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EXHIBIT D

NUCLEAR REGULATORY COMMISSION

Docket No. 50-413/414-OLA Official Exh. No. 28
In the matter of Duke Catawba
Staff ______ IDENTIFIED 7/14/04
Applicant ______ RECEIVED 7/14/04
Intervenor ✓ REJECTED ______
Cont'g Off'r ______
Contractor ______ DATE ______
Other ______ Witness ______
Reporter [signature]

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SECY-02

IRSN
Source Term Loca
PROGRAM
IN
THE PHEBUS FACILITY

Part 1 : The Safety Context
Part 2 : Source Term Part
Part 3 : The LOCA Part

A. Mailliat, J.C Mélis

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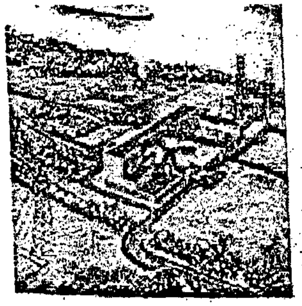
Evolution of the Market



The production and distribution of electric power experience a dual evolution

✓ Constructions and commissioning of nuclear power plants come to very low levels.
✓ Liberalisation and deregulation induce competition between operators on domestic and international markets.

The answer : utilities enhance the effectiveness of their facilities.

Evolution of the Reactors

✓Increase in reactor power: from 900 MWe to 1,400 MWe
✓Increase in fuel burn-up: from 33 GWday/tU to 60 GWday/tU
✓Introduction of new types of fuel - from UO2 to MOX
✓Introduction of new cladding and control rods
✓Fuel cycle lengthening from 1 year up to 1.5 or even 2 years
✓U5 enrichment increases, poison additions, etc

Formerly : Wide Margins

✓Nuclear safety models were designed with wide margins in order to cope with uncertainties both in the data base and the accident phenomenology knowledge

✓Conservative scenarios were taken as a basis for regulation and standard setting

Now : a Permanent Adjustment

A permanent need to reassess reactor safety studies

✓Due to continuous demands on plant and core operations

✓The increasing tendency of the operators to use best estimate codes and more realistic conditions for accident analyse

✓probabilistic safety studies and source term evaluations are being refined and call for reducing uncertainties on the consequences of some specific accidental situations e.g Air Ingress, Core Quenching

How much these margins are used by operators :

✓Are the criteria always appropriate ?

✓Are the accident estimates always correct ?

New Requirements

✓ data base extensions to broader conditions and materials

✓But also an adequate quantification of the data base uncertainties for safety task uses

Answer the future safety requests

Taking advantage of existing information
By continuation of available tools
Optimising the efforts and the costs

- **Updating and upgrading by continuation of the models**
  - Fuel behaviour, cladding properties, etc..
- **An optimized number of small-scale or semi-integral , out-of-pile or in-pile experiments**
  - Providing the additional data bases for model updating
- **A few integral in-pile experiments**
  - provide code assessments in terms of reactor applicability and simulation completeness.
  - Quantifying the calculation tool uncertainties and margins to criteria

According to this context, IRSN is preparing Source Term Loca new program the PHEBUS facility:

- *STL –LOCA Part* devoted to Loss Of Coolant Accident (LOCA)
- *STL –SourceTerm Part* for Severe Accidents

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Part 2

Source Term Loca Program-*Source Term Part*

A. MAILLIAT
B. CLEMENT
J.C. MELIS

CONTENT

- Severe Accident Pending Issues
- Rationale for Tests
- Source Term equipments

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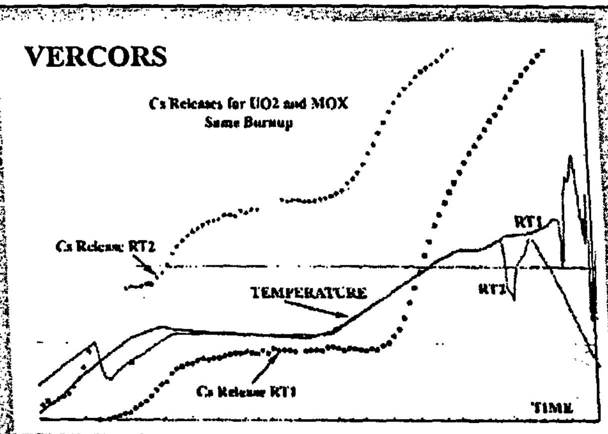
Pending Questions :

UO2 and MOX



Release rates for MOX fuel , High Burnup UO2

- ✓Fuel structure is different (RIM),
- ✓High burnup impacts fuel stoechiometry
- ✓Very high burnup in Pu rich clusters impacts on the FP distribution in the fuel phases
- ✓Pu-an U chemistries with regard to oxygen are different
- ✓FP release for observed earlier for MOX than for UO2 with the same Burnup
- ✓Low but significant release (<15%) of Nb, La, Eu, Mo, Ce, Np

Degradation process

- ✓Possibly a different degradation process like the so called fuel foaming which might take place especially with high burnup fuel. (400% swelling observed in ST1-2 tests in Sandia)
- ✓Mox fuel relocation was observed at a lower temperature than for UO2 (RT2 test)

Representative Data Base are rather poor in terms of results for high BU an Mox
UO2 (BU in GWd/tU) : VI-4 (47), VERCORS 6 (60), HT1 (49,4) RT1 (47.3) VEGA 4 (47)
MOX (BU in GWd/tm) : RT2 (45.6), RT7(43)

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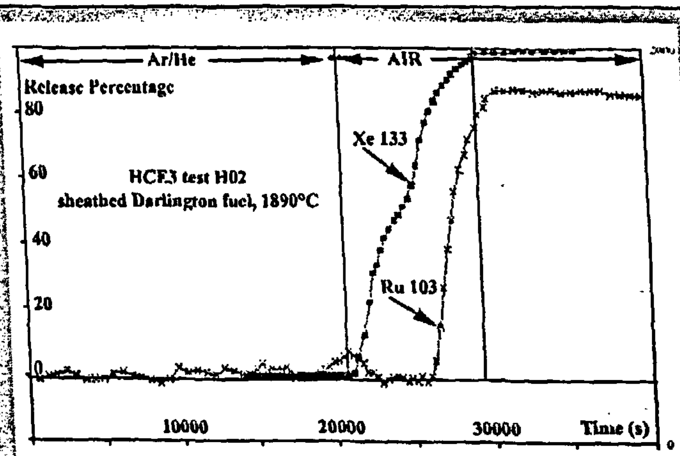
Pending Questions :

AIR



The Zircaloy oxidation process is very violent
Degradation should be increased

- ✓**Zircaloy oxidation kinetics is faster than with steam**
- ✓**Energy generation with air (755kJ/mole-Zr) is greater than with steam (459)**
  from single rod tests Dressmann (Germany) and bundle tests CODEX (Hungary)

Source term is modified

- ✓**Ru release is important**
- ✓**Ru and I can be revaporized when air is entering the primary circuit**
- ✓**Risk of RuO3 and RuO4 species in the containment in case of non equilibrium chemistry (Hot leg break)**

Pending Questions :

QUENCHING

- ✓How much corium will be involved in the interaction with water ?
- ✓which are the reacting corium properties ?

