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Fuel Failure Detection and Location in LMFBRs

Proceedings of an International Atomic Energy Agency Specialists' Meeting

International Working Group on Fast Reactors

Karlsruhe, Federal Republic of Germany, May 11-14, 1981

Editor: S. Jacobi
Institut für Reaktorentwicklung

Kernforschungszentrum Karlsruhe

KERNFORSCHUNGSZENTRUM KARLSRUHE
Institut für Reaktorentwicklung

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Kernforschungszentrum Karlsruhe GmbH, Karlsruhe

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Organization

Meeting Chairman:	S. Jacobi Kernforschungszentrum Karlsruhe Institut für Reaktorentwicklung
Co-Chairman:	W. Glauner Kernforschungszentrum Karlsruhe Projekt Schneller Brüter
Scientific Secretary of IWGFR:	V.F. Efimenko International Atomic Energy Agency, Vienna
Recording Secretaries:	G. Schmitz Kernforschungszentrum Karlsruhe Institut für Reaktorentwicklung
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CONTENTS

SUMMARY, CONCLUSIONS AND RECOMMENDATIONS

INTRODUCTION

The Specialists' Meeting on "Fuel Failure Detection and Location in LMFBRs" was held at the Kernforschungszentrum Karlsruhe, Federal Republic of Germany, on 11-14 May 1981. The meeting was sponsored by the International Atomic Energy Agency (IAEA) on the recommendation of the International Working Group on Fast Reactors (IWGFR). It was presided over by S. Jacobi (General Chairman) of the Federal Republic of Germany and was attended by 28 participants from Belgium, the Federal Republic of Germany, France, India, Japan, the Union of Soviet Socialist Republics, the United Kingdom, the United States of America and two international organizations, CEC and IAEA.

The purpose of the meeting was to review and discuss methods and experience in the detection and location of failed fuel elements and to recommend future development. The technical sessions were divided into five topical sessions as follows:

1. Reactor Instrumentation
2. Experience Gained from LMFBRs
3. In-pile Experiments
4. Models and Codes
5. Future Programs

During the meeting papers were presented by the participants on behalf of their countries or organizations. Each presentation was followed by an open discussion in the subject covered by the presentation. After the formal sessions were completed, a final discussion session was held and general conclusions and recommendations were reached. Session summaries, general conclusions and recommendations, the agenda of the meeting and the list of participants are given below.

SUMMARY

Session 1. Reactor Instrumentation

Following common subjects were covered in presentations of specialists from DeBeNe-countries, France, India, Japan, the Union of Soviet Socialist Republics, the United Kingdom and the United States of America:

1. Role of failed fuel detection systems as safety systems and monitoring systems

2. Design considerations for DND and cover gas monitoring systems, state-of-the-art of cover gas monitoring systems

3. Location techniques and systems (sodium sampling and gas tagging)

4. Inspection methods

5. Calibration methods

At the present time every LMFBR has a fuel failure detection and location equipment to achieve a high availability and it is recognized that DND monitoring is necessary for the safe operation of an LMFBR plant.

The Ge-Liγ-spectrometer is considered to be the most adequate system for cover gas activity measurements, and, from the point of view of French specialists, the further development of instrumentation for cover gas activity measurements does not need large efforts.

Different views were stated by specialists of different countries on the purpose of cover gas activity measurements. French specialists believe that the purpose is contamination measurements rather than characterization of fuel failures. Japanese specialists consider these measurements as a means for failure characterisation and for preventing serious consequences. German specialists emphasize the importance of the work on detection, location and characterisation of fuel failures during early stage of LMFBR programme development when there is insufficient experience of operation LMFBRs with defected fuel pins.

It was acknowledged by the participants of the meeting that DND monitoring is an efficient tool to detect and to follow the evolution of a cladding failure. Investigations of this method are carried forward in many countries.

Three methods are available now for location of failed fuel subassemblies: (1) sampling systems; (2) gas tagging and (3) sipping systems. The first two methods are well adapted to the removal of a failed fuel subassembly as soon as the DND signal reaches the determined level. The last method should be used if the reactor operation with several DND failures in the core is accepted. In this case a sipping system enables the contamination of the primary circuit to be limited.

The following items were agreed to be important for future international information exchanges:

1. Requirements for parameters of DND and cover gas monitoring systems (response time, sensitivity, scram/alarm levels)

2. Criteria for selection of various methods of cover gas monitoring

3. Upgrading sensitivity for DND systems

4. Performance of location systems used during a reactor operation (gas tagging, sodium sampling)

5. Performance of wet/dry sipping methods used during reactor shutdown

6. Consideration of cost reduction for location systems

7. Calibration methods

Session II. Experience Gained from LMFBRs

The discussion indicated that in the reactor under operation the DND and fission gas detection systems provide sufficient information about the moment the failure occurs, the number of failures, the failure type (gas leak failure or failure with fuel-coolant contact)*, the failure evolution and, approximately, the surface area of fuel-coolant interaction.

This information enables the primary circuit contamination by fission products to be estimated provided that well developed calculations methods and experimentally determined fission products release fractions are used. On the other hand, up to now there is no satisfactory information about the fuel release.

Practically all countries have an experience of operation with failed fuel pins, however, the questions of localising, characterisation and possible further irradiation of failed pins are also subjects of interest.

The fuel failures accompanied by the fission gas release were detected by the gas activity measurement systems and by DND systems. In most of the cases fission gas montiors indicated fuel failures earlier than DND monitors. The time of delay of the DND-signal in comparison with the fission gas detection depends on the composition of the fuel, the fuel element construction, irradiation manner (especially burn-up) and it is within the range from several hours to several weeks.

In the field of location of failures the tagging methods and the DND-location systems during reactor operation and dry sipping methods after reactor shutdown are considered as very successful.

The characterization of fuel pin failures is possible to some extent, but additional information from in-pile experiments and from reactor operation experience is needed. Two criteria for taking a decision about the reactor shutdown were presented - fission products contamination level and fuel escape out of the defected pin.

Different kinds of control systems are under development in many countries and the main objectives of such development are further improvements of the efficiency and time response of fuel failure detection systems, used in LMFBRs.

* Short designation: Leaker and DND Failure

Session III. In-Pile Experiments

Loops and reactors are used for experiments with failed fuel pins.

1. On loops:

Japanese specialists use FPL, SIL and CGTF loops for instrumentation qualification required to commission JOYO and MONJU reactors, as well as for the investigation of behaviour of fission products in sodium. Some results obtained can be used for any reactor (for example, temperature dependence of the diffusion of fission gas from sodium to the cover gas).

British, French and German specialists have obtained important results by irradiating defected fresh fuel pins in the SCARABEE loop instrumented with DND monitoring system. Concerning the recoil model, a much larger surface than that of the defect should be considered and the time of delay within the pin should be taken into account.

DeBeNe-specialists used the SILOE loop (in cooperation with CEA) to study the evolution of defective fuel pins in normal operating conditions and MOL loops to study the evolution of pin bundles in accidental conditions. In both cases very high DND signals were reached.

2. On reactors:

From EBR-2 US Specialists obtained results that can be compared with the SCARABEE results: k-factors for recoil model were more than 30 and delay time ("aging") was about 10 sec.

USSR philosophy of defected pins irradiation is related to the contamination consequences of failures. A computerized Ge-Li γ-spectrometer has been developed on BOR-60 for investigation of this problem. Special functioning conditions allow to measure on-line the contamination of sodium coolant by Cs and I isotopes at a level 10^4 to 10^5 less than the background level due to Na24.

In France a new DN counter able to work at a high temperature and γ-level is under development. Its sensitivity is expected to be less than that of the modular systems. Research work is devoted to the study of failure evolution up to very high DND signals and fuel release.

It was agreed that future in-pile experiments will concern the development of instrumentation (towards its simplification and decreasing cost) and the investigations of the evolution of the fuel failures up to massive fuel release.

Session IV. Models and Codes

Five papers were presented in this session: - 'Studies on Modeling to Failed Fuel Detection System Response in LMFBRs' by the Japanese delegates; ' Models and Codes' by the UK delegates; 'Delayed-Neutron Signal Analysis Techniques' by the US delegates; 'Plans MARGO, VOLGA and PARKA' by the French delegates; and 'Fuel Failure Detection and Location in Fast Breeder Reactors' by the German and Interatom delegates. Subjects discussed in the session were diverse in nature and ranged from empirical correlations of DN signal data on the one hand to discussion of a whole-system code for fission-product transport in LMFBR on the other.

Four papers dealt with various ways to interpret DN signals from breached pins, and only one with trying to model the behaviour of fission gas and DN precursors in a whole reactor system, whereby the expected response of detection instruments might eventually be determined. Only one paper attempted to treat real signals from an operating LMFBR, the others being limited to interpretation of signals in small test loops (where, of course, systems effects on signals are probably small and may therefore be ignored). This diversity of approach indicated there is much work yet to be done before signals from breached pins may be interpreted in real time for LMFBRs. Nevertheless, considerable progress has been made world-wide towards such an interpretation.

There was unanimous agreement at this session (and throughout the meeting) that DN precursors release from breached pins with exposed oxide fuel is not by recoil alone, but by a likely combination of recoil, knockout, and radiation-enhanced and thermal diffusion. These mechanisms combine to produce an enhancement of DN signal, or k factor, which is dependent on many local fuel parameters, including roughness, porosity, burnup, temperature and temperature gradient, and the presence of microcracks and sodium reaction products. Derived parameters, the release to birth ratio (R_i/B_i) and the alpha ratio [$-\ln(R_i/B_i)/(\ln\lambda_i)$] have proved useful to interpretation of DN release. Increase in DN signal and defect size on test pins in loops appear correlatable to a distinct change in the alpha ratio. There is hope that such a formalism may eventually be used to indicate the severity or character of a cladding breach.

Once DN precursors and fission-gas isotopes leave a breached pin, their activity which is eventually measured by the detection systems appears critically dependent on their transit path through the primary sodium and, for fission gas, on disengagement from sodium and movement through the cover-gas space. Mixing in both media is important. Modeling of this transport - which is important to determine the required sensitivity of failure detectors, and to set prudent shutdown limits for activity in operating plants - is at an early stage world-wide; it lacks much required data, and is very dependent on the particular reactor plant configuration. This is an area, perhaps, where most future work is needed.

As a major agreement of the session, and as an aid to fruitful exchanges in this area in the future, it was decided that a commonly agreed upon data set for DN-precursor and daughter isotopes should be adopted by all countries represented at this IAEA specialists' meeting - a possible set is attached.

A further proposal to aid a meaningful exchange of future information is to have a similar method used in all countries for the calibration of DN monitors. That is the use of a common type of recoil source of an agreed-upon composition or nature.

Session V. Future Programmes

Future activity should be aimed at the improved safety and performance of fast reactors.

Failed pin can potentially release fuel and it is necessary to know the effect of this on the reactor system, e.g. this could lead to flow reduction in the fuel bundle or to contamination of the circuit - the importance of this in the circuit needs further study. Also failed fuel can release fission products which could eventually cause significant contamination of the reactor circuits and thus complicate maintenance procedures. As fuel failures are to be expected from time to time, most of which will be innocuous, it is important that simple failures should be readily distinguished from these likely to cause problems. Further work is required to enable the in-reactor signals from failures to be known and characterised. This must be done statistically from a large data base using results both from in-pile loops experiments and from the reactors operation experience. To support and interpret those studies the development of models should be continued involving both theoretical and supplementary experimental work on fission product release taking into account chemical interactions of fuel, coolant, fission products and cladding. Also transport of fission products in accident condition requires further study.

This work is particularly important to assist in determining reactor operating policies, i.e., whether or not failed fuel should be removed immediately or at some future date.

Development of instrumentation to enable failed fuel detection and location to be made cheaper, simpler and more reliable is to be encouraged.

Systems for particular reactor designs must be developed to meet engineering considerations and sampling must be such that representative samples are obtained quickly and allow all reactor sub-assemblies to be adequately monitored.

The discharge of failed fuel has important economic aspects thus the operating policies must take this into account.

General Conclusions and Recommendations

1. It was recognized that the problems of the failed fuel characterization and the failed fuel management are connected with other fields and specialists in these fields should be involved to solve the problems. The evolution of the cladding failure, and methods for direct measurements of fuel release into the coolant could be investigated in cooperation with specialists on sodium chemistry and metallurgy. The policy of operating reactors with failed fuel should be discussed not only by specialists on fuel failure detection and location, but also by specialists on reactor safety, economics, operation and management, fuel performance.

2. It was recommended that a commonly agreed upon data set for DN precursor and daughter isotopes should be adopted by all countries represented at this specialists' meeting. A possible set was proposed by US specialists.

3. A fruitful exchange of the future information obtained in different countries could be organized if a standard method were used in all countries for the calibration of DN monitors, including using of a common type of a recoil source of an agreed upon composition or nature.

4. It was recommended by the participants of the meeting that a specialists' meeting on "Contamination Problems in LMFBRs including Contamination by Fuel" should be sponsored by the IWGFR in the near future.

5. It was also recommended that the specialists' meeting on "Fuel Failure Detection and Location and a Strategy of Failed Fuel Management in LMFBRs" be sponsored by the IWGFR in three or four years' time.

POSTSCRIPT

In accordance with the subjects of the meeting the participants elaborated during the Sessions I to III the following tables. Although they are tight connected to the items of the three sessions they are subtracted from the presentations and presented on the next pages as a further summarized result of the meeting.

TABLES

TABLE 1. SURVEY OF FAILED FUEL DETECTION INSTRUMENTATION

Related to Session I: Reactor Instrumentation

	DND transit time and sensitivity			Location					Scram
Reactor	Main pipe	By-pass	spec.pipe work	Na-probe	Gas strip.	Dry sipping	Wet sipping	other	Autom. DND scram
Rapsodie	–	6–16 s		7 subass.	no	Exp.		no	yes
Phenix	–	no	~30 s	yes	yes	no	no	no	yes
SPX	–	no	15 s	15 s	no	no	no	no	yes
KNK II	22 s 22 cps/cm^{2}R	no	no	no	no	yes	Exp.	no	yes
SNR 300	–	no	29 s 2 cps/cm^{2}R	no	no	no	yes	no	yes
FBTR	–	20–30 s	no	no	no	no	no	no	yes
JOYO	~44 s	no	no	no	no	no	Exp.	no	no
MONJU	<60 s	no	no	no	no	no(?)	yes(?)	no	yes
DFR	–	~20 s	no	no	no	no	no	no	no
PFR	–	no	25 s	26 s	yes	no	no	no	yes
CDFR	–	no	~60 s	20 s	yes	no	yes(?)	no	yes
EBR-II	–	18–20 s 16 cps/cm^{2}R	no	no	no	yes(2h)	no	no	no (administr. limit)
FFTF	27–28 s	no	no	no	no	no	no	no	no
CRBR	30–40 s	no	no	no	no	no	no	no	no
BR 10	–	no	no	no	no	yes	yes	no	no (adm. lim.)
BOR-60	~30 s 3400 cps/g (U-235)	no	no	no	no	yes	yes	no	no (adm. lim.)
BN-350	30 s	no	no	no	no	yes	yes	no	no (adm. lim.)
BN-600	–	~60 s	no	no	no	yes	yes	no	no

Cont. Tab. 1

Reactor	Cover gas									Blanket elements
	Ge or Ge(Li)	Variable sensit.	Gaschro-matogr.	Ion. chamb.	Mass-spectr.	Preci-pit.	Scint. counter	Tagging	Xe-adsorp.	
Rapsodie	no	Volume	yes	yes	yes	no	no	yes	no	DND?
Phenix	yes	Volume	yes	yes	yes	no	no	part.	no	DND?
SPX	yes		no	yes	no	no	no	no	no	DND?
KNK II	Exp.	Geom./Coll.	Exp.	no	no	Exp.	yes	no	Exp.	DND?
SNR 300	yes	yes	no	no	no	no	no	no	no	DND?
FBTR	no	Volume	yes	yes	no	no	no	no	no	DND?
JOYO	Exp.	?	no	yes	no	yes	no	Exp.	Exp.	DND?
MONJU	yes(?)	Volume	no	yes	yes	yes	yes(?)	yes	no(?)	DND?
DFR	no	Volume	no	no	no	no	yes	part.		
PFR	yes	Distance	no	no	no	yes	no	part.	no	DND
CDFR	yes	yes?	no	no	no	yes	no	part.	no	DND
EBR-II	yes	Vol./Geom.	no	no	yes	no	no	yes	yes	DND?
FFTF	yes	Abs./Coll.	yes	no	yes	no	no	yes	yes	no
CRBR	yes	Abs./Coll.	yes	no	yes	no	no	yes	yes	yes (Tag)
BR 10	yes	Coll.	no	yes	no	yes	yes	no	no	no
BOR-60	yes	Coll.	no	yes	no	yes	yes	no	no	no
BN-350	yes	Coll.	no	yes	no	yes	yes	no	no	no
BN-600	yes	Coll.	no	yes	no	yes	yes	no	no	no

TABLE 2. EXPERIENCE WITH FAILED FUEL ELEMENTS

Related to Session II: Experience from LMFBRs

Reactor	From Leaker to DND Failure						
	Number	Detected by	Leaker Phase min	Leaker Phase max	DND Phase min	DND Phase max	Geom. Area (cm^2) Recoil Area (cm^2R)
Rapsodie	50 failed subass.	Gas and DND	hours	< month	some hours	some days	
Phenix	> 4	Gas and DND	hours	< month	some hours	some days	
KNK II	2	Gas and DND	1.5 h	30 d	min.	18 d	$\leq$ 3.2 cm^2R
JOYO	no	no	no	no	no	no	no
DFR	?		-			10 months	
PFR	2 Exp.	Gas and DND	-	1 month	1 month	1 month	1 cm^2
EBR-II	~ 20	Gas, Tagging	hours	days	-	-	
BR-10	several	Gas and DND	hours	days		days	
BOR-60	10 %	Gas and DND	10 d	40 d	1 d	40 d	
BN-350	non	-	-	-	-	-	-
BN-600	non	-	-	-	-	-	-

Cont. Tab. 2

Reactor	From Leaker not to DND Failure				DND Failure without Leaker Phase					Repair
	Number	Detected by	Time min	Time max	Number	Detected by	Time min	Time max	Geom.Area Recoil Ares	
Rapsodie	20	Gas	?	?	2	DND	0	some hours at low power	-	yes, sometimes no
Phenix	> 2	Gas	-	several months	0	-	-	-	-	no
KNK II	none	-	-	-	none	-	-	-	-	no
JOYO	none	-	-	-	none	-	-	-	-	-
DFR	-	-	-	-	-	-	-	-	-	-
PFR	none	-	-	-	none	-	-	-	-	-
EBR-II	90 - 100	Gas + Tagging	days	months	~ 2	Gas + Tagging	hours	days	-	yes (2)
BR-10	?	-	-	-	?	-	-	-	-	-
BOR-60	> 100	Gas	-	up to 2 months	several	DND	-	-	-	-
BN-350	10	Gas	-	months	none	-	-	-	-	-
BN-600	none	-	-	-	none	-	-	-	-	-

TABLE 3. TEST FACILITIES
Related to Session III: In-pile Experiments

Country	Name of the facility	Abbr.	Location, Reactor	open closed	No. of pins (min/max)
France	exp. defect. fuel	-	Rapsodie	closed	1
	"	-	Phenix		1
	"	Scarabee	-		37
	"	Eros	-		
	"	Volga	-		
DEBENE	Thermopump	-	SILOE	closed	1
	Mol 7C	-	BR 2	closed	37
	exp. open fuel	-	KNK II	open	now 1
India			none		
Japan	Fission Product Loop	FPL	Toshiba Training Reactor TTR	closed	UO_2 balls
	Cover Gas Monitor Test Facility	CGMTF	JRR-3	open	
UK	Prototype Fast Reactor	PFR		open	19
USA	Breached Fuel Test Fac.	SLSF	EBR-II	-	9/37
	Fuel Performance Test Fac.		EBR-II	-	9/37
	Closed Loop Irradiation Ass.		FFTF	-	37
USSR	BR-10, BOR-60 and BN-350 also used for in-pile experiments				

Cont. Tabl. 3

Country	Surveillance Na/cover gas	Main Objectives
France	yes/yes	evaluation of exp. and nat.
	"	clad failures
	"	evaluation of accidental bundles
DEBENE	yes/yes	allround
	yes/no	DND
	yes/yes	allround
India	-	-
Japan	yes/yes(Ar)	to study fission product deposition
	no/yes(He)	to test cover gas monitors of various kinds
UK	yes/yes	
USA	yes/yes	to test monitors to understand signals, DND and fuel loss

TABLE 4. ABBREVIATIONS

ANL	=	Argonne National Laboratory, Argonne, Illinois
CEA	=	Commissariat a l'Energie Atomique
HEDL	=	Hanford Engineering Development Laboratory, Richland, Washington
IA	=	INTERATOM, Bergisch Gladbach, Germany
IAEA	=	International Atomic Energy Agency, Vienna
JAERI	=	Japan Atomic Energy Research Institute, Tokai-mura
KBG	=	Kernkraftwerk-Betriebsgesellschaft mbH, Karlsruhe
KfK	=	Kernforschungszentrum Karlsruhe
PNC	=	Power Reactor and Nuclear Fuel Development Corporation, Tokyo
RRCK	=	Reactor Research Centre Kalpakkam, India
SBK	=	Schnellbrüter Kernkraftwerksgesellschaft mbH, Kalkar, Germany
SCK/CEN	=	Studiecentrum voor Kernenergie/Centre Belge d'Etudes Nucleaires, Mol, Belgium
SRIAR	=	Scientific Research Institute of Atomic Reactors, Dimitrovgrad
UKAEA	=	United Kingdom Atomic Energy Authority

AMERICAN ACRONYMS

BFTF	-	Breached-Fuel Test Facility
CGCS	-	Cover-Gas Cleanup System
CGMS	-	Cover-Gas Monitoring System
CRBRP	-	Clinch River Breeder Reactor Project
DAS	-	Data Acquisition System
EBR-II	-	Experimental Breeder Reactor II
FERD	-	Fuel-Element Rupture Detector
FFTF	-	Fast Flux Test Facility
FFR	-	FERD-Flow Reduction (Test)
FPTF	-	Fuel Performance Test Facility
GLASS	-	Germanium-Lithium Argon Scanning System
OLDU	-	On-Line Distillation Unit
OLMS	-	On-Line Mass Spectrometer
RAPS	-	Radioactive Argon Processing System

A G E N D A

Monday, 11. May 81

8.30	Bus from the hotels to KfK (every following day also)	
9.00	Registration	
9.30	Welcome	W. Marth, KfK, Head Fast Breeder Project
9.45	Opening remarks	V.F. Efimenko, IAEA
10.00	"A REVIEW" and Approval of the agenda	S. Jacobi, KfK
	Technical remarks	W. Glauner, KfK
10.30	Coffee Break	
10.45	Session I: Reactor Instrumentation	Sess. Chairman: T. Miyazawa, Toshiba Vice-Chairman: F.E. Holt, HEDL

- Techniques and equipment for measuring fission products in reactors
- Methods for detection, localisation and characterisation of failed fuel elements
- Sensitivity of the monitors, alarm and scram levels
- In-pile calibration methods

Contributions from France, Germany and India with discussions

12.30	Lunch
14.00	Continuation of Session I
	Contributions from Japan, UK, USA, USSR with discussions
15.30	Coffee break
16.30	Session discussion
17.30	Reception at the KfK Casino
20.00	Bus to the hotels

Tuesday, 12. May 81

9.00 Session II: Experience from LMFBR´s

Sess. Chairman:
E.K. Yakshin, SRIAR

Vice-Chairman:
M. Relic, IA

- Results from measurements of existing and applied instrumentations to detect, localize and characterize pin leakers and open fuel failures
- Behaviour of fuel elements during these measurements
- Failure statistics, strategy for continued operation in respect to minimize shut-off-time

  with contributions from all participating countries (about 15´) and short discussions, starting with Germany

10.30 Coffee Break

12.00 Session discussion

12.30 Lunch

14.00 Session III: In-pile experiments

Sess. Chairman:
P. Michaille, CEA

Vice-Chairman:
E. Sakai, JAERI

- Research and development work on instrumentation and detection methods
- In-pile experiments and related fission product and fuel release results
- Signal interpretation
  with contributions of all participating countries (about 15´) and short discussions, starting with India

15.30 Coffee Break

16.30 Session discussion

17.00 Bus to the hotels

Wednesday, 13. May 81

9.00	Session IV: Models and Codes	Sess. Chairman: J.D.B. Lambert, ANL Vice-Chairman: D.B. Sangodkar, RRCK

- Modelling and code development of fission product release and transport in loops and reactor plants
- Explanation of DND-enhancement factor k
- Application of these codes to LMFBR´s

with contributions from all participating countries (about 15´) and short discussions, starting with Japan

10.30	Coffee Break
12.00	Session discussion
12.30	Lunch

14.00	Session V: Future Programs	Sess. Chairman: D.K. Cartwright, UKAEA Vice-Chairman: C. Berlin, CEA

- Research and development programs related to the session I to IV
- Objective and description of national and international programs
- Remaining questions

with contributions from all participating countries (about 15´) and short discussions, starting with United Kingdom

15.30	Coffee Break
16.30	Session discussion
17.00	Bus to the hotels

Thursday, 14. May 81 Session VI: Conclusions

9.00	Summary and recommendation from Session I	T. Miyazawa
9.20	Summary and recommendation from Session II	E.K. Yakshin
9.40	Summary and recommendation from Session III	P. Michaille
10.00	Summary and recommendation from Session IV	J.D.B. Lambert
10.20	Summary and recommendation from Session V	D.K. Cartwright
10.40	Closing of the meeting	S. Jacobi
11.00	Bus-tour through the Kernforschungszentrum	
12.00	Lunch	
13.30	Visit of KNK II	

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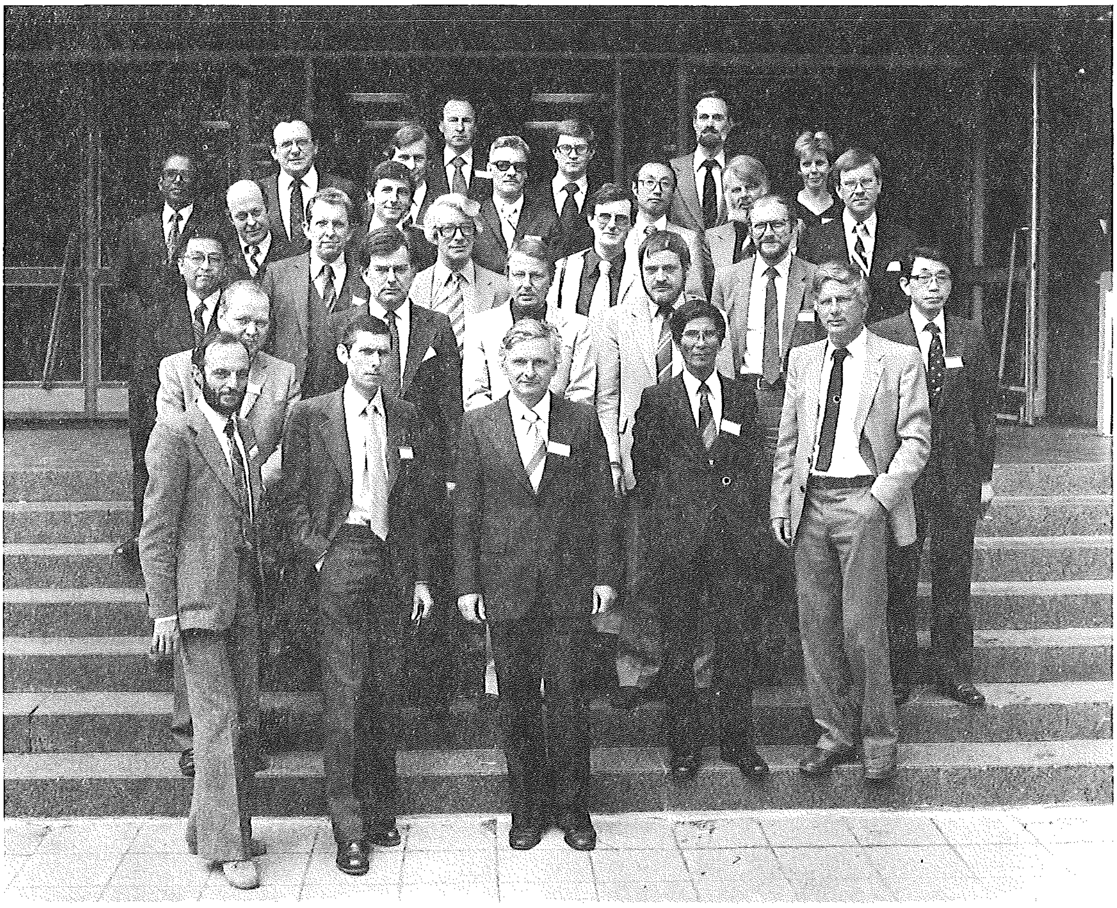

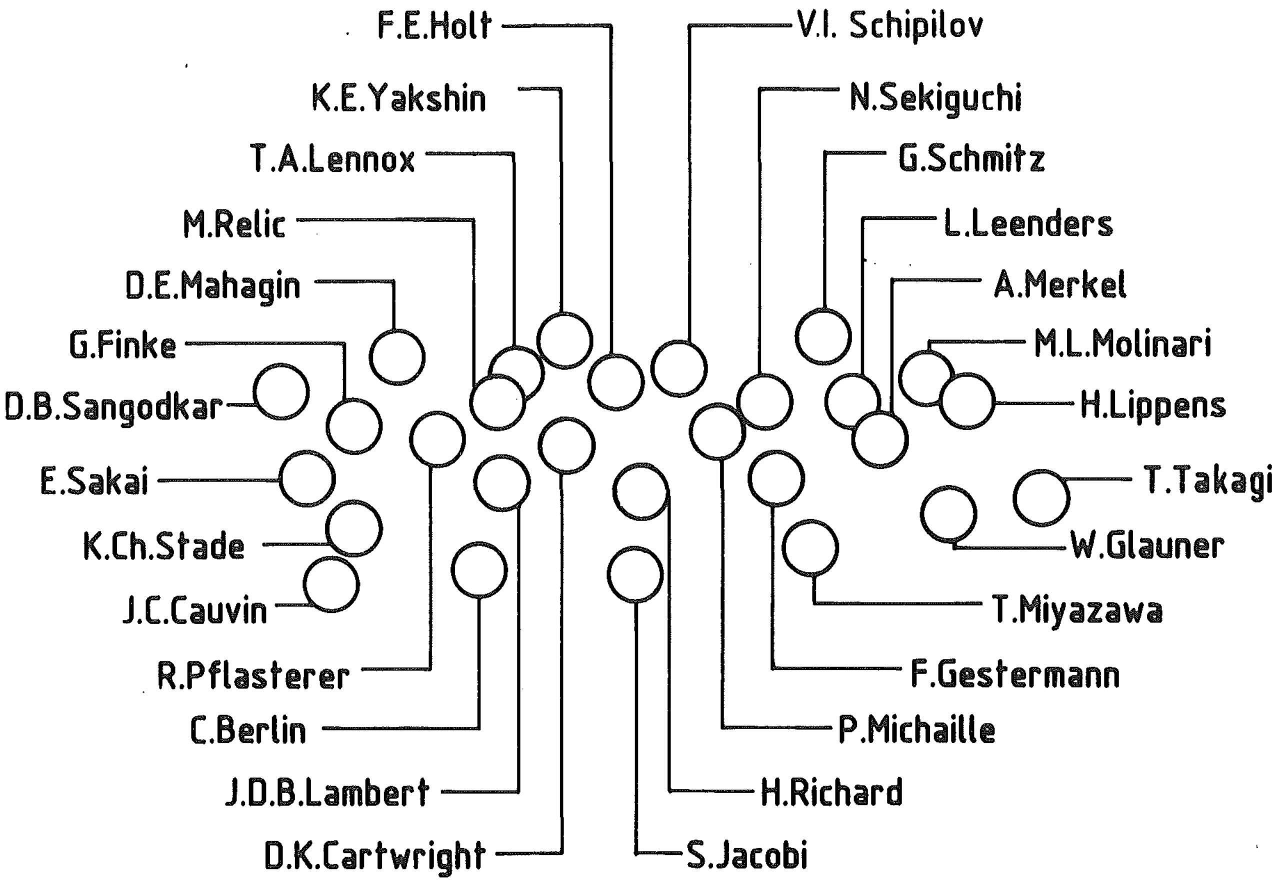
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DETECTION AND LOCATION OF FAILED FUEL ELEMENTS IN LMFBRS
A REVIEW

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AN EXPANDING FIELD OF WORK

Many years ago in the early days of reactor engineering the topic of "Failed Fuel Detection and Location" had been dealt with in most countries by just a few research workers who hardly maintained international contacts. Although, at the time, activities in this field had been considered as necessary already, they were not very much appreciated everywhere: They pointed at failures, more particularly fuel element failures, and these very failures should actually be avoided. Today, the situation is completely reversed. In every country engaged in the development of breeder reactors more or less large working groups have been established, extensive out-of-pile and in-pile experimental programs and intensive international contacts exist, in most cases provided for by agreements. Which have been the reasons for this course of development?

Cladding Tube Failures are Unavoidable

The finding has been generally accepted that cladding tube failures are unavoidable despite careful fabrication and testing methods and strictly specified conditions of operation. This means that cladding tube failures must be taken into account when designing a reactor, i.e.,

- devices must be available for detection of a cladding tube failure
- devices must be available for locating the failed fuel element and
- these devides or additional ones must allow to characterize the failure.

In this work literature references have been omitted intentionally because their number would necessarily be extremely high. The reader is referred to the following technical papers.

Number of Fuel Elements

The first small prototype reactors accommodated about 2 to 3 dozens of fuel subassemblies. The present goal of breeder reactor development is 1000 to 1300 MWe with up to 360 fuel subassemblies. Given this great number alone, certain precautions as regards failed fuel elements must be taken.

Rod Power and Burnup

Above all for reasons of better economy, the specific rod power and the burnup were increased continuously. This necessarily implied several higher loadings on the cladding tubes. On the other hand, mainly on account of neutron economy, the wall thickness of the cladding tubes had to be kept within limits and could not be increased to comply with the failure probability. Consequently, despite the improvement of the cladding tube quality, no major reduction of the number of cladding tube failures could be expected because of the increased rod power and burnup.

Loss of Fuel

In case of failed pins the fuel might get lost into the coolant, with the following potential consequences:

- Increase in the fission product background signals.
- Background signals approaching the limits of alarm generation.
- Contamination of the primary circuit and fuel, especially with plutonium. This results in more difficult conditions for maintenance and repair.
- Cooling disturbances in the bundle due to deposited fuel particles.

Also these aspects led to an intensification of activities performed in the field of "failed fuel detection".

Contamination with Fission Products

Although unavoidable in general, the emission of fission products from failed pins should be restricted because, eventually, it makes repair and maintenance conditions more difficult, in the same way as a loss of fuel. Also in this case successful detection and characterization of the failure may result in improvements for reactor operation.

Reactor Operation

A sufficiently precise information about the condition of failed fuel elements may lead to substantial advantages for the reactor plant: Each scram implies as serious thermal load on the system and is avoided, if possible, by the operator through slow reduction of the power.

This means the highest possible scram level and reliable alarm signaling accompanied by the most accurate possible characterization of the damage.

Economy

For reasons of economy and power reserve in the supply network it is the outspoken goal pursued by the utility company that a large plant can be operated until the next or the weekend after next despite damage to the fuel rods. Obviously, this is possible only if the most exact information possible is available concerning the condition of the core.

Safety

Last but not least, the requirements on reactor safety likewise exert a great influence on failed fuel detection and location: A major fuel pin failure may constitute a disturbance giving rise to a dangerous disturbance in cooling. Thus, from those responsible for reactor safety, the requirement was imposed of the quickest possible failure detection. This also implies a considerable extension of the scope of work.

STATE OF THE ART

The reasons indicated above gave rise to a multitude of interesting activities, including a very broad spectrum of different disciplines: reactor design, component technology, reactor operation, fuel fabrication and behavior, measuring technology, electronics and, last but not least, computer technology. A fascinating spectrum which never made work boring, and we always felt the close link to reality and that our activities are really necessary!

Delayed Neutron Detection (DND)

The DND method seems to be a well established one: It has been integrated in one way or the other in each LMFBR, either beside the main sodium pipe, or by a special pipework and in almost all cases as an information enhancing the protection against cooling disturbances. In many examples derived from in-pile experiments and also from reactor operation the very high reliability and availability of the DND method could be proved. On the other hand, the availability of the reactor was adversely influenced in several cases by the DND system: A DND scram which had not been necessary, as found out later, had shut down the plant.

Fission Products in the Cover Gas

The detection of fission products present in the cover gas is practiced by different, mostly diverse techniques (indicated below in alphabetical order):

- gamma spectroscopy using a solid state detector
- gas chromatography
- ionization chamber
- mass spectrometer for stable Xe isotopes, especially Xe-131 and Xe-134
- precipitator
- scintillation counter
- tagging
- Xe adsorption on activated carbon.

It became apparent in almost all cases that the integral methods, such as ionization chamber and precipitator, are the cheapest methods and also attain the highest availabilties whilst gamma spectroscopy, which is very expensive without any doubt, furnishes detailed information which can hardly be dispensed with.

Location and Preliminary Location of Failed Fuel Elements

Several methods have been applied so far for location or prelocation, i.e., determination of the subassembly group affected by the failure (indicated below in alphabetical order):

- Determination of burnup by means of Xe-131 and Xe-134 and of the half-lives of fission gases, respectively. Approximate group estimation.

- DND with individual sodium sampling. Very high investment cost, very good results of location.
- DND with sectorial sodium sampling. More or less satisfactory preliminary location depending on the design.
- Dry sipping. Many fuel element handling operations.
- Flux tilting. Approximate group estimation for smaller sized reactors.
- Tagging. Capital costs and recurrent core costs very high. With increasing number of fuel elements only approximate group estimation possible.
- Wet sipping. Often less successful.

CLASSIFICATION OF FAILED FUEL ELEMENTS

The previous operating experience has revealed two different types of failure:

- The leaker. A cladding tube failure accompanied only by the emission of fission gases and volatile fission products, but not generating a DND signal, which means that there is not yet a contact between fuel and sodium. This damage may prevail in the reactor over weeks or months until it develops into a DND failure by extension or by one or several secondary failures.
- The DND failure. There is a direct contact between fuel and sodium. Following the occurrence of the first DND signal - which has given the name to this type of damage - attainment of the scram level can be expected within days to weeks.

Although much experience with DND failures has been gathered all over the world, the determination of the geometric area of damage on the basis of the DND signals is still very inaccurate. In general, the accuracy is within plus/minus one order of magnitude. The cause for this relatively high inaccuracy are the following facts:

- At the moment when a failure occurs, the fuel may have a very different structure. Both the open porosity due to fabrication and the crack pattern and shape of the central channel influenced by operation greatly affect the release of the short-lived DN precursors. However, all these parameters are not very well known at the time of occurrence of the cladding tube failure.

- The diffusion controlled release mechanism is not yet accessible theoretically on account of the fuel structure mentioned before.
- Frequently bursts have been recorded of DN precursors which make a correlation of signal amplitude and free fuel surface no longer meaningful. Although this non-steady-state release is understandable as a phenomenon, it has not yet been described until now theoretically by a model.

TARGETS SET FOR THIS MEETING AND FOR FUTURE WORK

Besides the exchange of new results from research work and reactor operation, the following subjects should be paid particular attention:

Modeling and Codes

The discrepancies indicated above between the results of post-irradiation and the free damage surfaces calculated according to the recoil model must become smaller. A great help could be provided by the improved post-irradiation examination which is capable of still better quantifying the fuel crack pattern and of yielding measuring values about the open porosity. Efforts in this direction could start by compiling a table of all previous DND damage including the greatest possible number of fuel data, plant specific data as well as operating parameters concerning the failed fuel element.

Methods of Measurement

It seems meaningful to work out a table for the individual reactors, containing among others the following data:

- type and location of the fission product measuring points
- sodium transit time
- sensitivity
- type of location and preliminary location of failed fuel elements
- DND scram and other methods.

Fission Gas Bursts

When a cladding tube failure occurs, the activity concentration of the fission gases in the cover gas may rise by several orders of magnitude. In the absence of special precautions, this automatically results in the overflow of a gamma spectrometer equipped with a semi-conductor counter. To avoid this, the following procedures have been proposed so far:

- dilution of the gas concentration
- automatic collimator replacement
- use of several detectors with different sensitivities.

It is desirable to establish an intensive exchange of experience in this sector.

Contamination of the Primary System

As already mentioned, loop contamination plays a major part with a view to repair and maintenance work in case fuel element failure occurs on account of fuel and fission products. Regarding the continuation of reactor operation with a failed fuel element, the following question arises:

"How much loss of fuel and how much fission product contamination are still tolerable or reasonable?"

Operation in the Presence of Failed Cladding Tubes

It has been mentioned that it is desirable to continue reactor operation until a given date despite the presence of failed cladding tubes. The fission product signals measured under such operating conditions and to be measured in the future should be exchanged and discussed. This as well as the comparison with post-irradiation examinations carried out at a later stage may provide a major support in decision making regarding future operation.

Location of Failed Fuel Elements

In the 2. Chapter the usual methods were indicated for locating failed fuel elements together with the important advantages and drawbacks of these methods. It becomes evident from this listing that a convincing method does not yet exist. This statement is supported among others by the fact that different methods are still tried out in new reactors. Therefore, it seems reasonable to publish all experience - both positive and negative - accumulated with the individual methods or to exchange it. Maybe, a joint effort will bring the desired success.

PRESENTATIONS AND DISCUSSIONS

OF SESSION I:

REACTOR INSTRUMENTATION

IWGFR SPECIALISTS' MEETING ON
"DETECTION AND LOCALIZATION OF FAILED
FUEL ELEMENTS IN LMFBRs"

KARLSRUHE, 11-14 MAY 1981

FRENCH STATEMENT ABOUT DETECTION AND LOCALIZATION OF CLAD FAILURES IN LMFBRs

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I - REACTOR OPERATION -

The risk that the plutonium escapes from a fuel element when the clad is breached imposes a continuous surveillance of the fast reactors.

Since no direct and continuous means of measurement of released fuel is available in a reactor, the surveillance of the fuel is performed indirectly by surveillance of the fission products released when the clad is defected.

The French in-pile experience with RAPSODIE and PHENIX, supported by in-loop experiments, proves that, for detection and localization of the defected fuel subassemblies, DND is a means both efficient and sufficient. Roughly, a DND signal characterizes a failure at the level of the fuel column, and the higher the signal - the greater the defect. At relatively small levels of DND signals and durations of irradiation, no evidence of fuel release by the defected fuel elements could be shown. On the other hand, the experience of RAPSODIE and PHENIX proves that, as long as no DND signal is detectable, the contamination by gaseous, volatible and soluble fission products can be tolerated for a normal operation.

Therefore, at SUPER-PHENIX 1, detection and localization of the fuel elements' failures will be performed by DND. As soon as a DND signal will be detected by the surveillance system (DRG), the operator will start the rotating selectors of the localization system (LRG) that is able to localize the defected subassembly in about 30 minutes. During this time, a scram level on the DRG system will protect the operator of a possible fast evolution of the DND signal.

II - <u>R AND D WORK</u> -

It is oriented towards three main objectives.

The first is to minimize the number of reactor's shut-downs because of fuel element failures. Therefore, a gas plug inserted in the LRG system of PHENIX is being checked. By that way, the gas emitting fuel elements are localized while the reactor is under operation, and can be removed during a normal reactor's shut down before a possible evolution into DND stage.

The second is to make the surveillance instrumentation safer and cheaper. A simplification of the DRG system can be obtained with the use of integrated DND counters (high temperature fission chambers). They are now under development and testing. But their predictable sensitivity being less than the sensitivity of the existing DRG system, their use requires to bring the proof that the DND failures can be irradiated in a reactor up to higher DND levels than reached now. This is the aim of the plan MARGO, that is supported by experiments of pin failures irradiations as well in loops as in reactors (program VOLGA) and by the program PARKA that aims to understand better the neutronuclides' release mechanisms.

The third is to define the field covered by the DND surveillance system with respect to subassembly accidents. For this objective, in addition to the knowledge of the neutronuclides' release when the fuel temperature is very high, and when the fuel is disintegrated, it is necessary to study and modelize the transportation of the neutronuclides by a diphasic coolant.

EVOLUTION OF FFD - FFL INSTRUMENTATION ON FRENCH FAST REACTORS

by C. BERLIN

- PROGRESS IN THE KNOWLEDGE OF THE CAUSES AND IN THE EVOLUTION OF THE FAILED PINS,
- INCREASE OF THE SIZE OF THE POOL REACTORS,
- VOCATION OF THE REACTORS :

  RAPSODIE : EXPERIMENTAL REACTOR,

  PHÉNIX : DEMONSTRATION REACTOR CONSIDERED AS A TESTING STAND FOR VARIOUS FFD - FFL DEVICES,

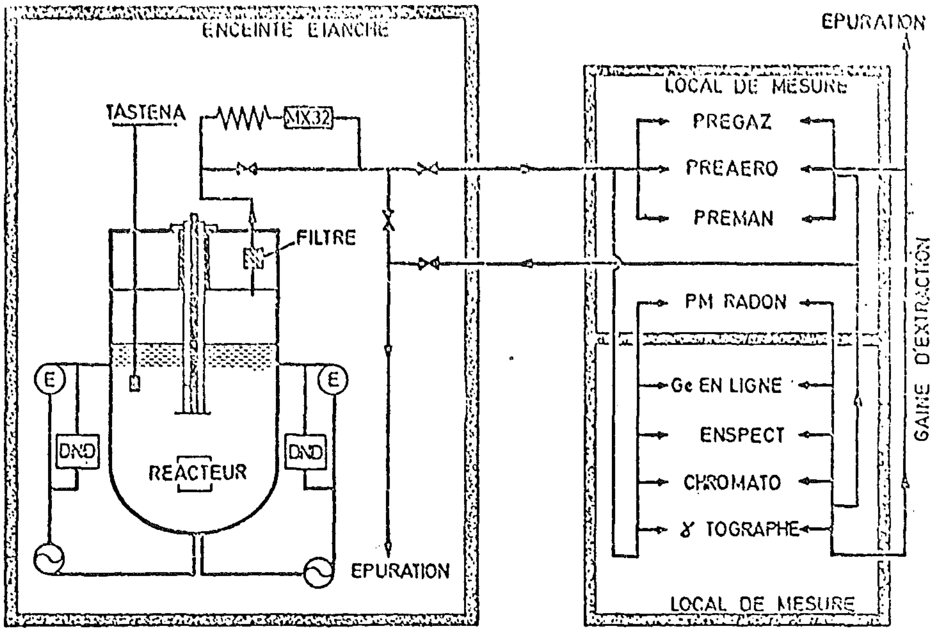
  SPX 1 : POWER STATION PLANT THAT DEMANDS A SIMPLIFIED AND RELIABLE INSTRUMENTATION,

- FOR SPX 2, THE OBJECTIVE IS TO MAKE THE INSTRUMENTATION SAFER AND CHEAPER

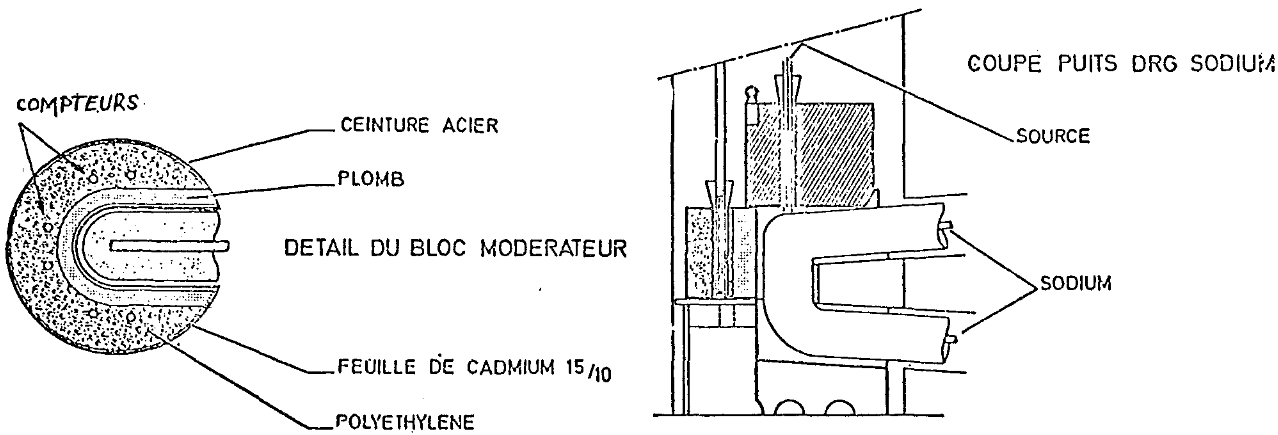
COMPARAISON DE L'INSTRUMENTATION RAPSODIE, PHENIX, SPX1.

FONCTION		APPAREILLAGE	RAPSODIE	PHENIX	SUPER PHENIX 1
DETECTION (fissiles) (fertiles ?)	GAZ	MX 32 + γ TO	X	X	X
	DND prélocalisation	DRG Globale	X 1/5	X 1/6	X 1/8
fissiles	LRG-Gn	Bouchon LRG	non	1	6 (option)
LOCALISATION en marche (fissiles)	GAZ	ENSPECT (traceurs, âge)	X	X (ass. expérimentaux)	non
		Bouchon LRG	non	X (mi-79)	non
	DND		SRG (7 assem.)	Bouchon LRG-G3	6 bouchons LRG-G1

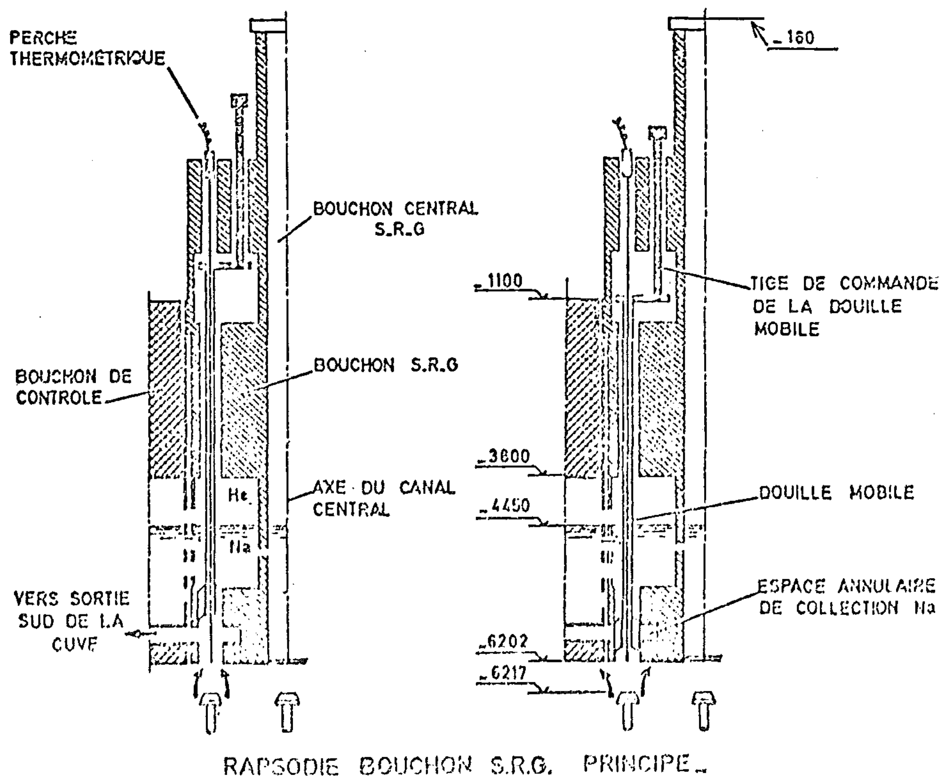
FFD-FEL
INSTRUMENTATION
OF RAPSODIE



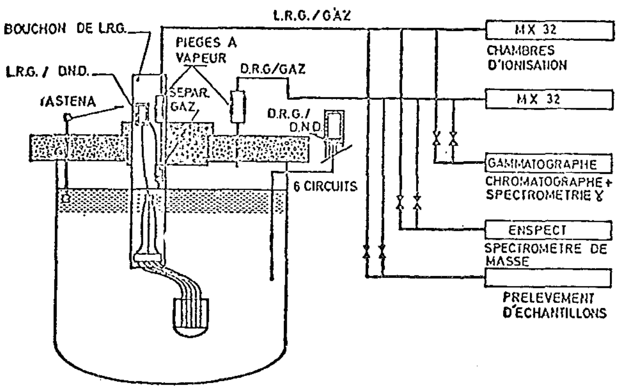
RAPSODIE : SYSTEME DRG_LRG



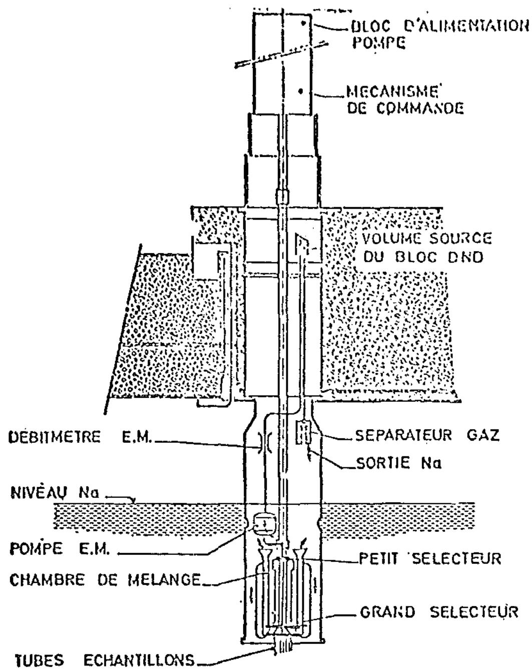
RAPSODIE DRG SODIUM.



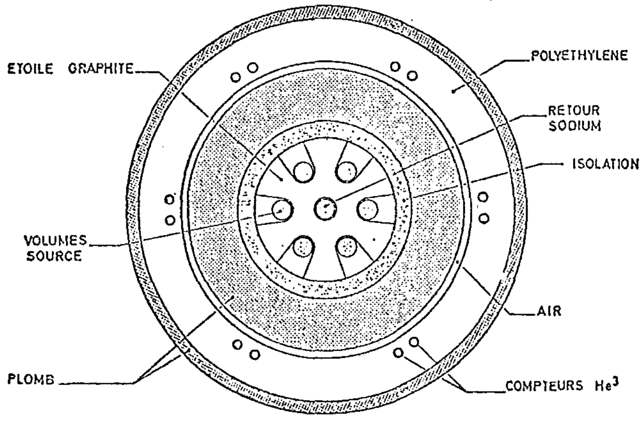
RAPSODIE BOUCHON S.R.G. PRINCIPE_



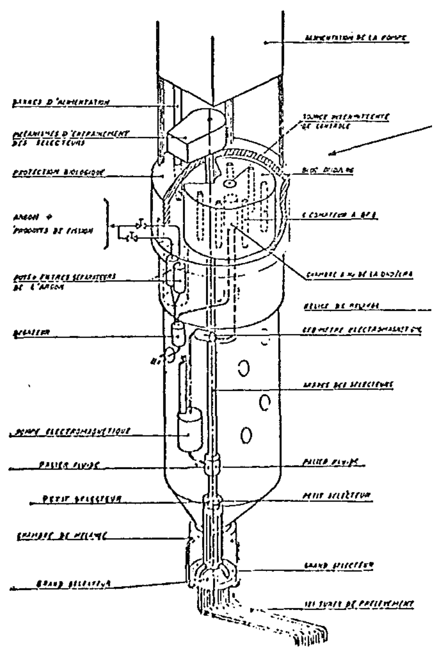
PHENIX_ SYSTEME DRG_LRG



PHENIX: BOUCHON_LRG

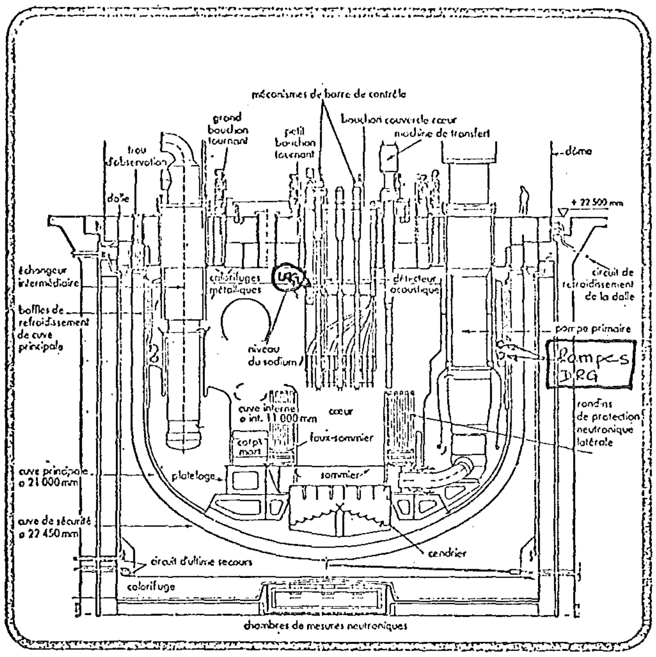


PHENIX: BLOC DND

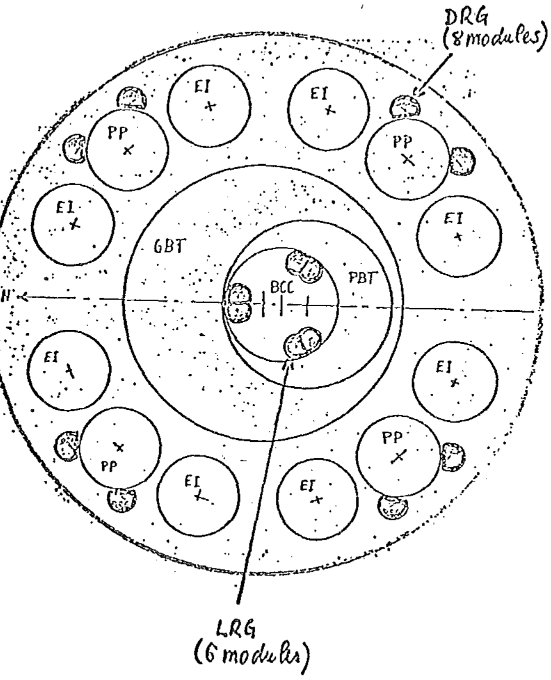


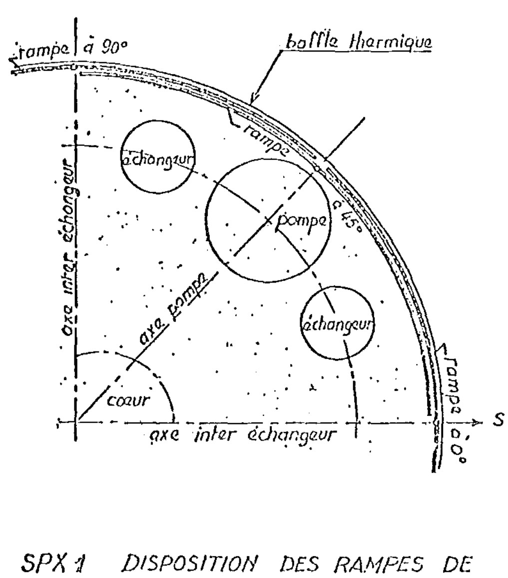
FFD-FFL INSTRUMENTATION OF PHENIX

FFD-FFL
INSTRUMENTATION
OF SPX1

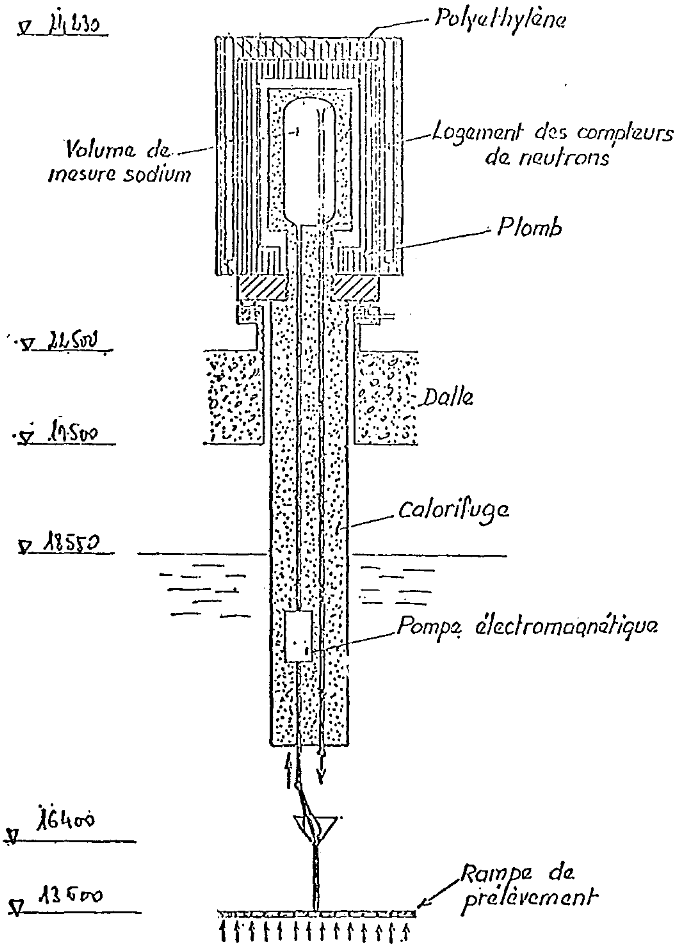


Implantation of SPX1 FFD-FFL INSTRUMENTATION

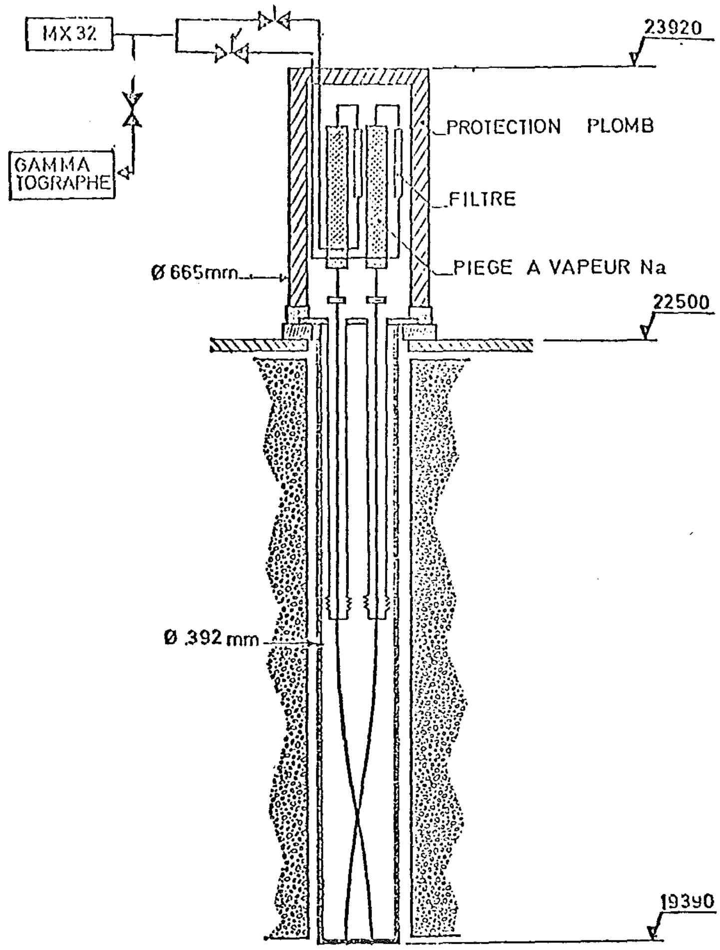




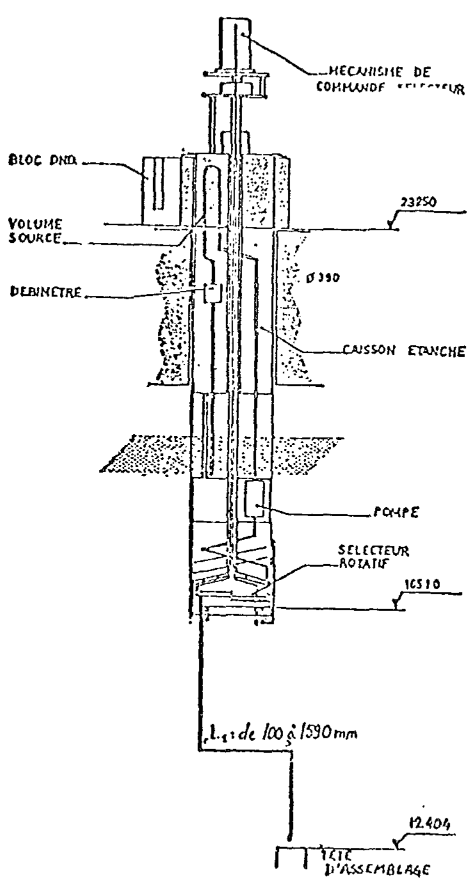
SPX 1 DISPOSITION DES RAMPES DE PRELEVEMENT DRG



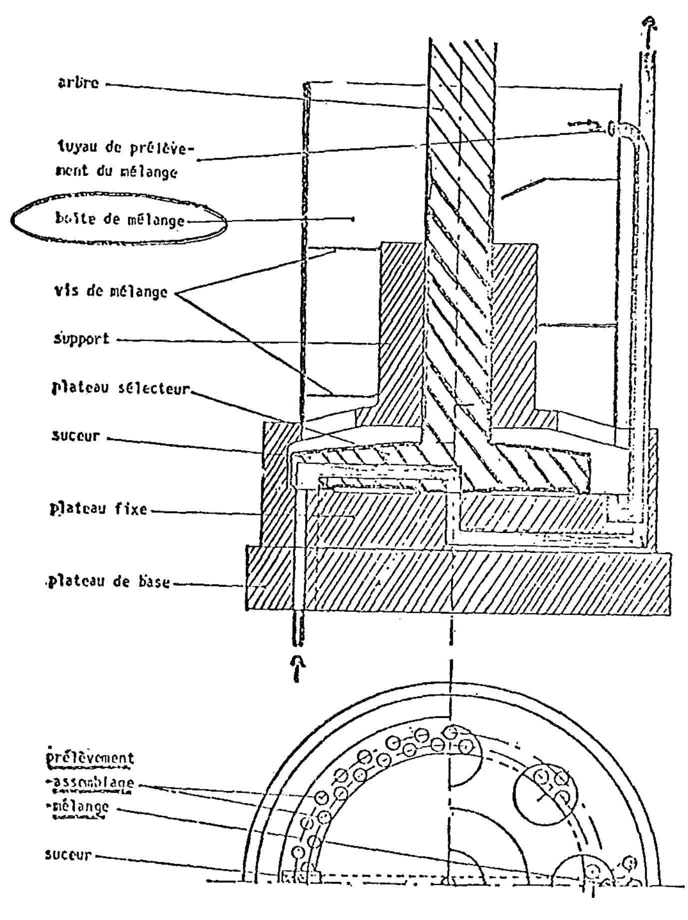
SPX1 - Schéma d'un module DRG cuve et de sa rampe de prélèvement



SUPER PHENIX
MODULE DRG/GAZ



SPX1 - Schéma d'un module LRG avec tuyau de prelèvement.



SPX1. Croquis d'un sélecteur LRG

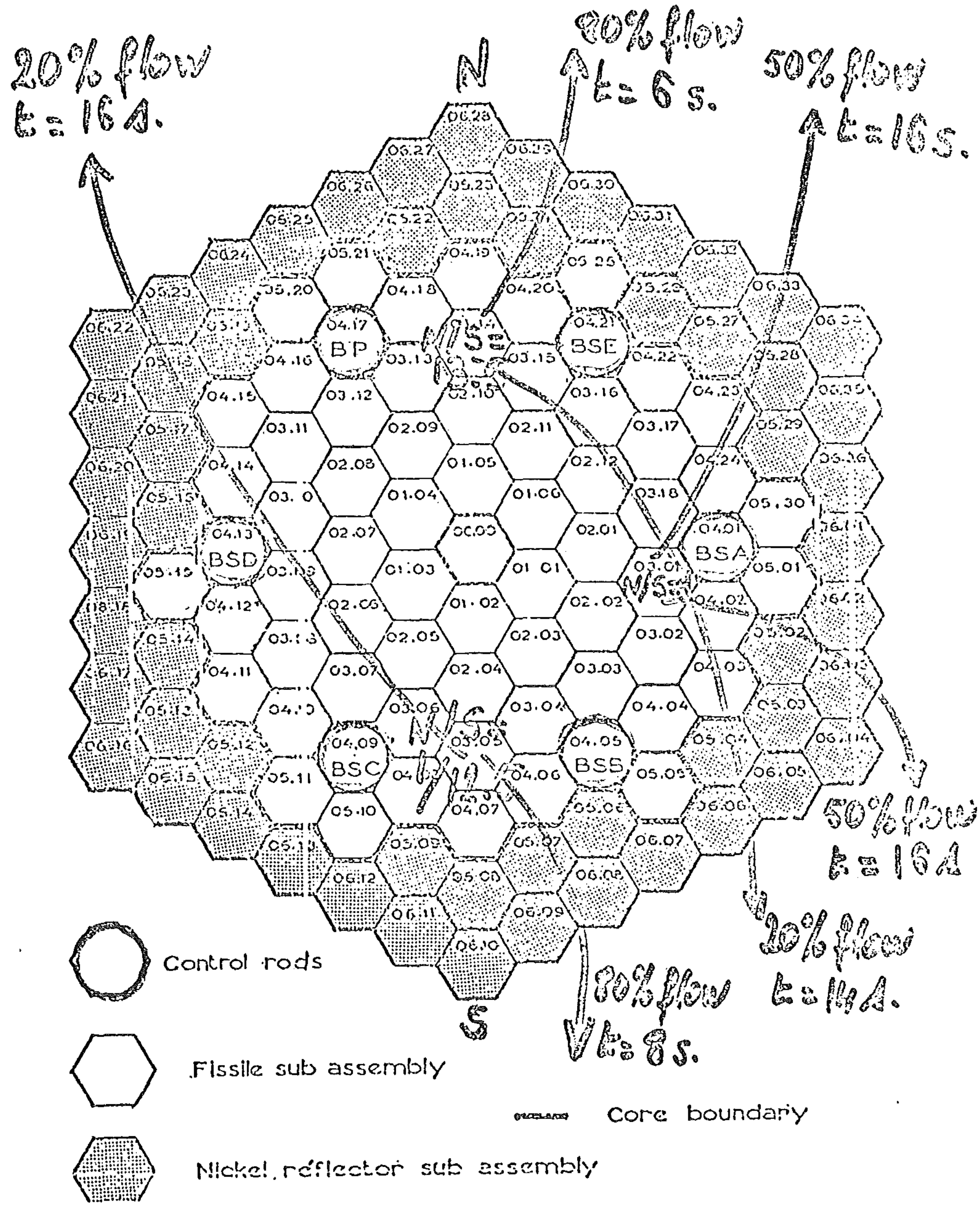
IN PILE CALIBRATIONS
by P. Michaille

YEAR	RAPSODIE	PHENIX	SUPER PHENIX
1970	U-Cr VARIOUS P, Q		
1973		U-Cr 21% P_N, 27% Q_N	
1977	U-Cr EVOL^{N}. IN STORAGE		
1981	U-Mo POWER STEPS		
1982		U-Mo P_N, Q_N PREP^{N}. OF SPX CALIB^{N}.	
1983			U-Mo VARIOUS P,Q UP TO P_N,Q_N VARIOUS LOCATIONS : - CORE I, CORE II - FERTILE - STORAGE DEMONST^{N}. OF INTEG^{D}. DND

RESULTS OF THE 1970 RAPSODIE CALIBRATION

- HYDRAULIC MOCK-UP ⟶ REPARTITION OF THE FLOW N/S

- IN-PILE CALIBRATION INTERPRETED BY THE RECOIL MODEL ⟶ TRANSIT TIMES N/S



RESULTS OF THE DND CALIBRATION OF RAPSODIE

INFLUENCE OF THE POWER : LINEAR

INFLUENCE OF THE FLOW-RATE : TRANSIT TIME $\# \ Q^{-1}$

INFLUENCE OF THE INLET TEMPERATURE : DECREASE OF A FACTOR 2 BETWEEN 400°C & 250°C

Discussion

N. Sekiguchi, PNC:

We have an interest in the durability of the rotating mechanics of DRG-LRG in Rapsodie or Phenix. What life do you anticipate or have you any experience on the life time of the equipment?

