Ministry of Defence

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Our Ref: FOI2019/08636

[redacted]

18 December 2019

Dear [redacted]

Thank you for your email requesting the following information:

File reference: AB38/2072

As confirmed previously, we have treated your correspondence as a request for information under the Freedom of Information Act 2000 and we can advise that the Ministry of Defence (MOD) holds information in scope of your request.

We attach the following document:

• File AB38/2072 – Chernobyl - Symposia

If you wish to complain about the handling of your request, or the content of this response, you can request an independent internal review by contacting the Information Rights Compliance team, Ground Floor, MOD Main Building, Whitehall, SW1A 2HB (e-mail CIO-FOI-IR@mod.gov.uk). Please note that any request for an internal review should be made within 40 working days of the date of this response.

If you remain dissatisfied following an internal review, you may raise your complaint directly to the Information Commissioner under the provisions of Section 50 of the Act. Please note that the Information Commissioner will not normally investigate your case until the MOD internal review process has been completed. The Information Commissioner can be contacted at: Information Commissioner's Office, Wycliffe House, Water Lane, Wilmslow, Cheshire, SK9 5AF. Further details of the role and powers of the Information Commissioner can be found on the Commissioner's website at https://ico.org.uk/.

Yours sincerely,

Defence Nuclear Organisation Secretariat

UNITED KINGDOM ATOMIC ENERGY AUTHORITY

Prefix Letters	G M Ballard - Chief Executive
	CHERNOBYL
Registered Number	AB 38/2072

Former File	Other relevant Files	Referred to	Date
(For Registry use only)	Reviewed: 24/1/01 Class DECLASSIFIED J.J.Clifton, UKAEA, Risley JJ Clifton		5 closed 1022

Referred to	Date	Referred to	Date

Authority for closing

CLOSED UNTIL 2019

Initials

Branch

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SAFETY AND RELIABILITY DIRECTORATE

SEMINAR

INTERNATIONAL APPROACH

TO

NUCLEAR SAFETY

(after THREE MILE ISLAND and CHERNOBYL)

UKAEA

ADMINISTRATIVE ARRANGEMENTS

Venue

The Pembroke Hotel
North Promenade
Blackpool
Lancashire
Tel: Blackpool (0253) 23434
Telex: 677469

Meals

Refreshments mid-morning and mid-afternoon will be provided. Lunch will be provided. For those delegates who are staying at the Hotel Evening Dinner will be provided.

Seminar Dinner

The Seminar Dinner will be held on the evening of Thursday, 9 June 1988, 1930 for 2000 hours.

Working Language

The Seminar will be conducted in English only and all other documents will be printed that way.

TIMETABLE

Wednesday, 8 June

OPENING

0900 to 0945	**NUCLEAR POWER PERSPECTIVE**
	J. H. Gittus
0945 to 1030	**SAFETY GOALS FOR NPP**
	A. R. Garlick *Presented by J. G. Tyror*
1030 to 1100	**COFFEE**
1100 to 1145	**ROLE OF PSA IN NPP SAFETY ASSURANCE**
	M. R. Hayns
1145 to 1230	**PAST PSA STUDIES AND APPLICATIONS**
	N. J. Holloway
1230 to 1400	**LUNCH**

Wednesday, 8 June (Cont'd)

DESIGN FOR SAFETY

1400 to 1445	DEVELOPMENT OF PROBABILISTIC SAFETY CRITERIA/PRINCIPLES *A. A. Debenham*
1445 to 1530	SAFETY PRINCIPLES FOR ADVANCE PLANT *D. Phillips/M. R. Hayns* *Presented by A. R. Taig*
1530 to 1600	TEA
1600 to 1645	ANALYSIS OF RBMK AGAINST UK SAFETY PRINCIPLES *P. Bonell*
1645 to 1730	REVIEW OF SIZEWELL DESIGN SAFETY FEATURES (SRD R281) *A. A. Debenham*

Thursday, 9 June

MAN-MACHINE INTERACTION

0900 to 0945 THE SIGNIFICANCE OF HUMAN ACTIONS FOR PLANT SAFETY

N. J. Holloway

0945 to 1045 STRATEGIES FOR MMI ANALYSIS IN PSA

P. Humphreys

1045 to 1115 COFFEE

1115 to 1200 HUMAN RELIABILITY ANALYSIS METHODS

T. L. Waters

1200 to 1230 THE ROLE OF SPECIAL PROCEDURES AND RELATED FEATURES IN THE FRENCH NUCLEAR SAFETY APPROACH

D. Vignon/J. P. Kus

1230 to 1400 LUNCH

ACCIDENT PHENOMENOLOGY

1400 to 1445	FUEL BEHAVIOUR IN SEVERE ACCIDENTS *D. McInnes*
1445 to 1530	DEVELOPMENTS IN MFCI MODELLING *I. Cook*
1530 to 1600	TEA
1600 to 1645	ASSESSMENT OF ASEISMIC DESIGN STANDARDS AND THE RELEVANCE TO NPP SAFETY *D. Phillips*
1645 to 1730	INFORMATION AND TOOLS REQUIRED FOR A FIRE PSA *M. Finucane*

Friday, 10 June

SOURCE TERM AND CONSEQUENCES

0900 to 0945	SOURCE TERM AND THE CHERNOBYL ACCIDENT *P. N. Clough*
09⁄ to 1030	AEROSOL BEHAVIOUR IN SOURCE TERM ANALYSIS *I. Dunbar*
1030 to 1100	COFFEE
1100 to 1145	MODELLING THE CONSEQUENCES OF REACTOR ACCIDENTS *B. Underwood*
1145 to 1230	CONSEQUENCES OF THE CHERNOBYL ACCIDENT *W. Nixon*
1230 to 1400	LUNCH

Friday, 10 June (Cont'd)

ACCIDENT RESPONSE

1400 to 1445	EMERGENCY PLANNING FOR NPP ACCIDENTS *G. C. Meggitt*
1445 to 1530	ACCIDENT RESPONSE IN FRANCE *J. J. Duco*
1530 to 1600	TEA
1600	PANEL DISCUSSION AND CONCLUSIONS

HSSC SYMPOSIUM ON CHERNOBYL "ONE YEAR AFTER"

LECTURE THEATRE, RISLEY

PROPOSED PARTICIPANTS

F R ALLEN	SRD
A R BAKER	RISLEY
A BALL	SRD
G M BALLARD	SRD
P BARR	SRD
I R BIRSS	RISLEY
T E BLACKMAN	WINFRITH
P G BONELL	SRD
B R BOWSHER	WINFRITH
D BRICKLIFFE	SRD
F BRISCOE	CULHAM
M L BROWN	SRD
J BUTLER	WINFRITH
R BURNUP	SRD
P T CAMPBELL	SRD
R CLASPER	SRD
P N CLOUGH	SRD
C B COLLINS	WINFRITH
I COOK	CULHAM
M A COOPER	SRD
C B COWKING	RISLEY
W CROSBIE	HARWELL
A A DEBENHAM	SRD
J N EDMONDSON	SRD
G T EDWARDS	SRD
J FEWSTER	RISLEY
A M GAMES	SRD
A GREEN	SRD
S R GRUBB	WINFRITH
A N HALL	SRD
S F HALL	SRD
D HARRIS	RISLEY
R HASLAM	RISLEY
M R HAYNS	SRD
D HENDERSON	DOUNREAY
A R HERRICK	RISLEY

J R HIND	RISLEY
W HURST	SRD
I HYMES	SRD
R G JACKSON	SRD
J JOWETT	SRD
M P KISSANE	SRD
J R MATTHEWS	HARWELL
R McKEAGUE	RISLEY
R N H McMILLAN	SRD
P D MICHELL	SRD
P A MORETON	SRD
W NIXON	SRD
K C O'DONNELL	SRD
P D PARSONS	SPRINGFIELDS
R A PATTIE	SRD
G PEARCE	RISLEY
R S PECKOVER	SRD
F PHILLIPS	SRD
B L PRESCOTT	SRD
M PRESTON	LHQ
S A RAMSDALE	SRD
N SHERIFF	RISLEY
N SIMPSON	SRD
R R SMITH	SRD
R THOMAS	RISLEY
D E J THORNTON	RISLEY
B TOMKINS	RISLEY
B TURLAND	CULHAM
G TYLER	DOUNREAY
B WALKER	RISLEY
E WALKER	SRD
D L WARD	DOUNREAY
R K WEBSTER	HARWELL
W H WHITLOW	RISLEY
A WILKINSON	SRD
R WILLIAMS	SRD
J WILSON	RISLEY
M WOOD	WINFRITH

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HSSC(87)SYMP 2

UNITED KINGDOM ATOMIC ENERGY AUTHORITY

HEALTH AND SAFETY STUDIES COMMITTEE

SYMPOSIUM

CHERNOBYL: ONE YEAR AFTER

Risley Lecture Theatre, Wednesday, 27 May 1987
at 9.45 am

PROGRAMME

9.45		Coffee		
10.00		Introduction	Dr J H Gittus	
	1.	Current Understanding of the Accident		
10.15		Comparative Safety Provisions	Dr G M Ballard	HSSC(87)P16
10.50		Accident Phenomenology	Dr A N Hall	HSSC(87)P17
11.25		Fission Product Release - Source Terms	Dr P N Clough	HSSC(87)P18
12.00		Accident Consequences - Fission Product Dispersion	Dr W Nixon	HSSC(87)P19
12.45		Lunch		
	2.	Effects of the Accident		
2.00	a.	The Authority's Reactor and Plant	Dr R S Peckover	HSSC(87)P20
2.30	b.	The Impact on the UK Nuclear Industry	Dr F R Allen	HSSC(87)P21
3.00	c.	The Impact on International Nuclear Safety Issues	Dr M R Hayns	HSSC(87)P22
3.30	3.	Discussion - Summary	Dr J H Gittus	

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Secretary

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Distribution

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GNSR Steering Committee
Symposium Participants

Not for Publication

HSSC(87)P24

CHERNOBYL : ONE YEAR AFTER

1. Introduction

An all day Symposium entitled 'Chernobyl : One Year After' was held for Authority Staff under the auspices of HSSC at Risley on 27 May 1987. Eight HSSC papers were produced for this symposium and formed the bases of the respective presentations. These papers are summarised here and a copy of the programme is attached.

2. Introduction by Dr M R Hayns (Representing Dr J H Gittus)

Dr Hayns started his introduction by pointing out how much our understanding of the accident at Chernobyl had increased over the preceding 13 months. In particular he referred to the information meeting in Vienna in August 1986 from where most of the basic information about the accident was obtained. In fact, little additional information has come from the Soviet Union since that meeting. The last 13 months has not only increased our understanding but enabled us to respond to the accident in a well thought-out way, in particular a detailed conparison with UK plant has been possible.

Dr Hayns went on to say that the symposium could be divided into two parts:- the morning session using comparisons between Soviet and UK safety provisions, what physical phenomena occurred during the accident and how our understanding of source terms and environmental impact have developed; and the afternoon session being much more interpretive covering implications for UKAEA plant, for the UK Nuclear Industry as a whole and for International Nuclear Safety Issues.

3. Comparative Safety Provisions - G M Ballard (HSSC(87)P16)

In this paper Dr Ballard gave a brief comparison between some aspects of the Russian RBMK reactor (as typified by the Chernobyl plant) and the standards, criteria, guidelines and principles which are commonly used for the UK plant. Reactor safety principles can be conveniently grouped under three main headings:

- inherent safety features
- engineered safety systems

- man-machine interaction

and these formed the three main headings of the paper.

3.1 Inherent Safety Features

A fundamental feature of all potentially hazardous plant is that they should be designed, as far as reasonably practicable, to provide inherently safe response to perturbations from normal operation.

a. A failure, malfunction or maloperation should produce no significant operational response in the plant.

b. A failure, malfunction or maloperation should produce a change in the plant state towards a safer condition.

This principle has been paramount in reactor designs in the UK and in particular has been applied to the issue of core reactivity behaviour.

While some reactor perturbations do require additional safeguards, the general UK design intent has been to provide a large measure of inherent safety by utilising physical properties of the reactor, to maintain stable operation.

Turning to the design of the RBMK reactor the situation is significantly different. In some circumstances close to the normal operating envelope (below 20% power) the combination of fuel and void reactivity coefficients resulted in a positive power coefficient, and hence an unstable core, with the possibility that minor perturbations could lead to large increases in power.

All reactors have some adverse characteristics which require engineered safeguards to be provided. However the RBMK did not take the maximum benefit from potential inherently safe design features.

3.2 Engineered Safety Systems

Where a reactor response to perturbation cannot be reasonably designed to be inherently safe then the NII guidelines indicate alternative design responses.

Following a failure, malfunction or maloperation the plant should be rendered safe by the action of engineered safeguards which are continuously available in the state required to control the fault.

Failing that, the plant should be rendered safe by the action of the engineered safeguards which are brought into service in response to the fault.

These guidelines are interpreted within the design of UK reactor plant to require an extensive investigation of potential reactor accident sequences, no matter how unlikely. The design of the safeguard systems such as the shutdown system then reflects the most onerous conditions within this design basis.

These provisions are supported as required by diverse systems for reactor shutdown and hold down, completely separated from normal operation control systems.

When viewed against UK practice some aspects of the design and operation of the RBMK engineered safety system appear inadequate.

The shutdown system itself employed multiple absorber rods which entered the core slowly (approximately 20 seconds for full insertion). The shutdown system had no diverse back-up and the rod system itself was capable of being rendered ineffective if the normal operational control rods were withdrawn too far out of the core.

The Russian designers understood that the reactor characteristics were unstable below 20% power and that in such circumstances the shutdown system would be inadequate. However there was no engineered interlock to prevent operation in such a condition or trip parameter to ensure adequate plant protection.

Finally, the design of the RBMK core is such that a complex instrumentation system would usually be required to detect the core state and provide reactor trip protection. In practice the Chernobyl instrumentation was almost certainly incapable of providing adequate trip information.

3.3 Man-Machine Interaction

Operational staff clearly see themselves as active elements in plant safety and unless this is recognised there exists the potential for a mismatch between man and machine. Even with automatic systems there remain some operational safety issues that are essentially human actions.

However this is an area which, while significant to safety, is poorly represented in meaningful guidelines in current documentation. Such guidelines as exist are rather unspecific and expressions of hope rather than based on a sound analysis, for example:

> "Unauthorised access to and interference with safety related structures, systems and components should be prevented by suitable measures",

and "Sufficient information should be made available to the operator at all times to enable an accurate appreciation to be made of the plant state..."

A review of the RBMK reactor would identify a number of areas in which the man-machine interaction had been inadequately recognised:

i The reliance on the operator to maintain the effectiveness of the shutdown system.

ii The absence of hardware interlocks to back-up the operating limit at less than 20% power.

iii The ability to simply override the reactor protection system.

iv The inadequacy of plant instrumentation to inform the operator of the plant state.

v The very high workload on the operator for normal plant control.

There exist a number of guidelines that encapsulate some of the principles of reactor design and operation. However these principles are extremely difficult to observe in practice, particularly when a further, perhaps overriding, requirement is that plant design should be as simple as possible. The increasing requirements for plant safety currently appears to involve increasingly complex plant systems with the parallel potential for misunderstanding and unexpected performance.

4. Accident Phenomenology - A N Hall (HSSC(87)P17)

4.1 Accident Progression

At a special IAEA meeting in Vienna in August 1986, the Soviets gave detailed information about the events leading up to the Chernobyl accident and presented the results of their analysis of the final stages of the power transient that resulted in the destruction of the reactor. Actions taken by the operators during preparations for a turbogenerator experiment had resulted in the reactor safeguards being disabled. At the start of the turbogenerator experiment, the coolant entering the core was only just below the boiling point and there was very little steam in the core, even at the fuel channel outlets. When the experiment began and main coolant pumps started to run down, the water in the core started to boil rapidly, tending to increase the reactivity of the core due to the reactor's positive void coefficient of reactivity. At first the automatic control rods were able to counteract this effect, but they soon reached the end of their range of effective operation and the reactivity and power then began to rise. As the reactor was at this time operating in an unstable regime where its power coefficient was positive, the power then began to run away. The excess reactivity rose so much that the reactor became prompt-critical, the fuel overheated and disintegrated, steam explosions occurred in individual fuel channels causing them to rupture and the resultant overpressure of the reactor vault caused the pile cap to be blown off.

4.2 Analysis of the Russian Data - Introduction

During the turbogenerator experiment, the centralised control system of the reactor was monitoring and recording various details of the state of the reactor. The data provided was found useful by the Soviets in validating a mathematical model of the state of the reactor from about four minutes before the experiment began to the time at which the reactor was destroyed. Western experts have confirmed the qualitative validity of the Soviet description of events.

Sufficiently detailed information about the Chernobyl reactor and the circumstances of the accident is now available for independent quantitative analyses of the course of the accident to be attempted. Analyses by teams from the USDOE and AECL have led to the identification of a number of possible phenomena that the Soviets neglected in their description.

4.3 The USDOE Team Analyses

The USDOE team divided the Chernobyl accident sequence into three stages, each of which was analysed using several computer codes. These stages were:

1. the initial power increase (from 1:12 to about 1:23:40)

2. the rapid power excursion (from about 1:23:40 to about 1:23:43)

3. the energetic events (after about 1:23:43)

a. The Initial Power Increase

Any quantitative analysis of the Chernobyl accident must take into account the information provided by the Soviets (Figure 1) so the USDOE team made a considerable effort to understand this. As a result, they were able to draw attention to two areas where particular care is needed in the interpretation of the figure: first, the control rod motion in the figure is hypothetical; second, the USDOE analyses have shown consistently that the pressure build up in a closed system suppresses a power excursion, so the Soviet curve M cannot represent the total steam leaving the system as stated in the Figure 1 key.

The USDOE calculated power excursion clearly establishes the type of transient displayed in Figure 1 ie a power excursion driven by the large positive void coefficient. It should be noted however, that a small transient positive reactivity insertion had to be assumed at the start of trip to produce this result.

b. The Rapid Power Excursion

Although the power surge that led to the destruction of the reactor could have occurred even had there been no positive reactivity insertion, it is clear that such a mechanism would have accelerated an already growing excursion. Several possible mechanisms were identified by the USDOE team:

a failure of the pressure tube transition joints

opening of the pressure relief valves or rupture disks in the pressure circuit;

anything else that released steam from the pressure circuit;

transient positive reactivity insertion on trip.

c. The Energetic Events

For preliminary calculations heat transfer from the pin to its surroundings was modelled by two limiting cases:

1. where the cladding outer surface remained at the steam saturation temperature of 573K;

2. where the pin was heated adiabatically.

The predicted areal fuel melt fraction at cladding failure is 57%, the liquid fraction of molten fuel is 10%. The average void fraction in the fuel channel at about this time is predicted to be significantly higher (72%) than the Soviet analytical result (40%), but in either case fuel failure would lead to forced intermixing of fuel materials and water, providing the potential for fuel coolant interactions. Failure of the fuel channel due to FCI pressurisation alone was not predicted, although local failures in the core region due to fuel impingement effects were considered likely.

d. Best Estimate of Accident Events

The calculations outlined above were all based on average channel conditions and uniform axial power, and it was found that, rather than a double-peaked power excursion occurring as determined by the Soviets, the noncoherency of the fuel failures caused the first power surge to escalate into a single large event. This was less severe in terms of peak power, energy deposition, fuel temperature and fuel vapour pressure than the Soviet's double peaked excursion, but the fuel vapour pressure was nonetheless predicted to augment significantly the destructive event.

4.4 The AECL Analyses

Physicists at Atomic Energy of Canada Ltd have reached the conclusion that the power surge was initiated by the movement of the control rods and their graphite followers when the trip button was pressed. The graphite followers displaced water from a region of high neutron flux near the bottom of the core as the control rods were inserted, resulting in a positive reactivity insertion. In their view, the positive void coefficient of the Chernobyl reactor was not a significant factor in the accident.

4.5 Discussion of Phenomenology

a. The Role of the Graphite Followers

When the control rods are fully withdrawn, the followers are at the axial centre of the core and as the control rods begin to move into the core, they displace water leading to local transient insertion of reactivity.

The USDOE view is that the movement of the graphite followers caused a local positive reactivity excursion in the lower part of the core that accelerated an already developing power excursion. The positive void coefficient/power coefficient is still regarded as the fundamental cause of the accident.

The AECL physicists believe, however, that the positive reactivity insertion associated with the graphite follower movement was the dominant cause of the accident as, in the absence of the follower effect, the excursion would have been sufficiently slow for the shutdown system to terminate it before significant damage occurred. Nonetheless, for this view to be self-consistent, the speed of shutdown would have to be assessed in terms of the capability to suppress a local excursion in the lower part of the core rather than in terms of the overall rate of reduction of reactivity.

It should be noted that a positive reactivity insertion in the lower part of the core might alternatively have occurred as a result of cavitation in the main circulating pumps injecting vapour bubbles into the bottom of the core.

b. Fuel-Coolant Interactions

The USDOE analysis of fuel-coolant interactions was based on the assumption that particulate fuel debris would be ejected into liquid coolant in the fuel channels, causing it to vaporise rapidly. The fuel channels would then fail near the top closure heads, but not within the core where failure would lead to overpressurisation of the reactor vault and displacement of the pile cap. This type of "steam explosion" could not, therefore, have been responsible for the destruction of the Chernobyl reactor.

c. Conclusions

The USDOE and AECL analyses generally confirm the description of the accident given by the Soviets at Vienna and are in general agreement with the UK view of the accident development. Those minor differences that have arisen are related to assumptions made in areas where inadequate detailed information is available and reflect legitimate differences of view between the experts in the field.

5. Fission Product Release - Source Terms - P N Clough (HSSC(87)18)

No new data has become available since the Vienna information meeting in August 1986. However, since then various reviews and reinterpretations have been performed by USDOE/NRC, AECL,CEA, JAERI, OECD/CSNI and SRD.

5.1 Magnitude of Release

The Soviet assessment of the activity release subsequent to the Chernobyl accident is given in Table 1. This assessment was based on measurements of ground deposition at 300 locations over several months. Their estimate is that 20MCi total was deposited in the 30km exclusion zone and a further 30-50MCi elsewhere in the Soviet Union. Deposits outside the USSR are ignored.

NRPB measurements throughout Europe carried out on behalf of the OECD/NEA and trajectory modelling at Imperial College have led to the conclusion that the Russians may have underestimated the I and Cs release by a factor of 2 or more.

Because they measured ground deposits, the Russian estimate of I and Cs is dominated by particulate forms where as measurements at remote locations were dominated by fine aerosol forms of I or Cs (and also Te, Ru). Using this data, back calculations of the source term indicate release fractions for I of 30-60% and of Cs of 20-40%. The uncertainties on these figures allow for a strong possibility that 100% of the I was released and that the sudden termination of release on 6 May 1986 was as much to do with depletion of inventory as to counter measures.

5.2 Composition of Release

The Soviet data is insufficient to give day by day releases of the different radionuclides but SRD have been able to perform an interpolation which is given in Figure 2. Four phases of the release:

1. an initial release on 26 April;

2. a declining release over the next few days;

3. a second large release on 3-5 May;

4. termination on 6 May.

In particular the large increase in I, Te and Ru release during stage 3 should be noted.

SRD's conclusions are that, on average 30-60% of the volatiles were released together with 3-6% of the non-volatiles. This latter figure is much higher than would have been expected.

5.3 Release Mechanisms

During stages 2 and 3 of the release it appears from the data that the non-volatile fission products Sr, Ba, Ru, Mo, La, Ce and the actinides all had very similar release fractions. Two release mechanisms have been suggested: oxidative and reductive, the latter being the Russian suggestion and latched onto by organisations in the USA.

a. Oxidative Release

Oxidation of UO_2 to U_3O_8 is the principal reaction during this process. Between 800K and 1000K this promotes I, Te or Ru vapour release but also spalling of U_3O_8 aerosol leads to indiscriminate release of non-volatiles. Above 1800K U_3O_8 vaporisation also leads to indiscriminate release. The process would have been terminated by N_2 injection. This oxidation process also accounts for the almost pure Ru particles found in Sweden: RuO_2 oxidises to RuO_3 and RuO_4 which on release then condenses back to RuO_2.

b. Reductive Release

This process is driven by reaction of UO_2 and CO. To proceed, however, it requires temperatures in excess of 2000K, very low oxygen partial pressures ($<10^{-3}$ bar) and direct contact between the fuel and graphite, all of which are rather unlikely. In addition, the process promotes the release of Ba and Ce while suppressing the release of Ru and Mo. This process doesn't , therefore, account for what has been observed.

5.4 Significance for LWR Source Terms

The Chernobyl release has very little significance for assessing possible Western LWR source terms since oxidative release mechanisms are only possible in very particular circumstances eg pressurised melt ejection.

The Russian response to the accident was very rapid and shows how ad hoc methods can be used to limit the source term. Engineered safety features to limit the source term (eg filtered vent containment) should be considered in the light of this experience.

6. Accident Consequences - Fission Product Dispersion - W Nixon and M J Egan (HSSC(87)P19)

This paper considered the time dependent pattern of the spread of contamination throughout Europe and presents preliminary estimates of the collective dose to various countries.

6.1 Atmospheric Dispersion Across Europe of Material Released from Chernobyl

Increased activity levels were first reported on 28 April in Finland and Sweden, where external dose rates in certain locations exceeded normal background levels by a factor of ten or more. On succeeding days elevated radioactivity concentrations were detected throughout Europe until almost complete coverage had been achieved by 3 May. Based upon reported measurements complemented by computer calculations, it has been possible to assemble a picture of the pattern of dispersion as it affected western Europe. The results of this work for 3 May 1986 are shown in Figure 3.

Material transported at very high altitudes (>1km) may have been responsible for the subsequent observations of elevated activity levels in countries bordering the Pacific Ocean.

6.2 Dosimetric Assessment for Western Europe

An estimate of the dosimetric impact on Western Europe may be obtained by utilising monitoring data collected by the various national agencies and consists of the mean (population-weighted) individual dose for various countries, based on estimated mean environmental concentrations.

Various pathways need to be considered:-

a. Inhalation Pathway

In general, significant elevated concentrations of activity in air were monitored for a few days. This data allows a mean, individual, committed effective dose equivalent to be calculated for the inhalation pathway.

b. Ingestion Pathway

Radioactivity transferred to man through incorporation in foodchains is available over a more extended period, so that monitoring data is unlikely to give a full picture of the average intake of activity via foods. Ingestion dose calculations are limited to contributions from milk, green vegetables and meat, and are based on mathematical models assuming consumption rates typical of the UK adult population.

d. Results

The total collective dose for Western Europe is estimated to be approximately 76,000 man Sv, broken down as follows:

Pathway	Collective Dose Man Sv	%
INHALATION	3600	5
INGESTION		
MILK	11000	14
VEG	15200	20
MEAT	12500	17
EXTERNAL	33300	44
TOTAL	75600	

Assuming a linear relationship between dose and risk of cancer, characterised by a cancer fatality risk coefficient of 1.25×10^{-2} per man Sv, the total number of cancer fatalities in Western Europe arising from the accident over the next decades is estimated to be just under 1000.

6.3 Dosimetric Assessment for Eastern Europe

Information on levels of radioactive contamination in Eastern Europe is relatively limited and an assessment of the dosimetric impact for these countries is therefore subject to considerable uncertainty. The total collective dose for Eastern European countries excluding the Soviet Union is estimated to be approximately 100,000 man Sv which corresponds to a total of around 1250 fatal cancers.

6.4 Consequences in the USSR

From among the on-site personnel, some 300 required hospital treatment. These included operating personnel, site emergency squads, and particularly, members of the fire brigades. Of those examined, 203 were diagnosed as suffering from acute radiation syndrome. A total of 29 fatalities has been reported from among those suffering from acute radiation effects. Two additional deaths are reported to have occurred in the immediate aftermath of the accident.

It has been emphasised that no individual member of the population away from the reactor site itself incurred a radiation dose above the threshold for clinically manifest symptoms of acute radiation syndrome.

Off-site people were instructed to shelter indoors on 26 April, and stable iodine tablets were distributed. The population of 45,000 were evacuated from Pripyat in 2 hrs 30 mins, commencing at 2.00 pm on 27 April. Individual exposure appears to have been kept below a level of 750 mSv. During the next few days a further 90,000 people from within a radius of about 30km around the plant were evacuated as the levels of concentration became more widespread.

The Soviet authorities have estimated the collective dose to the European part of the USSR from various pathways. For external exposure they calculate a collective dose of around 3×10^5 man Sv. For internal exposure resulting from consumption of foodstuffs contaminated with Cs, they estimate a figure of around 2×10^6 man Sv.

Using a linear dose-risk relationship, this implies around 7,500 fatal cancers in the European part of the USSR as a result of the accident, compared with the mortality rate from spontaneous cancer of 9.5×10^6 cases in the same population.

6.6 Comparison of UK Dose from Chernobyl with Background Radiation

The average annual dose in the UK from background is around 2 mSv; this may be compared with the estimated 50 year individual dose from Chernobyl of 0.05 mSv. The corresponding collective dose to the UK from background, over the next 50 years, is around 5×10^6 man Sv, which may be compared with the figure of 2.8×10^3 man Sv from Chernobyl.

An alternative way of comparing with background is to consider how the background dose rate varies throughout the UK. This variation can be up to around 1 mSv per year. Thus the 50 year individual dose from Chernobyl of 0.05 mSv corresponds to (say) living in East Anglia and having approximately a three-week holiday in Cornwall.

7. Review of Authority Reactor Safety Prompted by the Chernobyl Accident - R S Peckover (HSSC(87)P20)

The Authority has nine reactors on three sites - at Harwell, Winfrith and Dounreay. Of these the only ones with sufficient radioactive inventory to produce an accident even remotely approaching the scale of Chernobyl are the two prototype power reactors - PFR at Dounreay and SGHWR at Winfrith and the two closely similar materials testing reactors at Harwell - DIDO and PLUTO. This review concentrates on these reactors, though the others have not been ignored. The review has confirmed that our designs, procedures, operator training and emergency plans are soundly based.

7.1 Reactor Safety Environment

The arrangements for controlling the operation of reactors to ensure high standards of safety are generally similar on the three sites. There are five distinguishable elements - the reactor operation staff, the reactor safety committee, the safety documentation, SRD, and the Site Director.

The arrangements indicate that a good management environment has been established in which safety is taken seriously and vigilance is maintained to ensure continued high standards of safety for Authority reactors. By contrast, the development of the accident at Chernobyl suggested an absence of such an environment.

7.2 Core Reactivity Characteristics

At normal full power the large positive void coefficient of the RBMK reactor is more than compensated by the negative Doppler coefficient. However below about 20% of full power the power coefficient is positive, and the reactor is potentially unstable. This instability led to the large power surge and subsequent explosion. What of Authority Reactors?

a. SGHWR

Both components of the power coefficient (Doppler fuel and coolant void coefficients) are negative at all reactor power levels.

The only feasible way in which the void coefficient could be made positive would be the addition of at least 2 tonnes of light water to the 37 tonnes of heavy water moderator - essentially impossible without detection.

b. PFR

Sudden void production in the core centre would lead to a mild temperature transient because of the local positive void coefficient. This would be arrested by the increased effect of the negative Doppler fuel coefficient; prompt criticality would not occur.

Widespread boiling of the liquid sodium coolant would require an unusual fault since even at full power it is more than 300°C below its boiling point.

c. MTRs

For DIDO and PLUTO the void coefficient and overall temperature coefficient of reactivity are negative.

DIDO and PLUTO have an excess reactivity well above the 0.7% required for prompt criticality. However the reactivity available for initiating prompt critical events is relatively small and these events have a very low probability.

7.3 Reactor Shutdown Systems

The criticisms of the RBMK system are that it is clumsy to operate, there is little discrimination between control and shutdown, the shutdown rod response was too slow to handle a fast transient, and control rod withdrawal was not sufficiently physically constrained. The Russians accept that major deficiencies exist in this area and propose to rectify them.

What of Authority Reactors?

a. SGHWR

There are two distinct shutdown systems neither of which is used for reactor control.

The fast acting shutdown system operates in less than 1 second.

There are no means whereby the amount of negative reactivity injected by the system can be modified by any short-term actions on the part of the operators.

b. PFR

In PFR there are 5 control rods and 5 distinct shutdown rods. The maximum time to fully insert all 10 rods is 0.7 sec.

Although the shutdown rods are dedicated to a trip function and the APS has two independent systems, PFR does not have a true diversity in shutdown capability. Designs for a further "ultimate" shutdown device are currently under consideration.

c. MTRs

Some absorbers are used for both control and shutdown but a second diverse system used only for shutdown is provided. For both MTRs the control and shutdown absorbers are interlocked to ensure that criticality cannot be achieved until the shutdown absorbers have been raised. The overall time for the absorbers to drop from fully-out to fully-in is less than 0.6 sec.

7.4 Man/Machine Interface

Each Authority reactor has fast acting and reliable automatic protection systems (APS).

There is no hardware or software device which ultimately prevents the bypassing of parts of the APS (as was done at Chernobyl) in any of the AEA's reactors, indeed bypassing parts of the protection system on occasion is unavoidable. It is essential therefore that when parts of the APS are bypassed, it is done in a controlled and self-consistent fashion under circumstances which are well understood, are safe, and have been planned for in advance.

Bypasses of the APS are listed in the safety documentation and their use anticipated in the safety case. The overriding principle in designing controls on bypass is that the difficulty of bypasses should match the importance for safety.

7.5 Operator Training

Even when comprehensive operator aids and constraints are present to make inappropriate action very difficult in a well designed plant, the safety of the reactor depends on the operators - their integrity, their attitudes and their knowledge.

The rather cavalier actions of the Chernobyl operators seem to imply a casual attitude to operational procedures and a belief that the actions would not matter either from woeful ignorance or misplaced confidence.

In the Authority, the standard of training and authorisation applied to the staff operating reactors is considered to be high.

For each post, one may identify four stages: selection, pre-authorisation training, authorisation and post-authorisation training.

Once authorised, the responsible officers are subjected to a continuing training and refresher programme, with an annual review of performance and suitability for the job.

7.6 Experiments in Reactors

The Chernobyl accident occurred during an experiment which clearly distracted the focus of attention away from the safety of this civil power reactor.

The Russians have stated that the experiment was badly planned and its safety case was inadequate and not properly reviewed.

How does this compare with Authority practice and experience? The most important point is that the Authority has been running research, experimental and prototype reactors for many years. This means that a dominant strand in its reactor programme has been the carrying out of experiments and tests and inserting experimental rigs into its reactors. A large number of experiments have been successfully and safely carried out. This is not surprising because experiments are taken seriously and the systems for running the reactors are tailored to cope with a variety of experiments.

7.7 Fire

An important contribution to the dispersion of radioactivity from Chernobyl were the intense fire in the reactor core.

This raises two issues - the presence of combustible materials in the reactor complex, and the arrangements for preventing and combating fire. What is the situation for Authority reactors?

i. Combustible Materials

a. SGHWR

This reactor has a liquid moderator (heavy water); there are no conventionally combustible materials in the core region. Channel power is limited so that a clad temperature of 1200° (the onset of the zircaloy/water reaction) is not exceeded even for the worst design basis loss-of-coolant accident.

b. PFR

Because of the presence of sodium, PFR incorporates design features to prevent sodium fires.

The PFR fire precautions were formally reviewed about a year ago and found to be satisfactory after some modification to one system.

c. MTRs

The use of combustible materials within the reactor containment is carefully controlled.

A severe and highly improbable accident involving major loss of coolant and air ingress would be required before burning of the metal fuel became possible. The graphite reflector is in a steel secondary containment tank and is at a low temperature. Therefore the potential for graphite exacerbating an accident is low.

ii Fire Fighting

At each site there are full time, dedicated fire officers - properly equipped and trained to fire fighting standards. 1 the event of a fire the fire brigade, which is on duty at all times, is alerted immediately. As a precaution they patrol the site and in particular the risk areas. The fire brigade is regularly exercised, and also takes part in the emergency exercises which each site holds.

7.8 Emergency Planning

For a number of years a document summarising emergency plans at a site has been available in local County reference libraries. Recently, the UKAEA decided to publish its Emergency Handbooks and those for Dounreay, Harwell and Winfrith have now been published.

The scale of Chernobyl suggests that each site needs a fall-back operations centre off-site, 10 miles or so distant from site. Dounreay has Tollemache House in Thurso, Winfrith and Harwell are making appropriate arrangements.

8. The Impact on the UK Nuclear Industry - F Allen (HSSC(87)P21)

As yet, the full implications of the Chernobyl Accident are still working themselves through. However, we can already see some of these effects and the purpose of this paper was to review the current situation within "the UK industry", which will be interpreted widely as meaning all those bodies interested in promoting nuclear power in the UK.

8.1 The Immediate Response of the UK Industry

Immediately following the accident, Lord Marshall re-convened the Industry Task Force which drew together representatives of the industry. This provided the technical backing for a robust response to be made to the situation. Advice was provided to the Secretary of State and briefing material for staff (at all levels) who had contact with the media or the public. The Task Force also provided information which enabled industry staff, themselves, to be better informed as to what had occurred and what the implications might be.

Initial responses to the accident took a variety of forms. The design of the RBMK was such that no close parallel to the design of UK reactors could be discerned. However, attention was focused on operator training and on emergency planning. The latter concern led to the publishing by the CEGB, UKAEA and others of emergency plans for reactor installations. Also whilst the long term review of the Magnox stations had already been going on for some time, attention was focused on it.

8.2 The Watt Committee on Energy

The Watt Committee on Energy has set up a Nuclear Safety Sub-Group to look at the implications of Chernobyl for the UK. The objective was produce an informed but impartial view to be directed towards technical people outside the industry. This body has carried out studies over the last six months and has come up with some interesting, if predictable, recommendations concerned with increased emphasis on operators and operator training, research into relevant areas, international co-operation and staffing levels at NII.

8.3 Emergent Issues

Various issues have emerged as being of relevance to nuclear power in the UK. These include the following:

a. Human Error and Operator Training

This was an area which had been highlighted already by the TMI accident and so its importance had already been recognised, even before the Chernobyl accident.

b . Emergency Preparedness

The CEGB, UKAEA and other operators have now made public their emergency plans. In addition, there has been increased emphasis in the last 12 months on emergency exercises.

c. Containment

The proposed PWR at Sizewell will, of course, have a strong containment, but the existing Magnox and AGR reactors have no containment other than the primary pressure boundary.

d. The Proposed PWR at Sizewell

The PWR proposed for Sizewell 'B' has been seen to have a number of important safety features when compared to the RBMK design.

The effect of the Chernobyl accident on the debate surrounding the Sizewell proposals has not, therefore, been affected by technical issues raised by the Chernobyl accident. Indeed, the recent Greenpeace report "Chernobyl UK" attacks the UK gas reactors not the proposed PWR.

e. "Chernobyl UK" - The Greenpeace Report

To coincide with the anniversary of the accident, Greenpeace produced a report "Chernobyl UK" which sought to show that a accident like that at Chernobyl could occur in a UK gas reactor. The report focused on the positive moderator coefficient, common to both UK gas reactors and the RBMK, but which in fact played no significant role in the Chernobyl accident, and could not produce a fast power transient in any UK reactor.

The document represents a totally unfounded attack in which the basic safety characteristics of the UK gas reactors are grossly misrepresented.

f. Political Issues

Following Chernobyl, both Labour and Liberal/SDP parties adopted policies which are against nuclear power. This aspect, of course, is dominated by the coming General Election.

Clearly the outcome of the election could have a significant effect on the future development of the UK industry. This is potentially the most important impact of all and the one with, at the time of writing, the greatest uncertainty.

9. The Impact on International Safety Issues - M R Hayns (HSSC(87)P22)

Whilst it had been understood for many years that the long range transportation of radionuclides following a severe reactor accident would give rise to effects beyond transnational boundaries, the full appreciation of its impact was not crystallised until the accident at Chernobyl did just that. This served to emphasise the interdependence of neighbouring countries on the way in which they conduct their internal affairs relating to the safe design, construction and operation of nuclear power plants.

In this paper the response of the major international agencies to the accident is discussed individually.

9.1 The International Atomic Energy Agency (IAEA)

The IAEA is the only international response to the Chernobyl accident. The reasons for this include:

a. The IAEA is the only international collaboration in the relevant fields - it was therefore a natural choice for the Soviet Union for a channel of communication.

b. The IAEA have a long record of international collaboration in the relevant fields - there was an established hierarchy and administration available to deal with the problem.

c. The IAEA offered access not only to Eastern block countries but also developing nations and many aspects of the "international problem" concerning those countries too.

d. The IAEA was 'endorsed' as the preferred agency by the Tokyo Summit in May.

The principal activities of the Agency post Chernobyl were:

i The establishment and signing of the conventions on early warning and mutual assistance.

ii The "Information Meeting" in August at which the Soviet Union gave a very full account of both the causes of the accident and its effects.

iii The General Conference of the IAEA at which national statements concerning the internationalisation of nuclear safety were given at Ministerial level.

iv The Expert Meeting held in November at which the supplementary programme of the IAEA was reviewed and recommendations made to the Board of Governors meeting in December.

9.2 The Nuclear Energy Agency of the OECD

Following the Chernobyl accident great benefit was derived from the network of contacts established over many years within the framework of the Committee on the Safety of Nuclear Installations (CSNI) and the Committee on

Radiological Protection and Public Health (CRPPH). For the first week or two links set up earlier in various groups reporting to the CSNI provided an efficient channel for rapid and accurate information on radiation levels and radioactivity measurements in Northern Europe.

Early in May an extraordinary meeting of CSNI and CRPPH was held in Paris to pool information on the accident and a small ad hoc group was set up to prepare a report on Chernobyl for CSNI. The ad hoc group was asked to consider the following aspects:

Information needed from USSR to enable OECD countries to assess the accident

Brief description of RBMK reactors

Review of Soviet measures to upgrade safety following Chernobyl in the light of OECD safety philosophy

Review of post TMI-2 improvements in OECD countries to see what further should be done after Chernobyl

A final version is likely to be published by the NEA around the middle of the year.

Further measures adopted included strengthening the NEA Incident Reporting System with greater emphasis on identifying possible severe accident precursors and greater emphasis on containment and the modelling of accident consequences.

9.3 The European Communities

Activities within the European Community are of a different nature than those described for the other international agencies because of the legal structure of the EEC and to the fact that it funds and manages its own R&D programmes in the area of Health and Safety.

One of the Commission's major concerns following Chernobyl was the control of movement of foodstuffs in the Community. A complete ban of certain Eastern Bloc imports was imposed early in May and it was only after this that proper consideration was given to intervention levels for contamination of foodstuffs. There is clearly considerable disagreement on intervention levels within the Community which the commission are finding it difficult to resolve.

The EEC's R&D programme has not been significantly altered to account for any post-Chernobyl insights. Any impact on the general sense of priorities within the Framework R&D programme will depend on negotiations over the size of the Framework programme as a whole.

9.4 International Issues Raised by the Chernobyl Accident

Resolution of the technical issues involved in the overseas to the accident was very important task. Of perhaps greater importance to the long term 'international climate' on Nuclear Safety has been a series of ramifications from the accident which are less easy to resolve. These issues are illustrated by the following examples.

i International Regulation and Licensing

One of the immediate responses of the many Governments to the Chernobyl accident was to call for immediate imposition of binding international standards of nuclear safety. In the main the call for such a scheme has come from developed countries which have themselves no nuclear power programme. This is a very difficult issue since on the face of it, such a scheme might be appealing. However, the practicalities of the problem are that trying to find mutually acceptable international safety standards is like seeking the Holy Grail.

As a way forward, Mr Walker proposed a scheme whereby the IAEA should update its knowledge of existing inspectorate's operations, promote best practices and perhaps in due course, form an expert team which could be responsible to peer reviews. This is the 'Walker Initiative' and represents a middle line whereby regulatory systems and practices may be reviewed rather than attempting to define international regulations. This proposal met with a good deal of support at the IAEA and is likely to form the basis of future Agency activities in this area.

ii Pooling Nuclear Safety Information

Many of the developing countries see the international activities of the IAEA as a means whereby knowledge acquired by the developed countries can be made available to them. The accident at Chernobyl has highlighted a particular dilemma for the Western countries. On the one hand passing information freely to other countries can only be done at some cost, particularly in the time of relatively few experts. In the other hand it is clear that if other countries are to develop nuclear power then it is in all of our interests to ensure that it is done as safely as possible hence we should made such information available to them.

iii The Role of Developing Countries

Coupled to the previous point is the attitude to take towards the aspirations of developing countries for nuclear power. In many such countries the infra structure does not exist to support such an enterprise locally and there would remain a strong dependence on the nuclear vendor. In politically volatile regions this could have serious ramifications. However, through both multi and bi-lateral efforts the evolution of currently under developed countries into nuclear operating states can be envisaged with some confidence that the technology will be properly handled.

The evaluation of the idea of the need for a "safety culture" in the operation of nuclear plant is perhaps one of the most important lessons drawn from the Chernobyl accident and there is some evidence that it is being taken seriously.

10. Discussion

A lively discussion at the end of the meeting explored whether this accident was in any sense a "bounding case". In terms of release fractions, at least, it was decided that it was.

Other questions discussed were the usefulness or otherwise of expressing risk from non-nuclear sources of radiation as expected deaths and the desirability of collecting "near miss" data from nuclear plant.

The speakers are to be congratulated for contributing to an informative and stimulating meeting.

TABLE 1

SOVIET ASSESSMENT OF THE ACTIVITY RELEASE

FROM CHERNOBYL-4

	26 APRIL		6 MAY	
	MCi	%	MCi	%
Xe133	5	2.8	45	100
Kr85m	0.15	1.7	-	100
Kr85	-	-	0.9	100
I131	4.5	5.3	7.3	20
Cs134	0.15	3.0	0.5	10
Cs137	0.3	3.9	1.0	13
Te132	4	5.4	1.3	15
Sr89	0.25	0.37	2.2	4.0
Sr90	0.015	0.27	0.22	4.0
Ba140	0.5	0.38	4.3	5.6
Mo99	0.45	0.28	3.0	3.2
Ru103	0.6	0.45	3.2	2.9
Ru106	0.2	0.35	1.6	2.9
Zr95	0.45	0.34	3.8	3.2
Ce141	0.4	0.27	2.8	2.3
Ce144	0.45	0.51	2.4	2.8
Np239	2.7	0.38	1.2	3.2
Pu238	1(-4)	0.48	8(-4)	3.0
Pu239	1(-4)	0.50	7(-4)	3.0
Pu240	2(-4)	0.50	1(-3)	3.0
Pu241	0.02	0.42	0.14	3.0
Pu242	3(-7)	0.43	2(-6)	3.0
Cm242	3(-3)	0.43	2.1(-2)	3.0

FIGURE 1

USSR ANALYSIS OF THE INITIATION OF

THE ACCIDENT AT CHERNOBYL-4

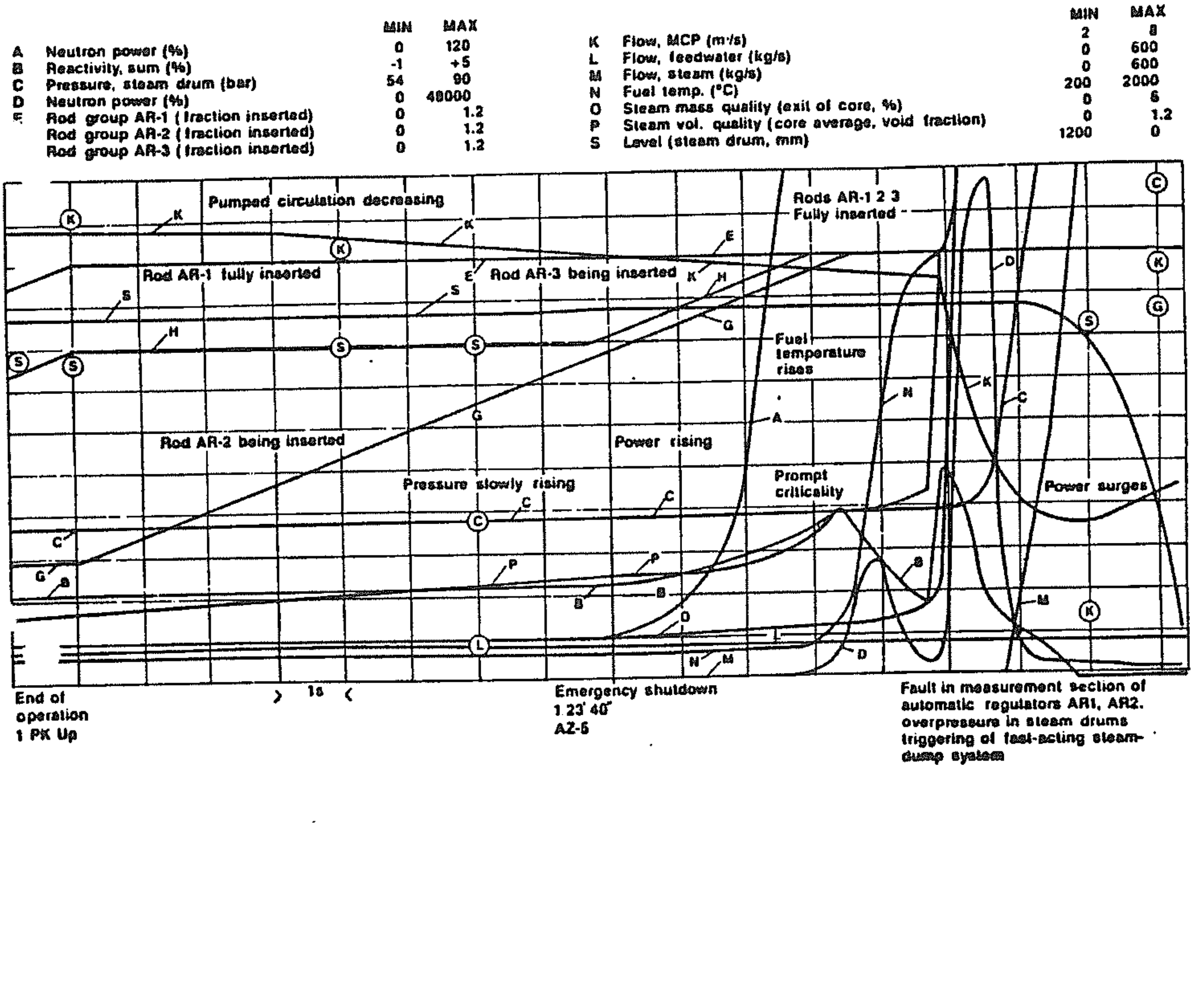


FIGURE 2

SRD INTERPOLATION OF ISOTOPE SPECIFIC

DAILY RELEASES (/MCi)

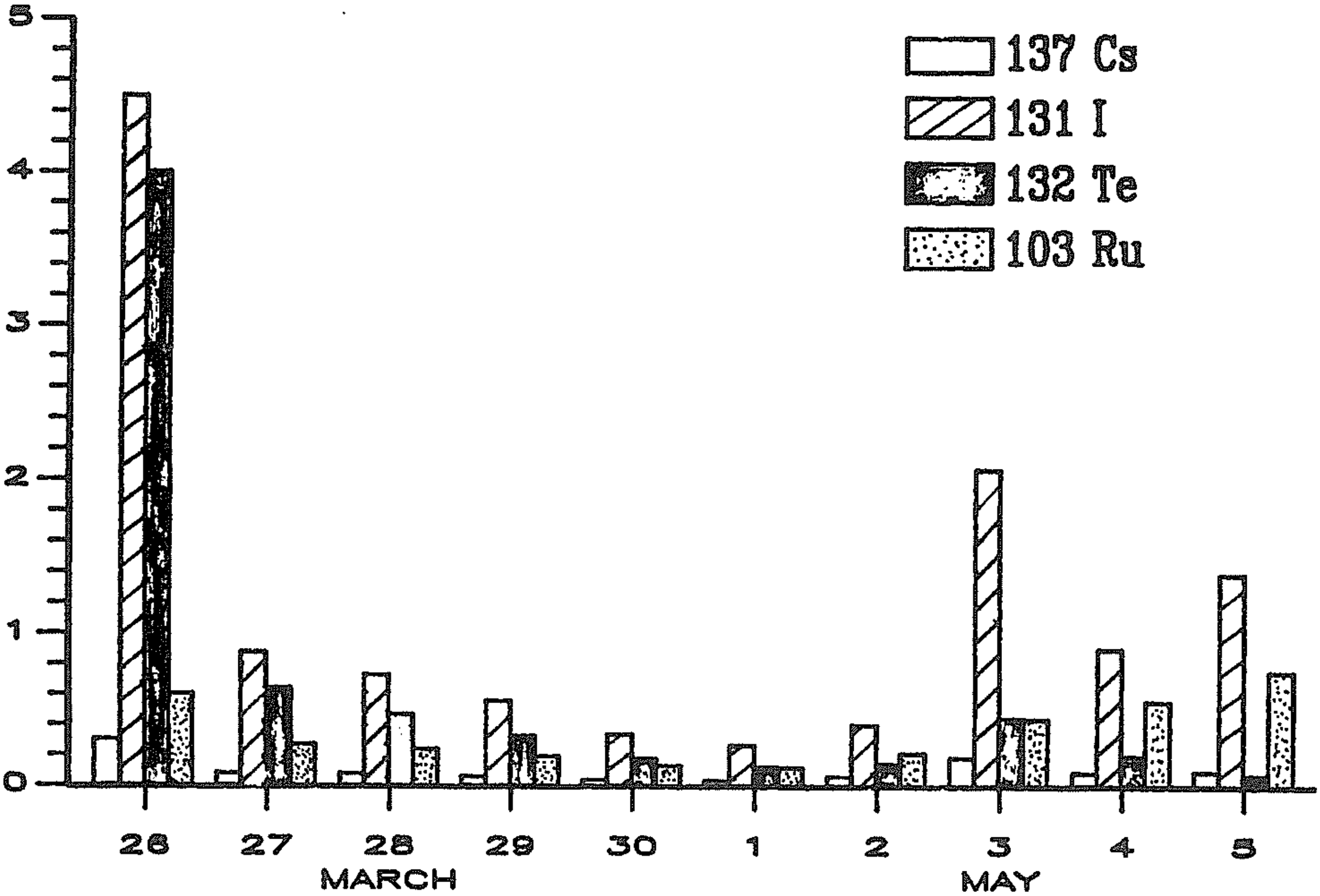
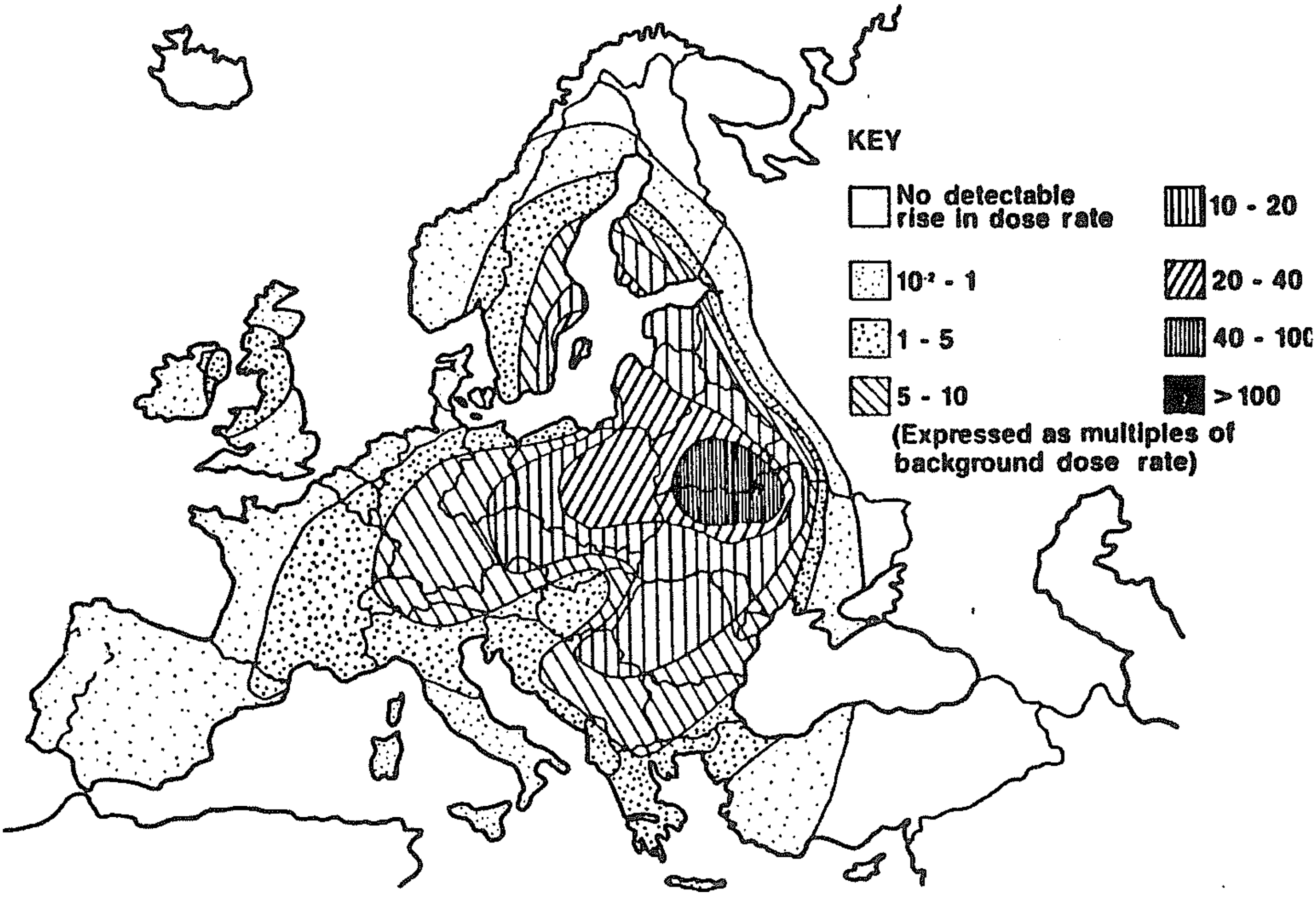


FIGURE 3

<u>RADIATION DISPERSION PATTERN</u>

<u>ACCROSS EROPE 3 MAY 1987</u>



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UNITED KINGDOM ATOMIC ENERGY AUTHORITY

HEALTH AND SAFETY STUDIES COMMITTEE

Comparative Safety Provisions

by

G M Ballard

SRD Culcheth
May 1987

Distribution:

Standard HSSC

HEALTH AND SAFETY STUDIES COMMITTEE
CHERNOBYL SYMPOSIUM

Comparative Safety Provisions

G M Ballard

1. In this paper a brief comparison is made btween some aspects of the Russian RBMK reactor (as typified by the Chernobyl plant) and the standards, criteria, guidelines and principles which are commonly used for the UK plant. Reactor safety principles can be conveniently grouped under three main headings:

   . inherent safety features

   . engineered safety systems

   . man-machine interaction

   and this general format will be followed in this paper.

2. A fundamental feature of all potentially hazardous plant is that they should be designed, as far as reasonably practicable, to provide inherently safe response to perturbations from normal operation. Such provisions may be difficult or impossible to achieve, eg a large passenger aircraft, or may be judged on unnecessary grounds, eg roll-on-roll-off ferries. However for nuclear power plant the NII guidelines clearly indicate a very strong hierarchy in the possible plant responses to a perturbation or fault. Following principle 29, in the HMNII safety assessment principles for nuclear power reactors, the two preferred responses are:-

   (a) A failure, malfunction or maloperation should produce no significant operational response in the plant

   (b) A failure, malfunction or maloperation should produce a change in the plant state towards a safer condition.

   In other words the plant design should, if possible, be insensitive to perturbations or faults but if a response is unavoidable then the deviation in plant state should be towards a safer plant state. This principle has been paramount in reactor designs in the UK and in particular has been applied to the issue of core reactivity behaviour. Thus as a general rule the intention has been to design a core such that the reactivity is insensitive to perturbations or that a safer state (ie reduced reactivity) is produced. For example in the AGR the reactivity is insensitive to changes in coolant conditions such as flow, pressure, temperature etc. For the PWR the reactivity is reduced by increasing coolant temperature or void fraction.

In all reactors the reactivity response to increasing fuel temperature is overall negative. Of greatest importance is the fact that the prompt power co-efficient, ie the combination of the various reactivity effects in a power perturbation, is always negative.

It should be noted however that a negative void reactivity coefficient, for example, is not automatically an inherently safe feature. While it ensures that increased voidage will lead to reduced core reactivity it also leads to a situation in which void collapse or coolant temperature reduction increases core reactivity. The potential therefore exists for damaging reactor transients, eg steam line break, which result directly from the negative void reactivity coefficient and which require engineered safeguards to protect the reactor from damage.

Thus, while some reactor perturbations do require additional safeguards, the general UK design intent has been to provide a large measure of inherent safety by utilising physical properties of the reactor.

3. Turning to the design of the RBMK reactor the situation is significantly different. The core is basically overmoderated (for neutron economy) so that in most circumstances an increase in the void fraction of the coolant will lead to an increase in the core reactivity. Under normal design circumstances this positive void coefficient could be counterbalanced by the negative fuel coefficient to produce a negative power coefficient and thus a stable reactor to minor perturbations. However in some circumstances close to the normal operating envelope (below 20% power) the combination of fuel and void reactivity coefficients resulted in a positive power coefficient. This produces an unstable situation in which rising power produces more coolant voidage, resulting in increased reactivity and thus faster rises in power - a runaway situation. The Russian designers recognised this situation but decided to accept it rather than alter the core design to eliminate this feature. All reactors have some adverse characteristics which require engineered safeguards to be provided. However the RBMK did not take the maximum benefit from potential inherently safe design features and in particular had a potential for an unstable situation close to the normal operating envelope.

4. Where a reactor response to perturbation cannot be reasonably designed to be inherently safe then the NII guidelines indicate alternative design responses:-

(c) Following a failure, malfunction or maloperation the plant should be rendered safe by the action of engineered safeguards which are continuously available in the state required to control the fault.

(d) Following a failure, malfunction or maloperation the plant should be rendered safe by the action of engineered safeguards which need to be brought into service in response to the fault.

For many purposes the reactor reactivity control and shutdown system is the most critical of the engineered safety systems, and would normally be expected to be designed to particularly high standards reflecting, for example, such NII guidelines as:-

NII 43 - The design aim should be to prevent any operating mode or fault sequence causing any safety related item to exceed safe limits. To this end:

(a) All fault sequences and combinations of fault sequences which might cause a radiation hazard should be identified, representative or bounding faults analysed, and appropriate monitoring and protective systems provided where necessary.

This guideline needs to be viewed in combination with another very important statement.

NII 121 - It must be recognised that unforeseen plant or protection system faults or maloperations may occur. Protection systems should reflect this aspect by, for example, the provision of reasonably practicable diversity and redundancy, both within each system and in the nature of each input and output.

These guidelines are interpreted within the design of UK reactor plant to require an extensive investigation of potential reactor accident sequences, no matter how unlikely. The design of the safeguard systems such as the shutdown system then reflects the most onerous conditions within the design basis of the plant. In addition the high reliability requirements are reflected in the appropriate use of diverse protection.

UK practice for all reactor systems is to provide automatic reactor protection based on either:

- Two diverse signals for detecting the fault condition or tripping the reactor

- One interlock to prevent the condition arising and one trip signal for detecting the condition and tripping the reactor

- Two diverse interlocks to prevent the condition arising.

These provisions are supported as required by diverse systems for reactor shutdown and holddown. The main shutdown system is usually a set of gravity-driven absorber rods which provide a fast acting response to any plant transient; typically AGR shutdown rods insert 5 niles in approximately 3 seconds. Further shutdown systems such as nitrogen injection, boron balls or beads and boron dust are provided in various combinations on Magnox and AGR stations.

A further aspect of UK guidelines on engineered safety systems is the intended separation from normal operation control systems. Thus the NII say:

NII 115 - "The required performance of components, subsystems and systems should be stated and shown to be adequate for the purpose of providing protection. Limits should be defined outside which components etc should not be operated and provision should be made to ensure that these limits are not infringed. It should be shown that the overall reliability of the protective system is adequate".

NII 120 - "When equipment has more than one function, one of which is to ensure nuclear safety, this equipment should be classed as protection equipment. The protective function should not be jeopardized by other functions.

5. When viewed against UK practice some aspects of the design and operation of the RBMK engineered safety systems appear inadequate.

The shutdown system itself employed multiple absorber rods which entered the core slowly (approximately 20 seconds for full insertion) but had been designed to provide a negative reactivity insertion of at least β(the delayed neutron reactivity) per second. However the shutdown system had no diverse back-up to guard against primary system failure and the rod system itself was capable of being rendered ineffective if the normal operational control rods were withdrawn too far out of the core. In addition it is not clear that the capability of the shutdown system had been evaluated against reasonable bounding reactivity transients that could occur on the RBMK design.

The Russian designers understood that the reactor characteristics were unstable below 20% power and that in such circumstances the shutdown system would be inadequate. However there was no engineered interlock to prevent operation in such a condition or trip parameter to ensure adequate plant protection. The design of the RBMK core was such that it was large and loosely coupled neutronically to the extent that local reactivity effects could occur semi-independently of whole core response. In consequence a complex instrumentation system would normally be required to detect the core state and provide reactor trip protection. In practice the RBMK's flux protection was sparse in core coverage and almost certainly incapable of providing adequate reactor trip information.

The overall situation was such as to probably be acceptable if everything on the plant operated as intended by the designers. However the cardinal rule of reactor design is to recognise that things will go wrong! Thus engineered safety must be based on an analysis of all potential reactor conditions that cannot be disregarded on reasonable probabilistic grounds.

A further example of the apparent failure of the RBMK design to recognise the likely requirement on its engineered safety concerns the consequences of pressure tube rupture. Nominally the RBMK was designed such that a single pressure tube failure, blowing down into the reactor vault, would be accceptable and within the relief capacity of the vault. However it seems far from clear that even a single pressure tube failure would be acceptable because of the narrow relief passages between the pressure tube and the moderator graphite. In the NNC review of the RBMK in 1975 concern was expressed that the moderator graphite might be displaced in a tube rupture event. Given these concerns the potential repercussions on the integrity of the core shutdown rods (located in the graphite) and the integrity of the vault upper lid (to which there were many pressure tube and control rod attachments) should have been considered.

6. Notwithstanding the inherent and engineered safety of modern reactor plant there remains some safety role of the man-machine interaction. However this is an area which, while significant to safety, is poorly represented in meaningful guidelines in current documentation. Such guidelines as exist are rather unspecific and expressions of hope rather than based on a sound analysis. Thus

NII 38 - "Unauthorised access to and interference with safety related structures, systems and components should be prevented by suitable measures".

NII 124 - "The protection system should be automatically initiated. No operator action should be necessary in a timescale of approximately 30 minutes. The design should however be such that an operator can initiate protection system functions and can perform necessary actions to deal with circumstances which might prejudice the maintenance of the plant in a safe state but cannot negate correct protection system action at any time".

NII 132 - "....Sufficient information should be made available to the operator at all times to enable an accurate appreciation to be made of the plant state so that all actions necessary in the interests of safety can be taken promptly and effectively....."

Within the UK there is some recognition of the potential significance of safety related actions by plant operations staff. However there remains a tendency to believe that because the use of automatic safety systems removes the <u>necessity</u> for human actions that therefore no actions will be taken - the operator is a passive element in plant control. However operational staff clearly see themselves as active elements in plant safety and unless this is recognised there exists the potential for a mismatch between man and machine. Even with automatic systems there remain some operational safety issues that are essentially human actions, eg

i) ensuring operation within the plant's normal design envelope

ii) ensuring the availability of safety systems during power operation

iii) control of test and maintenance schedules for protectic systems

iv) control of bypass/vetoes on reactor protection

7. A review of the RBMK reactor would identify a number of areas in which the man-machine interaction had been inadequately recognised.

i) The reliance on the operator to maintain the effectiveness of the shutdown system

ii) The absence of hardware interlocks to back-up the operating limit at less than 20% power

iii) The ability to simply override the reactor protection system

iv) The inadequacy of plant instrumentation to inform the operator of the plant state

v) The very high workload on the operator for normal plant control.

In a number of areas current UK practice would undoubtedly avoid many of these features of the RBMK design. However there is also a significant need to improve man-machine interaction issues particularly in areas such as

i) improved ergonomic design of controls and instrumentation

ii) improvements in the structure and analysis of operating procedures for abnormal plant states

iii) a systematic safety analysis of potential operator actions, the plant feedback and appropriate recovery procedures.

8. It is possible to highlight a number of guidelines and issues that encapsulate some of the principles of reactor design and operation.

   i) NII 43 - ...All fault sequences and combinations of fault sequences which might cause a radiation hazard should be identified ... bounding faults analysed, and appropriate monitoring and protective systems provided where necessary.

   ii) NII 121 - It must be recognised that unforeseen plant or protection system faults or maloperation may occur. Protection system design should reflect this aspect by ... provision of reasonably practicable diversity and redundancy...

   iii) NII 124 - The protection system should be automatically initiated. No operator action should be necessary in a timescale of approx 30 mins. The design should however be such that an operator can initiate protection system functions ... but cannot negate correct protection system action at any time.

However these principles are extremely difficult to observe in practice. This is particularly true when a further, perhaps overriding requirement, is that plant design should be as <u>simple</u> as possible. The increasing requirements for plant safety currently appears to involve increasingly complex plant systems with the parallel potential for misunderstanding and unexpected performance.

UNITED KINGDOM ATOMIC ENERGY AUTHORITY

HEALTH AND SAFETY STUDIES COMMITTEE

HSSC(87)P17

THE CHERNOBYL ACCIDENT - DIFFERENT VIEWS OF THE DOMINANT CAUSE

A N Hall

SRD Culcheth

SUMMARY

Quantitative analyses of the Chernobyl accident in the USA and Canada have raised some questions on the detailed development of the accident. It has been found that in order to replicate the Soviet analytical results, it is necessary to assume that a positive reactivity insertion occurred at about the time that the emergency shutdown button was pushed. A U.S. Department of Energy team has suggested that this might have been caused by the initial movement of the graphite followers that were suspended from the control rods in the Chernobyl reactor. Canadian physicists have suggested that this might have been the dominant cause of the accident, contrary to the Soviet view as expressed in Vienna and that formed in the UK. In this paper, these differing views are critically reviewed.

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Distribution:

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Quantitative analyses of the Chernobyl accident in the USA and Canada have raised some questions on the detailed development of the accident. It has been found that in order to replicate the Soviet analytical results, it is necessary to assume that a positive reactivity insertion occurred at about the time that the emergency shutdown button was pushed. A U.S. Department of Energy team has suggested that this might have been caused by the initial movement of the graphite followers that were suspended from the control rods in the Chernobyl reactor. Canadian physicists have suggested that this might have been the dominant cause of the accident, contrary to the Soviet view as expressed in Vienna and that formed in the UK. In this paper, these differing views are critically reviewed.

1. INTRODUCTION

At a special IAEA meeting in Vienna in August 1986, the Soviets gave detailed information about the events leading up to the Chernobyl accident and presented the results of their analysis of the final stages of the power transient that resulted in the destruction of the reactor [Ref 1]. Actions taken by the operators during preparations for a turbogenerator experiment had resulted in the reactor being placed in an unstable state at low power with many of the reactor safeguards disabled. At the start of the turbogenerator experiment, the coolant entering the core was only just below the boiling point and there was very little steam in the core, even at the fuel channel outlets. When the experiment began and main coolant pumps started to run down, the water in the core started to boil rapidly, tending to increase the reactivity of the core due to the reactor's positive void coefficient of reactivity. At first the automatic control rods were able to counteract this effect, but they soon reached the end of their range of effective operation and the reactivity and power then began to rise. As the reactor was at this time operating in an unstable regime where its power coefficient was positive, the power then began to run away. The excess reactivity rose so much that the reactor became prompt-critical, the fuel overheated and disintegrated, steam explosions occurred in individual fuel channels causing them to rupture and the resultant overpressurisation of the reactor vault caused the pile cap to be blown off.

During the turbogenerator experiment, the "Skala" centralised control system of the reactor was monitoring and recording various details of the state of the reactor. Although only those parameters considered important for the analysis of the experimental results were monitored at high frequency, the data provided was nonetheless found useful by the Soviets in validating a mathematical model of the accident that they constructed. Having satisfied themselves of the validity of their model, the Soviets published its predictions for the state of the reactor from about four minutes before the experiment began to the time at which the reactor was destroyed in Figure 4 of their report [Ref 1]. Western experts who have reviewed the Soviet report have confirmed [Refs 2 & 3] the qualitative validity of the Soviet description of events.

However, sufficiently detailed information about the Chernobyl reactor and the circumstances of the accident is now available for independent quantitative analyses of the course of the accident to be attempted. A U.S. Department of Energy team has assembled a suite of computer codes that it has used to analyse the entire accident sequence and assess the quantitative validity of the Soviet description of events. Comparison of the code predictions with the Soviet Figure 4 has led to the identification of a number of possible phenomena that the Soviets neglected in their description.

Furthermore, Atomic Energy of Canada Ltd (AECL) experts have also analysed the Chernobyl power transient. They have tentatively concluded that one of the possible phenomena identified by the USDOE team, a transient positive reactivity insertion upon reactor trip, was the dominant cause of the accident rather than just a minor factor affecting the timescale of the transient. These differing views are reviewed in this paper.

2. THE USDOE TEAM ANALYSES

The USDOE team divided the Chernobyl accident sequence into three stages, each of which was analysed using several computer codes [Ref 4]. These stages were:

(1) the initial power increase (from 1:23 to about 1:23:40);
(2) the rapid power excursion (from about 1:23:40 to about 1:23:43);
(3) the energetic events (after about 1:23:43).

Stages (1) and (2) were analysed using the MINET code [Ref 5], which was developed to simulate the thermal-hydraulic systems of a wide range of reactor types, and by using a code developed by ORNL to analyse the specific thermal-hydraulic behaviour of the Chernobyl reactor, known as CRAS-1 (Chernobyl Reactor Accident Simulator, version 1). Stage (3) was analysed using the FPIN-2 fuel pin behaviour code [Ref 6] combined with the RETRAN thermal-hydraulic code [Ref 7], plus the EPIC fuel-coolant thermal interaction code [Ref 8] and the ALICE-2 fluid-structure interaction code [Ref 9].

2a. The initial power increase

Any quantitative analysis of the Chernobyl accident must take into account the information provided by the Soviets in Figure 4 of their report, so the USDOE team made a considerable effort to understand this. As a result, they were able to draw attention to two areas where particular care is needed in the interpretation of the figure. First, the control rod motion in the figure is hypothetical, although it is clear that rods were being inserted from about 1:23:30 onwards. Second, the USDOE analyses have shown consistently that the pressure build up in a closed system suppresses a power excursion, so the Soviet curve M cannot represent the total steam leaving the system as stated in the Figure 4 key. The Soviets have stated informally that steam was indeed being released and this is supported by the pressure data recorded by the "Skala" control system and shown on Figure 4 (points C). Furthermore, the Soviet curve O appears to represent the total steam flow out of the turbine bypass valve plus that to turbogenerator number 8 rather than the steam mass quality as stated in the Figure 4 key.

2a.I MINET analysis

The MINET code simulates the thermal-hydraulic systems of a variety of reactor types. It contains a simple reactivity model, based on a single channel and point kinetics, which includes reactivity feedback due to fuel temperature changes. It requires only a few boundary conditions to perform a transient analysis: pump coastdown rate; feedwater flow rate; steam pressure. These were obtained from data on Figure 4.

The results of the MINET code analysis were found to be in good agreement with the recorded data from the "Skala" control system, often better in fact than the Soviet analytical results. The MINET calculated power excursion is reproduced in figure 1 and clearly establishes the type of transient displayed in the Soviet Figure 4, ie a power excursion driven by the large positive void coefficient. It should be noted however, that a small transient positive reactivity insertion of +0.00048 K at the start of trip had to be assumed to produce this result. The code also predicted that a considerable quantity of steam was released from the reactor during the last minute of the transient. The calculations failed just before 1:23:43 due to the rapid ejection of water from the core.

2a.II CRAS analysis

The Chernobyl power transient was also analysed using the CRAS-1 code, developed at ORNL, which was able to perform calculations beyond the stage at which the MINET calculations failed. The results were generally similar to those from the MINET calculations, as demonstrated by the predicted power transient reproduced in figure 2. It is seen, however, that once again a large (about 1$) transient (lasting about 3s) positive reactivity insertion at the start of trip is required to match the Soviet analytical results. Nonetheless, a power surge would still have occurred even in the absence of the reactivity insertion, albeit about 10s later.

2b. The rapid power excursion

Although the power surge that led to the destruction of the reactor could have occurred even had there been no positive reactivity insertion, it is clear that such a mechanism would have accelerated an already growing excursion. Several possible mechanisms were identified by the USDOE team:

- a failure of the pressure tube transition joints;
- opening of the pressure relief valves or rupture disks in the pressure circuit;
- anything else that released steam from the pressure circuit;
- transient positive reactivity insertion on trip.

The USDOE team favour the trip hypothesis and this is discussed later.

2c. The energetic events

The energetic events associated with the fuel failures and fuel-coolant interactions (FCIs) were analysed using a suite of computer codes (FPIN-2/EPIC/ALICE-2) that were developed at ANL to analyse transient overpower (TOP) accidents in liquid metal-cooled fast breeder reactors. The codes were modified to represent a single typical channel of the Chernobyl reactor, but as there was insufficient time to properly modify the FPIN-2 thermal-hydraulic models, RETRAN calculations were performed in combination with the FPIN-2 calculations.

2c.I Fuel failure analysis - FPIN-2/RETRAN calculations

The FPIN-2 code was used to model a single axial segment of a typical fuel pin. The power transient required as input for these calculations was determined from the Soviet Figure 4. For preliminary calculations, heat transfer from the pin to its surroundings was modelled by two limiting cases: (1) where the cladding outer surface remained at the steam saturation temperature of 573K; (2) where the pin was heated adiabatically. Comparing the results from these two cases with subsequent RETRAN calculations, it was found that at the predicted failure location near the bottom of the core, the

RETRAN fuel and cladding temperature histories were nearly identical to the FPIN results for case (2). The FPIN predictions of areal melt fraction, cavity pressure and cladding plastic strain for case (2) are reproduced in figure 3. This shows the rapid increase in cavity pressure that occurs once the fuel begins to melt, which is due to the reduction of available free volume inside the fuel pin by the 10% increase of fuel volume upon melting. Experiments have shown that this fuel expansion is accommodated by rapid localised deformation of the hottest cladding, which quickly leads to cladding failure. This is indicated in figure 3 by the peak in the cavity pressure. The areal fuel melt fraction at cladding failure is 57%, the liquid fraction of molten fuel is 10%. The average void fraction in the fuel channel at about this time is predicted by RETRAN to be significantly higher (72%) than the Soviet analytical result (40%), but in either case fuel failure would lead to forced intermixing of fuel materials and water, providing the potential for fuel-coolant interactions.

2c.II Fuel-coolant interactions - EPIC calculations

The EPIC computer code was used to calculate the fuel-coolant interactions and the fuel and coolant motion using the fuel conditions predicted by FPIN-2 at the cladding failure time. Its modelling is generally based on the physical picture that there is a molten fuel cavity in the pin: the cladding breach allows ejection of molten fuel, unmelted fuel and fission gas into the coolant channel; liquid coolant is present flowing through the channel at the breach elevation; and that the fuel pin structure remains nominally intact during the time scale of the FCI. To a large extent, it is a parametric code and initial cladding rip length (0.08 & 0.70 m) and fuel particle size (10, 100, 300 & 600 um radius) were specified as input. The molten fuel cavity was assumed to extend the whole length (3.5 m) of the fuel pin. Expansion of the interaction zone was only allowed in the upward direction, due to check valves in the inlet piping. The expansion was assumed to accelerate a simple incompressible liquid slug towards the top of the fuel channel.

The results of the EPIC runs are reproduced in figures 4 and 5. Fuel was ejected into the channel over a timescale of several tens of milliseconds. The best estimates of channel pressurisation and slug ejection velocity were 140-150 atmospheres and 40-50 m/s respectively.

2c.III Fluid-structure interactions - ALICE-2 calculations

In the ALICE-2 calculations, the fuel channel pressure tube was modelled as a thin tube and the top closure head was modelled simply as a closed end to the tube. The tube was allowed to deform radially under internal pressurisation without constraint imposed by the graphite heat transfer rings or moderator blocks. The top end cap was allowed to move axially under impact loading. The outlet from the fuel channel and internal structures were not modelled, so the strains in the tube wall near the top head were probably overpredicted by the code.

Three calculations were performed using the results from EPIC code runs for fuel particle radii of 100, 300 and 600 um to provide the loading function. In all three cases, the code predicted that channel failure would occur close to the top closure head, due to hammer pressure from the impact of the liquid slug. Failure of the fuel channel due to FCI pressurisation alone was not predicted, however, although local failures in the core region due to fuel impingement effects were considered likely.

The transient heating of the transition welds between the zirconium alloy pressure tubes and stainless steel outlet pipework as the power surge developed was also analysed using the INCIRT thermal-hydraulics computer code coupled to the previous RETRAN results. It was found that the average weld temperature was predicted to rise by 55 K over the last second of the power surge and was rising at a rate of 95 K/s at the end of this period. This is four orders of magnitude greater than the maximum rate of temperature change allowed in normal operation and might have led to thermal stresses that could have caused failure within the transition region.

2d. Best estimate of accident events

The calculations outlined above were all based on average channel conditions and uniform axial power. In reality, axial power peaking and variation in the power to flow ratio between different channels would cause fuel failures to occur at slightly different times. The USDOE team investigated the effects of noncoherent failures with some scoping calculations performed with the EPIC code. The core was divided into ten groups containing equal numbers of fuel channels for this purpose. The reactivity effects of fuel failure related coolant voiding and the Doppler effect were calculated explicitly and used to determine the changing reactor power.

It was found that rather than a double-peaked power excursion occurring as determined by the Soviets, the noncoherency of the fuel failures caused the first power surge to escalate into a single large event. This was less severe in terms of peak power, energy deposition, fuel temperature and fuel vapour pressure than the Soviet's double peaked excursion, but the fuel vapour pressure was nonetheless predicted to augment significantly the destructive event.

3. THE AECL ANALYSES

Physicists at Atomic Energy of Canada Ltd (AECL) have also been analysing the Chernobyl accident. They have reached the conclusion that the power surge was initiated by the movement of the control rods and their graphite followers when the trip button was pressed. Their conclusion is based on an analysis of the core neutronics prior to the power surge that indicates the graphite followers displaced water from a region of high neutron flux near the bottom of the core (see figure 6) as the control rods were inserted, resulting in a positive reactivity insertion. Without this effect, the rate of increase of system reactivity would have been very low. In their view, the positive void coefficient of the Chernobyl reactor was not a significant factor in the accident. Their analysis is based on assumptions about the fuel burn-up and positions of the control rods that have not yet been validated, however, and discussions to clarify the basis of these assumptions are planned.

4. DISCUSSION

4a. The role of the graphite followers

The graphite followers are 5 m long and suspended 1.2 m below the control rods. They move within dedicated control rod channels and are cooled by water in a circuit separate from the pressure circuit. When the control rods are fully withdrawn, the followers are at the axial centre of the core as shown in figure 6. As the control rods begin to move into the core, the lower ends of the followers displace water from the lower ends of the control rod channels and a local transient insertion of positive reactivity occurs, as the graphite is a poorer neutron absorber than the water. If the neutron flux profile were axially symmetric, the movement of the top end of the follower would initially cause a local transient insertion of negative reactivity at the top of the core of the same magnitude as the positive insertion at the bottom of the core and the overall reactivity of the core would not initially change. If the neutron flux also increased monotonically towards the mid-plane of the core, the overall reactivity of the core would then fall as the followers moved away from the mid-plane, as the lower end of the followers would be moving into a region of lower worth and the upper ends would be moving into a region of higher worth. Of course, the overall reactivity would also start falling due to the absorber rods entering the core.

In reality, the axial neutron flux profile was double-humped and greater in the upper half of the core than in the lower half [Ref 1]. This suggests that the overall effect of the graphite follower movement would have been to reduce the core reactivity, as the upper ends of the followers would have been in regions of higher worth than the lower ends. The RBMK core is neutronically large, however, and the neutronic behaviour in the lower part of the core could be effectively decoupled from the behaviour in the upper part. The USDOE view is that the movement of the graphite followers caused a local positive reactivity excursion in the lower part of the core that accelerated an already developing power excursion. The positive void coefficient/power coefficient is still regarded as the fundamental cause of the accident, however, as the excursion would have continued to develop even in the absence of the graphite follower effect.

The AECL physicists believe, however, that the positive reactivity insertion associated with the graphite follower movement was the dominant cause of the accident. In the absence of the follower effect, the excursion would have been sufficiently slow for the shutdown system to terminate it before significant damage occurred. Nonetheless, for this view to be self-consistent, the speed of shutdown would have to be assessed in terms of the capability to suppress a local excursion in the lower part of the core rather than in terms of the overall rate of reduction of reactivity and the AECL physicists have not yet reported such an assessment. One could equally well take the view that were it not for the positive void coefficient reducing the reactivity of the core when its power fell below that planned for the experiment and the coolant void fraction dropped, the control rods would not have been withdrawn so far in the first place and the follower effect would not have come into play.

Furthermore, the USDOE and AECL calculations are very sensitive to the assumptions made concerning the distribution of fuel burnup in the core and the positions of the control rods. The positions of the shortened bottom entry rods are particularly relevant to a local reactivity excursion near the bottom of the core, as their insertion would reduce the neutron flux in the very region where follower movement could cause a local reactivity insertion and might suppress an excursion in this region. It should also be remembered that to predict an excursion at all, the USDOE team had to allow steam to be released from the pressure circuit in their model. The AECL modelling in this respect is not clear.

It should be noted that a positive reactivity insertion in the lower part of the core might alternatively have occurred as a result of cavitation in the main circulating pumps [Ref 10]. This would have reduced pump efficiency and might have injected vapour bubbles into the bottom of the core, raising the reactivity through the positive void coefficient.

4b. Fuel-coolant interactions

The USDOE analysis of fuel-coolant interactions (FCIs) was based on the assumption that particulate fuel debris would be ejected into liquid coolant in the fuel channels, causing it to vaporise rapidly. The codes run to assess the consequences of this predicted that the fuel channels would fail near the top closure heads, but not within the core where failure would lead to overpressurisation of the reactor vault and displacement of the pile cap. This type of "steam explosion" could not, therefore, have been responsible for the destruction of the Chernobyl reactor. The codes were conservative in that internal structures that would have mitigated the load on the closure heads were ignored. Evidence that the closure heads did not fail would not therefore invalidate the analysis.

The transition welds between the zirconium alloy fuel channels and the stainless steel pressure tubes might have been weakened by the thermal transient during the power surge and might have failed following the loading of the closure heads. It is likely, however, that the limit on the rate of temperature change was related to thermal cycling considerations over the lifetime of the reactor and would not be relevant to a single transient.

The fuel-coolant interaction analysed in the USDOE report is discussed qualitatively in the UK report [Ref 3] and the conclusions reached are not dissimilar. The UK report, however, suggests that conventional (propagating) molten fuel-coolant interactions could have caused individual fuel channels to rupture, whereas the USDOE report does not discuss these. Nonetheless, the USDOE analyses show that about 60% of the fuel would have started melting at the time of pin failure and between 10% and 50% of this would have liquified. With the favoured value of 10% fuel liquified in the melting regions, the mass of molten fuel potentially available for fuel-coolant interactions would have been about 4 Kg. In the SUW series of FCI experiments at AEE Winfrith [Ref 11] in which 24 Kg masses of molten fuel simulant were released into pools of water, fuel-coolant interactions involving on average about 3 Kg of melt were routinely observed to generate mechanical yields in the range 100-1000 KJ. The mechanical energy required to rupture a fuel channel would have been of order 1 KJ, so a conventional FCI involving a small fraction of the molten fuel should have been capable of rupturing a fuel channel. To put these mechanical yields into perspective, the mechanical yield required to cause rupture of a PWR pressure vessel has been estimated [Ref 12] to be about 1000000 KJ.

5. CONCLUSIONS

The USDOE and AECL physicists have performed quantitative analyses of the Chernobyl accident that have revealed some discrepancies between the Soviet description and analysis of the accident. In particular, the power surge could not have occurred at all unless steam were released from the pressure circuit during the transient, contrary to the information provided by the curves of the Soviet Figure 4. Furthermore, the timing of the power surge as shown in Figure 4 can only be accounted for if a transient positive reactivity insertion occurred at about the time that the trip button was pushed. This might have been due to displacement of water near the bottom of the control rod channels by the graphite followers that are suspended from the control rods, or it might have been due to cavitation at the main circulating pumps causing the injection of vapour bubbles into the core. A further possibility that cannot be excluded is that the discrepancy in the timing of the power surge might be indicative of deficiencies in the Soviet modelling of the accident.

Nonetheless, the USDOE and AECL analyses generally confirm the description of the accident given by the Soviets at Vienna and are in general agreement with the UK view of the accident development. Those minor differences that have arisen are related to assumptions made in areas where inadequate detailed information is available and reflect legitimate differences of view between the experts in the field.

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22 May 1987

MINET Analysis of

Chernobyl Power Excursion

of 4/26/86

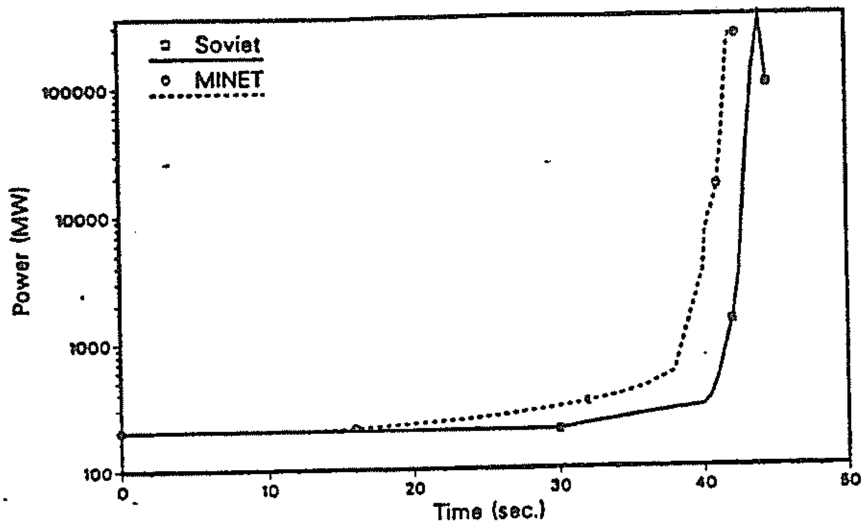


Figure 1. MINET calculated power excursion (from Ref 4, fig 11)

CRAS-1 ANALYSIS OF THE CHERNOBYL POWER EXCURSION
OF APRIL 26, 1986

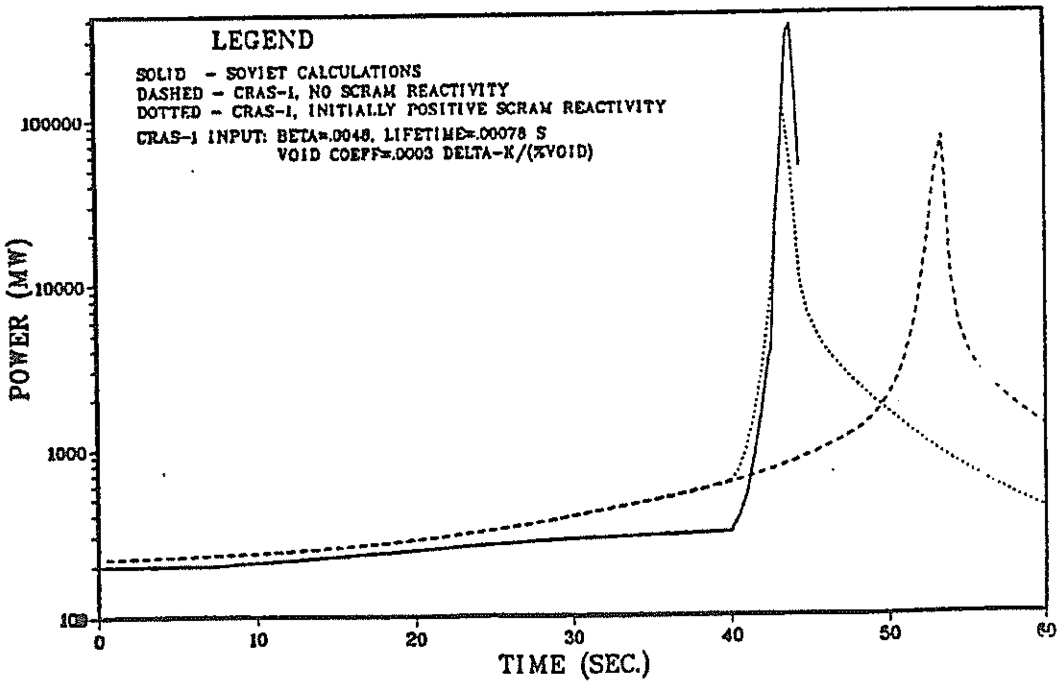


Figure 2. CRAS-1 calculated power excursion (from Ref 4, fig 12a)

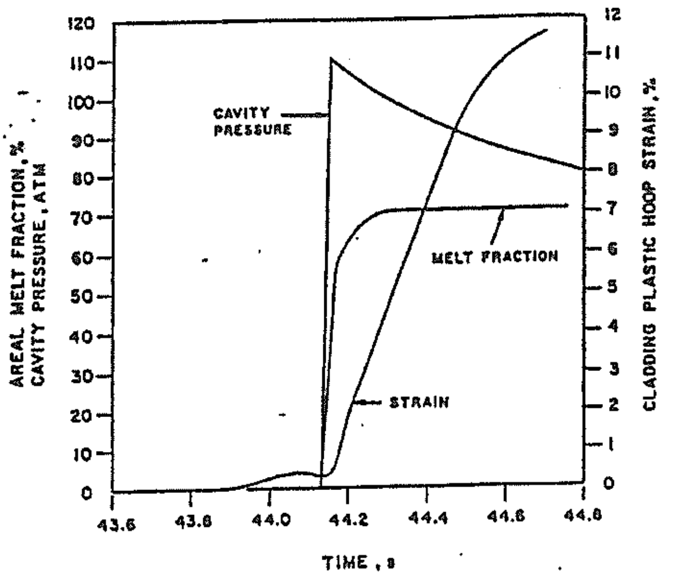


Figure 3. FPIN2 predictions for fuel pin response with adiabatic heating (from Ref 4, fig 24)

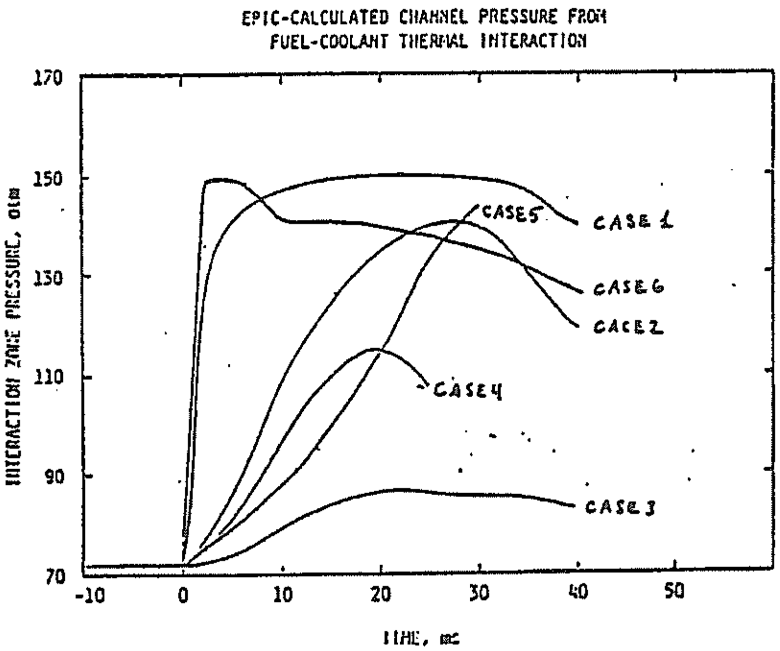


Figure 4. EPIC calculated channel pressure from FCI (from Ref 4, fig 28)

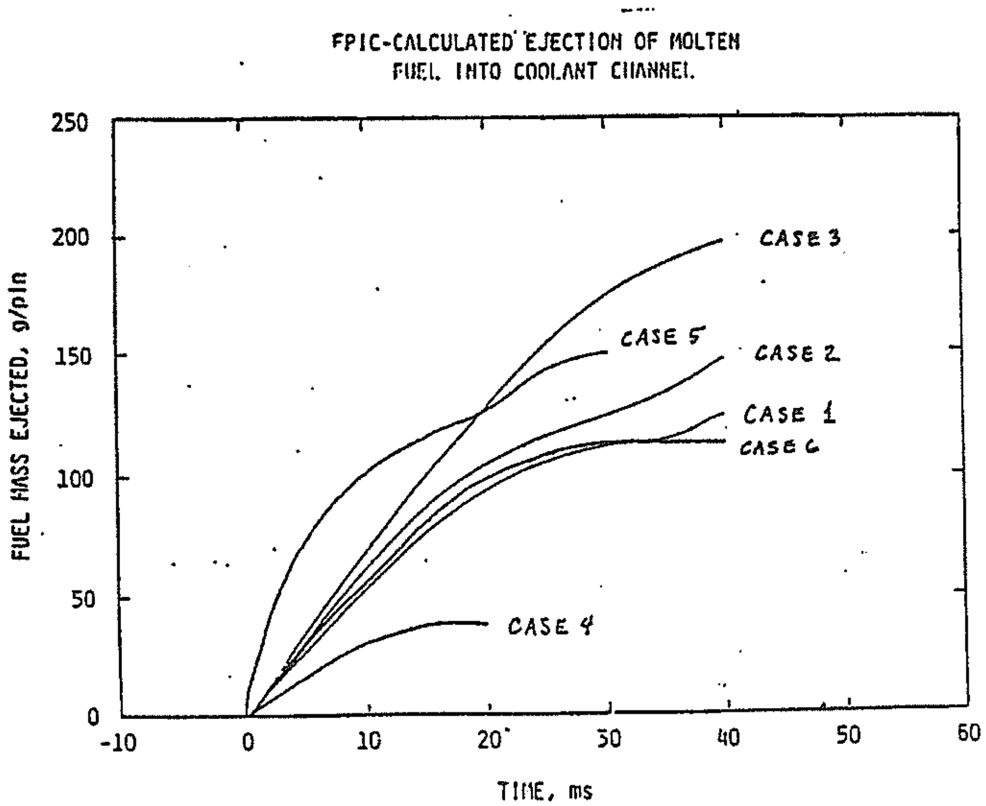


Figure 5. EPIC calculated ejection of molten fuel into fuel channel (from Ref 4, fig 29)

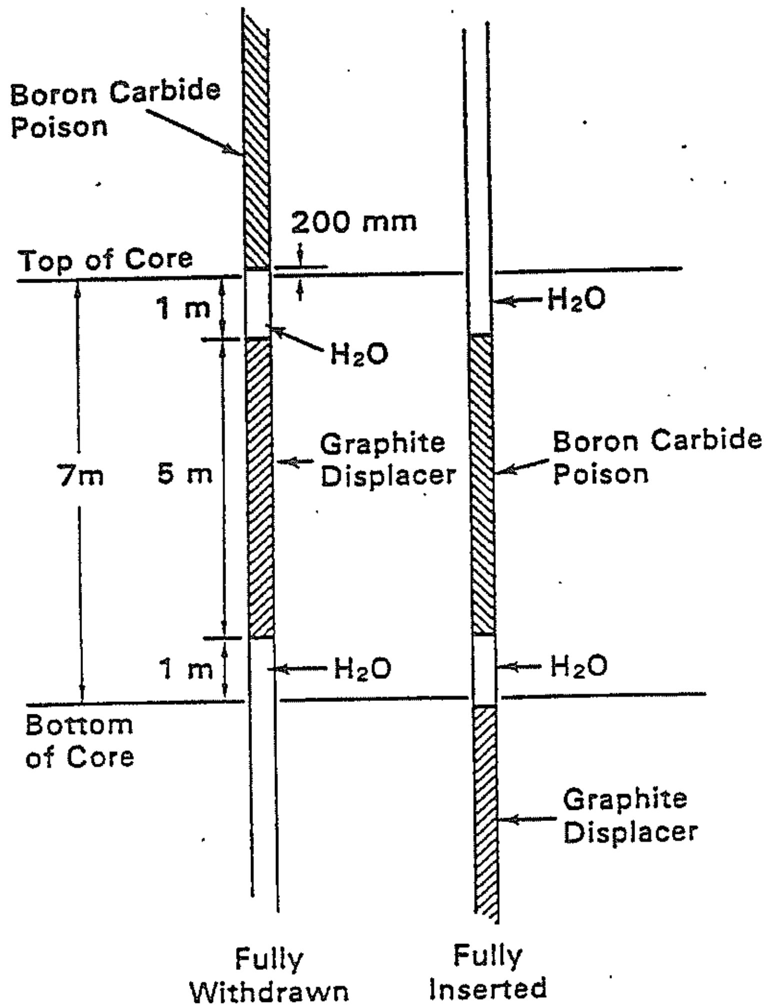


Figure 6. Locations of control rods and graphite followers in the core when fully withdrawn and fully inserted (from Ref 4, fig 13)

UNITED KINGDOM ATOMIC ENERGY AUTHORITY

HEALTH AND SAFETY STUDIES COMMITTEE

SYMPOSIUM ON

CHERNOBYL 'ONE YEAR AFTER'

Risley Lecture Theatre

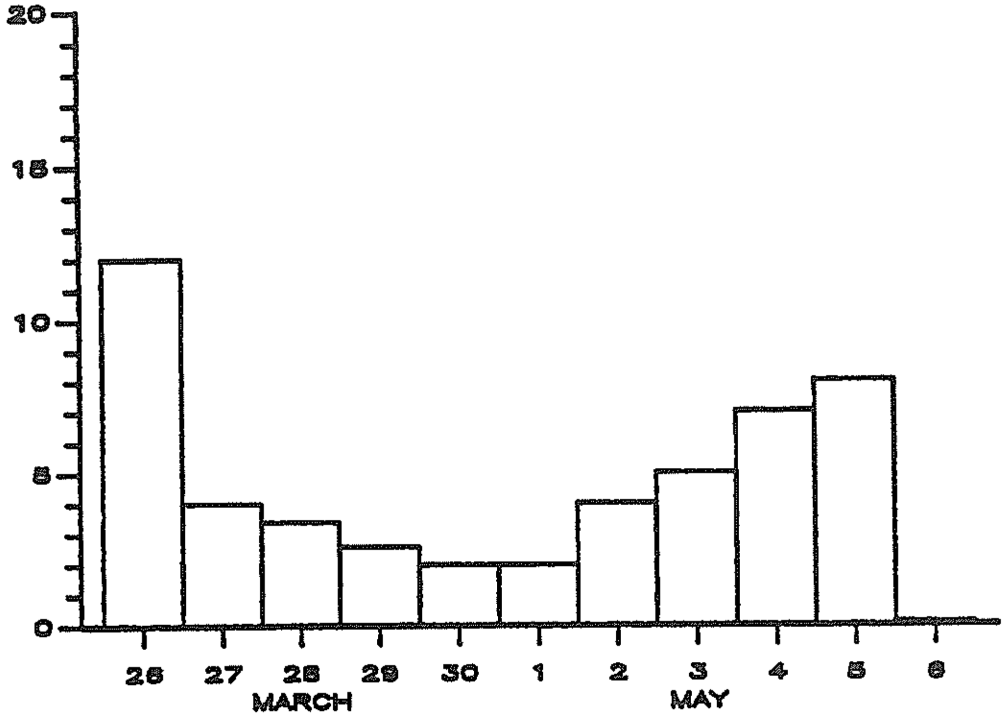
27 May 1987

Viewgraphs used by Dr P N Clough

FISSION PRODUCT RELEASE - SOURCE TERMS

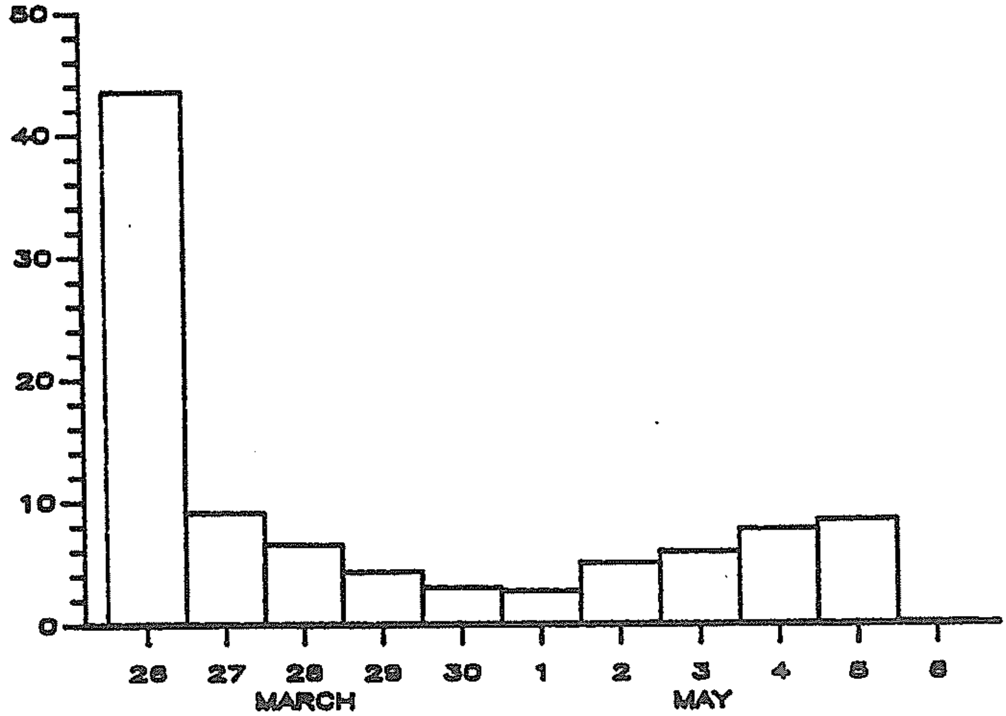
SOVIET DATA

RELEASES CORRECTED TO 6th MAY (/ MCi)



S.R.D. INTERPOLATION OF SOVIET DATA

TO GIVE ACTUAL DAILY RELEASES (/ MCi)



SOVIET ASSESSMENT OF THE ACTIVITY RELEASE FROM CHERNOBYL UNIT 4

	26 APRIL		6 MAY	
	MCi	%	MCi	%
Xe133	5	2.8	45	100
Kr85m	0.15	1.7	-	100
Kr85	-	-	0.9	100
I131	4.5	5.3	7.3	20
Cs134	0.15	3.0	0.5	10
Cs137	0.3	3.9	1.0	13
Te132	4	5.4	1.3	15
Sr89	0.25	0.37	2.2	4.0
Sr90	0.015	0.27	0.22	4.0
Ba140	0.5	0.38	4.3	5.6
Mo99	0.45	0.28	3.0	3.2
Ru103	0.6	0.45	3.2	2.9
Ru106	0.2	0.35	1.6	2.9
Zr95	0.45	0.34	3.8	3.2
Ce141	0.4	0.27	2.8	2.3
Ce144	0.45	0.51	2.4	2.8
Np239	2.7	0.38	1.2	3.2
Pu238	1(-4)	0.48	8(-4)	3.0
Pu239	1(-4)	0.50	7(-4)	3.0
Pu240	2(-4)	0.50	1(-3)	3.0
Pu241	0.02	0.42	0.14	3.0
Pu242	3(-7)	0.43	2(-6)	3.0
Cm242	3(-3)	0.43	2.1(-2)	3.0

- Magnitude of activity release
- Composition of released activity
- Mechanisms of release

- No new Soviet data

- Source Term reviews
  - USA (DOE,USNRC)
  - Canada (AECL)
  - France (CEA)
  - Japan (JAERI)
  - OECD/CSNI

Magnitude of activity release - did Soviets underestimate ?