✓How much steam and hydrogen are produced during the degradation process and especially during the core quenching ?

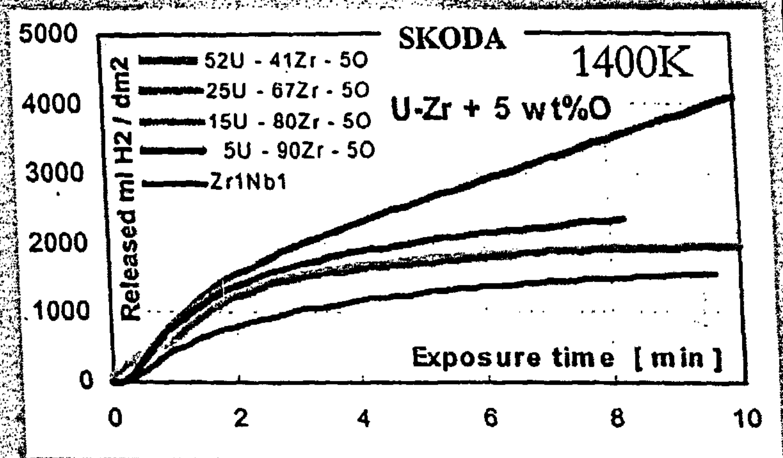
> There is not much information (PBF-ST, LOFT-FP2, PHEBUS-SFD-B9R) available for representative conditions of a severe accident, even if some material is available from separate effect tests or electrically powered tests (programmes CORA, QUENCH).

- ✓Are there any risk of a late pressurisation or steam explosion in the primary circuit ?
- ✓Induced containment bypass flow paths e.g SGTR due to primary circuit high pressure.

> 20-40 bars pressure peak observed in LOFT-LP-FP2 and TMI

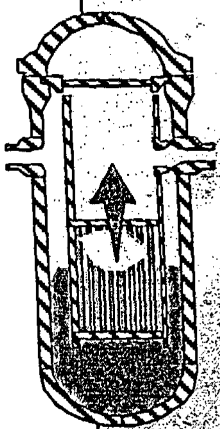
None or very few information is available about

- ✓ Additionnal releases from fuel induced by temperature escalation or fuel fragmentation (some evidence from LOFT-LP-FP2),
- ✓ Re-entrainment of previously deposited materials in and above the core by large steam flow



To build valuable data bases, results are needed from experiments which will have to reproduce the physical mechanisms implied in the degradation and release processes with enough correctness.

Three points are essential to obtain such a correctness.



- **Firstly, actual irradiated fuel (UO2 and MOx), with the appropriate burnup, is necessary.**
- **Secondly, for degradation correctness, the heat source to fuel has to be maintained despite its movements, melting and its metallurgical transformations.**
- **Finally, release rates and degradation mechanisms being so strongly related, they cannot be tackled separately.**

EQUIPMENT MAIN OPTIONS

Taking account the Phébus FP project experience, the main guide line for preparing this STLOC programmes and their associated equipment is to avoid any heavy decontamination operations which are time consuming and costly.

Such a constraint can be accommodated because the previously described objectives of the STLOC programmes do not require to investigate particular phenomena in the primary circuit (except the FP resuspension from core upper plenum) nor in the containment.

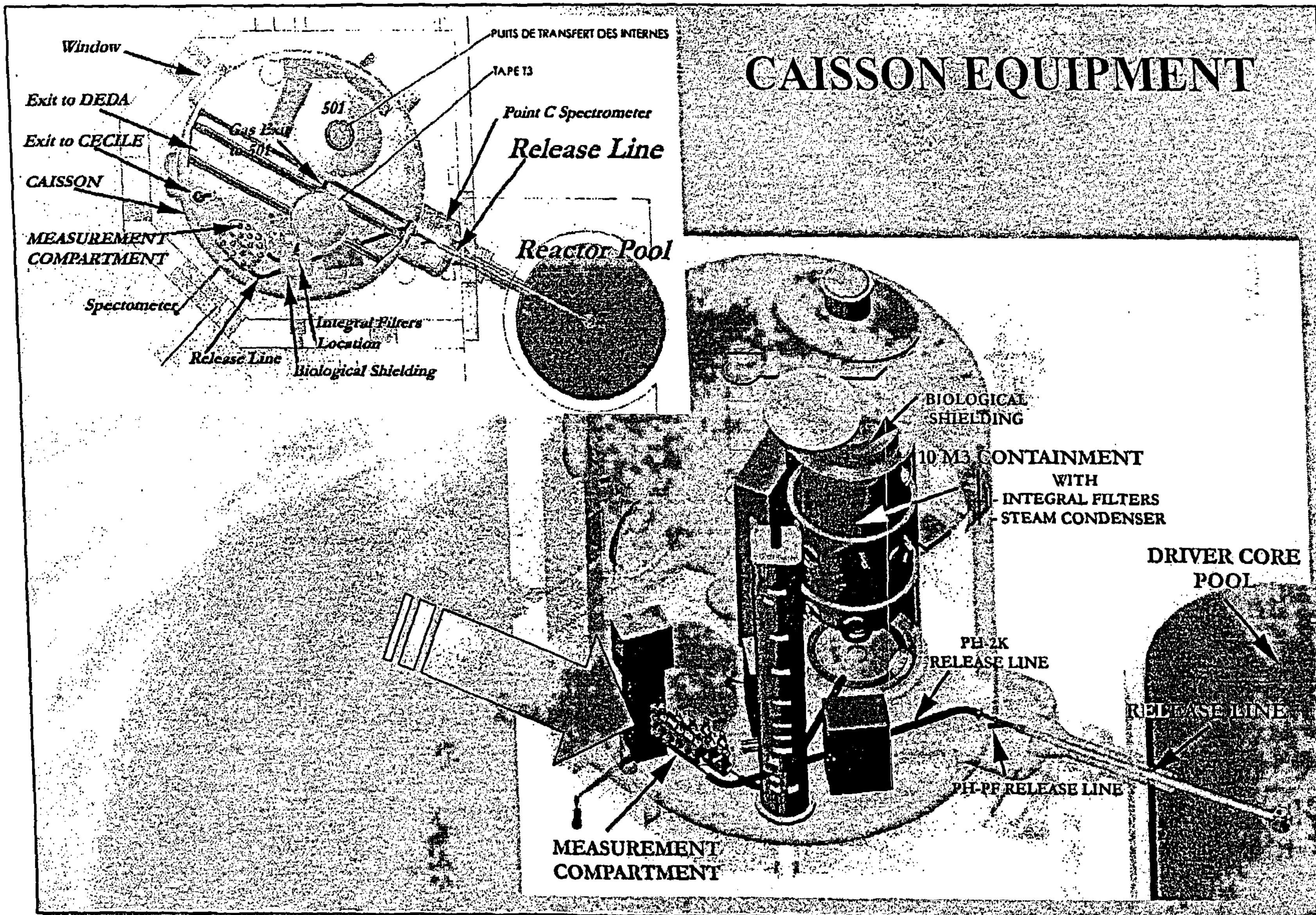
Therefore, the experimental equipment and instruments, by comparison with the Phébus- FP ones, can be

- ❑ reduced,
- ❑ simplified,
- ❑ designed in an integrated way.

