A Duke Energy Company

Presentation Overview

- Background
- Analytical Models
- Methodology Report
- Benchmark Analysis

Historical Perspective

- Duke core design methodology is used to support operation of 7 nuclear units
- Since 1982, Duke methodologies have been used to design and operate 60 fuel cycles
- This translates to approximately 68 EFPY of operating experience

MOX Fuel Characteristics

- Mixed oxide fuel pellets are 95% Uranium oxide
- 85% of the assemblies in the initial mixed cores will be standard Uranium fuel assemblies
- At least 60% of the assemblies in all mixed cores will be standard Uranium fuel assemblies
- In many respects, MOX fuel will have no impact of reactor operation or performance

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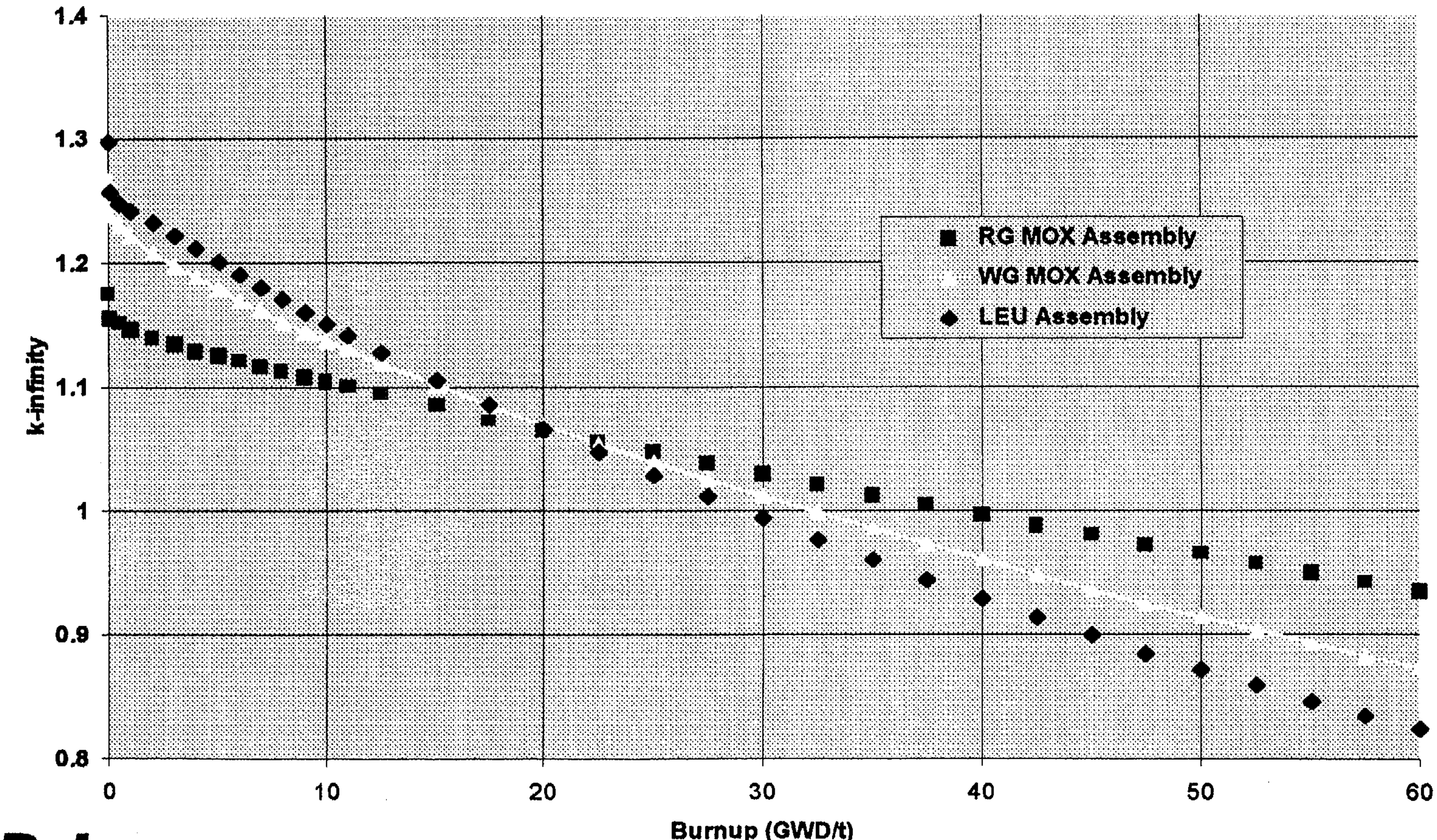
k-infinity vs. Burnup