Soviet estimates - deposited activity - 300 point grid

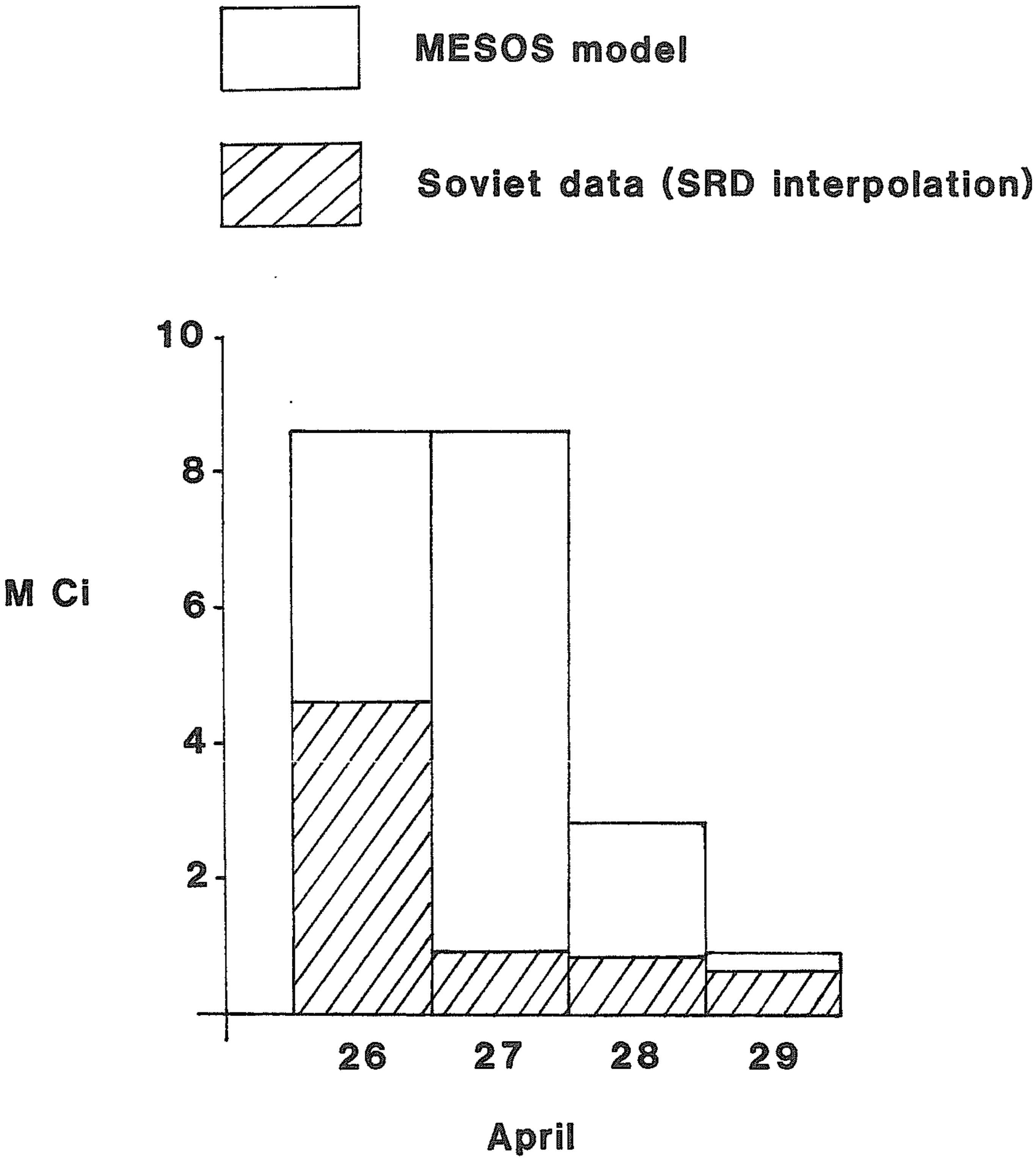
Within 30km zone	- 20 M Ci (6 May '86)
Remainder of USSR	- 10 - 30 M Ci (")
Total	30 - 50 M Ci (")

Neglected activity transported across Soviet border

Additional evidence - I131 and Cs 134/137

- MESOS trajectory modelling (H. Ap Simon, Imperial College)
- OECD/NEA Survey of radiological measurements (NRPB)

Estimates of I131 release - early stages



Totals MESOS 21 M Ci

Soviet 6.7 M Ci

Distribution of deposited activity (M Ci)

OECD/NEA survey

	USSR	W. EUROPE
I131	7.3	2.7
Cs 134/137	1.5	0.55

Adding contributions for E. Europe and sea areas ⟶ about 50% of released I, Cs crossed Soviet border

Chemical form of Iodine

Most air sampling - particulate only

Few sites - particulate plus gaseous

	Particulate %	Gaseous or desorbable %
Nurmijaari (Finland)	16	84
Budapest (Hungary)	30	70
Neuherberg (W. Germany)	18	82
Berkeley (U.K.)	20	80

Elevated radioactive plume

Conclusions on magnitude of release

- Volatile fission products I and Cs (probably also Te, Ru) formed high proportion of fine aerosols transported outside USSR

- Soviet measurments within USSR have underestimated release fractions

- True release fractions were

Iodine	30 - 60%
Caesium	20 - 40%

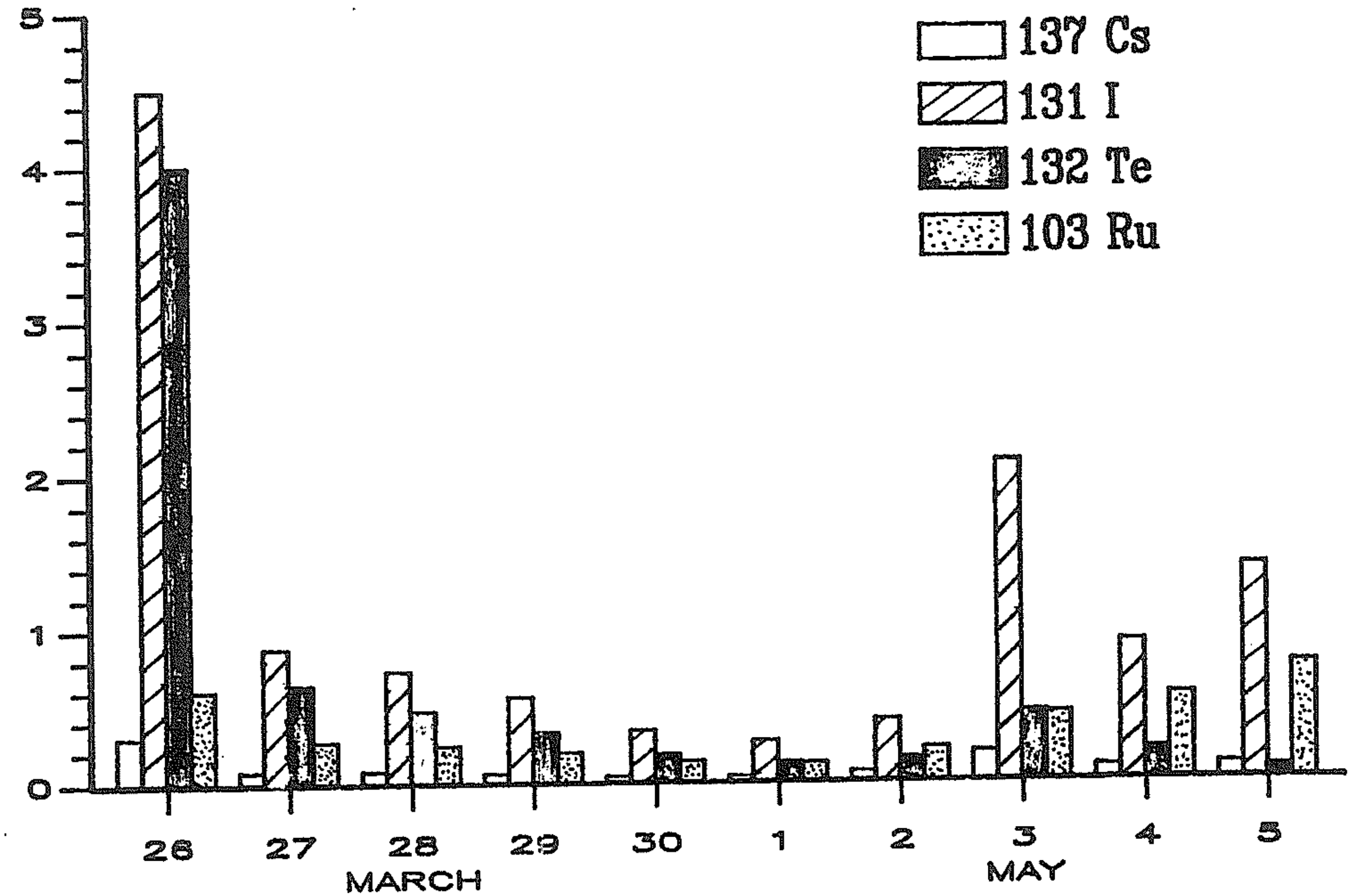
- Depletion of inventory may have been significant

CHERNOBYL CORE INVENTORY CALCULATIONS (MCi, INITIAL)

	SOVIET	FISPIN (UK)	ORIGEN 2 (FINLAND)
I131	85	79.7	79
Te132	74	110	119
Cs134	5.0	2.97	4.4
Cs137	7.7	6.45	6.0
Sr89	68	98.7	108
Sr90	5.5	5.4	5.0
Ba140	132	156	151
Mo99	(1634)	149	155
Ru103	132	115	109
Ru106	56	24.0	21.4
Ce141	150	151	146
Ce144	88	105	93
Zr95	131	158	150
Np239	711	1278	1367

AVERAGE B.U. 10,300 MWd/te

S.R.D. INTERPOLATION OF ISOTOPE SPECIFIC DAILY RELEASES (/ MCi)



Release mechanisms

Stages 2 and 3

Problem - Core very heterogeneous

Temperature range 800 - 2000K

Oxygen potential range -

Very high (air)

to Very low (graphite)

Why so little discrimination in release fractions of non-volatile fission products Sr, Ba, Ru, Mo, La, Ce, and actinides?

Oxidative release

- $3UO_2 + O_2 \longrightarrow U_3O_8$
- 800 - 1000K - promotes I, Te, Ru, vapour release
  - promotes U_3O_8 aerosol spalling - indiscriminate release
- > 1800K U_3O_8 vaporisation - indiscriminate release
- Almost pure Ru particles in Sweden

  $RuO_2 \xrightarrow{air, O_2} RuO_3, RuO_4 \xrightarrow{condense} RuO_2$

  fuel T > 1200K vapour T > 1000K aerosol
- Termination by N_2 injection

Reductive release

- $UO_2 + 4C \longrightarrow UC_2 + 2CO$
  Carburisation

- Requires $T > 2000K$

  O_2 partial pressure
  $< 10^{-3}$ atm

  Fuel-graphite contact

- Promotes release of Ba, Ce

- Suppresses release of Ru, Mo

Significance of Chernobyl source terms

- Very limited relevance to source terms for western LWRs

- Source term management

- Filtered venting of containment

D4.WN8

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UNITED KINGDOM ATOMIC ENERGY AUTHORITY

HEALTH AND SAFETY STUDIES COMMITTEE

SYMPOSIUM

CHERNOBYL: ONE YEAR AFTER

ACCIDENT CONSEQUENCES

W. Nixon and M. J. Egan

Safety and Reliability Directorate, Culcheth.

May, 1987.

Distribution:
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CHERNOBYL: ONE YEAR AFTER

ACCIDENT CONSEQUENCES

W. Nixon and M. J. Egan

This paper is comprised of:

(i) SECTION 7: ENVIRONMENTAL CONSEQUENCES, from "THE CHERNOBYL ACCIDENT AND ITS CONSEQUENCES", NOR 4200, March, 1987.

(ii) Copies of overhead slides to be used during presentation.

SECTION 7: ENVIRONMENTAL CONSEQUENCES

Following the accident at Chernobyl, radioactive material was swept across Europe resulting in increased dose levels. This section considers the time dependent pattern of the spread of contamination throughout Europe and presents preliminary estimates of the collective dose to various countries. Most of the work described here has been published elsewhere[1].

7.1 Atmospheric Dispersion Across Europe of Material Released from Chernobyl

Increased activity levels were first reported on 28 April from environmental monitoring stations in Finland and Sweden, where external dose rates in certain locations exceeded normal background levels by a factor of ten or more. On succeeding days elevated radioactivity concentrations were detected throughout Europe until almost complete coverage had been achieved by 3rd May. Based upon reported measurements conveyed through international bodies (IAEA, WHO and NEA), complemented by computer calculations, it has been possible to assemble a picture of the pattern of dispersion of the material released from the core of the damaged reactor, as it affected western Europe. The progression of this pattern with time is illustrated in Figures 1 to 6. The figures indicate how the external exposure rate varied across Europe from 28 April to 3 May. Note that the monitoring data used as a basis for these plots exhibits a marked patchiness, deriving from (patchy) rainfall patterns and, possibly, uncertainty in the environmental measurements. Such large variations over relatively short distances are not shown in the figures, so that they provide a general picture of the spread of the contamination across Europe. Note also that the figures may be subject to some slight revision as more monitoring information becomes available.

Referring to Figures 1 to 6, it can be seen that by the 28 April (Figure 1), radiation dose levels had increased in Scandinavia (as has already been noted above), resulting from the generally north-westerly trajectories prevailing at that time. By the 29 April (Figure 2), the contamination had spread further across the Scandinavian countries, with lateral dispersion increasing the width of the plume. On the 30 April (Figure 3), central

Europe was beginning to be affected, reflecting a trajectory which was initially north-westerly, but which subsequently veered westwards in the vicinity of the Baltic sea. (At this point it should be appreciated that Figures 1 to 6 are based upon measured dose rates, so that although by the 30 April the plume of material was largely over central Europe, dose levels were still relatively high in Scandinavia, reflecting earlier deposition of material from the plume). From the 1 to 3 of May (Figures 4 to 6), the plume spread to the west, north and south, essentially covering Europe. This further spreading of the plume was influenced by an anticyclone which was moving eastwards across central Europe. Indeed the contamination of the UK resulted from air being convected northwards behind the area of high pressure; the higher contamination in the north of the UK reflects greater rainfall rates during plume passage.

As far as Europe is concerned, from the 3 May the dose levels generally stabilised and fell. One of the more notable exceptions to this is Scandinavia, where increased air concentrations were observed around 8 May. This almost certainly represents material discharged during the latter part of the Chernobyl release (around 5-6 May).

Before leaving this discussion of the spread of radioactivity across Europe, it is worth noting the similarities between the contamination patterns of Figures 1 to 6 and those generated by long-range trajectory models. Many such studies have been performed and the interested reader is referred to the work of, for example, ApSimon et al[2].

In addition to the very wide dispersion brought about by the changing meteorology over the several days during which emissions from the damaged plant took place, it seems likely that, initially, material was distributed over a considerable range of elevation. Material transported at very high altitudes (> 1km) may have been responsible for the subsequent observations of elevated activity levels in countries bordering the Pacific Ocean.

7.2 Dosimetric Assessment for Western Europe

An estimate of the dosimetric impact on Western Europe from Chernobyl may be obtained by utilising the monitoring data collected and published by the various national agencies responsible for radiological protection. Such an assessment (necessarily preliminary in nature) is presented here.

At the outset it should be appreciated that, within any country of Western Europe, there is some variability in the measured concentrations of radioactivity, arising from the complicated patterns of atmospheric dispersion and rainfall during passage of the plume. Clearly any dosimetric assessment needs to take account of this distribution in relation to the distribution of population. Presented here are estimates of the mean (population-weighted) individual dose for various countries, based on estimated mean environmental concentrations. The (mean) dose estimates are, therefore, subject to a degree of uncertainty, up to around a factor of a few depending on the country, arising solely from this averaging process. Other sources of uncertainty are discussed below.

Dosimetric pathways contributing to radiation exposure include the inhalation of activity during passage of the plume, ingestion of contaminated foodstuffs and external irradiation from deposited activity. In addition to these, β-dose to the skin and external exposure to radiation from the passing cloud of activity can also contribute to total dose levels; however, these mechanisms are transient in nature and have been shown to make up only a very small fraction of the total effective dose to a representative individual[3]. Each of the pathways considered in the present analysis is discussed briefly below.

7.2.1 Inhalation pathway

In general, significant elevated concentrations of activity in air were present for a few days and direct measurements of ambient levels of the radiologically important nuclides (isotopes of Cs, I and Ru) were monitored. This data allows time integrated air concentrations to be estimated. Used in conjunction with standard values of inhalation rate and dose per unit intake for adults[4], a mean, individual, committed effective dose equivalent can be calculated for the inhalation pathway.

7.2.2 Ingestion Pathway

By contrast with inhalation exposure, radioactivity transferred to man through incorporation in foodchains is available over a more extended period, so that monitoring data is unlikely to give a full picture of the average intake of activity via foods. It is therefore necessary to turn to mathematical models representing the temporal pattern of appearance of radionuclides in different foods following an initial deposit. For milk and green vegetables, doses are estimated from representative measured peak concentrations of Iodine and Caesium activity in these foodstuffs, on the basis of those models used by NRPB in deriving reference levels for the introduction of countermeasures affecting food(4). Consumption rates typical of the UK adult population are assumed(5) and activity losses in preparation for consumption are neglected. Dosimetric calculations are integrated for a 50 year period following ingestion. In some cases, the absence of monitoring data for foodstuffs has necessitated the estimation of peak concentrations in milk and green vegetables from measured deposition levels. One advantage of using measured levels of contamination in foodstuffs with the models is that the effect of certain countermeasures is implicitly included; thus, if dairy cattle were removed from contaminated land or kept indoors for a period of time (eg in Scandinavia and the Low Countries), measured activity levels in milk will be low and reflected in assessed doses. While there is some evidence that other, mainly voluntary, countermeasures were introduced affecting normal food distribution and consumption patterns throughout Europe, these have been ignored in the dosimetric assessment. It might therefore be expected that the results of the calculations yield a slightly conservative estimate of the overall dosimetric impact.

In addition to the contribution to ingestion doses from milk and green vegetables, estimates are made of the intake of Caesium in supplies of beef and lamb. Here committed doses from ingestion are derived from reported ground contamination levels, using published data on the transfer of activity through foodstuffs(6). Again, typical UK adult consumption rates are assumed(5), with appropriate values for the committed dose equivalent per unit intake(7). Calculations are extended to include contributions to ingested activity arising in meat over the next 50 years.

Scoping estimates based upon the same suite of models as those used above(6) suggest that, by comparison, the contribution to mean individual committed doses from other foods (root vegetables, cereals etc) will be relatively small (well within bounds of uncertainty). This is due in part to the recorded absence of significant quantities of the radioisotopes of Strontium in environmental monitoring (usually considered to be important for foods contaminated by uptake from soil) and to the delay of a number of months between the accident and the cereal harvest. Ingestion dose calculations are therefore limited to contributions from milk, green vegetables and meat.

7.2.3 External exposure

External exposure to activity deposited within each country during passage of the plume will continue for several decades. During this time, the predominant contribution to external doses will be due to the decay of radioisotopes of Caesium. For the present assessment, the population-weighted average ground concentrations of these nuclides are used in conjunction with appropriate dose conversion factors taking account of decay and migration into the soil[4]. The dose calculations assume a shielding factor of 0.36 for protection by buildings and involve integration over a period of 50 years following the accident.

7.2.4 Note on calculations for UK

A small additional degree of sophistication is introduced into estimates of mean individual dose in the UK. It is well known that a fairly sharp division exists between contamination levels in the northern and north-western parts of the United Kingdom and the remainder of the country, due to different rainfall patterns at the time the plume was passing. Dosimetric calculations are therefore made separately, assuming average contamination levels characteristic of the two regions. A population weighting factor of 18% is applied to doses calculated for the 'north' and 82% to those estimated for the 'south'. The only exception to this method for averaging doses is applied in the case of lamb consumption, since a large proportion of the country's sheep farming is in the more heavily contaminated region. Indeed restrictions on sale and slaughter of lambs have been in force in certain of the worst affected regions. Mean individual dose to members of the UK population from consumption of lamb is therefore determined from the characteristic ground concentrations of Caesium reported for the 'north' only, ignoring any reduction from restrictions (these would not be predicted to be necessary on the basis of the models when average contamination levels are assumed) and taking into account that the UK is only 77% self-sufficient in mutton and lamb.

7.2.5 Results

Table 1 shows the results of a preliminary dosimetric analysis for Western Europe, in terms of the contributions to the total collective dose (integrated to 50 years) from the various pathways considered. It can be seen that the contribution from inhalation is less than 10%, while those from external irradiation and ingestion are roughly similar. The total collective dose to Western Europe is estimated to be approximately 76,000 man Sv.

Table 2 shows a preliminary estimate of the distribution of dose among West European countries; for each country the mean individual dose commitment and the total collective dose commitment is presented. In some cases, the contributions to total dose from each of the pathways differ somewhat from the general picture of Table 1; comments on such deviations and other considerations are included, where necessary, in Table 2. Finally, some authorities have provided estimates of average individual doses in their own countries, arising from the Chernobyl accident. The methods used differ in their degree of sophistication and in the time periods over which the dose is considered to be accumulated. However, where available, these national estimates have also been included in Table 2 for comparison.

As noted above, the total collective dose for Western Europe is estimated to be approximately 76,000 man Sv. Assuming a linear relationship between dose and risk of cancer (generally considered to be conservative for the very low levels of individual dose experienced here), characterised by a cancer fatality risk coefficient of 1.25×10^{-2} per man Sv, the total number of cancer fatalities in Western Europe arising from the Chernobyl accident over the next decades is estimated to be just under 1000.

7.3 Dosimetric Assessment for Eastern Europe

Information on levels of radioactive contamination in Eastern Europe is relatively limited and an assessment of the dosimetric impact for these countries is therefore subject to considerable uncertainty. Here, calculation of mean individual dose is performed by scaling from the values previously estimated for countries in Western Europe according to the ratio of activity concentrations in deposited material or foodstuffs, whichever data are available.

Table 3 summarises mean individual doses and total collective dose commitments estimated in this way for Eastern European countries excluding the Soviet Union. The total collective dose is estimated to be approximately 100,000 man Sv which, using a linear dose risk relationship in the manner described above, corresponds to a total of around 1,250 fatal cancers.

7.4 General comment on estimated doses

It must be appreciated that the dose estimates in Tables 1, 2 and 3 are subject to some uncertainty. Firstly, as already noted above, the use of a weighted contamination level for each country studied may give rise to uncertainty levels up to around a factor of a few. Secondly, the use of monitoring data (with its associated uncertainties), and the application of standard dosimetric models using UK consumption data may give rise to similar degrees of uncertainty. This should be borne in mind when considering the data in the Tables and when comparing the doses with those estimated by others. The uncertainty is clearly greater for Eastern as compared to Western Europe; indeed the former should be regarded as order of magnitude estimates.

The preliminary nature of the above dose calculations should also be noted; they have been performed using available data and may be subject to some revision as more information is obtained.

7.5 Consequences in the USSR

The following is a discussion of some of the impacts in the USSR resulting from the accident, based primarily on the presentations made by the Soviet delegation to the IAEA meeting in Vienna[8].

7.5.1 Emergency and medical response

At the reactor site, an immediate priority following the accident was attached to fighting a number of fires which had broken out in and around the reactor building and turbine hall, threatening the safety of the station's third reactor unit. Within an hour of the accident the fire team stationed at the plant, together with firemen from the nearby towns of Pripyat and Chernobyl, were beginning to bring the worst of these fires under control. It appears that there was some difficulty initially in accurately reporting the severity of the situation to personnel at the plant and to the relevant authorities in Moscow. Nevertheless, site emergency response initiated within the first two hours included arrangements for beds to be made available in local hospitals and the issue of stable iodine tablets to plant personnel. Furthermore, a specialist team had been called out and flown from Moscow within ten hours of the accident.

From among the on-site personnel, some 300 required hospital treatment. These included reactor and electrical plant operating personnel, site emergency squads and, particularly, members of the fire brigades. Of those examined, 203 were diagnosed as suffering from acute radiation syndrome arising from absorbed doses in the range 1-16 Gy (almost entirely from external radiation, in particular from debris on the site and airborne contamination). Two days after the accident, 129 casualties were flown to Moscow for specialist treatment and the remainder taken to Kiev. Despite efforts to provide bone marrow transplants for the worst affected victims (diagnosed irreversible depression of bone marrow function), only limited success was achieved. All but one of 13 patients receiving this support died. A total of 29 fatalities has been reported from among those hospitalised and diagnosed as suffering from acute radiation effects. Two additional deaths are reported to have occurred in the immediate aftermath of the accident. Based on prior understanding of human biological response to very high acute radiation exposures, a lethality substantially higher than that which was observed might have been expected, in the absence of therapeutic treatment, for the range of doses experienced by the casualties among the site personnel. The considerable medical care which was made available within a relatively short time, including blood transfusions, chemotherapy and antibiotic administration, together with techniques to prevent infection, appears to have been generally effective (within the limits imposed by the severity of injuries) in achieving an increased survival rate.

It has been emphasised that no individual member of the population away from the reactor site itself incurred a radiation dose above the threshold for clinically manifest symptoms of acute radiation syndrome. An emergency control centre, established in the town of Chernobyl with the support of State Committee for Atomic Energy and the specialist team from Moscow, directed the emergency response as it affected the population surrounding the reactor site. Precautionary evacuation of the town of Pripyat, less than 10 km from the site, was viewed as an immediate objective. However, the initial plume of radioactive material released from the damaged reactor had missed the town but contaminated evacuation routes laid down under existing emergency plans. During the morning of 26 April, the day of the accident, people were instructed to shelter indoors with windows and doors shut; schools and kindergartens were closed. Later that day stable iodine tablets were distributed by volunteers from house to house in the town. During this time, ad hoc evacuation plans were being devised so that by the next day, when radiation levels in the town had begun to rise sharply, the necessary resources (transportation, relocation centres, medical teams) had been organised. The population of 45,000 people were evacuated in 2½ hours, commencing at 2.0pm on 27 April. Individual exposure appears to have been kept below the upper dose intervention level of 750 mSv operated by the Soviet authorities as a reference for emergency planning. During the next few days, the emergency control centre supervised the gradual evacuation of a further 90,000 people from within a radius of about 30 km around the plant as the levels of contamination became more widespread. Substantial medical resources were deployed to monitor the total of 135,000 evacuees, many of whom were found to be wearing contaminated clothing requiring replacement. Nevertheless, no cases of acute radiation syndrome were diagnosed from among the evacuees and, correspondingly, none were hospitalized as a result of radiation-induced effects.

The emergency control centre was responsible for a variety of other aspects of the emergency response, including radiological monitoring both inside and outside the 30 km evacuation zone, food and water bans, control of the movement of cattle and decontamination. The effort appears to have been supported by the deployment of thousands of troops within the affected area, together with technical teams assembled from appropriate institutions throughout the Soviet Union. Among the results of this activity, tens of thousands of cattle were evacuated from the 30 km zone. The consumption of milk and other foodstuffs had to be banned over a considerable area, based on derived intervention levels of radioactive contamination. Access controls were established for personnel and vehicles moving within different areas of the 30 km zone. Special emphasis was placed on measures to reduce resuspension of activity by chemical fixation in more highly contaminated zones, and efforts were also made to prevent or reduce the effects of contamination of water bodies and groundwater supplies.

In summary, it is evident that the Soviet authorities were able to assemble a massive emergency response effort which was deployed rapidly. The chronology of the response indicates measures being applied progressively, first to mitigate early health effects at increasing distances from the plant, followed by efforts to control longer term exposure.

7.5.2 Comment on Soviet estimate of collective doses to the European part of USSR

The Soviet authorities have estimated the collective dose to the European part of the USSR from various pathways. For external exposure they calculate a collective dose of around $3x10^5$ man Sv. For internal exposure resulting from consumption of foodstuffs contaminated with Cs, they estimate a figure of around $2x10^6$ man Sv. Finally, their quoted estimate of the number of thyroid cancer fatalities appears to suggest a collective effective dose from I in milk of around 10^5 man Sv.

These figures are somewhat out of line with previous estimates of collective dose from atmospheric releases of Cs (eg from weapons fallout), which suggest that the contributions from foodchains and external exposure are roughly equal. However, it must be noted that Soviet scientists have been performing whole-body examinations of exposed people and find agreement between observed and calculated Cs levels in only about 3% of cases. The remaining 97% average about ten times lower than expected. This may result from some of the assumptions involved in their model calculations, relating to the rate of uptake of Cs by plants and the consumption habits of individuals. Thus, the contribution to foodchain doses from Cs may be up to a factor of ten lower than that quoted above[9] (ie 2×10^5 man Sv), yielding a net Soviet estimate (summed over all pathways) of order 6×10^5 man Sv. Using a linear dose-risk relationship, this implies around 7,500 fatal cancers in the European part of the USSR as a result of the accident. According to the Soviet report(8), the mortality rate from spontaneous cancer will give rise to 9.5×10^6 cases in the same population.

7.6 Estimated UK Doses in Perspective

It is important to set the results of the above dosimetric assessment in some perspective, to give an appreciation of the low levels of risk involved. This is achieved here by taking the estimated doses for the UK (50 year individual dose of 50μSv and collective dose of 2,800 man Sv) and comparing them, and the risks they represent, with other doses and risks. Before embarking upon this comparison, a general indication of the low dose levels can be obtained by appreciating that external dose rates in Europe from Chernôbyl are now so low that, in many cases, they cannot be distinguished from background levels.

7.6.1 Comparison of UK dose from Chernobyl with background radiation

The average annual dose in the UK from background is around 2 mSv; this may be compared with the estimated 50 year individual dose from Chernobyl of 0.05 mSv. The corresponding collective dose to the UK from background, over the next 50 years, is around 5×10^6 man Sv, which may be compared with the figure of 2.8×10^3 man Sv from Chernboyl. Clearly the dose from Chernobyl is very much less than that from background.

An alternative way of comparing with background is to consider how the background dose rate varies throughout the UK. This variation can be up to around 1 mSv per year. Thus the 50 year individual dose from Chernobyl of 0.05 mSv corresponds to (say) living in East Anglia and having approximately a three-week holiday in Cornwall.

7.6.2 Comparison of UK dose from Chernobyl with weapons testing

The current average annual individual dose from weapons fallout is around 10 μSv. Exposure from this source has been falling and will continue to fall in future years. On the assumption that the rate of decline in recent years will continue, the total collective dose to the UK from this source over the next 50 years will be around 11,000 man Sv. This may be compared with the collective dose from Chernobyl of 2,800 man Sv.

7.6.3 Comparison of UK risks from Chernobyl with smoking

Using the data quoted by Sir Walter (now Lord) Marshall et al (10), it can be shown that the UK risk posed by Chernobyl is equivalent to the compulsory smoking of less than 3/10000 of a cigarette per week (ie less than 2/100 of a cigarette per year) for 30 years.

7.6.4 Comparison of the cancer risk from Chernobyl with cancer statistics

Cancer deaths in England and Wales numbered 134,270 in 1983 and 140,101 in 1984; thus the variation between these adjacent years is just under 6000 (similar variations can be observed between other years). This may be compared with the 35 or so UK cancer fatalities predicted to result (using cautious risk factors) over the next 50 years from Chernobyl. Clearly, if this cancer risk is eventually expressed in the UK it will not be perceptible in the cancer statistics.

7.7 References

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TABLE 1

Dosimetric Assessment for Western Europe: Contribution by Pathway
(Note some rounding of numbers)

Pathway	Collective Dose Man Sv	%
INHALATION	3600	5
INGESTION		
MILK	11000	14
VEG	15200	20
MEAT	12500	17
EXTERNAL	33300	44
TOTAL	75600	

TABLE 2
Dosimetric Assessment for Western Europe
(Note some rounding of numbers)

Country	Mean Individual Dose (μSv)		Collective Dose (SRD Estimate) (Man Sv)	Comments
	National Estimate	SRD Assessment		
Austria		610	4600	
Belgium	120	140	1380	Recommendation to keep dairy cows indoors - not always adhered to.
Denmark	< 270	160	820	Cattle being fed on stored fodder; results in relatively low contribution of milk to total dose.
Finland	2100	280	1370	Cows not returned to pasture from winter feeding until 26 May. Milk contribution relatively small.
France	50	46	2500	
W Germany	70	250	15400	National estimate is for first year only. Large local variations eg Bavaria.

TABLE 2 (continued)

Country	Mean Individual Dose (μSv) National Estimate	Mean Individual Dose (μSv) SRD Assessment	Collective Dose (SRD Estimate) (Man Sv)	Comments
Greece		260	2500	Relatively high dose from food in comparison with external irradiation. Based on sparse data.
Italy	90	500	28600	National estimate is only for first two months and only considers certain foodchain pathways.
Netherlands	450	275	3950	Cattle taken indoors from 3 May to 8 May.
Norway		770	3200	'Mean' figures are not necessarily weighted according to population; difficult to extract weighted dose from data.
Portugal		0.4	4	Very limited data - much extrapolation.
Spain		1.2	45	Very limited data - much extrapolation.
Sweden		770	6400	Much variability across country. Mean dose is arithmetic mean for range in populated areas.

TABLE 2 (continued)

Country	Mean Individual Dose (μSv)		Collective Dose (SRD Estimate) (Man Sv)	Comments
	National Estimate	SRD Assessment		
Switzerland	330	300	1900	
United Kingdom		50	2800	

TABLE 3
Dosimetric Assessment for Eastern Europe
(Note some rounding of numbers)

Country	Mean Individual Dose (µSv)	Collective Dose (man Sv)
Albania	300	830
Bulgaria	700	6250
Czechoslovakia	600	9200
East Germany	500	8370
Hungary	1000	10700
Poland	1200	43700
Romania	600	13500
Yugoslavia	300	6800

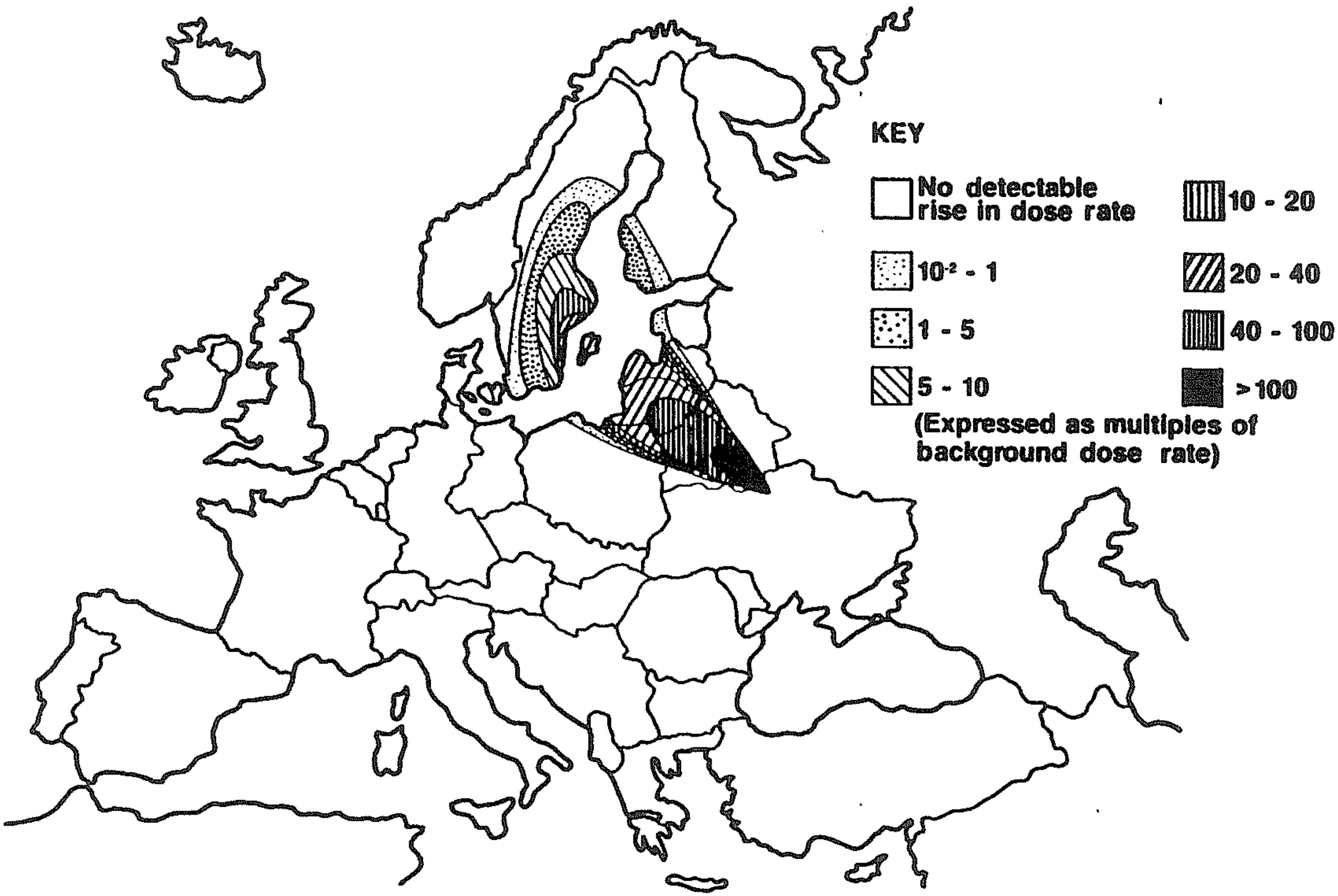


FIG.1 RADIATION DISPERSION PATTERN ACROSS EUROPE 28 APRIL 1986

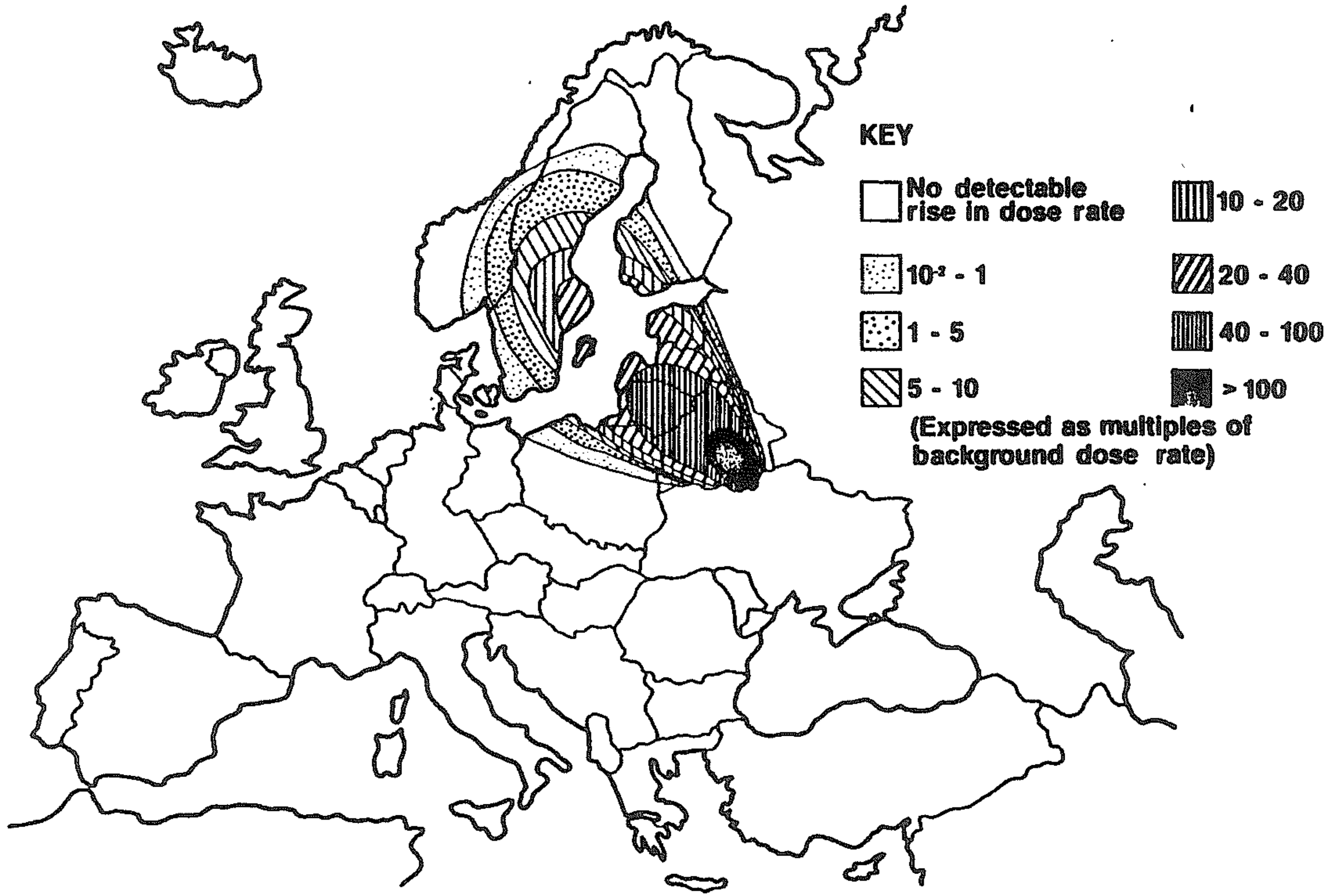


FIG.2 RADIATION DISPERSION PATTERN ACROSS EUROPE 29 APRIL 1986

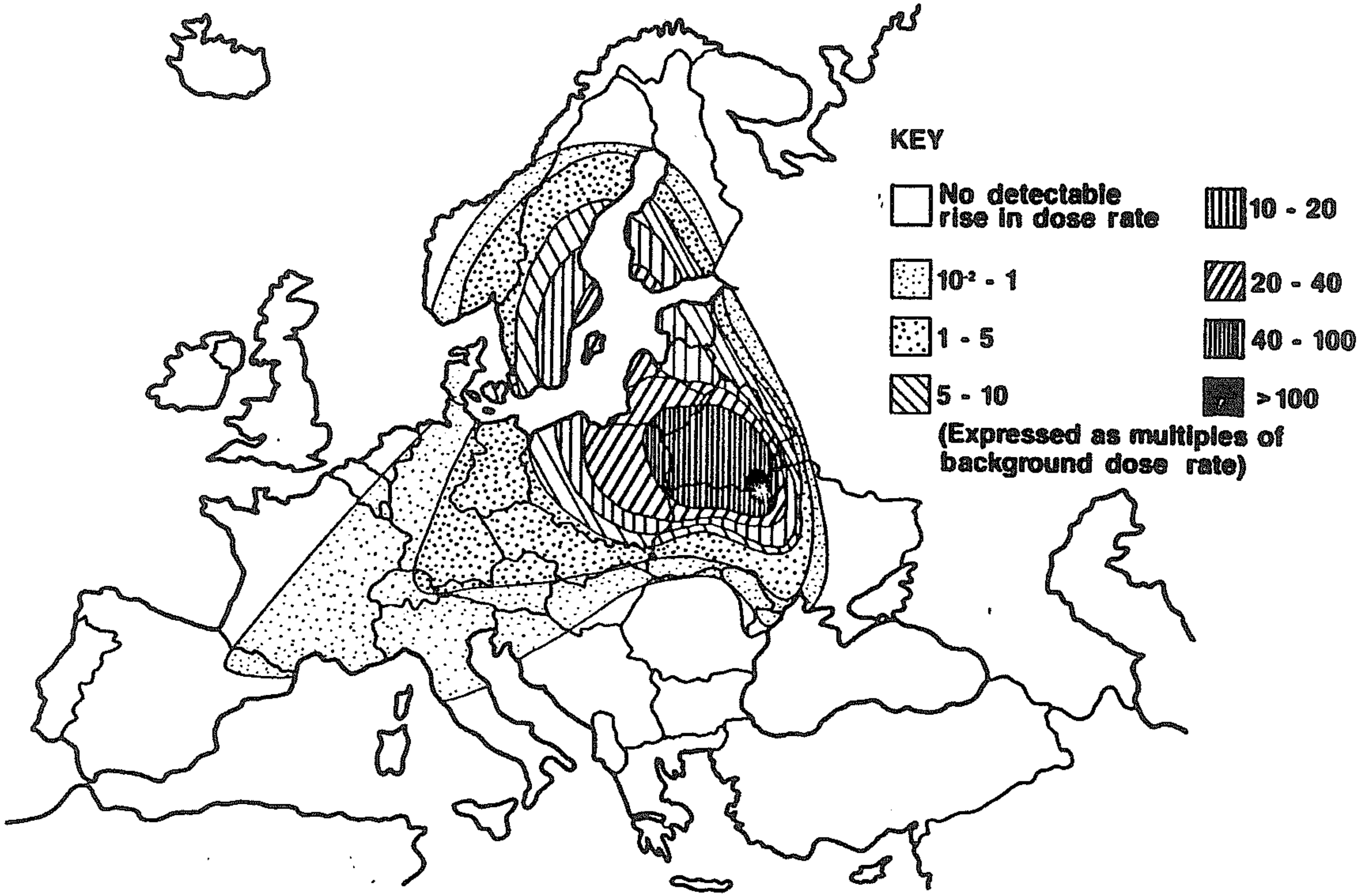


FIG.3 RADIATION DISPERSION PATTERN ACROSS EUROPE 30 APRIL 1986

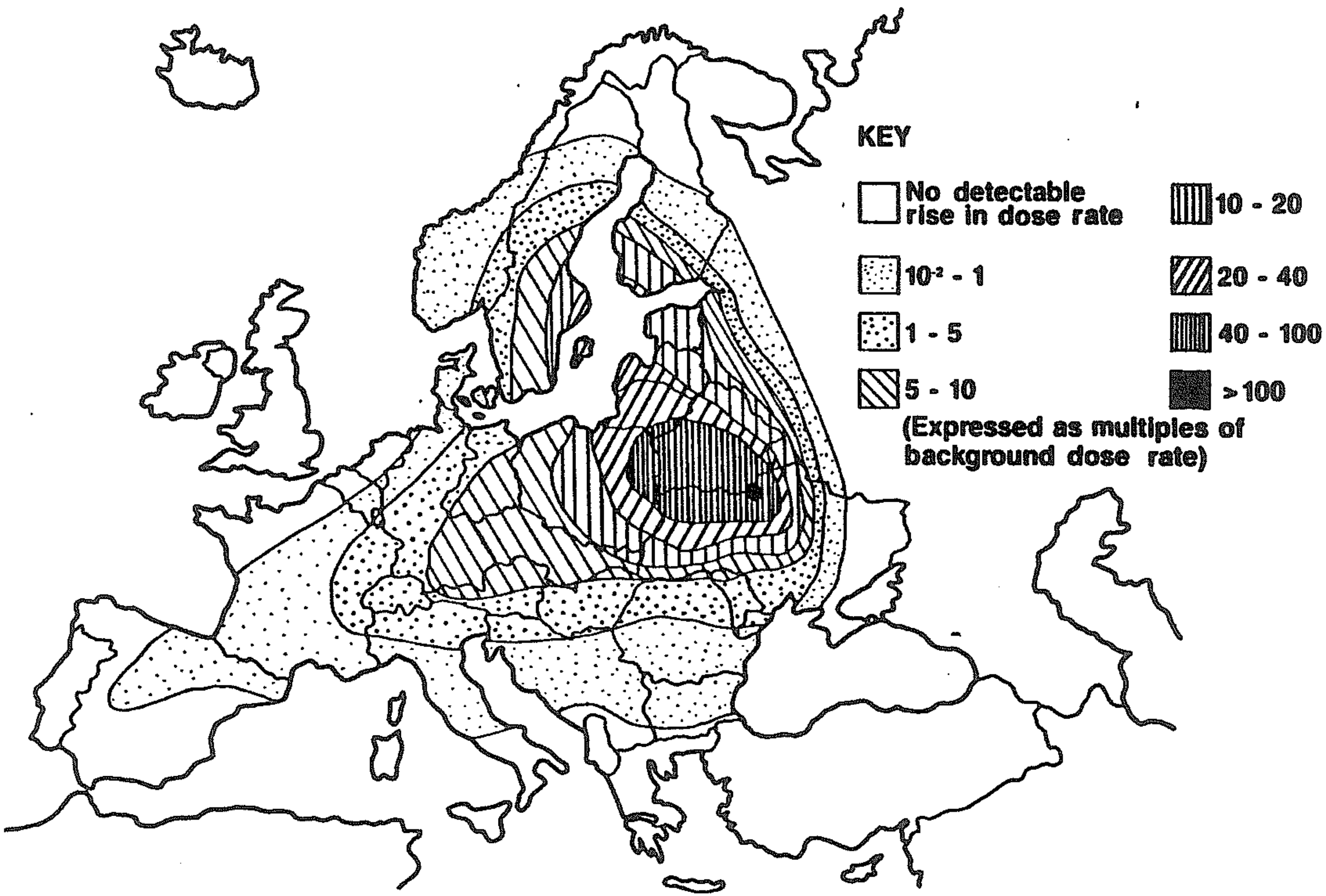


FIG. 4 RADIATION DISPERSION PATTERN ACROSS EUROPE 1 MAY 1986

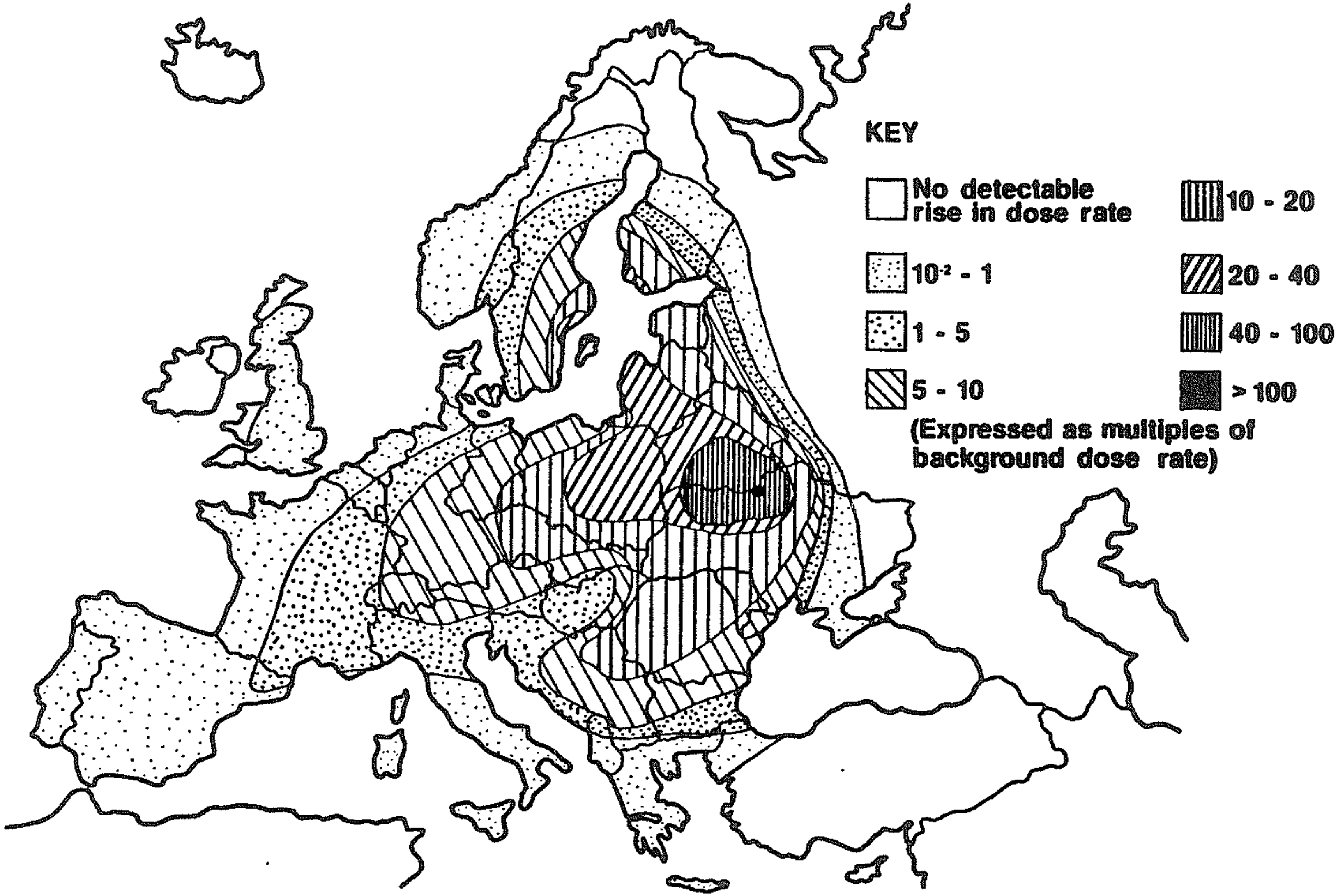


FIG.5 RADIATION DISPERSION PATTERN ACROSS EUROPE 2 MAY 1986

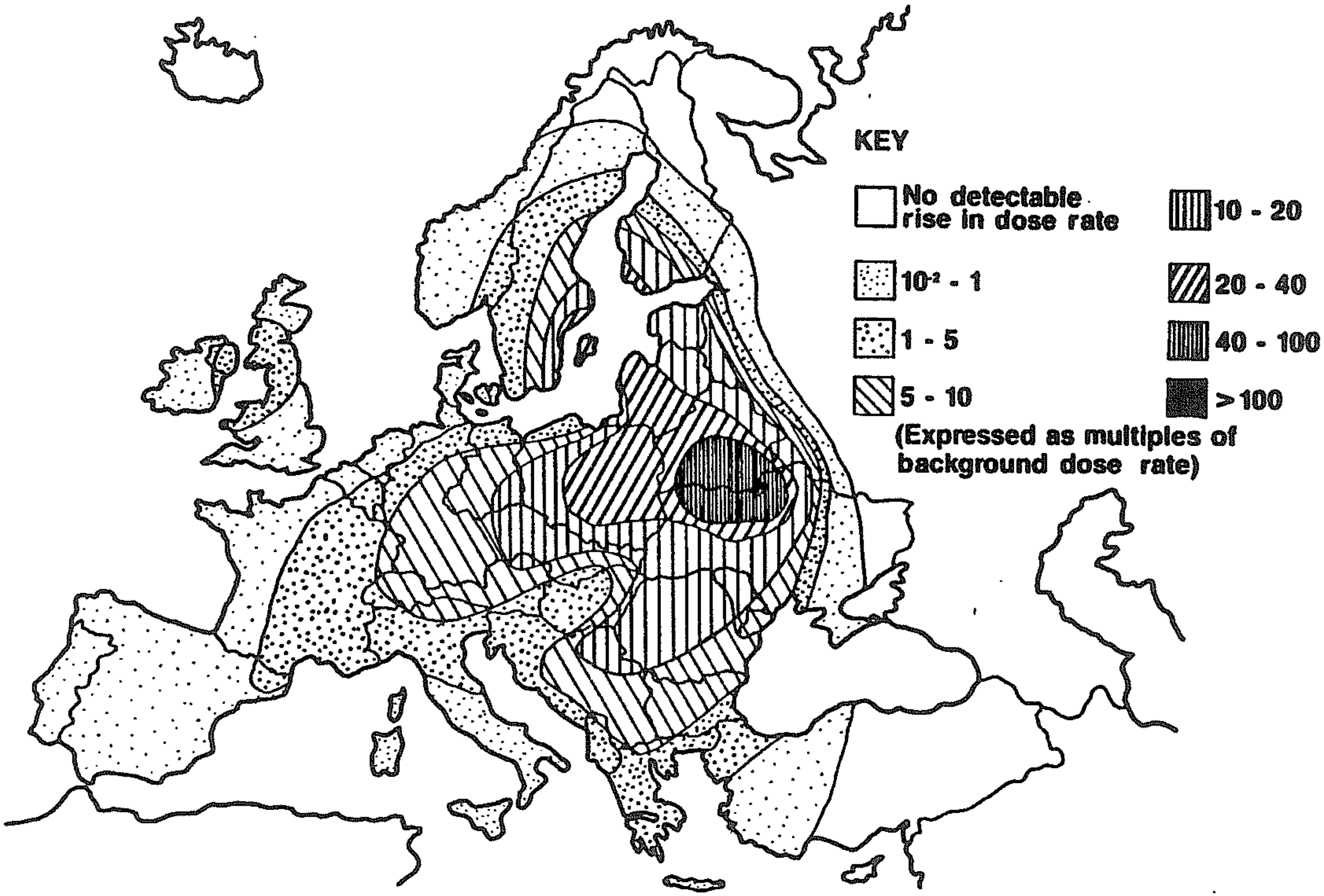
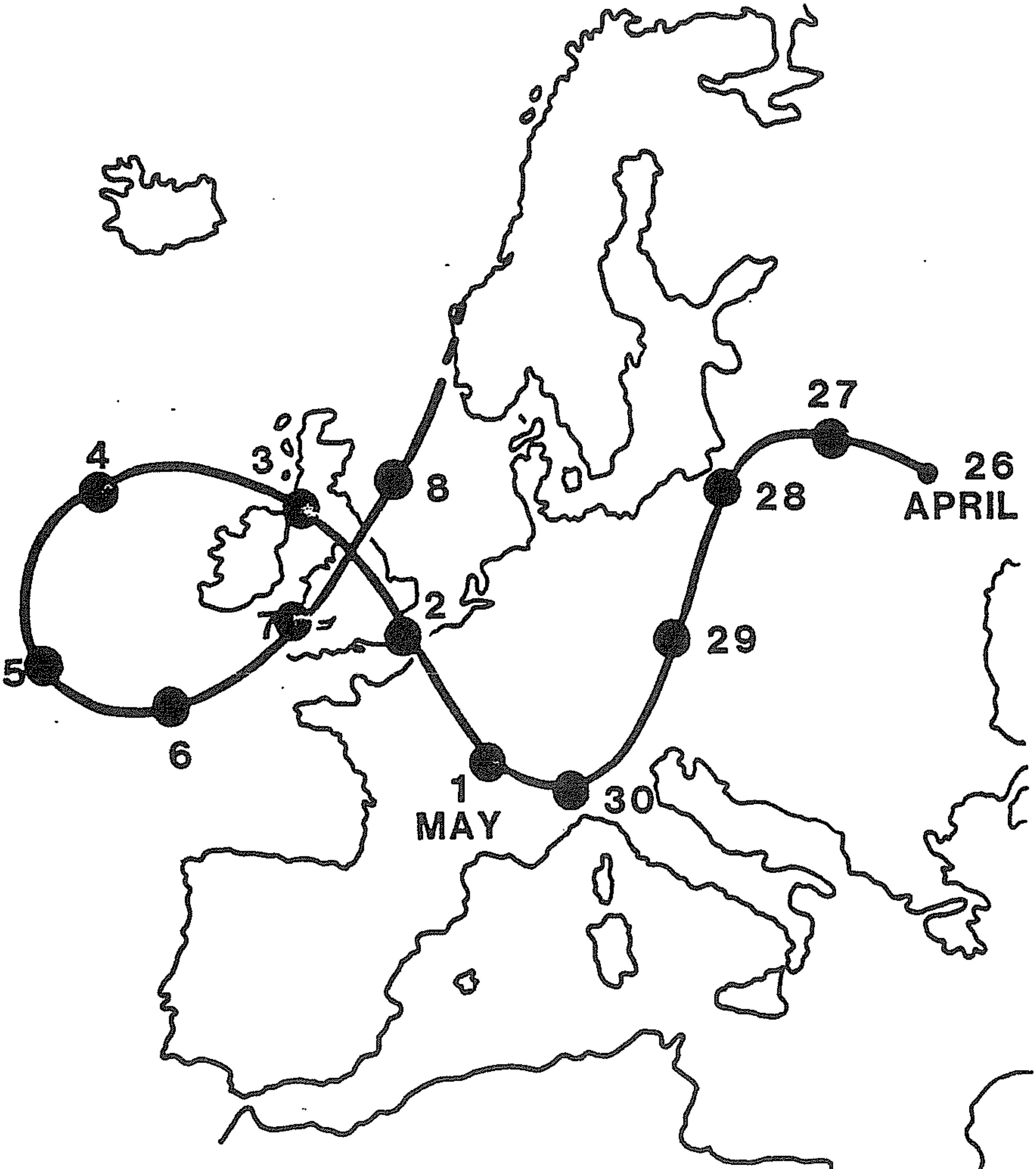
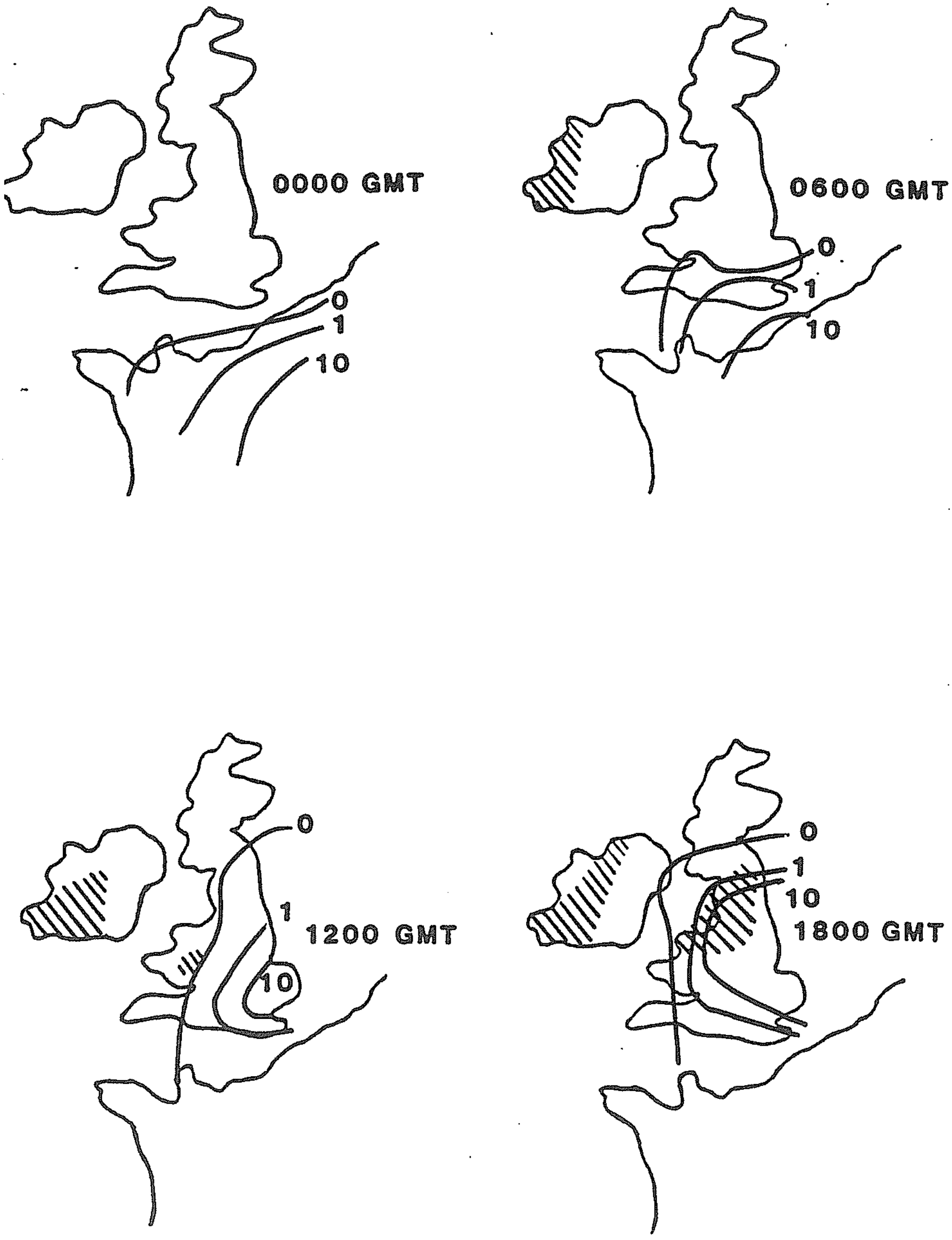


FIG. 6 RADIATION DISPERSION PATTERN ACROSS EUROPE 3 MAY 1986

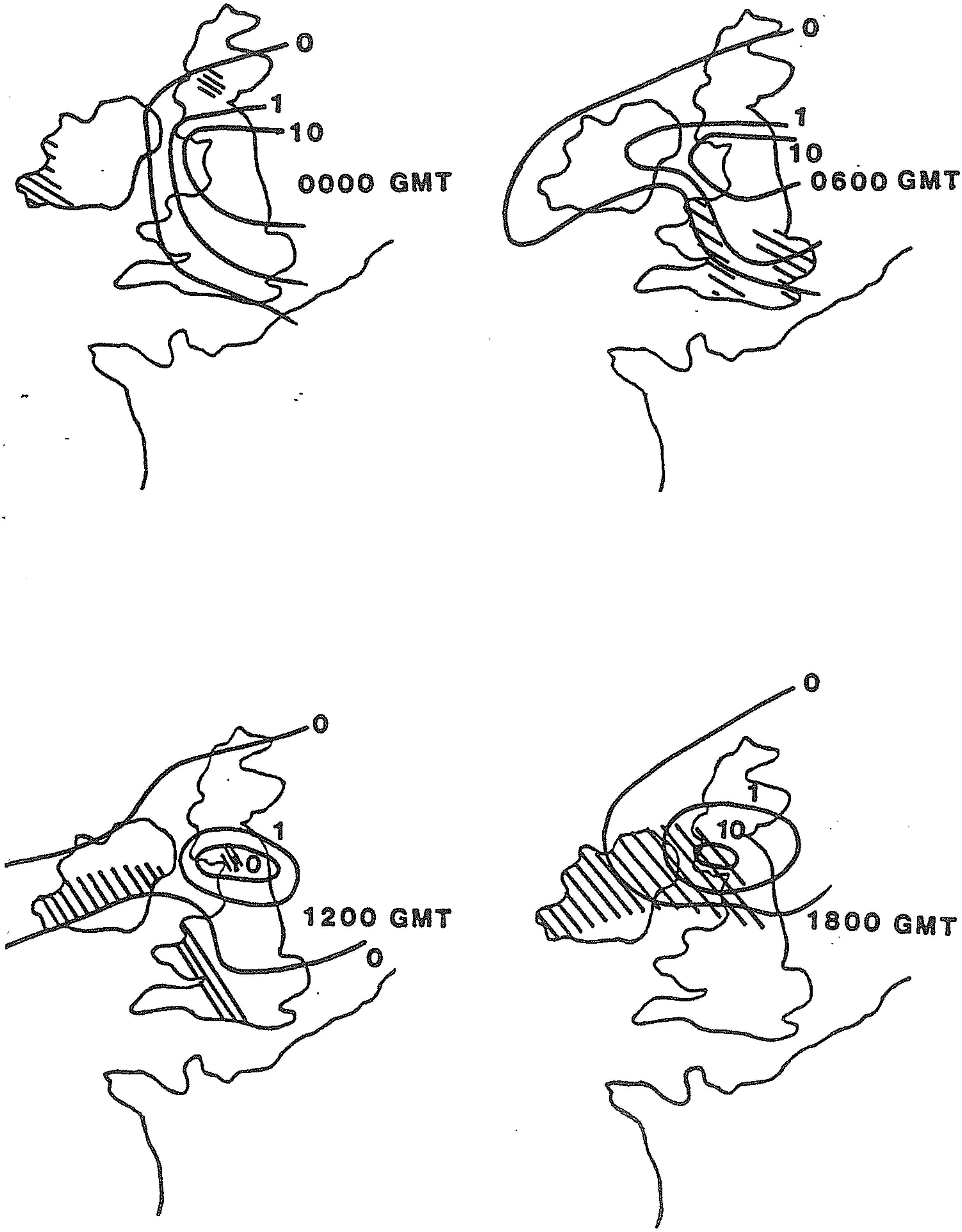
PATH OF CLOUD TO U.K.
(METEOROLOGICAL OFFICE)



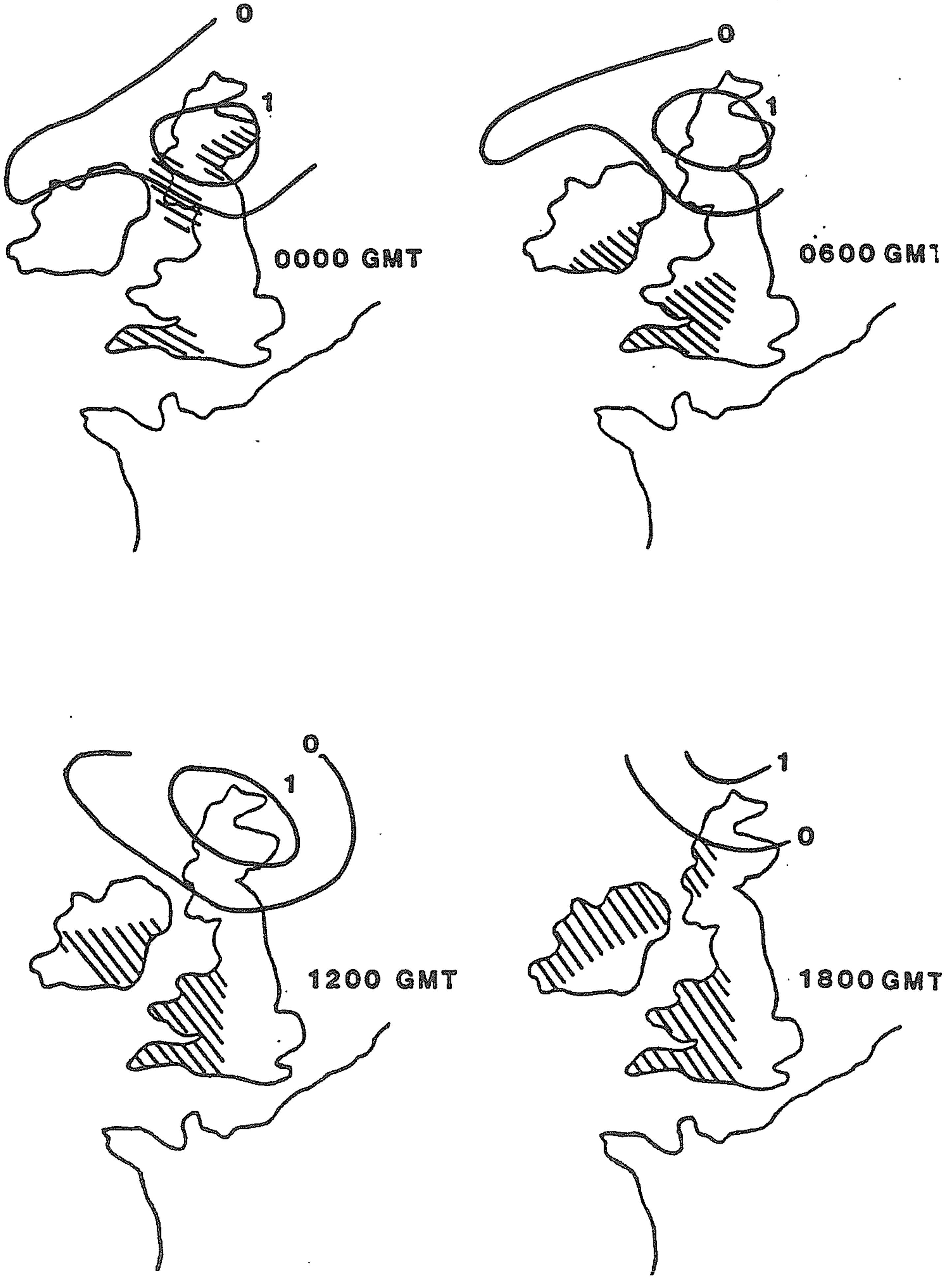
FRIDAY 2 MAY



SATURDAY 3 MAY



SUNDAY 4 MAY



DOSIMETRIC ASSESSMENT FOR WESTERN EUROPE

1 BASED ON MONITORING DATA

2 THREE PATHWAYS

(i) INHALATION

(ii) INGESTION

(iii) EXTERNAL EXPOSURE

3 INTEGRATED TO 50 YEARS

4 ISOTOPES OF I, Cs, Ru.

INHALATION

- TIME INTEGRATED AIR CONCENTRATIONS ESTIMATED, BASED ON MEASURED AIRBORNE LEVELS DURING PLUME PASSAGE

- USED IN CONJUNCTION WITH BREATHING RATE AND DATA ON DOSE PER UNIT INTAKE OF ACTIVITY

- ESTIMATE COMMITTED EFFECTIVE DOSE-EQUIVALENT

INGESTION

FOODSTUFFS CONSIDERED:-

MILK

GREEN VEGETABLES

MEAT

USE RESULTS OF MODELS FOR TIME-DEPENDENT TRANSFER OF ACTIVITY THROUGH FOODCHAINS TO MAN IN COMBINATION WITH

(i) MEASURED LEVELS IN FOODSTUFFS WHERE AVAILABLE

OR

(ii) MEASURED DEPOSITION LEVELS

YIELDS TOTAL ACTIVITY INGESTED BY MAN - CAN BE RELATED TO DOSE.

EXTERNAL EXPOSURE

BASED ON DEPOSITION LEVELS WITHIN EACH COUNTRY AND STANDARD MODELS FOR EXTERNAL EXPOSURE.

SHIELDING FACTOR FOR PROTECTION BY BUILDINGS, ETC - 0.25

CONTRIBUTIONS TO DOSE FOR

WESTERN EUROPE

SUMMED OVER ALL COUNTRIES

PATHWAY	MAN SV	%
INHALATION	3600	5
INGESTION		
MILK	11000	14
VEG.	15200	20
MEAT	12500	17
EXTERNAL	33300	44

TOTAL 75600 MAN SV

APPLYING RISK FACTOR OF 1.25×10^{-2} / MAN SV

- ESTIMATED NUMBER OF CANCER DEATHS ~950

DISTRIBUTION OF DOSE THROUGHTOUT

WESTERN EUROPE

	MEAN INDIVIDUAL DOSE (μSV)	COLLECTIVE DOSE (MAN SV)
AUSTRIA	610	4600
DENMARK	160	820
FINLAND	280	1370
FRANCE	46	2500
W. GERMANY	250	15400
ITALY	500	28600
NORWAY	770	3200
SWEDEN	770	6400
UK	50	2800

DISTRIBUTION OF DOSE THROUGHOUT EASTERN EUROPE

	MEAN INDIVIDUAL DOSE (μSV)	COLLECTIVE DOSE (MAN SV)
ALBANIA	300	830
BULGARIA	700	6250
CZECHOSLOVAKIA	600	9200
E. GERMANY	500	8370
HUNGARY	1000	10700
POLAND	1200	43700
ROMANIA	600	13500
YUGOSLAVIA	300	6800

TOTAL ~ 100 000 MAN SV

APPLYING RISK FACTOR OF 1.25×10^{-2} / MAN SV

- ESTIMATED NUMBER OF CANCER DEATHS ~ 1250

CONTRIBUTION BY PATHWAY TO DOSE ESTIMATED FOR UK

PATHWAY	MAN SV	%
INHALATION	170	6
INGESTION		
MILK	1470	53
VEG	190	7
MEAT	630	23
EXTERNAL	310	11

TOTAL 2770 MAN SV

- APPLYING RISK FACTOR OF 1.25×10^{-2}/ MAN SV
- ESTIMATED NUMBER OF CANCER DEATHS~35

REPRESENTATIVE RADIATION DOSES
(NRPB CALCULATIONS)

AREA OF THE UK AND POPULATION (MILLION)	INDIVIDUAL ADULT EFFECTIVE DOSE EQUIVALENT FOR THE FIRST YEAR (μSv)
CUMBRIA (0.48)	190
REST OF ENGLAND (46)	20
CLWYD & GWYNEDD (0.63)	190
REST OF WALES (2.2)	29
DUMFRIES & GALLOWAY (0.15)	190
REST OF SCOTLAND (5)	83
N. IRELAND (1.6)	97

COMMENT ON UNCERTAINTY

1 USE OF WEIGHTED MEAN ACTIVITY CONCENTRATIONS FOR EACH COUNTRY GIVES RISE TO UNCERTAINTY OF A FACTOR OF A FEW

2 OTHER SOURCES OF UNCERTAINTY

(i) UNCERTAINTIES IN MONITORING DATA

(ii) USE OF STANDARD DOSIMETRIC MODELS

(iii) USE OF UK CONSUMPTION DATA

THUS - DOSE ESTIMATES MAY BE UNCERTAIN BY FACTORS RANGING FROM AROUND 5 TO AROUND 10, DEPENDING ON THE COUNTRY

COUNTERMEASURES IN EUROPE

PROTECTIVE ACTIONS TAKEN INCLUDE :-

- ADVICE TO STAY INDOORS
- ADVICE THAT CHILDREN SHOULD NOT PLAY IN SAND
- RESTRICTIONS IN USE OF MILK
- DAIRY CATTLE TAKEN IN FROM GRAZING
- ADVICE NOT TO DRINK RAINWATER
- ADVICE NOT TO EAT FRESH SURFACE VEGETABLES
- RESTRICTIONS ON SALE OF MEAT
- ADVICE TO TAKE IODINE PILLS
- EXPLICITLY "NO ACTION"

EFFECTS OF COUNTERMEASURES

- MINIMAL IMPACT ON COLLECTIVE DOSES

- LARGER IMPACT ON CRITICAL GROUP DOSES
  E.G (NRPB CALCULATIONS)

  - LARGEST EFFECTS CALCULATED FOR GREECE AND ITALY:-

    EFFECTIVE DOSE REDUCED BY FACTOR OF ~ 2
    THYROID DOSE REDUCED BY FACTOR OF ~ 8

  - SMALLER EFFECTS NOTED ELSEWHERE
    - EG. UK CRITICAL GROUP DOSES REDUCED BY ONLY 6 TO 8%

EFFECTS OF COUNTERMEASURES

- PROBLEM

  - WIDE RANGE OF INTERVENTION LEVELS ADOPTED (ie. Bq/Kg or Bq/LITRE)

  - VARIED WIDELY FROM COUNTRY TO COUNTRY

  - NEED FOR INTERNATIONAL COLLABORATION ON DERIVED INTERVENTION LEVELS

CONSEQUENCES IN USSR

1. ON-SITE
2. WITHIN 30 KM ZONE
3. EUROPEAN PART OF USSR

ON - SITE PERSONNEL

- 300 PEOPLE REQUIRE HOSPITAL TREATMENT

- 203 DIAGNOSED AS SUFFERING FROM ACUTE RADIATION SYNDROME - FROM ABSORBED DOSES IN THE RANGE 1 - 16GY

- EXTENSIVE THERAPEUTIC TREATMENT
  - EG BLOOD TRANSFUSIONS
    CHEMOTHERAPY AND VARIOUS ANTIBIOTICS
    ANTIVIRAL DRUG
    BONE MARROW TRANSPLANTS
    (13 PEOPLE ALL BUT ONE DIED)
  - GENERALLY EFFECTIVE IN ACHIEVING INCREASED SURVIVAL RATE

- 29 FATALATIES FROM AMONG THOSE DIAGNOSED AS SUFFERING FROM ACUTE RADIATION SYNDROME

- 2 FURTHER FATALITIES REPORTED TO HAVE OCCURRED IN IMMEDIATE AFTERMATH OF ACCIDENT

WITHIN 30KM ZONE

DOSES

MOST INDIVIDUAL DOSES BELOW 250 mSv ALTHOUGH IN MORE CONTAMINATED AREAS MAY BE 300 - 400 mSv

- NO SYMPTOMS OF ACUTE RADIATION SYNDROME IN OFF-SITE AREAS

COLLECTIVE EFFECTIVE DOSE - 1.6×10^4 MAN Sv

- USING LINEAR DOSE-RISK, IMPLIES AROUND 200 EXCESS CANCER FATALITIES IN COMING DECADES

COLLECTIVE THYROID DOSE - 4×10^4 MAN Sv

- MAY BE AROUND 20 FATAL AND 400 NON-FATAL EXCESS THYROID CANCERS

FATALITY FIGURES REPRESENT INCREASE OVER BACKGROUND CANCER MORTALITY RATE OF LESS THAN 1%

WITHIN 30KM ZONE

COUNTERMEASURES

26 APRIL - PEOPLE IN PRIPYAT ASKED TO SHELTER, CLOSE WINDOWS, ETC.

- STABLE IODINE ISSUED

- PROJECTED DOSES APPROACHED INTERVENTION LEVELS (LOWER LEVEL - 250 mSv, UPPER LEVEL - 750 mSv)

27 APRIL, 2.0PM - 45,000 PEOPLE EVACUATED FROM PRIPYAT WITHIN 2.5 HOURS

OVER NEXT FEW DAYS - GRADUAL EVACUATION OF ANOTHER 90,000 PEOPLE WITHIN RADIUS OF ~30KM

EUROPEAN PART OF USSR

SOVIET ESTIMATES OF COLLECTIVE DOSE

PATHWAY		MAN Sv
EXTERNAL		3×10^5
Cs IN FOODSTUFFS		2×10^6
I IN MILK		1×10^5
	TOTAL	2.4×10^6

HOWEVER

Cs IN FOODSTUFFS MAY BE OVERESTIMATED BY ABOUT AN ORDER OF MAGNITUDE

EUROPEAN PART OF USSR

ASSUMING REDUCED VALUE FOR Cs IN FOODCHAIN PATHWAY

TOTAL COLLECTIVE EFFECTIVE DOSE - ~ 6×10^5 MAN Sv

- USING LINEAR DOSE - RISK, IMPLIES AROUND 7,500 EXCESS FATAL CANCERS IN COMING DECADES

COLLECTIVE THYROID DOSE - ~ 3×10^6 MAN SV

- MAY BE AROUND 1500 FATAL (INCLUDED IN ABOVE FIGURE OF 7,500) AND 30,000 NON-FATAL EXCESS THYROID CANCERS.

FATALITIES REPRESENT INCREASE OVER BACKGROUND CANCER DEATH RATE OF AROUND 0.05%.

U. DOSE IN PERSPECTI_E
COMPARISON WITH BACKGROUND

AVERAGE INDIVIDUAL ANNUAL DOSE - BACKGROUND - 2×10^{-3} SV

AVERAGE INDIVIDUAL 50 YEAR DOSE - CHERNOBYL - 5×10^{-5} SV

COLLECTIVE DOSE TO UK (50 YEAR) - BACKGROUND - 5×10^{6} MAN SV

COLLECTIVE DOSE TO UK (50 YEAR) - CHERNOBYL - 3×10^{3} MAN SV

UK DOSE IN PERSPECTIVE

COMPARISON WITH VARIATIONS IN BACKGROUND

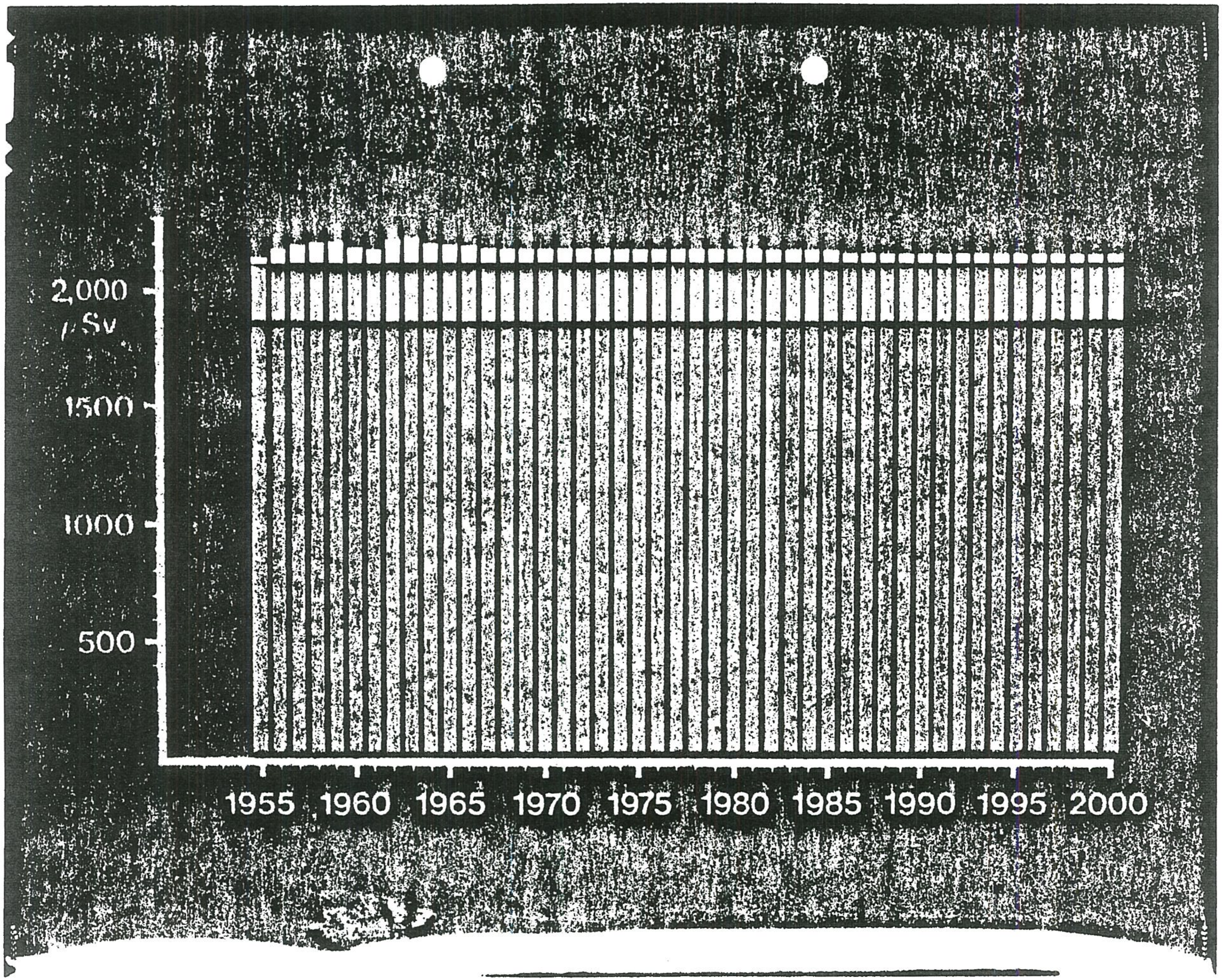
VARIATION THROUGHOUT UK - UP TO AROUND 10^{-3} SV PER YEAR

THUS AVERAGE 50 YEAR INDIVIDUAL DOSE FROM CHERNOBYL OF 5×10^{-5} SV IS EQUIVALENT TO (SAY) LIVING IN EAST ANGLIA AND HAVING APPROXIMATELY A THREE - WEEK HOLIDAY IN CORNWALL.

UK DOSE IN PERSPECTIVE
COMPARISON WITH CANCER INCIDENCE IN ENGLAND AND WALES

YEAR	TOTAL NUMBER OF CANCER DEATHS
1979	129638
1981	131691
1983	134270
1984	140101

COMPARE WITH ESTIMATED ~ 35 CANCER DEATHS FROM CHERNOBYL IN THE NEXT 50 YEAR


2,000
µSv
1500
1000
500
1955
1960
1965
1970
1975
1980
1985
1990
1995
2000

IN CONFIDENCE

HSSC(87)P20

UNITED KINGDOM ATOMIC ENERGY AUTHORITY

HEALTH AND SAFETY STUDIES COMMITTEE

REVIEW OF AUTHORITY REACTOR SAFETY PROMPTED BY THE CHERNOBYL ACCIDENT

by

R. S. Peckover

The Safety of Authority Reactors has been reviewed in the light of Chernobyl. A number of specific areas have been addressed highlighted by the deficiencies in design and operation of the RBMK. The review has confirmed that our designs, procedures, operator training and emergency plans are soundly based.

20 May, 1987

Distribution:

Standard HSSC
Symposium Participants

REVIEW OF AUTHORITY REACTOR SAFETY PROMPTED BY THE CHERNOBYL ACCIDENT

1. Introduction

As a result of the accident which occurred to the Unit 4 reactor at Chernobyl in the Ukraine in April 1986, some 31 people died either directly or as a result of receiving lethal radiation doses. Something like 50 million curies of radioactivity were released into the atmosphere corresponding to a few percent of the total fission product inventory. The accident has been extensively analysed, and is now reasonably well understood; it was caused by a combination of operator malpractice and design shortcomings. The operator failings magnified the effects of the design shortcomings, which must thus be viewed as the primary cause. The culpable combination of design features are found only in the RBMK reactor type which is unique to the Soviet Union. It follows that a Chernobyl-type accident could not happen to Authority reactors, or indeed to UK reactors. Nevertheless it is clearly prudent to re-examine the Authority's reactors in the light of Chernobyl. Reviews have been carried out both by site management and by SRD. This paper presents a summary of this activity.

2. The Chernobyl Accident and its Causes

A detailed account of the Chernobyl reactor accident is given in the paper "The Chernobyl Accident and its Consequences" by J. H. Gittus et al. To provide context here the main characteristics of the Chernobyl reactor and the course of the accident are briefly recounted in Appendix 1. The accident came about as a result of a combination of operator malpractice and design shortcomings whereby the reactor became unstable during the course of an experiment and a prompt critical excursion resulted.

The main design shortcomings which contributed to the accident were (i) a positive void coefficient of reactivity which allowed an unstable inherently unsafe operating regime, (ii) a slow and clumsily organized shutdown system. NNC reviewed the RBMK design 10 years ago and had a number of reservations, as detailed in the paper by Gittus et al, which implied that the RBMK design could not be licensed in the UK.

The actions by the operators revealed that (i) it was easy to over-ride vital protection systems - a further design weakness, (ii) the operators failed to appreciate the hazard of operating in the unstable regime - pointing to inadequate operator training, and (iii) the operators ignored operating instructions relevant to safety - pointing again to inadequate operator training and also to a poor management environment with regard to safety.

The accident occurred during an experiment which clearly distracted the focus of attention away from the safety of the reactor. The Russians have stated that the experiment was badly planned, and that its safety case was inadequate, and was not properly reviewed - pointing again to management shortcomings with regard to safety.

Two other issues highlighted by the accident are fires and emergency planning. The intense fire in the reactor core seems strongly to have influenced the dispersal of fission products in the atmosphere. An accident of the magnitude of Chernobyl put enormous demands on the emergency authorities and plans - it must be said that the Russians coped extraordinarily well.

Based on these comments about Chernobyl, it is convenient to consider the Authority reactors under the following headings:-

- Reactor Safety Environment
- Core Reactivity Characteristics
- Shutdown Systems
- Man/machine Interface
- Operator Training
- Experiments in Reactors
- Fire
- Emergency Planning

Each of these items form one of the following sections.

3. Authority Reactors

The Authority has nine reactors on three sites - at Harwell, Winfrith and Dounreay (see Appendix 2). Of these the only ones with sufficient radioactive inventory to produce an accident even remotely approaching the scale of Chernobyl are the two prototype power reactors - PFR at Dounreay and SGHWR at Winfrith - and the two closely similar materials testing reactors at Harwell - DIDO and PLUTO. This review concentrates on these reactors, though the others have not been ignored. GLEEP, NESTOR and DIMPLE are low powered reactors, generating at most a few tens of kilowatts, with radioactive inventories much less than 1000 curies of I_{131} equivalent. Appendix 3 provides some additional detail about these. ZEBRA and HECTOR are currently out of commission.

