September 13, 1983

SECY-83-374

POLICY ISSUE
(Information)

For: The Commissioners

From: William J. Dircks
Executive Director for Operations

Subject: FOREIGN DEVELOPMENTS OF INTEREST TO NRC

Purpose: To report the results of a staff study on agency problems in covering foreign developments of interest to NRC.

Discussion: In response to direction by the Commission to the staff in a memorandum from Samuel J. Chilk to William J. Dircks on August 13, 1982, the staff has conducted a study that addresses agency-wide problems in covering foreign developments of interest to NRC. The results of this study are fully described in Enclosure 1.

The Study finds that NRC has an extensive network of cooperation on international nuclear matters involving the exchange of technical reports, joint participation in foreign research projects, and representation in the technical activities of international organizations. The staff has been involved in a number of foreign safety activities that have been measurably valuable to our domestic regulatory responsibilities. These include assessments of control rod guide tube pin failures and problems with Westinghouse Model D steam generators, planned modifications of 10 CFR Part 20, participation in the Kuosheng (BWR/Mark III) test program and evaluation of the UK Sizewell B reactor design. These activities are discussed in detail in Enclosure 2.

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Additionally, the study notes that:

- Foreign incident data are received, reviewed and utilized by NRC in a timely and efficient manner.

- AEOD is currently investigating methods to make the foreign event data file more accessible to other NRC offices and to incorporate important foreign events into the sequence coding system.

- Current effort is adequate to keep informed on foreign developments in waste management and fuel cycle safety.

This study concludes that, while the agency receives a large amount of useful safety information from foreign documents, foreign travel and visits by foreigners to staff offices, documentation of important safety points identified at meetings and distribution of this foreign information have been irregular, and future utilization of foreign information in the NRC reactor regulation process should be improved. As foreign reactor designs continue to advance and become more independent of U.S. practice, the insights developed become more valuable to the NRC and merit greater attention.

In contrast to the safety area, an extensive and productive exchange of safeguards information has not been developed with other foreign countries, and the prospects for doing so are not promising. These safety and safeguards situations are fully described in Enclosure 1.

In conducting the staff study, a number of improvements were identified that should help to remedy existing problems and further stimulate the consideration and use of foreign information in the agency. These improvements are:

(1) NRR will develop procedures and responsibilities for the review and utilization of foreign safety information provided by IP or RES. The procedures and responsibilities will be specified based on experience with the review of the Sizewell B reactor. NRR will conduct, as necessary, analyses and follow-up to determine the application of foreign research results and design improvements to U.S. regulatory practice;

(2) IP will attempt to develop and implement a provision to exchange safeguards information in future and renewal international arrangements;

(3) IP will coordinate the preparation and distribution of trip reports for all foreign travel and summaries of all meetings with foreign delegations held in the U.S.;

(4) IP will centralize the receipt and distribution functions for all foreign documents received through our formal exchange and cooperation arrangements;

(5) IP will automate the indexing and searching of its collection of foreign documents to facilitate the retrieval and use of such information.

I am directing that items (1) and (2) be implemented immediately. Items (3) through (5) will involve new or increased activities for IP. Because of resource limitations for FY 1984, these three improvements will gradually commence in FY 1984, but full implementation will take until FY 1985.

I believe that implementation of these actions will result in substantial improvments in staff awareness and use of information relating to foreign developments.

William J. Dircks
Executive Director for Operations

Enclosures:
1. Staff Review of Use of Foreign Information
2. Examples of Important Foreign Safety Activities
3. Bimonthly Report from Kernforschungszentrum, Karlsruhe
4. CSNI Working Group Representatives

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ENCLOSURE 1

STAFF REVIEW OF USE OF FOREIGN INFORMATION

Background

NRC now has active regulatory information exchange and cooperation arrangements with most countries that have substantial nuclear power programs, either planned or in being, a total of 21 countries, including France, the Federal Republic of Germany, Japan, and the United Kingdom. Through these arrangements, we have established a formal channel for communications with foreign nuclear regulatory organizations to obtain prompt and reciprocal notification of reactor safety problems and developments that could apply to U.S. nuclear facilities, thereby forming a network for bilateral cooperation on nuclear safety and safeguards. These arrangements typically call for exchange of regulatory information in technical reports and newsletters and joint meetings focusing on nuclear safety problems.

In addition to our regulatory cooperative arrangements, NRC has concluded over 30 separ agreements in the area of nuclear safety research. These include broad bilateral exchanges of safety research information, research participation agreements, and participation in international research projects such as the Halden project in Norway and the Marviken project in Sweden. In addition to the exchange of information by reports and meetings, in some cases temporary assignments of NRC contractor personnel are made to foreign research programs and facilities.

As an outgrowth of some foreign interest in exchanging generic physical security information, the staff is developing agreements to exchange classified safety and safeguards information with other countries. One such agreement has been concluded with the FRG and similar agreements are under negotiation with Switzerland and the UK.

NRC participates in the safety-related activities of two international organizations, the IAEA and the NEA. We serve on many committees and working groups dealing with reactor safety, operational experience, radiation protection, spent fuel storage, waste management and safeguards matters. The staff has provided safety advice and assistance to regulatory and safety authorities of countries embarking on nuclear power programs and has continued its lead role in the IAEA nuclear power reactor safety standards program.

The staff's involvement in foreign activities included 198 separate trips to 23 foreign countries during the past fiscal year. In the same period, NRC has also funded foreign visits of 161 contractor personnel to exchange safety information. Additionally, NRC held nearly 200 policy and technical meetings in the U.S. with foreign delegations and individuals totalling nearly 500 visitors.

Each year, the RES staff organizes and hosts a Safety Research Information Meeting that addresses all aspects of NRC-sponsored LWR research and that is well known world-wide. Typically, for most technical areas, presentations are made by foreigners of their work that complements or relates to our programs.

Since the meeting is held locally, it provides a regular and convenient opportunity for large numbers of the staff and ACRS to become familiar with, and keep abreast of, foreign safety developments.

Over the years, NRC has benefitted from the knowledge of foreign incidents, plant operating experience, safety developments and research. Currently, the staff is involved in a number of topics with direct application to our domestic reactor program. Some of these topics include control rod guide tube pin failures, problems with Westinghouse Model D steam generators, modifications of 10 CFR Part 20, NRC participation in the Kuosheng 1 (BWR 6/Mark III) test program, and the UK Sizewell B reactor design (similar to SNUPPS). Each of these activities is described further in Enclosure 2. These examples are intended to illustrate that the staff has been and is actively involved in a wide range of activities with foreign countries, the information from which is directly applicable to current domestic safety issues.

As mentioned in the memorandum to Chairman Palladino from W. J. Dircks reporting on the recent NEA CSNI meeting, the Commissariat a l'Energie Atomique (CEA) of France has extended an invitation to assign an NRC engineer at one of their facilities. Because of the active French reactor program in Westinghouse LWRs, and their growing involvement in technical safety developments, the staff is giving consideration to making such an assignment.

The staff finds this extensive and active program of sharing information and results, exchanging technical experts and participating in international research projects to be a technically beneficial means of contributing to overall nuclear safety.

COLLECTION OF INFORMATION

To assess systematically the staff's activities relating to keeping informed of foreign safety developments, the following discussion is organized into the collection, documentation and dissemination, and utilization of foreign safety information. Under utilization, our practices are assessed according to the various functional activities performed in the agency.

By means of NRC's cooperative arrangements with foreign countries through our regulatory and research agreements and our participation in international organization activities, we have access to a large amount of foreign safety information. During the past year, we have received over 800 foreign documents through the agreement program. Information is also received on a person-to-person basis such as during our 198 foreign trips and the 200 meetings that NRC hosted. Regular exchange meetings between the ACRS and their counterparts in the FRG, France, and Japan, that have resulted in voluminous reports of important safety activities, are included in these trips and meetings. Information is also received through our participation in activities of international organizations and our attendance at international meetings such as those sponsored by our contractors, the American Nuclear Society, the European Nuclear Society, etc.

Firsthand, specialized knowledge is regularly obtained and reported by our 11 contract representatives located at 8 different foreign research establishments. NRC representatives are located at GRS-Munich, Materialprufungsanstalt-Stuttgart and Kernforschungszentrum Karlsruhe in the FRG; Cadarache C.E.N. in France; Halden Reactor Project in Norway; Tokai Research Center in Japan; Joint Research Center of the European Communities in Italy; and Winfrith Atomic Energy Establishment in the UK. They monitor and directly participate in activities covering a wide spectrum of technical safety topics such as LOCA thermal/hydraulics, fracture mechanics, fuel behavior under accident conditions, human factors engineering, core melt phenomenology, etc. An example of this type of in depth reporting is provided in Enclosure 3.