C. Berlin, CEA:
We expect that the PHENIX failed fuel localization device will work property during a long time in G_3 and G_n mode. Up to now, only the little selector value failed. The large one have been always working property. - Each week a complete selection is realized.

M. Relic, IA:

a) Have you experience in detection of gas failure by the localisation plug at PHENIX?
b) Why do you not apply this gas localisation at SUPER PHENIX?

C. Berlin, CEA:
a) The first gas FFL plug failed during the commissioning of Phenix
In 1980 a second new one was installed - up to-day we located two gas failures with the new FFL plug.

b) We think that only the DND failures have to be discharged out of core. So on SPX1, the detection and the localisation are by DND only.

D.K. Cartwright, UKAEA:

Are breeder assemblies on SPX sampled by a location system?

C. Berlin, CEA:
Only the fuel subassemblies are sampled by the localisation system on SUPER-PHENIX 1.

S. Jacobi, KfK:

You explained the results from your calibration tests in RAPSODIE. Could you please tell us about the influence of the reactor power on the transit time, signal level and repartition ratios of the sodium?

M. Michaille, CEA:
The answer is given in one of the tables. Power has a linear influence on the signal. The transit time varies roughly inversely proportionally to the flow rate. In Rapsodie, the flow rate is constant. But in PHENIX and SPX, power and flow rate vary.

REACTOR INSTRUMENTATION FOR FUEL FAILURE DETECTION AT KNK II AND SNR 300

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Kernforschungszentrum Karlsruhe
Institut für Reaktorentwicklung
J. Dauk, M. Relic
INTERATOM, Bergisch Gladbach
Federal Republic of Germany

A. FUEL FAILURE DETECTION AT KNK II

The equipment for fuel failure detection of the experimental fast breeder reactor KNK II can be devided up into the operational system and the test system. The experience of both instrumentations are important for the design of the fuel failure detection instrumentation at SNR 300.

Fig. 1 shows the instrumentation of the two systems. The operational system consists of the delayed neutron detection (DND) instruments, of a fission gas monitor and of a defect fuel element localization equipment.

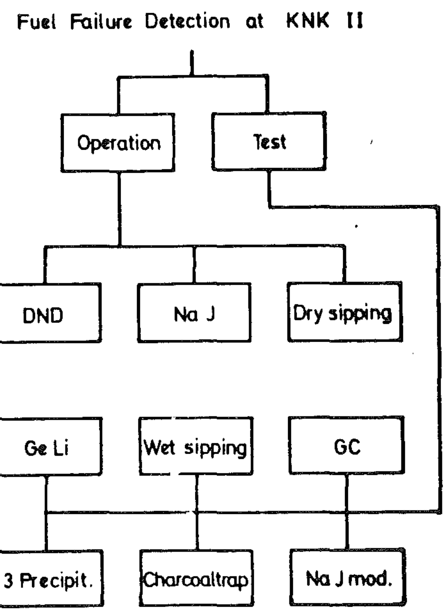

Fig. 1: Instrumentation for fuel failure detection at KNK II

KNK II is a loop reactor with two primary sodium loops. In order to detect the delayed neutrons from the whole core and to have a possibility of a fuel failure prelocalization both loops were equipped by a DND monitor. The DND monitors are classical systems with He-3 neutron counters. After more than 280 operational days no changement of its sensitivity and of its characteristics was noticed.

The background signal of the DND system at KNK II depends on the reactor power and gets up to about 100 cps for the full reactor power. The calculated detection sensitivity depends on the reactor power, too, as the transit time for the delayed neutrons from the defect fuel pin to the DND system decreases by reactor power increase. The countrates calculated by recoil model for 1 cm^2 fuel surface are given in table 1.

Table 1: KNK II: Transit time for delayed neutrons to the DND system and calculated countrates for 1 cm^2 free fuel surface (k = 1)

Reactor power [%]	Countrate for 1 cm^2 recoil surface [cps]	Transit time [s]
30	6	73
50	12,5	44
70	17,5	31
80	18,9	28
100	22,1	22

The in-pile calibration of the DND monitor is planned for the next experimental period.

The scram level is set at 2000 cps, this corresponds to a clad defect of about 8 cm^2 at reactor full power, taking into account the factor k = 10. The alarm level amounts to 1500 cps.

The released fission gas is detected in the argon cover gas by a NaJ(Tl) crystal. Only the Xe-133 activity is measured. In order to eliminate the influence of Ne-23 a delay line is installed between the reactor and the measuring point. The sensitivity of the monitor is 560 $cps/Ci/m^3$, when the measurement is done in presence of an argon specific activity of 1,1 Ci/m^3.

The localization of defect fuel elements is performed by a dry sipping method. After the reactor shut-down the fuel element is lifted up into the handling machine. The gas cooling is interrupted and the volume of the handling machine evacuated to about 0,1 bar. The evacuation time lasts approximately 8 minutes.

The activity of ^{85}Kr, ^{131}J and ^{133}Xe in the gas is measured by a NaJ(Tl) crystal. The sensitivity of the system is for $^{85}Kr \approx 7 \cdot 10^3$ $cps/Ci/m^3$ and for $^{133}Xe \approx 5{,}2 \cdot 10^5$ $cps/Ci/m^3$.

The test systems for fuel failure detection at KNK II can be devided up into equipments which are already working and devices which will be tested during the next defect fuel program at KNK II and during the next KNK II operation. The first group is constituted by an improved NaJ(Tl) instrumentation and by an on-line GeLi system. The other devices are precipitators, gamma gaschromatography, charcoal trap technique and wet sipping equipment for defect fuel element localization.

The on-line GeLi system and the wet sipping equipment are particularly interesting because they are foreseen for the instrumentation of SNR 300.

The GeLi system measures the cover gas activity continuously, that means, it takes the gamma spectra one after another without pauses and calculates the activity concentration immediately. In order to measure the activity in both cases at reactor operation without and with defect fuel elements a countrate range of 10^7 is necessary. For this reason a special collimator which will change the sensitivity of the system over a range of 10^4 is in construction.

The wet sipping equipment of KNK II is similar to that of SNR 300 and will be described later.

B. FUEL FAILURE DETECTION AT SNR 300

At SNR 300 following fuel failure detection systems are foreseen:

- Cover gas monitor with GeLi diode
- DND
- wet sipping device.

The system for the gas activity measurement is similar to the test device at KNK II. The sensitivity is adjusted by moving the GeLi diode in relation to the gas probe. An automatical setting of the sensitivity as a function of the total countrate is foreseen.

The main parts of the DND system are:

- The sodium "sniffing" pipe. This pipe surrounds the core and is situated between the pressure surge shielding tank and the dip plate, Fig. 2. Its diameter is about 5 m. The "sniffing" pipe is devided up into three segments of 120°, and each segment is connected to one pipe. The three pipes lead sodium samples from the respective core section out of the reactor tank. Fig. 3 shows some details of the construction. There are 60 sampling holes altogether. The sampling pipe is fixed at the pressure surge shielding tank.

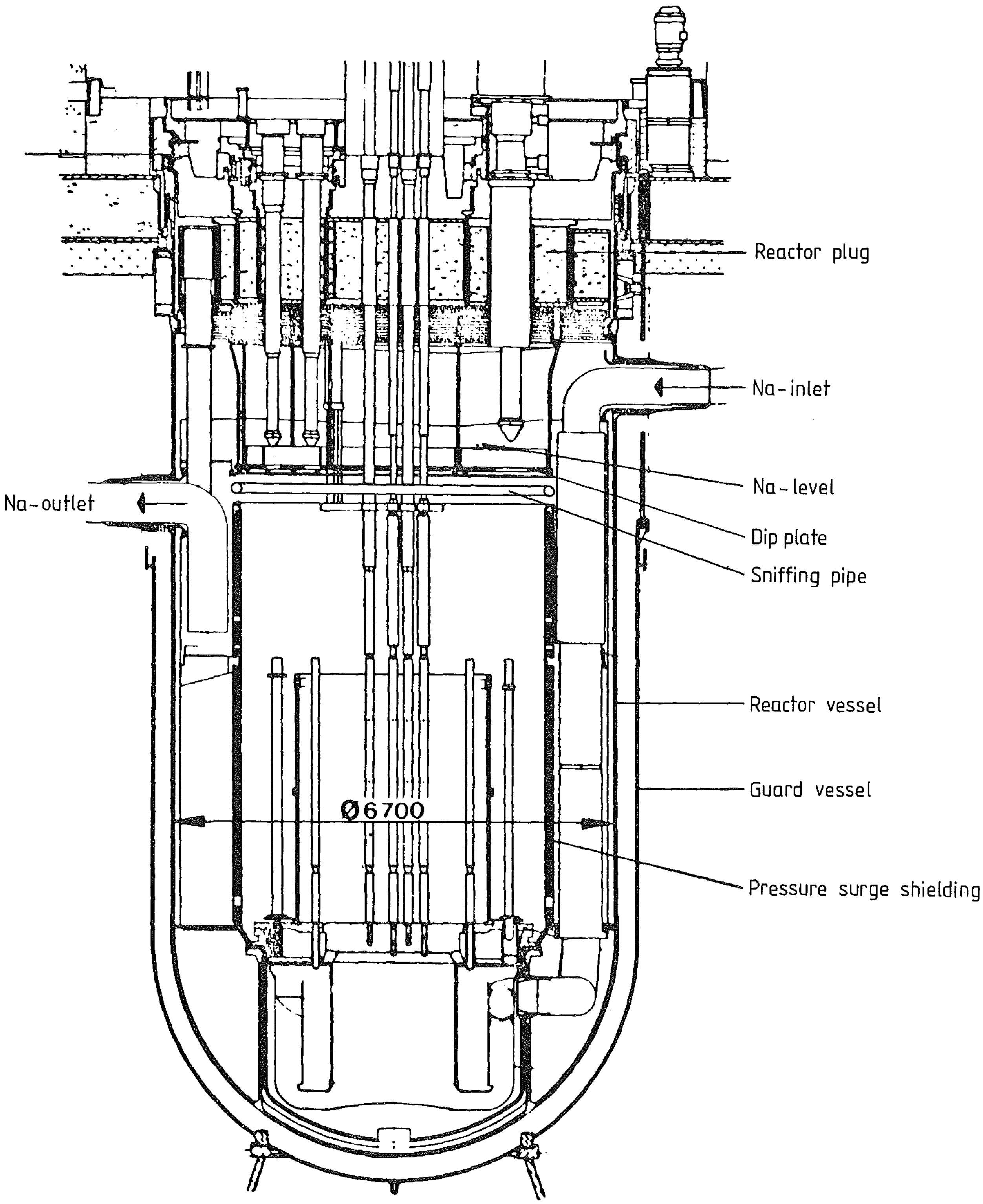

Fig. 2: SNR 300 reactor tank

- Connection pipes between the "sniffing" pipe and the DND monitor. The three sodium sampling pipes (50 mm diameter) coming from the three "sniffing" segments lead sodium from the reactor tank to the sodium level tank room. The sodium flow in these three pipes is controlled by sodium flow meters and valves before the connection to one pipe happens. The mass flow in the pipe amounts to about 100 m^3/h. This pipe leads sodium to the DND safety monitor, Fig. 4. On the other hand sodium samples from the three pipe segments can be analysed in the reactor operational DND monitor. The sodium samples are taken directly from the three pipes coming from reactor tank and chosen by automatical valves. The sodium of the three core parts can be analysed only one after another.

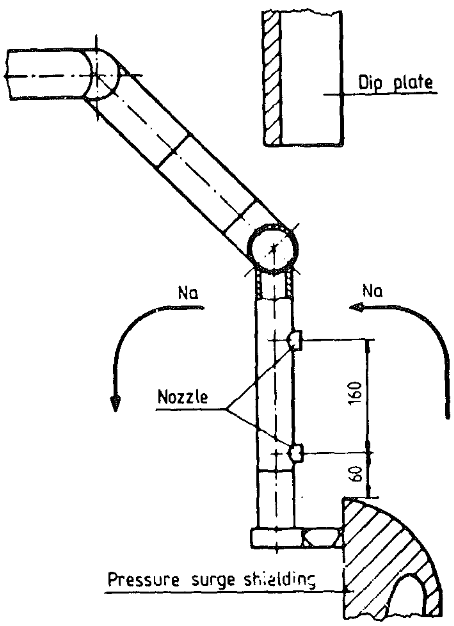

Fig. 3: SNR 300
DND sampling pipe

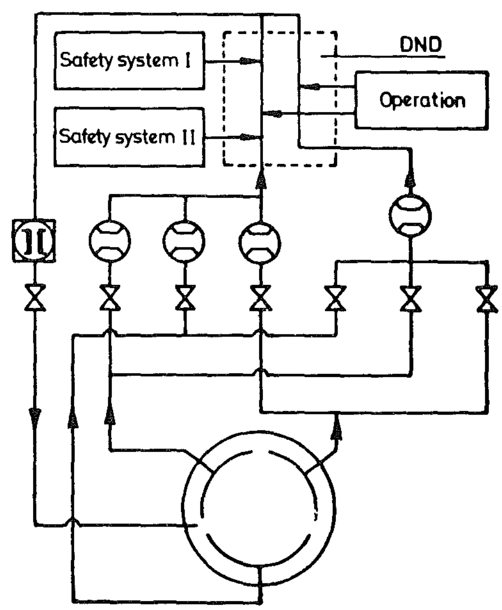

Fig. 4: SNR 300
DND monitor system

- DND monitor. It is a classical monitor with a neutron shielding of serpentine/concrete. For each channel two He-3 counters which can be exchanged during reactor operation, are taken. The design data are:

0,6 R/h gamma dose for reactor operation without defects

1,5 $cm^{-2}s^{-1}$ neutron flux for reactor operation without defects

0,5% efficiency for delayed neutrons.

In order to be sure that the sodium at the DND monitor is representative of the whole core and that the sodium in the three sampling pipes corresponds to the sodium of the given reactor section, water tests at the 4:1 tank model were performed. This model simulates the core pressure drop by blends at the core inlet. The delayed neutrons were simulated by the colour tracer "Methylenblau". It was injected at several core positions and its density was measured at positions around the reactor tank. Fig. 6 shows the relative colour concentration for a central injection, Fig. 5 for an injection at the core border position. The tests were performed for flow rates which correspond to a reactor power between 30% and 100%. The main conclusions from these experiments were:

. The survey of the three core sections is efficient.

. The 250 mm net of sampling holes around the core is sufficient.

. The transit time in the reactor tank lies between 2 s (for 100% flow rate) and 15 s (for 30% flow rate).

The total transient times between the core and the DND monitor are given in table 2.

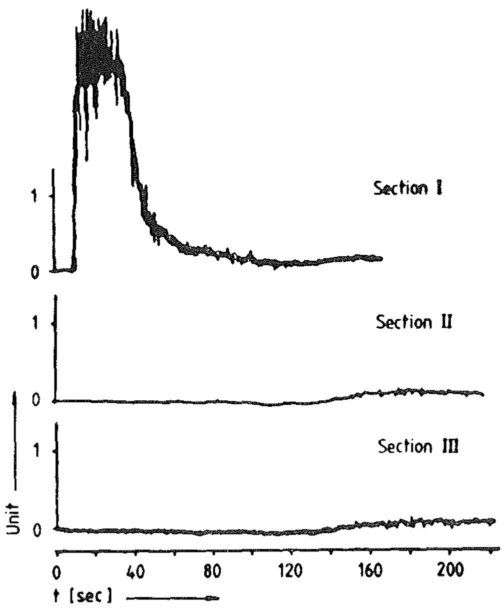

Fig. 5: SNR 300 tank model test
Excentral injection

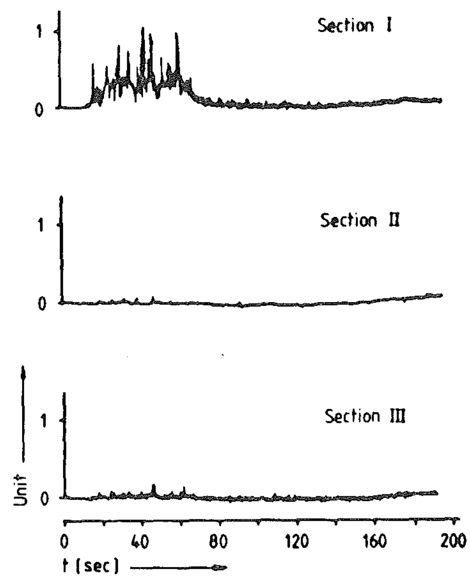

Fig. 6: SNR 300 tank model test
Central injection

<u>Table 2:</u> Transit time for delayed neutrons to the DND monitor (SNR 300)

Reactor power N [%]	Transit time t [s]
30	42
50	36
70	32
80	31
100	29

The countrate at the monitor Z_S was calculated by the recoil model and with the recoil factor k = 1. In the following table 3 the calculated values for the signal background Z_B (surface fuel contamination 10^{-8} g/cm^2, core neutrons, gamma influence) and the highest+) and the lowest+) countrates Z_s for defects of 1 cm^2 in the fuel area are given.

Table 3: Calculated performance of the DND monitor (SNR 300)

Reactor power	Background countrate	Countrate for 1 cm^2 defect	
N [%]	Z_B [cps]	Z_s [cps] (Min)	Z_s [cps] (Max)
30	31	0,82	1,62
50	37,1	0,99	1,95
70	41,9	1,12	2,21
80	43,2	1,16	2,28
100	46,1	1,23	2,43

+) Depends on the power of the defect fuel pin.

For the DND system of SNR 300 the lowest detectable countrate Z_{Min} [cps] is defined as the signal whose amplitude is four times the standard deviation of the background countrate.

At SNR 300 the localization of defective fuel elements is performed by individual examination of all fuel elements after reactor shut-down. The device used for this purpose is called Brennelement-Kontrollvorrichtung (BKV).

The BKV II uses the housing and the drives of the invessel handling machine. It contains an adapter tube that can be connected to the head of a fuel element in the core. Inside the tube a gas overpressure lowers the sodium level inside the tube nearly to the lower end of the tube.

After connection to the fuel element head (sealing according to the diving bell principle) the sodium flow through the fuel element is blocked. In this situation the sodium in the fuel bundle is - in the case of a fuel pin leak - enriched with fission gases. Afterwards the gas overpressure in the tube is compensated with the reactor cover gas plenum causing rising of the sodium from the fuel bundle into the tube. There the sodium is rinsed by a carrier gas stream (fresh argon) and the activity of the released gases is measured in the gas circuit unit, which is attached to the head of the in-vessel handling machine housing. The connection between the circuit and the tube (vertically moving about 500 mm) is made by means of a connection plug inside the housing. After passing the measuring circuit the carrier gas is introduced in the reactor cover gas plenum.

Discussion

J.D.B. Lambert, ANL:

What is the exact method of obtaining sodium for DND in the SNR 300?

M. Relic, IA:
It is a by-pass system with directly Na-sampling in the reactor tank.

K.Ch. Stade, KBG:

Have I correct understand you, there is only one electromagnetic pump in the DND-circuit of SNR 300?

M. Relic, IA:
No, we have two EM pumps at SNR 300.

FAILED FUEL ELEMENT DETECTION AND LOCATION IN FAST BREEDER TEST REACTOR

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REACTOR RESEARCH CENTRE
KALPAKKAM, INDIA.

1. Introduction

Fast Breeder Test Reactor (FBTR) is a 40 MW (Th) sodium cooled fast reactor under construction in India. The reactor design is based on the French reactor, RAPSODIE-FORTISSIMO. However, a sodium heated steam generator and a turbogenerator are added to generate about 16 MW of electrical power. Each fuel subassembly will have 61 mixed oxide fuel pins and the nominal core will have about 65 fuel subassemblies. The reactor has two primary coolant loops. Two delayed neutron detection (DND) blocks are provided, one on each primary loop. DN signals are incorporated in the reactor scram circuit. Gaseous fission product detection (GFPD) is done with the help of a gas flow ion chamber. An instrument for gamma spectrometry of cover gas after gas-chromatographic separation of its constituents is utilized for prediction of the age of a failed fuel element. This presentation gives a brief description of these systems and the development work done in support of design of FBTR.

2. Failed Fuel Element Detection

2.1 DND System

The DND system consists of two DND blocks, each positioned around a sodium pot connected in a by-pass primary coolant loop across the intermediate heat exchanger (IHX). Six boron coated counters are arranged in a hourse-shoe shape geometry in the moderator portion of the block. The shield portion of the block is constituted by 14 cm of lead thickness between the moderator and the sodium pot. The Lead shield attenuates the gamma flux on the detectors to $<$ 2R/h and also helps in reducing the probability of γ-n reactions in the moderator. Three signals are generated by summing the signals from a pair of detectors for actuating the reactor scram circuit in 2/3 mode.

2.2 GFPD System

Gaseous fission product detection is done using a gas-flow ion chamber in the Cover gas sampling circuit.

A theoretical assessment of the signal-to-background ratio which can be obtained with a gas-flow ion chamber of about 1 litre volume gave the following results:

Cause	Ion chamber current in amperes	Signal-to-background ratio
1. Background due to residual $^{41}Ar + ^{23}Ne$ activity (after a delay of 210 seconds)	4×10^{-11}	-
2. Background due to tramp fissile material in the core.	2×10^{-13}	-
3. Background due to ambient gammas.	Negligible*	-
4. Signal due to failure of Central pin after 10,000 MWd/T	1.3×10^{-8}	325
5. Signal due to failure of peripheral pin after 2000 MWd/T.	1.5×10^{-9}	37.5

*Note: The detector is predominently a beta detector, the specific ionization by gammas being less by a factor of 100.

The failures considered in this study are pinholes or microfissures involving release of all the accumulated fission gases in the gas plenum. Applying S/B=1 as the detectability criterion, it has been estimated that a release of 0.5% of accumulated fission gases from a central pin after 10000 MWd/T is detectable whereas for the peripheral pin with a burn-up of 2000 MWd/T the detectable fraction is 5%.

3. Localisation

Small loop-type reactors like FBTR cannot accommodate the required in-vessel equipment for individual coolant channel scanning for d.n. activity. Moreover, methods based on cover gas analysis are adequate. Their poor resolution might ultimately result in a longer delay in localisation.

Gas-chromatographic separation of the constituents of the cover gas followed by gamma spectrometry to determine the age of the failed element and subsequent localisation by substitution of the "suspects" is the only method of localisation adopted for FBTR.

Sampled Cover gas is first passed through a gas chromatographic column for separating the constituent gases viz. argon, neon, Xenon and Krypton. The separated gases eluting from the column are passed around a sodium iodide detector through a helical tube coiled around it. The activities of any six of the fission gas isotopes can be measured with the help of six single channel analysers. From the ratio of the activities of some selected isotopes it is possible to estimate the age of the failed fuel element. The ratio of the activity of an isotope which has not reached equilibrium to that of an isotope which has reached, equilibrium is a function of the age of the fuel element in the core.

4. Development work in support of Design

4.1 DND instrument channel

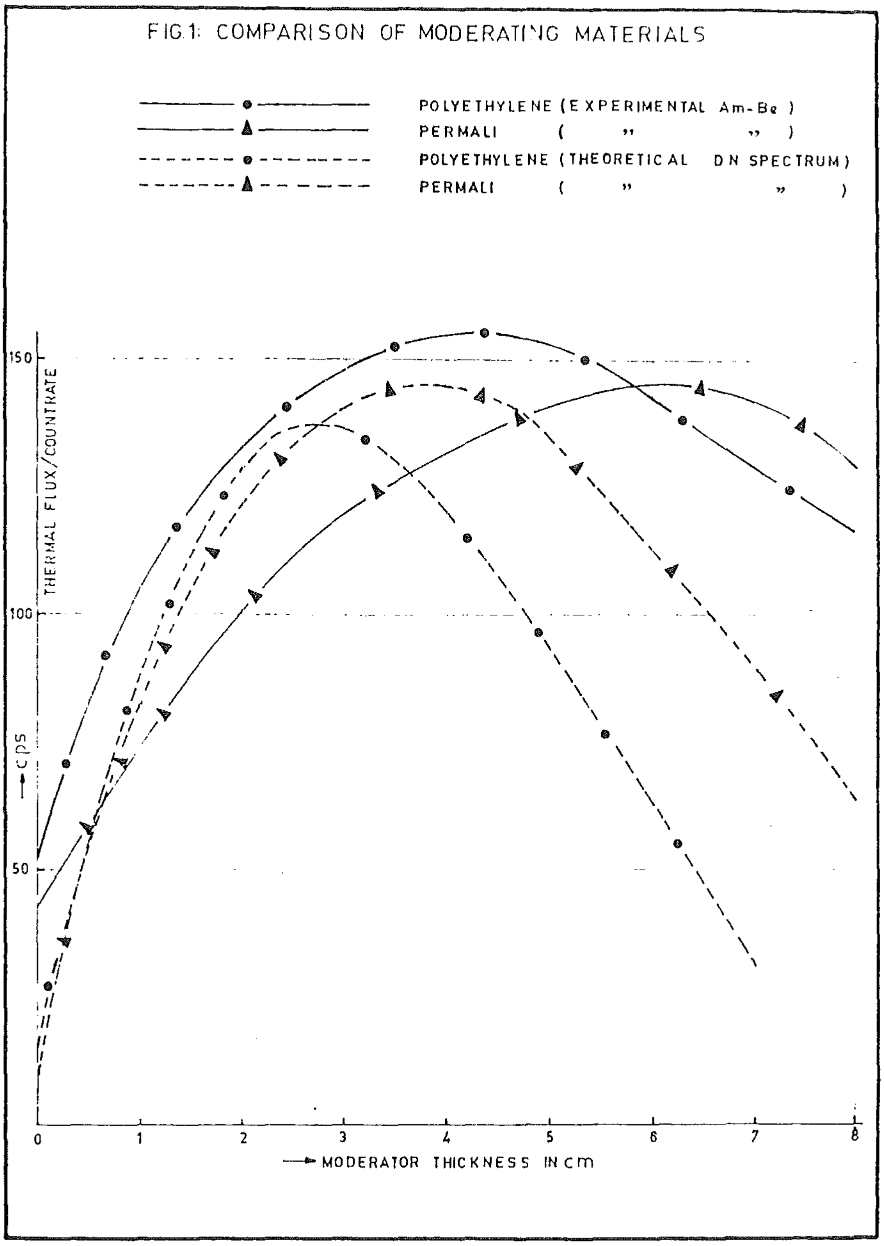
A complete instrumentation channel including detector has been developed and field-tested. The detector developed is a boron-coated counter of 4 Cps/nv sensitivity. The input stage of the instrument channel is operated in current-mode to reduce the effect of gamma pile up and to dispense with any electronics (pre-amplifier) close to the detector. Field tests in Zerlina have shown satisfactory performance of the channel upto a countrate of 10^5 cps with a 100 m cable. The detector plateau is not affected by a gamma flux of 10 R/h (a gamma flux of 2 R/h is expected at the detector location). These tests have established the sensitivity of the instrument channel. Life test on the detector is in progress.

4.2 Choice of moderator

At the design stage polyethylene was chosen as the moderator material in the DND block. This was changed to polyprolylene with a view to facilitating the machining of the block in the desired shape. Experimental comparison of the two material showed that there was no significant difference. However, later it was found that use of permali, a wood veneer product in place of polypropylene would be more cost-effective. It has been experimentally established (Fig.1) that the loss in countrate due to the use of permali in place of polyethylene is only about 15% and hence permali has been accepted.

5. Conclusions

The design basis of FBTR instrumentation and the modest experimental effort, executed and planned, point to a satisfactory performance of the system when implemented on the reactor.



Discussion

S. Jacobi, KfK:

Could you please give us more information about the composition of PERMALI?

D.B. Sangodkar, RRCK:

Permali is a wood veneer product made by compression of bonded thin sheets of a special type of wood found in South India. It's composition is 6.5% hydrogen, 48 % carbon and 45 % oxygen by weight. This material with a few percent of boron has been in use as a neutron shield. As moderator however, it is used in natural form without addition of boron.

M. Relic, INTERATOM:

What are the temperature limits of PERMALI?

D.B. Sangodkar, RRCK:

From the manufacturer's specifications it was found that PERMALI can withstand a temperature up to 110 °C without significant change in composition and physical properties.

Faild Fuel Detection and Location System in the Japanese LMFBRs

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Power Reactor and Nuclear
Fuel Development Corporation

T. Miyazawa
Toshiba
Corporation

ABSTRACT

Outline of FFDL system in the Japanese fast breeder reactor and the requirements to its function is described. Research and development on FFDL system, which have been implemented since 1968, are introducted and reviewed from the viewpoint of availability of JOYO and MONJU. These include basic experiments of fission product behavior, component development and testing, feasibility study of detection system response, and design consideration of several kinds of location methods. The discussion extends to the role of FFDL in other surveillance and monitoring instruments for core anomaly detection in LMFBR plant. One of major concerns on the development of FFDL is now concentrated especially on gas tagging and in-and ex-vessel sipping.

INTRODUCTION

Failed fuel detection (FFD) system in the Japanese experimental fast reactor, JOYO now operating at 75MWt, is comprised of a precipitator and a delayed neutron monitor. The FFD system, fortunately, has indicated no fuel failure in the reactor ever since the first criticality of JOYO in 1977, and no fission product radioactivity is observed in the primary cooling system by radiological analysis of sodium and cover gas samples, which are periodically extracted for examination of FP radioactivity contamination in the system. The FFD system has been well operated(1) without significant trouble for four years.

However, it is necessary to make the system modification for its sensitivity, and to be prepared an interpret model to the signal indication in relation with mode and size of fuel failure possibly encountered in LMFBR. Development works to in-vessel location system are also under way.

On the other hand, design of FFDL system for MONJU is currently under consideration. The detailed specification is being almost completed for the system of both overall detection and localization. Development of the FFDL system and components has been conducted by in-pile and out-of pile loop experiments since 1970. Among many achievements from these R&D works, special concerns are directed to tagging gas system development, outline of which is introduced in more detail in this paper.

REQUIREMENTS TO FFDL FUNCTION

The roles of FFDL system in LMFBR plant should be specified in general expression as follows;

- indication of failed fuel
- qunatitative determination of failure
- characterization of failure mode
- location of failed subassembly
- diagnosis of local anomaly in core
- identification and inspection of failed subassembly

FFD monitoring system is one of core diagnosis and surveillance system in LMFBR, and should fulfill its function complementarily with other instruments and systems having same purpose, such as reactivity meter, in-core flow and temperature sensors for local anormaly detection and acoustic sensor for sodium boiling detection.

Principal design requirements of overall detection system to detect and characterize fuel failure are generally expressed as follows;

- Fast response
- High sensitivity(background reduction)
- Reliability of system and components
- Maintainability (inspection and calibration)
- Diversity of sensors and system

FFDL system in JOYO and MONJU, and their function are described in Table 1 and Fig.1 and 2. Requirements of minimum detectable sensitivity of delayed neutron monitor which are shown in Table 2, come from a safety considerations for preventing pin failure propagation within one subassembly.

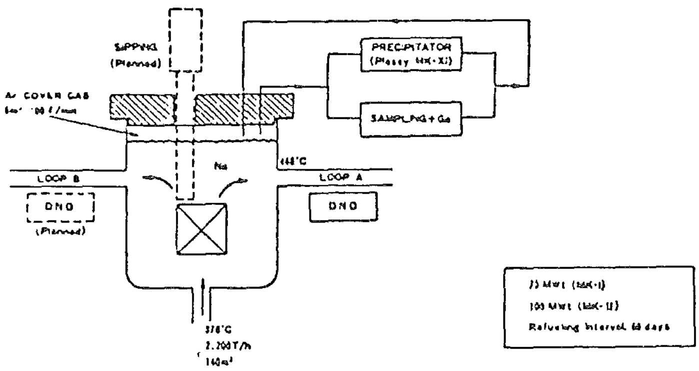

Fig. 1 Block Diagram of FFD & L Systems for JOYO.

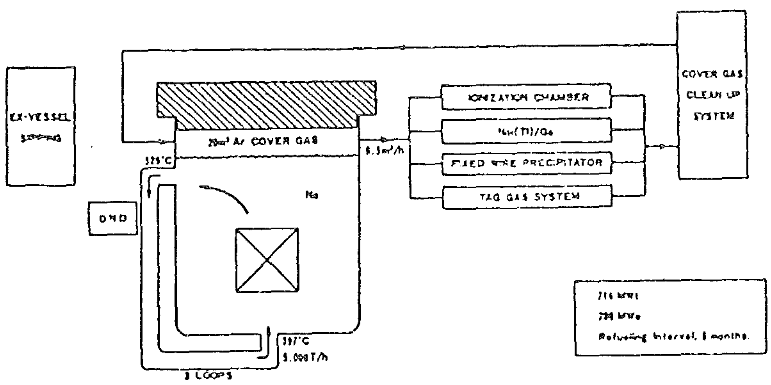

Fig. 2 Block Diagram of FFD & L Systems for MONJU (Planned)

However, the criteria should be reviewed based on several experiences in foreign countries and basic studies.

FFDL system must have function as an indicator to withdrawal of failed subassembly for refueling, if neccessary. Accordingly it is considered to be more important from the viewpoint of plant availability rather that of safety in LMFBR.

In-vessel selector valve sodium sampling system (S/V) had been discussed in the design of MONJU until 1976. Development works of S/V concerning mechanical performance have been continued in order to support the FFDL system design, by 1/4 scale model in-sodium test in 1974 and full mock up model in-sodium test in 1978. The system, however, has discarded from the MONJU design in 1977, because the mechanism of S/V has required too large space to be installed in a small reactor vessel. It is clarified and proven from this

Table.1 Failed Fuel Detection and Location System of JOYO and MONJU

	Function	Method	Equipment
JOYO	Overall detection of fuel exposure and release to coolant	Delayed neutron monitoring	One module
	Overall detection of fission gas leakage	Gaseous fission product monitoring	Moving-wire type precipitator
	Location of failed subassembly	In-vessel sipping (with rotating shield plug operation)	Coolant sampler, gas separator and gamma-ray detector
MONJU	Overall detection (and zone location) of fuel exposure and release to coolant	Delayed neutron monitoring (and triangulation)	One module for each 3 loops
	Overall detection of fission gas leakzge	Gaseous fission product monitoring	Fixed-wire type precipitator and gamma-ray spectrometer system
	Location of failed subassembly	Gas tagging Ex-vessel inspection	Gas absorber and gas mass spectrometer Gas separator and gamma-ray detector

Table.2 Requirements on Delayed Neutron Monitor

Item	JOYO	MONJU (design basis)
Purpose of Instrumentation	Monitor (Manual Scram)	Monitor and Reactor Safety System (Auto Scram)
Minimum Detectable Sensitivity	2.4∿4.7 gr fuel meat release from pins	50 gr fuel meat release from pins
Response Time	∿45 sec	within 60 sec
Maintenance of Detectors	Not specified	To be released in reactor operation

experience, that loop-type reactor in practical designing can not easily accommodate mechanisms and shieldings in its relatively small vessel.

Development of another location method based on sipping (2) is under way in order to provide JOYO and MONJU with a means for in-vessel inspection and ex-vessel examination, respectively.

Study of tagging gas method began in 1970, and several preliminary tests on capsule fabrication and irradiation, analysis of burn-up, and gas concentration procedure have been carried out at aiming to use JOYO MK-II. These efforts have been extended to the adoption for MONJU. The status and review of the development activity on tagging gas system (3) are emphasised in this article.

GAS TAGGING SYSTEM

1. Description of system

The gas tagging system concept is illustrated in Fig.3. The system is comprised of the tagging gas filling fuel pins, gas sampling and enriching equipment, gas mass spectrometer for isotopic ratio analysis and data processing computer. In designing a gas tagging system for LMFBR plants, fundamental parameters are selected as shown in Table 3. This system is actuated automatically by the failed fuel detection signal in cover gas system.

2. Gas Tagging System Development

To implement a gas tagging system for LMFBR it is necessary to research various subjects mentioned above.

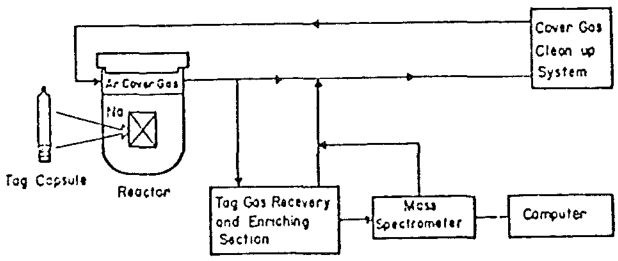

Fig. 3 Gas Tagging System Block Diagram

Table 3 Fundamental Parameter of Tagging Gas System

Filled Tagging Gas Volume	: 2 ml
Released Tagging Gas Volume in Ar	: 0.02ml
Tagging Gas Concentration in Ar	: 0.1ppb
Tagging Gas Concentration of Analysis	: 10ppm
Required Enrichment Factor	: 100,000

1) Tagging Gas Capsule

In developing the tagging gas capsule, gas and capsule volume should be as low as possible for reducing fuel size. Capsule materials should be compatible with fuel and the cladding material.

Three kinds of tagging gas capsule were fabricated for test, and its results were evaluated. These are (a) fusible alloy, (b) coil spring, and (c) bimetal categories. Descriptions of the three tagging gas capsules are shown in Fig.4.

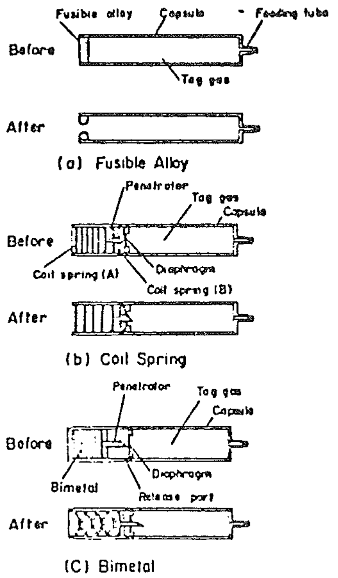

Fig 4 Tagging Gas Capsule

Leakage rate measurement for the welded section and metallographic examination were carried out to verify welding integrity. The result provided that the leakage rates were sufficiently small ($\approx 10^{-8}$atm.cc/sec). Tagging gas release tests were also carried out using an electric furnace. As a result, though the coil spring unit released tagging gas, depending on the heating rate, every tagging gas capsule released tagging gas at the expected temperature.

2) Tagging Gas Enrichment Equipment

Because the escaped tagging gas from a failed fuel pin is transfered into the cover gas via sodium coolant and diluted in the cover gas, the tagging gas concentration was estimated to be on the order of ppb or less. Thus, it is necessary to enrich the tagging gas about 100,000 times to be able to identify the tagging gas isotopes by conventional mass spectrometer.

Based on the preliminary investigation results, low temperature activated charcoal enrichment equipment with two activated charcoal columns has been fabricated, as shown in Fig.5. The enrichment process in this equipment is also shown in Fig.6. The Ar feed gas, containing Xe and Kr of low concentration, was introduced into the 1st stage activated charcoal bed. The 1st stage bed temperature was controlled to -183°C by liquid oxygen for an adequate period. In the next step, the 1st stage charcoal bed temperature was raised by changing the coolant from liquid oxygen to freon (CCl_3F) coolant. Maintaining an adequate temperature, He gas was flowed into the activated charcoal bed for Ar gas desorption by exchanging He adsorption. After this process; the 1st stage activated charcoal bed was heated up to 400°C by electric heater for Xe, Kr, He and Ar gases desorption with adequate time. The desorbed gas was introduced to the 2nd stage activated charcoal bed with a temperature of -183°C for easy handling of gas to mass analysis.

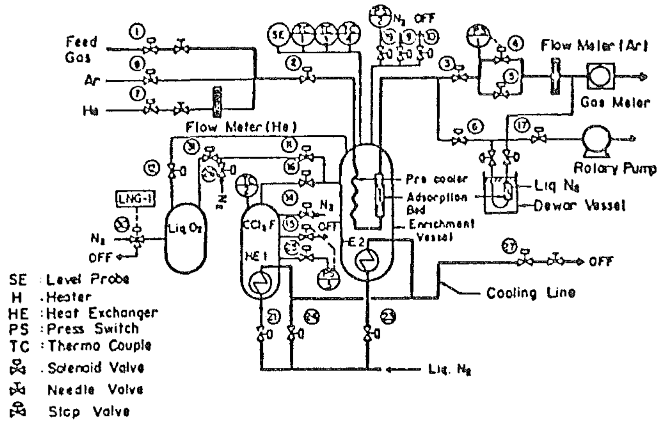

Fig 5 Tagging Gas Enrichment System Diagram

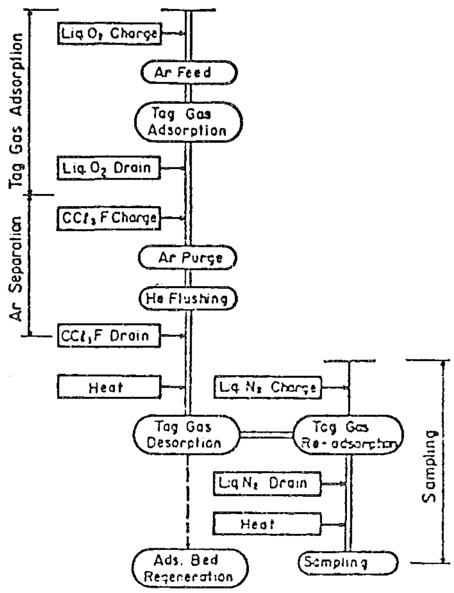

Fig 6 Tagging Gas Enrichment Process

Experimental results, as shown in Fig.7, 8 and 9, provided that Xe recovery rate was not strongly dependent upon He flushing parameters. However, Kr recovery rate and Ar residual amount were greatly affected by He flushing parameters.

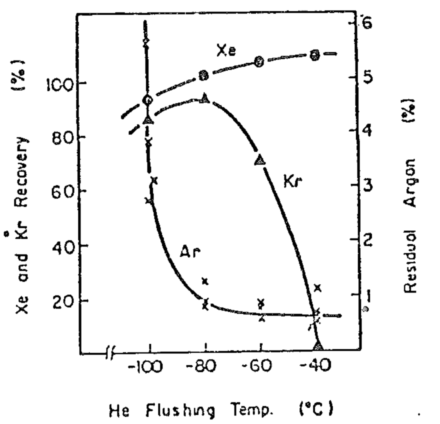

Fig 7 He Flushing Temperature Effect

Feed gas : 396ppb Kr, 410ppb Xe in Ar
Linear velocity : 15 cm/sec
Adsorption temp. -183°C (Liq O_2)
He flushing volume 320

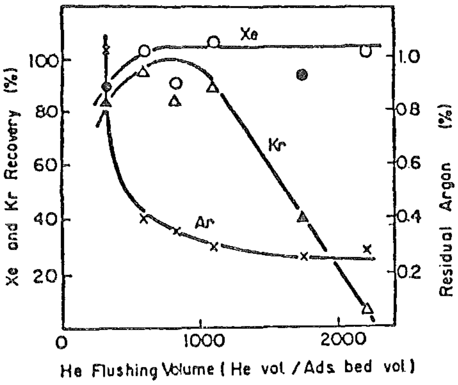

Fig. 8 He Flushing Volume Effect

Feed gas ○ Xe 410ppb, △ Kr 400ppb in Ar
● Xe 490ppb, ▲ Kr 80ppb in Ar
Linear velocity : 15 cm/sec
Adsorption temp. : − 183°C (Liq. O_2)
Flushing temp − 80°C
Flushing linear velocity : 4cm/sec

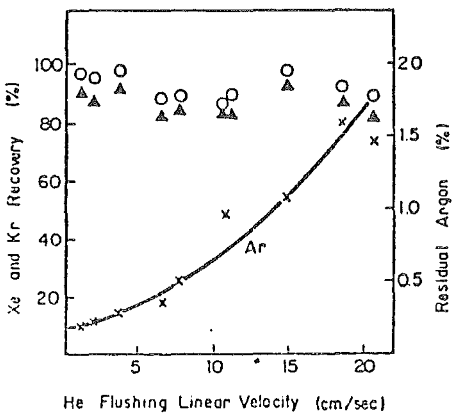

Fig 9 He Linear Velocity Effect

Feed gas : ▲ 480ppb Kr, ○ 490ppb Xe in Ar
Linear velocity 15 cm/sec
Adsorption temp −183°C (Liq O_2)
Flushing temp − 80°C
Flushing volume : 850

Thus, operational parameters for the developed equipment were selected as follows;

Adsorption temperature	:	-183°C
Sampling gas linear flow velocity	:	15 cm/sec
He flushing temperature	:	-80°C
He flushing volume	:	850 times activated charcoal bed volume
He flushing linear flow velocity	:	4 cm/sec

3) Computer Code "TAG"

Computer code "TAG" was developed to have two functions, "Burnup" and "Failure Detect" calculation. "Burnup" can determine the change in tag ratios as a function of reactor operation time. "Failure Detect" can locate the failed fuel subassembly. Fig. 10 shows a computer code "TAG" diagram.

ENDF/B-IV is a computer code for calculation of tagging gas burnup. In this calculation, neutron flux was divided into 26 energy groups.

Failure location calculation is carried out with regard to only Xe tag gas isotopic ratios. In the calculation, the following factors are assumed.

(1) Number of Xe tag ratios is 54 (18 of $^{129}Xe/^{126}Xe$ x 3 of $^{129}Xe/^{128}Xe$)

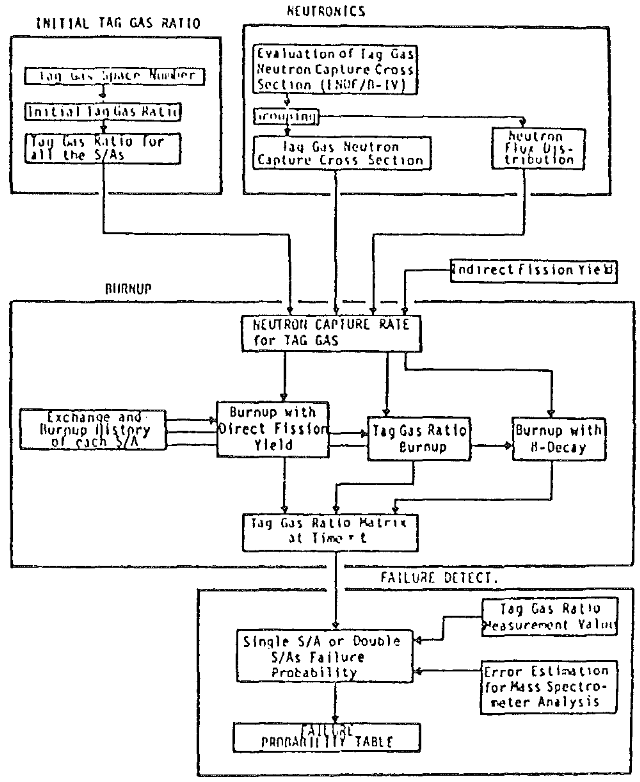

Fig. 10 Computer Code "TAG" Diagram

(2) Spacing factor ($r_{i+1}=r_1 S^i$);
for $^{129}Xe/^{128}Xe$, r_1=1.0, S=0.883
for $^{129}Xe/^{126}Xe$, r_1=100, S=0,8

(3) Reactor operation time ; maximum 740 days

(4) Neutron flux density corresponds to demonstration class FBR, such as MONJU

(5) Pin failure occurs after 740 days reactor operation

(6) Analyzed tag ratio ;
$^{129}Xe/^{128}Xe$ = 0.97
$^{129}Xe/^{126}Xe$ = 3.0

Fig. 11 shows the initial tag gas ratios and the ratios after 740 days reactor operation. It indicates that any tag ratios after 740 days reactor operation does not cross the next initial tag ratio, therefore tag ratio "uniqueness" is maintained for any assemblies.

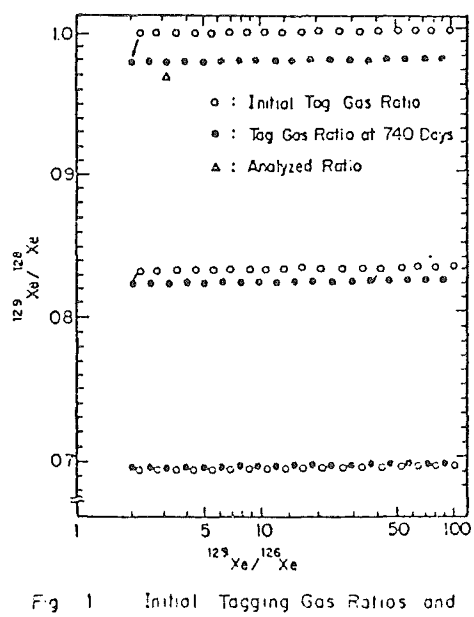

Fig 11 Initial Tagging Gas Ratios and Ratios after 740 Days Reactor Operation

3. Future R & D Program

Some improvements are required to establish the gas tagging systems described in this paper. The following plans will be carried out in the future.

(1) Tag gas capsule

- Establishment of a commercially available fabrication technique

(2) Tag gas enrichment system; Low temperature activated charcoal equipment

- Development of a large scale activated charcoal bed to manage a large quantity feed gas

- Development of the easy maintainance cooling system

(3) Computer code "TAG"

- Verification of the tagging gas burn up effect

- Investigation on tagging gas isotope production from a indirect fission yield

SUMMARY AND CONCLUSION

Several R & D works on Failed Fuel Detection and Location System and equipment have been reviewed in this article. As conclusions of this article, following items are summarized.

1) In MONJU, DND will be designed to have plant safety system based on previous R & D works, and some in-plant calibration should be considered.

2) Cover gas monitoring system function, in MONJU, is the monitoring to fuel failure including gas leaker. The system will be designed based on the performance test at CGMTF (4)

3) Gas Tagging Method is considered to be one of the most promissing method by reffering several works on the performance. (5)(6) The major problems are the tagging gas cost reduction and multiple failure pin location.

4) Sipping method will be applicable in any type of LMFBR. The problem of this method is the operation time reduction.

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2. T. Takagi, et al., "Sipping Methods for Localization of Failed Fuel Element in Japanese LMFBR", this meeting.

3. N. Sekiguchi, et al., "Gas Tagging System Development in Japan", this meeting.

4. E. Sakai, et al., "Summary of In-pile Loop Experiments Related to the Development of the Fuel Failure Detection Systems for LMFBR in Japan", this meeting.

5. C. Berlin, et al., "Les Systems de Detection des Ruptures de Gaine Dans Les Reacteurs Rapides", IAEA-SM-226/66, Canne, France, 24-28 avril 1978.

6. J.D.B. Lambert, et al., "Recent Improvements in Identifying Fission Product Sources in the Experimental Breeder Reactor II", Nucl. Tech. Vol.39 Aug. 1978.

Discussion

F.E. Holt, HEDL:

What is your DND countrate at full power operation?

T. Miyazawa, Toshiba:
The DND background countrate at full power is 2500 cps.

M. Relic, IA:

DND at Monju is the same as at Joyo?

T. Miyazawa, TOSHIBA:
No. We will modify the JOYO module.

M. Relic, IA:

Contribution of Ne-23 to the precipitator counter?

T. Miyazawa, TOSHIBA:
Precipitator countrate due to Ne-23 was about 70≃80 cps at full power
(About 60 % of total background).
But we are not sure the sensitivity on the precipitator to Ne-23 (cps/μCi/cc or cps/mR/H.)

D.K. Cartwright, UKAEA:

What is the contribution of (γ-n) neutrons on DN monitor?

T. Miyazawa, Toshiba:
Below 10 %. Thus we did not use lead shield to polyethylene for reducing photo-neutron contribution.

C. Berlin, CEA:

How many DND monitors are there on Joyo and Monju reactors?

N. Sekiguchi, PNC:
Only one module is equipped on JOYO now, but one module/each two loops will be equipped prior to the initial operation of JOYO-Mk-II (100 MWt).

In MONJU, installation of one module/each three loops is planned and expected to reinforce failed fuel location by DN signal triangulation.

(Please refer to the distributed paper of session V.)

THE DETECTION AND LOCATION OF FUEL FAILURES IN SODIUM COOLED FAST REACTORS

REVIEW OF UK WORK

COMPILED BY

D K CARTWRIGHT RNL
C V GREGORY DNE

WITH CONTRIBUTIONS FROM

T A LENNOX DNE
I CATHRO BNL CEGB
F A JOHNSON AERE
W R DIGGLE RNL
D MACDONALD NNC

UK presentation to the IWGFR Specialist Meeting on 'Detection and Localisation of Failed Fuel Elements in LMFBRS' Karlsruhe 11-14 May 1981

Summary

and

Part 1

REACTOR INSTRUMENTATION FOR FAILED FUEL DETECTION IN UK FAST REACTORS

SUMMARY

The detection and location of failed fuel elements is regarded in the UK as an important aid to the operation and safety of large fast power reactors. Although experience suggests that fuel failures will be infrequent the very large numbers of pins contained in present and future reactor cores means that some failures must be expected, whether as a result of manufacturing defects, prolonged atypical operation, or simply of exhaustion of the clad at high burn up.

Continued power operation with failed fuel pins in a number of fast reactors and for a range of severity of failures has shown that it should not normally be necessary to remove a defect fuel pin immediately the failure is detected. Nevertheless even if the pin is to remain within the core it is of benefit to the reactor operators if such defects are detected and located as early as possible to allow any implications upon future operations to be considered and the character and progress of the failure to be monitored. In particular it is of interest to identify whether the failure is likely to be 'quiescent', as the majority of 'endurance type' failures have proved to be, or whether subsequent release of fuel and fission products to contaminate the primary circuit may be expected. This emphasises the need to understand the characteristics of signals from different types of fuel failure, particularly if the reactor is to be operated with a number of failures in the core.

Work in the UK relevant to the detection and location of failed fuel in sodium cooled fast reactors is reviewed in an appropriate paper in each session. A description is given of the bulk system used in PFR for the detection of rare gases in the cover gas and for the detection of delayed neutrons in the primary sodium. Methods used, and proposed, for the location of failures are also described, including the detection of fission gases and delayed neutrons from individual subassemblies and the detection of radon tags in the reactor cover gas. An outline is also given of the tentative scheme proposed for CDFR with a brief discussion of design philosophy.

Experience with the PFR systems is reviewed including the detection and location of two failures in experimental pins.

The release of fission products from fuel and from fuel pins to the coolant is being studied at a number of laboratories in the UK and models for the release of fission products are being developed. It is evident that this release is complicated and depends on many factors including the type and state of the fuel, which affects diffusion of fission products, as well as fuel-coolant reactions, and the fuel-clad gap and coolant velocity, which affect the release by scouring of the fuel surface by coolant.

In addition to the continuation of these studies a substantial programme of work with natural and proposed defected pins has started on PFR. This programme which is principally to study the behaviour of defect fuel during operation and to characterise signals from failures, is also discussed.

REACTOR INSTRUMENTATION FOR FAILED FUEL DETECTION IN UK FAST REACTORS

In this section of the UK presentation the PFR reactor instrumentation is briefly described and a tentative instrumentation scheme for CDFR is presented.

(a) The PFR Monitoring Systems

The reactor was designed with systems to monitor both the cover gas and the primary sodium coolant for fission product activity. A design of the PFR primary circuit is presented as Figure 1.

The cover gas circuit is shown in Figure 2, the gas is monitored continuously using a precipitator of the same type used on AGRs (1) and shown in Figure 3. The precipitator enables some of the shorter lived fission product gases eg. Kr 88, Xe 138, to be monitored while suppressing the effect of other activities. Gamma spectrometry on the cover gas is also carried out virtually continuously to monitor for longer lived fission gases.

The original delayed neutron system was designed by Hackney & Wood (2). A bulk system continuously monitors samples taken from each of the six intermediate heat exchangers (IHX's); the six separate samples are mixed in a single IHX bulk monitor. The unit is contained in a separate thimble similar to the location monitor.

In addition a system was also provided to take samples continuously from the outlet of each subassembly. These samples could be mixed to provide a second bulk monitor - the subassembly, 'S/A', bulk monitor while at the same time one of the samples is selected by a rotating valve and is monitored separately by another delayed neutron monitor. In addition the location loop was fitted with a gas stripper, as described by Cartwright (3), to enable fission gas to be removed from the sodium for activity measurement with a gas precipitator similar to that used on the cover gas.

A new improved location loop (Fig 4) is shortly to be fitted to PFR, the Mk IV loop, which will provide the same detection facilities as the original loop but with the additional facility that the subassembly location and bulk systems can be operated independently. The new Mk IV loop uses an advanced sodium pump design.

A location facility has also been achieved by using a sparge pipe to inject argon into the exit sample pipe from the rotating valve. This acts as both a gas-lift pump and gas stripper and although only a small flow is obtained the efficiency is high. It was this system which was used to successfully locate failures in the two experimental fuel pin failures in PFR. Following the method successfully developed for DFR (4) provision is provided to monitor the cover gas for radon. Radon capsules are incorporated in a number of experimental pins and failure of such a pin releases radon to the cover gas so that these failures may be distinguished from these in the normal driver charge. Detection of radon in the cover gas is at present performed by sampling and subsequent measurements of the alpha activity. However a precipitator system has been developed and is currently being commissioned and will enable the cover gas to be monitored for radon on-line.

(b) The CDFR Proposed Monitoring Scheme

A scheme for CDFR has not yet been decided although a tentative scheme outlined below and in Figures 5 and 6 is being studied.

The cover gas will be monitored as on PFR to measure its activity and to indicate the overall state of the core.

The bulk coolant will be monitored for delayed neutrons using monitors which receive samples continuously pumped from the outlets of each heat exchanger. To ensure full coverage for safety two independent monitors are planned, each receiving four samples from each of the eight heat exchangers. A simple location system is also being considered to give an approximate guide to the position of a failure.

Such a system is considered satisfactory for detection of failures in the core but sensitivity for failures in the low rated breeder assemblies will be lower than for the core. To improve breeder detection sensitivity the possibility of monitoring the breeders separately is being studied. One proposal is to take individual samples from each breeder and to monitor the mixed breeder samples with a DN monitor. For this purpose a pipe would be brought down the centre of each breeder assembly from the outlet to a sample pipe connecting the diagrid to two monitors mounted in the fixed roof. Such a scheme could also be used to locate breeder faults either to an individual assembly or to groups of assemblies using a rotating sampling device for selection of the sample. Location of core failures is achieved similarly by sampling subassemblies in turn. In this case pipes from S/A outlets can be fed directly to monitors in the rotating shield. Six monitors are being considered each sampling 60 assemblies. The type of monitor has not been decided but DN and gas detection are both being considered.

A further possibility being explored to locate assemblies containing defect pins is to sample the sodium from subassemblies in turn for fission gas release after the reactor has been shut down. For this purpose a moving or 'roaming' probe is a possibility.

(c) Monitor Design

The DN monitors in PFR are all cylindrical in construction, the sample volume being surrounded by annuli of lead, polythene and then lead, with suitable heat barriers and steel supports. The inner lead annulus reduces the gamma radiation at a ring of BF_3 counters in the polythene annulus to an acceptable level for pile-up and damage purposes. The gamma activity is also such that gamma-neutron interactions generate a negligible background. Sufficient polythene is present outside the outer ring to reduce neutron leakage and to make the count rate due to external radiation negligible, while the outer lead reduces the external radiation to a radiologically acceptable level. With this design the count rate in the absence of failures is the result of contamination of the core by 'tramp' fuel. The basic sensitivities of the monitors are given in Table 1. These can be checked by a neutron source lowered into the monitor. The relative sensitivities for detection of fission products released from the core are, of course, also dependent on dilution of the signals, these are also given in Table 1.

The sensitivity to failures is deduced from calculations based on fission rates and on estimates of sample dilution etc from experiments and on the transit time to the monitors which is of the order 20-30 seconds.

To check the performance of the location monitor one subassembly has a known area of UO_2 exposed in an axial breeder position. The signal from similar foils has been measured previously in an in-pile loop.

The bulk monitors have 36 BF_3 counters connected in 3 groups to enable a 2 out of 3 trip system to be used, and the location monitor has 12 counters.

The precipitators used on the cover gas and location loops are standard 1 litre Plessey Mk X1 gas precipitators with NaI detectors (see Figure 3). The sensitivity can be varied according to the activity being measured, typically they are used with a soak time of 1 min followed by a count time of 1 min enabling a small increase, 10% or less of the background count rate, to be detected.

The gamma spectrometer is a standard Ge-Li detector connected to a multichannel analyser.[3] The counting volume is 135 cm^3 and the sensitivity can be adjusted, if this should become necessary, geometrically to give a suitable sensitivity.

The output signals are mostly connected to the PFR DRE computer and are presented to the operator directly or in other suitable forms. The bulk monitors have independent signals fed to safety circuits for alarm and trip purposes.

Monitor design for CDFR is still being considered, it will be based on experience gained with the PFR design.

REFERENCES

1. The Plessey Co. Mk X1 Precipitator - Commercial literature.

2. HACKNEY S and WOOD J. 1970. "Burst Pin Detection (BPD) Equipment". IAEA-SM-130/17.

3. CARTWRIGHT D K. (1974). "In-core Instrumentation and Failed Fuel Detection and Location". Proc IAEA Specialist Meeting. AECL 5124.

4. DIGGLE W R and SINCLAIR V. (1969). Private Communication.

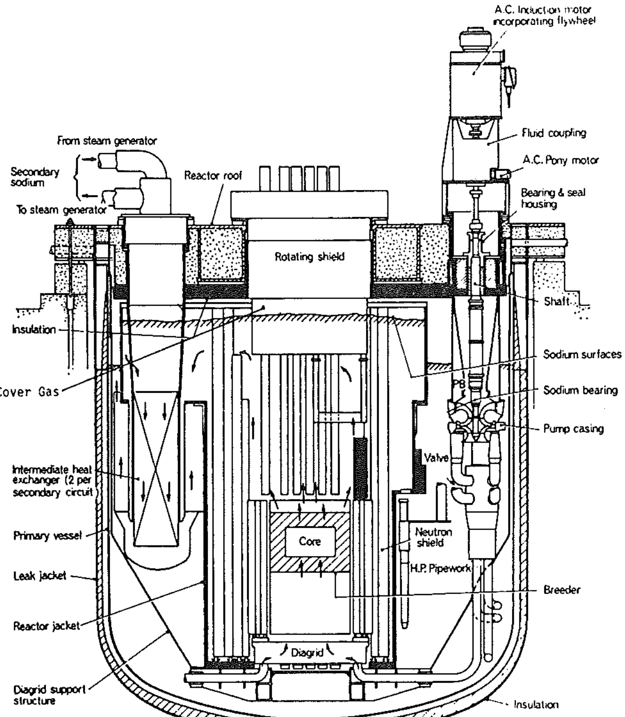

FIG. 1

SECTION THROUGH PFR

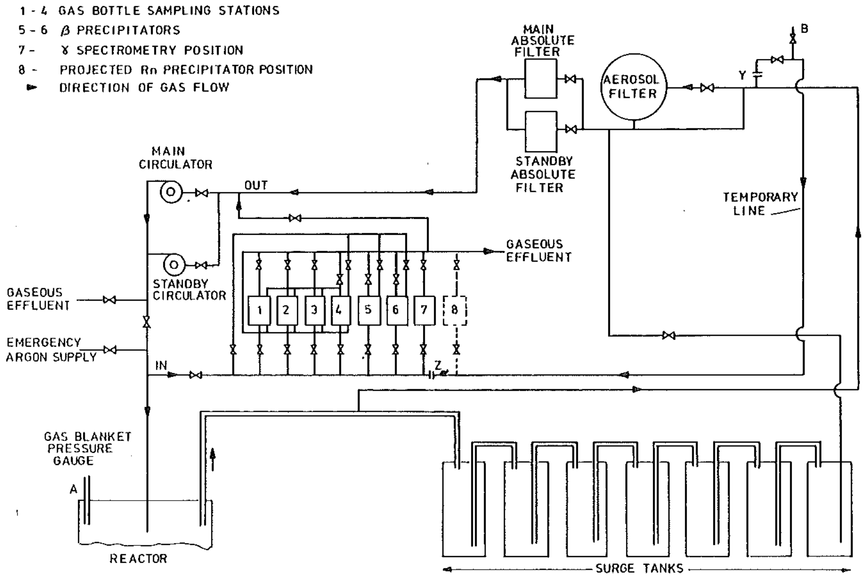

FIG. 2. PRIMARY ARGON GAS BLANKET SAMPLE SYSTEM FLOW DIAGRAM

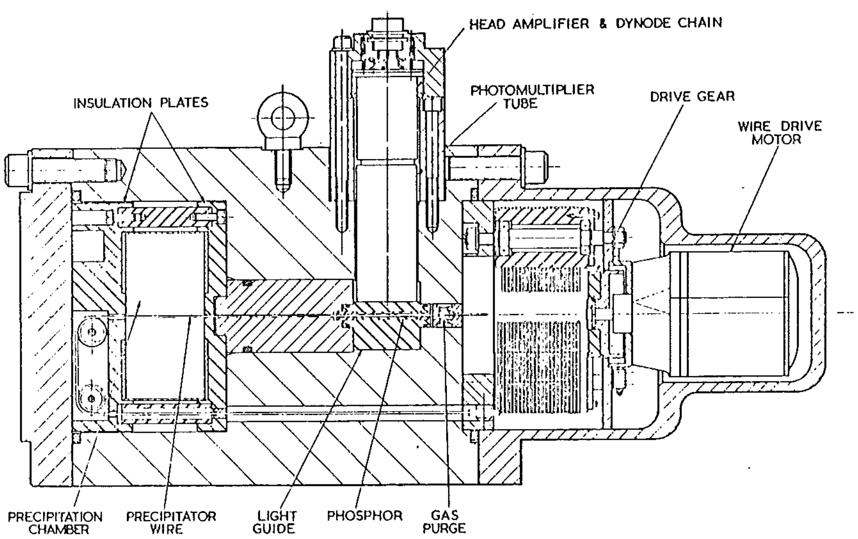

FIG. 3

PLESSEY MARK ELEVEN PRECIPITATOR

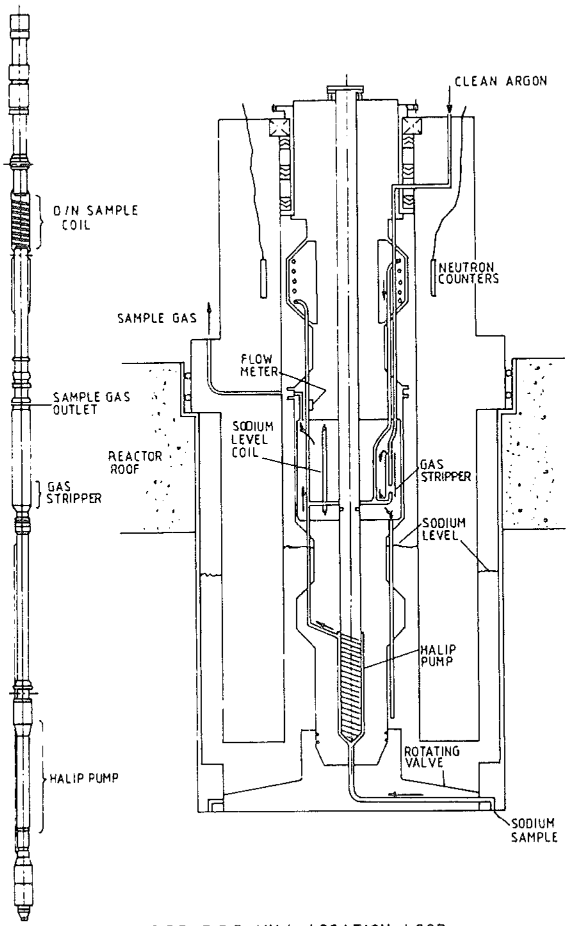

FIG.4. P.F.R. B.P.D. MK.4 LOCATION LOOP.