4. Reactor Safety Environment

It has always been Authority policy to give the highest priority to safeguarding both its own staff and the general public from the potentially harmful effects of both accidental releases of radioactivity and any other small releases which may occur during normal operation. The NII does not license Authority nuclear operations, nevertheless it is required that the Authority's safety requirements should be no less than those for licensed sites. These are in essence that the greatest care must be taken to ensure that a safe design is fully thought through and then followed through in the manufacture, construction, commissioning, operation and maintenance so that the safe design intention is not defeated, either deliberately or inadvertently.

The arrangements for controlling the operation of reactors to ensure high standards of safety are generally similar on the three sites. There are five distinguishable elements - the reactor operations staff, the reactor safety committee, the safety documentation, SRD, and the Site Director.

i) The Site Director is directly answerable to the Chairman of the Authority for the safety of the operations on his site. Reactor operation is not permitted until the Site Director is satisfied that the plant is safe to operate; his permission is signified by the issue of an Authority to Operate (ATO) which is issued annually or more frequently if necessary. The ATO specifies the conditions and restrictions under which the reactor may be operated. Periodically the Site Director arranges for the independent inspection of selected aspects of operational safety.

ii) The reactor has both design and operational safety documentation. Authority reactors have been designed to rigorous safety standards, and the designs thoroughly scrutinized. The operational safety documentation shows how the operations are intended to keep within the design conditions and describe changes, particularly ones that could affect safety. They also describe how experiments will be carried out safely. Documentation relating to safety includes Standing Orders for operators, and emergency operating procedures.

iii) Each reactor has an associated reactor safety committee (or working party) whose membership is predominantly independent of the reactor line management. Throughout the lifetime of the reactor, the safety committee vets changes in design, changes in operation and experiments proposed. It also reviews the safety and operating performance of the reactor and the safety documentation. It advises the Site Director on the safety of the reactor as appropriate and particularly prior to the issue of an ATO.

iv) The prime responsibility for reactor safety necessarily lies with the reactor operations staff themselves especially the Operations manager. There are strict operating instructions governing all safety aspects. There is detailed training of operations staff. The shift managers are not appointed until they have been fully trained, and tested to ensure they have adequate knowledge about normal and abnormal operation to run the plant safely.

v) The Director of SRD is responsible directly to the Chairman of the Authority for independent safety advice on the Authority's reactors and plant. He is also Authority Programme Director for GNSR - the Authority-wide research and development programme for General Nuclear Safety; one of its aims is to enable him to acquire information relevant to this responsibility. Prior to the re-issue of an ATO for major plant, the Site Director formally consults the Director of SRD as to whether he sees any reason why the ATO should not be issued. The Director of SRD is supported by the Authority Reactors and Plant Safety (ARPS) Project at SRD. A member of ARPS sits on each of the reactor safety committees. SRD also carries out independent safety audits of the Authority's reactors.

The arrangements just described indicate that a good management environment has been established in which safety is taken seriously and vigilance is maintained to ensure continued high standards of safety for Authority reactors. By contrast, the development of the accident at Chernobyl suggested an absence of such an environment.

5. Core Reactivity Characteristics

An RBMK reactor of the type in the Chernobyl accident has a large positive void coefficient of reactivity (i.e. the replacement of water by steam bubbles in the coolant channels would lead to an increase in reactivity). At normal full power this is more than compensated for by the negative Doppler coefficient of reactivity. However below about 20% of full power the power coefficient is positive, and the reactor is potentially unstable. The Chernobyl reactor was placed in this regime with little steam present; when the coolant pumps began to run down as expected in the experiment, the voidage rapidly increased and the power level rose culminating in a prompt critical excursion (i.e. a large power surge). What of Authority Reactors?

(a) SGHWR

SGHWR has a small, negative void coefficient at all power levels. This has been confirmed by experiment, calculation and by plant behaviour. Using the control systems available to them, the operators cannot cause the void coefficient to become positive during reactor operation.

The only feasible way in which the void coefficient could be made positive would be the addition of at least 2 tonnes of light water to the 37 tonnes of heavy water moderator - essentially impossible without detection.

Both components of the power coefficient (Doppler fuel and coolant void coefficients) are negative at all reactor power levels. The core is neutronically small and there are no localised flux instabilities. The neutron flux instrumentation is sufficiently robust and fast enough to be able to initiate fast shutdown successfully against the most rapid conceivable reactivity changes.

(b) PFR

It is well known that PFR has a positive void coefficient, but it is restricted to a small volume at the core centre and occurs only then under certain fault conditions. For the core as a whole, however, the void coefficient is negative, as it is in the upper part of the core where bubbles are most likely to form. Sudden void production in the core centre would lead to a mild temperature transient arrested by the increased effect of the negative Doppler fuel coefficient; prompt criticality would not occur.

Widespread boiling of the liquid sodium coolant would require an unusual fault since even at full power it is more than 300°C below its boiling point (and is virtually at atmospheric pressure). The core of PFR is neutronically stable both spatially and temporally. Neither does xenon poisoning give the problems experienced in some thermal reactors.

Besides these intrinsic safety features, there is a fast acting shutdown system which can be actuated by a diversity of parameters. The response of a fast reactor to an inadvertent power surge coupled with postulated failure of the automatic protection system has been the subject of extensive theoretical and experimental study for many years. For all save highly improbable events, the worst consequences can be withstood by the primary containment.

(c) MTRs

For DIDO and PLUTO the void coefficient and overall temperature coefficient of reactivity are negative. Indeed because the void coefficient is strongly negative, the flooding of an initially empty thimble with heavy water can add up to 0.35% reactivity to the reactor. An analysis of the consequences of this when the reactor is critical is included in the Safety Document.

Prompt criticality requires the addition of at least 0.7% reactivity to a critical core. DIDO and PLUTO have an excess reactivity well above this. However the reactivity is absorbed by a large number of thimbles, rigs, control absorbers and poisons so that the reactivity available for initiating prompt critical events is relatively small and these events have a very low probability. In all cases the transients would be rapidly terminated by the reactor protection system.

Sufficient negative reactivity margin is maintained throughout reactor shutdowns so that two separate failures or errors will not make the reactor critical. Moreover, the safety absorbers are raised and available throughout a shutdown period to terminate any unintentional criticality.

It should be noted that the normal power output from DIDO and PLUTO is more than a factor of 10 down from that of SGHWR and PFR.

6. Reactor Shutdown Systems

The shortcomings in the RBMK reactor control and shutdown system materially contributed to the accident. RBMK has a total of over 200 absorber rods, and they were used to perform a variety of functions. There was no discrimination except by means of administrative rules between groups of rods for control and for emergency shutdown. In an emergency shutdown, most of the rods were designed to motor into the core at a speed of 0.4 m/s, full insertion taking 15 to 20 seconds. Because of the xenon poisoning the operators had at one point only the equivalent of about 7 control rods in the core (compared with a minimum of 15 required and 30 recommended). Apart from the operational rules, which were violated, there were no physical constraints on control rod withdrawal. The control system operated in several modes depending on power level, with manual switching between modes and the need for the operator to carry out a number of essential auxiliary tasks such as resetting the set point power level (the failure to do so allowed the large downward power fluctuation prior to the test).

The criticisms of the RBMK system then are it is clumsy to operate, there is little discrimination between control and shutdown, the shutdown rod response was too slow to handle a fast transient, and control rod withdrawal was not sufficiently physically constrained. The Russians accept that major deficiencies exist in this area and propose to rectify them.

What of Authority Reactors?

(a) <u>SGHWR</u>

There are two distinct shutdown systems - the fast-acting liquid shutdown system (LSD) and the slower acting moderator drain. There are no solid withdrawable absorber rods.

The fast-acting shutdown system consists of twelve independent loops. Each is designed to be quickly flooded (in 1 second) with liquid poison (lithium borate in demineralised water). The spurious operation of any one of the twelve liquid shutdown loops would initiate an automatic reactor trip. At least eight loops must always be available, and if fewer than the full twelve are available, the testing frequency must be enhanced according to a rigorous algorithm.

The reactor power is controlled via the level of the borated heavy water moderator in the calandria; the level is changed by means of a pumping/draining circuit between the calandria and the drain tanks. In the event of a reactor trip, the moderator is automatically drained, acting as a secondary diverse shutdown system, and puts the reactor into a safe long-term shutdown state. The moderator drain inserts negative reactivity comparable to that in the LSD in somewhat less than one minute.

The systems of reactivity control and emergency shutdown are thus distinct in SGHWR, unlike those of the Chernobyl-4 reactor. There are no means, either manual or automatic, whereby the amount of negative reactivity injected by the system can be modified by any short-term actions on the part of the operators.

(b) <u>PFR</u>

In PFR there are 5 control rods and 5 shutdown rods consisting of boron carbide pellets clad in stainless steel. All shutdown and control rods are suspended by electromagnets; in the event of a trip all 10 rods are dropped into the core. The maximum time to fully insert the rods is 0.7 sec. If the time recorded for any rod drop exceeds a specified value, an investigation must be held prior to a return to power. The rods are regulatly exercised on load, to give additional assurance that they are free to fall on demand. For control rod withdrawal in manual mode there is a limitation by the system on the rate of reactivity increase (0.6 cents/sec). All trips and abnormal occurrences are formally recorded and formally investigated, and appropriate actions taken to prevent a recurrence.

Although the shutdown rods are dedicated to a trip function and are not used for control and the APS has two independent systems (one relay operated and one solid state), PFR does not have a true diversity in shutdown capability. Designs for a further "ultimate" shutdown device are currently under consideration.

(c) MTRs

The shutdown systems in DIDO and PLUTO are closely similar. DIDO has 6 coarse control absorbers of cadmium in stainless steel sheaths. These are used both for control and shutdown. There is a second diverse system - the VSR - which are vertical cadmium safety rods in hollow fuel elements. PLUTO uses 6 vertical absorbers for both control and shutdown. The second diverse system is provided by 3 coarse control absorbers. For both MTRs the control and shutdown absorbers are interlocked to ensure that criticality cannot be achieved until the shutdown absorbers have been raised. The overall time for the absorbers to drop from fully-out to fully-in is less than 0.6 sec.

7. Man/machine Interface

The essential principle for the safe operation of a nuclear reactor is that the capabilities of the coolant circuits should at all times match the heat generated sufficiently closely that the core temperature always remains below some safe maximum. To ensure this is so each reactor is provided with an Automatic Protection System (APS) which monitors relevant parameters and, if these move outside safe ranges, trips the reactor and where appropriate initiates emergency core cooling. Each Authority reactor has fast acting and reliable automatic protection systems. Provided the APS works as intended, there should be no possibility of a Chernobyl-type incident in an Authority reactor.

At Chernobyl, the operators reduced the usefulness of the Automatic Protection System by disabling several trip initiators, contrary to operating rules. It is clear that if they had not done so the accident would not have occurred. The interfaces between the operators and the safeguards systems on UKAEA reactors have been reviewed afresh in the light of Chernobyl.

There is no hardware or software device which ultimately prevents the bypassing of parts of the APS in any of the AEA's reactors, indeed the changing of some trip levels and the bypassing of others is a standard and necessary practice during reactor manoeuvring between high and lower power states. Moreover maintenance activities and the testing of trip effectiveness necessarily require that the bypassing of individual trip parameters is possible. Bypassing parts of the protection system on occasion is thus unavoidable. It is essential therefore that when parts of the APS are bypassed, it is done in a controlled and self-consistent fashion under circumstances which are well understood, are safe, and have been planned for in advance.

There are a number of anticipated bypasses of the automatic protection system. These are listed in the safety documentation and their use anticipated in the safety case. All AEA reactors rely upon two types of bypass - "conditional" and "unconditional". A conditional bypass requires that interlocks exist so that if the range of conditions for which the bypass is acceptable is exceeded the bypass is automatically removed. For unconditional bypasses, reliance is placed upon following strict operating instructions as to the order of operations and as to what is allowable. Such administrative control can be reinforced by the use of keys either applied directly to the protection circuits, or indirectly for allowing access to relay cubicles to control the alteration of plant state protection; systems involving keys are widely used in the Authority.

Interlocks are designed into plant to give added assurance of safety, and to promote troublefree operation. They achieve this by ensuring that operations are carried out only in an approved sequence or only when an approved set of conditions have been satisfied. Interlocks can be created using computer software, by electrical circuits and by mechanical means. All three types are used in Authority reactors. Interlocks can be associated with potential bypasses to ensure unsafe combinations of bypassing are excluded.

Given sufficient time, appropriate tools and the will, any set of reactor safeguards can be overcome. The aim must be to match the difficulty of wrong action (whether erroneous or wilful) with the importance for safety of its avoidance. The controls proceed, in increasing severity, from simple strict instructions, through the use of keys under single control, to keys with multiple control and spatial separation, to the use of interlocks and strong physical constraints.

At PFR, SGHWR and DIDO/PLUTO studies of the protection systems have been carried out by site staff and by SRD staff particularly concerned with reliability to see whether controls were appropriately constraining. Not only the automatic protection system itself but also the systems with which it is involved - shutdown systems, cooling circuits and guaranteed supplies were scrutinized. It is concluded that the current arrangements are generally satisfactory, with the stringent administrative controls backed up by a variety of physical controls that could not be readily over-ridden, as was the case at Chernobyl. Nevertheless for each reactor the review has identified a number of specific, though relatively minor, areas where consideration is being given to strengthening administrative control or providing additional physical protection. Obviously the number of existing bypasses should be kept to a minimum and should be reviewed regularly. Consideration is being given to making the regular justification of each bypass a formal obligation.

Two further concerns relevant to this topic are the clarity of the operational instructions, and display to operators of bypass states. Ambiguous instructions and uninformative displays can lead to operator error. Operational safety documentation, while broadly satisfactory is kept under continuous review and further active consideration is being given to bring the documentation into greater conformity with contemporary quality assurance systems. The ease of availability to operators of information on bypass states is also being examined, although the current situation is not considered unsatisfactory.

Even when comprehensive operator aids and constraints are present to make inappropriate action very difficult in a well designed plant, the safety of the reactor depends on the operators - their integrity, their attitudes and their knowledge. Selection and training of operations staff is addressed in the next section.

8. Operator Training

The Chernobyl accident involved violations of the operating rules by operators and by-passing various trips despite their importance for safety. These rather cavalier actions seem to imply a casual attitude to operational procedures and a belief that the actions would not matter - either from woeful ignorance or misplaced confidence. These conclusions have led the Russians to undertake improvements in the training of RBMK operating staff.

In the Authority, the standard of training and authorisation applied to the staff operating reactors is considered to be high. Because of the variety of the Authority's reactors, there is no unified training scheme for operators, but each site management ensures that the operators for its own reactors are properly trained.

Whatever titles they are actually given, each reactor needs and has Responsible Officers. These include the Plant Manager, his deputies, and professional engineers and technologists who are nominated as holding responsibility for the surveillance of specific areas of operating plant and the provision of technical and scientific support in respect of safe and efficient operation. These officers are selected on the basis of academic qualifications plus extensive training and working experience in their field plus their managerial capabilities. The Responsible Officers who supervise shift operations demonstrate that they have achieved the required standards before undertaking the duties of their posts and their official appointment and authorisation is implemented formally.

For each post, one may identify four stages: selection, pre-authorisation training, authorisation and post-authorisation training. Prior to selection several years of relevant experience are generally required. Criteria for selection have already been mentioned. The pre-authorisation training period involves (i) on-the-job familiarisation, (ii) instruction by professional staff, (iii) attendance at specified formal courses, (iv) instruction on the Reactor Simulator. For authorisation, a candidate will be interviewed by the Authorisation Panel of specified membership who will explore his general capabilities and specific knowledge and familiarity with the plant and its operations. Their conclusion will depend on the interview and continuous assessment reports of the pre-authorisation period. Once authorised, the responsible officers are subjected to a continuing training and refresher programme. As with all Authority staff a formal annual review of performance and suitability for the job is carried out under the Staff Reporting System.

The system just described is actually that for Winfrith. The other sites have analogous schemes differing in detail and nomenclature but not in essentials.

Since Chernobyl the operator training and authorisation procedures have been reviewed by the site managements and are kept under review by SRD. The broad conclusion is that the current arrangements work reasonably well. This however is an area where there is always room for improvement and vigilance against complacency is essential.

Two topics, currently under consideration, are concerned with increasing formality and written records to give greater quality assurance. These are (i) a formal system for scheduling and recording the training activities for each member of the reactor staff, and (ii) a formal system of positive endorsement annually as to the fitness of each officer for continued responsibility in his existing post.

The Winfrith simulator has been updated and is in regular use. The PFR simulator now appears rather dated - corresponding to computer technology of the early 1970s. A new simulator will be installed there during 1987, which will be realistic over a larger range of operations including many improbable malfunctions. It should thus provide good training and refresher experience for operators.

9. Experiments in Reactors

The Chernobyl accident occurred during an experiment which clearly distracted the focus of attention away from the safety of this civil power reactor. A number of extra staff were present; similar experiments in earlier years had not been totally successful. The conflicting requirements of the experiment and the grid proved detrimental. The grid request to delay the experiment led to the xenon poisoning build-up which contributed to the additional difficulty of controlling this reactor, and the desire to keep the reactor running led to the final fatal violation by the operators viz the disabling of the turbine generator induced reactor trip. The Russians have stated that the experiment was badly planned and its safety case was inadequate and not properly reviewed.

How does this compare with Authority practice and experience? The most important point is that the Authority is a research and development organisation which has been running research experimental and prototype reactors for many years. This means that a dominant strand in its reactor programme has been the carrying out of experiments and tests and inserting experimental rigs into its reactors. A large number of experiments have been successfully and safely carried out. This is not surprising because experiments are taken seriously and the systems for running the reactors are tailored to cope with a variety of experiments.

The situation at PFR will illustrate this point (the arrangements for SGHWR and the MTRs are broadly similar). In the design of PFR, the need to perform experiments has been taken into account. To ensure safety every experiment is dealt with by a formal clearance procedure. Each must be planned in detail and thorough safety documentation must be produced. The most significant experiments cannot be initiated without endorsement by the main PFR Safety Working Party (SWP). Lesser experiments are considered by an Experiments Sub Committee or the Senior Operator of the reactor depending on the characteristics of the particular experiment. Where the experiments are complex a senior member of the PFR operating team is appointed as the Test Controller to provide additional professional oversight during the test period. However the PFR Shift Manager is at all times responsible for the safe operation of the plant and is empowered to call a halt to any experiment if he is unhappy about the safety of the plant.

Where experimental rigs are loaded into the reactor, and this is commonly the case in MTRs, the safety standards for these experiments must comply with a series of stringent basic safety principles which ensure that any fault internal to the rigs themselves will be contained and thus not hazard the reactor. In addition major fissile rigs are assessed by PRA against criteria to ensure the risk to the operating staff and public is acceptably low. The scrutiny of experiments for the MTRs is carried out by a separate Reactor Experiments Safety Committee.

The Authority's detailed procedures and successful experience with experiments is in marked contrast with the events at Chernobyl.

10. Fire

One aspect of the Chernobyl accident which has added to the public's fear of nuclear power has been the widespread dispersal of fission products throughout Europe; necessitating foodstuff control as far away as Anglesey. An important contributor to this dispersal was the intense fire in the reactor core. This maintained a hot plume of radioactive gas rising for 10 days high in the atmosphere and then being transported by the winds to other countries until the dropping of materials from the air onto the reactor smothered the fire. More locally the explosion cast hot debris from the reactor onto neighbouring plant, causing according to the Russians over 30 fires, and hazarding the other three reactors on the site. It was in valiantly putting out these fires that the main loss of life seems to have occurred from the high radiation doses in the vicinity.

This raises two issues - the presence of combustible materials in the reactor complex, and the arrangements for preventing and combatting fire. What is the situation for Authority reactors?

(i) Combustible Materials

(a) SGHWR

This reactor has a liquid moderator (heavy water); there are no conventionally combustible materials in the core region. Above 1400°C the zircaloy/water corrosion reaction is exothermic, oxidizing the clad and pressure tubes and generating hydrogen which could ignite. The integrity of fuel cladding is an important element in the multi-barrier containment of the fission products and consequently channel power is limited so that a clad temperature of 1200°C is not exceeded even for the worst design basis loss-of-coolant accident - the so-called LOCASTAG.

(b) PFR

The coolant in PFR is sodium; if hot sodium comes into contact with sufficient air it catches fire. The combustion product, sodium hydroxide, would be a major hazard to reactor staff inside a containment containing a sodium fire. If released to atmosphere the vapour cloud would probably convert to sodium carbonate particles which would constitute a nuisance but not a major hazard.

Because of the presence of sodium, PFR incorporates design features to prevent sodium fires. Wherever possible close secondary containment with detection equipment is used to prevent leakage of hot sodium. For instance, the reactor vessel has a containment vessel around it. However, for some equipment this is not possible. Such is the case for the secondary sodium pipework, steam generator vessels, and secondary sodium pumps. These are contained in cells and any sodium leakage would fall through perforated plates into limited volumes where any fire would be rapidly extinguished passively by air exhaustion (as tests show). Additionally there are pressurised fire extinguishers containing graphex which very efficiently puts out local fires, and two graphex fire tenders to deal with larger fires (these are backed up by a large store of graphex).

The PFR fire precautions were formally reviewed about a year ago and found to be very satisfactory apart from the operation of an installed automatic electrical fire extinguisher system which had been modified. This automatic system has been changed to make it satisfactory.

(c) MTRs

The moderator is heavy water and the use of combustible materials within the reactor containment is carefully controlled. At sufficiently high temperatures ($\sim$ 1000K) the metallic uranium fuel would burn in air. The fuel is clad and the fuel elements are immersed in heavy water within the reactor tank. A severe and highly improbable accident involving major loss of coolant and air ingress would be required before fuel burning became possible. The graphite reflector is contained in a steel secondary containment tank. As this graphite is not intimately interspersed with fuel (as in RBMK) and is at a low temperature, the potential for graphite exacerbating an accident is low. Note that the potential peak temperature if the Wigner stored energy were released in either DIDO or PLUTO is only about 250°C.

Generally, fuel oil for the diesels which are located outside the reactor buildings represents a potential fire hazard as does hydrogen used in the alternators. Fire fighting and prevention measures have been instituted to ensure that the probability of a conventional fire hazarding the reactor is extremely small. These hazards have been considered in the safety documentation.

(ii) Fire Fighting

At each site there are full time, dedicated fire officers - properly equipped and trained to fire fighting standards. There are automatic fire detector systems in all the risk areas. In the event of a fire the local alarm sounds and a location alarm operates in the fire brigade control room. The fire brigade which is on duty at all times is therefore alerted immediately. As a precaution they patrol the site and in particular the risk areas. The fire brigade is regularly exercised, and also takes part in the emergency exercises which each site holds (see Section 11).

11. Emergency Planning

Authority sites have always had emergency procedures to deal with any accident which might occur, including a release of radioactivity. In the light of the emergency operations in the wake of the Chernobyl accident, it is natural to re-examine the UKAEA's own emergency plans. This is done below under three headings (i) On-site plans, (ii) Off-site plans, (iii) Public confidence.

(i) On-site plans

The details of the emergency plans for each site are contained in the site Emergency Handbook, and in associated documentation for the use of senior members of the site Emergency Organisation. The plans have been reviewed by SRD against criteria based on the NII's "Emergency plans for civil nuclear installations" and the IAEA's safety guide for emergencies at nuclear power plants. The plans for each site are structured differently but have been found to be adequately detailed and to contain all the broad elements which should be present in an emergency plan for a nuclear site. An industry-wide committee has also been set up under Mr. R. R. Matthews to consider Emergency plans post-Chernobyl. In detail there is considerable diversity between the different organisations and some convergence of approach is to be expected. Emergency exercises are regularly held with external participants and observers and analyses carried out afterwards.

(ii) Off-site plans

The off-site component of each plan is aimed at providing appropriate countermeasures in the event of an off-site release - iodate tablets, sheltering, evacuation etc. Each site is in liaison with its local community and off-site plans are made in consultation with the local agencies (local authorities, police, fire, health etc.) and included in the Emergency Handbook. Each site has a Local Liaison Committee which enables local representatives to raise points of concern and enable the site to inform and explain any relevant developments.

The scale of Chernobyl suggests that each site needs a fall-back operations centre off-site, 10 miles or so distant from site. Dounreay has Tollemache House in Thurso, Winfrith and Harwell are making appropriate arrangements.

(iii) Public Confidence

For a number of years a document summarizing emergency plans at a site has been available in local County reference libraries. Recently, the UKAEA decided to publish its Emergency Handbooks and those for Dounreay, Harwell and Winfrith have now been published. For emergency purposes the AEA sites at Windscale and Springfields are subsidiary to their BNFL neighbours. The winning back of public confidence in nuclear power is clearly of the greatest importance. Dissemination of information about emergency plans is a small element in this.

It is proper that our emergency plans are regularly reviewed and updated in the light of experience and improved technology. It may well be that we shall need in future revisions to make our Emergency Plans more consistent within a common approach - not especially for technical reasons but because the diversity does not inspire public confidence that our emergency plans are as good as can reasonably be achieved.

12. Conclusions

The safety of the main Authority Reactors has been reviewed in the light of Chernobyl. Conclusions which may be drawn from this review are

i) The arrangements for controlling the operation of the reactors provide an environment conducive to safety.

ii) The reactors are neutronically stable. SGHWR and the MTRs have a negative void coefficient, as does PFR except in a central region.

iii) The reactor shutdown systems are adequately fast, much faster than the RBMK; they are also much less clumsy to operate.

iv) There is necessarily the potential for the automatic protection systems and related safety systems to be bypassed in part. In most cases there is a good match between the strength of the constraints on operators and safety requirements. Some scope for improvement was identified.

v) The system of operator training is soundly based; increased formality and more written records may give greater assurance.

vi) The Authority has great experience in carrying out experiments in reactors; reactors and their operating procedures are designed with experiments in mind.

vii) While all the reactors have materials combustible at very high temperatures, only the sodium at PFR constitutes a conventional fire hazard; the fire precautions are correspondingly thorough.

viii) The existing AEA emergency plans are soundly based; as with all contingency plans there is always room for improvement.

APPENDIX 1

The Chernobyl reactor involved in the accident is a large civil power producing reactor of 950 MW(e). Its fuel is lightly enriched UO_2 clad in zirconium alloy. It is cooled by boiling (ordinary) water passing through banks of vertical pressure tubes. Moderation is provided by a matrix of graphite blocks which because of neutron and gamma heating are heated to quite high temperatures, up to 700°C. The reactor has a positive void coefficient of reactivity; at normal full power operation this is more than compensated for by the negative fuel coefficient, giving a net negative power coefficient. However below about 20% of full power the power coefficient becomes positive and the reactor becomes unstable. For this reason operating below 20% power was administratively restricted. The reactor control and shutdown rods consist of over 200 solid absorber rods. Of these 24 are dedicated to emergency protection and are motored in rather slowly with full insertion taking 15-20 seconds. The control system is arranged to operate over several different power ranges.

The accident occurred around one o'clock in the morning of 26th April. It occurred, ironically, during an experiment aimed to improve the safety of the plant. The objective of the experiment was to see whether the mechanical inertia in a turbine generator could be used to supply electricity for important station systems, including safety systems, immediately following a grid power failure. The experiment had been attempted twice before, and it involved the presence of additional engineers primarily concerned with the experiment rather than reactor operation. The planned experiment required the reactor to be at about 25% full power with one of its turbine generators shut down. The operators had started to reduce power 24 hours earlier since the experiment was originally scheduled for the previous afternoon. By 2 p.m. on the 25th April conditions for the experiment were almost established and the emergency core cooling system (ECCS) was disconnected. However the grid controller requested continued grid supply until 11 p.m. - this was complied with, though with the ECCS disconnected, contrary to operating rules (this violation did not have any significant effect on the accident, but may be a reflection of operators' attitudes towards plant safety).

After 11 p.m. the operators started again to move to test conditions, which involved changing to a different reactor control regime. The operator made a major error in failing to reset the set point of the automatic regulation system and was then unable to control properly the reactor power which then dipped to below 30 MW(t). The delay in the experiment led to a growth in the levels of the fission product ^{135}Xe, a neutron poison, and this made it very difficult for the operator to raise power because of the small reactivity margin available. By 1 a.m. on the 26th April the power was stabilized at 200 MW(t) - well below the power level initially proposed for the experiment, and in the range of the positive power coefficient (though this fact may not have been realized). At this power level, the coolant voidage was much reduced, and the coolant flow rate higher than anticipated - in a regime normally prohibited to avoid cavitation and vibration in the event of a pump trip. The primary coolant system was very close to boiling, though little steam was being generated. To avoid the reactor tripping in this condition, the operators over-rode some of the "trip" signals in unsuitable circumstances (another violation of operating rules). The low steam fraction in the core coupled with xenon poisoning necessitated having very few control rods within the core if power was to be maintained at 200 MW(t). In fact calculations have shown that only something like seven control rods were in the core, less than half the design "safe" minimum (15) and a quarter of the minimum number of 30 given in the operating instructions. The operator noticed at 1.22 a.m. that the reactivity margin was less than that of 15 control rods; he should have then tripped the reactor according to operating instructions but did not. The experiment required that the turbine generators should be tripped and to allow this without tripping the reactor, the protection was disengaged. This was not necessary and was a further violation of the operating rules.

At 1.23 a.m. the experiment was initiated, the pumps and the turbine generators started to run down, and the steam fraction rose sharply. The positive power coefficient caused a power excursion, and although the shift manager attempted a manual scram, the control rods were too far out to be effective on the required timescale. A prompt critical excursion then occurred leading to severe fuel damage, fuel channel disruption, and an explosion - a steam explosion from the contact of dispersed fuel with water. A hydrogen explosion probably followed. The explosions lifted the reactor roof off, releasing the fission products, and over 30 fires were started due to thermal radiation from the exposed core, and damage from the hot core fragments. The fires were put out by 5 a.m. but only as a result of heroic efforts by the firemen, many of whom have since died as a result of radiation exposure.

The reactor itself continued over the next 10 days to put out a hot plume of radioactive gas rising high into the atmosphere; this discharge was terminated when materials were dropped from the air onto the reactor, smothering the fire. The emergency planning must have been highly thought through for over 100,000 people were evacuated from the 30 km radius control zone established about Chernobyl in a remarkably short time.

APPENDIX 2

RESEARCH AND EXPERIMENTAL REACTORS

Name	Date of Start-up	Moderator/ Coolant	Fuel	Peak Neutron Flux n/cm² sec	Maximum Heat Output	Status
GLEEP (Harwell)	1947	Graphite/ air	Natural Uranium	10^9	50 kW (3kW norm)	Operating
DIDO (Harwell)	1956	Heavy water	Highly enriched uranium	about 2.5×10^{14}	26 MW	Operating
PLUTO (Harwell)	1957	Heavy water	Highly enriched uranium	about 2.5×10^{14}	26 MW	Operating
NESTOR (Winfrith)	1961	Water & graphite/ water	Highly enriched uranium	10^{11}	30 kW	Operating
DIMPLE (Winfrith)	1962	Water/ none	Uranium or plutonium	3×10^8	Less than 100 W	
ZEBRA (Winfrith)	1962	None/air	Uranium and/or plutonium	about 10^{10}	1 kW	Out of commission
HECTOR (Winfrith)	1963	Graphite/ CO_2 for heating	Highly enriched uranium-aluminium alloy	3×10^8	100W	Out of commission

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POWER REACTORS

Name	Date of start-up	Moderator	Coolant	Fuel	Net power per reactor
SGHWR (Winfrith)	1967	Heavy water	Water	Enriched UO_2	314 MW(h) 100 MW(e)
PFR (Dounreay)	1974	None	Liquid sodium	PuO_2/UO_2	600 MW(h) 250 MW(e)

APPENDIX 3

GLEEP

GLEEP, situated at Harwell, is a natural uranium fuelled, graphite moderated, air cooled reactor. It comprises of a lattice of 505 tons of graphite incorporating horizontal channels in which the fuel (30.7 tons of natural uranium rods clad in aluminium) is loaded. The reactor normally generates only 3 kw at a thermal flux of 10^9 n/cm² s at 28°C. It is air cooled and inherently safe by virtue of its low multiplication factor and high negative temperature coefficient. The Wigner energy level at present is approximately 1 calorie/gm and this level of stored energy corresponds to a potential temperature rise of 6°C. As the combustion temperature of graphite in air is above 700°C there is no real possibility of a fire being produced within the core. Installed fire fighting equipment includes bottled argon which can purge the shield and fuel channels, and as a last resort water can be pumped into the reactor. Combustible materials are not allowed to be taken into the reactor shield and gaseous flammable products introduced through the inlet plenum would not be ignited because of the low operating temperature of the fuel (28°C).

For GLEEP the void coefficient and overall temperature coefficient of reactivity are negative. Moreover GLEEP cannot become prompt critical - under its most reactive conditions its excess reactivity cannot exceed 0.49% (compared with the minimum adddition to a critical core of 0.7% needed to achieve prompt criticality). Steady state operation at 3 kw (its normal level) results in a whole core inventory of I_{131} of 80 curies.

The safety document for GLEEP concludes that even if the maximum excess reactivity available was inserted as a step input then a reactor trip from the maximum power trip of 50 kw (although in practice GLEEP is never operated above 3 kw nowadays) would result only in a 0.6° kw increase in fuel temperature.

Although inconceivable the safety case now postulates that one fuel element has a serious clad failure at 50 kw and the consequence is calculated to be 1.3 m Rem effective dose equivalent to a member of the public most at risk (2.5% of the ERL for members of the public). The max I equivalent inventory is approximately 160 curies.

Note: The NESTOR safety document considers an aircraft crash and if a similar worst case analysis was applied to GLEEP it would show a maximum release of 155 curies at a frequency of 2 x 10^{-7} per year.

NESTOR

A 30 kw ARGONAUT type thermal reactor. It is graphite reflected and light water moderated. Fuelled with 80% enriched uranium in plate type elements, clad with aluminium. The total uranium mass is 4.7 Kg and the total radionuclide inventory is less than 1000 curies I equivalent.

Deterministic and probabilistic calculations discussed in the reactor safety document show that no conceivable reactor accident sequence can lead to an uncontrolled release of radioactivity. Protection against external building fires includes the provision of fire detectors and for the introduction of CO_2 into the core vault together with conventional light water firefighting apparatus. In view of these precautions and safeguards and the concrete shielding around the reactor, it has been concluded that the graphite could not reach the high temperature required for ignition ($\sim$700°C).

The safety document for NESTOR concludes that only a severe aircraft crash on the reactor would have the potential to release sufficient radioactivity to affect the public. A frequency of 10^{-6} per year under absolute worst case analysis could release at most 700 curies of Iodine (the total core inventory).

DIMPLE

Dimple is a flexible zero-power water moderated reactor located at Winfrith. It was used ten to twenty years ago to validate water reactor calculational methods and has now been refurbished to undertake a similar role in the criticality field and to pursue an experimental programme relevant to the manufacture, transport, storage and reprocessing of reactor fuel.

The risks to personnel and to plant which could arise during normal and under accident conditions have been examined by a Safety Document. No hazards have been identified which endanger personnel and the analysis demonstrates that no accident sequence can lead to a major off-site hazard. In fact the worst transient analysed would result in a whole core Iodine inventory of less than 5 curies although it is not conceivable that this could be released in practice.

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CHERNOBYL: ONE YEAR AFTER
IMPACT ON THE NUCLEAR INDUSTRY

F R ALLEN

SRD, CULCHETH

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CHERNOBYL: ONE YEAR AFTER

IMPACT ON THE UK NUCLEAR INDUSTRY

Introduction

1. The Chernobyl accident was an extremely severe accident to a civil nuclear station - the most severe accident which had occurred in the history of nuclear energy. Clearly, its impact on the nuclear industry could be very great; in fact, prior to the accident, many would, I expect, have conjectured that such an event would move public opinion so severely and irrevocably against nuclear power that the industry would not have survived.

2. As yet, the full implications are still working themselves through. However, we can already see some of these effects and the purpose of this paper is to review the current situation, within "the UK industry" which will be interpreted widely as meaning all those bodies interested in promoting nuclear power in the UK.

The Immediate Response of the UK Industry

3. Immediately following the accident, Lord Marshall re-convened the Industry Task Force which drew together representatives of the industry, including, from the UKAEA, Dr Hicks, Dr Gittus (and Mr Collier). This provided the technical backing for a robust response to be made to the situation. Advice was provided to the Secretary of State and briefing material for staff (at all levels) who had contact with the media or the public. The Task Force also provided information which enabled industry staff, themselves, to be better informed as to what had occurred and what the implications might be.

4. The UKAEA played a key role in this, both by the participation of senior staff in the Task Force discussions and by providing technical assessment and briefing. A significant achievement was the detailed analysis of the accident by Dr Gittus, et al, published as NOR 4200(1), This has become the definitive UK analysis of the accident.

5. Initial reponses to the accident took a variety of forms. The design of the RBMK was such that no close parallels to the design of UK reactors could be discerned. However, atention was focussed on operator training and on emergency planning. The latter concern led to the publishing by the CEGB, UKAEA and others of emergency plans for reactor installations. Also whilst the long term review of the Magnox stations had already been going on for some time, attention was focussed on it.

6. Public concern was also focussed on the apparently haphazard and uneven way in which radiological intervention levels varied throughout Europe. This has led to efforts to harmonise these which are still going on.

The Response of the Public

7. The impact on the UK industry cannot be divorced from movements in public opinion. Clearly, public opinion has moved against nuclear power since the accident. However, this has not precipitated any hasty decisions so far. The very great differences in design between the RBMK and UK reactors seem to have been appreciated by the public. So, too, does the assertion that the 'safety culture' in the UK would prevent the chain of operator faults which occurred at Chernobyl. Thus far, public opinion seems not to have moved disastrously against nuclear power. However, clearly the outcome of the General Election could have a major impact on this in the future.

8. An interesting initiative has been the setting up by the Watt Committee on Energy of a Nuclear Safety Sub-Group to look at the implications of Chernobyl for the UK. The objective here is to produce an informed but impartial view to be directed towards technical people outside the industry. This body has carried out studies over the last six months and has come up with some interesting, if predictable, recommendations concerned with increased emphasis on operators and operator training, research into relevant areas, international co-operation and staffing levels at NII.

Emergent Issues

9. Various issues have emerged as being of relevance to nuclear power in the UK. These include the following.

Human Error and Operator Training

10. As noted above, this is an area which is relevant to all reactor systems to a greater or lesser degree. Following the accident, existing programmes in this area were given additional impetus - for example the work under the GNSR programme on human error was expanded to some extent. However, this was an area which had been highlighted already by the TMI accident and so its importance had already been recognised, even before the Chernobyl accident.

Emergency Preparedness

11. As noted above, the CEGB, UKAEA and other operators have now made public their emergency plans. In addition, there has been increased emphasis in the last 12 months on emergency exercises.

12. The control of offsite events is clearly an area of political as well as technical and logistic difficulty. It now seems likely that the Government Technical Adviser will take charge of an incident very rapidly (in the past this might not have occurred until 24 hours had elapsed). Also, his presence

may be augmented by a member of the government soon afterwards. This will bring the control of the incident more directly under the civil authority.

Containment

13. The RBMK had what is acknowledged to be a weak containment system. The proposed PWR at Sizewell will, of course, have a strong containment, but the existing Magnox and AGR reactors have no containment other than the primary pressure boundary. This is justified because

- The coolant is single phase
- The coolant volume is large
- Loss of coolant occurs relatively slowly and without any abrupt change in heat transfer capability
- The coolant is normally essentially inactive
- Catastrophic failure of the pressure boundary is so unlikely that the risk involved is negligible.

The Proposed PWR at Sizewell

14. The PWR proposed for Sizewell 'B' has been seen to have a number of important safety features when compared to the RBMK including

- Negative void coefficient
- Negative power coefficient
- Very reliable ECCS
- Strong containment
- Lack of dependence on operator action

The effect of the Chernobyl accident on the debate surrounding the Sizewell proposals has not, therefore, been affected by technical issues raised by the Chernobyl accident. Indeed, the recent Greenpeace report "Chernobyl UK" attacks the UK gas reactors not the proposed PWR.

"Chernobyl UK" - The Greenpeace Report

15. Within the last month, to coincide with the anniversary of the accident, Greenpeace produced a report "Chernobyl UK" (2) which sought to show that an accident like that at Chernobyl could occur in a UK gas reactor. The report focussed on the positive moderator coefficient, common to both UK gas reactors and the RBMK. However, the conclusions drawn by Greenpeace are wrong because

- The moderator coefficient played no significant role in the Chernobyl transient.
- The Chernobyl transient was caused by the water cooling which is absent from the UK gas reactors.
- The Chernobyl transient was too fast for the control rods to stop.
- In the UK gas reactors, any power changes due to the moderator coefficient are slow and easily stopped by the control rods.
- The control rods in the UK gas reactors are very reliable and are backed by secondary devices.

16. The Greenpeace document, therefore, perhaps not surprisingly, represents a totally unfounded attack in which the basic safety characteristics of the UK gas reactors are grossly misrepresented.

Political Issues

17. Following Chernobyl, both Labour and Liberal/SDP parties adopted policies which are against nuclear power. This aspect, of course, is dominated by the coming General Election.

18. Clearly the outcome of the election could have a significant effect on the future development of the UK industry. This is potentially the most important impact of all and the one with, at the moment, the greatest uncertainty.

Summary

19. Initially the UK industry responded rapidly and produced a robust defence of nuclear power in the UK. Whilst public opinion moved against nuclear power, this move has not (so far) been extreme or precipitated any major changes of policy. However, clearly there is a major task in re-building public confidence in which the UKAEA must play its part well.

20. Various technical issues have emerged which are the subject of programmes of work which, by an large, represent slight changes in direction or resource level for existing activities. A significant issue is emergency response and here changes in the role of the government representatives are taking place.

21. The greatest potential impact comes from the attitudes of the various political parties and so hinges on the result of the election on 11th June. It is here that, currently, the greatest uncertainty arises.

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Chernobyl: One Year After

The Impact on International Nuclear Safety Issues

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Introduction

Whilst it had been understood for many years that the long range transportation of radionuclides following a severe reactor accident would give rise to effects beyond transnational boundaries, the full appreciation of its impact was not crystallised until the accident at Chernobyl did just that. This served to emphasize the interdependence of neighbouring countries on the way in which they conduct their internal affairs relating to the safe design, construction and operation of nuclear power plants.

Nuclear safety and radiological protection has a long history of international collaboration. Many organisations have established fora for discussions on these topics and there are a number of multi-national and bi-national arrangements which also play a role. All of these activities have been affected in one way or another by the need first to respond to the accident and then to reconsider priorities and work programmes in the light of further discussions. This paper reviews the general overseas position post-Chernobyl. To do this we first briefly review the chain of events following the accident, the mutual assistance which took place and the evaluation of the spread of fallout, particularly through Europe. Then we outline the activities in the major international agencies. The IAEA* have been at the centre of the activities and this receives the most attention, however, the work of the NEA* of the OECD* and the CEC* is important and is also discussed. Finally, some aspects of multi-national and bi-national collaboration are outlined. The next section considers the overseas position from the point of view of the now established consensus on the topic areas of greatest importance now that time for reflection has allowed the lessons from Chernobyl to be assessed. The final section addresses the prospects for international collaboration, changing world views of nuclear safety issues and the UKAEA's role in this new environment.

Initial Collaboration and Assistance

When the fallout from the accident was first detected in Sweden the nature and scale of the accident began to become apparent. Any accident giving rise to such large amounts of deposited activity would of necessity, have been very serious indeed. International collaboration began immediately through personal

* IAEA - International Atomic Energy Authority
  NEA - Nuclear Energy Agency
  OECD - Organisation for Economic Co-operation and Development
  CEC - Community of European Communities

contacts established through previous joint work. The contacts made through the NEA proved very effective in this and in the immediate aftermath of the accident, data on fallout in various countries was channelled to UK national bodies through the AEA. During the first few days of the accident it was vitally important to have as authoritative an understanding of what might have been happening at the site as possible. To many countries the need to establish whether any emergency response would be required was very important, whilst for others, like the UK, the most pressing need was to advise the Foreign Office on potential risks to UK citizens abroad or those planning visits to the area. It was necessary for example, to advise the FCO on whether further major releases were likely due to the core melting through the concrete of the reactor cavity. The advice given was that it was not likely and this was based not only on our own analysis but also on discussions with colleagues in the USA, Germany and Sweden. This informal exchange and collaboration continued for the first few weeks after the accident but was eventually transplanted by the activities of the major international agencies. The activities of the major international agencies in the immediate aftermath of the accident is now a matter of history. To give an overall impression of the events of last summer we have drawn together a chronological table of the more important activities - these are shown in Table 1.

In order to give some structure to this paper we now discuss briefly the response of the major international agencies to the accident individually.

<u>The International Atomic Energy Agency (IAEA)</u>

The IAEA dominates the entries in Table 1, this reflects the prominence given to this agency as the principal focus for international discussions. The reasons for this include:

a. The IAEA is the only international agency to which both East and West contribute - it was therefore a natural choice for the Soviet Union for a channel of communication.

b. The IAEA have a long record of international collaboration in the relevant fields - there was an established hierarchy and administration available to deal with the problem.

c. The IAEA offered access not only to Eastern block countries but also developing nations and many aspects of the "international problem" concerning those countries too.

d. The IAEA was 'endorsed' as the preferred agency by the Tokyo Summit in May.

In the UK the Department of Energy and the Foreign Office, following item (d), indicated that the IAEA was the preferred route for UK input to the international scene. The authority was heavily involved in supporting HM Government officials in many of the IAEA activities which followed.

Table 1 indicates the order of events of Agency activities. The most important of these were:

i) The establishment and signing of the conventions on early warning and mutual assistance

ii) The "Information Meeting" in August at which the Soviet Union gave a very full account of both the causes of the accident and its effects.

iii) The General Conference of the IAEA at which national statements concerning the internationalisation of nuclear safety were given at Ministerial level.

iv) The Expert Meeting held in November at which the supplementary programme of the IAEA was reviewed and recommendations made to the Board of Governors meeting in December.

The two conventions were agreed in a very short timescale; the texts are given in Appendix I. The capability of the Agency to implement these conventions was given top priority in the new work under the supplementary programme proposals.

The Information meeting provided a forum for the Soviet Union to present detailed analysis of the events leading to the accident and of their recovery efforts afterwards. The very frank admissions by Acedemician Legasov who led the Soviet delegation have been frequently quoted; particularly his admission that it was a combination of design and administrative errors as well as errors and malpractice on the part of the operators which established the conditions which made such an accident possible. The INSAG* acted as "referees" at this meeting and provided a definitive account of the accident - and a general endorsement of the Soviet Union's analysis. This was subsequently published as IAEA Safety Series Document 1.

The General Conference provided a forum for all member nations of the IAEA to express their views. Naturally there was a very wide range - Austria and Denmark on the one hand were adamantly opposed to the further development of nuclear power, especially in their neighbours where they had little or no control and

*INSAG - International Safety Advisory Group was set up to advise the Director General of the IAEA on safety matters. The UK representative is Dr B Edmondson, CEGB.

rather little confidence either judging by the speeches. On the other hand, France and Japan reiterated their confidence in and dependence on nuclear power for their economic and social development. The Secretary of State for Energy gave a speech very supportive of nuclear power but calling for the development of mechanisms for "reviewing the regulators". This, the "Walker Initiative" is outlined in more detail below.

The Expert Meeting was called to review and endorse a 'supplementary' programme of work proposed by the Secretariat to reflect the general consensus on the need to strengthen various activities under the aegis of the Agency. Even though the financial cost of this programme was apparently quite modest ($2.03M) because of the sensitivity of some issues to some countries (for example, the follow up investigation on the people affected by radiation were considered to be the Soviet's own responsibility and they were unwilling to give the Agency specific responsibilities in this area), and because the cost o. supporting Agency activities has to be borne by the contributing national bodies, the deliberations were detailed and often difficult. A work programme was agreed, the principle topics are listed in Appendix 2.

Subsequently the Board of Governors met and gave approval to the supplementary programme with various caveats indicating the recommendation that there should not be a proliferation of new committees or hierarchy within the Agency. Plans for the implementation of the programme are now being developed by the secretariat of the Agency.

The Nuclear Energy Agency of the OECD

Following the Chernobyl accident great benefit was derived from the network of contracts established over many years of collaboration within the framework of the Committee on the Safety of Nuclear Installations (CSNI) and the Committee on Radiological Protection and Public Health (CRPPH). For the first week or two links set up earlier in various groups reporting to the CSNI provided an efficient channel for rapid and accurate information on radiation levels and radioactivity measurements in Northern Europe.

Early in May an extraordinary meeting of CSNI and CRPPH was held in Paris to pool information on the accident and to discuss future action centred on the NEA. It was decided that the emphasis should be on detailed technical analysis and debate, since IAEA was clearly identified as the channel for exchanges on broader issues. Specifically, a small ad hoc group was set up to prepare a report on Chernobyl for CSNI. The ad hoc group was asked to consider the following aspects:

Information needed from USSR to enable OECD countries to of links as are available to individual member countries. The EEC's R&D programme has not been signficantly altered to account for any post-Chernobyl insights and in that follows the same general line as is being taken by many other countries. Any impact on the general sense of priorities within the Framework R&D programme has yet to become clear as the situation is at present dominated by negotiations over the size of the Framework programme as a whole.

Commentary

The technical discussions within the various international agencies have led to a general consensus as to which areas should receive additional effort in the future as a result of analysis of the Chernobyl accident. It is not surprising that these are consistent with recommendations already made within the authority for programme revisions. In the main, international expert opinion concurs with our view that no new phenomena, or insights into reactor design safety principles were revealed by the accident or the analysis of it which followed. Furthermore, it has been generally agreed that the RBMK design was very unlikely to have received a licence to operate in any Western country. The topics identified in the supplementary programme of the IAEA (Appendix 2) provide an overall list of the areas identified for further work.

Individual countries have made their own responses to the accident. To illustrate the nature of the responses the examples of Switzerland, Sweden and the US are used. No attempt is made here to dtail all of the different countries responses.

Switzerland is perhaps a special case as it is a small country with very high pressure on land utilisation. It does have a nuclear industry and is well endowed with measurement and other emergency response facilities. The principal impact on Switzerland involved the realization that any largescale accident would pose extremely difficult evacuation/relocation problems. For any small country it is not immediately clear where people can be evacuated or relocated to. This was exacerbated for the Swiss by their cultural tradition of a very strong sense of 'belonging to the land'. To Switzerland any accident of this sort either within their boundaries or in their neighbours would involve unacceptable consequences.

Sweden as one of the worst affected countries outside the Soviet block and one with a record of environmentalist action against nuclear power it is not surprising that the Swedish Government is the only one to date which has puublished an overall "Government" view. This is contained in "After Chernobyl - consequences for energy policy, nuclear safety, radiation and environmental protection". It was produced by the Committee on Nuclear Safety and the Environment under the auspices of the

Swedish Ministry of Industry. The document covers a wide range of topics, some of those are illustrated below.

1. It is concluded that no re-assessment of the technical risk assessment of Swedish plant is needed (because of large differences between the RBMK and Swedish reactors).

2. The 'world risk' from nuclear plant is dominated by a few types, particularly those with inadequate containment. Eastern Europe is specifically mentioned and because of the lack of confidence in other countries capabilities the Swedes note the need for emergency preparedness in Sweden for reactor accidents abroad.

3. A range of 'safety enhancing' measures are identified including:

a. Man-machine interface
b. Reactor physics
c. Strength of pressure vessels
d. Hydrogen control
e. Earthquakes
f. Accident Management including fire fighting
g. International collaboration

4. Re-scheduling the close down of Swedish Reactors.

The current situation in Sweden is that nuclear power is to be phased out by 2010. The first response to Chernobyl was to consider accelerating this phase out, perhaps on as short a timescale as two years. The general conclusion is that the geo-political conditions in Sweden preclude an early phase out of nuclear generated electricity. However, an earlier date of 2005 is now proposed but not formally adopted. It is the intent of the Swedish ministry to spend the intervening time developing low impact technologies for use in Sweden, and for export, which would either reduce the need for electricity or assist in generation.

5. Health effects -, no detailed conclusions are drawn but the report does point out that the average dose due to Radon in Sweden is now 2-7 mSv/yr due, in the main, to improved insulation for energy conservation. This is to be compared with less than 1 mSv as the average incremental dose from Chernoybl.

<u>The United States of America</u>

Compared with the response of Sweden, that of the USA is fragmented and there are no clear lines of policy guidance. The following illustrate some of the activities.

1. Relevance of Chernobyl. A US Department of Energy team

have analysed the accident in detail (DOE /NE 0076). Their conclusions are very similar to those of the UK analysts but the paper restricts its treatment to the technical issues.

2. There have been no indications of plans for additional plant closures, although it has been said that the poor climate in the USA has not been improved. The USNRC has now published its review of the accident (NUREG 1250).

3. The US DoE operate the 'N' reactor at Hanford which is dedicated to Plutonium production. Superficially it is similar to RBMK in that it is a pressure tube, graphite moderated boiling water reactor. A special commission considered the continued operation of that plant: it was decided to shut it down for 6 months whilst its operations and safety are reviewed. This is currently underway.

Commentary

Resolution of the technical issues involved in the overseas response to the accident was a very important task. However, within the limitations of resources and practical feasibility, this has essentially been completed. Of perhaps greater importance to the long term 'international climate' on Nuclear Safety has been a series of ramifications from the accident which are less easy to resolve. These issues are illustrated by the following examples.

1. International Regulation and Licensing

One of the immediate responses of the many Governments to the Chernobyl accident was to call for immediate imposition of binding international standards of nuclear safety. Much of the subsequent debate has centred around this issue. In the main the call for such a scheme has come from developed countries which have themselves no nuclear power programme. They feel threatened by their neighbours who have and believe that an international regulatory framework would allow the highest standards of safety (ie. those they themselves would impose) to be imposed upon their neighbours. This is a very difficult issue since on the face of it, such a scheme might be appealing. We would like to believe that our standards are perfectly adequate and should be imposed on others but would not subscribe to the opposite view. Furthermore, the practicalities of the problem are that trying to find mutually acceptable international safety standards is like seeking the Holy Grail. Countries have evolved different practices, have quite different siting problems and even cultural differences which would colour any risk benefit balances. Consequently, all previous attempts at such a system have always ended up with "the highest common set of standards". These are in reality the minimum set to which everyone could agree and are quite ineffective in providing for adequate safety practice. A further difficulty in setting up such a system is the practical one of finding sufficient well trained staff to take on such a task. There are constant calls for "independent reviews" (a good recent example occurs in draft European Parliament Resolutions from the Committee on Energy, Research and Technology - Appendix 3) of nuclear safety but it is doubtful that the expertise exists to implement such reviews.

As a way forward, Mr Walker proposed a scheme whereby the IAEA should update its knowledge of existing inspectorate's operations, promote best practices and perhaps in due course, form an expert team which could be responsible for peer reviews. This is the 'Walker Initiative' and represents a middle line whereby regulatory systems and practices may be reviewed (initially at least on a voluntary basis) rather than attempting to define international regulations. This proposal met with a good deal of support at the IAEA and is likely to form the basis of future Agency activities in this area. The full text of Mr Walker's speech is in Appendix 4.

2. Pooling Nuclear Safety Information

Many of the developing countries see the international activities of the IAEA as a means whereby knowledge acquired by the developed countries can be made available to them. This has led to considerable dissension between the IAEA and what it calls "regional" agencies. Thus, the NEA and the CEC are considered as "clubs" for the rich and powerful and consequently they should not be supported, unless a means could be found to allow the developing countries to access the information being shared. The mood of the IAEA, led by the less developed countries, (in particular Argentina) has been strongly to oppose the operation of these 'regional agencies'. The accident at Chernobyl has highlighted a particular dilemma for the Western countries. On the one hand passing information freely to other countries can only be done at some cost, particularly in the time energy of relatively few experts. The Soviet Union made this point themselves. If this effort is to send information 'one way' then there is a natural reluctance to follow this course. On the other hand it is clear that if other countries are to develop nuclear power then it is in all of our interests to ensure that it is done as safely as possible hence we should make such information available to them.

3. The Role of Developing countries

Coupled to the previous point is the attitude to take towards the aspirations of developing countries for nuclear power. In many such countries the infra structure does not exist to support such an enterprise locally and there would remain a strong dependence on the nuclear vendor. In politically volatile regions this could have serious ramifications. However, through both multi and bi-lateral efforts the evolution of currently under developed countries into nuclear operating states can be envisaged with some confidence that the technology will be properly handled. An example is India, where a homegrown nuclear industry is supported by the education of many specialists overseas. Another example although lower on the development list is Indonesia. Here the Authority has direct experience, along with other UK nuclear interests, in negotiating contracts concerning support for a German supplied research reactor. We have recently been notified that the Indonesians are considering a programme of 5 or 6 nuclear plant and that we (UK limited) could be tendering for contracts. The important points here are not that this is a business opportunity (a fact which should not be ignored, however) but that through the efforts of the IAEA other Western countries and ourselves, the Indonesians are actively seeking support to ensure that any plant they build will be safe. The evaluation of the idea of the need for a "safety culture" in the operation of nuclear plant is perhaps one of the most important lessons drawn from the Chernobyl accident and there is some evidence that it is being taken seriously.

Conclusions

The accident at Chernobyl has had a significant impact on international appreciation of nuclear safety. The potential for actions in one country to seriously affect the health and well being of its neighbours has been clearly underlined. Much of the activity in the principal international agencies has taken place with this in mind. Whilst technical programmes reflecting experts views on what might be done to provide more supportive research have been agreed, it is still too early to predict how the various nations will react to their heightened awareness of being embedded in a world where they are not masters of their own environment.

Table 1

Chronological Order of International Activities Post-Chernobyl

Date		Organisation
26 April	Initiation of accident	
27/28 April	Radiation first detected outside Soviet Union	
5 May	Tokyo Summit	
6 May	Discharge of radioactive materials ceases	
8/9 May	Joint meeting of the CSNI (Committee for the Safety of Nuclear Installations) and the CRPPH of the NEA	NEA
26/27 June	European Council meeting, the Hague, recommending that general contamination tolerance levels be determined on a scientific basis very quickly. This was to use the framework of Article III of the Euratom treaty.	EEC
July/ August	Drafting of International Conventions on mutal assistance and early warning	IAEA
20 August	Commission of the European Community advice to the Council for application of Chapter III of the Euratom Treaty 'Health and Safety'. Including a report of the specialist group on tolerable dose levels in food. ('Article 31' experts).	EEC
25/29 August	Chernobyl Post-Accident Review meeting Vienna	IAEA
24/26 Sept	Meeting of the General Council of the IAEA - Ministerial level contributions on post-Chernobyl IAEA activities.	IAEA
3-7 November	Expert meeting to discuss IAEA supplementary programme	IAEA
8 December	Board of Governors meeting	IAEA

Annex 1

Texts of the International Conventions

CONVENTION ON ASSISTANCE IN THE CASE OF A NUCLEAR ACCIDENT OR RADIOLOGICAL EMERGENCY

THE STATES PARTIES TO THIS CONVENTION,

AWARE that nuclear activities are being carried out in a number of States,

NOTING that comprehensive measures have been and are being taken to ensure a high level of safety in nuclear activities, aimed at preventing nuclear accidents and minimizing the consequences of any such accident, should it occur,

DESIRING to strengthen further international co-operation in the safe development and use of nuclear energy,

CONVINCED of the need for an international framework which will facilitate the prompt provision of assistance in the event of a nuclear accident or radiological emergency to mitigate its consequences,

NOTING the usefulness of bilateral and multilateral arrangements on mutual assistance in this area,

NOTING the activities of the International Atomic Energy Agency in developing guidelines for mutual emergency assistance arrangements in connection with a nuclear accident or radiological emergency,

HAVE AGREED as follows:

Article 1

General provisions

1. The States Parties shall cooperate between themselves and with the International Atomic Energy Agency (hereinafter referred to as the "Agency") in accordance with the provisions of this Convention to facilitate prompt assistance in the event of a nuclear accident or radiological emergency to minimize its consequences and to protect life, property and the environment from the effects of radioactive releases.

2. To facilitate such cooperation States Parties may agree on bilateral or multilateral arrangements or, where appropriate, a combination of these, for preventing or minimizing injury and damage which may result in the event of a nuclear accident or radiological emergency.

3. The States Parties request the Agency, acting within the framework of its Statute, to use its best endeavours in accordance with the provisions of this Convention to promote, facilitate and support the cooperation between States Parties provided for in this Convention.

Article 2

Provision of assistance

1. If a State Party needs assistance in the event of a nuclear accident or radiological emergency, whether or not such accident or emergency originates within its territory, jurisdiction or control, it may call for such assistance from any other State Party, directly or through the Agency, and from the Agency, or, where appropriate, from other international intergovernmental organizations (hereinafter referred to as "international organizations").

2. A State Party requesting assistance shall specify the scope and type of assistance required and where practicable provide the assisting party with such information as may be necessary for that party to determine the extent to which it is able to meet the request. In the event that it is not practicable for the requesting State Party to specify the scope and type of assistance required, the requesting State Party and the assisting party shall, in consultation, decide upon the scope and type of assistance required.

3. Each State Party to which a request for such assistance is directed shall promptly decide and notify the requesting State Party directly or through the Agency whether it is in a position to render the assistance requested, and the scope and terms of the assistance that might be rendered.

4. States Parties shall within the limits of their capabilities identify and notify the Agency of experts, equipment and materials which could be made available for the provision of assistance to other States Parties in the event of a nuclear accident or radiological emergency as well as the terms, especially financial, under which such assistance could be provided.

5. Any State Party may request assistance relating to medical treatment or temporary relocation into the territory of another State Party of people involved in a nuclear accident or radiological emergency.

6. The Agency shall respond, in accordance with its Statute and as provided for in this Convention, to a requesting State Party's or a Member State's request for assistance in the event of a nuclear accident or radiological emergency by:

(a) making available appropriate resources allocated for this purpose;

(b) transmitting promptly the request to other States and international organizations which, according to the Agency's information, may possess the necessary resources; and

(c) if so requested by the requesting State, co-ordinating the assistance at the international level which may thus become available.

Article 3

Direction and control of assistance

Unless otherwise agreed:

(a) the overall direction, control, co-ordination and supervision of the assistance shall be the responsibility within its territory of the requesting State. The assisting party should, where the assistance involves personnel, designate in consultation with the requesting State, the person who should be in charge of and retain immediate operational supervision over the personnel and the equipment provided by it. The designated person should exercise such supervision in cooperation with the appropriate authorities of the requesting State;

(b) the requesting State shall provide, to the extent of its capabilities, local facilities and services for the proper and effective administration of the assistance. It shall also ensure the protection of personnel, equipment and materials brought into its territory by or on behalf of the assisting party for such purpose;

(c) ownership of equipment and materials provided by either party during the periods of assistance shall be unaffected, and their return shall be ensured;

(d) a State Party providing assistance in response to a request under paragraph 5 of article 2 shall co-ordinate that assistance within its territory.

Article 4

Competent authorities and points of contact

1. Each State Party shall make known to the Agency and to other States Parties, directly or through the Agency, its competent authorities and point of contact authorized to make and receive requests for and to accept offers of assistance. Such points of contact and a focal point within the Agency shall be available continuously.

2. Each State Party shall promptly inform the Agency of any changes that may occur in the information referred to in paragraph 1.

3. The Agency shall regularly and expeditiously provide to States Parties, Member States and relevant international organizations the information referred to in paragraphs 1 and 2.

Article 5

Functions of the Agency

The States Parties request the Agency, in accordance with paragraph 3 of article 1 and without prejudice to other provisions of this Convention, to:

(a) collect and disseminate to States Parties and Member States information concerning:

(i) experts, equipment and materials which could be made available in the event of nuclear accidents or radiological emergencies;

(ii) methodologies, techniques and available results of research relating to response to nuclear accidents or radiological emergencies;

(b) assist a State Party or a Member State when requested in any of the following or other appropriate matters:

(i) preparing both emergency plans in the case of nuclear accidents and radiological emergencies and the appropriate legislation;

(ii) developing appropriate training programmes for personnel to deal with nuclear accidents and radiological emergencies;

(iii) transmitting requests for assistance and relevant information in the event of a nuclear accident or radiological emergency;

(iv) developing appropriate radiation monitoring programmes, procedures and standards;

(v) conducting investigations into the feasibility of establishing appropriate radiation monitoring systems;

(c) make available to a State Party or a Member State requesting assistance in the event of a nuclear accident or radiological emergency appropriate resources allocated for the purpose of conducting an initial assessment of the accident or emergency.

(d) offer its good offices to the States Parties and Member States in the event of a nuclear accident or radiological emergency;

(e) establish and maintain liaison with relevant international organizations for the purposes of obtaining and exchanging relevant information and data, and make a list of such organizations available to States Parties, Member States and the aforementioned organizations.

Article 6

Confidentiality and public statements

1. The requesting State and the assisting party shall protect the confidentiality of any confidential information that becomes available to either of them in connection with the assistance in the event of a nuclear accident or radiological emergency. Such information shall be used exclusively for the purpose of the assistance agreed upon.

2. The assisting party shall make every effort to coordinate with the requesting State before releasing information to the public on the assistance provided in connection with a nuclear accident or radiological emergency.

Article 7

Reimbursement of costs

1. An assisting party may offer assistance without costs to the requesting State. When considering whether to offer assistance on such a basis, the assisting party shall take into account:

(a) the nature of the nuclear accident or radiological emergency;
(b) the place of origin of the nuclear accident or radiological emergency;
(c) the needs of developing countries;
(d) the particular needs of countries without nuclear facilities; and
(e) any other relevant factors.

2. When assistance is provided wholly or partly on a reimbursement basis, the requesting State shall reimburse the assisting party for the costs incurred for the services rendered by persons or organizations acting on its behalf, and for all expenses in connection with the assistance to the extent that such expenses are not directly defrayed by the requesting State. Unless otherwise agreed, reimbursement shall be provided promptly after the assisting party has presented its request for reimbursement to the requesting State, and in respect of costs other than local costs, shall be freely transferrable.

3. Notwithstanding paragraph 2, the assisting party may at any time waive, or agree to the postponement of, the reimbursement in whole or in part. In considering such waiver or postponement, assisting parties shall give due consideration to the needs of developing countries.

Article 8

Privileges, immunities and facilities

1. The requesting State shall afford to personnel of the assisting party and personnel acting on its behalf the necessary privileges, immunities and facilities for the performance of their assistance functions.

2. The requesting State shall afford the following privileges and immunities to personnel of the assisting party or personnel acting on its behalf who have been duly notified to and accepted by the requesting State:

(a) immunity from arrest, detention and legal process, including criminal, civil and administrative jurisdiction, of the requesting State, in respect of acts or omissions in the performance of their duties; and

(b) exemption from taxation, duties or other charges, except those which are normally incorporated in the price of goods or paid for services rendered, in respect of the performance of their assistance functions.

3. The requesting State shall:

(a) afford the assisting party exemption from taxation, duties or other charges on the equipment and property brought into the territory of the requesting State by the assisting party for the purpose of the assistance; and

(b) provide immunity from seizure, attachment or requisition of such equipment and property.

4. The requesting State shall ensure the return of such equipment and property. If requested by the assisting party, the requesting State shall arrange, to the extent it is able to do so, for the necessary decontamination of recoverable equipment involved in the assistance before its return.

5. The requesting State shall facilitate the entry into, stay in and departure from its national territory of personnel notified pursuant to paragraph 2 and of equipment and property involved in the assistance.

6. Nothing in this article shall require the requesting State to provide its nationals or permanent residents with the privileges and immunities provided for in the foregoing paragraphs.

7. Without prejudice to the privileges and immunities, all beneficiaries enjoying such privileges and immunities under this article have a duty to respect the laws and regulations of the requesting State. They shall also have the duty not to interfere in the domestic affairs of the requesting State.

8. Nothing in this article shall prejudice rights and obligations with respect to privileges and immunities afforded pursuant to other international agreements or the rules of customary international law.

9. When signing, ratifying, accepting, approving or acceding to this Convention, a State may declare that it does not consider itself bound in whole or in part by paragraphs 2 and 3.

10. A State Party which has made a declaration in accordance with paragraph 9 may at any time withdraw it by notification to the depositary.

Article 9

Transit of personnel, equipment and property

Each State Party shall, at the request of the requesting State or the assisting party, seek to facilitate the transit through its territory of duly notified personnel, equipment and property involved in the assistance to and from the requesting State.

Article 10

Claims and compensation

1. The States Parties shall closely cooperate in order to facilitate the settlement of legal proceedings and claims under this article.

2. Unless otherwise agreed, a requesting State shall in respect of death or of injury to persons, damage to or loss of property, or damage to the environment caused within its territory or other area under its jurisdiction or control in the course of providing the assistance requested:

(a) not bring any legal proceedings against the assisting party or persons or other legal entities acting on its behalf;

(b) assume responsibility for dealing with legal proceedings and claims brought by third parties against the assisting party or against persons or other legal entities acting on its behalf;

(c) hold the assisting party or persons or other legal entities acting on its behalf harmless in respect of legal proceedings referred to in sub-paragraph (b); and

(d) compensate the assisting party or persons or other legal entities acting on its behalf for:

(i) death of or injury to personnel of the assisting party or persons acting on its behalf;

(ii) loss of or damage to non-consumable equipment or materials related to the assistance;

except in cases of wilful misconduct by the individuals who caused the death, injury, loss or damage.

3. This article shall not prevent compensation or indemnity available under any applicable international agreement or national law of any State.

4. Nothing in this article shall require the requesting State to apply paragraph 2 in whole or in part to its nationals or permanent residents.

5. When signing, ratifying, accepting or acceding to this Convention, a State may declare:

(a) that it does not consider itself bound in whole or in part by paragraph 2;

(b) that it will not apply paragraph 2 in whole or in part in cases of gross negligence by the individuals who caused the death, injury, loss or damage.

6. A State Party which has made a declaration in accordance with paragraph 5 may at any time withdraw it by notification to the depositary.

Article 11

Termination of assistance

The requesting State or the assisting party may at any time, after appropriate consultations and by notification in writing, request the termination of assistance received or provided under this Convention. Once such a request has been made the parties involved shall consult with each other to make arrangements for the proper conclusion of the assistance.

Article 12

Relationship to other international agreements

This Convention shall not affect the reciprocal rights and obligations of States Parties under existing international agreements which relate to the matters covered by this Convention, or under future international agreements concluded in accordance with the object and purpose of this Convention.

Article 13

Settlement of disputes

1. In the event of a dispute between States Parties, or between a State Party and the Agency, concerning the interpretation or application of this Convention, the parties to the dispute shall consult with a view to the settlement of the dispute by negotiation or by any other peaceful means of settling disputes acceptable to them.

2. If a dispute of this character between States Parties cannot be settled within one year from the request for consultation pursuant to paragraph 1, it shall, at the request of any party to such dispute, be submitted to arbitration or referred to the International Court of Justice for decision. Where a dispute is submitted to arbitration, if, within six months from the date of the request, the parties to the

dispute are unable to agree on the organization of the arbitration, a party may request the President of the International Court of Justice or the Secretary-General of the United Nations to appoint one or more arbitrators. In cases of conflicting requests by the parties to the dispute, the request to the Secretary-General of the United Nations shall have priority.

3. When signing, ratifying, accepting, approving or acceding to this Convention, a State may declare that it does not consider itself bound by either or both of the dispute settlement procedures provided for in paragraph 2. The other States Parties shall not be bound by a dispute settlement procedure provided for in paragraph 2 with respect to a State Party for which such a declaration is in force.

4. A State Party which has made a declaration in accordance with paragraph 3 may at any time withdraw it by notification to the depositary.

Article 14

Entry into force

1. This Convention shall be open for signature by all States and Namibia, represented by the United Nations Council for Namibia, at the Headquarters of the International Atomic Energy Agency in Vienna, and at the Headquarters of the United Nations in New York from................. until its entry into force or for twelve months, whichever period is longer.

2. A State and Namibia, represented by the United Nations Council for Namibia, may express its consent to be bound by this Convention either by signature, or by deposit of an instrument of ratification, acceptance or approval following signature made subject to ratification, acceptance or approval, or by deposit of an instrument of accession. The instruments of ratification, acceptance, approval or accession shall be deposited with the depositary.

3. This Convention shall enter into force thirty days after consent to be bound has been expressed by three States.

4. For each State expressing consent to be bound by this Convention after its entry into force, this Convention shall enter into force for that State thirty days after the date of expression of consent.

5. (a) This Convention shall be open for accession, as provided for in this article, by international organizations and regional integration organizations constituted by sovereign States, which have competence in respect of the negotiation, conclusion and application of international agreements in matters covered by this Convention.

(b) In matters within their competence such organizations shall, on their own behalf, exercise the rights and fulfil the obligations which this Convention attributes to States Parties.

(c) When depositing its instrument of accession, such an organization shall communicate to the depositary a declaration indicating the extent of its competence in respect of matters covered by this Convention.

(d) Such an organization shall not hold any vote additional to those of its Member States.

Article 15

Provisional application

A State may, upon signature or at any later date before this Convention enters into force for it, declare that it will apply this Convention provisionally.

Article 16

Amendments

1. A State Party may propose amendments to this Convention. The proposed amendment shall be submitted to the depositary who shall circulate it immediately to all other States Parties.

2. If a majority of the States Parties request the depositary to convene a conference to consider the proposed amendments, the depositary shall invite all States Parties to attend such a conference to begin not sooner than thirty days after the invitations are issued. Any amendment adopted at the conference by a two-thirds majority of all States Parties shall be laid down in a protocol which is open to signature in Vienna and New York by all States Parties.

3. The protocol shall enter into force thirty days after consent to be bound has been expressed by three States. For each State expressing consent to be bound by the protocol after its entry into force, the protocol shall enter into force for that State thirty days after the date of expression of consent.

Article 17

Denunciation

1. A State Party may denounce this Convention by written notification to the depositary.

2. Denunciation shall take effect one year following the date on which the notification is received by the depositary.

Article 18

Depositary

1. The Director General of the Agency shall be the depositary of this Convention.

2. The Director General shall promptly notify States Parties and all other States of:

(a) each signature of this Convention or any protocol of amendment;

(b) each deposit of an instrument of ratification, acceptance, approval or accession concerning this Convention or any protocol of amendment;

(c) any declaration or withdrawal thereof in accordance with articles 8, 10 and 13;

(d) any declaration of provisional application of this Convention in accordance with article 15;

(e) the entry into force of this Convention and of any amendment thereto; and

(f) any denunciation made under article 17.

Article 19

Authentic texts and certified copies

The original of this Convention, of which the Arabic, Chinese, English, French, Russian and Spanish texts are equally authentic, shall be deposited with the Director General of the International Atomic Energy Agency who shall send certified copies to States Parties and all other States.

IN WITNESS WHEREOF the undersigned, being duly authorized, have signed this Convention, open for signature as provided in paragraph 1 of article 14.

ADOPTED by the General Conference of the International Atomic Energy Agency meeting in special session at Vienna on the .... day of .... one thousand nine hundred and ....

GC/8/Rev.3/Corr.1
15 August 1986

CONVENTION ON EARLY NOTIFICATION OF A NUCLEAR ACCIDENT

THE STATES PARTIES TO THIS CONVENTION,

AWARE that nuclear activities are being carried out in a number of States,

NOTING that comprehensive measures have been and are being taken to ensure a high level of safety in nuclear activities, aimed at preventing nuclear accidents and minimizing the consequences of any such accident, should it occur,

DESIRING to strengthen further international co-operation in the safe development and use of nuclear energy,

CONVINCED of the need for States to provide relevant information about nuclear accidents as early as possible in order that transboundary radiological consequences can be minimized,

NOTING the usefulness of bilateral and multilateral arrangements on information exchange in this area,

HAVE AGREED as follows:

Article 1

Scope of application

1. This Convention shall apply in the event of any accident involving facilities or activities of a State Party or of persons or legal entities under its jurisdiction or control, referred to in paragraph 2 below, from which a release of radioactive material occurs or is likely to occur and has resulted or may result in an international transboundary release that could be of radiological safety significance for another State.

2. The facilities and activities referred to in paragraph 1 are the following:

(a) any nuclear reactor wherever located;
(b) any nuclear fuel cycle facility;
(c) any radioactive waste management facility;
(d) the transport and storage of nuclear fuels or radioactive wastes;
(e) the manufacture, use, storage, disposal and transport of radioisotopes for agricultural, industrial, medical and related scientific and research purposes; and
(f) the use of radioisotopes for power generation in space objects.

Article 2

Notification and information

In the event of an accident specified in article 1, (hereinafter referred to as a "nuclear accident"), the State Party referred to in that article shall:

(a) forthwith notify, directly or through the International Atomic Energy Agency (hereinafter referred to as the "Agency"), those

States which are or may be physically affected as specified in article 1 and the Agency of the nuclear accident, its nature, the time of its occurrence and its exact location where appropriate;

(b) promptly provide the States referred to in sub-paragraph (a), directly or through the Agency, and the Agency with such available information relevant to minimizing the radiological consequences in those States, as specified in article 5.

Article 3

Other Nuclear Accidents

With a view to minimizing the radiological consequences, States Parties may notify in the event of nuclear accidents other than those specified in article 1.

Article 4

Functions of the Agency

The Agency shall:

(a) forthwith inform States Parties, Member States, other States which are or may be physically affected as specified in article 1 and relevant international intergovernmental organizations (hereinafter referred to as "international organizations") of a notification received pursuant to sub-paragraph (a) of article 2; and

(b) promptly provide any State Party, Member State or relevant international organization, upon request, with the information received pursuant to sub-paragraph (b) of article 2.

Article 5

Information to be provided

1. The information to be provided pursuant to sub-paragraph (b) of article 2 shall comprise the following data as then available to the notifying State Party:

(a) the time, exact location where appropriate, and the nature of the nuclear accident;
(b) the facility or activity involved;
(c) the assumed or established cause and the foreseeable development of the nuclear accident relevant to the transboundary release of the radioactive materials;
(d) the general characteristics of the radioactive release, including, as far as is practicable and appropriate, the nature, probable physical and chemical form and the quantity, composition and effective height of the radioactive release;
(e) information on current and forecast meteorological and hydrological conditions, necessary for forecasting the transboundary release of the radioactive materials;
(f) the results of environmental monitoring relevant to the transboundary release of the radioactive materials;
(g) the off-site protective measures taken or planned;
(h) the predicted behaviour over time of the radioactive release.

2. Such information shall be supplemented at appropriate intervals by further relevant information on the development of the emergency situation, including its foreseeable or actual termination.

3. Information received pursuant to sub-paragraph (b) of article 2 may be used without restriction, except when such information is provided in confidence by the notifying State Party.

Article 6

Consultations

A State Party providing information pursuant to sub-paragraph (b) of article 2 shall, as far as is reasonably practicable, respond promptly to a request for further information or consultations sought by an affected State Party with a view to minimizing the radiological consequences in that State.

Article 7

Competent authorities and points of contact

1. Each State Party shall make known to the Agency and to other States Parties, directly or through the Agency, its competent authorities and point of contact responsible for issuing and receiving the notification and information referred to in article 2. Such points of contact and a focal point within the Agency shall be available continuously.

2. Each State Party shall promptly inform the Agency of any changes that may occur in the information referred to in paragraph 1.

3. The Agency shall maintain an up-to-date list of such national authorities and points of contact as well as points of contact of relevant international organizations and shall provide it to States Parties and Member States and to relevant international organizations.

Article 8

Assistance to States Parties

The Agency shall, in accordance with its Statute and upon a request of a State Party which does not have nuclear activities itself and borders on a State having an active nuclear programme but not Party, conduct investigations into the feasibility and establishment of an appropriate radiation monitoring system in order to facilitate the achievement of the objectives of this Convention.

Article 9

Bilateral and multilateral arrangements

In furtherance of their mutual interests, States Parties may consider, where deemed appropriate, the conclusion of bilateral or multilateral arrangements relating to the subject matter of this Convention.

Article 10

Relationship to other international agreements

This Convention shall not affect the reciprocal rights and obligations of States Parties under existing international agreements which relate to the matters covered by this Convention, or under future international agreements concluded in accordance with the object and purpose of this Convention.

Article 11

Settlement of disputes

1. In the event of a dispute between States Parties, or between a State Party and the Agency, concerning the interpretation or application of this Convention, the parties to the dispute shall consult with a view to the settlement of the dispute by negotiation or by any other peaceful means of settling disputes acceptable to them.

2. If a dispute of this character between States Parties cannot be settled within one year from the request for consultation pursuant to paragraph 1, it shall, at the request of any party to such dispute, be submitted to arbitration or referred to the International Court of Justice for decision. Where a dispute is submitted to arbitration, if, within six months from the date of the request, the parties to the dispute are unable to agree on the organization of the arbitration, a party may request the President of the International Court of Justice or the Secretary-General of the United Nations to appoint one or more arbitrators. In cases of conflicting requests by the parties to the dispute, the request to the Secretary-General of the United Nations shall have priority.

3. When signing, ratifying, accepting, approving or acceding to this Convention, a State may declare that it does not consider itself bound by either or both of the dispute settlement procedures provided for in paragraph 2. The other States Parties shall not be bound by a dispute settlement procedure provided for in paragraph 2 with respect to a State Party for which such a declaration is in force.

4. A State Party which has made a declaration in accordance with paragraph 3 may at any time withdraw it by notification to the depositary.

Article 12

Entry into force

1. This Convention shall be open for signature by all States and Namibia, represented by the United Nations Council for Namibia, at the Headquarters of the International Atomic Energy Agency in Vienna, and at the Headquarters of the United Nations in New York from .... until its entry into force or for twelve months, whichever period is longer.

2. A State and Namibia, represented by the United Nations Council for Namibia, may express its consent to be bound by this Convention either by signature, or by deposit of an instrument of ratification, acceptance or approval following signature made subject to ratification, acceptance or approval, or by deposit of an instrument of accession. The instruments of ratification, acceptance, approval or accession shall be deposited with the depositary.

3. This Convention shall enter into force thirty days after consent to be bound has been expressed by three States.

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5.(a) This Convention shall be open for accession, as provided for in this article, by international organizations and regional integration organizations constituted by sovereign States, which have competence in respect of the negotiation, conclusion and application of international agreements in matters covered by this Convention.

(b) In matters within their competence such organizations shall, on their own behalf, exercise the rights and fulfil the obligations which this Convention attributes to States Parties.

(c) When depositing its instrument of accession, such an organization shall communicate to the depositary a declaration indicating the extent of its competence in respect of matters covered by this Convention.

(d) Such an organization shall not hold any vote additional to those of its Member States.

Article 13

Provisional application

A State may, upon signature or at any later date before this Convention enters into force for it, declare that it will apply this Convention provisionally.

Article 14

Amendments

1. A State Party may propose amendments to this Convention. The proposed amendment shall be submitted to the depositary who shall circulate it immediately to all other States Parties.

2. If a majority of the States Parties request the depositary to convene a conference to consider the proposed amendments, the depositary shall invite all States Parties to attend such a conference to begin not sooner than thirty days after the invitations are issued. Any amendment adopted at the conference by a two-thirds majority of all States Parties shall be laid down in a protocol which is open to signature in Vienna and New York by all States Parties.

3. The protocol shall enter into force thirty days after consent to be bound has been expressed by three States. For each State expressing consent to be bound by the protocol after its entry into force, the protocol shall enter into force for that State thirty days after the date of expression of consent.

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Denunciation

1. A State Party may denounce this Convention by written notification to the depositary.

2. Denunciation shall take effect one year following the date on which the notification is received by the depositary.

Article 16

Depositary

1. The Director General of the Agency shall be the depositary of this Convention.

2. The Director General of the Agency shall promptly notify States Parties and all other States of:

(a) each signature of this Convention or any protocol of amendment;
(b) each deposit of an instrument of ratification, acceptance, approval or accession concerning this Convention or any protocol of amendment;
(c) any declaration or withdrawal thereof in accordance with article 11;
(d) any declaration of provisional application of this Convention in accordance with article 13;
(e) the entry into force of this Convention and of any amendment thereto; and
(f) any denunciation made under article 15.

Article 17

Authentic texts and certified copies

The original of this Convention, of which the Arabic, Chinese, English, French, Russian and Spanish texts are equally authentic, shall be deposited with the Director General of the International Atomic Energy Agency who shall send certified copies to States Parties and all other States.

IN WITNESS WHEREOF the undersigned, being duly authorized, have signed this Convention, open for signature as provided in paragraph 1 of article 12.

ADOPTED by the General Conference of the International Atomic Energy Agency meeting in special session at Vienna on the .... day of .... one thousand nine hundred and ....

Appendix II

Principal Topics Identified for Action Post-Chernobyl in IAEA Supplementary Programme

1. Man-machine interface
2. Development of methods of probabilistic safety assessment
3. Quality assurance
4. Fire fighting and control in irradiation environments
5. Medical handling of highly irradiated persons
6. Epidemiological studies of affected population
7. Methods for the training of reactor operators
8. Methods for review of safety, eg. OSART and ASSERT missions
9. Emergency preparedness - measurements, communications management
10. Review safety objectives and principles.

Appendix III

UNOFFICIAL TRANSLATION by the Secretariat(original German) of the

DRAFT MOTION FOR A RESOLUTION

The European Parliament,

- having regard to the Commission's communications to the Council (Com(86) 276, 327 and 607 fin.);
- having regard to the report by the Russian State Committee to IAEA concerning use of nuclear power, following the reactor accident at Chernobyl;
- having regard to the report of the Committee on Energy, Research and Technology (Doc A2 - /86);

A. whereas the civilian use of nuclear power in Europe makes a considerable contribution to the electricity and energy supply and will continue to do so in the near future,

B. whereas the design characteristics of the RBMK reactor involved in the Chernobyl accident, in particular the "positive power-coefficient", are not found in the Western nuclear reactors,

C. whereas the operation of nuclear reactors must be carried out under optimal safety proceduers,

1. recalls and confirms previous resolutions by the European Parliament, as well as the requests made therein to the Commission- namely, for

   (i) establishment of international standards for design, containment and safety monitoring for nuclear reactors,

   (ii) establishment of a permanent system for monitoring and alarm in case of a nuclear accident,

   (iii) establishment of an international and independent inspectorate, also responsible for examination of operating licences,

2. points out that in case of an accident like the one at Chernobyl, the powers of the responsible authorities in the Community, (i.e. the Commission) are, not withstanding the provisions of the Euratom Treaty, insufficient,

3. regrets nevertheless that the Commission, in the months which followed the reactor accident at Chernobyl, took no initiatives - or at best inadequate measures - to remedy the apparent shortcomings of the Euratom Treaty and to improve the preparedness of the Community,

4. demands in this connection a detailed report on the state of technical safety of all nuclear power-stations in the EC,

5. demands that if safety shortcomings are found in any nuclear power-station in this study, then either further safety measures must be introduced or they must be closed,

6. points to the necessity of further studies in reactor safety,

7. calls for the European Community to take a more active role in international fora, in particular in the framework of the IAEA, when establishing general procedures, safety standards and norms for the construction and operating of reactors, as well as for inspections and regulations,

8. stresses that the nuclear community is international by nature, which is why problems also have to be met at international level,

9. considers the IAEA the best suited organisation on the international level for these purposes as Eastern European states also belong to it,

10. stresses the great importance of education and continuing training for operating staff, up to the most recent state of the art,

11. instructs its President to send this resolution to the Commission, the Council and the International Atomic Energy Agency.

Appendix IV

SPEECH TO THE FIRST SESSION OF THE IAEA SPECIAL GENERAL CONFERENCE BY THE RT HON PETER WALKER MBE MP, UNITED KINGDOM SECRETARY OF STATE FOR ENERGY

Living standards throughout much of the World have been transformed during this century. We all know that that transformation would have been impossible without bountiful supplies of energy. A century in which the population has quadrupled and manufacturing and industrial activity has developed at a pace never previously envisaged has resulted in this century being the first in the history of mankind when significant energy shortages have occurred and the prospect of the supply of energy not meeting the demand has become a real possibility.

If we examine objectively the known reserves of finite energy resources such as oil, gas and coal we must all be aware that the next century could be a century where the improvement of living standards could be halted.

The world is vigorously examining all of the alternative forms of energy: the sun, the wind, the tides, geothermal energy. Yet every one of these examinations shows and indicates that whilst these sources may each make a contribution it would be a totally inadequate contribution to what the world will require.

If one decided that in spite of these projections we would erradicate nuclear energy, then the dimension of the problem would take on staggering proportions. Already a third of the European Community's electricity is generated by nuclear power. The Soviet Union, the United States, many major countries depend upon nuclear power as a major contributor to their existing energy supplies which is vital to their future. Cast aside this option and the pressures upon other energy sources would be such as to cause considerable economic upheaval.

Nuclear power has considerable attractions in terms of efficiency and in terms of environmental considerations. The one question-mark is whether or not the world can use the benefits of nuclear power with confidence as to its safety. It is the task of this Conference and this Agency to see that that is done.

The Chernobyl accident frightened the world. The Chernobyl accident illustrated that any major accident would be international in its impact and not confined to territorial boundaries. The Chernobyl accident made it clear that this Agency had to produce the agreements, the understandings, the practices and the international collaboration which would guarantee that the world could reap the benefits of nuclear power in safety.

Our prospects of achieving that aim have been considerably enhanced by the manner in which the Soviet Union have provided the facts to the international community. They have not covered up, they have not concealed what went wrong out of national pride, they have given us an objective analysis of what went wrong.

They have illustrated that there were mistakes in design, there were mistakes in engineering and there were mistakes in management. This ruthless analysis means that not only can they embark upon their future expansion of the use of nuclear energy, fulfilling the highest safety requirements, but that we can pursue policies internationally to see that safety is procured.

The widespread impact of Chernobyl illustrates clearly that the IAEA is the Agency in which action must take place. Of course every country will wish to pursue programmes of its own. The European Community will wish to discuss amongst its members the mutual problems of nuclear energy. But when it comes to laying down a regime for broader international collaboration on safety it must be done effectively by this Agency.

I wish to commit the British Government to our fullest support of the Agency in its work. We are ready to sign the Conventions that are before us. Ratification will be needed in the normal way, but in the meantime we shall apply both Conventions as from today. The British Government will also voluntarily inform the IAEA and any states affected of any other serious accident involving the United Kingdom's military facilities or equipment, so that the nuclear industry in both the military and energy sectors will carry out their responsibility to provide the information needed by others.

The British Government is anxious to see a general system of compensation in respect of nuclear accidents and we would support a binding international regime to provide that compensation.

I wish to make to this Conference, on behalf of the British Government, five essential proposals which we hope can be implemented in the months ahead.

Firstly, we must agree and perfect an international warning system of any accident. There are lessons to be learned from Chernobyl. But this Agency must see that there is put into place throughout the world an immediate warning system that is effective and complete.

Secondly, we must have a system in which we organise the widest exchange of experience. The recent meeting of international experts was creative and constructive. There must be a continuing programme with a complete exchange of views in areas such as operator training, protective systems against operator error and all of the methods for preventing and detecting radioactive releases.

Thirdly, this Agency must look in depth at all of the existing regulatory regimes. The Agency has done important work on the development of a code of practice for regulations which has certainly been helpful to those embarking on nuclear programmes for the first time. The time has come to extend this work. The Agency should up-date its information about the details of all existing national regulatory systems with a clear definition as to the powers of nuclear inspectorates, their objectives, their involvement in reactor licensing, their involvement in design approval, and the methods of risk assessment. Having collated information - that will be available to us all - the Agency will be in a strong position to see that arrangements of one regulatory authority that have benefit are transmitted to other authorities where they do not exist so they can swiftly apply them.

The third stage would be the possibility of the Agency having a team who could co-ordinate a peer review of international regulatory systems with the aim of bringing about a constructive exchange as to how those regulatory systems could reach the highest possible standards. Certainly my country would welcome such a role for this Agency.

Fourthly, this Agency must develop the skill and expertise that it can examine the quality of nuclear inspectorates. It is not possible to organise an international inspectorate or indeed even a European inspectorate, due to the diverse forms of reactor, problems of language and due to the importance of a nuclear inspectorate being constantly close to what it has inspected. But this Agency could organise the expertise to provide those inspectorates with suggestions and advice which could improve their performance.

Fifthly, we must support the Convention for Mutual Assistance in the Event of a Nuclear Accident. I know that the Soviet authorities benefitted from some international help at Chernobyl and I believe that this Agency must put into operation a system in which international help would be immediate and effective.

We must provide the vision and leadership to create for coming generations conditions in which they can enjoy a form of energy that is safe, that is of immense economic benefit and that is environmentally more benign than any other form of energy known to man. If we succeed in our regulatory and safety requirements nuclear energy will not pollute the air that we breath as other fuels have always polluted our air. It will not pollute our lakes or our forests.

The provision of nuclear energy, like every other great service to mankind, has its challenges and its dangers but if we succceed by international collaboration it has immense benefits for the future of the human race.

We must make real and important progress, so that this Agency is operating a system of international safety in which the world has confidence and trust.

24 September 1986

Chernobyl file C11

МЕЖДУНАРОДНОЕ АГЕНТСТВО
ПО АТОМНОЙ ЭНЕРГИИ

МЕЖДУНАРОДНАЯ КОНФЕРЕНЦИЯ
ПО ПОКАЗАТЕЛЯМ И БЕЗОПАСНОСТИ
ЯДЕРНОЙ ЭНЕРГЕТИКИ

Вена, Австрия, 28 сентября – 2 октября 1987 года

IAEA-CN-48/63

АВАРИЯ НА ЧЕРНОБЫЛЬСКОЙ АЭС:
ГОД СПУСТЯ

АВАРИЯ НА ЧЕРНОБЫЛЬСКОЙ АЭС: ГОД СПУСТЯ

В.Г. Асмолов, А.А. Боровой, В.Ф. Демин, А.К. Калугин,
И.И. Кузьмин, В.М. Кулаков, В.А. Легасов, Г.Л. Лунин,
Н.Н. Пономарев-Степной, А.Н. Проценко, В.К. Сухоручкин,
А.А. Хрулев, О.Я. Шах
Институт атомной энергии им. И.В. Курчатова

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А.А. Абагян, В.М. Дмитриев, А.А. Шкурпелов
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А.П. Поваляев
Госагропром СССР

С.И. Авдюшин, Ю.А. Израэль, В.Н. Петров, В.В. Писарев
Госкомгидромет СССР

1. ВВЕДЕНИЕ

Делегацией советских экспертов на специальном совещании, проведенном МАГАТЭ 25 — 29 августа 1986 г. в Вене, была представлена информация об аварии на Чернобыльской АЭС и ее последствиях [1]. Эта информация содержала результаты исследования причин аварии, а также описание и предварительный анализ эффективности первоочередных мероприятий, проведенных с целью ограничения и ликвидации ее последствий по данным, полученным до 1 августа 1986 г.

В последующий период усилия были сосредоточены на следующих направлениях.

1. Продолжение работ по ликвидации последствий аварии, в том числе:

— завершение проектирования и сооружения объекта Укрытие, обеспечивающего надежную защиту окружающей среды от попадания в нее радиоактивных веществ из разрушенного блока и от радиоактивного излучения;

— дальнейшая дезактивация территории ЧАЭС, зданий и помещений I, II и III энергоблоков и населенных пунктов в зоне, подвергшейся радиоактивному загрязнению;

— введение в эксплуатацию I и II энергоблоков ЧАЭС;

— завершение мероприятий по социально-бытовому обеспечению эвакуированного населения и его трудоустройству;

— проведение необходимых медико-санитарных мероприятий по обеспечению безопасности населения и охране его здоровья.

2. Разработка программы и организация долгосрочных исследований по изучению отдаленных последствий аварии, а также мероприятий по их ограничению и ликвидации, в том числе:

— мониторинг радиоактивного загрязнения окружающей среды;

— определение необходимости и проведение дальнейших дезактивационных работ;

— проведение научно-исследовательских контрольных и профилактических работ в Укрытии;

— проведение научных исследований по изучению долговременных последствий радиоактивного загрязнения биосферы.

Все научные исследования координируются специально созданным советом при АН СССР.

3. Разработка и внедрение мер по повышению безопасности действующих АЭС.

4. Рассмотрение планов дальнейшего развития ядерной энергетики и возможностей повышения уровня ее безопасности, включающее разработку концепции ядерных реакторов нового поколения и расширение научных исследований по всем аспектам оценки и обеспечения безопасности ядерной энергетики.

В представляемом докладе рассматриваются ход и результаты работ по указанным направлениям.

2. СООРУЖЕНИЕ УКРЫТИЯ

К числу важнейших мер по ликвидации последствий аварии относилось сооружение Укрытия, которое должно обеспечить долговременную консервацию аварийного блока.

Объект Укрытие по своему назначению и функциям не является ни хранилищем ядерного топлива, ни могильником высокоактивных отходов, ни каким-либо другим объектом, ранее встречавшимся в ядерной технологии. Необходимость создания Укрытия потребовала разработать основные положения, определяющие назначение объекта и требования к нему, сформулировать концепцию его безопасности.

Безопасным состоянием объекта Укрытие является состояние, при котором поддерживаются условия, исключающие:

— возникновение самоподдерживающейся цепной реакции;

— нарушение условий теплосъема, приводящих к плавлению остатков топливных масс;

— образование взрывоопасных концентраций водорода.

Основное назначение Укрытия:

— предотвращение выхода в окружающую среду радиоактивных веществ из поврежденного реактора;

— защита прилегающей территории от проникающего излучения.

Основные требования к проекту Укрытия:

— сведение к минимуму времени строительства при использовании простых, надежных и апробированных средств;

— сохранение функций Укрытия при возможном воздействии различных природных явлений (ураганы, землетрясения и т.п.), которые могут происходить на площадке ЧАЭС;

— обеспечение отвода остаточного тепловыделения и радиолизного водорода;

— минимизация доз облучения у строителей при возведении сооружения;

— обеспечение доступа в консервируемые помещения с невысокими уровнями радиации для проведения научных исследований;

— возможность контроля и диагностики состояния активной массы.

Сразу после аварии были начаты радиационные и температурные измерения внутри и за пределами здания реактора, исследования состояния сохранившихся элементов конструкции реактора и реакторного здания, которые легли в основу временной системы контроля тепловых и радиационных параметров.

Основным способом проведения измерений в помещениях разрушенного блока была избрана радиационная и теплофизическая разведка, что позволило установить различного типа датчики, составить карту степени загрязненности помещений и приблизиться к помещениям с большими скоплениями топлива. Следует отметить, что опыт использования как отечественных, так и зарубежных робототехнических устройств показал их малую пригодность для решения основных задач разведки и дистанционного выполнения работ в сложных по конфигурации помещениях блока, при наличии разрушений и завалов, а также значительных полей гамма-излучения.

Начиная с мая в пространстве над реакторной шахтой провошему развитию математических моделей, описывающих процессы охлаждения. Были проанализированы несколько возможных механизмов этих процессов. Трассерные эксперименты, проведенные на аварийном блоке, показали, что адекватной реальной ситуацией является модель фильтрационного охлаждения, позволившая естественным образом объяснить основные особенности изменения теплового режима активной зоны после аварии [4].

Строгое математическое исследование нелинейной системы уравнений показало, что при квазистационарном режиме фильтрационного охлаждения в завале не возникают (с вероятностью, близкой к единице) зоны очень интенсивных локальных перегревов, даже при наличии интенсивных локализованных источников тепла. Вынос тепла из активной зоны фильтрационным потоком обеспечивает эффективное охлаждение всей массы завала.

Исследование условий существования квазистационарных режимов охлаждения позволило определить критерий существования режима квазистационарного фильтрационного охлаждения. В число определяющих параметров критерия входят интегральная мощность источников тепла и характеристики проницаемости завала. Условие квазистационарного охлаждения может оказаться нарушенным при уплотнении завала и соответствующем уменьшении его проницаемости.

Показано, что отсутствие квазистационарного решения приводит к нестационарному процессу, который был назван "сухим кипением". Этот процесс сопровождается мелкими разрывами сплошности и приводит к разрыхлению завала, а тем самым к увеличению его проницаемости. В результате система переходит в новое состояние, в котором квазистационарный режим охлаждения стабилизируется.