EQUIPMENT MAIN OPTIONS

- SIMPLIFIED CIRCUITS
  - ♦NO MORE NEEDS FOR 700°C ON THE LINES
  - ♦NO MORE NEEDS FOR ONE WEEK RE-IRRADIATION
  - ♦NO MORE EXPERIMENTAL NEEDS IN THE CONTAINMENT
- A SIMPLIFIED SAMPLING STRATEGY
  - ♦ONE SAMPLING LOCATION ABOVE THE BUNDLE IN THE TEST TRAIN
  - ♦A SECOND SAMPLING LOCATION IN THE CAISSON INCLUDING ALL THE REQUIRED EXPERIMENTAL MEASUREMENTS
- DELAY AND COSTS
  - ♦A REDUCED TIME GAP BETWEEN TESTS
  - ♦A REDUCED INVESTMENT COST FOR EACH TEST

CAISSON EQUIPMENT



CAISSON EQUIPMENT

MAIN ASPECTS OF THE MEASUREMENT COMPARTMENT

The measurement compartment (MC) is located in the caisson and includes all the experimental measurements.

The MC is included in a 150°C prototype furnace

The MC includes 16 sampling instruments with standardised connectors removable through remote operations

Each sampling instrument is equipped with commercial self sealing low pollution quick disconnect coupling

After sampling removals, sleeves replace the instruments and the decontamination is performed.

The MC can be transferred without any dismantling through the equipment lock T3

The STL-SoureTerm programme includes 2 MCs, one under preparation and calibration while the other one is under operation in the caisson for a test

MEASUREMENT STRATEGY

Highlights on the measurement strategy for the STL-SourceTerm tests.

Fuel degradation measurements will be basically the same as for the Phébus-FP:

- ❑ A number of on-line sensors
- ❑ Sophisticated in-situ non destructive techniques (?-spectrometry, absorption and emission computed tomography).
- ❑ Detailed destructive examinations will also be performed.
- ❑ A special attention will be paid to measure transient hydrogen and steam productions during the core reflooding.

Realeases measurements will be performed through :

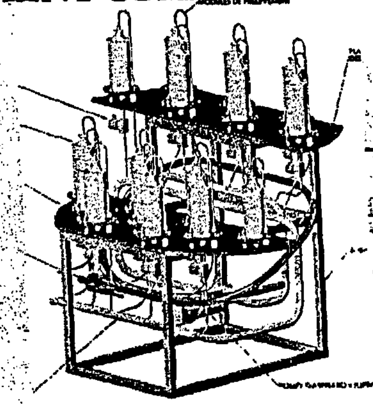
- ❑ Thermal gradient tubes, above the fuel, aiming at the determination of fission products deposition and re-entrainment during reflooding.
- ❑ The Measurement Compartment including a number of sequentially operated filters in order to measure both the transient and overall releases and impactors for aerosol particle sizing.

The basic technique will be ?-spectrometry, both on line and on the samples after their transfer to the CECILE hot cell.

For non-? emitters, chemical analyses will be performed to complete the experimental data base.

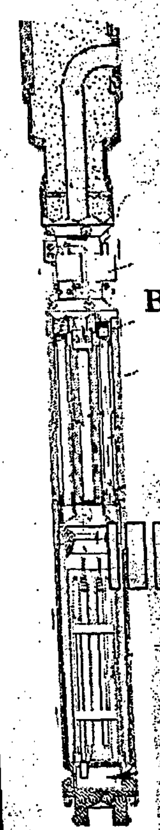
MEASUREMENT STRATEGY

MEASUREMENT COMPARTMENT



- ON LINE
  - → GAMMA SPECTROMETRY
- SEQUENTIAL
  - → RELEASE RATE
  - → RESUSPENTION RATE
- POST TEST
  - → GAMMA SPECTROMETRY
  - → DESTRUCTIVE EXAMINATIONS
  - → CHEMICAL ANALYSES

BUNDLE AND UPPER PLENUM



- CONVENTIONAL
  - → PRESSURE
  - → TEMPERATURES
  - → MASS FLOW RATES
  - → POWER
  - → HYDROGEN, STEAM CONTENTS
- SEQUENTIAL
  - → RELEASE RATE
  - → DEPOSIT RATE
  - → RESUSPENSION RATE
- POST TEST
  - → GAMMA Spec, TOMOGRAPHY
  - → DESTRUCTIVE EXAMINATIONS
  - → CHEMICAL ANALYSES

Part 3

Source Term Loca Program-*LOCA Part*

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CONTENT

- Rationale for LOCA Safety
- Pending Issues
- Rationale for LOCA Researches
- IRSN Future Programme

Design Basis Accident : BASIC REQUIREMENT

After any LOCA transient a core geometry which preserve its coolability must be guaranteed

It means :

➔Shattering of the fuel rods has to be avoided

➔Core coolability has to be maintained

Application of these basic requirements imposes two different actions

To derive Criteria

Check that Criteria are not violated for reactor LOCAs

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RATIONALE FOR LOCA SAFETY

TO DERIVE CRITERIA

It means : to know the quantities which control the cladding residual ductility back to cold conditions and core coolability, the values not to exceed.

Such information e.g. is the famous criteria 1 and 2 on Peak Clad Temperature (PCT) and Equivalent Clad Reacted (ECR)

TO CHECK THAT CRITERIA ARE NOT VIOLATED FOR REACTOR LOCAs

It means : To demonstrate, through calculations tools, that whatever is the kind of LOCA transient, nowhere in the core, the criteria values are exceeded

For providing such correct estimates we are facing two main needs in terms of models

- Models for thermal and mechanical behaviours of the fuel rods in the reactor geometry during LOCA transients
- Models for thermal-hydraulic behaviour of the coolant during LOCA transients

RATIONALE FOR LOCA SAFETY

In the context of a permanent need to reassess reactor safety studies imposes by reactor evolution, new types of fuel and cladding, increase of burnup ... IRSN is revisiting LOCA studies to check

- The adequacy of criteria
- That criteria are not violated for Reactors
- The available margins

On the next slides we will explore

- *the main pending issues which affect both criteria derivation and calculations correctness*

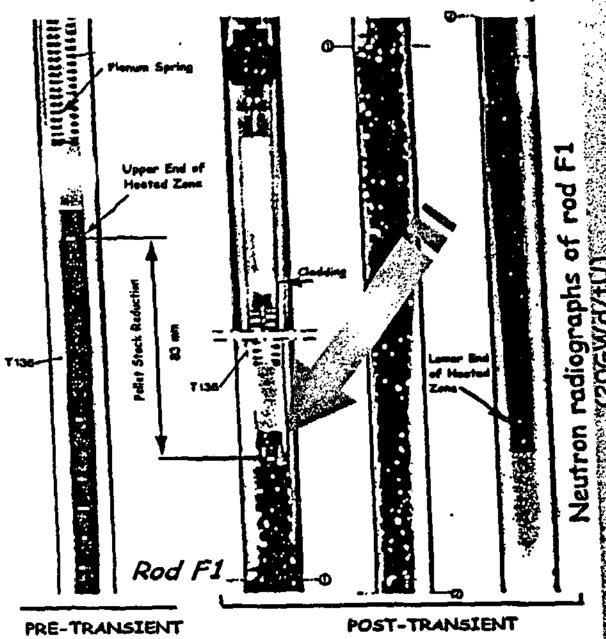
Then, from a comparison between the pending issues and the existing research programs