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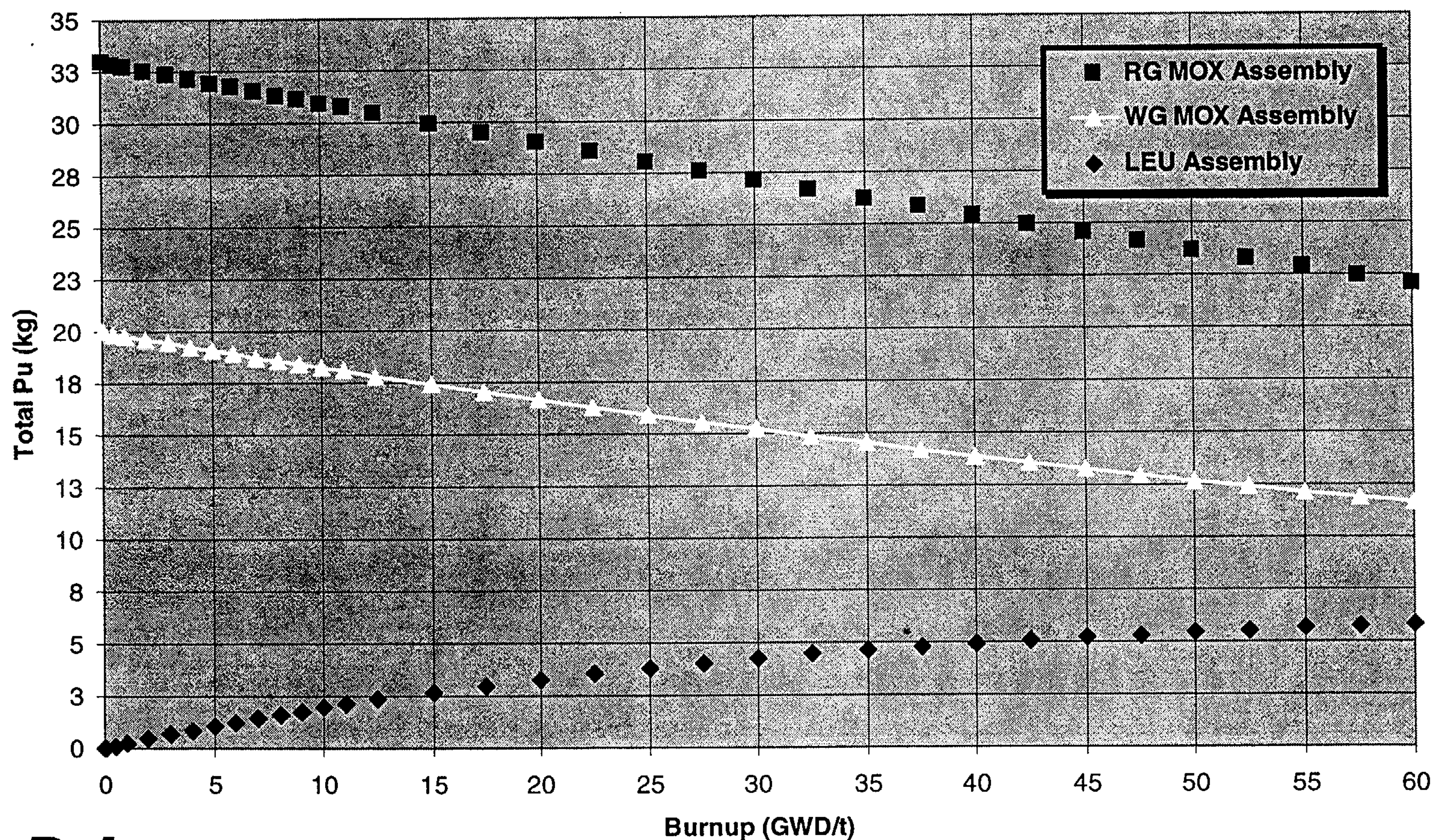
Plutonium Mass vs. Burnup