Для оценки параметров процесса охлаждения аварийного блока на основе математической модели фильтрационного охлаждения была создана программа численного расчета полей температур и других теплофизических характеристик в завале. Были проведены расчеты соответствующих полей при различных распределениях источников тепла в завале, которые коррелируют с данными экспериментальных измерений.

Учитывая исключительную важность сооружаемого объекта, для принятия окончательного решения было проработано и рассмотрено большое количество вариантов строительных конструктивных решений. Все они сводились к двум принципиальным направлениям:

первое — над разрушенным энергоблоком возвести арочное перекрытие пролетом 230 м или выполнить купольное сводчатое и консольное перекрытие пролетом до 120 м;

второе — выполнить перекрытия из конструктивных элементов пролетом 55 м, используя в качестве опор для них сохранившиеся стены и перекрытия задания.

Проработки и технико-экономические расчеты показали, что работы по первому направлению потребуют 1,5 — 2 года, тогда как работы по второму направлению позволят значительно сократить сроки строительства и расход материалов. Поэтому второе направление было принято в качестве основного решения, заложенного в проект.

Проектная документация по мере готовности выдавалась на строительство, где при необходимости уточнялась или дополнялась бригадой авторского надзора с учетом складывающихся конкретных условий. При разработке проектной документации были найдены также инженерные решения, позволившие максимально сократить трудовые затраты и сроки строительства в условиях сложной радиационной обстановки.

Объемно-пространственная структура Укрытия образована рядом каскадно-поднимающихся блоков, размеры и очертания которых определены особенностями элементов ограждающих конструкций. Между III и IV энергоблоками предусмотрена бетонная разделительная стена (при этом максимально использованы существующие стены). В машинном зале между II и III энергоблоками сооружена металлическая разделительная стенка.

Одновременно с сооружением Укрытия осуществлялась большая программа по созданию системы контроля за состоянием объекта и определения его влияния на радиационную обстановку на площадке ЧАЭС и за ее пределами. Были сформулированы технические требования и исходные данные для создания информационно-диагностического комплекса (ИДК), предназначенного для контроля и диагностики состояния топливной массы, элементов конструкции, радиационной обстановки, а также для контроля технологических систем объекта Укрытие.

ИДК состоит из:

— системы контроля и диагностики;

— информационно-вычислительного центра;

— системы связи с внешними пользователями.

Система контроля и диагностики включает в себя подсистемы:

— технологического контроля (ТК);

— радиационного дозиметрического контроля (РДК);

— систему диагностики (ДС) физико-механического состояния активной массы и конструкций сооружений.

Задача подсистемы ТК — контроль работы вентиляционного оборудования, обеспечивающего теплоотвод, и параметров теплоотводящей среды.

Подсистема РДК должна обеспечивать контроль за состоянием и переносом радиоактивности внутри сооружения и выбросом радиоактивных продуктов в окружающую среду.

Функцией ДС является определение физико-механического состояния активной массы, возникающих вибраций, смещений элементов конструкций сооружения и разрушения за счет внутренних процессов.

Для обеспечения контроля и диагностики состояния Укрытия проводятся измерения температуры в объеме под перекрытием центрального зала и на верхней поверхности шахты реактора, элементов нижней опорной плиты, поверхностей перекрытий в бассейне-барботере. С целью уточнения распределения и определения интенсивности источников тепловыделения измеряется тепловой поток в доступных точках подреакторных помещений и на верхней поверхности разрушенной активной зоны. Интенсивность гамма-излучения измеряется во всех помещениях, где осуществляются контроль состояния работающего оборудования, его обслуживание и ремонт. Кроме того, измеряются поля гамма-излучения в большинстве доступных помещений здания, а также в объеме под перекрытием и на верхней поверхности разрушенной активной зоны. Проводится постоянный контроль концентрации H_2, CO и H_2O в воздухе.

С целью обнаружения маловероятной самоподдерживающейся цепной реакции установлены нейтронные датчики и ведется контроль появления короткоживущих изотопов йод-131 на выходе системы вентиляции. Разработаны и реализованы мероприятия по ядерной безопасности законсервированного IV блока, которые обеспечивают возможность аварийного гашения процесса в случае развития цепной реакции деления в шахте реактора за счет ввода жидкого поглотителя нейтронов.

Для контроля за механической стабильностью топливной массы и элементов конструкции установлены виброакустические датчики, обеспечивающие регистрацию виброускорений, виброскоростей и виброперемещений.

Информационно-диагностический комплекс надежно обеспечивает контроль за состоянием объекта, который 1 декабря 1986 г. передан эксплуатирующей организации на техническое обслуживание.

Анализ имеющихся экспериментальных данных о мощностях дозы гамма-излучения, температуре и тепловом потоке указывает на стабильность состояния топливной массы. Мощность дозы гамма-излучения снижается в соответствии с распадом топлива. Среднесуточный суммарный выброс радиоактивных продуктов деления не превышает 3 мКи*. На территориях, прилегающих к г. Припять, за период август — сентябрь не обнаружено образования новых пятен радиоактивного загрязнения или увеличения интегрального загрязнения территории. Снижение интенсивности гамма-излучения соответствовало теоретическим оценкам.

По завершении работ на объекте Укрытие разрушенный энергоблок перестал являться источником повышенного выделения аэрозольной активности, как вследствие выноса ее за счет вентиляции, так и в результате ветровой эрозии.

Фотография Укрытия приведена на рис. 2.1. Изменения интенсивности гамма-излучения внутри Укрытия и температуры в наиболее "горячей" точке в одном из подреакторных помеще-

* Для работающего реакторного блока мощностью 1000 МВт предельно допустимый выброс смеси долгоживущих радионуклидов составляет 15 мКи/сут [6].

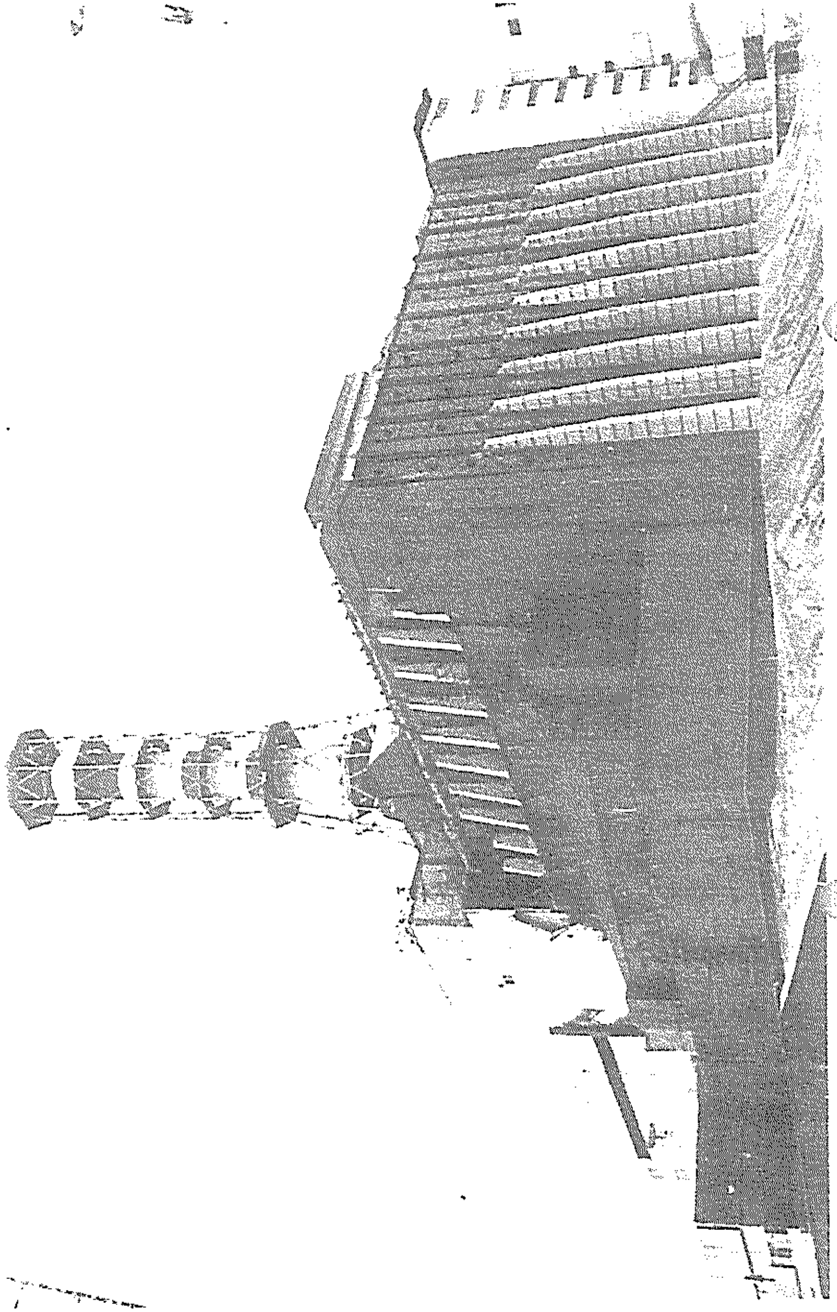

РИС. 2.1. ВНЕШНИЙ ВИД УКРЫТИЯ ПОСЛЕ ЗАВЕРШЕНИЯ ЕГО СТРОИТЕЛЬСТВА (ФОТО)

ний, приведенные к их значению на 1 января 1987 г., показаны на рис. 2.2.

В настоящее время в соответствии с долговременной программой научных исследований проводятся и планируются работы по следующим основным направлениям:

— уточнение количества и размещения ядерного топлива внутри блока;

— определение механического и физико-химического состояния топлива;

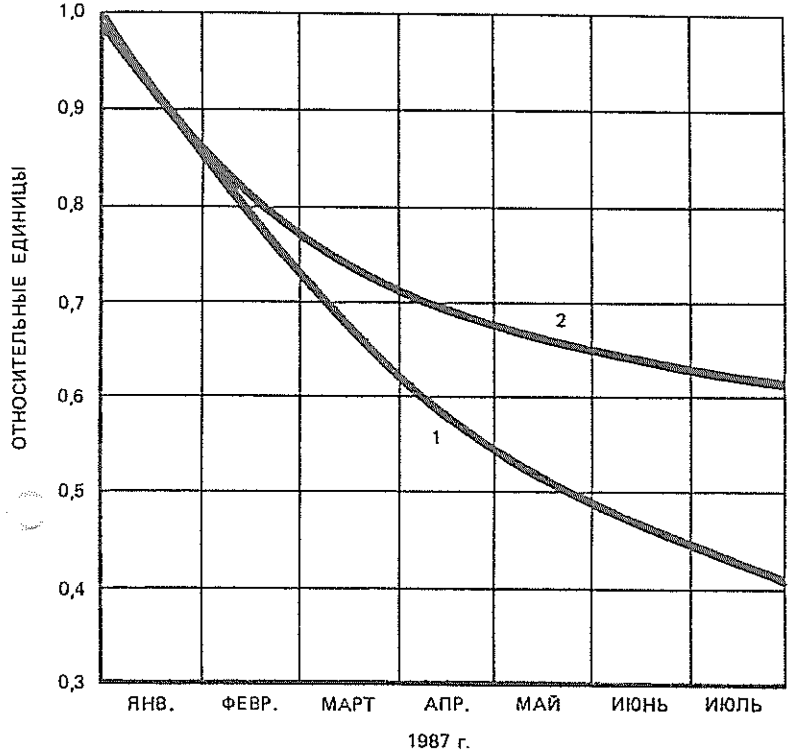

РИС. 2.2. ИЗМЕНЕНИЯ ИНТЕНСИВНОСТИ ГАММА-ИЗЛУЧЕНИЯ ВНУТРИ УКРЫТИЯ (1) И ТЕМПЕРАТУРЫ В НАИБОЛЕЕ "ГОРЯЧЕЙ" ТОЧКЕ В ОДНОМ ИЗ ПОДРЕАКТОРНЫХ ПОМЕЩЕНИЙ (2)

— исследование нейтронными методами размножающих, поглощающих и замедляющих свойств скоплений материалов, содержащих ядерное топливо;

— изучение свойств конструкционных материалов в полях гамма-излучений при взаимодействии с остатками топливной массы.

3. ВОЗОБНОВЛЕНИЕ ЭКСПЛУАТАЦИИ БЛОКОВ ЧЕРНОБЫЛЬСКОЙ АЭС

Вопросы возобновления эксплуатации I, II и III блоков Чернобыльской АЭС, проведения необходимых для этого работ были в ряду важнейших в плане ликвидации последствий аварии и решались параллельно с работами по консервации IV блока.

После аварии на IV блоке I и II блоки остались в нормальном работоспособном состоянии и были остановлены в 1 ч 13 мин и в 2 ч 13 мин соответственно 27 апреля.

Третий блок, который технически связан с IV блоком, был остановлен в 3 ч 26 апреля. Было произведено нормальное расхолаживание всех остановленных блоков.

После расхолаживания реакторы I, II и III блоков были переведены в глубоко подкритическое состояние путем ввода в активную зону всех стержней СУЗ и загрузки в реакторы I и II блоков по 20 дополнительных поглотителей (ДП), а в реактор III блока — 200 стерженьков-поглотителей в центральную трубку ТВС. Контроль за нейтронным потоком осуществлялся штатной аппаратурой.

Для отвода остаточного тепловыделения все технологические каналы (ТК) и контур многократной принудительной циркуляции (КМПЦ) оставались заполненными водой. Остаточное тепловыделение снималось в режиме естественной циркуляции. Температура воды в активной зоне поддерживалась на уровне 20 — 80° С, температура графита — 30 — 90° С.

Восстановительные работы были начаты с дезактивации основных и вспомогательных зданий и сооружений энергоблоков, находящегося в них оборудования и рабочих мест персонала, а также прилегающей территории.

Наибольшие уровни загрязнения имели отдельные горизонтальные участки поверхностей машинного зала [до 10^6 β-част./(см^2.мин)], так как его загрязнение происходило через разрушенную кровлю IV блока. Мощность дозы гамма-излучения в загрязненных помещениях I и II блоков на 20 мая 1986 г. составляла 10 — 100 мР/ч, машинного зала — 20 — 600 мР/ч.

Дезактивация проводилась с использованием специальных растворов, состав которых подбирался с учетом отмываемого материала, характера и уровня загрязнения поверхности. Применялись струйные и пароэжекционные методы, методы сухой дезактивации с помощью полимерных покрытий. Часть помещений и оборудования дезактивировалась вручную протиркой ветошью, смоченной дезактивирующими растворами.

Контроль за эффективностью дезактивации осуществлялся прямым замером мощности дозы гамма-излучения и методом "мазка". В результате дезактивации уровни загрязнения поверхностей помещений и оборудования в основном были снижены до нормативных требований.

На I и II блоках работы по дезактивации были завершены в начале третьего квартала 1986 г.

Продолжаются дезактивационные работы на III блоке, что ведет к дальнейшему улучшению радиационной обстановки на действующих блоках.

Завершается съем загрязненной мягкой кровли на крышах зданий блоков. При этом успешно используются специальные клеющие средства, которые дистанционно наносятся на загрязненные участки кровли и затем снимаются с помощью кранов. В результате выполнения части запланированных работ мощность дозы в машинном зале III блока к концу июля 1987 г. резко снижена и составила 7 — 50 мР/ч.

В настоящее время после завершения сооружения объекта Укрытие и проведения комплекса работ по дезактивации территории станции на I и II блоках радиационная обстановка окончательно стабилизировалась и практически доведена до установленных норм.

Для измерения температуры воды использовались дополнительные термопары, установленные в центральные отверстия тепловыделяющих сборок на левой и правой половинах реактора. Температурный режим графитовой кладки и КМПЦ контролировался штатными температурными каналами.

Заданный температурный режим реактора и КМПЦ обеспечивался включением в работу системы продувки и расхолаживания (СПИР). Графитовая кладка периодически продувалась азотом либо сухим воздухом с влажностью не более 0,5 г/м^3. Контур СУЗ после полного расхолаживания реакторов был обезвожен.

Все необходимое вспомогательное оборудование I и II блоков поддерживалось в состоянии готовности к работе. Системы вентиляции до проведения работ по дезактивации воздуховодов и вентиляционного оборудования и монтажа дополнительной установки по очистке приточного воздуха находились в отключенном состоянии. Система пожаротушения поддерживалась в состоянии готовности к работе в автоматическом режиме. Система дозиметрического контроля I и II блоков была включена в работу в полном объеме. Электрические схемы собственных нужд обеспечивали нормальное электропитание с готовностью принять нагрузку любых механизмов, задействованных в режиме ожидания до пуска. Системы машинного зала поддерживались в законсервированном состоянии. Контроль за состоянием реакторых установок и оборудования блоков вел оперативный персонал АЭС.

Поддерживание систем и оборудования блоков в работоспособном либо законсервированном состоянии обеспечивало возможность в дальнейшем в короткие сроки выполнить восстановительные работы на блоках и ввести их в эксплуатацию.

Длительное пребывание I и II блоков в остановленном состоянии, воздействие радиационного облучения и дезактивирующих веществ потребовали тщательной ревизии и диагностической проверки всего основного и вспомогательного оборудования и систем автоматики, проведения ремонта и комплекса пусконаладочных работ.

Объем работ по подготовке и проведению пуска I и II блоков определялся "Программой комплексного опробования и пуска I и II блоков ЧАЭС" и соответствовал требованиям, предъявляемым к пуску вновь вводимых блоков. В соответствии с этой программой в предпусковой период была проведена поузловая проверка систем блока, включавшая в себя проверку работоспособности арматуры, КИП, мнемосхем, защит, блокировок, сигнализации, АВР, включение системы в работу, доведение качества рабочей среды до эксплуатационных норм, проверку срабатывания систем и механизмов по сигналам аварийных защит. По результатам ремонтно-восстановительных работ и поузловой проверки составлялся акт о готовности системы к пуску блока и общий акт о готовности оборудования, систем, технической документации и персонала ЧАЭС к проведению пуска блока.

В предпусковой период первоочередное внимание уделялось подготовке эксплуатационного персонала. Учитывая наличие вокруг станции зоны с радиоактивным загрязнением, важно было создать условия для проживания персонала. Решение этих вопросов было найдено в организации эксплуатации ЧАЭС по вахтовому методу, суть которого заключается в следующем:

— в период вахтовых (рабочих) дней оперативный и эксплуатационный персонал проживает в вахтовом поселке, расположенном за пределами 30-километровой зоны;

— выходные дни эксплуатационный персонал проводит в городах Киеве и Чернигове, где им предоставлены необходимые условия.

Продолжительность рабочего дня для оперативного персонала принята равной 12 часам (с 8.00 до 20.00 и с 20.00 до 8.00), для эксплуатационного персонала и всех сотрудников станции (как и для всех работающих в 30-километровой зоне) — 10 часам (с 9.00 до 19.00). Продолжительность вахты для оперативного персонала принята равной 5 дням с 7 выходными днями, для всех остальных сотрудников продолжительность вахты и отдыха по 15 дней.

Вахтовый метод в условиях радиационной обстановки в 30-километровой зоне оправдал себя и позволил создать условия как для работы, так и для отдыха персонала.

С окончанием строительства города эксплуатационников — Славутича в 1988 г. будут восстановлены нормальные условия жизни и работы коллектива станции.

Специфические задачи пуска I и II блоков заключались в проверке и уточнении характеристик и режимов работы оборудования после длительной стоянки и в проверке эффективности выполненных мероприятий по повышению безопасности АЭС с реакторами РБМК. Для решения этих задач пуск блоков проводился в три этапа:

1-й этап — формирование активной зоны и проведение физического пуска;

2-й этап — освоение мощности блока на уровне 700 МВт (эл.) с проведением комплексного опробования работы оборудования систем блока;

3-й этап — освоение уточненной номинальной мощности.

Целью мероприятий по повышению безопасности прежде всего было уменьшение парового эффекта реактивности и увеличение быстродействия аварийной защиты. Для достижения этих целей и с учетом особенностей реактора перед физическим пуском 50% стержней были введены в активную зону в положение 1,4 м по указателю положения (УП), количество укороченных стержней-поглотителей увеличено до 32, а их перемещение ограничено интервалом от 1,2 до 3,5 м по УП.

В процессе физического пуска вместо части ТВС устанавливались ДП. Оперативный запас реактивности при эксплуатации реакторов установлен равным 43 — 48 стержням. Сформированная начальная загрузка реактора I блока содержала 1648 ТВС (из них 124 свежие), 30 ДП, 14 ТК с водой и 1 заглушенную ячейку, а начальная загрузка реактора II блока — 1610 ТВС (из них 313 свежих), 81 ДП, 2 ТК с водой. Количество ДП в реакторе II блока было увеличено в целях дальнейшего уменьшения значения парового коэффициента реактивности.

По окончании физического пуска реакторов были подготовлены и включены в работу в соответствии с требованиями технологического регламента системы и оборудование блоков и начато проведение работ по настройке и испытаниям оборудования, проверке технологических параметров, освоению мощности 700 МВт (эл.) и поочередному пробному выходу ТГ-1 (3), ТГ-2 (4) на мощность 500 МВт (эл.).

В процессе подъема мощности и комплексных испытаний на уровне 700 МВт (эл.) на I и II блоках существенных недостатков в работе основного и вспомогательного оборудования не обнаружено. Качество воды КМПЦ и КОСУЗ, питательной воды соответствовало нормам. Регуляторы блока обеспечивали поддержание заданных регламентом значений технологических параметров. Характеристики активной зоны реактора оставались удовлетворительными.

Распределение энерговыделения в рабочих каналах измерялось с достаточной достоверностью и легко поддерживалось в пределах уставок отклонений. Радиохимический состав теплоносителя КМПЦ соответствовал требованиям регламента. Радиационная обстановка в помещениях позволила ввести в эксплуатацию I и II блоки.

Полученные при комплексном опробовании блоков на мощности 700 МВт (эл.) данные позволили сделать вывод о возможности подъема мощности I и II блоков ЧАЭС до проектного номинального значения 1000 МВт (эл.) и выполнения 3-го этапа пуска блоков. Подъем мощности производился ступенями по 10%N с проведением комплексного опробования оборудования блоков на каждой ступени.

Значения технологических параметров и физические характеристики реактора при работе на номинальной мощности удовлетворяют требованиям технологического регламента. Нарушений в работе оборудования, препятствующих эксплуатации I и II блоков на номинальном уровне мощности, не было выявлено. На основании этого I и II блоки ЧАЭС работают на номинальной проектной мощности.

Управление реакторами I и II блоков после осуществления мероприятий по повышению безопасности не вызывало трудностей. Этому способствовало то, что на ЧАЭС в технологическую цепочку подготовки и выбора перегрузок включена ЭВМ ЕС-1035.

Результаты пуска I и II блоков ЧАЭС могут быть суммированы следующим образом.

1. Принятые меры значительно повысили безопасность на I и II блоках ЧАЭС. Возросла начальная скорость ввода отрицательной реактивности до $\geqslant 0{,}5\ \beta$/с. По результатам измерений скоростной эффективности стержней аварийной защиты принято решение несколько уменьшить глубину

погружения регулирующих стержней в активную зону — с 1 — 2 до 0,7 м, что снижает деформацию высотного поля и в то же время сохраняет приемлемые скоростные характеристики АЗ.

Уменьшилось значение парового коэффициента реактивности. К настоящему времени количество ДП на I блоке также увеличено до 80.

2. В текущем году начинается загрузка реакторов ТВС с обогащением 2,4%, что позволит уменьшить значение парового коэффициента реактивности до нуля.

4. ЛИКВИДАЦИЯ ПОСЛЕДСТВИЙ АВАРИИ, ОБУСЛОВЛЕННЫХ РАДИОАКТИВНЫМ ЗАГРЯЗНЕНИЕМ ПРИРОДНЫХ СРЕД

Радиоактивное загрязнение площадки ЧАЭС, прилегающего региона привело к необходимости проведения комплекса мероприятий по ликвидации последствий аварии.

В материалах [1] содержится описание этого комплекса. Основное внимание в них было уделено срочным неотложным мероприятиям (оценка радиационной обстановки, эвакуация населения, изоляция источника радиоактивных выбросов, активная дезактивация и др.).

После выполнения этих мероприятий и стабилизации обстановки в регионе ЧАЭС появилась возможность перейти к регулярной работе по ликвидации последствий аварии и мероприятиям долгосрочного характера.

4.1. Контроль за радиоактивным загрязнением природных сред в регионе ЧАЭС

Регулярный сбор и представление информации о радиационной обстановке в зоне аварии и на территории всей страны были начаты 26 апреля 1986 г. Контроль за радиоактивным загрязнением природных сред осуществлялся и продолжает осуществляться организациями Госкомгидромета СССР совместно с организациями Минздрава, Госагропрома, АН СССР, Министерства обороны СССР, ГКАЭ СССР и др.

С учетом масштабов аварии действовавшая система радиационного контроля была значительно расширена путем привлечения дополнительных групп специалистов (несколько тысяч человек) и техники. В частности, для контроля за радиоактивностью атмосферы были привлечены авиационные подразделения Министерства обороны СССР.

Сбор данных осуществлялся на действующих постоянных или временных станциях, постах наблюдения, самолетами или вертолетами радиационной разведки, экспедициями, передвижными группами и т.п.

Эти данные включают:

результаты гамма- и бета-радиометрии и спектрометрии загрязненных площадей;

анализ проб воздуха, воды, почвы, растительной массы;

анализ проб радиоактивных выпадений.

После решения неотложных краткосрочных задач система контроля за радиоактивным загрязнением трансформировалась в постоянно действующую систему мониторинга в районе радиоактивного загрязнения. Эта система дополнилась проведением научных исследований по радиоэкологии и миграции радиоактивных веществ в природных средах (включая пищевые цепочки), по прогнозам изменения радиоактивного загрязнения, дозам облучения объектов живой природы и населения. В рамках Академии наук, Минздрава, Госагропрома и других ведомств страны созданы специальные научные организации по выполнению долгосрочных научных программ в регионе ЧАЭС.

Программа научно-исследовательских работ по контролю радиационной обстановки в зоне ЧАЭС включала:

1) создание комплексного метода радиационного контроля земной поверхности, объединяющего:

— спектрометрическую и дозиметрическую разведки (полупроводниковая полевая спектрометрия);

— радиохимические и ядерно-физические методы анализа проб почвы, воды, аэрозолей;

— установление корреляционных соотношений между содержаниями плутония и стронция-90 и содержанием церия-144, определяемого по гамма-спектру образцов;

2) создание новых методов регистрации плутония и строн-

ция на основе низкофоновых установок с жидкими сцинтилляторами;

3) разработку метода непрерывного дистанционного контроля радиационной обстановки по измерению свечения воздуха в УФ области спектра;

4) создание банка оцененных данных по почвенным загрязнениям, а также системы экспертной оценки, обработки и анализа получаемой информации о радиационной обстановке.

Информация о радиоактивном загрязнении легла в основу принятия решений по обеспечению радиационной безопасности населения и хозяйственной деятельности на загрязненной территории. К этим решениям относятся:

— эвакуация и реэвакуация населения ряда населенных пунктов загрязненной зоны;

— дезактивация территории, зданий и т.п.;

— выделение зон отчуждения или ограничения производственной деятельности;

— защита гидросферы от радиоактивного загрязнения.

Данные о радиоактивном загрязнении использовались также для уточнения общего количества, динамики и радионуклидного состава радиоактивных выбросов аварийного блока ЧАЭС.

4.2. Радиоактивное загрязнение местности в ближней зоне, примыкающей к ЧАЭС

Ближняя и дальняя зоны радиоактивных загрязнений сформировались с 26 апреля по 7 — 8 мая 1986 г. Особенности их формирования определялись динамикой и высотой выброса, метеорологическими условиями в регионе ЧАЭС и в отдаленных районах по направлению распространения загрязненных воздушных масс.

Данные о радиоактивных выбросах и радиоактивном загрязнении, полученные в первые несколько месяцев после аварии, приведены в докладе советской делегации на совещании экспертов МАГАТЭ в августе 1986 г. [1].

За прошедший с того времени год проведен большой объем работ по уточнению и детализации картины радиоактивного загрязнения в ближней и дальней зонах радиоактивных выпадений. Создан и постоянно пополняется банк данных о радио-

активном загрязнении в регионе ЧАЭС. Банк содержит также данные о радиационном фоне до 26 апреля 1986 г.

Радиационный фон до аварии в регионе ЧАЭС характеризовали следующие данные:

— мощность экспозиционной дозы гамма-излучения 0,01 — 0,015 мР/ч;

— плотность загрязнения цезием-137 и стронцием-90 (за счет глобальных выпадений от ядерных испытаний) 0,1 и 0,07 Ки/км^2 соответственно.

Радиоактивные выпадения в результате аварии на ЧАЭС привели к загрязнению природной среды. Некоторое представление об этом дают рис. 4.1 и 4.6 (см. также [1]).

На загрязненной территории можно выделить три ветви радиоактивного следа — северную, южную и западную.

После прекращения радиоактивных выбросов аварийным блоком изменение радиоактивного загрязнения определялось в основном радиоактивным распадом, ветровым переносом, смывом и переносом дождевыми и паводковыми водами (после таяния снегов), диффузией в почву и т.п.

К осени 1986 г. из-за распада относительно короткоживущих радионуклидов ($T_{1/2} \leqslant$ несколько месяцев) первый процесс стал играть меньшую роль. Окончательное формирование радиоактивного загрязнения природных сред закончится практически в течение 1988 г.

В ближней зоне ЧАЭС в период после аварии регулярно осуществляются измерения гамма-полей, аэрогамма-съемка загрязненной местности. Карта распределения уровней радиации на местности на 1 мая 1987 г. приведена на рис. 4.1.

По данным распределения мощностей доз гамма-излучения на местности в различные времена после аварии оценено суммарное количество радиоактивных продуктов, выпавших на ближнем следе, и изменение этого количества во времени за счет радиоактивного распада и других факторов. На рис. 4.2 показано изменение во времени мощности дозы гамма-излучения суммарного количества радиоактивных продуктов на ближнем следе, которое хорошо согласуется с аналогичными данными, полученными на основе анализа радиоактивных продуктов почвенных проб, отобранных в западных и северных областях ближнего следа [3].

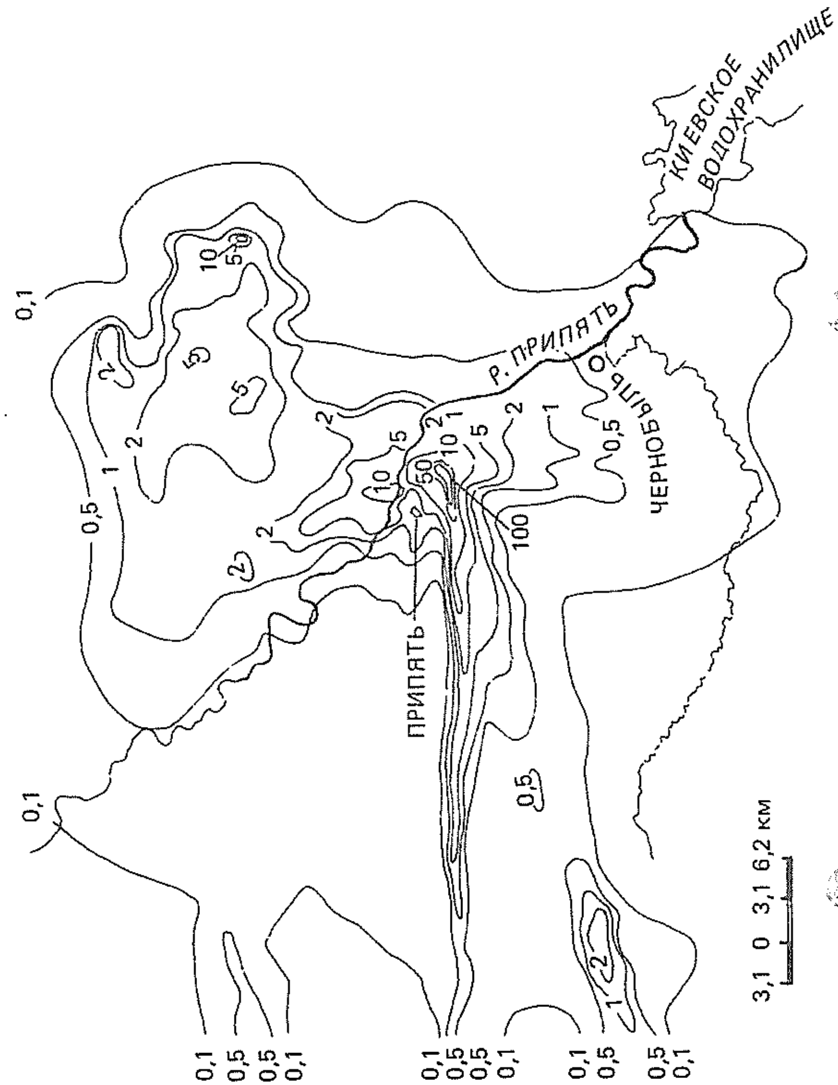

РИС. 4. КАРТА ГАММА-ПОЛЯ (мР/ч) НА 1 МАЯ 198

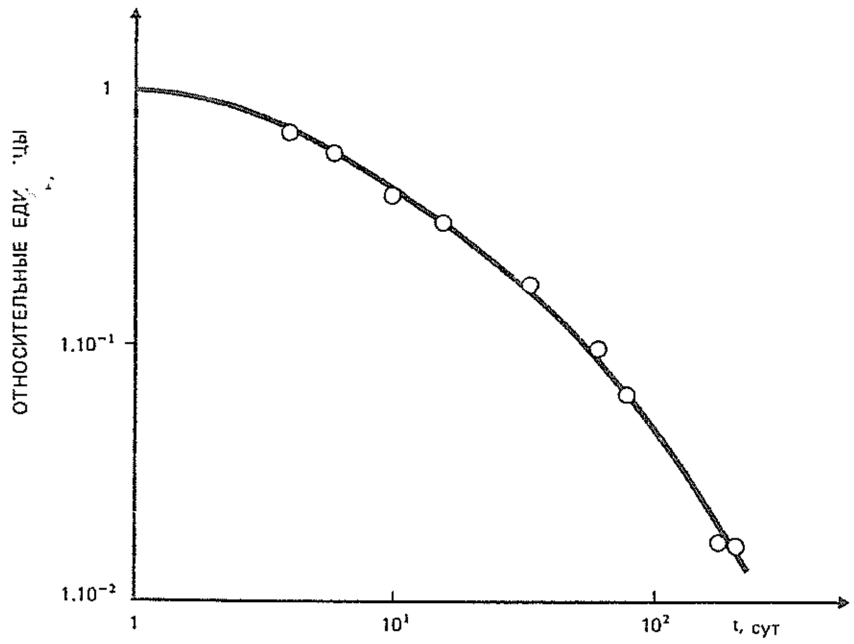

РИС. 4.2. ИЗМЕНЕНИЕ МОЩНОСТИ ДОЗЫ ГАММА-ИЗЛУЧЕНИЯ РАДИОАКТИВНЫХ ПРОДУКТОВ НА БЛИЖНЕМ СЛЕДЕ ВО ВРЕМЕНИ ПО ДАННЫМ АЭРОГАММА-СЪЕМКИ

Суммарное количество гамма-радиоактивных продуктов на ближнем следе через год после аварии уменьшилось примерно в 55 раз и составляет $2{,}7.10^6$ Р.м^2/ч на 1 мая 1987 г.

Миграция радионуклидов в почву приводит к уменьшению мощности P_γ экспозиционной дозы гамма-излучения от радиоактивных выпадений. Как показывают наблюдения, к осени 1986 г. на типичных легких дерново-подзолистых и супесчаных почвах глубина миграции достигала 0,6 — 1,2 см. Это должно приводить к ослаблению мощности дозы P_γ на высоте 1 м в 1,5 — 2,5 раза. Этот эффект подтверждался прямыми измерениями на местности [2].

Площади, ограниченные изоуровнями мощностей доз, на 1 мая 1987 г. равны: Р = 1,0 мР/ч — 500 км²; Р = 2,0 мР/ч — 280 км²; Р = 5,0 мР/ч — 70 км²; Р = 10 мР/ч — 20 км²; Р = = 20 мР/ч — 8,0 км²; Р = 50 мР/ч — 3,0 км². Через год после аварии площади, ограниченные указанными изоуровнями мощностей доз, уменьшились от 50 до 150 раз. На рис. 4.3 показана взаимосвязь площадей радиоактивных выпадений на ближнем следе и изоуровней мощности дозы, ограничивающих площадь, и с учетом приведения данных аэрогаммасъемок, проведенных в различные времена после аварии, к одному времени, а именно к 29 мая 1986 г. согласно данным рис. 4.2 [5].

Радиоизотопный состав ближних выпадений определяется радионуклидами, приведенными в [3], за исключением короткоживущих. После распада относительно короткоживущих радионуклидов наибольшую радиобиологическую значимость имеют такие радионуклиды, как цезий-134, -137, а также изотопы стронция и плутония. Построены карты загрязнения этими радионуклидами, а также цирконием-95, ниобием-95, рутением-103, лантаном-140 и др.

Исследования изотопного состава на радиоактивном следе показали, что существенное фракционирование радиоизотопов наблюдается на расстояниях, превышающих 15 — 30 км; например, на северном следе обнаружено значительное обогащение радиоактивных выпадений цезием-137 (в 10 и более раз).

Загрязнение цезием имеет "пятнистый" характер. Это обусловлено как динамикой выброса, так и неравномерностью выпадения дождей в зонах прохождения радиоактивного облака. Имеются участки с плотностью загрязнения по цезию-137 (+ цезий-134) до 20 — 30 Ки/км², а в отдельных местах порядка 80 Ки/км².

Количество цезия-137, выпавшего на ближнем следе, по данным аэроспектральной съемки и анализа проб почв, составляет ~ 0,2 МКи. Эта величина несколько меньше, чем та,

которая приведена ранее [3]. Она получена путем использования данных отношений плотности радиоактивных выпадений данного радионуклида σ (Ки/км²) к мощности дозы Р (мР/ч) для различных секторов на ближнем следе.

На ближнем следе плотность загрязнения плутонием достигает 0,1 — 1,0 Ки/км². В непосредственной близости к промышленной площадке местами наблюдается плотность, превы-

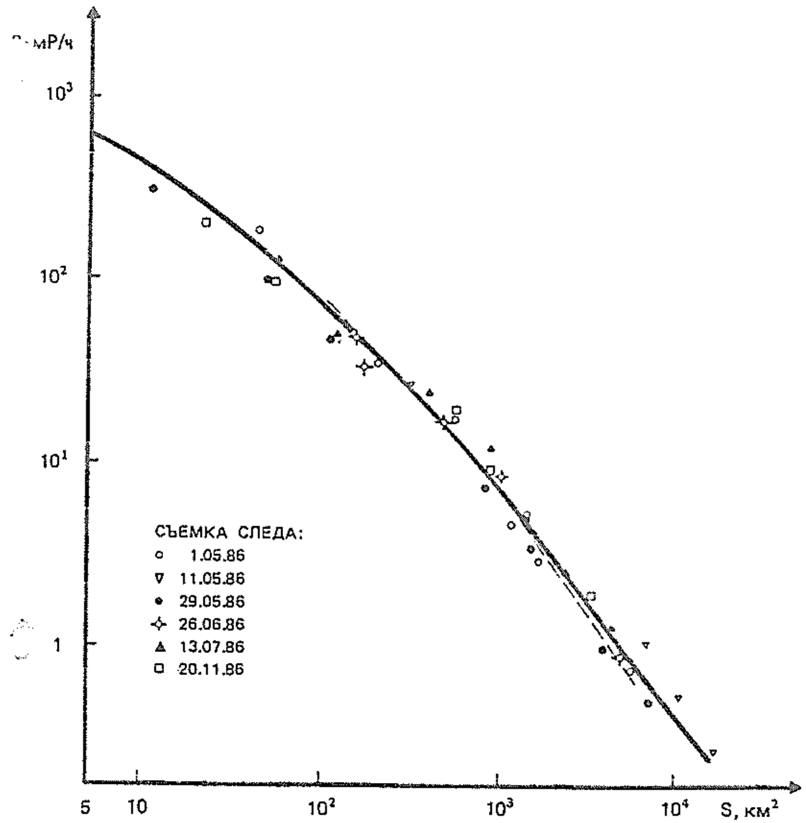

РИС. 4.3. ВЗАИМОСВЯЗЬ ПЛОЩАДИ РАДИОАКТИВНЫХ ВЫПАДЕНИЙ НА БЛИЖНЕМ СЛЕДЕ И ИЗОУРОВНЯ МОЩНОСТИ ДОЗЫ (ДАННЫЕ ПРИВЕДЕНЫ НА 29.05.86 г.): ——— – СРЕДНЯЯ ПО ИЗМЕРЕНИЯМ, – – – – РАССЧИТАННАЯ

щающая 10 Ки/км². Уровень загрязнения местности плутонием относительно быстро убывает с расстоянием. Следует отметить, что плутоний, стронций и ряд других долгоживущих радионуклидов в настоящее время находятся в составе частиц топлива. Это необходимо учитывать при анализе экологических и биологических последствий, а также при рассмотрении миграции этих радионуклидов в различных средах.

4.3. Метеорологические данные о направлениях ветра в период аварии и динамика выхода радиоактивных продуктов, выпавших на ближнем следе

Метеорологическая информация в период основного выхода радиоактивных продуктов в атмосферу после аварии включала данные шаропилотных наблюдений о направлениях и скоростях ветра в аэропортах городов Киев (Жуляны, Борисполь), Мозырь, Гомель, Чернигов и данные радиозондирования в г. Киеве с 26 апреля по 1 мая 1986 г. По специальной программе из первичных наблюдений вычислялись средние направления и скорости ветра в слое от поверхности земли до заданной высоты. На рис. 4.4 представлены рассчитанные значения средней скорости и направлений ветра в слоях 0 — 500 и 0 — 1000 м за время наблюдений в течение пяти суток после аварии. Эти данные использовались для расчета переноса частиц в атмосфере в слоях 0 — 500 и 0 — 1000 м.

Анализ метеорологических данных о направлениях ветра, представленных на рис. 4.4, показывает, что в течение пяти суток с 26 по 30 апреля 1986 г. направление переноса воздушных частиц в слое от уровня земли до 1000 м изменилось на 360°, фактически описав полный круг. Учет отмеченных метеорологических особенностей переноса радиоактивных веществ из зоны реактора в сопоставлении с характером распределения ближних радиоактивных выпадений позволил получить дополнительные характеристики динамики выброса [5].

Данные о ежесуточном выбросе радиоактивных веществ в атмосферу из зоны реактора приведены в [1]. Относительное изменение этого выброса в первые пять суток хорошо аппроксимируется зависимостью

$$Q(t) = 0{,}32e^{-0{,}28t}, \quad t = 0,\ 1, \ldots, 4 \text{ сут.}$$

Формирование радиоактивных выпадений в ближней зоне закончилось в первые 4 — 5 суток. В последующие дни, как показала проводившаяся регулярно аэрогамма-съемка радиоактивного следа, суммарное количество гамма-радиоактивных продуктов на следе монотонно убывало согласно распаду суммы радионуклидов (см. рис. 4.2). Ежесуточный выброс радиоактивных веществ в атмосферу, аппроксимируемый приведенной экспоненциальной зависимостью, включает весь спектр радиоактивных частиц, которые обусловили как ближние и региональные, так и глобальные выпадения.

Штриховкой на рис. 4.4 показаны секторы (в градусах) направления ветра в слоях от уровня земли до 500 и 1000 м, в которых происходили перенос и выпадение частиц из струи в разные временные интервалы после аварии. В соответствии с этим на карте распределения уровней радиации в ближней зоне (до 80 км) были выделены секторы (230 — 320°, 320 — 20°, 20 — 90°, 90 — 220°), в которых оценены суммарные количества гамма-радиоактивных продуктов (мР.км²/ч) на 29 мая 1986 г. [5]. Суммарное количество гамма-радиоактивности на следе на указанную дату составляет $4{,}4 \cdot 10^4$ мР.км²/ч. На рис. 4.5 в виде гистограммы приведены данные о почасовых выпадениях гамма-радиоактивных веществ на ближнем следе. Сплошной линией показано относительное изменение почасовых выпадений с 26 апреля по 1 мая 1986 г.

Таким же способом, используя отношения плотности радиоактивных выпадений данного радионуклида σ (Ки/км²) к мощности дозы P (мР/ч) для различных секторов (запад, север, юг), приведенные в [3], были определены выпадения отдельных радионуклидов в последующие дни после аварии. В табл. 4.1 приведены результаты расчетов относительного выброса и выпадений суммы гамма-радиоактивных веществ и отдельных радионуклидов на ближнем следе за первые пять суток.

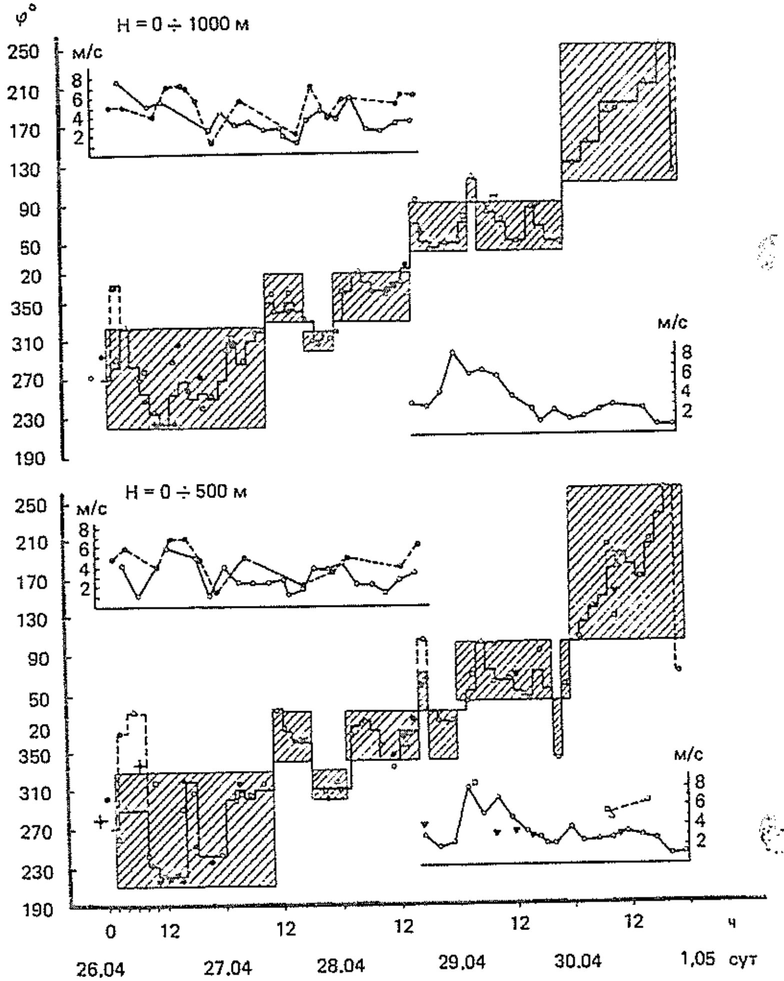

РИС. 4.4. СРЕДНИЕ В СЛОЕ 0 – 500 И 0 – 1000 м ЗНАЧЕНИЯ НАПРАВЛЕНИЯ И СКОРОСТИ ВЕТРА С 26.04 ПО 1.05.86 г. В РАЙОНЕ, ПРИМЫКАЮЩЕМ К ЧАЭС. ✦ – КИЕВ (РАДИОЗОНД), ○ – КИЕВ, АЭРОПОРТ: ● – БОРИСПОЛЬ, △ – МОЗЫРЬ, □ – ГОМЕЛЬ, ▼ – ЧЕРНИГОВ

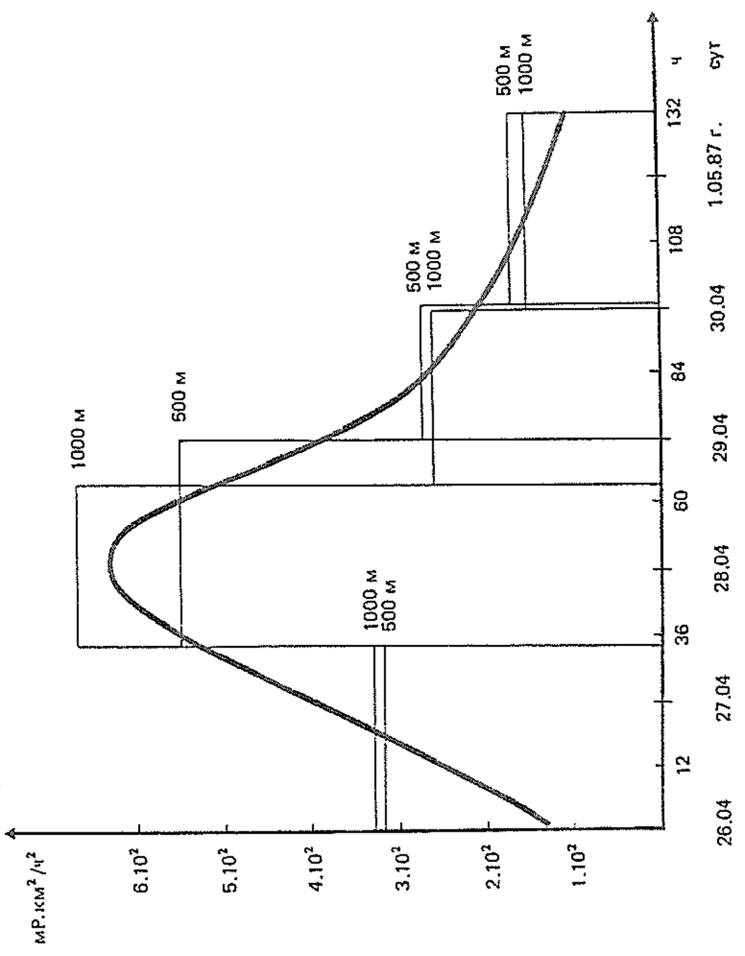

РИС. 4.5. ПОЧАСОВЫЕ ВЫПАДЕНИЯ ГАММА-РАДИОАКТИВНЫХ ВЕЩЕСТВ НА БЛИЖНЕМ СЛЕДЕ

Таблица 4.1. Относительное распределение выброса радиоактивных веществ за первые пять суток и их выпадений на ближнем следе (апрель 1986 г.)

Число	Полный выброс [1]	Выпадение радиоактивных веществ на ближнем следе					
		Сумма радиоактивных веществ			Отдельные радионуклиды		
		H = = 1000 м	H = = 500 м	средн.	церий-144	цезий-137	цезий-134
26	0,32	0,17	0,17	0,17	0,19	0,1	0,09
27	0,24	0,29	0,25	0,28	0,28	0,3	0,31
28	0,19	0,29	0,29	0,29	0,3	0,4	0,42
29	0,14	0,14	0,15	0,14	0,12	0,18	0,15
30	0,11	0,11	0,14	0,13	0,11	0,02	0,03

4.4. Радиоактивное загрязнение местности на территории СССР

В мае 1987 г. проведена повторная аэрогамма-съемка и аэроспектральная съемка территории СССР. Приведенное на рис. 4.6 распределение гамма-поля на территории СССР по изоуровню мощности дозы 0,05 мР/ч на 10 июня 1986 г. [3] в 1987 г. уже нельзя представить в виде замкнутых изолиний, гамма-поле прослеживается в виде множества отдельных незамкнутых пятен.

Суммарное количество гамма-радиоактивных веществ, выпавших на территории СССР за пределами ближнего следа, по данным аэрогамма-съемки, на конец мая 1987 г. оценивается $(6 \div 9) \cdot 10^6$ Р.м^2/ч по сравнению с $1{,}2 \cdot 10^8$ Р.м^2/ч на начальный период после аварии [3]. Суммарная величина выпадений в ближней и дальней зонах составляет $(9 \div 12) \cdot 10^6$ Р.м^2/ч, или около 4% от суммарного количества радиоактивных продуктов в реакторе на это время.

По данным спектральной гамма-съемки, количество цезия-137 и цезия-134 на территории СССР (в дальней зоне) составляет примерно 0,6 МКи. Суммарное количество цезия-137 в ближней и дальней зонах на территории СССР оценивается

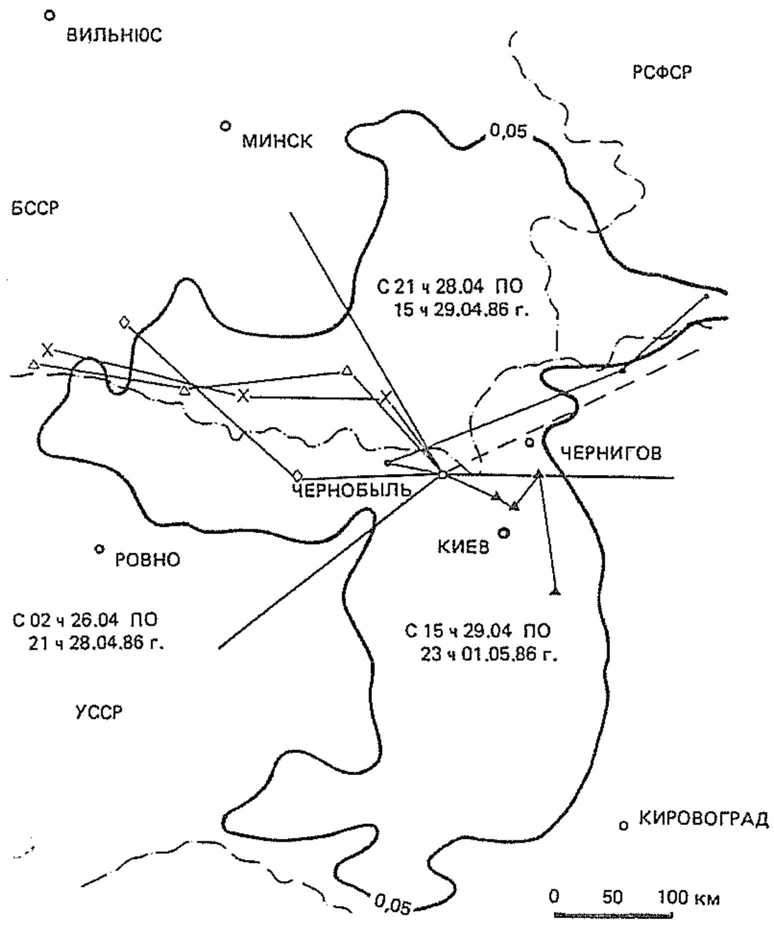

РИС. 4.6. РАСПРЕДЕЛЕНИЕ ГАММА-ПОЛЯ НА ТЕРРИТОРИИ СССР ПО ИЗОУРОВНЮ МОЩНОСТИ ДОЗЫ 0,05 мР/ч НА 10 ИЮНЯ 1986 г. И СЕКТОРЫ РАСПРОСТРАНЕНИЯ ТРАЕКТОРИЙ ЧАСТИЦ НА УРОВНЕ 925 мб (X——X – ОТ 3 ч 26.04.86 г., ◇——◇ – ОТ 15 ч 26.04.86 г., △——△ – ОТ 3 ч 27.04.86 г., •——• – ОТ 15 ч 27.04.86 г., ▲——▲ – ОТ 3 ч 29.04.86 г., – – – – ОТ 15 ч 29.04.86 г.)

менее 0,8 МКи, или около 10% от образовавшегося количества цезия-137.

4.5. Радиоактивное загрязнение рек и водохранилищ

Основные источники загрязнения рек и водохранилищ, а также уровни концентраций радионуклидов в воде с момента аварии на ЧАЭС до июля 1986 г. приведены в [3].

В рассматриваемый период времени радиоактивность поверхностных вод в основном определялась изотопами цезия и стронция. Изотопы рутения, церия, циркония-95 и ниобия-95 в пробах воды обнаруживались в малых концентрациях, не постоянно и в основном сорбированными на взвешенных частицах.

В табл. 4.2 приведены результаты наблюдений изменения концентрации изотопов цезия в пробах воды Киевского водохранилища, а также рек Припять и Днепр с июля 1986 г. по май 1987 г., из которых следует, что концентрации цезия в воде за указанный период снизились более чем в 20 раз. Особенно резкое снижение концентраций цезия наблюдалось в летне-осенний период 1986 г. В пробах воды, отобранных осенью, до 70% активности радионуклидов цезия находилось в сорбированном состоянии на взвешенных частицах. Содержание взвеси в воде и как следствие увеличение радиоактивности, таким образом, связано с гидрометеоусловиями исследуемых водоемов. Наиболее очевидно такая зависимость наблюдалась в Киевском водохранилище, в которое разгружают взвесь реки Припять и Днепр.

В осенний период на водохранилище наблюдается увеличение волнения и как следствие вторичное загрязнение воды в результате ветрового взмучивания верхнего слоя загрязненных илов. Экспериментальные исследования, подтверждаемые расчетами, показали, что даже полное взмучивание обменного слоя загрязненных илов, которое может иметь место при сильных штормах, не увеличит дозовые нагрузки, определяемые суммой всех радионуклидов, в предположении использования воды для питья, более чем до 5 — 10% от принятых в СССР норм.

В течение зимы 1986/87 г. в условиях устойчивой мороз-

Таблица 4.2. Средние концентрации цезия (10^{-11} Ки/л) в пробах воды Киевского водохранилища, а также рек Припять и Днепр, отобранных в июле 1986 г. — мае 1987 г.

Водный объект	Июль 1986 г.		Октябрь 1986 г.		Апрель — май 1987 г.	
	цезий-137	цезий-134	цезий-137	цезий-134	цезий-137	цезий-134
Киевское водохранилище	(20 — 50)	(10 — 20)	(1 — 3)	(0,5 — 1,5)	(0,4 — 1,2)	(0,2 — 0,6)
р. Припять (ниже Чернобыля)	(40 — 50)	(15 — 25)	(2 — 5)	(1 — 2)	(2 — 5)	(1 — 2)
р. Днепр (с. Теремцы)	(1 — 1,4)	(0,4 — 0,6)	(0,5 — 0,6)	(0,2 — 0,3)	(0,4 — 0,6)	(0,2 — 0,3)

ной погоды и отсутствия существенных оттепелей. радиоактивность вод Днепровского каскада изменялась незначительно. К концу зимы суммарная бета-активность воды в Киевском и Кременчугском водохранилищах приближалась к $(1 \div 2).10^{-11}$ Ки/л. Определяющий вклад в загрязненность воды вносили цезий-137 и стронций-90, концентрации которых в воде составляли $(1 \div 4).10^{-11}$ и $(0{,}1 \div 4).10^{-11}$ Ки/л соответственно.

В условиях слабой проточности водохранилищ в зимние месяцы слабо изменялся уровень загрязненности донных грунтов. Уменьшения происходили главным образом за счет распада циркония-95, ниобия-95, а также изотопов рутения и церия.

При подготовке к половодью весной 1987 г. на реках и водотоках, протекающих по загрязненной территории, с целью уменьшения смыва радионуклидов было сооружено более 100 защитных и фильтрующих дамб.

Половодье весной 1987 г. носило плавный, растянутый характер без резких паводочных волн, в результате чего заметного повышения концентраций радионуклидов в реках и водохранилищах не произошло. Концентрации цезия в пробах воды, взятых в конце апреля — мае 1987 г. (см. табл. 4.2), оставались на уровне концентраций в осенних пробах либо немного уменьшились.

В табл. 4.3 приведены усредненные концентрации цезия-137 и цезия-134, измеренные осенью 1986 г. и весной 1987 г. на р. Днепре и на основных его притоках, протекающих по территории БССР.

Из таблицы видно, что концентрации значительно ниже ПДК. В осенний период до 70% цезия переносилось взвесью. В период стокообразующих дождей (31 августа и 1 сентября) количество цезия, сорбированного на взвесях, возрастало до 80 — 90%, что связано с процессами эрозии загрязненных водосборов.

Весной 1987 г. уровни загрязнения указанных рек практически не изменились. Соотношение концентраций цезий-137/стронций-90 колебалось в различные периоды в диапазоне $1 \div 10$.

Таблица 4.3. Сезонные концентрации изотопов цезия (10^{-11} Ки/л) в водах р. Днепр и основных его притоков на территории БССР

Река	Осень 1986 г.		Весна 1987 г.	
	цезий-137	цезий-134	цезий-137	цезий-134
р. Днепр (на участке Жлобин — Лоев)	(0,5 — 0,8)	—	(0,5 — 0,8)	—
р. Сож (в районе г. Гомель)	(5 — 12)	(2 — 6)	(5 — 7)	(2 — 3)
р. Беседь (с. Светиловичи)	(2 — 10)	(1 — 3)	(3 — 10)	(2 — 3)
р. Припять (в районе г. Мозырь)	(0,5 — 1)	—	(1 — 1,4)	—

4.6. Медико-санитарные мероприятия

Первоочередные мероприятия по защите здоровья лиц, участвовавших в противоаварийных работах, и населения зоны загрязнения описаны в [1].

Здесь дается общая характеристика медико-санитарных мероприятий, проведенных до середины 1987 г.

Осуществление широкомасштабных санитарно-гигиенических мероприятий, направленных на обеспечение радиационной безопасности населения загрязненных зон, было начато в первые дни после аварии. В них участвовали в дополнение к существовавшим на местах медицинским учреждениям до 400 специализированных бригад (медики, дозиметристы и др.), около 5 тыс. медицинских работников, включая студентов и выпускников медицинских вузов.

В результате проведенных мероприятий:

— всеми видами медицинского обследования был охвачен почти 1 млн. человек, из них 700 тыс. (включая 216 тыс. детей) с применением углубленных дозиметрических и лабораторных методов исследования; в стационарных условиях обследовано 32 тыс. человек, из них 12,3 тыс. детей;

— йодная профилактика проведена 5,4 млн. человек, из них 1,7 млн. детей;

— разработаны и реализованы рекомендации об организации летней оздоровительной кампании для детей и беременных женщин за пределами зараженных территорий;

— выполнены оценки и прогноз радиационной обстановки в районах радиоактивного загрязнения; на основе полученных данных разработаны рекомендации по мерам защиты здоровья населения, включая эвакуацию;

— создана и реализована комплексная система контроля за уровнем облучения и состоянием здоровья людей, привлеченных к работам по ликвидации последствий аварии, и персонала I и II блоков ЧАЭС, запущенных вновь в эксплуатацию.

Систематически проводилась санитарно-просветительная работа среди населения.

Для ведения регулярной работы по оказанию специализированной медицинской помощи, осуществления последующих наблюдений и исследований создан Всесоюзный научный центр радиационной медицины АМН СССР в г. Киеве.

Принятые меры позволили:

— максимально снизить воздействие радиоактивных веществ, особенно йода-131, на население;

— предотвратить переоблучение лиц, принимавших участие в ликвидации последствий аварии после 27 апреля 1986 г.;

— не допустить возникновения вспышек инфекционных заболеваний и пищевых отравлений в 30-километровой зоне и вне ее;

— обеспечить защиту здоровья персонала I и II блоков ЧАЭС.

Согласно выполненным оценкам, коллективная доза S внешнего облучения населения СССР, обусловленного радиоактивным загрязнением окружающей среды в результате аварии на ЧАЭС, составила около 10^7 чел.-бэр за первый год после аварии. Ожидаемая коллективная доза S^c облучения населения страны будет менее $3{,}3.10^7$ чел.-бэр. Эта оценка получена с учетом всего комплекса осуществленных и намечаемых мероприятий по обеспечению радиационной безопасности населения. Следует отметить, что около 60% значения S^c приходится на внешнее гамма-облучение людей выпавшими на местность радионуклидами после аварийного выброса, около 38% — на внутреннее облучение за счет перорального поступления радионуклидов (в основном цезия-137) и лишь около 2% — на облучение

от облака выброса и внутреннее облучение организма за счет ингаляции.

Среднедушевая ожидаемая доза облучения населения страны составит около 120 мбэр. Это даст прибавку всего около 2% к дозе от естественного радиоактивного фона, равной в среднем примерно 100 мбэр в год. Расчет радиологических последствий аварии на ЧАЭС, основанный на концепции беспороговой линейной зависимости доза — эффект, показывает, что дополнительная смертность от рака может составить лишь около 0,01% от уровня смертности от спонтанного рака. Эта добавка абсолютно необнаружима на фоне флюктуаций естественной онкологической смертности населения.

Также незначительны будут и ожидаемые генетические радиологические последствия.

Принятые меры по контролю за потреблением пищевых продуктов позволили во много раз снизить дозы внутреннего облучения населения. Отметим, что в отсутствие такого контроля дозы внутреннего облучения за счет потребления местных продуктов могут быть в 10 раз выше доз внешнего облучения.

Предполагая, что контроль пищевых продуктов в необходимом объеме будет продолжаться и в будущем, можно принять, что дозы внутреннего облучения населения будут на уровне доз внешнего облучения.

4.7. Мероприятия Госагропрома. Радиоэкологические исследования

В физико-географическом отношении загрязненная территория (ЗТ) расположена в юго-западной части Восточно-Европейской равнины и частично в Припятском Полесье (водосборный бассейн р. Припять), к которому с востока примыкает Приднепровская низменность. В целом рельеф рассматриваемого района равнинный, максимальные высоты не превышают 200 м. Климат умеренно-континентальный: с теплым летом и сравнительно мягкой зимой, среднегодовое количество осадков колеблется в пределах 500 — 650 мм, примерно 2/3 осадков выпадает в теплое время года.

В почвенном покрове южных районов Белоруссии преобладают дерново-подзолистые и торфяно-болотные почвы, в юговосточных районах встречаются дерново-подзолистые, суглинистые и супесчаные почвы.

Район Полесья (южные районы Гомельской области, северные районы Киевской и Житомирской областей) характеризуется широким распространением заболоченных дерново-подзолистых песчаных и супесчаных почв в сочетании с крупными массивами низинных торфяников. Почвы легкого механического состава занимают 58% площади. Все дерново-подзолистые почвы Полесья отличаются невысоким естественным плодородием, как правило, кислые (pH 4,5 ÷ 5,5), слабо обеспечены минеральными питательными веществами (в том числе калием, фосфором, магнием). Расчлененность рельефа небольшая, однако микрорельеф, особенно в Белорусском Полесье, сильно выражен, что совместно с заболоченностью определяет мелкоконтурность с/х угодий. Распаханность территории составляет около 25%. До 50% с/х площадей занято под естественными кормовыми угодьями (злаково-осоковые луга). Такая природная среда сформировала специфический тип с/х производства. В регионе развито молочное и мясное скотоводство (до 60 коров на 100 га). Значительное место занимают посадки картофеля (около 8% площадей), кормовых культур (35 — 40%), зерновых (около 50%), льна-долгунца (до 5%).

Основной массив лесов на ЗТ находится в районах Полесья, где лесистость достигает 70%. Основная доля (63%) видового состава лесов приходится на хвойные породы (сосна), остальная часть — лиственные породы (дуб, граб, береза, ольха).

Южнее Украинского Полесья начинается зона лесостепи (южная ветвь следа), в почвенном покрове которой преобладают оподзоленные черноземы, серые и светло-серые оподзоленные почвы на лессовых отложениях. Преобладающая порода в лесах Украинского Полесья — сосна с примесью березы и дуба, в лесостепных районах — небольшие массивы лесов из дуба, граба, липы.

Из анализа данных, характеризующих распределение ЗТ по видам хозяйственного освоения, видно, что примерно половина загрязненных земель Белоруссии приходится на с/х угодья (41 — 50%) и до 52% — на природные комплексы (леса, болота, водные объекты). На Украине доля ЗТ, приходящаяся на природные ландшафты, уменьшается от 46% на севере (Чернобыльский район) до 10 — 12% в южных районах. В целом примерно половина площади ЗТ приходится на природные комплексы — леса, болота, неудобья, а равнинные открытые участки местности практически полностью заняты агросистемами. Из этого следует вывод, что для оценки и прогноза экологических последствий радиоактивного загрязнения рассматриваемой территории миграционные процессы в природных экосистемах требуют столь же детального изучения и контроля, как и в агросистемах.

Радиоактивному загрязнению подверглась большая часть с/х угодий внутри 30-километровой зоны и примерно 2,0 млн. га за ее пределами (по состоянию на август 1986 г.). Специалистами Госагропрома совместно с другими организациями проводились оценка и прогноз радиологической обстановки в сфере агропромышленного производства, контроль за радиоактивным загрязнением продукции агропромышленного производства и ее потреблением и др.

Полученные результаты анализировались с учетом существующих научных данных по сельскохозяйственной радиологии, радиоэкологии, миграции радиоактивных веществ по пищевой цепи, норм радиационной безопасности и т.п. На основе анализа принимались решения по мерам сохранения природных и материальных ресурсов отрасли, обеспечению качества и предотвращению потерь продукции в связи с ее радиоактивным загрязнением, обеспечению рентабельности с/х производства в зонах умеренного радиоактивного загрязнения при соблюдении требований радиационной безопасности населения.

В зависимости от уровня радиоактивного загрязнения пищевых продуктов принимались решения:

— о полном запрете на их потребление в пищу и использовании в корм скоту или отправке на переработку;

— об изменении технологии их хранения, переработки и пути использования;

— о разрешении на употребление в пищу при непревышении уровня загрязнения, установленного нормами радиационной безопасности.

Вся загрязненная территория была разделена на несколько зон (в первые месяцы по уровню гамма-излучения, а затем по содержанию в почве цезия-137; содержание других радионуклидов в контролируемой Госагропромом зоне не учитывалось, так как их биологическая активность невысока).

При уровне загрязнения выше 40 $Ки/км^2$ по цезию-137 наложен запрет на использование земель для с/х производства. Рекомендовано передать их в Гослесфонд для организации специального заповедника.

В зонах с меньшим уровнем загрязнения допускается с/х производство с теми или иными ограничениями и рекомендуемыми мероприятиями в зависимости от характера и уровня загрязнения. К таким мероприятиям относятся:

— изменение структуры посева с/х культур и направлений животноводства;

— проведение на пахотных, сенокосных и пастбищных угодьях специальных агромелиоративных мероприятий (внесение в почву в повышенных дозах минеральных удобрений, извести, сорбентов (глиняной суспензии, цеолитов в верхний загрязненный слой с последующей запашкой и др.).

Эти мероприятия направлены на снижение перехода радионуклидов из почвы в продуктивную часть урожая.

Об эффектах воздействия радиоактивного загрязнения на природную среду

Эффекты прямого радиационного воздействия на растительные и животные сообщества в виде поражения хвойных лесов и заметных изменений численности почвенной фауны проявились в ограниченной зоне сильного радиоактивного загрязнения на расстоянии нескольких километров от ЧАЭС.

Наибольшую чувствительность к радиоактивному загрязнению проявили, как и предполагалось, сосновые леса.

Летальные эффекты облучения сосен визуально проявились к концу лета 1986 г. Площадь погибшего лесного массива, примыкающего с запада к промышленной площадке ЧАЭС, составила 400 га [2].

Лиственные древесные породы (представленные в зоне сильного радиоактивного загрязнения вокруг ЧАЭС в основном березой, осиной, дубом) практически не пострадали, поскольку их радиационная устойчивость в 10 раз выше, чем у хвойных пород. Не было обнаружено в этой зоне видимых морфологических изменений и у травянистых растений.

Вне этой небольшой зоны видимых эффектов радиационного поражения флоры и фауны не отмечалось. Прогноз состояния здоровья животных, находящихся на загрязненной территории, удовлетворителен.

Основное внимание в зонах умеренного и слабого радиоактивного загрязнения уделялось изучению миграционных характеристик радионуклидов в природных экосистемах.

В различных физико-географических и ландшафтных условиях была заложена сеть так называемых ландшафтно-геохимических полигонов для мониторинга содержания радионуклидов в компонентах природных экосистем.

Более подробно некоторые вопросы этого раздела рассмотрены в специальных докладах советских специалистов и в работах [2, 3].

5. ВЫВОДЫ ИЗ АВАРИИ НА ЧЕРНОБЫЛЬСКОЙ АЭС И ДАЛЬНЕЙШЕЕ РАЗВИТИЕ ЯДЕРНОЙ ЭНЕРГЕТИКИ В СССР

5.1. Общая часть. Роль ядерной энергетики в Энергетической Программе СССР

Авария на Чернобыльской АЭС заставила специалистов во всем мире еще раз критически рассмотреть как планы по развитию ядерной энергетики, так и меры по обеспечению ее безопасности.

Понятно, что в Советском Союзе уроки Чернобыля изучались с особой тщательностью. Мы пришли к следующим основным выводам.

1. Причины, вызвавшие аварию на ЧАЭС, связаны в первую очередь с ошибками персонала станции, нарушениями им установленных регламентов эксплуатации АЭС. Эти причины сами по себе не имеют специфически ядерного характера и поэтому не могут считаться фатальными для развития ядерной энергетики.

2. В результате анализа аварии не было обнаружено каких-

либо физических явлений, которые ранее не изучались в рамках анализа безопасности теоретически и/или экспериментально. Анализ показал, что безопасность ядерных энергетических установок всех типов может быть повышена хорошо известными физическими и техническими методами и с помощью более тщательного учета человеческого фактора.

3. Анализ последствий аварии в Чернобыле показывает, что хотя ущерб от нее весьма значителен как с точки зрения потерянных человеческих жизней, так и с экономической точки зрения, однако он сравним с ущербом от других проанализированных крупных промышленных и транспортных аварий.

4. Если ядерные энергетические источники будут заменены на традиционные, риск для здоровья населения и окружающей среды значительно возрастет.

5. Причины, потребовавшие развития ядерной энергетики в Советском Союзе, не исчезли, наоборот, со временем они будут становиться все более существенными.

К этим причинам в первую очередь относится необходимость:

— ликвидации географических диспропорций в добыче и потреблении топлива;

— вытеснения нефти и газа в энергетике и оптимизации структуры топливно-энергетического баланса страны;

— экономии трудовых ресурсов.

Кроме того, проводимая модернизация промышленности в европейской части страны требует увеличения производства электроэнергии и, разумеется, более эффективного ее использования. Строительство городов в северной климатической зоне невозможно без производства тепла и электроэнергии с использованием ядерных источников, в противном случае мы не сумеем разрешить транспортные и экологические проблемы.

Таким образом, анализ, которому мы подвергли свое отношение к ядерной энергетике после аварии на ЧАЭС, не привел к изменению наших принципиальных позиций. Мы по-прежнему уверены в необходимости ее развития как для экономики Советского Союза, так и для мировой экономики в целом. Наши планы по вводу ядерных энергетических мощностей существенно не изменяются и будут уточнены в новой редакции Энергетической Программы СССР.

Однако авария в Чернобыле, как и аварии на атомных электростанциях в других странах, показывает, что вопросы безопасности в ядерной энергетике до конца еще не решены. Уроки этих аварий для нас и всего мирового сообщества состоят прежде всего в том, что возникающая в процессе научно-технической революции новая сложная техника требует внимательнейшего отношения к вопросам ее безопасности и надежности, не прощает халатного и неквалифицированного обращения.

В Советском Союзе после аварии на ЧАЭС принят комплекс , как организационного, так и технического характера, направленный на существенное повышение безопасности ядерной энергетики.

В качестве первоочередных были разработаны и осуществлены технические решения, исключающие возможность повторения аварии, подобной чернобыльской, на реакторах типа РБМК.

По результатам анализа аварии разработан комплекс мероприятий по повышению безопасности АЭС всех типов. Этот комплекс включает как реализацию ранее предусмотренных, так и осуществление новых мер, связанных в основном с последними достижениями науки и техники, накопленным опытом эксплуатации, например, по совершенствованию диагностики состояния металла трубопроводов и оборудования, более широкому использованию устройств автоматического управления технологическими процессами. Проводится критический анализ вопросов, связанных с размещением АЭС.

Проведены ревизия и оценка состояния расчетных и экспериментальных исследований по обеспечению безопасности АЭС и выработаны меры по их расширению, совершенствованию и интенсификации.

Совершенствуются расчетные программы анализа безопасности поведения АЭС во всевозможных переходных и аварийных режимах, включая непроектные, развиваются моделирующие системы и комплексы.

Расширяются исследования по возможности создания реакторов с пассивными системами безопасности — так называемых реакторов с внутренне присущей безопасностью, активные зоны которых не могут разрушиться ни при каких авариях.

Усиливаются исследования по количественно-вероятностному анализу безопасности, анализу риска от ядерной энергетики, разработке концептуальных и методологических основ оптимизации радиационной безопасности и сравнения радиационной опасности с другими видами опасностей от промышленной деятельности.

Существующая в СССР система надзора и нормативно-технических документов охватывает все основные вопросы обеспечения безопасности АЭС и продолжает совершенствоваться. Под эгидой Госатомэнергонадзора в 1985 г. в СССР создан Сводный перечень и план разработки правил и норм в области атомной энергетики, координирующий и направляющий деятельность всех ведомств по разработке и систематизации соответствующей научно-технической документации.

Существующие нормативные требования, связанные с безопасностью, в основном не нуждаются в пересмотре. Однако их практическая реализация требует более тщательного контроля. Необходимо поднять качество подготовки и переподготовки персонала, усилить контроль со стороны конструкторов и проектировщиков за качеством изготовляемого оборудования, монтажом и проведением пусконаладочных работ и их ответственность за последующую эффективность и безопасность эксплуатируемых АЭС.

В целях повышения уровня руководства и ответственности за развитие ядерной энергетики, улучшения эксплуатации атомных электростанций образовано общесоюзное Министерство атомной энергетики.

Намечен ряд мероприятий по усилению государственного надзора за безопасностью в ядерной энергетике. Принимаются меры по повышению ответственности персонала за качество эксплуатации АЭС.

5.2. Меры по повышению безопасности АЭС с реакторами РБМК

Важнейшей первоочередной задачей после аварии на ЧАЭС была выработка и осуществление технических решений, которые позволили бы устранить на действующих и строящихся АЭС с реакторами РБМК их особенности, наиболее существенно повлиявшие на развитие аварии и обусловившие увеличение ее масштабов.

За истекший год основные представления о развитии аварии не претерпели существенных изменений.

Продолжались исследования аварийного процесса на интегральных одномерных и трехмерных моделях энергоблока с реактором РБМК-1000. В рамках данных моделей были дополнительно учтены такие факторы, как неоднородность физических характеристик в объеме реактора, возможность двустороннего истечения теплоносителя при повышении давления в активной зоне, транспортное запаздывание в трубопроводах, сжимаемость пароводяной смеси.

Урок Чернобыля заставил считаться с тем, что нарушения регламента могут быть самыми непредсказуемыми. Поэтому в первую очередь необходимо было исключить возможность неконтролируемого разгона реактора при нарушениях технологического регламента.

С этой точки зрения наиболее существенным следует считать, во-первых, положительный паровой коэффициент реактивности α_φ и соответствующий положительный эффект реактивности при обезвоживании активной зоны и, во-вторых, недостаточное быстродействие АЗ при нарушении требования технологического регламента эксплуатации о минимальных запасах реактивности в переходных и стационарных режимах. Расчеты по разным моделям дают сходные результаты, например, показывают быстрое заглушение реактора при регламентном запасе реактивности (15 стержней) на момент сброса АЗ.

Как известно, перед аварией запас реактивности на IV блоке ЧАЭС был существенно меньше регламентного, и не исключена возможность ввода положительной реактивности в первые секунды после нажатия кнопки АЗ-5. Оценки, выполненные по одномерным моделям реактора, показали, что при изменении расчетного исходного высотного поля в пределах различий между показаниями разных датчиков и наложении возможных их погрешностей (до 25% при мощности реактора 200 МВт) значения вводимой положительной реактивности могут меняться в пределах $(0 \div 1{,}5)\beta$.

ростной эффективности АЗ на начальном участке движения стержней сразу после аварии было принято решение зафиксировать стержни в верхнем положении на глубине 1,2 м в активной зоне (см. рис. 5.1,а,б). Однако это привело к искажению высотных полей и необходимости снижения мощности на 10 — 15%. В настоящее время конструкция стержня изменена за счет удлинения соединительного звена между стержнем и "вытеснителем". Это дало возможность уменьшить высотные перекосы энерговыделения (см. рис. 5.1,в).

Оперативный запас реактивности, компенсируемый стержнями СУЗ, увеличен до 43 — 48 стержней ручного регулирования на РБМК-1000 и до 53 — 58 на РБМК-1500, что существенно повышает быстродействие АЗ на начальном участке опускания стержней, находящихся в средней (по высоте) части активной зоны и обладающих большой дифференциальной эффективностью. В результате начальная скорость ввода отрицательной реактивности стержнями по сигналам АЗ составляет не менее 0,5 $\beta_{эф}$/с. Кроме того, в СУЗ внесены и другие усовершенствования, повышающие надежность и безопасность работы реактора:

— увеличено число укороченных стержней-поглотителей до 32 на РБМК-1000 и до 40 на РБМК-1500, вводимых в активную зону снизу;

— реализована схема ввода укороченных стержней-поглотителей в активную зону по сигналам АЗ;

— обеспечена индикация на цифропоказывающем устройстве запаса реактивности в любом состоянии реактора;

— предусматривается автоматическая остановка реактора при снижении запаса реактивности до 30 стержней ручного регулирования.

Штатные сервоприводы стержней СУЗ модернизируются, что позволит уменьшить время полного ввода стержней по сигналам АЗ с 18 — 20 до 10 — 12 с.

Разрабатывается быстрая аварийная защита, внедрение которой на действующих энергоблоках АЭС должно обеспечить ввод отрицательной реактивности до 3$\beta_{эф}$ за 2 — 2,5 с.

При определении первоочередных мероприятий, направленных на снижение α_{φ}, были использованы расчетные данные,

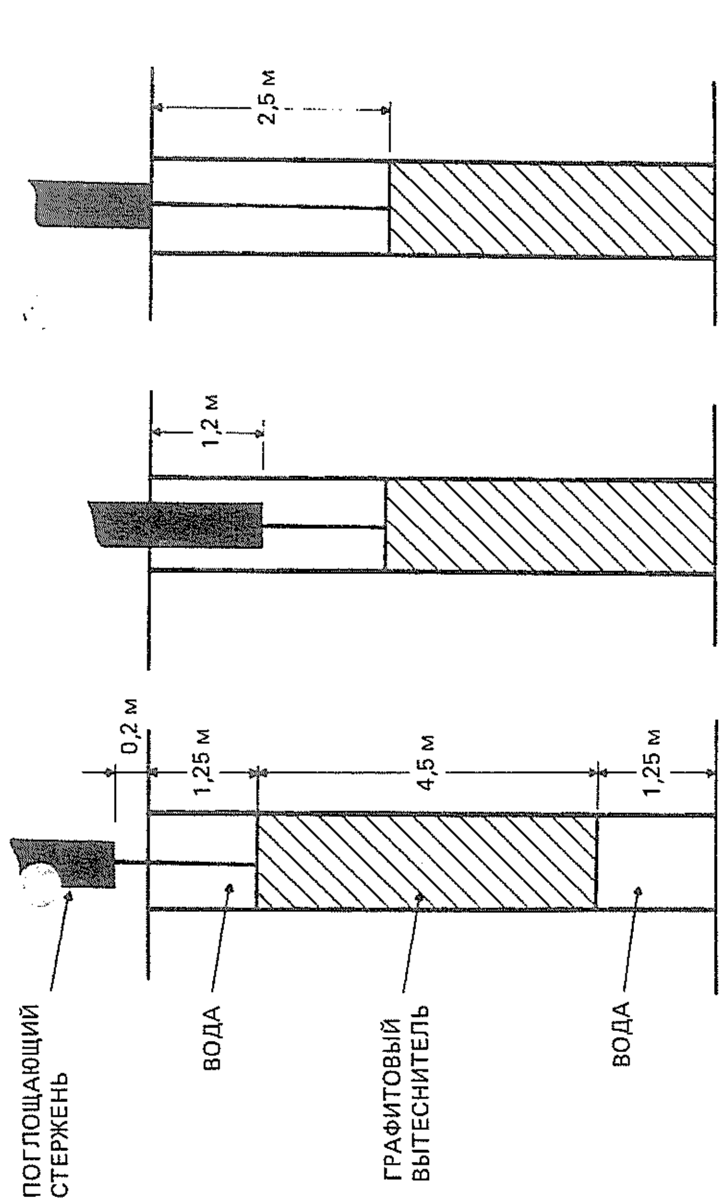

РИС. 5.1. КРАЙНЕЕ ВЕРХНЕЕ ПОЛОЖЕНИЕ РЕГУЛИРУЮЩЕГО СТЕРЖНЯ АЗ ОТНОСИТЕЛЬНО АКТИВНОЙ ЗОНЫ РЕАКТОРА, ПРИНЯТОЕ ДО АВАРИИ (а), ПОСЛЕ АВАРИИ (б) И В НАСТОЯЩЕЕ ВРЕМЯ (в)

уточняющие характер изменения парового коэффициента реактивности при обезвоживании реактора. Влияние дополнительных поглотителей и числа стержней СУЗ, находящихся в активной зоне, на α_φ было уточнено в процессе специальных исследований, проведенных на реакторах Чернобыльской и Смоленской АЭС в октябре — ноябре 1986 г.

По результатам этих исследований было принято решение увеличить количество ДП в активной зоне РБМК-1000 до ~80.

Некоторые типичные фактические значения α_φ, полученные при измерениях на различных блоках АЭС с реакторами РБМК, приведены в табл. 5.1.

Таблица 5.1. Измеренные значения α_φ

Блок	Дата измерения	Число ДП	Запас реактивности (стержней РР)	α_φ
ЛАЭС-I	30.03.87 г.	80	43	1,0 ± 0,2
ЛАЭС-II	13.03.87 г.	79	46	0,8 ± 0,1
ЛАЭС-III	08.05.87 г.	80	42	1,1 ± 0,2
ЧАЭС-II	21.11.86 г.	81	43	1,0 ± 0,2
КАЭС-IV	25.05.87 г.	82	43	0,9 ± 0,4

Кроме снижения α_φ за счет установки дополнительных поглотителей и увеличения минимально допустимого запаса реактивности на стержнях СУЗ, приняты и другие меры, направленные на повышение безопасности. В частности, на блочном щите управления смонтирована дополнительная световая панель автоматики, фиксирующая выведение из работы АЗ реактора по каждому из сигналов. Вмешательство оперативного персонала в работу светового табло (гашение сигнала) полностью исключено.

Еще одним направлением по обеспечению безопасности действующих энергоблоков является существенное расширение внутризонного контроля энерговыделения как по высоте, так и по радиусу активной зоны. С этой целью разработаны проекты и изготовлена партия специальных малогабаритных детекторов энерговыделения, которые устанавливаются на действующих энергоблоках. Разрабатывается проект модернизации существующей системы диагностики и регистрации параметров энергоблока, позволяющий идентифицировать и установить характер развития аварийных ситуаций, а также фиксировать действия оперативного персонала, с выделением усовершенствованной системы в отдельный, не зависимый от имеющейся информационно-вычислительной системы комплекс с надежным автономным питанием. Большое внимание уделяется работе по созданию систем специального ультразвукового и акустико-эмиссионного контроля состояния металла трубопроводов в процессе эксплуатации.

В текущем году существенно расширен перечень проектных и запроектных аварий, которые подлежат анализу при обосновании безопасности энергоблоков с реакторами РБМК, и такой анализ проводится в настоящее время с учетом изменений физических характеристик реактора и других описанных выше изменений.

Дальнейшие мероприятия по повышению безопасности РБМК связаны с увеличением обогащения топлива. Как следует из расчетных и экспериментальных исследований, увеличение обогащения топлива подпитки с 2 до 2,4% позволяет дополнительно уменьшить паровой коэффициент реактивности. Реакторные испытания 146 ТВС обогащением 2,4% проведены на Ленинградской АЭС, в результате принято решение о переводе РБМК-1000 на такое топливо (табл. 5.2). Применение в активной зоне умеренного числа дополнительных поглотителей (до 80) при загрузке топливом обогащением 2,4% позволит снизить α_φ до значения, меньшего $\beta_{эф}$. В то же время при загрузке активной зоны топливом обогащением 2,4% и 80 дополнительных поглотителях глубина выгорания топлива будет даже несколько превышать глубину выгорания топлива 2%-ного обогащения при прежних условиях эксплуатации.

Для строящихся реакторов рассматривается возможность снизить паровой эффект реактивности, уменьшив количество графита в ячейке и реакторе, в частности, за счет срезания ребер графитовых блоков. Уменьшение количества замедлителя

Т а б л и ц а 5.2. Основные характеристики РБМК-1000 при использовании топлива различного обогащения

Характеристика	Начальное обогащение, %											
	2	2,4	3	2	2,4	3	2	2,4	3	2	2,4	3
Число дополнительных поглотителей в реакторе, шт.		0			30			80			110	
$\alpha_\varphi, \beta_{эф}$	4,5	3,2	1,3	3,4	2,1	0,2	1,6	0,2	−1,5	0	−2,6	−5,1
Глубина выгорания топлива, МВт.сут/кг	22,3	28,8	37,6	20,7	27,1	36,0	18,0	24,5	33,4	16,3	22,7	31,6
Максимальная линейная нагрузка, Вт/см	315	350	390	305	340	380	285	320	360	270	300	340

в активной зоне реактора за счет срезания ребер графитовых блоков позволит получить требуемый паровой коэффициент без изменения геометрического шага каналов в активной зоне (размер катета основания отсекаемой треугольной призмы должен составлять 5 — 7 см), снизить начальную мощность свежезагружаемой ТВС, увеличить вовлечение в топливный цикл урана-238. Чувствительность глубины выгорания к количеству графита в ячейке невелика (изменение размера катета основания отсекаемой треугольной призмы на 1 см вблизи оптимального значения эффективного шага приводит к уменьшению глубины выгорания топлива на 0,04%).

5.3. Меры по повышению безопасности АЭС с реакторами ВВЭР

Реакторы типа ВВЭР являются ведущими и перспективными в Энергетической Программе СССР, их уровень безопасности соответствует международным требованиям, но и на АЭС с ВВЭР проводятся мероприятия по повышению безопасности, необходимость которых была вызвана:

— анализом фактического состояния корпусов в отношении их способности сопротивления хрупкому разрушению;

— выводами из аварии на АЭС "Три-Майл Айленд";

— анализом аварии в Чернобыле.

На основании результатов исследования образцов-свидетелей корпусной стали и данных фактического состава металла корпуса, в первую очередь сварного шва, расположенного в районе активной зоны, был предложен ряд конструктивных и технологических изменений, которые позволяют без опасений реализовать проектный ресурс корпусов. Были разработаны меры как по уменьшению вероятности попадания относительно холодной воды на корпус реактора, так и по снижению флюенса нейтронов на корпус.

Ряд серьезных мероприятий по повышению безопасности ВВЭР в условиях возникновения малых и средних течей был осуществлен после аварии на АЭС "Три-Майл Айленд".

В проектах ВВЭР-1000 была реализована система принудительного снижения давления в первом контуре, облегчающая подачу раствора борной кислоты в контур при неблагоприятном развитии аварии с течью. Предусмотрены дополнительные возможности выброса водорода, если бы он скопился под крышкой реактора и в коллекторах парогенераторов, и дренирования U-образного участка в холодной нитке главного циркуляционного трубопровода.

Авария в Чернобыле заставила еще раз обратиться к вопросам о способности активных зон реакторов атомных электростанций за счет внутренних защитных свойств предотвращать повреждение ядерного топлива и о достаточности мер по обеспечению безопасности населения, если такая авария, как плавление активной зоны, с высокой вероятностью исключаемая проектными средствами, все-таки случится.

Для АЭС с реакторами ВВЭР-440 первого поколения принято решение о реконструкции с введением дополнительных пассивных и активных систем охлаждения активной зоны и установкой на трубопроводах большого диаметра акустико-эмиссионных и шумовых систем диагностики.

Для проектов ВВЭР рассматриваются следующие направления повышения их безопасности:

— разработка активных зон с повышенной внутренней безопасностью;

— увеличение надежности основного и вспомогательного оборудования АЭС;

— обеспечение пассивного отвода остаточных тепловыделений от активной зоны в условиях длительного полного обесточивания блока, включая потерю источников надежного питания переменного тока (дизелей);

— оснащение АЭС автоматизированной системой управления технологическим процессом, имеющей повышенную надежность и включающую в себя современные системы диагностики состояния основного оборудования АЭС.

Мероприятия по повышению безопасности ставят целью исключить возможность серьезного повреждения активной зоны из-за плавления топлива или недопустимой скорости выделения энергии во всех аварийных ситуациях с вероятностью 10^{-5} — 10^{-6} 1/(год.реактор), при этом вероятность выхода радиоактивных продуктов за пределы защитных барьеров, с превышением допустимого уровня облучения населения и загрязнения окружающей среды, не должна превышать 10^{-7} 1/(год.реактор).

5.4. Совершенствование подходов к обеспечению безопасности ядерной энергетики

Анализ причин и последствий аварии на ЧАЭС не выявил необходимости пересмотра общей концепции обеспечения безопасности ядерной энергетики, которая как в СССР, так и в других странах базируется на многобарьерной системе изоляции от окружающей среды радиоактивных веществ, находящихся в ядерном реакторе, и комплексе инженерных и организационных мер, обеспечивающих безопасную работу АЭС.

В СССР были интенсифицированы исследования по дальнейшему совершенствованию превентивных мер безопасности и мер по уменьшению и ликвидации последствий аварии. Определен ряд областей приложения усилий для наиболее эффективного повышения безопасности ядерной энергетики. В частности, многочисленные научные исследования, выполненные в период после аварии, показали следующее.

1. Безопасность АЭС должна совершенствоваться на пути оптимизации взаимодействия трех ее главных элементов — технических средств, эксплуатационных процедур и персонала. Здесь в первую очередь необходимо указать, что авария на ЧАЭС выявила недостатки в действовавшей концепции человек — машина, когда при управлении инженерными устройствами предпочтение отдавалось действиям человека-оператора. Анализ аварии показал, что ряд систем безопасности должен функционировать исключительно на основании сигналов технических систем контроля параметров установки, а не по указаниям операторов. Примером реализации такого подхода на реакторах РБМК может служить установленная на них автоматизированная система расчета оперативного запаса реактивности с выдачей сигнала аварийной остановки реактора при уменьшении запаса реактивности ниже заданного уровня.

Другой вывод из аварии на ЧАЭС, касающийся рассматриваемой области, — необходимость более глубокой подготовки операторов АЭС и оснащение их компьютеризованными системами-помощниками, т.е. системами, которые могли бы подсказать оператору возможные причины отклонений в работе установки, вели бы его в поиске причин и средств устранения неисправностей. Одним из средств построения таких систем может быть создание по возможности полных и непротиворечивых банков знаний наиболее опытных операторов, их приемов анализа причинно-следственных зависимостей отклонений реактора от нормального состояния. В СССР проводятся работы в этом направлении. Эти работы сопровождаются также совершенствованием технических средств, снабжающих оператора информацией о работе АЭС и развитии аварийных ситуаций.

В плане оптимизации отношений человек — машина в ядерной энергетике задача по повышению квалификации персонала, совершенствованию методов его обучения смыкается с задачей создания более простых в управлении реакторов, обеспечения оптимальных условий работы операторов.

Сюда в первую очередь относятся:

— вопросы дальнейшей оптимизации распределения ответственности между оператором и техническими средствами в принятии решений по управлению реактором;

— проблема рационального представления оператору текущей информации о функционировании реактора.

И самое главное в долгосрочном плане — повышение уровня внутренне присущей реакторам безопасности, вплоть до создания реакторных систем, физико-химические и конструкционные свойства которых не позволят произойти аварии при любых ошибках оператора, нарушениях регламента эксплуатации или выходе из строя элементов оборудования.

Вопрос оптимизации отношений человек — машина актуален для всех современных технологий. Во многих из них, как и в ядерной энергетике, он все еще далек от своего окончательного решения. Работа в этой области требует значительных материальных и людских затрат как на пути теоретического осмысливания проблемы, так и для проведения экспериментов и создания надежных и полных математических моделей, описывающих функционирование сложных систем. Поэтому важное значение приобретает международная кооперация усилий в этом направлении.

2. Безопасность ядерной энергетики должна совершенствоваться на основе разработки строго научно обоснованных целей и критериев безопасности. В этой области основополагающим является ответ на вопрос о том, какой уровень безопасности является приемлемым. До последнего времени как в СССР, так, по-видимому, и в большинстве других стран подавляющая доля средств, расходуемых на безопасность в ядерной энергетике и в других областях промышленности, затрачивалась на совершенствование технических систем контроля и упреждения аварийных ситуаций. Аварии на ЧАЭС, на АЭС "Три-Майл Айленд" в США, на химических предприятиях (например, в городах Бхопале и Базеле) и т.д. показали, что, несмотря на принятые меры безопасности, всегда может произойти что-то непредвиденное либо из-за серии механических поломок, либо в результате ошибок оператора. Представляется, что один из главных уроков, который следует извлечь из этих аварий, состоит в осознании необходимости оптимизации распределения затрат на предотвращение аварий и ограничение и ликвидацию их последствий. Основой для такой оптимизации может быть определение величины приемлемого уровня безопасности.

Из-за одностороннего внимания к средствам предотвращения аварий в обществе возникает чрезмерная успокоенность относительно возможности крупных аварий и в результате промышленность технически и организационно не всегда оказывается готовой к ограничению и ликвидации последствий крупномасштабных аварий.

Так, при аварии на ЧАЭС оказалось необходимым многие технические и технологические решения принимать уже в ходе ликвидации аварии в экстремальных условиях. Потребовалось проведение срочных и широкомасштабных экспериментальных работ, которые могли и должны были бы быть выполнены заблаговременно. При аварии выявился недостаток измерительных средств, рассчитанных на эксплуатацию в широком диапазоне измеряемых параметров. На начальном этапе работ по ликвидации последствий аварии практически отсутствовали средства, позволяющие дистанционно производить пробоотбор в аварийных условиях, совершать другие необходимые технологические операции. В настоящее время в СССР решается задача о создании специального оборудования для ликвидации крупных аварий.

К этому же кругу вопросов относится и проблема оперативной готовности использования возможностей фундаментальной и прикладной науки для целей ликвидации последствий крупномасштабной аварии. Необходимо указать на важную роль в ликвидации и снижении масштабов аварии информационно-решающей, прогнозирующей и рекомендующей системы, включающей в себя комплекс имитационных моделей: модель экономики региона, математические модели различной степени сложности, позволяющие описать поведение химических, радиоактивных примесей в атмосфере, почве, открытых водоемах и грунтовых водах, миграцию примесей по пищевым цепочкам.

Задача такой информационно-решающей системы — оперативно обеспечить специальные силы по ликвидации аварии информацией о возможных сценариях развития послеаварийной ситуации, о соответствующем влиянии этих сценариев на экологию и экономику региона и, наконец, оперативно представить возможные варианты решений по уменьшению ущерба от аварии (например, по планам эвакуации населения).

Необходимость значительного расширения масштабов исследований и моделирования аварийных ситуаций была выявлена и при анализе эффективности имевшихся на ЧАЭС аварийных планов.

Таким образом, авария на ЧАЭС поставила вопрос о необходимости ускорения широкомасштабных теоретических и экспериментальных исследований, направленных на изучение сценариев серьезных аварий на АЭС. С этой целью:

— усилены исследования по количественно-вероятностному анализу безопасности, анализу риска от ядерной энергетики, разработке концептуальных и методологических основ оптимизации радиационной безопасности и сравнения радиационной опасности с другими видами опасностей от промышленной деятельности;

— проведены ревизия и оценка состояния расчетных и экспериментальных исследований по обеспечению безопасности АЭС и выработаны меры по их расширению, совершенствованию и интенсификации;

— совершенствуются расчетные программы анализа безопасности поведения АЭС во всевозможных переходных и аварийных режимах, включая непроектные, развиваются моделирующие системы и комплексы.