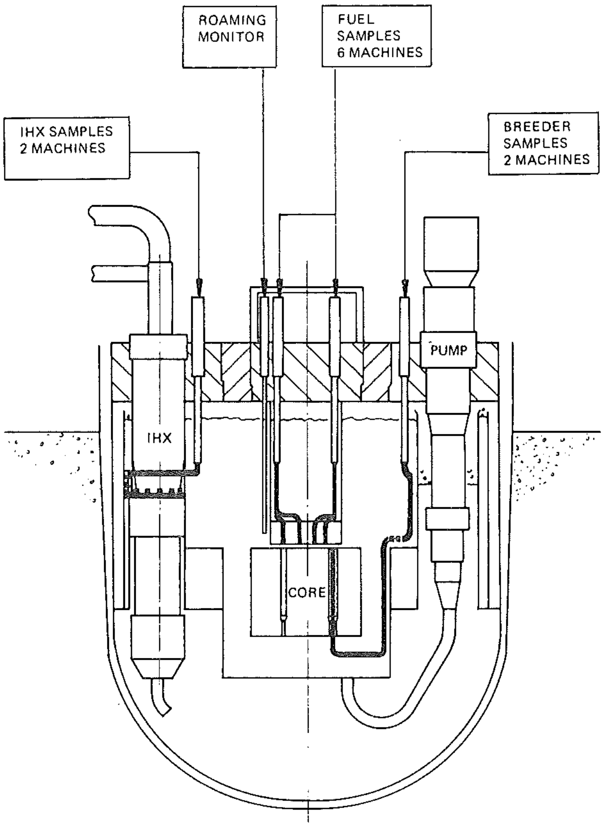

FIG 5 SITING OF PROPOSED CDFR BPD UNITS AND TYPICAL SAMPLE PIPE RUNS

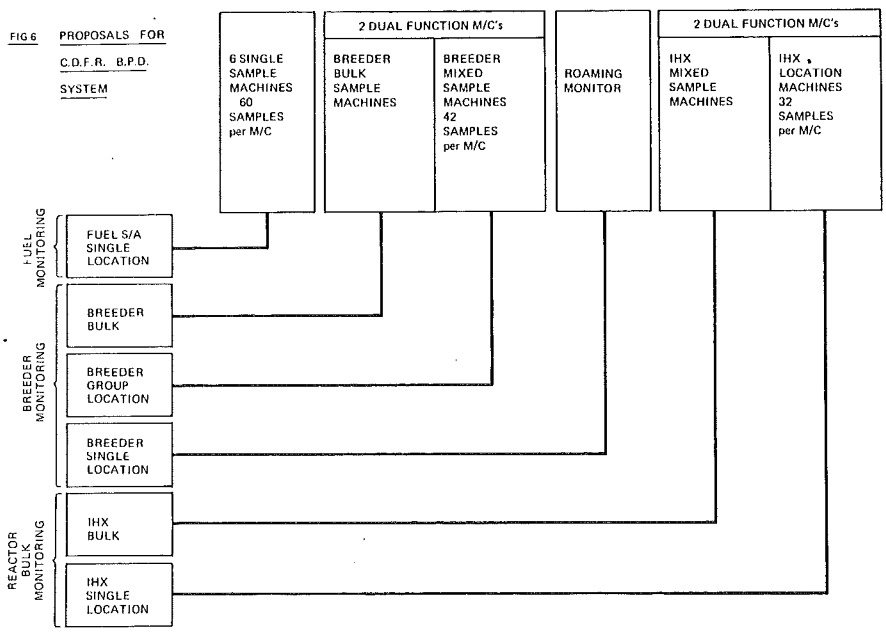

FIG 6 PROPOSALS FOR C.D.F.R. B.P.D. SYSTEM

TABLE 1. Sensitivities of PFR DN Monitors

	IHX	Subassembly	Location
Basic sensitivity $c/n/cm^3$	100	100	5
Relative sensitivity for failure detection	1	1	10

Discussion

D.B. Sangodkar, RRCK:

Your slide on DND location showed a sampling pipe coming from the bottom. What exactly is the arrangement for sampling?

D.K. Cartwright, UKAEA:

Sodium is directed from each S/A to impinge on the neighbouring control or shut off position. Sample pipes run down inside these positions and a sample pump takes samples to the monitor.

F. Gestermann, IA:

How many times do you need to examine the whole core with your FFDL System

a) under continuous operation without pin failure?

b) during location procedure?

D.K. Cartwright, UKAEA:

For DN monitoring the core can be scanned in about 1 hour, with the precipitator about 3 hours.

P. Michaille, CEA:

Can you give some details about the calibration with UO_2?

D.K. Cartwright, UKAEA:

The UO_2 fission source is a small cylinder of UO_2 with high U235 content. It is mounted in a breeder pin position at the outlet of fuel assembly modified to enable this to be done.

Fuel Failure Monitoring Systems in U.S. Breeder Reactors

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J.D.B. Lambert
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ABSTRACT*

The fission-gas activity monitors, the delayed-neutron detectors, and the methods to measure primary-sodium activity are described for the Experimental Breeder Reactor-II (EBR-II), the Fast Flux Test Facility (FFTF), and the Clinch River Breeder Reactor Plant (CRBRP) in the United States. Operating experience with these systems is briefly described.

INTRODUCTION

The U.S. breeder reactor program presently has one operating facility, a new test reactor, and a prototype commercial plant which is nearing the end of its design phase. Each reactor has operating, or planned systems for the detection and monitoring of fuel-pin failures; they are briefly described herein. Parallel papers (1-3) describe U.S. experience in identifying failures.

The operating facility is of course EBR-II, located in the state of Idaho and operated since 1964 by Argonne National Laboratory. It has been used for numerous tests of fuel, absorber, cladding, and structural materials. The reactor also supplies 18 MW of electricity to the local grid, and has been recently cited for a State award as a "co-generation" facility.

The new test reactor is FFTF, which achieved its initial ascent to full power (400 MWt) in December 1980. FFTF is operated by Westinghouse Hanford Company at the Hanford Engineering Development Laboratory, in the state of Washington. FFTF will soon become the major facility in the United States for irradiation of full-size LMFBR fuel pins and components.

CRBRP is the U.S.'s first prototype of a commercial LMFBR. Its design is near completion, but its construction schedule is not firm at this time. CRBRP will be operated by the Tennessee Valley Authority, a publicly-owned utility, and will generate 380 MWe of power.

Slightly different fuel-failure monitoring systems are employed on each of these reactors. In part, these differences reflect the different reactor designs; and in part, different operating philosophy toward fuel-pin failures. For example, EBR-II is a small hybrid-pool reactor; FFTF and CRBRP are larger-size loop reactors. Cover-gas activity due to Ne-23 and Ar-41 is less important at EBR-II than it will be at the other two; the cover-gas monitors differ accordingly.

Similarly, the initial operating philosophy at EBR-II and FFTF has been to detect, locate, and remove subassemblies containing a breached pin. In contrast at CRBRP "gas leakers" will remain in-core and will only be removed when sodium-fuel contact is evident or suspected. However, EBR-II has also recently been used for fuels endurance tests, and several intentional breached-pin tests. Reactor operation has then continued until an administrative limit on delayed-neutron (DN) activity has been exceeded. The results of such tests in the future (4) may allow a more relaxed attitude toward fuel-pin failures.

*Work supported by the U.S. Department of Energy.

COVER-GAS ACTIVITY MONITORS

At EBR-II

The main cover-gas activity monitor now in use is the Germanium-Lithium Argon Scanning System, or GLASS (5) which was developed from previous systems (6). Argon is extracted from the cover gas, aged for ∿4 m. to allow Ne-23 activity to die away, and measured in a 25-mL chamber above a coaxial Ge(Li) gamma detector. Signals are fed into a dedicated NOVA computer which acts as a 1024-channel analyzer. The dynamic range of the system may be increased by 100 by dilution with fresh argon to give an overall signal-to-background capability of 10^4. The activities of Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135, Xe-135m, and Xe-138 are routinely monitored by GLASS.

During normal full-power operation there is a small cover-gas activity due to the fissioning of 2-5 mgm. of "tramp" uranium. This background has remained remarkably constant and is believed due to some fixed contamination in-core which resulted from use of reprocessed driver-fuel pins over 1964-68. Background activity is entirely swamped by fission gas released from breached pins. For example, Fig. 1 shows a typical change in cover-gas activities from failure of an experimental driver-fuel pin. Such failures are characterized by an initial increase in Xe-135m activity; this arises from decay of iodine in the sodium bond that escapes before stored fission gas.

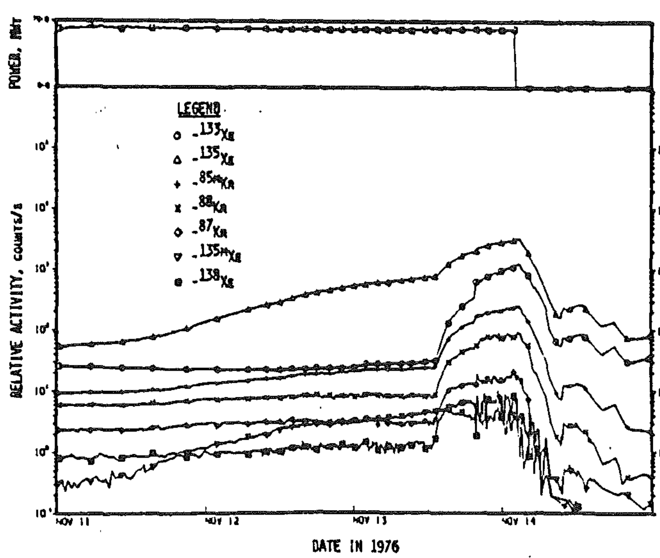


Fig. 1. GLASS Activity Data during a Metal Driver-Pin Failure

At FFTF and CRBRP

As earlier indicated, the likely faster disengagement of entrained gas from sodium in FFTF and CRBRP, and the higher neutron fluxes than at EBR-II, will give substantially higher Ne-23 and Ar-41 activities. Also at CRBRP, "leakers" will remain in-core to give potentially much greater intrinsic cover-gas activity. Thus, two refinements in monitoring will be required over the GLASS at EBR-II: a means to drastically reduce the unwanted neon and argon activities in the sampling stream; and a device to increase the range of activities which can be monitored in it.

The monitoring systems at both reactors will be similar (Fig. 2). As at EBR-II, the stream of active cover gas will be first aged for ∿5 min. to reduce as far as possible the Ne-23 activity. But then it will be passed through a column of activated charcoal held at about 1°C, which has at its end a 5-mm. thick planar germanium diode detector. This collimator/diode arrangement reduces the impact of Ar and Ne activities. Tests at Hanford (7) have shown that no more than 3 gm. of charcoal will produce an enhancement of Xe-133 (81 KeV)/Ar-41 (1293 KeV) of $\sim 3\times10^4$; and $\sim 10^6$ for Xe-133/Ne-23. A similar arrangement will be used at CRBRP. Between the detector and the column is an absorber/collimator device (Fig. 3) which contains five tungsten cylinders located in a steel disc, driven by a stepping motor. Four of the cylinders are collimators, with two containing absorbers in the collimator holes; the fifth contains a calibration source. Change in collimator position is triggered by count rate (8).

The activities of Xe-133, Xe-135, Xe-135M, and Xe-125 will be routinely monitored at FFTF; Xe-125 is the activation product of the stable tag isotope Xe-124, and

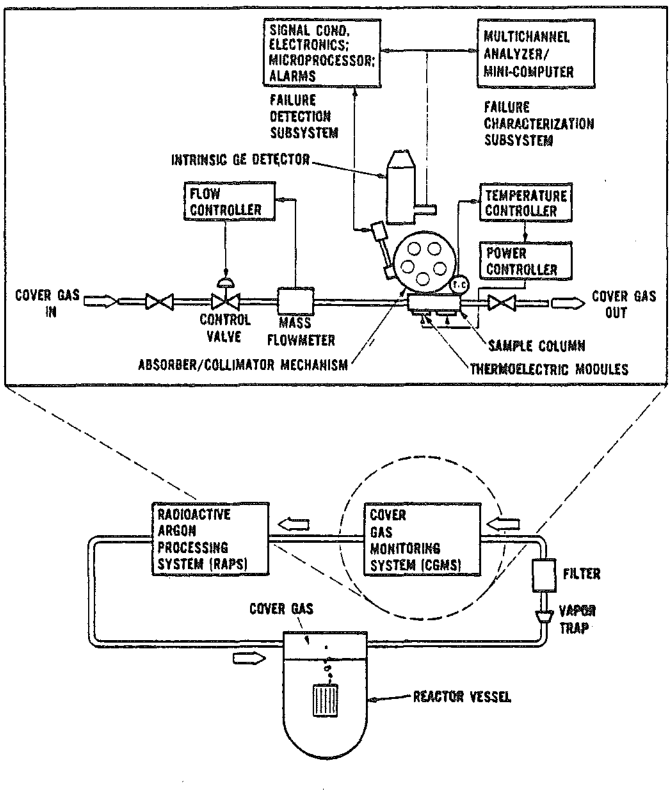


Fig. 2. The FFTF Cover-Gas Monitor

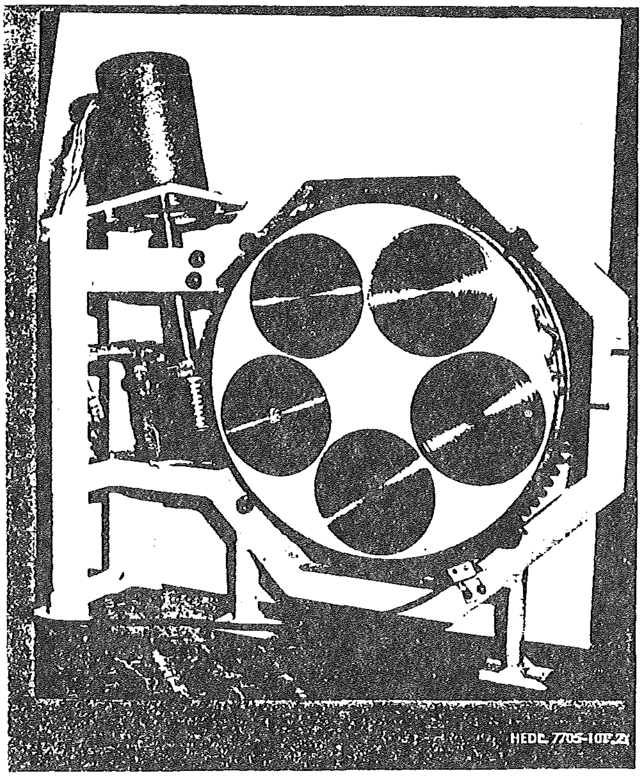


Fig. 3. The Rotatable Collimator/Absorber in the FFTF Cover-Gas Monitor

will be used to indicate failure of a control rod.

The FFTF system was of course functional during the steady rise and hold at full power in December 1980, a length of time sufficient to reach equilibrium in short-lived gas activities; none was detected in measurable amounts. This gratifying result indicated that there was essentially no tramp fissile material in the active core region, a conclusion confirmed by the absence of delayed-neutron (DN) activity in the primary sodium.

DELAYED-NEUTRON MONITORS

The monitoring of DN activity in primary sodium is similar at all three reactors, although because of their different geometries the monitoring stations are located differently. At EBR-II the primary sodium outlet pipe is inaccessible. Therefore a sample stream of sodium is pumped from the discharge of the single intermediate heat exchanger (IHX) to a detection station (FERD) outside of the primary vessel, and back in again to the primary tank. At FFTF and CRBRP there are three detection stations, one on each of the primary outlet pipes, upstream of the pumps. Figure 4 shows the planned arrangement of DN detection systems on CRBRP (9).

In all three systems BF_3 proportional counters are or will be used; they are lead-shielded against primary sodium activity and moderated; several detectors are ganged together to yield appropriate sensitivity. Details of the recently upgraded FERD system at EBR-II are given in a parallel paper (2). At FFTF and CRBRP the detectors are housed as close as possible to the large-diameter primary sodium pipes. The time between DN's entering primary sodium in-core and their detection by the counters is 18-20 s. in EBR-II, 27-28 s. in FFTF, and 38-40 s. in CRBRP. The DN detectors are not in the plant protection system at any of the three reactors, but there is or will be an administrative shutdown limit

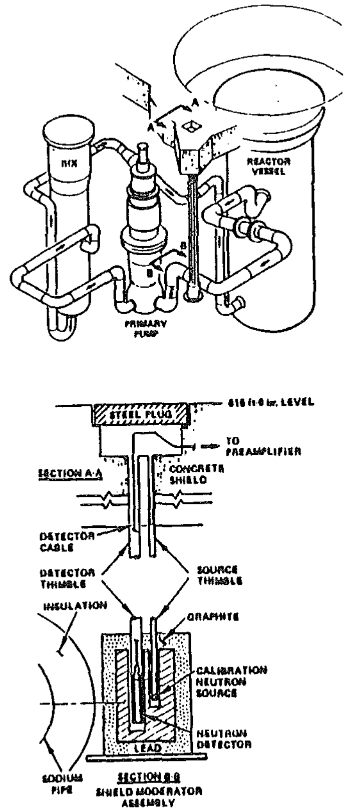


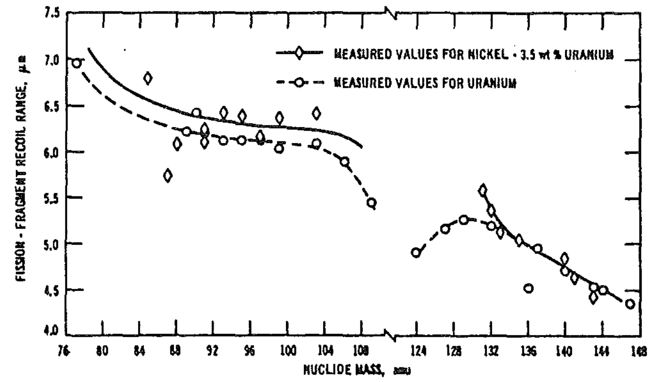
Fig. 4. Configuration of DN Monitors at CRBRP

on DN signal count rates.

At initial full power in EBR-II the background DN signal count rate is 55 cps, rising to ~60 cps over 2-3 days due to build-in of Na-24 activity and (γ,n) reaction in the Benelex moderator. Approximately 40 cps are due to tramp uranium, the remainder being due to the (γ,n) effect, counter off-set, and other minor factors. The tramp count rate is a component which changes promptly with reactor power change, as has been observed in recent power-cycling tests of the driver fuel. The history of DN signals over and above this well-defined background is reviewed in another paper to this meeting.

CALIBRATION AND TESTING OF MONITORING SYSTEMS

The monitoring systems at EBR-II--GLASS and FERD--were calibrated in 1977 (10), and again in 1978 (after an upgrade of the FERD). To do this a recoil source in the form of a dilute uranium-nickel alloy was irradiated in-core for a full reactor run. The recoil source strength of the alloy had been previously measured in the known flux of the thermal reactor CP-5 (Fig. 5a). The equilibrium activities of fission-gas isotopes in the cover gas could then be related to their known source terms, allowing the counting efficiency to be directly calibrated; similarly for FERD (2). Fig. 5b shows how the isotope activities approached equilibrium over the first 1-2 days irradiation. The overall source strength was chosen to allow good counting statistics on both GLASS and FERD. A similar source test is being investigated for the FFTF, using the same alloy material. An added feature of such a calibration will be to proof-test the concept of DN source location by triangulation, using the signals from the detectors on the three primary outlet pipes (11).



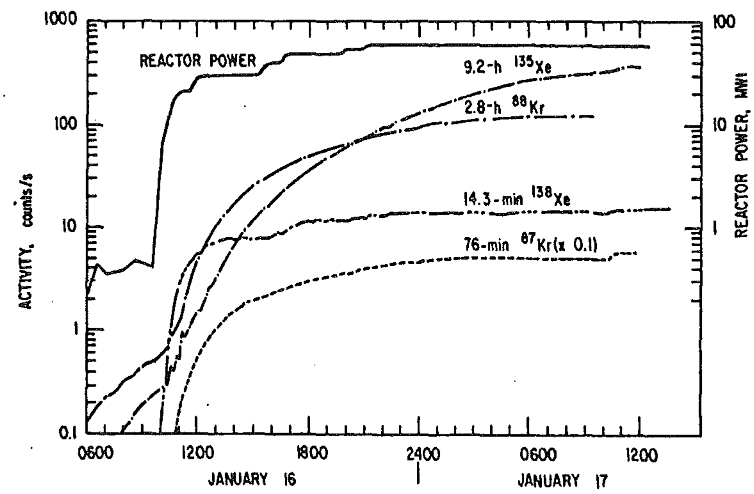


Fig. 5. EBR-II Fission-Product Source Test
a: Recoil range in source material
b: Gas activities approach equilibrium

Meanwhile, the cover-gas monitoring systems for FFTF and CRBRP have been fully tested at EBR-II, making use of the frequent fission-gas releases from breached pins to finalize counter geometry, and other operating characteristics. The FFTF cover-gas monitor has supplied a backup, alternative view of gas releases at EBR-II. In particular, the system has often allowed identification of Xe-125 release in-core, an isotope which GLASS is not set up to monitor. Tests on the FFTF system have been successfully concluded; tests of the CRBRP system are now underway.

PRIMARY SODIUM ANALYSIS

At all three reactors primary sodium has or will be monitored routinely for oxygen, carbon and the trace impurities associated with normal operations; and for fission products and possibly fuel resulting from fuel-pin failures. Although several methods of sampling have been developed at EBR-II (12) the main one used until recently has been a simple overflow sampler located in the sodium purification cell. In this device up to four 10-mL Pyrex beakers are filled to overflowing with

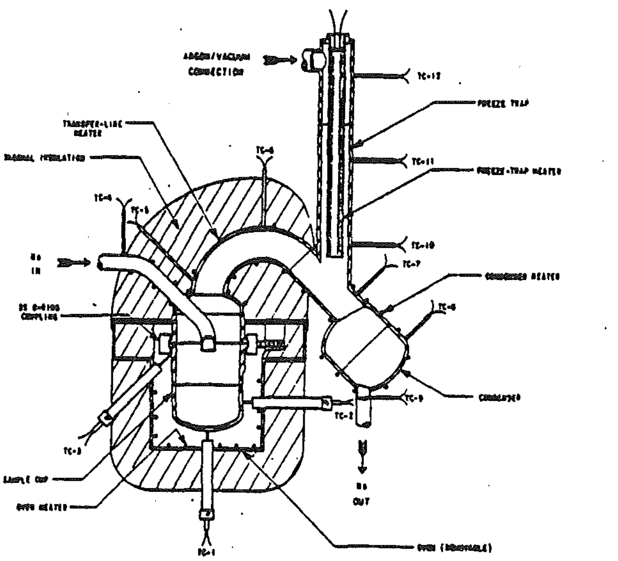


Fig. 6. The On-line Distillation Unit at EBR-II for Analysis of Primary Sodium

sodium in an inert atmosphere; after 5 days cooling the beakers are removed and analyzed. Similar procedures will be adopted at FFTF and CRBRP (13).

Throughout the course of EBR-II history plutonium has never been detected by the overflow sampler to the limit of detectability of $5x10^{-12}$ gm/gm. Similarly, only once or twice has uranium been detected above the detectability limit of $1x10^{-9}$ gm/gm. This experience tends to confirm that exposure of metal or ceramic fuels to primary sodium has been minimal.

With advent of limited breached-pin testing a more sensitive device--the on-line distillation unit or OLDU (14)--was installed and made operational in 1978. In the OLDU (Fig. 6) a sample of about 150 gms of primary sodium is heated and volatalized. Non-volatile impurities are left in the sample cup and may be analyzed immediately, since cooling of the Na-24 activity is not required. With the OLDU operational, plutonium has been detected on one or two occasions above its detectability limit. The highest value occurred in February 1979, when a value of $\sim 1x10^{-10}$ gm/gm of plutonium was measured. This value followed the endurance failure of a fuel pin with a particularly brittle cladding, although no fuel loss was apparent by visual examination or weight loss. The value declined to its detection limit over the next ten days operation, indicating that the fuel was probably dispersed as minute particulates which had gradually settled out in the cold primary-sodium tank.

SUMMARY AND CONCLUSIONS

Fission-gas and delayed-neutron monitors have been developed and thoroughly tested over the years at EBR-II for immediate use at FFTF and eventual deployment on CRBRP.

Cover-gas activity monitors use rugged, well-tested γ-ray detectors to monitor up to seven of the individual fission-gas isotopes released from breached pins. Operating experience has shown these systems to be extremely sensitive and reliable devices, being capable for example of readily detecting the presence of minute amounts of "tramp" uranium in EBR-II. Steps must be taken at FFTF and CRBRP to reduce the high Ne-23 and A-41 activities which arise from loop sodium operation, and to handle the potentially high cover-gas activities expected at these plants. Proof testing of these systems has been carried out at EBR-II.

The method of DN monitoring is the same at all three U.S. reactors, and is similar to monitoring on LMFBR's world-wide. It involves the detection of DN activity in primary sodium by lead-shielded, moderated BF_3 detectors, and associated electronics. EBR-II differs from FFTF and CRBRP in having a by-pass loop of fresh primary sodium to interrogate in this manner; at the loop reactors, of course, monitoring can take place on the hot leg of the three primary outlet pipes. The considerable experience gained at EBR-II (described at this meeting) should be directly of benefit to the other two reactors. The same may be said for the analysis of primary sodium.

The main conclusion to be drawn from U.S. experience to date is that the detection and monitoring of fuel-pin failures in LMFBR's is quite well developed. A parallel paper addresses the state-of-the-art with regard to the identification, or location, of fuel-pin failures.

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Discussion

E.K. Yakshin , SRIAR:

How many sodium probes have been distilled at this method?

F.E. Holt, HEDL:
About 100.

C. Berlin, CEA:

How is plutonium measured in the primary sodium of EBR-II?

F.E. Holt, HEDL:
By distillation of 150 gms of primary sodium & examination of the residue.

C. Berlin, CEA:

Could you define the units of the last pictures you showed us?
(Pu in primary sodium versus to time after release).

F.E. Holt, HEDL:
Units are 10^{-12}gm/gm of sodium.

F. Gestermann, IA:

Do you have any other system to locate pin failures, if the tag-system has not found the tag-gas?

F.E. Holt, HEDL:
At EBR-II they have developed a number of nonspecific methods which were described on Tuesday, a.m. For FFTF-CRBRP there are not systems other than gas tagging for location.

V.F. Efimenko, IAEA:

What is your experience in location of failures in blanket region?

F.E. Holt, HEDL:
No experience to date.

J.C. Cauvin, CEA:

How long is the delay to can make the first control of subassembly by dry sipping?

J.D.B. Lambert, ANL:
∿ 15 days, depending on the subassembly type.

M. Relic, IA:

Have you done experiments to measure fuel element temperature due to heat decay?

J.D.B. Lambert, ANL:
Yes.

D.K. Cartwright, UKAEA:

How long does dry sipping take per assembly?

J.D.B. Lambert, ANL:
Approximately 2 hours.

A. Merkel, IA:

Could you give us the limit for the decay heat during dry sipping?

J.D.B. Lambert, ANL:
It's 1 to 5 kW.

F. Gestermann, IA:

Is it true that in America reactors the DND-system has no influence on the automatic SCRAM system?

F.E. Holt, HEDL:
That is true. EBR-II's DND System was initially connected to its Plant Protection System but was removed when that plant began its Run-To-Cladding-Breach program. FFTF+CABRP DND systems are not connected to the Plant Protection System.

J. Dauk, IA:

Do you have special purification procedures concerning covergas for tagging?

F.E. Holt, HEDL:
A cover gas cleanup system includes a cryogenic still which will remove Xe and Kr from the argon cover gas.

THE RESULTS OF TESTING AND EXPERIENCE IN THE USE OF LEAK DETECTION METHODS IN FAST REACTORS

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Part 1

Introduction

Part 2

PERFORMANCE OF THE DETECTION SYSTEMS AND THE EQUIPMENT USED IN LMFBR'S IN THE USSR

A b s t r a c t

Briefly reviewed are the available LMFBR failed fuel elements detection systems. Presented are the main characteristics (and instrumentation) of delayed neutron and fission gas monitors as well as the instrumentation for failed fuel subassembly control during the reactor shut-down. The features of delayed neutron and fission product release during and following a fuel pin failure are described. Present-day and future research aimed at leak-detection methods improvement are covered. It is concluded that the available failed fuel detection systems prevent the development of failure into a wide-open crack leading to fuel-coolant contact and resulting in contamination of the primary circuit exceeding the permissible limits.

I. INTRODUCTION

Economic efficiency of nuclear power plants operation may be increased by achieving higher fuel burnups and decreasing the number of refuelling outages. On the other hand this results in higher number of failed fuel and longer periods of operation with failed fuel elements present in the core. In this case, operating the power plant with failed fuel elements in the core will lead to deterioration of radiation situation, prolonged maintenance, higher costs of protection measures and decontamination.

For every nuclear power plant the limits on the number and the extent of fuel defects are imposed. Achieving these limits must be followed by the reactor shutdown and failed fuel subassemblies localization and discharge.

As an example, for the BOR-60 plant normal operation with 0.1-0.2 % of failed fuel pins was set allowable in case of removal of long-lived ^{137}Cs and ^{134}Cs nuclides. The fission product activity within the coolant was in this case comparable to that of ^{22}Na and corrosion products and did not affect significantly the radiation situation.

The radiation safety of a nuclear power plant may be ensured provided the activity detection of such long-lived radionuclides as ^{131}I, ^{134}Cs, ^{137}Cs, ^{95}Nb, ^{140}Ba both in the coolant and in the circuit will be carried out as well as with the knowledge of the relationship between the release fractions of the above radionuclides and the measured values of delayed neutrons and fission gas activities.

For the operated nuclear power plants the possibility has been provided to continue the operation with the occurrence of pin (gaseous) leaks in the fuel cladding. An increase in the delayed neutron detector signal is considered to be an indication of direct fuel-coolant contact requiring reactor shutdown to discharge failed fuel subassemblies. However, the BOR-60 experimental data shows that even at constant delayed neutron signal the activity buildup for such radionuclides as ^{137}Cs, ^{134}Cs, ^{131}I occurs up to undesirable level. The release fraction of ^{137}Cs was 20-40 % on the average in BOR-60.

The knowledge of fission product release, regularities of fission product removal from the circuit through the clean-up systems combined with continuous control over their level in the circuit make possible operating the power plant with an imposed limit of failed fuel elements.

In the USSR the following three techniques are used for defected fuel pins detection and localization in the core :

- delayed neutron control in the primary circuit,
- fission gas control in the cover gas,
- fuel subassemblies control in the core during reactor outages.

In addition, sodium and cover-gas samples are taken from the primary circuit to control fission product concentrations.

The reactor is properly instrumented and equipped for every of the above control techniques. The control equipment and technology are improved. Fission gas detection in cover gas was formerly accomplished with gas-discharge counters and scintillation detectors. Now Ge(Li) detectors are used as well as the electrostatic precipitation technique for fission gas decay daughters : ^{88}Rb and ^{138}Cs. Subassembly detection was initially carried out only from fission gas control. In the last few years nongaseous fission products are also measured for this purpose.

Selective detection of fission products in the coolant against the ^{24}Na background under reactor operation holds considerable promise for the purpose.

The information of this paper is to a large extent based on the BOR-60 data. This is due to the fact that the BOR-60 reactor was specially created for investigations, testing fuel materials, fuel element design and its performance under real power plant conditions.

2. PERFORMANCE OF THE DETECTION SYSTEMS AND THE EQUIPMENT USED IN LMFBR'S IN THE USSR

Delayed neutron detection system makes possible continuous control and is designed to guard against the accidental fuel element failure and considerable contamination of the circuit when a fuel element failure leads to a direct fuel-coolant contact.

Neutron detectors (corona counters) are placed into the polypropilene moderator in the vicinity of the primary loop. The detectors may operate at 100°C and in gamma-radiation fields up to 1500r/hr at a maximum irradiation dose up to $5 \cdot 10^7$rad and 10^{12}n/cm^2. Maximum sensitivity of the detectors is (0.6-1) pulses·cm^2/hr. The time of coolant transport from the core to the delayed neutron detectors location is about 30s. The system sensitivity is about 50pps/cm^2 of fuel-coolant contact surface area. Initial background of the system in BOR-60 reactor was determined from the leak neutrons (about 100pps).

Cover fission gas control technique is also well known. So it will be outlined briefly here.

Cover gas is taken from the reactor to a special loop containing a sodium cold trap, a pump and measuring section. For detecting beta-counters are used mounted along a stainless steel tube or on rubidium and cesium electrostatic precipitation cell. The delay time is chosen such that the short-lived ^{23}Ne may be sufficiently decreased. To better the measurement information NaI(Te) crystal gamma spectrometers and Ge(Li) detectors are used.

The method is characterized by high signal-to-background ratio and sensitivity. However, the procedure of the analysis of the detected radionuclides activity dynamics requires to be improved.

Sampling of cover gas and primary sodium is applied for detection equipment calibration, for measuring both cover gas low active fission gases and solid fission products (I, ^{140}Ba, ^{140}La, ^{134}Cs, ^{137}Cs, ^{95}Zr radionuclides etc.) in sodium. Sampling is accomplished with special sampling devices. The samples are analyzed with Ge(Li) detectors.

The delayed neutron signal on the reactor control panel may be used to come to a decision about an unscheduled reactor shutdown.

Three methods of subassembly detection during the reactor schutdown are known. In the USSR reactors two procedures are used: the failed fuel subassembly detection both in-situ and after their discharge into a flask. In future it is suggested to detect failed fuel subassemblies on their removal into the gas. Fig. 1 shows a flow diagram of a special BOR-60 failed fuel subassembly detection facility. The cover gas is passed through the activated charcoal absorber prior to detection. Then a special tube is lowered and placed over a core fuel-less channel for bubbling the coolant. Pure argon is supplied to the lower end of the tube and the coolant enclosed in the tube is degassed to decrease the measurement background. During the detection the lowered tube is tightly placed on the upper end of the subassembly and sodium is displaced with argon. After fitting the tube on the fuel bundle wrapper sodium coolant is additionally displaced. In such a condition the detected subassembly is held for a period required to heat it. Then the tube gas section is connected to the measuring loop. The wrapper and lowered tubes are filled with sodium. The sodium in the lowered tube is bubbled with argon over the subassembly. The bubbling time depends on the time of the tube gas change and sodium degassing. The blow-gas is passed through a particulate filter and a measuring cell equipped with beta-detectors.

The conclusion about failed fuel pins is drawn from the fission product activity in the blow-gas. The fuel subassemblies are heated due to fuel residual heat. This imposes limits on the schedule of the onset of the fuel subassembly detection. It is usually initiated in 6-10 days after the reactor shut-down and assures the fuel subassemblies temperature rise up to 600-650°C in 3-5 min. The fuel subassemblies heating time is calculated to within 10-15%.

After all the fuel subassemblies are checked those with suggested failed pins are selected. To this end, the criterion is used of successive exclusion of those resuits which fall out of the normal distribution rule for the obtained data set. It is suggested that a significant deviation of blow-gas fission product activity levels are due to the failed fuel subassemblies. With a great number of failed fuel subassemblies their selection by the above criterion becomes less trustworthy.

In the BOR-60 plant the reliability of true failed fuel subassembly selection was increased due to non-volatile fission product measuring in addition to that of fission gases.

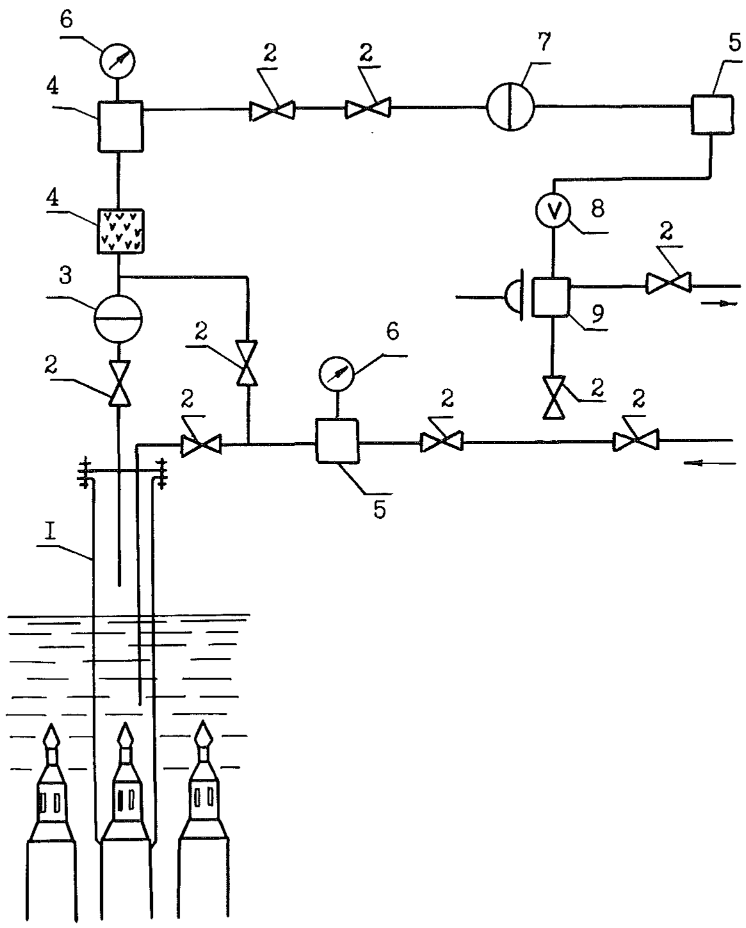

Fig. 1 Flow-diagram of failed fuel subassembly detection system
1 - movable tube, 2 - valve, 3 - particulate filter, 4 - sodium cold trap, 5 - dosage vessel, 6 - gas gauge, 7 - particulate filter, 8 - flowrate meter, 9 - measurement vessel.

Discussion

D.K. Cartwright, UKAEA:

Are systems on BN 350 and BN 600 similar to the BOR 60 you described?

E.K. Yakshin, SRIAR:
These reactors have similar systems.

D.E. Mahagin, HEDL:

In BOR 60 is there a level of coolant contamination by fission products that limits the power level or causes shutdown?

E.K. Yakshin, SRIAR:
We are working on this question - status will be discussed by V.I. Schipilov at this meeting.

P. Michaille, CEA:

Did you observe fuel loss from defected pins?
Did you measure some solid fission products on the circuits (Zr, RU)?

E.K. Yakshin , SRIAR:
We measured Ru-106,Ba-La-140, Nb-95, Ce-141. They were predominately measured on the surfaces.

J.D.B. Lambert, ANL:

You say you watch DND and fission gas activity together, is this to watch for other failures?

E.K. Yakshin , SRIAR:
Yes, but only in BR5 reactor, and did not in BOR 60.

D.K. Cartwright, UKAEA:

Please describe the technique used for locating failures in BOR 60.

E.K. Yakshin, SRIAR:
The answer to this question is in my paper. I omitted it in my presentation to save time.

M. Relic, IA:

a) Can you control the temperature of the fuel element during your localisation?
b) What can you say to the tightness between the location tube and the fuel element head?

E.K. Yakshin, SRIAR:
a) Only for experimental subassemblies and for typical fuel subassembly partly.
b) The tightness is controlled by level of argon pressure. We say O.K. when the level is constant.

P, Michaille, CEA:

When you say that the DND threshold correspond to 1 cm^2 of failure, how did you calibrate it?

E.K. Yakshin, SRIAR:
This is a calculated data.

F. Gestermann, IA:

Is your DND-system only capable for controlling the development of pin failures or is there any automatic SCRAM-level?
When do you decide to shut down the reactor?

E.K. Yakshin, SRIAR:
In our DND-system we have alarm and automatic SCRAM levels. They depend on DN background.

PRESENTATIONS AND DISCUSSIONS

OF SESSION II:

EXPERIENCE FROM LMFBR's

EXPERIENCE OF FAILED FUEL DETECTION AND LOCALIZATION AT KNK II

by

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ABSTRACT

The German KNK II experimental fast breeder reactor has two different systems for detecting faulty fuel elements, namely the measurement of delayed neutrons (DND system) and the examination of the primary cover gas for radioactive noble fission product gases. The control system for the primary cover gas is installed in a separate circuit and consists of two NaI(Tl) detectors with amplifiers and rate meters, an on-line γ-measuring station with high purity germanium detector, multi-channel analyzer and computer, and a sampling station.

Since KNK II was commissioned in October 1977, there have been total of two failed fuel elements. The first defect occured on April 1, 1979 and was discovered by a sudden, step rise of fission product activity in the primary cover gas. When the plant was subsequently shut down, increased signals were also observed in both DND monitors.

The second fuel element defect starts on May 21, 1980 as a "leaker", i.e., with an increased gaseous fission product activity in the primary cover gas without, at the same time, giving rise to a DND signal. The leaker phase lasted for 30 days of reactor operation at 96% power until, on August 19, 1980, the leaker developed into a fault generating a DND signal; accordingly the reactor was scrammed after the DND limit had been exceeded.

Prior to localization and unloading of the second failed fuel element, the plant had been operated with this "open" fuel rod defect at 40% power for two weeks in order to allow the development of the DND signal and the gaseous fission product activity to be followed under this conditions.

Especially the following techniques were used after the second fuel failure for pre-localization of the defective fuel element concerned:

(1) Determining the $^{131}Xe/^{134}Xe$ ratio in the primary cover gas (distinction between plutonium bearing test zone and the non-plutonium bearing driver zone).

(2) Operation with skewed absorber position ("flux tilting") at 15% reactor power (to narrow the area with the defective fuel element by way of different signal levels of the DND monitors at the two primary sodium outlet pipes).

(3) Comparison of the two DND signals at the primary sodium outlet pipes.

(4) Precise monitoring of the outlet temperature and calculating of the burnup of the elements in the plutonium bearing test zone.

Testing this pre-localization methods were so successful that the defective fuel element causing the second fault was found within three days after start the normal localization technique by a modified dry sipping technique.

This report describes the experience accumulated in the detection, pre-localization and localization of two defective fuel elements in KNK II.

1. INTRODUCTION

KNK II went first critical on October 10, 1977 and attained its full power of 58 MWth (= 20 MWe) on March 3, 1979.

The KNK II core (Fig.1) consists of a central UO_2/PuO_2 test zone with 7 fuel elements surrounded by a UO_2 driver zone with 22 fuel elements. The fuel of the test zone contains 30% Pu and 83/93% enriched uranium. The uranium in the driver zone is enriched to 37%. A summary of the core and fuel element data is shown in Table 1. The reactor is controlled by five control rods; three additional absorber rods provide redundant reactor protection.

The fist fuel element failure occurred on April 1, 1979 after 65.4 full power days and an average burnup of the elements in the test zone of 11,6 MWd/tHM [1].

The second failed fuel element intitially occurred May 21, 1980 after 158 full power days and an average burnup in the test zone of 27,5 MWd/tHM, appearing as a "leaker" in the test zone. After another 30 full power days the leaker developed into a failure with a DND signal during normal start up operation.

The reactor was then operated for another two weeks at 40% reactor power with this defective fuel element which, when localized by the dry sipping technique, had thus attained an average burnup of 39,1 MWd/tHM.

KNK II has two systems for failed fuel element detection. The first system consists of two DND (= Delayed Neutron Detector) monitors, one of each is attached to the two main primary sodium pipes (Fig.2) downstream of their outlet from the reactor vessel. Each monitor contains eight ^{3}He counters connected in pairs to an amplifier. Three pairs make up three redundancies in the DND system, one pair serves for standby purposes. The counting tubes are enclosed in a moderator block made of polyethylene and are strongly shielded against γ-radiation and core neutrons. Each DND measuring station has a γ-dose rate monitor and is equipped with a temperature sensor. If a limit value of the DND counting rate of 2000 cps is exceeded, the reactor will be scrammed. Alarm is given by 1500 cps.

The second system consists of measuring the fission product gas concentration in the primary cover gas. For this purpose, the KNK-II facility has a separate gas circuit (Fig.3) for failed fuel detection. It contains two NaI(Tl) detectors equipped with amplifiers, single-channel discriminators and ratemeters, an on-line γ-measuring station with high purity germanium detector, multi-channel analyzer and microprocessor as well as a gas sampling station ("grab sample"). The amplifier and the discriminator are set to the 80 KeV γ-peak of ^{133}Xe. The single-channel discriminators of the four amplifiers of the second NaI(Tl) detector are set to ^{133}Xe, ^{135}Xe, ^{87}Kr, and ^{41}Ar.

II. BEHAVIOUR OF THE FAILED FUEL DETECTION SYSTEMS DURING KNK II OPERATION WITH INTACT FUEL ELEMENTS

II.1 DND Monitor

KNK II had been operated for 65.4 full power days without any failed fuel elements until April 1, 1979.

In that period of operation, the background counts rates of the DND monitors were determined at various power levels (30 - 100%). This indicated that the measured background signal was one order of magnitude below the level precalculated under conservative assumptions. Out-of pile experiments were conducted to find that the sensitivity of the KNK II DND monitors was 0.0023 counts per emitted neutron. Including KNK II parameters and an assumed surface contamination on the KNK II fuel elements of 10^{-7} g/UO_2/cm^2, the recoil model according to [2] was used to calculate the expected count rate as a function of various reactor power levels when operating the plant with intact fuel elements. The result is shown in Fig.4. For comparison, also the count rates measured at the respective power levels are plotted after reaching the ^{24}Na sasuration activity. This also shows that the actual surface contamination of the fuel elements of the KNK II/1 core is considerably lower, amounting to approx. 1.4×10^{-9} g UO_2/cm^2.

In KNK II, the sodium flow is constant up to 30% reactor power. It is only above this level that the flow increases parallel with the reactor power. The transport times from the reactor core to the positions of the DND monitors drop from 73 sec at 30% reactor power to 22 sec at 100% power.

Analysis of the development as a function of time of the DND signals after reactor scram allowed the follwing contributions to the bulk background signal at 96% reactor power to be determined:

(γ,n) reaction of ^{24}Na	:	73 cps
surface contamination	:	26 cps
core neutrons	:	11 cps
bulk count rate	:	110 cps

For KNK II it was provided the reactor to be shut down as soon as a geometric fault area of 8 cm^2 was exceeded. Taking into account a background count rate of 110 cps, this results in a shutdown level of 2000 cps. The resultant detectable geometric fault areas for alarm and shutdown levels for KNK II are listed in Table 2.

II.2 Cover Gas Monitoring

The gaseous fission product concentration in the primary cover gas was measured both with γ-spectrometer and by means of gas samples at various reactor power levels up until April 1,1979. The results are summarized in Table 3. It should be noted that the saturation concentration for ^{133}Xe and ^{133m}Xe when operating with intact fuel elements has not yet been reached because of the lengths of the periods of operation so far. The maximum levels so far measured under these operating conditions were $33.0 \times 10^3 kBq/m^3$ for ^{133}Xe and $1.4 \times 10^3 kBq/m^3$ for ^{133m}Xe.

The reason for these gaseous fission product concentrations in the primary cover gas when operating with intact fuel elements is the surface contamination of the fuel rods ("background fission source"). The fission gas release from sodium into the cover gas was calculated for KNK II from the measured gaseous fission product concentrations. The release rate determined amounts to $A = 1.6 \times 10^{-5} s^{-1}$, which corresponds to a delay in the leakage of 17 h.[3]. The surface contamination determined from the gaseous fission product concentration agrees with the values calculated from the DND signal.

Fig.5 is a plot of the dependence of ^{41}Ar and ^{135}Xe activity concentrations in the KNK II primary cover gas when operating with intact fuel elements. For ^{85m}Kr and ^{88}Kr, this dependence has a similar shape. The non-linear dependence of the gaseous fission product activity on reactor power above 80% could be due to the major differences in the sodium flow rates between the test zone and the driver zone and to the well-known KNK II gas transport problem, respectively [3].
The first operating experience with the cover gas monitoring system for failed fuel element detection dates back from the operating phase of KNK I (KNK with a thermal core until 1974). At that time, there were no on-line γ-spectrometers. It was found that the measured count rates of the first NaI(Tl) crystal in that system were falsified considerably at the 80 KeV peak (^{133}Xe) by the high ^{23}Ne activity. This disturbance was reduced by the installation of a delay vessel between the reactor plenum and the NaI(Tl) device, which increased the transit time from the reactor plenum to measuring device from 6 to 15 minutes. Since then, the spurious influence has been removed, but it is no longer possible now both with gas samples and the on-line spectrometer, to measure the concentration of ^{23}Ne and very short-lived fission products.
At 97% reactor power and intact fuel elements, the first NaI(Tl) detector produces a counting rate of 2.0×10^2 cps, which corresponds to a ^{133}Xe concentration of $33.0 \times 10^3 kBq/m^3$ in view of the efficiency of that measuring device.

The second NaI(Tl) detector in the cover gas monitoring system under the same reactor conditions shows a counting rate of 2.5×10^2 cps for ^{41}Ar and 1.3×10^2 cps each for ^{85m}Kr, ^{87}Kr and ^{135}Xe.

III. BEHAVIOUR OF THE FAILED FUEL DETECTION SYSTEMS IN THE PRESENCE OF A FAILED FUEL ELEMENT

III. 1 First Fuel Failure of April 1, 1979

On April 1, 1979, the counting rates of the first NaI(Tl) detector of the cover gas monitoring system rose to > 10^4 cps within 25 minutes, and those in the four channels of the second NaI(Tl) detector rose by a factor of 60 - 100 in the same time (Fig.6). The on-line γ-spectrometer only supplied one more measurement after the increase of the primary cover gas activity, because the upper limit of the counting rate of the detector was subsequently exceeded. Two gas samples could only be extracted in a makeshift way through the feed line to the gaschromatograph because,as a consequence of a leak in the rotating plug of the reactor vessel the containment could not be entered for increased radioactivity and thus the gas sampling station of the cover gas monitoring system was no longer accessible. Both gas samples had such high radioactivities (D_γ = 2 r/h) that they could measured and evaluated in an off-line Ge(Li)-spectrometer in the laboratory only after serveral days of decay time. From the results it was possible to calculate among other facts that in this fault characterized by a sharp burst of fission product gas 9.4×10^9 - 11.1×10^9 kBq of ^{133}Xe and 6.3×10^9 - 7.4×10^9 kBq of ^{135}Xe were released from the defective fuel rod within 25 minutes. Under the assumption of only one rod being defective, this release provided a first indication of the element involved being located in the area of the plutonium bearing test zone.

Fig 7. shows the development of the DND signal in this defect. Simultaneous with the fission gas burstobserved, only the DND monitor at the primary sodium outlet pipe 1 ("DND east") shows an increase in the counting rate by approx. 1-2 cps. 1h later, the first DND burst was observed in the two monitors. Because of the cover gas activity in the containment, the reactor was stepwise shut down by gradual power reduction 1.5 h later. Thirty minutes after the onset of power reduction the largest DND burst was found in the two monitors, which was followed by three other signal peaks at intervals of 1.5h each. For subseqent pre-localization it was very important that the signal of the DND monitor coming from primary sodium circuit 1 (DND east) was 4.2 times higher than that coming from primary sodium circuit 2 (DND west). Calculations performed to determine the geometric fuel area (recoil model,K=10) indicated values of 1 - 1.2 mm^2.

In the meantime, the defective fuel element has been examined in the hot cells of the Karlsruhe Nuclear Research Center; one pin of this assembly was partly filled with sodium. The exact location of the defect has not yet been found.

III.2 Second Fuel Failure of May 21 and August 19, 1980 respectively

On May 21, 1980 a rise in the cover gas activity was observed during 100% power operation. It reached a factor of 10 below the concentrations measured after the first fuel failure. This was the first "leaker" experienced in the KNK II operation. By August 19,1980 there had been no increase in the DND signal but only for the cover gas activity. After the beginning of the leaker, the following period of operation at 96% power for 30 days exhibited two phases with different patterns of behaviour of the leaker. The first phase lasted for 16 days and was characterized by periodic gaseous fission product bursts at intervals of 1 -1.5 days. Fig.8 shows the counting rates in this phase as measured by NaI(Tl) detector set at ^{133}Xe. Fig.9 shows a section of the ^{135}Xe concentration curve as measured by the on-line γ-spectrometer. In this phase, a total of 2.0 x 10^9 kBq ^{133}Xe and 3.2 x 10^9 kBq ^{135}Xe were released.

In the second phase, which lasted for two weeks under the same reactor conditions, there were only four prolonged gaseous fission product bursts (some of them lasting up two days). Fig.10 shows the typical development of the gross ^{133}Xe counting rate as measured by the first NaI(Tl) detector in this phase. The activity released in this second phase amounted to 1.3 x 10^9 kBq of ^{133}Xe and 2.2 x 10^9 kBq of ^{135}Xe. By the end of the leaker phase, this had accumulated to activity release of 3.3 x 10^9 kBq of ^{133}Xe and 5.4 x 10^9 kBq of ^{135}Xe. These levels indicate that the fission product gas inventory generated was largely blown off in the first phase.

All detectors of the cover gas monitoring system including the gas sampling station worked well und permitted continuous balancing of the fission product gas quantities released.

On August 19, 1980, the plant was scrammed thirty minutes after having attained 95% power, because the DND signal limit setting in the circuit 1 monitor (DND east) had been exceeded. The signal developments in the two DND monitors are shown in Fig. 11. The curves indicate that the clear increase in the DND signal began 30 minutes before reaching the set limit and that the signal of the monitor in circuit 1 was slightly higher than that of circuit 2.

IV. BEHAVIOUR OF THE FAILED FUEL DETECTION SYSTEMS OVER TWO WEEKS OF OPERATION WITH DND SIGNALS

A probable geometric fault area of 0.5 cm^2 was calculated for the second failed fuel element from the DND signal by means of the recoil model using a K-factor of 10. This relative small failure and the objective to observe the development of the DND signal as a function of time for subsequent correlation between DND signal and fault size (following post-irridation examinations) led to the decision to continue reactor operation for two or three weeks at 40% rated nuclear power with open fuel (i.e., with DND signal) up until a DND signal of approx. 1000 cps. The development of the DND signal in the two weeks of this operation mode is plotted in Fig.12. Raising the plant to 40% power led to a short DND burst. The signal then stabilized for four days at 80 - 100 cps and within another five days rose to 400 and 600 cps, respectively. At this level, the signal remained constant until the end of this operating phase. The geometric fault area calculated for this counting rate under the same conditions was found to be 1.3 cm^2.

The cover gas monitoring system only showed some short minor fission product gas burts in the first 3 - 4 days of this operating phase. Afterwards, the fission product gas concentration in the cover gas stabilized at a constant level.

V. EXPERIENCE IN KNK II IN THE PRE-LOCALIZATION OF DEFECTIVE FUEL ELEMENTS

Three methods of pre-localization of defective fuel elements are used at KNK II:

(1) Determining the $^{131}Xe/^{134}Xe$ ratio in the primary cover gas.
(2) Operation with skewed absorber position ("flux tilting") at 15% nuclear power while observing the DND signals of the monitors at the two primary sodium outlet pipes (DND east and DND west).
(3) Comparison of the DND signals from primary circuits 1 and 2.

Method (1) can be used already if the fault is a leaker; method (2) and (3) require a DND signal resulting from the size of the fault.

V.1 Determination of $^{131}Xe/^{134}Xe$ Isotope Ratio in the Primary Cover Gas of KNK-II

For the determination of Xenon isotope ratios in the KNK primary cover gas a combination of gas chromatography and mass spectrometry was used. With this procedure up to six selected Xenon isotopes are monitored continuously and the results processed by a lab data system. For gas concentrations in the 10 - 30 vpb range (i.e. 10^{-7} volume percent range) isotope ratios may be determined directly with a standard deviation better than $\pm$ 5%.
Taking into consideration the fission yields, neutron energies, irradiation and sampling history the ratio of $^{131}Xe/^{134}Xe$ allows a clear distinction of fuel element failures from PuO_2 containing test elements and enriched uranium driver elements.

The technique was applied to the two fuel element failures of KNK II. During the 1979 failure a $^{131}Xe/^{134}Xe$ ratio of 0.437 $\pm$ 0.001 and 0.439 $\pm$ 0.004 was observed. Interpretation based on EBR-II experimental data [4] suggested a failure in the driver zone. More accurate data (iodine-131-decay) after the dry sipping confirmed the defect of a test element (calculated ratio 0.422 for test element, 0.383 and 0.375 for driver element).

In 1980 during the appearance of a leaker the isotope ratio was measured over a period of serveral months. The $^{131}Xe/^{134}Xe$ ratios are shown in Table 4 including concentration values for ^{131}Xe. From the data (ratio 0.46 ± 0.02 the leaker could be clearly assigned to the test zone. The corresponding isotope ratio during "dry sipping" was found to be 0.444 ± 0.002 and 0.46 ± 0.02, respectively.

V.2 Operation with Skewed Absorber Position (Flux Tilting) at 15% Nuclear Power

Arrangement of the five absorber rods of the reactor control system in the core and the geometric arrangement of the two primary sodium outlet pipes with the DND monitors relative to the cross section of the reactor core (Fig.13) allows the regions of test zone and driver zone to be subdivided into four quadrants by skewing the power distribution, in which quadrants more DND emitters are released in case of a fault under the respective skewed absorber position, which will then lead to different signal levels in the DND monitors of primary circuits 1 and 2.

The result of this flux tilting operation after the second fuel failure with four different skewed absorber positions is plotted in Fig.14. The results clearly pointed at a defective fuel rod in the south-eastern sector of the core, which was later confirmed by localization.

V.3 Comparison of DND Signals in Primary Circuits 1 and 2

The geometric arrangement of the two primary sodium outlet pipes mentioned above, with the DND monitors positioned relative to the reactor core, can lead to different DND signals in the two monitors, depending on the position of the defective fuel element and the sodium flow distribution above the reactor core. This has also been found in the two KNK-II fuel failures encountered so far, however with major deviations in the difference between the two signal levels; the conclusions resulting from this finding with respect to pre-localization were confirmed in subsequent localization.

VI. EXPERIENCE IN FAILED FUEL LOCALIZATION AT KNK II

Faulty fuel elements are localized by a modified dry sipping method (Fig.15). Localization is possible only with the reactor shut down. The fuel element to be tested is pulled out of the core above the sodium level, flushed with hot argon and then drawn into a separate flask, capable of evacuation. After cooling with argon, this procedure is interrupted briefly and the flask with the fuel element is evacuated to 0.2 bar. The gas evacuated is collected and measured with NaI(Tl) detectors and, by way of gas sample with subsequent evaluation, counted in the laboratory in a Ge(Li) detector with multi-channel analyzer. Handling and measuring a fuel element takes four hours. The activity concentrations of ^{85}Kr and ^{133}Xe are measured in the collecting fission gas; in the grab samples, also ^{131m}Xe is measured.

After the first fuel element defect, 11 driver elements and 4 elements of the test zone had to be unloaded, also for insufficient pre-localization, until, 11 days after the start of the examination the failed fuel element was localized in position 203 of the test zone (Fig.16).
The pre-localization process described under V. above produced good results. After the start of the localization test the second failed fuel element had been indentified in position 202 of the test zone. The activity concentrations measured in this case by dry sipping are listed in Table 5.

VII. CONCLUSIONS

The instrumentation for fuel failure detection at KNK II proved to be useful. According to the experience from the first fuel element failure the systems were improved so that during reactor operation with the second fuel failure more detailed informations were available.

Amendments in the pre-localization resulted in a localization time of two days for the second failure compared to eleven days for the first failed bundle.
A pre-localization is difficult as long as the fuel element failure is a fission gas leaker only.
So far, it is necessary to remove each fuel bundle from the core into a special flask for a dry sipping test. However, a new device is in preparation according to the method described in [5], which will avoid the removal of the tested fuel bundle from the core.
At present, the cover gas monitoring system is extended and will be equipped with three different precipitators, a Xe-adsorption system including a NaI(Tl) detector, a gas chromatograph also including a NaI(Tl) detector, and a new on-line γ-spectrometer with an instrinsic germanium detector, automatically operated collimator, multichannel analyzer and a computer.

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Proceedings of the International Conference on Fast Reactor Power Stations
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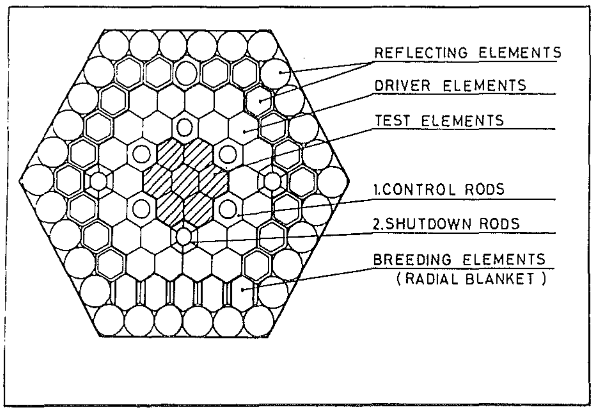


Fig. 1: Core KNK II

Power	55 MW_{th}
Sodium Velocity	
Test Zone	5.5-6.0 m/s
Driver Zone	2.2-2.4 m/s
Total Sodium Flow through the Core	996 t/h
Av.Core Diameter	624 mm
Core High	600 mm
<u>Test Zone</u>	7 Elements
Pins per Element	166/211
Fuel	164 Kg PuO_2(30% Pu)/UO_2(63/93% ^{235}U)
Pin Outer Diameter	6.0 mm
Pin Inner Diameter	5.2 mm
Av.Fast Neutron Flux	1.4×10^{15} n/cm^{2}s
Power Density	183 MW/t_{HM}
<u>Driver Zone</u>	6/16 unmoderated/moderated Elements
Pins per Element	121/102
Fuel	525 Kg UO_2(37/41% ^{235}U)
Pin Outer Diameter	8.2 mm
Pin Inner Diameter	7.2 mm
Power Density	60 MW/t_{HM}

Table I: CORE DATA KNK II/1

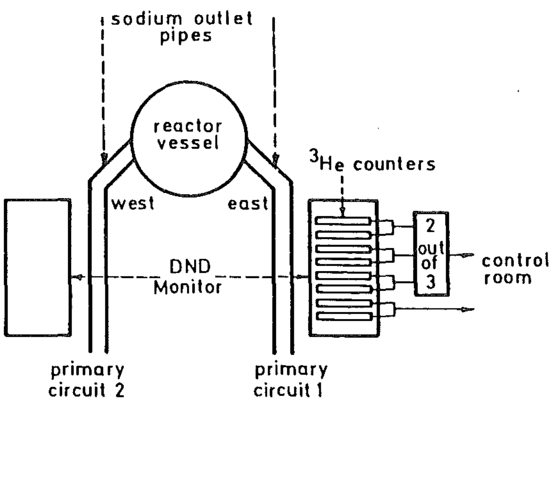


Fig. 2: DND Monitoring System KNK II

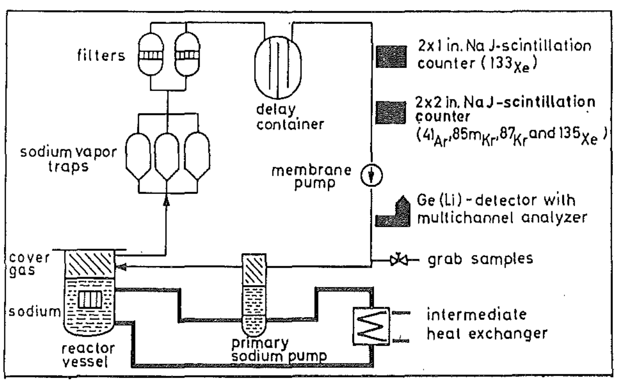


Fig. 3: Failed Fuel Detection System on the Cover Gas KNK II

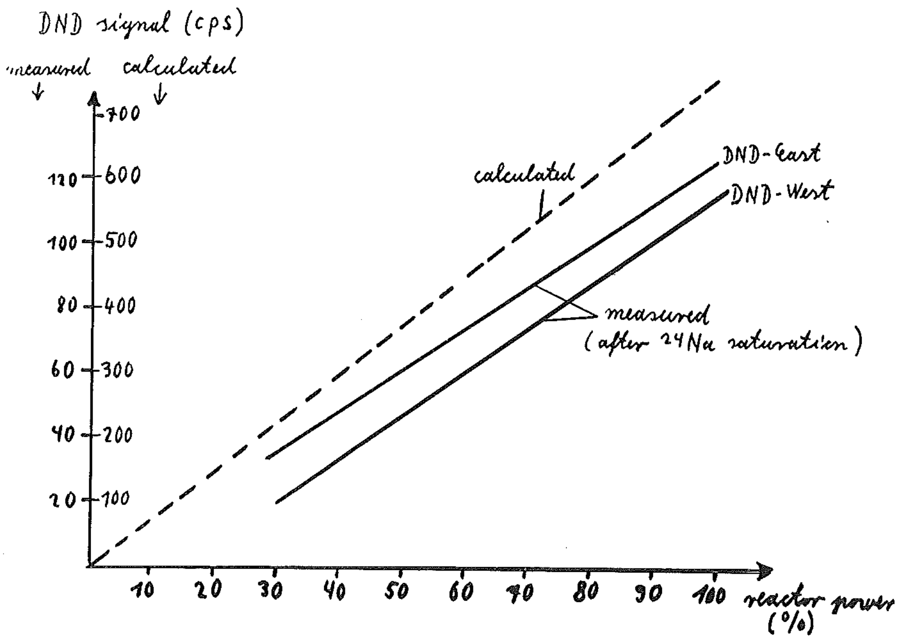


Fig. 4: DND-Background Signal vs. Nuclear Power a) calculated b) after ^{24}Na Saturation measured

Nuclear Power (%)	Background Count Rate (s^{-1})	Count Rate at 1 cm^2 Fault Area (s^{-1})	Fault Area regarding Alarm Level (cm^2)	Fault Area regarding Limit Value (cm^2)
30	27	60	[illegible]	33
40	40	95	[illegible]	21
50	54	125	12	16
60	67	150	10	13
70	81	175	[illegible]	11
80	94	189	[illegible]	10
90	107	210	[illegible]	9
100	120	221	[illegible]	9

Table II: DETECTABLE FAULT AREAS WITH DND MONITORS KNK II

Nuclear Power MW_{th}	(%)	^{41}Ar	^{135}Xe	^{133}Xe	^{133m}Xe	^{85m}Kr	^{87}Kr	^{88}Kr
		kBq/m^3						
21.5	(37.0)	$2.8x10^6$	$3.4x10^4$	$2.1x10^4$	$8.2x10^2$	$2.9x10^3$	$3.0x10^3$	$3.8x10^3$
36.0	(62.0)	$4.1x10^6$	$5.1x10^4$	-*)	-*)	$6.0x10^3$	$6.5x10^3$	$7.7x10^3$
47,6	(82.0)	$5.4x10^6$	$7.9x10^4$	$2.2x10^4$ **)	$1.1x10^3$ **)	$8.3x10^3$	$7.4x10^3$	$1.0x10^4$
56.3	(97.0)	$7.0x10^6$	$1.0x10^5$	$3.3x10^4$ **)	$1.4x10^3$ **)	$1.2x10^4$	$9.5x10^3$	$1.4x10^4$

*)not measured
**)no saturation

Table III: PRIMARY COVER GAS ACTIVITY DURING OPERATION WITH INTACT FUEL ELEMENTS

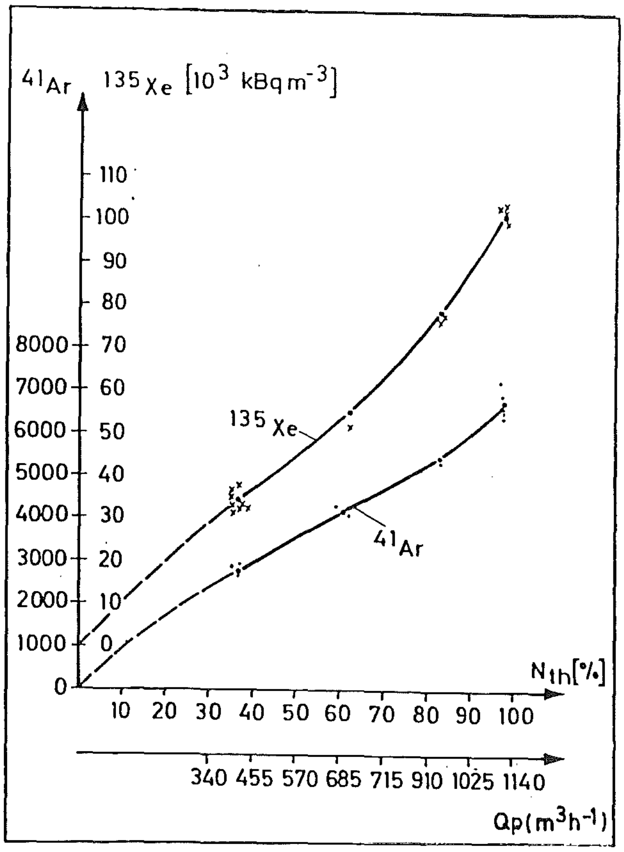


Fig. 5: ^{41}Ar and ^{135}Xe Activity vs. Nuclear Power

Date	Time	Pressure in the Dry-Sipping Flask (ata)	Subassambly	^{133}Xe	^{85}Kr	^{131m}Xe
				kBq/m³		
October 13,1980	15^{53}	0.18	YN-205-IA Test Zone	4.2×10^2	n.d.*)	n.d.*)
October 14,1980	10^{46}	0.19	YN-202-IA Test Zone	5.7×10^6	1.1×10^7	5.5×10^5

*) n.d.= not detectable

Table V: ACTIVITY CONCENTRATIONS IN THE COLLECTED DRY-SIPPING GAS

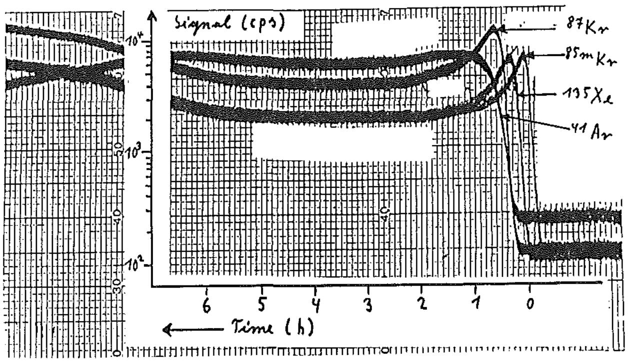


Fig.6: Cover Gas Signal April 1, 1979

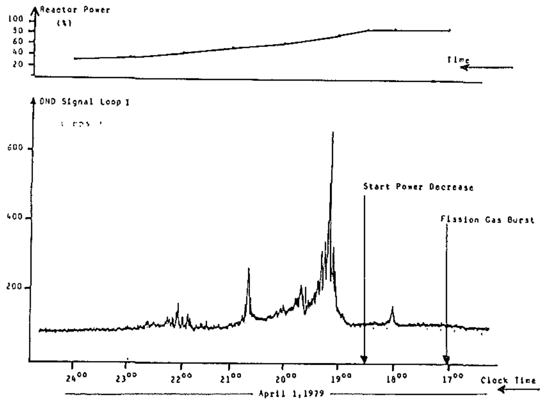


Fig.7: DND Signal Behaviour April 1,1979

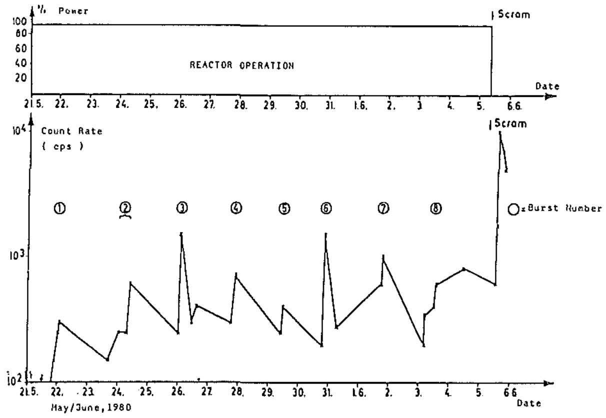


Fig.8: ^{133}Xe Count Rate during First "Leaker" Phase May/June 1980

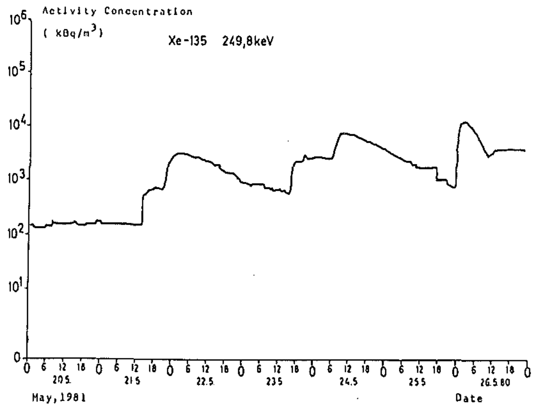


Fig.9: ^{135}Xe Concentration during First "Leaker" Phase May 1980 with periodic Bursts

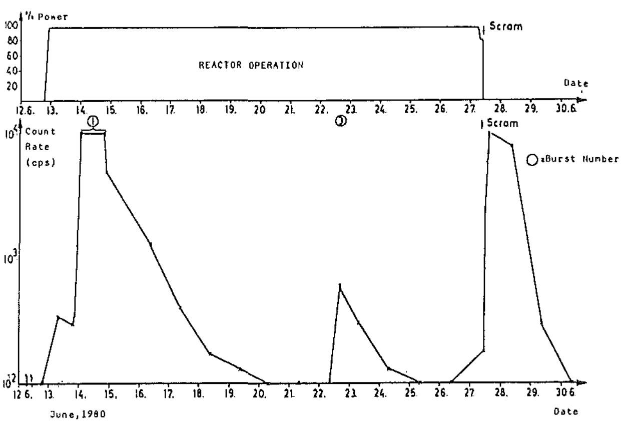


Fig.10: ^{133}Xe Count Rate during Second "Leaker" Phase June 1980

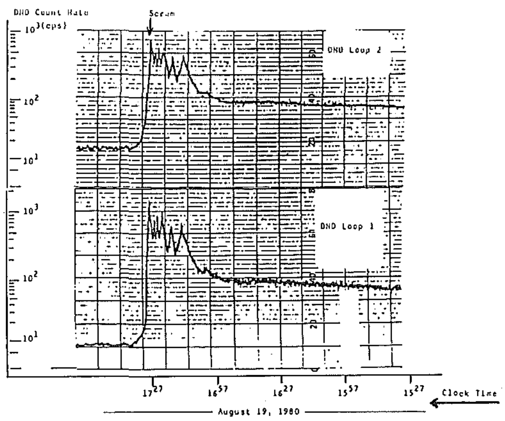


Fig.11: Reactor Scram by DND Signal

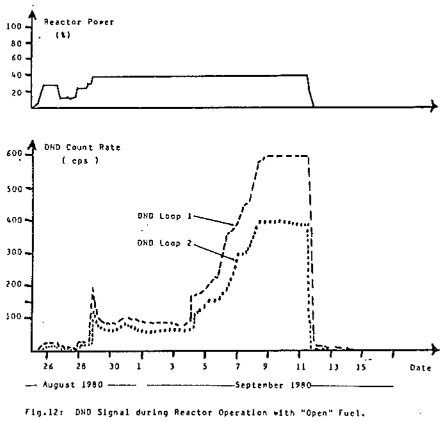
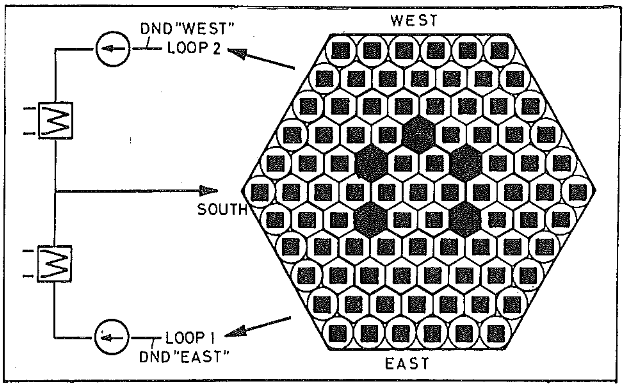


Fig.12: DND-Signal during Reactor Operation with "Open" Fuel

*) not measured

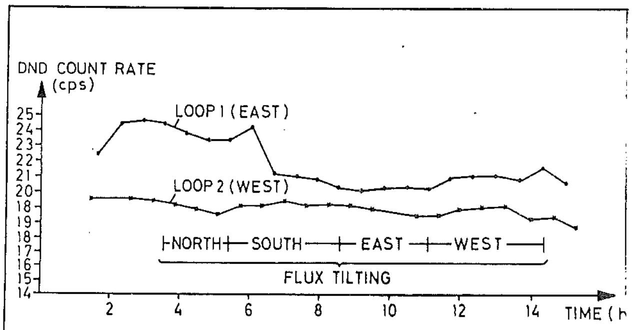
Sampling time	Ratio $^{131}Xe/^{134}Xe$	Concentration ^{131}Xe (vpm)
May 23,1980	below detection	< 0.002
May 27,1980	below detection	< 0.002
May 28,1980	0.36 $\pm$0.1	0.002
June 9,1980	0.468 $\pm$0.004	0.155
June 12,1980	0.478 $\pm$0.008	*)
July 18,1980	0.5 $\pm$0.1	0.028
August 19,1980	0.403 $\pm$0.02	*)
Sept. 10,1980	0.456 $\pm$0.02	0.009
Sept. 11,1980	0.46 $\pm$0.02	0.023
Sept. 12,1980	0.46 $\pm$0.02	0.023

Table IV: $^{131}Xe/^{134}$XENON RATIO IN THE PRIMARY COVER GAS DURING LEAKER PHASE



Contol Rod

Fig.13: Location DND monitors and Flux Tilting



North = Reduced Power in North Sector etc.