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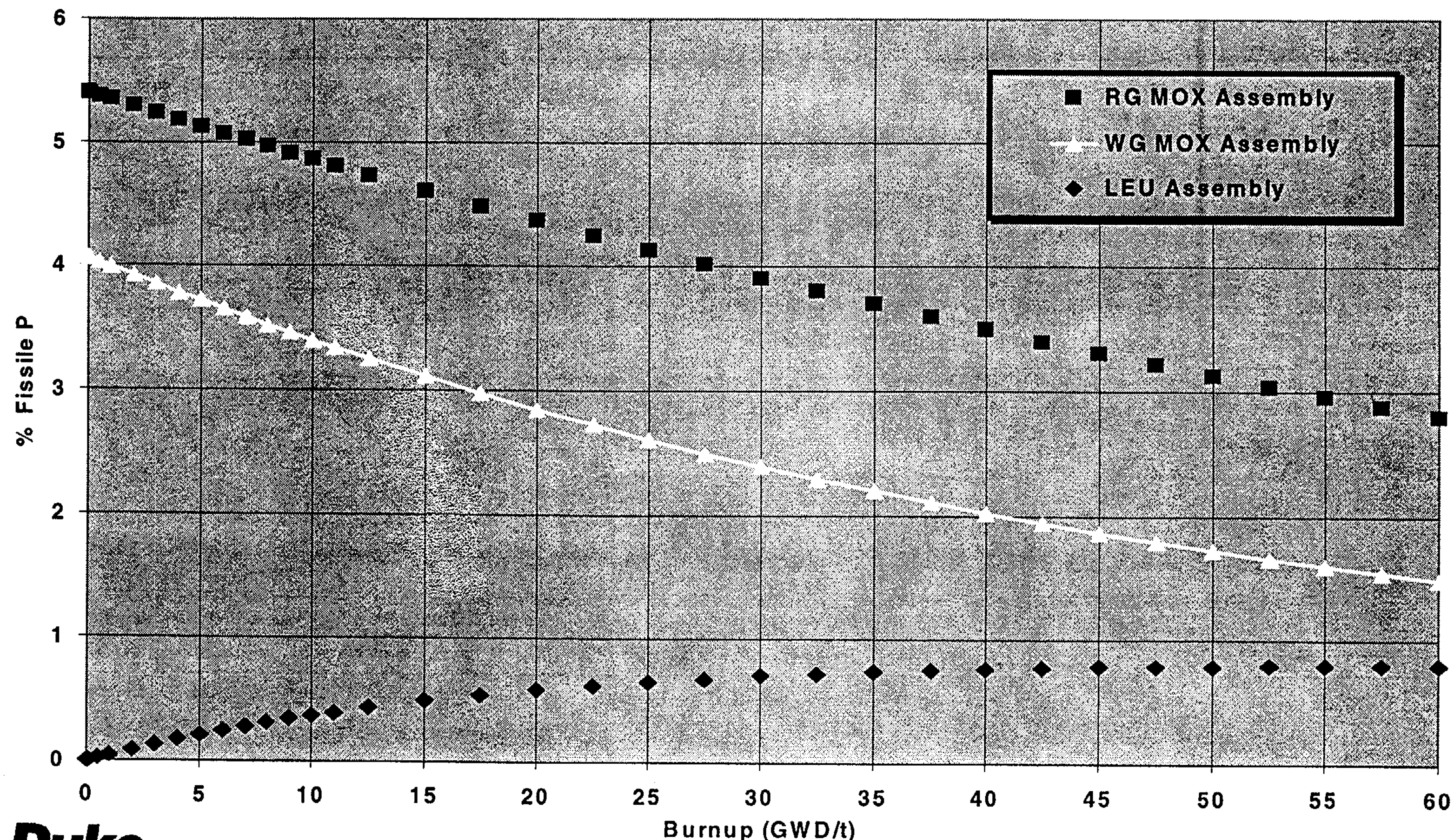
Fissile Pu Versus Burnup



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Core Physics Methodology

- Current methods use CASMO-3, SIMULATE-3, and SIMULATE-3K
- Updated methods will use CASMO-4 and modified versions of SIMULATE (SIMULATE-MOX)
- Changes made to analytical models to accommodate MOX fuel extend or enhance existing models
- CASMO/SIMULATE core models are used to support more than 75 PWR's in 11 countries worldwide

Core Physics Methodology Report

- **Details of the methodology will be provided in a topical report by August 2001**
- **Format and content of the new report will be similar to previously approved reports**
- **The methodology report will :**
  - describe the analytical models,
  - describe the reload core design process,
  - document comparisons of model results to measured data

Core Physics Benchmark Analyses

- Total benchmark effort models 29 fuel cycles and 30 critical experiments
- Benchmark analysis is comprised of 3 major components
  - comparison to measured data from recent McG/Cat LEU fuel cycles
  - comparison to measured data from European reactor utilizing MOX fuel
  - comparison to measured pin by pin power distributions from critical experiments containing MOX fuel pins

McGuire / Catawba Benchmark Analysis

- Includes comparisons to 10 most recent fuel cycles
- Comparisons to measurements
  - BOC HZP physics test
  - Core reactivity letdown versus cycle depletion
  - Assembly power distributions versus cycle depletion
  - EOC HFP temperature coefficient

St Laurent B1 Benchmark Analysis

- 3 loop Westinghouse type reactor utilizing 17x17 fuel
- Model first 12 fuel cycles, 8 cycles containing MOX fuel
- MOX core fractions up to 30% of 157 assemblies
- Comparisons to measurements
  - BOC HZP physics test
  - Core reactivity letdown versus cycle depletion
  - Assembly power distributions versus cycle depletion

Critical Experiment Benchmarks

- SAXTON
  - Near weapons grade fuel material
- EPICURE
  - 17x17 array of reactor grade MOX fuel
  - 3 radial zones of fuel enrichment
  - AIC, B_4C , and SS poison pins
- ERASME
  - 11 % Pu enrichment
  - Several arrays containing B_4C poison pins
- B&W
  - LEU fuel
  - widely modeled experiments
  - used to support currently approved methodologies