Finally, foreign safety information is available to the staff from the NRC Library. Although the number of foreign books, journals, and reports is limited, summaries of materials in the published literature are widely available because of the worldwide coverage of various data bases such as Energy and Nuclear Science Abstracts. Documents of interest that are identified by scanning the abstracts can then be procured.

Unlike the safety area, an extensive exchange of safeguards information has not been developed by the staff with countries having substantial nuclear programs. The exchange of safeguards information from technical reports and documents has been minimal, and foreign in-country visits for physical protection reviews seldom provide significant material on the safeguards activities of the countries.

Based on our study, the staff concludes that the agency receives and has at its disposal a large volume of information on foreign safety developments. Furthermore, because of our wide network of both formal and informal channels of communication and the responsiveness of these contacts, we are reasonably confident of obtaining current data if we request it. However, very little information on foreign safeguards developments is available. Actions that will attempt to improve this situation are discussed later on page 15 of this enclosure.

DOCUMENTATION AND DISSEMINATION OF INFORMATION

The documentation and dissemination of foreign information within NRC are not as effective as the collection of information. Most foreign documents are received by the agency via either IP or RES. Documents received in this manner are distributed to interested offices, and all documents received are announced in the Weekly Information Report.

Although trip reports are encouraged by the NRC manual for foreign travel, they are not consistently submitted. Also, few written summaries of the 200 meetings hosted by NRC in the past year were prepared. Documents are often exchanged at these foreign and domestic meetings, but they are neither widely distributed nor added to the foreign document files. Consequently, foreign information received in both foreign and domestic meetings is not regularly documented or distributed.

Over the past few years, the number of documents being handled has been increasing, and with the addition of trip reports and meeting summaries, the indexing and retrieval of these materials will soon grow beyond manual capability. As contractor to NRC, the Nuclear Operations Analysis Center (NOAC) of Oak Ridge National Laboratory reviews and abstracts foreign research and operating incident reports, and files the abstracts in automated data bases. For the future, we will need to utilize some form of automated indexing and search methods for all foreign information reports that is similar to the process currently used at NOAC. The staff is investigating accomplishing such automated processing within NRC, but depending on the outcome this task may have to be contracted out.

Presently, the receipt, storage, filing and distribution of foreign documents are done both by IP and RES. RES receives directly foreign documents sent to NRC as part of our research exchange program, and IP receives all other documents. Although the collection and distribution of these documents is deemed adequate, RES does not believe that they are performing these activities as vigorously and regularly as desirable. Furthermore, RES believes that it would be better to centralize these collection and distribution functions in IP because IP is organized to perform these activities more effectively, and the various staff and Commission offices normally associate foreign documents with IP rather than RES. In considering this concept of document centralization, the other program offices agreed that it is more consistent and desirable from a use standpoint to centralize these activities in IP.

A number of improvements, which are listed below, have been identified to correct the deficiencies mentioned above and to extend the automation process to all foreign report activities. All of these specific improvements involve new or increased activities for IP. Because of resource limitations in FY 1984, these improvements will gradually commence in FY 1984 with full implementation taking until FY 1985.

The specific improvements to be adopted are:

1. IP will monitor and coordinate the preparation of trip reports for all foreign travel and will ensure wide distribution of completed reports. In the interim, Office Directors will continue to oversee and be responsible for preparation of trip reports for staff foreign travel.

2. IP will develop procedures to institute a process of documenting important matters from all meetings with foreigners held in the U.S., and will monitor this activity, also. In the interim, steps will be taken to require the staff to prepare summaries of all meetings with foreigners held in the U.S.

3. IP will automate the cataloging and searching of foreign-related reports to facilitate the retrieval and use of such information.

4. The collection, announcing, distribution and storage functions for foreign research reports will be transferred from RES to IP. By taking this action all foreign report activities will be centralized in IP.

UTILIZATION OF INFORMATION

Incident Information

A major source of foreign power reactor incident information for the staff is the Incident Reporting System (IRS) of the NEA. This System is the result of an NRC initiative presented initially to the NEA in 1979. Since then, it has developed into a regular and reliable system of exchanging international incident data. Since 1980, over 270 separate reports have been exchanged via this system. In addition, the NRC receives a large quantity of foreign incident reports under the provisions of our bilateral agreements. These documents are received from countries and areas that do not belong to NEA, such as Korea and Taiwan.

In order to process this large quantity of information, AEOD has established a program at NOAC to screen and catalog foreign incidents and operational experiences. This program is designed to screen all foreign operational experience information received by the NRC since 1975 to select events that are significant and relevant to U.S. LWRs. NOAC indexes and abstracts the significant events into a protected, computerized file which can be searched using keywords and coded fields. AEOD is currently investigating methods to make this file more accessible to other NRC offices, as well as methods for incorporating important foreign events into the sequence coding system.

In addition to the program described above, incident information is routinely reviewed and evaluated by AEOD, IE, NRR, and NMSS when appropriate. Problems with control rod guide tube pin failures and the Westinghouse Model D steam generators, which are discussed in more detail in Enclosure 2, are examples of the significant value foreign operating information has been to the NRC staff. Consequently, the staff concludes that foreign incident data are received, reviewed, and utilized in a timely and efficient manner.

Research and Standards

The Office of Research is a party to over 30 foreign cooperative agreements varying from broad information exchange arrangements to direct participation in specific research projects. Some of these collaborations, such as the Coordinated Analytical and Experimental Study of the Thermohydraulic Behavior of ECC (2D-3D) with the BMFT in Germany and JAERI in Japan, and the Severe Fuel Damage program actively under negotiation with many foreign countries, are intended to result in multi-million dollar savings of NRC research funds.

In administering this important program, large numbers of foreign trips for both staff and contractors occur annually and large numbers of foreign research documents are received. Over the years there has been effort on the part of the staff to become knowledgeable about safety research projects throughout the world in order to participate wisely in joint projects and to initiate and guide NRC efforts in a complementary manner with programs underway in foreign countries. This practice has been reinforced by Commission guidance over the past few years. In the research area, there has been a continuing awareness and utilization of knowledge of foreign safety research programs and recognition of the value of cooperative activities.

A systematic review and cross-comparison of worldwide regulations, standards of practice, and guides used in regulation were made as a major part of the IAEA Nuclear Power Plant Safety Standards (NUSS) Program. Technical review committees working under the guidance of a senior advisory group were established in each of the main areas of the Program. Within these broad areas, approximately 60 topics were addressed. The topics covered the whole range of nuclear power safety regulation. NRC participation included membership on the senior advisory group and on the technical review committees. These reviews followed a detailed procedure to ensure that all relevant national practices and standards-related documents were considered in preparing IAEA safety guides. All appropriate national documents were collected, collated, and supplemented by questionnaires. The material was reviewed in depth by working groups of experts, differences identified, and their importance assessed. In turn, the results of the work group efforts were reviewed in depth by experts of the participating countries.

In the course of these systematic comparisons, potentially significant differences between countries were sometimes identified in evolving public policy areas. Two areas of emerging U.S. policy divergence were seen to be in radiation protection and reactor siting. Our participation in the IAEA-NUSS program and our interfaces with the International Commission on Radiation Protection (ICRP) showed us that other countries were being more explicitly responsive to ICRP 26 recommendations. This contributed to the staff decision to revise

10 CFR Part 20 to be consistent with ICRP 26 and helped decide the approach taken to doing so (see Enclosure 2). On siting policy, the NRC Siting Policy Task Force Report approached the relationship of demographics and engineered safety features in a new formulation which was not well understood in other countries. This led to international discussions at the technical and policy levels to explain the U.S. approach, to determine how different it was from prevailing practices and to help U.S. and foreign regulators determine whether policy adjustments were necessary in future national siting regulations. Review of siting policy in the U.S. has now been deferred pending resolution of the source term reevaluation.

Reactor Regulation

An examination of NRR's past response to important foreign safety information has led to the following general conclusions. Response to foreign operating experience applicable to U.S. plants has been quite good. The NRR staff has responded rapidly, assessing the generic impact on domestic plants, initiating internal studies, negotiating exchange agreements, and working closely with foreign parties to arrive at technical solutions. The application of foreign research results and design improvements to U.S. regulatory practice has been more uneven. When there is knowledge of an existing U.S. problem, foreign information is incorporated easily and rapidly; when there is no such knowledge, foreign developments receive less attention.

Use of foreign information requires resource commitment, not only the manpower and travel funds to collect information, but the manpower to review the foreign information and to provide analyses and follow-up. Much of the valuable information gathered by NRR staffers on overseas trips frequently had only a peripheral connection with the original purpose of the trip. The amount of timely, useful foreign information obtained is a function of the amount of direct foreign contact by technical staff members. Many time-sensitive issues do not appear in formal reports or documents for a period of months or even years. Such information is available much earlier if one knows it exists, and its existence is frequently identified by informal, personal contact.