Fig.14: Result of Flux Tilting Operation at 15% Nuclear Power

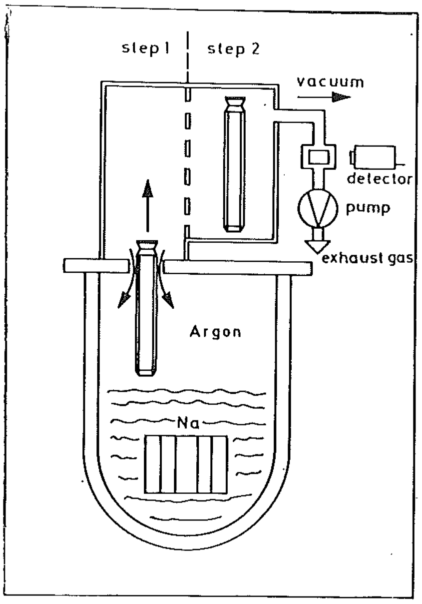


Fig.15: Detection of Defective Fuel Elements

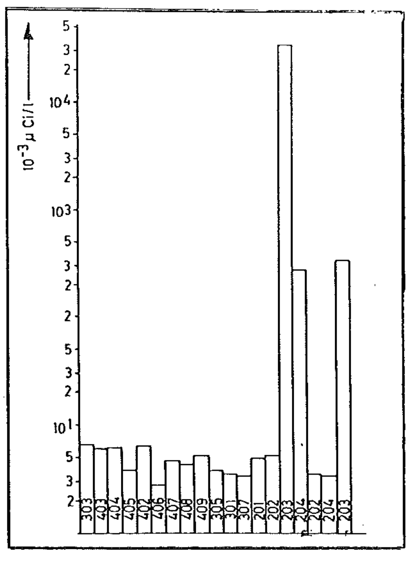


Fig.16: ^{133}Xe Concentration for Tested Elements

Discussion

P. Michaille, CEA:

When you speak of a geometrical size of a failure, which assumptions do you use and on what are they based?

H. Richard, KBG:
We use the recoil model with a k-factor of 10.

Some Experiences of Fuel Failure Detection System in JOYO

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H. Taniyama and Y. Nara
Power Reactor and Nuclear
Fuel Development Corporation

ABSTRACT

The Japan Experimental Fast Reactor, JOYO, has two different types of Fuel Failure Detection (FFD) systems, the Delayed Neutron Monitoring (DNM) system and the Cover Gas Monitoring (CGM) system. Operational results of both systems are summarized as follows:

1) The total background of the DNM system was reduced to about 1/70 of its initial value by additional installation of polyethylene shielding. At 75 MW of reactor thermal output, background is 2500 cps, which satisfies the design requirement. About 8.5% of the background is due to the photoneutrons from the polythylene shielding and the remainder is due to the leaking of core neutrons through the primary piping penetration.

2) The total background of the CGM system at 50 MW and 75 MW is about 80 cps and 120 cps, respectively. From the results of this 50 MW power up test, about 60% of the total background is contributed by Ne-23, 20% by Ar-41, 5% by Na-24, and the other 15% is from unidentified sources.

3) Neither the DNM nor the CGM system detected signals from tramp fuel in the sodium.

4) The transfer time of Ne-23 from the core to the CGM system is about 130 seconds, which agrees well with the design value.

5) The transfer fraction of Ne-23 from sodium to argon cover gas is about 1.4%.

INTRODUCTION

The Japan Experimental Fast Reactor, JOYO, has two kinds of Fuel Failure Detection (FFD) system. These are (1) Delayed Neutron Monitoring (DNM) system, and (2) Cover Gas Monitoring (CGM) system.

Since the first criticality of JOYO on April 24, 1977, both systems have operated satisfactorily.

SYSTEM DESCRIPTION

(1) Delayed Neutron Monitoring (DNM) system

The DNM system is installed beside one of the two reactor outlet primary pipes as shown in Figure 2.1. The system consists of the neutron shielding of polyethylene and boral, lead blocks, graphite blocks, and two kinds of neutron detectors, BF_3 proportional counters and boron lined counters.

In order to assure a broad sensitivity range for the DNM system, the BF_3 counters having a high neutorn sensitivity were employed to detect a small amount of fission product emission and the boron lined detectors were installed for a large breach of the fuel element.

The polyethylene neuron shielding was installed after the reactor critical test to reduce the background of the core neutrons leaking through the primary piping penetration.

The transfer-time of sodium from the core to the DNM system is estimated as about 30 seconds from the results of the thermal transient test in the 50 MW power up test.

(2) Cover Gas Monitoring (CGM) system

The CGM system consists of piping systems, two vapor traps, a compressor, and a precipitator. The reactor cover gas is extracted from the reactor gas plenum and driven to the precipitator through one inch piping by a compressor.

The precipitator has three nozzles, a sample gas inlet, a clean argon gas purge line and a gas outlet nozzle. Clean argon gas flows from scintillator side to the chamber through the wire guide hole to prevent the sample gas flowing to the detector and to reduce the background.

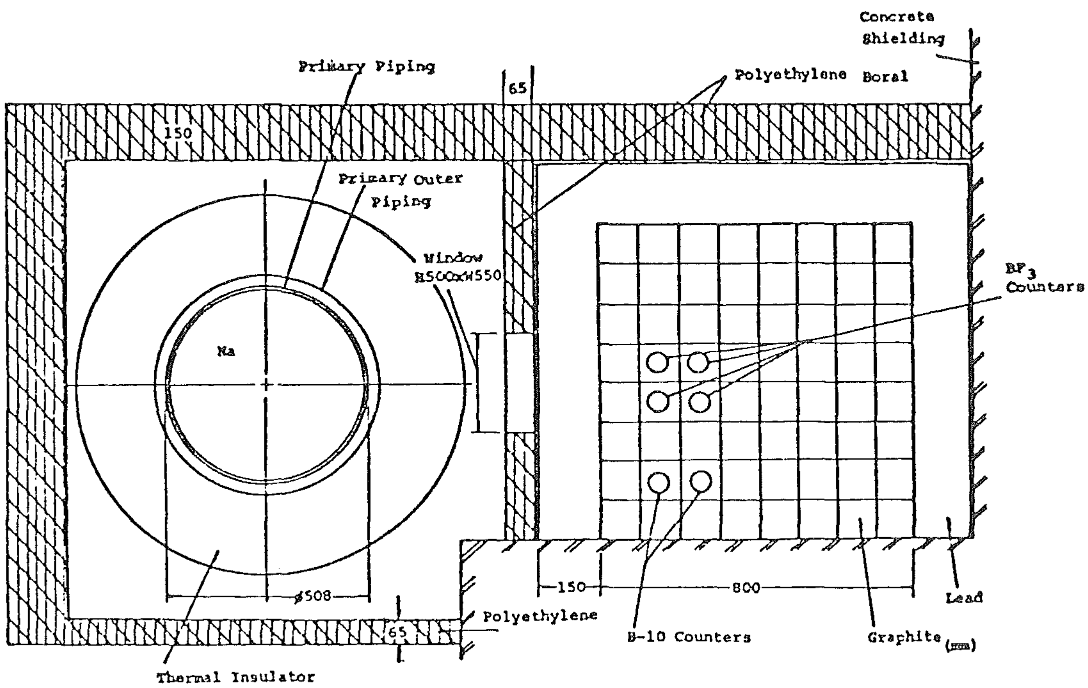


Fig. 2.1 Delayed Neutron Monitoring Block

In JOYO, the gas flow rate of main piping in the CGM system is 100 l/min., and the flow rates of the sampling gas from main line to the precipitator and of the purge gas to the precipitator are 250 cm^3/min. and 50 cm^3/min., respectively.

The CGM system diagram is illustrated in Fig. 2.2.

Main background of this detection system is from gases, Ne-23 and Ar-41, which come to the scintillator through the penetration hole for the collection wire, and from Na-24 which are carried by cover gas and precipitate on the wire.

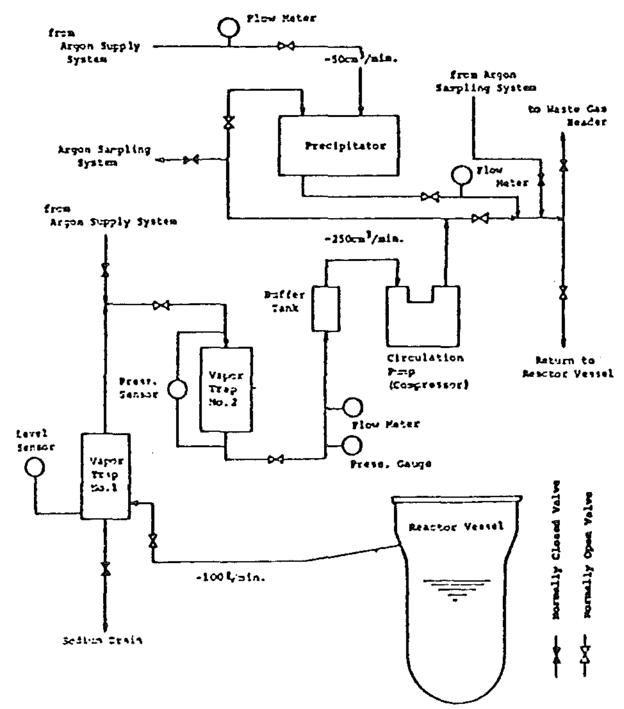
In addition to the CGM system mentioned above, a gamma area monitor is installed in the FFD room to detect and record the gamma dose rate continuously. This measurement is also used for the analysis of the reactor cover gas activity performance and of the CGM system function.

OPERATION EXPERIENCE

(1) Delayed Neutron Monitoring (DNM) System

The count rate vs reactor power during the low power nuclear test period is shown in Fig. 3.1. This background level was much larger than anticipated.

To reduce the background, a modification was done to the system before the power up test by installation of the polyethylene shielding illustrated in Fig. 2.1.



ig. 2.2 Block Diagram of Fuel Failure Detection Cover Gas Monitoring System

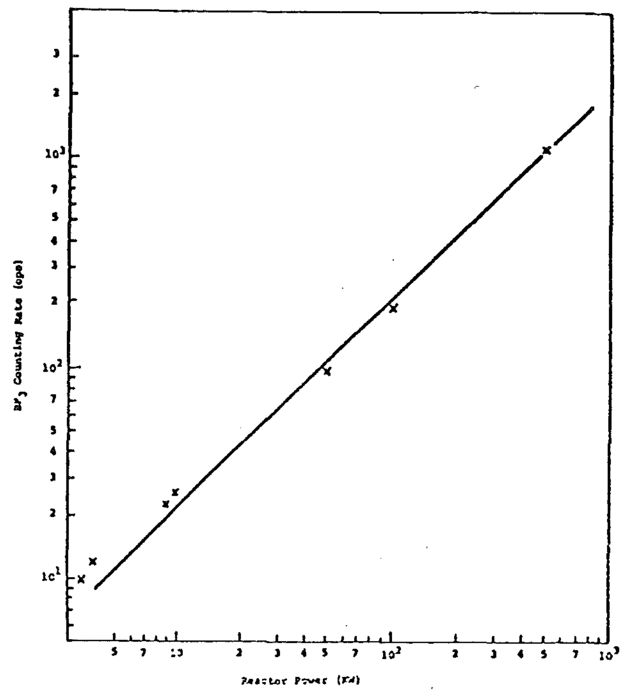


Fig. 3.1 DNM BF_3 Counting Rate vs. Reactor Power during Low Power Nuclear Test

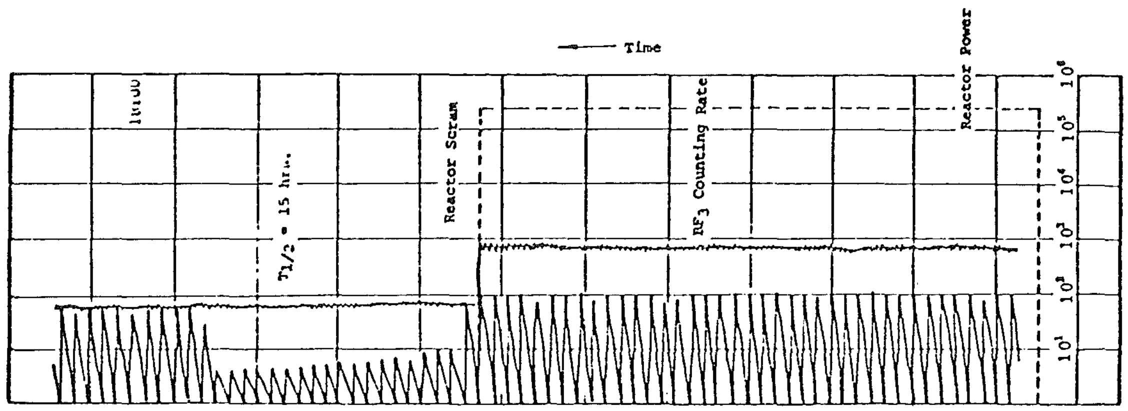


Fig. 3.2 BF_3 Decay Curve after Reactor Reactor Scram at 25 MW on June 2, 1978.

After the modification, the background was reduced by 1/70, and is 2500 cps at 75MW reactor power, about one half of the expected level at the system design stage.

About 8.5% of the background is due to the photoneutrons from the polyethylene shielding and/or the concrete of the building structure, and the remainder is due to the leaking core neutrons.

The DNM responses after reactor scram during the low power nuclear test is shown in Fig. 3.2. The contributions of the photoneutrons were measured at the reactor scram tests, showing a postscram decay that coincides with the 15-hour half-life of Na-24.

Measurements and analysis have indicated neither any release of fission products from the fuel elements nor signals from tramp fuel nuclides.

(2) Cover Gas Monitoring (CGM) System

1) Background characteristics

The background characteristics of the CGM system are affected by system operation conditions. Operating experience of these past two years indicates that the deviation of total background is less than 30%.

The total background at 75MW of reactor thermal power is about 120 cps.

Changing the CGM system operation conditions indicated that no core fission products were released to the reactor cover gas.

About five days are required for the count rate to reach the saturation value. Analysis revealed that this is because of Na-24 build up (15 hr half-life).

2) Ne-23 Background and Transfer Time

The major background nuclide is Ne-23, produced by sodium activation, Na-23(n,p) Ne-23. To identify the background isotopes, the precipitator responses were taken during the reactor scram tests and the 50 MW power up test, in 1978.

The precipitator count rate decreases to approximately 40% about two minutes after reactor scram. The decayed isotope had a short half life of about one minute and was assumed to be Ne-23.

A delay from reactor scram to count rate decrease is considered to be the transfer time of Ne-23 from the reactor core to the precipitator. This is therefore the minimum response time of the JOYO CGM system. The soak/count interval of the precipitator is set to 10 minutes.

The measured transfer time, about 130 seconds, exhibits good agreement with the design value of 125 seconds calculated by the assumption that gas is transported by piston flow.

The transfer rate of Ne-23 from sodium to the cover gas is estimated by the production and detection ratios of Ne-23 to Ar-41, and the decay term calculated from the transfer time. The detection ratio of Ne-23 to Ar-41 was measured in the precipitator, preliminary test. The calculated result shows a transfer fraction of 1.4% till the end of the first 50 MW duty cycle.

3) Background by Other Isotopes

The decay curve of typical precipitator background following a reactor scram test is shown in Figure 3.3.

The Ar-41 and Na-24 contributions are summarized as about 20% and 5%, respectively, although they vary from test to test.

The analysis indicated no background by fission products of fuel elements and tramp fuel materials in sodium.

4) Background Increase after Reactor Scram

Both the precipitator count rate and the gamma dose in the FFD room increase about 30 minutes after reactor scram, or perhaps decrease rapidly just after the scram and return partially after about 30 minutes. The reason for and isotopes causing this activity change (increase of decrease) have not been found.

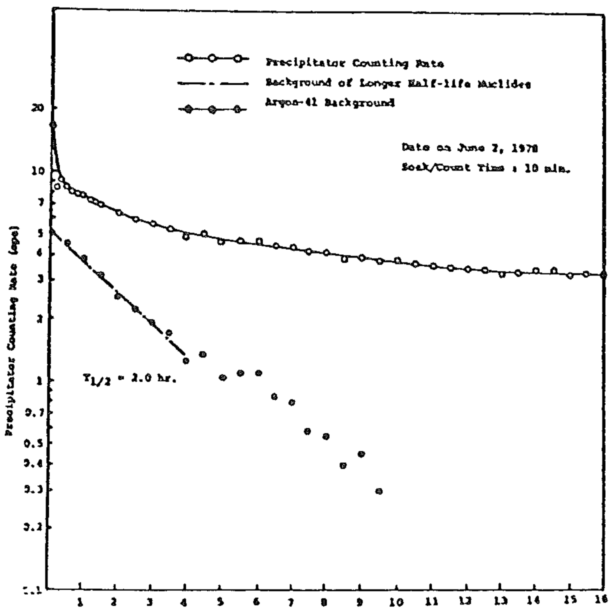


Fig. 3.3 Decay Curve of ^{41}Ar, ^{24}Na and Others at 25 MW Reactor Scram Test.

CONCLUSION

Operating experience of the Fuel Failure Detection (FFD) system for the Japan Experimental Fast Reactor, JOYO, is summarized as follows:

1) Two systems using the DNM method and the CGM method, have been operated satisfactorily and have verified their functional requirements.

2) Since the first criticality in 1977, no fission product has been released in JOYO.

3) The background of the DNM system at 75MW of reactor thermal power is about 2500 cps, about 8.5% of which is by the photo-neutrons from the polyethylene shielding and the rest of which is by the core neutrons leaking out through the primary piping penetration.

4) The background of the CGM system through the first 50 MW duty cycle was about 40 cps. It has changed to about 80 cps in the second 50 MW of reactor power.

5) The major components of the CGM background are Ne-23, Ar-41, and Na-24. Contributions of Ne-23, Ar-41, and Na-24 are about 60%, 20% and 5%, respectively. The remainder of 15% is unidentified isotopes.

6) The background due to the tramp fuel mateirals in sodium is so small that it has not been detected by either the DNM or the CGM.

7) The transfer time of Ne-23 from core to the CGM system was measured several times. The average was 130 seconds, which shows good agreement with the design value calculated on the basis of piston flow.

8) The transfer rate of Ne-23 from sodium to argon reactor cover gas is estimated to be about 1.4%.

Discussion

S. Jacobi, KfK:

In one of the viewgraphs a cross section of the DND monitor of Joyo was shown. What is the reason to use graphite for moderation of the delayed neutrons in contrast to use polyethylene to moderate the core neutrons?

Mr. Miyazawa, TOSHIBA:
At first, we designed by following the requirement on fire protection in the reactor building. In the second step, we were permitted to use the polyethylene as core neutron shielding with stainless steel cover.

THE DETECTION AND LOCATION OF FUEL FAILURES IN SODIUM COOLED FAST REACTORS

REVIEW OF UK WORK

COMPILED BY

D K CARTWRIGHT RNL
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WITH CONTRIBUTIONS FROM

T A LENNOX DNE
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UK presentation to the IWGFR Specialist Meeting on 'Detection and Localisation of Failed Fuel Elements in LMFBRS' Karlsruhe 11-14 May 1981

Part 2

EXPERIENCE FROM THE PROTOTYPE FAST REACTOR AT DOUNREAY

EXPERIENCE FROM THE PROTOTYPE FAST REACTOR AT DOUNREAY

In this section of the UK presentation the signals from tramp fuel contamination, a defected pin, and two failures in experimental pins are discussed.

Background Signal Levels

The signals for the IHX bulk DN monitor and the precipitator in the absence of known activity sources other than contamination of the core are given in Table 1. The data are consistent with a fuel level equivalent to less than 1 mg of Pu 239 at the core centre.

The tramp DN signal is proportional to power and to the number of IHX's in use as would be expected.

The cover gas precipitator activity is sensitive to core outlet temperature and increases by 50% if all the IHXs are not in use. Variation of flow through the DN monitor has shown that stray and gamma-n neutrons made little contribution to the count rate. These measurements and those from reactor transients show that the mean time from the core to the IHX DN monitor at full reactor flow is 25 $\pm$ 3 seconds.

The activity of the cover gas is principally caused by Ne23 and A41. The Ne23 activity is very variable while the A41 level corresponds to a gas entrainment level of the order 5 x 10^{-3}%. Ne23 measurements have been used to measure the transit time from the cover gas to the monitor as 7 minutes whereas the transfer of fission products from the sodium to the gas has an inverse time constant of about 0.7 hr^{-1}.

Signals from an artificially defected pin

A defect pin with a 0.25 mm diameter hole at the top of the plenum has been in the core since June 1978. No increase in signals was initially observed with the defect in the core except when the pump speed was reduced, then gas signals were recorded. Subsequently occasional increases in gas activity under steady running have been attributed to release from this pin as shown in Figure 1. Activities are given in Table 2 - no increase in DN activity has been observed.

Signals from Fuel Failures

Two fuel failures have been detected in PFR, both arising from highly rated experimental pins containing vibro fuel. Both defects were initially detected by increases in the cover gas of long lived Xe133 and Xe135 although at first it was not clear if these increases were due to the defected pin. DN signals were not observed until many days after the initial detection by fission gas signal. In the case of the first failure the delayed neutron signal rose to several times background over several weeks. The signal was noisy and increased non-linearly with power. The signal from the second failure increased more rapidly so that after several weeks it was 25 times background, and was again non-linear with power. Oscillation of reactor power and rod drop experiments suggested that during irradiation the effective transit time of the signal from the second defect pin increased from 15 to 26 seconds.

At the time of failure peaks giving an increase of several orders of magnitude in cover gas activity occurred but after about 20 days these reduced to a high but steady signal. During this period, after about 10 days the DN signal increased as shown in Figure 2.

The failures were located using the loop and sparge pipe. The first failure gave a count-rate of about 10 times background. The second failure was situated in a DMSA cluster and the location signal was therefore concentrated; correcting for this the count rate was equivalent to about 5 times background. The orientation of the loop was checked using the calibration foil and the defect pin.

Examination of the second failure has shown one of the pins to have a series of axial cracks with a total area of about 1 cm^2 whereas the signal if based upon an effective exposed recoil area is equivalent to about 10 times this value.

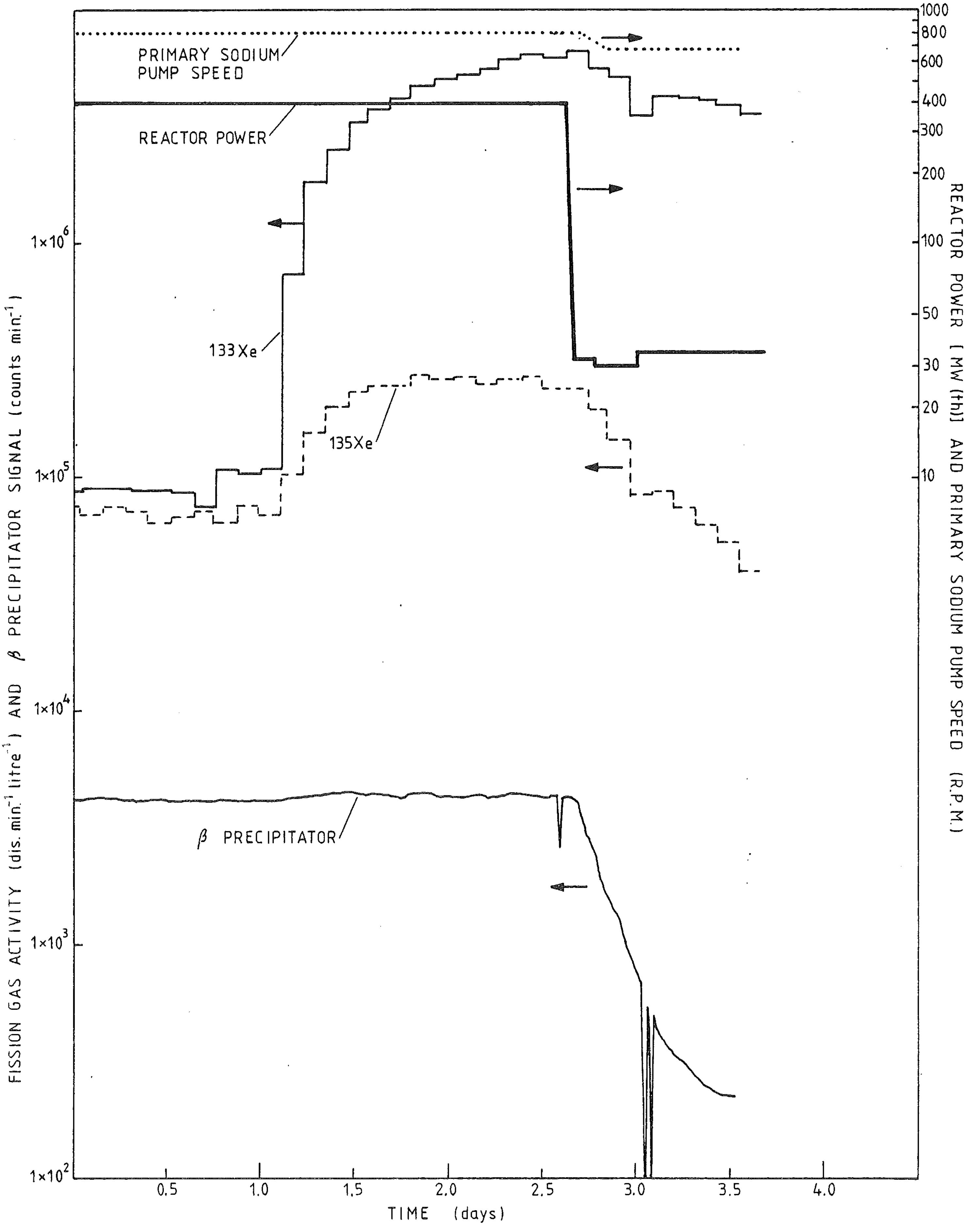

FIG. 1. COVER GAS ACTIVITY WITH DEFECT PIN IN REACTOR.

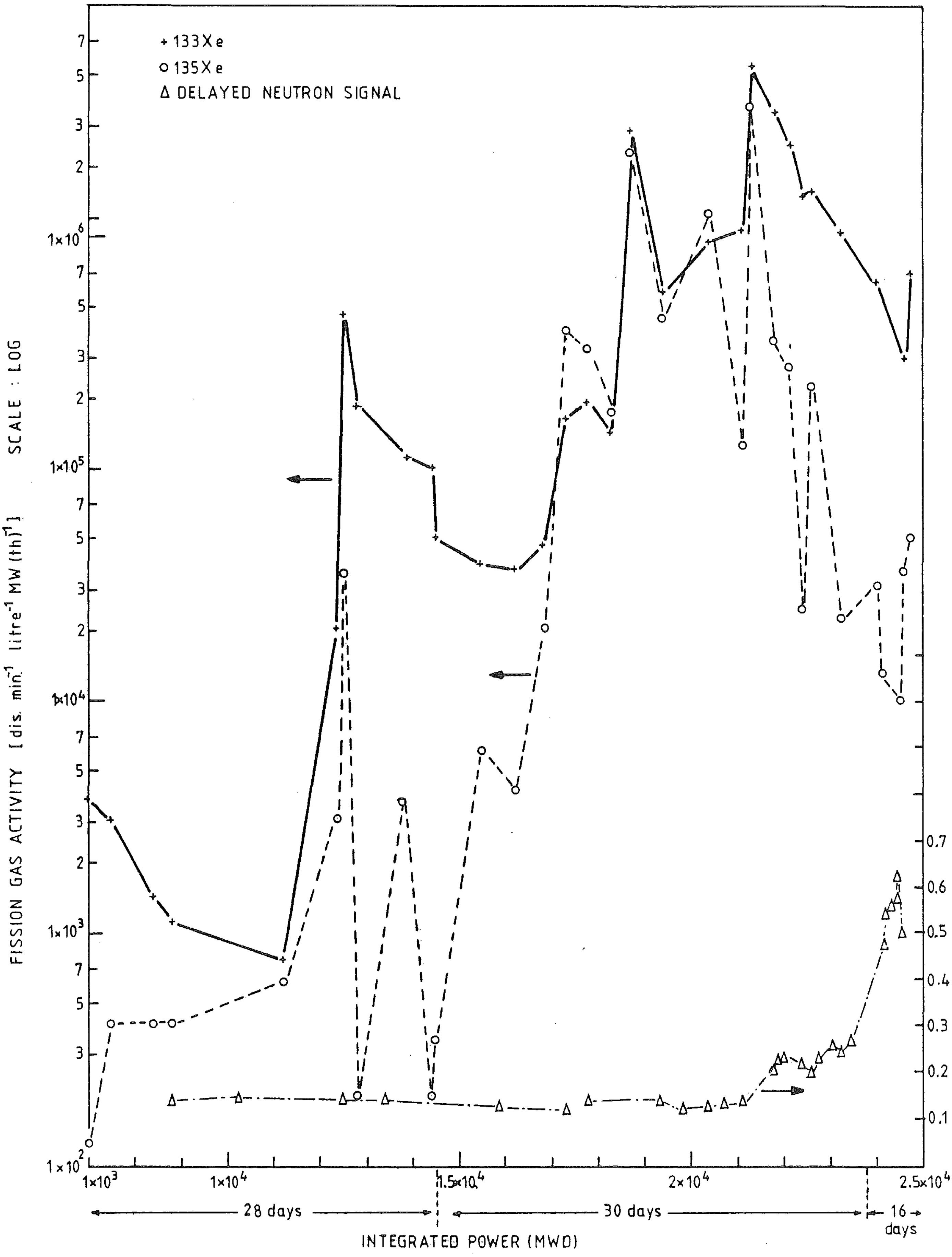

FIG. 2. HISTORY OF SIGNALS DURING SUB ASSEMBLY FAILURE.

TABLE 1

Signals from Tramp Fuel at Full Power

IHX DNM	Cover Gas Precipitator	Gamma Spectrometer				
count/s	counts/min	microcuries/m^3				
		Xe133	Kr 85m	Kr 88	Xe135	Kr87
4	5,300	80	4.5	9.0	37	7.3

TABLE 2

Activities attributed to Defect Pin

Gamma Spectrometer				Cover gas Precipitator
microcuries/m^3				counts/min
Xe133	Kr 85m	Kr 88	Xe135	
11.10^3	$0.2.10^3$	$0.1.10^3$	14.10^3	$1.3.10^5$

Discussion

P. Michaille, CEA:

How did you stop the reactor in the case of the DND signal of fig. 10? Which is the level of the DND scram?

T.A. Lennox, UKAEA:
The reactor was shutdown manually to permit refuelling. The scram level is adjusted according to the proposed operating conditions.

M. Relic, IA:

a) What sensitivity has your precipitator for Ne-23?
b) What is the transit time of gases from core to the precipitator?

T.A. Lennox, UKAEA:
b) The transit time to the precipitator from the cover gas is about 7 minutes.
a) Since the transit time to the precipitator is long Ne23 does not influence the precipitator signal. We have measured the total contribution of the gamma background to the precipitator signal by stopping the precipitator wire. This contibution is very low about 5 % of the total signal.

C. Berlin, CEA:

About your artificial experiment, you show a picture where the gas signals are represented you presume that the precipitator signal slightly increase relatively to the large evaluation of the 133 Xe activity - What is your conclusion?

T.A. Lennox, UKAEA:
The precipitator is sensitive to the level of Kr-88 activity. In the figure presented the small change in precipitator signal implies a small change in the level of Kr-88 activity i.e. the gas release is of "aged" gas containing the long lived isotope Xe-133, Xe-135 principally.

U.S. Experience in Identifying Fuel-Pin Failures

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ABSTRACT*

A general description is given of the methods used at EBR-II to identify fuel-pin failures over sixteen years of operation. The recent successful use of on-line mass spectrometric analysis to identify xenon tags is discussed in detail.

INTRODUCTION

Experimental fuel pins have been irradiated in EBR-II since 1964 and cladding breaches have developed quite often to release radioactive fission gases to the reactor primary system. Because the reactor cover was not originally built to be completely leak tight, until recently every breached pin had to be identified as quickly as possible and its subassembly removed from the core (1).

The installation of a cover-gas cleanup system (CGCS) in 1977 (2) removed this prior need for rapid shutdown and discharge of failures--to the betterment of plant factor (which now averages above 70%). But with an increasing number of failures in endurance tests of fuel pins of aggressive design, this change in operating philosophy has laid a heavy emphasis on distinguishing between multiple xenon tags in the reactor cover gas. Thus, although all the methods used over the years at EBR-II to identify failures will be briefly reviewed, most time will be spent here in describing recent experience with on-line mass spectrometry of the xenon tag-beds of the CGCS. Parallel papers (3-5) describe the detection systems at EBR-II, and the experience gained with delayed-neutron signals and their analysis.

FAILURE INCIDENCE & CONCOMITANT IDENTIFICATION

As the use of EBR-II as a test facility has grown and changed with time so has the incidence of fuel-pin failures varied (Fig. 1). Before about 1972, failures were rare and were due to defective closure welds (6), and the like. Although xenon tags were included in experimental pins from 1969 onwards (7), none of these early failures had them. So this initial period, which might be called Phase I of EBR-II operations, was characterized by lengthy searches for sources of fission-gas release.

From 1973 onwards, intentional endurance or run-to-clad-breach (RTCB) tests gave an increasing number of failures. But generally these failures were well separated in time and released xenon tags. Experience was gained in correcting xenon-tag compositions for changes due to in-reactor exposure, and identification was quite rapid (8). This period might be called Phase II of EBR-II operations.

From 1977, or thereabouts, when reference FFTF fuels tests approached goal exposure and RTCB tests of advanced fuel-pin designs began, the incidence of failures sharply increased; run-bevond-clad-breach

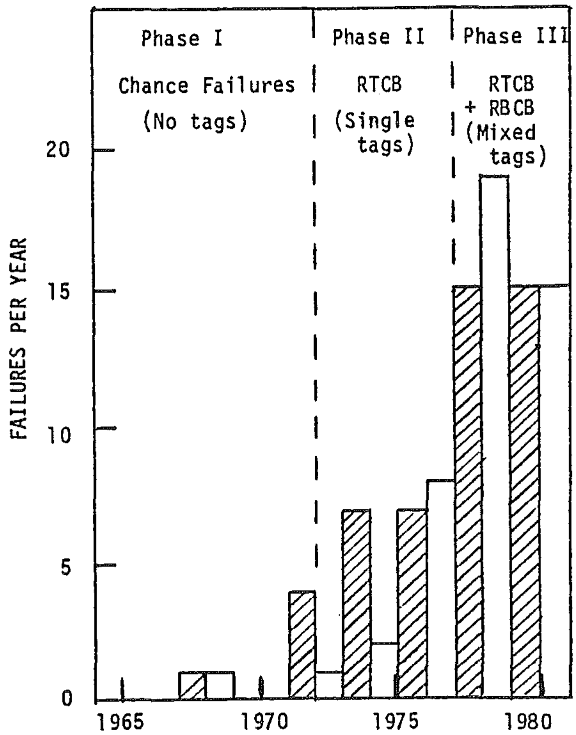

Fig.1 Incidence of Fuel Failures in EBR-II over 1964-1980

(RBCB) tests also started. During this period, which may be called Phase III and includes the present, an average of 10-15 failures have occurred per year. Phase III has been characterized by an increasing number of occasions when several failures have been in-reactor at the same time. Although this has caused no problems per se, it has complicated identification of failures, as we shall later describe.

METHODS USED FOR IDENTIFICATION

Twelve methods have been developed at EBR-II for the identification of fuel-pin failures; they are listed and briefly described in Table I. The first six methods in this table were used in Phase I of EBR-II operations when there was no accompanying release of a xenon tag. With one exception, these methods involved the measurement of fission-gas activity in the cover gas and were non-specific in nature. The sixth method predicted failure based on experience; it was of use for homogeneous groups of fuel pins (such as the EBR-II metal driver) in which there was a well-defined mode of failure (9). Identification frequently involved many reactor shutdowns and startups, and 5-10 days of lost operating time. The acid test of identification was always the absence of short-lived fission-gas activity on resumption of full power.

*Work supported by the U.S. Department of Energy.

TABLE I. Methods Used to Identify Sources of Fission-product Releases in EBR-II

Method	Purpose	Advantages	Disadvantages
1. Fission-gas volume	Identifies suspects by gas release	Eliminates low-burnup suspects	Usually limited applicability
2. Ratio $^{134}Xe/^{133}Xe$	Discriminates between metal and oxide	Identifies type and burnup of suspect	Ratio changes for same element; affected by fuel and breach geometry
3. Normalized excursion parameter	Discriminates betweem metal and oxide	Rapid	Empirical; uses release characteristics that may give ambiguous results
4. ^{135m}Xe behavior	Indicates release of bond sodium	Rapid	None; occasionally overlooked
5. Weibull failure analysis	Ranks suspects by failure probability	Predicts breach in advance; helps rank otherwise equal suspects	Assumes common mode of failure; limited by previous experience
6. Flux-tilting test	Narrows down suspects to a section of the core	Easy to perform	Suspect must be adjacent to control rod; only positive response is meaningful
7. Ratio $^{134}Xe/^{128}Xe$	Determines burnup level of untagged element	Eliminates suspect with too high or low burnup	For small release, natural background contamination can be significant
8. Ratio $^{131}Xe/^{134}Xe$	Discriminates between metal and oxide	Uses stable high-yield isotopes of xenon	Can be affected by tag in low-burnup elements
9. Xenon tag	Identifies suspects by tag composition	Limits choice to one to three suspects	Exposure changes in tag; sometimes small tag releases; contamination
10. Fission-gas and tag volumes	Discriminates suspects with similar tag compositions	Ranks xenon-tag suspects	As above; also, early tag volumes were variable
11. Lift-and-hold test	Identifies suspects by gas release	Confirms suspect subassembly in fuel handling	Shutdown required; only positive response meaningful; time-consuming
12. FUM isolation test	Identifies suspects by gas release	Confirms suspect subassembly at operator's convenience; minimizes interference from cover-gas activity	As above; can tolerate only low decay-heat level in discharged subassembly

The remaining methods of identification shown in Table I were in general developed and used in Phases II and III of EBR-II operations. The predominant method was, of course, the identification of a xenon tag from a failure. However, all methods were used if and when appropriate. For example, the Xe-131/Xe-134 ratio has proved useful in distinguishing between failure of a uranium-fissioning metal driver-fuel pin and failure of a plutonium-bearing experimental pin.

At EBR-II tags have consisted of a $\sim$1 mL volume of a unique mixture of the stable xenon isotopes: Xe-124, Xe-126, Xe-128, and Xe-129, which is added to all pins in a given subassembly. The tags were blended from three components: natural xenon, natural xenon which is 5% enriched in Xe-124, and pure Xe-128. All told, 140 tag compositions have been produced, with Xe-129/Xe-124 ratio increments of 1.25, and Xe-128/Xe-124 ratio increments of 4. For FFTF and CRBR, where there will be a need for a much greater number of tags, stable Kr-78, Kr-80, and Kr-82 will also be employed (10,11).

mal to such operation, have been encountered with the system, but none seriously affecting its availability.

The On-line Mass Spectrometer

A7.5-cm radius 60° sector mass spectrometer was installed on-line with the tag-trap analysis system in 1977. Although the instrument functioned well and reliably, it had an inadequate resolution for Xe-133 (an important isotope), and was also difficult to maintain at optimum performance. In May 1980, therefore, it was replaced with a new, 15-cm. radius spectrometer which had been specially designed to have an abundance sensitivity of greater than 30,000 at 133 AMU; to be sensitive to analysis of very small samples; and to be more easily maintained.

The performance of this new instrument has been extremely gratifying: it has allowed, for example, the routine analysis of background samples which contain less than 0.002 mL xenon tag in the $\sim 10^7$ mL cover-gas volume, and in which Xe-124 can be measured with a precision of 1% at a mole fraction of 0.0015 of the xenon isotopes. This value exceeds the best sensitivity which could be obtained by the 'manual' method by an order of magnitude (the 7.5-cm. instrument was worse by a factor of four or five).

The mass spectrometer is operated automatically by the tag-trap system computer, and is therefore required to be on-line whenever the reactor is at power. The instrument has operated with a minimum of downtime. Routine replacement of the source filament is required every 5 months; and the turbomolecular pumps have had slight bearing problems. The most serious but easily remedied problem has been the removal of hydrocarbons following system shutdowns.

RECENT EXPERIENCE WITH FAILURES

This somewhat reduced sensitivity of the first on-line mass spectrometer in failure identification was more than offset by the system's ability to produce frequent analyses of the cover-gas composition. During releases the changes in cover-gas composition towards the xenon tag of a current 'leaker' could be easily discerned, a situation which added an extra degree of confidence in diagnosis. Nowhere was this capability better illustrated than during the fission-gas release from subassembly X294 in July 1978. At that time there were two previous failures in the storage basket, and the on-line tag analysis had just become fully operational. Release from the helium-bonded nitride pin in X294 began in the early morning of July 20 and continued throughout that day. Table III shows how the cover-gas composition changed steadily from a 70/30 mixture of the tags from the two failures in the basket, in the early morning, through to essentially 100% of the X294 tag by the early afternoon.

The ability of the CGCS to: control activity; purge previous tag contaminants; and allow frequent sampling, when combined with the fact that if there are several leakers in-core then only one will, in general, be dominant at one time, has been the combination of factors which has allowed EBR-II operations to continue under "multiple-leaker" conditions. Under such conditions it has been possible to track the mixing of xenon tags in the cover gas, and to not lose the ability to identify new failures.

This ability for on-line identification at EBR-II was well shown during July and early August 1980 when, during the course of a three-week period, five failures occurred: one in a driver-fuel pin, two in mixed-oxide pins, and two in mixed-carbide pins. Figures 3 and 4 relate the history of identification during that busy period. Figure 3 plots two isotope ratios in the cover gas: Xe-129/Xe-124; and Xe-129/Xe-128. The figure shows the significant changes which occurred in one or both of these ratios at times of fission-gas release from the failures. Figure 4 plots the same data in one of the planes of "xenon-tag ratio" space. Again the shifts in the cover-gas composition over that period show how each of the failures, in turn, was predominant over the others. Two shutdowns only were required to remove four of the five failures (the fifth--subassembly X280 --remained in-core for the remainder of the run at the experimenter's request). The lost reactor time for all five failures was approximately 40 hours; a considerable improvement even over affairs during Phase II of operations when single failures were discharged.

TABLE III
Failure in Subassembly X294 as Followed by
On-line Mass Spectrometer (7.5-cm. magnet)

Time of Sample	Mole Fraction of Xenon: Xe-124	Xe-126	Xe-128	Xe-129	Fraction of X294	Tag Volume (mL)	^{133}Xe (nCi/mL)
[1]0105	0.0243	0.0117	0.3310	0.6330	0.0	0.0006	7
0255	0.0252	0.0132	0.3290	0.6326	0.01	0.004	36
0445	0.0277	0.0130	0.2809	0.6784	0.21	0.005	41
0636	0.0344	0.0163	0.1716	0.7777	0.72	0.009	84
0826	0.0391	0.0175	0.1236	0.8198	0.91	0.030	305
[2]0850	0.0409	0.0182	0.1165	0.8244	0.92	0.036	350
1016	0.0391	0.0178	0.1128	0.8304	0.96	0.054	536
1206	0.0396	0.0179	0.1068	0.8358	0.97	0.072	703
1337	0.0403	0.0185	0.1048	0.8364	1.00	0.100	1022
X294 Tag	0.0416	0.0181	0.1061	0.8345			

[1]Background composition: 70/30 mixture of X273 and X138 E tags.
[2]Data from a manual tag sample.

When a failure occurred, the xenon contents of about 0.5 m^3 of argon or (1/20-th. of the cover-gas volume) were absorbed on a cooled charcoal trap, later released by heating, and collected in a shielded vial for analysis on a laboratory mass spectrometer. In this manual mode, sampling was limited to once every ten hours, and was complicated by personnel exposure. Nevertheless, considerable accuracy was obtained, and released tag volumes of 0.04 mL could be readily used for identification.

Although shifts in tag composition due to in-core exposure were early anticipated, the magnitude of the shifts was not. Major changes were caused by burnout of Xe-124 by neutron capture, and production of Xe-128 by an (n,γ) reaction on fission product I-127. During several tag releases in 1975 these effects caused considerable confusion in identification and extra lost operating time. But empirical correlation and, later, the results of tag-exposure tests and calculations have steadily removed this uncertainty due to composition change. Computer programs are now used to update the compositions of all xenon tags in-reactor for each and every run. Table II compares the as-loaded, corrected, and measured mole fractions of the tag isotopes in three subassemblies of different burnup and fluence. As can be seen, the changes in tag composition can be substantial after high exposure.

TABLE II
Mole Fractions of Tag Isotopes
in Three EBR-II Subassemblies

1. Subassembly X342B
Fuel Burnup: 3.9 at.%; Fluence: $4x10^{22}$

	Xe-124	Xe-126	Xe-128	Xe-129
As-Loaded	0.0159	0.0077	0.0742	0.9023
Corrected	0.0143	0.0077	0.0811	0.8969
Measured*	0.0139	0.0076	0.0819	0.8966

2. Subassembly X367
Fuel Burnup: 13 at.%; Fluence: $9.3x10^{22}$

As-Loaded	0.0177	0.0086	0.3010	0.6726
Corrected	0.0152	0.0087	0.3076	0.6684
Measured*	0.0154	0.0088	0.3108	0.6658

3. Subassembly X332
Fuel Burnup: 6.4 at.%; Fluence: $12.3x10^{22}$

As-Loaded	0.0273	0.0127	0.0938	0.8662
Corrected	0.0177	0.0127	0.1474	0.8220
Measured*	0.0206	0.0133	0.1535	0.8126

*Measured in the cover gas at failure

During Phase II most failures were well separated in time so that the tag from one failure did not interfere with the tag from a second failure. An exception occurred in June 1976 when failure of metal driver-fuel pins (in subassembly X208) closely followed failure of a mixed-oxide pin (in subassembly X138D). Figure 2 shows how, on successive days, the cover-gas composition changed from the X138D tag to that of the X208 tag. Maps such as the one shown in Fig. 2 are frequently used to aid diagnosis, although decisions are based on computer generated "figures-of-merit" of the tags in-reactor with the measured composition. Even if a failure is discharged from the core to the storage basket (as in this case) its tag will still be a contaminant for the 10-30 days of cooling, because the storage basket and the primary tank share a common atmosphere. Such contamination was mitigated by judicious purging of the

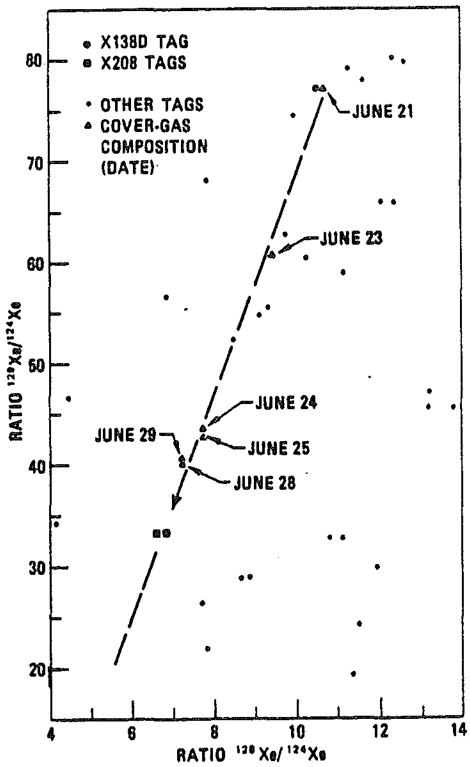

Fig. 2. Shift in Measured Tag-Isotope Ratios due to Mixing of an Old Tag (Subassembly X138D) with a New Tag (Subassembly X208)

cover gas with fresh argon, and controlled venting of activity to the atmosphere under favorable conditions.

THE CGCS IDENTIFICATION SYSTEM

The ability to simultaneously purge (and store) fission gas from the cover gas and to automatically obtain tag samples for analysis was made possible by installation of the CGCS in early 1977. In this system a cover-gas flow of up to 0.3 m^3/minute is passed through a liquid-nitrogen cooled cryostill to remove xenon and krypton, reheated, and returned to the cover-gas space. A small flow of 0.03 m^3/minute is passed directly through three sets of primary and secondary charcoal absorber beds, a sample product cylinder, a vacuum system, and an on-line mass spectrometer. All gas processing and mass spectrometer operations are controlled by a dedicated NOVA computer. Further details of the CGCS may be found in reference 2.

In the tag-trap system xenon is collected on a primary absorber bed at -100°F. It is then transferred to a secondary absorber bed for concentration, where it is stripped of excess argon. Finally, it is cryopumped into the sample cylinder and, eventually, into the mass spectrometer for analysis. Intermediate steps remove excess argon and regenerate the absorber beds. Two modes of operation may be used: a single sample mode (which requires 220 minutes to complete); or consecutive sampling every 110 minutes. All three sets of primary and secondary beds are used. Again each sample process and analysis is complete 220 minutes after sampling began. Either method is available to the reactor operator, and results are made available in the reactor control room via the data acquisition system (DAS) computer.

The tag-trap system has been routinely and reliably operated since early 1978 through many cycles. Very few component failures have occurred considering the large number involved. A few valve and heater problems, nor-

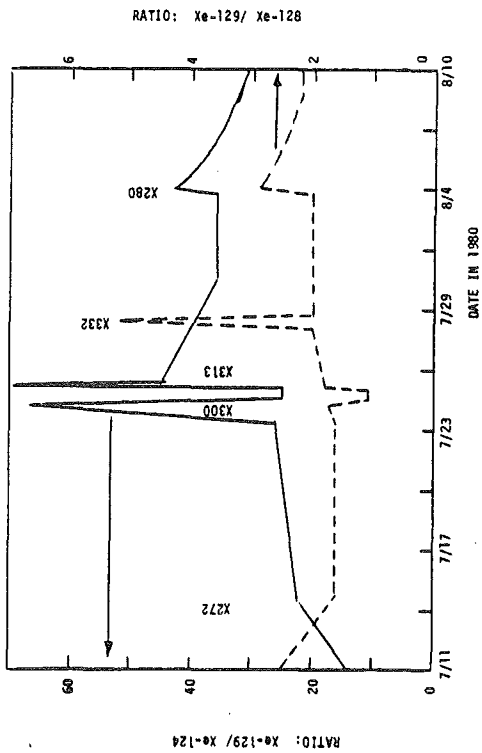

Fig. 3. Change in Cover-Gas Isotope Ratios: July-August 1980

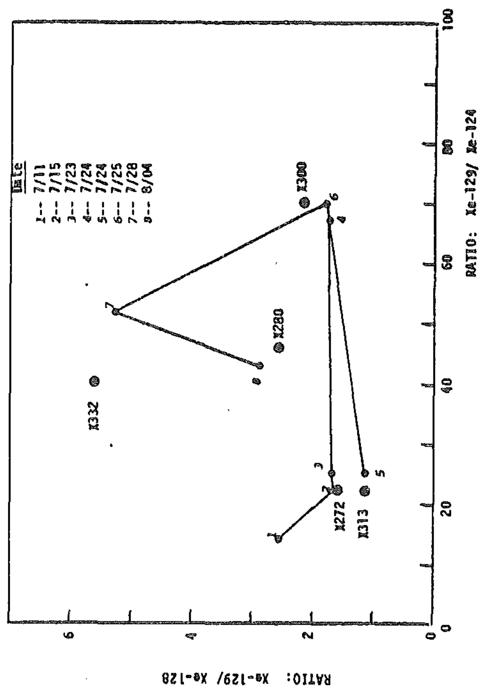

Fig. 4. Shifts in Tag-Ratio Space during July-August 1980 which Indicated Five Fuel-Pin Failures

SUMMARY AND CONCLUSIONS

Identification of failures in experimental fuel pins in EBR-II has evolved over 16 years of operation from an initial time-consuming activity of considerable uncertainty to the present routine procedure of on-line analysis of xenon tags. Shutdown of the reactor to remove failures is now a matter of operational convenience, or for programmatic reasons, rather than to limit cover-gas activity. This improvement in identification is reflected--in part--by a commensurate increase in plant factor which, in 1980, was 77.1% despite fifteen fuel-pin failures.

Extensive experience with xenon tagging over the last 6-8 years has shown this method to be a viable means of identification even when the number of tags in use exceeds 50, and when up to five fission-gas leakers in-core are intermittently releasing their fission gas and tag to the reactor cover-gas space. Such experience augurs well for the judicious use of tagging in large LMFBR's. The high reliability of the systems at EBR-II for control of cover-gas activity and for on-line mass spectrometry, coupled with the ancillary methods of source identification developed through the years, adds extra confidence to this overall conclusion.

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Discussion

W. Koop, SBK:

Do you think that each element in a commercial fast reactor will have a different Xe-composition in the tag?

J.D.B. Lambert, ANL:
No. The ∿ 300 pins in one subassembly will all contain the same tag. Each subassembly then has a different tag.

M. Relic, IA:

a) After tag indication will the fuel elements be discharged without other confirmation methods?
b) What is the realibility of the automatical mass spectrometer?

J.D.B. Lambert, ANL:
a) Yes
b) Very high 99.999 %.

F. Gestermann, IA:

You have the back-up method of lift and hold and hold and blow to confirm if you have taken the right element. How much time do you need before beginning this procedure and what was your success rate?

J.D.B. Lambert, ANL:
Two hours after shutdown minimum, i.e., during normal fuel handling. Success rate varies, about 50-75 %.

J. Dauk, IA:

Did you make consideration concerning the cost, whether the cost is more on the side of installation of covergas system and mass spectrometer or on the side of putting in the tags into the fuel pins?

J.D.B. Lambert, ANL:
CGCS is needed whether you have tags or not, but intend to operate with gas leakers. The mass spectrometer is a small additional cost. Major cost is in tagging the fuel pins, but could be reduced considerably by <u>not</u> using tag capsules.

S. Jacobi, KfK:

We got an impressive picture about the success of tagging. But defected pins indicated by tagging may be undangerous leakers or serious DND failures. How do you localize the DND failure especially in the case that there are some failures at the same time inside the core?

J.D.B. Lambert, ANL:
By DN triangulation technique using the DN detector on the 3 or 4 loops of the reactor.

N. Sekiguchi, PNC:

My question may be a little apart from your presentation now. Why don´t you use on-line gas mass spectrometer for tagging gas analysis in FFTF?

J.D.B. Lambert, ANL:
A matter of cost.

S. Jacobi, KFK:

The figure with the failure probability versus burn up give the impression that no failures happened with burn up lower than 10 %. Because this is surprising it seems we need some additional remarks to this figure.

J.D.B. Lambert, ANL:
The figure is correct but refers to the sodium-bonded metal driver-fuel pins of EBR-II.

D.K. Cartwright, UKAEA:

You have located 5 failures in the reactor, present at the same time. Was this lucky or would you think this could always be done?

J.D.B. Lambert, ANL:
I am fairly certain this could be done routinely, it depends on our experience that at any one time one of the failures will predominate.

J. Dauk, IA:

Do you have experiences with mixed Kr-Xe tags in EBR II?

J.D.B. Lambert,ANL:
No, but we soon will at FFTF.

THE RESULTS OF TESTING AND EXPERIENCE IN THE USE OF LEAK DETECTION METHODS IN FAST REACTORS

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Part 3

OPERATING EXPERIENCE WITH FAILED FUEL DETECTION AND LOCALIZATION SYSTEMS

At present an experience of LMFBR operation with failed fuel elements has been gained sufficient for the analysis. Data has been obtained about the dynamics of fission product release from the fuel elements with pin leakers and developed failures in cladding.

The experience in the use of detection systems summarizes information about fission product release fractions both in the very moment of failure and in the process of steady fission product release as well as that of trustworthiness of failed fuel subassemblies detection.

Further more investigated factors will be considered and summarized.

The BOR-60 plant operation at typical heat rates and fuel burnups permitted the mass testing of standard fuel elements to be carried out. A great number of cases with fuel failures resulted in fission gas release has been analyzed. The fuel element testing conditions may be discribed by the following parameters.

The standard BOR-60 fuel subassemblies comprize 37 UO_2 fuel pins (6 mm dia.) with 12 cm^3 plenum. The average ^{235}U mass per fuel pin is 56 g. Maximum temperature of 0.3 mm thick OX16H15M3 steel cladding of fuel pins was 700°C. The reactor outlet sodium temperature at 60 MW rating was 530-550°C, the maximum reactor heat was equal to 1100 kW/l (550 w/cm). The achieved burnup was in excess of 12%. Initial defects in fuel element cladding were detected at burnup of 7.5%.

Fission gas behaviour in the reactor cover gas was studied in more detail in 1972-1973. Fig.2 shows the curve of ^{133}Xe and ^{135}Xe activity variation during the above period. During the years which followed the above activity behaviour was the same.

Fuel element failure is followed by a steep rise in fission gas activity. This is due to fast gas release through the clad defect into the coolant. A failure formed at high gas pressure inside the fuel cladding results in practically complete fission gas release from the cladding. Therefore, the fission gas release fraction in cover gas is equal to fission gas release from the fuel considering the delay time in moderator. Then the fission gas activity in cover gas decreases. In some tens of days of the reactor steady power operation the activity smooth increase is observed. An essential fission gas activity decrease follows after a little drop in the reactor power. This may be due to the process of plugging the fuel element defect by sodium [2] . The analysis of a great number of cover gas activity bursts has revealed that ^{133}Xe and ^{135}Xe release fractions when one fuel element fails are $(5.3 \pm 1)\%$ and $(1.3 \pm 0.6)\%$, respectively [3].

When a fuel element failure occurs in the course of slow reactor shutdown, the fission gas activity attains its maximum in 5-10 min. after the shutdown. Then it decreases in accordance with the decay constants. For this case the fission gas release fraction in cover gas is equal to the release fraction from the fuel elements. The measured fission gases have no time to decay.

The evaluation of release fractions of short-lived gaseous radionuclides has been carried out in the experiments, when they were managed tobe measured just after the occurrence of a fuel element failure. The activity rate values with one failed fuel element for ^{133}Xe, ^{133m}Xe, ^{135}Xe, ^{85m}Kr, ^{88}Kr, ^{87}Kr radionuclides were respectively, 1800, 30, 600, 40, 85 and 40 GBc. Table 1 lists the release fractions from the fuel.

Solving the equations of fission gas diffusion in the fuel under irradiation [4] yields in a release relation of release fractions with square root of the decay constant. The experimental values for the relative release fractions are in good agreement with calculations. This leads to a conclusion that the BOR-60 fuel release is governed by a diffusion-controlled process.

Following a burst the ^{133}Xe and ^{135}Xe activities in cover gas decrease approximately in accordance with their half-lives.

Table I

Fission Product Release from Failed Fuel Elements in BOR-60 Power Plant

Nuclides	Fission product release fracture, %	
	experiment	calculation
^{133}Xe	3.4 - 5.4	6.4
^{133m}Xe	3.6	4.9
^{135}Xe	1 - 2.5	2.5
^{85m}Kr	0.5 - 0.8	1.2
^{88}Kr	0.1 - 0.2	0.3
^{87}Kr	0.2 - 0.3	0.7
^{137}Cs	20 - 40	35
^{134}Cs	40	-
^{131}I	2 - 9	5.1
^{133}I	4.4	2.2
^{140}Ba	0.01-0.07	0.16
^{95}Zr	-	0.05
^{95}Nb	0.02-0.05	0.43
^{103}Ru	-	0.11
^{141}Ce	-	0.08

Note : fuel burnup is 7-12 %1
fuel element linear power is 520 W/cm

Delayed neutron signal increase in the studued cases occured in 10-40 days after fission gas burst. Solid fission product release from fuel pins were small except for ^{137}Cs. The obtained release fraction data is in a good agreement with literature values. A gradual manner of defects development is also supported by EBR-II results [6] and by loop experiments with predefected fuel elements [7] .

Such a time shift of different stages of fission product release is characteristic not for all the kinds of fuel element defects. Sometimes a gas burst coincided with delayed neutron signal increase [8] . It may be suggested that these variations are due to different size of defects and/or their distribution along the fuel element height.

In the BR-10 plant operated with PuO_2 fuel elements at up to 10 % h.a. burnup the occurrence of pin (gaseous leaks)resulted in the pattern of the initial and subsequent fission gas and delayed neutron release similar to that in the BOR-60 reactor.Such a pattern appears to be not dependent on the burnup level.In two cases, when fuel pins failed in fuel subassemblies at 11,8%plutonium burnup and in 2 fuel subassemblies at 12,6% plutonium burnup, the delayed neutron release fraction increased twofold and a sharp rise in fission gas activity was observed [11] . The delayed neutron detection and the equilibrium fission gas activity were found to be non-linearly depend on the reactor power. The examination has revealed failed fuel subassemblies. Two of them were found to release fission gas during the detection (at 100 and 200°C). Six subassemblies released fission gas only in excess of certain boundary temperature which was in the range from 380 to 500°C for different subassemblies. On the failed fuel subassemblies discharge delayed neutron flow decreased up to the initial level.

The delay neutron detection, unlike the fission gas control, does not result in so true information on the very moment and the number of fuel element failures. Here are some features of variations in the delay neutron monitor signal:

. In absence of failed fuel elements the monitor signal is proportional to the reactor power. It is stable at constant reactor power.

. With failed fuel element present in the core the monitor signal reading is not stable at constant reactor power; short--term fluctuation in the signal occurs during fuel element failure.

. An increase in delayed neutron detector signal is most often observed in some period of time after a fission gas activity burst.

. A change in reactor power leads to more pronounced variations in the detection system signal in presence of failed fuel subassemblies in the core.

Delay neutron signal analysis presents some information of failed fuel subassemblies in the core. However, it is usually the case with wide-open failures. Fuel pins with gaseous leaks are not detected with this technique. A great difficulty during the detection is presented by background originating from the initial contamination of fuel element cladding and consequent contamination of the core.

Thus the control system for fission gas activity and delayed neutrons in the coolant provide the operating staff with the required information on fuel element cladding condition. The fission gas activity is a means of determining the moments and the number of fuel element failures. The delayed neutron detection in the coolant informs about the fuel-coolant contact surface area in case of wide-open failures. This information is required to assess the primary circuit contamination with radioactive products and to prevent accidents due to several open failures or a single but severely damaged fuel element.

Further the experience gained with subassembly detection system operation during the reactor shutdown **will be considered.**

Blow-gas activity is measured with beta-counters. As an example of such measurements, Fig.3 shows the activity rate variation during degassing the sodium over the core, over a failed fuel subassemblies and over a blanket subassemblies. A minor activity rate for the blanket subassembly is due to the coolant degassing.

From the very beginning of using this technique it was revealed that several failed fuel subassemblies were not detected during the checks and left in the core. Some of the detected failed subassemblies were left in the core for further irradiation. In most cases (but not always), again they induced a higher gas activity rate. This fact has caused further improvement of the technique.

12 oxide fuel subassemblies of those which were detected to release ^{133}Xe have been examined in the hot cell. All of them have 1-2 and rarely up to 5 failed fuel pins. However of 19 checked "intact" fuel subassemblies which did not reveal ^{133}Xe release during the detection 3 subassemblies (i.e.16%) did have failed fuel pins. The origin of this phenomenon is not readily apparent. It should not be excluded from consideration that cladding of some fuel subassemblies might be defected after their discharge in the course of cleaning, transportation and dismantling.

Uncertainties occuring in failed fuel subassembly detection show that the following factors are required to be considered during the check and its results interpretation:

- the extent of fuel heating,
- defects size and location,
- fuel burnup level,
- uncertainties which may occur in the course of procedures.