- *needs for additional researches will be deduced*

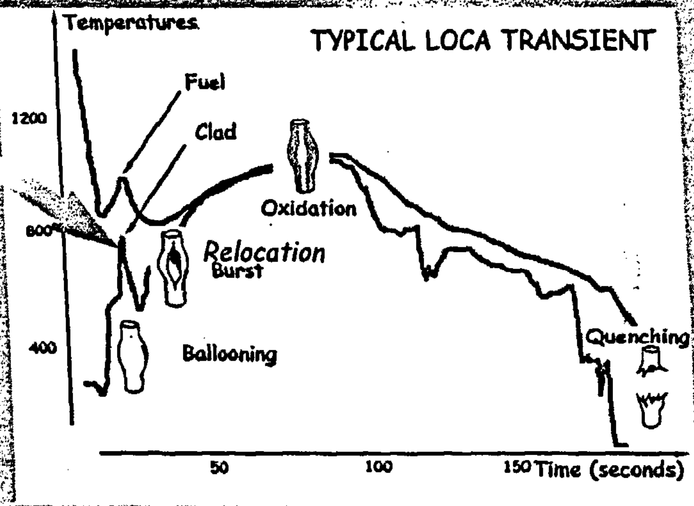
Finally,

- *IRSN STLOC program proposal regarding LOCA will be summarized*

Pending Issues : Fuel Relocation



*See references 1 to 4



Consequences

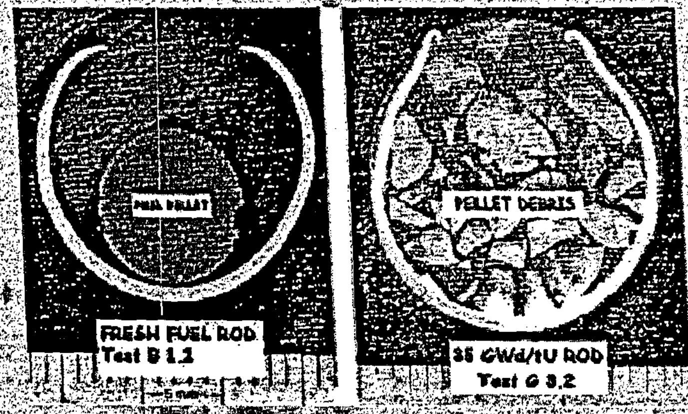
- The increase of power
- The heat transfer modification due to fuel accumulation
- The steam access inside the balloon

Impact on

- Peak Clad Temperature
- Final oxidation ratio and the consequences for quenching and post quench embrittlement
- Hydrogen uptake and the consequences for quenching and post quench embrittlement

- Note that this question is particularly important for end-of-life MOX fuel where power generation is not reduced, unlike for UO2 fuel.

Pending Issues : Fuel Relocation



*See references 1 to 4

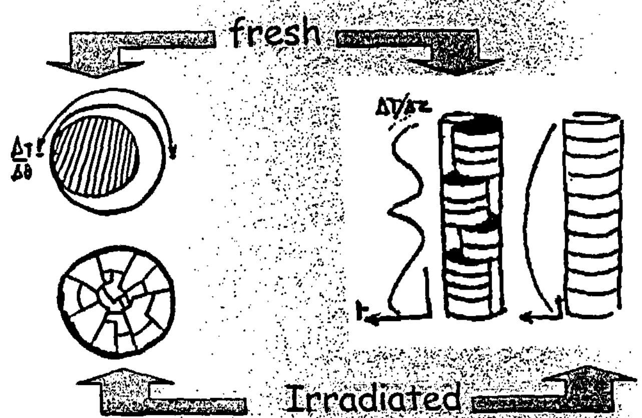
- ✓ Instant of fuel movement at high burnup ?
- ✓ Are they any delay due to fuel-clad bonding ?
- ✓ Filling ratio of clad balloon at high burnup, with fragmentation of UO2 rim or MOX agglomerates ?
- ✓ Fragment sizes ?

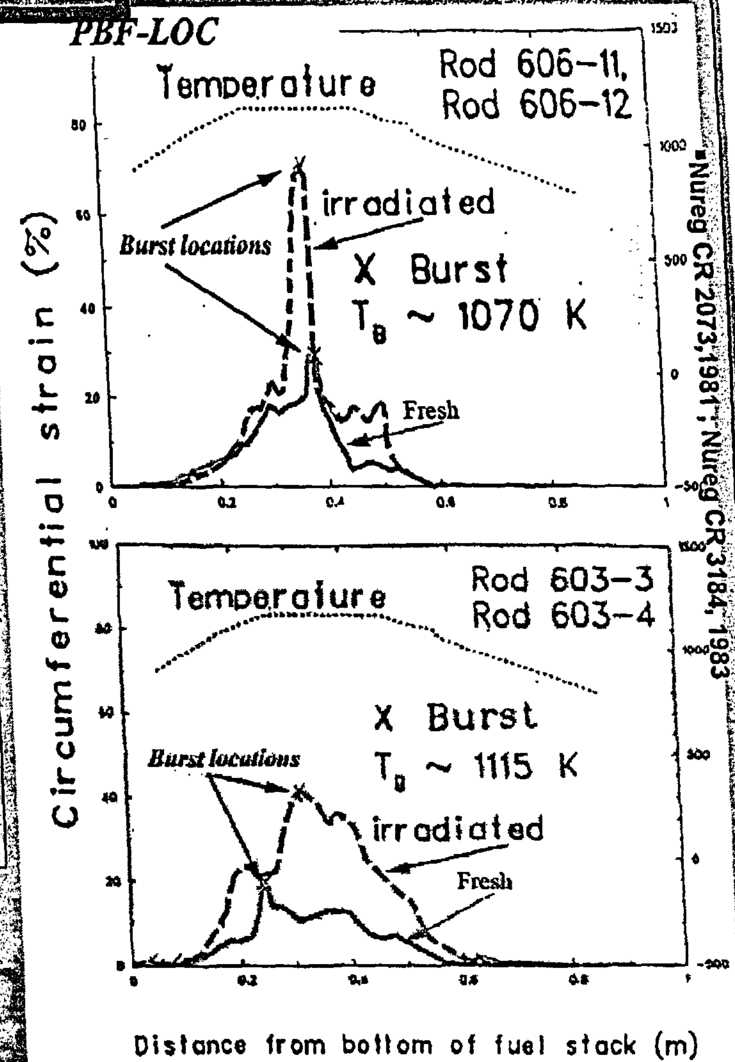
Pending Issues : BUNDLE BLOCKAGE GEOMETRY

▪See references 4 to 6

What is the bundle blockage geometry for irradiated bundles ?