Understanding and utilizing foreign information on issues of regulatory policy require more than a review of foreign literature. Insight into foreign safety policy requires continuing personal contact. Some of the hardware requirements of a regulatory policy are obtainable through document exchanges. However, it is difficult to work from hardware modifications or additions to the safety policy on which the hardware is based. Likewise, written foreign regulations describe the tangible consequences of applying regulatory policy, but not the fundamental tenets underlying the policy.

In licensing the Sizewell B plant, the British are reviewing and modifying the U.S. SNUPPS design. This situation provided a unique opportunity to examine the conclusions and thought processes of a totally different group of competent engineers in reviewing a PWR plant design of the type used in the U.S. As described in Enclosure 2, NRR has recently completed a major analysis of the British design features.

NRR plans to develop a formal, internal system that will assign responsibility for following specific foreign developments. As described above, foreign information related to well-defined, actively pursued technical areas is usually handled well. However, issues not clearly identified within a single technical specialty or not associated with ongoing work or issues tend to be pursued less vigorously. For these issues, a central point of identification, assignment of responsibility, and follow-up will be made as necessary.

NRR will assign a point of contact for all foreign safety development activities. Based on the experience of the Sizewell Review, future NRR procedures and responsibilities will be developed for the review of foreign safety information provided by the receiving offices, IP or RES.

Recent changes in NRC's representation and participation in the OECD Nuclear Energy Agency (Paris) have served to involve NRR more closely in international technical activities on reactor safety. The NEA Committee on the Safety of Nuclear Installations (CSNI) has always had a very active program of safety cooperation and information exchange, but that program has been oriented towards safety research. In the last year, CSNI reorganized its program under five principal working groups. In order to assure broader staff involvement and access to CSNI information, members of NRR are now the lead representatives of two working groups, and the backup representatives for the remaining three groups (see Enclosure 4). These assignments, combined with W. Dircks' chairmanship of the CSNI, D. Eisenhut's participation in the CSNI Licensing Subcommittee, and other NRC participation at the CSNI parent committee level should help assure an informative and productive NRR involvement in CSNI.

Physical Security and Safeguards

As required by the Nuclear Non-Proliferation Act for the export of U.S. nuclear materials, the staff keeps current on international physical security matters through foreign country visits and interchanges. Additionally, the review of threat data provided by Executive Branch Agencies is analyzed on a continuing basis for possible relevance to domestic protection activities.

Over the years, the staff has informed the Commission of the difficulties in obtaining information on international safeguards and states' systems of accounting and control for nuclear material. A good summary of the history in obtaining this type of information was provided in SECY-81-170, which addressed Criterion 1 Determinations in Export Licensing. As noted, there are institutional and practical difficulties in obtaining some types of information, and where information of a generic nature has been obtained, it has been of limited value.

Nevertheless, the staff will undertake some limited measures to attempt to enhance our abilities to obtain safeguards information. Presently, some of our cooperative, bilateral arrangements contain a provision for the exchange of information on regulatory procedures for the safeguarding of nuclear facilities. For future cooperative arrangements and as renewals of our existing arrangements come due, IP will offer to include a provision for the mutual exchange of such safeguards information. Informally, we have been advised by the State Department that they see no problems with such a change to the scope of our cooperative arrangements.

Because of the sensitivity of some physical security and safeguards-related information and material, certain types of information are not widely disseminated throughout the staff and are limited in use to relatively few individuals with a need to know.

Waste Management

The primary means of following foreign safety developments in the waste management area is through participation in both IAEA and NEA programs. The staff regularly participates in the Radioactive Waste Management Committee of the NEA and the Committee on the Code of Practices for Low-Level Waste Management at Power Plants of the IAEA. Also, foreign participation is regularly included at NRC-sponsored symposia such as those on Site Characterization and Monitoring for a high-level waste repository, and Facility Design, Construction, and Operating Practices for low-level waste disposal sites. In the past year, Canadian, French, and Philippine representatives attended these symposia.

Another source of information on foreign developments in the waste management area is the International Source Books compiled by Pacific Northwest Laboratory. These books outline the status of national waste management programs in many foreign countries. Finally, information on international nuclear waste programs is routinely obtained from the exchange of documents, from visitors to NRC, and from foreign trips by the staff.

The staff believes the current levels of effort are adequate to keep informed of foreign developments, and they should be continued in the future.

Fuel Cycle

The staff participates in a number of fuel cycle international activities, including the NEA Committee on Radiation Protection and Public Health, IAEA experts' groups on transportation, irradiated fuel storage, global environmental monitoring, and ICRP Committee 4. To the extent practicable, the staff visits foreign fuel cycle facilities of special interest. For example, during the past year a visit was made to the British Nuclear Fuels Limited facility at Sellafield, UK, to view and discuss BNFL technology that will be incorporated into the Westinghouse fuel fabrication plant in South Carolina.

Based on our past experiences, the staff concludes that we are not missing information on significant foreign fuel cycle safety developments.

ENCLOSURE 2

EXAMPLES OF
IMPORTANT FOREIGN SAFETY ACTIVITIES

Westinghouse Steam Generator Preheater Degradation

In October 1981, Duke Power informed the staff of problems with Westinghouse Model D steam generators that had been identified at a foreign plant and were applicable to the McGuire plant. The problem was associated with tube wear and eventual failure in the preheater section, apparently caused by flow induced vibration. Concurrently, IP began to obtain relevant operational data for the Ringhals 3 plant in Sweden and the Almaraz 1 unit in Spain. Subsequently, the staff has participated in tests and evaluations of possible steam generator modifications being performed in Sweden. Additionally, the staff has assisted the Yugoslavian authorities in evaluating a fix proposed by Westinghouse that is being evaluated at the Krsko reactor. Staff members have visited and met with representatives of the utilities and regulatory agencies for each of these power plants.

Based largely on this foreign experience and data, NRR and Duke agreed on a test program that allowed continued operation at McGuire. Subsequently, both McGuire 1 and Summer completed steam generator modifications along the lines suggested by the European experience and returned to power. A number of other U.S. PWRs are undergoing or plan to undergo similar modifications.

Control Rod Guide Tube Support Pin Failures

The first two support pin failures occurred at Japanese facilities; the first at Mihama 3 in 1978, and the second at Ohi 1 in 1980. Purportedly, these failures had limited applicability to U.S. plants because the support pins were neither designed, manufactured, nor supplied by Westinghouse. Subsequently, the Japanese investigated support pins at another plant. For the first time, this investigation revealed stress corrosion cracking in Westinghouse-supplied pins. However, since the cracks were small and only found in a few pins and since no pin failures had been experienced in domestic plants, the phenomenon was considered, again, to have limited applicability to domestic plants.

Between March and September of 1982, additional support pin failures occurred at four French facilities. In contrast to the failures experienced in Japan, the French failures involved support pins supplied by Westinghouse. The loose part associated with the Gravelines incident was found in a check valve that permits water to flow from an accumulator to the reactor core in the event of a LOCA. The loose part associated with the Fessenheim incident did considerable damage to a steam generator. Although the broken part consisted of the bolt section and attaching nut, only the nut was found.

On May of 1982, the acoustic monitors at North Anna 1 detected loose parts in two steam generators. The noise detected by the acoustic monitors was initially attributed to loose steam generator tube plugs by both Westinghouse and the licensee. However, IE detected similarities between the North Anna Unit 1 event of May and the earlier Fessenheim event. Consequently, IE queried both Westinghouse and the licensee regarding the possibility that the North Anna event was due to a support pin failure. Although the initial responses were that such likelihood was very remote, it was later determined that two support pins had indeed failed at North Anna 1. As at Fessenheim, only the nuts were found.

IE convened a meeting with Westinghouse on June 2, 1982, to discuss the safety implications of these failures on domestic plants and issued an Information Notice on July 23, 1982. Subsequently, the Atomic Safety Licensing Board Panel and Appeals Board Panel were notified of the safety implications of support pin failures.

More recently, the staff has scheduled a meeting with Westinghouse and the Westinghouse Owners' Group. The purpose of this meeting is to obtain information from Westinghouse regarding the safety implications and generic aspects of the support pin failures at North Anna Unit 1 and the units in France. Also, IE is reviewing the more recent foreign reports on this subject to keep abreast of the corrective measures being implemented in France and Japan.

Participation in Kuosheng 1 Test Program

In 1981, the Taiwan Power Company (TPC) conducted inplant safety relief valve testing (SRV) at their Kuosheng 1 plant. Kuosheng 1 is the first BWR 6/Mark III NSSS/containment design to become operational. NRC participated in these tests including review of the test program plan and assistance in preparation of the test procedures. This test was specially instrumented, and the utility added measurements to the normal test program at NRC's request. The entire test program was primarily funded by TPC with very little cost to NRC for our participation.