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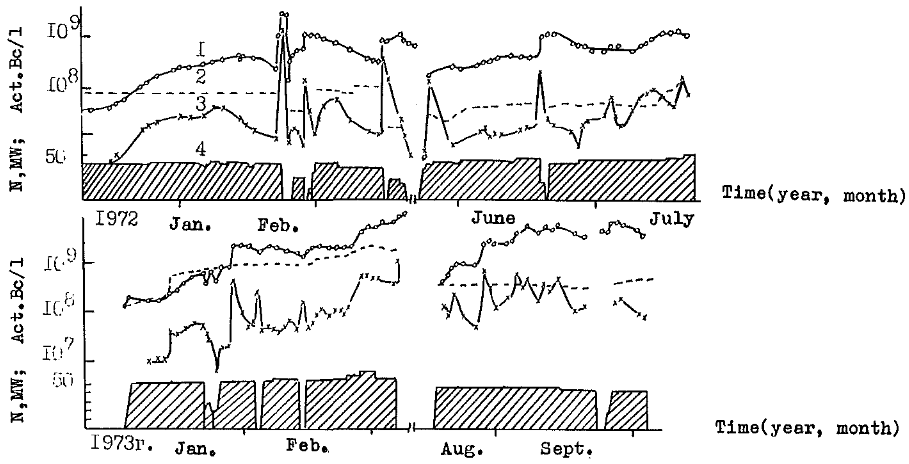


Fig. 2 Cover gas activity and delayed neutron signal versus BOR-60 operation time
1 - ^{133}Xe, 2 - delayed neutrons, 3 - ^{135}Xe, 4 - power history.

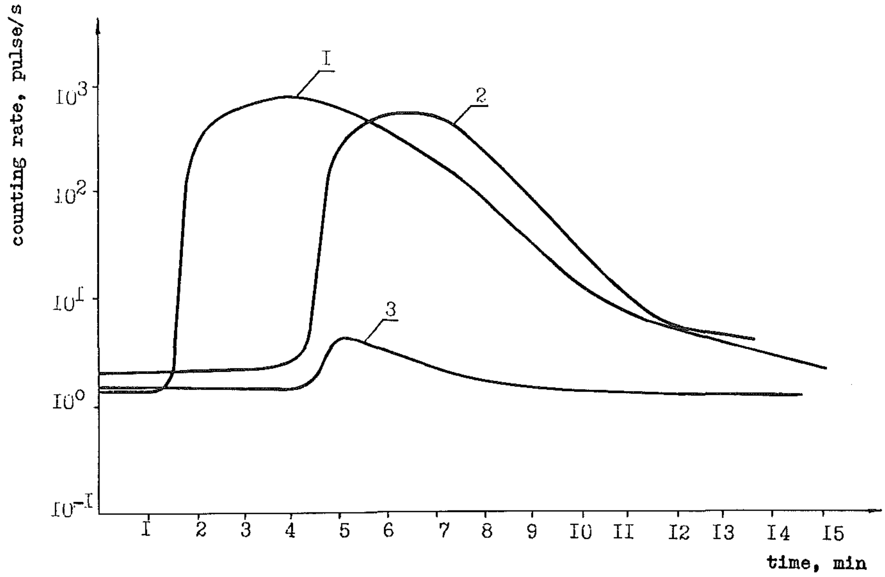


Fig. 3 Blow-gas activity variation :
1 - during argon bubbling of sodium, 2 - from a failed fuel subassembly, 3 - from a "dummy" blanket subassembly.

Discussion

J.D.B. Lambert, ANL:

At zero reactor power are failed subassemblies discharged?

E.K. Yakshin, SRIAR:
They were discharged this time in the fig. 2, but not that time (in February)

D.K. Cartwright, UKAEA:

What is efficiency of location technique for detection of failures?

E.K. Yakshin, SRIAR:
It reachs now up to 90 %.

J.D.B. Lambert, ANL:

Do you get an increased DN signal from open failures after restart of the reactor?

E.K. Yakshin, SRIAR:
After discharge the failed fuel subassemblies the DN signal decreased sometimes with a factor of 3 and more, but not to previous level.

A. Merkel, IA:
Did you find that fission gas release from mixed oxide fuel can be explained generally by diffusion theory, even if the fuel is at high temperature?

E.K. Yakshin, SRIAR:
We have experience on uranium oxide; statistic with respect to mixed oxide are not good, now.

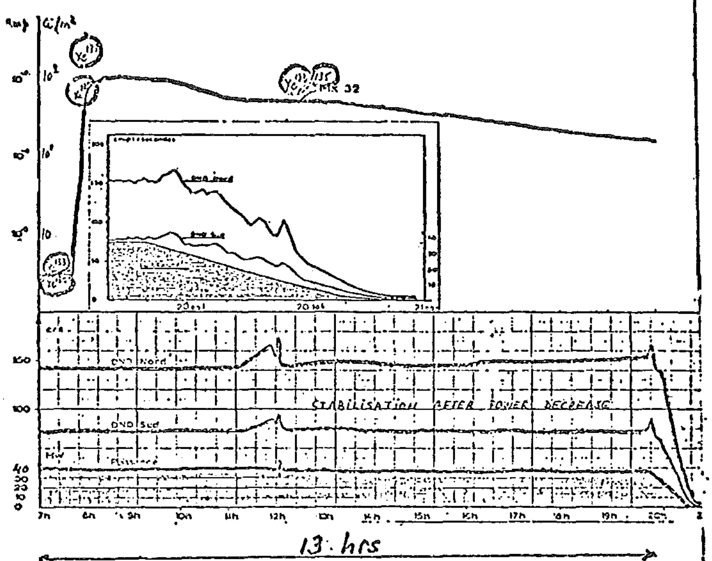
EXAMPLES OF LOCALISATION IN RAPSODIE

by P. Michaille

1/ FORTUNATE CASE

EVOLUTION OF THE SIGNALS

- HIGH GAS BURST ⟶ MEASUREMENT OF THE AGE
  MEASUREMENT OF THE TAG
- $3\frac{1}{2}$ HOURS AFTER : DND SIGNALS
  STABILISATION BY SLIGHT POWER DECREASE FOR 7 HOURS
  ⟶ MEASUREMENT OF THE N/S RATIO
- MANUAL REACTOR SHUT-DOWN



LOCALIZATION IN RAPSODIE - EXAMPLE N°1 - DND & GAS SIGNALS

RESULTS

AGE

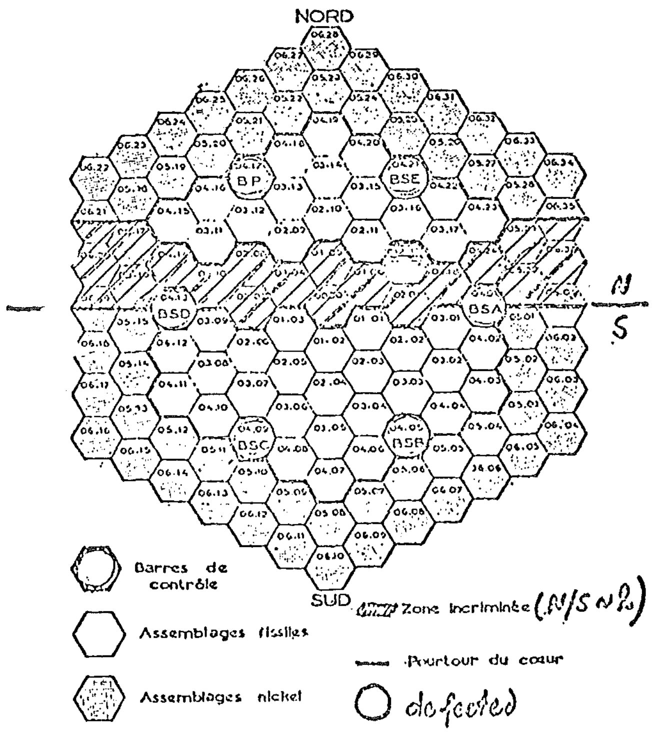
REAL	:	303 JEPC
MEASURED :	131/133	: 279
	132/133	: 288
	134/133	: 288
	136/133	: 270

TAG

ISOTOPIC RATIO	MEASURED	THEORETICAL
129/124	169	162
129/126	212	206
129/128	LT 6	12 FISSION PRODUCT

N/S DND RATIO : 2.2 CENTRE N (SEE FIGURE)

TIME LOSSED FOR THE IRRADIATIONS : 2.5 DAYS

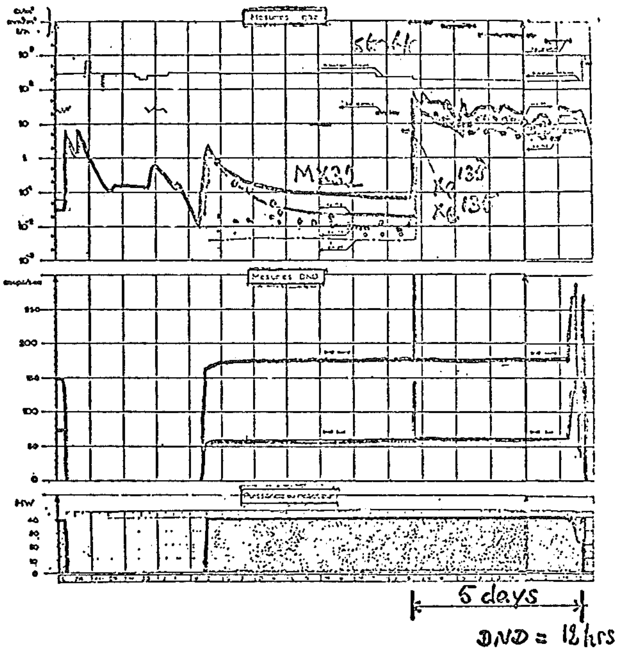


LOCALIZATION IN RAPSODIE

(2) UNFORTUNATE CASE

EVOLUTION OF THE SIGNALS

- SEVERAL PRIOR FAILURES IN THE CORE (GAS SIGNALS AT START-UP)
- HIGH DND PEAK CONCOMITANT WITH IMPORTANT GAS BURST (ACTIVE & STABLE GASES)
- RAPID DND EVOLUTION 5 DAYS AFTER



RESULTS

- AGE : MEASURED = 710 DAYS ; REAL = 961 DAYS
- TAG : NOT FOUND (NOT TAGGED)
- N/S DND RATIO = 1.45   EXPECTED LOCATION = CENTRE N
  REAL LOCATION = DEEP SOUTH

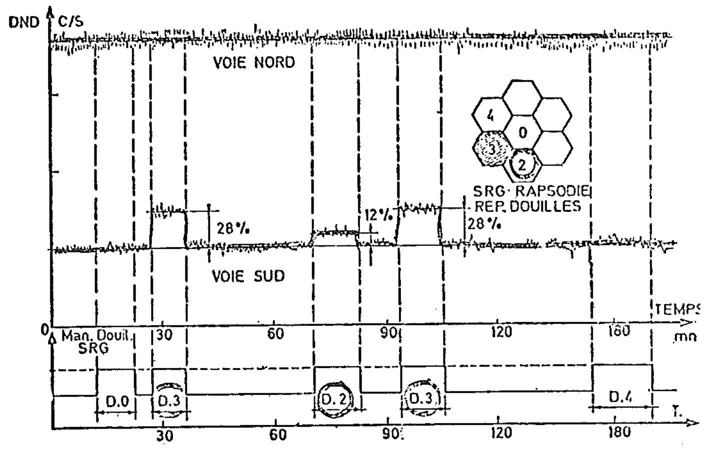
TO FIND THE DEFECTED SUBASSEMBLY, 4 CORE RE-ARRANGEMENTS AND REACTOR START-UPS WERE NEEDED

HENCE A LOSS OF 26 DAYS

(3) NORMAL CASE

LOCALISATION BY DND DEVICE : SRG

RAPSODIE: LOCALISATION D'UNE RUPTURE A L'AIDE DU SRG



METHODS OF LOCALISATION
CONCLUSIONS FROM THE RAPSODIE OPERATION

·WITH THE PARTITION FACTORS, IF ONE FACTOR FAILS, MUCH TIME IS NEEDED FOR LOCALISING

THE CAUSES OF ERROR ARE NUMEROUS:

FOR THE GAS MEASUREMENTS
- PRESENCE OF SEVERAL FAILURES IN THE CORE,
- SLOW RELEASE OF FISSION PRODUCTS (MALFABRICATION DEFECTS)

FOR THE DND MEASUREMENTS:
THE HYDRAULIC OF RAPSODIE IS VERY SENSITIVE TO THE ENVIRONMENT (CAPSULES HEAD)

THEREFORE, INDIVIDUAL SAMPLING DEVICES HAVE BEEN DEVELOPPED:

AT RAPSODIE : SRG
AT PHENIX : LRG

PHENIX - LOCALIZATION OF THE FIRST CLAD FAILURE

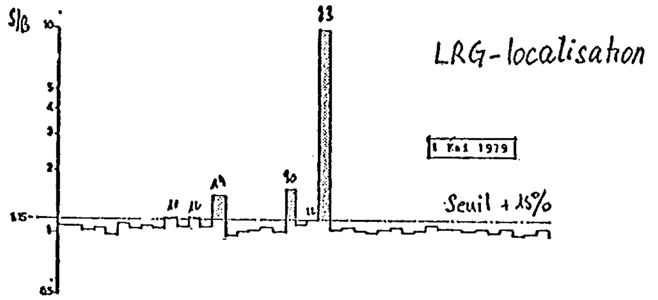
by C. Berlin

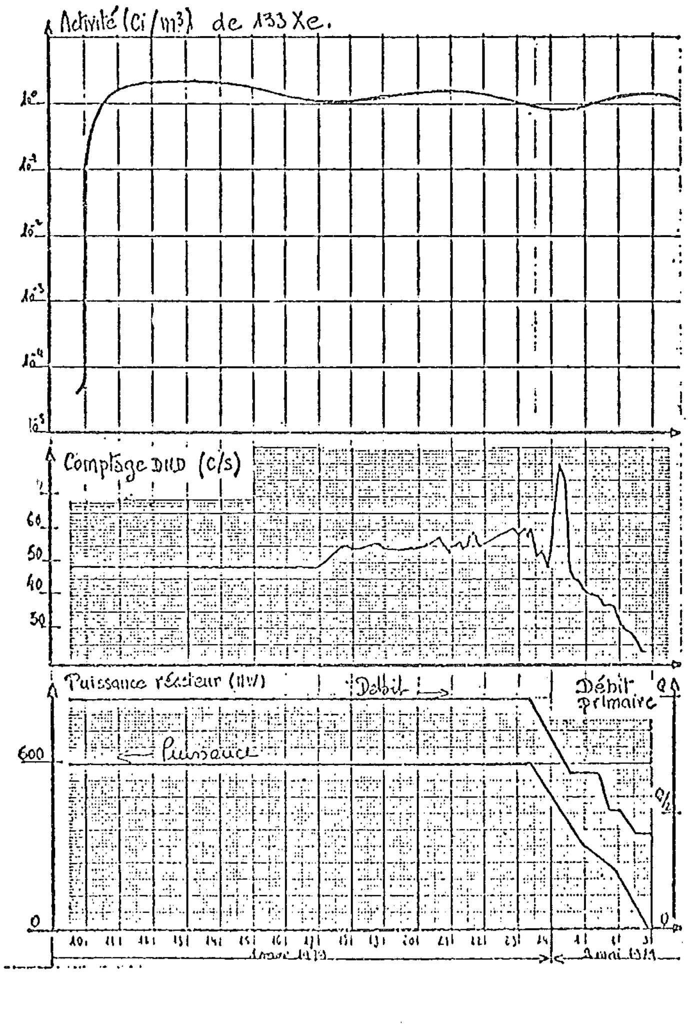
EVOLUTION OF THE SIGNALS

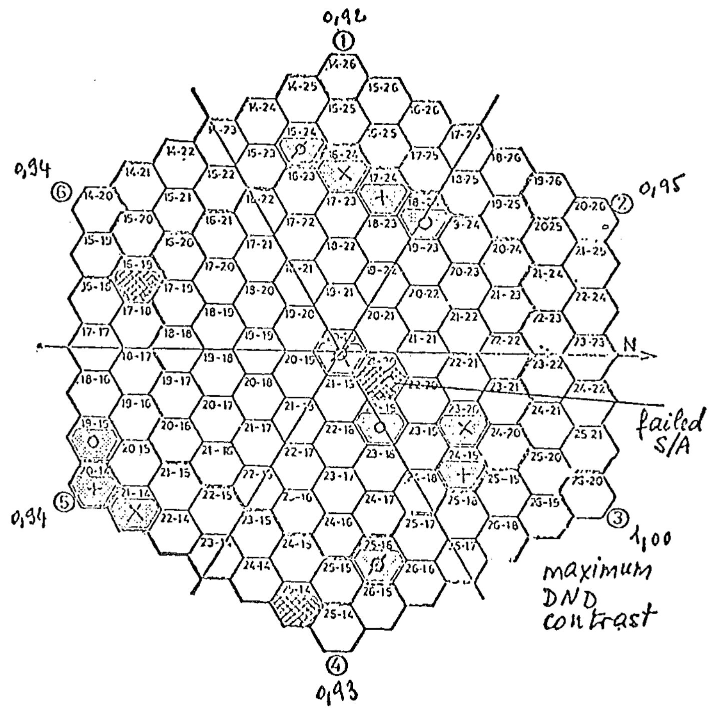
- Gas signals at the beginning of the run, (slightly decreasing during 30 days)
- High gas burst : measurement of the age,
- 8 hours after : DND signals,
- Reactor at full power during 6 hours :
  - research of the failed subassembly with the FFD device (G3 mode),
  - measurements of the prelocalization ratios,
- Manuel reactor shut-down

RESULTS

- FFL : three subassemblies are suspected (see figure)
- Sorting : 1/ hydraulic consideration see figure
  - 2/ prelocalization : sector n°3 (see figure)
  - 3/ age : old subassembly
    - 134Xe/133Xe : ~ 120
    - large quantity of 85 Kr.







Position G3 défectueuse
defected possible positions (G3)

①②③④⑤⑥ : secteur de prélocalisation DND/G.

PHENIX - TEST OF THE GAS-FFL DEVICE

by C. Berlin

- SETTING UP A NEW GAS-FFL DEVICE (MID OF 1980)

- TWICE, THE GAS-FFL INSTRUMENTATION ALLOWS THE LOCALIZATION OF GAS FAILURES (REACTOR ON POWER)

- EXAMPLE :

  * EVOLUTION OF THE GAS-SIGNALS IN THE PLENUM OF THE REACTOR (SEE FIGURE)

  * LOCALIZATION

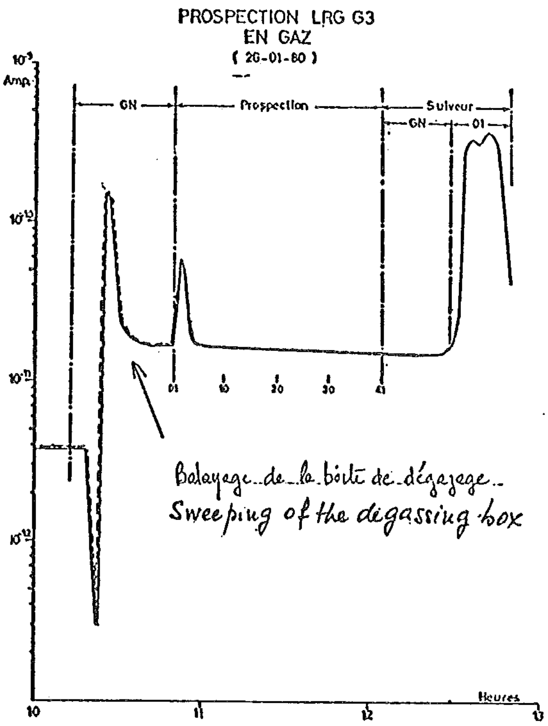
    - SWEEPING OF THE DEGAZOR DURING 15'

    - MEASUREMENT DURATION (IONIZATION CHAMBER) 2 MINUTES PER EACH POSITION OF THE SELECTOR VALVE (TRANSIT TIME = 15')

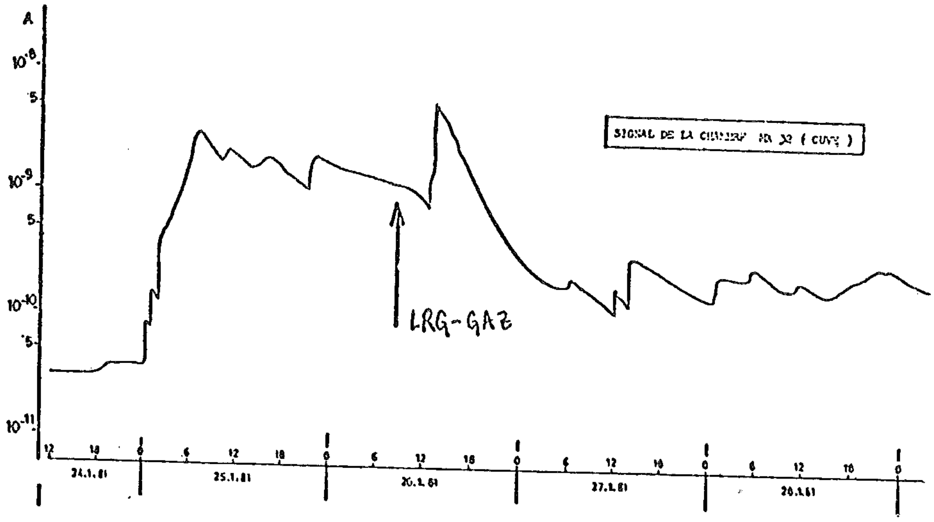
      GOOD ESTIMATE OF THE DEFECTIVE POSITION (G3)

    - CONFIRMATION BY A PROSPECTION STEP BY STEP

    - SORTING : AGE 134XE / 133XE
      OLD SUBASSEMBLY



EVOLUTION DES SIGNAUX GAZ



Discussion

D.B. Sangodkar, R R C K :

You have used U-Cr or U-Mo alloy for calibration of DND. Are the results directly applicable to oxide fuel?

P. Michaille, CEA:
No. Up to now, we make a translation to oxide by calculation (mean free path of the neutronuclides, especially).

J. Dauk, IA:

Do you know the reduction of the transport time by your SR6-system or is the explanation only qualitatively?

P. Michaille, CEA:
No measurement has been made as yet. One may expect a reduction of the transit time from 26 up to 6 sec if one considers that the main part of the delay due to the instrumented plate just above the core.

PRESENTATIONS AND DISCUSSIONS

OF SESSION III:

IN-PILE EXPERIMENTS

SUMMARY OF IN-PILE LOOP EXPERIMENTS RELATED TO THE DEVELOPMENT OF THE FUEL FAILURE DETECTION SYSTEMS FOR LMFBR'S IN JAPAN

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ABSTRACT

Many extensive works performed in Japan on the development of the fuel failure detection systems for LMFBR's since 1972 using two sodium in-pile loops(SIL and FPL) and one cover gas loop(CGMTF) are introduced and briefly reviewed.

INTRODUCTION

Table 1 summarizes the in-pile loop experiments carried out in the fuel failure detection(FFD) system development program for Japanese LMFBR's designing. The loop names, their locations, their reactors where the loops were installed and their main objectives as well as references[1 - 5] are indicated. The main objectives were to test various FFD systems for the Japanese first experimental LMFBR "JOYO" and the second prototype LMFBR "MONJU" for designing and licensing purpose, and also to understand fission product(FP) behavior in sodium and cover gas from the view point of FFD design and safety.

All the three in-pile loops located in the thermal research reactors. The SIL(Sodium In-pile Loop)[1,2] and the FPL(Fission Product Loop)[3 - 5] used only new uranium metal plates, UO_2 pellets or UO_2 balls as fission sources which were immersed in sodium flow. The CGMTF(Cover Gas Monitor Test Facility) used only helium cover gas containing rare gas FP's produced from the residual uranium contamination in the reactor as a result of some fuel failures in the past.

SIL EXPERIMENT[1 , 2]

The SIL, a sodium in-pile loop located at Japan Research Reactor-2(JRR-2) in Tokai Research Establishment, Japan Atomic Energy Research Institute as shown in Fig.1, contained 8.5 liter sodium in which uranium metal plates[1] or UO_2 pellets[2] had been irradiated by a thermal neutron flux of 5×10^{10} n/cm^{2}s at the reactor full thermal output power of 10MW. The calculated fission rates from the former and the latter fissile materials were 3.5×10^{11} and 2×10^{11} fissions/s, respectively. The cover gas was helium. A delayed neutron monitor, a moving-wire type precipitator(Plessey MK-XI) and a cover gas reservoir - Ge(Li) gamma-ray spectrometer had been tested. The counting efficiency of the delayed neutron monitor(Fig.2) was measured in a simulation experiment using a 2Ci Am-Be neutron source embedded in a 20cm cubic lead block which produced a simulated delayed neutron spectrum as shown in Fig.3. The obtained counting efficiency was 4.6×10^{-4} cts/n for the graphite moderator. Using this counting efficiency with the obtained counting rate of the delayed neutron monitor, the FP release fraction from fuel can be evaluated as shown in Table 2. When these fractions were compared with those calculated from the neutron flux with a recoil length of 5μm, the fissuring factor became 1.28 for the metallic uranium plates and 3.5 for UO_2 pellets. These fissuring factors are reasonable since the experimental conditions limit the FP release only due to recoil mechanism. The release fractions induced from sodium and cover gas gamma-ray spectrometries[6] are also listed and show the same order of the magnitude.

From the cover gas gamma-ray spectrometry, the FP release fraction from fuel f and the FP transfer rate from sodium to cover gas η were calculated using a method shown in Table 3. For Kr and Xe, η's calculated were 7.7×10^{-3}min^{-1}($T_{1/2}$=90min) and 5×10^{-3} min^{-1}($T_{1/2}$= 138min), respectively.

The delayed neutron counting rate was measured as a function of the temperatures of the irradiation section and the main cooler. The results are shown in Figs. 4 and 5, which show some adhesive natures of delayed neutron precursor nuclides on the stainless tube wall.

The effect of the SIL operational conditions on the counting rates of the delayed neutron monitor, the cover gas gamma-ray spectrometer and the precipitator is summarized in Table 4.

The results obtained in the SIL experiment as far as testing of FFD systems concerns show nothing so peculiar and the test results of the delayed neutron monitor and the precipitator were used to obtain the licensing of "JOYO". The precipitator was transferred to "JOYO" and are being used as an actual cover gas monitor.

FPL EXPERIMENT[3 - 5]

The FPL(Fission Product Loop), a sodium in-pile loop located at Toshiba Training Reactor(TTR) in Nuclear Engineering Laboratory, Toshiba Corporation, had 4.8Kg sodium with natural UO_2 balls of 150g exposed to a thermal neutron flux of 1×10^{10} n/cm^{2}s at 100kW reactor thermal power. A simplified diagram of the FPL is shown in Fig.6. The fission rate was calculated as. 1.4×10^{10} fissions/s. Argon was used as the cover gas in the expansion tank. Two delayed neutron monitors, a newly-developed fixed-wire type precipitator and a Ge (Li) gamma-ray spectrometer were used to obtain their performance as a function of the sodium flow rate and temperature as well as the reactor power.

The newly-developed precipitator system[5] are constituted with three tubes as shown in Fig.7, and the tubes act as FP collectors collecting positively-charged nuclides on the central wires during a soak time with cover gas flow and also act as proportional counters during the following counting period with pure-Ar counting gas flow. The proportional counters count β-particles emitted from the nuclides collected on the wires. In the system as shown in Fig.8, the gas flow was controlled to the three FP collector/proportional counter tubes by a switching valves. A precipitation voltage of -200V and a proportional counter voltage of +1500V were switched sequentially. The average precipitation efficiencies for Rb and Cs were found to be (65 ± 15)% and the counting efficiency of Au-198 β-particles(962keV) 0.73 ± 0.04 cts/β.

Figure 9 shows an example of the precipitator response to a reactor power change. The delayed neutron counting rates are also shown. From the other experiments, no accumulation of Rb and Cs was found on the wire although the reason is not exactly understood. In order to prepare the method of eliminating any accumulated long-lived nuclides on the wire, a test was performed by heating the wire at 400°C for 300s in 0.1 torr vacuum and resulted in 20% reduction in the precipitated nuclides. This type of the precipitator is under consideration in designing the FFD systems of "MONJU".

The other group in Toshiba Corporation[4,5] made an extensive study on FP behavior in sodium and cover gas using the FPL. They found the cover gas flow rate had no effect on the release rates of FP gas from sodium to cover gas; the release fractions of Kr and Xe to from sodium to cover gas depended strongly on sodium temperature(Fig.10); and the release fractions of Kr and Xe increased as sodium flow rate increased(Fig.11).

They also found two kinds of degassing halflives as shown in Figs. 12 and 13. One was 80 to 200min calculated from the release fractions after reactor shut down and also from log(R/B) plot as a function of log λ_i. One of the examples is shown in Fig.12. The other was 17 to 32min calculated from the increasing release rates after the reactor start up, and found to be independent of nuclide halflife and to depend on sodium temperature. The latter was caused by the diffusion through the liquid sodium-cover gas interface as shown in Fig.13. For Xe-135 and Xe-135m release, Xe atoms produced by the decay of iodine absorbed on the colder wall of the loop was found not to be transported to the cover gas; the same phenomenon was also observed in the SIL experiment (Table 4).

CGMTF EXPERIMENT

The CGMTF, a cover gas monitor test facility located at Japan Research Reactor-3(JRR-3) in Tokai Research Establishment, JAERI, was built to test various types of cover gas monitors using the reactor helium cover gas which contained gaseous FP's of about 0.01uCi/cm^3 for each of the nuclides and Ar-41. A cover gas reservoir-Ge(Li) on-line gamma-ray spectrometer system, a moving-wire type precipitator, a fixed-wire type precipitator, a room temperature charcoal chamber for high-sensitivity cover gas gamma-ray spectrometry and a membrane unit for enriching gaseous FP's are being tested. Tables 5 and 6 show the examples of the results obtained from the room temperature charcoal chamber and from the membrane unit. These results will be reflected in the design of the FFD systems and also of the tag gas system of "MONJU"[7].

CONCLUSION

The experiments carried out in two sodium in-pile loops(SIL and FPL) and one cover gas loop(CGMTF) have resulted in a very informative basis for testing, designing, licensing the FFD systems for Japanese LMFBR's "JOYO" and "MONJU", and also for understanding the basic mechanisms of FP behavior in sodium and cover gas. Operations of the SIL and the FPL had been terminated in 1976 and 1979, respectively. The second FPL is under construction for studying FP behavior in primary coolant systems of LMFBR's. The operation of the CGMTF will be continued for testing reliability of the performance of FFD components and systems until the next year(1982).

The experimental LMFBR "JOYO" has no experience on fuel failure up to now and also no tramp fuel has been detected[8]. However, it is prerequisite to maintain any FFD system performance at any time and to understand the correspondence of various FFD signals to fuel faillure in any FBR plant. In this sense, an appropriate means of in-pile calibration of the FFD systems should be considered in order to confirm the designed counting efficiencies of the "JOYO" FFD systems[9].

ACKNOWLEDGMENT

The authors wigh to express their thanks to many people who cooperated with the works described in this paper, especially to Messrs. Y.Mimoto, K.Nakamoto, K. Imani, H.Rindo, N.Mitsutsuka, Power Reactor and Nuclear Fuel Development Corporation; Mr.M.Katagiri, Japan Atomic Energy Research Institute; Mr.M.Fujino, Hitachi Ltd.; and Mr.K.Kubo, Toshiba Corporation.

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Table 1

SUMMARY OF IN-PILE LOOP EXPERIMENTS RELATED TO THE DEVELOPMENT OF THE FUEL FAILURE DETECTION SYSTEMS FOR LMFBR IN JAPAN

Eiji Sakai(JAERI), Tatsuo Miyazawa(Toshiba Corp) and Nobutada Sekiguchi(PNC)

Loop name	SIL(Sodium Inpile Loop)	FPL(Fission Product Loop)	CGMTF(Cover Gas Monitor Test Fac.)
Location	JAERI-Tokai	Toshiba Corp.-Kawasaki	JAERI-Tokai
Reactor	Japan Research Reactor 2	Toshiba Training Reactor	Japan Research Reactor 3
Year	1972 - 1976	1974 - 1979	1978 - 1982
Object	1)FP behavior in sodium from the point of view of safety (by Hirano, et al) 2)Test of FFD systems for licensing "JOYO" i)DNM ii)Moving-wire precipitator β-detection γ-spectrometry iii)Ge γ-spectrometry (by Sakai,Fujino,et al.)[A,B]	1)FP behavior in sodium and cover gas from the view point of safety and FFD (by Mitsutsuka,et al)[C,D] 2)Development and test of FFD systems for "MONJU" i)Fixed-wire precipitator ii)Ge γ-spectrometry iii)DNM (by Miyazawa,et al)[E]	1)Development and test of FFD system for "JOYO" i)On-line Ge spectrometry ii)Room temperature charcoal trap + i) iii)Cooled charcoal trap + i) iv)Fixed-wire precipitator v)Membrane unit + i) (Sakai,Kubo, et al)

References; A. Sakai,Katagiri,et al,IEEE Trans.,NS-23(1),363(1976)
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C. Mitsutsuka,et al,Int.Conf.Liq.Metal Tech.in Energy Production,CONF-760503-P1,284(1976)
D. Mitsutsuka,et al, ibid.,291
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Table 2 Comparison of fission product release fraction obtained from various methods

Run number of sodium circulation exp	No.2 - No.10 (20%-enriched metal U)		No.11 - No.17 (90%-enri.UO_2)
Sodium temperature outside of fuel(°C)	520(typical)		500(typical)
Expansion tank temperature(°C)	372(typical)		340(typical)
Sodium flow rate(ℓ/min)	3.0(typical)		3.0(typical)
Total sodium circulation time(h)	2739		1606
Delayed neutron counting rate(cps)	No.2	No.10*	
	1884.3	13000	2300
Background counting rate(cps)	6.2		9.0
Detection sensitivity($cps/cm^2/MW_{th}$)	3.3		13.5
S/N ratio	302.9		254.6
Minimum detectable surface area of fuel	$0.186cm^2$ (4.9mm dia)		$0.062cm^2$ (2.9mm dia)
Fission product release fraction from fuel			
Calculated from recoil length of 5μm	2.8×10^{-3}		2.25×10^{-3}
From delayed neutron counting rate	3.6×10^{-3}	2.5×10^{-2}*	7.70×10^{-3}
Fissuring factor k	1.28	8.93*	3.5
From cover gas gamma spectrometry			5.0×10^{-3} (Kr-85m,87,88) 1.5×10^{-3} (Xe-135m,135,138)
From sodium gamma spectrometry		1.8×10^{-2}* (Rb-89,Te-134,I-134, Xe-138,Rb-138)	4.2×10^{-3} (Kr-90) 6×10^{-3} (I-136)

*Post-irradiation examination revealed that the surfaces of the fuel plates were oxidized, eroded and corroded and also some cracks by erosion penetrated through the plates.

Table 3

A METHOD TO CALCULATE F.P. RELEASE FRACTION FROM FUEL (F) AND F.P. TRANSFER RATE FROM SODIUM TO COVER GAS (η) USING F.P. CONCENTRATIONS IN COVER GAS

C_k^i : Kr-i (or Xe-i) concentrations in cover gas (Ci/cc)

C_k^j : Kr-j (or Xe-j) concentrations in cover gas (Ci/cc)

f_k : Kr (or Xe) release fraction from fuel

η_k : Kr (or Xe) transfer rate from sodium to cover gas (s^{-1})

$$\eta_k^i \equiv \eta_k^j$$

Ai, Aj, Bi, Bj : Constants calculated from decay constant, sodium flow rate and sodium volume.

Ci, Cj: Constants calculated from sodium flow rate and sodium volume

Ki, Kj: Constants calculated from decay constant, sodium flow rate and cover gas flow rate.

Di : Constant calculated from cover gas flow rate and cover gas volume

F : Amount of Kr (or Xe) generated in fuel in unit time.

$$\frac{C_k^i}{C_k^j} = \frac{A_j\,(B_j + \eta_k) - C_j}{A_i\,(B_i + \eta_k) - C_i} \cdot \frac{K_j}{K_i}, \qquad \longrightarrow \eta_k$$

$$C_k^i = D_i\, K_i\, \eta_k\, f_k\, F \qquad \longrightarrow f_k$$

Example

f(Kr-85m,87,88) = 5.0×10^{-3}

f(Xe-135m,135,138) = 1.5×10^{-3}

η(Kr) = $7.7 \times 10^{-3} min^{-1}$ ($T_{1/2}$=90min)

η(Xe) = $5 \times 10^{-3} min^{-1}$ ($T_{1/2}$=138min)

Table 4

THE EFFECT OF SIL OPERATIONAL CONDITIONS
ON FFD COUNTING RATES

Sodium inpile loop	90%-enriched UO_2 pellets (5mm dia. x 5mm x 24 pieces, U-235 8.01g, surface 16.30cm²), JRR-3 10MW, fuel outside 7.3 x 10^{11} nth/cm²s, fuel inside 1.64 x 10^{11} nth/cm²s, 1.97 x 10^{11} fissions/s, Irradiation section 500°C, Main cooler 400°C, Expansion tank 380°C, Main sodium 3l/min, Cold trap 200°C, Cold trap sodium 0.3l/min, Total sodium 8.5l		
Fuel failure detector	**Delayed neutron monitor**	**Cover gas gamma-ray spectrometer**	**Precipitator**
Detector	50 x 50 x 60cm³ graphite moderator + BF_3 counter (73cps/nv), Detection efficiency 4.6 x 10^{-5} c/n, 1st(56s), 2nd(22s), 3rd(6.2s) groups	100cm³ gas reservoir + 54cm³, 5% Ge(Li) detector, Kr-85m, 87, 88, Rb-88, Xe-135m, 135, 138, Cs-138	Precipitation chamber 1017 cm³ Rb-88(17.8min) Cs-138(32.2min) Collection efficiency of Rb-88 is almost 100%
Delay time	fuel→ expansion tank inlet 13.1s (3l/min)	fuel→ reservoir inlet 45min (100cc/min)	fuel→ chamber inlet 45min (100cc/min)
Long-term change in counting rate	Stable – UO_2 pellets were stable.	Stable – UO_2 pellets were stable. Sometimes, the counting rates of only Xe-135 (9.2h), Kr-85m (4.4h), Kr-88 (2.8h) increased. → Release of F.P. from F.P. trapping point.	Stable – UO_2 pellets were stable. Sometimes, the counting rate increased → Release of F.P. from F.P. trapping point.
Counting rate dependence on temperatures of irradiation section, main cooler, expansion tank	Counting rate decreased as temperatures decreased → F.P. adhered on tube wall.	Peak counting rates decreased as temperatures decreased. → F.P. adhered on tube wall. F.P. transfer rate may depend on temperature.	Counting rate decreased as temperatures decreased. → F.P. adhered on tube wall. F.P. transfer rate may depend on temperature.
Counting rate dependence on sodium flow rate	Counting rate increased as flow rate increased.	Counting rate increased a little as flow rate increased	
Counting rate dependence on cold trap temperature	Constant counting rate for 270 – 150°C.	Counting rates of only Xe-135m and Xe-135 decreased to 1/7 and 1/3 as temperature decreased from 270 to 150°C. → I-135 adhered on cold trap wall.	Constant counting rate for 270 – 175°C.
Counting rate dependence on cold trap sodium flow rate	Constant counting rate for 0.14– 0.9l/min		Counting rate decreased a little as flow rate increased.
Counting rate dependence on cover gas flow rate		For 200→100→50cc/min cover gas change, counting rates of Xe-138(17 min), Cs-138 (32.2min) decreased and those of Kr-85m(4.4h), Kr-87 (78min), Kr-88 (2.8h), Rb-88(17.8min) Xe-135 (9.2h) increased. →Short halflife F.P.s decayed.	For 200→100cc/min cover gas change, counting rate decreased at first, then recovered the original counting rate. This reason is not yet known. For 100→50cc/min change, counting rate decreased.
Counting rate dependence on purge gas flow rate			Counting rate decreased as purge gas flow rate increased. → F.P. concentration became thinner.
Effect of adding natural Kr, Xe gas	No effect	Counting rates of Kr-85m, 87, 88 decreased a little	Counting rate decreased a little
F.P. release fraction from fuel (Calculation from 5um range of F.P. gives 2.2 x 10^{-4})	7.7 x 10^{-7}	7.3 x 10^{-4} (from Kr concentration) 4.3 x 10^{-4} (from Xe concentration	
F.P. transfer rate from sodium to cover gas at expansion tank		Kr 7.7 x 10^{-3} min^{-1} Xe 5 x 10^{-3} min^{-1}	

Table 5 Rare gas nuclide concentrations in JRR-3 cover gas and in charcoal-filled chamber(TSURUMI COAL 2GM, 50g, 100cm³, room temperature), observed and calculated enrichment factors, and observed adsorption coefficients

JRR-3 cover gas flow rate was 1 liter/min. Pressures at vacant chamber and at the charcoal chamber outlet were 0.6kg/cm²G and 0.15kg/cm²G, respectively.

Nuclide	Decay constant λ (min^{-1})	Concentrations (uCi/cm³) Cover gas in vacant chamber*1)	In charcoal chamber*2)	Observed enrichment factor	Observed adsorption coefficient, k(cm³/g)*3) Piston flow	Complete mixing	Calculated enrichment factor*4) Piston flow	Complete mixing
Kr-85m	0.00265	0.01345	0.25	18.9	37.9	38.0	12.5	12.5
Kr-87	0.00908	0.03358	0.54	16.5	33.2	33.5.	12.4	12.4
Kr-88	0.0039	0.03752	0.65	22.7	45.6	45.8	12.5	12.4
Xe-133	0.00022	0.01079	4.7	423.5	851	855	124.8	124.6
Xe-135m	0.044	0.002518	3.4	136.4	415	675	96.3	80.9
Xe-135	0.00127	0.05956	2.7	460.4	949	978	124	123
Xe-138	0.0485	0.01391	6.9	469.7			93.9	78.2
Ar-41	0.0063	0.04079	0.11	2.5	5.0	5.0	1.25	1.25

*1) 1980/6/26 Run#99, 1 liter/min, 0.6kg/cm²G. The listed concentrations were for 0kg/cm²G and obtained by calculating (1/(1+0.6))xobserved concentrations.

*2) 1980/6/26 Averaged saturated concentrations at 1 liter/min and 0.15kg/cm²G. The listed concentrations were for 0kg/cm²G and obtained by calculating (1/(1+0.15))xobserved concentrations.

*3) $k=-(F/\lambda W)\ln[1-A+A\exp(-\lambda V/F)]$ for piston flow model; $k=(AV/W)/[1-(\lambda V/F)(A-1)]$ for complete mixing model

*4) Calculation was made using k=25cm³/g for Kr, 250cm³/g for Xe and 2.5cm³/g for Ar.

Table 6 Rare gas radioactive nuclide concentrations in inlet gas and in enriched gas of membrane unit observed enrichment factors, corrected enrichment factors and enrichment factors obtained from non-active standard gases.

Operation conditions of the membrane unit were 5Kg/cm²G pressure, 5 liters/min feed gas flow rate and 4.75 liters/min reject gas flow rate. The gas was JRR-3 cover gas which consisted of He carrier gas including Ar-41 and rare gas fission products. The experiments were conducted on June 6, 1980.

Nuclide	$T_{1/2}$ (min)	Concentrations (uCi/cm³) Inlet gas(MC-2)	Enriched gas(MC-3)	Observed enrichment factor Enriched con./ Inlet concentration	Corrected enrichment factor *	Enrichment factor obtained from non-active standard gas measurement
Kr-85m	268.8	1.478x10^{-2}	5.520x10^{-2}	3.735	3.947	6.5 for 4950ppm Kr in He
Kr-87	76	1.746x10^{-2}	8.284x10^{-2}	4.716	5.732	
Kr-88	170.4	3.781x10^{-2}	1.851x10^{-1}	4.896	5.341	
Xe-135m	15.6	2.700x10^{-3}	7.834x10^{-3}	2.901	7.508	14.9 for 522ppm Xe, 13.1 for 1040ppm Xe, 10.4 for 5090ppm Xe in He
Xe-135	546	3.930x10^{-1}	2.924x10^{-1}	7.440	7.440	
Xe-138	14.1	1.890x10^{-2}	4.860x10^{-1}	2.57	7.359	
Ar-41	109.8	6.78 x10^{-2}	1.610x10^{-1}	2.375	2.719	

* Assuming average time between inlet and enriched outlet of 21.4min which was calculated from the same enrichment factor of 7.44 for Xe-135m, 135 and 138.

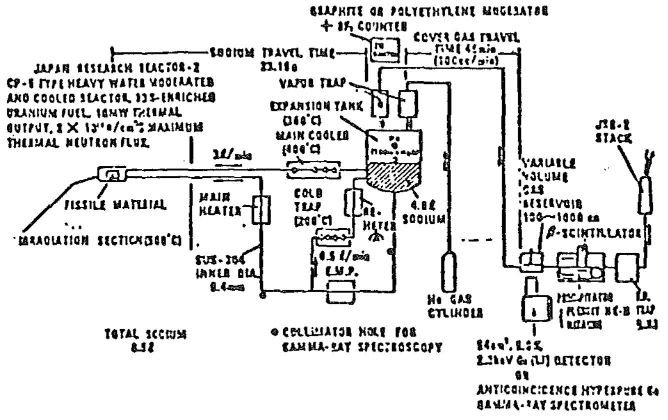
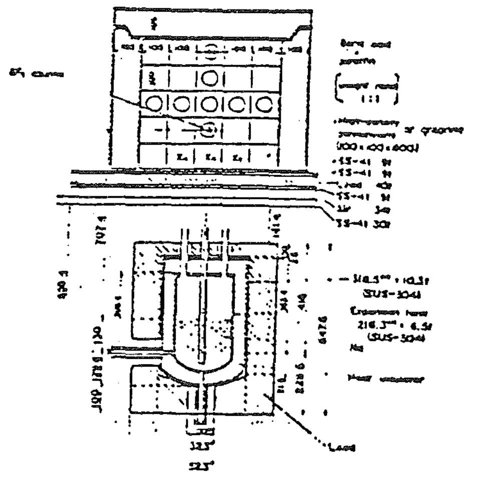


FIG. 1 SCHEMATIC DIAGRAM OF SODIUM IN-PILE LOOP



BF$_3$ counter
Twenty Century
40EB-70/50G
73cps/nv

Fig.2 Delayed neutron monitor

Run-number of sodium circulation		No.2 ~ No.10	No.11 ~ No.17
Fuel (FP source)		20% enriched metallic	90% enriched UO_2
	Dimension, number	37.6cm x 15.9mm x 1mm 2 plates; 37.6mm x 18.9mm x 1mm 2 plates	O.D. 9mmd, 2mmt, 24 pellets
	U-235 weight (g)	9.18	8.01
	Total surface area (cm^2)	56.44	16.96
	JRR-2 Power (MWt)	10	10
	Thermal neutron flux outside[1] (n/cm² sec) ϕ_0	5.4×10^{10} (measured)	7.3×10^{10} (measured)
	Thermal neutron flux inside[2] (n/cm² sec) ϕ_{th}	2.44×10^{10} (calculated)	1.64×10^{10} (calculated)
	Fissions/sec[3]	3.47×10^{11} (calculated)	1.97×10^{11} (calculated)
	Heat generation (W)	11.2	6.4
	Fission product release fraction[4]	0.28%	0.225%
	Total sodium circulation time (h)	2739	1606
	Burn-up	0.022% or 41MWD/T of U	0.007% or 65MWD/T of U

1) Calculated from the saturated Na-24 activity measurement. Fast neutron flux was negligibly small since no 440keV gamma-rays from Na-23 was observed.

2) $\bar{\phi}_{th} = \phi_0/(1+\Sigma_t\bar{l})$, Σ_t = the sum of the macroscopic total cross section of U-235 ($15.424cm^{-1}$), U-238 ($0.0282cm^{-1}$) and oxygen atoms ($0.218cm^{-1}$) = $15.67cm^{-1}$.
$\bar{l} = 4V_f/S_f$ 3) fissions/s = $\bar{\phi}_{th}\Sigma_f V_f$ 4) $f = S_f R/(4V_f)$, R = recoil length = 5.0μm

Delayed neutron counting efficiency obtained from the simulation experiment

$= 4.6 \times 10^{-4}$ counts/n

(Graphite moderator)

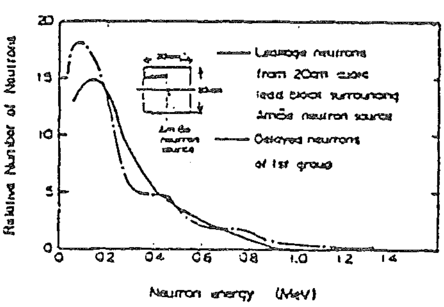


Fig.3 Energy spectra of leakage neutrons from lead block surrounding AmBe source and of first group delayed neutrons

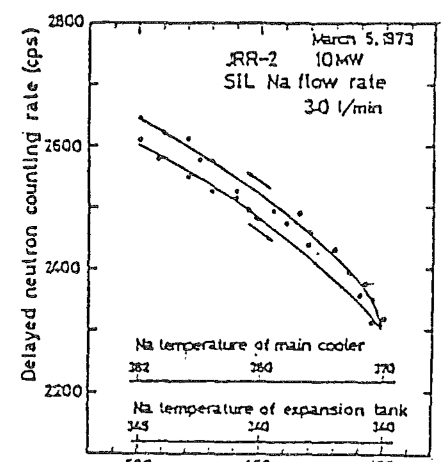


Fig.4 Delayed neutron counting rate vs. sodium temperature at irradiation section

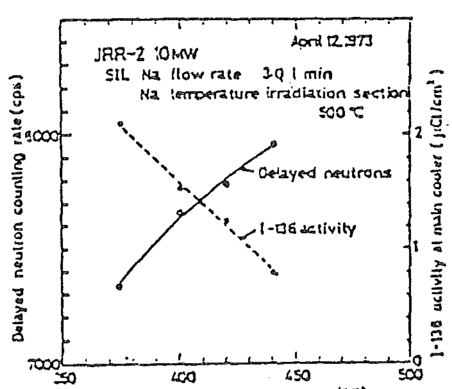


Fig.5 Delayed neutron counting rate vs. main cooler sodium temperature

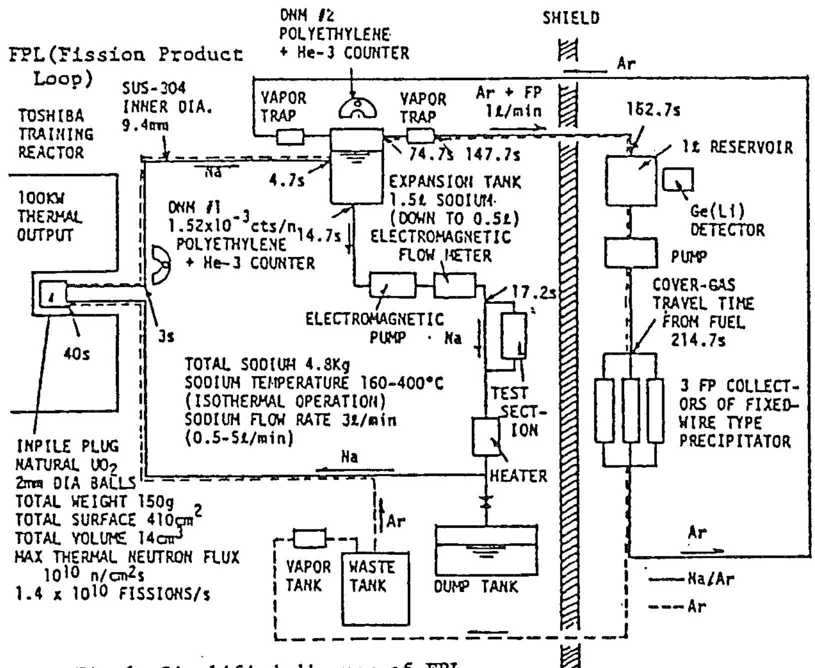


Fig.6 Simplified diagram of FPL

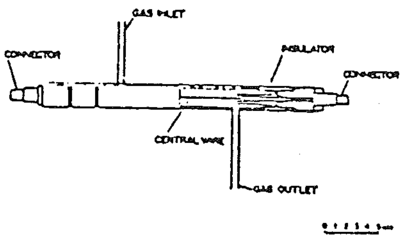


Fig.7 Schematic drawing of FP collector/proportional counter

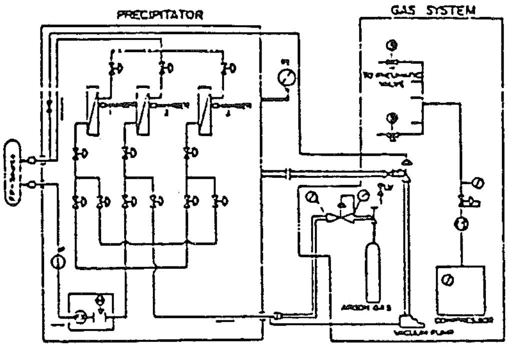


Fig.8 Gas flow diagram of the fixed-wire precipitator

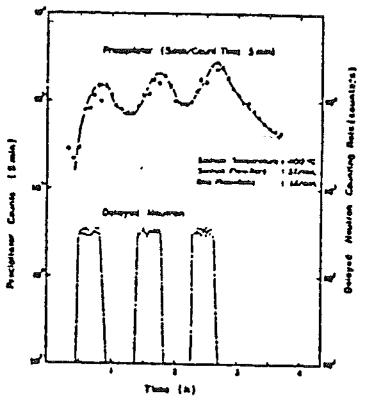


Fig.9 Responses of fixed-wire precipitator and delayed neutron monitor on reactor power

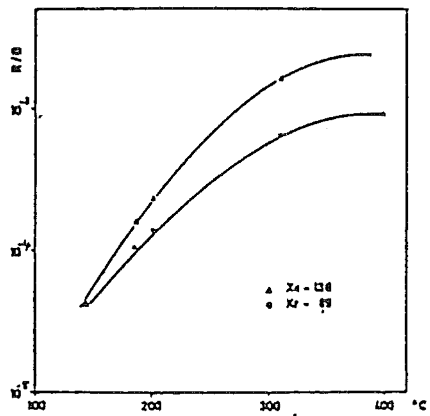


Fig.10 Effect of sodium temperature on the release fraction

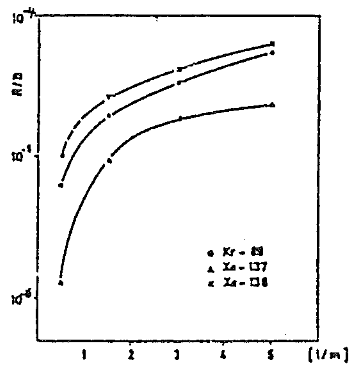


Fig.11 Effect of sodium flow rate on the release fraction. Sodium temperature 260°C.

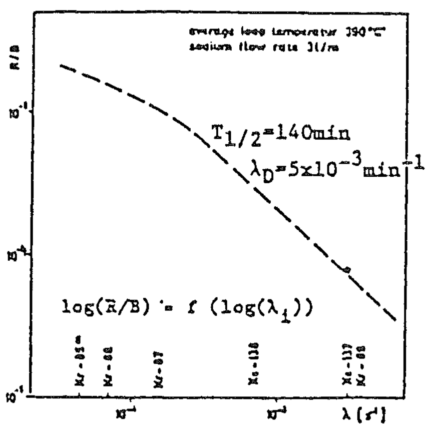


Fig.12 Effect of the decay constants on the release fraction

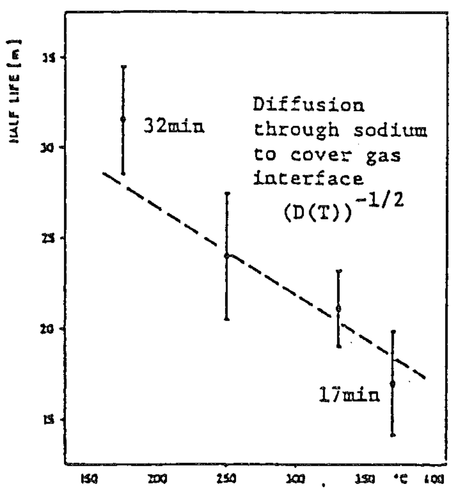


Fig.13 Effect of sodium temperature on the degassing halflife

THE DETECTION AND LOCATION OF FUEL FAILURES IN SODIUM COOLED FAST REACTORS

REVIEW OF UK WORK

COMPILED BY

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UK presentation to the IWGFR Specialist Meeting on 'Detection and Localisation of Failed Fuel Elements in LMFBRS' Karlsruhe 11-14 May 1981

Part 3

IN-PILE EXPERIMENTS IN THE UK

IN-PILE EXPERIMENTS IN THE UK

Experiments with artificiallyy defected pins carried out in the CEA Scarabee loop are described together with subsequent investigations in the Universities Reactor, Risely, to study the influence of a number of parameters on signal size.

Measurements of the signals from artifically defected new fuel pins were made in 1973-74 under contract with CEA in the CEA in-pile sodium loop Scarabee (1, 2) shown in Figure 1. These experiments were originally planned to measure the emission from fuel pins containing UO_2 hollow fuel pellets with slit defects in the fuel section of the can parallel to the axis. In addition experiments to determine means of encouraging emission for location purposes were planned. However at a fuel rating of 160 w/g the delayed neutron release from each of the four pins with 0.25 x 6 mm long slits indicated the equivalent of a recoil area 1000-4000 times that of the superficial area of the slit, and was in the range 15-60 cm^2 effective recoil area with a transit time of 46 secs. With a shorter transit time the recoil area was smaller showing that the emission contained an excess of longer lived precursors. In fact measurements with one pin in which fission product data were also analysed were consistent with a release model which assumes that the area in the vicinity of the defect releases recoils directly to the sodium while recoils from the rest of the fuel are released with a delay of 40 seconds. Although water model tests described below have shown that some sweeping of the fuel can occur, this is only in the immediate vicinity of the defect and thus the additional signal is expected to arise from diffusion. The experiments in Scarabee were made over only a few hours and thus the burn-up of the fuel was negligible although some sodium attack caused break-up of fuel into grains and some chemical interaction occurred.

So that a better understanding of the experiments described above could be obtained some simple experiments were made in a water loop in the Universities Reactor, Risley, and the effect of slit size, slit shape and fuel/can gap examined. The rig is shown in Figure 2. The experiments were performed with a fuel rating of 10^{-3} w/g, so that diffusion was negligible. The results confirmed that the signal is enhanced over that expected from slit area alone. A 6mm long slit, area 1.5 mm^2, gave an effective recoil area of 6 cm^2 which compares with the 3-7 cm^2 obtained in Scarabee at 25 w/g. However, these water experiments show that the whole signal came from the rapidly swept area around the defect and from about 1 cm above and below the defect as shown in Figure 3. This suggests that although sweeping of the fuel by coolant is a major mechanism for fission product transfer the details of release from the fuel may be different at high power to that at low power. This could result from enhanced diffusion of fission products or different hydraulic conditions when the fuel is at high temperature. However, the main observed difference is that the signal is increased at high power and is relatively independent of defect area in the range tested.

Experiments to attempt to check the effect of closing the fuel-clad gap were made in the Universities Reactor by swaging cans on to fuel. These showed that whereas for slits 0.25mm wide with lengths up to 6mm the signal was reduced by a factor of 10 compared with those of clearance fit, for longer slits the ratio decreased so that at 37 mm no reduction occurred as shown in Figure 4.

Further experiments were made with circular defects. With a 3mm dia hole the signal was similar to a slit of the same length while the signal with smaller holes fell approximately linearly to about 1/40 of the signal for a 0.25mm hole. On the other hand two 0.25mm holes separated axially by 37mm gave the same signal as a 37mm x 0.25mm slit, thus confirming that the length of the defect and not the area is the more important parameter.

More recently experiments have been started to try to get some indication of how fission product release depends on the state of the fuel, and in particular if vibro fuel behaves differently to pelleted fuel. Although the first results show considerable scatter it is evident that DN signals from new low rated vibro fuel are about 5-10 times those from whole pellets. Signals from chipped and broken fuel pellets give intermediate signals.

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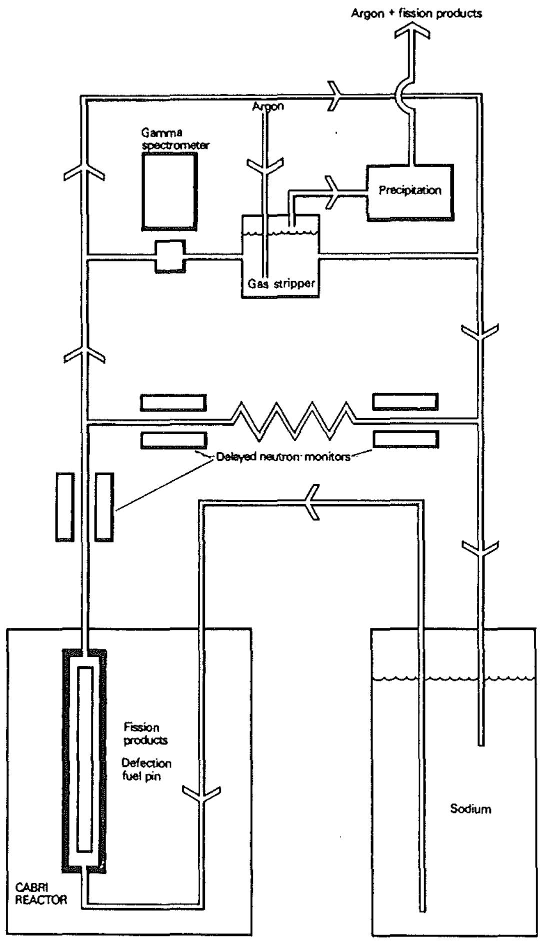
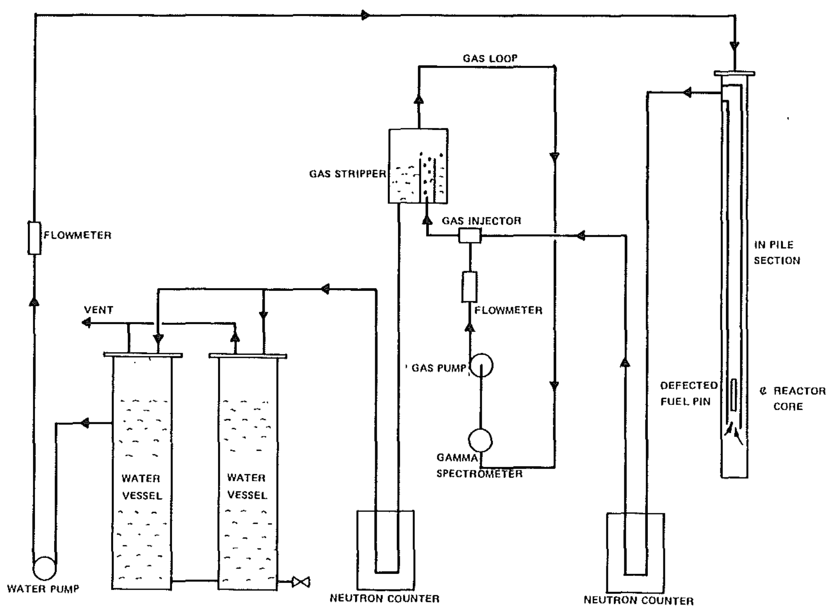
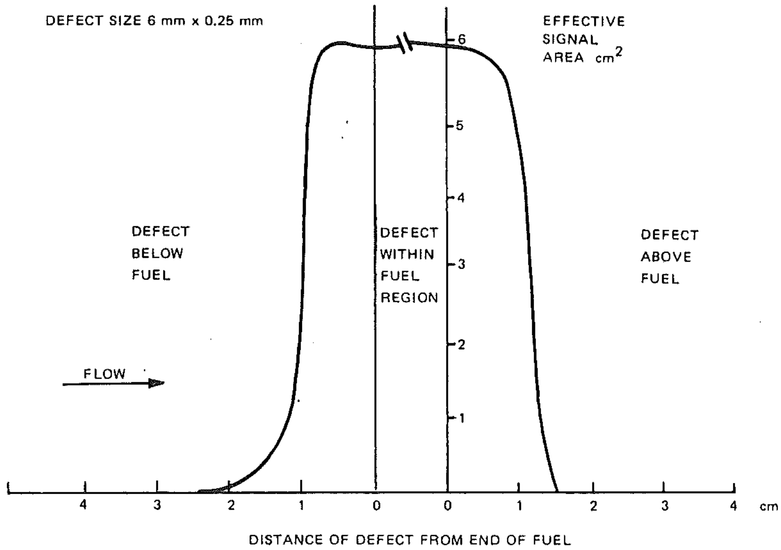


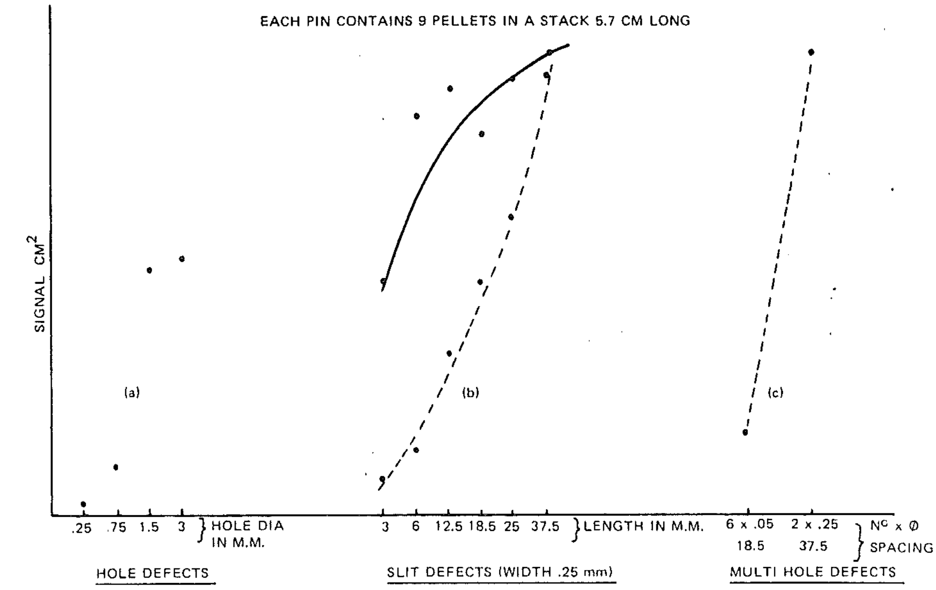
THE CEA SODIUM LOOP SCARABEE (SIMPLIFIED SCHEMATIC) FIG. 1.



BPD FISSION PRODUCT EMISSION RIG UNIVERSITIES REACTOR FIG 2



DELAYED NEUTRON SIGNAL FROM A SLIT AS FUNCTION OF DEFECT POSITION FIG 3



AFFECT ON SIGNAL OF TYPE & SIZE OF DEFECTS FIG 4

U.S. Experience of Delayed-Neutron Monitoring

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ABSTRACT*

A history is given of the delayed-neutron signals observed at EBR-II during sixteen years of operation. Special tests to calibrate the monitors are described and the character of signals from some recent special breached-pin tests are discussed.

INTRODUCTION

The detection of delayed neutrons (DN's) released during the decay of fission products carried by the primary sodium of an LMFBR has long been recognized as a means to detect fuel-pin failures. Such a system, known as the fuel-element rupture detector, or FERD, was originally installed on EBR-II and has been in use since 1964. This paper describes the considerable experience gained with FERD since that time. Two parallel papers (1,2) describe the methods also developed at EBR-II for failed-pin identification and for DN signal analysis.

DESCRIPTION OF FERD SYSTEM

The FERD system interrogates primary sodium from the outlet of the intermediate heat exchanger (IHX). Approximately one eightieth of the IHX discharge is collected by a rectangular scoop and funneled into a 5-cm. diameter pipe of the loop to FERD. An electromagnetic pump draws sodium through this pipe to the detectors at 6.3 L/s during normal operation, and then discharges it back to the primary tank. The transit time for sodium from a failure in-core to the FERD detectors is 18-20 s. In its original configuration, the FERD loop passed through a 91-cm. cube of graphite surrounded by lead shielding. Three independent counting channels were provided and several different types of detectors were used. The axis of the detectors was perpendicular to the sodium, and they were cooled with forced air. The FERD in this initial configuration was calibrated by Smith and others (3) in 1968-9, using two exposed driver-fuel pins as a source of DN's.

After that calibration, the detectors were replaced with pairs of Reuter-Stokes BF_3 proportional counters for each of the three channels. An effort was also made to reduce gamma flux from sodium and to enhance moderation by placing 5 cm. of lead and 2.5 cm. of polyethylene around the sodium pipe. This FERD configuration was recalibrated in 1977 by Strain and others (4) with tubes of a dilute uranium-nickel alloy as the source of DN's. This calibration showed that FERD signals were slightly unstable during the first day or two after reactor startup, but that reproducible mean count rates were obtained thereafter.

FERD underwent a substantial modification in March 1978. The graphite moderator was replaced with Benelex Type 401 because of its relatively high hydrogen content and the 5-cm. diameter BF_3 detectors were replaced by longer, 2.5-cm. diameter detectors laid along the axis of the pipe. Two channels were used for operations and each contained seven, ganged BF_3 detectors; a third contained six 2.5-cm. diameter boron-lined detectors; and the fourth, six BF_3 detectors. This FERD configuration was calibrated in April 1978 with the same alloy tubes that had been used a year previously.

HISTORY OF SIGNALS BEFORE 1976

The first DN signal from a fuel-pin failure occurred on December 30, 1973. This signal, accompanied by some fission gas, rose rapidly from a background of ~60 cps to near 120 cps, a value which caused the reactor to trip. The signal remained high for 60 s, then rapidly declined (Fig. 1a). The source was later identified as a sodium-bonded carbide pin and the signal was assumed due to release of DN-rich bond sodium to the primary sodium before, during and after reactor shutdown.