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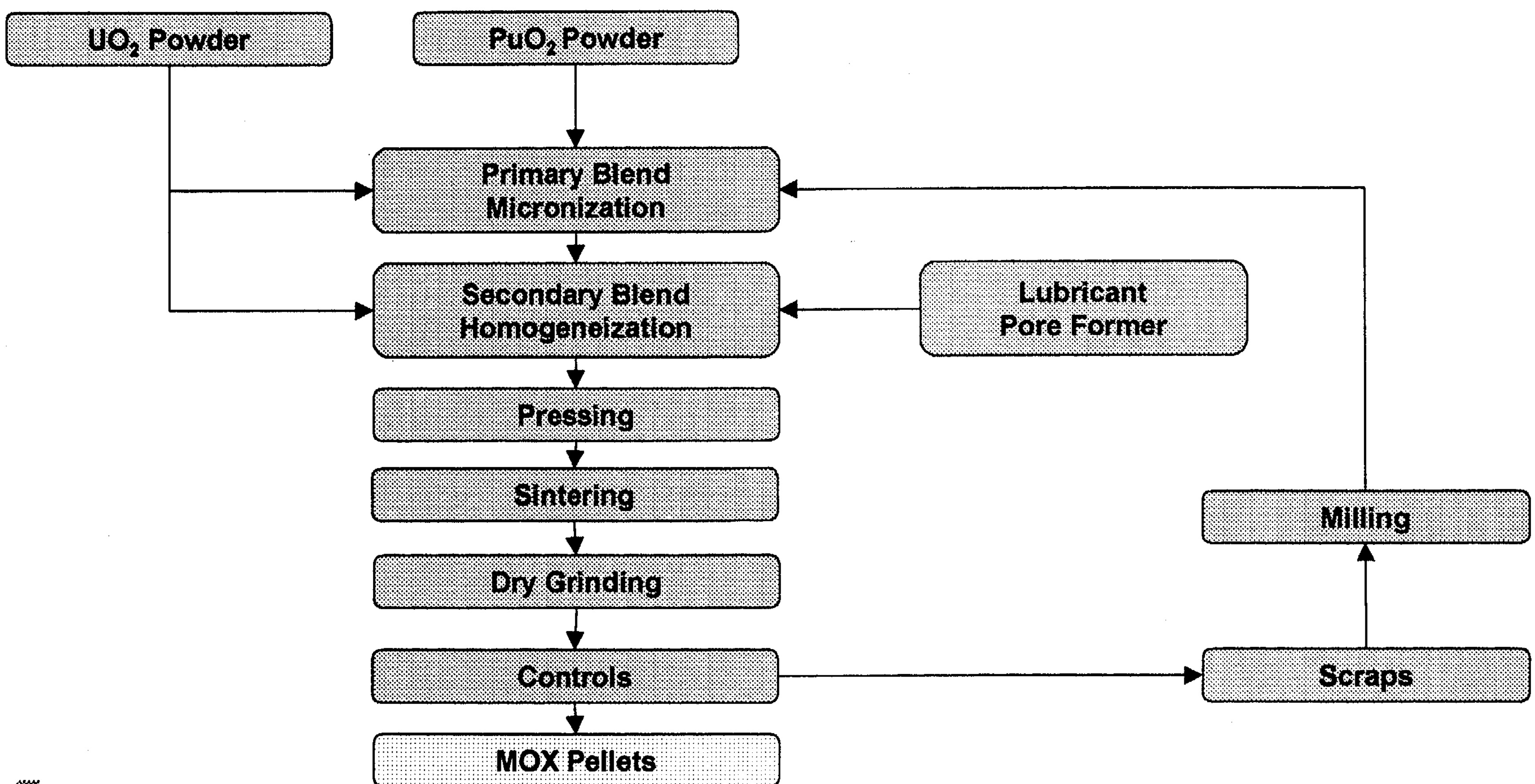
Summary