The test at Kuosheng provided fullscale, confirmatory testing of a reactor plant about to become operational. The results of these tests contributed directly to resolving the many concerns NRC has had regarding hydrodynamic loads associated with the suppression pool design of a Mark III containment; they were directly utilized in the resolution of task action plan A-39, "Unresolved Safety Issue on Safety Relief Valve Pool Dynamic Loads." The testing confirmed the adequacy of the GE methodology to calculate and predict loadings on structures and equipment. Also, it provided vibration measurements of piping inside the containment during SRV actuation. Finally, the tests measured the dynamic response of electrical and mechanical equipment and components due to SRV loadings. Although the test data are still being evaluated, we can say that these tests contributed substantially to our understanding and acceptance of the Mark III containment design for operation in the U.S.

Development of Revisions of 10 CFR Part 20

A substantive revision of NRC standards for protection against radiation, 10 CFR Part 20, is underway currently. As drafted, the rule will be quite different in approach from the current rule utilizing new methodology, updated biology and physiology, and an expanded exposure data base. In connection with this revision, NRC is benefiting from participation with the ICRP, the IAEA, and the NEA. The ongoing activities of these organizations have contributed to obtaining a sound understanding of the principles and issues involved. ICRP publications 26, 27, 30, and 32 are being directly utilized in preparation of the revised rule. The staff believes that our international activities in radiation protection influenced the current approach to protection and has been very valuable in the planned reshaping of our regulations.

Sizewell B Reactor Design

Sizewell B is the first PWR planned for construction and operation in the United Kingdom, The Sizewell B reactor plant is basically a SNUPPS plant; i.e., it consists of a Westinghouse four loop nuclear steam supply system and a Bechtel balance of plant layout. However, modifications and adaptations of this basic design are being made for installation in the UK. Some of these modifications include installing additional safety systems. Because the UK has the opportunity to take a fresh look at a current U.S. design, the staff believes it is valuable to know and understand the UK design and the design bases.

In late September 1982, two members of the staff visited the UK and held a series of discussions with the various UK parties involved in the Sizewell B effort. Discussions were held with representatives of the Central Electricity

Generating Board, the national utility; the UK Atomic Energy Authority, which is providing specialized guidance on specific technical topics such as probabilistic risk assessment; the Nuclear Installations Inspectorate, the regulatory body in the UK; and the National Nuclear Corporation, the architect engineer and constructor in the UK. Based on the results of these preliminary discussions, the staff sent a team of technical specialists to the UK in December to discuss in depth specific areas of interest to the NRC. A report stating the findings of the staff evaluation (NUREG-0999) was published, and subsequently discussed with the ACRS. Additionally, we reviewed the safety assessment principles of the NII, the design safety criteria and guidelines of the CEGB, and the probabilistic risk assessment used in the safety case for Sizewell B.

ENCLOSURE 3

May-June 1982

I. Introduction

The Nuclear Regulatory Commission (NRC) Karlsruhe Sponsored Delegates represent the interests of the NRC and the Department of Energy (DOE) at the Kernforschungszentrum, Karlsruhe (KfK). Our primary duties include: (a) reviewing, summarizing, and reporting of Light Water Reactor (LWR) safety research being conducted at KfK; (b) performing technical functions in support of the KfK staff, including analysis and reporting of selected experimental results; and (c) reporting developments in LWR safety research being performed in Western Europe pertinent to the NRC and DOE Nuclear Research Programs. Specifically, we report on work performed in the following major areas: (a) Core Melt Research, (b) Zircaloy-Uranium-Oxygen High Temperature Interaction Research, (c) Severe Fuel Damage Computer Code Development, and (d) Fission Product Release Research.

Attachment 1 is a paper prepared by S. Hagen for the 6th International Conference on Zirconium in the Nuclear Industry held June 28 - July 1, 1982 in Vancouver, Canada, titled "Out-of-pile Experiments on the High Temperature Behavior of Zry-4 Clad Fuel Rods". Attachment 2 is a paper prepared by S. Leistikow for the same conference titled "Comparison of High Temperature Steam Oxidation Kinetics under LWR Accident Conditions: Zircaloy 4 versus Austenitic Stainless Steel". The paper prepared by P. Hofmann, D. Kerwin-Peck, and P. Nikolopoulos for the same conference titled "Physical and Chemical Phenomena Associated with the Dissolution of Solid UO_2 by Molten Zircaloy-4" was transmitted in the March-April report. We provided technical and editorial review of these three papers. Attachment 3 is a preliminary summary by A. Fiege of the severe fuel damage investigations being performed at KfK within the framework of the Nuclear Safety Project. Attachment 4 is a KfK formal report (in English) titled "SSYST, a Code-System for Analyzing Transient LWR Fuel Rod Behavior Under Off-Normal Conditions", by H. Borgwaldt and W. Gulden. Attachment 5 is a KfK formal report (in English) titled "Thermal and Mechanical Behavior of PWR Fuel Rod Simulators for LOCA Experiments" by V. Casal, S. Malang, and K. Rust. Attachments 6 through 12 are papers (in German) which were presented at a conference sponsored by the Kerntechnische Gesellschaft (KTG) on Release and Transport of Fission Products During

Severe Hypothetical Loss-of-Coolant Accidents in Light Water Reactors held June 8-9, 1982, at KfK. The chairman's summary remarks and the titles of the individual papers have been translated in Section II of this report. ttachment 13 is a KfK internal report (in German) by K. Wiehr titled ...IKA Material Test". The summary and conclusions have been transl... in Section IV of this report. Attachment 14 is a KfK internal report (in German) by H. Neitzel titled "Investigation of Coolant Channel Blockage in Rod Bundles During PWR LOCA Experiments". The summary has been translated in Section V. Attachment 15 is a KfK formal report titled "Material Properties under Accident Conditions: Densities of Core Melts, Interfacial Energies Between Solid Uranium Oxide and Liquid Metals" by P. Nikolopoulos and G. Ondracek. The report comprises three journal article reprints (two in German, one in English). Attachment 16 is a KfK formal report (in English) titled "Oxidation of Zircaloy 4 Tubing in Steam at 1350 to 1600°C" by A. Aly. The conclusions have been summarized in Section VII. Attachment 17 is a KfK formal report (in German) titled "Concrete Crucible Experiments with Thermite Melts" by D. Perinic et al. The report contains many interesting figures and photographs for those interested in core melt investigations. Attachment 18 is a KfK internal report (in German) titled "Experiment Matrix for the BETA Facility for the Interaction of a Steel Melt with Concrete - Status as of 9/17/81" by H. Alsmeyer et al. Table 2 of this report is our translation of the table describing the experiment matrix.

The balance of this report includes, by section: (II) Dr. Schikarski's summary remarks on the KTG Conference on Fission Product Release and Transport, (III) REBEKA 5 Test, (IV) REBEKA Material Test, (V) Investigation of Coolant Channel Blockage in Rod Bundles During PWR LOCA Experiments, (VI) UO_2/Zircaloy-4 Interaction Behavior and Reaction Kinetics Studies at High Temperatures, (VII) High Temperature Zircaloy-4/Steam Oxidation Experiments, and (VIII) BETA Facility.

II. Summary Remarks* on the Release of Fission Products in Hypothetical Severe LWR Loss-of Coolant Accidents, W. O. Schikarski, Laboratorium für Aerosolphysik und Filtertechnik I, Kernforschungszentrum Karlsruhe GmbH, KTG Topical Meeting "Release and Transport of Fission Products in Hypothetical Severe LWR Loss-of-Coolant Accidents" Schule für Kerntechnik, Kernforschungszentrum Karlsruhe, June 8-9, 1982 (translation from German) (Debbie)

A. Problem Statement

In addition to the design basis accidents, hypothetical severe LWR loss-of-coolant accidents have been of particular interest to the technical community, as well as to the public. Especially in the US, experiments were performed early on to address the question, what is the worst case occurrence when a large LWR power plant fails such that a considerable fission product release results. The WASH-740 report /1/, which was published in 1957, the WASH-1400 Risk Study (Rasmussen Study) /2/, which was published in 1975, and the German Risk Study /3/, which was just published in 1980, should be remembered.

It was obvious that the different specialists from the areas of material science, reactor physics and dynamics, radio chemistry, aerosol physics, etc., were increasingly interested in particular questions from these studies. Therefore, numerous international groups have been engaged in the problems of core melt dynamics, melt-concrete interaction, aerosol behavior, fission product chemistry, and containment behavior. For the last 10 years, the Kernforschungszentrum Karlsruhe has performed investigations, within the framework of the Nuclear Safety Project, which are dedicated to examination of core melt accidents.

How can the status of knowledge be summarized today?

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* The following personal statement of the author was discussed at the conference and, in general, approved.

B. Status of Knowledge

The status of knowledge can be summarized (without repeating individual statements from the meeting) in a few sentences:

1. We must differentiate between risk and consequence. Risk, as is well-known, is the product of probability and consequence. The probability of the occurrence of these hypothetical accidents has been shown to be very small, so that the risk in the above-mentioned studies for very severe consequencas is also small.
2. A new result is that the consequences of these accidents has, up to now, been considerably overestimated.
3. The areas in which the consequences have been overestimated are:
   a. aerosol behavior,
   b. fission product chemistry, and
   c. thermodynamics of the containment system.