Approximately a year later the reactor scrammed a second time as the result of a DN signal (Fig. 1b). The

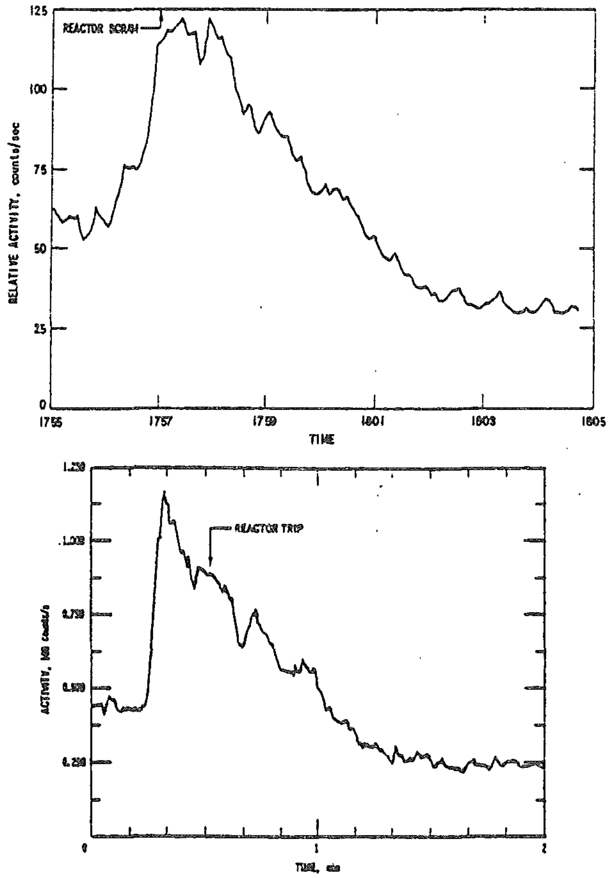

Fig. 1. Transient DN Signals from Fuel Failures
a: During release of bond sodium
b: During rapid release of fission gas

*Work supported by the U.S. Department of Energy.

source of this signal was identified as a mixed-oxide fuel pin which had failed in the fuel-column region, a fact later confirmed by leak checking the pin while it was internally pressurized under alcohol in a hot cell. The lack of a plateau in the DN signal and the very small size of defect found on the pin suggested that the signal arose from DN precursors which had escaped in the vapor phase during rapid release of fission gas. This DN signal was the first ever to be detected from a mixed-oxide fuel pin in EBR-II.

Later in the same reactor run, EBR-II was again scrammed as the result of a DN signal accompanying a rapid release of fission gas. The shape of the FERD response was different this time in that the count rate reached a plateau and remained there for a short time after shutdown (Fig. 2a). Subsequent examination in the hot cell revealed that the signal came from a mixed-oxide fuel pin which had developed a small split in its cladding due to wire-wrap wear (5). The initial flat portion of the signal probably represented release of DN precursors from fuel exposed to the coolant, and was the first example of such a phenomenon. The signal magnitude was much greater than could be explained by recoil release of DN's from the very small area of fuel exposed by the cladding split.

During March 1975, a fourth DN signal was recorded by FERD, as shown in Fig. 2b. The reactor was operated

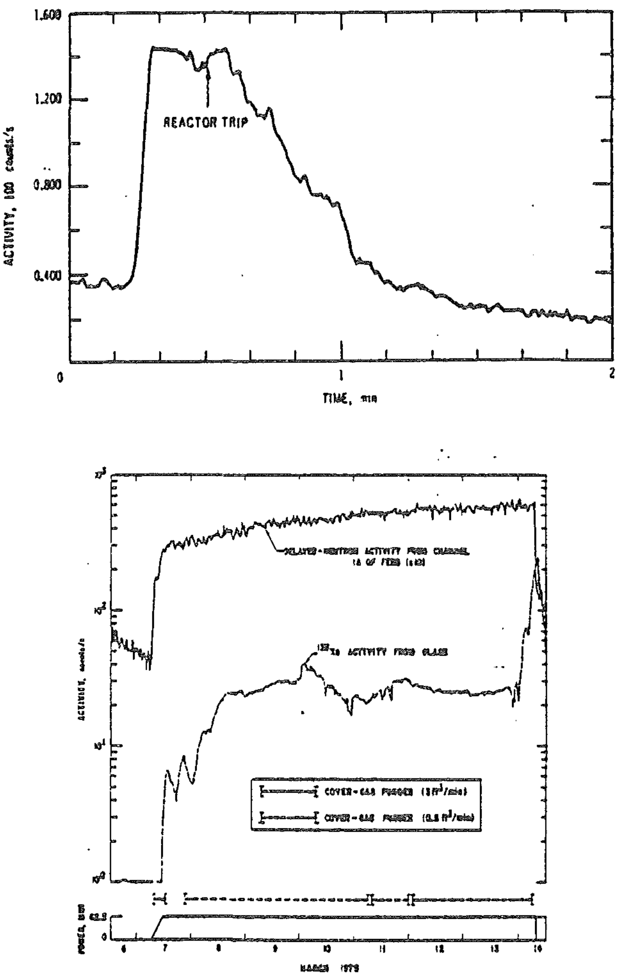

Fig. 2. DN Signals from Exposed-Fuel Failures
a: From a mixed-oxide fuel pin
b: From several metallic driver pins

for about seven days with a DN signal significantly above background levels, but less then the scram limit. The subassembly with the failure was identified finally as an instrumented subassembly containing untagged metallic driver-fuel pins (6). Postirradiation examination showed that in fact six pins had breached, three of them exhibiting quite large failures. The DN signals were consistent with a recoil release only from the exposed metal fuel. However, the possibility that the postirradiation handling of the subassembly had enlarged the breach sites could not be ruled out.

A fifth DN signal occurred in July 1975, about five days after the start of a reactor run. This DN signal, which was similar to the one in Fig. 1b, was accompanied by a particularly rapid release of short-lived fission gas and a xenon tag. The failure was easily identified as being on a mixed-oxide fuel pin with a bottom plenum which had just been loaded and which had operated at a very high cladding temperature. In this instance, high fuel temperatures, a bottom plenum, and low burnup of fuel had allowed plenum gas stored below the fuel column to be swept unimpeded over the fuel surface to the failure, bringing with it volatile DN precursors.

FERD SIGNALS SINCE 1977

In May 1976 Fryer developed the basis for and performed a FERD-flow reduction (FFR) test to determine the transit time for DN precursors from the core to the FERD detectors (7). Concurrently, the case was successfully made to remove FERD from the automatic shutdown system after analyses (8) had shown: (a) reactivity measurements were a prompter method for dealing with a core perturbation which could have safety implications; and (b) propagative fuel-pin failures would likely take place at rates slow enough that manual shutdown would effectively limit circuit contamination. FERD was taken out of the scram system for run 87 (January 1977). This act freed FERD from its former dedicated function, and calibration and testing of the system began in earnest.

Operation with a Recoil Source

The irradiation of the uranium-nickel source in EBR-II during run 87 provided new insight into the behavior of FERD. The system was shown to respond in a sensitive and linear fashion to the strength of a recoil source, and in way entirely consistent with that of the independent cover-gas monitor GLASS. Figure 3a shows the DN signal count rate versus reactor power for two core locations. The short-term fluctuations in the count rate of ±10% which were observed at all steady power levels (Fig. 3b) were almost certainly due to a variable mixing of sodium in the upper plenum of the reactor, and cannot be eradicated. However, averaging the count rate recorded by the data acquisition system (DAS) showed stable mean count rates over very long periods of time.

The FFR technique was proved as a method to determine the average "age" of DN's emitted from a source, while transit times also appeared to differ slightly for the two core locations. Similarly, movement of the control rods adjacent to the source was also proved as a possible means of source diagnosis. For the exercise reactor power was reduced slightly and the pair of control rods next to the source raised while the opposite pair of control rods was lowered (to mainain reactor power). This maneuver tilted flux in the vicinity of the source. The procedure was reversed to tilt flux the opposite way. Whether this technique could be used in other reactors, or for non-recoil sources could not of course be determined.

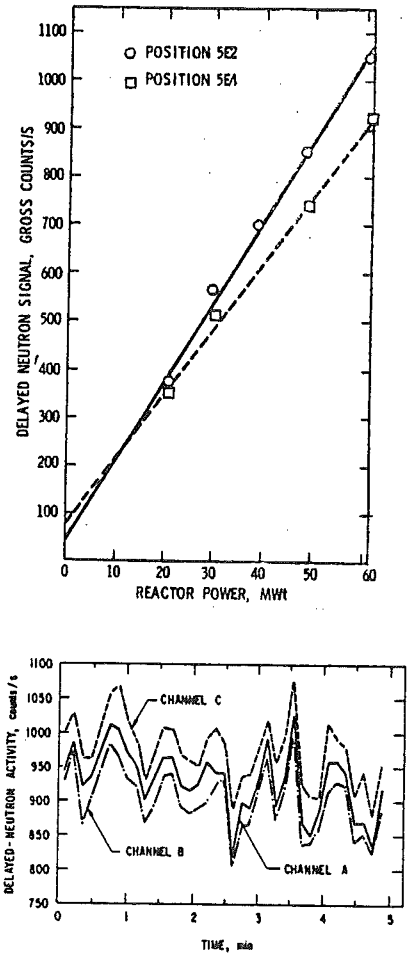

Fig. 3. FERD Response to a Recoil Source in EBR-II
a: Signal linearity with reactor power
b: Fluctuations due to sodium mixing

<u>Operation with Predefected Pins</u>

Four predefected fuel pins were irradiated in 1977: two pins contained unirradiated UO_2 and two pins contained mixed-oxide fuel at medium burnup. The first UO_2 pin had a 0.75-mm. diameter hole in the cladding in the fuel-column region but gave no discernible DN signals in-core. The second UO_2 pin had an axial cladding slit which measured 1 mm. by 20 mm. The preirradiated mixed-oxide pins were machined in the hot cell to give slits in the cladding of about 0.8 mm. wide and 22 and 58 mm. long respectively. The DN signals from the second pin of each type will be described here; further details of all the pins may be found in reference 9.

During rise to power with the UO_2 pin the DN signal count rate increased in an essentially linear fashion to a full-power value of 55-60 cps above background, as shown in Fig. 4a. The count rate stayed constant through ten days of irradiation. The pin was then relocated to a higher-power position in EBR-II. The DN signal count rate on the second ascent to power is also shown. Very little change in signal occurred over the next 33 days irradiation.

The DN signal count rate at full power was about 25 times the value expected due to recoil release from the geometrical area of defect. The cause of enhancement is not precisely known. However, fission-gas release was such as to suggest that the UO_2 was sodium-logged at power. A FFR test also indicated an "aging" of the DN precursors of 12-18 s. Thus, enhancement and aging in this instance may have been due to release from a large surface area of fuel via a sodium bond.

The lower curve in Fig. 4b shows that the DN signal behavior of the second mixed-oxide pin during initial ascent to power was different and almost exponential with increase in power. Just before full power was achieved the reactor was scrammed for an unrelated reason; the count rate was then $\sim$425 cps. On the next rise to power (upper curve) a similar non-linear rise in count rate occurred, but at a higher level than the first time. The full-power DN count rate was $\sim$575 cps.

During the next 2-1/2 hours operation the average count rate increased to $\sim$650 cps. However, the actual signals showed considerable variation, and the shutdown limit of 800 cps on FERD was soon exceeded and the reactor tripped. The fuel pin had extended its defect considerably, increasing its geometrical area from 0.35 to 0.71 cm^2. The final count rate was equivalent to about 100 times the recoil release of DN's from the geometrical area of defect. Increase in signal between the first and second rise to power could be well explained by defect extension. The cause of enhancement of

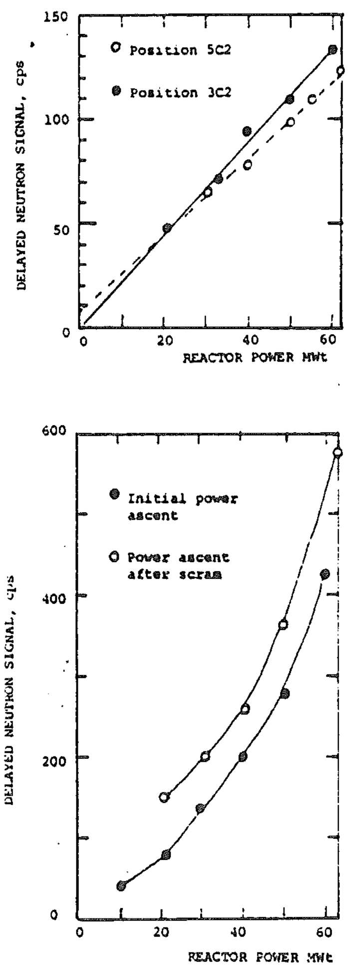

Fig. 4. DN Signals from Predefected Fuel Pins
a: UO_2 pin in two core locations
b: $(UPu)O_2$ pin during two power ascents

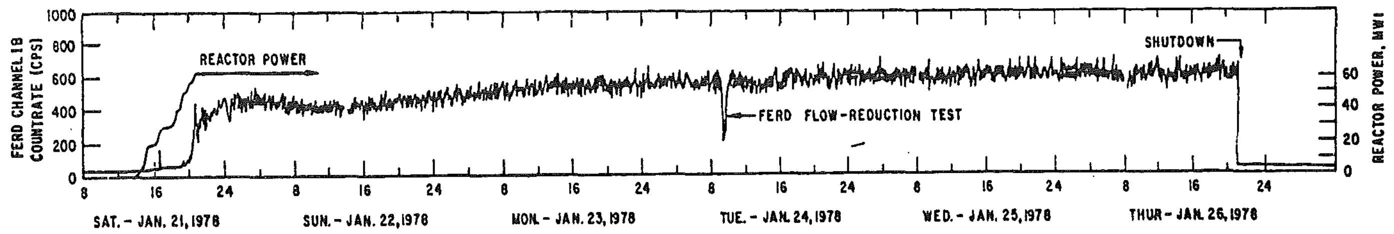

Fig. 5. DN Signal Count Rate during First Natural Breached-Pin Test in EBR-II

the DN signal is again unknown but could have been due to a number of factors. For example, both the surface temperature of the fuel and its burnup were higher for this pin than for the UO_2 pin, which may have enhanced the release of bromine DN precursors to the primary sodium (10).

Typical Behavior of Natural Failures

The first continued irradiation of a natural breach was carried out in 1978. The pin, which contained mixed oxide fuel at 11 at.% burnup, had developed a small leak of fission gas (and tag) in mid 1977. Its subassembly was kept under sodium in the EBR-II storage basket while procedures and approval for performing the test were obtained. Its subsequent DN signal behavior, which is believed typical of natural failures, will be described here; other details of the test appear in reference 9.

Figure 5 shows the DN signal count rate from the breached pin during the course of its 5-day irradiation in EBR-II. Significant DN activity was first observed at 40 and 50 MWt, peak count rates of 500-600 cps being achieved. At full power (62.5 MWt) averaged count rates were at 500 cps, and declined slowly for the next 12 hours of operation. The signals slowly increased with time over the next 4-1/2 days. The prescribed DN shutdown limit was then exceeded and the reactor tripped. A FFR test was performed towards the end during a temporary plateau in the count rate increase. It showed a significant aging of the DN precursors, of about 8 s.

Postirradiation examination of the fuel pin showed a 4-cm. long split of the cladding toward the top of the fuel-column region; the maximum width of split was 1 mm., corresponding to a geometrical area of defect of 0.21 cm^2. As before, the magnitude of the DN signal was several hundred times that expected from recoil release from the exposed fuel surface (even allowing for a much roughened and cracked fuel surface at the defect site). Once again, therefore, fuel surface temperature and burnup, as well as perhaps fuel density, appear to be important factors in release of DN's.

SUMMARY AND CONCLUSIONS

During endurance test of fuel pins carried out to 1976 there were five times when genuine DN signals from fuel-pin failures were recorded by FERD. During that period FERD was included in the reactor's automatic shutdown system. It was removed from the scram circuits in 1977, enabling breached-pin tests to start. As a result interest in DN monitoring was revived and FERD was recalibrated; the system was upgraded a year later. Freedom to operate under these conditions has allowed greater EBR-II utilization and has also led to a better understanding of both the emission of DN precursors and the "systems" effects on their transport and detection. For example, experience has shown that transient DN signals may accompany several types of initial failure but that sustained DN signals only emanate from exposed fuel.

Artificially increasing the transit time for DN's to the detectors can supply additional information on the "age" of the DN precursors. Similarly, experience has shown that exposed oxide fuel will give much bigger signals than would be expected by recoil release from the surface area of defects. This fact augurs well for detection and monitoring breached fuel pins in a large LMFBR. However, the relationship between the magnitude of DN signal and the area and condition of exposed fuel is not now known. This relationship must be studied in depth and be understood before significant breached-pin operation may be contemplated in future reactors. A parallel paper (11) discusses what work needs to be carried out in this and related areas.

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Discussion

D.B. Sangodkar, RRCK:

In all the cases described in your presentation a DND enhancement is observed. I remember to have read a report that when fuel was irradiated in the form of tiny granules in a perforated container in EBR-II there was no enhancement. How do you explain this difference?

J.D.B. Lambert, ANL:
Difference in fuel form, absence of temperature gradient, and low fuel temperature

S. Jacobi, KfK:

One of the table indicates that there was no DND signal till 1973: Does this mean there was no DND failure or the DND system was not sensitive enough during this time (sensitivity was improved by some steps)?

J.D.B. Lambert, ANL:
No DND failures before 1973

W. Glauner, KfK:

Did I understand, that you don´t know exactly the release mechanism? You come to this point later under "Models, Sess. IV"?

J.D.B. Lambert, ANL:
Correct.

S. Jacobi, KfK:

Could you give an explanation for the negative peak in the DND signal plot from the data aquisition mashine, because the peak is out of normal statistics?

Mr. Lambert, ANL:
A spurious data point, I think.

M. Relic, IA:

a) Did the composition of the groups of DN-precursors show that there is a minor quantity of short lived fission products then predicted in the literature?

J.D.B. Lambert, ANL:
Yes, the DN signals appeared to be "aged".

P. Michaille, CEA:

Is the flow rate constant at EBR 2 when power is changed?
Did you examine the age of the DN bursts?

J.D.B. Lambert, ANL:
Yes, yes.

THE RESULTS OF TESTING AND EXPERIENCE IN THE USE OF LEAK DETECTION METHODS IN FAST REACTORS

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V.I.Polyakov, D.I.Starozhukov, E.K.Yakshin

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Part 4

RESEARCH AND DEVELOPMENT WORK ON INSTRUMENTATION AND DETECTION METHODS

When applying delayed neutron detection method in large power plants the signal/background ratio becomes insufficient for detecting the occurence of single fuel element failures. The following disadvantages of the delayed neutron detection system should be noted:

- short transportation time is not easily assured when designing this system,
- delayed neutron signal does not provide information about fission product concentrations in the coolant.

In few cases a pronounced Cs activity buildup in the circuit and radiation situation deterioration were observed while the delayed neutron signal remained constant.

Fission gas detection system provides information about gaseous leak fuel elements availibility and permits the evaluation of failed fuel element number. However , this information does not allow to assess a primary circuit contamination.

At present adequate experimental data are not available on the release fractions of such fission products as ^{95}Nb, ^{131}I, ^{134}Cs, ^{137}Cs, ^{140}Ba as a function of fuel type, defect size, temperature, reactor operation time and of fission gas and delayed neutron monitor signals. This prevents the complete knowledge of defects occurence and development and activity buildup in the circuit from being obtained based only on the above system signals. Therefore the efforts of investigators throughout the world were directed to ensure higher detection system sensitivity to definite fission products.

The developed methods may be subdevided into two directions: those, based on physical and chemical extraction and concentration of fission products from coolant and those based on the improvement of input properties and selectivity of detection devices. As a rule, many of them are too complicated and inadequately prompt in service.

A theoretical analysis has been carried out of potentialities of the latter approach- through adequate instrumentation - to decrease the lower detection threshold of gamma--spectrometers. The following tasks have been posed:

. the choice of semiconductor detectors for a particular measurement geometry in a single-crystal gamma-spectrometer;

. the choice of a spectrometer capable of operating at maximum counting rates with no reduction both in energy resolution and in the efficiency.

. improvement of instrumentation to provide possibility of long-term measurements;

. to reduce the interference background and Compton distribution contribution under the studied peak of total absorption by anticompton schielding; the use of summing Compton spectrometers.

The studies have revealed that thin composite planar detectors in the summing spectrometer scheme are more sensitive compared to a single crystal detector. Fig.4 shows the summing Compton spectrometer block-scheme. The spectrometer was tested in the BOR-60 primary circuit by-pass sodium loop. A Ge(Li) duod consisting of two planar detectors with a thin berillium layer between them. The total sensitive volume was 12cm^3. In the apparatus gamma-spectrum for the measurement time of 150min. the total absorption peaks were obtained for

^{131}I (364 and 637 KeV) and ^{137}Cs (662 KeV) on the ^{24}Na back - ground. The useful-to-interference activity ratio was about 10^{-4}.

The detector design and size optimization, improvement of spectrometric path input properties at a less counting loss, the choice of energy thresholds make it possible to reduce the lower detection threshold.

To detect ^{137}Cs on ^{24}Na background, for example, within 20 %, with a conventional single-crystal Ge(Li) detector a longer holding period (about 70-80 hrs) is required at mass ^{137}Cs activity in the sodium coolant up to 10^{8} Bc/kg.

Further the detection of fission products in cover gas is considered.

The data gained during BR-10, BOR-60 power plants operation reveal (as is mentioned above) that an oxide fuel ele - ment failure is moreoften, followed by the same pattern of fission gas activity burst. Each time about the same activity amount is released to the primary circuit. These events identification is easy due to high signal-to-background ratios.

In most cases fission gas release through a defect is fast because the fission gas pressure inside the cladding by the time of its failure is about 2-5 MPa while the external coolant pressure being an order of magnitude lower. In the moment a failure occurs a gas fraction (large bubbles) comes fast from the circulating coolant into the cover gas. The rest fission gas breaks into small bubbles and release from the coolant after an average delay-time of 8 - 10 hrs.

Fission gas release through a defect doesn't remain constant. The sodium coolant fills fuel/cladding gap in the defect vicinity both on failure induced pressure drop and after a power decrease as well as during long-term reactor outage. Radioactive fission gases release from such a fuel element is very weak. Here the measured fission gas activities levels are due to the release of easily dissolved iodine and bromine precursor-nuclides through the sodium plugging. In some reactor operation days fission gas releases through the defect in the form of bubbles as a result of pressure increase inside a fuel element. The delay time of gases inside failed element during the bubble mode release is determined from the plenum volume and the rate of fission gas release through the defect.

These factors were taken as a base of counting techniques for the determination of a failed fuel elements number on the basis of the dynamics pattern of fission gas release into the cover gas.

Instrumentation allowed to carry out rather frequent (with 0.5 h intervals) measurements of ^{133}Xe, ^{135}Xe, ^{88}Kr radionuclides. Their release fractions in cover gas are listed in Table 2. Gamma-ray detection efficiency in 80-250 KeV energy range considering collimation was $(2\text{-}4)\cdot 10^{-3}$ pulses per quantum from 1 cm^3 of gas. At radionuclide concentrations higher than $5\cdot 10^5$ Bc/l the measurement statistical error was no more than 5 % for 300 s measurement time.

Failed fuel element detection was carried out by one of the following indications :

- the increase in ^{133}Xe activity per day operation is more than by 50 %,
- the increase in ^{88}Kr and ^{135}Xe activities per a day operation is more than twice,
- the increase in beta-counting from the beta detector per a day operation is more than twice.

Table II

^{133}Xe, ^{135}Xe and ^{88}Kr Release from BOR-60 Reactor Failed Fuel Elements into the Cover Gas

Detection mode		radionuclide release fraction, %		
		^{133}Xe	^{135}Xe	^{88}Kr
in the moment of fuel element failure	during the reactor shut - down	5.3	1.3	0.5
	at steady reactor power	5.1	0.9	0.2
bubble mode release	at steady reactor power	4.4	0.25	0.02
Detection error, %		± 20	± 50	+100 - 70

The best precision in failed elements number determination may be achieved for ^{133}Xe. However, a background for further failures detection grows with this radionuclide buildup. The release fractions of ^{135}Xe and ^{88}Kr gases in the cover gas are low due to their decay inside the fuel element and in the coolant. The signal-to-background ratios for the above radionuclides are fairly higher. Thus, at the relative measurement error of 10 % it is difficult to detect the next new failure against the ^{133}Xe background from 10 failed elements at steady gas release process. These limits for ^{135}Xe and ^{88}Kr are , respectively, 40 and 200 failed elements in the reactor core.

The assessment shows that one fuel element failure may be with any assurance detected against the background activity caused by BOR-60 core contaminated with 1 g of ^{235}U. Continuous cleanup of cover gas from radioactive fission gas favours better counting efficiency based on ^{133}Xe data. With the decontamination factor of 100 the efficiency increases by a factor of 20.

Trustworthiness of the method used for counting the failed fuel element number was verified by fission product measurements in sodium, fuel subassembly examination during the reactor shutdown, selective material investigations of subassemblies. As was seen from material investigations during one of the reactor campaigns the mean number of failed fuel elements per subassembly was 2-3, i.e. (5-8) % of all the elements in a subassembly. The mean fuel burnup was 7.3 % h.a. In such a situation, in the average 0.1-0.2 % of failed fuel elements should be present in the core by the end of the reactor campaign. This was in agreement with the results of fission product detection in the reactor under operation.

The ultimate goal of failed fuel detection in the reactor under operation consists in more precise defining the level of the primary circuit fission products and fuel contamination. Fission gas measurements provide adequate information about cover gas contamination.

Recently a new approach to main fission product detection in sodium against the ^{24}Na background has been found. The approach is suggested to be promising and its potentialities are being studied at present. It is hoped that this method will allow to carry out continuous detection of caesium and iodine radionuclides in sodium and thereby to restrict primary circuit contamination in excess of permissible limits at both slow and rapid fission products activity increase.

In the last few years to localize failed BOR-60 fuel subassemblies the blow-gas is analyzed not only for ^{133}Xe, but also for those fission products deposited on particulate filter. This allowed to better the trustworthiness of failed fuel subassemblies detection. The activity measurements are carried out using Ge(Li) pulse analyzer detector. The decision about the fuel discharge results from the mathematic analysis of ^{133}Xe, ^{131}I, ^{137}Cs, ^{95}Nb, ^{132}Te nuclides release data. The fuel subassembly features (burnup, irradiation objective, fuel composition etc..) are considered.

In many cases ^{137}Cs and ^{133}Xe were detected simultaneously in blow-gas while ^{131}I and ^{132}Te being not always detected. As an example, one fuel subassembly comprised two failed fuel pins and was identified due to ^{133}Xe and ^{137}Cs detection. Another subassembly comprised one failed fuel pin with minor failure of cladding and was detected due to all above nuclide activity inventory. One more subassembly comprised four fuel pins with pin defects (with no cracks) but only ^{137}Cs release was detected during the check. Fig. 5 shows, as an example, a diagram of such discrepancy check.

The reasons of the results discrepancy are unknown. They may be due to the coolant contamination level, the extent of fuel cladding failure, the defect plugging with sodium etc.. Studying the above reasons is required to better the assurance in adequate removal of failed fuel elements from the reactor.

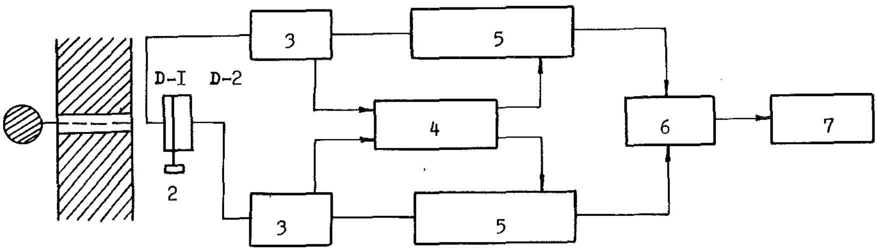

Fig. 4 Ge(Li) gamma-spectrometer block-scheme :
1 - sodium coolant tubing, 2 - Ge(Li) - duode,
3 - fast charge-sensitive preamplifier, 4 - linear gate, 5 - linear amplifier, 6 - summator, 7 - pulse-height analyzer

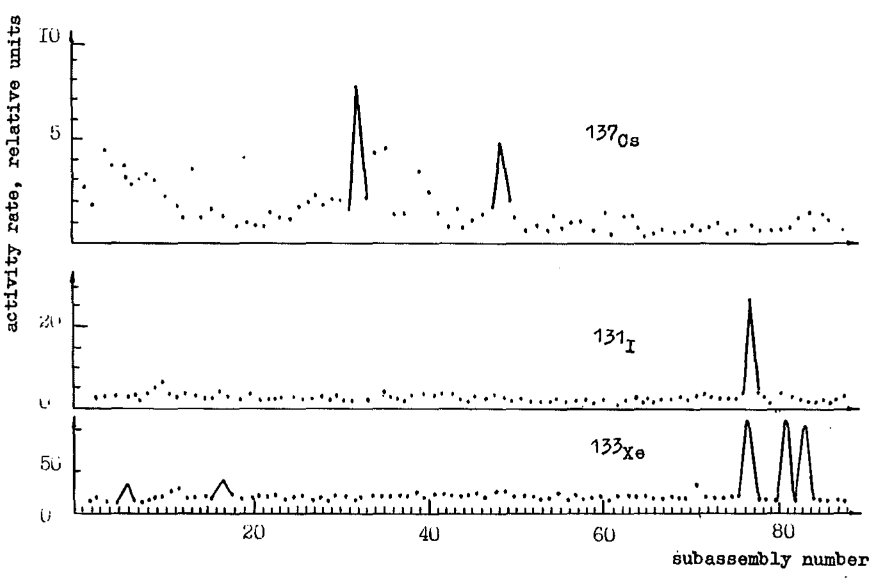

Fig. 5 Subassembly detection chart

Discussion

C. Berlin, CEA:

Could you explain us, how you do an on-line measurement of caesium and iodine activities in primary sodium?

E.K. Yakshin, SRIAR:
The method and facility are now patented. We hope to explain our method in some time in future.

K.Ch. Stade, KBG:

Do you measure Ne-23 in the primary cover gas? How much is the concentration?

E.K. Yakshin, SRIAR:
We have measured Ne-23 concentration once several years ago. The value is constant at steady power.

M. Relic, IA:

Have you measured Cs-134 in your cold trap (in the gas circuit)?

E.K. Yakshin, SRIAR:
In the cold trap of fission gas detection system Cs-134 was not measured. We have not seen a sense of such measurement.

SOME FEATURES OF DATA ACQUISITION AND PROCESSING USING BOR-60 FAILED FUEL DETECTION SYSTEM

S.M. Glushak, N.V. Krasnoyarov, E.V. Krainov,
D.I. Starozhukov, V.I. Shipilov

A b s t r a c t

Based on the multichannel analyzer/mini-computer M-6000 and the multichannel analyzer IN-96/computer Multi-6 a computerized measurement system has been designed at the BOR-60 reactor.

Gamma-activity from 9 nuclides in sodium flow and 9 nuclides in cover gas are being measured at 15 minute intervals by Ge(Li) detectors under reactor operation. The data on the nuclide behaviour with time are systematized and described by a mathematical model with the mini-computer mentioned above.

Information about the fission product behaviour in sodium and cover gas, the data on the delayed neutrons, power, temperature, flowrate and pressure changes within the circuits over the reactor campaign is accumulated at the Multi-6 computer. A software for a convenient access to the data bank has been developed.

After the reactor shut-down the computerized measurement system hardware and software makes the localization of the defected fuel pin subassembly possible.

INTRODUCTION

The BOR-60 research reactor fuel pins are tested to high burnups,therefore,is a great probability for their clad failure.

The reactor operates with the limited and controlled number of the defected fuel pins. Fission products and some portion of fuel release from the defects into the coolant and cover gas resulting in the primary circuit contamination.

To limit the consequences of such release to a permissible time, to estimate the size of all the defects to a high accuracy and to assess the extent of contamination. The BOR-60 reactor is provided with special equipment and devices for detection and characterization of the defects as well as for estimation of the primary circuit contamination level.

To measure the fission product activity various methods or the semiconductor gamma-spectrometry are widely used. A high activity of ^{24}Na doesn't permit the ordinary gamma-spectrometry measurements of relatively low activities from fission products just in sodium flow under the reactor operation. To solve this problem a series of tests of the Ge(Li) anticompton- and Ge(Li) summing compton spectrometers [1,2] was carried out. The composite Ge(Li) detector of a special configuration was expected to solve this problem. But the difficulties in the production of such detectors force to search another ways for solving the problem.

The fission product measurements carried out at the operating BOR-60 reactor by a standard Ge(Li) gamma-spectrometer are provided. In this connection a decision has been made upon the development of hardware and software complex at BOR-60 for acquisition and processing data on fission product gamma - activity behaviour in sodium and cover gas.

CONTROL SYSTEM DESCRIPTION

The computerized measurement system which block-diagram is presented in Fig.1, is based on the multichannel analyzer with a mini-computer M-6000 [3] and the programmed multi - channel analyzer IN-96 with a Multi-6 computer (Intertechnique production, France). The data exchange between M-6000 and Multi-6 computers is realized by the perforated tape in the off-line mode.

The M-6000 computer base multichannel analyzer operates around-the-clock during the whole reactor campaign.

Two Ge(Li) detector gamma-spectrometers record fission product gamma-radiation in sodium and cover gas through the lead collimator by a set of 12 successive measurements. The time for setting up one gamma-spectrum in the M-6000 computer multichannel analyzer is about 3 to 5 minutes. The gamma-peaks corresponding to 9 nuclides both in sodium and cover gas are processed for each spectrum. The measurements are performed at 15 minutes intervals.

The algorithm for processing the current data on each nuclide in a series of 12 measurements provides the reduction of unimportant data (for example, on a stationary process) and the detail description of the transient processes. To this end the data obtained for each nuclide in a series of 12 measurements are described by a mathematical model using the Chebyshev's polinomial regression [4] . As a result of statistical processing the valuable polynomial coefficients pointing to the essential changes in the controlled parameter are transferred to the printer for subsequent input into the Multi-6 computer. By this means an efficient elimination of the redundant information is provided. From the memorized data one can recover the controlled nuclide behaviour with time in any time interval to a sufficient accuracy.

A comprehensive information about the controlled nuclide behaviour in sodium and cover gas, about the reactor power transients and the delayed neutron number variations as well as the data on temperature, pressure and flowrate within the circuit are accumulated in the Multi-6 computer.

A special software has been developed which enables :

• to maintain the data bank in working state, that is

to perform the data input, to search and correct the errors ;
- to classify the data according to various criteria;
- to select the data according to the required criteria;
- to send the data to various peripherals (display, graphecon, printer) in a convenient form (tables,diagrams);
- to prepare the data for the following analysis, i.e. to form a restricted data bank for the accelerated access to the required information and for its vivid presentation.

The input data analysis permits to detect new defects, to follow their propagation and to control the fission product contamination level.

After the reactor shut-down a search for the defected fuel pin subassemblies is performed. Based on the recorded data on the volatile fission product behaviour in the exhausting gas a conclusion is drawn about the occurrence of the defected fuel pins within the subassembly. The analysis of probes is carried out by means of the Ge(Li) detector and IN-96 computer multi - channel analyzer. The gamma-peaks on 15 nuclides in each probe are then processed. The software providing the preparation for the experiment, gamma-spectra recording and processing, and the analysis of the experimental data has been developed. The deviation of the measurement results above the established level for some fuel subassemblies may testify to the presence of failed fuel pins,this fact being the basis of the analysis. Some criteria based on the evaluation of the root-mean-square error are also used for the analysis.

MEASUREMENT RESULTS AND CONCLUSIONS

The developed hardware and software provides the effective detection of the following fission products: ^{135}Xe(249.6 KeV), ^{131}I(364.5 KeV), ^{133}I(529.5 KeV), ^{137}Cs(661.6 KeV), ^{132}I(772,7 KeV), ^{134}Cs(795.8 KeV), ^{136}Cs(818.5 KeV), and ^{24}Na(1368.5 KeV) as well as the following gaseous fission products in cover gas: ^{133}Xe(81 KeV), ^{85m}Kr(151.3 KeV), ^{88}Kr(196.1 KeV), ^{135}Xe(249.6 KeV), ^{87}Kr(402.7 KeV), ^{138}Xe(434.5 KeV), ^{88}Rb(1836.1 KeV), ^{135m}Xe(527 KeV) and ^{41}Ar(1293.6 KeV).

Fig.2 presents the example of the fission-product gamma-spectra in sodium. The activity levels for ^{134}Cs, ^{136}Cs, ^{137}Cs, ^{138}Cs isotopes are, respectively, 8.4, 2.2, 183, 140 KBc/g.

The example for fission gas gamma-spectra in cover gas is given in Fig.3. At a fixed moment there are several fuel pins with failed clads in the core remained from previous reactor campaign.

The main features of the BOR-60 failed fuel detection system are as follows:

- effective measurements of fission product gamma-activity levels simultaneously in primary sodium and in cover gas just under the reactor operation,
- the controlled nuclide activity changes with time are described by the mathematical model to shorten the data for storage, with these data one can restore with sufficient accuracy any nuclide behaviour at any time interval,
- during each reactor campaign the measured data bank with a convenient access is built-up.

The described control system operates at BOR-60 research fast reactor during a year and a half. Over this period the actual problems for safe and economic reactor operation have been solved using this system. On this basis the development of simple, effective and low cost failed fuel detection systems for large sodium cooled fast reactors has been continued.

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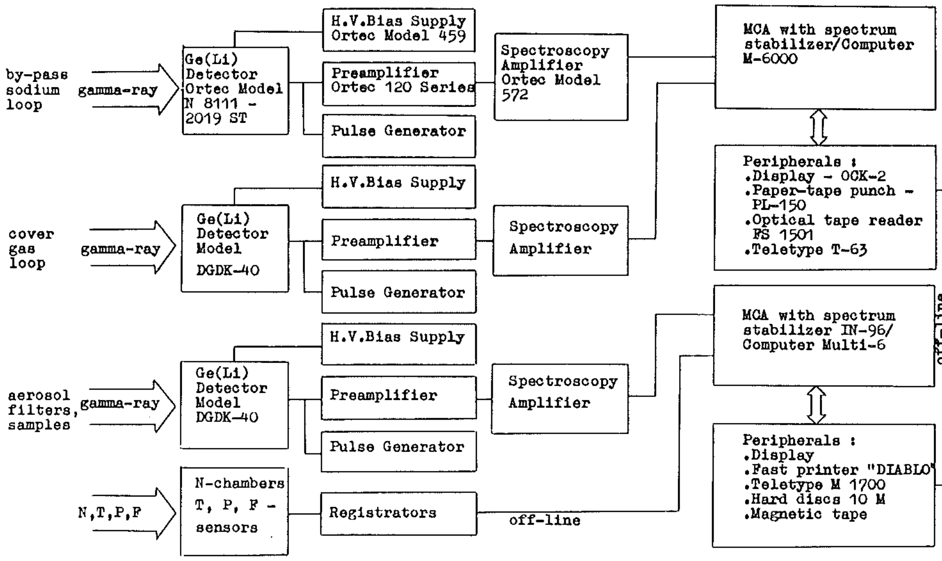


Fig.1 Block diagram of electronic system

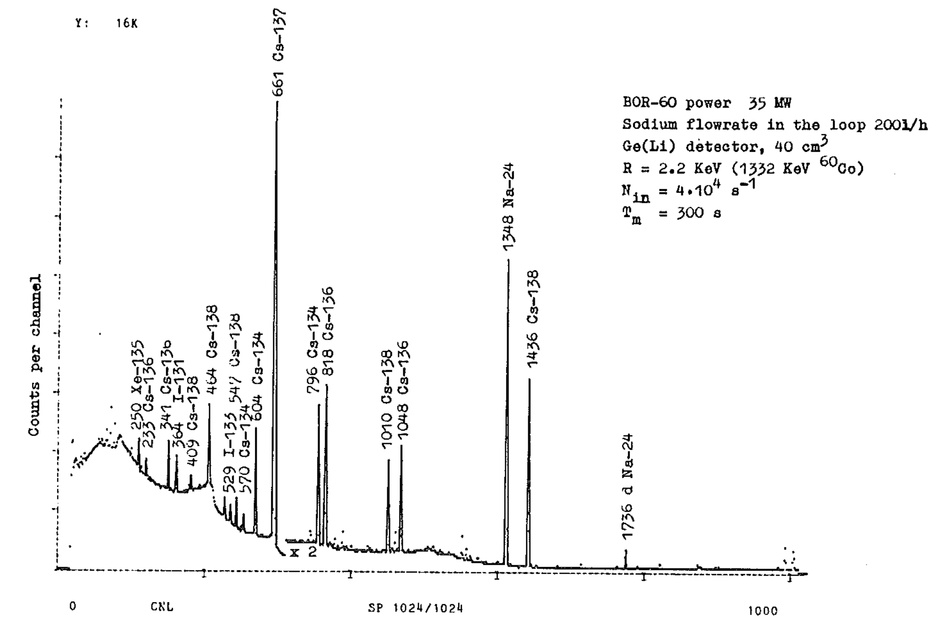


Fig. 2 Gamma-ray height distribution of by-pass sodium loop fission products measured by Ge(Li) gamma-ray spectrometer with MCA/Computer M-6000

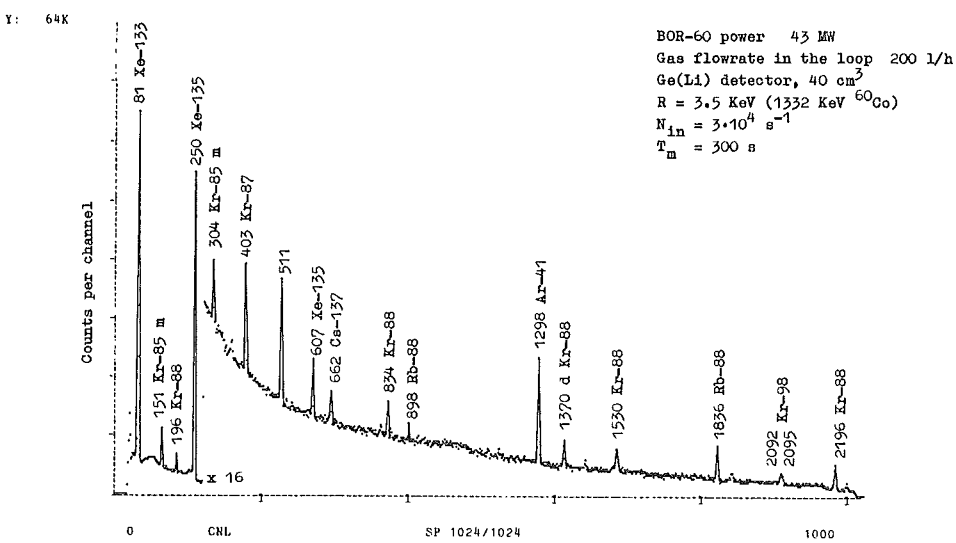


Fig. 3 Gamma-ray height distribution of cover gas fission products measured by Ge(Li) gamma-ray spectrometer

Discussion

C. Berlin, CEA:

1) What is the activity of the Na-24 in BOR-60 when the reactor is at full power?
2) Considering that the activity of the Na-24 is 10^4 to 10^5 higher than the activity of Cs-137, could you explain us how you do to measure on line reactor BOR-60 at full power the activity of Cs-137 (and I-131).

V.I. Schipilov, SRIAR:

1) When the BOR-60 reactor operate on full power the activity of the Na-24 is about 20 mCi/g.
2) We have the special method and facility to measure the relatively low gamma-activities from fission products in sodium flow under the reactor operation. We want to study this method more detail. This experiments are well under way.

INTEGRATED DND SYSTEM

by C. Berlin

AIM — SIMPLIFICATION OF THE DND INSTRUMENTATION,
— SHORTER RESPONSE TIME,
— NO PRIMARY SODIUM ABOVE THE ROOF SLAB,
— INCREASED REDONDANCY,
— LESSER COST

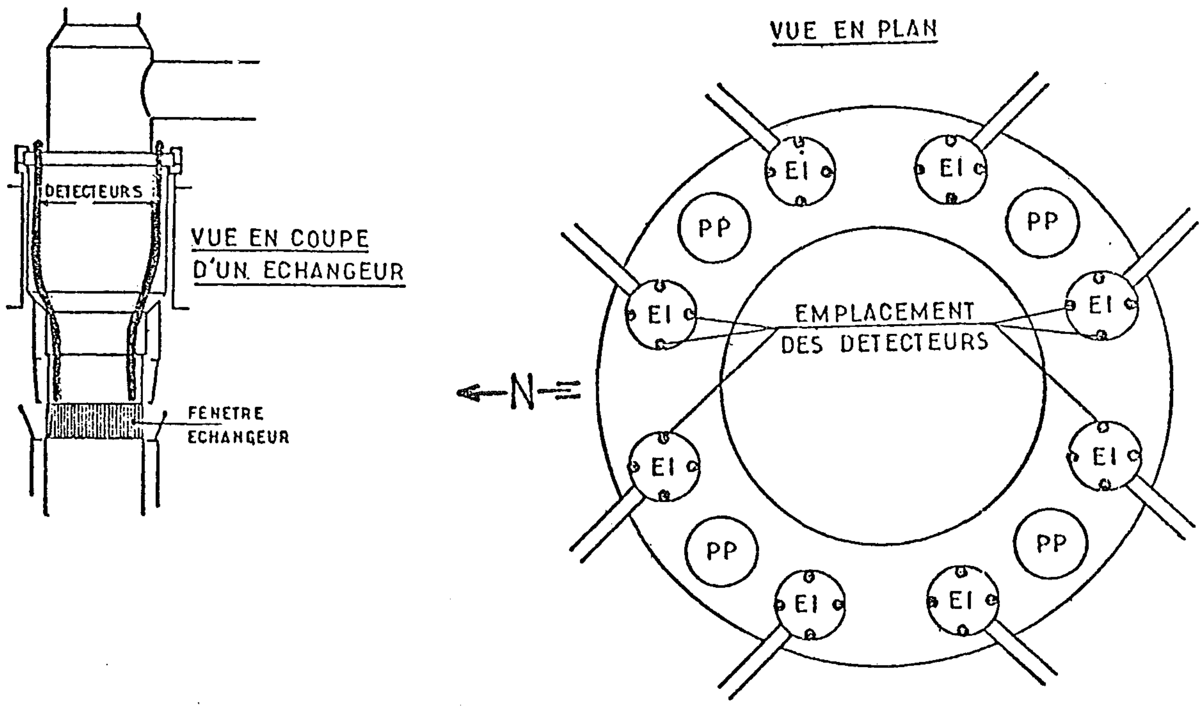
PRINCIPLE — DELAYED NEUTRONS COUNTING INSIDE THE
THE PRIMARY SODIUM WITHOUT MODERATOR.

— BACKGROUND ?

VALIDATION AT SPX 1

— MEASUREMENT OF THE BACKGROUND
— MEASUREMENT OF THE DND SIGNAL FROM 1 CM^2R D'U-MO
— RESPONSE TIME
— QUALIFICATION OF THE DESIGN CODES

} SENSITIVITY



SPX1 – DND intégrée expérimentale
Implantation dans les échangeurs intermédiaires

HIGH TEMPERATURE FISSION CHAMBERS DEVELOPMENT

by C. Berlin

AIM : INTEGRATED DND SYSTEM

WORKING CONDITIONS : TEMPERATURE : 550 - 600°C
GAMMA DOSE RATE : 10^5 RAD H^{-1}
NEUTRON FLUX : A FEW ENS OF N CM^{-2} SEC^{-1}

COUNTER CHARACTERISTICS : STAINLESS STEEL
ENRICHED URANIUM : (90 % U^5).
PURE ARGON
TWO SEPARATED WIRES (SIGNAL AND HIGH VOLTAGE),

FIRST PROTOTYPE :

TEST IN LABORATORY T~ 25°C
GAMMA DOSE RATE: 1.5 10^4 A 4 10^5 RAD H^-
NEUTRON FLUX: 25 N CM^{-2} SEC^{-1}

TEST AT PHENIX

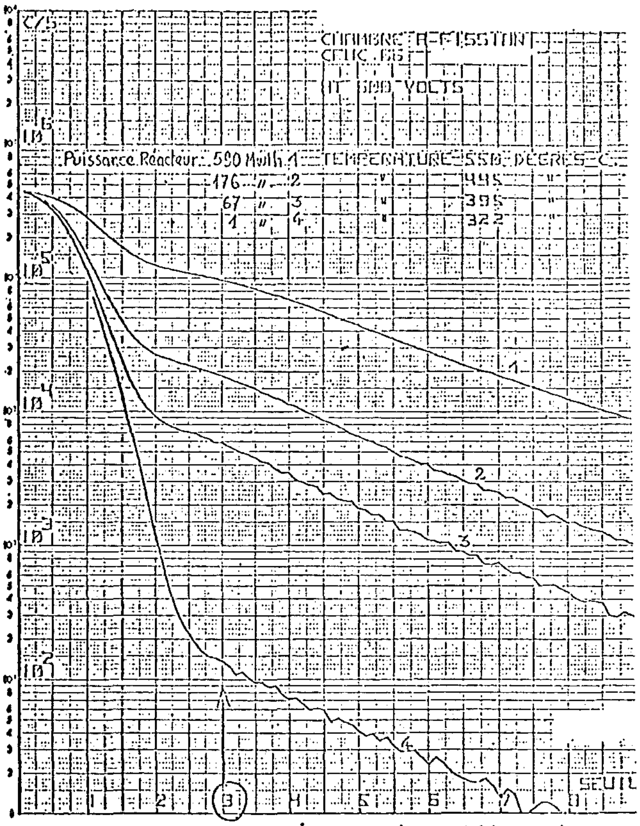
✱ REACTOR AT ZERO POWER
T = 250°C
GAMMA DOSE RATE = 0

✱ REACTOR AT FULL POWER
T = 550°C
GAMMA DOSE RATE: 3 10^4 RAD H^{-1}
NEUTRON FLUX: 10^5 A 6 10^5 N CM^{-2} SEC^{-1}

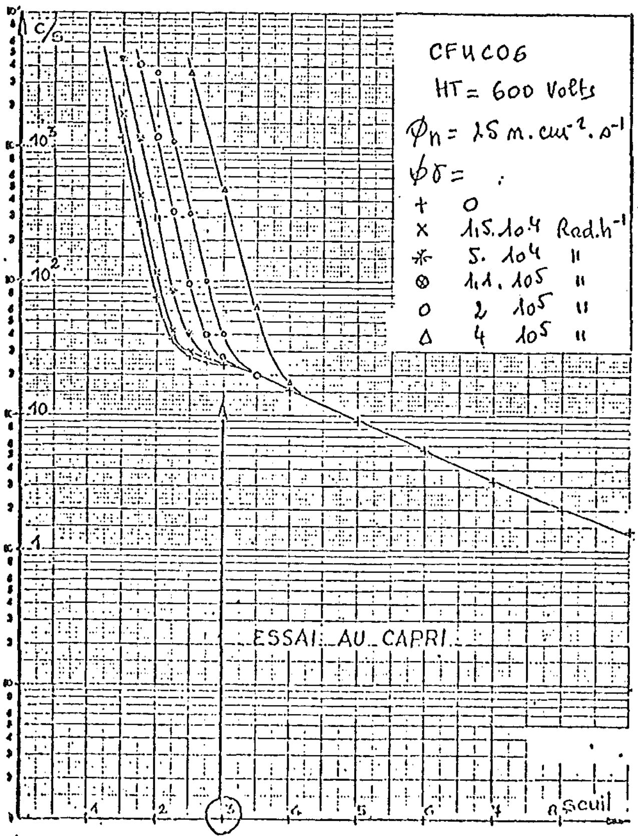
DEFECTIVE INSULATION

SECOND PROTOTYPE : (WITH NEW DESIGN)

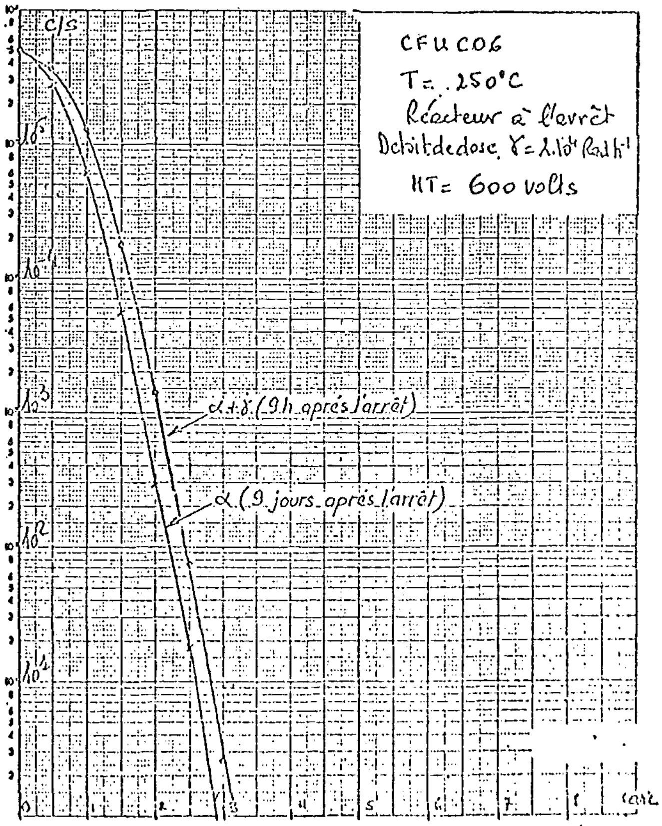
✱ BEGINNING OF THE TEST AT PHENIX : MID OF MAY 1980



Influence of the neutronic flux (Phénix)



Influence of the gamma flux



Influence of the gamma-flux (Phénix)

Discussion

D.K. Cartwright, UKAEA:

What is sensitivity of the high temperature neutron counter you described?

C. Berlin, CEA:
The measured sensitivity of the first prototype is 1c/s for 1 n cm^{-2} s^{-1}.
This is the expected sensitivity.

J. Dauk, IA:

How do you intend to measure delayed neutrons in the high background ($\sim 10^5$ $cm^{-2}s^{-1}$) with your high temperature fission counter?

C. Berlin, CEA:
According to integrated DND purpose we need to know the actual neutron flux coming directly from the core of the reactor. We intend to measure it on Super-Phenix at the start up.

W. Glauner, KfK:

New fission counters:
At the future detector position: you'l have more fast or intermediate flux?

C. Berlin, CEA:
The new fission counters will be placed in the IHX. At this place the flux is not a fast one because the long distance from the core. It is an intermediate flux I suppose.

P. Michaille, CEA:

Could you please give the dimensions of the high temperature fission chamber?

C. Berlin, CEA:
The demensions of the first prototype are about diameter $\simeq$ 45 mm, active length $\simeq$ 150 mm.

IN-PILE EXPERIMENTS WITH FAILED FUEL ELEMENTS

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Kernforschungszentrum Karlsruhe
Institut für Reaktorentwicklung

M. Relic

INTERATOM, Bergisch Gladbach
Federal Republic of Germany

A. EXPERIMENTS AT THE SILOE REACTOR IN GRENOBLE, FRANCE

Considering the poor experience with failed fuel elements an in-pile experimental program was started by Kernforschungszentrum Karlsruhe GmbH (KfK) in collaboration with Centre d'Etude Nucléaire Grenoble. The tasks of this program can be summarized as follows:

- Study of the behaviour of failed LMFBR fuel pins under normal reactor conditions.
- Characterization of defect type and its further development by nuclear instrumentation.
- To assess consequences of failed fuel operations (maintenance, shielding, cleaning, etc.) and
- to specify a core and plant instrumentation as a surveillance system to prevent undesired reactor conditions.
- To provide specific data and approve models and codes.
- To develop a basis for economical plant operation with defined classes of defective fuel pins.

The test program includes measurements with fresh and high burn-up pins.

The irradiations are performed in SILOE reactor in Grenoble (France). This reactor is a water swimming pool reactor in which several different experimental rigs can be inserted. The experimental devices used for this experimental program are called "thermopump" sodium loops. The "thermopump" sodium loops are designed for experimental irradiations of one fuel pin. The essential features of this rig are

- pin power levels of up to 60-70 kW
- thermal or epithermal neutron flux
- sodium flow from 2 to 6 meters/second
- sodium temperature regulation
  (total sodium quantity in the loop about 1,5 l)
- good accessibility for experimental instrumentation.

The experimental instrumentation includes the following measurement system:

- Continuous gas ionization chambers
- Gamma spectrometry of gas samples by GeLi
- Delayed neutron detection (DND)
- On-line gamma spectrometry of sodium by GeLi
- Off-line gamma spectrometry at selected parts of the loop or of the fuel pin by GeLi
- Neutrography.

Three experiments have been done and evaluated until now. Table I shows the most important data about the fuel pins before irradiation.

Table I: Data of the test pins for the experiments S2, S3 and S4 of the defect fuel program

	S2	S3	S4
		Mixed oxyde $(U,Pu)O_2$	
$\frac{Pu(fiss.)}{Pu(tot.)+U(tot).}$		27,6 %	
U isotopic composition		83% U5 (weight)	
Relative density		86,8% TD	
Stoechiometry		1,98	
Outside diameter		6 mm	
Material		1.4970 cold worked	
Pellets length		10 mm	
Pellets outside diameter		5,09 mm	
Fissile length		320 mm	
Clad thickness		0,38 mm	
Burn-up (max.)	0	10,4 %	9,7 %
Material dosis	0	45 dps	45 dps
Kind of artificially machined defect	Slit, open 35 mm x 1 mm	Slit, Zn plugged 5 mm x 0,05 mm	Slit, open 35 mm x 1 mm
Defect position		upper fissile column	

The SILOE-irradiation data and the failure development are summarized in Table II.

Table II: Data of the irradiation experiments S2, S3 and S4 of the defect fuel program

	S2	S3	S4
Max. linear power	420 W/cm	350 W/cm	350 W/cm
Max. clad midwall temperature	620°C	600°C	610°C
Irradiation time	47 d	23 d	21 d
Irradiation time at max. linear power	25 d	18 d	9 d
Diameter increase at the irradiation end	27 %	12,5 %	12 %
Secondary cracks	no (only enlargem.)	yes (several)	yes (prolongation till 110 mm)
Geometrical defect area	~0,9 cm^2	~2 cm^2	~2,5 cm^2
Release of fissile material	<4 mg	≤125 mg	<1,4 g
Release of Cs 137	40 %	<85 %	~80 %
Release of I 131	10÷20%	~12 %	~30 %

The evolution of the failures during the irradiation could be followed by means of the continuous measurements of the released fission products and by means of neutrographies and of gamma scans during irradiation stops. The measured data give some good information concerning the failure development during the first days and the following steady state conditions.

Regarding the information got from the applied instrumentation and the utility of this instrumentation for reactor application following statements can be made.

On-line measurements of fission gases by a gamma ionization chamber

The signals during the irradiation show a continuous fission gas release superposed by fission gas bursts or bubbles. The frequency and the amplitude of these bursts increase normally during the failure evolution time and can reach a number of ten or more bursts per hour. After the failure stabilization the burst frequency is again low. This behaviour of the fission gas release allows to distinguish the evolution from the steady state condition of a failure.

Fission gas measurements by GeLi in the cover gas

The composition of the released fission gas isotopes gives more information about the clad failure than the bulk gamma detection. Especially an emission changement will be seen clearly. During the failure evolution the short-lived isotopes increase more rapidly than the long-lived isotopes and afterwards at the equilibrium conditions the isotope emission is between the initial type and the evolution type. The augmentation of the short-lived isotopes during the failure enlargement points to a gas emission from deeper fuel layers (e.g. from the central hole) through new fuel cracks. Afterwards the formation of sodium/fuel reaction products stops partly this rapid gas release.

Delayed neutron detection (DND)

The signals of delayed neutrons show well the evolution of the pin failure. The defect area calculated with recoil model differs about factor 10 to 100 from the geometrical failure surface. This factor depends on the reactor power and/or on the fuel temperature, Fig. 1.

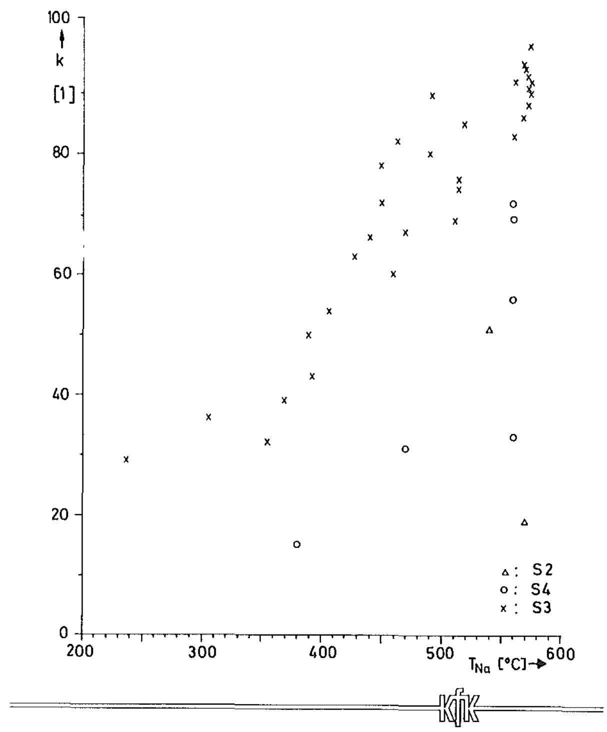

Fig. 1: S2, S4 and S3 calculated k-Factors versus the Sodium Temperature

Therefore for the characterization of the failure evolution the DND signals must always be taken at the same pin power and temperature. Although an improvement of the calculation method is necessary, the DND system is able to give the information concerning the degree of the clad failure. The threshold of a specific recoil area which could lead to an adverse operation (e.g. loss of fuel, etc.) was not found, as such conditions have not occurred.

B. EXPERIMENTS AT THE BR 2 REACTOR IN MOL, BELGIUM

The fuel element behaviour due to local off-normal cooling conditions and the possible pin-to-pin failure propagation are of special interest in the safety analysis of LMFBR's. In a joint program called "Mol 7C" the Kernforschungszentrum Karlsruhe GmbH (KfK) and the Centre d'Etude de l'Energie Nucléaire/ Studiecentrum voor Kernenergie, Mol, Belgium, are performing related experiments in a sodium loop of the BR 2 reactor. The test section contains a 37-pin bundle of fresh UO_2-fuel with an artificial local blockage of steel or fuel involving about 11 pins. After some days of preirradiation the transient test phase was initiated by interruption of cooling in the zone of local blockage /‾1,2_7. This program will be continued with preirradiated mixed oxide fuel from KNK II.

The photographs and autoradiographies taken during the post irradiation examinations of Mol 7C have shown fuel fragment blockages, Fig. 2. The observed residual sodium flow through the blockage is of importance for the cooling of the fuel inside the blockage as well as for transportation of DN precursors to the DND monitor outside the reactor.

The DND signals from the experiments reveal typical pattern during the formation of the defects as well as subsequently at constant reactor power, Fig. 3. During the first one to two seconds the DND signals increased rapidly by some orders of magnitudes. This is contributed to the failure and subsequent melting of the claddings.

Then after a rest of a few seconds the next rise of the signals by a factor 5 to 20 indicated the partial disintegration of the pins. After this the signals were about constant which means that the generated fuel fragment blockages were stable during the following reactor power operation of 48 min (Mol 7C/1), 6 min (Mol 7C/2) and 5 days (Mol 7C/3).

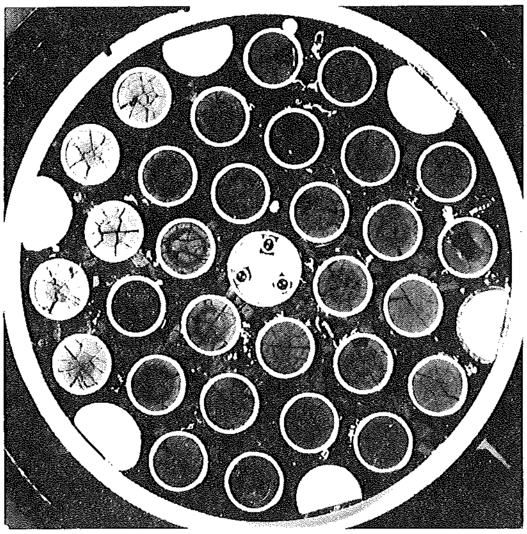

Fig. 2: Mol 7C

Fuel Fragment Blockage

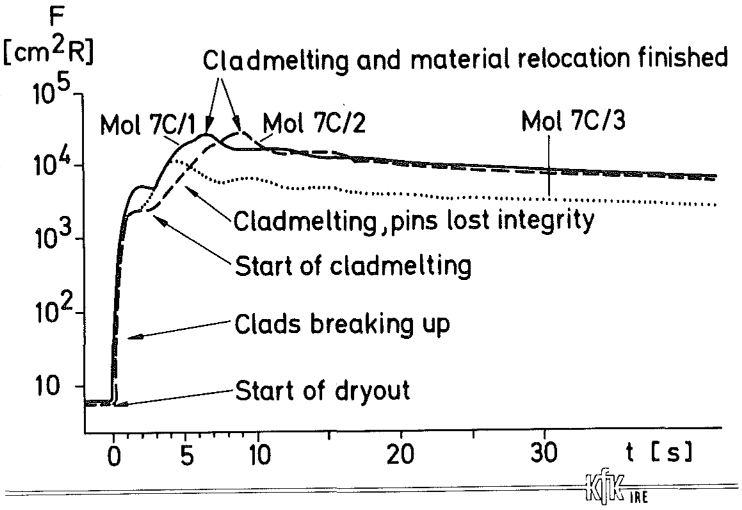

Fig. 3: DND Signals of Mol 7C. Log. Chamber Channel

The experiments demonstrated that the cooling of the whole fuel bundle was not interrupted, though several pins had failed. The main results regarding DND signals are:

- Depending only on the sodium transit time from the core to the DND monitor, the DND system gives a prompt response about the events inside the core.

- The DND system is qualified to generate a scram signal.

- The counter equipment with a wide measuring range avoided overflow by passing over to large failures.

- In a reactor a short-time memory storing with high speed the DND signals, the covergas signals and the main operational signals about 5 min before and 5 min after the scram would be helpful in signal interpretation.

C. EXPERIMENTS AT THE TOSHIBA-FISSION-PRODUCT-LOOP FPL

Instrumentation tests were performed at the Toshiba-Fission-Product-Loop FPL. It was interesting to look for the sensitivity of the KfK's (Kernforschungszentrum Karlsruhe) precipitator and its feasibility under sodium loop conditions. The KfK precipitator has the advantage that it has no moving parts and its handling is easy, Fig. 4 [3,4].
The main interest of the test was:

- Measurement of the precipitator's sensitivity for fission gases degasing out of the sodium.

- Influence of sodium vapour.

- Study of parameters as transit time, gas pressure, gas flowrate and temperature.

- Reliability.

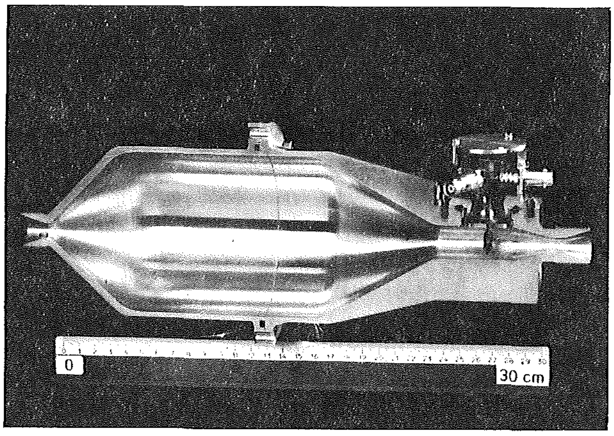

Fig. 4: The KfK-Precipitator.
At the Outlet of the Ar-Volume the Miniature-GM-Counter

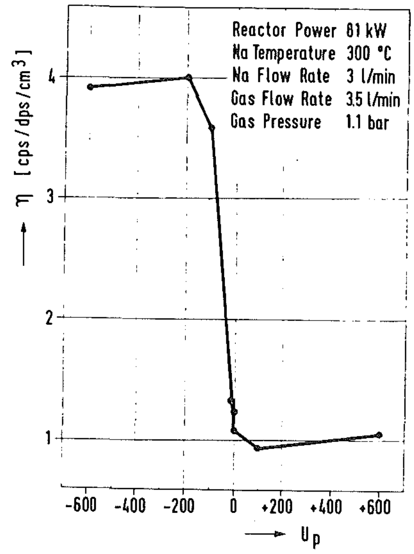

Fig. 5: Detection Efficiency η versus Precipitation Voltage U_p

The experimental program was defined together with the Toshiba company, the tests were performed by the Toshiba company. In Table III the data which were measured under water conditions and the FPL results are compared. Taking into account the uncertainty of the gamma spectrometry technique with NaJ(Tl)-crystal which was used for the measurements under water conditions, the collected activity on the Geiger-Müller-surface is substantially identical in both cases.

Table III: Analysis of the precipitate on the Geiger-Müller-surface [4,5]

	H_2O loop, 1967 [4] NaJ(Tl)-measurements			FPL, 1979 [5] GeLi measurements		
Nuclide	Cs-138	Rb-89	not analysed	Cs-138	Rb-89	Rb-88
Half-life (min)	32,3	15,4	-	32,2	15,4	17,8
Percentage of isotope in the precipitate activity in the time of precipitation	54,2%	29,4%	16,4%	46%	16,5%	37%

The detection sensitivity for the fission products is also comparable:

11 cps/decay/s.cm^3 for H_2O [4]
4 cps/decay/s.cm^3 for Na.

Fig. 5 shows the relation between the precipitation voltage and the detection efficiency. Already at low voltages the detection efficiency reaches its saturation value.

The good detection efficiency and the troubleless precipitator operation though the sodium vapour rests at FPL led to the decision to install the KfK precipitator in the experimental by-pass of the KNK-II cover gas system.

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PRESENTATIONS AND DISCUSSIONS

OF SESSION IV:

MODELS AND CODES

Studies on Modeling to Failed Fuel Detection System Response in LMFBR

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ABSTRACT

Failed Fuel Detection (FFD) system with Fission Products (FP) detection is considered to be the most promissing method, since FP provides direct information against fuel element failure.

For designing FFD system and for evaluating FFD signals, some adequate FFD signal response to fuel failure have been required. But few models are available in nowadays.

Thus Power Reactor and Nuclear Fuel Development Corporation (PNC) had developed FFD response model with computer codes, based on several fundamental investigations on FP release and FP behavior, and referred to foreign country experiences on fuel failure.

In developing the model, noble gas and halogen FP release and behavior were considered, since FFD system would be composed of both cover gas monitoring and delayed neutron monitoring.