Hot and cold points are less likely due to the fuel stack reorganisation during the irradiation.





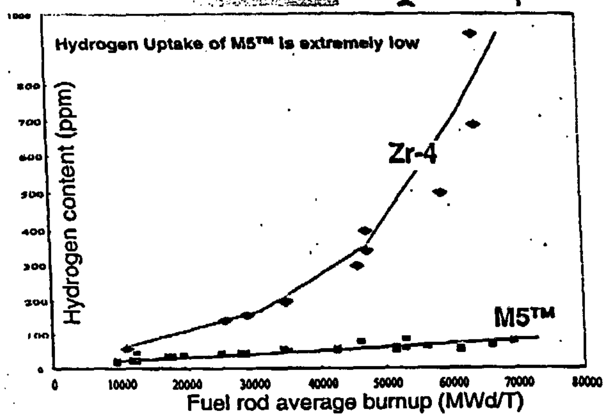
Higher the symmetry bigger the balloons

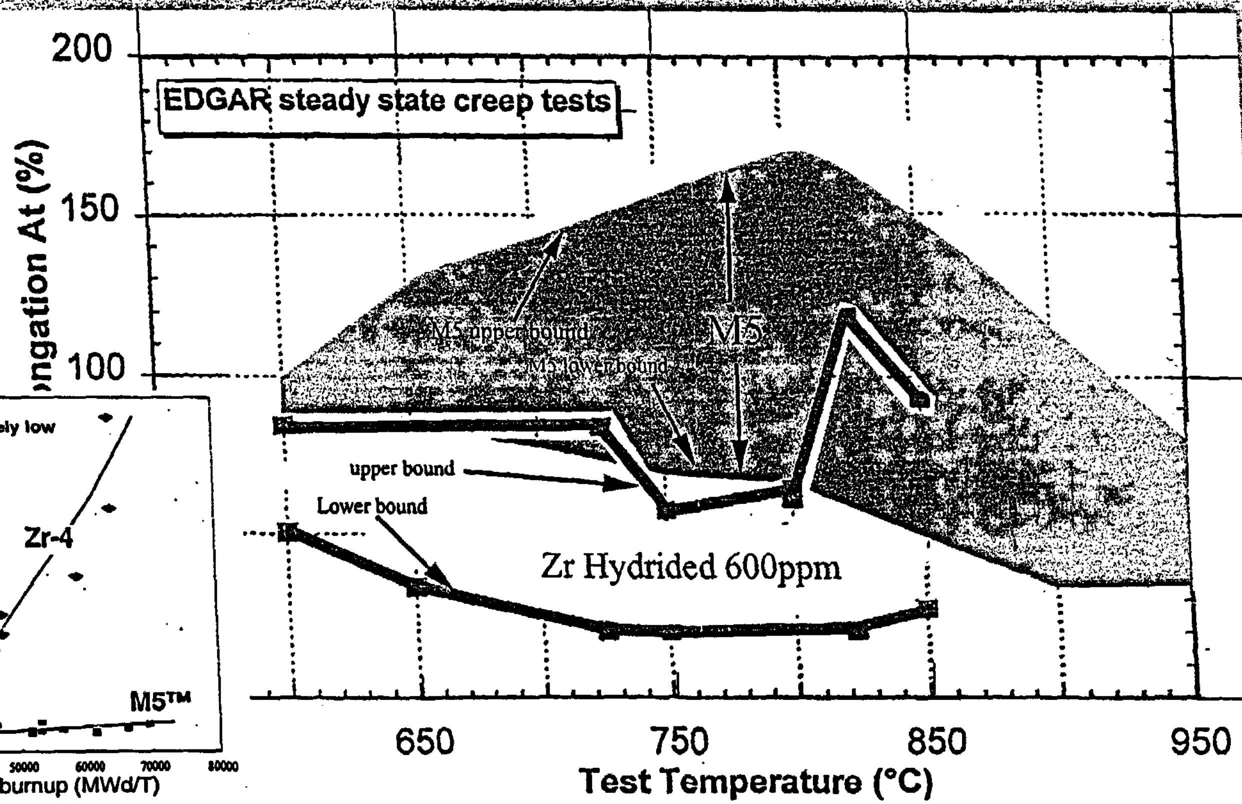
Pending Issues : **BUNDLE BLOCKAGE GEOMETRY**

Corrosion and hydrogen uptakes are lowered for modern clad alloys : ductility is better kept. Lower the corrosion, lower the associated H uptake, better the ductility, bigger the balloons

higher blockage ratio will be likely for modern alloys

▪See references 7 to 9



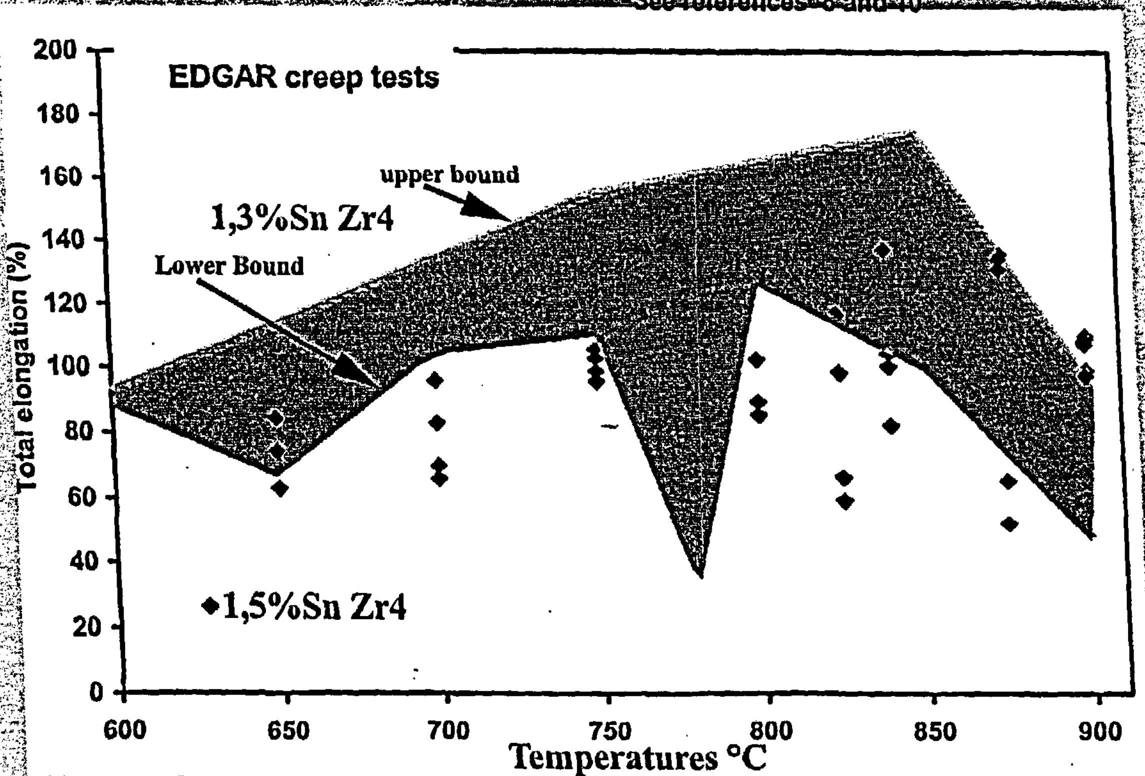


Pending Issues : BUNDLE BLOCKAGE GEOMETRY

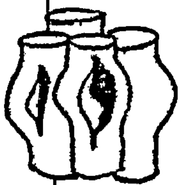
Same Tendency is observed with low tin Zr alloy: lower the tin content, bigger the balloons