   The last area is more indirect since better knowledge has led to a distinct lengthening of the time up to containment failure /4/.
4. Also, the probability of special occurrences, e.g., steam explosion or H_2 detonation, is considerably lower than previously thought /4/.
5. Since for many people the concept of probability or frequency of occurrence is not easily understood, and so that the consequences of an accident can be understood by the public, it should also be established that the number of early deaths which is now estimated is three orders of magnitude smaller than the 10,000 (all release categories) estimated earlier /3/. A similar result holds for late deaths.
6. The analytical models which are used to calculate aerosol and iodine behavior are apparently consistent; verification with respect to realistic scenarios has not yet been completed. This will require two to three years. Resolution of the uncertainties which are present in the models will lead to a further reduction in consequences, as a correction to the higher values.
7. Should this new data on the release during severe hypothetical accidents be confirmed in the next year, the following would be established:
   a. the probability of occurrence of nuclear reactor accidents of all kinds is extremely low;
   b. the extent of personal injury is limited; and

c. when all possible consequences have been considered and summed, the remainder is very probably so small that the associated "remainder risk" is essentially a zero risk.

C. References

1. Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants, US Atomic Energy Commission, WASH-740 (1957)

2. Rasmussen, N. C., Reactor Study - An Assessment of Accident Risks in US Commercial Nuclear Power Plants, US Nuclear Regulatory Commission, WASH-1400 (1975)

3. German Risk Study - Nuclear Power Plants (in German), published by the BMFT, Verlag TÜV-Rheinland (1980)

4. Papers presented at this meeting (in German):

- Prof. Dr. H. Unger, Institute for Reactor Energy and Energy Systems, University of Stuttgart: "Overview of the Scenarios of Hypothetical Severe LWR Loss-of-Coolant Accidents"

- Dr. H. Albrecht and Dipl.-Phys. H. Wild, Institute for Radiochemistry, Kernforschungszentrum Karlsruhe: "Release Source Terms for Hypothetical LWR Accidents"

- Dr. K. Hassman, Kraftwerksunion (KWU), Erlangen: "Thermodynamics in the Containment for Different Core Melt Scenarios"

- Dipl.-Phys. H. Bunz and Dr. W. Schöck, Laboratory for Aerosol Physics and Filter Techniques I, Kernforschunszentrum, Karlsruhe: "Transport and Retention of Aerosols in the Containment During Core Melt Accidents"

- Prof. Dr. H. J. Ache, Institute for Radiochemistry, Kernforschungszentrum, Karlsruhe: "Fission Product Chemistry"

- Dr. J. P. Hosemann, Project Nuclear Safety, Kernforschungszentrum, Karlsruhe: "Special Occurrences During an Accident"

- Prof. Dr. A. Bayer, Institute for Neutron Physics and Reactor Techniques, Kernforschungszentrum, Karlsruhe: "Estimation of the Radiation Load and Resulting Damage After Accident-Dependent Activity Releases"

III. REBEKA 5 Test (Scott)

The REBEKA 5 LOCA simulation test was performed June 3 by F. J. Erbacher and K. Wiehr. The primary objectives of the test were to investigate the influence of rod-to-rod mechanical interactions and bundle size on the ballooning and flow blockage behavior of zircaloy cladding. The bundle contained a 7 x 7 array of pressurized, electrically-heated rods. As shown in Figure 1, the pressure and temperature history of the bundle was similar, but not identical, to that of REBEKA 3. The average burst pressure of 68 bar was somewhat higher and the average burst temperature of 795°C slightly lower than REBEKA 3 (51 bar and 830°C, respectively). Table 1 compares the main test data from both experiments. Figure 2 gives the burst strains and locations for all 49 rods and Figure 3 shows the deformation profiles of the inner 25 rods as a function of axial elevation. Figure 4 illustrates the average burst strains of the three rings of simulators plus the center rod. As can be seen, very little difference between the average strains was observed. Cross section photographs of the entire bundle are shown at five elevations in Figures 5 through 9. Elevations are from the top of the bundle in millimeters (1910, 1950, 1980, 2000, and 2110 mm, respectively). The elevation of maximum coplanar blockage was 2000 mm.

The major conclusions from the test are that:

1. The results are consistent with the REBEKA 1 through 4 and single rod test results.
2. Substantial rod-to-rod mechanical interactions occurred.
3. Failure propagation, which would result in increased flow blockage, was not observed.

4. The continuous, cosine-shaped axial power prof...e did not enhance the coplanarity of bursts. (Earlier tests used a stepped power profile.)
5. The deformation and flow blockage behavior are independent of bundle size.

These test results are preliminary. Evaluation of the test data is continuing.

IV. REBEKA Material Test (translation of Attachment 13 summary and conclusions) (Scott)

The purpose of the test was to deform a 5 x 5 zircaloy cladding bundle as severely as possible through rod-to-rod mechanical interaction. The "REBEKA Material Test", a purely material examination experiment, was therefore carried out under completely atypical reactor boundary conditions. The important parameters were a nearly constant cladding temperature of 750°C, constant internal pressures, and a decay heat of about 0.25% which only served to cover the heat losses from the test section.

The results were deformation times from pressurization to burst of 3.5 to 7 minutes, relatively large circumferential burst strains up to 89%, and a resulting maximum coolant channel blockage of the inner 9 rods of 84% . A coplanarity of the maximum strains of 8 of the 9 inner rods occurred.

The test confirms the strong dependence of plastic deformation of zircaloy on temperature (large differences in deformation times of individual cladding despite relatively small deviations in cladding temperature). The large differences in deformation times prevented the maximum possible cladding mechanical interaction, and thus the formation of "square cladding" and a coolant channel blockage near 100% . Despite good insulation of the test section, a radial temperature profile was developed which resulted in azimuthal temperature differences and also reduced the maximum possible cladding deformation.

The apparent coplanarity of the short, balloon-shaped bursts of 8 of the 9 inner rods was test-dependent and is completely atypical for cladding deformation during the refill and flood phases of a LOCA.

The experiment showeu that it is practically impossible to produce cladding deformation in a bundle which could severely reduce the coolant channel area and lead to serious impairment of emergency coolability.

V. Investigation of Coolant Channel Blockage in Rod Bundles During PWR LOCA Experiments (translation of Attachment 14 summary) (Scott)

The degree of blockage of plastically-ballooned fuel rod cladding was calculated. A relationship was derived that gives the degree of blockage as a function of the measured circumferential cladding strain. The use of this relationship for burst and nonuniformly strained cladding is discussed in the report.

VI. UO_2/Zircaloy-4 Interaction Behavior and Reaction Kinetics Studies at High Temperatures (Debbie)

The paper prepared by P. Hofmann, P. Nikolopoulos, and myself titled "Physical and Chemical Phenomena Associated With the Dissolution of Solid UO_2 by Molten Zircaloy-4" was presented by Peter Hofmann in Vancouver. A copy of this paper was transmitted in the March-April report. Three journal article reprints by P. Nikolopoulos and G. Ondracek have been published as a KfK formal report titled "Material Properties under Accident Conditions: Densities of Core Melts, Interfacial Energies Between Solid Uranium Oxide and Liquid Metals" (Attachment 15). The third article is in English and discusses the interfacial energies of thirteen UO_2-liquid metal systems as functions of temperature. The liquid metals considered are Fe, Co, SS 1.4970, Au, Ni, Al, Cu, Pb, Na, In, Bi, Sn, Sb.

The solid UO_2/solid zircaloy interaction experiments continued. The Arrhenius diagram of reaction layer growth rate versus reciprocal temperature shows a possible acceleration of the reaction kinetics from 1500°C to the melting point of zircaloy, compared to the reaction kinetics from 900 to 1500°C. The reaction layer growth rates for the experiment series at 1300 and 1400°C for 3 to 150 minutes agree quite well with Hofmann's 1978 work. The higher temperature measurements have been causing difficulties and some of the experiment series (several reaction times at one temperature) have been repeated. The maximum temperature series (1700°C) was performed as closely as possible to the melting point of zircaloy (1720 ± 20°C as given by the cladding manufacturer) without achieving melting. We are therefore fairly certain that the temperatures of the experiment series performed at 1300, 1400, and 1700°C are correct within a reasonable uncertainty (± 50°C). The temperature uncertainty for the series performed at 1500 and 1600°C is somewhat larger.