3. Безопасность ядерной энергетики должна совершенствоваться на основе оптимального выбора местоположения АЭС и других промышленных предприятий. Стремление к наибольшей экономичности, к максимальному использованию однажды сделанных вложений в промышленность, транспортные коммуникации, социально-бытовую сферу какого-либо региона вызывает насыщение этого региона различными предприятиями, в том числе и АЭС, без должного изучения их взаимодействия. И может случиться так, что авария на одном из них и не была бы значительной по последствиям, если бы не ее воздействие на соседний объект с возможным многократным усилением поражающих факторов. Эффект от возможного взаимовлияния разных объектов в зависимости от их мощности и плотности размещения становится все более существенным, и экономический ущерб от аварийных последствий, вызванных близостью различных предприятий, может превысить выгоды, связанные с близостью сырьевой базы или транспортными удобствами. Чтобы задачи размещения решались оптимально, необходимо управляемое совместное действие специалистов разного профиля, способных прогнозировать воздействие разнохарактерных факторов, в том числе неспецифичных для данного конкретного производства, самое широкое использование методов математического моделирования.

В СССР эта деятельность по выбору площадок для АЭС и других промышленных объектов до аварии на ЧАЭС была дифференцирована по отраслям и типам производства. В настоящее время поставлена задача разработки единого подхода в этой области. Она потребует существенного увеличения масштаба расчетных и экспериментальных работ в этом направлении.

4. Безопасность ядерной энергетики должна совершенствоваться не только на собственном эксплуатационном опыте, но и на опыте, полученном при эксплуатации сложных систем в других отраслях промышленности. Сравнительный анализ аварии на ЧАЭС с другими крупными авариями как в ядерной энерге-

тике (например, авария на американской АЭС "Три-Майл Айленд"), так и в других отраслях промышленности (например, аварии на хранилище сжиженного газа в Мексике в 1984 г., на химическом предприятии в индийском городе Бхопале в том же году и т.д.) позволяет выявить очевидное подобие их причин. Масштабы крупных промышленных аварий главным образом определяются общепромышленной тенденцией роста единичных мощностей технологических блоков.

Развитие современных технологий, направленных на повышение уровня жизни населения, в то же время приводит к появлению возможности крупных аварий с разрушительными последствиями для окружающей среды, тяжелыми последствиями для общества. Это требует глубокого осмысливания и энергичных действий, направленных на совершенствование технологических процессов и промышленных структур с позиций безопасности. Это возможно только на основе объединения исследований по безопасности в единую научную дисциплину промышленной безопасности, способную найти не только качественно новые направления по повышению безопасности современных технологий, но и определить общие принципы и методы создания технологий следующего поколения. В рамках такой единой научной дисциплины, изучающей проблемы безопасности в промышленности, должна совершенствоваться и безопасность в ядерной энергетике.

6. ЗАКЛЮЧЕНИЕ

Авария на IV блоке ЧАЭС потребовала мобилизации значительных сил и средств для ограничения и ликвидации ее последствий. В этой работе участвовали многие министерства, ведомства и научные организации СССР. Правительства и различные организации ряда стран предложили свою помощь. Эта помощь с благодарностью была принята.

На первом этапе потребовались немедленные действия по предотвращению развития аварии и защите здоровья населения, персонала АЭС и лиц, участвовавших в противоаварийных работах.

В конце 1986 г. было завершено строительство Укрытия и

осуществлен пуск I и II блоков ЧАЭС. Среди других работ этого же периода следует отметить необходимые мероприятия по обеспечению нормальных условий жизни эвакуированного населения, медико-санитарные и сельскохозяйственные мероприятия, дезактивационные работы на площадке АЭС и в 30-километровой зоне, организацию и осуществление радиационного мониторинга.

Осуществление крупномасштабных работ по ликвидации последствий аварии в короткие сроки стало возможным в результате способности советского общества мобилизовать имеющиеся средства, сконцентрировать их на целевых задачах и обеспечить высокую организацию проводимых мероприятий.

Авария на ЧАЭС потребовала критического анализа состояния безопасности ядерной энергетики с целью достижения более высокого ее уровня.

Прежде всего, предметом анализа стали причины, ход и последствия аварии, противоаварийные мероприятия и их эффективность. Были разработаны и реализованы первоочередные мероприятия по повышению безопасности АЭС с РБМК. В более широком плане были рассмотрены все аспекты обеспечения безопасности, в том числе:

— технические средства;
— управление и подготовка персонала;
— организационное и нормативное обеспечение;
— научно-техническое обеспечение;
— аварийные планы и средства их обеспечения.

На основе этого анализа был разработан долговременный план совершенствования безопасности ядерной энергетики, включающий в себя работы по всем отмеченным направлениям. В этом плане значительное внимание уделяется вопросам совершенствования управления и взаимодействия человек — машина. К ним относятся совершенствование средств автоматики и контроля, информационного обеспечения, подготовки персонала, особенно к действиям в аварийных условиях. Разрабатываются программы и организуются долгосрочные исследования по изучению отдаленных последствий аварии, а также мероприятий по их ограничению и ликвидации.

Поставлена задача совершенствования научно-методических основ оценки, анализа и управления безопасностью.

Большое значение придается работам по созданию ядерных энергетических реакторов нового поколения с так называемыми внутренне присущими свойствами безопасности.

Причины и масштабы больших аварий на АЭС нельзя считать исключительным свойством ядерных установок. Как и крупные аварии в неядерных областях, они, главным образом, определяются общепромышленной тенденцией роста единичных мощностей технологических блоков, вовлечением в производство большого количества разнообразных вредных веществ, усложнением системы управления, недостатками во взаимодействии человек — машина и т.п.

Критический анализ, которому была подвергнута ядерная энергетика после аварии на ЧАЭС, не привел к изменению наших позиций в отношении развития ядерной энергетики в СССР и в мире в целом. Наши планы по вводу ядерных энергетических мощностей существенным образом не изменились.

Однако авария на ЧАЭС, как и другие аварии в ядерной и неядерной областях, указывает на необходимость повышения уровня безопасности в ядерной энергетике и других отраслях промышленности. Уроки этих аварий для нас и всего мирового сообщества состоят прежде всего в том, что возникающая в процессе научно-технической революции новая сложная техника требует внимательнейшего отношения к вопросам ее безопасности и надежности, не прощает халатного и неквалифицированного обращения.

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СПИСОК СОКРАЩЕНИЙ

АВР	— автоматическое включение резерва
АЗ	— аварийная защита
АЗ-5	— сигнал к вводу в активную зону всех регулирующих стержней аварийной защиты
АМН	— Академия медицинских наук
АН	— Академия наук
АЭС	— атомная электростанция
БССР	— Белорусская Советская Социалистическая Республика
ВВЭР	— водо-водяной энергетический реактор
ГКАЭ	— Государственный комитет по использованию атомной энергии СССР
Госагропром	— Государственный агропромышленный комитет СССР
Госатомэнергонадзор	— Государственный комитет СССР по надзору за безопасным ведением работ в атомной энергетике
Госкомгидромет	— Государственный комитет СССР по гидрометеорологии и контролю природной среды
ДП	— дополнительный поглотитель
ДС	— система диагностики
ЗТ	— загрязненная территория
ИАЭ	— Институт атомной энергии им. И.В. Курчатова
ИДК	— информационно-диагностический комплекс
КАЭС	— Курская АЭС
КИП	— контрольно-измерительные приборы
КМПЦ	— контур многократной принудительной циркуляции
КОСУЗ	— контур охлаждения СУЗ
ЛАЭС	— Ленинградская АЭС
Минздрав	— Министерство здравоохранения СССР
МПЦ	— многократная принудительная циркуляция
ПДК	— предельно-допустимая концентрация
РБГ	— радиоактивные благородные газы
РБМК	— реактор большой мощности канальный
РДК	— радиационный дозиметрический контроль
РР	— ручной регулятор
РСФСР	— Российская Советская Федеративная Социалистическая Республика
СПИР	— система продувки и расхолаживания
СУЗ	— система управления и защиты
с/х	— сельскохозяйственный
ТВС	— тепловыделяющая сборка
ТГ	— турбогенератор
ТК	— технологический канал
Т	— технологический контроль
Уп	— указатель положения
УССР	— Украинская Советская Социалистическая Республика
УФ	— ультрафиолетовый
ЧАЭС	— Чернобыльская АЭС
ЭВМ	— электронно-вычислительная машина

СОДЕРЖАНИЕ

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V G Asmolov and others

***

1 Introduction

A delegation of Soviet experts presented information on the Chernobyl nuclear power station accident and its consequences at the IAEA conference in Vienna on 25-29 August 1986 [1]. This information contained the results of investigations into the causes of the accident, and also a description and preliminary analysis of the effectiveness of the initial measures taken to limit and remove the consequences, based on data received up to 1 August 1986.

. the following period efforts were concentrated on the following:

1 Continuation of work to remove the consequences of the accident, including the following:

- completion of designing and building of the entombment to provide positive protection of the environment from radioactive substances entering it from the destroyed reactor unit and from radiation;
- continued decontamination of the power station site, buildings and containments of units 1, 2 and 3 and centres of population in the zone which had suffered radiactive contamination;
- bringing into operation units 1 and 2 of the power station;
- completion of measures to meet the social and living needs of the evacuated population and its specialised employment;
- the implementation of medical and public health measures to ensure the safety of the population and the maintenance of its health.

2 The development of programmes and the organisation of long-term investigations to study indirect consequences of the accident, and also measures to limit them and remove them, including the following:

- monitoring the radioactive contamination of the environment;
- determining the need for further decontamination work and carrying it out;
- conduct of exploratory inspections and preventive work on the entombment;
- conduct of scientific research to study the long-term consequences of radioactive contamination of the biosphere.

All scientific work is being co-ordinated by a specially formed committee of the USSR Academy of Sciences.

3 The development and introduction of measures to increase the safety of operating nuclear power stations.

4 The consideration of plans for the further development of nuclear power and the scope for increasing its level of safety, including the development of a new generation of reactor design and the extension of scientific research into all aspects of assessing and ensuring the safety of nuclear power.

This paper reviews the progress and results of work in these areas.

2 Building of the entombment

The building of the entombment, whose job is to provide the long-term protection of the damaged reactor unit, was one of the most important measures taken to remove the consequences of the accident.

The entombment is not intended as a store for nuclear fuel, and it does not function as such; neither is it a repository for highly active wastes, nor is it any other kind of structure previously met with in nuclear technology. The need to build the sheltering structure required the development of rules defining its purpose and the demands placed upon it, and the formulation of its safety design.

The safe state of the entombment is a state in which conditions are maintained which prevent the following:

- the development of a self-sustained chain reaction;
- the disruption of heat removal leading to the melting of fuel residues;
- the formation of explosive concentrations of hydrogen.

The basic purpose of the entombment is:

- to prevent the release of radioactive substances from the damaged reactor into the environment;

- the protection of the adjacent area from prompt radiation.

The basic requirements of the entombment design are:

- to reduce the building time to a minimum by using simple, reliable and approved facilities;
- to preserve the functions of the entombment under the possible action of natural phenomena (hurricane, earthquake, etc), which could occur at the Chernobyl nuclear power station site;
- to ensure the removal of residual heat and radiolytic hydrogen;
- to minimise the irradiation dose on building workers during construction;
- to ensure access for the performance of scientific investigations to low radiation level enclosures undergoing conservation;
- to permit monitoring and diagnosis of the state of the active mass.

Immediately after the accident, radiation and temperature measurements were begun inside and outside the reactor building, together with investigations of the condition of elements of the reactor structure and reactor building which had been preserved and on which the temporary system of thermal and radiation parameter monitoring was based.

The basic means adopted for carrying out measurements in enclosures of the destroyed reactor unit were radiation and thermal surveying which made it possible to prescribe a different type of sensor, to draw a map of the degree of contamination of the enclosures and to approach enclosures with large accumulations of fuel. It should be noted that experience of the use of both Soviet and foreign robot devices showed them to be a not very suitable solution to the problem of surveying and remote performance of work in enclosures of the unit with a complicated configuration, in the presence of bble and obstructions, and also considerable gamma radiation fields.

Starting in May, temperature and gamma radiation measurements were carried out in the space above the reactor cavity using sensors lowered from helicopters. At the beginning of August special sensors (diagnostic beacons) were set up on the rubble of the core in the region of the upper slab position and on the edge of the rubble. Gamma radiation fields, conductive and convective thermal fluxes, air temperature, and vertical and horizontal air velocities were measured. The temperature variations observed depended on wind speed and air movement. Air velocities measured were 0.8-1.0 m/s vertical, and 0.5-0.8 m/s horizontal, temperature 30-50°C at different measuring points, conductive thermal flux up to 200 W/m^2, convective thermal flux up to 10 kW/m^2. The intensity of gamma radiation was over 10^4 R/h at the approximate centre of the reactor. A total of 9 diagnostic beacons were placed by helicopter. In the region of the reactor foundations thermocouples and gamma sensors in tubes were installed channels in the space around the core via services channels. Measurements showed the irradiation field to be from 10^3 to 10^5 R/h, which confirmed the presence of fuel, and the temperature varied approximately from 30 to 50°C depending on the circulation of air along the tubes.

At the end of Septmber (before the start of covering up), a further 4 diagnostic beacons were placed on the rubble with the aid of construction cranes. Velocities of inflowing currents of air through artificially formed passages in the pressure suppression pond were measured with anemometers.

The number of measuring points steadily increased as access was gained to different parts of the damaged reactor.

Measurements made in the diagnostic beacon programme not only provided efficient monitoring, but also permitted an estimate of the quantity of nuclear fuel remaining in the reactor building on the basis of its power release. This estimate agreed with data on radioactive discharges and fall-out in the locality as a result of the accident, according to which the reactor cavity and unit lV enclosures contained about 96% of the fuel of the full reactor inventory.

ıe "thermal location" of the basic sources of power release showed that their major part was concentrated in the reactor cavity and the enclosures under the reactor.

lnvestigations of the enclosures under the reactor revealed significant quantities of melted sand (SiO_2) containing up to 2% of the fuel mass. This finding confirmed the preliminary estimate of the temperature during the accident (~2000 K).

Solution of the constantly arising problems provided unique operating experience under the abnormal conditions of the reactor unit, and different diagnostic methods were tried out which served as a basis for further investigations.

The measurements carried out permitted a reliable check on the thermal conditions of the fuel mass while the building work was going on.

The experimental information obtained was important for the design of the ventilation system and of the entombment as a whole.

Calculations showed that with a power release of ~2 MW and a mean heat-up of the air of ~(30-40)°C, the flow of air through the reactor zone G ~50-60 kg/s, which agreed with the results of direct measurements. This must result in a pressure drop vertically through the reactor cavity of 2-6 mm water column.

To validate the ventilation system using experimental data, special tests were carried out on a mock-up which showed the correctness of the following ventilation systems: natural inflow of air into the enclosures under the reactor (pressure suppression pond) and forced extraction ventilation with discharge to atmosphere via special filters. lmplementation of this system requires the free access of air to the fuel mass and removal of the heated air which has passed through the reactor cavity. The ventilation system adopted means that with continued operation of the entombment, as the residual power naturally drops it will be possible to transfer to natural circulation of air through the fuel mass, and ultimately to a closed system of air circulation within the sheltering structure.

The accumulated experimental data led to the further development of mathematical models describing the cooling processes. Several possible mechanisms of these processes have been analysed. Tracer experiments made on the damaged reactor unit have shown that a model of filtered cooling is equivalent to the real situation, providing a natural explanation of the basic features of the change in the thermal conditions of the core after the accident [4].

A rigorous mathematical investigation of the system of nonlinear equations showed (with a probability approaching unity) that under quasi-steady conditions of filtered cooling, no zones of very intense local overheating occurred in the rubble, even with intense localised sources of heat. Removal of heat from the core by a filtered flow ensures effective cooling of the whole mass of rubble.

Investigation of the conditions for the existence of quasi-steady cooling has led to the determination of the criterion for the existence of quasi-steady filtered cooling conditions. The parameters which determine the criterion include the integrated power of heat sources and the permeability characteristics of the rubble. The conditions for quasi-steady cooling may be sturbed by compacting of the rubble and the corresponding reduction in its permeability.

It has been shown that the absence of a quasi-steady solution leads to a nonsteady process called "dry boiling". This process is accompanied by small explosions in the solid material leading to loosening of the rubble and so to an increase in its permeability. As a result, the system goes over to a new state in which quasi-steady cooling conditions are stabilised.

In order to estimate the parameters of the cooling process of the damaged reactor, a program for the calculation of temperature fields and other thermal characteristics of the rubble was written on the basis of a mathematical model of filtered cooling. Calculations were made of corresponding fields with different distributions of heat sources in the rubble, which correlate with experimental measurements.

Because of the extreme importance of the structure being built, for the final solution a large number of versions of the solution for the building structure were developed. These all reduced to 2 main versions:

1 to raise an arched roof with a span of 230 m over the damaged reactor unit or to make a domed, arched cantilever roof with a span of 120 m;

2 to make a roof with structural elements having a span of 55 m, using for their support the retained walls and roofs of the building.

Analyses and feasibility calculations showed that the work on the first version would take 1½ to 2 years, while work on the second would permit a considerable reduction in the building time and consumption of materials. The first version was therefore adopted as the basic design incorporated in the plan.

As the design documentation became ready it was issued to the building group where, if necessary, detail was added or additions made by a team under the designer's supervision taking into account concrete conditions as they arise. In developing the design documentation, engineering decisions were also found which permitted a maximum reduction in the effort required and the construction times under the complex radiation conditions.

The spatial structure of the shelter is formed by a series of blocks rising in steps whose dimensions and shape are determined by the features of the elements of the enclosing structures. A concrete separating wall is provided between units 3 and lV (with maximum use being made of the existing walls). ln the turbine building between units 2 and 3 a metal separating has been built.

At the same time as the entombment was being built, a large programme was conducted to create a system for monitoring the state of the structure and to determine its influence on the radiation level of the Chernobyl nuclear power station site and beyond its boundaries. The technical requirements and initial data were formulated for the creation of an information and diagnostic system, designed for the monitoring and diagnosis of the state of the fuel mass, components of the structure, the radiation level, and also for monitoring the engineering systems of the entombment.

.ıe information and diagnostic system consists of the following:

- the monitoring and diagnostic system;
- the information and computer system;
- communications with external users.

The monitoring and diagnostic system includes the following subsystems:

- process monitoring;
- radiation monitoring;
- the diagnostic system for the physical and mechanical state of the active mass and of the building structures.

The task of the process monitoring subsystem is to monitor the ventilation ·uipment which removes the heat, and the parameters of the cooling medium.

The radiation monitoring subsystem must monitor the level and transfer of radioactivity inside the structure and the discharge of radioactive products to the environment.

The function of the diagnostic system is to determine the physical and mechanical state of the active mass, and to detect vibrations which develop, the displacement of structural components of the structure, and damage due to internal processes.

Monitoring and diagnosis of the state of the entombment are achieved by measuring the temperature of the space under the roof of the central hall and on the upper surface of the reactor cavity, of components of the core support plate, and of covering surfaces of the pressure suppression pond. ln order to determine more accurately the distribution and intensity of sources of heat, the thermal flux is measured at accessible points of the enclosures under the reactor and on the upper surface of the damaged core. The intensity of gamma radiation is measured in all enclosures where the state of the operating equipment is monitored and where it is serviced and repaired. ln addition, the gamma radiation field is measured in most of the accessible enclosures of the building, and also in the space under the cover and on the upper surface of the damaged core. The concentration of H_2, CO and H_2O in the air is constantly monitored.

In order to detect any very improbable self-sustaining chain reactions, neutron detectors have been mounted, and the occurrence of the short-lived isotope iodine-131 at the outlet of the ventilation system is monitored, Measures have been developed and implemented for the nuclear safety of the protected reactor unit IV, which guarantee the emergency suppression of the process in the event of the development of a chain reaction in the reactor cavity due to the introduction of a liquid neutron absorber.

In order to monitor the mechanical stability of the fuel mass and of structural components, acoustic vibration detectors have been installed which record acceleration, frequency, and changes in vibration.

The information and diagnostic system provides positive monitoring of the state of the structure which was handed over to the operating organisation on 1 December 1986 for technical servicing.

..nalysis of experimental data on gamma dose rates, temperature and thermal flux indicates the stability of the fuel mass. The gamma dose rate is decreasing in accordance with the decay of the fuel. The mean daily total discharge of radioactive fission products does not exceed 3 mCi*. In the zone adjacent to the township of Pripyat, no formation of new spots of radioactive contamination or increase in the integrated dose for the zone was found for the period August-September. The reduction in intensity of gamma radiation was in accordance with theoretical estimates.

When work on the shetlering structure was completed, the damaged reactor unit ceased to be a source of high rates of release of aerosol activity, both as a result of its being removed by ventilation, and as a result of wind erosion.

Figure 2.1 is a photograph of the entombment. Figure 2.2 shows the changes in gamma radiation intensity inside the entombment and changes in the temperature of the "hottest" spot in one of the enclosures under the reactor, adjusted to the value on 1 January 1987.

the present time, in accordance with the long-term programme of scientific research, work is being carried out and planned in the following areas:

- accurate determination of the quantity and location of nuclear fuel inside the reactor unit;
- determination of the physical and chemical state of the fuel;
- investigation, using neutronics methods, of the breeding, absorbing and moderating properties of the aggregate of materials containing nuclear fuel;
- study of the properties of structural materials in gamma radiation fields in reactions with fuel mass residues.

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*For a 1000 MW operating reactor unit the maximum permitted discharge of long-lived radioactive impurities is 15 mCi/day [6].

3 Restarting of the Chernobyl nuclear power station reactor units

Questions of restarting operation of units 1, 2 and 3, and the work necessary for this were among the most important in the plan for removing the consequences of the accident and were solved in parallel with work on the protection of unit 4.

After the accident to unit 4, units 1 and 2 remained in a normal serviceable condition and were shut down at 0113 h and 0213 h respectively on 27 April.

Unit 3, which has engineering connections with unit 4, was shut down at 0326 h on 26 April. Normal shutdown cooling was carried out on all the shut down units.

After shutdown cooling, units 1, 2 and 3 were brought to the deep subcritical condition by inserting control rods into all their cores and charging 20 additional absorbers into units 1 and 2, and 220 absorber rods into the central fuel subassembly tube of unit 3. The neutron flux was monitored by the standard equipment.

For the removal of the residual heat all the fuel channels and the recirculation circuit were left full of water. The residual heat was removed by natural circulation. The water temperature in the core was maintained at 20-80°C, and the graphite temperature at 30-90°C.

Recovery work was started with decontamination of the basic and auxiliary buildings and plant of the reactor units, of the equipment and workplaces of the personnel in them, and of the adjoining land.

The highest level of contamination was of the individual horizontal sections of surfaces of the turbine hall [up to 10^6 β-particles/cm^2.min], since its contamination was due to destruction of the roof of unit 4. The gamma dose rate in the contaminated enclosures of units 1 and 2 on 4 May 1986 was 10-100 mR/h, and in the turbine hall 20-600 mR/h.

Decontamination was carried out using special solutions whose composition was determined by the material being washed, and the nature and level of contamination of the surface. Liquid jet and steam jet methods, and dry decontamination with the aid of polymer coatings were used. Some of the enclosures and equipment were decontaminated by manual wiping with cloths soaked in decontamination solutions.

Effectiveness of decontamination was montiored by direct measurement of the gamma dose rate and by smear testing. As a result of decontamination, levels of decontamination of surfaces in enclosures and on equipment were reduced to regulation levels.

Decontamination work on units 1 and 2 was completed at the beginning of the third quarter of 1986.

Decontamination is continuing on unit 3, and this will lead to the further improvement of radiation levels on the operating reactor units.

The removal of the contaminated soft covering on the roofs of the reactor units is being completed. Special adhesives are being successfully used for this, which are remotely applied to the contaminated section of the covering, with subsequent removal with the aid of cranes. As a result of the fulfilment of a part of the planned work, the dose rate in unit 3 turbine building was sharply reduced at 7-50 mR/h by the end of July 1987.

At the present time, with completion of building of the entombment and performance of all work on decontamination of the power station land, the radiaton levels on units 1 and 2 have finally stabilised and have been virtually returned to regulation levels.

Water temperature was measured with additional thermocouples mounted in the central openings of the fuel subassemblies in the left and right hand halves of the reactor. Thermal conditions of the graphite stacking and recirculation circuit are monitored using the standard temperature channels.

The set thermal conditions of the reactor and recirculation circuit were provided by bringing into operation a blowdown and shutdown cooling system. The graphite stacking was periodically purged with nitrogen or with dry air with a maxiumum moisture content of 0.5 g/m^3. The control rod circuit was drained after complete shutdown cooling.

All necessary auxiliary equipment of units 1 and 2 was maintained in a state of readiness for operation. The ventilation systems remained disconnected until the air lines and ventilation equipment had been contaminated and additional equipment for cleaning inflowing air had been installed. The fire extinguishing system was maintained in a state of readiness under automatic control. The unit 1 and 2 radiation monitoring system was taken out of operation over the complete volume. The power station internal load was supplied from the normal supply source with readiness to take over the load of any mechanisms on standby. The turbine building system was maintained in a protected state. The power station operating staff monitored the state of the reactors and reactor unit equipment.

Maintaining the sytems and equipment of the reactor units in a state ready for operation or in a protected state made it possible for recovery work to be undertaken in the future and for them to be brought into operation at short notice.

With units 1 and 2 spending a long period shut down, the effects of radiation and of decontaminants required careful examination and diagnostic checking of all the basic and auxiliary equipment and of the automatic systems, and the performance of repairs and commissioning work.

The work on preparation and start-up of units 1 and 2 was specified in the "Programme of integrated trials and start-up of units 1 and 2 of the Chernobyl nuclear power station" and it corresponded with the requirements for newly introduced reactor units. In accordance with this programme, in the pre-startup period, the reactor systems were tested component by component, including checking of the serviceability of valves and fittings, instumentation and control systems, mimic flow diagrams, shielding, interlocks, signalling systems, automatic standby start-up, the switching on of systems, bringing working fluids up to regulation quality, and checking the tripping of systems and mechanisms by emergency protection signalling systems. On the basis of the results of repair and recovery work and the component-by-component checking, the document of readiness of the reactor systems for start-up was drawn up, and also the general document of readiness of the equipment, systems, technical documentation and personnel of the Chernobyl nuclear power station for start-up of the reactor.

In the pre-startup period the main attention was paid to the preparation of the operating staff. Because of the existence around the power station of zones of radioactive contamination, it was important to create proper living conditions for the personnel. These problems were solved by the organisation of a duty system, the essentials of which were as follows:

- in the duty (working) days period, strategic and operating personnel live in the duty housing complex located beyond the 30 kilometre zone;

- days off are spent by the operating personnel in the towns of Kiev and Chernigov, where appropriate conditions are created for them.

The working day for strategic personnel is 12 hours (0800 to 2000 and 2000 to 0800), and for the operating personnel and all power station staff (as for all those working in the 30 kilometre zone) it is 10 hours (0900 to 1900). For strategic staff the duty system is 5 days on and 7 days off, while for all other workers it is on a 15 day cycle.

The duty system justified itself under the radiation conditions in the 30 kilometre zone, and made it possible to create appropriate conditions both r work and rest.

When building of the operators' town of Slavutich is completed in 1988, normal working and living conditions will be resumed for the power station team.

The specific problems in the start-up of units 1 and 2 lay in checking and accurately determining the characteristics and operating conditions of the equipment after prolonged shutdown, and in checking the effectiveness of the measures taken to improve the safety of nuclear power stations with RBMK reactors. To solve these problems, the reactors were started up in 3 stages:

1st stage	-	formation of the core and carrying out physical start-up;
2nd stage	-	raising reactor power to 700 MWe with performance of integrated testing of the reactor systems;
3rd stage	-	fine adjustment up to rated power level.

ɜ aim of the measures for increasing safety was mainly to reduce the steam reactivity effect and to increase the rate of response of the emergency protection system. To achieve these aims, and taking into account the features of the reactor, before the physical start-up 50% of the rods were inserted in the core to the 1.4 m position with respect to the position indicator, the number of short absorber rods was increased to 32, and their displacement was limited to 1.2 to 3.5 m with respect to the position indicator.

During the physical start-up, additional absorbers were mounted in place of some of the fuel subassemblies. The operational reactivity margin during reactor operation was set at 43-48 rods. The developed initial reactor charge of unit 1 was 1648 fuel subassemblies (of which 124 are new), 30 additional absorbers, 14 fuel channels with water and one plugged subchannel; the initial charge of unit 2 is 1610 fuel subassemblies (of which 313 are new), 81 additional absorbers, and 2 fuel channels with water. The number of additional absorbers in the unit 2 reactor was increased in order further to decrease the steam coefficient of reactivity.

On completion of the physical start-up, the reactor systems and equipment were prepared and brought into operation in accordance with engineering regulations, and work was carried out on adjusting and testing equipment, checking process parameters, bringing up to a power of 700 MWe, and the test running of turbo-generator 1(3) and turbo-generator 2(4) up to a power of 500 MWe.

During power raising and integrated testing to 700 MWe, no significant faults in the operation of the basic and auxiliary equipment were found on units 1 and 2. The water quality in the recirculation circuit and control rod coolant circuit, and the feedwater were up to standard. The reactor controls maintained the process parameters at the regulation values. The reactor core parameters remained satisfactory.

The power density distribution in the fuel channels was measured with sufficient accuracy and was easily maintained within the set limits. The radiochemical composition of the recirculation circuit coolant was in accordance with regulations. The radiation levels in the enclosures permitted units 1 and 2 to be brought into operation.

The data obtained in the integrated trials of the reactors up to a power of 700 MWe showed that it was possible to raise the power of units 1 and 2 of the ernobyl nuclear power station to the design rated value of 1000 MWe and to carry out the third stage of their start-up. Power was raised in stages of 10%N with the integrated testing of the reactor equipment being carried out at each stage.

The values of the process parameters and physical characteristics of the reactor when operating at rated power meet the requirments of the technical regulations. No malfunctions to the equipment which could have prevented operation of units 1 and 2 were found. On this basis units 1 and 2 of the Chernobyl nuclear power station are operating at rated power.

No problems were encountered in controlling units 1 and 2 after the measures had been taken to increase safety. This was facilitated by the fact that an EC-1035 computer forms part of the process chain for the preparation and selection of refuelling at the Chernobyl nuclear power station.

The results of the start-up of units 1 and 2 of the Chernobyl nuclear power station may be summarised as follows:

The measures taken considerably increased the safety of units 1 and 2. The rate of insertion of negative reactivity increased to $\geq 0.5\beta/s$. From the results of measurements of the velocity effectiveness of the emergency control rods, it was decided to reduce slightly the depth of insertion of the control rods in the core from 1-2 to 0.7 m, which reduces the deformation of the axial field and at the same time retains acceptable velocity characteristics of the emergency control rods.

The steam reactivity coefficient was reduced. At the moment, the number of additional absorbers in unit 1 has also been increased to 80.

2 In the current year a start is being made with refuelling the reactors with fuel subassemblies with 2.4% enrichment, which enables the steam coefficient of reactivity to be reduced to zero.

4 Removing the consequences of the accident arising from radioactive contamination of the natural environment

The radioactive contamination of the Chernobyl nuclear power station site and adjacent region called for a package of measures to remove the consequences of the accident.

Reference [1] contains a description of this package. The main attention was

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devoted to urgent and essential measures (estimation of the radiation levels, evacuation of the population, isolation of the source of radioactive discharges, active decontamination, etc).

After these measures had been taken and the situation in the region around the power station had stabilised, it became possible to start regular work on removing the consequences of the accident and measures of a long term character.

4.1 Monitoring the radioactive contamination of the natural environment in the region of the Chernobyl nuclear power station

The regular collection and presentation of information on radiation levels in the accident zone and over the whole country was started on 26 April 1986. ,nitoring the radioactive contamination of the natural environment was carried out and is being continued by organisations of Goskomgidromet [State hydrogeological committee] of the USSR together with organisations of Minzdrava [Ministry of health], Gosagroprom [State agricultural industry], the Academy of Sciences, Ministry of Defence, GKAEA [State committee on utilisation of atomic energy] of the USSR, etc.

Because of the scale of the accident the existing radiation monitoring was considerably extended by involving additional groups of specialists (several thousand) and technicians. The aviation division of the Ministry of Defence was involved, in particular, for monitoring the radioactivity of the atmosphere.

The data was collected by active permanent or temporary stations, observation posts, reconnaissance aeroplanes or helicopters, expeditions, mobile groups, etc.

The data include:

results of gamma and beta radiometry and spectrometry of contaminated areas;

analysis of samples of air, water, soil, and vegetable matter;

analysis of samples of radioactive fall-out.

After essential short-term problems had been solved, the radioactive contamination monitoring system was transformed into a constantly active monitoring system in the region contaminated by radioactivity. This system supplemented scientific investigations into the radioecology and migration of radioactive substances in the natural environment (including the food chains), into the forecasting of changes in radioactive contamination, and the radiation dose on natural objects and the population. Special scientific organisations have been created within the framework of the Academy of Sciences, Gosagroprom and other departments of the country to carry out long term scientific programmes in the Chernobyl nuclear power station region.

The programme of scientific research to monitor the radiation levels in the Chernobyl region included the following:

1 the development of a complex method for radiation monitoring of the ground surface, incorporating the following:

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- spectrometric and dosimetric surveying (semiconductor field-effect spectrometry);

- radiochemical and nuclear physics techniques of analysing soil, water, and aerosol samples;

- the establishment of correlations between the content of plutonium and strontium-90 and the content of cerium-144, determined from the gamma spectra of specimens;

2 the development of new techniques of recording plutonium and strontium on the basis of low background apparatus with liquid scintillators;

3 the development of a technique for the constant remote monitoring of radiation levels by measuring the luminescence of air in the ultraviolet region of the spectrum;

4 the creation of a data bank of assessments of soil contamination, and also a system for the expert assessment, processing and analysis of information received on radiation levels.

Information on radioactive contamination formed the basis of decisions on the guaranteeing of the radiation safety of the population and economic activity in the contaminated territory. These decisions include the following:

- the evacuation and re-evacuation of the population from a number of centres of population in the contaminated zone;

- the decontamination of the area, buildings, etc;

- establishment of exclusion zones or zones where productive activity is limited;

- protection of the hydrosphere from radioactive contamination.

Data on radioactive contamination was also used for the more accurate determination of the total quantity, dynamics and radionuclide composition of radioactive discharges from the damaged nuclear power station.

4.2 Radioactive contamination of the immediate locality adjacent to the Chernobyl nuclear power station

Near and far zones of radioactive contamination were established from 7-8 May 1986. The features of their formation were determined by the dynamics and height of discharge, meteorological conditions in the region of Chernobyl, and in remote regions by the direction in which the contaminated air masses are spreading.

Data on radioactive discharges and radioactive contamination obtained in the first few months after the accident are given in the report of the Soviet delegation to the IAEA meeting of experts in August 1986 [1].

In the year since then much work has been done to determine more accurately and develop a detailed picture of the radioactive contamination in the near and far zones of radioactive fallout. A data bank on radioactive contamination in the Chernobyl nuclear power station area has been created and is being constantly added to. The data bank also contains data on the radiation background up to 26 April 1986.

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The background radiation before the accident in the Chernobyl region was characterised by the following data:

- gamma exposure dose rate 0.01-0.015 mR/h;
- density of caesium-137 and strontium-90 contamination (due to global fallout from nuclear testing) 0.1 and 0.07 Ci/km^2 respectively.

Radioactive fallout as a result of the Chernobyl accident led to contamination of the natural environment. Figures 4.1 and 4.6 give some idea of this (see also [1]).

The radioactive track in the contaminated region has 3 branches - north, south and west.

..ter radioactive discharges from the damaged reactor had been stopped, the change in radioactive contamination was basically determined by radioactive decay, wind transfer, washout and transfer by rain and floodwaters (after melting of snow), diffusion in the soil, etc.

By autumn 1986 the first process began to play a smaller role due to decay of relatively short-lived radionuclides with a half-life of no more than a few months. Radioactive contamination of the natural environment will virtually come to an end during 1988.

After theaccident, in the near zone to Chernobyl regular measurements are being taken of gamma fields, and aerogamma plots of the contaminated locality. Figure 4.1 is a map of the distribution of radiation levels in the locality on 1 May 1987.

From data on the gamma dose rate distribution in the locality at different times after the accident, the total quantity of radioactive products which had fallen on the near track was estimated, plus the change in this quantity with ' ne due to radioactive decay and other factors. Figure 4.2 shows the change w.th time in the gamma dose rate from the total quantity of radioactive products on the near track, which is in good agreement with similar data obtained from the analysis of radioactive products from soil samples, taken from the western and northern areas of the near track [3].

The total quantity of gamma radioactive products on the near track one year after the accident has decreased by approximately 55 times, being $2.7 \cdot 10^6$ R·m^2/h on 1 May 1987.

The migration of radionuclides in the soil leads to a reduction in the R_γ rate of the exposure dose from radioactive fallout. As observations show, by autumn 1986, the depth of migration had reached 0.6-1.2 m in typical light soddy podzolic and sandy loam soils. This must lead to a weakening of the R_γ dose rate at a level of 1 m by a multiple of 1.5 to 2.5. This effect was confirmed by direct measurements in the locality [2].

Areas bounded by isodose rate lines on 1 May 1987 were as follows:

R = 1.0 mR/h	-	500 km^2
R = 2.0 mR/h	-	280 km^2
R = 5.0 mR/h	-	70 km^2
R = 10 mR/h	-	20 km^2
R = 20 mR/h	-	8.0 km^2
R = 50 mR/h	-	3.0 km^2

By one year after the accident, the areas bounded by the above isodose lines had decreased by a multiple of 50 to 150. Figure 4.3 shows the relationship between the areas of radioactive fallout on the near track and the isodose rate lines bounding the area, and taking into account the data of the aerogamma plots carried out at different times after the accident adjusted to the same time, viz, 29 May 1986, in accordance with the data in Figure 4.2 [5].

The radioisotope composition of near fallout is determined by the radionuclides given in [3], except for short-lived ones. After the decay of relatively short-lived radionuclides, such radionuclides as caesium-134 and caesium-137, and also isotopes of strontium and plutonium have the greatest radiobioligical significance. Contamination maps have been drawn for these radionuclides, and also for zirconium-95, niobium-95, ruthenium-103, lanthanum-140, etc.

Investigations of the isotope composition on the radioactive track have shown that considerable fractionation of radioisotopes is observed at distances beyond 15-30 km; for example, on the northern track considerable enrichment in the radioactive fallout of caesium-137 was observed (by a factor of 10 and higher).

Caesium contamination has a patchy character. This is due both to the dynamics of the discharge, and to the nonuniformity of rainfall in the regions through which the radioactive cloud passed. There are sections where the density of caesium-137 (plus caesium-134) contamination is up to 20-30 Ci/km^2, while in individual places it is of the order of 80 Ci/km^2.

The quantity of caesium-137 which has fallen on the near track is ~0.2 MCi (from aerospectrum plots and the analysis of soil samples). This figure is less than that given earlier [3]. It was obtained from the ratio of the density of radioactive fallout of a given radionuclide σ (Ci/km^2) to the dose rate R (mR/h) for different sectors of the near track.

On the near track the density of plutonium contamination is 0.1-1.0 Ci/km^2. In the direct vicinity of the industrial area, there are places where a density exceeding 10 Ci/km^2 is observed. The level of plutonium contamination of the locality rapidly decreases with distance. It should be noted that plutonium, strontium and a number of other long-lived radionuclides now form a component of fuel particles. This must be taken into account in the analysis of the ecological and biological consequences, and also in the consideration of the migration of these radionuclides in different media.

4.3 Meteorological data on wind direction during the accident and the dynamics of the release of radioactive products falling on the near track

Meteorological information during the period of the main release of radioactive products into the atmosphere after the accident included data from pilot-balloon observations of the direction and speed of the wind at airports in the towns and cities of Kiev (Zhulyany, Borispol'), Mozyr', Gomel', and Chernigov, and radiosonde data from Kiev from 26 April to 1 May 1986. A special program was used to calculate from the initial data the mean wind direction and speed in a band from the ground surface up to a given height. Figure 4.4 gives the calculated values of the mean wind speed and direction in bands of 0-500 and 0-1000 m for the time of observation over a period of 5 days after the accident. These data were used to calculate the transfer of particles in the atmosphere in bands of 0-500 and 0-1000 m.

Analysis of the meteorological data on wind direction in Figure 4.4 shows that during the 5 days from 26 to 30 April 1986, the direction of transfer of air particles in the band from ground level up to 1000 m changed by 360°, in fact describing a full circle. These meteorological features of the transfer of radioactive substances from the reactor zone when combined with the nature of the distribution of near radioactive fallout provided additional characteristics of the dynamics of the discharge [5].

Reference [1] gives data on the daily discharge of radioactive substances into the atmosphere from the reactor zone. A good approximation of the relative change in this discharge in the first 5 days is given by the equation

$$Q(t) = 0.32e^{-0.28t}, \quad t = 0, 1, \ldots, 4 \text{ days}$$

Radioactive fallout in the near zone came to an end in the first 4-5 days. In the subsequent days, as was shown by the regular aerogamma plots of the radioactive track, the total quantity of gamma radioactive products on the track decreased monotonically in accordance with the decay of the total radionuclides (see Figure 4.2). The daily discharge of radioactive substances to the atmosphere, approximated by the above exponential equation, includes the full spectrum of radioactive particles which resulted in both near and regional, and global fallout.

Figure 4.4 shows shaded wind direction sectors (in degrees) in bands from ground level to 500 and 1000 m in which the transfer and fallout of particles from the flow occurred at different time intervals after the accident. In accordance with this, sectors (230-320°, 320-20°, 20-90°, 90-220°), in which the total quantity of gamma radioactive products ($mR \cdot km^2/h$) have been estimated for 29 May 1986 [5], have been marked on the map showing the distribution of radiation levels in the near zone (up to 80 km). The total quantity of gamma radioactivity on the track on that date is $4.4 \cdot 10^4$ $mR \cdot km^2/h$. Figure 4.5 shows as a histogram the hourly fallout of gamma radioactive substances on the near track. The relative change in hourly fallout from 26 April to 1 May 1986 is shown by the solid line.

In the same way, using the ratio of density of radioactive fallout of a given radionuclide σ (Ci/km^2) to the dose rate R (mR/h) for different sectors (west, north, south), given in [3], we determined the fallout of individual radionuclides in the days following the accident. Table 4.1 gives the results of calculations of the relative discharge and fallout of the total of gamma radioactive substances and of individual radionuclides on the near track for the first 5 days.

In May 1987 another aerogamma plot was made and an aerospectrum plot of the territory of the USSR. The gamma field distribution shown in Figure 4.6 for the 0.05 mR/h isodose for 10 June 1986 [3] can by 1987 no longer be shown by complete isodose lines, and the gamma field is traced in the form of many separate unconnected spots.

The total quantity of gamma radioactive substances falling on the territory of the USSR beyond the limits of the near track, is estimated from the aerogamma plot to be $(6\text{-}9) \cdot 10^6$ $R \cdot m^2/h$ for the end of May 1987, compared with $1.2 \cdot 10^8$ $R \cdot m^2/h$ for the initial period after the accident [3]. The total fallout on the near and far zones is $(9\text{-}12) \cdot 10^6$ $R \cdot m^2/h$, or about 4% of the total quantity of radioactive products in the reactor at this time.

From the spectrum gamma plot, the quantity of caesium-137 and caesium-134 on the territory of the USSR (in the far zone) is approximately 0.6 MCi. The total quantity of caesium-137 in the near and far zones on the territory of the USSR is estimated to be less than 0.8 MCi, or about 10% of the quantity of caesium-137 formed.

4.5 Radioactive contamination of rivers and reservoirs

The basic sources of contamination of rivers and reservoirs, and also the concentration of radionuclides in water from the time of the Chernobyl accident up to July 1986 are given in [3].

In this period the radioactivity of surface waters is basically determined by caesium and strontium. Isotopes of ruthenium, cerium, zirconium-95 and niobium-95 in water samples were found in small concentrations, not constantly, and mainly adsorbed on suspended particles.

Table 4.2 gives the results of observations of the change in concentration of isotopes of caesium in water samples from the Kiev reservoir, and also from the rivers Pripyat and Dnieper from July 1986 to May 1987, from which it can seen that the concentration of caesium in the water for the period indicated decreased by a factor of more that 20. A particularly sharp decrease in caesium concentration was observed in the summer-autumn period of 1986. In water samples taken in the autumn, up to 70% of the activity of caesium radionuclides was in the adsorbed state on suspended particles. The content of suspended matter in the water and, consequently, the increase in radioactivity, is thus connected with the hydrometeorological conditions of the reservoirs investigated. The most obvious example of this relationship is observed in the Kiev reservoir into which suspended matter from the rivers Pripyat and Dnieper is discharged.

In the autumn, an increase in agitation is observed in the reservoir, and as a consequence, secondary contamination of the water as a result of wind disturbance of the upper layer of contaminated silt. Experimental investigations, confirmed by calculations, showed that, even the complete disturbance of the exchange layer of contaminated silt, which may occur in severe storms, does not increase the dose rates, determined as the total of all radionuclides, assuming the use of water for drinking, by more than 5-10% of the accepted USSR standards.

During the winter of 1986/87, under conditions of constant frost and the absence of any significant thaws, the radioactivity of the waters of the Dnieper cascade changed only slightly. By the end of the winter, the total beta activity of the water of the Kiev and Kremnechug reservoirs approximated to $(1\text{-}2)\cdot10^{-11}$ Ci/ℓ. The main contribution to the contamination of the water was from caesium-137 and strontium-90, whose concentrations in the water were $(1\text{-}4\cdot10^{-11}$ and $(0.1\text{-}4)\cdot10^{-11}$ Ci/ℓ respectively.

Under conditions of weak flow in reservoirs in the winter months there was little change in the contamination of the base ground. Decreases occurred mainly due to decay of zirconium-95, niobium-95, and also isotopes of ruthenium and cerium.

In preparation for the spring floods of 1987, over 100 protective and filtering dams were constructed on rivers and streams flowing through contaminated territory in order to reduce the washout of radionuclides.

The spring flood of 1987 was smooth and extended without any severe flood waves, as a result of which there was no increase in concentration of radionuclides in the rivers and reservoirs. The concentration of caesium in water samples taken at the end of April-May 1987 (see Table 4.2) remained at the level of the autumn samples or was slightly less.

Table 4.3 gives the mean conentrations of caesium-137 and caesium-134, measured in the autumn of 1986 and in the spring of 1987 on the river Dnieper and its main tributaries flowing through the territory of the Belorussian SSR.

It can be seen from the table that the concentration is considerably less than the maximum permitted. In the autumn period, up to 70% of the caesium was transferred as a suspension. In a period of rain which formed a run-off (31 August and 1 September), the quantity of caesium adsorbed on suspended matter increased to 80-90%, which is connected with processes of erosion of contaminated drainage basins.

In the spring of 1987 there was virtually no change in the level contamination of the above rivers. The ratio of the concentration of caesiu-137 to rontium-90 varied in different periods over the range 1-10.

4.6 Medical and public health measures

The initial measures for the protection of the health of individuals taking part in work to clear up the results of the accident and the population of the contaminated zone are described in [1].

Here we give the general characteristics of the medical and public health measures caried out up to mid-1987.

Broad scale public health and medical measures directed at ensuring the radiation safety of the population of the contaminated zones were started in the first days after the accident. Taking part, in addition to the medical institutes on the spot, were up to 400 specialised teams (medical staff, dosimetric staff, etc), and about 15 thousand medical workers, including students and graduates of medical institutes of higher education.

e measures taken had the following results:

- nearly one million people were subjected to medical examination, of whom 700 thousand (including 216 thousand children) underwent thorough dosimetric examination plus examinations requiring laboratory techniques; under stable conditions 32 thousand people, including 12.3 thousand children, are examined;

- 5.4 million people, including 1.7 million children, were given iodine treatment;

- recommendations were drawn up and implemented for the organisation of a summer health campaign for children and pregnant women beyond the boundaries of the contaminated areas;

- estimates and forecasts of radiation levels in the regions of radioactive contamination were made; on the basis of the data obtained, recommendations were drawn up for measures to protect the health of the population, including evacuation;

- a comprehensive system was created and implemented for monitoring the radiation level and health of people brought in to remove the consequences of the accident, plus personnel of units 1 and 2 of the nuclear power station which had again been brought into operation.

Health education was systematically conducted among the population.

For the provision of regular specialised medical assistance and the performance of subsequent observations and investigations, the All-union Scientific Centre for Radiation Medicine of the Academey of Medical Sciences of the USSR was established at Kiev.

The measures had the following results:

- reduced to a maximum the effects of radioactive substances on the population, particularly of iodine-131;

- prevented the re-exposure of individuals taking part in removing the consequences of the accident after 27 April 1986;

- prevented any outbreaks of infectious illnesses and food poisoning in and beyond the 30 kilometre zone;

- protected the health of personnel of units 1 and 2 of the nuclear power station.

According to estimates, the collective dose S of external radiation of the population of the USSR from contamination of the environment as a result of the Chernobyl accident was about 10^7 man ~~m~~rem for the first year after the accident. The expected collective radiation dose S^c of the population of the country will be less than $3.3 \cdot 10^7$ man ~~m~~rem.

This estimate takes into account the whole package of implemented and intended measures for the radiation protection of the population. It should be noted that about 60% of the value of S^c is due to external gamma irradiation of people by radionuclides falling on the locality after the emergency discharge, about 38% is due to internal irradiation from peroral intake of radionuclides (mainly caesium-137), and only about 2% is due irradiation from the discharge cloud and internal exposure of organs from inhalation.

The mean individual expected radiation dose of the population of the country is about 120 mrem. This will give an addition of about 2% overall to the dose from natural background radiation, equal on average to approximately 100 mrem per year. Calculation of the radiological consequences of the Chernobyl accident, based on the concept of the no-threshold linear relationship between dose and effect, shows that the additional mortality from cancer may amount to only about 0.01% of the mortality level from spontaneous cancer. This addition is absolutely undetectable against the background of fluctuations in natural oncological mortality of the population.

The expected genetic radiological consequences will be similarly insignificant.

The measures to monitor the consumption of food products have reduced the internal exposure dose of the population by a high factor. It should be noted that, in the absence of such monitoring, the internal exposure dose due to consumption of local products could be 10 times higher than the external exposure dose.

Assuming that the required amount of monitoring of the consumption of food products will continue in the future, it can be assumed that the internal exposure dose of the population will be at the same level as the external exposure dose.

4.7 Gosagroprom measures. Radio-ecological investigations

In the physical geography respect, the contaminated territory lies in the south-west part of the east European plain and partly in the Pripyat Polessie (the water catchment basin of the river Pripyat), whose eastern part adjoins the Dnieper depression. In its overall topography the region is a plain whose maximum elevation does not exceed 200 m. The climate is moderately continental, with warm summers and comparatively mild winters, the mean annual precipitation ranging from 500 to 650 mm, with about two thirds falling in the warm part of the year.

The soil covering of the southern regions of Belorussia has a soddy podzolic and peat bog character, while that the south-eastern regions is soddy podzolic, loam and sandy loam.

The Polessie region (southern region of the Gomel oblast [district], northern regions of the Kiev and Zhitomir oblasts) is characterised by widely distributed soddy podzolic sandy, and sandy loam soils combined with large areas of low peat bogs. Light textured soils occupy 58% of the area. All of the soddy podzolic soils of the Polessie are distinguished by low natural fertility, as a rule acid (pH 4.5-5.5), with a low content of mineral nutrients (including calcium, phosphorus, magnesium). The topography is not rugged, although the micro-topography, particularly in the Belorussian Polessie, is strongly expressed, which, combined with the swampy state results in the low contours of the agricultural land. About 25% of the territory is arable. Up to 50% of the agricultural area is under natural fodder (grass-sedge meadows). This natural environment has formed a specific type of agricultural production. Dairy and meat cattle breeding (up to 60 cows per 100 ha). An important place is occupied by potatoes (abot 8% of the area), fodder (35-40%), grain (about 50%), and long-stemmed flax (up to 5%).

The bulk of the forests in the contaminated territory are in the Polessie regions (70% forested). The main forest species (63%) are conifers (pine), with the remainder being broadleafed (oak, hornbeam, birch, alder).

A zone of forest steppe begins with the southern Ukraine Polessie (southern branch of the radioactive track), whose soil cover has podzolised chernozems, and grey and light grey podzolised soils on loess deposits. The predominant species in the forests of the Ukraine Polessie is pine with admixtures of birch and oak, while in the forest steppe regions there are small areas of oak, hornbeam and lime woods.

An analysis of the data characterising the distribution of the contaminated territory in terms of agricultural management shows that approximately a half of the contaminated ground of Belorussia is agricultural (41-50%), and up to 52% natural (forest, marsh, bodies of water). In the Ukraine, the fraction of contaminated territory belonging to the natural landscape decreases from 46% in the north (Chernobyl region) to 10-12% in the south. Overall, approximately a half of the area of the contaminated territory is natural (forest, marsh, not cultivable), while the flat, open sections of the locality are virtually completely occupied by agriculture. This leads to the conclusion that the estimation and forecasting of the ecological consequences of the radioactive contamination of the territory requires that as much detailed study and monitoring be devoted to migration processes in the natural ecosystems as in the agricultural systems.

A large part of the agricultural land inside the 30 kilometre zone has suffered radioactive contamination, and approxiamtely 2.0 million hectares beyond its boundary (August 1986 data). Specialists from Gosagroprom together with other organisations have estimated and forecast the radiological situation in agricultural production, and have monitored agricultural products and their consumption, etc, for radioactive contamination.

The results have been analysed taking into account existing scientific data on agricultural radiology, radio-ecology, migration of radioactive substances along the food chain, radiation safety standards, etc. On the basis of the analysis, decisions have been taken on measures for the preservation of the natural and material resources of the industry, ensuring the quality of production and preventing its loss as a result of its radioactive contamination, and ensuring the profitability of agricultural production in zones of moderate radioactive contamination while observing the radiation safety requirements of the population.

Depending on the level of radioactive contamination of food products, decisions were taken as follows:

- complete prohibition of consumption as food or use as cattle fodder or for processing;

- a change in the technique of storage, processing or use;

- permission to use as food provided the level of contamination laid down in radiation safety standards is not exceeded.

The whole contaminated territory was divided into several zones (in the first months, on the basis of gamma radiation, and then on the basis of the caesium-137 content of the soil; the content of other radionuclides in the zone monitored by Gosagroprom was not considered, since they have a low biological activity).

With a contamination level above 40 Ci/km^2 of caesium-137 the use of the land for agriculture is prohibited. It is recommended that such land be transferred to Goslesfond [State forestry resources] for the setting up of a special reserve.

In areas with a lower level of contamination, agricultural production is permitted with various limitations and recommended measures depending on the nature and level of contamination. These measures include the following:

- a change in the sowing pattern of crops and in the trends of animal husbandry;

- on arable, meadow and pasture land the carrying out of special agricultural improvements (addition to the soil of increased doses of mineral fertilisers, lime, sorbents (clay suspensions and zeolites) to the upper contaminated layer, followed by ploughing in).

These measures are aimed at reducing the transfer of radionuclides from the soil to the productive part of the harvest.

The effects of radioactive contamination on the natural environment

The direct effects of radiation on the plant and animal communities in the form of damage to confiferous forests and noticeable changes to the number of soil fauna appeared in the limited zone of strong radioactive contamination at a distance of several kilometers from the Chernobyl nuclear power station.

As might be assumed, the most sensitive to radioactive contamination were pine forests.

Lethal effects of the irradiation of pine trees appeared at the end of the summer of 1986. The area of forest which had died adjoining the nuclear power station site on the west side was 400 hectares [2].

Broad leafed tree species (represented in the zone of severe radioactive contamination around the nuclear power station mainly by birch, aspen and oak) were virtually unaffected, since their resistance to radiation is 10 times greater than that of conifers. Neither were any morphological changes to rbaceous plants detected in this zone.

Outside of this small zone no visible effects of radiation damage to flora and fauna were detected. The prognosis for the state of health of animals in the contaminated territory is satisfactory.

The main attention in the zones of moderate and weak radioactive contamination was devoted to studying the migration characteristics of radionuclides in natural ecosystems.

A network of so-called landscape-geochemical polygons was laid down under various physical geography and landscape conditions in order to monitor the radionuclide content of components of the natural ecosystems.

More detailed questions of this section are dealt with in special reports of Soviet specialists and in references [2 and 3].

5 Conclusions on the Chernobyl accident and further development of nuclear power in the USSR energy programme

5.1 General. Role of nuclear power in the USSR energy programme

The Chernobyl accident compelled specialists throughout the world once more to examine critically both the plans for the development of nuclear power, and measures to ensure its safety.

The Chernobyl lessons have obviously been studied carefully in the Soviet Union. We have come to the following conclusions.

1 The causes of the Chernobyl accident are connected in the first place with errors of the power station personnel, and with infringements by them of prescribed power station operating rules. These causes do not by themselves have a specifically nuclear character and so cannot be considered as fatal to the development of nuclear power.

2 The analysis of the accident did not reveal any physical phenomena which had not been previously studied theoretcially or experimentally as part of the safety analysis. The analysis showed that the safety of nuclear power plants may be increased by well known physical and technical techniques and with more careful account being taken of the human factor.

3 The analysis of the results of the Chernobyl accident show that although the damage it caused is highly significant both from the point of view of loss of human life, and from the economic point of view, it is comparable with the damage caused by other large industrial and transport accidents which have been analysed.

4 If nuclear sources of power were to be exchanged for traditional ones, the risk to the health of the population and the environment would increase considerably.

5 The reasons demanding the development of nuclear power in the Soviet Union have not disappeared: on the contrary, they will become all the more important with time.

These reasons relate in the first place to the following necessities:

- to remove the geographical disproportion between the production and consumption of fuel;

- to displace oil and gas from power engineering and to optimise the fuel and power balance of the country;

- to save manpower resources.

In addition, the updating of industry in the European part of the country demands an increase in the production of electrical power and, naturally, its more efficient use. The building of towns in the northern climatic zone is impossible without the production of heat and electricity using nuclear sources, otherwise, we will not be able to solve the transport and ecological problems.

Thus the analysis to which we subjected our attitude to nuclear power after the Chernobyl accident has not led to any change in our main positions. We are still convinced of the necessity for its development both for the economy of the Soviet Union, and for the world economy as a whole. Our plans for the introduction of nuclear power plants have not basically changed and will be refined in the new USSR power programme [Energeticheskaya Programma SSSR].

However, the Chernobyl accident, like that in nuclear power stations in other countries, shows that questions of safety in nuclear power have still not been finally settled. The lessons of these accidents for us and the whole world community consist above all in that a new complex technology arising in the process of the scientific and technical revolution demands the most careful treatment of its safety and reliability, and a negligent and unskilled treatment is not excused.

In the Soviet Union after the accident a package of both organisational and technical measures was adopted, aimed at significantly increasing the safety of nuclear power.

Among the first were the development and implementation of technical solutions to exclude the possibility of the repetition of an accident on RBMK reactors similar to that at Chernobyl.

As a result of the analysis of the accident, a package of measures was adopted to increase the safety of nuclear power stations of all types. This package includes both the implementation of measures already planned and the carrying out of new ones basically involving the latest advances in science and technology, operating experience, eg, of the improvement of the diagnosis of the condition of the metal of pipework and equipment, and the wider use of the automatic control of engineering processes. A critical analysis is being conducted of questions involving the siting of nuclear power stations.

An examination and evaluation has been conducted of the state of theoretical and experimental investigations into the safety of nuclear power stations and measures have been worked out for their extension, improvement and intensification.

Computer programs are being improved for the safety analysis of the behaviour of nuclear power stations under all possible transient and accident conditions, including beyond-design-basis conditions, and modelling systems and complexes are being developed.

Investigations are being extended into the possibility of building reactors with passive safety systems - so-called inherently safe reactors, whose cores cannot disintegrate in any accident.

Investigations into quantitative probability safety analysis are being intensified, and also the analysis of risks from nuclear power, the development of conceptual and methodological bases for optimising radiation safety and the comparison of radiation hazards with other types of hazards from industrial activity.

The system of supervision and standard specifications in the USSR embraces all the basic questions of nuclear power station safety and continues to be improved. In 1985, under the aegis of Gosatomenergonadzor SSSR [State committee for the supervision of the safety of work in nuclear power engineering of the USSR] a Master check list and plan for developing rules and standards in nuclear power [Svodnyi perechen' i plan razrabotki pravil i norm v oblasti atomnoi energetiki] was drawn up, which co-ordinates and directs the activities of all departments developing and systematising the corresponding scientific and technical documentation.

The existing standard specifications requirements relating to safety do not basically need reviewing. However, their practical implementation demands more careful monitoring. The quality of training and of retraining of personnel must be raised, and the quality control of equipment manufacture, installation and commissioning must be strengthened by the constructors and designers, and also their responsibility for the subsequent effectiveness and safety of the operating nuclear power stations.

In order to increase the level of leadership and responsibility for the development of nuclear power, and to improve the operation of nuclear power stations, an all-union Ministry of Atomic Power Engineering [Ministerstvo atomnoi energetiki] has been formed.

A number of measures to improve government supervision of safety in nuclear power engineering are intended. Measures are being taken to increase the responsibility of personnel for the operating quality of nuclear power stations.

5.2 Measures to enhance the safety of nuclear power plants using RBMK reactors

See Canadian translation

AECL - SP2-F4 87-12654

5.2 MEASURES TO ENHANCE THE SAFETY OF NUCLEAR POWER PLANTS USING RBMK REACTORS

The first and most important task after the Chernobyl accident was to develop and implement technical measures which would make it possible to eliminate - at existing plants with RBMKs and at plants still under construction - those features which had contributed most to the development of the accident and which had made it such a serious accident. During the year past our main ideas concerning the development of the accident have undergone no substantial change. Investigations of the accident process on one-dimensional and three-dimensional integral models of a power generating unit using an RBMK-1000 reactor have continued. Factors such as the non-uniformity of certain physics characteristics within the reactor, the possibility of a two-way coolant leakage accompanying an increase in core pressure, transport lags in the piping and the compressibility of the steam-water mixture have also been taken into account in these models.

One lesson of Chernobyl is that violations of the operating rules can take the most unpredictable forms; this is something we must always bear in mind. The first essential task, then, was to exclude the possibility of an uncontrolled power excursion due to violations of the rules. From this point of view the most important feature to take into account was, first, the positive void coefficient of reactivity, α_φ, and the corresponding positive reactivity effect when the core is deprived of water; secondly, the insufficiently fast action of the AZ (emergency protection system) when the rules governing operation with minimum excess reactivity - in both transient and steady-state regimes - are violated. The calculations with the different models give similar results: for example, they indicate fast shutdown of the reactor if the prescribed excess reactivity (15 rods) is observed at the moment when the emergency system operates.

As we know, before the accident the excess reactivity in Unit 4 of the Chernobyl plant was substantially below that required by the rules, and the possibility was not excluded that positive reactivity would be introduced in the first few seconds after the AZ-S (emergency protection) button was pressed. Analyses with one-dimensional models of the reactor have shown that if the calculated initial vertical field varied within limits set by the differences between the indications of the various sensors and if the possible errors in these readings were applied (up to 25% for reactor power of 200 MW), the positive reactivity introduced could vary within the range from zero to 1.5 β.

To completely exclude the possibility of local increase in reactivity, or the insertion of positive reactivity during control rod drop and to increase the worth of the AZ rods over the first sector of rod movement, it was decided immediately after the accident to fix the rods in the raised position at a depth of 1.2 metres into the core (see Fig 5.1(a) and (b)). However, this led to a distortion of the vertical fields and made it necessary to reduce the power by 10 to 15%. At present the design of the rods has been changed: the connecting link between the rod and the "displacer" has been lengthened so that it is now possible to reduce vertical misalignments of energy release (see Fig 5.1(c)).

The working excess reactivity compensated by the SUZ (control and protection system) rods has been increased to 43-48 manually operated regulating rods on the RBMK-1000 and to 53-58 rods on the RBMK-1500; this substantially improves the fast action of the AZ (emergency protection system) over the initial sector of rod insertion for rods located in the middle (vertically) of the core which have a large differential worth. As a result the initial rate of negative reactivity insertion by the rods in response to AZ signals is at least $0.5\beta_{eff}/s$. Furthermore, certain other improvements have been made in the SUZ which increase the reliability and safety of reactor operation:

- The number of short absorber rods inserted into the core from beneath has been increased to 32 on the RBMK-1000 and to 40 on the RBMK-1500.

- A scheme has been developed for the insertion of short absorber rods into the fore in response to AZ signals;

- The digital excess reactivity display now gives readings for any and all states of the reactor; and

- Automatic shutdown of the reactor will now occur when the excess reactivity is reduced to 30 manually operated rods.

The original servo drives of the SUZ rods have been modernised so that the time required for full rod insertion in response to the AZ signals can now be reduced from 18-20 s to 10-12 s.

A fast-acting emergency protection system is now being developed which is to be introduced on existing power station units and which should make it possible to introduced up to $3\beta_{eff}$ of excess reactivity in 2-2.5 seconds.

In deciding on the first steps that should be taken to reduce α_φ, calculational data were used which clarified the nature of the change in the void coefficient of reactivity when the reactor loses water. The influence of the additional absorbers and of the number of SUZ rods positioned in the core on α_φ was studied in special experiments carried out at the Chernobyl and Smolensk nuclear power plants in October and November 1986.

As a result of these investigations, it was decided to increase the number of additional absorbers (AA) in the core of the RBMK-1000 to $\sim$80.

A number of typical α_φ values actually measured on various units using RBMK reactors are given in Table 5.1.

Apart from reducing α_φ by the insertion of additional absorbers and increasing the minimum permissible excess reactivity on the SUZ rods, a number of other measures aimed at improving safety have been taken. In particular, an additional illuminated indicator panel has been installed in the control room, which fixes the withdrawal from operation of the reactor's emergency protection system in each signal. Interference of the operating personnel in the functioning of this panel (extinguishing of the signal for example) is completely impossible.

Another approach to the improvement of safety at existing units lies in a substantial enlargement of in-core power density control both in the vertical direction and over the radius of the core. Special miniaturized energy release detectors have been developed for this purpose and are being installed on existing units. Another project is in progress which aims at modernising the existing system of diagnostics and parameter recording: this should make it possible to identify and determine the nature of accident situations as they develop, and also to establish the actions of operating personnel. This improved system is to be removed to a separate complex which is quite independent of the existing plant computer and is provided with reliable autonomous power feed. A good deal of attention is also being given to the development of special ultrasound and acoustic-emission systems for monitoring the state of the metal in the piping during operation.

During the current year the list of design accidents and "super-design" accidents subject to analysis in the course of the safety assessment of power plants with RBMK reactors has been substantially enlarged; these analyses now make allowance for changes in the physics characteristics of the reactor and other changes described above.

Other measures aimed at enhancing the safety of RBMKs are associated with increasing the fuel enrichment. Both calculations and experiments have indicated that an increase in the enrichment of make-up fuel from 2 to 2.4% allows a further reduction of the void coefficient of reactivity. Reactor tests on 146 fuel assemblies enriched to 2.4% have been carried out at the Leningrad nuclear power plant, and as a result the decision was taken that the RBMK-1000 reactor should go over to this type of fuel (Table 5.2). The use of a moderate number of additional absorbers in the core (up to 80) along with fuel enriched to 2.4% should make it possible to reduce α_{ϕ} to a value below that of β_{eff}. At the same time the use of 2.4%-enriched fuel in the core and 80 additional absorbers will yield a fuel burnup actually somewhat in excess of the burnup obtained with the 2%-enriched fuel, other operating conditions remaining the same.

For reactors at present under construction, we are considering the possibility of reducing the void coefficient of reactivity by limiting the amount of graphite in the pile and in the reactor - in particular by shearing off the ribs of the graphite blocks. Reducing the amount of moderator in the core by cutting off the fins of the graphite blocks should make it possible to obtain the desired void coefficient without changing the geometric spacing of the channels in the core (the cathetus at the base of the triangular prism is to be 5-7 cm), to reduce the initial power of freshly loaded fuel assemblies, and to enhance the utilisation of uranium-238 in the fuel cycle. The burnup is not very sensitive to the amount of graphite in the cell (changing the cathetus of the triangular prism base by one centimetre close to the optimum value of the effective spacing should reduce fuel burnup by only 0.04%).

Table 5.2. Principal characteristics of the RBMK-100 with fuels of different enrichment

Characteristic	Initial enrichment, %											
	2	2,4	3	2	2,4	3	2	2,4	3	2	2,4	3
Number of additional absorbers in reactor		0			30			80			110	
$\alpha_{\varphi}, \beta_{eff}$	~~4,5~~ 20·7 mk	3,2	1,3	3,4	2,1	0,2	1,6	0,2	−1,5	0	−2,6	−5,1
Fuel burnup, MWd/kg	22,3	28,8	37,6	20,7	27,1	36,0	18,0	24,5	33,4	16,3	22,7	31,6
Maximum linear load, W/cm	315	360	390	305	340	380	285	320	360	270	300	340

Table 5.1. Measured values of α_φ

Unit	Date of measurement	Number of additional absorbers	Excess reactivity (manually operated rods)	α_φ
Leningrad I	30 March 87	80	43	1,0 ± 0,2
Leningrad II	13 March 87	79	46	0,8 ± 0,1
Leningrad III	8 May 87	80	42	1,1 ± 0,2
Chernobyl II	21 November 86	81	43	1,0 ± 0,2
Kursk IV	25 May 87	82	43	0,9 ± 0,4

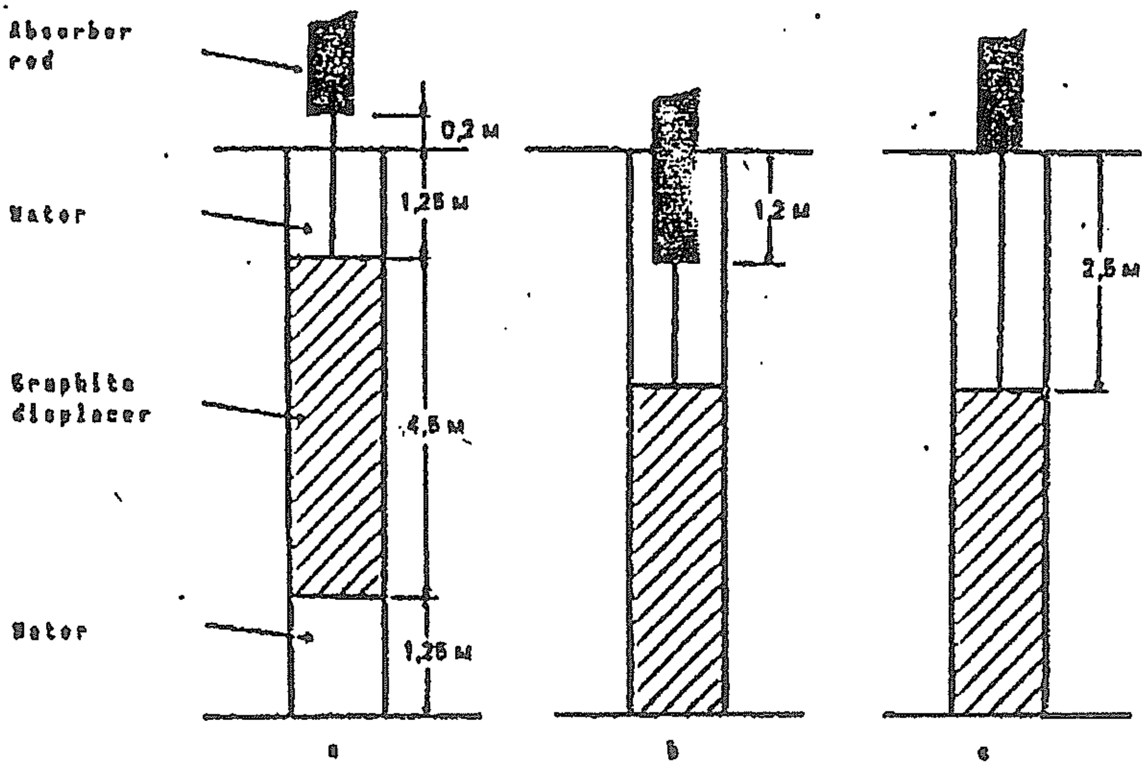

Fig. 5.1. Extreme upper position of regulating rod of the emergency protection system (AZ rod) relative to the reactor core before (a) and after (b) the accident, and at the present time (c).

5.3 Measures to enhance the safety of nuclear power stations with VVER reactors

The VVER reactor is a leading and promising reactor type in the USSR power programme. Its safety level meets international requirements, but measures to increase its safety, too, are being taken for the following reasons:

- an analysis of the actual state of the pressure vessels in relation to their ability to withstand brittle fracture;
- conclusions drawn from the Three Mile Island accident;
- an analysis of the Chernobyl accident.

On the basis of test coupon investigations of pressure vessel steel and data on the actual condition of the pressure vessel metal, primarily of welds in the core region, a number of design and engineering changes have been proposed which will enable the design life of the pressure vessels to be achieved thout hazard. Measures have been developed both to reduce the probability of relatively cold water contacting the reactor vessel, and to reduce the neutron fluence on the pressure vessel.

A number of serious measures to increase the safety of VVER reactors with the development of small and medium-sized leaks were carried out after the Three Mile Island accident.

In VVER-1000 designs a system has been implemented for the forced reduction of pressure in the primary circuit which facilitates the injection of boric acid solution into the circuit in the event of the detrimental development of a loss of coolant accident. The additional facility is provided of the discharge of hydrogen should it accumulate under the reactor closure head and in the steam generator headers, and the draining of the U-section in the cold leg of the reactor coolant pipe.

The Chernobyl accident again compelled us to turn our attention to questions of the ability of the cores of nuclear power station reactors to prevent damage to the fuel by means of internal protective devices, and to questions the adequacy of measures to ensure the safety of the population if, despite the high probability that it would be prevented by design devices, an accident nevertheless occurred such as a core meltdown.

For nuclear power stations with VVER-440 reactors of the first generation, the decision was taken to rebuild with the introduction of additional passive and active core cooling systems and the fitting of acoustic emission and acoustic noise diagnostic systems on large diameter pipework.

The following directions for increasing the safety of VVER designs are being considered:

- the development of cores with increased internal safety;
- improving the reliability of the power station basic and auxiliary equipment;
- ensuring the passive removal of decay heat under conditons of complete loss of grid, including loss of standby AC power supplies (diesels);

- providing nuclear power stations with automated control systems for the basic process, with increased reliability and including up-to-date systems for diagnosing the condition of the basic power station equipment.

The aim of measures to increase safety is to prevent the possibility of serious damage to the core from fuel meltdown or an unacceptable rate of power release in all emergency situations with a probability of 10^{-5}-10^{-6}1/year·reactor, and at the same time the probability of release of radioactive products beyond the protective barriers, with the permitted levels of radiation to the population and contamination of the environment being exceeded, must not exceed 10^{-7}1/year·reactor.

5.4 Improvement of approaches to ensuring the safety of nuclear reactors

The analysis of the causes and consequences of the Chernobyl accident did not reveal the need for a re-examination of the general concept of ensuring the safety of nuclear power which, both in the USSR and in other countries, is based on a multiple barrier system of isolating from the environment radioactive substances in the nuclear reactor, together with a package of engineering and organisational measures ensuring the operating safety of the nuclear power station.

In the USSR investigations into the further improvement of preventive safety measures and measures to reduce and remove the consequences of accidents have been intensified. A number of areas have been specified for the application of effort to give the most effective increase in nuclear power safety. In particular, the many scientific investigations carried out during the period after the accident have shown the following.

1 The safety of nuclear power stations must be improved by optimising the interaction between 3 of their main elements: equipment, operating procedures, and personnel. Here in the first place it is necessary to show that the Chernobyl accident revealed shortcomings in the current man-machine concepts when, in the control of engineering equipment, preference was given to man-operator actions. Analysis of the accident showed that a number of safety systems must function exclusively on the basis of signals from the technical systems for monitoring plant parameters, and not from indications from the operators. An example of the implementation of such an approach on RBMK reactors would be the fitting of an automatic system for calculating the operating reactivity margin with the generation of a reactor emergency shutdown signal if the reactivity margin is reduced below a set level.

Another conclusion from the Chernobyl accident affecting the area under consideration is the need for more thorough training of nuclear power station operators and provision of computerised aids, ie, systems which could prompt the operator with the possible causes of departures from normal running of the reactor, and could guide him in his search for the causes and their elimination. One way of constructing such systems could be the creation of banks of nonconflicting knowledge, as complete as possible, from the most experienced operators, and their procedures for analysing the cause and effect relationships for detartures of the ractor from the normal state. Work in this direction is proceeding in the USSR. This work is accompanied by improvements to the technical facilites which provide the operator with information on the running of the nuclear power station and the development of emergency situations.

In the plan for optimising man-machine relationships in nuclear power, the task of raising the qualification of personnel and the improvement of methods of instruction are linked to the task of building reactors which are simpler to control, and providing the optimum working conditions for the operators.

This includes in the first place:

- questions of the further optimisation of the distribution of responsibility between the operator and the technical facilities in making decisions on reactor control;

- the problem of presenting the operator in the best way with current information on the functioning of the reactor.

And the most important in the long term plan is to increase the level of inherent safety of reactors, up as far as building reactor systems whose physical, chemical and structural properties would not permit an accident to occur with any operator error, infringement of the operating rules or failure equipment.

The question of optimising man-machine relationships is of topical importance for all state-of-the-art technologies. In many of them, as in nuclear power, the question is still far from its final solution. Work in this area demands considerable material and human effort both by way of the theoretical interpretation of the problem, and for carrying out experiments and building reliable and complete mathematical models to describe the functioning of complex systems. For this reason, international co-operation in this direction is important.

2 Nuclear power safety must be improved on the basis of the development of strictly scientifically based safety aims and criteria. Basic in this area, is the answer to the question of what level of safety is acceptable. Until recently both in the USSR and, apparently, in other countries, the majority of resources devoted to safety in nuclear power and in other areas of industry have been spent on improving the technical systems for monitoring and preventing emergency situations. The accidents at Chernobyl, Three Mile Island in the USA, at chemical plants (eg, Bhopal and Basle), etc, have shown that, in spite safety measures taken, something unforeseen may always happen, either through a series of mechanical failures, or as a result of operator error. It is suggested that one of the main lessons to be extracted from these accidents consists of the realisation of the need to optimise the distribution of expenditure on the prevention of accidents and the limitation or elimination of their consequences. Basic for such optimisation may be the determination of the acceptable level of safety.

Because of the one-sided attention paid to accident prevention, society has become too reassured with regard to the possibility of serious accidents, and as a result, industry has not always shown itself ready technically and organisationally to limit and eliminate the results of large-scale accidents.

Thus, in the Chernobyl accident, it became necessary to take many technical and engineering decisions during the course of eliminating the accident under extreme conditions. It was necessarry to carry out urgent and wide-scale experimental work which could have been and should have been carried out beforehand. In the accident, shortcomings appeared in the measuring equipment designed for operation over a wide range of measured parameters. In the initial stage of eliminating the consequences of the accident, there were virtually no facilities for remote sampling under accident conditions or for carrying out other necessary engineering operations. The problem of making special equipment for eliminating serious accidents is currently being solved in the USSR.

This group of questions includes the problem of the routine readiness to use the facilities of basic and applied science to eliminate the consequences of large-scale accidents. It is necessary to point out in eliminating and reducing the scale of accidents the important role of information-and-decision, predicting and recommending systems, including the group of simulation models: a model of the economy of the region, and mathematical models of varying degrees of complexity which can describe the behaviour of chemical and radioactive impurities in the atmosphere, soil, open areas of water and ground waters, and the migration of impurities along the food chains.

The task of such information-and-decision systems is the efficient provision of effort to eliminate the accident by means of information on possible scenarios of the development of the post-accident situation, information on the corresponding influence of these scenarios on the ecology and economics of the region and, finally, to present as efficiently possible versions of solutions to reduce the damage from the accident (eg, plans for the evacuation of the population).

The neccesity for a considerable extension of the scale of investigation and modelling of emergency situations was also revealed by the analysis of the effectiveness of emergency plans existing at Chernobyl.

Thus, the Chernobyl accident raised the question of the need to accelerate wide-scale theoretical and experimental investigations directed at studying scenarios of serious accidents at nuclear power stations. With this aim, the following actions have been taken:

- investigations have been reinforced into quantitative probability safety analysis, analysis of risks from nuclear power, development of the conceptual and methodological bases for optimising radiation safety, and the comparison of radiation hazards with other forms of hazard from industrial activity;

- the state of theoretical and experimental investigations into nuclear power station safety has been examined and evaluated, and measures have been developed for their extension, improvement and intensification;

- computer programs are being improved for analysing the safety behaviour of nuclear power stations under all possible transient and emergency conditions, including beyond-design-basis, and modelling systems and complexes are being developed.

3 Nuclear power safety must be improved on the basis of the optimum choice of nuclear power station sites and of other industrial plants. The attempt to achieve the greatest economies, the maximum utilisation of investments once made in industry, communications, and the social life of any region causes that region to become saturated with different plants, including nuclear power stations, without any obligatory study of their interaction. And it could happen that an accident at one of them would not have important consequences were it not for the effect on adjacent plants with the possible multiplication of injurious factors. The effect of the possible mutual influence of different entities as a function of their power and density of siting becomes all the more important, and the economic damage from accident consequences caused by

the closeness of different plants may exceed the advantage resulting from the closeness of the raw materials base or of transport facilities. The optimum solution of siting requires the controlled joint action of specialists in different areas, able to predict the results of different factors, including those not specific to a given type of production, and the widest use of mathematical modelling techniques.

In the USSR, before the Chernobyl accident, this activity regarding the choice of nuclear power station sites and other industrial plants was differentiated by branch of industry and type of production. The question has now been raised of developing a unified approach in this area. It requires a considerable increase in the scale of mathematical and experimemtal work in this direction.

4 Nuclear power safety must be improved not only from our own operating experience, but also from that of the operation of complex systems in other branches of industry. A comparative analysis of the Chernobyl accident with other serious accidents both in nuclear power (eg, the American nuclear power ·ation Three Mile Island), and in other branches of industry (eg, the ..quefied gas tank accident in Mexico in 1984, and the chemical plant in the Indian town of Bhopal in the same year, etc) will show the obvious similarity of their causes. The scale of large industrial accidents is mainly due to the general industrial tendency to the growth in unit power of process units.

The development of modern technologies aimed at increasing the standard of living of the population, at the same time leads to the possibility of serious accidents with destructive consequences for the environment, and serious consequences for society. This requires profound interpretation and energetic action directed at improving technological processes and industrial structres from the position of safety. This is possible only on the basis of the amalgamation of safety investigations into a single scientific discipline of industrial safety, able to find not only qualitatively new directions for increasing the safety of modern technologies, but also to define the general principles and techniques for creating the next generation of technologies. Within such a single scientific discipline studying the problems of safety in industry, the safety of nuclear power must also be improved.

6 Conclusions

The accident to unit 4 of the Chernobyl nuclear power station called for the mobilisation of considerable efforts and resources for the limitation and eliminaion of its consequences. Many ministries, departments and organisations of the USSR have participated in this work. The governments and different organisations of a number of countries offered their help. This help was gratefully accepted.

The first stage required immediate action to prevent the development of the accident and to protect the health of the population, the power station personnel and of individuals involved in combatting the accident.

At the end of 1986 building of the entombment was completed and units 1 and 2 of the Chernobyl nuclear power station were started up. Among other work during this period, mention must be made of measures required to ensure normal living conditions of the evacuated population, medical, public health and agricultural measures, decontamination on the power station site and in the 30-kilometer zone, and the organisation and implementation of radiation monitoring.

The performance of large scale work to eliminae the consequences of the accident in a short time was possible because of the ability of Soviet society to mobilise available resources, and to concentrate them on desired tasks and to provide a high level of organisation of the measures undertaken.

The Chernobyl accident demanded the critical analysis of the state of safety in nuclear power with the aim of raising its level.

Above all, the object of analysis was the causes, progress and consequences of the accident, the measures to combat the accident, and their effectiveness. Initial measures were developed and implemneted to increase the safety of nuclear power stations with RBMK reactors. The broader plan examined all aspects of ensuring safety, including:

- technical facilites;
- organisational and standard specification provision;
- scientific and technical provision;
- emergency plans and resources for their implementation;

On the basis of this analysis a long-term plan was developed for improving the safety of nuclear power, including work along all the lines mentioned. In this plan, considerable attention is devoted to questions of improving control and man-machine interaction. This includes improving facilities for automation and monitoring, provision of information, and training of personnel, particularly with regard to action in emergencies. Programmes are being developed and long-term investigations organised to study the remote consequences of accidents, and also measures to limit and eliminate them.

The question has been raised of improving the scientific and technical bases of safety evaluation, analysis and control.

Great importance is being attached to work towards the building of the new generation of nuclear power reactors of the so-called inherently safe type.

It is impossible to consider the causes and scale of serious accidents in nuclear power stations as exclusively a property of nuclear installations. As th serious accidents in non-nuclear areas, they are mainly determined by the general industrial tendency to the growth of the unit power of process units, the involvement in production of a large number of different harmful substances, the complication of control systems, and shortcomings in man-machine interaction, etc.

The critical analysis to which nuclear power was subjected after the Chernobyl accident did not lead to any change in our positions in relation to the development of nuclear power in the USSR or in the world in general. Our plans to introduce nuclear power plants have not changed significantly.

However, the Chernobyl accident, like other accidents in the nuclear and other areas, indicates the necessity to raise the level of safety in nuclear power and in other branches of industry. The lessons of these accidents for us and for the whole world community consist mainly in that a new complex technology arising in the process of the scientific and technical revolution demands the most careful treatment of questions of its safety and reliability, and negligent and unskilled treatment is not excused.

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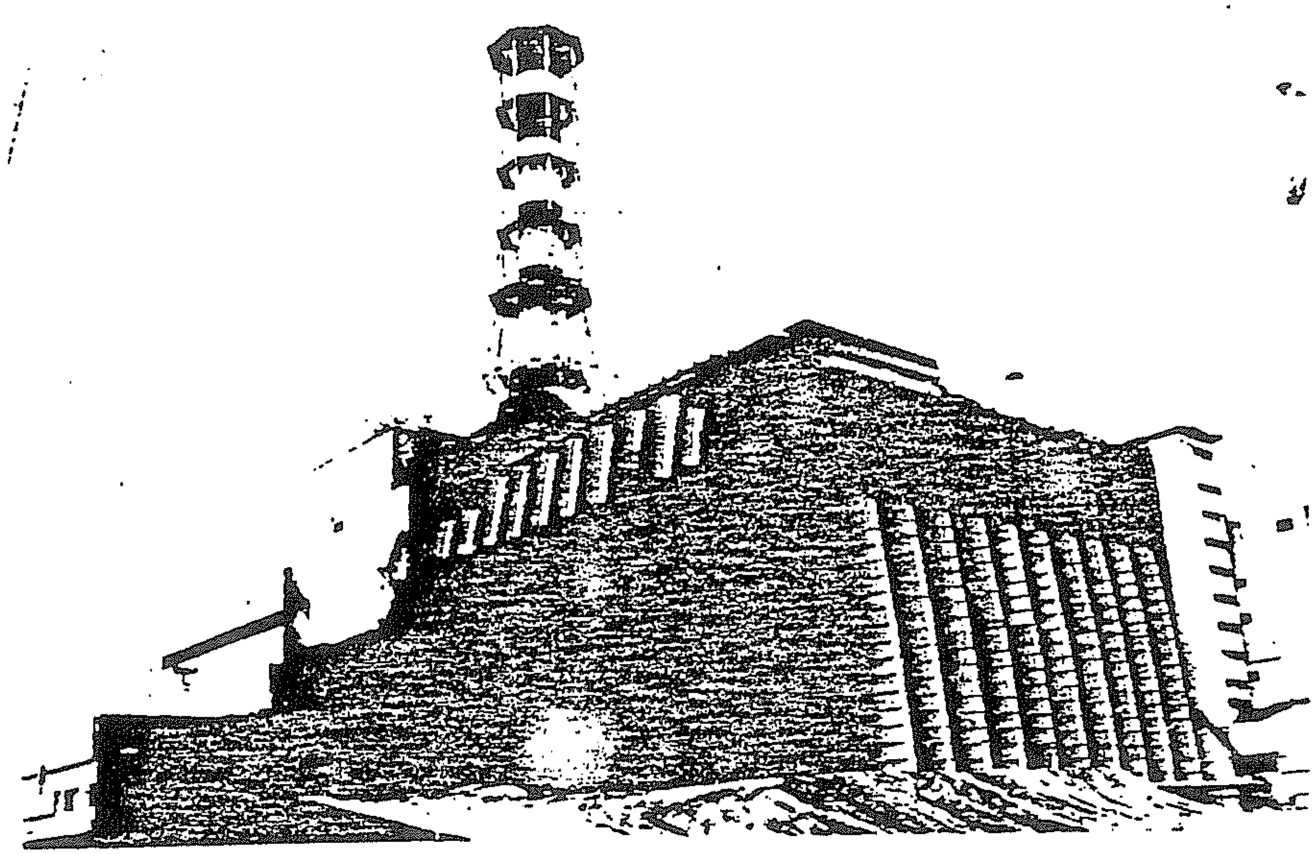

РИС. 2.1. ВНЕШНИЙ ВИД УКРЫТИЯ ПОСЛЕ ЗАВЕРШЕНИЯ ЕГО СТРОИТЕЛЬСТВА (ФОТО)

Figure 2.1 Outside view of entombment after its completion (photograph)

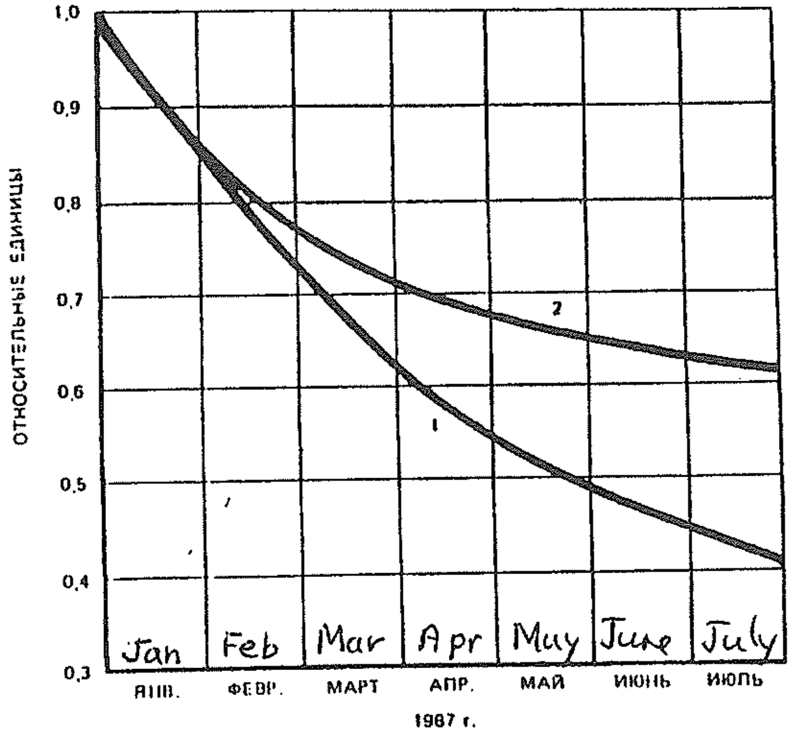

РИС. 2.2. ИЗМЕНЕНИЯ ИНТЕНСИВНОСТИ ГАММА-ИЗЛУЧЕНИЯ ВНУТРИ УКРЫТИЯ (1) И ТЕМПЕРАТУРЫ В НАИБОЛЕЕ "ГОРЯЧЕЙ" ТОЧКЕ В ОДНОМ ИЗ ПОДРЕАКТОРНЫХ ПОМЕЩЕНИЙ (2)

<u>Figure 2.2</u> Variation of intensity of gamma radiation within the entombment (1) and temperatures in the "hottest" spot in of the enclosures under the reactor (2)

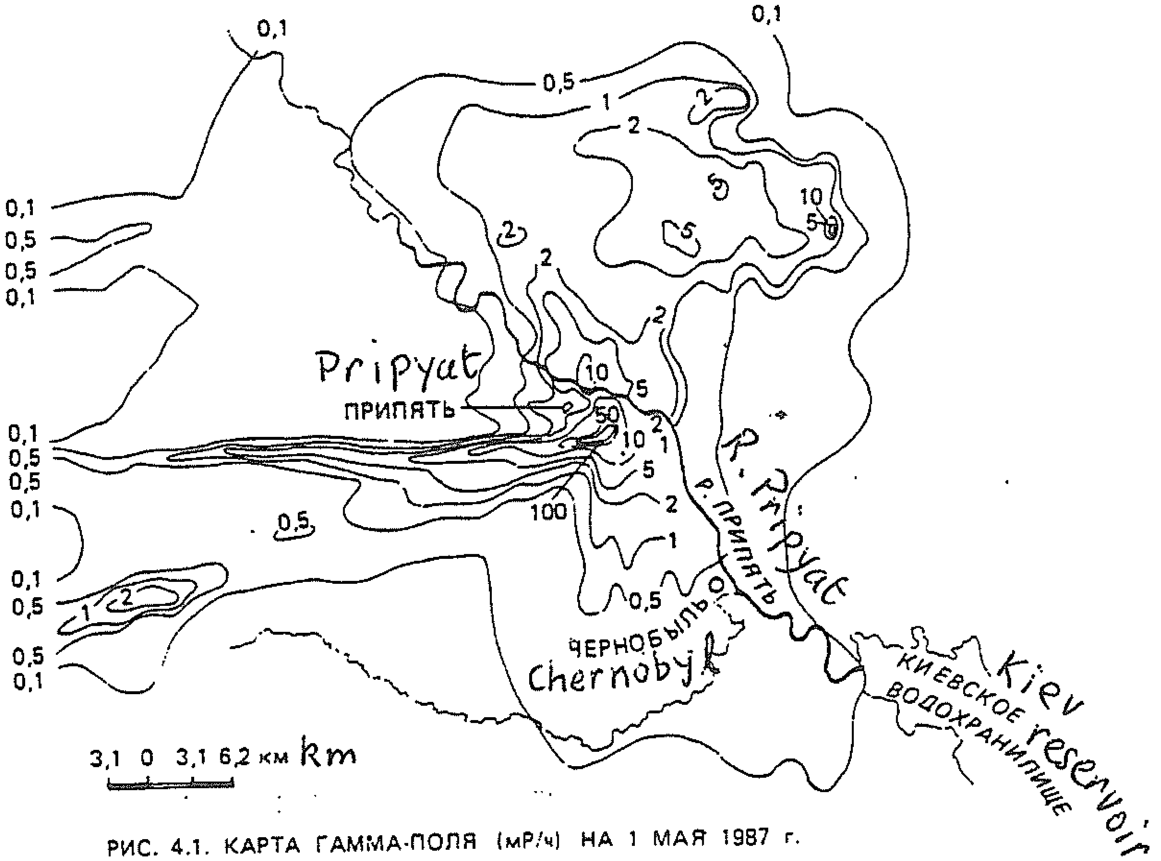

РИС. 4.1. КАРТА ГАММА-ПОЛЯ (мР/ч) НА 1 МАЯ 1987 г.

Figure 4.1 Gamma field map (mR/h) as at 1 May 1987.

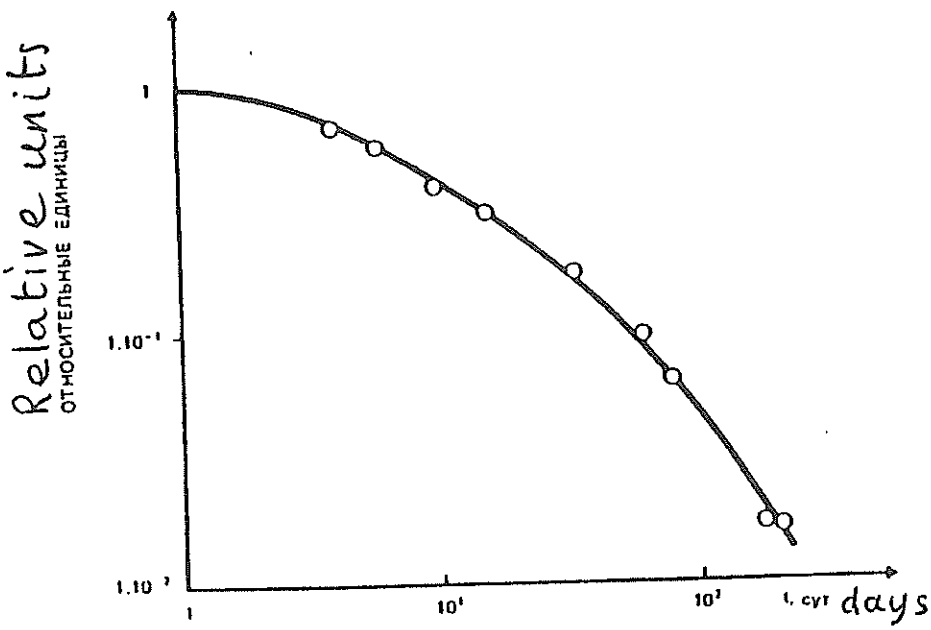

РИС. 4.2. ИЗМЕНЕНИЕ МОЩНОСТИ ДОЗЫ ГАММА-ИЗЛУЧЕНИЯ РАДИОАКТИВНЫХ ПРОДУКТОВ НА БЛИЖНЕМ СЛЕДЕ ВО ВРЕМЕНИ ПО ДАННЫМ АЭРОГАММА-СЪЕМКИ

Figure 4.2 Variation of gamma dose rate of radioactive products on the near track during the time of the aerogamma plot data.

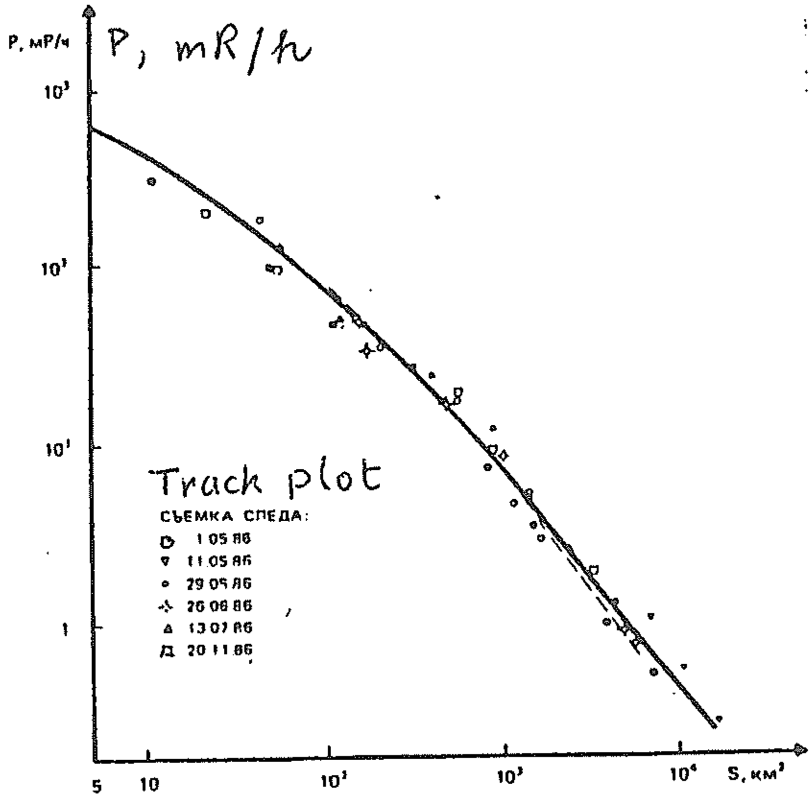

РИС. 4.3. ВЗАИМОСВЯЗЬ ПЛОЩАДИ РАДИОАКТИВНЫХ ВЫПАДЕНИЙ НА БЛИЖНЕМ СЛЕДЕ И ИЗОУРОВНЯ МОЩНОСТИ ДОЗЫ (ДАННЫЕ ПРИВЕДЕНЫ НА 29.05.86 г.): ——— – СРЕДНЯЯ ПО ИЗМЕРЕНИЯМ, – – – – РАССЧИТАННАЯ

<u>Figure 4.3</u> Interrelation between area of radioactive fallout on the near track and isodose lines (data adjusted to 29.05.86).