The developed model can provide typical fuel failure response and detection limit which depends on various background signals at cover gas monitoring and delayed neutron monitoring.

According to the FFD response model, we tried to assume fuel failure response and detection limit at Japan experimental fast reactor "JOYO".

The detection limit of JOYO FFD system was estimated by measuring the background signals.

Followed on the studies, a complete computer code has been now made with some improvement.

On the paper, the details of the model, out line of developed computer code, status of JOYO FFD system, and trial assumption of JOYO FFD response and detection limit.

INTRODUCTION

Failed fuel detection (FFD) system has important role for LMFBR safety and availability aspect. Among various detection system, most promising method is the detection to radiation from fission products (FP) which escape from failed fuel pin into coolant or diffuse into gas space through coolant. The most conventional FFD system for loop type LMFBR is composed with delayed neutron detection (DND) subsystem and cover-gas monitoring (CGM) subsystem, as shown Fig.1. To FFD system, some fuel failure signal responce estimations have been required for setting automatic warning or reactor trip level, or operator decision making to the plannings continious plant operation beyond fuel failure. For this purpose, PNC (Power Reactor and Nuclear Fuel Corp.) has planed to make theoretical models which can descrived the FFD signal responce at loop type LMFBR.

For making adequate model, PNC has reffered several in-pile loop experiments(1) and other countries experiences on FP release and behavior in LMFBR coolant system at fuel failure. Making models on FP release from failed pins and FP behaviror in reactor coolant system, rial calculation on DND and CGM responces have been carried out in assuming fuel failure at Japan experimental fast breeder reactor JOYO(2). And also the minimum detection limits of each subsystem were estimated with using measured back ground signals(2).

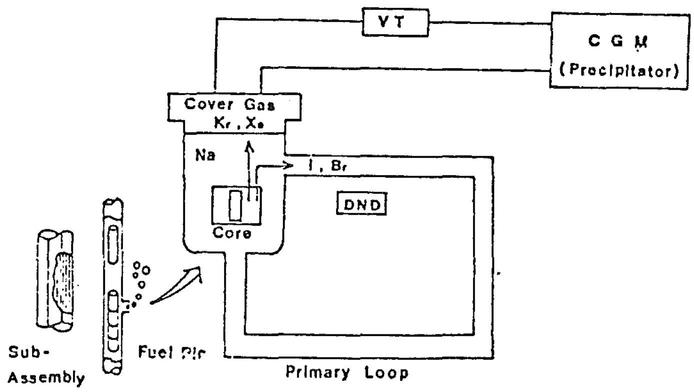

Fig.1 Block Diagram of "Loop Type LMFBR" FFD System

MODELING TO FP RELEASE AND BEHAVIOR

(1) Modeling concept

Because FP release from failed fuel pins and FP behavior in reactor coolant system were most important subjects on modeling FFD system responce, FP transportation process were divided into following 5 steps.

i) FP release from fuel pellets into fuel pin gap or plenum.

ii) FP relese from failure pin into coolant.

iii) FP behavior in reactor coolant plenum.

iv) FP transportation in primary coolant pipe.

v) FP transportation from coolant into reactor cover gas plenum.

On the other hand, fuel failure modes were classified into two codes. The one is constant FP release with time and the other is time dependent FP release. The constant release can be used for gas leaker, and time dependent release can be used for burst failure such like cracking growth at fuel pin surface.

(2) FP release model at fuel failure

FP inventry balance in fuel pin with failure was defined as Eq.(1).

$$\frac{dG_k(t)}{dt} = R_k(t) - L_k(t) - \lambda_k G_k(t) + \lambda_{k'} G_{k'}(t) \quad ...... \quad (1)$$

Where, $G_k(t)$: kth FP inventry in fuel pin gap and plenum (atoms)

$R_k(t)$: kth FP release rate from fuel pellets into pin gap or plenum (atoms/sec.)

$L_k(t)$: kth FP escape rate from fuel pin at fuel failure only for $t \gtrsim T$ (atoms/sec.)

λ_k : kth FP decay constant (1/sec.)

k : FP identification suffix

k' : identification suffix for the precurser of kth FP

For solving Eq.(1), R_k and L_k should be given.

a) FP escape from failure pin

Regarding to escape rate $L_k(t)$, three analytical models have been made. To the constant release model, $L_k(t)$ can be written as Eq.(2).

$$L_k(t) = \lambda_d G_k(T) \quad \ldots\ldots (2)$$

Where, λ_d : FP escape constant for all gaseous FP (1/sec.)

$G_k(T)$: kth FP inventry in fuel pin gap and plenum (atoms)

To time dependence FP release, two different process were considered. The first model was that FP escape rate was equal to FP release rate from fuel pellets as shown in Eq.(3)

$$L_k(t)=R_k(t) \quad \ldots\ldots (3)$$

The second model was defined as Eq.(4) with using equivalent gas escaping flow rate.

$$L_k(t) = \frac{W_L(t)}{V_p(t)} G_k(t)/\sum_k G_k(t) \quad \ldots\ldots (4)$$

Where, $W_L(t)$: Equivalent gas escaping flow rate (atoms^3/sec.)

$V_p(t)$: Total gas volume in fuel pin gap and plenum (atoms^3)

Moreover, the equivalent gas escaping flow rate W_L can be given by theoritical models which were introduced by T.C. Chawle et al(3) and R.E. Murata.(4) acording to the failure location along fuel pin. The failure locations were considered to be at pin plenum region and fuel column region as shown in Fig.2.

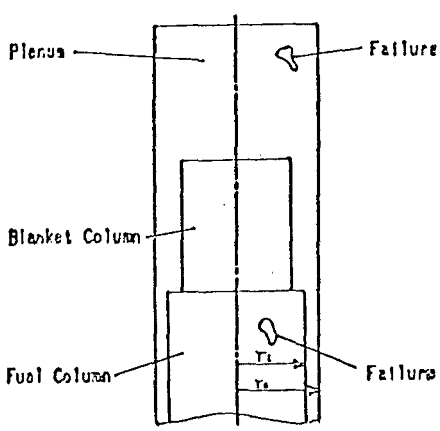

Fig.2 Schematic View of Failure Location

For a failure at pin plenum region, the escape rate W_L is given by Eq.(5). and (6).(3)

$$W_L = C_d S_d \rho \left(\frac{P_o}{P_i}\right)^{1/\gamma} \left[\frac{2\gamma}{\gamma-1}\left(1-\left(\frac{P_o}{P_i}\right)^{\gamma-1/\gamma}\right) RTg_c\right] \quad \ldots\ldots (5)$$

for

$$\frac{P_o}{P_i} \geqq \left(\frac{2}{\gamma+1}\right)^{\gamma/(\gamma-1)}$$

and

$$W_L = C_d S_d \rho \left(\frac{2}{\gamma+1}\right)^{\gamma/(\gamma-1)} \left(\frac{2\gamma}{\gamma+1} RTg_c\right)^{1/2} \quad \ldots\ldots (6)$$

for

$$\frac{P_o}{P_i} < \left(\frac{2}{\gamma+1}\right)^{\gamma/(\gamma-1)}$$

Where, P_i: FP gas pressure inside fuel clad (atm)

P_o: Coolant pressure (atm)

γ : Ratio of specific heat (C_p/C_v)

C_p: Specific heat of gas at constant pressure (cal/cm^3)

C_v: Specific heat of gas at constant volume (cal/cm^3)

C_d: Coefficient of discharge (-)

S_d: Breach area at pin plenum (cm^2)

R : Gas constant (atm·cm^3/mol)

ρ : FP and He mixture gas density (atom/cm^3)

T : Gas temperature (°K)

g_c: dimentional conversion factor (dyne/cm^2·atm)

For a failure at fuel column,(4) following equation can be used for laminar flow.

$$W_L = \frac{\pi D_o D_h^3 (P_\ell^2 - P_o^2)}{128 \mu RT\ell} \quad \ldots\ldots (7)$$

Where, D_h: $2(r_0-r_1)$, diametrical gap (cm)

r_0: annulus outer radius (cm)

r_1: annulus inner radius (cm)

D_0: $= 2r_0$

ℓ : flow annulus length (cm)

μ : gas viscosity (atm·sec/cm^2)

P_1: pressure at outlet from annulus (atm)

P_0: pressure at inlet to annulus (atm)

For turbulent flow region, the equivalent flow rate can be given with using Funning friction factor of f.

$$W_L = \frac{\pi D_o D_h^{3/2}}{4(\ell f RT)^{1/2}} (P_\ell^2 - P_o^2)^{1/2} \quad \ldots\ldots (8)$$

In case of $L_k(t)=R_k(t)$, theoretical model will be handled in the next paragraph.(5)

(b) FP release from fuel pellet into pin gap

For making FP release model, recoil process and diffusion process have been considered. To the recoil process, R_{kr} can be given as Eq.(9).

$$R_{kr} = \frac{R_r \beta}{4} K_f \quad \ldots\ldots (9)$$

Where, R_r: kth FP recoil range (cm)

β : kth FP generation rate per unit volume (atms/cm^3 sec)

K_f: Roughness factor

To the diffusion process, Booth diffusion model(5) was used for both stable and radioactive FP release from fuel pellets.

For radioactive FP,

$$R_{kd} = 3\left(\frac{D'}{\lambda_k}\right)^{1/2}\left\{\coth\left(\frac{\lambda_k}{D'_k}\right)^{1/2} - \left(\frac{D'_k}{\lambda_k}\right)^{1/2}\right\} \quad \ldots\ldots (10)$$

Where, D'_k: D_k/a^2 (1/sec.)

a : Equivalent sphere radius (cm)

D_k : FP diffusion constant (cm^2/sec)

On the other hand, stable FP release rate can be given in Eq.(11).

$$R_{kd} = 1-\frac{6}{90D'_kT_t} + \frac{6}{\pi^4 D'_kT_t}\sum_{n=1}^{\infty}\frac{1}{n^4}\exp(-n^2u^2D'_kT_t) \quad \ldots(11)$$

Where, T_t: Fuel irradiated time

Thus $R_k(t)$ can be given as

$$R_k(t) = R_{kr}(t) + R_{kd}(t) \quad \ldots\ldots\ldots\ldots\ldots\ldots\ldots(12)$$

(3) FP behavior model in the coolant

FP behavior in coolant system was considered with three steps. The first step was the behavior in a fuel subassembly with fuel failure, the second step was the behavior at coolant plenum, and the 3rd step was the transportation from coolant system into cover gas plenum. The FP transportation concept is shown in Fig.(3).

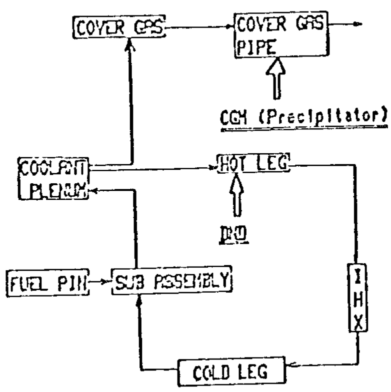

Fig.3 FP Transport

The released FP concentration in fuel subassembly can be written as Eq.(13) with an assumption that entire volume is divided into small cells.

$$\frac{d}{dt}C_{sk}dV = \int(L_K + \lambda_{(k-1)}C_{s(k-1)} - \lambda_k C_{sk})dV + \int(1-k_{dk})C_{sk}\vec{V}_1 dS_1 + \int C_{sk}\vec{V}_2 dS_2 \quad \ldots\ldots (13)$$

$$\frac{d}{dt}C_{ck}dV = (W_s C_{sk} + \int \lambda_{(k-1)}C_{c(k-1)} - \lambda_k C_{ck} + \lambda_k C_{ck}(t-\tau))dV - \mathrm{div}(C_{ck}V) - K_{ck}C'_{ck} \quad \ldots\ldots\ldots\ldots(14)$$

Where, C_{ck}: FP concentration in primary coolant (atoms/cm^3)

W_s : Sodium flow rate in on fuel assembly (cm^3/sec)

k-1: kth FP precursor identification

τ : Coolant circulating time (sec)

$\vec{V}$: Coolant flow velocity with vector form (cm/sec)

K_{ck}: kth FP transport coefficient from sodium into covergas (cm/sec)

C'_{ck}: FP concentration at contact surface between coolant and cover gas

For solving this equation, flow velosity distribution should be given.

The flow velocity distribution was solved by solving the combination equation of the mass continuity equation and the motion equation (Navier-Stokes Equation) under adequate boundary conditions. For this purpose, modified SOLA code(6) was used. For obtaining DND signal response of conventional system as shown in Fig.1, the FP concentration in the primary coolant pipe was required. The FP concentrations in the primary pipe was considered to be changed only by decay but not by plate-out.

On the other hand, FP concentration in cover gas space can be led as Eq.(15) by using transport coefficient of K_{ck} in Eq.(14), and (16).

$$\frac{dN_k}{dt} = \int k_c C'_{sk}dS_p + \lambda_{(k-1)}N_{(k-1)} - k_s\frac{S_G}{V_G}N_k - \lambda_k N_k - \frac{W_G}{V_G}S_v N_k \quad (15)$$

$$C_{Gk} \equiv \frac{N_k}{V_G} \quad \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots \quad (16)$$

Where, N_k: FP atoms number in cover gas (atoms)

S_p: Contact area between sodium and cover gas (cm^2)

k_s: FP plate out coefficient at cover gas system wall (cm/sec)

S_G: Total cover gas system wall(cm^2)

V_G: Cover gas volume (cm^3)

W_G: Cover gas sampling flow rate (cm^3/sec)

S_v: Sampling nozzle area (cm^2)

(4) Response on failed fuel signal

a) CGM response

Assuming the CGM to be composed of moving wire type precipitator, the response model was led as shown in Eq.(17).

$$M_c \doteq \epsilon_p\epsilon_c \sum_k \frac{\lambda_k}{\lambda_{k+1}-\lambda_k}\int_0^{t_s} C_{Gk}W_G\,dt \times (e^{-\lambda_k t_p} - e^{-\lambda_{k+1}t_p})e^{-\lambda_{k+1}t_w} \quad \ldots\ldots\ldots\ldots (17)$$

Where, M_c: Count rate by a precipitator (cps)

ϵ_p: Precipitation efficiency

ϵ_c: Counting efficiency

λ_{k+1}: Daughter nuclide decay constant of kth FP (1/sec)

t_s: Soaking time (sec)

t_p: Gas transit time of chamber (sec)

t_w: Required time of wire moving (sec)

b) DND response

The DND response was led as shown in Eq.(18)

$$M_D = \eta_D \sum_k C_{ck}e^{-\lambda_k t_d}V_D Y_k \quad \ldots\ldots\ldots\ldots \quad (18)$$

Where, M_D: DND count-rate (cps)

η_D: DND efficiency (cps/atoms/cm^3)

t_d: Transit time from inlet of primary coolant pipe to DND location (sec)

V_D: Measuring sodium volume (cm^3)

Y_k: kth FP branching ratio

TRIAL CALCULATION

Using the developed model, FFD signal response was estimated with assuming fuel failure at JOYO.

Major parameters which were used for calculation are listed in Table(1), (2), (3), (4). And typical flow pattern which was used the FP concentration calculation is shown in Fig.(4). The trial calculations provided typical response to leaker failure and burst failure as shown in Fig.(5), (6), and (7).

In these figures, the back ground levels are shown for obtaining the detection limit of each subsystem.

TABLE 1 Fuel Data

Fuel Compotion (fission rate)

^{235}U	22.29 %
^{238}U	7.21 %
^{239}Pu	59.11 %
^{241}Pu	11.39 %

Fuel Dimention

Pellet radius	0.2315 cm
Clad inner radius	0.24 cm
Clad outer radius	0.275 cm

Fission Rate

3×10^{10} fission/watt

TABLE 2 Parameters to FP release

CONSTANT	VALUE
Equivalent spere radius	5×10^{-4} cm
FP range	10^{-3} cm
Gas viscosity	5.334×10^{-10} atm sec/cm^2
Ratio of specific heat	1.66
Coefficient discharge	0.5

TABLE 3 Diffusion Constant

Temperature : 1400 °C

Noble gas	7.990×10^{-15}
Halogen	1.598×10^{-14}
Alkali	5.133×10^{-4}

TABLE 4 Deposition Rate and Transport Coefficient

	Halogen	Noble gas	Alkali
K_{wk} *	2.449×10^{-6}	———	2.763×10^{-5}
E_w *	11.187	———	8.359
k_c **	2.763×10^{-4}	2.763×10^{-2}	2.763×10^{-2}

*) $k_w = k_{wo}\, e^{-E_w/RT}$

E_w = Activation Energy (Kcal/mol)

R = Gas Constant (Kcal/mol K)

T = Temperature (K)

**) k_c = Transport Coefficient (/cm)

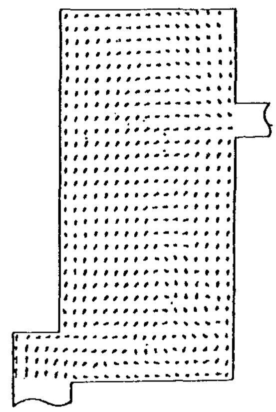

Fig.4 Calculated Flow Distribution

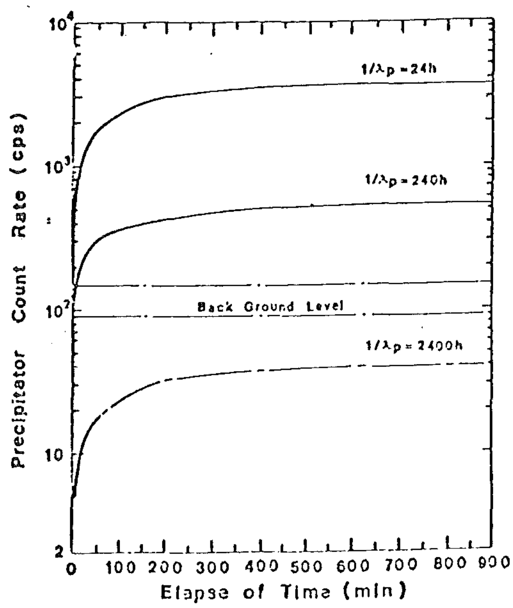

Fig.5 CGM Precipitator Responce to Leaker

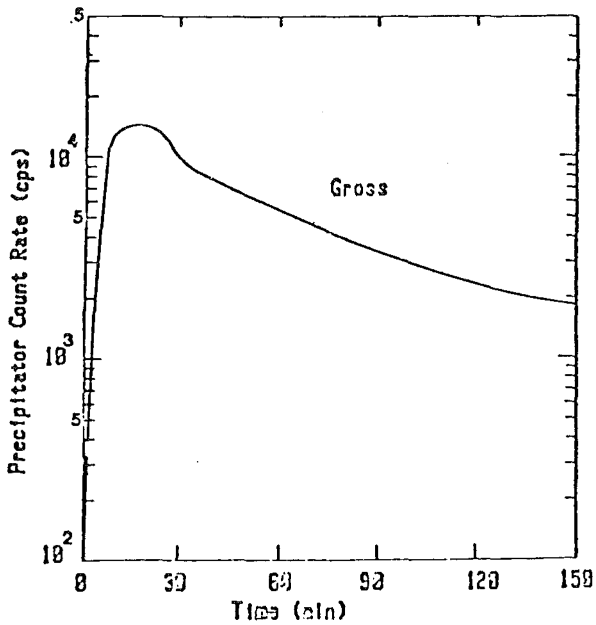

Fig.6 CGM Precipitator Responce to Burst Failure

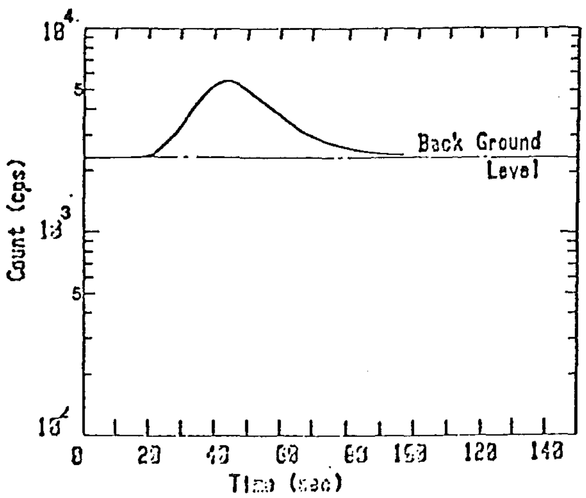

Fig.7 DND Responce to Burst Failure

CONCLUSION

Based on the theoretical model, PNC has a plan to make a computer code for FFD response with some improvement. For improving the model, more detail data on FP release from failed fuel pins, and FP behavior in reactor coolant system.

In order to solve this problem, a lot of experimental investigations have been planed in Japan.

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Discussion

F.E. Holt, HEDL:

Question to Fig. 5: What is the significance of 24 h , 240 h , etc. ?

T. Miyazawa, Toshiba:
The term λp is the escape rate of gas from the fuel pin.

J.D.B. Lambert, ANL:

Do you have plans to check in-pile for the parameters to be used in your code?

T. Miyazawa, Toshiba:
Yes. One is for FPL, the other is for JOYO with stable isotope.

F.E. Holt, HEDL:

Your model is theoretical and will be modified as you obtain data?

T. Miyazawa, Toshiba:
Yes.

THE DETECTION AND LOCATION OF FUEL FAILURES IN SODIUM COOLED FAST REACTORS

REVIEW OF UK WORK

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PART 4

MODELS AND CODES FOR THE RELEASE OF FISSION PRODUCTS USED IN FAILED FUEL DETECTION STUDIES IN THE UK

MODELS AND CODES FOR THE RELEASE OF FISSION PRODUCTS USED IN FAILED FUEL DETECTION STUDIES IN THE UK

The mechanisms guiding the release of fission products from fuel are being studied at a number of establishments in the UK. Detailed codes describing the release are being constructed from controlled experiments on small fuel samples. This note reviews this work.

From experience with defected pins it is evident that the fission product emission is greater than that expected solely from recoil fission products over the area of the defect. Long lived fission gases obviously accumulate in the plenum of the pin, some if not most of these escape at pin failure followed by a subsequent steady release. The initial release could carry with it some fission products but normally the escape is expected to be too slow to carry the short lived DN precursors, for example, to the coolant. The subsequent release of about 1 cm^3 per day also probably has little effect on release of short lived species, but significant activity from long lived species such as Xe133 and Xe135 will probably escape and remain in equilibrium in the cover gas.

Release from Fuel

The rate of release of both long and short-lived species from polycrystalline fuel is a function of the effective surface area of the fuel, which depends on burn-up, as well as the fuel rating and fuel temperature (itself a function of rating). Most of the primary fission product atoms come to rest within the fuel crystallites and diffuse towards the grain boundaries by a mechanism which depends on thermal activation, with enhancement from irradiation effects and delays arising from temporary trapping in precipitated fission gas bubbles. During the time taken to reach the grain boundaries, the primary fission products may decay radioactively and the resulting members of the fission chain continue to diffuse towards the surface. The resultant emission of a given isotope may thus be considered as a summation of a number of distinct but related processes involving its precursors as well as itself.

The mechanism of release from the grain boundaries of the fuel to the free volume of the pin can be shown to involve the passage of volatile atoms along narrow tunnels, between the grains, which are formed during irradiation by the interlinkage of fission gas bubbles on grain boundaries and along grain edges. The process is aided by the formation of large voids and cracks within the fuel.

The release of short-lived fission gases from thermal and fast rector fuels has been studied in the UK by the UKAEA and CEGB. At DNE, experiments have been conducted with a vented pin in the DFR (1). The results were compared with experiments carried out in the thermal reactor DIDO at AERE (2).

More detailed experiments have been made by BNL and AERE (3, 4) on small fuel samples to obtain the temperature and rating dependancies of the release (5) and to study the effects of fuel structure and burn-up.

For short-lived isotopes of a given chemical species ($t_{½}$ <20 h), under specified conditions of temperature, fuel rating and burn-up, the release rate, R, can be represented as a fraction of the rate of production, B, by the expression

$$(R/B) \qquad (D / \lambda)^{½}$$

where D is the effective diffusion coefficient for the chemical species and λ is the decay constant of the isotope. The total rate of emmision, including the diffusion and emission of precursors of different chemical form, can be calculated using the method of Friskney and Speight (6). In comparing the measured and derived values of the total emission rates for a range of isotopes with different decay constants, it has become evident that the effective diffusion rates of different chemical species are different: in particular, the value for Br is likely to be about 300 times greater than that for Kr and the value for I about 10 times greater than that for Xe. These larger values should be used in calculating the emission of the DN precursors Br89 and I137.

The rate of release of a given isotope from the fuel in a whole pin under specified conditions has been computed by dividing the fuel volume into a series of axial and radial zones, assigning the appropriate values of temperature, rating and burn-up to each section and calculating the release from that section on the basis of measured release rates. The contributions from each section can be summed for the whole pin, or alternatively axial zones may be treated separately. Axial diffusion to the defect in a pin may be calculated using a simple model. It is calculated that the quantity of an isotope with disintegration constant λ which reaches the defect is that which is born within a diffusive distance $(D/\lambda)^{½}$ of the defect. (D is the diffusion coefficient). If coolant has not entered the pin then D is the diffusion coefficient of the gas in the voids, although this should be modified if significant diffusion is through cracks in the fuel.

The picture is further complicated by pressure fluctuations at the defect, and by the ingress of sodium which is aided by these fluctuations, and in the case of delayed neutron precursor bromines and iodines by their affinity for other fission products, particularly caesium.

The chemistry is such that caesium which accumulates near to the cool cladding would be expected to be washed out taking any halogens chemically bonded to it. Subsequently the sodium, depending on the oxygen potential, can react with the fuel forming complex uranates or plutanates depending on the fuel, (7). Being less dense than the fuel these compounds cause swelling of the can and unless they are washed away release of fission products will be affected by diffusion through these complexes.

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Discussion

S. Jacobi, KfK:

What is the main part of the diffusion release:

- Diffusion through the solid state lattice?
- Diffusion along the grain boundaries?
- Diffusion through open porosity?
- Others?

D.K. Cartwright, UKAEA:
I think that diffusion through the solid lattice is a main part of the release, but this must depend on the details of the failure.

J.D.B. Lambert, ANL:

Are there stable (R/B) for all isotopes in irradiated fuel after 1-2 % burn up?

D.K. Cartwright, UKAEA:
Yes for rare gases, for other conditions constant.

P. Michaille, CEA:

When you interpret the DND signals in terms of diffusion, do you take into account all the fuel of the pin or only a part of it at the vicinity of the defect?

D.K. Cartwright, UKAEA:
We calculate the release from that part of the pin within a diffusion distance from the defect.

P. Michaille, CEA,

Have you an explanation for the fact the bromides and iodides have such a greater diffusion coefficient than noble gases?

Comment from D.E. Mahagin, HEDL

The insolubility of noble gases in solids inhibits their diffusion through the solid relative to other gases than can be dissolved in the solid.

Delayed-Neutron Signal Analysis Techniques

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ABSTRACT*

The experimental and analytical techniques developed at EBR-II to characterize delayed-neutron signals are described. These techniques include raw data averaging, special flow-reduction tests on the delayed-neutron detector loop, and the development of a special code (GIRAFFE) to unfold the character of the signals.

INTRODUCTION

Ever since the first failure of an experimental fuel pin in 1967 it has been hoped that the character of fission-gas releases from failures--their fine structure as it were—could be used to infer information about the failures themselves. However, apart from some very qualitative statements about the type of fuel involved--oxide or metal, and its burnup--high or low (1), attempts to extract further information about failures from their gas release met with surprisingly little success. Attention then was focused on delayed-neutron (DN) monitoring as a means to garner additional information during reactor operation. It was argued that fuel directly exposed to primary sodium would release the very short-lived DN precursors to it by recoil; the magnitude of the signal then could be directly correlated with the size (or the severity) of defect, provided allowance were made for fission rate, counter efficiency, and a few other parameters. In truth, little happened in the first 12 years of EBR-II operation to change this belief, as described in a parallel paper (2). The small number of DN signals from natural failures during that period were curtailed by scram of the reactor, when the signals would rapidly die away.

When the DN detector FERD was removed from the EBR-II trip circuit in 1977, in order to allow breached-pin tests to start, a number of new and revealing aspects of DN monitoring became apparent. During tests to calibrate FERD, "systems" effects—such as the mixing of sodium in the upper plenum, and the transit time from core to detector—were first observed and appreciated. Soon afterwards, the DN signals from initial breached-pin tests made it abundantly clear that release from breached pins could not be by recoil only. Signals were many times the value to be expected by such release from the geometrical area of exposed fuel. The "enhancement factor" was born in the U.S.; workers abroad had observed the same phenomenon (3,4). This paper describes a history of the techniques developed to date at EBR-II to extract as much information as possible from DN signals regarding the nature of cladding breaches.

INSIGHT GAINED BY SPECIAL TESTS

The DN detector FERD and the cover-gas activity monitor GLASS were calibrated in 1977 (5), and again in 1978, by using a fission-product source (FPS) which gave release by recoil only. In 1977 the FPS was irradiated in two core locations: 5B2 and 5E4; 5B2 is closer to the primary outlet pipe than 5E4 (Fig. 1). Response of the FERD was sensitive and linear to reactor power change, or fission rate, as described elsewhere (2). The FERD signals under all constant-power conditions, however, exhibited short-term cyclic fluctuations of ±15% about the mean count rate on all three (independent) channels. The fluctuations appeared to be in phase (Fig. 2), although their magnitude and (common) frequency varied.

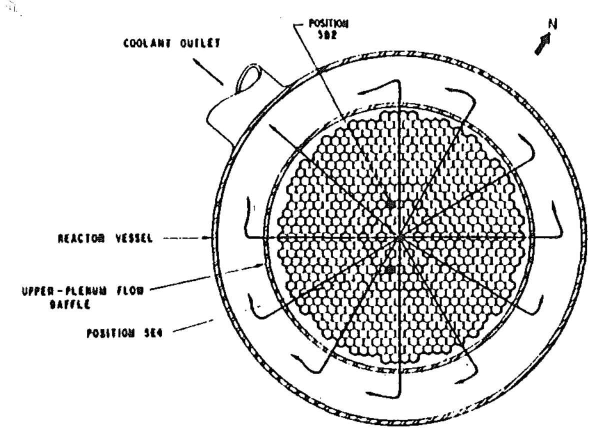

Fig. 1. Flow in Upper Plenum of EBR-II

Similar fluctuations had been always observed in the background DN signals, but were ascribed to normal variation due to the low count rates; the fluctuations in signal from the FPS were much more than could be explained by counting statistics. Attempts were made to correlate the fluctuations with variation in a number of reactor parameters, such as primary flowrate, flow in the FERD loop, fine-scale power variations, even beats between the two primary pumps; none was successful. It was found later that part of the fluctuations resulted from the coupling between the FERD loop and the IHX discharge. When the FERD-loop flow was reduced slightly, the DN signal count rate went up and the fluctuations decreased; bulk sodium from the tank was clearly diluting the signal strength at full-rated flow in the FERD loop. This effect was peculiar to EBR-II; clearly it could be, and was minimized.

The remaining fluctuations could only be ascribed to varying mixing of sodium in the upper plenum, and therefore to varying transit paths for DN precursors to follow. Although the movement of sodium is radially outward through the circumferential flow baffle (Fig. 1) and around the periphery to the outlet pipe, this whole region has much turbulence. A similar, but perhaps less marked effect will occur for reactors with more than one outlet pipe. The phenomenon was ineradicable,

*Work supported by the U.S. Department of Energy.

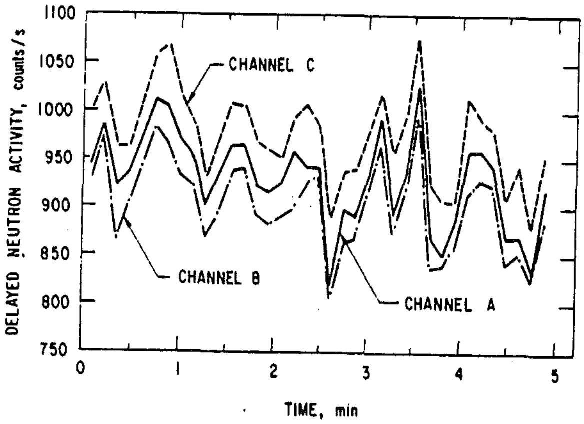

Fig. 2. Fluctuations in DN Signals from the FPS

but since it equally affected all signals we hoped it could be ignored. In practice, all DN signals and related parameters (flow, reactor power...) during FERD tests were collected on fast magnetic tape at two readings per second, and data averaging later used to produce mean count rates corrected for small variations in reactor power. Very stable DN signals were then apparent. There is now an historic file of DN tapes which can be used and re-used as analytical techniques improve.

FERD-Loop Flow Reduction Reduction (FFR) Tests

Further insight was gained while the FPS was operated at full power by reducing the FERD-loop flow in steps and recording the DN signals. Reducing the flow artificially increases the transit time to the detectors (Fig. 3a); the corresponding DN signals are shown in Fig. 3b. The activities (A_1,A_2) recorded at two flow rates (transit times in the FERD loop of t_1 and t_2) may then be related to the unknown transit in-core, T_{tr}, by the relationship:

$$\frac{A_1}{A_2} = \frac{\sum_{i=1}^{n} R_i Y_i \lambda_i \exp[-\lambda_i(T_{tr} + t_1)]}{\sum_{i=1}^{n} R_i Y_i \lambda_i \exp[-\lambda_i(T_{tr} + t_2)]} \quad , \ \ldots.(1),$$

where R_i = release rate for nuclide i,

Y_i = delayed-neutron yield for nuclide,

λ_i = decay constant for nuclide i,

and n = number of isotopes included in model (currently nine).

This equation was used to determine the in-core transit time. R_i values were obtained from recoil ranges determined in CP-5; Y_i values were obtained from Meek and Rider (6), with probabilities of DN emission from the data of Izak-Biran and Amiel (7).

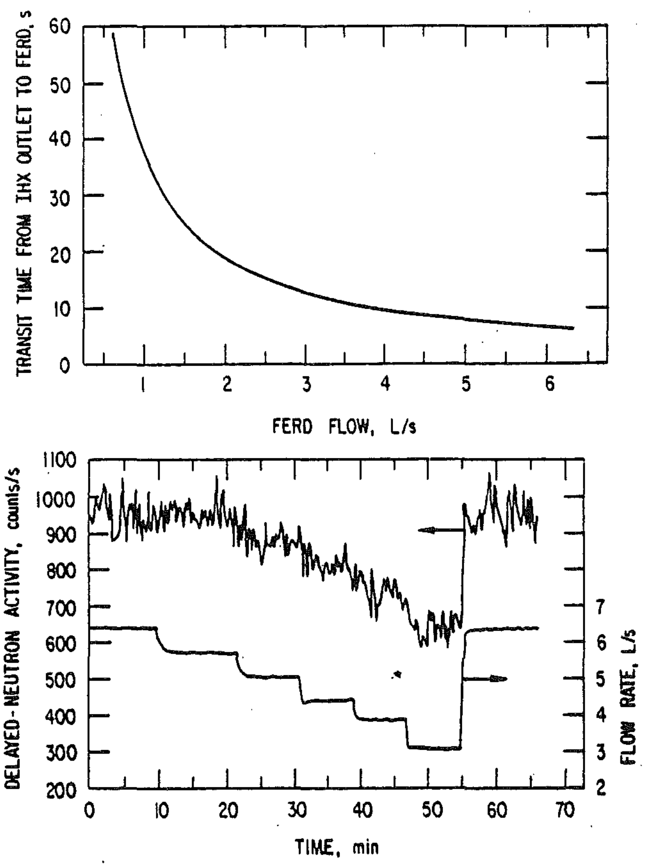

Fig. 3. FFR Tests in EBR-II
a: Transit time in FERD loop
b: Data from first test

Equation (1) was used to determine the in-core transit times for DN precursors for the two core locations that were used for the FPS irradiation. Transit times of 13.1 and 14.7 sec. were obtained for locations 5B2 and 5E4, respectively. A similar determination had been made by Fryer (8) somewhat earlier, using tramp uranium in-core as the source of DN precursors; that value was 15.3 sec. (a value with a lower precision because of the low signal count rates involved). Thus, it appeared that despite sodium mixing in the upper plenum the in-core transit time varied in a plausible manner with reactor core location. An interesting and possibly useful phenomenon in failure location.

CALCULATIONAL MODEL FOR NON-RECOIL SOURCES

When the first breached-pin tests were performed in EBR-II, the magnitude of the DN signals from them clearly indicated that release of DN precursors was not predominantly by recoil. Equation (1) could not, therefore, be used because it assumes instantaneous, or recoil release to primary sodium, with decline in DN activity occurring in the sodium in transit, not in the fuel itself. Nevertheless, FFR tests were performed and data gathered during the predefected UO_2 test and the first test of a naturally breached mixed-oxide fuel pin (2,9). In fact, using the estimates of transit time for DN's in-core obtained from the FPS tests, application of equation (1) revealed an "aging" or apparent holdup of DN precursors of 10-20 sec.

In view of this result, we went back to first principles to arrive at a model for non-recoil sources which treats parent and daughter species separately and assumes that the release of both is controlled by an Arrhenius rate equation with an effective escape coefficient λ_e. The units of λ_e are sec^{-1}, and its value is inferred from FFR tests. Incorporation of the Arrhenius model enabled us to predict DN activity in the FERD loop by the following equation:

$$A(T_{tr},\lambda_e) = \sum_{i=1}^{N} \frac{\lambda_d^i \lambda_e}{F} \times \left\{ \lambda_p^i S_p^i \frac{\exp[-\lambda_p^i(T_{tr}+t)] + \exp[-\lambda_d^i(T_{tr}+t)]}{(\lambda_d^i - \lambda_p^i)(\lambda_p^i + \lambda_e)} + \frac{\left(\frac{\lambda_p^i}{\lambda_p^i + \lambda_e} S_p^i + S_d^i\right)}{\lambda_d^i + \lambda_e} \exp[-\lambda_d^i(T_{tr}+t)] \right\} , \quad (2)$$

where

F = bulk sodium flow rate, m^3/s

λ_p, λ_d = decay constants for parent and daughter isotopes of species i, s^{-1}

S_p, S_d = fission production rates for parent and daughter isotopes of species i, atoms/s

T_{tr} = transit time from core to inlet of FERD loop, s

t = decay time within FERD loop, s

N = number of isotope groups treated ($N = 9$)

The decay time in the FERD loop is of course varied by the FFR technique. The values of the unknown model parameters are then determined with a systematic numerical algorithm that minimizes a non-linear error function of the form:

$$E(T_{tr},\lambda_e) = \sum_{k=1}^{J} \left\{ W_k(T_{tr},\lambda_e)[A_{Ek} - A_{mk}(T_{tr},\lambda_e)] \right\}^2 \quad \ldots(3)$$

where

A_{Ek} = experimentally measured DN activity at flow step k

A_{mk} = model-predicted activity [i.e., Eq. (2) at flow step k with parameter set (T_{tr},λ_e)

W_k = weighting function

J = total number of FERD flow steps.

Unfortunately , data that can be obtained from a given FFR test are not sufficient to permit solution of Eq. (3) for T_{tr} and λ_e simultaneously. The procedure, therefore, is to perform a test first with a source having a known value of λ_e. (For these tests, a recoil fission source was used, for which $\lambda_e = \infty$). Data from these tests define, through minimization of Eq. (3) the value of T_{tr}. This value is then used as an input parameter for subsequent tests in which λ_e is unknown. The reciprocal of λ_e is a physically useful parameter representing the holdup time for DN precursors in the fuel (T_h). Typical values of T_h have been 10–23 sec. for oxide fuel, and less than 3 sec. for carbide fuel.

The above model has been incorporated into a computer code GIRAFFE (10) which may be used to calculate T_{tr}, T_h, and other potentially useful diagnostic parameters. Figure 4 illustrates how the new model is used to obtain information from an FFR test involving five flow steps. The Y axis represents the measured count rates corrected for tramp background. The X axis represents the corresponding count rates predicted from the analytical model formulated above. Line I is the least-squares best fit between measured and predicted DN count rates. The Y intercept (γ) is equal to the flow-independent contribution to the measured DN signal from (γ,n) reaction in the moderator. Earlier methods (8) required this contribution to be measured directly.

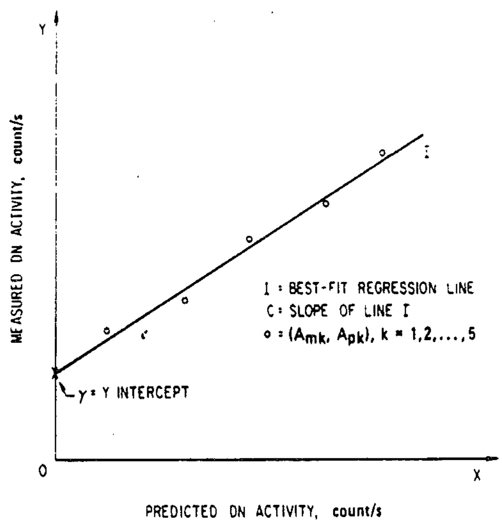

Fig. 4. Plot of five pairs of measured and predicted DN activities and best-fit line from GIRAFFE regression algorithm.

The residuals, or deviations of the measured data from the best-fit line, will indicate deficiencies in the model or anomalous data due to a drift in the counting system characteristics. The correlation coefficient, XFIT, provides a quantitative measure of the effectiveness of the model in predicting the outcome of actual FFR tests.

The slope C of the best-fit Line I contains potentially useful physical information. Theoretical considerations reveal that C is equal to $V_{Na}\alpha\varepsilon\rho V/F$, where V_{Na} is the sodium volume "seen" by the detectors, α is the fission rate in the source, ε is the efficiency of the detector arrangement (a dimensionless quantity), ρ is the fuel density, V is the fuel volume from which the DN precursors are emitted, and F is the primary-sodium flow rate.

The slope C can be used to define the equivalent recoil area of a non-recoil source in terms of the strength of the fission-product source (FPS) irradiated several times in EBR-II (5). We note first that three of the parameters in an FFR test do not change, namely, F, ε, and V_{Na}. Thus, for a given RBCB experiment we can write

$$(S_{rc})_{RBCB} = C \times \frac{\alpha_{FPS} \times \rho_{FPS} \times W_{FPS}}{\alpha_{RBCB} \times \rho_{RBCB} \times W_{RBCB}} \times (S_{rc})_{FPS} \quad \ldots(4)$$

where W is the recoil distance in the source material, and S_{rc} is defined to be the "equivalent recoil area" for the fuel. For non-recoil sources, S_{rc} provides a measure of the square centimeters of the source material that would produce a given detector response if direct recoil were the only mechanism for release.

Defined in this manner, the equivalent recoil area provides an absolute basis for comparison of different breached-pin tests, and to establish meaningful limits for assuring reactor safety and maintainability. Present practice for fast reactors that have DN monitoring systems is to set reactor-scram limits in terms of the magnitude of the DN signal. The signal magnitude, however, is a sensitive function of the isotopic holdup time for the source and the transit time to the detector. Thus, more meaningful information is conveyed, in terms of how severe or benign an exposed-fuel test may be, by a statement of its equivalent recoil area, since that value automatically accommodates differences in the release rate and transit time for the DN emitters. Typical values of S_{rc} have ranged from 7 to 91 cm^2.

The enhancement factor Ω, defined as the ratio of S_{rc} to actual area S of fuel beneath a defect, requires a knowledge of the defect area on a breached pin. Only once has this been determined accurately: for the case of the first naturally breached mixed-oxide fuel pins irradiated in EBR-II. Its enhancement factor was found to be approximately 235.

Direction of Future Research

The Arrhenius model incorporated into GIRAFFE to account for isotopic removal terms in the transport equations was not derived from physical principles. It is, rather, an empirically based model that "adjusts" the effective removal rate in a manner that best agrees with the observed flow-dependent signal behavior. Use of the resulting escape-rate coefficient in the general transport and decay model provides a particularly tractable technique for accommodating differences in the relative composition of parent and daughter species which result from gross differences in the overall holdup time, no matter how that holdup time arises physically. In this regard, use of the Arrhenius equation represents a considerable improvement over the earlier recoil model.

However, further research is needed to better understand the interplay of the complex physical mechanisms which are responsible for transport of fission products and their progeny from the fuel matrix to the sodium under the various conditions that attend sustained operation with exposed fuel in a fast reactor. Ongoing research at EBR-II is focused on reducing the considerable uncertainty that currently exists in this area (11).

SUMMARY AND CONCLUSIONS

Significant progress has been made at EBR-II in both DN monitoring and in DN signal analysis since 1977, when the DN detector FERD was removed from the trip circuit of the reactor. Irradiation of a recoil source in two core locations gave indication of a different transit time for DN precursors for different core locations. Similarly, sodium mixing in the upper plenum of the reactor was found to always produce cyclic fluctuations in DN signals which cannot be eradicated: their effect, however, appears not to be serious.

The technique of artificially altering the transit time for DN precursors to reach the detectors has proved a useful technique, in enabling the aging or holdup of DN precursors in exposed ceramic fuel to be determined. The formalism for doing this is based on an (assumed) Arrhenius release rate for the precursors, whose reciprocal is the effective holdup time in the fuel. The release rate is determined empirically by matching the activities measured at different total transit times with those calculated by means of a computer code GIRAFFE; correlation between measured and predicted activities has been good to date.

Further development of present techniques might involve allowing for different release rates for the various DN precursors considered, in line with their different measured diffusion rates in fuel and likely different volatilities. The acid test for such a future model would be comparison between measured and predicted DN signals in tests of stable defects where the temperature and/or fission rate is varied.

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Discussion

D.B. Sangodkar, RRCK:

You have said that by changing the flow in the FERD loop you can deduce the transit time from the defected fuel to the outlet of IHX. Could you use this information for localisation of failed fuel?

J.D.B. Lambert, ANL:
To a small extent, but this is not its major purpose.

P. Michaille, CEA:

1. Did you investigate all the transit times shown in the fig. 3 of your paper?
2. Which is the Reynolds number for the longest transit times (60 s on the curve)?

J.D.B. Lambert, ANL:
1. To zero flow; but mostly at 4-5 steps.
2. I don't know, but flow is turbulent.

W. Glauner, KfK:

The gap in the curve f.p. recoil range vs. nuclide number: what are the isotopes there? Is it just Bromine or Iodine?

J.D.B. Lambert, ANL:
The gap is for the low yield f.p.'s, I hope I and Br are not amongst them.

V.F. Efimenko, IAEA:

Do you have any experimental information on the change of flow rates distribution in the core subassemblies while changing the flow rate in the main piping? What is the accuracy of measuring flow rate in a subassembly?

J.D.B. Lambert, ANL:
Sorry, no. We always operate at full primary flow.

W. Glauner, KfK:

Do you think that the DND-fluctuation is also an indication for temperature fluctuations in the upper Na-plenum, which are undesirable for thermal stress reasons?

J.D.B. Lambert, ANL:
At EBR-II are limit the differences between the mixed-mean outlet temperatures of subassemblies to less than 40-50 °C. But these small differences will have an effect on upper-plenum mixing.

Comment from S. Jacobi, KfK to J.D.B. Lambert, ANL:

A comment about the fluctuations of the DND signals measured in the EBR II: Operating the KNK II 1980 with a natural defected pin fluctuations of the DND signals were observed in the following manner: Although the defect was in the eastern part of the core sometimes the DND signal east was decreasing during the DND signal west increased and inversely. Our only explanation for this are fluctuations of the hydraulic conditions in the upper sodium plenum.

M. Relic, IA:

I find that the linearity between measured and predicted DN activity in your fig. 4 is excellent. Can you give us some comments about this plot?

J.D.B. Lambert, ANL:
It is an illustration not actually real data.

D.K. Cartwright, UKAEA:

What fission product range do you use to predict recoil areas from defects?

J.D.B. Lambert, ANL:
For mixed-oxide fuel 10 μm.

PLAN MARGO

by P. Michaille

PROGRESSIVE ENHANCEMENT OF THE CONTINUOUS DND LEVELS

TOLERATED IN THE REACTORS RAPSODIE & PHENIX

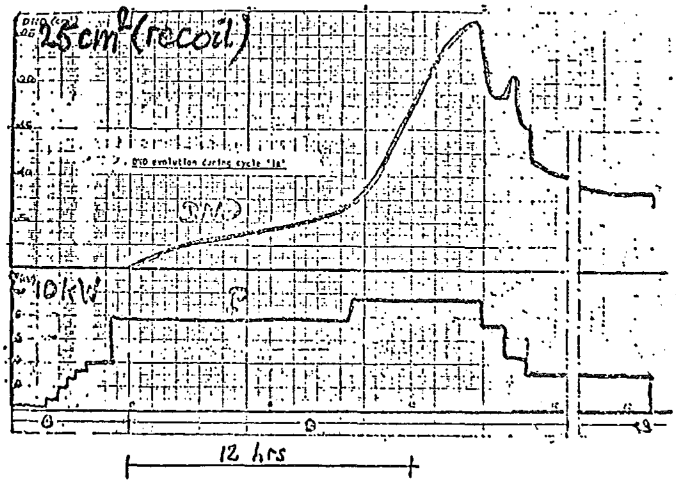
BASED ON : – THE IN-PILE MEASUREMENTS & PIE (FUEL RELEASE)
- THE VOLGA PROGRAM
  - IN-LOOP PRECURSORS
  - IN-PILE CONFIRMATION

IN-LOOP VOLGA PROGRAM

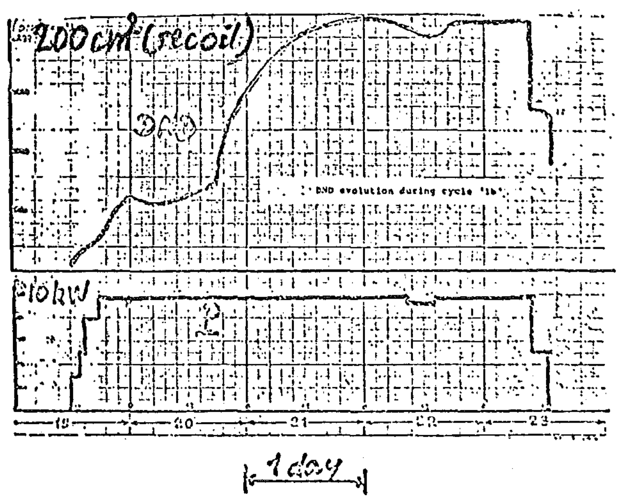
COMPARISON OF THE EVOLUTION OF THE DND & GAS SIGNALS,

OF THE CLADDING BREACH, & OF THE RELEASE OF FUEL

N°	PARTICIPANTS	B.U.	TYPE	INITIAL DEFECT	INSTRUMENTATION
IV	F-D	HIGH	SNR	30 x 1 MM	SIMPLE LOOP
III	F-D	HIGH	SNR	2.5 x 1 MM	SIMPLE LOOP
I	F-D	HIGH	SPX	NOT THRU	SIMPLE LOOP
VI	F	HIGH	SPX	NOT THRU	DOUBLE LOOP
V	F-D	LOW	SPX	Ø 1 MM	DOUBLE LOOP
0	F-I	FERTILE	PX		DOUBLE LOOP



IN-LOOP EXPERIMENT N°III – EVOLUTION OF THE DND SIGNALS UP TO 25 CM²(RECOIL)



IN-LOOP EXPERIMENT N°III – EVOLUTION OF THE DND SIGNALS UP TO 200 CM²R

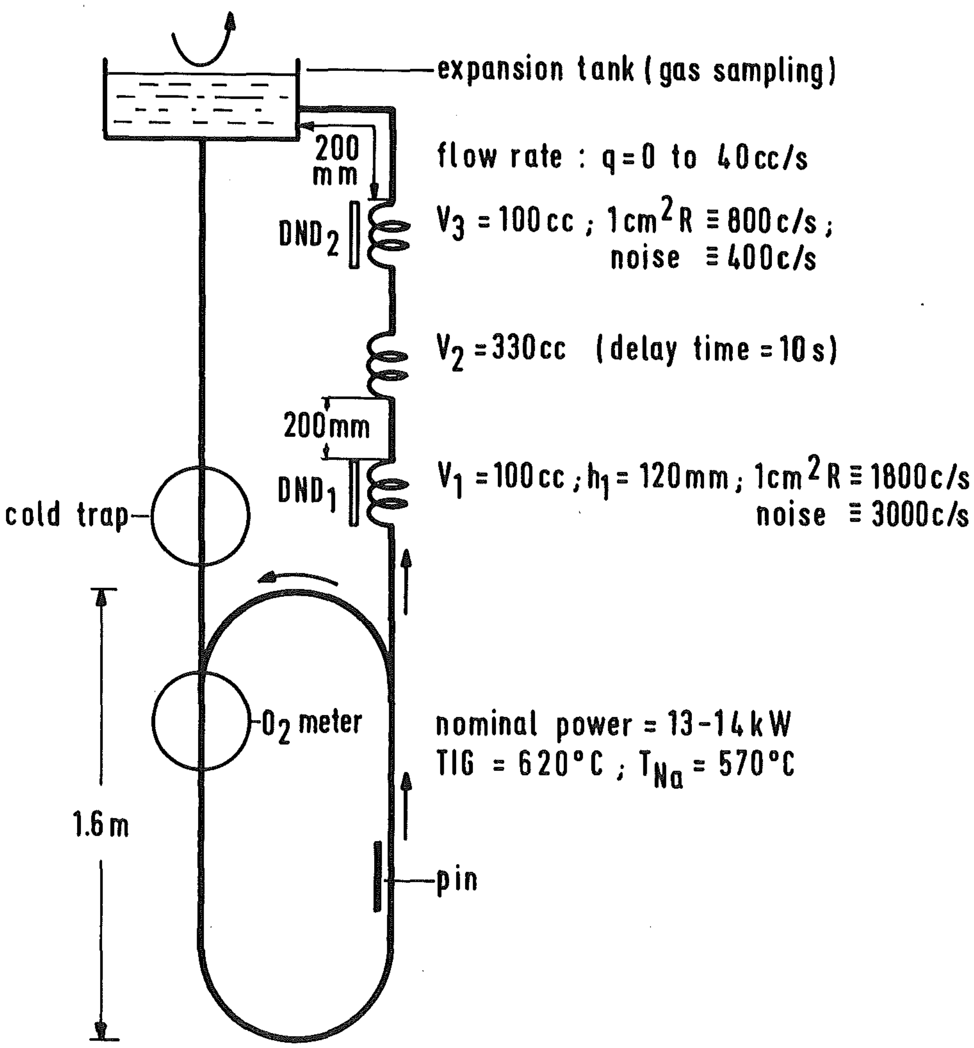
PROGRAMME PARKA

by P. Michaille

STUDY OF THE AMPLITUDE & COMPOSITION OF THE DND SIGNAL

SOURCE	OBJECTIVE
COLD METALLIC TUBE	CALIBRATION SOURCE
COLD CERAMIC SAMPLE	EFFECT OF SURFACE
SODIUM BOND	TEMPERATURE-ACTIVATED MECHANISMS
GAS BOND	INTERPRETATION OF THE DND PEAKS

Instrumentation : Double Circuit Loop



Design of the Double Circuit Loop

Discussion

D.K. Cartwright, UKAEA:

In experiment III of your Siloe work you stated that the increase in DN recoil area was supported by gas measurements. Does this mean that the gas release depends on the size of the slit?

P.A. Michaille, CEA:
Not of the size of the slit but of the conductance of the defect. We observe a few hours after the DND increase a decrease of the $|\alpha|$ slope ($|\alpha| = \frac{\log R/B}{\log \lambda}$) from ~ 1.0 to 0.5 that was interpreted to a good access to the central hole of the pin.

J.D.B. Lambert, ANL:

When you quote R/B, B is calculated, yes?

P. Michaille, CEA:
Yes.

Comment from G. Schmitz, KfK to P. Michaille, CEA:

The radioactive fission products are measured in the sodium or in the cover gas. Then the release rates are calculated. Also the birth rates in the total fuel pin are calculated. At last the R/B-ratios are plotted versus the appropriate decay constants.

N. Sekiguchi, PNC:

Have you developed any computer code which is able to interpret FP and fuel material behavior not only within fuel pin but also in coolant and loop in order to analyse the results from your in-pile loop?

P. Michaille, CEA:
We have a code for fission gas release from the fuel which is reviewed from recent measurements at very high burn-up. For the fission products transformation, we can't make efforts in the loop because we think it is different from pool reactors. Therefore, we make measurements at Phenix.

D.E. Mahagin, HEDL:

What exactly is the MARGO program?

C. Berlin, CEA:

It is not a program.

It is a plan to make concrete the will to go ahead with failures considering the DND levels reached, as well from natural causes as from artificial defects.

J.D.B. Lambert, ANL:

Are the counters easily replaceable?

C. Berlin, CEA:

Yes.

FUEL FAILURE DETECTION AND LOCATION IN FAST BREEDER REACTORS

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ABSTRACT

In case of failures of fuel claddings different mechanisms of fission product releases must be taken into account such as

- Releases into the coolant by recoil.
- Releases into the coolant by diffusion.
- Releases by recoil into open fuel pores and on into the coolant.

The physical models are described and mathematical correlations between release rates, birth rates and decay constants of different radioactive nuclides are given. Then a graphical method to analyse a given case of fission product release is described and different equations for the fissuring factor k are given depending on the predominant release mechanism. Three inpile experiments with artificially defected fuel pins have been analysed with the result, that in two cases there was a diffusion controlled release mechanism associated with J diffusion corefficients between $2 \cdot 10^{-7}\ cm^2s^{-1}$ and $6 \cdot 10^{-6}\ cm^2s^{-1}$. In the third case the release mechanism was controlled by recoil into open fuel pores and on into the coolant associated with an open porosity surface of 864 cm^2 in contrast to a geometric defect area of 2 cm^2. At last the code FICTION I (Fission Product Signal Calculation) is described. This code calculates the expected count rates of the DN detectors in KNK II, on the one hand, and in the SILOE loops, on the other hand.

INTRODUCTION

If leaks occur in the claddings of individual fuel rods during operation of LMFBR's fission products and possibly, also fuel may be released into the coolant.[1-11] When analyzing such phenomena, different mechanisms of fission product releases must be taken into account.[3, 1o, 12, 13, 14, 15]

- Releases into the coolant by recoil.
- Releases into the coolant by diffusion.
- Releases by recoil into open fuel pores and on into the coolant.
- Releases by diffusion into the plenum and on into the coolant.
- Releases into the coolant by knock-out mechanisms.
- Releases by knock-out mechanims into open fuel pores and on into the coolant.

RELEASES INTO THE COOLANT BY RECOIL

In releases of radioactive fission products caused by recoil mechanisms it is assumed that fission products formed in a layer of the thickness L_i right underneath the defect area F are spontaneously released into the coolant by recoil in a fraction δ.[12] After an equilibrium condition has become established, the following equation holds:

$$R_i^R = \delta \cdot F \cdot L_i \cdot \lambda_i \cdot \nu_i \cdot S \qquad (1) \text{ Model A}$$

R_i^R	$[s^{-2}]$	Activity release rate of nuclide i into the coolant by recoil.
δ	$[1]$	Geometric escape coefficient.
F	$[cm^2]$	Geometric defect area.
L_i	$[cm]$	Recoil length of nuclide i.
λ_i	$[s^{-1}]$	Decay constant of nuclide i.
ν_i	$[1]$	Cumulative fission yield of nuclide i.
S	$[s^{-1} cm^{-3}]$	Specific fission rate.

RELEASES INTO THE COOLANT BY DIFFUSION

If radioactive fission products are released by diffusion, the fission products generated are assumed to migrate through the fuel up to the fuel rod defect and on into the coolant as a result of a concentration gradient.

Under the additional assumptions of a linear, one-dimensional geometry and the absence of any repercussions on diffusion processes in the fuel of the concentration of fission products in the coolant, it follows, after an equilibrium condition has become established.[13]

$$R_i^D = F \cdot \nu_i \cdot S \cdot \sqrt{\lambda_i} \cdot \sqrt{D_i} \qquad (2) \text{ Model B}$$

R_i^D	$[s^{-2}]$	Activity release rate of nuclide i into the coolant by diffusion.
D_i	$[cm^2 s^{-1}]$	Diffusion coefficient of nuclide i in the fuel region.

RELEASES BY RECOIL INTO OPEN FUEL PORES AND ON INTO THE COOLANT

For releases of radioactive fission products by recoil mechanisms into open fuel pores and on into the coolant, the following model is assumed to be valid: Fission products formed in a layer of the thickness L_i and an area F^P, which surrounds the open porosity, are released spontaneously into the free pores by recoil in a fraction δ. Fission product noble gases also released carry part of the fission products from the open porosity through the rod defect into the coolant. An Arrhenius type release is assumed to prevail.[10] After an equilibrium has been established, it is found that

$$R_i^{RP} = \frac{\lambda_i \cdot \lambda_e}{\lambda_i + \lambda_e} \cdot \delta \cdot L_i \cdot F^P \cdot \nu_i \cdot S \qquad (3) \text{ Model C}$$

R_i^{RP}	$[s^{-2}]$	Activity release rate of nuclide i by recoil into the open porosity and on into the coolant.
λ_e	$[s^{-1}]$	Escape coefficient.
F^P	$[cm^2]$	Surface of open porosity in the region of the fuel rod defect.

The other three release mechanisms mentioned above are not at present treated mathematically and numerically.

PLOTTING INDIVIDUAL RELEASE MECHANISMS

When analyzing the measured results obtained in operation with defective fuel rods, or fuel rods artificially made defective, in reactors or in-pile test loops, it is advisable to convert the measured fission product activity concentrations to the ratio of radioactivity release rates to radioactivity birth rates (R/B ratios) of the individual fission product nuclides, taking into account the fuel rod and systems data. Since the functional relationships between the R/B ratio and the decay constants of different radioactive fission product nuclides vary as a function of release mechanisms, it is possible to determine the specific type of fission product release by plotting these relationships. In an equilibrium case, it applies to

Model A:

$$\frac{R_i^R}{B_i} = \frac{R_i^R}{\lambda_i \cdot \nu_i \cdot S \cdot V_B} = \frac{\delta \cdot L_i \cdot F}{V_B} \qquad (4)$$

B_i	$[s^{-2}]$	Radioactivity birth rate of nuclide i in the fuel rod.
V_B	$[cm^3]$	Fuel volume of the defective rod.

Model B:

$$\frac{R_i^D}{B_i} = \frac{F \cdot \sqrt{D_i}}{V_B} \cdot \lambda_i^{-1/2} \tag{5}$$

Model C:

$$\frac{R_i^{RP}}{B_i} = \frac{\delta \cdot L_i \cdot F^P}{V_B} \cdot \frac{\lambda_e}{\lambda_i + \lambda_e} \tag{6}$$

Fig. 1 shows three typical plots:

On the abscissa, the decay constant λ_i of the individual nuclides has been plotted on a logarithmic scale. On the ordinate, also on a logarithmic scale, the R_i/B_i ratio of the respective nuclides has been plotted.

Curve A shows the expected behavior of the R/B ratio for case A: release into the coolant by recoil. The following values were used for calculation: $\delta = 0.25$[12]; $L_i = 6 \times 10^{-4}$cm (this value was assumed to be identical for all the nuclides considered); $F = 2\ cm^2$ and $V_B = 2\ cm^3$. This curve is a horizontal straight line.

Curve B shows the expected behavior of the R/B ratio for case B: release into the coolant by diffusion. For this case, the following values were used for calculation: $F = 2\ cm^2$; $D = 1 \times 10^{-6}\ cm^2 s^{-1}$ (this value was assumed to be identical for all the nuclides considered), and $V_B = 2\ cm^3$. This curve is a straight line with the slope of -1/2.

Curve C shows the expected behavior of the R/B ratio for case C: release by recoil into open fuel pores and on into the coolant. For this case, the following values were used for calculation: $\delta = 0.25$; $L_i = 6 \times 10^{-4}$ cm; $F^P = 400\ cm^2$; $V_B = 2\ cm^3$; $\lambda_e = 1 \times 10^{-3} s^{-1}$. This curve shows three typical sections. Section I is characterized by $\lambda_e \gg \lambda_i$. As a consequence, Eq. (6) changes into

$$\frac{R_i^{RP}}{B_i} = \frac{\delta \cdot L_i \cdot F^P}{V_B} \quad \text{in case of } \lambda_e \gg \lambda_i \tag{7}$$

In this section, the curve is a horizontal straight line. Section II is characterized by λ_e and λ_i being of comparable orders of magnitude. This is a part of the curve with decreasing slope, as λ_i increases. Finally, Section III is characterized by $\lambda_e \ll \lambda_i$. As a consequence, Eq. (6) changes into

$$\frac{R_i^{RP}}{B_i} = \frac{\delta \cdot L_i \cdot F^P}{V_B} \cdot \lambda_i^{-1} \text{ in case of } \lambda_e \ll \lambda_i \qquad (8)$$

In this section, the curve is a straight line with the slope of -1.

K-FACTOR

The fissuring factor, k, has been defined in[7] as the ratio of the effective recoil area to the geometric defect area, in which case fission product release by recoil was assumed. For a specific nuclide it can be expressed as follows:

$$k_i = \frac{R_i^G}{R_i^R} \qquad (9)$$

k_i [1]	Fissuring factor relative to nuclide i.
R_i^G [s^{-2}]	Measured radioactivity release rate of nuclide i.
R_i^R [s^{-2}]	Radioactivity release rate of nuclide i calculated by means of a recoil model.

If the real radioactivity release is diffusion controlled, R_i^G can be expressed as follows in accordance with Eq. (2):

$$R_i^G = F \cdot \sqrt{D_i} \cdot \lambda_i^{1/2} \cdot \nu_i \cdot S \qquad (10)$$

With Eqs. (10) and (1), Eq. (9) is transformed into

$$k_i = \frac{1}{\delta \cdot L_i} \cdot \left(\frac{D_i}{\lambda_i}\right)^{1/2} \qquad (11)$$

Eq. (11) can be converted into

$$D_i = L_i^2 \cdot \delta^2 \cdot \lambda_i \cdot k_i^2 \qquad (12)$$

In deriving Eq. (11) it has been assumed that the fuel will not change chemically after the occurrence of the defect. However, this condition does not always exist if sodium is used as a coolant because, when certain oxygen concentrations in the fuel and the sodium are exceeded, part of the fuel will be converted into Na_3MO_4 or similar compounds. The letter M in this case stands for metal, i.e., U or Pu. Since also the density of the fissile material taken into account changes in this transformation. also different specific fission ratios must be substituted in Eq. (10) and (1). For this general case one obtains:

$$k_i = \frac{S^R}{\delta \cdot L_i \cdot S^B} \cdot \left(\frac{D_i^R}{\lambda_i} \right)^{1/2} \tag{13}$$

S^R [$s^{-1} cm^{-3}$]	Specific fission rate in the Na_3MO_4 reaction layer.
S^B [$s^{-1} cm^{-3}$]	Specific fission rate in MO_2 fuel.
D_i^R [$cm^2 s^{-1}$]	Diffusion coefficient of nuclide i in the reaction layer.

Eq.(13) can be changed into:

$$D_i = L_i^2 \cdot \delta^2 \cdot \lambda_i \cdot \left(\frac{S^B}{S^R} \right)^2 \cdot k_i^2 \tag{14}$$

However, if the actual radioactivity release is controlled by recoil into open pores, it holds for R_i^G according to Eq.(3):

$$R_i^G = \frac{\lambda_e}{\lambda_i + \lambda_e} \cdot \delta \cdot L_i \cdot F^P \cdot \lambda_i \cdot \nu_i \cdot S^B \tag{15}$$

With Eq. (15) and (1), Eq. (9) changes into

$$k_i = \frac{\lambda_e}{\lambda_i + \lambda_e} \cdot \frac{F^P}{F} \tag{16}$$

If this type of release prevails, it is assumed that there has been no chemical conversion of the fuel or that such (Na_3MO_4) reaction layer has been reduced again.

RESULTS OF THE FIRST SILOE EXPERIMENTS

Fig. 2 to 4 show R/B values of the radioactive I isotopes calculated from measured values and experimental data.[14] When calculating the R/B value of I-137 from the count rate of the DND detectors, a recoil type release mechanism was assumed. For this reason, this R/B value is underestimated in other types of releases. More precise calculations are planned for the future. Fig. 3, in addition, shows R/B values of radioactive Kr and Xe isotopes.

As in Fig. 1, the abscissa is a logarithmic plot of the decay constant, λ_i, of the different nuclides, whereas the ordinate is a logarithmic plot of the R_i/B_i ratio of the corresponding nuclides.

The relative positions of the different R_i/B_i values in Fig. 2 and 3 indicate a diffusion controlled release mechanism. The excessively high R_i/B_i value of I-132 can be explained by the fact that the Te-132 precursor nuclide has a much longer half-life and is probably also released by diffusion. Table 1 indicates k(I-137) factors and I diffusion coefficients calculated at specific points in time. Table 1 replaces Table 4 in [14] because, in that older Table, Eq. (12) had been used instead of Eq. (14) in calculating the diffusion coefficients. Moreover, the k(I-137) factors had to be recalculated for the S4 experiment.

The positions of the different R_i/B_i values in Fig. 4 indicate a release mechanism by recoil into open pores and on into the coolant. For $\lambda_e >> \lambda_i$, i.e., $\lambda_i < 3 \times 10^{-3} s^{-1}$, a R_i/B_i value of 1.56×10^{-2} is found. With the data of $\delta = 0.25$. $L(I) = 4.7 \times 10^{-4}$ cm and $V_B = 6.51$ cm^3, it follows from Eq. (7): $F^P = 864$ cm^2 as against a defect area of $F = (2.0 \pm 0.4)$ cm^2 as determined optically after irradiation. From the R/B(I-137) value it is possible to estimate λ_e from Eq.(6) and (7) as $\lambda_e > 7 \times 10^{-3} s^{-1}$.

CODES ESTABLISHED TO DATE

To calculate the expected count rates of the DN detectors in KNK II, on the one hand, and in the SILOE loops, on the other hand, the FICTION I (Fission Product Signal Calculation) code was written in the FOCAL-8 programming language for a PDP-8 computer made by the Digital Equipment Corporation.[16] Recoil of the fission products into the coolant was assumed as the release mechanism. In addition, the k-factor was programmed. The following formulae were used:

$$N_i = \delta \cdot L_i \cdot F \cdot k \cdot \lambda_i \cdot \nu_i \cdot \mu_i \cdot S \qquad (17)$$

N_i [s^{-2}] Neutron activity release rate of nuclide i into the coolant.