Transient experiments have been performed up to approximately 2000°C at heatup rates of 1, 2, 5, and 10 K/s, a one second hold time, and the same cooldown rate as heatup rate. The external overpressure was 60 bar. The results clearly show that the amount of molten cladding runoff depends on the heatup rate. A slower heatup rate allows zircaloy and UO_2 to interact so that oxygen uptake by the zircaloy occurs, raising the melting point of the cladding and decreasing the total amount of melt at a given temperature. At 2 K/s, very little melting occurred, indicating that the cladding had essentially completely reacted to α-Zr(O) (melting point about 2000°C). The specimens tested at 5 and 10 K/s are shown in Figure 10. Note in the figure that the exposed fuel stacks have been "glued" together by the runoff of molten material. That is, the uptake of oxygen by molten zircaloy results in a melt which wets the UO_2 pellets very well, flowing into the dishing between pellets and into pellet cracks. Although the fuel stack is very brittle, it actually has more structural integrity after the test.

VII. High Temperature Zircaloy-4/Steam Oxidation Experiments (Debbie)

The results of oxidation experiments performed between 1350 and 1600°C in steam have been published as a KfK formal report (Attachment 16). The conclusions are summarized below:

1. The oxidation reaction in this temperature range obeys a parabolic rate law. The oxide layers which form are uniform, adherent, and protective up to complete cladding wall consumption.
2. A discontinuity in the temperature dependence of the reaction rate was observed at 1550°C and is attributed to the phase transformation of ZrO_2 from tetragonal to cubic at this temperature.
3. Severe embrittlement of the underlying metal was indicated by typical brittle fracture of the oxygen-stabilized α-zircaloy region.

4. The dependence of microhardness on dissolved oxygen concentration in the transformed β-zircaloy matrix and oxygen-stabilized α-zircaloy region has been established.
5. Compressive stresses generated in the oxide are compensated by tensile stresses in the underlying metal.

VIII. BETA Facility (Debbie)

Pre-BETA test M103 was performed June 24. A sketch of the experiment setup is shown in Figure 11. One hundred kilograms of thermite powder was ignited in a concrete crucible, and it was planned that an additional 900 kg be added at 30 kg/minute. The primary purpose of this method was to maintain the high temperatures in the crucible over an extended period of time to test the integrity of the crucible. When a large mass is ignited all at once, the temperature is very high at first (T ≈ 3000°C) and then rapidly falls off unless the melt is heated in some way. For example, in the BETA facility the melt will be inductively heated, with the metallic phase as susceptor. A secondary purpose for testing this pouring method is that, since the oxide phase lies above the metallic phase in the melt, the melt could possibly be heated by adding powder to the oxide phase rather than inductively heating the metallic phase. This would allow the metallic phase to solidify before the oxide phase, which is what is calculated to occur. However, in test M103 the violent splashing of the melt in the crucible sealed the opening of the funnel shortly after the pour started. Only approximately 100 kg of additional powder was added.

Gas analyses of the pre-BETA tests are performed by the Kraftwerkunion (KWU), Erlangen. The report on the gas analysis of test M103 should be available in July. No analyses of the chemical composition of the melts will be performed due to lack of funding. No tests will be performed in July and August.

Main characteristics	REBEKA 3		REBEKA 5	
main test objective	influence of thermohydraulics		influence of: • rod-to-rod mechanical interactions • bundle size	
bundle size	5 x 5 (inner 3x3 pressurized)		7 x 7 (all pressurized)	
axial power profile	stepped		cosine shaped	
flooding rate, cold	2.69 cm/s		2.66 cm/s	
rod power (decay heat)	20 W/cm		20.8 W/cm	
average burst pressure	51 bar	inner 3x3 rods	68 bar	inner 5x5 rods
average burst temperature	830 °C		795 °C	
average burst strain	44 %		52 %	
max. flow blockage	52 %		52 %	

REBEKA 3 - REBEKA 5

Table 1

comparison of main test data

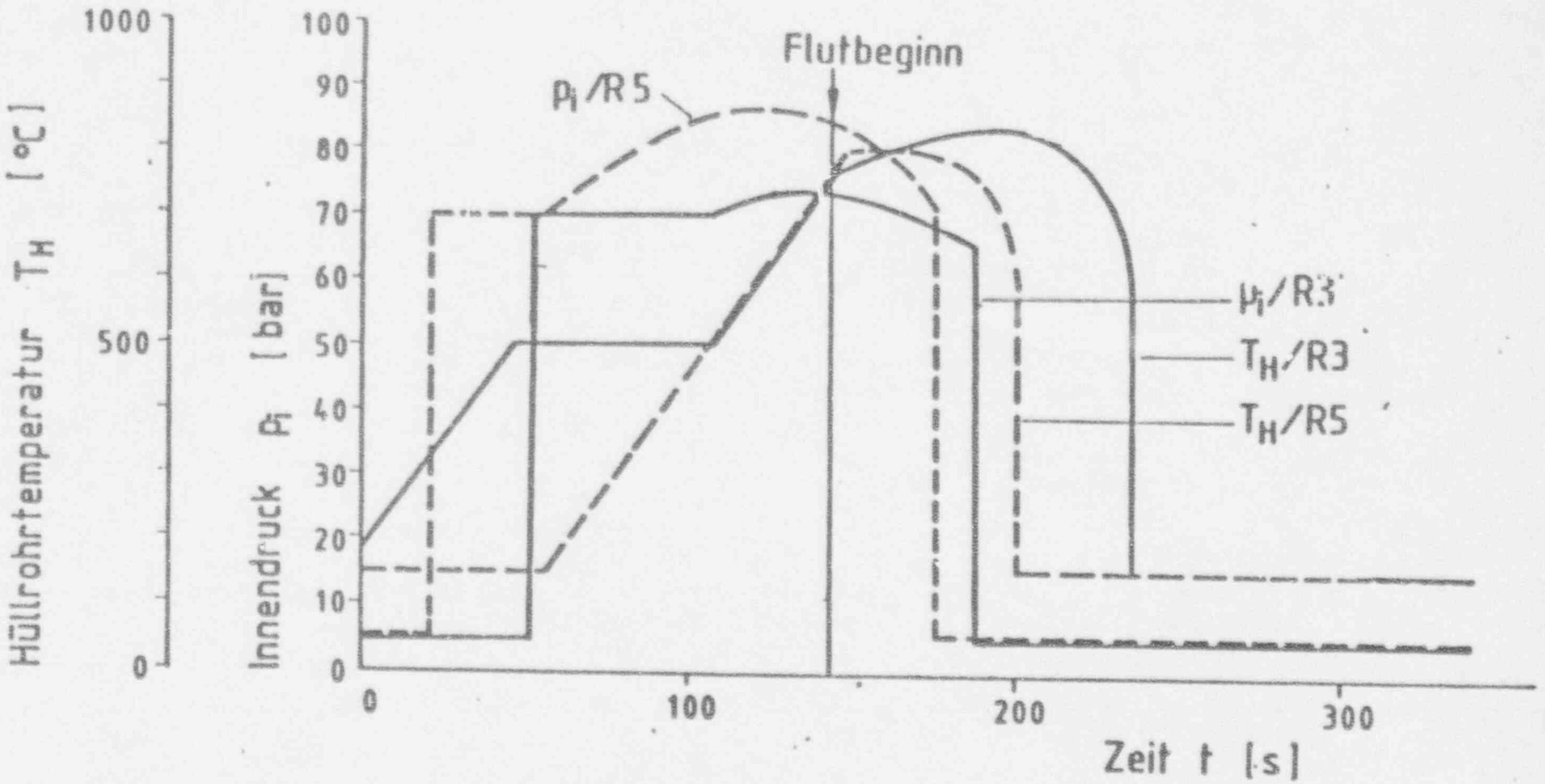
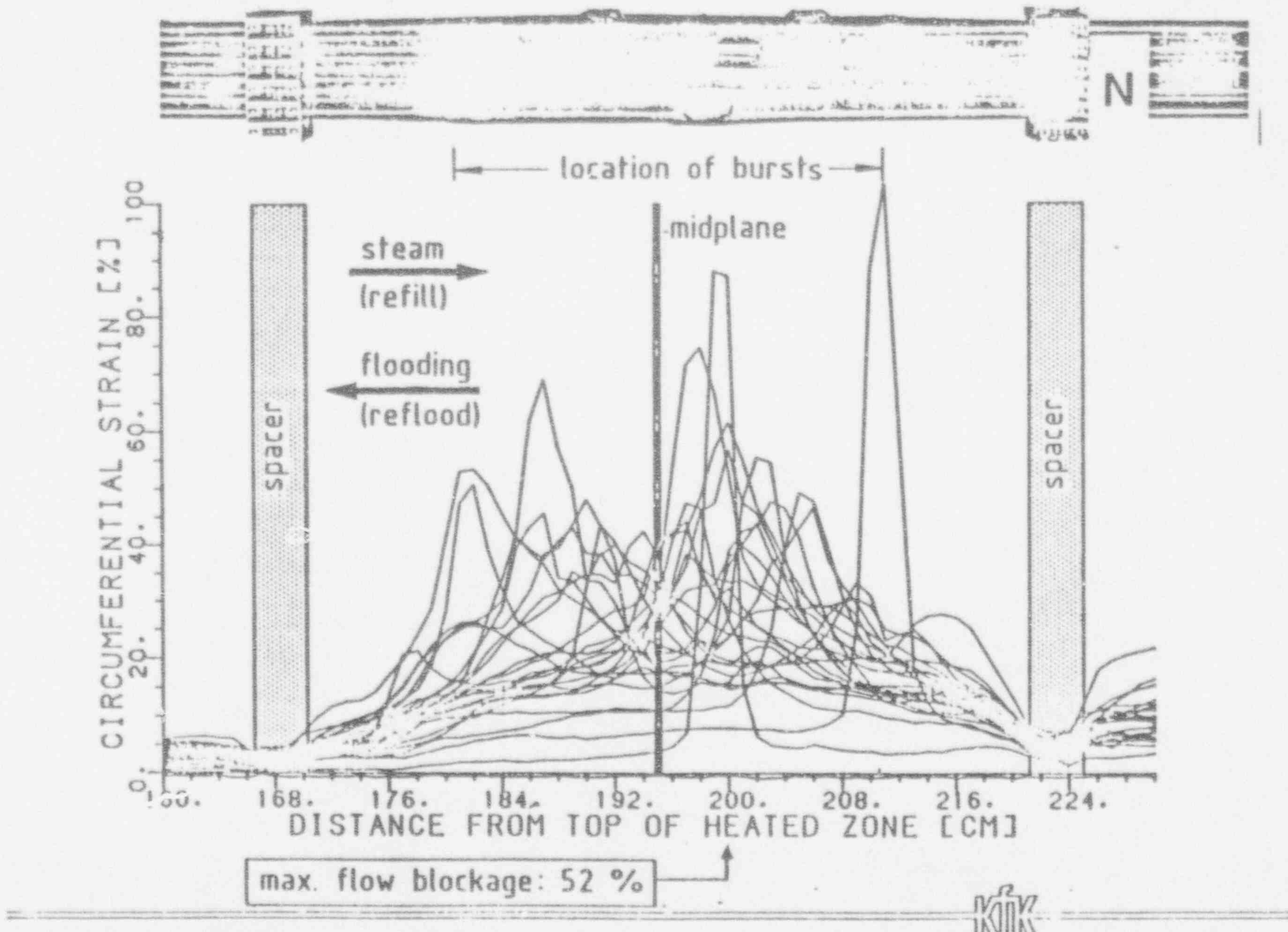