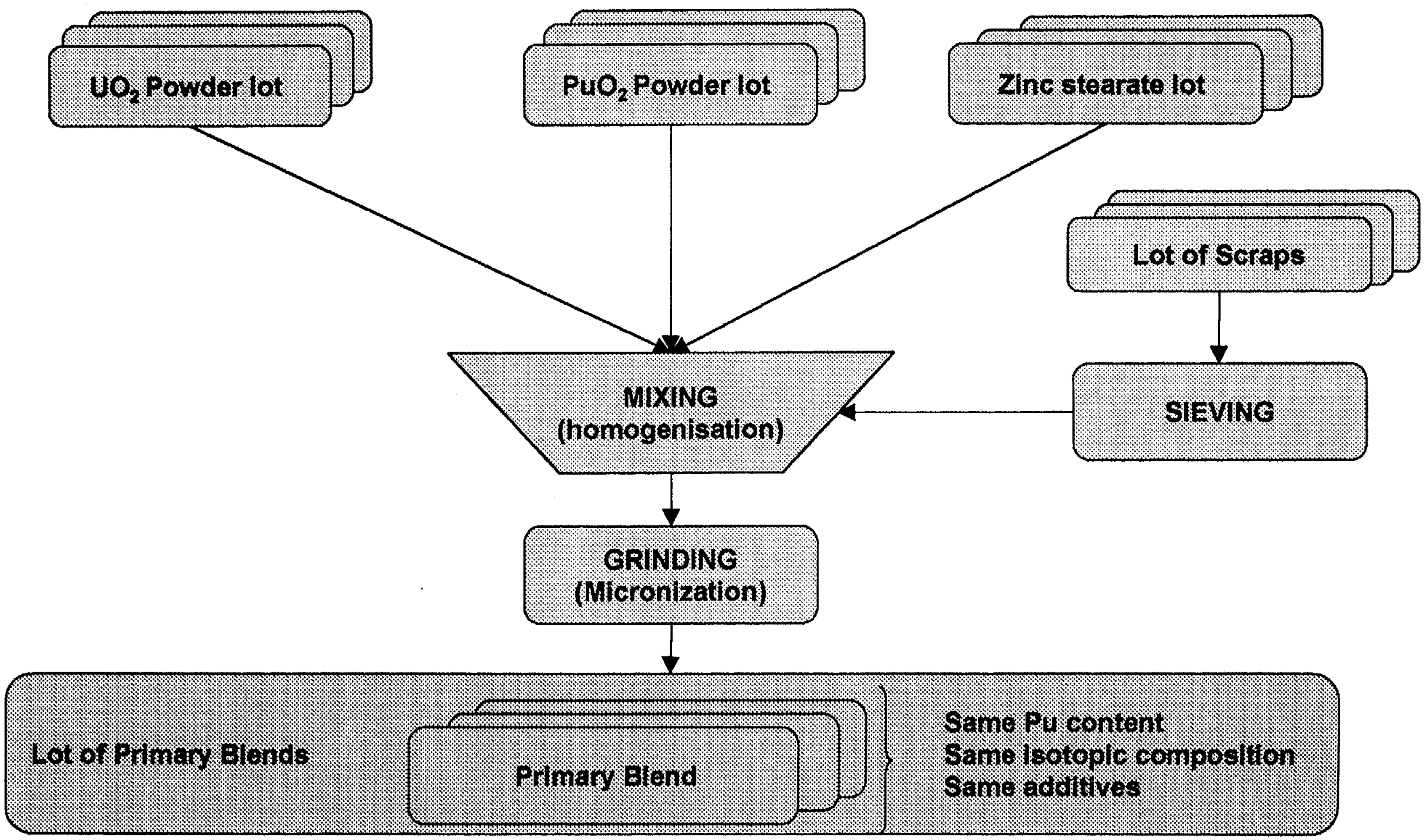
- The "MOX" methodology extends previously approved methods to accommodate MOX fuel
- The "MOX" methodology will make use of widely used analytical models
- Model validation and documentation is ongoing
- Anticipate submittal of methodology report in August 2001

MOX Pellet Fabrication

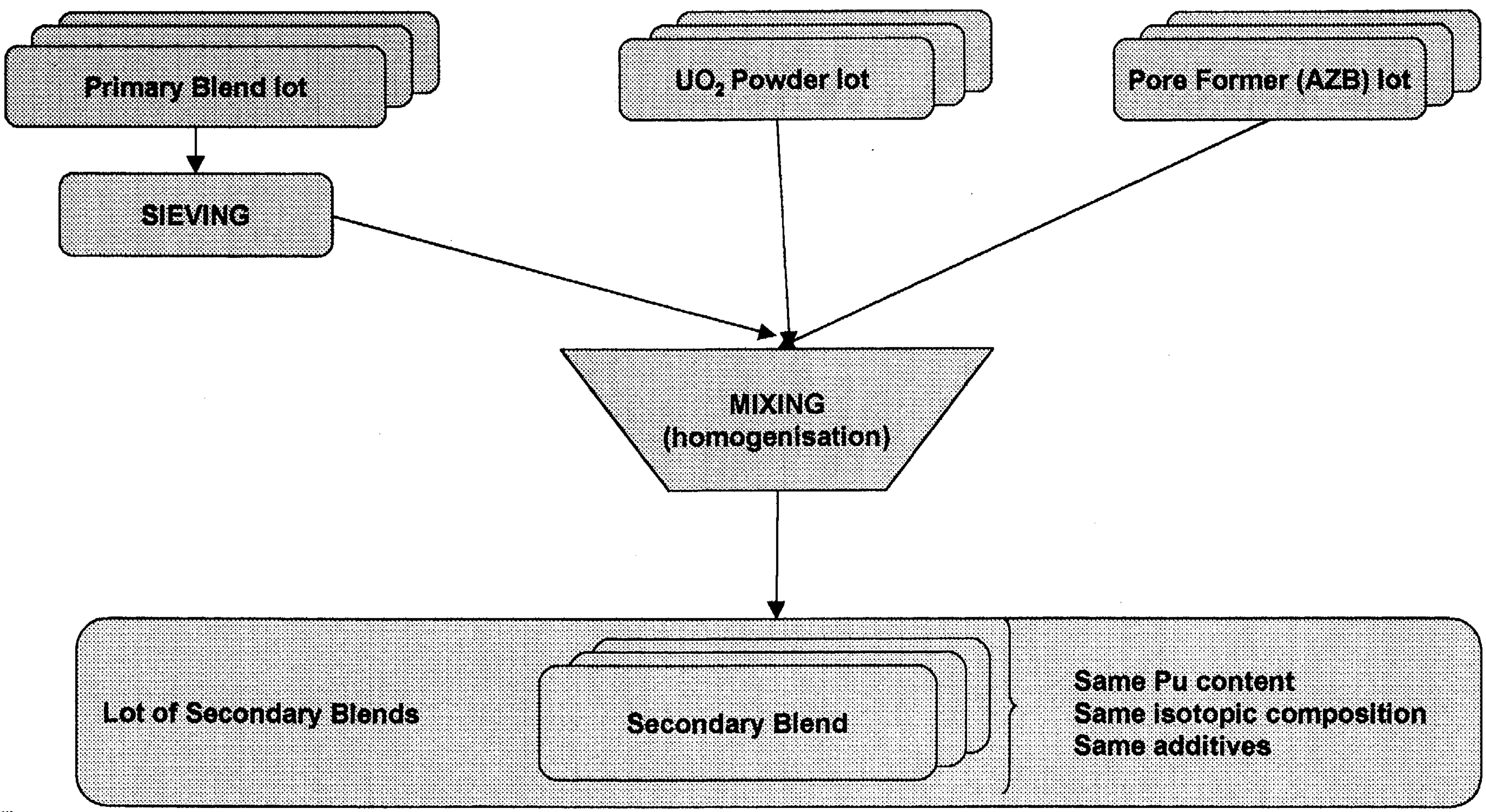
MIMAS PROCESS ("Micronization Master blend")