higher blockage ratio will be likely for modern alloys

*See references 8 and 10



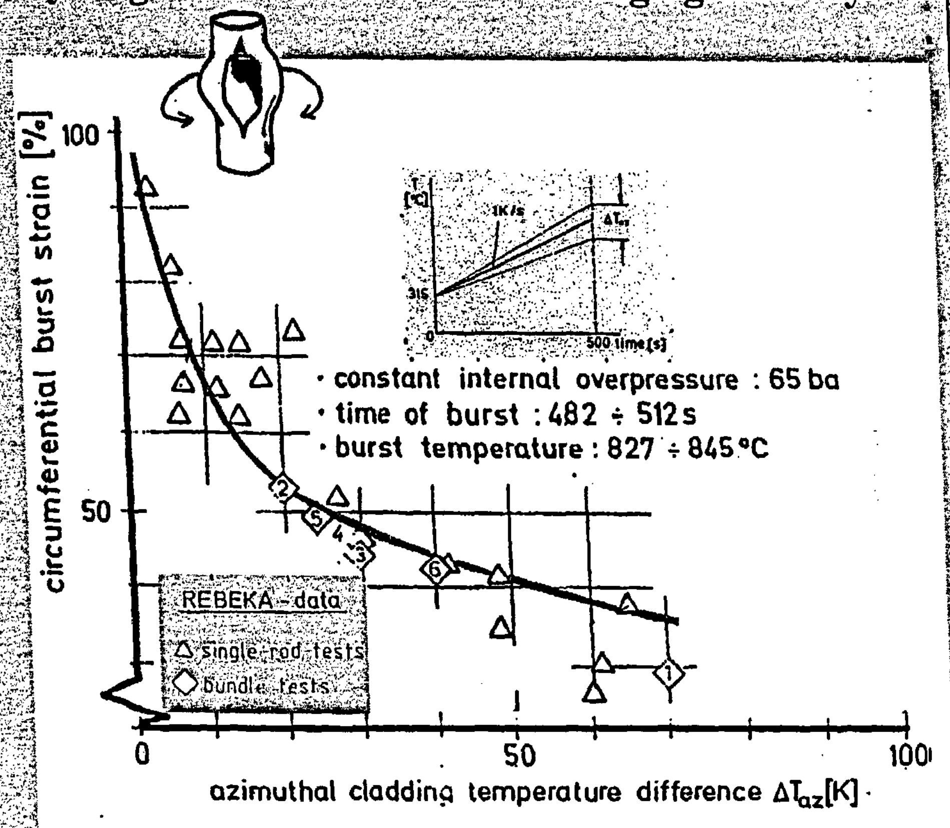
Pending Issues : **BUNDLE BLOCKAGE GEOMETRY**



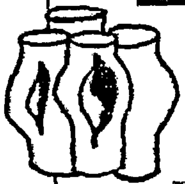
A BUNDLE geometry is necessary to get realistic bundle blockage geometry :

It provides correct azimuthal temperature field around the tested fuel rod

Temperature field correctness is crucial to produce realistic balloon size



Pending Issues : BUNDLE BLOCKAGE GEOMETRY



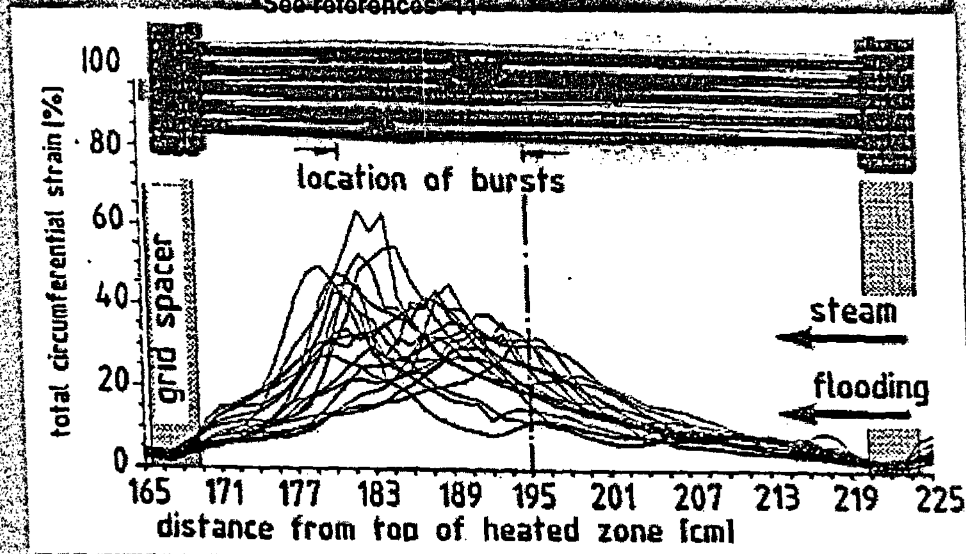
A BUNDLE geometry is necessary to get realistic bundle blockage geometry :

The blockage ratio is not the linear addition of the single rod observations :
N times the single rod balloon geometry will produce unrealistic too high flow blockage.

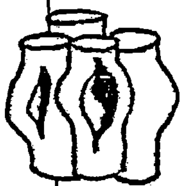
*See references 11

A BUNDLE geometry

provides the correct no coplanar balloons distribution and a lower blockage ratio

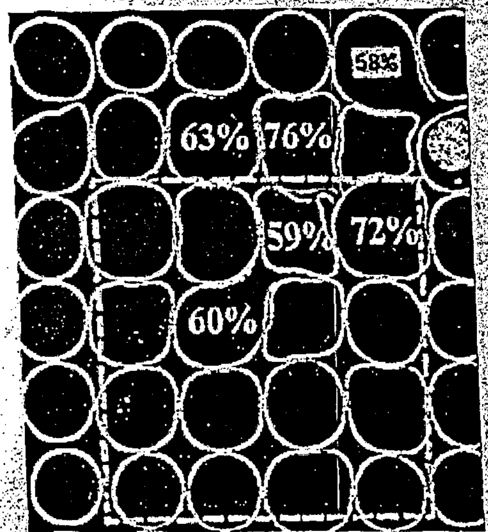


Pending Issues : BUNDLE BLOCKAGE GEOMETRY



A BUNDLE geometry is necessary to get realistic Bundle Blockage geometry :