Figure 1 REBEKA 3 - REBEKA 5

Vergleich der Testverläufe

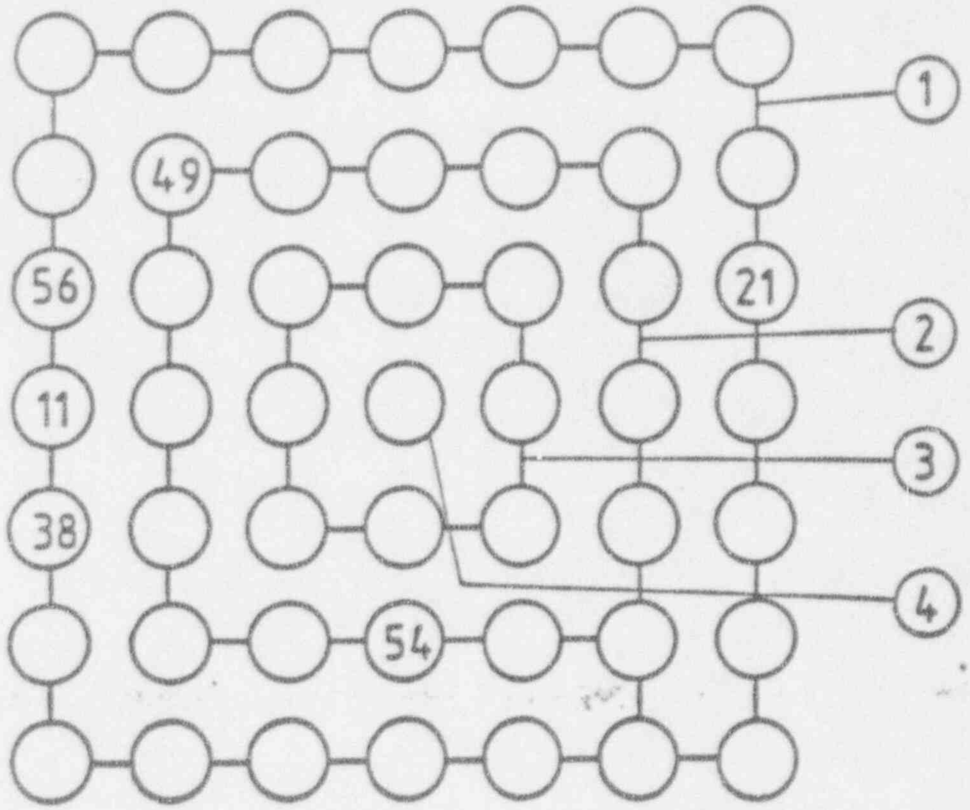
Figure 2. REBEKA 5

burst strain and burst location



igure 3 REBEKA 5

axial deformation profile of the inner 25 rods



(1) Außenring: 35,3 % (incl. der 4 nicht geborstenen Hüllen 56, 11, 38 und 21)
36,6 % (ohne die 4 nicht geborstenen Hüllen)

(2) Zwischenring 54,7 % (ohne Stab 54: kein Innendruck)
51,3 % (mit Stab 54)
51,2 % (ohne Stab 54 und 49)
(49 besitzt sehr lokale Beule)