μ_i [1] Emitted neutrons per nuclide disintegration.

$$C = \eta \cdot \rho^N \cdot V^P \cdot \sum_{i=1}^{10} \left(\frac{N_i}{Q} \cdot e^{-\lambda_i \cdot t^1} \right) \qquad (18)$$

C [s^{-1}] Count rate of DN-detector

η [1] Neutron sensitivity of DN-detector

V^P [cm^3] Volume of Na sample vessel

Q [g/s] Sodium flow.

ρ^N $[g/cm^3]$ Sodium density.

t^1 $[s]$ Transit time of sodium from the defect point to the DN detector.

For short sodium recirculation times it holds that

$$C^R = \eta \cdot \rho^N \cdot V^P \cdot \sum_{j=0}^{m} \cdot \sum_{i=1}^{10} \left(\frac{N_i}{Q} \cdot e^{-\lambda_i (t^1 + j \cdot t^u)} \right) \qquad (19)$$

C^R $[s^{-1}]$ Count rate of DN detector taking account of sodium recirculation

t^u $[s]$ recirculation time of sodium

with

$$m = \frac{5 \cdot \ln 2}{t^u \cdot \lambda \, (Br\text{-}87)}$$

The variable input quantities include F, K, S, η, ρ^N, V^P, Q, t^1 and t^u. The following nuclides are taken into account: Br-87, Br-88, I-137, Br-89, Rb-93, I-138, As-85, Br-90, Rb-94 and I-139. The fixed parameters are taken from: δ = 0.25 and L_i from [12], ν_i and μ_i from[17].

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Date	Exp	Power per rod length [W/cm]	Na-temp [°C]	I-diff coeff [$cm^2 s^{-1}$]	k-factor for I-137 [1]	F [cm^2]	Days in operation
17 3 76	S2	213	540	$2{,}9 \; 10^{-6 \pm 0{,}5}$	51	0,7	20
19 3.76	S2	413	570	$4{,}1 \; 10^{-7 \pm 0{,}5}$	19	0,7	22
28.5 77	S4	381	560	$1{,}2 \; 10^{-6 \pm 0{,}5}$	33	1,8	2
3.6 77	S4	431	560	$5{,}9 \cdot 10^{-6 \pm 0{,}5}$	72	2,5	4
9 6 77	S4	394	560	$5{,}4 \; 10^{-6 \pm 0{,}5}$	69	2,5	6
27 6 77	S4	163	380	$2{,}5 \cdot 10^{-7 \pm 0{,}5}$	15	2,5	13
3.7 77	S4	309	470	$1{,}1 \cdot 10^{-6 \pm 0{,}5}$	31	2,5	19
7 7 77	S4	397	560	$3{,}5 \cdot 10^{-6 \pm 0{,}5}$	56	2,5	23

Table 1: Diffusion Coefficients and Calculated k-Factors in the SILOE S2 and S4 Experiments

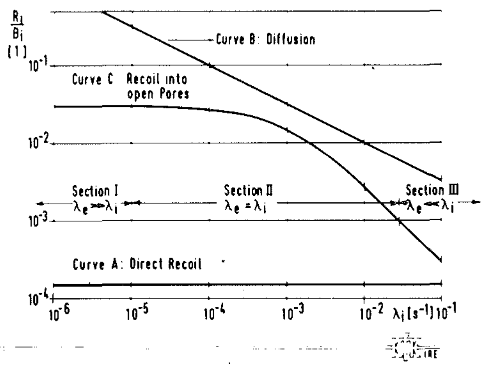

Fig. 1: Examples of Analyses of Fission Product Release Mechanisms

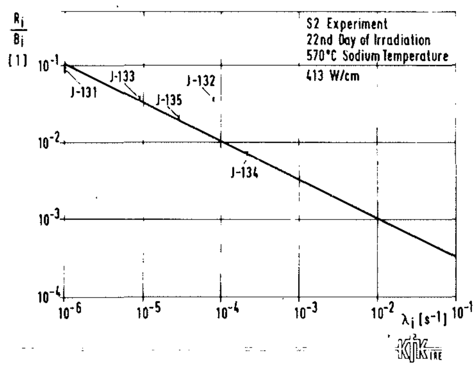

Fig. 2: R/B-Ratios of the J Isotopes versus their Decay Constants

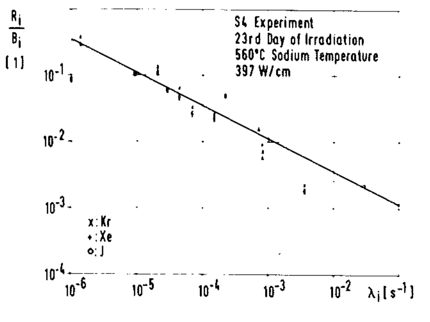

Fig. 3: R/B-Ratios of the Kr, Xe and J Isotopes versus their Decay Constants

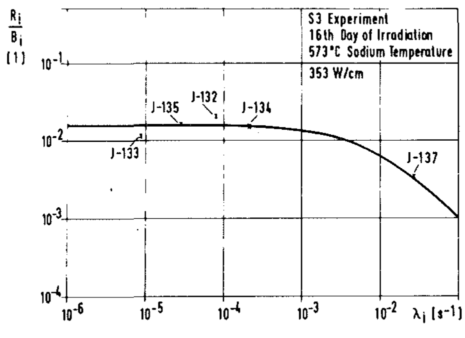

Fig. 4: R/B-Ratios of the J Isotopes versus their Decay Constants

Discussion

P. Michaille, CEA:

Is it possible to take into account for the model as well fission gas as volatile (iodines) since M. Mahagin said that diffusion mechanisms are different?

G. Schmitz, KfK:
I think that the presented models are applicable for releases of fission gases as well as for volatile fission products. But the controlling parameters may be different.

J. Dauk, IA:

1. Do you think that your model is applicable to reactors, considering that the isotopes you measured are important as precursers in reactor?
2. Did you measure the isotopes by on line γ-spectrometry in sodium?

G. Schmitz, KfK:
1. Yes.
2. During our SILOE-experiments we measured the activity-concentrations of J-131, J-132, J-133, J-134 and J-135 in the sodium by a gamma-telescope. This telescope did not work continuously.

M. Relic, IA:

You have in your model two unkown parameters F^p and λ_e. How can you calculated them separately?

G. Schmitz, KfK:
The two parameters F^p and λ_e have to be calculated simultaneously.

P. Michaille, CEA:

Is it possible to use your code to predict the DND signal for a given defect?

G. Schmitz, KfK:
In principle yes. In case of a defect with a $Na_3(U, Pu)O_4$ layer I would prefer the model "releases into the coolant by diffusion". In case of a defect without such a layer I would prefer the model "release by recoil into open fuel pores and one into the coolant".

M. Relic, IA:

How do you calculate F^p and λ_e from your experimental curves?

G. Schmitz, KfK:
Regarding to fig. 4 in case of $\lambda_i < 10^{-5}\ s^{-1}$ the R/B-ratio is about $1.5 \cdot 10^{-2}$ and constant. Taking equation (7) you can calculate F^p. Then you can calculate λ_e by equation (6), taking the values of R/B and λ for J-137 out of fig. 4.

Comment from H. Feuerstein, KfK to G. Schmitz, KfK:

I wish to point the fact that all gas measurements in the covergas must also consider the fact of the degassing delay from the loop sodium to the covergas.

D.K. Cartwright, UKAEA:

Is the escape coefficient empirical or is it derived from physical parameters?

G. Schmitz, KfK:
The escape coefficient is empirical.

PRESENTATIONS AND DISCUSSIONS

OF SESSION V:

FUTURE PROGRAMS

Overview of Japanese Status and Future Program for Development of FFDL System

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Mitsubishi Electric Corporation

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ABSTRACT

The future program of FFDL system development in Japan is outlined with the achievements resulted from the activities on the subject over these 12 years. Major concerns are directed to evaluation of FFD response in relation to the behavior of failed pin, and development of reliable locating system from the viewpoint of compatibility of plant operational safety and availability. Another interest is taken in sipping method, as one of means of location of failed subassembly. The program of its development is described for a particular attention for an important future work in this field in Japan.

INTRODUCTION

The Japanese experimental fast reactor, JOYO, reached its first criticality on April 1977. The 50 MWt normal operation began in October 1978 and successfully completed in February 1979. Normal operation of 75 MWt is at present underway without any fuel failure as well as major components trouble, and will be continued until the end of 1981. The MK-II program of a core of 100 MWt power will be started in the beginning of 1982.

Meanwhile, the project of the proto-type fast breeder reactor, MONJU, 280 MWe of power, is going on with great effort in PNC in cooporation with the utilities and the manufacturers. The licensing procedure for start of construction is now being intensively conducted. The final principal design specification is also currently under consideration, on the basis of the supporting research and development works. The major part of the works will be achieved within a couple of years.

The development program of FFDL system (1) with other components and system in the area of instrumentation and control in LMFBR has been planned, inplemented, and succeeded in line and accordence with the operation and the design of the Japanese LMFBRs in these 10 years.

Several future under takings are projected in the scope of FFDL system, as presented in Table 1 and 2. It is a matter of practical importance to set up a distinct guide line for continuation of operation of a reactor with failed fuels. By accumulating many experiences of fuel failure in the reactor, a proper guide line for plant operation will be obtained. On the other hand, however, a study of FP release from failed pin and transport in primary cooling system is considered to support quantitative analysis of the relation between FFD response and fuel failure. In this meaning, Japanese approach to the subject at present is directed to computer code and modelling of FP behavior. In-pile loop testing program in JOYO Mk-II is expected to verify these analyses based on theoretical investigation and the other in-pile experiments. As another future program of FFDL system development, major interest is paid for sipping method as well as for gas tagging. Development of in- and ex-vessel inspection system based on sipping method(2) is discribed in a little detail in this paper.

CODE AND MODELLING

Computer Codes for analysis of FP behavior and FFD response have been developed (3) and run to be tested in the case of JOYO and MONJU. As one example of codes, an outline of FPT-1 is introduced as follows.

The code is used to analyze release and transport process of FP gas released from the fuel pellets into liquid metal coolant and to cover gas region. This is based on the FP gas diffusion model by Booth, evaluating a release rate of FP gas from fuel pellets. It is constructed by modelling FP behaviors which greatly affect the response characteristics. These are shown as follows:

- Rate of FP released from pellet.
- Rate of FP released from fuel pin
- Rate of FP, released in the coolant, transported to cover gas
- Clean up or trapping rate of FP in cover gas system.

This is able to be used for calculating response of FFD system that detects fuel failure by monitoring cover gas radioactivity.

Table. 1 Planned Program of FFDL System Development in Japan.

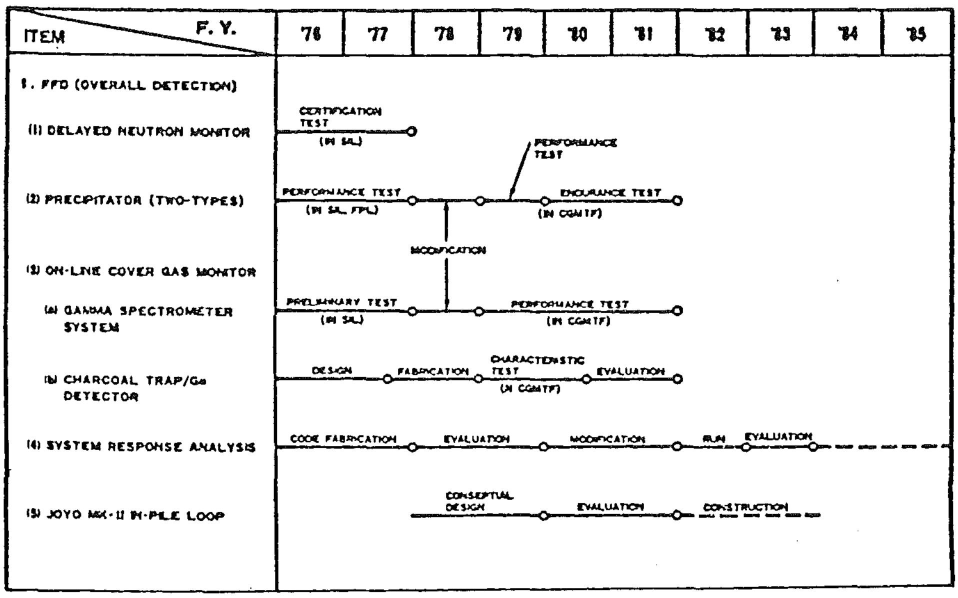
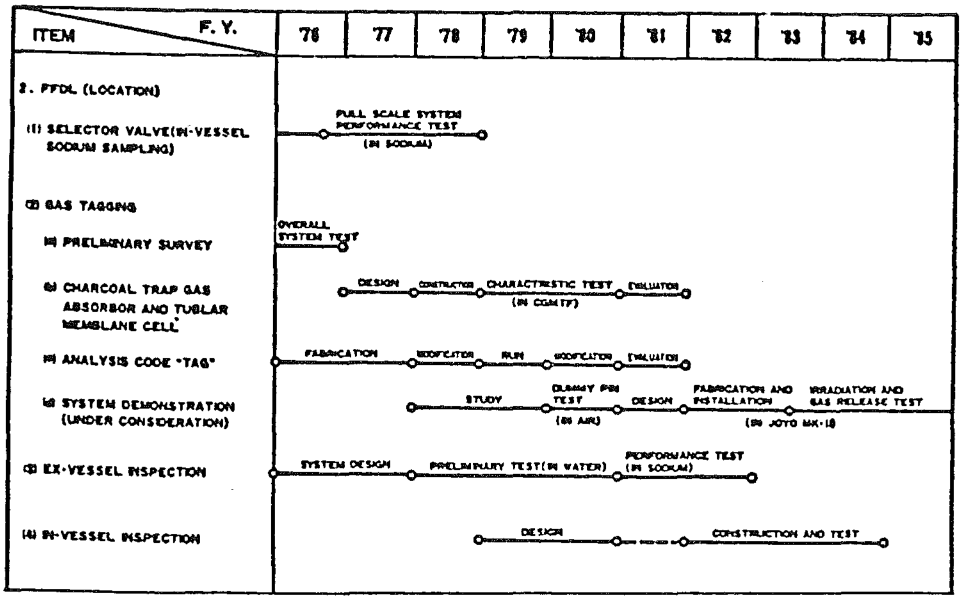


Table. 2 Planned Program of FFDL System Development in Japan (Cont'd)



The code is verified and well compared with the experimental results of GETR in the U.S., where fuel failure is artificially produced in an in-pile sodium loop.

It is expected and recommended that more experimental results are obtained and examined to improve reliability of input parameters not only to the code but also to other ones which are under developed, while introducing more detailed analytical models, since release and transport phenomena of FP are still left much to be investigated.

The computer codes, being well modified, will be of helpful to evaluate operation guide and system design concerning FFD in JOYO and MONJU.

TAGGING GAS IRRADIATION AND SYSTEM VERIFICATION TEST

It is necessary from technical aspect to make intensive study of feasibility and on the other hand, cost evaluation of tagging gas system. There are several uncertainties on the feasibility of tagging gas system in LMFBR. It is, therefore, important and desirable to verify system performance and enduarance by overall system test and by accumulating experiences in actual fast reactor. Prior to and apart from verification of the system, uncertainties which are particularly influencial to system capability must be fully investigated.

Isotopic change of Xe and Kr due to burn-up and behavior of realeased gas from failed pin in primary system, which are left to be studied will be analysed and evaluated, by using the irradiated test subassemblies and the other special device in JOYO Mk-II. Whether the future program will be implemented or not, however, are still under consideration.

SIPPING METHODS

Two types of inspection system based on sipping method and the like have been designed, and are being fabricated for test in sodium one is in-vessel failed fuel detection and location system(IV-FFDL) which is used to scan all the subassembly with operation of rotating shield plug during reactor shutdown.

Another type is named ex-vessel failed fuel detection system(EV-FFD), the purpose of which is expected to identify the failed subassembly out of reactor vessel after localizing failed fuel in reactor vessel.

1. In-Vessel Failed Fuel Detection and Location System (IV-FFDL)

The method of IV-FFDL is to suck up the coolant sodium (Na) from the upper part of the fuel subassembly (S/A) and extract the gaseous fission products (FP) contained therein by Argon gas in order to measure its radioactivity and check whether there is any failed fuel element in the S/A.

1) Principle of Detection

Liquid Na flows on the Fuel Exchange Mode, and the pressure is given thereby keeping a balance with the gas pressure within the fuel pin as shown in Fig. 1.

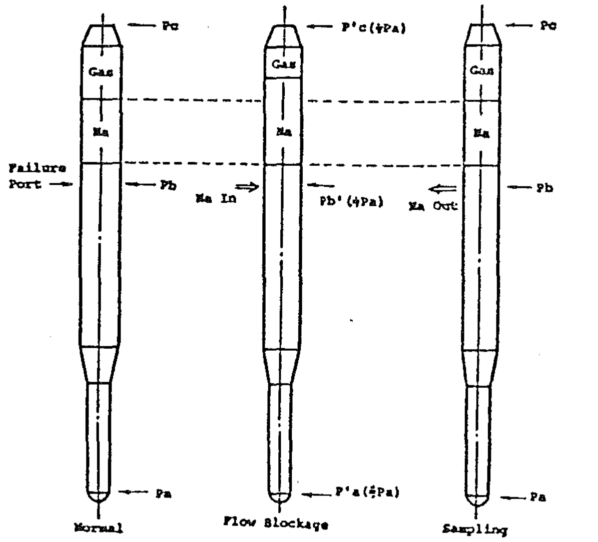


Fig 1. The Pressure Applide on the Failed Fuel Pin

(1) With the sipping port fixed on the top of the S/A, Ar gas pressurized and then the flow blocked, it will turn out as follows; (following to Fig. 1)

$$Pa' = Pa,\ Pb' \fallingdotseq Pa > Pb,\ Pc' \fallingdotseq Pa > Pc$$

The volume of the gas plenum is then reduced to 1 - (Pb/Pb') with the subsequent entry of Na to the extent that the gas plenum is reduced to balance the pressure.

(2) When the sipping port is lifted from the top of the S/A, the pressure balance of the fuel pin is restored to the initial state. It is, therefore, presumed that the same volume of Na as is entered when the flow is blocked will be released. It is also presumed that the released Na includes FP gas which is mixed in the pin.

(3) By lifting the sipping port from S/A and, at the same time, sampling Na into the tank of the device, FP gas of the failed pin can be induced into the device.

(4) FP gas is to be extracted from Na by flowing the Carrier gas (Ar) into the sampled Na. A fairly good percentage of the extraction is expected. For example, according to the ANL experiment, it is reported that 99% of Xe can be extracted in 55 seconds when 1ℓ/min Argon is blown into 500 cc Na.

(5) The failure can be detected by measuring β & γ radiation of carrier gas after inducing carrier gas which includes FP gas into the gas sampling chamber of the radiation detector.

The detection procedure is, first of all, to measure the background level (B) of radioactivity of rare gases in Na at the place. Where there is no posibility of failure, such as, at the radial blanket S/A, and then to measure the radiation (C) of the centre S/A one by one.

Judgement regarding to the failure is made by a comparison of the above two with each other. The following is the formula of the judgement criterion.

$$C - B > n\sqrt{C} + \varepsilon B$$

n ; The statistical weight and the case is $n \geq 1$ but for the first time being make it $n = 1$, and find an appropriate value after some accumulated experience.

ε ; Errors in measurement, that is to say, the uncertainty with regard to the Na sampling volume or the reproducibility of extraction of FP gas by Ar gas. The tentative value used is $\varepsilon = 0.05$

2) Quantitative Study of the Sipping Method.

(1) Estimation of the Background

The background includes FP from the failure of one pin, and FP from natural Uraniums in the coolant Na and contaminating to the surface of the fuel pin.

- FP gas from the failure of a pin

On the condition that 10% each inventory of Xe and I is transfered into the coolant, the FP gas background is obtained as following.

In the case of rare gas FP, they are carried into the coolant Na by its flow to be spread evenly and are decayed or transfered to the cover gas of reactor. Of course they are supplemented by the creation from the parent nuclide.

The quantitative calculation shows the presense of FP gas in the coolant after two days from shutdown. The background is about 10 Ci in total.

Fig 2 shows variation of the background in relation to the progressing days after the shutdown.

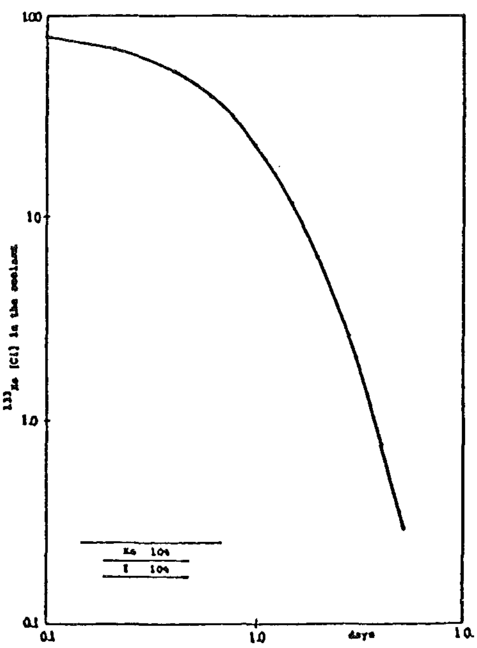


Fig 2. Decay Curve of Xe in the Coolant

- FP from Tramp Fuels

Fission materials tramped in the coolant and contaminated on surface of the fuel pins are roughly estimated to be less than 10 mg corresponding to ^{235}U by the information of the other experiments.

Then this contribution is negligible small compared to FP from the failure of a pin.

(2) The Effect of the Blockage of S/A Flow.

By contacting the sipping port on the top of the S/A, flow of Na is blocked and then the pressure rise at the failure port is resulted. It is estimated that the pressure change is about 0.1 kg/cm^2 which is effective for releasing FP gas from failed fuel pin.

On the other hand, the temperature rise of the fuel pin accompanied with the blockage of the coolant flow is estimated to be about 0.25°C. This rise does not effect on the healthy of fuel pin.

2. Ex-Vessel Failed Fuel Detection System

Ex-Vessel Failed Fuel Detection System (EV-FFD) is out-of-core sipping system which is adopted in Japanese prototype LMFBR MONJU design.

Main objective of EV-FFD is to identify a failed fuel subassembly out of fuel subassemblies which were suspected as failed in core by tagging gas FFDL system.

1) System design

Schematic diagram of this system is shown in Fig. 3 and 4.

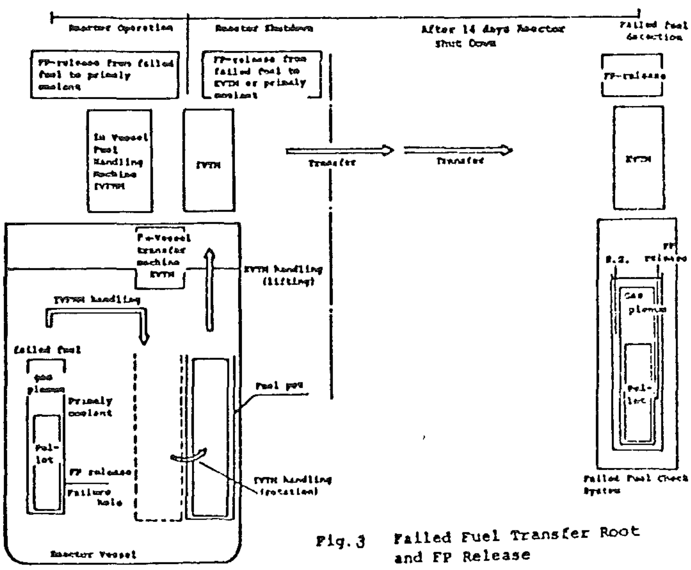


Fig. 3 Failed Fuel Transfer Root and FP Release

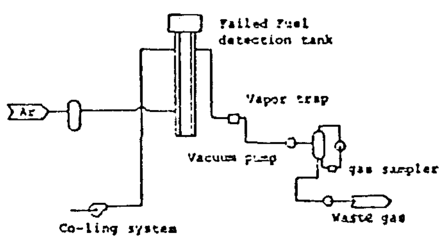


Fig 4 Shematic Diagram of EV-FFD

EV-FFD mainly consists of failed fuel detection tank, cooling system, argon gas supply system and monitoring system.

The fuel subassembly maintaining within sodium filled pot, which was detected as suspicious by tagging gas FFDL, is transferred from reactor core to EV-FFD port located in ex-containment by Ex-vessel transfer machine and inserted into detection tank.

By retaining cover gas of detection tank to negative pressure and heating sodium filled pot, it is expected to release gaseous fission product (^{133}Xe) from gas plenum of the fuel subassembly.

Identification of failed fuel subassembly is determined by analyzing radioactivity from the sampled gas using NaI detector and nuclide spectrum analyzer.

2) Research and development

Simulating FP gas (N_2) release experiments

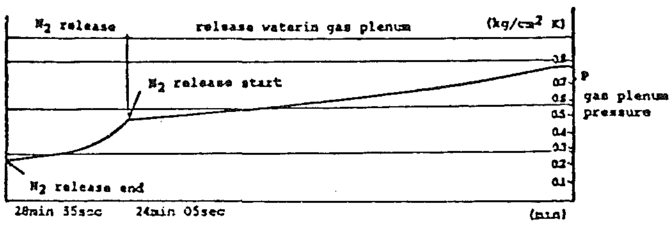
The simulating FP gas release experiments were carried out to evaluate the effectiveness of EV-FFD. The first series of experiments were performed with simulating FP gas release through a small failure hole (diameter: 0.2, 0.5, 1.0, 2.0, 4.0 mm) in water.

The second series of experiments were performed with simulating FP gas release through less small failure hole (diameter: 0.04, 0.06, 0.08, 0.2, 0.5mm) in water. Results obtained are as follows,

(1) Differential pressure of 0.68 ∿ 0.75 kg/cm² was necessary to release the simulating FP gas of model fuel pin gas plenum into water.

(2) Time of about 4 ∿ 30 minutes was required to release the simulating FP gas of model fuel pin gas plenum into water as shown in Fig.5.

Experimental results shown that sipping method of EV-FFD will be effective to identify failed fuel subassembly out of suspicious fuel subassemblies.



Failure hole ø 0.04 mm

Clad and dummy fuel pellets Gap 0.11 mm

Fig. 5 Simulating FP gas (N_2) release time

3) Investigation and future program.

EV-FFD will be effective to identify failed fuel subassembly from the result of simulating FP gas release experiment in water. But it is necessary to estimate the following items.

(1) Influence of sodium surface tension and viscosity.

(2) Volume of FP gas which is trapped in the fuel pin bundle after escaping failed fuel pin.

Simulating FP gas release experiment in sodium will be carried out to estimate above mentioned items.

SUMMARY AND CONCLUSION

In order to analyze FFD signal response to fuel failure an intensive study in modelling of FP behavior will proceed in the future. The code, which represent FP behavior from its birth in fuel and to transport to primary cooling system, is expected to be verified by in-pile experiments with artificial FP source in JOYO Mk-II.

The experiences of fuel failure in the actual LMFBR plants will produce the most interesting and meaningful information on the study of FFDL. Accordingly, it is much beneficial to release and accumulate the detailed data concerning the experiences of fuel failure in LMFBRs in every country.

Location and inspection of failed subassembly is quite important from the view point of plant availability. Sipping method as well as gas tagging will be tested for verification of its feasibility in LMFBR. The future programs are now being considered and examined.

REFERENCE

1. N. Sekiguchi, "Current Status of Japanese Research and Development Activity and Program Plan on Failed Fuel Detection and Location Systems for LMFBR," this meeting.

2. T. Takagi, et al., "Sipping Methods for Localization of Failed Fuel Element in Japanese LMFBR," this meeting.

3. T. Miyazawa, et al., "Studies on Modeling to Failed Fuel Detection System Response in LMFBR," this meeting.

Discussion

D.B. Sangodkar, RRCK:

Why do you propose to develop individual sampling of sodium for DN monitoring and gas-tagging at the same time?

N. Sekiguchi, PNC:
Selector value sampling sodium device for DN monitoring has been developed for MONJU. But at the detailed design stage of the vessel, it was determined to be impossible to be applied, because of very narrow space of the upper plate for its installation.
This is the major reason that we have started to develop gas tagging system.

V.F. Efimenko, IAEA:

Are you going to use ultrasonic under sodium viewing for location of fuel failures?

N. Sekiguchi, PNC:
No, we are not.

F.E. Holt, HEDL:

You feel gas tagging will not be reliable with many failures in core?

N. Sekiguchi, PNC:
That is true. Some experience will be required.

F.E. Holt, HEDL:

How long does it take to obtain Na sparge and count in JOYO sipping system?

N. Sekiguchi, PNC:
The entire core, 67 assemblies will take $\sim$ 24 HRS.

F. Gestermann, IA:

Can you precise your system of in-core inspection of fuel elements, especially the sodium sampling and degazification?

N. Sekiguchi, PNC:

The principle of our sipping method is described in the paper "Sipping Methods for Localization of Failed Fuel Element in Japanese LMFBR". That was distributed yesterday.

P. Michaille, CEA:

Can you explain me what you expect more with a precipitator than with a germanium for gases measurements?

N. Sekiguchi, PNC:

Although γ-spectrometer is able to give more information concerning characterization of failed fuel, precipitator is better with respect of reliability, maintenance and cost.

J. Dauk, IA:

Will your tagging system be an on-line system?

N. Sekiguchi, PNC:

Yes.

THE DETECTION AND LOCATION OF FUEL FAILURES IN SODIUM COOLED FAST REACTORS

REVIEW OF UK WORK

COMPILED BY

D K CARTWRIGHT RNL
C V GREGORY DNE

WITH CONTRIBUTIONS FROM

T A LENNOX DNE
I CATHRO BNL CEGB
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UK presentation to the IWGFR Specialist Meeting on 'Detection and Localisation of Failed Fuel Elements in LMFBRS' Karlsruhe 11-14 May 1981

Part 5

THE FUTURE PROGRAMME FOR FAILED FUEL DETECTION STUDIES IN THE UK

THE FUTURE PROGRAMME FOR FAILED FUEL DETECTION STUDIES IN THE UK

The main object of future work is to attempt to relate fission product signals to fuel behaviour so that by observing these signals the state of the core can, as far as possible, be known.

To this end the main UK programme is mounted in PFR where the behaviour of fuel and the related fission product signals will be studied from both naturally occurring failures if and when they occur and from artificially defected pins. The latter cover a range of defect sizes and fuel types including pelleted and vibro fuel and both core and breeder designs are to be studied. This programme in part utilises the demountable subassemblies (DMSA's) which enable up to 6 clusters of fuel each containing 19 pins to be irradiated in special core lattice positions. These lattice positions in addition to the normal bulk monitoring facilities have sampling pipes to the location monitor which enables the mixed mean sample of the DMSA and two individual clusters to be monitored.

This programme will thus enable the fission products released from fuel failures to be correlated with fuel behaviour as determined by post irradiation examination. In addition the fission gas and DN precursors released will be correlated with the release of fisson products which can potentially contaminate components and possibly restrict maintenance work.

In support of this work models for the release of fission products are being developed involving the use of in-pile loops. This work includes the study of fuel coolant interactions.

Work related to the design of CDFR is also being carried out to ensure that the monitor design is adequate and that the monitors will receive representative samples of coolant with reasonable time delays.

Discussion

F. Gestermann, IA:

You intend to irradiate artificially defected pins in the FFR up to long times. How is this comparable with your automatic SCRAM level of DND? Is the level higher than expected for one breached fuel pin after long irradiation or can the level be adjusted?

T.A. Lennox, UKAEA:
The SCRAM level is adjusted according to the operation proposed. It is expected that prolonged operation with failures in the failed fuel rig will be acceptable since e.g. this rig has a strenghtened wrapper and can be monitored by the location loop. The core can be scanned automatically by the location loop during power operation. These factors and the relevant safety argument will be taken into account in deciding the SCRAM level.

A. Merkel, IA:

To which amount is the dissolving silverseal contributing to the radioactive contamination in the sodium?

T.A. Lennox, UKAEA:
Only a few milligrams of silver are used and the dilution by the bulk sodium is large. The Ag-110 production is negligible.

D.E. Mahagin, HEDL:

What is the endurance goal to which failed fuel testing is directed?

T.A. Lennox, UKAEA:
The UKAEA wishes to keep fuel in the reactor until the next scheduled shutdown.

Breached-Pin Testing in the U.S.

D.E. Mahagin and J.D.B. Lambert

Hanford Engineering Development Laboratory

Argonne National Laboratory

ABSTRACT*

Experience gained at EBR-II by the late 1970's from a significant number of failures in experimental fuel-pin irradiation forms the basis of a program directed towards the characterization of breached pins. The questions to be answered and the issues raised by further testing are discussed.

INTRODUCTION

The design and manufacture of LMFBR pins have produced a remarkably reliable product in recent years, both in the U.S.A. and abroad. There is, however, no absolute means of guaranteeing that cladding breaches will not occur during the expected service lifetime of the fuel. In recognition of this fact, most countries have instituted development programs to characterize the performance of breached pins and to minimize the effects of this type of behavior on overall reactor operations. This paper summarizes work in America in this area.

EXPERIENCE

Other papers to this meeting (1-4) relate the considerable experience obtained to date in the US on the detection, monitoring, and identification of fuel-pin breaches. By the same token, equivalent experience has also come from learning to accommodate the consequences of failures in a power-producing LMFBR. Part of that experience derived from retrofitting EBR-II to control the fission-gas and cesium contamination from breached pins; part came from having to justify the safety of continued operation with them; and part came from trying to interpret their condition from the fission-product signals that emanated from them--particularly delayed-neutron (DN) signals. We briefly relate this experience, because it bears directly on how LMFBR's may be operated with failures; and points to the questions that must be answered by further testing.

Contamination Control

EBR-II was not built to accommodate fuel-pin failures, and a small but persistent leakage of cover gas around seals in the reactor head required shutdown whenever a gas leaker occurred. A conserted effort was made in the mid 1970's to reduce this leakage, and to design and construct a cover-gas cleanup system, or CGCS (5). The CGCS was rated to deal with the continuous (and conservatively high) release from up to 12 breached mixed-oxide pins in-core, and to cryogenically store their fission gas. Two factors were found after the CGCS was operational: gas leakers do not release all of their fission gas at once; and primary sodium acts as an excellent holdup device to reduce activity of short-lived gas isotopes. Consequently, no problems have been encountered with fission-gas contamination with up to five breached pins in core together; and this with a residual cover-gas leakage of ~0.2 L/min. Figure 1 shows how the CGCS reduces gas activity after a failure.

About 3-5 Ci of long-lived cesium activity have also been released from each and every failure. Although no immediate problems occurred because of this activity, by 1977 primary sodium activity was at ~350 nCi/gm, and future problems were envisioned with frosting of the upper internal structures of the reactor. Tests were therefore performed and a nuclide trap developed (6) to limit this activity buildup. It was installed upstream of the primary cold trap in 1978. The trap reduced the cesium activity by two orders of magnitude, and has maintained it at that level ever since (Fig. 2).

Over 1978-80 fifteen failures per year were encountered at EBR-II: more than one might reasonably expect in a commercial plant. And yet the major sources of primary circuit contamination--fission gas and cesium (7)--have been very adequately controlled, and experience that suggests there should be no insurmountable problems in controlling contamination from breached pins in a large LMFBR.

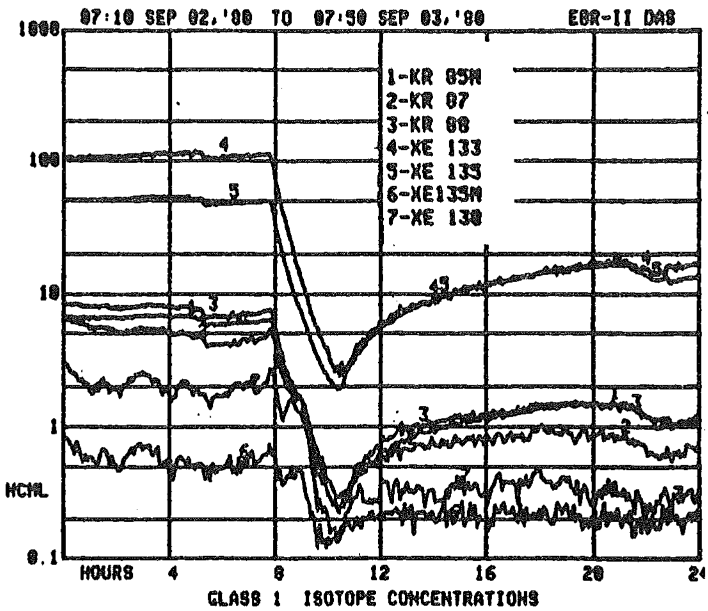

Fig. 1. Reduction in Activity from a Leaker in EBR-II by the CGCS

*Work supported by the U.S. Department of Energy.

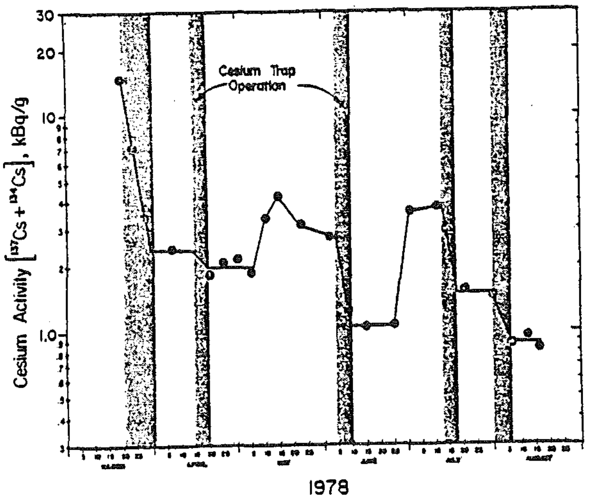

Fig. 2. Nuclide Trap Control of Cesium Activity in EBR-II Primary Sodium

Failure Monitoring

When breached-pin tests were first contemplated in EBR-II there was still a lingering fear that rapid fuel-pin failure propagation might occur as a consequence of continuing to operate breached pins. At that time also, the DN detector FERD was in the scram circuit of the reactor and set to trip at twice its normal background level.

It was argued strongly and successfully from the benign nature of the ~50 fuel-pin failures which had already been encountered at EBR-II; from safety appraisals (8); and from foreign experience (9), that rapid propagation of failure was extremely unlikely under normal reactor conditions. Similarly, the FERD--which receives signals 18-20 secs. after any event in-core--would be slower to respond than other devices, such as reactivity feedback (10). FERD was therefore taken out of the trip circuit and an administrative limit placed on DN signals. This limit was first set to be the full-power signal plus 25%. Since that time this limit has been raised in increments. At all times test results have been fed back to the review groups so that they are aware of progress achieved. This cautious 'bootstrap' approach to operating EBR-II has worked well.

When testing started it was also thought that DN signals from breached pins would be by recoil only, and that exposed fuel, therefore, might be difficult to monitor during operations. Results from initial tests (2) showed this to be quite untrue for ceramic fuels. Enhancement (or k factors) one or two orders of magnitude above recoil release have been observed from all tests of ceramic pins to date. (Metal fuels however appear to have a genuine recoil release of DN's, at least on the limited data obtained in EBR-II).

Although the cause of DN signal enhancement is not known, and will be a phenomenon for further study, it has made easier the monitoring of failures in-reactor. It has, for example, given some reassurance that the degradation of breached pins is slow, not rapid. Indeed the shape of the DN signal with time for the first naturally-breached pin test in EBR-II (see Fig. 5 in reference 2) was such as to indicate that breached mixed-oxide pins may become stable under steady reactor conditions.

Fuel-Sodium Reactions and Contamination

Results from early thermal-reactor tests (11), laboratory studies (12), and thermochemical modeling (13) all indicated that the phenomenon of fuel-sodium reaction swelling (FSRS) would be the primary driving force in the deterioration of breached pins in oxide-fueled LMFBR's. The problem in trying to predict the consequences, however, was the inability to test under oxygen potentials of fuel and sodium; temperature gradients and fission-product inventory; and cladding ductilities which were in any way typical of an actual operating pin.

The results from the initial predefected and naturally failed mixed-oxide fuel-pin tests (14) indicated that we were generally correct in our understanding of what governs behavior of breached pins. For example, Fig. 3 shows radial sections of one of the predefected pins. The interaction between primary sodium and mixed-oxide fuel appears to stop at about the 1100°C isotherm --in agreement with theory. On the other hand, it was found that the simple act of shutdown and restart of the reactor (for an extraneous reason) was sufficient to cause a ratchetting of the DN signal from a pin. Was this due to mechanical interaction on restart, additional ingress of sodium and reaction, or both? Clearly, the behavior of breached pins, and their modeling, are complex, and must be systematically studied.

During these initial tests there was no indication of fuel loss from the breached pins, either by weight loss (not accurate), visual examination (sensitive), or by sampling of the primary sodium (very sensitive). However, such sampling was on the cold-leg only, and thus would not indicate if minute amounts of fuel had come adrift and caught in the IHX, or simply settled in the cold tank and never been detected. At least, loss of fuel was slight enough as to pose no serious consequence to reactor operations; a reassuring fact.

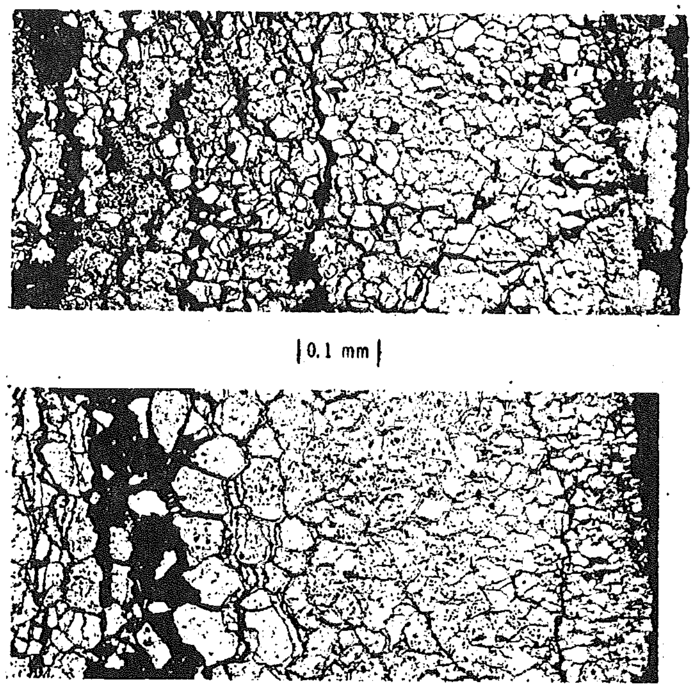

Fig. 3. Fuel-Sodium Reaction in a Predefected Mixed-Oxide Fuel Pin Tested in EBR-II

ADDITIONAL WORK

It is clear from the proceeding discussion that there are three basic areas involved in a thorough understanding and modeling of breached-pin operation: (i) the kinetics of sodium-fuel reaction swelling--these will govern the deterioration of breached oxide pins; (ii) the release, transport, and deposition of contamination from breached pins--these will govern the consequences of such operation on the reactor plant; and (iii) the characteristics of DN release from breached pins--these will be ultimately used to monitor the progression of items (i) and (ii), and to control reactor operations. All these activities must have a common goal: definition of the limits which must be applied to breached fuel operation in LMFBR's.

The program should proceed on two fronts: laboratory studies and modeling, and in-reactor tests of breached pins. The ex-reactor work is mainly focused on developing a detailed, physically realistic FSRS model which can be used to interpret the reactor tests, and to extrapolate to large-plant conditions. Such a model is required to evaluate the potential for slow pin-pin failure propagation and blockage formation as a consequence of reaction product buildup. In its present form (15) the FSRS model will predict the rate of cladding diametral strain and opening at the initial breach site in a typical mixed-oxide pin at nominal power. Models for axial extension/cracking of the primary defect, sodium ingress, and subsequent axial transport and metastatic reaction swelling, and fuel release from an increasing area of exposed fuel are currently under development.

A combination of experimental and analytical evaluation is being used to derive the source term for oxide fuel release into sodium. This information is needed to estimate primary sodium system contamination. Measurement of the limiting solubilities of urania and plutonia in sodium at 550-800°C (16) have been made, and the aim of current experiments is to characterize fuel particulate formation at the oxide-sodium interface. To accomplish this, the separate influences of fuel stoichiometry, temperature, oxygen impurity concentration, and sodium flowrate will be examined in a small-scale experimental loop.

Additional EBR-II tests should be of at least three types: (i) tests to identify the primary fuel/blanket variables which influence breached-pin performance; (ii) tests to study pin performance and fuel loss in order to validate FSRS models; and (iii) tests specially designed to study DN release phenomena in breached pins. The first type of test can, in general, be carried out sequentially in open-core positions. The second two types of tests are best performed in special-purpose facilities which may be instrumented.

The first of these special facilities--the breached fuel test facility, or BFTF (17) is shown schematically in Fig. 4. The facility enables the discharge from a test subassembly to be taken through the reactor cover, past thermocouples and flowmeters to a DN detector in the small rotating plug of the reactor. Here the flow of sodium is reversed and brought back down to the upper plenum of the reactor. On its upward travel part of the sodium flow is through the wall of a cylindrical filter, and part of the flow over the surface of rings of structural material. In this way, deposition of fuel and fission products on hot-leg sodium surfaces may be studied. Also DN signals from a breached pin may be monitored with a minimum of delay (5-10 secs.). This facility has been installed in EBR-II and calibrated with a recoil source, and successfully used once (18). It functions admirably. It permits very detailed study of pin behavior, DN signals, and contamination, all at the same time.

The second special facility--the fuels performance test facility, or FPTF--is somewhat similar to the BFTF. It also mates with the top end of a test subassembly and allows its discharge to be individually monitored by a DN detector in the small rotating plug. It differs from the BFTF in having no deposition-sampler section. Instead, it has a variable flow-valve which allows, during reactor operation, for the flow in the subassembly beneath it to be altered. It is also equipped with fast-response thermocouples, acoustic detectors, and flowmeters. With this facility, therefore, it will be possible to perform constant power/variable temperature, variable power/constant temperature tests, and combinations thereof. It will be used in conjunction with predefected pins to study in detail the characteristics of DN release from defects. It is hoped that by using the FPTF the temperature and power sensitivity of DN release from defects may be finally elucidated.

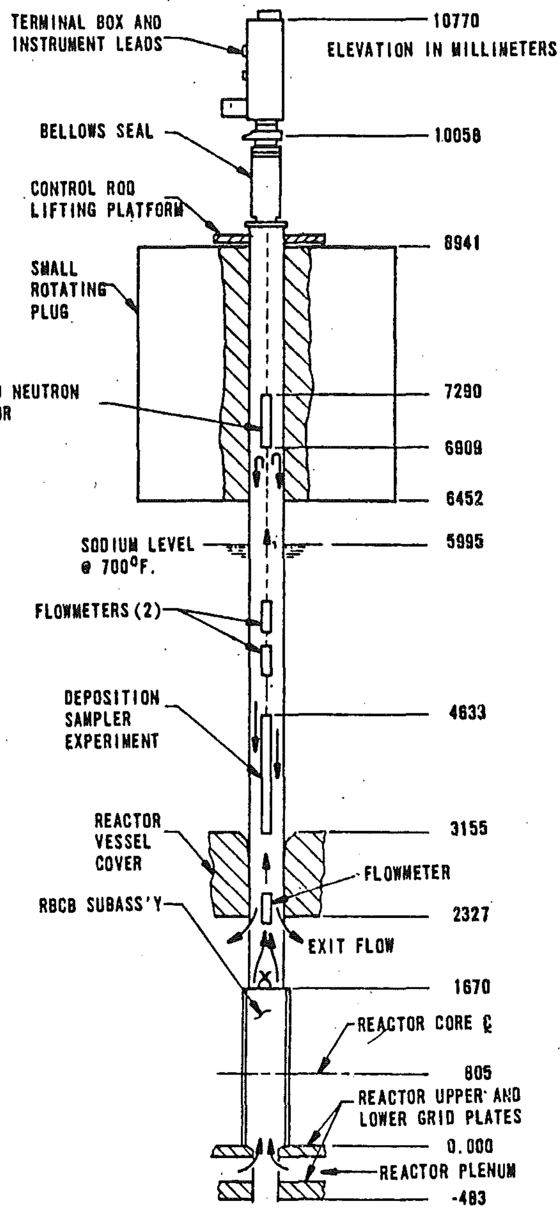

Fig. 4. The EBR-II Breached-Fuel Test Facility

SUMMARY AND CONCLUSIONS

Sporadic fuel failures will likely occur in commercial LMFBR's despite a very high quality control of fuel pins. Extensive experience gained at EBR-II from fuel-pin endurance testing suggests that simple "gas leakers" may be easily tolerated if systems are available to control fission-gas contamination and the buildup of cesium activity in primary sodium. An important question that needs to be answered by further development work is whether oxide pins with fuel exposed to primary sodium can be tolerated for limited periods of operation. What limits should be placed on this potentially efficient mode of operation? What safety concerns may be raised by such continued operation?

Results of preliminary testing in EBR-II suggest that breached oxide pins may only slowly deteriorate through fuel-sodium reaction under steady reactor conditions. Similarly, they seem easy to detect by delayed neutron monitoring, due to an enhancement of DN release whose cause is unknown; they also appear to lose insignificant amounts of fuel or fission products (other than gas and cesium). But our experience base is limited, and many factors remain uninvestigated.

To accomplish further development in this area, we have implemented an integrated program involving irradiation tests (several in special in-core facilities); laboratory simulation; and computer modeling of breached fuel-pin performance. As an adjunct, work on detection systems and definition of the consequences of fission-product contamination of the primary circuit is also proceeding. We intend that the culmination of this work will be guidelines for the operation of future LMFBR's in the presence of fuel failures; and the necessary support for licensing such plants.

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Discussion

P. Michaille, CEA:

The radial temperatures you showed on the figure of a defected pin cross section are calculated by a code or measured by any experimental method?

D.E. Mahagin, HEDL:
Calculated by a code that considers pin operating history.

P. Weimar, KfK:

For new computer Code:
Is this code able to describe nonuniform swelling of Breached Pins up to swelling rates of e.g. 40 %?

D.E. Mahagin, HEDL:
The code only describes pin behaviour after a breach has occurred; it appears to yield results that are consistent with experimental data.

THE RESULTS OF TESTING AND EXPERIENCE IN THE USE OF LEAK DETECTION METHODS IN FAST REACTORS

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Part 5

PLANNED RESEARCH AND FURTHER DEVELOPMENT OF DETECTION METHODS

Our knowledge about the main fission product release in the moment of and following a fuel element failure are un - adequate. The available possibilities of measuring the released activity under reactor operation pose the task to study quantitative and qualitative regularities of fission product release and to relate them to fission gas and delayed neutron measurement results. Quite a number of the studied volatile fission products of different half-lives may indicate the extent of fuel element failure and burnup.

The improvement is required of failed subassembly detection and localization systems and the criteria for the identification of such subassemblies. On the other hand, efforts are made to standardize all the procedures in gas and aerosol sampling process. This must decrease the scatter in the results of measurements associated with intact fuel subassemblies. On the other hand the existing failed subassembly detection system in the BOR-60 power plant is enhanced by other in- and out-of-pile devices for measurements verification. The sodium fission product contamination background signal is lower for these devices. The out-of-pile system may be vacuumed and its heating temperature may be controlled. Combined with hot cell examination of a portion of fuel subassemblies the above checks should improve the criteria of failed fuel assemblies identification and better its reliability.

CONCLUSION

1. Used in LMFBR's methods, instrumentation and equipment for failed fuel subassemblies detection and localization assure safe fast reactor power plants operation even in situations when a limited number of failed fuel subassemblies are present in the core.

2. In the reactor under operation the delayed neutron and fission gas detection systems provide information required to gain knowledge about the occurrence of fuel element failure, the moment the failure occurs, the number of failures and the fuel-coolant contact surface area. This information is sufficient to make approximate estimate of the primary circuit fission product contamination using the developed calculation methods and fission product release fractions. The efficiency of both detection systems is adequate to signal from the occurrence of a single wide-open cladding failure or some fuel element gaseous leaks in the situation when failed fuel elements are present in the core.

3. The method of out-of-pile failed fuel subassemblies detection assures all the failed fuel element removal from the core. The in-pile failed fuel detection allows to maintain in the core no more than 0.05% of failed fuel elements and provides adequate purity of the coolant and the circuit.

4. Further improvements are necessary in the informativity of the methods and the efficiency of the primary circuit contamination detection to control the defects development, to predict more precisely the primary circuit fission product contamination, to determine the number of failed fuel elements and assure safe reactor operation radiation situation.

FUTURE PROGRAMS

by P. Michaille

DETERMINATION OF THE FIELD OF EFFICIENCY OF THE DND SYSTEM IN CASE OF SUBASSEMBLY ACCIDENT

- RELEASE RATES BY FUEL
  - AT VERY HIGH TEMPERATURES
  - DISPERSED BY THE PRESSURE OF THE OCCLUDED FISSION GASES
- MECHANISMS OF TRANSPORTATION BY A DIPHASIC COOLANT

GERMAN RESEARCH AND DEVELOPMENT PROGRAM ON FAILED FUEL DETECTION AND LOCATION IN LMFBRs

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Introduction

Within the framework of R&D acitivities on the subject of "Failed Fuel Element Detection and Location", DEBENE[1)] plan a major number of acitivities, which constitute the direct extension of work performed so far and must be classified under the two main headings of safety and availability of LMFBR's. Of course, it is not always possible to make separate direct assignments either to aspects of safety or those of availability; in most cases, close connections exist to both topics. The projects outlined below are partly directly associated with the KNK II or SNR-300 facilities, partly they are applicable to fast breeder development in general.

Roughly half of the activities are being carried out jointly with foreign partners under various agreements for cooperation, such cooperation is always mentioned specifically.

1. Reactor Instrumentation

In Session I, the on-line gamma spectrometer installed in KNK II has been described. As mentioned on that occasion, the spectrometer furnishes both gamma spectra and calculated radioactivity concentrations of individual fission products, from which it compiles daily and weekly plots, respectively; see Fig. 1a and b.

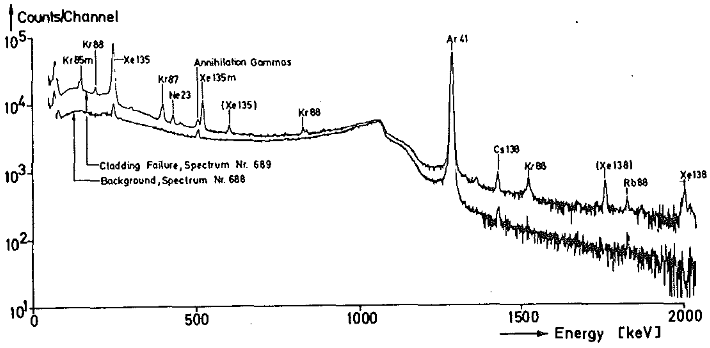

Fig. 1a: Gamma spectrum

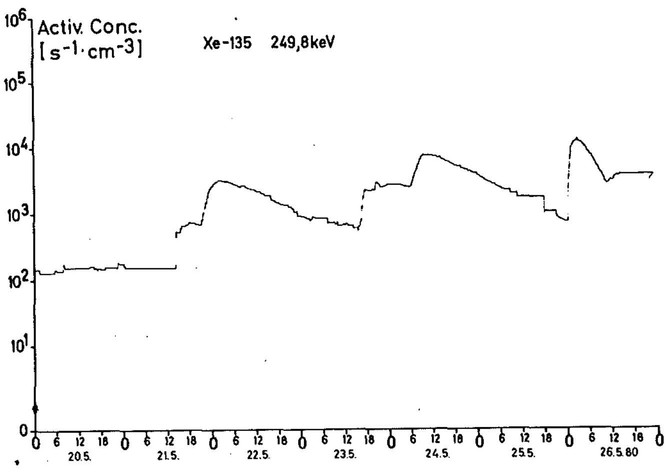

Fig. 1b: Weekly plot of Xe-135

1) DEBENE = Deutschland - Belgium - Netherlands

Both plots, and also the computer printout referred to in the above paper, are not yet regarded as the optimum of what is possible. For assessments of fault propagation it is not only the daily and weekly plots of individual nuclides and the instantaneous pictures of the fission product spectra, but especially their variation as a function of time plotted in a three-dimensional representation, which is deemed to be very helpful. The abscissa in that representation is either the half-life, $t_{1/2}$, or the decay constant, λ; the ordinate is either the radioactivity concentration, A, or the release-to-birth ratio, (R/B). A parameter to be included is time, Fig. 2, for which random values are indicated (Fictitious values). The plot of R/B as a function of λ is the most sophisticated, but also the most elegant solution and, at the same time, would permit information to be obtained about the mechanism of fission product escape, i.e., recoil, diffusion and others.

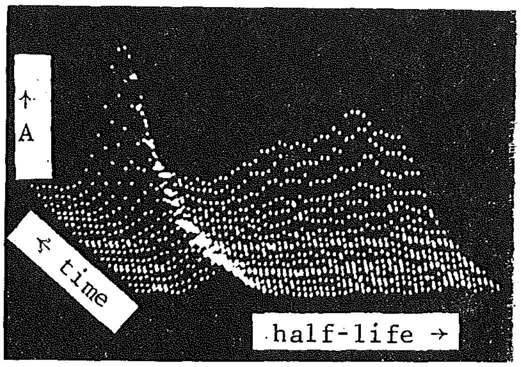

Fig 2.: Plot of the variation as a function of time of the fission product spectrum. For explanations, see text 1.

It is a well known fact that the localization of defective fuel rods has not yet been solved satisfactorily. For this reason, a novel detector for beta-radiation has been proposed. That detector could be installed at the fuel element outlet and could detect in particular the high energy beta-particles of the very shortlived fission products. The working principle of the detector is based on the passage of fission products beta-particles through the detector housing to the collector and measurement of the resultant equalizing current, Fig. 3.

Estimates carried out so far have indicated a sensitivity of approx. 10^{-10}A for 1 cm^2 of free fuel area (= recoil area) for a detector of L = 10 cm length and D = 3 cm diameter. Spurious effects caused by Na-24, gamma and neutron radiations could be eliminated.

The DN-monitors used so far have relatively large dimensions because of the necessary shielding against gamma radiation, thermal insulation and moderator. Although this kind of shielding against core neutrons is indispensable, a study is intended to show that detectors less sensitive to temperature and gamma radiation allow smaller DN-monitors to be built. However, neutron sensitivity is not to be reduced too much compared with the present type of monitors.

2. Operation of LMFBR's

In Session II, the flux tilting measurements in KNK II were mentioned in the DEBENE presentation, especially in Fig. 14 of the presentation by H. Richard et al. In the course of those flux tilting measurements, changed statistical variations of the DN count rate were found when moving absorber rods at certain rod positions. Additional studies are to indicate the way in which the height of the lower absorber rod edge correlates with the axial position of a fault in a fuel element. This may lead to a possibility of three dimensional localization of the fault.

On the same subject, the current post-irradiation examinations of the first defective KNK II fuel element and the post-irradiation-examinations to be started in a few weeks of the second defective KNK II fuel element should be mentioned. These post irradiation examinations serve the following main objectives with respect to detection and localization of faulty fuel rods:

- geometric position of fault
- size of fault
- type of fault
- presence or absence of secondary cracks
- correlation between size of fault and DND-signal.

3. In-pile Experiments

Several experiments with open fuel samples are planned in KNK II. A first series of four experiments will begin with only one fuel pellet and an artificial fault of 1 mm^2, terminating with 12 pellets and several artificial faults of a total of 80 mm^2.

The need to obtain official licenses for this series of experiments was the reason for starting the tests at the very low fuel inventory of only one pellet. As the inventory increases and the leakage areas grow, it will be demonstrated that the risk to reactor operation brought about by those experiments is negligibly small.
Here are the real objectives of these experiments:

- General studies of fuel rod behaviour
- Fission product releases
- Fuel loss
- Formation of Na_3MO_4
- Fission product transport
- Detection of fission products by means of DND and in the cover gas
- Measurements of sodium transit times and response times of measuring chains
- Correlation of leak size with the measured fission product signals
- Verification of computer codes.

According to the present timetable, the first experiment of this type is to be started in July 1981. After completion of the first series of experiments, measurements are to be performed on uranium metal to calibrate the DND monitors.

The first three Mol 7C-experiments on studies of disturbances of coolant flow in a 37-rod bundle have been conducted successfully with SCK-CEN Mol, Belgium; see Session III and /1, 2/. As in those earlier experiments, the Mol 7C/4 and Mol 7C/5 experiments presently in preparation and planned for 1982 will use fuel with medium burnup for extensive measurements of delayed neutrons. The modification of the loop design due to burnup also requires a greatly changed DND-monitor.

The successful cooperation carried out so far between the Toshiba

Company of Tokyo and the Institute of Reactor Development of the Karlsruhe Nuclear Research Center will be continued. In the next few years, the FPL fission product loop of Toshiba in the TTR (Toshiba Test Reactor) will be used for extensive DND parameter studies about the behaviour of DN precursors in sodium. Moreover, releases of DN precursors from a fuel pebble bed to be considered as a blockage are to be studied.

The series of experiments on defective fuel rods conducted in the SILOE reactor in Grenoble jointly by the CEA, France, and KfK, Germany, is to be continued. Also in the future especially those faults will be simulated which have been found to be not yet entirely avoidable in practice, i.e., in reactor operation. In addition to be continuous improvement of methods of detection of fuel rod defects, it is especially the permissible length of reactor operation with a defective fuel rod which is to be determined. Because of the potential plutonium contamination of reactor systems, the loss of fuel from a defective rod is in the foreground of studies.

4. Models and Codes

The quantitative interpretation of DND-signals is still highly problematic. The following points will be given special attention in the future:

- k-factors, their dependence on fuel structure, burnup and leak size
- Temperature dependence of fission product emission
- Influence of open porosity

The Mol 7C experiments have shown that open porosity represents a very important parameter influencing signal generation. The porosities involved may be either open pores in the narrower sense of the term (according to definition) or continuous inner surfaces resulting from thermal overloading. Measurements of open porosities are planned after the experiments within the framework of post-irradiation examinations.

For theoretical treatment of the DND-signals, the computer code referred to in Session IV is to be expanded further. New theoretical and experimental findings are to be used for this purpose, as is shown in Table 1. In a second step, the gas signals are to treated.

5. References

/¯1_7 S. Jacobi: "Behaviour of Fuel Fragment Blockages - Detection by Delayed Neutrons and Problems of Signal Interpretation"
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Sun Valley, August 2-6, 1981

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Sun Valley, August 2-6, 1981

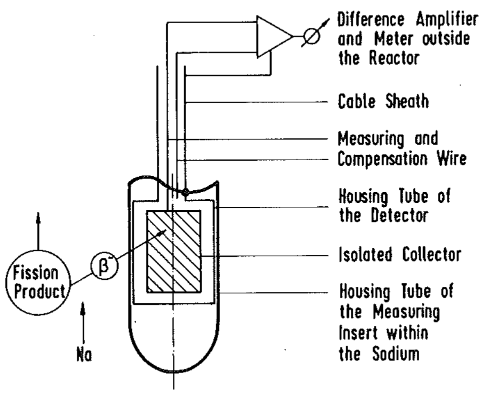

Fig. 3: Removable Beta Detector above the Outlet of the Fuel Subassembly for Location of Defect Fuel Elements

Table 1. **FICTION Code**

Fission Product Signal Calculation

Birth rates — Basis: ORIGEN

↓

Empirical migration values of fission products in mixed oxide fuel

↓

Open porosity

↓

Empirical release values of fission products through the surface of the fuel

↓

Free fuel surface in contact with the sodium as a function of time = f(t)

↓

Oxygen concentration in the sodium and the fuel → Thickness of Na_3MO_4 = f(t)

Fuel loss → Concentration of fuel in sodium = f(t)

↓

DND signal = f(t)

later: cover gas signals = f(t)

k-factor = f(t)

Shutdown criteria)1

Reactor instrumentation criteria)1

Recommendations to reactor operation)1

)1 Not part of the code, but application

Discussion

C. Berlin, CEA:

What is the size of the ß-detector?

S. Jacobi, KfK:
Proceeding from the design of the measurement equipments at the outlet of the subassemblies of SNR 300 for the thermocouples and for the flow meter which is now canceled I decided an outer diameter of 30 mm and a length of 100 mm. But these dimensions are changeable according to the requirements of the reactor observing that the sensitivity of the detector is proportional to its surface.

L. Leenders, S.C.K./C.E.N.:

Concerns: parasitic ß-contributions in the ß-detector
Question: How do you take the ß-activation of the detector itself and of its cladding tube into account?

S. Jacobi, KfK:
In contrast to the self powered neutron detectors I have the advantage that both electrodes that means the collector and the cladding tube are composed of the same material. So in case of the same size of surfaces of the collector and the cladding tube a perfect compensation exists for the beta current from the collector to the cladding tube and from the cladding tube to the collector. If the very small differences of the sizes of these both electrodes must be taken into account it is possible to compensate this by small grooves on the surface of the collector.

C. Berlin, CEA:

Have you any results about tests in the laboratory of the ß-detector?

S. Jacobi, KfK:
Simulating the betas of the fission products by a source of P-32 I used different kinds of material and different sizes of the detector. Using the measured current of electrons to calculate the source strength a sufficient conformity was obtained in comparison to the source strength calculated by the irradiation conditions producing the source.

C. Berlin, CEA:

Where the ß-detector will be localized in the Mol 7C loop?

S. Jacobi, KfK:
Inside the test train downstream of the bundle with a transit time of the sodium of about 1 to 1,5 s.

A. Merkel, IA:

Question due to the ß-detector situated at subassembly outlet: How do you eliminate the Na24- and the Ne23- influence?

S. Jacobi, KfK:
- Na 24 is eliminated by wall thickness of the detector.
- Ne 23 is not eliminated. Ne 23-concentration in the sodium is expected to be low enough to classify it as a background effect by which operation of the detector can be tested.
  In addition, the Ne 23-level may be used as a neutron flux monitoring signal to measure the power of the subassembly, but this is in the future.

PROGRAMME FOR DEVELOPMENT OF FAILED FUEL ELEMENT DETECTION AND LOCATION SYSTEMS FOR INDIAN FAST REACTORS

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1. Introduction

Indian fast reactor programme began with the start of construction of Fast Breeder Test Reactor at Kalpakkam. This reactor is in an advanced stage of construction. The instrumentation for failed fuel element detection and localisation in FBTR has been described in an earlier presentation at this Meeting. It is recognised that the instrumentation of FBTR, particularly that for localisation of failed fuel will not be adequate for a large power generating LMFBR. From considerations of resolution and speed, d.n. monitoring of individual coolant channel samples appears promising and hence this technique is chosen for development as a system for localisation of failed fuel in future Indian fast reactors. The programme planned to develop such a system for FBR-500, a 500 MW(e) LMFBR, to be set up in India in the next decade is described in this presentation.

2. Future Programme

2.1 Programme for FBTR.

2.1.1 Commissioning of instrumentation systems

One of the tasks to be pursued in the immediate future is that of commissioning and calibration of FBTR systems. Pre-commissioning tests are proposed to be restricted to source-checks on both the DND and GFPD systems. The DND system will be checked both before and after filling of sodium by a neutron source of strength 10^6 n/sec. The gas-flow ion chamber of the GFPD system will be checked with a gamma source of 5 millicurie placed external to the detector. Its sensitivity for radioactive gas flowing through it will be checked in the beginning of power operation of FBTR by measurement of the $^{41}Ar + ^{23}Ne$ background in the cover gas by an independent laboratory measurement of the activity of a cover gas sample.

2.1.2 Calibration

The calibration of both these systems will be done by introducing a bare uranium rod in the core.

2.1.3 Experimentation of additional systems

Gas-tagging which can be incorporated with relative ease in small as well as large fast reactors was considered for FBTR.

The void space in FBTR fuel pin was found to be too small to permit about one cm^3 of gastag without affecting its linear rating. However, it is possible to tag the experimental fuel assemblies which will have much larger void space. It is difficult to justify the enormous expenditure on the gastags and related detection equipment for this limited purpose. We are now considering the use of solid radio-active tracers such as ^{226}Ra and ^{228}Th in the experimental fuel pins with a view to detecting their gaseous daughter products, Radon and Thoron, in the Cover gas. Since the choice of the tracers is restricted, this system is proposed to be used in combination with the gamma-chromatograph so that the age of the failed fuel element and the simultaneous detection of the tracer will together permit localisation of a failed experimental assembly with enhanced precision.

2.2 Programme for FBR-500

In the next phase of fast reactor programme in India, construction of FBR-500, a 500 MW(e) LMFBR, is planned. While the design of the reactor is still in the conceptual stage, emphasis is being laid on the research and development needed to support the design and construction of the reactor. Included in this R & D programme is the development of a system for identification and localisation of failed fuel elements. Based on published literature and our own theoretical assessment, it has been decided to pursue development of a system based on d.n. monitoring of sodium sampled from individual coolant channels. From considerations of resolution and speed of localisation, this system appears to be the most promising for this application. The experience from Phenix and PFR is also reported to be good. The main disadvantages of the system are the need for intervention into the reactor vessel and for continued operation of the reactor albeit at lower power for identifying the failed element.

Theoretical assessment of the sensitivity of such a system indicates that it is adequate for FBR-500.

Development programme already planned in this field includes development of a selector valve for sodium sampling. Hydraulic performance of a prototype valve will be tested in water and in sodium at different temperatures. Apart from reliability which is of prime importance, other factors such as dilution of sampled sodium by bulk sodium, cross-contamination of adjacent channels at the level of selector valve and transit time which have a bearing on the sensitivity and selectivity of the system are planned to be studied.

3. Conclusions

Indian programme in the field of failed fuel element detection and localisation is aimed at augmenting the instrumentation of FBTR for localisation and development of a d.n. monitoring system of localisation for future large fast reactors.

Discussion

C. Berlin, CEA:

Will the FBR-500 be a pool or a loop reactor?

D.B. Sangodkar, RRCK:

FBR-500 will be a pool reactor.

S. Jacobi, KfK:

Sorry that I did not understand clear enough the "disadvantages of the system" mentioned in your presentation.

D.B. Sangodkar, RRCK:

One disadvantage is that the sampling equipment including selector valve, pump etc. have to be inside the reactor vessel which calls for their qualification for operation in a hostile environment. Another disadvantage is of course, that the reactor has to be operated for localisation. A problem may arise when the reactor cannot be started in the case of a large DND signal implying a large defect.