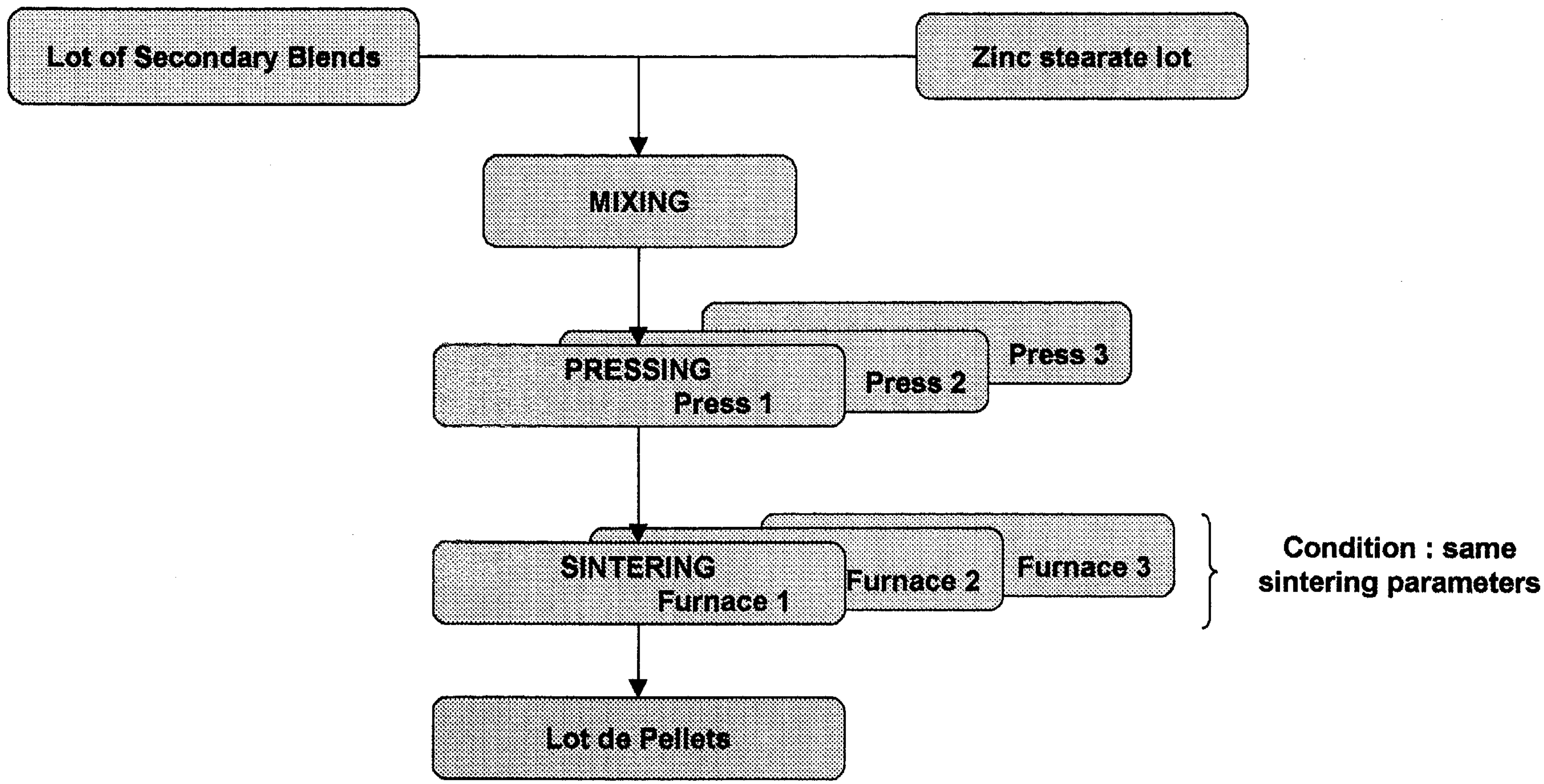
PRIMARY BLEND



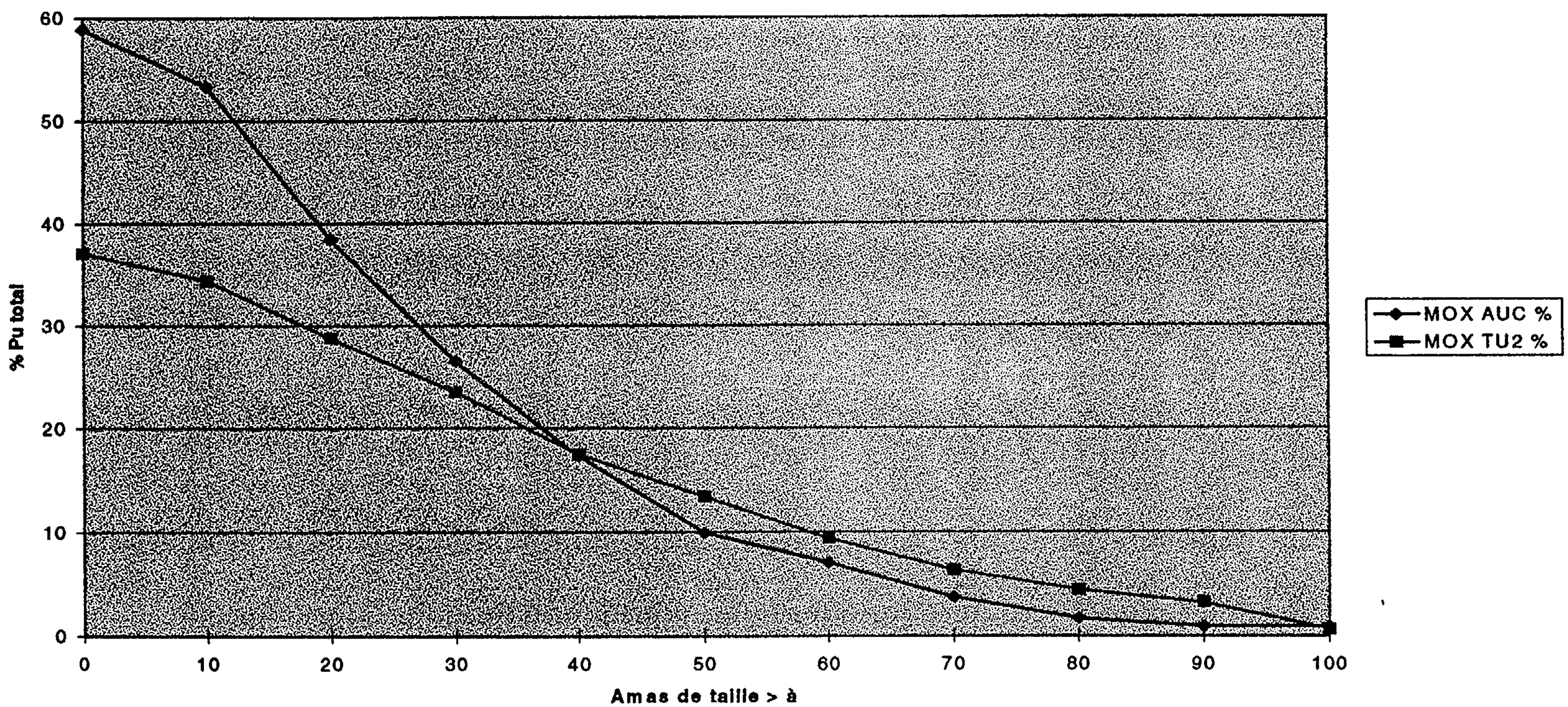
SECONDARY BLEND



PELLETIZING



As-Fabricated Pu Distribution



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Lead Assembly Program

NRC/DCS Meeting

October 12, 2000

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Lead Assembly Baseline Plan

LA fabrication complete		Jul-03
Irradiate M2C16		Oct-03 - Mar-05
Perform Poolside PIE		Mar-05
Irradiate M2C17		Apr-05 - Sep-06
Perform Poolside PIE		Sep-06
Decision to proceed with fuel fabrication		Oct-06
Certify Completion of Fuel Qualification		Oct-06
Batch irradiation	C2C16	Oct-07

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Lead Assembly Examinations

- Poolside Post Irradiation Examination (PIE)
  - Follows 1st and 2nd cycles of operation
  - Projected bunrup of 44,000 MWd/MThm, peak rod
  - Basis for batch implementation
  - Inspections
    - Fuel assembly growth
    - Fuel rod growth
    - Fuel rod oxide
    - RCCA drag force
    - Fuel rod integrity