_______________ mean of measurements; ------------ calculated

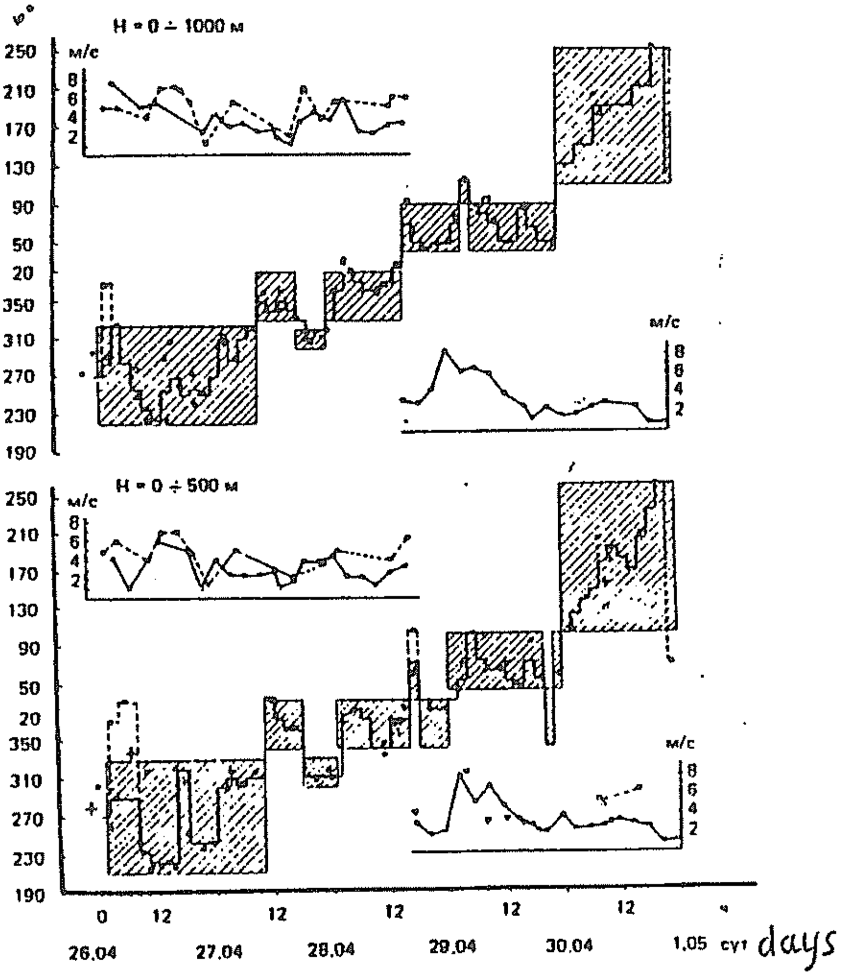

РИС. 4.4. СРЕДНИЕ В СЛОЕ 0 – 500 И 0 – 1000 м ЗНАЧЕНИЯ НАПРАВЛЕНИЯ И СКОРОСТИ ВЕТРА С 26.04 ПО 1.05.86 г. В РАЙОНЕ, ПРИМЫКАЮЩЕМ К ЧАЭС. ✦ – КИЕВ (РАДИОЗОНД), ○ – КИЕВ, АЭРОПОРТ: ● – БОРИСПОЛЬ, △ – МОЗЫРЬ, □ – ГОМЕЛЬ, ▼ – ЧЕРНИГОВ

Figure 4.4 Mean values for wind speed and direction in the 0-500 and 0-1000 m ł ds from 26.04 to 1.05.86 in the region adjacent to the Chernobyl nuclear power station.

Kiev (radiosonde) - solid circle with cross superimposed
Kiev (airport) - hollow circle
Borisopol' - solid circle
Mozyr' - hollow triangle
Gomel' - hollow square
Chernigov - solid triangle

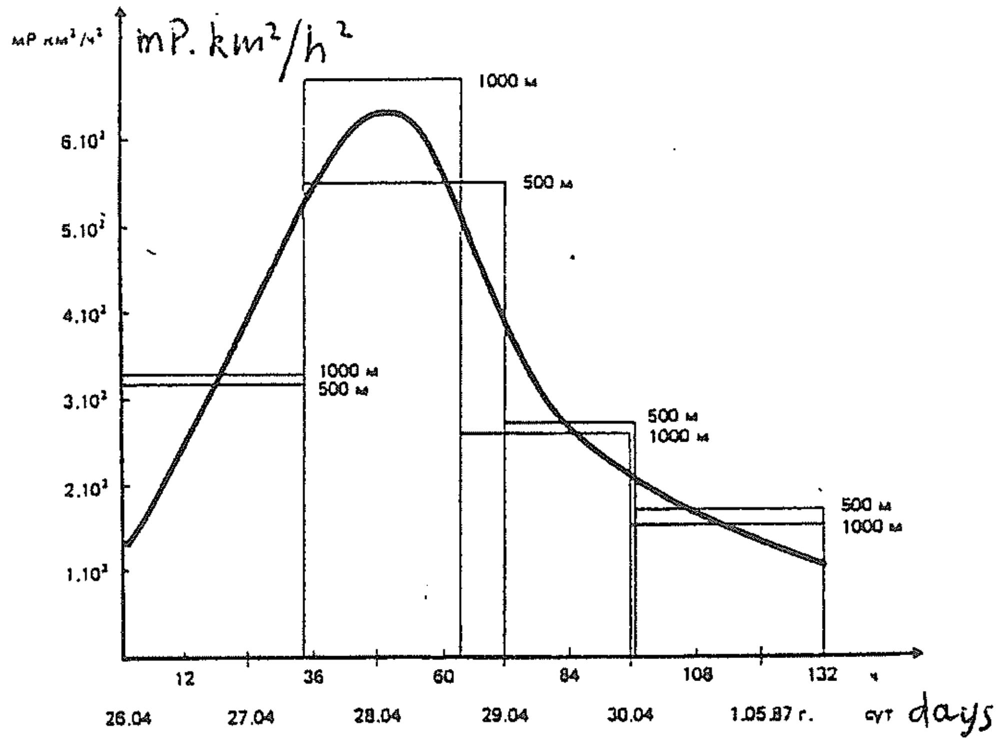

РИС. 4.5. ПОЧАСОВЫЕ ВЫПАДЕНИЯ ГАММА-РАДИОАКТИВНЫХ ВЕЩЕСТВ НА БЛИЖНЕМ СЛЕДЕ

Figure 4.5 Hourly fallout of gamma radioactive substances on the near track

K ·y: 1 - Number; 2 - full discharge; 3 - fallout of radioactive substances on n. ar track; 4 - total radioactive substances; 5 - individual radionuclides; 6 - mean; 7 - cerium-144; 8 - caesium-137; 9 - caesium-134.

Т а б л и ц а 4.1. Относительное распределение выброса радиоактивных веществ за первые пять суток и их выпадений на ближнем следе (апрель 1986 г.)

1 Число	2 Полный выброс [1]	3 Выпадение радиоактивных веществ на ближнем следе					
		4 Сумма радиоактивных веществ			5 Отдельные радионуклиды		
		Н = 1000 м	Н = 500 м	средн. 6	церий-144 7	цезий-137 8	цезий-134 9
26	0,32	0,17	0,17	0,17	0,19	0,1	0,09
27	0,24	0,29	0,25	0,28	0,28	0,3	0,31
28	0,19	0,29	0,29	0,29	0,3	0,4	0,42
29	0,14	0,14	0,15	0,14	0,12	0,18	0,15
30	0,11	0,11	0,14	0,13	0,11	0,02	0,03

Table 4.1 Relative distribution of discharge of radioactive subsatnces in the first 5 days and their fallout on the near track (April 1986).

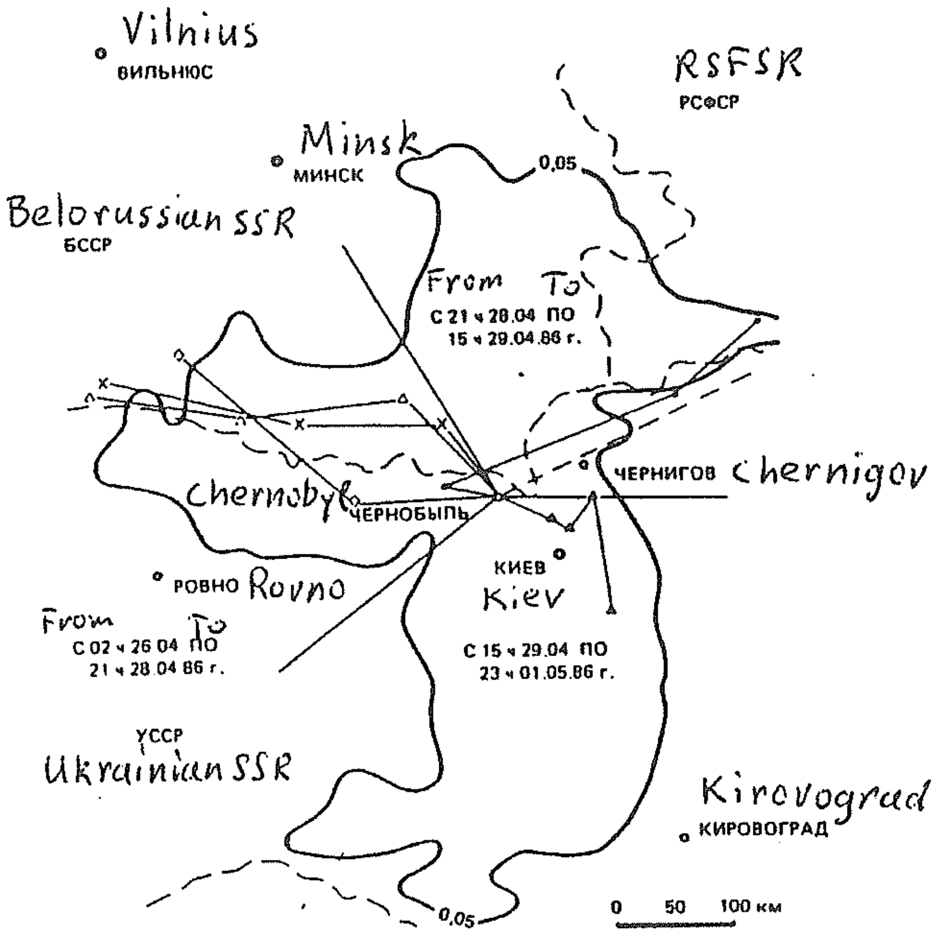

РИС. 4 6. РАСПРЕДЕЛЕНИЕ ГАММА-ПОЛЯ НА ТЕРРИТОРИИ СССР ПО ИЗОУРОВНЮ МОЩНОСТИ ДОЗЫ 0,05 мР/ч НА 10 ИЮНЯ 1986 г И СЕКТОРЫ РАСПРОСТРАНЕНИЯ ТРАЕКТОРИЙ ЧАСТИЦ НА УРОВНЕ 925 мб (х——х – ОТ 3 ч 26 04 86 г., ——— – ОТ 15 ч 26.04 86 г., △——△ – ОТ 3 ч 27.04.86 г., •——• – ОТ 15 ч 27.04 86 г., ▲——▲ – ОТ 3 ч 29.04.86 г., – – – – ОТ 15 ч 29 04 86 г.)

<u>Figure 4.6</u> Distribution of gamma field on the territory of the USSR for the 0.05 mR/h isodose on 10 June 1986 and particle trajectory distribution sectors at the 925 mb level.

x————x from 3 h on 26.04.86
———— from 15 h on 26.04.86
Δ————Δ from 3 h on 27.04.86
———— from 3 h on 29.04.86
- - - - - - from 15 h on 29.04.86

Таблица 4.2. Средние концентрации цезия (10^{-11} Ки/л) в пробах воды Киевского водохранилища, а также рек Припять и Днепр, отобранных в июле 1986 г. — мае 1987 г.

Водный объект 1	2 Июль 1986 г.		3 Октябрь 1986 г.		Апрель — май 1987 г. 4	
	5 цезий-137	6 цезий-134	7 цезий-137	8 цезий-134	9 цезий-137	10 цезий-134
11 Киевское водохранилище	(20 — 50)	(10 — 20)	(1 — 3)	(0,5 — 1,5)	(0,4 — 1,2)	(0,2 — 0,6)
12 р. Припять (ниже Чернобыля)	(40 — 50)	(15 — 25)	(2 — 5)	(1 — 2)	(2 — 5)	(1 — 2)
13 р. Днепр (с. Теремцы)	(1 — 1,4)	(0,4 — 0,6)	(0,5 — 0,6)	(0,2 — 0,3)	(0,4 — 0,6)	(0,2 — 0,3)

<u>Table 4.2</u> Mean concentration of caesium (10^{-11} Ci/ℓ) in water samples from the Kiev reservoir, and the rivers Pripyat and Dnieper, taken July 1986 - May 1987.

<u>Key</u>: 1 - Water body; 2 - July 1986; 3 - October 1986; 4 - April-May 1987; 5 - caesium-137; 6 - caesium-134; 7 - caesium-137; 8 - caesium-134; 9 - caesium-137; 10 - caesium-134; 11 - Kiev reservoir; 12 - Pripyat (below Chernobyl); 13 - river Dnieper (from Teremets).

Таблица 4.3. Сезонные концентрации изотопов цезия (10^{-11} Ки/л) в водах р. Днепр и основных его притоков на территории БССР

Река 1	2 Осень 1986 г.		3 Весна 1987 г.	
	цезий-137 4	цезий-134 5	цезий-137 4	цезий-134 5
6 р. Днепр (на участке Жлобин — Лоев)	(0,5 — 0,8)	—	(0,5 — 0,8)	—
7 р. Сож (в районе г. Гомель)	(5 — 12)	(2 — 6)	(5 — 7)	(2 — 3)
8 р. Беседь (с. Светиловичи)	(2 — 10)	(1 — 3)	(3 — 10)	(2 — 3)
9 р. Припять (в районе г. Мозырь)	(0,5 — 1)	—	(1 — 1,4)	—

Figure 4.3 Seasonal concentrations of caesium (10^{-11} Ci/ℓ) in waters of the river Dnieper and its main tributaries on the territory of the Belorussian SSR

Key: 1 - river; 2 - Autumn 1986; 3 - Spring 1987; 4 - caesium-137; 5 - caesium-134; 6 - river Dnieper (Sozh-Loev section); 7 - river Sozh (in the Gomel region); 8 - river Besed' (from Svetilovicha); 9 - river Pripyat (in the Mozyr' region).

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UNITED KINGDOM ATOMIC ENERGY AUTHORITY

HEALTH AND SAFETY STUDIES COMMITTEE

SYMPOSIUM ON 'CHERNOBYL'

Risley Lecture Theatre, Monday, 13 October, 1986, at 10.15 am

P R O G R A M M E

10.15		Coffee	
10.30	J H GITTUS	Introduction	
	P G BONELL	Description of the RBMK Reactors	HSSC(86)P33
	A N HALL	Degraded Core Accidents: What Happened at Chernobyl	HSSC(86)P34
	P N CLOUGH	Source Terms and Related Characteristics	HSSC(86)P35
12.30		Lunch	
14.00	W NIXON	Environmental Consequences	HSSC(86)P36
	B C CARPENTER	Soviet Emergency Response	HSSC(86)P37
	H J TEAGUE	International Perspective	
15.15		Discussion	
	J H GITTUS	Closing Remarks	
15.45		Tea - FINISH	

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HEALTH AND SAFETY ~~STUDIES~~ COMMITTEE

SYMPOSIUM ON "CHERNOBYL" - 13 October 1986

The Soviet Emergency Response

B.C. Carpenter, Head of Authority Health and Safety Secretariat, London HQ

Introduction

In the immediate aftermath to the accident, the UK media gave the strong impression that the response in Russia had been extremely lax and slow - that the local towns did not know of the accident and their populations had been left, without advice, to their fate; that Moscow had not been informed; that there was no clear plan on how to overcome the technical problem. The information presented at the Post Accident Review meeting in Vienna gave the opposite impression and indeed indicated that the Russians had produced a very impressive level of response which other countries might be hard-pressed to emulate. This note will describe under a number of headings the emergency response made to the Chernobyl accident as reported by the Russians.

Response times are indicated taking A (the accident occurrence time) as 1.24 am on 26 April.

Fire Fighting

Fire had broken out in some 30 places, including on the roof of the reactor building (or what was left of it) at a height of 71 metres. Fighting the fire was seen as the first hazard priority, particularly in view of the threat to Unit 3 of the reactor.

The fire unit stationed at the plant was informed more or less immediately after the accident occurred. Thereafter the sequence was:

A + 7 minutes - Firemen from the nearby towns of Chernobyl and Pripyat called to the nuclear plant.

A + 1 hour - Worst of the concentrations of fire under control.

A + 3½ hours - The fire was extinguished. Graphite was still burning within the reactor. A specialist team was put together which dropped some 5,000 tonnes of compounds of boron, dolomite, sand, clay and lead on to the reactor from helicopters between 27 April and 10 May.

Medical Response

Sequence of events

A + 7 minutes - Medical centre at Chernobyl plant informed of accident.

A + 20 minutes - 2 additional teams of medical staff left Pripyat for the site. Arrangements being made which resulted in 115 beds being available in local hospitals.

A + 45 minutes - 29 personnel admitted to plant medical centre for initial treatment.

A + 1½ hours - Iodate tablets issued to patients and other plant personnel.

A + 4 hours - First symptoms of nausea and vomitting appearing in some patients.

A + 4½ hours - 108 people admitted into local hospitals.

A + 19 hours - Iodate tablets issued house-to-house in Pripyat.

A + 1½ days - Some 350 people had received careful medical examinations including some 1,000 blood analyses

A + 2 days - Of 203 patients diagnosed as suffering from acute radiation syndrome, 129 are flown to Moscow for specialist treatment and the remainder taken to Kiev.

Those suffering from acute radiation syndrome were categorised in 4 severity levels (No. 4 being the most serious). The reports on these patients given at the Post Accident Review Meeting were as follows:

Syndrome Category	No. of Patients	No. of Deaths	Dose Estimate (in Grays)
4	22	21	6-16
3	23	7	4-6
2	53	1	2-4
1	105	-	1-2

We understand that at least an additional 2 deaths have occurred since those figures were produced, making the total 31 deaths.

By A + a few days some 450 medical teams had been assembled both to help at hospitals and to carry out checks among evacuees. The teams amounted to the following personnel deployment:

1,240	doctors
920	nurses
360	doctors' assistants
2,720	medical assistants drawn from personnel in secondary education
720	students from medical institutes

<u>The Central Emergency Organisation</u>

A + 5 hours - Moscow had been informed of the accident and under an established call-out system an expert team was being assembled.

A + 9½ hours - A team of 26 experts was flown in a specially arranged aeroplane from Moscow

From the information given in Vienna, it seems that there was a single emergency control centre set up in the town of Chernobyl, 16 kilometres from the site. The centre was headed by a senior official of the State Committee for Atomic Energy who had all the powers required to demand any resources and ~~to the~~ effort required. Other organisations represented included the State Committee on Hydrometeorology and Environmental Protection, the Ministry of Health, the Academny of Sciences and the State Committee for Agricultural and Industrial Affairs. The support team in the area near the plant included some 1,000 technicians with a range of hardware found necessary, including a large number of concrete mixers (concrete was presumably laid on some of the worst contaminated surfaces over which people needed to go.

There were clearly thousands of troops deployed to the stricken area to carry out a number of support activities.

The central emergency organisation directed the remainder of the emergency response, including decisions on evacuation, food and water bans, movement of cattle and decontamination, as well as directing the necessary action taking place at the plant.

The possibility of rainfall in the region was seen as an additional hazard (in that it would bring down further concentrations of activity). In the event, there was hardly any precipitation for several weeks, although to help this, the emergency controllers organised the dropping of chemicals to scatter clouds.

Evacuation

The largest concentration of population near the plant was 45,000 people in the town of Pripyat. The initial plume missed the town and the situation observed was that:

- the population of Pripyat was (at 1.24 am) in a "dormitory" state which meant that they were, effectively, sheltering.

- the evacuation routes from the town under emergency plans had, however been contaminated by the plume.

It was considered preferable, therefore, to leave the Pripyat population where they were at first.

On the following day (27 April) the wind changed and the plume was directed towards Pripyat. It was then decided to evacuate at A + $1\frac{1}{2}$ days (2 pm on 27 April). The 45,000 people were evacuated (presumably by the army) in $2\frac{1}{2}$ hours. One problem which occurred was the need to provide clothing for the evacuees. In many cases the clothing they wore was found to be contaminated.

When the emergency control centre decided on a control zone of 30 kilometres, the remainder of the 135,000 people in that zone were evacuated within a few days.

The decision was also taken to evacuate some tens of thousands of cattle from the 30 km zone. This was done, using several hundred lorries devoted to this purpose. The Russians said that they operate ERLs indicating (as in the UK) Upper and Lower levels of dose predictions within which decisions are made. An individual dose prediction of 25 rem (whole body) would make them think about evacuation. By the time the prediction was 75 rem the decision to evacuate should have been taken.

Monitoring Resources

It seems from what the Russians reported that emergency plans included the capability of assembling teams to monitor the environment. This included the quite rapid deployment of 2 U 14 aeroplanes plus helicopters and land vehicles.

The monitoring priorities were:

- to make recommendations to protect the people in the vicinity of the plant;

- to assess doses in the 30 km zone;

- to assess doses in areas outside the 30 km zone considered likely to provide useful information on the likely further spread of the problem;

- to assess doses at greater distances.

By the time the full monitoring resources had been assembled, there were more than 7,000 teams from appropriate technical institutions all over Russia.

Comment

The Russians were evidently able to assemble a massive emergency response effort which was deployed rapidly. It seems reasonable to speculate that this was greatly helped by the fact that there is a comprehensive civil defence system in Russia which is taken seriously. This is backed up by a capability for a large number of military personnel to be quickly deployed to the stricken area. Furthermore the population seemingly displayed a much smaller desire to take protective action into their own hands than would be likely in the West. Additionally, there was apparently no pressure from the media and of course none from Parliament. All of these factors helped the emergency controllers to concentrate on their main tasks.

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UNITED KINGDOM ATOMIC ENERGY AUTHORITY

HEALTH AND SAFETY STUDIES COMMITTEE

SYMPOSIUM ON 'CHERNOBYL'

ENVIRONMENTAL CONSEQUENCES

W Nixon, M J Egan and I R Brearley

Safety and Reliability Directorate, Culcheth

October 1986

Distribution:
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Symposium Participants

1 Introduction

Following the accident at Chernobyl, radioactive material was swept across Europe resulting in increased dose levels. This paper considers the time dependent pattern of the spread of contamination throughout Europe and presents estimates of the collective dose to various countries.

2 Atmospheric Dispersion Across Europe of Material Released from Chernobyl

Increased activity levels were first reported on 28 April from environmental monitoring stations in Finland and Sweden, where external dose rates in certain locations exceeded normal background levels by a factor of ten or more. On succeeding days elevated radioactivity concentrations were detected throughout Europe until almost complete coverage had been achieved by 3rd May. Based upon reported measurements conveyed through international bodies (IAEA, WHO and NEA), complemented by computer calculations, it has been possible to assemble a picture of the pattern of dispersion of the material released from the core of the damaged reactor,as it affected western Europe. The progression of this pattern with time is illustrated in Figures 1 to 6. The figures indicate how the external exposure rate varied across Europe from 28 April to 3 May. Note that the monitoring data used as a basis for these plots exhibits a marked patchiness, deriving from (patchy) rainfall patterns and, possibly, uncertainty in the environmental measurements. Such large variations over relatively short distances are not shown in the figures, so that they provide a general picture of the spread of the contamination across Europe.

Referring to Figures 1 to 6, it can be seen that by the 28 April (Figure 1), radiation dose levels had increased in Scandinavia (as has already been noted above), resulting from the generally north-westerly trajectories prevailing at that time. By the 29 April (Figure 2), the contamination had spread further across the Scandinavian countries, with lateral dispersion increasing the width of the plume. On the 30 April (Figure 3), central Europe was beginning to be affected, reflecting a trajectory which was initially north-westerly, but which subsequently veered westwards in the vicinity of the Baltic sea. (At this point it should be appreciated that Figures 1 to 6 are based upon measured <u>dose rates</u>, so that although by the 30 April the plume of material was largely over central Europe, dose levels were still relatively high in Scandinavia, reflecting earlier deposition of material from the plume). From the 1 to 3 of May (Figures 4 to 6), the plume spread to the west, north and south, essentially covering Europe. This further spreading of the plume was influenced by an anticyclone which was moving eastwards across central Europe. Indeed the contamination of the UK resulted from air being convected northwards behind the area of high pressure; the higher contamination in the north of the UK reflects greater rainfall rates during plume passage.

As far as Europe is concerned, from the 3 May the dose levels generally stabilised and fell. One of the more notable exception to this is Scandinavia, where increased air concentrations were observed around 8 May. This almost certainly represents material discharged during the latter part of the Chernobyl release (around 5-6 May); additionally, this material may have been responsible for the relatively high contamination levels in Lapland.

Before leaving this discussion of the spread of radioactivity across Europe, it is worth noting the similarities between the contamination patterns of Figures 1 to 6 and those generated by long-range trajectory models. Many such studies have been performed and the interested reader is referred to the work of, for example, ApSimon et al (1986).

In addition to the very wide dispersion brought about by the changing meteorology over the several days during which emissions from the damaged plant took place, it seems likely that material was distributed over a considerable range of elevation. Strong directional shear in the wind over the depth of the atmospheric boundary layer, in which much of the transport occurs, would lead to further lateral dispersion of the plume for activity discharged at a given time during the release. Material transported at very high altitudes (> 1km) may have been responsible for the subsequent observations of elevated activity levels in countries bordering the Pacific Ocean.

3 Dosimetric Assessment for Western Europe

An estimate of the dosimetric impact on Western Europe from Chernobyl may be obtained by utilising the monitoring data collected and published by the various national agencies responsible for radiological protection. Such an assessment is presented here.

At the outset it should be appreciated that, within any country of Western Europe, there is some variability in the measured concentrations of radioactivity, arising from the complicated patterns of atmospheric dispersion and rainfall during passage of the plume. Clearly any dosimetric assessment needs to take account of this distribution in relation to the distribution of population. Presented here are estimates of the mean (population-weighted) individual dose for various countries, based on estimated mean environmental concentrations. The (mean) dose estimates are, therefore, subject to a degree of uncertainty, up to around a factor of a few depending on the country, arising solely from this averaging process. Other sources of uncertainty are discussed below.

Dosimetric pathways contributing to radiation exposure include the inhalation of activity during passage of the plume, ingestion of contaminated foodstuffs and external irradiation from deposited activity. In addition to these, β-dose to the skin and external exposure to radiation from the passing cloud of activity can also contribute to total dose levels; however, these mechanisms are transient in nature and have been shown to make up only a very small fraction of the total effective dose to a representative individual (Fry et al, 1986). Each of the pathways considered in the present analysis is discussed briefly below.

3.1 Inhalation pathway

In general, significant elevated concentrations of activity in air were present for a few days and direct measurements of ambient levels of the radiologically important nuclides (isotopes of Cs, I and Ru) were monitored. This data allows time integrated air concentrations to be estimated. Used in conjunction with standard values of inhalation rate and dose per unit intake for adults (NRPB, 1986), a mean individual committed effective dose equivalent can be calculated for the inhalation pathway.

3.2 Ingestion Pathway

By contrast with inhalation exposure, radioactivity transferred to man through incorporation in foodchains is available over a more extended period, so that monitoring data is unlikely to give a full picture for the average intake of activity via foods. It is therefore necessary to turn to mathematical models representing the temporal pattern of appearance of radionuclides in different foods following an initial deposit. For milk and green vegetables, doses are estimated from representative measured peak concentrations of Iodine and Caesium activity in these foodstuffs, on the basis of those models used by NRPB in deriving reference levels for the introduction of countermeasures affecting food (NRPB, 1986). Consumption rates typical of the UK adult population are assumed (NRPB, 1980) and activity losses in preparation for consumption are neglected. Dosimetric calculations are integrated for a 50 year period following ingestion. In some cases, the absence of monitoring data for foodstuffs has necessitated the estimation of peak concentrations in milk and green vegetables from measured deposition levels. The importance of using direct measurements of activity in food for the dosimetric assessment, wherever possible, is demonstrated most clearly for those countries (particularly Scandinavia and the Low Countries) where restrictions were employed in the first weeks following the accident regarding the feeding of dairy cattle. In some cases these involved removing herds from pasture grazing, while in others a delay was introduced to the date at which cattle would normally have been returned from winter feeding. Activity levels measured in milk in these countries were well below those which would be predicted on the basis of initial ground deposition concentrations. While there is some evidence that other, mainly voluntary, countermeasures were introduced affecting normal food distribution and consumption patterns throughout Europe, these have been ignored in the dosimetric assessment. It might therefore be expected that the results of the calculations yield a fairly conservative estimate of the overall dosimetric impact.

In addition to the contribution to ingestion doses from milk and green vegetables, estimates have also been made of the intake of Caesium in supplies of beef and lamb. Here committed doses from ingestion are derived from reported ground contamination levels, using published data on the transfer of activity through foodstuffs (Linsley et al, 1982). Again, typical UK adult consumption rates are assumed (NRPB, 1980), with appropriate values for the committed dose equivalent per unit intake (Charles et al, 1982). Calculations are extended to include contributions to ingested activity arising in meat over the next 50 years.

Scoping estimates based upon the same suite of models as those used above (Linsley et al, 1982), suggest that, by comparison, the contribution to mean individual committed doses from other foods (root vegetables, cereals etc) will be relatively small (well within bounds of uncertainty). This is due in part to the recorded absence of significant quantities of the radioisotopes of Strontium in environmental monitoring (usually considered to be important for foods contaminated by uptake from soil) and to the delay of a number of months between the accident and the cereal harvest. Ingestion dose calculations are therefore limited to contributions from milk, green vegetables and meat.

3.3 External exposure

External exposure to activity deposited within each country during passage of the plume will continue for several decades. During this time, the predominant contribution to external doses will be due to the decay of radioisotopes of Caesium. For the present assessment, the population-weighted average ground concentrations of these nuclides are used in conjunction with appropriate dose conversion factors taking account of decay and migration into the soil (NRPB, 1986). The dose calculations assume a shielding factor of 0.36 for protection by buildings and involve integration over a period of 50 years following the accident.

3.4 Note on calculations for UK

A small additional degree of sophistication has been introduced into estimates of mean individual dose in the UK. It is well known that a fairly sharp division exists between contamination levels in the northern and north-western parts of the United Kingdom and the remainder of the country, due to different rainfall patterns at the time the plume was passing. Dosimetric calculations are therefore made separately, assuming average contamination levels characteristic of the two regions. A population weighting factor of 18% was applied to doses calculated for the 'north' and 82% to those estimated for the 'south'. The only exception to this method for averaging doses was applied in the case of lamb consumption, since a large proportion of the country's sheep farming is in the more heavily contaminated region. Indeed restrictions on sale and slaughter of lambs have been in force in certain of the worst affected regions. Mean individual dose to members of the UK population from consumption of lamb is therefore determined from the characteristic ground concentrations of Caesium reported for the 'north' only, ignoring any reduction from restrictions (these would not be predicted to be necessary on the basis of the models when average contamination levels are assumed) and taking into account that the UK is only 77% self-sufficient in mutton and lamb.

3.5 Results

Table 1 shows the results of the dosimetric analysis for Western Europe, in terms of the contributions to the total collective dose (integrated to 50 year) from the various pathways considered. It can be seen that the contribution from inhalation is less than 10%, while those from external irradiation and ingestion are roughly similar. The total collective dose to Western Europe is estimated to be approximately 76,000 man Sv.

Table 2 shows the estimated distribution of dose throughout Europe; for each country the mean individual dose commitment and the total collective dose commitment is presented. In some cases, the contributions to total dose from each of the pathways differ somewhat from the general picture of Table 1; comments on such deviations and other considerations are included, where necessary, in Table 2. Finally, some authorities have provided estimates of average individual doses in their own countries, arising from the Chernobyl accident. The methods used differ in their degree of sophistication and in the time periods over which the dose is considered to be accumulated. However, where available, these national estimates have also been included in Table 2 for comparison.

As noted above, the total collective dose for Western Europe is estimated to be approximately 76,000 man Sv. Using conservative assumptions, comprising a linear relationship between dose and risk of cancer extending to the very low levels of individual dose experienced in this case, characterised by a cancer fatality risk coefficient of 1.25×10^{-2} per man Sv, the total number of cancer fatalities in Western Europe arising over the next decades is estimated to be just under 1000.

The contributions from the various dosimetric pathways to the total collective dose estimated for the UK (2800 man Sv, see Table 2) are shown in Table 3. Using the above cancer fatality risk coefficient, the total collective dose implies that, as a result of Chernobyl, there will be 35 or so cancer fatalities in the UK in the coming decades. This figure is set in some perspective below.

4 Dosimetric Assessment for Eastern Europe

Information on levels of radioactive contamination in Eastern Europe is relatively limited and an assessment of the dosimetric impact for these countries is therefore subject to considerable uncertainty. Here, calculation of mean individual dose is performed by scaling from the values previously estimated for countries in Western Europe according to the ratio of activity concentrations in deposited material or foodstuffs, whichever data are available.

Table 4 summarises mean individual doses and total collective dose commitments estimated in this way for Eastern European countries excluding the Soviet Union. The total collective dose is estimated to be approximately 100,000 man Sv which, using a linear dose risk relationship in the manner described above, corresponds to a total of around 1,250 fatal cancers.

5 General comment on estimated doses

It must be appreciated that the dose estimates in Tables 1 to 4 are subject to some uncertainty. Firstly, as already noted above, the use of a weighted contamination level for each country studied may give rise to uncertainty levels up to around a factor of a few. Secondly, the use of monitoring data (with its associated uncertainties), and the application of standard dosimetric models using UK consumption data may give rise to similar degrees of uncertainty. Thus the estimates in Tables 1 to 4 may be uncertain by factors ranging from around 5 to around 10, depending on the country. This should be borne in mind when considering the data in the Tables and when comparing the doses with those estimated by others.

6 Dosimetric Assessment for USSR

SRD is currently performing a dose assessment for the USSR, using an estimate of the source term from Chernobyl and information on air parcel trajectories from the plant during the release. The trajectory data was kindly provided by Dr H ApSimon of Imperial College, London. Atmospheric dispersion and wet and dry deposition along each trajectory are being accounted for and information on the distribution of population and agriculture is being used to evaluate the dosimetric impact.

At the time of writing this paper no results were available. If, however, any results are obtained before the HSSC Symposium they will be presented at the meeting.

7 Comment on Soviet estimate of collective doses to the European part of USSR

The Soviets have estimated the collective dose to the European part of the USSR from various pathways. For external exposure they calculate a collective dose of around $3x10^5$ man Sv. For internal exposure resulting from consumption of foodstuffs contaminated with Cs, they estimate a figure of around $2x10^6$ man Sv. Finally, their quoted estimate of the number of thyroid cancer fatalities appears to suggest a collective effective dose from I in milk of around 10^5 man Sv.

These figures are somewhat out of line with those calculated by us above, which suggest that the contributions from foodchains and external exposure are roughly equal. However, it must be noted that the Soviets have been performing whole-body examinations of exposed people and find agreement between observed and calculated Cs levels in only about 3% of cases. The remaining 97% average about ten times lower than expected. This may result from some of the assumptions involved in their model calculations, relating to the rate of uptake of Cs by plants. Although relatively rapid uptake may be valid for certain regions of European USSR, it is unlikely to be appropriate for all regions. Thus, the contribution to foodchain doses from Cs may be up to a factor of ten lower than that quoted above, yielding a net Soviet estimate (summed over all pathways) of order $6x10^5$ man Sv; using a linear dose-risk relationship, this implies around 7,500 fatal cancers. Note that if this is the case, then the relative contribution made by external exposure and foodchain pathways would be consistent with that evaluated for the rest of Europe above.

8 Estimated Doses in Perspective

It is important to set the results of the above dosimetric assessment in some perspective, to give an appreciation of the low levels of risk involved. This is achieved here by taking the estimated doses for the UK (50 year individual dose of 50μSv and collective dose of 2,800 man Sv) and comparing them, and the risks they represent, with other doses and risks. Before embarking upon this comparison, a general indication of the low dose levels can be obtained by appreciating that external dose rates in Europe from Chernobyl are now so low that, in many cases, they cannot be distinguished from background levels.

a Comparison of UK dose from Chernobyl with background radiation

The average annual dose in the UK from background is around 2 mSv; this may be compared with the estimated 50 year individual dose from Chernobyl of 0.05 mSv. The corresponding collective dose to the UK from background, over the next 50 years, is around 5×10^6 man Sv, which may be compared with the figure of 2.8×10^3 man Sv from Chernboyl. Clearly the dose from Chernobyl is very much less than that from background.

An alternative way of comparing with background is to consider how the background dose rate varies throughout the UK. This variation can be up to around 1 mSv per year. Thus the 50 year individual dose from

Chernobyl of 0.05 mSv corresponds to (say) living in East Anglia and having approximately a three week holiday in Cornwall.

b Comparison of UK dose from Chernobyl with weapons testing

The current average annual individual dose from weapons fallout is around 10 μSv. Exposure from this source has been falling and will continue to fall in future years. On the assumption that the rate of decline in recent years will continue, the total collective dose to the UK from this source over the next 50 years will be around 11,000 man Sv. This may be compared with the collective dose from Chernobyl of 2,800 man Sv. Thus the long term cancer risk from Chernobyl is somewhat less than that from weapons fallout.

c Comparison of UK risks from Chernobyl with smoking

Using the data quoted by Sir Walter Marshall et al (1983), it can be shown that the risk posed by Chernobyl is equivalent to the compulsory smoking of less than 1/100 of a cigarette per week (ie less than half a cigarette per year).

d Comparison of the cancer risk from Chernobyl with cancer statistics

Cancer deaths in England and Wales numbered 134,270 in 1983 and 140,101 in 1984; thus the variation between adjacent years is just under 6000 (similar variations can be observed between other years). This may be compared with the 35 or so cancer fatalities predicted to result in the UK (using cautious risk factors) over the next 50 years from Chernobyl. Clearly, if this cancer risk is eventually expressed in the UK it will not be perceptible in the cancer statistics.

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TABLE 1

Dosimetric Assessment for Western Europe: Contribution by Pathway

Pathway	Collective Dose Man Sv	%
INHALATION	3600	5
INGESTION		
MILK	11000	14
VEG	15200	20
MEAT	12500	17
EXTERNAL	33300	44
TOTAL	75600	

TABLE 2

Dosimetric Assessment for Western Europe: Distribution of dose between countries

Country	Mean Individual Dose (μSv) National Estimate	Mean Individual Dose (μSv) SRD Assessment	Collective Dose (SRD Estimate) (Man Sv)	Comments
Austria		610	4600	
Belgium	120	47	460	Recommendation to keep dairy cows indoors - not always adhered to.
Denmark	< 270	160	820	Cattle being fed on stored fodder; results in relatively low contribution of milk to total dose.
Finland	2100	280	1370	Cows not returned to pasture from winter feeding until 26 May. Milk contribution relatively small.
France	50	46	2500	
W Germany	70	250	15400	National estimate is for first year only. Large local variations eg Bavaria.

TABLE 2 (continued)

Country	Mean Individual Dose (μSv) National Estimate	Mean Individual Dose (μSv) SRD Assessment	Collective Dose (SRD Estimate) (Man Sv)	Comments
Greece		260	2500	Relatively high dose from food in comparison with external irradiation.Based on sparse data.
Italy	90	500	28600	National estimate excludes doses from external irradiation. Large variation across country.
Netherlands	450	355	5100	Cattle taken indoors from 3 May to 8 May. Relatively low foodchain dose.
Norway		770	3200	'Mean' figures are not necessarily weighted according to population; difficult to extract weighted dose from data.
Portugal		0.4	4	Very limited data - much extrapolation.
Spain		1.2	45	Very limited data - much extrapolation.
Sweden		770	6400	Much variability across country. Mean dose is arithmetic mean for range in populated areas.

TABLE 2 (continued)

Country	Mean Individual Dose (µSv)		Collective Dose (SRD Estimate) (Man Sv)	Comments
	National Estimate	SRD Assessment		
Switzerland	330	300	1900	
United Kingdom	70	50	2800	

TABLE 3

Dosimetric Assessment for UK: Contribution by Pathway

Pathway	Collective Dose Man Sv	%
INHALATION	170	6
INGESTION		
MILK	1470	53
VEG	190	7
LAMB	560	20
BEEF	70	3
EXTERNAL	310	11
TOTAL	2770	

TABLE 4
Dosimetric Assessment for Eastern Europe

Country	Mean Individual Dose (μSv)	Collective Dose (man Sv)
Albania	300	830
Bulgaria	700	6250
Czechoslovakia	600	9200
East Germany	500	8370
Hungary	1000	10700
Poland	1200	43700
Romania	600	13500
Yugoslavia	300	6800

NOTE

Figures 1 to 6 indicate the estimated variation of external exposure rate throughout Europe in the days following the accident (see Key for units and notation). The figures show the general distribution, ie they do not include local variations due to, eg heavy rainfall. They also represent an interpretation of available measurements around Europe - some interpolation was necessary.

KEY	INCREASE GIVEN AS MULTIPLES OF BACKGROUND DOSE RATE
	$10^{-2} - 1$
	$1 - 5$
	$5 - 10$
	$10 - 20$
	$20 - 40$
	$40 - 100$
	> 100
	NO DETECTABLE RISE IN DOSE RATE

FIG 1
28 APRIL 1986

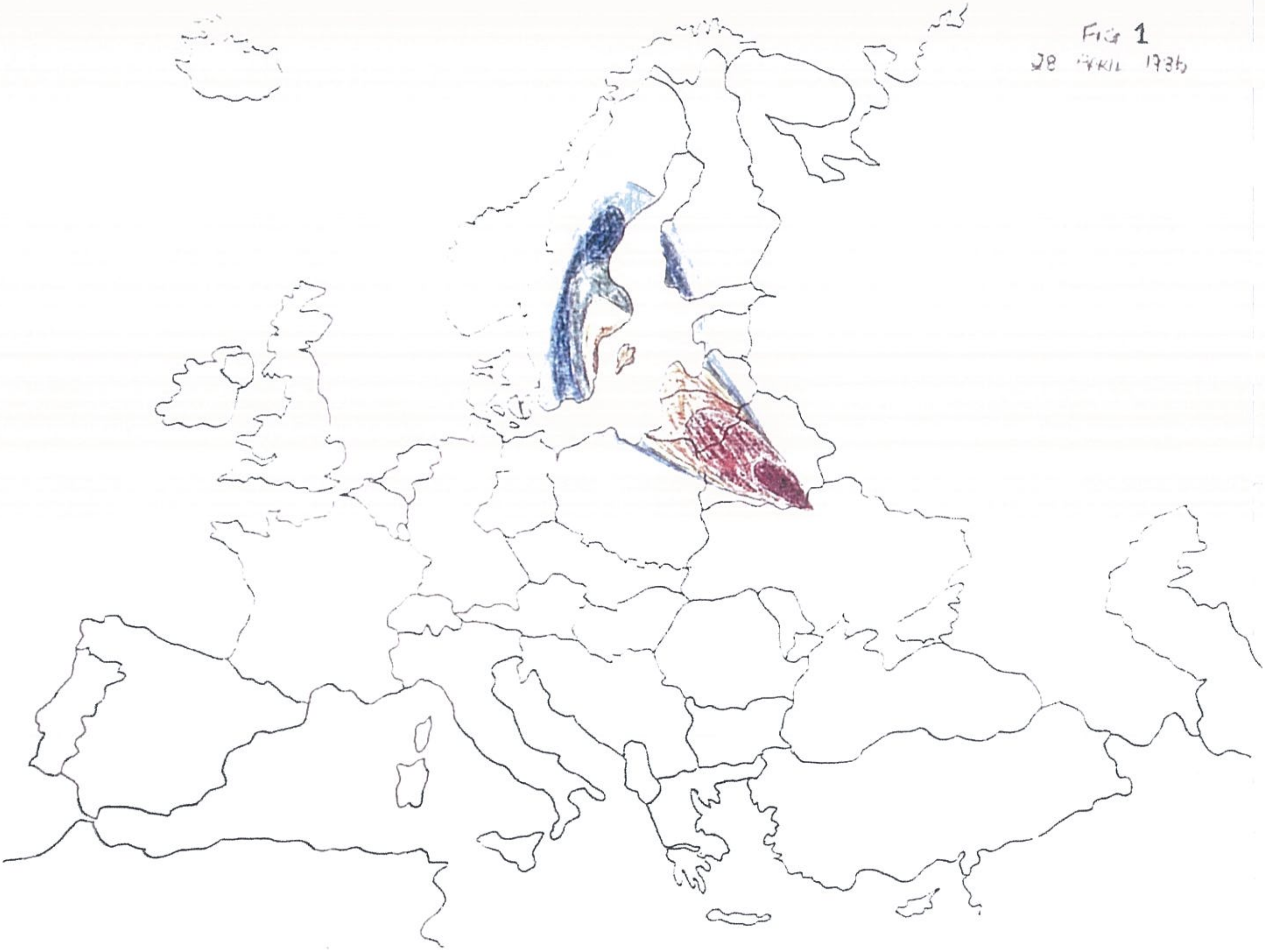

FIG 2
29 APRIL 1986

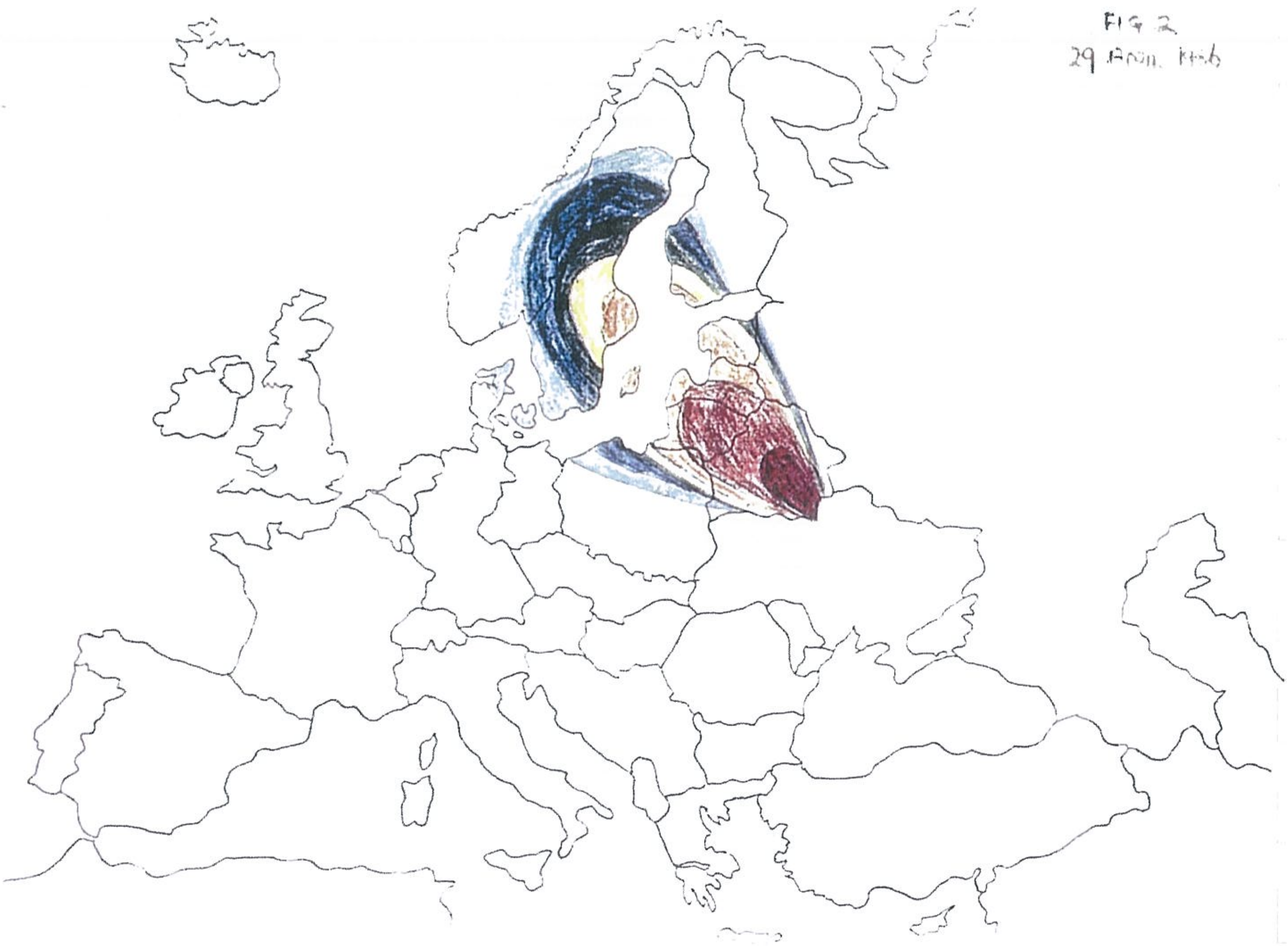

Fig 3
30 APRIL 1986

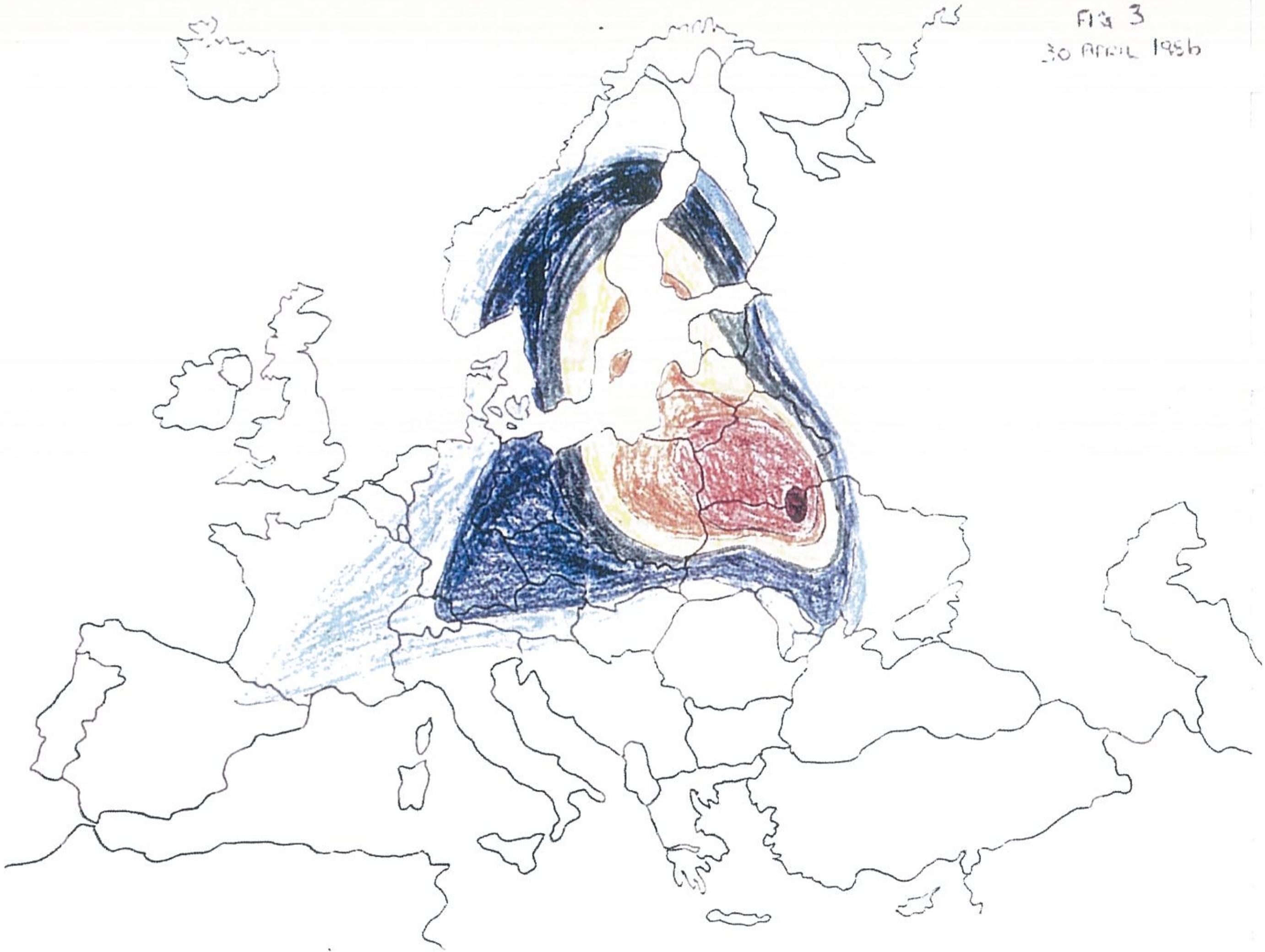

Fig 4
1 MAY 1986

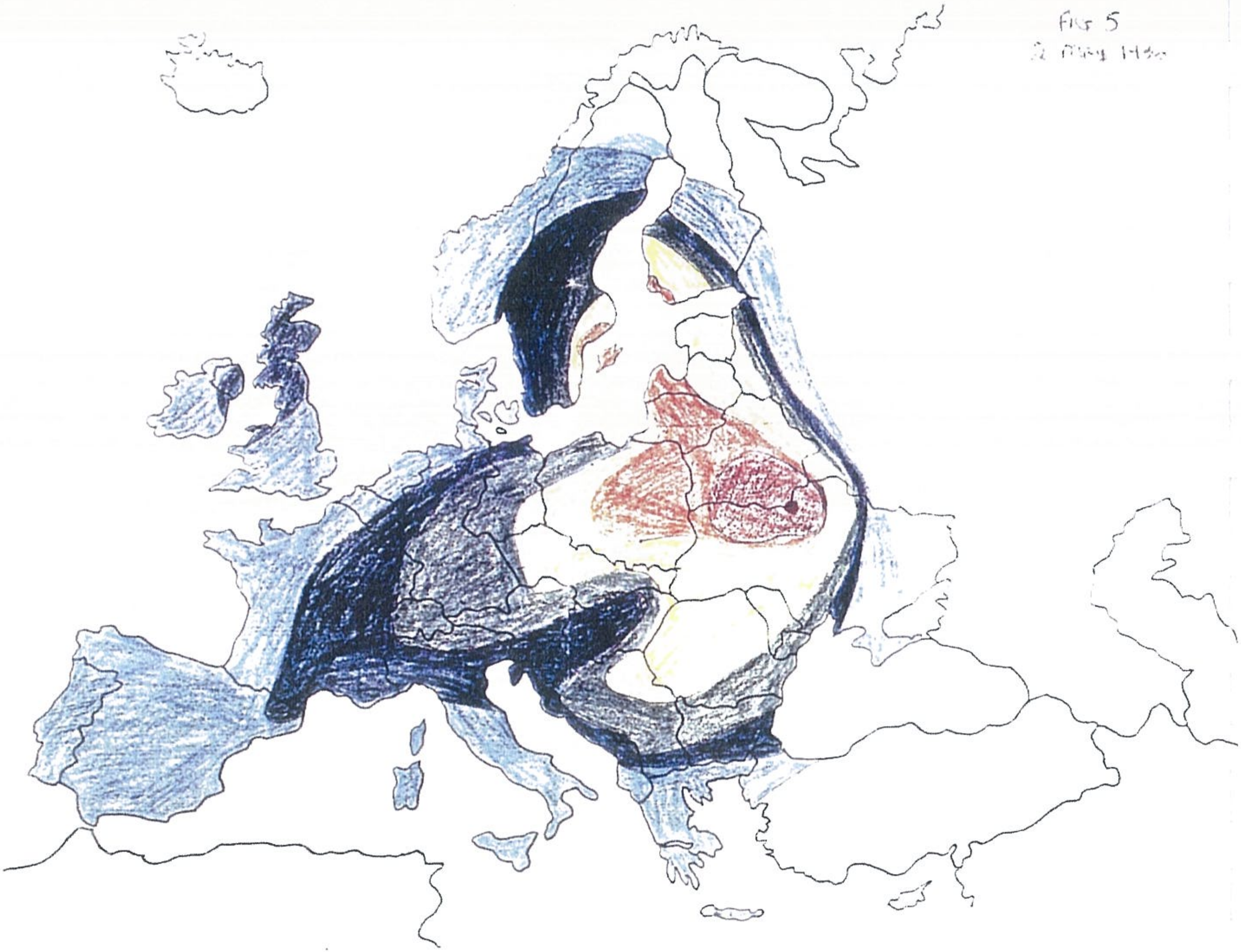

Fig 5
2 May 1986

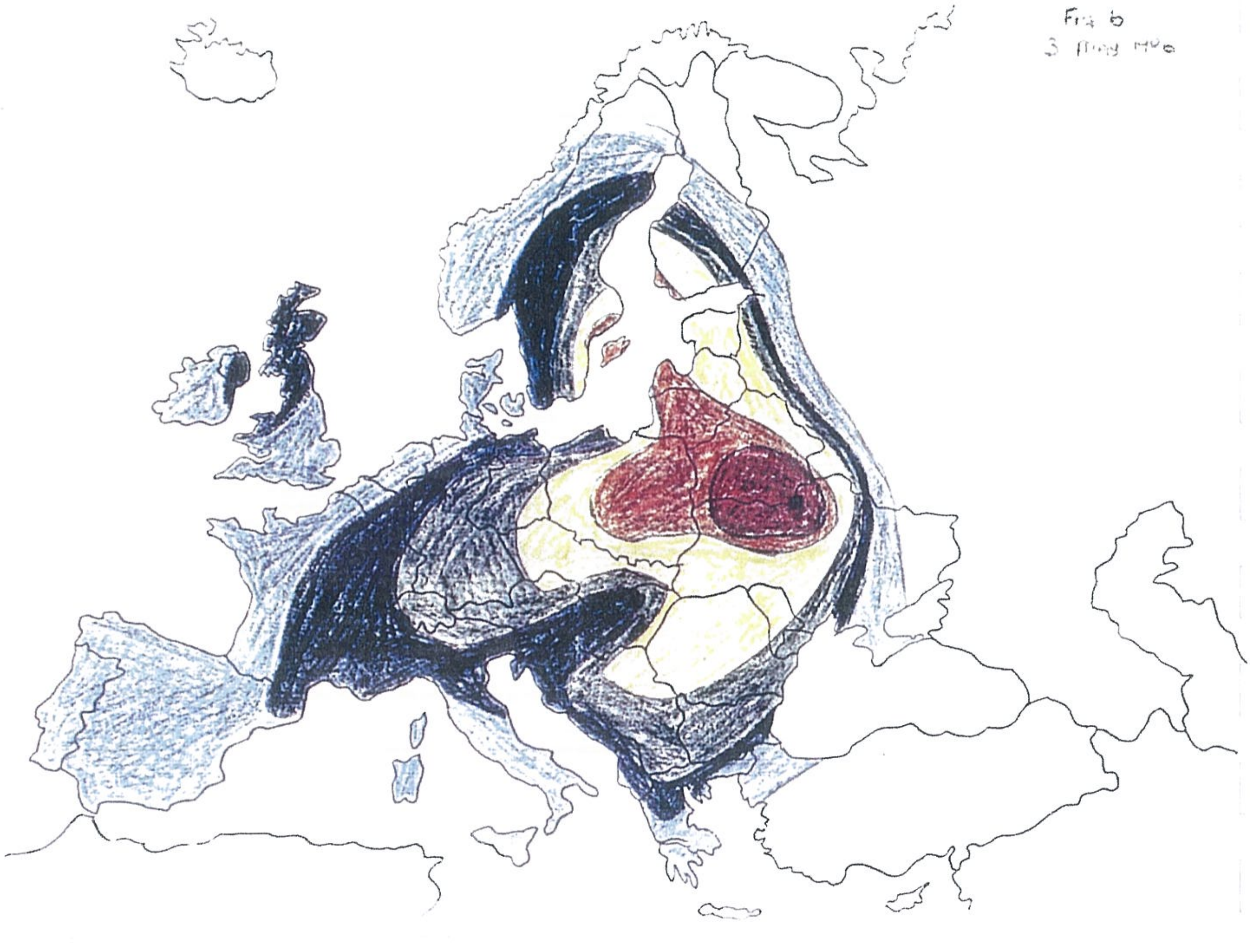

Fig 6
3 May 1986

17/10

HSSC(86)P35

UNITED KINGDOM ATOMIC ENERGY AUTHORITY

HEALTH AND SAFETY STUDIES COMMITTEE

THE CHERNOBYL ACCIDENT - SOURCE TERMS AND RELATED CHARACTERISTICS

P N CLOUGH

SRD, Culcheth

October 1986

Distribution:

Standard HSSC
Symposium Participants - Chernobyl Accident

SYNOPSIS. The Chernobyl reactor accident gave rise to a large release of radioactivity to the environment. The detailed characteristics of this release in terms of radionuclide composition, timing, and energy of release, that is the source term, are discussed in this paper. It is important to be able to establish this source term as precisely as possible in order to relate it to the consequences of the accident. The evidence presented by the Russians at the Vienna meeting, 25-29 August, is reviewed, and discussed in relation to evidence from Western European sources. Important phenomena in the progression of the accident which influenced the source term are examined.

INTRODUCTION

1. When, following the first indication of abnormally high levels of airborne radioactivity in Scandinavia on the 28 April 1986, it was confirmed that a major nuclear reactor accident had occurred at Chernobyl in the USSR, a key question was 'How bad is it?'. It could be quickly established that core of an RBMK reactor of the type at Chernobyl, Unit 4, would contain several thousand million curies of radioactivity ('core inventory'). The severity of the accident consequences was clearly closely related to the fraction of this escaping, to the environment. The information which we now have on the detailed activity released and associated characteristics important for determining the consequences of the Chernobyl accident, that is, the source term, is the subject of this paper. The definition of the source term involves three main ingredients:-

(a) the quantities of specific radionuclides released to the environment. Some 54 individual radionuclides are usually considered to be important for consequence assessment, of which the most prominent are isotopes of iodine, caesium and tellurium. The released activities of these are conventionally expressed as fractions of their inventories in the core at reactor shutdown ('release fractions').

(b) The timing of release, which includes both the start and duration of release relative to shutdown. A warning time for countermeasures may also be required for consequence modelling.

(c) the energy and height of release, which related to the rise and dispersion of the radioactive plume transported from the reactor site.

2. This definition of the source terms was largely established in the formative study of the risks due to severe accidents in Light Water Reactors (LWR) in the USA, the Reactor Safety Study (1). In this study it was found possible to represent the wide range of severity of potential activity releases in terms of a number of so-called release categories, each with a characteristic source term. Two such source terms for the Surry 1 Pressurised Water Reactor (PWR) are quoted in Table 1. The important radionuclides are grouped into seven groups, the members of each being similar in chemical and physical characteristics, and thus exhibiting similar release behaviour. Each group is represented by a leading element. Thus, Cs covers all relevant isotopes of caesium and rubidium, whilst La covers isotopes of some 8 lanthanide and 4 actinide elements, including plutonium. For each group, the corresponding release fractions in the source terms are shown. Clearly, the most severe source term PWR1 has the highest release fractions, which for the more volatile nuclides (Xe, I, Cs, Te groups) correspond to large proportions of the core inventory. Such a source term will, unless the reaction is sited very remotely, result in very serious consequences. For a PWR such as Surry 1, the containment building must fail almost immediately, or be by-passed in some way, for such a source term to arise. If the containment holds out even for a few hours, a significantly less severe source term results. Indeed, if the containment does not rupture above ground at all, but only suffers basemat failure, a much reduced source terms is expected, represented by PWR6. Such a source term will generally result in much less serious environmental consequences. For comparison, the best estimate which the Russians have been able to provide of the release fractions in the Chernobyl accident up to the 6 May when activity release was finally terminated are also quoted in Table 1. This shows that the Chernobyl accident lies towards the upper end of the spectrum of source terms severity. However, it must be pointed out that the protracted release at Chernobyl (11 days) differs markedly from the much shorter PWR release periods.

THE RUSSIAN ACCOUNT OF THE SOURCE TERM

3. The Russians have provided a relatively detailed account of their own reconstruction of the source term in the official documents provided to the IAEA Experts Meeting on the Chernobyl accident, Vienna, 25-29 August 1986. The main features of this will be considered first, and then compared with our own attempts to reconstruct the source term on the

basis of much more remotely collected and scant information. The Russian account contains two main components. Firstly, there is a broad history of the total activity release from the plant which is related to the variations in the known or assessed state of the core throughout the release period. Secondly, a detailed breakdown of activity releases by specific isotopes at certain stages of the accident is provided, supplemented by various isotope-specific sampling measurements at different locations and times. The latter information is chiefly of interest as exemplifying the type of data the Russians have employed in their source term reconstruction.

4. According to the Russian account, the activity release can be roughly divided into four stages. The day-by-day releases of activity which they estimate are shown in Fig 1. It is important to note that these values are all adjusted for radioactivity decay to day 10 of the accident (6 May). Thus, the actual release on day 0 (26 April) is quoted as 20-22MCi by the Russians, but by day 10 only 12 MCi of this remained in the environment following decay. The stages of release were:-

5. Stage 1 (day 0). This was the initiating in-core explosion at 01:24 on the 26 April. Significant quantities of fuel were ejected in the transient which blew off the pile cap, and these were accompanied by enhanced releases of the volatile fission products iodine, caesium and tellurium. The estimated isotope-specific activity releases on the 26 April are shown in the first column of Table 2 (2), but it is not clear what the contribution of the initial explosion was compared with the fire which followed.

6. Stage 2. (Days 0-6). Following the initial explosion, the core top was left fully exposed to the atmosphere, and it appears that on intense graphite fire rapidly developed. High levels of activity release were associated with this during the first day, attributed by the Russians at least in part to the re-entrainment of fuel material embedded in the graphite during the initial excursion. During the 27 April, dumping of materials onto the core debris from the air began to extinguish the fire, and by the 10 May some 5000 te had been dropped. In addition to boron compounds and lead, the materials included dolomite, clay and sand specifically intended to trap and filter out the active species from the core. This approach was reasonably effective, because during the 27 and 28 April the activity release rate was substantially reduced, and remained at a reduced level until the 2 May. However, release during this period was still very significant, and it is unclear whether the in-core fire was fully extinguished.

7. Stage 3 (Days 7-9). A marked increase in activity release occurred on the 3 May, which continued through the 4 and 5 May. This is attributed to the decay heat pushing the temperature of the core debris (fuel and moderator) to elevated levels, ultimately in excess of 2000°C, following, the effective insulation of the pile up by dumped material.

During the early part of this stage, strong enhancement of the volatile fission product contribution to activity release, notably of iodine, was observed. This is clear from the data reproduced in Table 3 (3), which shows the isotope-specific activity distribution in air samples from above the reactor at various dates. However, by the 5 May when the temperature peaked, the released activity assumed a composition close to that of the fuel itself. There is speculation in the Russian account that reduction of the UO_2 by the graphite moderator at these elevated temperatures may have contributed to the fuel-like release. Continued graphite combustion is also referred to, which is incompatible with fuel carbidisation.

8. Stage 4 (Day 10). On the 6 May, the release rate rapidly fell to effectively zero. This is attributed to a combination of factors which somehow brought about a rapid cooling of the fuel debris. A key feature seems to have been the injection of nitrogen at a high rate underneath the core, which simultaneously cooled the debris and stifled any residual graphite fires. Additionally, there is reference to special measures taken to promote the formation of more refractory fission product compounds by the introduction of further materials into the filter bed, but this is not enlarged upon. The precise reason why activity release fell so rapidly on the 6 May remains something of a mystery.

9. The total activities contributed by specific isotopes released from Chernobyl and still present in the environment on the 6 May, according to Russian estimates, are shown in the second column of Table 2. These values allow for decay between reactor shutdown and the 6 May. Thus, although 4 MCi of Te 132 was released on the 26 April, the residue of this plus all subsequently released Te 132 only amounted to 1.3 MCi in the environment on the 6 May. This exemplifies a problem in representing a protracted activity release such as that at Chernobyl by means of release fractions as is conventional in source term methodology. The release fraction approach was developed in the context of releases of short duration occurring at most a few hours after reactor shutdown. One approach for Chernobyl is to quote cumulative release fractions at the end of release on the 6 May. These would be given for each isotope by dividing the environmental burden at that date by the calculated inventory in the intact core at that date, allowing for decay in the core since shutdown. Such values calculated by the Russians are quoted in the third column of Table 2, and utilised in Table 1. However, these fractions are of limited value for consequence assessment, especially as the isotopic composition of the release varied substantially with time. What is needed, as a minimum, is the day-by-day actual activity releases due to each radiologically important isotope. In this context, the concept of release fractions loses much of its usefulness.

10. A further problem in establishing release fractions relates to defining the detailed core inventory of the

Chernobyl reactor at shutdown. This is also important for assessing the isotopic composition of the many environmental samples which have been measured, and relating these to the mechanisms of release from the core. The Russian documentation provided at Vienna contained no direct core inventory information. However, the final column of Table 3 quotes Russian results for the relative activities contributed by a set of important radionuclides at shutdown. To obtain absolute data, and to provide a cross-check on various aspects of the Russian analysis, the UKAEA core inventory code FISPIN has been run to estimate the Chernobyl inventory. Full-power operation up to 12 hours before the accident was assumed, followed by the power history specified during the experimental period which led up to the accident. The quoted average fuel burn-up of 10,3000 Md/te was employed, with no allowance for burn-up variations across the core. A Magnox-type neutron spectrum was assumed in the absence of details of the Chernobyl neutronics. In view of these simplifications, the results of this FISPIN calculation must be taken as indicative rather than definitive of the Chernobyl inventory. The FISPIN relative activities for the set of nuclides in Table 3 are compared with the Russian values in Table 4. The agreement is not very good even for the major fission products such as I131 and Cs137. Large discrepancies exist for Ru106 and Cs134, although the latter which is formed by neutron capture is rather sensitive to the assumed neutron spectrum.

11. As an indication of whether the FISPIN or Russian data on inventories are more reliable, Table 5 compares the predicted ratios of isotopes of the same element with the measured ratios in airborne samples taken from Table 3. Clearly, the release fractions of isotopes of the same element must be identical, so that the measured ratios reflect the ratios in the core. The FISPIN Cs134/Cs137 ratio is about 40% higher than the average sample value, but the Russian result is a factor of 3.3 too large. For the ruthenium and cerium isotopes, the FISPIN results tend to under-predict the ratios, and the Russian values to over-predict them, although in neither case are the discrepancies really serious in view of the measurement uncertainties. On the whole the FISPIN results give a reasonably good account of the data on isotopic ratios, and are used as the basis for discussion of the sampling measurements below. However, a further anomaly in the Russian reporting of the activity release relating to inventory and decay should be pointed out. The total activity release for the 26 April in the first column of Table 2 amounts to just over 20 MCi. Correction for decay to the 6 May by the Russians results in the 12 MCi value which appears Fig 1 for day 0. An independent decay calculation for the set of nuclides quoted in Table 2 gives a decay factor of 3. Thus, the Russian estimate for release on the 26 April of 20 MCi compares with 36 MCi back-calculated from Fig 1.

RADIONUCLIDE SAMPLING DATA FROM WESTERN EUROPE AND THE USSR

12. Radioactivity sampling data from Scandinavia was supplied to SRD soon after the enhanced levels in Finland and Sweden were detected following the Chernobyl accident and an exercise to establish the source term began. Subsequently, widespread sampling data from around Western Europe came in. The absolute levels of activity measured were important for estimating the magnitude of the source term. Based on the Scandinavian data, the atmospheric dispersion and consequences modelling code CRACUK was applied to back-calculate the likely size of the release once the site of the accident had been confirmed. Release fractions of about 5% for the volatile fission products iodine and caesium were deduced for the release towards Scandinavia. These values compare favourably with the 5-10% release fractions for these fission products quoted by the Russians for the 26 April, when the plume was directed towards Scandinavia.

13. Some of the Western European data contained radionuclide specific activity measurements, and these were particularly important for trying to reconstruct the state of fuel and core damage, and thus to establish details lacking at that time on the nature of the Chernobyl accident. The volatile fission products (noble gases, iodine, caesium, tellurium) are released from UO_2 fuel at lower temperatures, typically 1500-2000°C, than the more refractory fission products such as barium, strontium, ruthenium, lanthanum, cerium, and the actinides. The latter require temperatures well in excess of 2000°C to undergo significant release. Moreover, certain fission products are sensitive to the oxidizing conditions around the fuel. For example, tellurium becomes strongly bound to the zirconium in the fuel cladding when first released from the UO_2, but if the cladding itself becomes oxidized, the tellurium is freed. Similarly, ruthenium as an element is highly refractory, but in oxidizing conditions, much more volatile oxides can be formed. Enhanced releases of tellurium and ruthenium would indicate oxidizing conditions in the core, such as would be associated with an in-core fire.

14. Now that the Russian account of the accident is available, it is interesting to re-examine the Western sampling data and to relate it to the phenomenology of the accident as described by them. Radionuclide specific sampling data collected in Finland, Sweden, Denmark, the Netherlands and Hungary over the two weeks following the accident gave a remarkably consistent picture of the radionuclide spectrum in the release. The averaged results of air sampling measurements are shown in the first column of Table 6. The values quoted are the release fractions for individual isotopes relative to the release fraction for Cs137. The FISPIN-calculated core inventories were used to generate these fractions. Thus,

Relative Release Fraction for X =

$$\frac{\text{Activity of X in sample}}{\text{Activity of X in core}} \Big/ \frac{\text{Activity of Cs137 in sample}}{\text{Activity in Cs137 in core}}$$

All activities are corrected for decay back to the time of reactor shutdown. It is evident that the release fractions of the volatile fission products inferred from these samples are one to two orders of magnitude higher than those of the more refractory species.

15. It is now apparent that the activity transported to Scandinavia during the two days following the start of the accident must have been associated with the initial peak of release on the 26 April (Stage 1 plus the first day of Stage 2). A change of wind from north westery to south westery then transported the residue of the same material towards the Low Countries. This accounts for the similarity in radionuclide composition observed in these regions, and indeed later in the UK. One may compare this composition with that in air samples above the reactor on the 26 April. (Table 3). The relative release fractions for a selection of isotopes are shown in the second column of Table 6, normalised to Cs134 since Cs137 was not measured. The ratio of volatile to more refractory nuclides is much reduced relative to the remote samples, although there is still significant enhancement of the volatile content compared with the whole fuel composition. The activity quoted by the Russians for Te132 includes a contribution by I132, which partially accounts for the anomalously high relative release fraction assigned to Te132.

16. The differences in remote and local sample compositions may have arisen during the initial stages of the accident as follows. High fuel temperatures were undoubtedly reached in the initial reactivity transient. Russian estimates of the temperature history of the fuel throughout the accident are shown in Fig 2. The initial transient, which generated sufficient pressure to rupture most of the pressure tubes and severely disrupt the pile up, drove fuel temperatures in the zones of peak reactivity to levels where significant fuel fragmentation occurred. At these temperatures, the volatile fission products were almost entirely released from the fragmented fuel. Moreover, other zones of the fuel which did not get sufficiently hot to fragment nonetheless became hot enough to release large fractions of their volatile fission product contents. In the initial explosive burst, a small amount of fuel was ejected from the core ($\ll 0.5\%$ interpolating from Table 2), a proportion of which was in the aerosol size range. This was accompanied by a larger fraction of the vapours of the volatile fission products (a few percent). In the rapid cooling process on exiting the core, the vapours of the volatile species condensed. A proportion condensed onto the fragmented fuel, most of which was of a particle size such that it settled out fairly rapidly from the atmosphere. However, condensation onto fine fuel particles, possibly

accompanied by self-nucleating condensation, produced a fraction of very fine aerosol heavily enriched in the volatile fission products. It was this material which was carried over 1,000 km to Scandinavia and contributed heavily to the samples in Western Europe.

17. In terms of this interpretation, peak fuel temperatures in the hottest zones of the core must have reached or exceed 3000K in the initial transient. In several respects it more closely resembles the initiation of the Hypothetical Core Disruptive Accident in an LMFBR than any type of accident anticipated for Western designs of PWR or BWR. The Russian estimate of 1800K for the fuel temperature at the outset is certainly too low, although this may refer to the average over the core, rather than the hottest region.

18. The Russians have provided no information on the relative contributions of the initial burst and the subsequent in-core fire to the total activity release on the 26 April. The nature of the fire or fires in the core is still unclear, and is discussed in more detail below. From Fig 2, the fuel temperature appears to have fallen quite rapidly to about 1000K over this day, before any effective countermeasures were begun. It is likely that the air sample quoted in Table 3 was taken some hours after the initial explosion. A possible mechanism for continuing activity release in an oxidizing fire regime at relatively low temperatures is the oxidation of UO_2 to U_3O_8. Large amounts of fragmented, unclad, UO_2 pellets must have been present in the core debris after the initiating excursion. Oxidation of UO_2 to U_3O_8 in air promotes activity release by both chemical and physical mechanisms. The chemical mechanism accelerates the release of certain fission products, notably iodine, tellurium and ruthenium from the fuel matrix and is effective over a wide temperature range about 800K (4). The physical mechanism is due to the production of aerosol-size U_3O_8 particulate which is readily spalled away (5). This is effective over the temperature range 800-1000K but at higher temperatures larger particular U_3O_8 is formed which is less mobile.

19. A strongly oxidizing regime in the core at a temperature of around 1000K with ventilation generated by the in-core fires could thus have led to a steady release of U_3O_8 aerosol accompanied by enhanced releases of the volatile fission products. A composition corresponding to whole fuel enriched with volatile fission products is evident both in the air samples from above the reactor (Table 3 and 6), and in ground samples close to the site (Table 7 (6)). The vapours of the volatile species would deposit onto the aerosols rapidly upon cooling after leaving the core. The Russians themselves have attributed this release to re-entrainment of fuel impacted into the graphite as the latter burned, but such an explanation appears unnecessary. During the 26 and early part of the 27 April, the release was unconfined. Thereafter, it became increasingly attenuated as filtering materials were dumped on the core debris. However, these filtering materials

were far from 100% effective, since substantial releases persisted throughout Stage 2 of the accident. The filter bed itself probably became very hot, and was thus ineffective for trapping vapours of the volatile species. Moreover, under some conditions the proportion of U_3O_8 particles formed by UO_2 oxidation in the 1μm size range can be significant. Such small particles would be poorly trapped in a coarse filter.

20. The chief evidence for strongly oxidizing conditions in the core comes from the high release fraction of tellurium, comparable with iodine and caesium. As previously noted, this indicates extensive oxidation of the zirconium alloy fuel cladding. High temperature oxidizing conditions would also lead to the decomposition of caesium iodide, which is thought to be the predominant chemical form of fission product iodine in oxide fuel. There is no evidence in any of the Russian sampling data for a separation of caesium and iodine in the release, the release fraction for iodine relative to caesium being close to unity. However, much of the remote air sampling data showed an iodine relative release fraction well below unity (Table 5). This sampling was based mainly on particulate filters, but in one instance where gaseous iodine was also trapped on charcoal (Nurmijarvi, Finland), some 85% of the total was detected in this form. It thus appears that by the time the released iodine reached Western Europe, both particular and gaseous forms were present, and that the low apparent relative release fraction in air samples is an artifact of the sampling method. Infact, ground samples in Western Europe yield on iodine relative release fraction close to unity. It is not clear whether both forms of iodine were released from the site, but the chemistry regime in the core makes this probable. However, there is no evidence for enhanced ruthenium release relative to the other non-volatile species in the Russian sampling data, and only a slight indication of this effect in the remote sampling data. The oxidative mechanism is not, therefore, fully consistent with the observations.

21. Although the radionuclide composition through Stage 3 of the release remained fairly similar to that through Stage 2 on average, there is evidence that on the 3 May a sharp temporary increase in the relative contributions of the volatile fission products iodine, caesium and tellurium occurred. The Russians attribute the overall increase in activity release during Stage 3 to rising core temperatures (Fig 2), as the decay heat finally made an impact on the massive thermal sink represented by the graphite moderator. The U_3O_8 spalling mechanism discussed above should have become less effective for fuel release as the temperature rose. Increased release rates for the volatile fission products are fully expected if the core heated up to temperatures in excess of 1500°C, particularly since regions of previously unheated fuel probably became involved. The burst in the volatile species on the 3 May may well have been due to re-vaporization of material previously condensed in the filter bed as this

also heated up. However, the high level of release of material with composition akin to fuel is difficult to explain. There has been speculation that regions of the debris became hot enough for fuel liquefaction to have occurred. Direct vaporization of fuel and refractory fission products would then be a possible origin of the enhanced releases. However, such a mechanism would be expected to sharply discriminate between species such as barium, for which reducing conditions promote release, and ruthenium for which oxidizing conditions are more effective. There is no evidence for such discrimination in the sampling data. (Table 3).

22. The termination of the release (Stage 4) remains something of a mystery. The most significant feature appears to have been the injection of nitrogen to displace the air in the core. It is possible that graphite combustion was still contributing significantly to pushing up core temperatures at this stage, and that the extinction of this was an important contributory effect. However, the decay heat state represented a major energy source, and an additional effect related to the termination of oxidative mechanisms which were promoting release during Stage 3 seems necessary to account for the abruptness of the ending of release.

23. Much of the foregoing discussions has been based on the available air sampling data, since this provides the greatest detail with respect to the history of the release. The Russians have also provided a good deal of information on ground samples from close to the site. Indeed, the Russian estimates for the absolute magnitude of the release fractions appear to be based largely on extensive ground sampling measurements. Such samples have provided the main source of information on actinide releases in the accident, although the Russian data relates to measurements after the termination of the release, so that a cumulative picture of the radionuclide deposition at the sampling location is given. There is a large variability in the radionuclide composition found in different soil samples, but a representative set of relative release fractions found for samples within a 30km zone north of the reactor is given in the first column of Table 7. This combines data quoted in several tabulations by the Russians (6). On this evidence, the relative release of plutonium was substantially lower than that of other refractory oxide constituents in the fuel, such as cerium and zirconium. This result is inconsistent with the much higher relative release of Np239 indicated by the air sampling. It is also at odds with the limited independent evidence on the nature of the deposited activity from within the USSR. The second column of Table 6 shows the composition of the activity on the shoe of a student who returned from Kiev (80 km from Chernobyl) to the UK on the 2 May, measured by the NRPB. Here, excluding the actinides, the composition is very close to that of whole fuel. The actinides appear to be present in something of an excess over their expected core inventories, but in view of

the uncertainties in calculating the latter, undue significance should not be attached to the actual numbers in the table. The important point is that at least equal release fractions are indicated for the actinides as for the other non-volatile radionuclides. The quoted Russian soil sampling data may be unrepresentative in this respect. The sample variability has already been noted, and a marked variability has also been observed in the composition of individual radioactive particles which have been analysed in the USSR and elsewhere. For example, particles which were composed almost entirely of ruthenium were detected in Sweden (7), whilst particles of almost pure caesium or cerium were found close to the site. There is no ready explanation for the occurrence of such particles of almost unique composition, although formation via selective vapour condensation processes is a possible origin.

ENERGETICS OF THE RELEASE

24. The initial explosive release of activity from the core appears to have been a very energetic event. It is outside the cope of this paper to attempt any assessment of the energy release associated with Stage 1. A high rate of energy release must have continued at least into the 27 April, since the Russians have quoted a plume height in excess of 1200 m on that day at 5-10 km from the plant. Brief consideration will be given as to whether chemical energy sources from combustion could have contributed significantly to the energetics of the release processes. The yardstick of comparison is the decay heat in the core debris, which remain very substantial throughout the whole of the release period. For example, on the 27 April the decay heat was around 15 MW, and even at the termination of release on the 6 May it was still at a level of around 7 MW. If efficient mechanisms existed for converting much of this energy into sensible heat of the plume, it would be unnecessary to seek other sources to account for the higher plume rise.

25. Nonetheless, two chemical reactions which may have supplemented the decay heat substantially are the combustion of zirconium, present in large amounts in fuel cladding and the reactor pressure tubes, and of the graphite moderator:-

$$Zr\,(c) + O_2(g) \rightarrow ZrO_2(c) \quad \Delta H^\circ(1000K) = -\ 1091kJ$$

$$C\,(c) + O_2(g) \rightarrow CO_2(g) \quad \Delta H^\circ(1000K) = -\ 395.5kJ$$

The oxidation of zirconium alloy fuel cladding in air undergoes breakaway to a regime of parabolic kinetics at temperatures above 800°C, whilst graphite combustion breaks away at even lower temperatures of 500 - 600°C. However, zirconium oxidation is favoured both thermodynamically and kinetically over graphite oxidation at the temperatures of 1000°C or higher associated with the early stages of the accident. The zirconium was also concentrated in the fuel channels, where ventilation of the disrupted core was presumably best. Some of it was oxidized by steam in the initial transient, generating the hydrogen which was responsible for early ex-core explosions and fires. However, oxidation of the residual metal by air probably played an important part in sustaining high fuel temperatures at the outset. A simple adiabatic calculation suggests that the complete oxidation by air of the zirconium content of a fuel channel could have heated the UO_2 fuel to a temperature of around 4000°C. Thus, even allowing for appreciable heat loss, a strong chemical heating mechanism was available. Moreover, combustion of the zirconium in the core in air at the rate of 10% per day could have provided about 1 MW of sensible heat in the exit gases. It is more likely, though, that the zirconium was consumed very rapidly once air entered the core, contributing strongly to the initial energy spike.

26. Once the zirconium was locally exhausted in regions of the core, graphite combustion provided a further chemical energy source. This reaction releases less than 40% of the energy available for zirconium oxidation per mole of oxygen consumed, and the adiabatic temperature limit for graphite combustion in air is close to 2000°C. Burning of the graphite moderator at a rate of 1% per day would have generated a sensible heat release of 7.6 MW, which is comparable with the decay heat. There is no information available from the Russians as to their estimates of the total loss of graphite in the in-core fires.

CONCLUSIONS

27. The major characteristics of the source term in the Chernobyl accident appear to have been established by the Russians. Some 10-20% of the core inventory of the volatile fission products iodine, caesium and tellurium had been released from the core over the 11 day period, 26 April to 6 May, together with 3-4% of the refractory fuel constituents, although ± 50% uncertainty bounds are quoted for these values. The Russian account of the source term is consistent with independent evidence from Western sampling measurements in most respects, although some anomalies remain. The Chernobyl source term differed in important respects, especially the length of the release period, from the types of severe accident source terms which have been considered for reactors of Western design. This feature, and the high initial energy release, were important for consequence mitigation.

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Table 1. Source Term Estimates for the Surry 1 PWR compared with Chernobyl

	Start	Duration	Fraction of Core Inventory Released						
Surry 1			Xe	I	Cs	Te	Ba	Ru	La
PWR 1	2.5hr	0.5hr	0.9	0.7	0.4	0.4	0.05	0.4	3(-3)*
PWR 6	12hr	10hr	0.3	8(-4)	8(-4)	1(-3)	9(-5)	7(-5)	1(-5)
Chernobyl**	0hr	11d	0.8	0.2	0.1	0.15	0.04	0.05	0.02

* 3(-3) ≡ 3×10^{-3}

** quoted uncertainty in Russian sources ± 50%

Table 2. Soviet assessment of the radioisotopic composition of the release from Chernobyl Unit 4 (ref 2)

Isotope	Activity of release, MCi		Fraction of activity released from the core by 06.05.86,* %
	26.04.86	06.05.86	
Xe133	5	45	Possibly up to 100
Kr85m	0.15	-	"
Kr85	-	0.9	"
I131	4.5	7.3	20
Te132	4	1.3	15
Cs134	0.15	0.5	10
Cs137	0.3	1.0	13
Mo99	0.45	3.0	2.3
Zr95	0.45	3.8	3.2
Ru103	0.6	3.2	2.9
Ru106	0.2	1.6	2.9
Ba140	0.5	4.3	5.6
Ce141	0.4	2.8	2.3
Ce144	0.45	2.4	2.8
Sr89	0.25	2.2	4.0
Sr90	0.015	0.22	4.0
Pu238	$0.1.10^{-3}$	$0.8.10^{-3}$	3.0
Pu239	$0.1.10^{-3}$	$0.7.10^{-3}$	3.0
Pu240	$0.2.10^{-3}$	1.10^{-3}	3.0
Pu241	0.02	0.14	3.0
Pu242	$0.3.10^{-6}$	2.10^{-6}	3.0
Cm242	$0.3.10^{-2}$	$2.1.10^{-2}$	3.0
Np239	2.7	1.2	3.2

* Evaluation error ±50%.

Table 3. Relative activity of radionuclides in the air above the Chernobyl nuclear power plant (ref 3)

Nuclides	April 26	April 29	May 2	May 3	May 4	May 5	Relative Shutdown Inventory
^{95}Zr	4,4	6,3	9,3	0,6	7,0	20	3,6
^{95}NB	0,6	0,8	9,0	1,3	8,2	18	3,8
^{99}Mo	3,7	2,6	2,0	4,4	2,8	3,7	3,9
^{103}Ru	2,1	3,0	4,1	7,2	6,9	14	3,9
^{106}Ru	0,8	1,2	1,1	3,1	1,3	9,6	2,1
^{131}I	5,6	6,4	5,7	25	8,2	19	2,3
$^{132}Te + ^{132}I$	40	31	17	45	15	8,6	6,4
^{134}Cs	0,4	0,6	0,6	1,6	0,6	–	0,6
^{136}Cs	0,3	0,4	0,5	0,9	–	–	0,I
^{137}Cs	–	–	1,4	3,7	1,3	2,2	0,4
^{140}Ba	3,2	4,1	8,0	3,3	13	12	3,8
^{140}La	11	4,7	15	2,3	19	17	4,0
^{141}Ce	1,4	1,9	7,6	0,9	6,4	15	3,6
^{144}Ce	1,6	2,4	6,1	–	5,1	11	3,4
^{147}Nd	1,4	1,7	2,5	–	2,1	5,4	1,4
^{239}Np	23	3,0	11	0,6	2,8	6,8	56,7

Table 4. Comparison of FISPIN and Soviet Shutdown Inventories

	Activity (FISPIN) MCi	% of subtotal	Soviet % of subtotal	Soviet / FISPIN ratio
^{95}Zr	156	5.6	3.6	0.64
^{95}Nb	159	5.7	3.8	0.66
^{99}Mo	148	5.3	3.9	0.74
^{103}Ru	116	4.1	3.9	0.95
^{106}Ru	23	0.84	2.1	2.5
^{131}I	81.1	2.9	2.3	0.79
$^{132}Te + ^{132}I$	221	7.9	6.4	0.81
^{134}Cs	4.05	0.144	0.6	4.17
^{136}Cs	2.16	0.077	0.1	1.30
^{137}Cs	6.48	0.231	0.4	1.72
^{140}Ba	154	5.5	3.8	0.69
^{140}La	159	5.7	4.0	0.70
^{141}Ce	148	5.3	3.6	0.68
^{144}Ce	105	3.7	3.4	0.92
^{147}Nd	59.5	2.1	1.4	0.67
^{239}Np	1319	45.0	56.7	1.26
Subtotal	2866			

Table 5. Comparison of calculated and measured isotope ratios in Chernobyl release

	Calculation		airborne above reactor			
	FISPIN*	Soviet	day 0	day 6	day 7	day 8
Cs $\frac{134}{137}$	0.63 (0.63)	1.49	-	0.43	0.43	0.46
Ru $\frac{106}{103}$	0.21 (0.24)	0.54	0.38	0.27	0.43	0.19
Ce $\frac{144}{141}$	0.70 (0.84)	0.94	1.14	0.80	-	0.80

* value at shutdown; day 10 value in brackets.

Table 6. Comparison of relative release fractions from remote and local airborne samples

Element	Isotope	Relative Release Fractions: W European Samples	Relative Release Fractions: Chernobyl 26.4.86
Cs	137	(1.0	1.0
	134	(	
I	131	(	
	132	(0.35 - 0.50	0.72
	133	(	
Te	132	0.55 - 0.90	1.84
Ba	140	0.035 - 0.050	0.21
La	140	0.005	
Ru	103	0.055 - 0.120	0.19
	106		0.34
Ce	141	0.005	0.09
	144		
Nb	95	0.005	
Np	239	0.002 - 0.014	0.19

Table 7. Comparison of relative release fractions from deposition samples in Chernobyl and Kiev regions

Isotope	Relative Release Fractions: Average Chernobyl Soil Samples	Relative Release Fractions: Kiev UK Student Shoe*
Cs137	1.0	1.0
134	1.52	0.74
I131	0.89	1.22
Te132	1.00	1.47
Ba140	0.14	-
Zr85	0.12	1.82
Ru103	0.17	1.02
106	0.16	1.38
Ce141	0.13	-
Np239	-	1.02
Pu238	0.010	4.21
Pu(239 + 240)	0.007	3.10
Pu241		1.95
Am241		11.2
Cm242		4.57
Cm244		8.89

* measured by NRPB - private communication

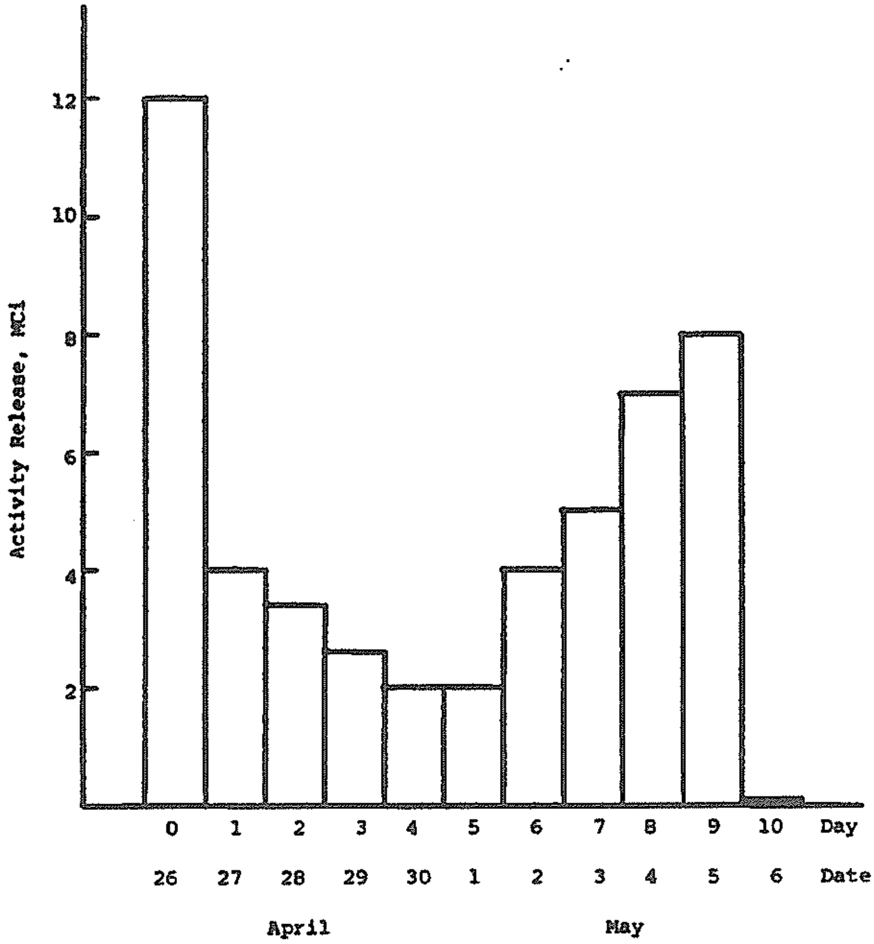

FIG 1 Soviet account of day-by-day activity release, corrected to 6 May

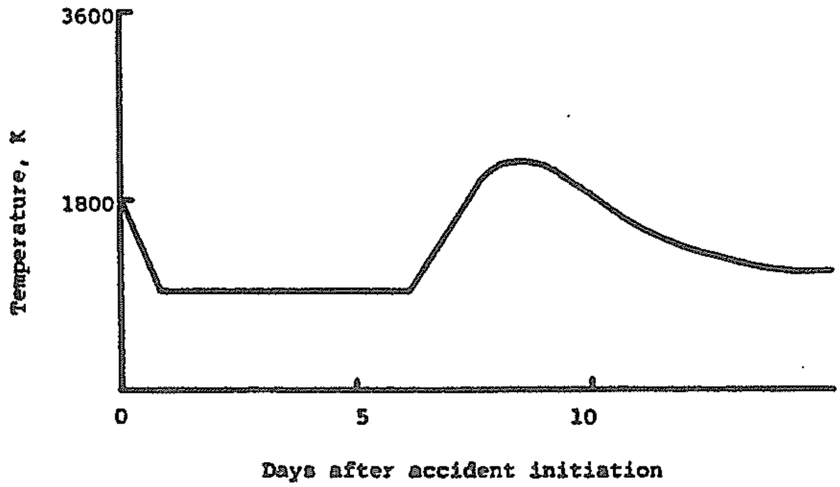

FIG 2 Soviet estimate of temperature history of Chernobyl fuel

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THE CHERNOBYL ACCIDENT - WHAT HAPPENED

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SUMMARY

This paper describes the events preceding the accident at Chernobyl and the way that they led to the initiation of the accident. The development of the accident post-initiation is also discussed.

INTRODUCTION

The information on the accident at Chernobyl Unit 4 provided by the Russians at the IAEA Experts' Meeting in Vienna in late August this year has greatly clarified the events leading up to the accident (see Ref 1). A detailed chronology and analysis of operator actions and plant state up to the moment of fuel fragmentation has been presented and a general appreciation of subsequent events obtained, although owing to the complex nature of the accident, some details of the underlying physics are still uncertain.

Several features of RBMK operation played a role in the development of the accident at Chernobyl and so normal operation is briefly outlined first to provide a point of reference. The development of the accident is then described and discussed. In probabilistic safety assessments, nuclear reactor accident sequences are usually broken down into a number of steps such as: accident initiation; cessation of fission by reactor shutdown; provision of cooling to avoid core degradation; and containment of fission products released from the pressure circuit. It is recognised however, that some initiating events might be so severe that the engineered safeguards of reactor shutdown and core cooling would be unable to cope with the conditions arising, although the reactor design should make such events highly improbable. The steps listed above would not then provide an appropriate framework for the description of the accident. The initiating event at Chernobyl appears to have fallen into the latter class, so rather than consider the accident in terms of the steps above, it will be discussed under the headings: accident initiation; failure of shutdown system; core degradation and pressure circuit failure; and response to the accident.

1 Normal operation of a reactor

Most power stations generate electricity via a thermal process, that is heat is used to boil water and raise steam that is fed to turbogenerators to generate electricity. In nuclear power stations, the heat source is a nuclear reactor in which fission of the nuclei of atoms of certain heavy elements (known as fissile elements) is induced by collisions with neutrons. Fission of a nucleus releases a relatively large amount of energy (about 200 MeV), which is divided amongst the products of the fission. These are principally two large fragments of the nucleus, which are themselves nuclei, plus neutrons and gamma rays. Over 80% of the energy released manifests itself as kinetic energy of the fission fragments and this is rapidly converted to heat in the fuel as the fragments collide and slow down.