Radial interactions between adjacent fuel rods impact the balloon size and the bundle Blockage geometry



need for row(s) of guard rods to ensure representative thermal and mechanical interactions with neighbour rods as recommended in 1981 by J. Broughton (INEL), in consideration of ORNL MRBT B5/B3 experiments)

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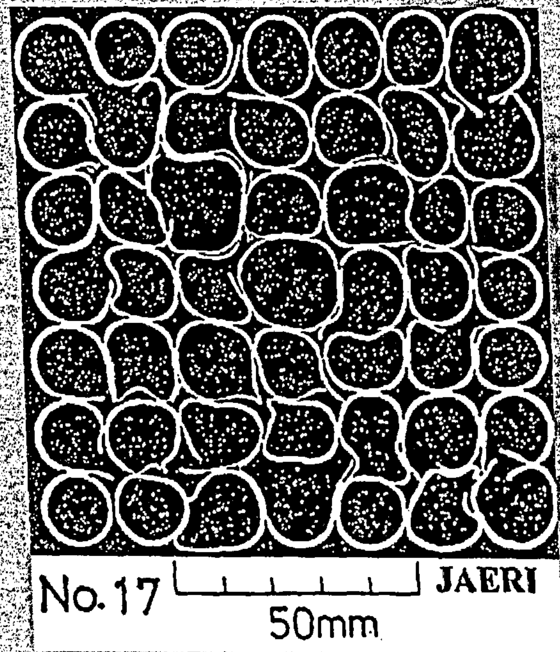
Pending Issues : HEAT SOURCE ACCUMULATION

What is the maximum bundle blockage ratio that leaves coolable an irradiated rods bundle ?

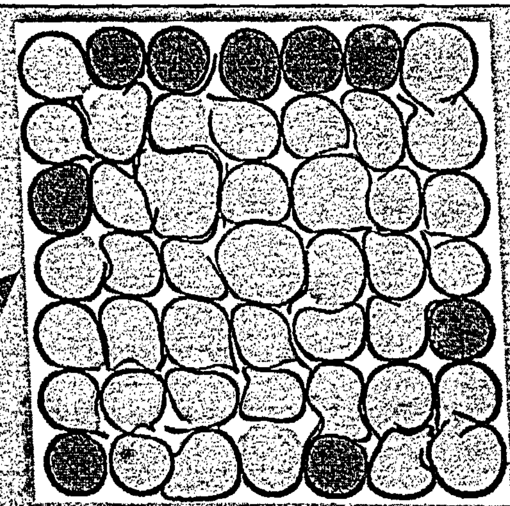


▪See references 12

The 90% value derived from results of flooding experiments (FLECHT-SEASET-SEFLEX-FEBA et al) on unirradiated rods arrays is questionable since these experiments ***did not take account of any fuel relocation and associated effects (lineic and surfacic power increase, gap reduction)***

Pending Issues :

BLOCKAGE COOLABILITY

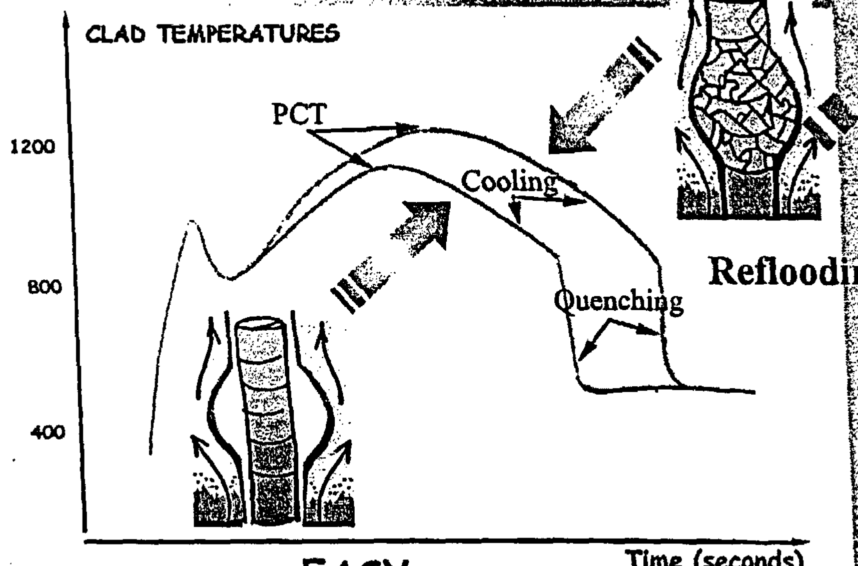


▪See references 12

NOT SO EASY

REAL SITUATION

Reflooding requires to quench the cladding and a part of the fuel column



EASY

FLECHT-SEASET-SEFLEX-FEBA et al

Reflooding requires to quench the cladding only

Pending Issues :

QUENCHING

Are they any needs to modify the criteria related to post quench ductility if, during quenching, additional mechanical stresses are induced by fuel rods blockage in the assembly ? See reference 14

Cooling

Which is the residual ductility of the cladding after quenching ? See references 13

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... FOR LOCA RESEARCHES

	AVAILABLE OR PLANNED PROGRAMS
TO DERIVE CRITERIA	
Post Oxidation ductility	ANL
Unconstraint Quenching	TAGCIR HYDRAZIR CINOG
Quenching with Constrained rods	JAERI
TO CHECK THAT CRITERIA ARE NOT VIOLATED FOR REACTOR LOCAs	
Cladding Burst	EDGAR ANL JAERI HALDEN
Relocation	HALDEN in pile ANL out of pile
Bundle Blockage Geometry And Heat Source Accumulation	STLOC
Cladding Oxidation	TAGCIR HYDRAZIR CINOG ANL JAERI
Bundle Coolability	If necessary according to STLOC results

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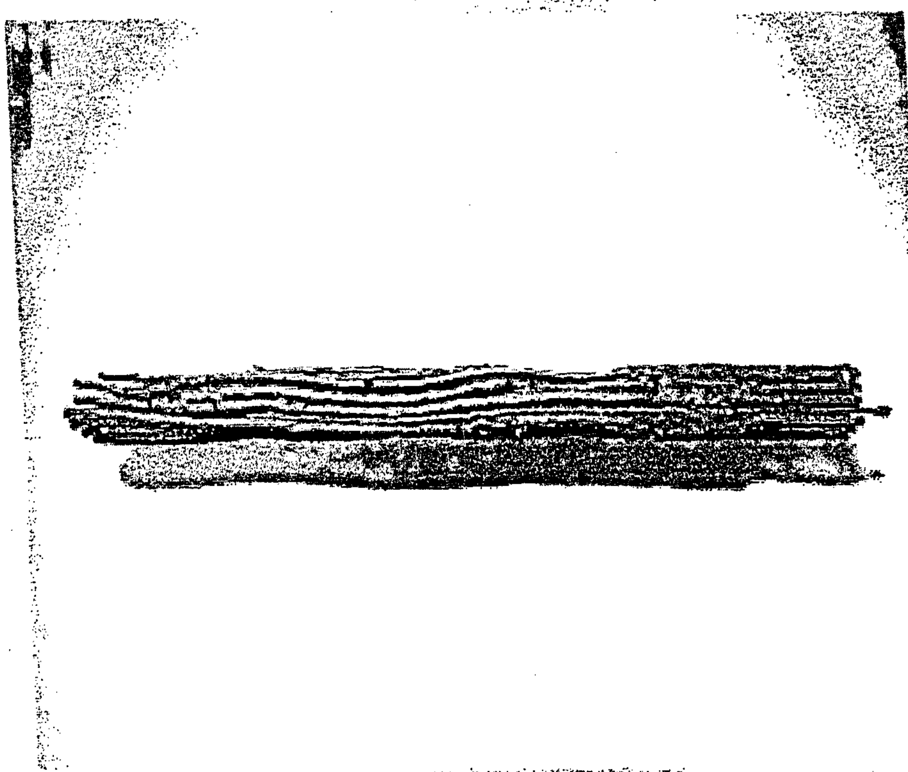
Movie produced from transmission tomography data (Test PHEBUS FPT1)