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Lead Assembly Examinations

- Hot Cell Examination
  - Following 3rd cycle of operation
  - Projected burnup > 50,000 MWd/MThm
  - Basis for future burnup improvement
  - Inspections
    - Fission gas release
    - Fuel clad metalography
    - Fuel pellet ceramography
    - Pellet-cladding interaction
    - Burnup analysis
    - Burnup distribution

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Lead Assembly Fabrication

- Fuel Qualification Plan
  - Fabricate two lead assemblies at LANL
  - Irradiate in McGuire 2, Cycle 16, 17 starting in October, 2003
- Current Plan - Alternate Fuel Qualification Study

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Alternate Fuel Qualification Study

- Evaluate
  - Fabrication of Lead Assemblies in Europe (Eurofab)
  - Fabrication of Lead Assemblies at the MOX Fuel Fabrication Facility (MFFF)
- Recommendation due to DOE October 20
- Projected decision date - by January 1, 2001

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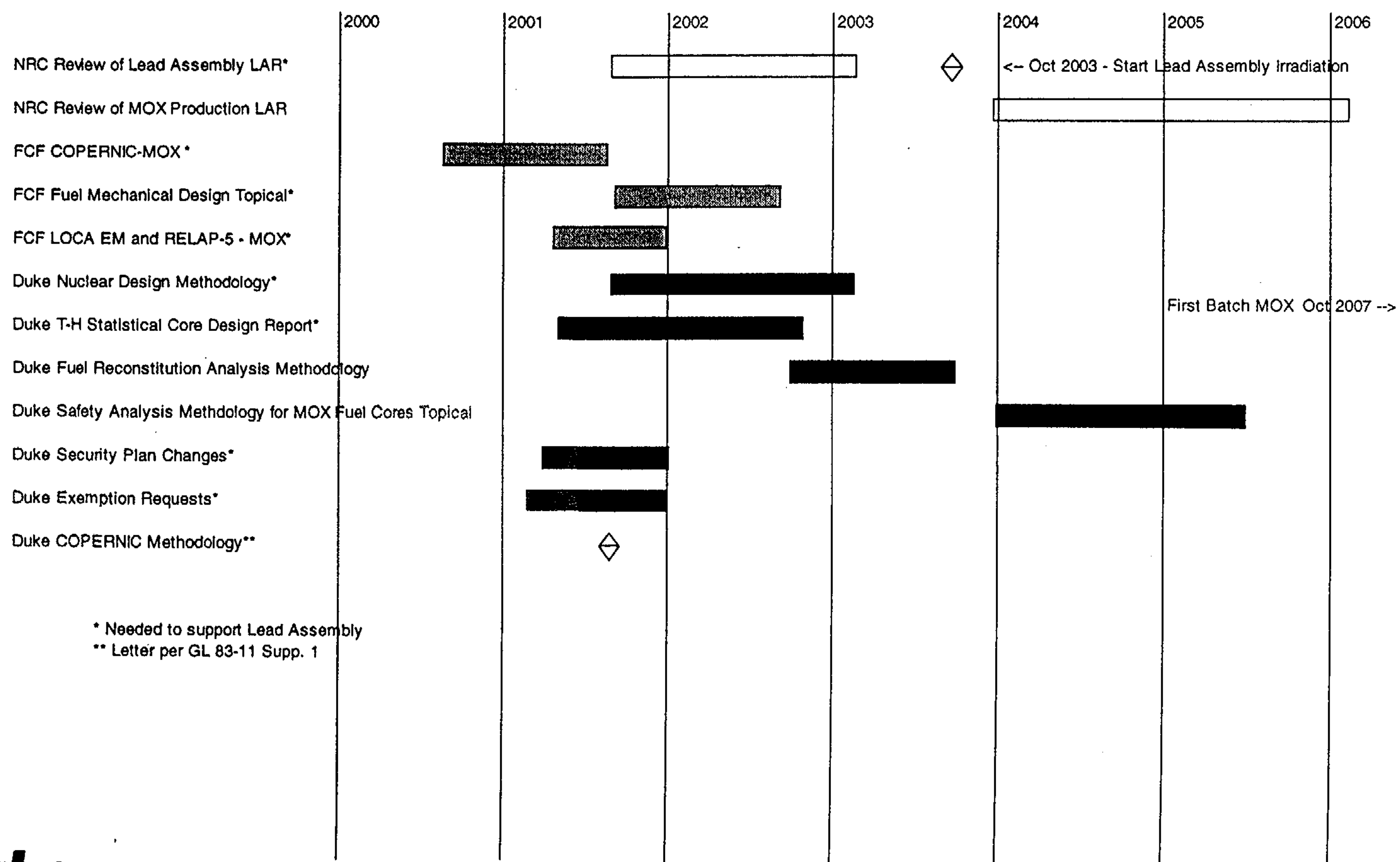
Alternate Fuel Qualification Study

- Eurofab
  - Weapons-grade plutonium
  - Polished at French facility
  - Fabrication of four lead assemblies in Europe
  - Fabrication processes match MFFF and mission reactor fuel
  - Commercial shipment to U.S. port, shipped via SGT to McGuire
  - Maintains lead assembly and fuel qualification schedules published in the Fuel Qualification Plan

Overall Licensing Schedule



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