Additional heat is produced by the absorption of neutrons and gamma rays in the reactor, and by radioactive decay of fission products. Neutrons released in fissions are themselves responsible for inducing further fissions, and so a "chain reaction" is established in which fissions at one time provide the conditions for further fissions at a subsequent time.

In a steady state, the rate of fission at any given time must be equal to that at any preceding time, which in practice means that the number of neutrons in the reactor must be the same from one generation to the next. The reactor is then said to be "critical". On average however, 2½ neutrons are emitted in each fission, so just under half of the neutrons produced must induce fissions in the reactor. The remainder are either absorbed within the reactor, or escape from it.

All the components of the core of a reactor absorb neutrons. Only a small proportion (about 2%, depending on the fuel enrichment) of the heavy elements in the fuel are fissile and some neutrons are absorbed in the fuel without causing fissions. These are mainly captured by the isotope of uranium, ^{238}U, which is a major constituent of the fuel. Some of these captures result in the formation of the plutonium isotope ^{239}Pu, which is itself fissile and so the depletion of the initial charge of fissile elements is to some extent compensated by the production of new fissile material during reactor operation. Furthermore, capture by ^{238}U of neutrons with energies less than about 1 KeV results in an effect with an important bearing on reactor stability. So called "resonances" between low energy neutrons and ^{238}U nuclei, cause efficient capture of neutrons whose energies lie within certain very narrow bands. As the fuel temperature rises, the widths of these bands increase due to thermal motion of the absorbing nuclei, which increases the probability of neutron capture. Consequently if a small disturbance were to raise the power of a reactor, the fuel temperature would rise, neutron capture in the fuel would increase and so the rate of fission in the reactor would tend to decrease. A reduction of the rate of fission would reduce the reactor power and so this neutron capture effect would tend to stabilise the reactor against disturbances in power. This effect is expressed technically through the Doppler coefficient of reactivity.

Neutrons are also absorbed in structural materials, the coolant, moderator and control rods. Except for absorption in control rods, such absorption is disadvantageous and materials are chosen to minimise neutron absorption, subject to other design requirements. At Chernobyl, light water was used as coolant and graphite as the moderator. In some reactors eg pressurised water reactors, light water plays the dual role of both coolant and moderator. Although a moderator absorbs neutrons, its principal function in a reactor is to minimise non-productive neutron capture. It does this by scattering the high energy neutrons produced by fission so that their energies are rapidly reduced to values below those at which resonant capture in the fuel could occur. This reduces non-productive capture in the fuel, allowing a chain reaction to be sustained in fuel with low fissile material content. The combination of water coolant and graphite moderator in the Chernobyl reactor was disadvantageous for reactor stability however, since the graphite alone could effectively moderate the nuclear reaction and the main neutronics effect of the water was therefore as an absorber. Such a reactor is said to be "over-moderated". Consequently an increase of the volume of steam in a fuel channel, expressed through the void fraction, reduced neutron absorption in the channel and tended to increase the reactivity and power of the reactor. This is expressed technically through the positive void coefficient of reactivity. Since a small disturbance increasing the power would increase the void fraction in fuel channels and this would tend to increase the power further, the positive void coefficient exerts a destabilising influence on the reactor. Operating above 20% of full power however, the Doppler coefficient dominates in RBMK designs and the reactor could be operated stably. Below this level, the reactor would enter an unstable regime. This was recognised by the designers and so operation below 20% full power was forbidden.

The control rods provide control of reactor power by allowing neutron absorption in the core to be varied. This is necessary for reactor start-up and shutdown and to compensate for local and core-wide variations of reactivity during operation. The control rods are able to perform this function by virtue of the existence of both "prompt" and "delayed" neutrons in the chain reaction. About 99.5% of the neutrons released in nuclear fission are released essentially as soon as fission occurs and are known as prompt neutrons. They slow down and diffuse in the graphite moderator and are absorbed causing further fissions in a few milliseconds. If a reactor were critical on prompt neutrons alone, changes of power would occur on a timescale much shorter than the characteristic response timescale of any power control system and reactor control would be impossible. Fortunately, the 0.5% of neutrons that are not emitted promptly are emitted after delays of up to about one minute and provided that a reactor is operated so that is is critical only when both prompt and delayed neutrons are accounted for, reactor control presents no difficulties. Reactors are therefore designed to avoid prompt criticality or limit its consequences.

2 Accident initiation

It appears that at Chernobyl, an initiating event too severe for the engineered safety systems and containment occurred. The design of the reactor core itself made such an event possible, but in recognition of this, operating instructions had been written to prevent the relevant conditions arising. Unfortunately, these operating instructions were violated, opening up the possibility of the severe initiating event. The performance of an experiment then produced the necessary trigger for the event to occur.

Ironically, the experiment that triggered the initiating event was designed to improve the safety of the plant. The objective was to see whether the mechanical inertia in a turbogenerator, isolated from both its steam supply and the grid, could be used to supply electricity via the station distribution system to the main reactor circulation pumps. In essence what was being attempted was to use the turbine generator as a mechanical "flywheel" coupled to the pumps electrically.

When disconnected from the grid and steam supply, a turbine generator unloaded would take about 15 minutes to come to rest from 3000 rpm but when coupled to the pump motors might provide a few tens-of-seconds supply. Even so, given the rapid coast down of the pumps without this provision, the long 'scram' and diesel start times, this "flywheel" effect would have provided a valuable margin in the safety case.

The experiment had been attempted twice before in 1982 and 1984. On the latter attempt following isolation of the generator from the grid the voltage level in the unit system fell rapidly and the operators were unable to arrest the drop by manual control of the voltage regulator. The fall in voltage resulted in the pump motors slowing down much faster than the generator. The reactor had been tripped at the start of the previous experiments however, so the reactor had shutdown successfully on those occasions.

For the experiment on the 26 April an automatic voltage regulator acting on the generator excitation current had been fitted with the aim of maintaining the voltage level in the unit system so that the pump motors ran down in step with the main generator at synchronous speed drawing upon the stored kinetic energy of the turbine generator.

The planned experimental initial conditions required the reactor to be at about 25% full power with one of its turbine generators shut down The other turbine generator would supply two main circulating pumps in each loop. The remaining two pumps in each loop and the auxiliary plant were to be fed from the grid.

The experiment had been badly planned, the safety case was inadequate, had not been properly reviewed and, as we shall see below the operators failed to achieve the chosen plant conditions, departed from the laid down procedures and violated several operating rules.

2a Chronology of the accident sequence

25 April 1986

01.00 Commencement of power reduction for maintenance shutdown.

13.05 50% power level (1600 MW(th)) achieved. Turbogenerator number 7 was disconnected from the grid and all house load was transferred to the still operating number 8 unit.

14.00 In accordance with the programme for a turbogenerator experiment, the reactor's ECCS was disconnected. Power was then held at 50% as the start of the experiment was delayed by a request from the controller in Kiev to keep supplying electricity to the grid. The ECCS was not switched back in violation of the operating rules.

23.10 The power reduction programme was resumed. The aim was to perform the test with the reactor at between 700 and 1000MW. On going to lower power, that set of reactor control rods used to control the power of the reactor at high powers (called the Local Automatic Rods), was switched out and a set of rods called the Automatic Rods switched in. However, the latter were set to continue power reduction rather than hold the power level as required and the operator was unable to stop the power of the reactor falling to 30MW(th).

<u>26 April</u>

01.00 Operator succeeded in stabilizing the reactor at 200MW(th). The combination of low coolant voidage at low power and the positive void coefficient made it difficult to achieve this level and the operator did so only by removing further control rods from the core. His reactivity reserve at this time was well below the limits laid down in the regulations.

01.03 and 01.07 One additional main circulation pump was switched into each coolant circuit. This made a total of 8 working. Following the experiment, four pumps would remain in operation, four having run down during the experiment.

Switching in these pumps increased the flow rate into the core. Since the reactor was already at low power, the hydrodynamic resistance was very low and the flow rate of water through the core was very high. Some pumps were operating beyond their permitted operating regimes. The increased flow caused a reduction in steam formation and a consequent fall in pressure in the steam drums. By this time essentially solid water was being circulated through the reactor. This action reduced the coolant subcooling in the pressure circuit and brought the whole circuit close to boiling.

01.19 Operators then tried to increase the water level in the steam drums by using the feedwater pumps - the reactor should have shut down on low water level in the steam drums but the operators had disengaged that signal. Because the water was replacing steam, the reactivity continued to drop and the operator compensated by removing manual control rods from the core. This reduced the reactivity reserve still further.

01.19.58 Turbogenerator bypass valve closed. Steam was no longer being dumped in the condenser and the fall of pressure caused by the feedwater flow lessened.

01.22 Operator reduced feedwater flow. No cooled water was then going into the reactor.

01.22.30 A print out of the reactivity reserve was produced - it was only 6-8 rods and immediate shut down was required but the operators continued (there is no automatic shut down on this signal).

01.23.04 Experiment began; the regulating valves to turbogenerator number 8 were closed. Reactor power was still about 200MW(th). The shutdown signal for loss of two turbogenerators had been blocked by the operators to permit a re-run of the experiment if the first were not successful.

The power of the reactor began to rise slowly.

01.23.40 The operator ordered full emergency shutdown. Three seconds later, there were high power and short period alarms. All control and shutdown rods were motored into the core but they were too late. Not all rods reached their low stops, the foreman 'unlatched' rods to fall under their own weight. Shocks were felt.

01.24 Observers outside the reactor at about this time reported two explosions about 3-4 seconds apart. Burning lumps of material and sparks were thrown into the air - some landed on the turbine hall and started fires.

2b Information derived by the Russians

RBMK reactors are equipped with the "Skala" centralised control system, which periodically records conditions in the reactor. For the turbogenerator experiment, only those conditions important for the analysis of the experiment results were recorded at high frequency. Nonetheless, data from the "Skala" system, combined with instrument readings and reports from the reactor operators, has enabled the Russians to validate a computer model

2c Discussion of the accident initiation

It is now seen that the initiating event in the accident at Chernobyl Unit 4 was a reactivity excursion. This resulted from the operators deliberately violating the operating regulations during preparations for the turbogenerator experiment and from the fact that the reactor was therefore operating with a positive power coefficient.

The void coefficient of RBMK reactors becomes more positive as absorbers are withdrawn from the core. At the time of the turbogenerator experiment, most rods had been withdrawn to counter the reduction of reactivity resulting from having too much water in the core. The void coefficient was therefore positive and unusually large at the time of the test. Furthermore, as there was little steam in the core, the potential existed for a large change of voidage and reactivity. The Doppler coefficient that normally dominated the void coefficient was insufficient because the reactor was at low power and the fuel temperature was consequently low, resulting in the power coefficient being positive. Consequently, there was a rapid escalation of power as the pumps ran down and steam generation increased. Then as most absorber rods were withdrawn from the core, the speed with which they effected shutdown on demand was significantly reduced and the reactivity excursion continued after shutdown was initiated.

There are also additional factors that might possibly have contributed to the reactivity excursion by accelerating the development of voids in the coolant. Firstly, coolant flow in the fuel channels at the start of the turbogenerator test would have been in the bubbly flow regime. As the pumps ran down and the rate of steam generation increased, a flow pattern transition from bubbly flow to annular flow might have occurred (see Appendix 1), rapidly increasing the void fraction and hence reactivity of the core. Secondly, parallel channel instabilities might have occurred, causing rapid voiding of some channels although this could have been compensated by reduced voidage in other channels. Finally, interactions between the pumps that continued to operate and those that were running down might have caused the latter to stop prematurely, resulting in a more rapid flow reduction than anticipated. The pressure head developed by the pumps that continued to operate would oppose the continuation of flow through the pumps that were running down.

Finally, it is noted that the reactivity excursion drove the core into a prompt critical state. The possibility of a thermal reactor becoming prompt critical in an accident has been recognised for many years and discussed in textbooks on nuclear reactor safety. The rate of power increase of a graphite moderated reactor would jump by about two orders of magnitude once it became prompt critical.

3 Failure of shutdown system

A crucial step in an accident is the shutdown of the fission reactions in the core. This action, known as either reactor trip or scram, is the first of a series of safeguards intended to prevent accident progression to core degradation. It causes the rate of heat generation in the core to fall to less than 5% of the operating value within a minute of shutdown and should place the heat generation rate well within the capability of the coolant to accept heat from the core.

The Russians have informed us that the operator at Chernobyl Unit 4 ordered a full emergency shutdown at 1:23:40 on April 26. All control and shutdown rods were then motored into the core. Not all of them reached their lower stops however, so the operator uncoupled them so that they could fall under their own weight. During this period (of tens of seconds), a number of shocks were felt. Nuclear fission in the core then ceased. At about this time, the pile cap was blown off and the reactor building was extensively damaged.

It is clear that emergency shutdown was initiated and nuclear fission in the core ceased. It appears unlikely however, that shutdown was due to insertion of control and shutdown rods. As the reactor went prompt critical, these could not have been inserted rapidly enough to terminate the reactivity excursion. In the view of experts at the IAEA Meeting, disintegration and dispersal of fuel in the accompanying explosions must have contributed to the shutdown. It therefore appears that major damage occurred before the reactor was shutdown. The premature arrest of the motion of control and shutdown rods into the core seems to support this. Furthermore, control rods were lifted when the pile cap blew off but fission still ceased, pointing to fuel dispersal and core damage as the reasons for shutdown.

4 Core degradation and pressure circuit failure

The Russians have told us that fuel degradation occurred while there was still water in the fuel channels, due to a heat transfer crisis at the surface of the fuel rods. The fuel then disintegrated and rapidly generated steam by mixing with the water present in the channels. The abrupt pressure pulse that this generated ruptured fuel channels and resulted in the 3m thick pile cap and floor above it being blown off and core materials being ejected into the atmosphere. Part of the emergency core cooling system capability was destroyed by the explosions.

It is known that pressure tubes failed in the accident at Chernobyl and their manner of failure is possibly of secondary importance for an appreciation of the consequences of their failure. Nonetheless, to demonstrate the more robust response of other reactor systems were a similar fuel disruption event to occur, it is necessary to understand the manner of failure at Chernobyl.

There are several related phenomena that could have caused disruption of the fuel, rupture of fuel channels containing water and the expulsion of the pile cap. These are summarised below. The feature common to all of these is that fuel fragments and its surface area for heat transfer increases, allowing very rapid steam generation.

A mechanism for fuel disruption was proposed at the IAEA Meeting. When high burn-up fuel is heated slowly, the pressure exerted by volatile fission products trapped on grain boundaries within it increases and this can cause the fuel to swell. If the fuel is heated very rapidly however, tensile stresses resulting from the pressure build-up cannot be relieved sufficiently quickly and the fuel can disintegrate into a fine powder, blown apart by the pressure of the volatile fission products. At Chernobyl, the fuel was heated rapidly. It has been suggested that if the heating were rapid enough, the fuel might have disintegrated prior to melting and been expelled into surrounding water, resulting in a very rapid burst of steam generation.

The first mechanism for rupture of fuel channels and expulsion of the pile cap is the steam explosion. In this case, fuel and clad would melt, mix with water remaining in the fuel channel and then fragment into fine particles, rapidly transferring heat to the water and causing an abrupt rise of pressure. The general geometry of an RBMK fuel channel is reminiscent of some of the configurations that have been used in experiments to promote steam explosions.

The second phenomenon that might have ruptured fuel channels is the "steam spike". As for steam explosions, this requires fuel to melt and mix with water. Steam spikes differ from steam explosions however, in that heat transfer occurs over a more extended period (perhaps tens of seconds rather than milliseconds) and steam spikes are not accompanied by the shock waves and fine fuel fragmentation that are characteristic of steam explosions.

The simplest of these three explanations appears to be the steam explosion in single fuel channels - indeed, a steam explosion has occurred in an experiment (Ref 2) under conditions almost identical to those in Chernobyl Unit 4. The Russians have said that the fuel temperature reached 3000 C, which is above the melting point of the oxide fuel, and so molten fuel would have been available to participate in steam explosions. Furthermore, the participation of molten zirconium alloy clad in steam explosions would have resulted in its fine fragmentation and rapid hydrogen generation. A chemical explosion was thought to have occurred three to four seconds after the pile cap was blown off, which would have required the rapid generation of combustible substances such as these. Indeed, in steam explosion tests involving molten aluminium, chemical explosions involving the rapid oxidation of finely divided aluminium have often been observed.

Steam spikes appear to be a less likely explanation of fuel channel rupture. As they occur over a more extended period than steam explosions, time is available for pressure relief within the system. Furthermore, heat transfer from the fuel to coolant becomes inefficient once the pressure has risen above the critical pressure of the coolant. For PWRs, the result of this is that the maximum pressure generated by a steam spike is assessed to be not much greater than the critical pressure of water. As the fuel channels at Chernobyl Unit 4 contained water during the reactivity excursion, the strength of the pressure tubes would have been close to normal and rough estimates (Appendix 2) indicate that they could have withstood the critical pressure of water. This suggests that a steam spike would not have ruptured fuel channels.

It should also be noted that the timescale of the reactivity excursion at Chernobyl was relatively long and the characteristic timescale for power escalation was probably greater than the thermal penetration time of the clad. The clad temperature would, therefore, probably have risen in unison with the fuel temperature. The temperature at which fuel disintegration might occur is thought to be slightly higher than the clad melting temperature and much greater than the temperature at which rapid exothermic clad oxidation would occur. Consequently, it appears likely that clad would have melted and mixed with water in a fuel channel before the fuel disintegrated. A steam explosion in a fuel channel involving molten clad alone could be sufficiently energetic to rupture it.

The Russian analysis of the accident (Figures 1a-1c) indicates that the fuel overheated and disintegrated while still surrounded by water. Had the fuel channels dried out in some locations however, the pressure tubes might have weakened and failed due to thermal radiation from the fuel and fuel relocation leading to contact between the fuel and pressure tubes. The integrity of the pressure tubes would be in doubt once they reached a temperature of about 700C. Simple scoping calculations of transient heat transfer indicate that if heat were generated in the fuel at three times the operating rate, the fuel would have risen in temperature to the clad melting temperature (about 1850C) by the time the pressure tube wall had reached 700C. At higher powers, fuel melting would occur before pressure tube failure. As the power in the Chernobyl reactor surged to 100 and 440 times the operating value, the pressure tubes would have been expected to fail as a result of fuel relocation creating hot-spots on the pressure tube walls rather than thermal radiation from fuel had the fuel channels dried out.

The core inerting system at Chernobyl Unit 4 was designed to relieve the pressure that would have resulted from a single channel rupture. Following the rupture of several channels, the core space was pressurised and the pile cap was blown off. The energy required to cause this damage might have been supplied by one of the three fuel-coolant interaction mechanisms described above or by the thermal energy already stored in the coolant in the fuel channels. Thermodynamic models of fuel-coolant interactions make no assumptions about the process of fuel fragmentation. A recently developed thermodynamic model can be used to estimate the mechanical yield of any of the three types of fuel-coolant interaction described above. Assuming that the mass of fuel participating in these interactions was equal to the mass of fuel expelled from the reactor vault (about 4% - similar to the mass in the smallest critical configuration in the core), mechanical work of order 1 GJ could have been done by fuel-coolant interactions (see Appendix 3). Even in the absence of a large contribution to mechanical yield from fuel-coolant interactions, the coolant in the fuel channels might do mechanical work of order 1 GJ in escaping from ruptured channels. These yields compare with rough estimates of the work done blowing off the pile cap in the range 0.2 to 2.0 GJ (Appendix 3).

Following the explosions, the core cooling systems and all fuel channels were destroyed. The accident then developed as a loss of coolant accident affecting the whole core.

5 Response to the accident

At Chernobyl, the containment was breached within seconds of accident initiation. Witnesses outside Unit 4 reported that burning lumps of material and sparks shot into the air above the reactor. It was likened to a fireworks display. Some of the burning material fell onto the roof of the turbine hall and started a fire that put Unit 3 at risk. The top of the core was exposed to the atmosphere and was seen later glowing red-hot in a video recording shown at the IAEA Meeting. The Russians dropped various materials onto the core to mitigate fission product releases, in the following order: 40 Te of boron carbide was dropped first to ensure shutdown; then 800 Te of dolomite (limestone) was dropped, as this releases carbon dioxide as it heats up and decomposes, which would starve the core of oxygen; 2400 Te of lead was then dropped, in the hope that this would melt, run through the core debris and carry heat away; finally, clay and sand were dropped to filter and retain fission products escaping from the core. Altogether, about 5000 Te of material was dropped onto the core.

Following the accident, the fuel temperature varied in a non-monotonic manner (see Figure 2). At the moment of the explosion, the fuel temperature had reached 3000 C (at least locally). Immediately afterwards, the effective temperature of fuel remaining in the core was assessed to be 1300-1500 C and this fell over several tens of minutes as heat diffused into the graphite. Subsequently, the fuel temperature increased again over a period of several days due to fission product decay and the thermal insulation provided by the materials dropped onto the core. Around 4-5 May, the fuel temperature reached a maximum of about 1900 C and then began to decrease, apparently as a result of improved core cooling by circulating air and nitrogen injection.

Long term control measures have also been instigated by the Russians. The pressure suppression pools beneath the reactor have been drained and a flat heat exchanger embedded in concrete has been placed below the building foundations. Unit 4 will eventually be entombed.

The state of the core following the explosions is uncertain. It is known that the pressure tubes leading from the tops of the fuel channels separated from them when the pile cap was expelled, but the state of the pressure tubes leading to the bottom of the core is unknown. Fuel would not have been lifted out of the core by the rising pile cap, because the force required to rupture the fuel channels would have caused the pile cap to accelerate so rapidly that the tubes supporting the fuel assemblies would also have snapped. Similarly, the control and shutdown rods might also have been expected to remain in the core, although the Russians reported that these were lifted when the pile cap was blown off. Being no longer suspended from the top of the core, the fuel and absorber rods would have tended to fall towards the bottom of the core. Fuel would not, however, have been able to migrate into the lower (inlet) pressure tubes unless molten or fragmented.

The core temperature would then have gradually risen, due to decay heating and oxidation of core components. The materials dropped onto the core would have thermally insulated the core and inhibited ingress of air. The core temperature of about 1900 C on 4-5 May would have been expected from decay heating of a well insulated core (Appendix 4). Dropping lead onto the core would not have had a great effect on this heat-up, as the melting point of lead is relatively low (328 C) and its thermal capacity would have been about an order of magnitude less than that of the graphite (Appendix 4). It would, however, have plugged the inlet piping if this were still intact and prevented cooling by circulating air or gases along the pipes.

The Russian explanation of the arrest and subsequent fall of core temperature on 4-5 May is somewhat puzzling. They have said that the temperature began to fall due to the formation of a stable convective air flow through the core into the free atmosphere and pumping nitrogen into the space beneath the reactor. A scoping calculation of free convection through a fuel channel, open at both ends, indicates that the fuel would have had to heat the air to about 1000 C to create a sufficient draft to remove the decay heat (Appendix 5). At this temperature, the zirconium alloy clad and graphite would have burned and exacerbated the heat rejection problem. Furthermore, the fuel channels were not open to the atmosphere at both ends: the core was covered by about 8m of dolomite, sand and clay, and lead that had melted and run through the core might have plugged the inlets. Thus, still higher fuel temperatures would have been required. To remove the decay heat by gaseous nitrogen injection would have required an injection rate of about 0.5 Te per minute.

A more likely explanation would have been that the zirconium alloy pressure tubes and/or stainless steel inlet pipework melted on 4-5 May and fuel debris fell from the graphite onto the concrete floor of the reactor vault. Concrete melts at about 1400 C, so core-concrete interactions could have brought about the fall in core temperature. Lead boiling could also have played a role (boiling point of lead = 1740 C). The Russians have said however, that the core debris did not interact with the concrete and that the maximum temperatures in the reactor vault are now only several hundred degrees Celsius.

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KEY TO THE CURVES ON FIG. 1 (from Figure 4 of the Soviet Report [Ref. 1])

		MIN	MAX			MIN	MAX
A	Neutron power (%)	0	120	K	Flow, MCP (m^3/s)	2	8
B	Reactivity, sum. (%)	-1	-5	L	Flow, Feedwater (kg/s)	0	600
C	Pressure, steam drum (bar)	54	90	M	Flow, steam (kg/s)	0	600
D	Neutron power (%)	0	48000	N	Fuel temp. (°C)	200	2000
E	Rod group AR-1 (fraction inserted)	0	1.2	O	Steam mass quality (Exit of core, %)	0	6
G	Rod group AR-2 (fraction inserted)	0	1.2	P	Steam vol. quality (Core average, void fraction)	0	1.2
H	Rod group AR-3 (fraction inserted)	0	1.2	S	Level (steam drum, mm)	-1200	0

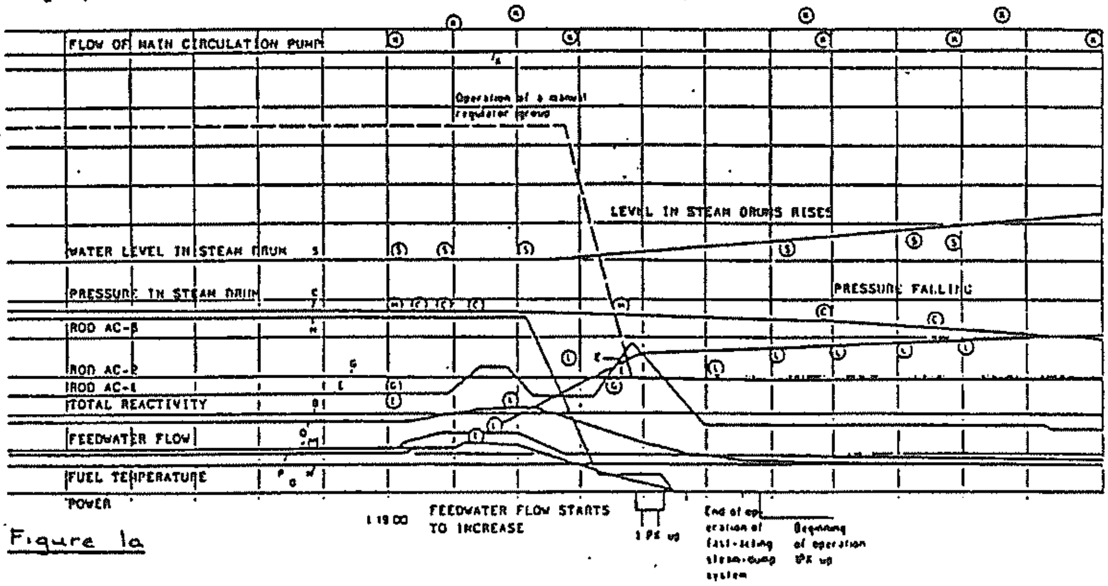

Figure 1a

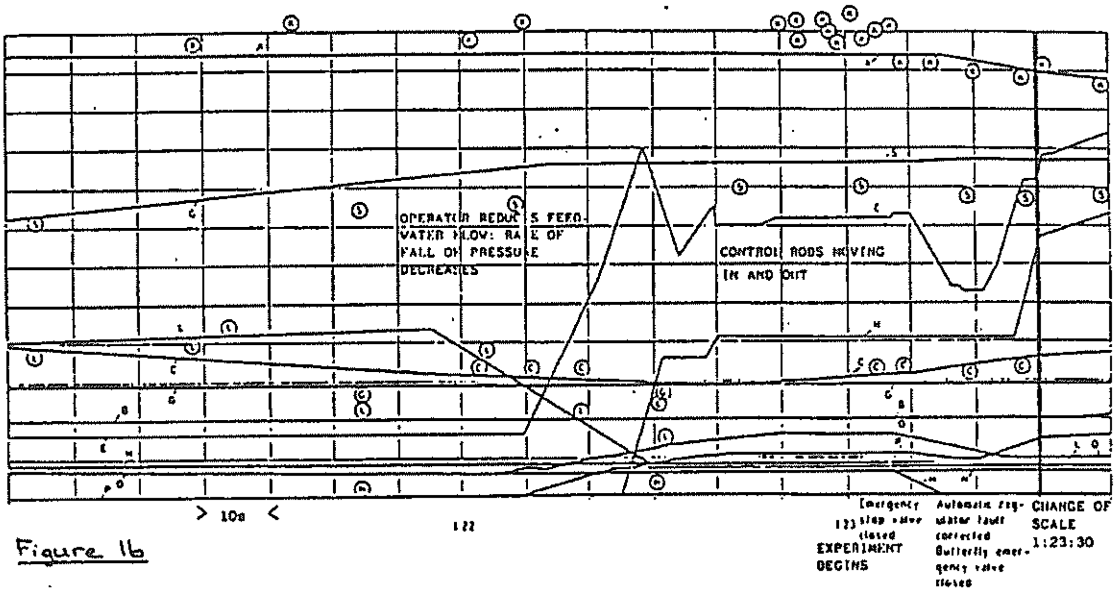

Figure 1b

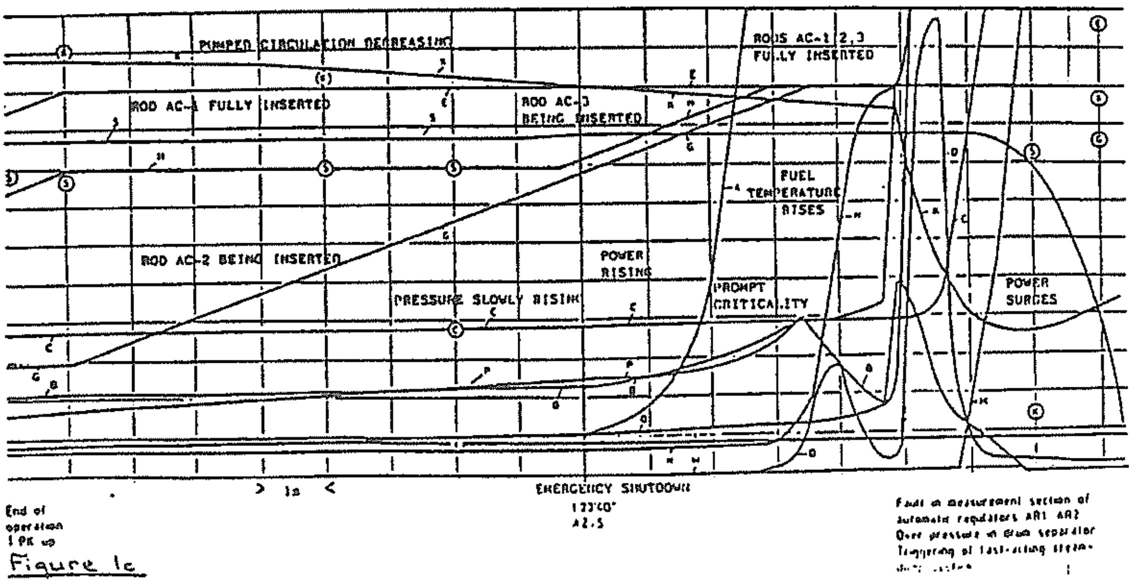

Figure 1c

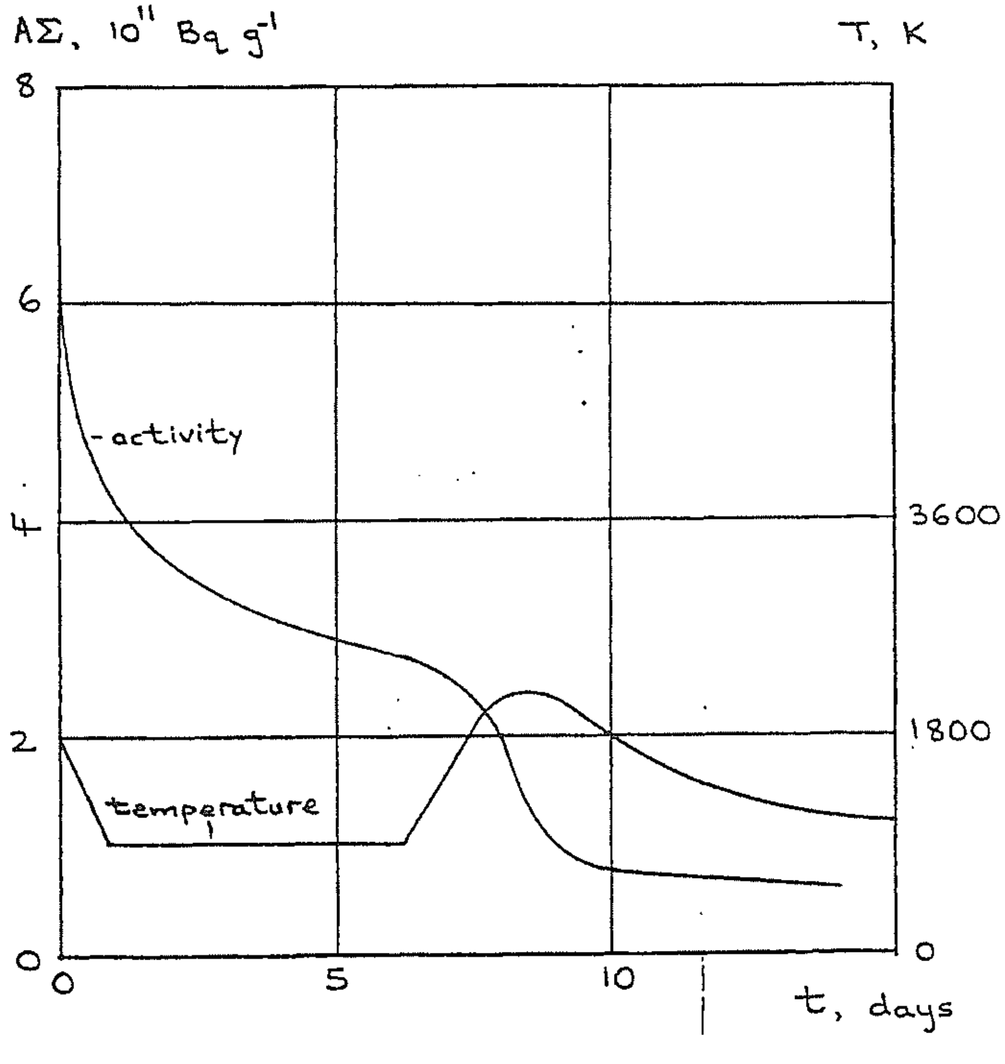

Figure 2   Variation of activity and temperature of the fuel with time

Appendix 1 - Flow Pattern Transition

The pattern of flow in a channel of a boiling water reactor usually develops as illustrated in Figure A1.1. Subcooled liquid enters the channel inlet and after flowing some distance up the channel, the liquid starts to boil on the surfaces of the fuel rods. Travelling further up the channel, the whole cross-section of the flow contains vapour bubbles and the flow is then described as bubbly. Still further up the channel, the flow consists of a liquid film containing bubbles flowing up the surfaces of the fuel rods and a vapour core containing entrained droplets of liquid. This is known as annular flow. In normal operation, the coolant flow conditions at the channel exits of boiling water reactors are well beyond the transition from bubbly flow to annular flow.

If the exit conditions were close to the transition point however, a flow pattern instability could occur (Ref A1-1). A small reduction of inlet flow would result in increased vapour generation and a transition from bubbly flow to annular flow might occur near the top of the channel. The resistance to flow of annular flow is less than that of bubbly flow, so the inlet flow would increase and vapour generation would subsequently decrease. A further transition from annular flow back to bubbly flow would then occur. The flow resistance would increase again, the inlet flow would decrease and the cycle would then be expected to repeat itself.

In an RBMK reactor, coherent transitions in several channels might be expected, as the increased flow to the inlet of a channel in which a transition from bubbly to annular flow occurred would reduce the flow to neighbouring channels and encourage a similar transition in those. Furthermore, the transition from bubbly flow to annular flow would increase the void fraction in a channel and locally raise the reactor power through the positive void coefficient. This would also encourage a transition from bubbly flow to annular flow in neighbouring channels and might furthermore change the nature of the instability from oscillatory to excursive.

Conditions in the channels of Chernobyl Unit 4 at the start of the turbo-generator test will now be estimated. From Figure 4 of Ref 1, the coolant mass quality at the core exit was 0.78% and the volumetric flowrate at the core inlet was 7.7 $m^3\ s^{-1}$. The flowrate is taken to be the total flowrate in one loop: if it were the flowrate per pump, the reactor power would be insufficient to obtain the exit mass quality, even with zero subcooling at the core inlet; furthermore, the total flowrate per loop in normal operation is about 6.9 $m^3\ s^{-1}$. The circuit pressure at the start of the test was 6.3 MPa, so the steam temperature in the steam drums would have been 279C. Equating the heat generated in one loop with that required to raise the temperature of the inlet water to the boiling point and generate an exit mass vapour quality, X, of 0.78%, it is found that

$$\Delta T = \left(\frac{\dot{Q}}{\dot{m}} - Xh_{fg}\right)/c_p \qquad (A1.1)$$

where ΔT is the initial subcooling of the water, $\dot{Q}$ is the heat generated in one loop, $\dot{m}$ is the mass flowrate per loop, h_{fg} is the latent heat of vapourisation of water and c_p is the specific heat of water at constant pressure. As $\dot{Q} = 10^8$W, $\dot{m} = 5.79 \times 10^3$ kg s^{-1}, $X = 7.8 \times 10^{-3}$, $h_{fg} = 1.55 \times 10^6$ J kg^{-1} and $c_p = 5.2 \times 10^3$ J kg^{-1} K^{-1}, ΔT is found to be 0.99C.

The superficial steam flow velocity, v_s, at a channel exit is easily shown to be given by

$$v_s = (\dot{Q} - \dot{m}c_p\,\Delta T)/(\rho_s\,Ah_{fg}) \qquad (A1.2)$$

where ρ_s is the steam density and A is the flow area of channels in one loop. As $\rho_s = 32.5$ kg m^{-3} and $A = 2.00\ m^2$, the superficial steam velocity at the channel exits is given by

$$v_s = 0.99 - 5.1 \times 10^{-5}\,\dot{m}\ . \qquad (A1.3)$$

$v_s = 0.69$ m s^{-1} at the start of the test and would have increased towards 0.99 m s^{-1} as the pumps ran down.

The critical velocity, v_c, for transition from bubbly flow to annular flow is given by the formula

$$v_c = A' \{ g\sigma (\rho_\ell - \rho_s)\}^{1/4} / \rho_s^{1/2} \qquad (A1.4)$$

A' is a constant (see below), g is the acceleration due to gravity, σ is the surface tension of the water and ρ_ℓ is the density of the liquid water. There is no generally accepted value for A': a value of 3 is quite common (eg Ref A1-2), although a value of 2 has been proposed in a discussion of entrainment of liquid droplets (Ref A1-3). Taking $\sigma = 1.91 \times 10^{-2}$ N m^{-1} and $\rho_\ell = 753$ kg m^{-3}, $v_c = 1.20$ m s^{-1} (A' = 2) or $v_c = 1.79$ m s^{-1} (A' = 3). It therefore appears that the coolant conditions at the channel exits would have approached those at which transition from bubbly flow to annular flow would have occurred as the pumps ran down. A flow pattern transition in the fuel channels might therefore have contributed to the rapid channel voiding and reactivity excursion at Chernobyl Unit 4.

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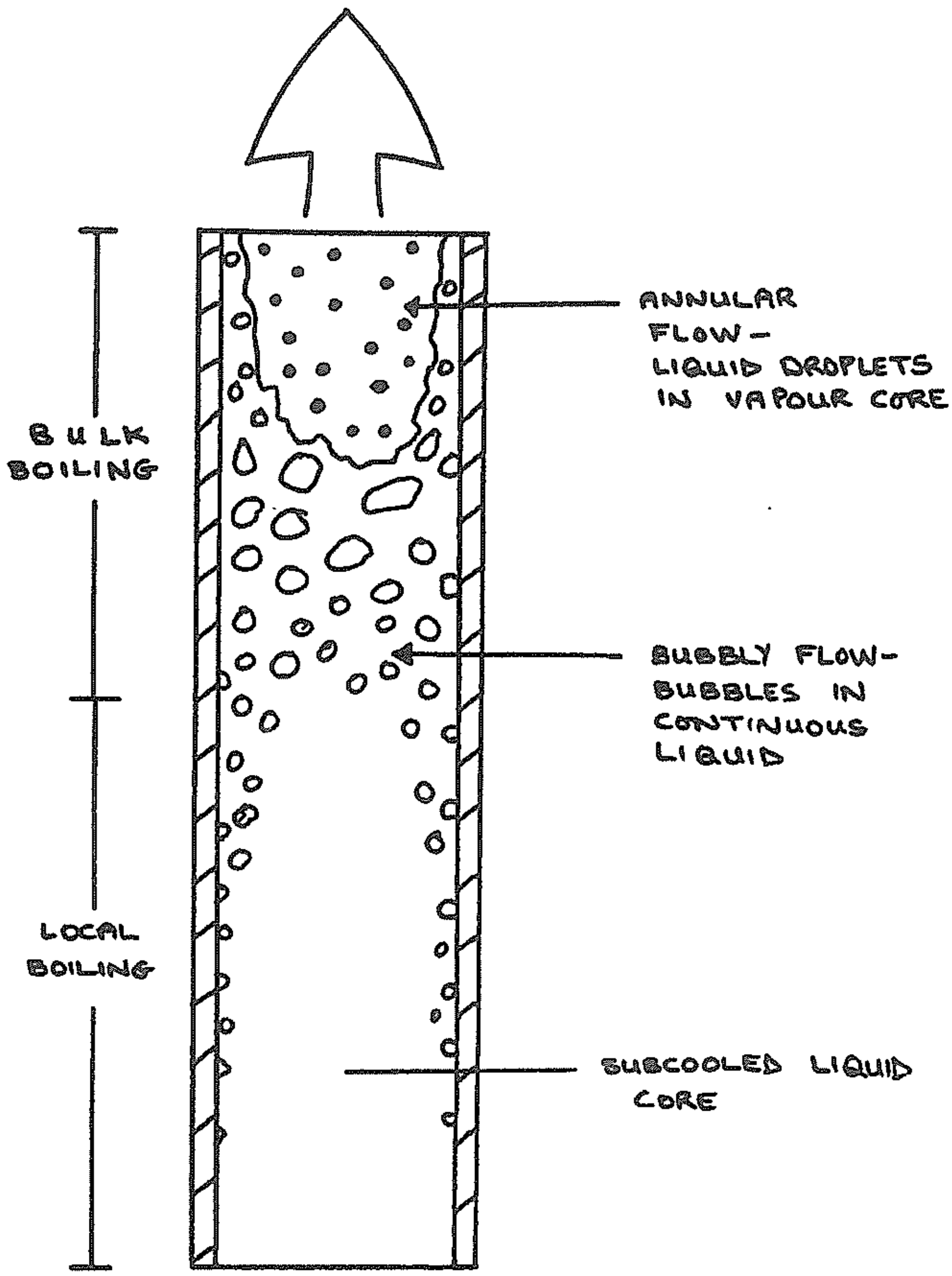

FIGURE A1.1 FLOW IN A CHANNEL OF A BOILING WATER REACTOR

Appendix 2 - Pressure Tube Failure by Steam Spike

Assessments of steam spikes in PWRs have found that the maximum pressure achieved is not much greater than the critical pressure of water, because heat transfer to water becomes significantly less effective after the transition from two-phase conditions below the critical pressure to monophase conditions above that pressure (see Ref A2-1). In this Appendix, the stress in the wall of a zirconium alloy pressure tube in an RBMK reactor is calculated for a circuit pressure equal to the critical pressure of water. This is compared with the ultimate tensile strength (UTS) of the material under normal operating conditions.

The hoop stress in a tube of radius, r, wall thickness, t, pressurized to a gauge pressure, P, is given by

$$\sigma_h = rP/t \ . \qquad \text{(A2.1)}$$

The longitudinal stress is half the hoop stress. As the mean radius of the pressure tube is 42mm and its wall thickness is 4mm, the hoop stress would be 232 MPa when the internal pressure was equal to the critical pressure of water (22.1 MPa).

The material properties of the zirconium alloy are expected to be very similar to those of zircaloy 4. In normal operation and as long as water remained in the fuel channels during the test on the turbogenerators at Chernobyl Unit 4, the pressure tube temperature would have been about 300C. At this temperature, the UTS of zircaloy is about 317 MPa (Ref A2-2). This is about one third greater than the hoop stress in the pressure tubes for an internal pressure equal to the critical pressure of water and suggests that a steam spike would not have ruptured fuel channels.

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Appendix 3 - Energy Required to Expel the Pile Cap

In this Appendix, the potential mechanical yields of fuel-coolant interactions and of expansion of coolant escaping from ruptured fuel channels are estimated. These are compared with estimates of the work done to expel the 3m thick pile cap and its attachments from the reactor vault, as occurred at Chernobyl.

A3.1 Mechanical yield of fuel coolant interactions

Rupture of fuel channels at Chernobyl Unit 4 appears to have been due to one of three types of fuel-coolant interactions. The three types of fuel-coolant interaction described in the main text differ in the extent and manner of fuel fragmentation. Thermodynamic models of fuel-coolant interactions do not, however, require knowledge of the mechanism of fuel fragmentation and may therefore be used to estimate the mechanical yield of all three types of fuel-coolant interaction.

Most thermodynamic models of fuel-coolant interactions are based upon that of Hicks and Menzies (Ref A3-1) and provide upper limits to the mechanical yield. These models are not particularly useful however, as the upper limits that they provide are very much greater than the mechanical yields measured in experiments. There is, however, a new thermodynamic model of these interactions (Ref A3-2) that provides qualified lower limits to the mechanical yield that have been found to be in excellent agreement with the results of the SUW (Scale-Urania-Water) experiments at AEE Winfrith (Ref A3-3). This latter model will be used to estimate the potential yield of fuel-coolant interactions.

Ref A3-2 presented the results of calculations of mechanical yield for infinite systems in which the final system pressure was equal to the initial system pressure and for finite systems in which the final pressure exceeded the initial pressure. At Chernobyl however, the final pressure, being atmospheric, was less than the initial pressure of the fuel-coolant mixture as mixing occurred prior to circuit failure. The method of Ref A3-2 could be extended to assess the yield of interactions at Chernobyl, but that will not be done here. Instead, the results of calculations for an infinite system at atmospheric pressure and zero coolant subcooling will be used to estimate the efficiency of fuel-coolant interactions at Chernobyl. It is simply noted that the elevated initial pressure and coolant temperature would lead one to expect efficiencies somewhat higher than that found for an infinite system at atmospheric pressure.

For an infinite system at atmospheric pressure and zero coolant subcooling, Ref A3-2 predicts that the mechanical efficiency of fuel-coolant interactions based upon the mass of fuel that actually participates would be about 7%. The mass of fuel that participated in fuel-coolant interactions at Chernobyl is assumed to be equal to the mass of fuel ejected from the reactor vault, as except for the simple steam spike thought to be unlikely to have caused fuel channel rupture (see previous Appendix), the fuel-coolant interactions are expected to be accompanied by fine fuel fragmentation and the fine particles would be swept from the reactor vault by the escaping fluids. About 4% of the fuel was ejected from the reactor vault. Therefore, the overall efficiency of the fuel-coolant interactions referred to the total mass of fuel in the core is estimated to be about 0.3%. The energy stored in the fuel at a temperature of 3000C would be about 4×10^{11}J, so the mechanical yield of fuel-coolant interactions is estimated to be about 1.2 GJ.

A3.2 Mechanical yield of expanding coolant

The volume of water in the fuel channels is estimated to be 29 m^3. At 6.3 MPa, this water would have a mass of 2.2×10^4 kg if solid. The heat required to raise the temperature of this mass of water from the saturation temperature at atmospheric pressure to the saturation temperature at 6.3 MPa at constant volume would be about 1.5×10^{10}J. The potential mechanical yield resulting from expansion of this water can be estimated by noting that the work done during the expansion would be the same as that occurring if the same amount of heat had been transferred to the water from fuel in a fuel-coolant interaction. Therefore, assuming an efficiency of about 7% as indicated above, a mechanical yield of about 1.1GJ would be expected from expansion of coolant escaping from the fuel channels. Voids in the coolant in the channels would reduce the estimated yield, but the participation of coolant flowing into the channels from other parts of the pressure circuit would raise the yield.

A3.3 Work done expelling the pilecap

The mass of steel in the plates forming the outer surface of the pilecap is estimated to be 176 Te. The mass of serpentine infill is estimated to be 1065 Te. The mass of the outlet pipes leading from the top of the core out of the reactor vault is estimated to be 136 Te. The floor of the reactor hall is constructed of blocks of iron-barium-serpentine cement stone (density 4.0 Te m^{-3}) surrounded by metal cases containing pig iron shot and serpentine and is estimated to have a mass of 808 Te. Therefore, the total mass of the pilecap, its attachments and the reactor hall floor that it supports is estimated to be 2185 Te. To expel these structures from the top of the core would require raising them by at least about half the diameter of the pilecap ie 8.5m. The energy required to raise these structures 8.5m would be 182 MJ.

Furthermore, ejection of the pilecap would require the zirconium alloy pressure tubes beneath the pilecap and the stainless steel outlet pipes to be broken. The energy required to cause axial rupture of a pipe is estimated to be (Ref A3-4)

$$U = 6\pi\varepsilon\, \sigma_{UTS}\, (rt)^{3/2} \qquad \text{(A3.1)}$$

where ε is the maximum strain supportable by the material, σ_{UTS} is the ultimate tensile strength, r is the pipe radius and t is the pipe wall thickness. Taking ε = 40% and σ_{UTS} = 317 MPa for the zirconium alloy pressure tube (at300C), U_{Zr} = 5.2 kJ per tube, ie 8.6 MJ to rupture 1661 pressure tubes. Taking ε = 40% and σ_{UTS} = 483 MPa for the stainless steel outlet pipes, $U_{s.st}$ = 6.8 kJ per pipe ie 11.3 MJ to rupture 1661 pipes. Therefore, the energy required to rupture the pressure tubes and outlet pipes and lift the pilecap and associated structures by a distance equal to its radius would have been about 202 MJ. This compares well with the estimate of 200 MJ given by French experts at the IAEA Experts' Meeting in Vienna. Note that the pilecap was mounted on 16 roller supports and upward motion was only resisted by its mass and the pipework connected to it.

The estimate above of the work done expelling the pilecap is likely to be a lower bound. With the two pressure circuits venting into the space beneath the pilecap through ruptured fuel channels, the volume available to the coolant would approximately double as the pilecap rose through its own thickness (3m). A significant driving force would therefore be maintained on the pilecap as it rose through its own thickness. The gauge pressure required beneath the pilecap to do 202 MJ of work as it rose 3m would be about 0.31 MPa, which is small compared to the initial pressure (6.3 MPa) in the pressure circuit. Furthermore, a gauge pressure of 0.31 MPa would not provide a force sufficient to cause axial rupture of the zirconium alloy pressure tubes. These have a total cross-sectional area of about 1.75 m^2 and taking the ultimate tensile strength of the alloy to be 317 MPa, a force of 5.55×10^8 N would be required to break the tubes. The gauge pressure required beneath the pilecap to raise it and break the pressure tubes would therefore be about 2.7 MPa. Acting on the pilecap as it rose by 3m, this pressure would do 1.7 GJ of work.

A larger pressure beneath the pilecap is also supported by the fact that only 4% of the fuel was ejected from the reactor vault. The fuel assemblies were supported from the pilecap and might have been expected to be lifted out of the core by the pilecap as it rose. Of course, damage to the graphite might have held the fuel assemblies in place or fuel might have melted and become detached from the supports. Nonetheless, they might have remained in the core because the pilecap accelerated so rapidly that the zirconium alloy tubes supporting the fuel assemblies snapped under the stress imposed by the acceleration. The central support tubes in the fuel assemblies were of 15mm diameter, 1.25mm wall thickness. The mass, m_f, of UO_2 in a fuel assembly was about 130 kg. To accelerate the fuel at a rate, $\ddot{x}$, equal to that of the pilecap would require a force of $m_f\ddot{x}$ to be transmitted through the central support tube (neglecting the 'dead-weight' of the fuel). The corresponding axial stress in the support tube would be equal to its ultimate tensile strength at an acceleration of 144 m s^{-2}. The gauge pressure required to accelerate the pilecap at this rate would have been about 1.5 MPa. Thus, the pressure required to cause failure of all of the zirconium alloy pressure tubes and the pilecap to rise would have caused the pilecap to accelerate so rapidly that the tubes supporting the fuel assemblies would have snapped and the fuel would have remained in the core.

These scoping calculations suggest that the work done expelling the pile cap and its associated structures would have been in the approximate range 0.2 - 2 GJ.

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Appendix 4 - Heat-up of the core

A4.1 Heat conduction from the reactor vault

The reactor vault was surrounded by concrete walls 2m thick and had a concrete floor and pilecap. Although the pilecap was blown out in the accident, the top of the core was subsequently covered with dolomite, sand and clay that could have formed a layer 8m thick. To estimate heat conduction from the reactor vault through the concrete and materials covering the core, it is assumed that heat is conducted through a slab of concrete (thermal conductivity 1.4 W m^{-1} C^{-1}) 2m thick with a surface area of about 1200m^2. In a steady state, the heat lost by conduction through the walls and materials covering the core would therefore be about 0.8MW per 1000C temperature difference across the walls. Ten days from shutdown, the decay heating rate in the core would have been about 6MW, so the temperature inside the reactor vault would have had to exceed that outside by 7500C for steady state conduction to remove the decay heat. This shows that conduction heat losses through the walls would have been negligible and they would have been effective thermal insulators.

Heat would also have been lost from the reactor vault by thermal conduction along the pipes leading to and from the core. These were made of stainless steel, which has a higher thermal conductivity (about 20 W m^{-1} C^{-1}) than concrete. The total area for heat conduction along the pipes would have been relatively small however (about 2.6m^2), and the conduction path would have been relatively long (about 10m). The heat lost by conduction along the pipes would therefore only have been about 5 kW per 1000C temperature difference between the core and outside the reactor vault, which is insignificant.

A4.2 Isothermal adiabatic core heat-up

The core at Chernobyl Unit 4 originally consisted of 1700Te of graphite perforated by fuel channels containing UO_2 and a zirconium alloy. Following the explosion in the core, all of the fuel channels were ruptured and the core cooling systems were destroyed. The fuel rod components were already at elevated temperatures immediately following the explosion. Therefore to estimate the time for the whole core to reach 1900C, the maximum temperature measured by the Russians, only the thermal capacity of the graphite will be considered.

The specific heat of graphite varies significantly with temperature. A mean value over the temperature range 300-1900C is about 2×10^3 J kg^{-1} C^{-1}. The heat capacity of the graphite was therefore about 3.4×10^9 J C^{-1} and the heat required to raise its temperature from an initial value of about 350C (following operation at low power) to 1900C would have been 5.3×10^{12}J, ie 1647 full power seconds.

In addition to the reactor core, the reactor vault contained an annular tank containing water that surrounded the core and acted as a biological shield. If this tank were still intact following the explosion, a general temperature rise in the core would have been inhibited until its contents had boiled off. The mass of water in this tank is estimated to be about 7×10^5 kg and the heat required to boil this off would be about 1.6×10^{12}J ie 494 full power seconds. Thus, the total heat required to raise the core temperature to 1900C would have been about 6.9×10^{12}J, ie 2141 full power seconds.

From Ref A4-1, the integrated decay heat between 5 and 30 days after shutdown may be shown to be approximately a power law of the form

$$\text{IRDH} = \alpha t^{\beta} \text{ full power seconds ,} \qquad \text{(A4.1)}$$

where IRDH is the integrated decay heat, t is the time from shutdown in seconds and α and β are constants given by 0.378 and 0.648 respectively. Allowing for 4% of the fuel having been ejected from the core by the explosion, it is found that decay heat would supply the 6.9×10^{12}J required to raise the core temperature to 1900C in 6.6×10^5 s ie about 8 days.

The Russians measured core temperatures of about 1900C on the 4th and 5th of May, which were 8 and 9 days after the start of the accident, in good agreement with this estimate. Of course, the estimate does not account for the twelve hours operation at half power prior to the accident, but the error introduced by this is not expected to be large.

A4.3 The effect of the lead on the heat-up

Lead has a specific heat of 0.031 cal/g/C (at 25C), melts at 328C and has a latent heat of fusion of 5.9 cal/g (Ref A4-3). The heat capacity of the 2400 Te of lead dropped onto the core was therefore about 3.1×10^{8} J C^{-1}, which is an order of magnitude smaller than the heat capacity of the graphite, estimated earlier to be about 3.4×10^{9} J C^{-1}.

If the temperature of the lead were initially about 20C, the heat absorbed raising it to its melting point and melting it would be about 1.6×10^{11} J. Decay heat in the core could have provided this quantity of heat within an hour or two of shutdown and it would be difficult to distinguish the cooling effect of the lead from that of heat diffusion into the graphite at that time. This quantity of heat is small compared to the total energy released by fission product decay within one week of shutdown (6.5×10^{12} J), so melting of the lead would not be expected to affect significantly the long term heat up of the core.

The lead would absorb more heat from the core if it remained in contact with it following melting and did not run away. The heat required to raise the temperature of all the lead to its boiling point (1740C) would be about 6.0×10^{11} J. This is, however, still small compared to the heat required (about 4.9×10^{12} J) to raise the graphite from 300C to the same temperature. Nonetheless, boiling of the lead (latent heat of vapourisation = 42.5 k cal/g mole) would absorb a significant amount of heat (about 2.1×10^{12} J) and boiling of lead might perhaps have contributed to the stabilisation of core temperatures at about 8 days after the accident.

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Appendix 5 - Free Convection of Air through Fuel Channels

A fuel channel is assumed to be open to the atmosphere at both ends and the geometry of the channel and fuel is assumed to be unaltered by any core degradation. To investigate free convection of air through an open channel, the Boussinesq approximation is used, that is density changes are neglected except where they contribute to buoyancy forces. Therefore the buoyancy of air in a channel, due to heat transfer from fuel, is equated with the resistance to flow through the channel of air. The resistance to flow is taken to be solely that arising from the fuel rods and channel wall ie the effects of supports and spacer grids are neglected. Collective effects due to many fuel channels allowing free convection in close proximity are not considered.

Assuming that the air temperature rises linearly with height in a fuel channel, the buoyancy force is given by

$$F_b = \frac{\mu P g \ell A}{R} \left(\frac{1}{\theta_e} + \frac{1}{\theta_a - \theta_e} \ln [\theta_a/\theta_e]\right) \qquad (A5.1)$$

where μ is the effective molecular weight of air, P is the ambient pressure, g is the acceleration due to gravity, ℓ is the height of the channel, A is its flow area, R is the universal gas constant and θ_e and θ_a are the external air temperature and air temperature at the top of the channel respectively.

The head lost to friction in steady turbulent flow is given by (Ref A5-1).

$$h_f = \frac{f \ell}{m} \frac{\bar{u}^2}{2g} \qquad (A5.2)$$

where $\bar{u}$ is the mean flow velocity, m is the hydraulic radius of the channel (given by the flow area divided by the perimeter in contact with the fluid) and f is the friction factor. The force resisting flow in a steady state is therefore

$$F_r = \frac{f\,R\,\ell}{2mA\mu P}\left\{\frac{\dot{Q}}{c_p(\theta_a-\theta_e)}\right\}^2 \left(\frac{\theta_a+\theta_e}{2}\right) , \qquad (A5.3)$$

where $\dot{Q}$ is the rate of heat transfer to air in the channel and c_p is the specific heat of air at constant pressure. Inserting the values for these quantities listed in Table A5.1, it is found that 10 days after the accident, $F_b = F_r$ when θ_a = 1320 K = 1047 C. The maximum fuel temperature clearly must have exceeded this value and burning of the zirconium alloy and graphite in and around fuel channels would have been expected.

Dumping material on top of the core could have extended the "chimney". As both F_b and F_r are proportional to ℓ, the maximum fuel temperature would not have altered if the resistance per unit length to flow through the material had been similar to that in channels and no heat had been lost to the dumped materials. The temperature of air escaping from the top of the dumped materials would, however, have been greater than 1000C. If the materials did absorb heat from the airflow, the temperature of escaping air would be reduced and the buoyancy force would also be reduced. For a steady state, the maximum fuel temperature would then have to exceed the value found above.

References

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TABLE A5.1

μ	=	0.0288
P	=	10^5 Pa
g	=	$9.81\ m\ s^{-2}$
ℓ	=	7.0m
R	=	$8.314\ J\ mole^{-1}\ K^{-1}$
θ_e	=	293 K
m	=	2.13×10^{-3}m
A	=	$2.27 \times 10^{-3}\ m^2$
c_p	=	$1.02 \times 10^3\ J\ kg^{-1}\ K^{-1}$
f	=	0.005
$\dot{Q}$	=	3.7×10^3W (1 channel, 10 days after accident)

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THE USSR POWER REACTOR PROGRAMME AND THE DESCRIPTION OF RBMK REACTOR

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THE USSR POWER REACTOR PROGRAMME AND THE DESCRIPTION OF THE RBMK REACTOR

ABSTRACT

This paper introduces the USSR Power Reactor programme and describes the RBMK reactor involved in the accident at Chernobyl.

USSR POWER REACTOR PROGRAMME

The nuclear power programme in the USSR is provided by two main types of reactors, the (RBMK) pressure tube reactors and the (PWRs) pressurised water reactors.

Pressure tube reactors of the RBMK type have been operating in the USSR since 1954 with a small 5 MW unit, with development to 600 MW in 1958, expanding to units of 1000 MW and 1500 MW capacity. Table 1 shows the large RBMK units now in service and also under construction in the USSR.

The reactors have had a high availability particularly at Unit 4 of the Leningrad and Unit 2 of the Chernobyl Nuclear Power Plants.

The USSR also has a large PWR programme which currently produces 44% of their nuclear power but will by 1991 produce 61%, Tables 2 and 3. Nuclear power is essential to the USSR economy and it presently forms about 15% of their power production with 26 GWs. The USSR have under construction 36 GWs up to 1991 of both RBMK units and PWRs which will give 62 GWs of nuclear power (Fig 1).

Following the accident at Chernobyl 4, the USSR have stated that they will continue with their nuclear power programme. They have proposed some alterations to the RBMK reactors to improve the safety of the plants.

TABLE 1

Large RBMK Units in Service and Under Construction in USSR

Status at 31.12.85	Station	Unit Output MWe(net)	No of Units	Commercial Operation
In Service	Leningrad	950	4	1974-1981
	Kursk	950	3	1976-1983
	Chernobyl	950	4	1978-1984
	Smolensk	950	2	1983-1985
	Ignalina	1450	1	1984-
Under Construction	Kursk	950	1	1986-
	Ignalina	1450	1	1986-
	Chernobyl	950	2	1987-1989
	Smolensk	950	2	1988-1989
	Kostroma	1450	2	1988-1989

TABLE 2

PWR Units in Service in USSR

Status at 31.12.85	Station	Unit Output MWe (net)	No of Units	Commercial Operation
In Service	Novo Veronoezh	265	1	1964-
		338	1	1970-
		410	2	1972-1973
		953	1	1981-
	Kola	440	4	1973/75- 1982/1984
	Armenia	370	2	1976-1980
	Rovno	420	2	1981-1982
	Nikolaiev	953	2	1984-1985
	Kalinin	953	2	1984-1985
	Bala Kovo	953	1	1985-
	Zaporozhe	953	1	1985-

TABLE 3

PWR Units Under Construction in USSR

Status at 31.12.85	Station	Unit Output MWe (net)	No of Units	Commercial Operation
Under Construction	Zaporozhe	953	5	1986-1991
	Khmelnitski	953	4	1986-1990
	Nicolaiev	953	2	1987-1989
	Aktash	953	2	1987-
	Tatar	953	1	1987-
	Volgodonsk	953	4	1987-1990
	Rovno	953	2	1988-1990
	Bashkir	953	2	1988-1989
	Odessa	953	2	1988-1990
	Balakovo	953	2	1989-1990
	Nizhinekamsk	953	1	1989-

DESCRIPTION OF RBMK REACTOR

Introduction

A more complete description of the Chernobyl RBMK reactor can be found in the Vienna papers. This paper gives only a brief description and concentrates on those parts of the reactor which help describe the accident. In addition to the factual description of the RBMK, some comments on the RBMK design features have been made and the Soviets' proposals for improving the RBMK design are given.

The RBMK Reactor

The Chernobyl nuclear power station is 60 miles north of Kiev on the Pripyat River. There are 4 x 1000 MW(e) RBMK reactors on the site and 2 more under construction (Figs 2 and 3). Tables 4, 5 and 6 give reactor and fuel specifications.

The RBMK reactor is a direct cycle boiling water pressure tube, graphite moderated reactor (Fig 4) developed from the USSR's first nuclear power plant commissioned in 1954 at Obninsk. The concept of the reactor design is unique to the USSR.

The reactor core is 12 m diameter x 7 m high built of graphite blocks with vertical channel pressure tubes, 88 mm internal diameter and 4 mm thick, made of Zirconium alloy, passing through the graphite. These tubes contain the fuel and control rods. The graphite structure forming the core and reflector is supported on a welded metal structure. There is concrete shielding 3 m thick above the core and 2 m thick below the core. Water tanks provide the inner radial biological shielding and there is an annulus filled with sand between these water tanks and the outer concrete of the reactor vault. The space immediately round the graphite core is an hermetically sealed steel structure filled with a helium-nitrogen inerting blanket (40% H_2 60% N_2) at atmospheric pressure.

The heat generated in the moderator by the energy of fission (approximately 160 MW - 5% at full power) is transferred to the pressure tubes by radiation and conduction via 'piston ring' type graphite rings, one mounted tightly on the fuel channel and the other fitted tightly in the graphite column (Fig 5). The maximum design temperature in the graphite stack is 700 degrees C, although the normal operating temperature in the graphite is lower.

There are 1661 fuel channels in the core and 211 absorber channels. Each of the 1661 fuel channels contains 2 fuel assemblies held together by a central tie and suspended from the top of the channel tube by a ball-type plug that seals the top of the channel tube. The total length of the assembly is about 10 m.

The fuel assemblies consist of 18-pin clusters, each pin in the form of enriched (2%) uranium dioxide pellets encased in zirconium alloy tubing (13.6 mm outside diameter x 0.825 mm thick). The maximum power from each channel is 3.25 MW (Fig 6).

Primary Circuit

The fuel is cooled by boiling light water at 70 bar pressure. The water enters the channel at 270 degrees C and leaves it as a 2-phase mixture with a mass mean steam quality content of 14%.

The coolant circuit shown in Fig 7 consists of 2 identical parallel loops. Each loop consists of 2 steam drums linked by steam and water lines to the reactor fuel channels. Four main water circulating pumps, 3 for normal use and one on standby, are provided for each loop.

The dry steam from the steam drums passes to one of 2 x 500 MW(e) turbine generators. Condensers are provided and to return the condensate to the steam drums a feed train of 5 electrically driven pumps is used.

Reactor Control and Protection

The control and protection system of the reactor is based on the movement of 211 solid absorber rods. The system ensures: automatic maintenance of a set power level; rapid reduction in power by the automatic control rods and radial controllers; emergency stoppage of the chain reaction by the scram rods; compensation for reactivity fluctuations when the reactor is heated up and brought up to power; and control of the power density distribution through the core.

The channels for the control and shut-down rods and for the in-core flux instrumentation pass through vertical holes in the graphite blocks and are to the same design as the fuel channel tubes. Radial flux monitors are provided in over 100 channels and axial flux profiles are monitored in 12 channels.

The system for reactor control and protection uses 211 absorber rods made up as follows:

(a) 139 manually operated control rods (RR) for radial power shaping.

(b) 24 automatically operated rods for power variations (AR).

(c) 12 local automatic controls (LAR) and 12 Local Emergency Shutdown rods (LAZ) uniformly distributed throughout the core.

(d) 24 shortened absorber rods (USP) for flux profiling, introduced from below the core.

The emergency shut-off rods are motor driven at a speed of insertion of 0.4 m/sec. Full insertion takes 15 to 20 seconds.

The control rod channels are separately cooled by water, not part of the fuel cooling system.

The control system is arranged to operate over 3 power ranges:

1. from sub-critical to 0.1% power.

2. an automatic protection regulation system from 0.1% to 5% power.

3. 2 automatic power controllers from 5% to 100% power (including the Local Automatic Control (LAR) system).

The RBMK reactor has a positive 'void coefficient'. Reduced coolant density results in an increase in neutron population and hence to an increase in power. This is particularly important in the RBMK reactor design at lower power operation when there is a marked positive coefficient. Operating the reactor below 20% power was prohibited.

Containment

The containment of the reactor includes provision for confining radioactive leakage from the fuel through a leaking channel tube into the reactor vault and also leakage from the cooling water system in a loss of cooling accident. Containment is not provided for the steam and water lines, the upper part of the fuel channels and the part of the downcomers, which are located in the steam separator area.

The main reactor containment is achieved by using sealed reinforced concrete volumes lined with steel, together with the suppression pool. The reactor vault is a steel vessel. Penetrations through the containment volumes and doors have special arrangements to allow checking for leak tightness.

The system carries out its functions under conditions of a single failure of any passive component having moving parts.

(a) Leakage of fission products - through the channel tube into the vault

The reactor vault is a steel vessel surrounding the graphite with the channel tubes passing through and welded to the vault top and bottom plates. Water leaking from the channel tubes will flow into the vault and the graphite.

The vault space is normally at atmospheric pressure filled with a helium/nitrogen mixture. The gas space adjacent to each channel tube is continuously monitored for tube leak.

The reactor vault space is protected to ensure that the permissible pressure is not exceeded (1.8 Kg/cm^2abs) in an accident situation involving rupture of one fuel channel. This is achieved by allowing the steam and gas mixture from the space to flow into the steam and gas discharge compartment and subsequently into the pressure suppression pool.

(b) Leakage of fission products from the main cooling water pipes

Confinement is provided for radioactive releases during accidents involving failure of any piping of the reactor cooling circuit, except the steam-water lines, the upper part of the fuel channels and the part of the downcomers which are located in the steam drum separator area.

The system of confinement uses a number of strong reinforced concrete steel-lined boxes surrounding the major cooling water components. Figs 4 and 8 show these main compartments and their connections to the suppression pool. These compartments contain all those reactor circuit elements that may be damaged in accidents for which the system is designed and they are connected via distribution corridors and valves (release and non-return) to the 2-storey, condensation-type pressure suppression system.

Refuelling

The RBMK reactors are designed to be refuelled at full load, approximately 2 fuel assemblies per day. Fig 4 illustrates a refuelling machine at the power station.

Shutdown Heat Removal

In common with all large nuclear reactors, the residual power production in the core when the reactor has been shut down is fairly substantial. For example, after a day it is 0.4% of the nominal power (Nnom), ie 12.8 MW. After 30 days, this falls to 0.12% Nnom and then remains virtually constant for a long time. A cooling system is provided.

Reactor Emergency Cooling System (ECCS)

An emergency core cooling system (ECCS) is provided as a protective safety system designed to draw off the residual heat from the core by feeding an appropriate volume of water into the reactor channels in the event of accidents which damage the main core cooling system. Associated with such accidents are ruptures in the large-diameter main cooling water pipelines, as well as ruptures in the steam pipes and in the feedwater pipelines.

The ECCS comprise 3 independent sub-systems, each of which ensures not less than 50% of the required output. Each sub-system includes a fast-acting part and a part providing prolonged after-cooling. The fast-acting part is in the form of water filled nitrogen pressurised accumulators which operate at a pressure of 10 MPa, and also a pumped water supply for a period of 45-50 secs.

For long-term afterheat removal, 3 ECCS channels comprising 2 groups of pumps are capable of supplying water at approximately 500 t/hr. The water is drawn from the pressure suppression pool where it is cooled by service water.

Comments on the Design

From this discussion of the design and operational features of the RBMK reactor, the following factors contributed to the accident and could have been avoided by more appropriate design:

i Positive Void Coefficient of Reactivity

The reactor was designed such that boiling and steam formation in the coolant channels could lead to an increase in the reactivity of the core. In order to mitigate this problem administrative rules of operation were applied to ensure that

(a) the positive void coefficient was within acceptable bounds,

(b) the reactor shutdown system could add negative reactivity at a sufficient rate to control possible transients.

These rules relating to rod positions were not subject to any design control such as interlocks or reactor trip.

ii Reactor Shutdown System

There was only one system provided for reactor shutdown and this comprised a number of rods being driven into the core at relatively slow speed. The efficiency of the shutdown system in terms of rate of negative reactivity insertion relied upon administrative rules governing the operation of the control rods. In certain circumstances, the shutdown system would provide a very slow negative reactivity insertion.

The positioning of the control rods outside the main coolant pressure boundary rendered the system liable to disruption in case of pressure tube rupture. No fast acting diverse shutdown system existed.

iii Pressure Boundary and Containment

The reactor coolant high pressure boundary is close to the fuel and may thus be directly affected by fuel overheating and explosion. In case of rupture of several pressure tubes the remaining containment was weak and ineffective. Failure of this containment exposes high temperature graphite to air ingress and the possibility of graphite fires.

iv Protection System

Acceptably safe reactor operation required that the operators observe various administrative rules concerning the reactor operation and the reactor protection system. It was possible to breach these rules without causing a reactor trip and no physical interlocks were provided to ensure correct adherence to the rules.

Overall, the reactor design is characterised by the necessity to control an inherently unstable system by the use of administrative rules on reactor operation.

Recommendations for Improving Nuclear Power Safety by the Soviets

Scientific and Technical Aspects

Following the Chernobyl accident the Soviets have reviewed the status of their theoretical and experimental research on nuclear safety and have evaluated the measures for extending, improving and intensifying the work.

Computer programs for analysing the safe behaviour of nuclear power plants are being improved to include transient and accident regimes not anticipated at the design stage. Research into the possibility of building so-called 'intrinsically safe' reactors is being expanded. There will also be an expansion of research on quantitative probabilistic safety analysis.

Organisation and Technical Measures

With regard to organisation and technical measures, the Soviets have compared their existing documents relating to nuclear power plant design and operation, with similar foreign documents; and they claim that this comparison does not reveal any major difference in these documents between the USSR and other countries. In general, they state that their nuclear safety standards currently in force do not require revision. However, more careful verification of the implementation of the standards in practice is necessary. The quality of training and retraining of staff needs to be improved. The design and construction staff must verify more carefully the quality of plant components during manufacture and assembly, and their adjustment during commissioning.

Since the Chernobyl accident organisational measures have been taken to improve the safety of all nuclear power plants in the USSR including the RBMK type. The Soviets intend to pay due regard to the international safety work and to expand the co-operation on safety aimed at preserving the health of the public and protection of the environment.

Priority Measures for Improving the Safety of RBMK Reactors

Alteration to the limit stop switches of the control rods will be made to ensure that all rods are permanently inserted into the core to at least a depth of 1.2 m.

As a temporary measure the number of absorber type control rods constantly present in the core will be increased to 70-80, thereby reducing the void coefficient to a permissible value. Later on, this measure will be replaced by conversion of RBMK reactor fuel to an initial enrichment of 2.4% and insertion of additional absorbers in the core. This will ensure that a positive overshoot of reactivity does not exceed conditions for prompt criticality for any change in coolant density.

The operation of the reactor below 20% power under steady state operation is already prohibited and will, in future, be protected against by a trip barrier.

Reactivity reserve margin and power level will also have trip protection.

A decision on the use of a fast acting shutdown system a using solid, liquid or gas medium, has still to be made.

Reference

1. "The accident at the Chernobyl nuclear power plant and its consequences". Information compiled for the IAEA Experts Meeting, 25-29 August 1986, Vienna, by the USSR State Committee on the Utilization of Atomic Energy.

TABLE 4

General Specification

Thermal power, MW	3200
Electrical power (at generator terminals), MW	1000
Core diameter, m	11.8
Core height, m	7
Lattice pitch, mm	250 x 250
Number of channels in lattice	2044
made up of:	
- fuel channels	1661
- control and safety system channels	211
Number of channels outside lattice	18
made up of:	
- temperature channels	17
- gas sampling channels	1
Constant uranium dioxide charge, t	204
Uranium, enrichment, %	2.0
Maximum design channel power, kW	3250
Mean power of fuel channel, kW	1850
Power of most highly loaded channel, kW	2700
Coolant flow, t/hour	37.5 x (10 to the 3)
Mean bulk steam content	0.15
Saturated steam temperature, deg C	284
Coolant temperature at fuel channel inlet, deg C	270
Saturated steam pressure in drum separators, kgf/cm sq	70
Feedwater temperature, deg C	160
Maximum graphite temperature, deg C	750
Burn-up MWD/kg uranium	18.5

TABLE 4 Continued

Mean channel power rating MW/te	15.4
Peak channel power rating MW/te	22.4
Coolant circuit	Two parallel loops, 4 pumps per loop. Coolant enters the fuel channels from below (supplied by individual feeder pipes) and the steam-water mixture from the top of the channels passes along individual riser pipes to steam drums (2 drums per loop). The coolant pressure at the steam drums is 68.6 bar (994 psi). Feedwater temperature is 160 degree C.
Refuelling	On load, up to 5 channels/ 24 hours.
Turbine generators	2x500MWe capacity each at the generator terminals.
Reactor building	(See Fig 4) the reactor core is in a concrete vault and the main primary circuit components (piping, pumps, steam drums) are in separate cells with concrete biological shielding round them. In the bottom of the reactor building is a 'bubbler pond' (suppression pool) into which steam can be discharged if it cannot be passed to the turbine condenser.

TABLE 5

Characteristics of RBMK-1000 Fuel Sub-Assembly and Fuel Element

Distribution of fuel elements in fuel sub-assembly	2 rows of 6 and 12
Spacer grid	Stainless steel cellular type
Supporting central rod	Zr alloy with 2.5% Nb
Length of fuel element	3644 mm
Weight of uranium dioxide (mean)	3.59 kg
Length of fuel column	3430 mm
Volume of gas collector	17.4 cubic cm
Filler gas	Helium at 1 atm
Fuel element cladding	Zr alloy with 1% Nb in fully annealed condition
External diameter of cladding	13.6 mm
Wall thickness of cladding (min)	0.825 mm
Diametral gap between fuel and cladding	0.18-0.38 mm
Fuel enrichment	2.0%
Fuel density	> or = 10.3 g/cubic cm
Height of fuel pellet	15.0 mm
Diameter of fuel pellet	11.52 mm
Volume of indentation on pellet	3%

TABLE 6

Thermal Parameters of RBMK-1000 Fuel Sub-Assembly and Fuel Element

Maximum power of fuel channel	3250 KW
Coolant pressure - Pump outlet	82.7 kgf/sq cm (79.8 bar)
Pressure in Steam Drum -	70 kgf/sq cm (67.6 bar)
Coolant temperature - at inlet	265 degrees C
- at outlet	284 degrees C
Maximum steam content	20.1 wt.%
Maximum velocity of steam-water mixture	20 m/sec
Rate of flow of coolant through fuel at maximum power	27,950 kg/hour
Maximum thermal flux from surface of element	83 W/square cm
Peak linear thermal power	385 W/cm
Maximum fuel temperature	2100 degrees C
Mean burn-up	18,500 MWD/t uranium
Duration of operation of fuel element at rated power	1190 days


FINLAND
NORWAY
SWEDEN
Leningrad
(RBMK-1000)
4 units
UNION OF SOVIET
SOCIALIST REPUBLICS
Kostroma
(RBMK-1500)
4 units
Ignalina
(RBMK-1500)
4 units
Smolensk
(RBMK-1000)
6 units
Chernobyl
(RBMK-1000)
6 units
Kursk
(RBMK-1000)
(4 units)
UNITED
KINGDOM
DENMARK
NETHERLANDS
BELGIUM
FRG
GDR
POLAND
CZECHOSLOVAKIA
SWITZERLAND
AUSTRIA
HUNGARY
FRANCE
ITALY
ROMANIA
YUGOSLAVIA
SPAIN

Fig.1.

Fig 1

CHERNOBYL UNITS 1 - 4

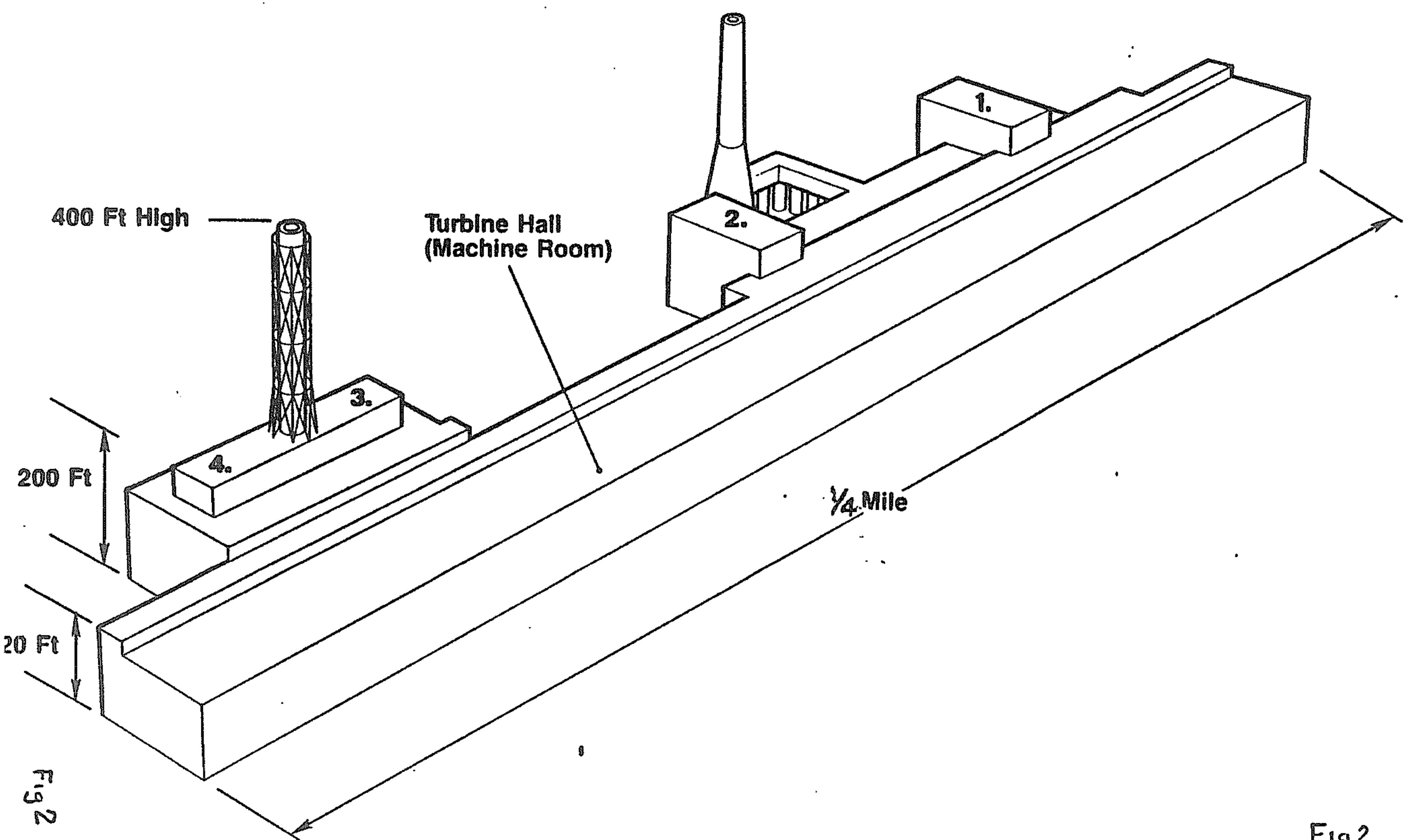


Fig 2

Fig 2

PLAN OF CHERNOBYL 3 AND 4

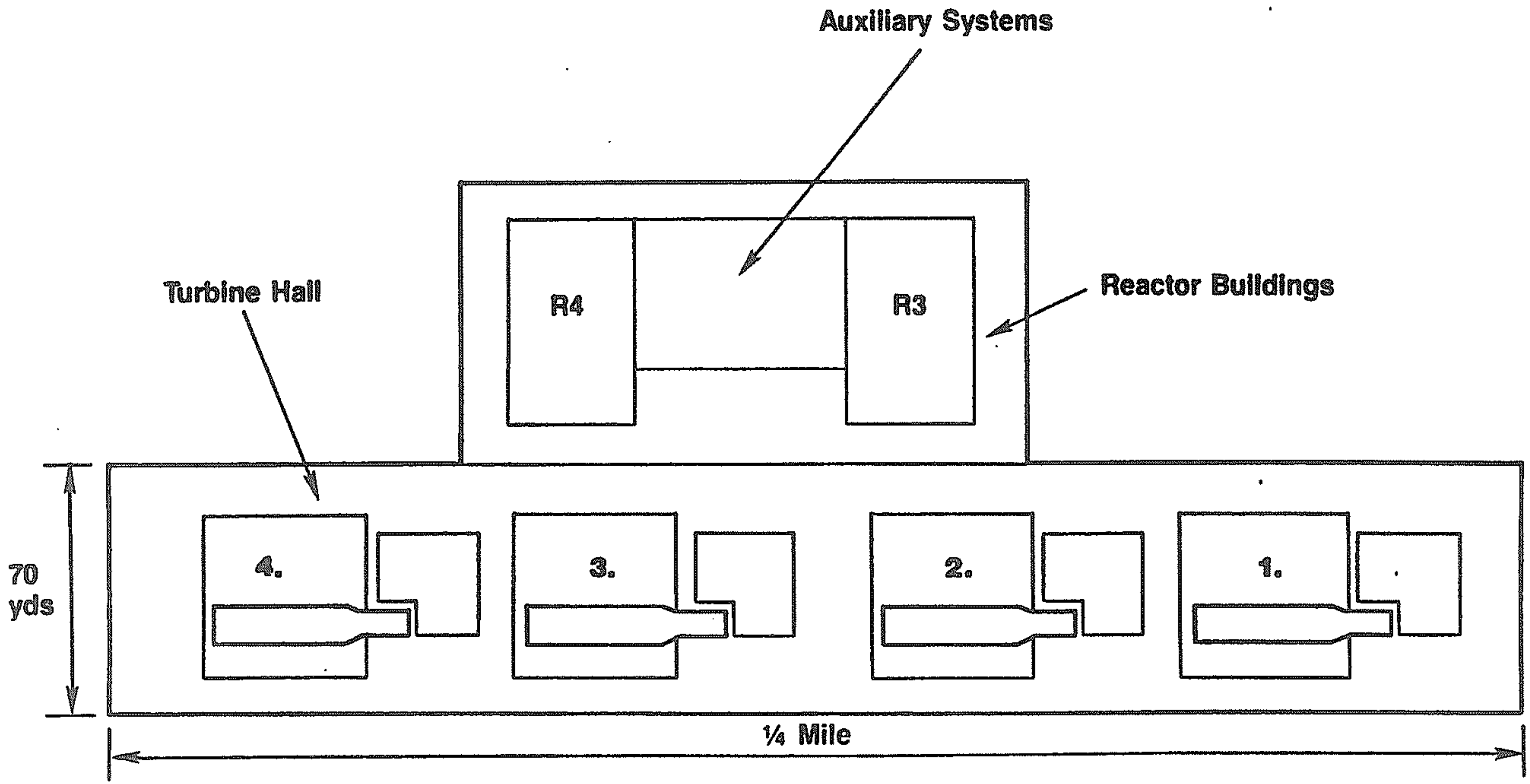


Fig 3

Fig 3

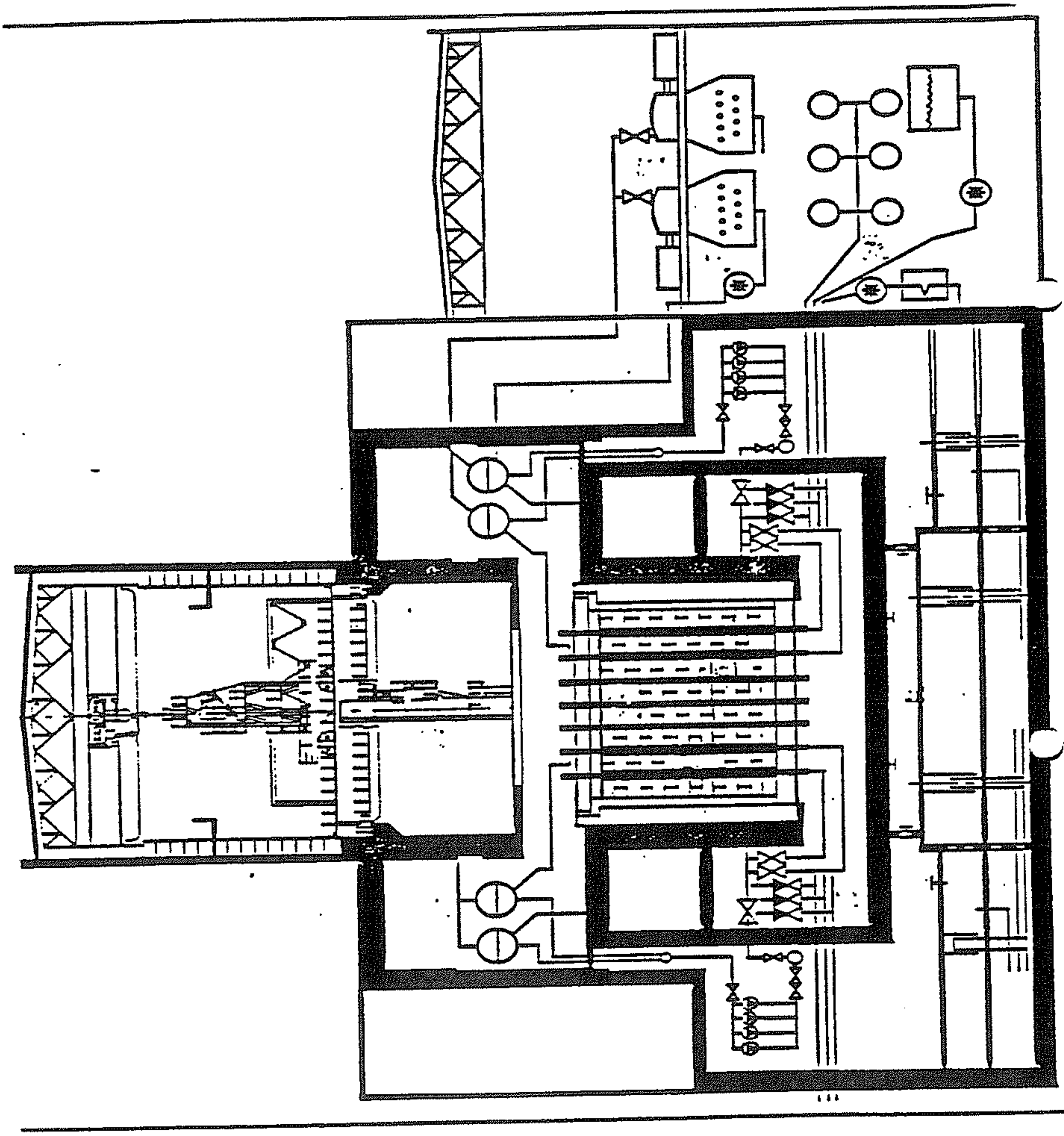


FIG.4

OUTLINE DIAGRAM OF THE RBMK REACTOR

FIG.4

ARRANGEMENT OF PRESSURE TUBES IN REACTOR CORE

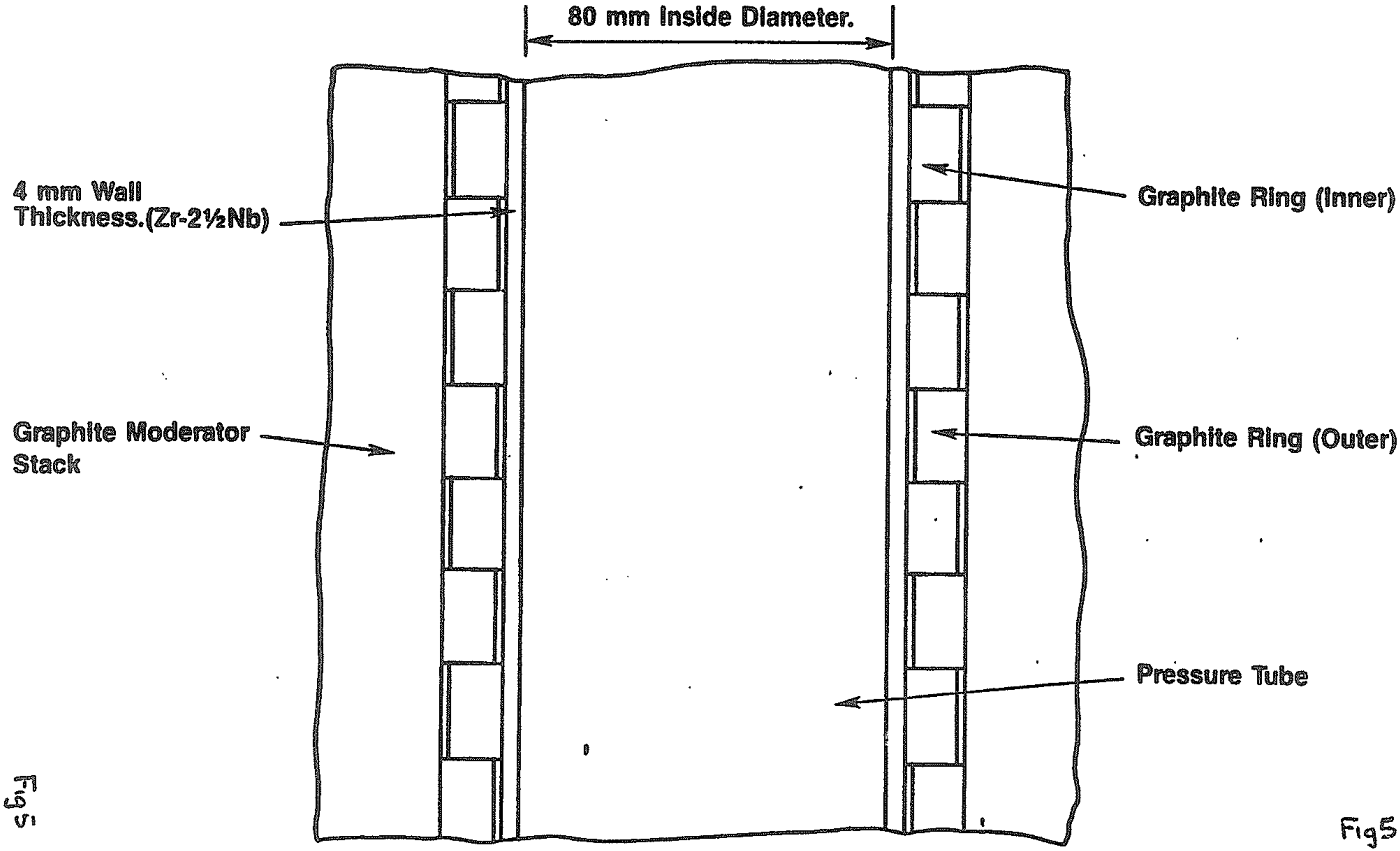
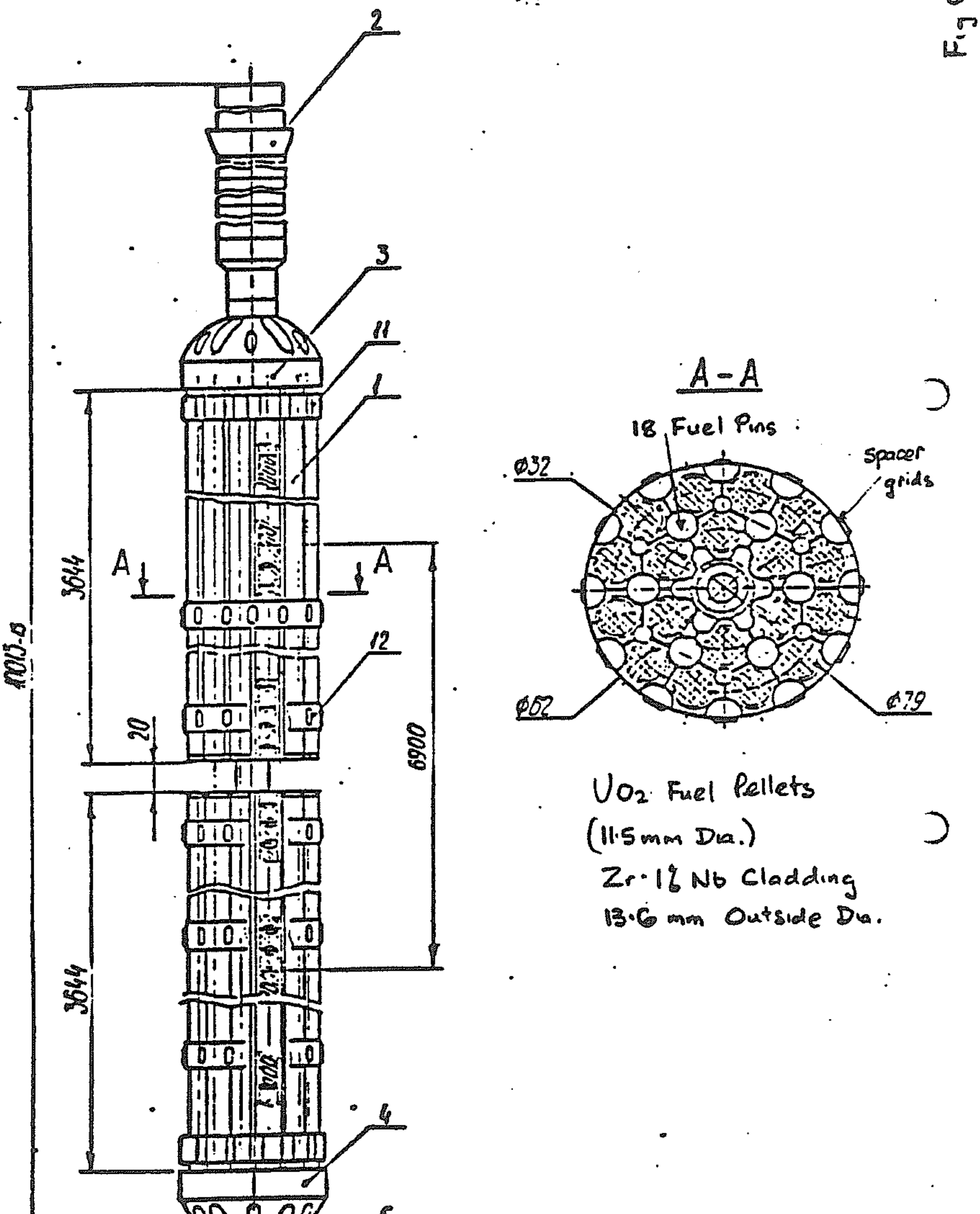


Fig 5

Fig5



FUEL ASSEMBLY.