STLOC



Bundle Blockage Geometry
And Heat Source Accumulation

STLOC

STLOC will be an unique source of data for bundle geometry at burst and the heat source addition induced by relocation. It is also an integral test which includes the major phenomena of the LOCA transient

TWO TESTS IN THE PHEBUS FACILITY

- 9 High burn up rods with a ring of 16 fresh fuel rods
- Temperature ramp from low power under steam condition

A FIRST TEST

Cladding for which a maximum blockage is expected (M5, Low tin)

Fuel UO2 Hight Burnup impact of microstructure (>=60 Gwd/tU local)

A SECOND OPEN TEST (to be defined)

- Cladding Zr low tin / Fuel UO2 end of 1st cycle
- Cladding Zr / Fuel MOX 52 GWd/tM
- Back-up Test for Relocation with bundle geometry

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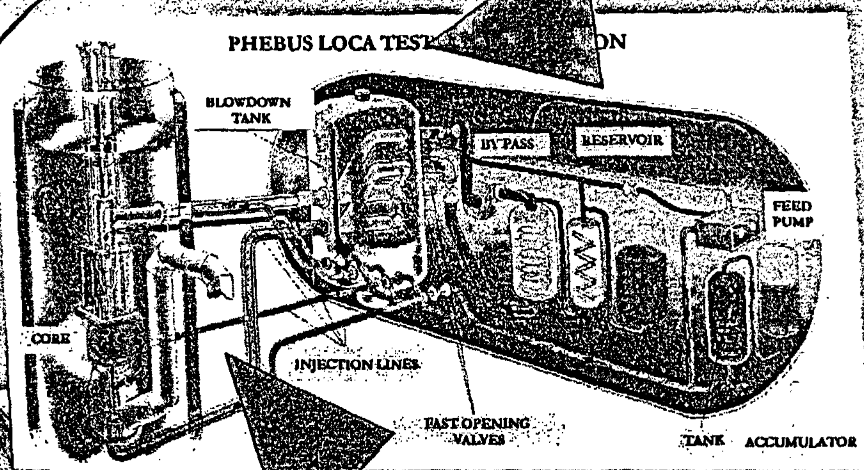
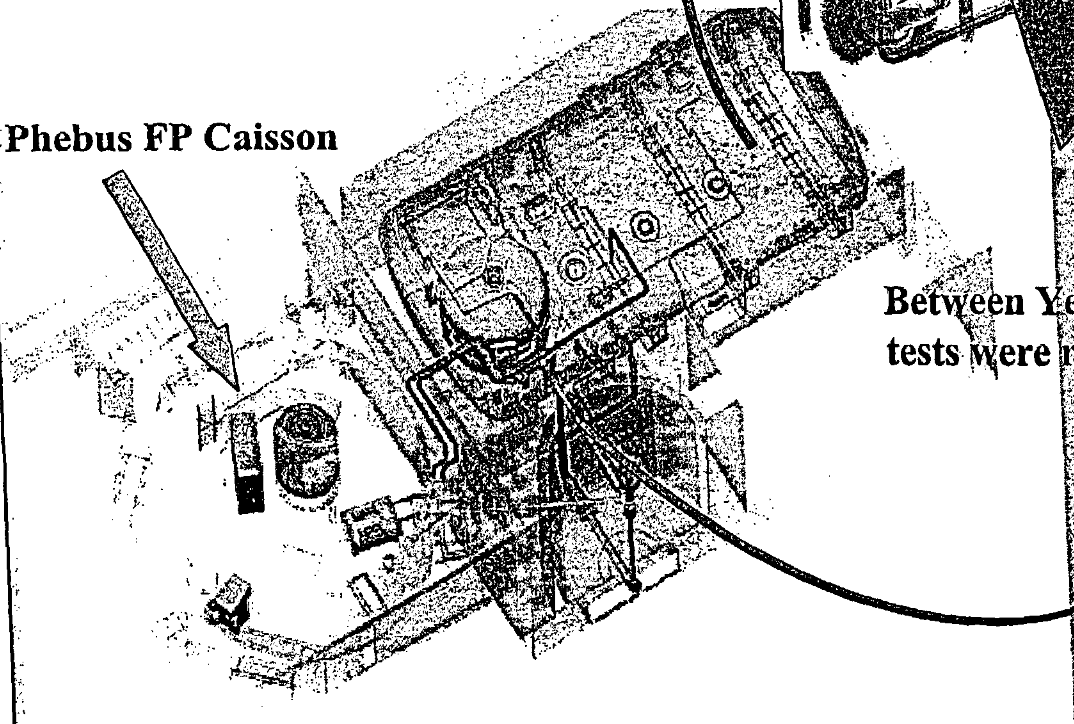
LOCA IN-PILE TESTS

Such a programme should take place in the PHEBUS Facility. By this way IRSN will take advantage of the know-how accumulated when the previous LOCA programme with fresh fuel was run



Phebus FP Caisson



Between Years 76 and 83 20 LOCA bundle tests were run in the PHEBUS Facility

- In-pile investigations at the Phébus facility, J. Duco, M. Réocreux, A. Tattegrain et al, 5th Int. Mtg. Thermal Reactor Safety, Karlsruhe, Sept.10-13,1984
- A study of fuel behavior in PWR design basis accident: an analysis of results from the PHEBUS and EDGAR experiments, M. Réocreux, E. Scott de Martinville, Nuclear Engineering and Design 124 (1990), p363-378

1//LWR fuel rod behaviour in the FR2 in-pile tests simulating the heat-up phase of a LOCA, E. H. KARB et al, KfK 3346- March 1983

2//Relocation of fuel fragments in ballooned fuel rods. NRC memorandum, M. L Picklesimer, March 30th,1981

3//NRC Generic Safety Issue N° 92, Fuel Crumbling during LOCA, April 1983

4//Behaviour of irradiated fuel during LOCA, G. Hache, First mtg. OECD/NEA/CSNI Task Force on fuel behaviour, 19-20th june,1997, Paris, France

5// PBF LOCA test LOC-6 fuel behaviour report, J. M. Broughton et al, Nureg/CR-3184, April 1983

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