Fig 6

RBMK-1000 Reactor Normal and Emergency Cooling Systems

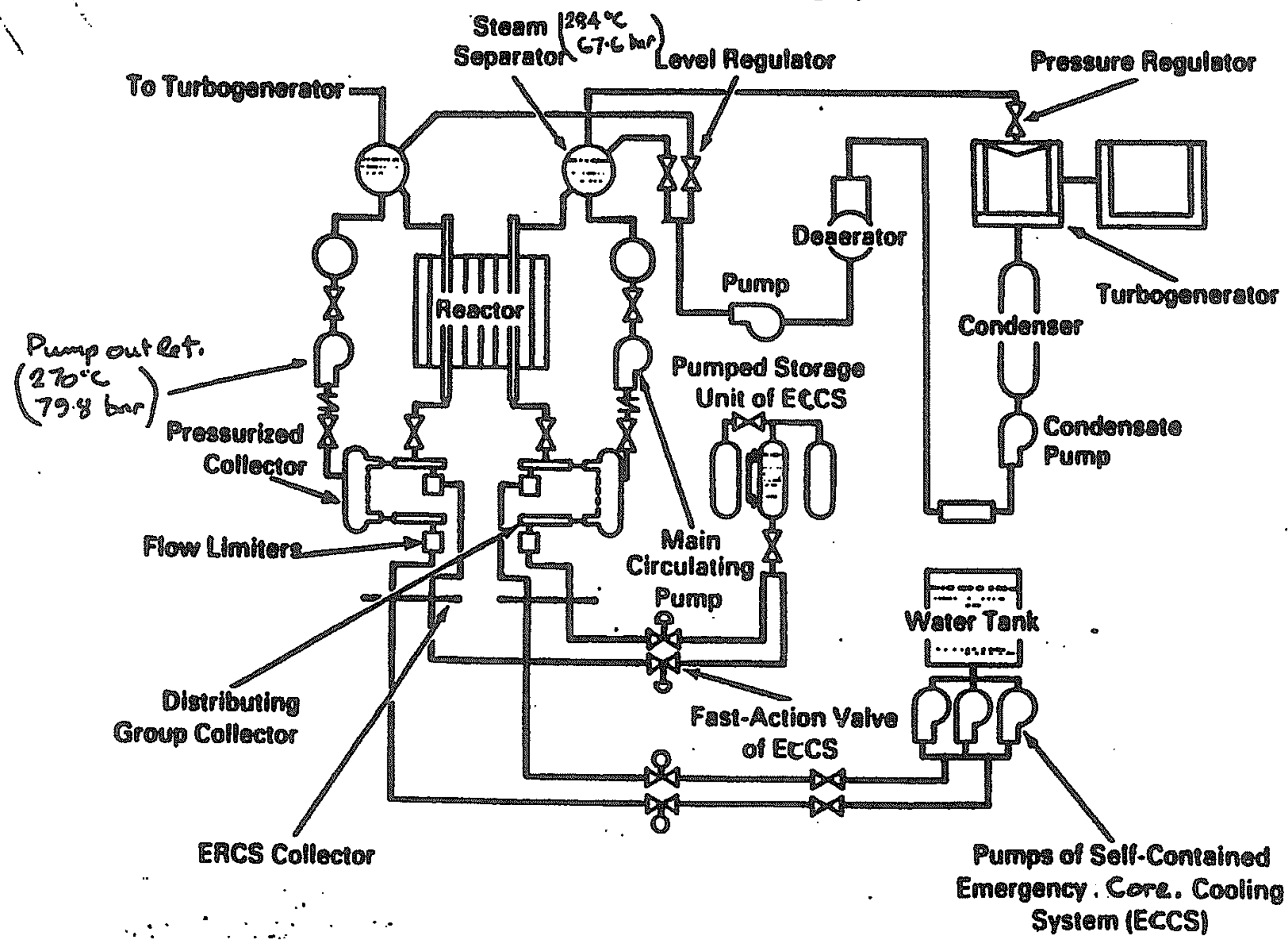
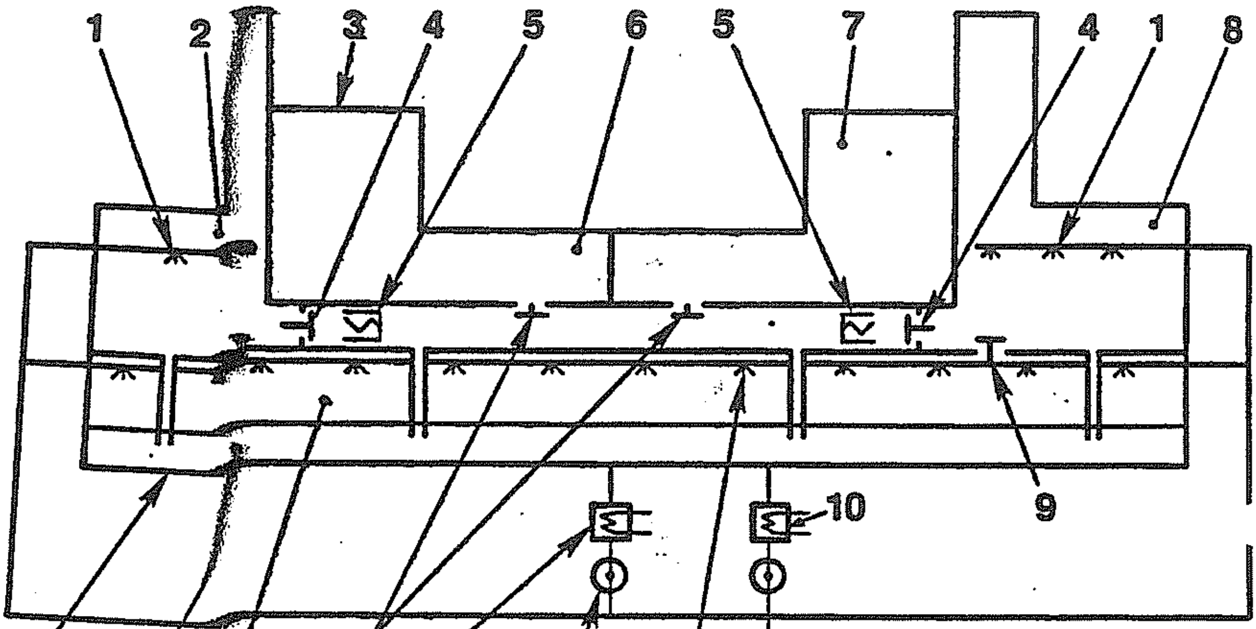


Fig 7

Fig 7

Fig 8



Schematic diagram of the system for retaining and localizing radioactive products in the event of an accident involving an RBMK-1000 type reactor

1, 11	-	Sprinklers
2, 8	-	Left and right hand halves of the hermetically sealed chambers
3, 7	-	Left and right hand halves of the rooms housing the lower water lines
4	-	Valve panels in the partitions separating the chambers and the corridors
5	-	Surface type condensers
6	-	Steam distribution corridor
9	-	Relief valves
10	-	Heat-exchanger
12	-	Pump
13	-	Check valves
14	-	Air space above sparge pond
15	-	Depth to which sparge pond is filled with water
16	-	To emergency reactor cooling system collectors.

Fig 8

6147/20.10 G M Ballard

Note for the Record

Saturday 23 August

Much of the discussion now centres around why the Russians were doing this particular experiment. Everyone seems to be agreed that given the condition of the reactor and the mistakes made by the operators, as indicated in the summary document, the accident was inevitable and we do not believe that the Russians are hiding anything in that part of the explanation. No-one however, is convinced that they understand why they found it necessary to demonstrate that the main feed pumps could be fed with electricity from the turbo alternator as it ran down after the steam supply had been cut off. Discussions with Lord Marshall have revealed several plausible explanations for this and the purpose of this note is to record a range of these prior to the possibility of questioning the Russians, that way it might become clearer. At least this will indicate how well we did in trying to understand the nature of the RMBK reactors and its operating regime. The following are possible scenarios for why they would wish to do this experiment.

1. A simple and appealing reason is that the motivation for the experiment was economic and this was based upon the idea that by using the main feedwater pumps, post-trip that the reactor could be controlled down to a lower power level rather than being tripped all the way down to zero power. The reasons for this would be as follows:

> On a normal loss of all grid supplies the turbo alternator is first of all tripped, ie. disconnected from the grid because no load can be supplied. Immediately the steam supply to the turbo alternator must be bypassed otherwise the turbo alternator would overspeed and destroy itself. The steam which the reactor is producing is bypassed and dumped into the condenser tanks. At the same time the automatic shutdown systems insert the control rods and the shutdown rods and bring the reactor to zero power. Because the reactor protection system is relatively slow acting, ie. it takes about 20 seconds for complete insertion of the shutdown rods then the requirement to dump steam is quite high during this early part of the shutdown. If the main pumps could be kept running then this steam dump could be extended over a long period of time, ie. many tens of seconds and instead of automatically tripping the reactor down to zero power it might be possible to reduce power in a controlled way down to a low power level rather than zero power. The advantages of this would be that instead of shutting down fully, the reactor could be kept at a low power but a power high enough to ensure that the build up of fission product poisons would not inhibit taking the reactor back to power on a short timescale demand. At the moment if the reactor is taken to zero power then rapid build up of short lived fission products, particularly Iodine and the daughter product Xenon, means that any spare reactivity in the core is used up because these fission products act as neutron absorbers and therefore the operators have to wait approximately two days for these fission products to decay away before the reactor

can be brought back to power. If the reactor were held at say, about 7% power, then we believe that an equilibrium will be established between the rate at which these fission products are destroyed by burn out and the rate which they are produced as daughters of the decay chain. With the reactor brought to this hot shutdown state then if the grid were re-established within a few hours the reactor would be in a position to go back on line immediately. Clearly the economic advantages of being able to do that rather than having to wait for a couple of days are enormous and could explain that there was a very strong motive indeed for being able to demonstrate the possibility of this controlled power set back mode.

2. It is not clear from the documents that we have as to whether the rundown of the turbine generating electricity for the pumps is required as part of the safety case for the reactor covering situations of loss of grid. If it is required as part of the safety case then the ramifications for this experiment are quite serious. It would mean that they had been able to demonstrate the validity of an important item in the safety case. It is not clear whether this experiment was being done to demonstrate that the equipment on this particular plant, ie. Chernobyl Unit 4 was operating as required or whether it was something that was generic to all RBMK reactors. The document says that changes had been made to the voltage regulator on the turbo alternator to prevent the voltage from falling too rapidly during the run down of the generator. If this were a local problem, ie. that the voltage regulator on Unit 4 was in some way faulty, and had to be repaired and then after the repair they had to do this test to demonstrate that it worked properly in order to satisfy the safety regulations then this would be a slightly less serious case with ramifications for other RBMK reactors only if they discover that their voltage regulators were faulty too. I personally do not subscribe to that because the document does indicate that they had tripped Turbine 7 prior to the accident and if the test had failed using Turbine 8 they would have tried again with Turbine 7. This would indicate that the voltage regulators on both of those turbines must have been altered to allow this change in the way the power run down was managed.

3. Another possibility is that the normal situation when a loss of grid trip occurs is the following:

When the alternator trips out then a demand is placed upon the back up diesel generator sets to provide electricity to run for example the pumps on the emergency core cooling circuits and the decay heat removal circuits. However, a demand to start these diesels might go out at the moment the alternators trip but it is well known that it takes some time before these diesels are up to speed and able to generate full power. It is thought that times of the order of several minutes might be involved between the signal to start and the ability to put power onto the house load buses. What happens in the meantime? If the turbine is tripped and is no longer providing house load, then the main pumps will run down under their own inertia, the flow through the core will slow down, at the same time the rods are only being inserted relatively slowly into the core so would expect a build up of pressure in the circuit even though steam may be dumped into the condensate tank as the means of removing energy from the system. Under these conditions it would be very likely indeed that the safety release valves would be required to lift to regulate the pressure in the circuit as a matter of course during such events. We know for example, that safety relief valves in this case, PORVs, do lift in BMW pressurised water reactors as a normal response to loss of grid type trips. This was certainly the case in the Three Mile Island reactor and indeed it was failure of the PORV to re-seat that was the cause of that accident. By analogy we would expect the Russians to have discovered that safety relief valves are very unreliable when it comes to re-seating after opening. They could, therefore, have been experiencing some difficulty in regulating these valves in this kind of trip situation. This would be exacerbated by the fact that we believe the grid is very dicy in that part of Russia and therefore power cuts would come rather often. If the valves did not re-seat properly then they would have to use the feedwater supply to the steam drums to make up the circuit to prevent a loss of coolant accident. This would not be a very satisfactory situation if it happened regularly and there would be considerable motivation to see if by running the pumps down, at the same time that the turbine runs down, could provide sufficient flow through the core to prevent this pressure build up and thereby prevent safetys lifting. This could be a conceivable motivation for doing these tests. It would of course apply to all RBMK reactors.

4. Another possibility is that the safety case made for protection during loss of trip transients is made only for trip from full power. Since these reactors are primarily base load plant, this would be the normal situation. However, it is possible that in some circumstances it would be required to trip the reactor on loss of grid from lower power levels. Because of the characteristics of the boiling water direct cycle reactor, tripping from low power is actually more difficult that tripping from high power because the thermodynamic properties of the circuit are such that it is closer to saturation when tripped from lower power than from high power. It is possible therefore that they were trying to establish that for these set of trip situations use of the main circulation pumps would be valuable in maintaining adequate cooling margins.

The discussions centred around these 4 main guesses at what could have motivated the Russians to do tests like this on a reactor of the RBMK type. It is still considered virtually incredible by everybody that the operating team could have deliberately overridden so many safety trips and circuits and got the reactor into such a dangerous state before operating the test. It is for those reasons that we believe the motivation for doing the test must have been very strong. We cannot discount of course, that the motivation for the operators was quite different from the motivation seen globally by the Central Moscow people and that all they wanted to do was to get this test over with so that they would not be told off next morning that they had failed to do what they were supposed to do. It is quite likely that it was nothing more esoteric than that.

Proposals for an International Effort to Design and Build an Inherently Safe Reactor

In the Russian document one of the indications they give for where they will put their effort in the future is into the development of reactor systems embodying more inherent safety features. We, of course are doing our own review of this area, both within the HSSC context and for the fast reactor with the job that I am doing on the IFR. However, it does seem that an international effort along the lines adopted for the fusion programme to develop an inherently safe reactor could be of great benefit. Not only would it be beneficial from the technical point of view, but for many other socio-political reasons as well. Therefore, after discussing this with John we have noted down a paper which we will present to Morphet with the idea that Peter Walker might consider including this in his speech to the Special Board Meeting later in the month of September. To start the ball rolling the following is an attempt at a draft outlining why such an international effort to produce a demonstration inherently safe reactor could be of benefit.

Prospects for International Collaboration on Reactor Design in the Post-Chernobyl Environment

It is undeniable that the accident at the Chernobyl plant has changed, possibly for ever, the appreciation and perception of nuclear power, both for the public and for the technical community. It is equally undeniable that it has not changed in any way the needs for a secure and environmentally acceptable means of generating electricity if the further development of all the world's peoples is to be assured. If nuclear power is to continue to offer a credible form for energy needs then the lessons from Chernobyl will not only have to be learned by the technical community but be seen to have been learned by the public and by decision makers. In the short term this will place considerable burdens on the technical community in re-appraising the safety of their own plant and underpining any developments with appropriate research. Because of the very large investment in existing plant and the strategic needs of some countries (especially France and Japan) a rundown or even a complete cessation of nuclear electricity generation is not seen as a credible alternative. Chernobyl has demonstrated vividly that these are not parochial matters - what does it matter if we abandon nuclear power if the French decide to expand operations and where for example, Gravelines is closer to London than the proposed Sizewell 'B' PWR? International activities are already planned in this shorter term time frame - they have been commented on separately*. This paper examines the need and justification for longer term programmes of work with the aim of providing optimal reactors including inherent safety features on an international basis. The analogue of the long term development of fusion power in this context is useful.

The Proposition

The international community including all political divisions, ie.

East/West and North/South pool technical expertise to plan, design and construct a nuclear power reactor which embodies optimal use of inherently safety features. Such a design will be intended for commercial use in all countries with minimum adaptation for specific national practices.

Advantages of Such an Approach

1. The design would benefit from the widest possible expertise as input.

2. Individual nations would be assured that nuclear plant being constructed in their neighbouring countries (or even rather distant neighbours) are based on a design of agreed and acceptable safety).

3. Development costs would be shared but the benefits would accrue to all.

4. Nuclear expertise world-wide would be to a greater extent "transferable", thus assuring back up in technology for emerging as well as developed nations.

5. Commitment to an international design would have a unifying effect.

6. Requirements for fuel supply could then be as diverse as possible.

Disadvantages

1. International projects are notoriously difficult to organise.

2. National pride must be overcome.

3. If, there were to be a design fault and it was not discovered until many plants had been built, this would be the Grand-daddy of all common mode failures!

The Way Foreward

1. Current knowledge and understanding of reactor design with special emphasis on inherent safety be brought up to date with an international review group.

2. Internationally agreed safety goals be approved against which any new design would be judged. The implementation and the realisation of these generic goals into design rules or codes of practice to be for national bodies - but subject to international peer review in the first instance.

3. Identification of research and development needs to support the design (or maybe a few alternative designs) chosen. Internationalisation of the R&D support to include a range of countries contributing: this should not be a club for the developed countries alone.

4. Prototype construction in a chose country. I believe there

would be considerable political advantages for choosing an Eastern Block country for the construction of the first prototype. Not least amongst these reasons is the fact that the problem over hard currency could be overcome if the contribution from the Eastern Block was in kind on their own territory. in concrete and steel.

5. Development of an internationalised architect engineering capability to assist in construction including quality assurance of plant in all countries. This could be a development of existing arrangements within the IAEA.

Timescale

Any new reactor system requires time to develop. However, because so much experience exists in reactor physics materials, design, etc, a full cycle starting from a clean sheet should not be necessary. It is envisaged that a prototype reactor could be built and demonstrated so that the new design would be available for commercial operation by the end of the century.

* Eg, briefings have been given on CEC, NEA and IAEA activities to the Department of Energy. Further detailed commentary will be provided to cover such matters as the need for various retrofitting options, eg. filtered vented containments, provision of transients for operating staff, emergency planning, etc.

Sunday 24 August

Further briefing meetings were held with Lord Marshall on Sunday morning with the intention of first of all formulating the 3 questions which we believe are absolutely crucial to understanding the causes of this accident. It also touched briefly upon the policy line which Lord Marshall envisaged when dealing with the media. Overnight Brian Edmondson and John Appleby had arrived bringing with them the first copy of the acredited IAEA translation of the Russian documents. There was much discussion concerning the nature of the controllability of this reactor. Many rather technical points were raised and essentially they included the following:

1. The uncontrollability increases as burn up increases because the positive void co-efficient increases. The reason that the positive void coefficient increases is that as the Uranium 235 is transmuted into Plutonium then when the water moderator is removed the neutrons can "see" the Plutonium better than before and hence increase the reactivity of the core.

2. When operated at low power there is a higher positive void coefficient and the reason for this is that removal of moderator exposes neutrons to more absorber and therefore less reflection from the moderator back into the fissionable material. Furthermore, at low power the margin to saturation is much lower than at high power.

3. At low power still the hydraulic state of the reactor is described as floppy. This means that the flow through the core being reduced over that at full power means that relatively small changes in steam quality or voidage can cause large sections of the core to change rapidly the moderating capabilities due to changes in the amount of water/steam in that part of the core. It is for that reason that we believe it would be better to keep the main circulation pumps running off the turbines as long as possible to keep as far away from this unstable hydrodynamic state as possible by keeping the core flow rate up.

4. The Doppler effect is lower at low power because the fuel is at lower temperature and therefore would have to heat up considerably before the negative reactivity insertion due to increased temperature came into effect.

5. The high flow rate involved in the conditions prior to the accident reduces the temperature gradients around the circuit to the point where the circuit has been operated more like a pressurised water reactor than a boiling water reactor. Thus, all of the circuit is brought close to its boiling point at the same time.

6. Even though the neutron power of the reactor may be reduced the channel power stays higher because heat from the graphite can be returned into the fuel channels.

Diagrams in the full Russian document, ie. 2.6.3/4/5 indicate values for power and flow parameters during this kind of situation. In all of these situations which require an AZ5 type

RPS response, involve the steam pressure in the separator (steam drum) to rise well over its normal value but what we do not know is whether it rises high enough to trip the safety relief valves or not. In the text (p80) it does indicate that steam relief valves would lift and there is a possibility that they would not reseat. Thus supporting one of the contentions that we had put to Lord Marshall yesterday as to the motiviation for wanting to do this kind of test.

The questions which have now been agreed upon boil down to the following:

1. Just why did they do the experiment?

   a. Was it to justify part of the safety case?
   b. Was it to improve plant availability, eg. to demonstrate that keeping the main circulation pumps operating would avoid safety valves lifting?
   c. Was it to indicate that a safety case which had been developed for trip from full power would also work if the reactor were tripped under the same external circumstances but at low power and/or with a high burn up core?

2. We understand it is difficult for operators to maintain reactivity margins for RBMK reactors under all conditions. Did they regularly need to override the reactivity margins as indicated on their computer output as an advisory warning to shut down? The reason behind this question is to try and establish whether the operators had become so familiar they thought with operating the reactor via movements of any of the sets of control rods available to them that they had become used to ignoring advice which did not automatically trip the reactor but warned them that their reactivity margin was dangerously low. This draws upon the comment in the paper that the operators had "lost their sense of danger".

3. If the safety case for this test was poor as admitted in the document, what was the case for the previous tests like? The corollary to that question did the previous accident at the Kursk plant in 1980 indicate that the test was going beyond or was likely to go beyond the design safety intent for the reactor?

Concerning relations with the media, Walter made it very clear that everyone in the UK delegation was charged with doing their bit to talk to the press when asked about policy and briefing his comments essentially were: if you guys didn't know how to deal with the press then you would not have been chosen to be here. Which seemed on the one hand a great show of confidence but on the other slightly worrying. Several aspects were raised.

1. The relationship between what we have now learned and what was stated both on 13 May in the CEGB's document to the Secretary of State and previously in the NNC assessment of the RBMK reactors in 1977. What we want to do is to go through the cause and course of the accident and highlight where our previous assessment was correct and why the arguments raised then as to why it could not happen here are either still valid or indeed reconfirmed by what was said in the previous assessments.

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1. The relationship between what we have now learned and what was stated both on 13 May in the CEGB's document to the Secretary of State and previously in the NNC assessment of the RBMK reactors in 1977. What we want to do is to go through the cause and course of the accident and highlight where our previous assessment was correct and why the arguments raised then as to why it could not happen here are either still valid or indeed reconfirmed by what was said in the previous assessments. This should include all

those items which were not anticipated previously and to be able to assess the impact of that neglect on the previous conclusions and recommendations. In particular it will be required to show that the fact that no account had been made of the possibility that a coherent team of operators under a supposed team leader could make so many catastrophic errors and override so many established safety procedures. The important thing though is to identify the phenomena and design aspects of the plant which were important and identify those with what was said previously. The idea being to demonstrate that we had pretty well worked it out previously and all we are learning here in Vienna are the details. Whether that will be borne out I do not know, personally I find that the understanding of what could happen with RBMKs changes and improves every time we sit down and discuss it.

The one question which we believed we would have great difficulty in answering from the media was, are Russian reactors really safe? This sort of question really is a catch 22 because we cannot answer for Russia, they are not operated on our soil and yet on the other hand we cannot say, no, they are not safe for obvious reasons. Walter's view was that we should argue that whilst the RBMK reactors would not be licensable in this country because of their characteristics, what they do is to present a greater challenge to the operator than do those in the West. The question then of course is, are Russian operators so well trained that they can handle these rather difficult beasts adequately? In this case the answer is clearly, no they are not, and the Russians recognise the need to modify both the reactor for ease of handling, and the training and management provided to and by the operators.

Notes and Comments on the Plenary Session, Monday 25 August

In the main the proceedings today involved Legasov presenting the summary document as an extended lecture. Apart from the Chairman of the session Romech, he was the only speaker and it is generally agreed that he did a very good job indeed in presenting a clear and apparently frank resume of that summary document.

The opening speeches were quite good. They contained much of the sentiments that we would have expected to have heard, including the important role of the IAEA in all these matters, the requirement for nuclear power based upon moral and ethical grounds concerning the fact that three quarters of the world's population are currently under developed and short of food, and therefore concluded that there will be a pressing need for a long term nuclear programme. He did not of course address how we get over the short term difficulty of public acceptability following this accident but I think that was understood by everybody there. He indicated that there was a need for a redoubling of the amount of energy required in the world and the only viable source was nuclear coupled with full and appropriate utilisation of natural resources. He was rather explicit in saying that our present ways of producing electricity by burning coal and oil were killing our forests. Fortunately I was not stood anywhere near Lord Marshall at this point and could not see his face.

The aims of the meeting were described briefly as:

a. To improve nuclear safety
b. To improve our understanding of the physics and chemistry, better to be able to utilise nuclear power safely.
c. To improve international collaboration in the areas of safety.

We were told that the report will be made available to the public after the Board of Governors has seen it. This refers of course to the report which will be prepared by the Secretariat and their experts following the meeting, not the Russian report which was made available for the meeting. That report now seems to have been given to anybody who wanted it, including the media.

The head of the Russian delegation, Legisov gave an introductory statement concerning the nature of the accident, how dreadful this was for Russia and the people involved and how it highlighted the fact that economic development within Russia was dependent upon nuclear power and therefore they had to make sure that what they were doing in the future was safe. Quite a bit of information was given exposing Russia's economic difficulties in utilisation of their own natural resources. For example, it was stated that coal has to be hauled 4,000 km to burn in power stations in Eastern Russia where the power is needed and that this meant that some 40% of all Russia's railway transport system was devoted solely to the movement of coal. This was all done to give an idea of the importance that Russia places on the development of nuclear power and how in fact this accident is not going to stop their programme although they accept it will cause them economic difficulties and a slowing down for some time.

The frank nature of Legaslov's lectures soon became apparent. The first thing that he considered; the reasons why the RBMK reactors were chosen for the generation of electricity in the USSR. He said that the original decisions were made on the basis of the fact that even though they were well aware that this design has serious limitations concerning margins of safety, they really had no other alternative because at that time, and this is the best part of 30 years ago, they had no capability of manufacturing large pressure vessels such as would be needed to implement a pressurised water reactor programme. Furthermore, considerable expertise was available in operating the RBMK generic design because it was used to produce Plutonium for the Russian weapons programme very early on. He then listed the disadvantages of the design which were known when the decisions were made to utilise this as a major source of electricity generation.

1. Positive void coefficient. This was described as being due to the phase shift in the coolant which we would say more simply is the fact that if the coolant boiled then the reactivity went up. This was recognised as a problem from the very earliest times.

2. In this design it was recognised that there was a very high sensitivity of the neutron field to very small changes in reactivity. This would make overall control of the reactor very

difficult.

3. There was a large amount of potential chemical activity tied up in the fuel and the cladding and the graphite blocks which made up the moderator. This was a potential source of unwanted chemical reactions of an adverse kind.

4. Using the direct cycle boiling water reactor principle meant that the turbine was driven with slightly radioactive steam with all the consequences concerning operator dose that that would entail.

Because of all these disadvantages, great responsibility was placed upon the setting of what was called the technological norms of the system. What he meant by this was that setting the normal operational parameters of the reactor had to be done with great care to ensure that these unstable features of the design were kept well away from the operating regimes. He then went on with a description of the parameters of the reactor which included a list of these technological norms. These included things like the pressure in the channels, steam quality, numbers of control rods, types of control rods, and the like. All this is contained in the document and there is no point in trying to reproduce it here.

One series of points that came out very clearly here was that they knew that this reactor was going to be very difficult indeed to control and that it would place great responsibility on the operators to drive it properly. One indication of this is a provision of automatic power set back situations so that instead of having to drive the reactor down manually to certain power levels this was all done by pre-arranged computer operations. In addition to this in order to try and improve the availability to the reactor only certain combinations of plant failures were led the reactor trip down to zero power. There was a whole series of automatically pre-set reductions in power which were determined by the particular pieces of equipment which were available to control the plant. By that I mean that if, for example, certain pumps failed then instead of tripping the reactor down totally they would only trip that part of the reactor out which the pump was involved with so they might lose a whole half of the reactor and hence one turbine but the other half would be kept in operation. This made the control system exceedingly complicated indeed.

One very important point here was that he said a decision was made very early on that when the reactors were designed, they would place more confidence in the human operators than in automated control systems. Considering what we were told about the cause of the accident later, this sounded like a rather poor decision. However, he explained that this was done at a time when automated control systems were not well developed and certainly were not considered to be very reliable. Therefore, it was apparent that that was a reasonable decision to be taken at that time.

A great deal was made of the fact that under these technological norms there was a requirements that the operative reactive margin should never be less than 30 effective rods. In any circumstances where the reactivity margins were reduced below that then only the Chief Engineer for the site can give permission for operation below 30 rods. If the margin goes below 15 rods then no-one and in his words "not even the Prime Minister" can approve operation of the reactor. He made this point very strongly which is not surprising since later on it is found that the operators were actually running the reactor with no more than about 6-8 effective rods reactivity spare - crucial point in the development of the accident. As an aside I should say that over lunch the point concerning the meaning of 30 effective rods in this context was discussed and no satisfactory answer was found. There seems to be a combination of 2 quantities here which are mixed up in this global parameter called 30 effective rods. First of all it does not mean 30 control rods must be out of the reactor and available to be put in. What it does mean is that there must be available, 30 effective rods worth of reactivity available at any time during operation. This can be totalled up by taking into account reactivity worths of partially inserted rods were they to be inserted further. However, it is one thing to have an overall reactivity worth sum done, the other part of it concerns the rate at which rods can be put in. Much was made of the fact that the rods must go in fast enough at least to be able to accommodate one beta per second of reactivity. In that case as argued that the positive void co-efficient could always be overcome by the control rods. It is interesting to note that later on we are told that one of the fixes for the safe operation of the reactor was to increase the reactivity margin requirements from 30 to 80 rods. They argued that this would then be able to protect them against any possible voidage in the core. That means for example that prior to this the 30 rod criterion was insufficient to be able to control all possible positive void co-efficient scenarios.

At this point Alegasov had an aside from his written notes which seemed to me to be quite important. He posed the question "Why was such a standard provided for within the rules and not designed out at that time"? What is meant by that is if you have a reactor with these unstable characteristics there are two things that you can do: you can either make it obligatory that the operators always keep the reactor in a condition so that it is very far away from getting into trouble or you can actually make design changes which will ensure that it is physically impossible for the reactor to get into those situations. In this case a decision was made that they would rely upon the operator to keep the reactor well away from these difficult situations. This of course is coupled with the requirements on reactivity margins mentioned above. It is also where he suggested that in those days operators were considered more reliable than automated systems. However, what he did not say was why they had dismissed the design decision available to them at that time to go for a higher enrichment in the fuel and more absorbers in the core. The result of that design decision would have been to make the reactors rather less economic, ie. more expensive in their use of fuel but a lot safer. My own suspicion is that this is a sign of the Russian mentality coming through and their drive to try and squeeze the last drop out of anything regardless of the safety of their population. I really believe that that attitude has now changed but it was probably prevalent 30 years ago when these decisions were originally made.

The provision of ECCS is one of those changes to the safeguards equipment provided on these plants. In addition, the bubbling pond or as we would call it, the pressure suppression pool, is also a new innovation. It is interesting that Alegasov indicated that the location of this pool on the Chernobyl Units 3 and 4 was unusual in that it was placed beneath the reactor. He indicated that there had been some debate about the location of this pond there because of worries about the generation of Hydrogen either by radiolysis or by oxidation of Zirconium which could then find its way to this area, ignite and present a hazard to the reactor structure above it. At this point he indicated that it was his personal view that this location of the suppression pool had indeed been very advantageous concerning the outcome of this particular accident. This was to do with such things as the use of this volume for concrete in-fill, and as a volume which provided access to reactor compartments which had not been damaged by the accident.

He then addressed the issue of containment. The most surprising thing here was not that he addressed this issue but the way in which he did it. That is his opening remarks acknowledged the debate going on in the West concerning containment capability and performance and indeed the arguments going on in the West as to whether the RBMK reactor did or did not possess a containment in the strict sense of the word. He then went on to describe the actual reactor systems indicating that high pressure steam circuits were all enclosed in special cells with a limit of pressure on them of 0.54 MPa (about 72 psi). That particular strength is larger than that of a PWR containment building but of course these cells only enclose rather small parts of the pressure circuit and he admitted that the core itself was so large that it was impossible to provide a civil engineering structure to surround it which would have a pressure capability that would mean it could be likened to the containment on pressurised water reactors. He said that the design basis for these containment structures was for pressurisation from within but not to be able to withstand detonation. Nevertheless, it was informative to note that these compartments did survive the detonation in the main even though they had only been designed to withstand the less rapid pressurisation due to, for example, pipe breaks.

The explanations of the accident and its background then continued but at this point following the text became rather more difficult as Alegasov included all sorts of odd comments on aspects of the reactor accident. He indicated at this point that the fire fighting efforts had been extremely effective and that flammable materials such as Hydrogen and Oxygen cylinders and various oil storage tanks close to the reactor had not in themselves, burned. Indeed he said that to this day there were still Hydrogen cylinders in the turbine hall which had been uneffected.

There followed an odd discussion in which he attempted to describe the justification for the experiments. He said that the design was such that mechanical energy could be used to maintain the house load. In addition, this could assist the working of the reactor in emergency conditions. If this then could be demonstrated to be the case then the house load could be maintained until the diesels could take over. Calculations along these lines were incorporated into the design. However, previous experiments on RBMK reactors in 1982 and 1984 showed that the voltage on the house buses was not maintained during turbo alternator rundown. Therefore it was decided that it was needed to improve the exciter on the generator to maintain appropriate voltage to run the house load. I did not understand whether this as a statement concerning the Chernobyl Unit 4 plant alone or whether it referred to all RBMK reactors. This is one of the points that might come out in the questioning in the working groups either tomorrow or Wednesday. In order to understand the pressures felt by the operating crew he indicated that the work on the fourth unit was such that with it being shut down for scheduled maintenance, these tests would have to be carried out during that shutdown process or they would have to wait for another 12 months for the next scheduled shutdown before they could do the test again. This statement leaves more questions unanswered than it asked. For example, if it was so important to run these tests, that they could not possibly wait for a year to do it again, then just why were they running the reactor in a situation where either the safety case had not been established and these tests were being done to prove the safety case with their new exciter, or the tests were considered so important in increasing the availability of the plant that they had to be done even though it clearly endangered the plant, yet, of course, they were not that important that the reactor could be taken off line and the test done later rather than waiting a whole year for the next scheduled shutdown. I still do not understand this point and have not been satisfied either by what the Russians have said or indeed the discussions with the experts in Walter Marshall's briefing group. That is a purely personal view at this time.

Further revelations included the fact that these tests were actually performed by experts in the electrical circuitry, it was their job to look after the turbo alternator and they knew nothing at all about reactors. The question arose as to why these tests were not discussed with a resident Reactor Physicist, the Local Safety Committee or indeed the Station Manager. The answers seem to be that nobody involved with the tests thought that they would interfere with the reactor. Indeed, the overall Site Manager appeared to agree with this logic. It is interesting as an aside to note that Mr Potter, when interviewed against Brian Edmondson on Newsnight when the Russian documentary was viewed, also expressed the view that he could not understand how these experiments could interfere with the reactor. It just shows you that even a supposed engineer, unless he knows about reactor systems in detail, should not be trusted with decisions about a combination of alternators and reactors when direct cycle nuclear power is being discussed.

An interesting aside was that we were told that the reactor designers, whoever they were, were asked their opinion on this test but they just did not get around to providing an answer in time before the tests were performed. One can imagine the bureaucracy is such that the design office could not be bothered to answer a request from some parochial site about some two-bit test that they were trying out and just did not get round to answering it even though presumably they were aware of the maintenance schedule for that reactor. However, even this poorly compiled programme would have been safe if the series of mistakes and overt overriding of safety systems had not been made by the operating team. Clearly that team was neither competent nor fully prepared for the test and this was admitted by the Russians.

One strange aside here was that the emergency core cooling system was cut off, made inoperative when turbine number 7 was taken out of service the previous day. It transpires that this was both incorrect and unnecessary. The reason they did it was that feedwater supplies were part of the ECCS and since these pumps were included in the baseload of the in-house supplies, then this was shutoff when the test was performed to see whether the turbine could keep the main circulation pumps running. This seems most obscure to me as it would have been perfectly acceptable to keep both feedwater and main circulating pumps operating during this test.

It is stated that when the reactor was reduced in power down to 1600 MW in which situation it would be running just the one turbine number 8 and the ECCS had been shutoff when the despatcher asked for the reactor to be maintained on power and not switched down presumably because he needed the electricity for the grid, the ECCS system was not switched on again. That means for many this reactor operated without one of its most important safeguards systems in operation. This was clearly a gross error on the part of the operators.

At 23.10 the power level should have been reduced to 1000 MW and the test begun. They could not meet this power level because at this time there had to be a shift in the use of the various control systems. This is very complicated. The local automatic shutoff rods were taken out of use because they only operate at the higher power regimes of reactor operation. At that point the automatic controllers should have taken over. This did not work. There is no explanation as to why not. However, another bank of automatic controllers were brought in to play. However, at this point yet another most important error was made. That is that this new bank of controllers were not synchronised for the power level required. By this I understand that these automatic controllers could be pre-set to take the reactor to a particular power level. However, the operator failed to pre-set this power level and the power in the reactor dropped to about 30 MW. Following that, attempts to increase power were fraught with difficulty, the automatic control rods were exhausted, ie. they were wound out of the core as far as they would go and the operators then raised power by removing all manual control rods out of the core by hand. This permitted the power to be stabilised at about 200 MW. The reactor was then being controlled not by any controllable moderator or absorber but by neutron absorbtion in the Xenon poison. At that point there was essentially no reactivity safety margin in the reactor. The rest of the accident sequence is then described and all of this is quite in line with what we understood before and I do not think that anything new was presented in the talks. It is clear that the pumps were overspeeding and should have tripped on overspeed, that didn't happen. It was clear that because the pumps oversped the water quality through the core was such that there was essentially no steam. With no steam the steam pressure dropped and indeed there should have been a trip on low steam pressure level. At the same time, feedwater was being fed into the steam drum to try and keep the circuit volume up, the reactor ought to have been tripped on low steam drum water level. The feedwater supplies were up at 4 times their normal level which ought to have been a warning to anybody but no notice was taken of that. The fact that cold water was being introduced in the down comers, by the use of the feedwater supply system simply made the temperature gradient across the core that much more critical. Finally when the alternator check valves were closed that should have tripped the reactor because it would have meant two turbines out of use but that of course had already been inhibited by the operators. At this point the steam pressure dropped and the water level dropped: at 1.22.30 the excess reactivity margin was violated and a computer printout was presented to the operators from the computer to the effect that they only had 6 to 8 effective rods reactivity available. This is to be compared with the absolute minimum limit of 15 and the advisory limit of 30 which were part of the programmed operating instructions. It is difficult to understand, and the speaker made this point, why they then continued with the test, even after the computer printout had said that they had violated all safety criteria. However, with the alternator check valves closed and the reactor still at power, even though it should have been tripped, the sequence of events which followed seems inevitable. At 1.23.40 the operators manually inserted all control and safety rods into the core by pressing button AZ-5. The story of the accident then makes fateful reading. The rods were seen to fall into the reactor but heavy shocks were felt. Instrumentation indicated that the rods had not hit bottom and at 1.24 two explosions according to outside observers occurred. The first explosion made large quantities of brightly burning material ejected into the night sky landing on the buildings and surroundings. At that point the accident had been initiated, the reactor had exploded and the subsequent series of events all followed. It was then stated that in trying to get an explanation of what happened, there were 3 possible reasons why the operators actually pushed the emergency scram button at that point. The actual statement made in the meeting was "while the operators were still alive they offered the following explanations". This was the first and only indication that we have had that in fact all the operators on the shift that night died as a result of the accident. The three reasons were:

1. That they noticed that the instruments showed the power going up at such a rate that they wanted to stop it.

2. That the test seemed to have worked all right and that they were just trying to trip the reactor in the normal way.

3. They could see the control rods being moved by the automatic systems and the operators knew by then that they were in real trouble and were scramming the reactor to avoid the obvious repercussions.

Consistent with explanations we have already had from other source, the very rapid power insert generated steam at such a rate that the reactor can could not maintain it and eventually in a few seconds the pile cap, even though it was 27 m across and 9 m deep, was blown off and the building above the reactor destroyed.

Having exposed the core to the air, oxidation set in Zircaloy steam reaction began, Hydrogen was evolved which then exploded and/or burned and caused extra damage to the reactor and made the problems of the recovery crews particular fire fighters even more difficult.

All of these aspects of the accident sequence were brought home very clearly by the film that was shown by the Russians. This film as I can make out had not been seen anywhere before and it included remarkable close up pictures of the damage to the plant and the release of fission products in the hours and days just after the accident. Goodness knows what kind of dose the helicopter pilots and cameramen received who took these shots because they were awesome and dramatic. It was easy to see the core of the reactor exposed to the air. Graphite could be seen glowing red to white hot and burning and smoke and vapours pouring out of the core. These films must have been taken before the helicopters dropped their loads of Boron. sand, limestone and lead on top of the core. Perhaps even more remarkable were the video films shot from within the building itself just after the accident where the cameraman clearly was standing within 20 or 30 m of the exposed core for a second or two to take the shots he had of the destruction of the pile cap and the reactor building. These were extremely awesome pictures indeed and if ever anybody wanted to re-inforce the enormity of this accident then this film would certainly do it for them. The film itself was not a dramatisation, it was not a documentary, it was just a series of video shots, taken by people clearly sent in by the Government investigating team to make a record of the scene. There was little commentary to go with it. I suspect that this film had not been shown in Russia and was indeed not shown whilst the TV and other visual media were present. I was particularly impressed by that film.

In summary to all of these parts of the reactor accident sequence there were 3 important admissions by the Russians concerning the causes of the accident. These were:

1. That the design was faulty in its conception and faulty in its realisation in having such a complicated and sensitive system for a reactor with a positive void coefficient.

2. That the people in charge of the reactor programme did not take on board early enough in their considerations the requirements on the human operators to be able to understand and drive this system safely.

3. There was an almost incomprehensible series of operator errors that led to the situation where the reactor was in such a difficult state that this accident was virtually inevitable.

There was some discussion concerning the immediate resonse to the accident, a great deal was made of the devotion of the firemen to putting out the 30 separate fires which had been started as a result of the initial explosion. Most of the deaths from radiation were amongst these firemen. It is clear that these men knew what they were doing and knew the dangers from fighting these fires next to a reactor which had just exploded. This was quite a remarkable performance on their part. It as interesting that the operators in the control room were said to have passed on the correct information to Moscow mere minutes after the first explosion had occurred. They had advised Moscow correctly in terms of one of the four types of damage into which they categorise reactor accidents. However, even though they had given correct initial information they subsequently gave incorrect and inconsistent information. It is said that they then informed Moscow that they could control the reactor and everything was all right even though it was clear that the reactor no longer existed. It is not known where this confusion came from or indeed what the conditions in the control room must have been like at that time. The Moscow centre immediately instigated their emergency response team who flew down to Chernobyl, arriving at 8.00 pm on 26th. By this time the fire in the machine room (turbine hall) was out but a separate fire had begun in the graphite of the core. There was an odd description concerning radiation levels in the surrounding areas and the associated evacuation plans. By 10.00 pm that evening, something like 18 hours after the accident, it is stated that the radiation levels in Pripiat were still very low. The reason used to explain this is that the first radioactive release actually missed the town. Advice went out to people living in stone houses that they should stay where they were and not try to be evacuated because it transpired that their original evacuation plan meant that people who were living in currently relatively safe areas would have to be transported via very dangerous areas to get them away. If they had implemented that plan they claim that the casualties would have been much greater. New routes had to be devised the next day which avoided the major contaminated areas. However, this seems to have been a very difficult decision in the end as the town of Pripiat itself is now so heavily contaminated that it is not inhabitable and will not be so in the foreseeable future. This is the main reason why units 1 and 2 have not yet been put back on line, ie. because there is nobody living there to run them.

There was much discussion about how it was decided to bombard the reactor with the various materials that helicopters dropped, things like the Dolomite limestone, the lead, clay and sand, etc. They seem to have worked out very carefully why these materials were used and the document does actually go into some detail for the justification of dropping each of these materials. If that was done at the time then it was quite impressive. We do not know of course whether this had the benefit of hindsight. Eventually they had to stop dropping these materials when they reached about the 5,000 ton level because they were very worried that the loading on the structure holding up the core such that it might all collapse. Then followed a rather straighforward reading of the document concerning the actual measured calculated releases in which there was a release on the first day which then dropped but it then built up again until the maximum release was on day 9. The explanations given for this seem quite reasonable and they concern both the filtration effect of all this material dropped on top, cooling effects of putting Nitrogen through the core and the like. We will have to wait until source term experts have been able to look at the detailed document rather than just the summary before we can really tell whether these figures look reasonable. At present I have no reason to believe that they should be incorrect. The Russians were very adamant that the maximum temperature reached in the core anywhere was 2000^{O}. They said that the maximum temperature actually measured was indeed 2000 degrees and that they would consider this to be an upper limit. This may be some bone of contention in the future when it is debated as to just what lease there was and how this was correlated with the conditions of temperature in the core at the time. The tables giving integrated activity releases have been discussed by us previously and nothing new seemed to come out of the presentations on the day.

It was interesting that some of the experience of the accident as described concerned the use of instrumentation under very high dose rates and in particular it was stated that infra red thermal radiation techniques could not be used in conditions where the radiation field was of the order of 100.000 r per hour. Personally I am not surprised that kind of radiation dose would knock out anything that relied upon sensitive energy levels in solids. Interestingly enough they said they had more success by going back to old valve operated electrical devices and optical devices. It was interesting that these admissions of radiation levels contrasted sharply with those quoted in the documents which all seemed to be rather late in the accident sequence and therefore gave no impression of the radiation dose levels which had been sustained on the site in the early days. It as stated that because of radioactive decay, decontamination procedures, etc, the dose rates around the reactor had been reduced from between 10 and 100,000 r/h to about 1 r/h now. (Note that the annual legal limit for radiation workers in the UK is 5 r/year). It was stated that only in the reactor vaults are the radiation readings now very high. Diagnostic devices have been placed in the reactor cavity to monitor the temperature and radiation levels and other things such as the air flow rate through the core. It was stated that:

1. No more radioactive release was occurring from the core. (He actually said "virtually no more release" indicating that aerosols were still be emitted to atmosphere but only the order of tenths of curies per day.)

2. Presently temperatures above 200-300^{O}C are not found. Temperatures in areas around the core and open to people are less than 45^{O}C and stable. Instrumentation is now available to give individual radiation levels in the various compartments of the reactor building.

The accident affected the other 3 units as well. Units 1, 2 and 3 were shut down, that is they were brought to cold shutdown but the time lag was a bit worrying. Unit 3 was not shutdown until 5.00 am on the day of the accident, units 1 and 2 some 12 hours later. The fact that the ventilation systems on these reactors pulled in radioactivity and contaminated the internals does not make the clean up any easier. It was stated that units 1 and 2 are entirely viable, all of the equipment is intact and the intention is to put them back into operation by the end of the year. The main problem seems to be building the town for the staff to live in since the town of Pripiat is no longer habitable. No prognosis was given for unit 3 and it was stated that the analysis was not yet complete but there was a further need to bring radiation levels down. Around the actual site itself there seem to be a rather complex situation where there are areas of very high radiation levels and low ones. It was implied that they were still going through a rather complicated clean up process on the site. Around the site 3 zones were identified.

1. Zone 1, total isolation within which no normal work was going on. Zone 2, where partial economic/industrial activity was being undertaken but with careful supervision. Zone 3, where strict monitoring was imposed but everything else was being done as usual. No indication was given as to the radiation levels or the nature of the hazard in those areas. The final numbers for the seriousness of the accident were given, that 203 people were seriously injured, 31 died, the total collective dose was 9×10^6 rem in 1986 and 29×10^6 rem as the integraded collective dose for 50 years. This would indicate approximately 3000 cancer deaths in the Soviet Union as a result of the accident.

2. There were then some indications of what steps were being taken to guard current RBMK reactors against this kind of accident in the future. The first of these was that the control rods would no longer be able to be withdrawn outside the active zone of the core, the minimum withdrawal distance would be to within 1.2 m of the top of the core, the net effect of this of course would be to reduce the size of the core from 7 m deep to 5.8 m deep but it would indicate that the control rods would be closer to the high flux regions should they be wanted in a hurry.

3. The reactivity margins had been increased in order to guard against the positive void coefficient instead of having 30 equivalent worth of rods available this would be increased to 80. This was an interim measure as the file solution would be increase the enrichment of the fuel to 2.4% from 2% and to include more absorber into the core such that there would be no situation where more than one beta could be inserted into the core for any change at all in coolant density. This is their answer to the positive void coefficient problem.

4. Additional instrumentation and "technical means" would be included, this presumably is engineered safeguards of various kinds which were not illucidated. Also organisational means to improve procedures leading to situations as well as operator training, management practices and the like.

Currently some RBMK reactors are described as not working whilst these measures are being implemented. Those plants where it has been done are now working again.

It was stated that it is too early yet to estimate the economic consequences of the accident since there is still the requirement to discuss the long term implications of these safety changes to all the RBMK in the Soviet Union.

Comments on Tuesday 26 August

Special Session A - Plant Design and the Accident Sequence Evolution

Overall the day was patchy. The problem was that a number of the speakers simply read the relevant passages from the annexes to the main document and no new information was given which was not already contained in it. On the other hand some speakers provided some insights into the nature of the accident and the nature of their work which were not explicitly provided in those annexes. This made the generation of a short descriptive and factual piece for John's daily report quite difficult. The way we organised it was that I attended this session, Alan Eggleton and Barry Carpenter session B which covered emergency planning and medical matters and the like. These were then fed to John Gittus who produced the report in time to distribute it the next morning.

The first speaker began by outlining the background to nuclear power in the Soviet Union indicating that they still planned a large increase in the contribution to be made by nuclear power to their overall energy needs. He highlighted that there were 3 uses for nuclear power:

1. The generation of electricity
2. Generation of electricity and the use of the waste steam to provide thermal energy for either industrial processing or other similar uses.
3. Purely for the generation of thermal energy to be used for district heating.

He outlined the fact that the energy requirements of the Soviet Union were somewhat different from the West in that a great deal of their energy was used for space heating because of the severe winters and that they had this difficulty of the distance between their natural hydrocarbon reserves and the centres of population. It was stated that they planned to begin building and 800 MW fast breeder reactor during the next 5 year plan. Overall a very optimistic and positive picture of the future use of nuclear power was given in these opening presentations.

A rather frank and illuminating talk was given on the history of nuclear power in the USSR. A man called Kurchartov was the leader of the civil nuclear power business in the forties and indeed the Institute in Moscow still bears his name. They built the first electricity generating reactor at Obnisk near Moscow which came on line in June 1954. This reactor was built in 4 years and was based on the pressure tube graphite moderated reactors being used then for the production of Plutonium. The reactor produced 5 MW (e) from 30 MW (th).

The development of a viable commercial scale electricity generating reactor was started in the fifties with the adoption of the RBMK as the design. The reasons that RBMK were chosen seemed to be that the Soviet Union at that time did not have the capability to build the very large pressure vessels required for a pressurised water reactor system of sufficient size to be economically viable. In addition, it was stated that the RBMK reactors required no specific industrial nuclear infra structure to support them. In other words all the pumps and pipework and all the rest were something that was not special to nuclear power plant. Eventually the water reactors were produced, the first one was brought on line in 1964 at the Novavonish power station of the WWER design and it was of 760 MW (th), 200 MW (e).

When the decision was taken to utilise the RBMK design it was done so in the light of the known problems with this reactor. This was emphasised by Mr Legasov during the Monday sessions and there is no point in going through again what the problems with that design were. It is worth noting that when Lord Marshall held a press conference the previous day, he said that the problems with the design as recognised and outlined in the talks were indeed foreseen in the 1977 NPC document where 5 aspects of the design were criticised. This is consistent with our view of the problems with a reactor of this kind.

There were one or two interesting remarks concerning the development of the RBMKs. First, at a site called Belagersk RMBKs were utilised using nuclear super heating. This meant that the dried steam was passed through the core again for further heating, very analogous to the rather standard designs of the super heating boiler. This had not been follwed up however, although no reason was given for that. It was also stated that this plant also used the thermal power from the waste steam for heating. It is interesting to wonder where that steam went because with a direct cylce reactor it must have been at least mildly radioactive. This plant, it was stated, was used to prove the design concept of the RBMKs.

Because of the fact that no specific industrial infrastructure had to be produced to support them there was a rapid programme of building of RBMK reactors and 15 plants were built in the 1970s, they now have some 100 years of reactor operation behind them.

It was stated that these plant have a very high availability and numbers like 90-91% were quoted. These numbers have to be treated with some suspicion until the basis upon which they are made is known because it depends whether planned shutdowns are included in that or whether that figure is 90% of the planned availability outside the shutdown periods, for example.

The design parameters of the reactor were then gone through in some detail, these are all in appendix 2 and need not be repeated. However, the limiting parameters for the reactor were given and it was stated that they were rather below the maximum that could be obtained and that efforts had been made to decrease this margin. For example, the present channel power was 2,600 kw. The maximum theoretical channel power could be between 3000 and 3,200 kw. The maximum graphite temperature was presently 600°C but this could be raised to 750°C (although in this context it was stated that the graphite temperature could rise to 750 during certain transients when the Helium/Neon mix was not matched to reactor power properly). There were several other parameters which were then given to indicate that the reactors were being operated well away from maximum capability. It was stated that the safety margins were about 10% of the operating parameters. Tests had been made at 107% of nominal operating power in order to demonstrate that the reactor could be run closer to its critical parameters. Personally I would have thought that was rather dodgy as those margins do not seem that large to me, especially with a reactor which is known to have difficult control characteristics. However, the Russians say they are still trying to increase the power by such means as improved heat exchanger intensities, whatever that meant. I think this is improving the heat transfer in the channels or having some kind of agitating device in it. This was the reason that the power output could go up to 1500 MW in the Ignalinsk station which was commissioned in 1983. The size of this plant is the same as that for 1000 MW plants.

Finally in this presentation they reconfirmed that they believed that the operating experience with these plants proved the correctness of the basic design solutions. However, they reiterated that they had not fully accounted for the human factor and they did not include appropriate technological protection at the design stage. It is a matter of conjecture whether experience would indicate the correctness of the basic design solution adopted but that is a matter for judgement.

The following was a description of the accident sequence, this was based closely on annex 2 and much of it was a straightforward presentation of the provisions on the reactor for the various design aspects pertinent to the accident. Thus, there were long presentations on the emergency procedures on the plant, highlighting the different levels of trip which are provided. These start with a level 5 in which this is a full emergency shutdown, all the rods are inserted in the reactor at once, this is the now famous AZ-5 button which instigates it, down through various other power set back type situations. Apparently because of the rather complex numbers of pumps and pipes on this reactor they believe that there are certain situations where a full trip is not required and so thay have several intermediate phases. In discussion later with various people including Derek Smith, it transpires that on AGRs there is only one such situation and that is where there is circulatory failure in one quadrant and in that case only that quadrant is tripped and the reactor can continue on 75% power. Speaking to the PWR people and to Enno Hicken from GRS they expressed the view that whilst they would like to be able to have these partial trips, as yet no-one had been able to pursuade their regulators that they were acceptable. This is particularly true for the Germans with their BWR plants which are probably more analogous to the RBMK in this context than the PWR itself.

Explanations of the thermalhydraulic aspects of the plant were explained, all of this was basically straightforward information as contained in the annexes. It was interesting that the flow adjustment through the fuel channel had to be made twice during the life of the channel so as to regulate the flow to keep its margin from critical heat flux approximately the same during the lifetime of the element. This of course, being due to burn up effects.

Natural circulation has been experimented with widely to demonstrate that this reactor has this as a reliable safety feature. Tests have been done on actual reactors and upon test rigs and with analytical models. They now believed that the reactor could operate at between 25% and 30% of full power totally relying upon natural circulation and with a good safety margin. He stated that when the coolant circulation pumps are cut the reactor goes into natural circulation automatically and indeed the channels do not go beyond the critical heat flux requirements to keep the thermal hydraulic situation stable. It was stated that this would even be the case if the maximum power loading was increased to 3.200 MW per channel but this would seem to indicate that main circulation pump rundown is required in the safety case for that particular situation. This was just an impression gained during the lecture and it was not stated.

On the graphite temperature control, the system using Helium and Nitrogen was described. The maximum graphite temperature was stated to be between 600 - 620°C. What surprised me was that as the power of the reactor is altered then the mixture Helium - Nitrogen has to be changed so as to keep the graphite temperature reasonably constant. For example, it was stated that when the power drops from 80 -60% then the temperature might go up but would never exceed 750°C. The fact that the gas mixture has to be controlled to match the reactor power because of the heat deposited in the graphite by the neutrons, indicates that this reactor is even more complicated than we had originally thought because that system must of course require controls to couple it to reactor power.

It was stated that Chernobyl Units 3 and 4 had provided on them more safety systems than previous reactors of the RBMK design. For the protective systems, essentially two different types of protection were provided. First of all, protection against overheating by an emergency core cooling system and secondly protection against excess pressure by the provision of an array of safety relief valves. The ECCS system was essentially a 3-loop system the requirements of which meant that the flow rates had to be of the order of 500 tons per hour in the initial phases, reducing to 100 tons per hour later in the sequence. These flow rates were provided by electrically driven pumps connected to the diesels with the valves being operated from the guaranteed uninteruptable supply. One of these channels utilised the electricity from the run down turbine to drive the feedwater pumps as part of the ECCS system. It was this that was complicating our understanding of which pumps were being driven by electricity from the run down turbine and whether the experiment was to demonstrate a requirement in the safety case or simply to demonstrate the additional margins available if the circulation pumps could be run on the electricity from that source. It was stated that the feedwater pumps could be kept running for 45 seconds as the turbine ran down and this would provide a useful backup to the initial 500 ton per hour flow rate required during the first 40 seconds after shutdown. The water supply for this ECCS system is obtained from the pressure suppression pool.

Protection against excess pressure was provided by a rather complicated set of safety relief valves which opened in sequence but the important point was that the total capability of these safety valves was equal to the nominal steam output capability of these safety valves was equal to the nominal steam output capability of the reactor. These valves were vented through the pressure suppression pool and had the capability to be forcibly opened if they failed to open under a command. I presume by forcibly they meant manually. This completed the description of the basic design parameters of the reactor, the afternoon session then went into the accident sequence itself. On the accident sequence and development the speaker began with quite an interesting personal expression of impressions on the accident. First of all he commented on the psychology of the team which could have led to them disobeying so many of the safety instructions. First of all he said that this was the end of the week, it was a Friday night and traditionally in Russia this is seen as the start of the weekend and people would be not in the best pyschological state. I would add that it was also early in the morning when the accident happened and that they had been messed about with all day because the local dispatcher in the Kiev area had asked them to keep the reactor on line even after they had begun the procedures for reducing the power. It has not been made clear whether the same team stayed on duty all the while from 1.00 pm on 25th through to 1.00 am on 26th, whether there was a shift change or whether in fact the original shift team went home and came back again. This could have a bearing on the psychological state of the team. Further, this was the beginning of the May day holiday and I am sure that they would have been very keen to have got the tests overwith and their holiday started. This is the biggest holiday in the Soviet Union. In addition, he also stated that Chernobyl Unit 4 had won awards for the best plant in the Soviet Union. By best he meant best in terms of availability. There is some question therefore that the team may have been overconfident and felt that they were on top of any situation as they were the best team in the whole country. These thoughts could well be behind the comment in the summary document that the operators had lost their sense of danger. All of these comments serve to indicate that the central investigating team seem to find it quite difficult to understand how the operating team could have got themselves into this situation and why they would think it all right to override so many of these safety protection systems.

In addition to the above, it was also stated very clear that on the face of it these tests had nothing to do with the reactor system itself. They were concerned only with whether the new wiring on the exciter of the turbine would provide high voltage electricity during turbine rundown. So far as the reactor is concerned all that did was to spin the turbine to give them the kinetic energy to start the tests. For this reason the test was supervised by an Electrical Engineer and not a Reactor Physicist. This Electrical Engineer evidently was giving orders to the reactor operating team. Another interesting part of this personal view concerned the response of the emergency team. The Russians clearly have quite well prepared plans to put together teams of experts to fly anywhere in the Soviet Union where reactor accidents might occur. The operators telephoned the Moscow Emergency Centre minutes after the accident with the code words appropriate to a nuclear accident with radiation and fire. This triggered the emergency team, they were called out of bed presumably and they were ready to go by 2.30 am. This is only one hour after the accident. Whilst still in Moscow they tried to get more information on the accident as the team only knew the bare bones at that time. This is where the operators confused the issue by saying that they thought they had the reactor under control. Legasov mentioned this yesterday and commented that they thought the reactor under control when in fact the reactor was totally destroyed. This emergency team then travelled to the station to see visually what had happened. When they arrived the firemen had already extinguished the fire. From the buildings on the site they could see lumps of graphite and fuel lying around on the ground. This must have been a most impressive sight! At the same time that the emergency team went down a Government level committee was set up and took a firm hand in organising recovery operations. Clearly very authoritative people had to be brought in to requisition enormous numbers of vehicles and work people to set in place the kind of recovery operation that happended. The scale of it and the time over which it took will of course be examined in detail by the specialists but to me it seemed quite impressive given the circumstances.

Concerning the time development of the accident sequence, two presentations were given, the first one essentially concentrated on the fact, ie. the state of the plant as it was and what the operators actually did. The second one combined the data available on the state of the plant with a quite complicated mathematical model which then gave a simulation of the response of the important accident parameters during the sequence of events. The latter is quite complicated and is all presented in great detail in the appendix so here I would just reiterate the essential features of the accident sequence as presented by the first speaker. The important points from the calculations were to indicate the magnitude of the energy depositions late in the energy sequence and to confirm the thermalhydraulic aspects, especially how close to saturation the system was when the test was initiated.

On 25 April, all units on the site were on line and producing electricity. The test had been timed so that the grid requirements would be at a minimum when the reactor was taken off line for a planned maintenance outage. The purpose of the test was again stated to be to examine the possibility of using the mechanical energy of the rota to sustain the houseload during a loss of grid incident. This feature is already used in one of the sub systems feeding the ECCS as mentioned above it is used to drive the feedwater pumps for the initial flow requirements on that system. It would be usual for such tests to be allowed on these reactors during a planned shutdown. Two tests had been performed previously and it was found that the voltage dropped off too quickly to be able to utilise all the stored energy in the rota. It was not made clear, and I still do not know, whether this rapid dropping off of the voltage was because of a fault in the exciter of this particular turbo alternator set or whether it was a generic fault which came into play when the main circulation pumps were on the houseload as well as the feedwater pumps. Either way the new voltage regulator was to be tried in these tests. It is emphasised again that there was no need for the reactor to be on power for these experiments. Because of this the proper safety procedures were only gone through in a very cursory manner indeed, the safety documentation only stating that if difficulties were encountered then they would have to rely upon the normal operating manual. This is another feature which could have effected the psychology of the team. They just did not believe that these tests were really important so far as the reactor was concerned.

The chronology of the accident has been written down now many times. At 1.00 am on 25, a 7th alternator turbo set was switched off and the power reduced to 1600 MW. All electrical houseload was switched to the bus bars of the 8th turbo alternator. At this time the ECCS was switched off. The reason for this is that in testing these emergency modes they did not want a lot of cold water flooding into the reactor unnecessarily. Not only was the ECCS automatic system blanked off but it was done manually as well. I do not quite understand what this manual arrangement was but it could have been for example that manually operated valves were closed. Instead of the tests proceeding the reactor was kept on line at 1600 MW because the Kiev distribution centre needed the power. The ECCS was not switched back on during this period and the reactor operated for many hours without the ECCS operational.

It is believed that the reason that the reactor was not shut down for the experiment was that if the first attempt at the experiment was unsuccessful then the reactor could be used to power up the turbine again and a second test attempted. This would tie in with the over confident psychology of this team that believed it had full control of the reactor and it was going to make a good job of these tests as a matter of team pride.

When the power was reduced the local automatic rod system, the ALR was switched off at low power. This is permitted as reactor control goes to the automatic system in those conditions. However, a mistake made was that these automatic rods were not synchronised for power requirements, the operator could not control the reactor power manually and the power fell to below 30 MW. It took them a long time then to get the reactor back up to 200 MW by that time the Xenon poisoning was so bad that this power level could only be attained by overriding the permitted reactivity margins. At this point the psychology of the team must have been dominant to have forced them to go on with the test.

A further main circulation pump was put into the circuit on each half of the reactor making 8 in all, this meant that the flow rate through the core was inexcess of permitted levels, that is 56-58000 m^3/h and some individual pumps were up to 8000 m^3/h which is a rate at which it could cause mechanical problems such as vibration due to cavitation. By increasing the flow the operators were reducing steam pressure because water was replacing steam in the circuit as the steam was not produced in the core because the flow rate was too high for that power level to boil the water in the channels. With the power only at 200 MW and the flow in this condition the reactor was unstable and there were strong fluctuations in steam pressure and water level. This meant that more trips had to be overriden to keep the reactor operating. At 1.23.10 am a computer printout showed that the reactivity margins were only 6-8 rods. The minimum is 30 and even then operation is only allowed by permission of the station Chief Engineer. They should have shut down at that time. At 1.24.04 the check valves to the alternator were closed and the test was initiated. The reactor carried on at 200 MW power. the reactor should have stopped at this point because with two alternators out it should have tripped but of course this trip had also been blocked. Some 30-40 seconds later the operators notice that the power is starting to go up slowly. At 40 seconds a level 5, ie. maximum trip on AZ 5 is initiated. All the rods start to go down, explosions are heard and the rods do not go down to their lowest stops. From external observers two explosions are reported at 1.24 am about the 4th unit. Burning materials were shot into the air landing on adjacent buildings and the turbine hall causing about 30 independent separate fires to start.

The reasons that AZ5 was pressed at this time are not really known because all of the operators have since died. There are 3 possibilities.

1. That the increase in power was noticed and this was cause for concern and they operated the trip as a panic measure.

2. That they believed that the test was over and that they could then shut the reactor down which they would do by pressing this button.

3. That the test was not going well and they tripped to re-start.

Detailed analysis of the accident indicates that when the alternator check valve was closed to start the test the reactor was in the worst possible state that it could be for such a transient. Because of the Xenon poisoning and the solid water flow through the core, both of which would decrease reactivity by increasing moderation and absorbtion, the control were as far out as they could possibly be. Indeed it was stated that even when initiated for the first 6 seconds they would not be able to have any effect at all on the neutron flux. The positive void coefficient would have been at its worst because the channels would be full of water and therefore have the most potential for reactivity insertion on voiding. The calculations presented indicated that the power went up exponentially to something like 400 times nominal power in about 1 second. It was a prompt critical excursion and the energy release was described by the speaker who explained the analytical techniques as mad and crazy.

The interpretation of what actually happened was that the energy deposited in the fuel raised its temperature to around 3000°C. The amount of energy deposited was about 300 calories per gramme and at that energy deposition rate fission gases and volatiles within the fuel would have exploded from within, small particles of very hot fuel would then be ejected into the fuel channels where any steam present or even if there were water, would very very rapidly produce high steam pressure, that would fail the pressure tubes and the enormous amount of steam would blow all of the top of the reactor off. Because of this very high energy depostion small particles of fuel were thought to have been burried in the graphite moderator and were only released later as the graphite burned.

Following this first explosion due to steam pressure the reactor would be exposed to air and various exothermic reactions leading to combustible gases being produced particularly Hydrogen from the Zirconium steam reaction and carbon monoxide from the graphite water reaction. Since there was only 2-3 seconds between the two explosions it is not at all clear that I would agree that the second was due to explosion of combustible gases. It is possible that the first explosion produced a hole in the pile cap and the like but this was not big enough to vent the steam pressure entirely which built up again and had a second larger explosion. However it is possible that these combustible gases did explode but this seems rather academic to try and decide between those two causes of the second explosion. There is no real reason to dibelieve the Russians when they account for it by means of this chemical explosion. They say they will be performing an extensive programme of research better to be able to describe the experiment but they believe that the main causes are explained by their current understanding and it is all down to the void coefficient and the speed of operation of the reactor protection system.

It is interesting to note that the explosion occurred in the upper part of the reactor. Presumably this is because the channels started to void at the top first and it was there that the prompt critical excursion was initiated. If the explosion had occurred in the lower part of the reactor there seems to be no doubt that the vast majority of the core would have been blown out of the reactor. As it is a considerable amount of fuel was blown out and was found lying on the ground and all was aerosolised and dispersed in the environment. However, if the explosion had occurred lower down the accident would have been very much worse. The reason that it occurred at the top of course is that the flow in this reactor is from bottom to top and therefore the top part of the channel would be closer to saturation than the bottom. Reactors with down flowing coolant or even horizontal coolant therefore would have suffered a much worse accident.

The next presentations were concerned with the releases from the reactor and again all of this material is included in the appendices so there is no point in going through it all again. The analysis by the Russians indicate that there were four phases to the release.

1. Mechanical release by the explosion. Thus the nature of the radioactive isotopes emitted were essentially those found in fuel and can be explained by mechanical entraining.

2. The release decreases due to the material dropped on top of the reactor. Again the composition seems to be similar to that of fuel, the argument being that hot air associated with burning graphite was entraining particulate and emitting it into the atmosphere.

3. At about 6 days into the accident they had to stop dropping material on top of the reactor cavity because there was a serious possibility that they would collapse the structures below the core. At this point the core was blanketed, the temperatures rose and the more volatile species, Iodine, Caesium and Telurium were

driven off differentially compared with the composition of fuel. This was caused by heating to temperatures up to maybe 15 - 1700°C. There was however, still a considerable fraction of material which was typical of core rather than heated core, indicating perhaps that not all of the core at this time was undergoing as much heating as other parts.

4. There was a sharp reduction in release after 6 May and this is put down to, first of all special measures taken to provide cooling when liquid Nitrogen was pumped into the bottom of the reactor and secondly, to the formation of more refractory chemical compounds with the volatile fission products by the materials which had been dropped on the reactor.

The actual amounts of release were given as about 12 million curies on the first day and an integrated total of about 50 million curies. Some 15-20% of the volatile Iodine and Caesium released and much less fractions of the less volatile species. There is some debate as to whether we would agree entirely with these numbers but in the main they are close enough to our estimates to make us believe that the Russians have done a reasonable job and are not trying to play down the size of the accident.

Further lectures were given on the on-site consequences in this session and a lot of this was concerned with radiation levels in various parts of the building and the site. All of these data are given in the annexes. It was stated that fuel was in the suppression pool because various pipes connecting the core with the pressure suppression pool had broken off and therefore the radiation levels were very high. However, the temperatures in that region were relatively low, ie. about 45°C. The argued that the coolability of the core had been enhanced by the fact that there was air being convected around the core through this pressure suppression region.

Instrumentation had been placed in the core debris by helicopters lowering various instrument packages on wires into the core. One wonders what dose the helicopter pilots and operators must have got. All this instrumentation allows them to be able to say that the current heat output is about 1+ MW, they know what the temperatures and air flow rates are and that these results will be used to define the design parameters for the entombment process which they are currently engaged in. Construction of the tomb will be done against essentially 4 requirements:

1. To protect the adjacent site and to allow the other reactor units to be operated again.

2. To remove residual heat from about the fuel and the collapsed part of the reactor.

3. To ensure that the monitoring for temperature and the production of gases such as Hydrogen (by radiolosis) can be carried out safely.

4. To be able to lay contingency plans if things were to go wrong after the entombment had been completed.

Quite a bit of information was available concerning the survivability of the plant and indeed many of the cells which were designed to withstand internal pressures from ruptures in the steam circuit had survived the explosion.

Finally the speaker reiterated the measures to be taken on other RBMK reactors to ensure their future safe operation. This was essentially the same as was given by Legasov yesterday and includes 4 measures:

1. Control rods will be premanently inserted 1.2 m into the core so that they are much closer to where they are needed in a hurry. This will have the effect of essentially making the core smaller by making its depth less.

2. The reactivity margin requirements will be increased to 80 rods over the current 30. This will have the effect of decreasing the void coefficient by about a factor of 2.

3. In the longer term they will use higher enrichment fuel and more absorbers to reduce the void coefficient still further. They stated that fuel with this enrichment had already been tested before the accident because they wanted to get rid of the positive void coefficient. The new enrichment level will be 2.4%.

4. Some 10 channels will be used for rapid safety rod insertion. By this they mean a special system where the reactivity can be controlled within one or two seconds. They have not yet decided on the material that would be used for these special control rods. Other interesting features of the day: Lord Marshall had finalised what questions the British delegation would be putting to the Russian delegation. He said that he had had meetings with both Blix and Legasov and in those they had identified questions which would be acceptable and would indicate that the UK wished to make a positive contribution in the future. This was because the Americans, French and Germans were all being somewhat reticent about making overmoves to be seen to be active on organising future activities. Lord Marshall speculated on the reasons for these which all sounded very reasonable. The suggestion is that there will be an international conference called to discuss the interface between the operators and the designers in terms of man-machine interface and problems of control. It was felt that there would also be a proposal for a longer term programme of work rather than just a conference. This as accepted and welcomed by Legasov. The second item was to propose a conference on decontamination so that we could learn from the Russian experience but at the same time would be able to offer our own efforts as a quid pro quo. Again this was welcomed by Legasov and indeed he wished to see this conference called at an early time. The third question concerned epidemeology and here care was needed so that it was made clear that the USSR were the only people who really could do the epidemeology following this accident but that interested international bodies could help with this and there was a suggestion that the IAEA might be called upon to provide some kind of organisational focus for this internation assistance to make sure that the epidemeology was done properly and the maximum information obtained from the data available.

The other parallel sessions were covered by Barry Carpenter and Alan Eggleton and their reports will have to suffice to fill in on what happened in those sessions.

First Workshop 1, Session A, Chaired by B Edmondson

Actual Accident Itself and Sequence of Events in the Shorter Term

Edmondson emphasised the nature of these workshops saying that they were supposed to be two way in the way of exchanges between the Western and Eastern experts. Further the aims were to discuss international collaboration in areas which were mutually identified as both important and feasible for such studies. It was obvious that detailed questions could not be handled in this kind of format but indicated that the Russians had already agreed that some kind of arrangement would be sought whereby these questions could in fact be answered. There were a great number of questions, Edmondson approximating these to 350 for his sessio alone.

Edmondson said there were 3 basic types of question which he would take during his workshop, they were:

1. Clarification of information already given by the Russians, information which could be cleared up very quickly.

2. To exchange relevant experiences so that research or previous events which had gone on in the West could be transmitted to the Russians.

3. Identification of the broad requirements for nuclear plant safety internationally.

Six areas of interest were identified which would form the structure of the workshop, these were:

1. The causes of the event.
2. The specifics of the phenomena in the accident evolution.
3. Remedial action taken and the design of plant to mitigate operator error.
4. Control of procedures for special experiments.
5. The training of operators including such topics as lines of delegated authority.
6. To learn lessons for our plant about such an extreme accident.

Edmondson began by trying to establish whether the assembled experts were satisfied with the information that had been given to us by the Russians. It is particularly important he said that we understand properly what the Russians have given us and therefore the experts who had been hired by the IAEA were to make sure that their understanding was reviewed by the Russian experts so there could be no difference between us.

The first expert was John Young from Berkeley Nuclear Laboratories who was concerned mainly with reactivity and neutron physics matters. He opened with a remark that Fermi had once said that there would no nuclear industry without delayed neutrons. Hopefully we will never test that argument again. His overall presentation essentially said that he agreed that the conditions as described by the Russians would have led to the nuclear excursion as described. Only the doppled effect and leakage were on the side of introducing negative reactivity. It was clear that a prompt critical nuclear excursion had occurred. He introduced some arguments to indicate perhaps that not all of the details as given by the Russians, were entirely correct. He particularly emphasised the difficulty of making arguments concerning events in parts of the core rather than the whole core. Nevertheless there was general agreement that the Russians conclusions were accurate. It was important to recognise that this was a most crucial issue. The reason being that all subsequent remedial action had been based on this appreciation of the accident. Clearly they had got it wrong then they could easily have made a mess of recovery.

Even though the agreed format of these workshops was that the experts would examine the questions posed, filter them and put them into a form to represent the delegates, it was strange that the first thing done after Youngs presentation was to throw open to the floor, for questions. The first one which came from a Yugoslavian sought to have a history of all other accidents particularly other accidents involving a nuclear excursion. This is precisely the question that had been so carefully avoided by going through this process of having the experts interface between the delegations and the Russians.

Following this there was a rather difficult period in the workshop when neither the experts nor the Russians were asking or answering questions but were rather making statements. Statements concerning the nature of the core at the time of the accident were then made but these were quite similar to those that had been presented earlier. The new information really concerned first of all the fact that the reactor shut itself down, they believe due to homogenizing the fuel and the graphite in that part of the core which underwent the explosion. That is the disintegration and explosion of the core separated the fissionable material to the point where the chain reaction could not keep going. It was a good job it did too because the steam pressure lifted the entire pile cap off and laid it down next to the reactor cavity. The explanation of how things went during the explosion is difficult but it is believed that the actual nuclear excursion took place in the lower part of the core and that approximately 30% of the core was involved. The reason for this is that the control rods had started to enter the core and therefore the flux was depressed towards the lower half and the voiding would go rapidly down the channel, overtake the control rod advance and cause the prompt critical excursion in the lower third of the core. It seems to me odd that in that case more of the core wasn't blown out in the explosion. As it is data for the amount of core was promised for later. Graphite blocks were blown out of the building and there would also have been thrown around inside the reactor cavity and this itself would also have served to shutdown the reactor.

The Russians argued that there had been insufficient work in the general area of the physical and chemical processes surrounding such a violent event at high temperatures. Particularly they highlighted aerosol generation in these conditions and said they were looking to the IAEA to institute some work in this general area.

It was not clear what work would be necessary to follow up on the reactivity transients. It was pointed out that test reactors had been investigating neutron pulses of many thousand times that involved in the Chernobyl accident for very many years. It was really thought that it was application of the current understanding that was more important. It was pointed out that the Chernobyl accident inserted about 1+ beta (dollars) in 2 seconds. When you consider that experiments have been done on small scale high flux reactors when thousands of dollars were inserted in a few milliseconds then you begin to get a feel for the rate at which the accident went. The important thing here of course is that in a very very large core like Chernobyl there is nowhere for the energy to go other than into heating the fuel and disrupting it. These experiments were on very small amounts of material where the energy dissipation was easy by radiation.

Turning to administration arrangements and operator action and the like the expert was John Brown from Ontario Hydro. His opening remarks essentially said that no matter how good the documentation for the operators the ultimate task that guarantees safety lies with the men actually running the machine. These people must be provided with the most appropriate tools. He then went through a detailed account of how Ontario Hydro trains its operators. This included all the requirements for junior operator, second operator, first operator, etc. Brian Edmondson then chipped in with an overview of UK treatment in these areas. Lars Hogberg from Sweden then remarked that in Sweden the idea of having so much safety documentation was to encourage people to get used to going through the steps required for safety justification so that it became second nature. G Petrangeli from Italy offered a moment of light relief when he said that in Italy they had made it a criminal offence to make mistakes on operating reactors. For some reason this seemed to be amusing to most delegates. Bacher from EDF emphasised the point that there was a great difference between intentions and applications in the field. The French have put a lot of effort into operator training and into the development of simulators. All operators have to go through twice a year, work on simulator training for big accidents. A radiological protection person and safety engineer is included in each operating team. This technical area was one where people were much more willing to participate in. There was a suggestion of the establishment of data banks to share out information on previous accidents and incidents. Brown said that the question is first to ask what kind of data collection exists because previously they have found with other plants that you had to be very careful in both collecting and storing the information in an appropriate form. The question was asked of the Russians whether they had had any precursors of any kind on any other plants prior to the accident. The Russians said very clear that they had never had anything like this before and they had never seen any precursors that could remotely have led to this accident. Personally I wonder about that but this is what the Russians said. They then indicated that there operator training and the like was just as stringent as that in the West and they considered it to be very important, however, it was agreed that further exchanges of experience in this area would be beneficial and the Chairman was asked to take up this idea of collaboration.

On the human machine interface, the Russians said, rather interestingly, that there was a moment during the accident evolution when the operator did not recognise quickly that the situation was very marginal. A great deal of information was being presented to the operator but the really crucial points weren't highlighted and drawn to his attention. The Russians said that first of all it is impossible to have someone holding the operators hands all the time. They also outlined the optimum degree of automation. When these plants are on baseload an atmosphere of calm settles in and a sense of passivity takes over. There seems to be a need to generate an active state in the minds of the operators so that they are in a proper condition to react to any situation. Personally I think that this accident offered nothing in the way of information in that area since the reactor was actually being operated under particular test conditions, it was not simply churning along on baseload happily when something quite extraordinary happened.

The Cuban delegate introduced the suggestion that the IAEA be charged with hosting a great deal of collaborative work on operator training, etc and this should include the possibility of international acreditation of operators. This was noted by the Chairman but of course is quite a dramatic suggestion in terms of getting agreement between countries to aggregate their responsibility to an international body.

More information was given at this point concerning the site being constructed to house the operators when the reactors go back on line. This will not be permanently occupied but will be a shift site. Staff will live there for 2 weeks before going back to their permanent station. In this way the staff will be cycled. Units 1 and 2 are expected to come back on line by the end of the year and this is held up mainly by the construction of this town. Unit 3 will take longer because it shared some apparatus with Unit 4.

A delegate from Israel asked the question I suppose that everybody wanted to but darent, which was essentially, is it the underlying philosophy of the Soviet State driving operators to meet production targets that caused the operators to override safety rules in order to successfully complete a test? The Russians answered this very calmly by saying that because each unit is shut down for planned preventative maintenance, so called medium maintenance every year, and every fourth year it is shutdown for capital or large maintenance. They use this as a defence that safety does not come second and production first.

Frescona of Ontario Hydro then led the discussions on design questions. Particularly on safety system design philosophy and the design of plant to mitigate errors. The areas concerning safety systems design philosophy were broken into 3 parts.

1. Criteria used for separation including redundancy and

diversity of the reactor protection system.

2. Criterion for the choice of level of the trip signal

3. Design to stop operators dismantling safety systems during operation.

Birks of Canada indicated that they had had a reactivity accident in the NRX reactor in 1952 which in some ways parallelled that at Chernobyl. However, the results were very much less. They felt it fortunate to have had their accident before the start of a general programme of civil reactors. The lessons they had learned from this were:

1. A shutdown system must be powerful enough to guarantee shutdown under all conditions.

2. These systems are entirely separate from any of the other controlling systems.

Derek Smith of NNC then gave quite a long rundown of British philosophy in reactor protection system design. This included the idea of diversity and redundancy in terms of means of shutdown as well as signals. He also introduced the concept of interlocks. At this point we had great difficulty with translation because somehow the Russians did not understand interlock and some time was lost in trying to explain to them what it was. It seemed at that time that the Russians did not actually use interlocks on their safety systems. One would have thought that if they had read the Western literature then the word would have been familiar to them and at least they would have understood it. An Italian delegate asked the straighforward question as to whether keys or locked cupboards were used in their systems. The Russians seemed totally bewildered by this idea and clearly they had no idea of the use of mechanical means to stop operators from switching out certain safety systems when the reactor was operating.

John Taylor of EPRI read a short statement about control rooms. He said they had become more complex as time has gone by and we must be seeking ways of making them simpler. The control room should be seen as an aid to the operator and not as an enemy. He got very close to describing what we would call "expert systems" in the control room including such things as computerised information control which would highlight points of particular interest and the like. The answer came back from the Russians saying that they too were using computers but I think that they misunderstood the question because their answer was concerned with using computers to analyse plant rather than having it in the control room helping as an expert system. He did comment that they recognise the problem because for older plant the control bores were already saturated with instruments and the operators had a very difficult task to differentiate between important and relatively unimportant numbers. It was suggested that a comparison of the differences between national approaches could form part of future IAEA activities in this area. On operational aspects, again it was suggested that the IAEA could help but it was very important that the operators themselves were actually invited along to the meetings. Finally there was a suggestion that work already undertaken by the UNIPEDE group within the CEC in the area of operator actions could form the focus for any future work in this area. That was the end of the Thursday afternoon session of Working Group A, Session 1.

Working Group A, Session 2, Chaired by P Tanguy

The expert in this area was Tom Cress from Oak Ridge National Laboratories and the topic was radioactive release issues.

Cress reiterated the four stages of the release as he understood it to make sure that the Russians agreed with the interpretation. These 4 stages were:

1 The initial burst/explosion and the mechanical dispersion of fuel into the environment.

2 Release associated with the burning of the graphite moderator.

3 An increasing release due to decay heat being exacerbated by the insulation effect of the materials dropped onto the core.

4 The sharp reduction in release on 6 May, reasons for which were given as a combination of possible improved cooling of the core by Nitrogen and the production of more refractory chemical compounds with the fission products.

The releases were quantified from ground contamination levels measured in the 30 km region around the site. The composition of the fall out was very similar to that of the basic fuel itself, somewhat enriched in Iodine and Caesium, ie. the more volatile species. This meant there were two types of release in this initial burst, first the normal vapourisation of volatile species due to overheated fuel and second the mechanical release because of physical entrainment of particles. It is believed that the temperature did not get high enough to vapourise the actinides. Mechanical releases of this kind had not previously been studied in any detail because of the accident scenarios of other reactors did not call for them. It is an area where more research might be required.

It was interesting that there was a conjecture that the UO_2 might have been oxidising to U_3O_8 during the graphite burning part of the accident. The Russians seemed to confirm that U_3O_8 had been found. There was also a conjecture that the fuel had reacted with the graphite to form Uranium Carbide which could also cause aerosolisation. Cress then asked seven specific questions:

1. Is this picture of the 4 phases agreed by all? The answer to this was yes.

2. Where there any chemical composition measurements made on samples around the site particularly for oxygen and carbon? The answer to this was essentially no, they had done chemical composition for other compounds but not for those and that this process was still underway.

3. Are there any estimates of the chemistry of the very small fuel particles? The work that they had already done on chemical speciation showed that the chemical forms were highly differentiated and the chemical and physical transformations were directed to the non-volatile forms and this reduced the release. With the in-fill which had been put on top of the reactor many chemical forms of an unusual nature had been created and that these had still to be analysed.

4. Have aerosol measurements been made - size distribution, scanning electron microscopes, etc? It is very difficult to abstract whether an answer was obtained on this but they did say that the size of particles varied greatly from less than 1 micron to tens of microns. Research was still going on in particular into the graphite which was blown out but since this mainly came from the reflector areas, penetration by fission products was not great.

The amount of material mechanically disrupted from the core was given as the following:

Between .3 and .5% of the fuel was deposited on the site. Up to 20 km away from the site some 1+ - 2% of the fuel was found. Between 20 - 30 km, 1 - 1+% of all the fuel which was in the reactor. When you consider that this adds up to approximately 4% of the fuel and there was 200 tons of fuel in the reactor at the time, this means that some 8 tons of high burn up fuel, including Plutonium actinides and other fission products were scattered about the countryside. About 1 tone would have fallen on the site according to these figures.

There was some comment on Caessium 134 - 137 ratio with an average burn up of 10.3 MW days per tonne a ratio of 2 was found but this would be different in different parts of the reactor. This is a point that needs following up and I personally do not understand why the ratio is different when the half life of Caesium is the order of 20 odd years. There were several questions concerning $Carbon_{14}$, the Russians made a great play of saying that it was not a biological hazard but that was not the point of the question, the point of the question was whether they had measured $Carbon_{14}$ because it would give them some sort of handle on the mechanical damage processes in the graphite. The Russians never did catch hold of that question and it went by.

The topic then turned to stablilisation and entombment. Here Dana Powers of Sandia National Labs was the questionner. He gave a short introduction indicating that a great deal of work has been going on in the management of severe accidents, particularly in the area of core concete interaction. However, there is now a very pressing need to understand the Russian success in stopping the graphite fires and the value and mechanisms of the material which was dropped on top of the core.

There was no core concrete interaction in this accident he suggested because of the dispersal of the fuel and the air cooling. The Russians said that they had been developing refractory cements of the Magnesia and Aluminium Oxide type which compared with developments in the West for making concrete which would be capable of withstanding temperatures up to the melting point of the fuel. There was a suggestion that this was an area where there could be more exchanges of information.

Questions concerning the dropping of material onto the debris led to the Russians giving the following information concerning the details of what they did. Some 5000 tons of material were dropped on the reactor between 27 April and 10 May. The main body of this material was dropped between 28 April and 2 May. The clearly had legistic problems getting enough helicopters together but certainly they managed to do it. The first material dropped on to the core was 40 tons of Boron Carbide. This was done to make absolutely sure that the chain reaction had ceased. This was followed by 800 tons of Dollamite (I presume they mean limestone). The reason that this dropped was because the high temperature reaction of Dollamite is to give off CO_2 and this would act as a blanketing agent to stop the graphite burning. After this, 2400 tons of lead were dropped on to the core debris. The reason for this was that heating up the lead and particularly the latent heat of melting would take a lot of heat out of the core and as the lead melted it would run down to the bottom and thereby remove heat. The rest of the material was sand and clay which was put there to act as a filter for aerosols. Concerning the special tomb which was going to be built around the reactor as part of the recovery processes, there was additional questioning here but mainly it concentrated on the criteria which would be used to build the building. Much of this was the same as had been given in the previous accounts but there were one or two interesting new items. They said that the building constructed to withstand natural events to a probability of 10^{-4}. This is the criteria which is used normally for construction in this area against earthquakes, temperatures, winds, etc. The dose levels which were aimed for were 1 millirem/hour at the side walls and 5 millr/h on the top. This would be on completion of the building and one year later it would have decreased to a value of 5 times less than this. There was much further discussion concerning the criteria which came into play when they decided whether to go for open or closed ventilation systems but most of this is contained within the document anyway. The decision was of course to go for an open system. The main reason being Hydrogen control and the build up of activity in the air in the tomb.

There was considerable interest in what had happened to the spent fuel pond during the accident. This pond is there to store burned up fuel before it is taken off after a suitable cooling period. This is right along side the reactor and therefore could have made the accident worse by including its activity in the release. The emphatic answer came that it had not been damaged by the accident, was still full of water and was under control. There were only 100 fuel assemblies in it with about the same burn up as the rest of the core.

The workshop then went back to working group 1 under chairmanship of Brian Edmondson to reconsider the question of design and the safety basis of the RBMKs. Again Friscora from Canada was the appropriate expert. Questions here concerned the nature of the design basis and to set things going he gave quite a long description of how things are done in the West. One of the specific questions which arose was how was the margin of 30 equivalvent rods arrived at in their design requirements. Did they (a) shake the flux (b) determine the rate of addition of shutdown , (c) simply guarantee the amount of reactivity required to shutdown. The Russians answer that the main aim of the 30 rod margin is to guarantee the needed speed of negative reactivity insertion to offset the positive void coefficient. Thus, it needs to reduce the void reactivity at a rate of 1.5 beta per second.

Questions were asked concerning the scope of the accident analysis, how were design basis events chosen and how were they used to determine the sizing of the emergency systems. In particular how were the reactivity insertion rates chosen. The answer was that in the USSR evaluation of safety begins with events which might effect the unit as a whole. The range of events associated with failures of individual components are examined and then those which have the most important affect on the system are chosen and that second list is agreed with the appropriate inspectorate bodies. This process is the same for all reactor systems, not just RBMKs. The list includes various sorts of events. It includes reactivity insertions, failure of equipment. the effect of the stopping of one or two turbo generators, the failure of main circulation pumps, the non-closing of one of the check valves and loss the the house electricity load. Also looked at were groups of failures concerning the breach of pipework including water, steam and steam/water pipes, fuel channels, feedwater pipes, group distribution headers and downcomers. The size failure was up to 90 cm in diameter. Seismic effects were taken into account in the design and fires in any of the machine rooms of the plant were also in the list. This list is then approved and agreed. From it a volume of technological justification for safety is produced. All initiating events are mentioned in this volume. From that volume is then produced working documentation.

Applications of probabilistic methods were discussed and the Russians said that as they gathered more data, data banks were being set up and probabilistic assessments had been made for the RBMK 1500 reactor. In answer to the question concerning the nature of the quantitative safety criteria in use, one example was given and that was that the reactor protection system had to be reliable to 10^{-7} per year.

This was the end of the workshops and the summing up indicated that the generic nature of human factors had been brought out as most important, a point that surprised me since Brian Edmondson made it and that is hardly beneficial to the position that we had all discussed previously. The Russians have admitted that it was a combination of human error, plant design and applying management which had led to the accident and we all believe that it is very important to maintain the combination of human error and design error at the same time. Human error of course applies to all plant everywhere whereas we believe this particular accident could only have occurred on an RBMK reactor even though, of course, there was a strong element of human failure in it.

Notes from the Friday Session, Final Plenary Meeting

Brian Edmondson began by saying the usual things about exchange of information and a few platitudes about how good it was of the Russians to tell us all this stuff. He then summarised that the accident had been due to the following 4 features:

a. Inadequate written procedures.
b. Lack of authorisation
c. Lack of permission to proceed
d. Attitude of the staff

All these strongly being focussed on the human factors element. He went on to say that there was little doubt about the basic issues. All the experts were agreed on the basis nature of the accident, its cause and its course. Naturally the details will have to await further examination but that can only be done in the fullness of time.

A large question had been raised concerning operator performance, lines of authority, etc. The accident did not arise solely from fault procedures. The staff lost their sense of danger and their previous high performance rating could have been a contributing factor.

Edmondson introduced the idea that we needed to have a nuclear operating safety culture and this was one of the lessons of the Chernobyl accident. There were questions concerning alternative routes forward, for example, training and appreciation versus penalties as means of getting operators to act properly.

He emphasised that the information flow to operators was most important and they could be swamped with unimportant detail. He suggested exchanges in the area of the man machine interface with possible collaboration with UNIPEDE but with a special plea that the operators should be represented in these exchanges.

The whole area of training of operators was highlighted as a possible area for international exchange.

The design factor in all this was important and he welcomed the fact that the known design deficiencies were going to be corrected in the future. Personally I am not sure that the go far enough or fast enough or whether the RBMK reactor can ever be made as safe as the thermal reactors that we operate.

In the future he said that the possibilities for answering the very large number of questions through other channels would be investigated and that the highlights of this workshop had included the interaction between design and operators, the needs for operator training and operator self-awareness in particular and all this revolved around his idea of a safety culture.

Pierre Tanguy then summarised working group A, session 2. His first remarks said that in addition to the now well known violations of safety practices there was also a very trivial human operator error burried in all this accident sequence and that was the failure of the operator to set the automatic regulator when the power was reduced. I am not quite sure why he chose to open up with this remark as it was rather out of context. He then proceeded to reiterate much of what we had heard already concerning the nature of the explosion, that it was due to a reactivity insertion of about 1.5 beta in less than 1 second (beta = 0.5% at that time). The control system did not act for at least 6 seconds and the explosion included about 30% of the channel volume in the lower part of the core. The summary of the reactivity issues was that he felt that there was a need for benchmarking of reactivity and thermal hydraulic codes against this accident. He then discussed in some detail the measurements being made on the debris using the instrument packages and that supplementary information on this had been made available. This is reported in the notes of that session.

Nine issues were identified as having been covered by this working group.

1. Core behaviour in the first seconds of the explosion - fuel disruption.
2. Fuel relocation after the explosion.
3. Graphite behaviour.
4. Fire fighting
5. Damage in the spent fuel storage pond
6. Behaviour of Unit 4 personnel
7. Mechanisms associated with radioactive release.
8. Stabilisation of the core
9. Prospects for entombment

He then went on to discuss some of these in more detail, particularly the fire fighting aspects and highlighted the proposals for action in this area. This included the development of international codes and standards, the development of technical means for fire fighting such as fire fighting robots, lifting mechanisms, the design of light protective clothes for a radioactive environment, etc and call upon the IAEA to organise a conference on fire protection, particularly for nuclear power plants.

He then went on to discuss in some detail the phenomena associated with the accident. He said that the phenomena were very complex and we were still at the preliminary stages of evolving our understanding of it. There was some question in his mind as to whether the second explosion was due to hydrogen or to a second power excursion. There seemed to be some disagreement with the Soviets on this point but it was seen to be not very important. The first explosion of course was reasonably well understood, the time constant for the energy input was of the order of 1 second, this is relatively low as far as reactivity insertions are concerned, but because it was a very large core indeed and there is nowhere for the energy to go other than heating up the fuel very quickly indeed.

He reiterated some of the details such as the fact that about 25% of the graphite had been burned, that is 250 tons, some was blown out of the reactor altogether and that there was general agreement on the 4 phases of the release as described. He indicated the interesting fact that it seemed that UO_2 had been oxidised to U_3O_8 during the accident. He emphasised the efficiency of the materials which had been dropped on to the core and highlighted the importance of accident management in this context. His recommendations were for exchanges in the internation community on:

1. Fire protection
2. Experiments and analytical research on severe accidents
3. Code validation, including parametric studies using the data for the initiating events leading to reactivity insertion
4. Chemical and physical aspects of release

In his conclusions he said that we must always remember that the first safety priority as accident prevention but we still have to look at mitigation and this led the research in source terms back into the forefront. Next Raboult introduced the work of Working Group 3 which included atmospheric dispersion, decontamination and emergency measures. He said that there had been very lively discussions in his group, particularly on decontamination methods and the handling of wastes arising. A great deal of very useful and valuable information had been gathered concerning the efficiency of decontamination processes, this could be very efficient if the surfaces were smooth and not absorbent, but that for surfaces in which the Caesium is fixed at the surface chemically are very much more difficult to decontaminate. He believed that it would be very fruitful to have further international collaboration on experience in decontamination as the Russians had had so much perhaps unfortunate, experience themselves. On the question of waste disposal a simple answer was that they had collected all the water in large tanks and chucked all the solids into a large hole they happened to have on the site.

In emergency measures, he mentioned 4 topics of importance:

1. Evacuation
2. The use of Iodine to block thyroid
3. The emergency reference levels used
4. The emergency response levels

He described the criteria for evacuation at 45 rem whole body minimum to 75 rem maximum, ie. above 75 evacuation was compulsory, below 75 it was not necessary. In comparison he said that in the UK the upper limit was 50 - 100 rem and the lower limit was 10. It was interesting that Iodine tablets had been widely handed out and that little side effects had been noticed. The predicted doses to the population were greater than 75 rem hence emergency evacuation was undertaken. He made the very strong point that it required very strict discipline and central organisation to integrate all the requirements in the evacuation process which included tens of thousands of cattle. For the Russians to say they needed strong central authority must indicate that it was a very difficult problem indeed. He also emphasised the multi-disciplinary nature of the requirements for organising such emergency procedures. Finally he said that confusion in Europe over emergency reference levels for interdiction of foodstuffs shows the need for the development of a national scientific basis for these levels. He indicated the IAEA should take on work in this area and to harmonise with what is going on in the CEC. He did mention that this would be an extremely difficult problem.

The final Working Group 4, chaired by Benninson was concerned with meteorological problems, dosimetry and medical aspects. The discussion covered 6 basic topics:

1. The estimation of source terms from the dose results

2. Movement of nucleides in the environment

3. Calculation of doses to the public

4. Medical and biological state of the people on the site, ie. non-stechastic effects

5. The use of Iodine in blocking thyroid cancer

6. For the future, the requirements to consider stechastic effects and the epidemiology associated with it. In particular the following 4 recommendations were made:

1. A mechanism was needed for expanding the area over which environmental data was considered, in other words the fallout data from Europe and Russia should be put together better to account for dispersion in the air, the role of food chains and dispersion in the various water environments. The invitation seemed to be to the IAEA to take this on.

2. The need to improve assessments of collective dose both internal and external. There was a possibility that there could be collaboration here between the IAEA and the UN committee on radiation doses.

3. That there should be an international workshop probably in collaboration between the IAEA and the world health organisation to select methodologies for monitoring the epidemiological studies which would follow from the examinations of the large numbers of people who had received low doses.

4. We should share information on the non-stechastic effects for the people who received very large doses on site including the special schemes devised for medical treatment and the theraphies associated with them.

That was the end of the summaries by the Working Group Chairmen, the Chairman of whole conference Rometch, then made his summary. He went over much of the ground again but he did usefully provide a list of 13 areas where recommendations were being made to the IAEA for further international effort. These were:

1. Severe accidents, their physics and chemistry and evolution
2. The man machine interface, particularly things like information flow, ergonomics, etc.
3. The balance between automation and human action in control room situations.
4. Exchange of experiences on operator training and practices