(3) Innenring: 48,8 %
(4) Zentralstab: 49,2 %

$\Big\}$ 9 Innenstäbe: 48,87 %

KfK

Figure 4 REBEKA 5

Mittelwerte der maximalen Dehnungen der Zr-Stabhüllen

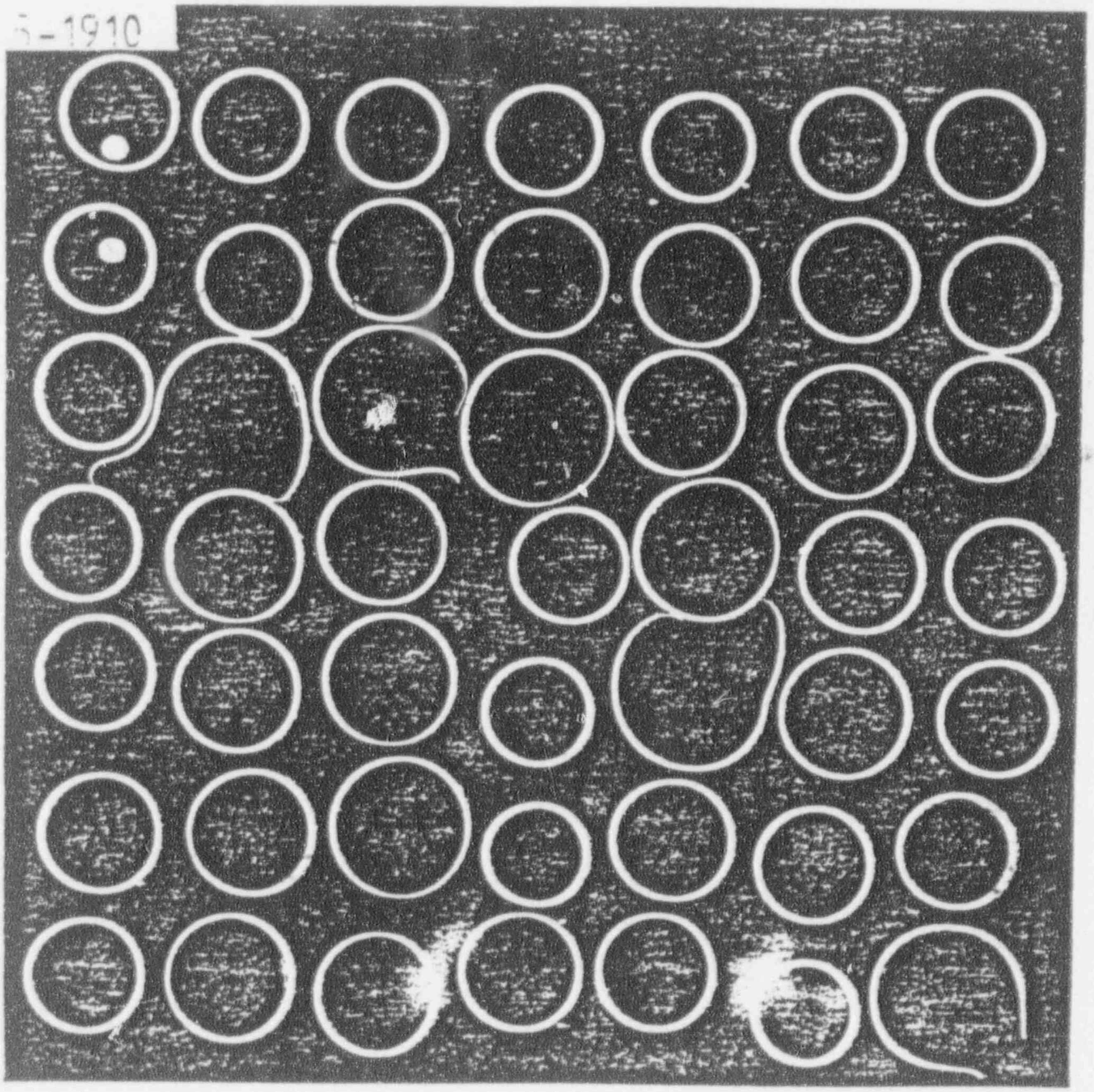


Figure 5. REBEKA 5 cross section at 1910 mm from top of bundle

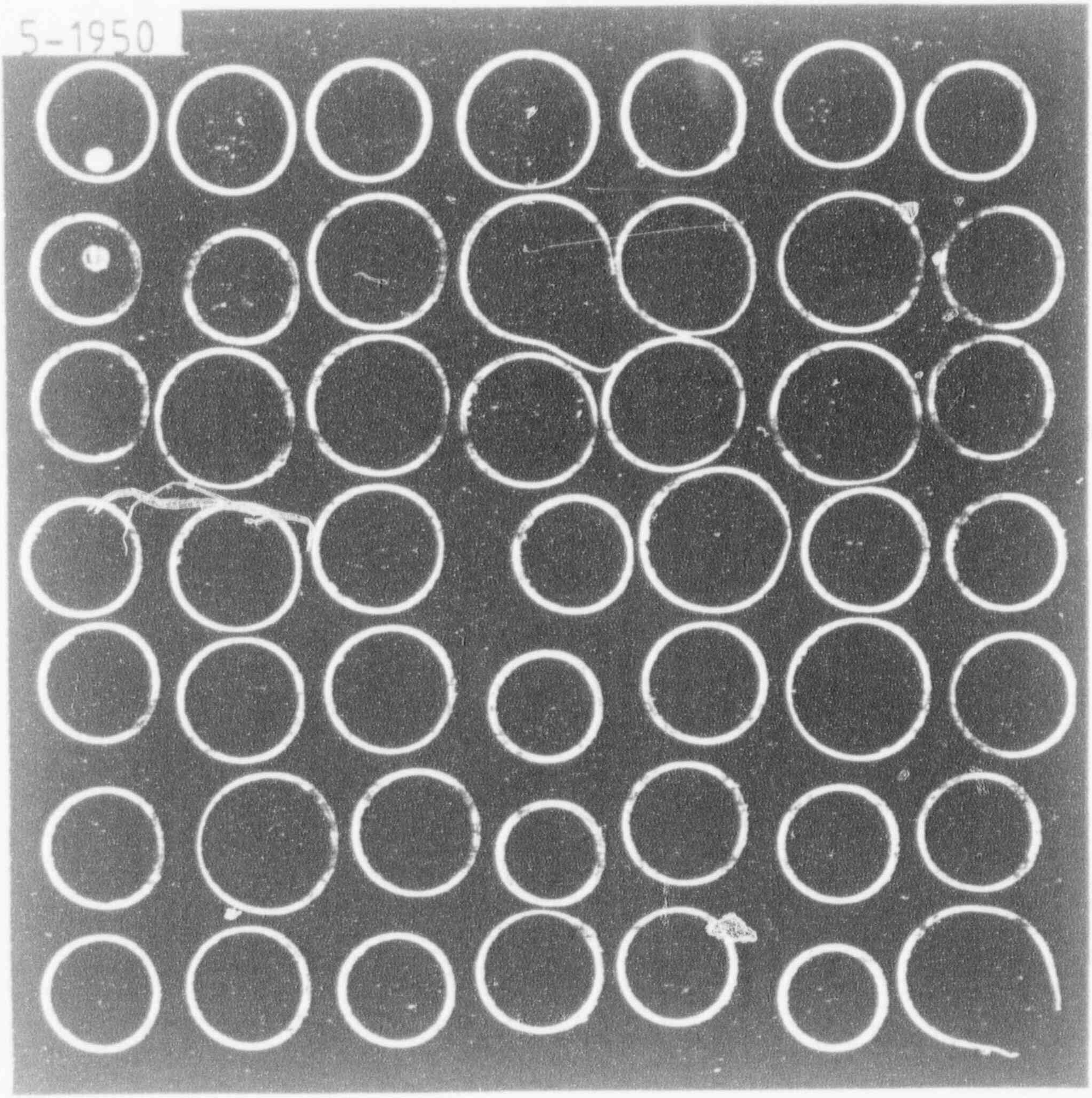


Figure 5. REBEKA 5 cross section at 1950 mm from top of bundle

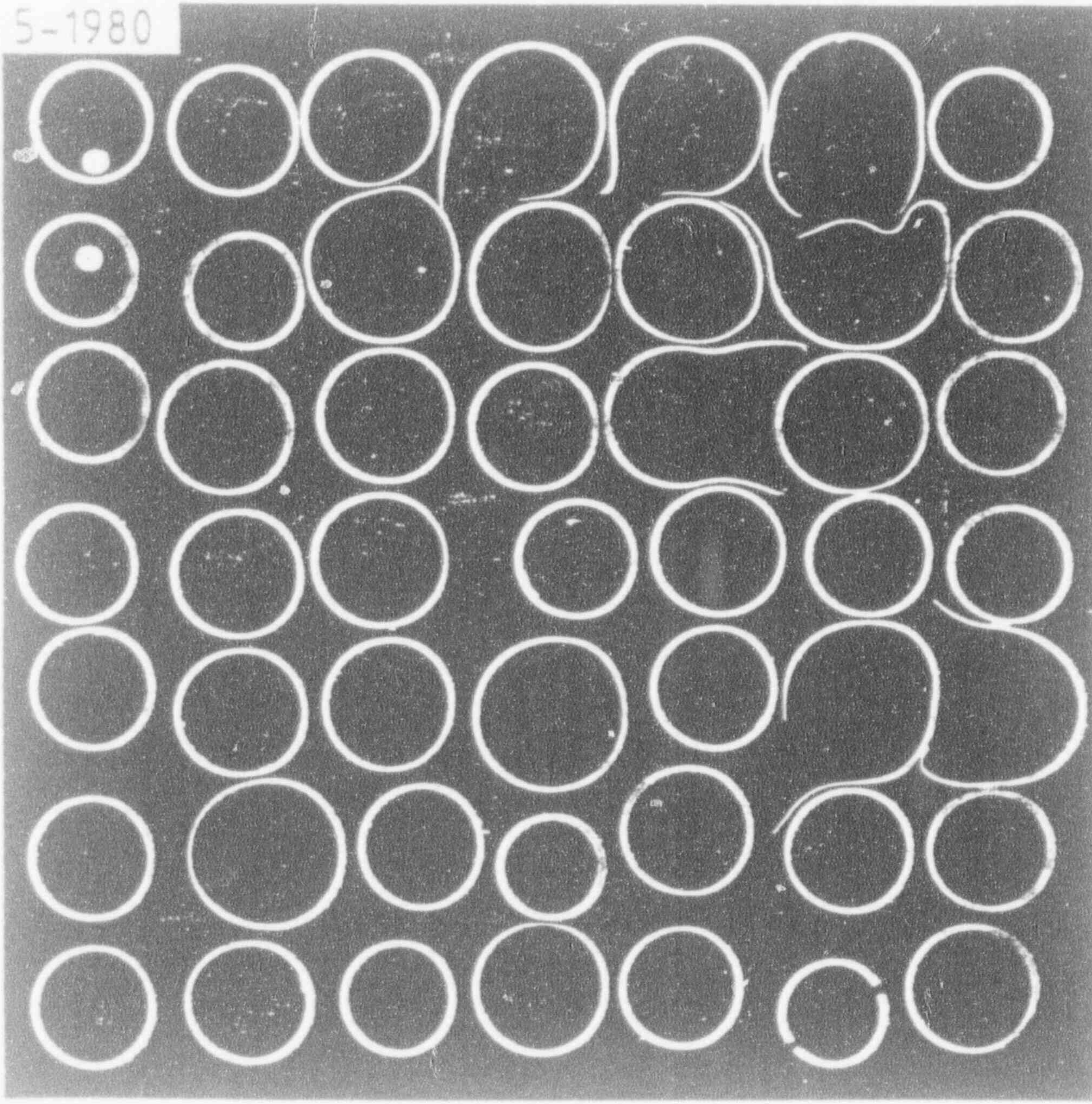


Figure 7. REBEKA 3 cross section at 1980 mm from top of bundle

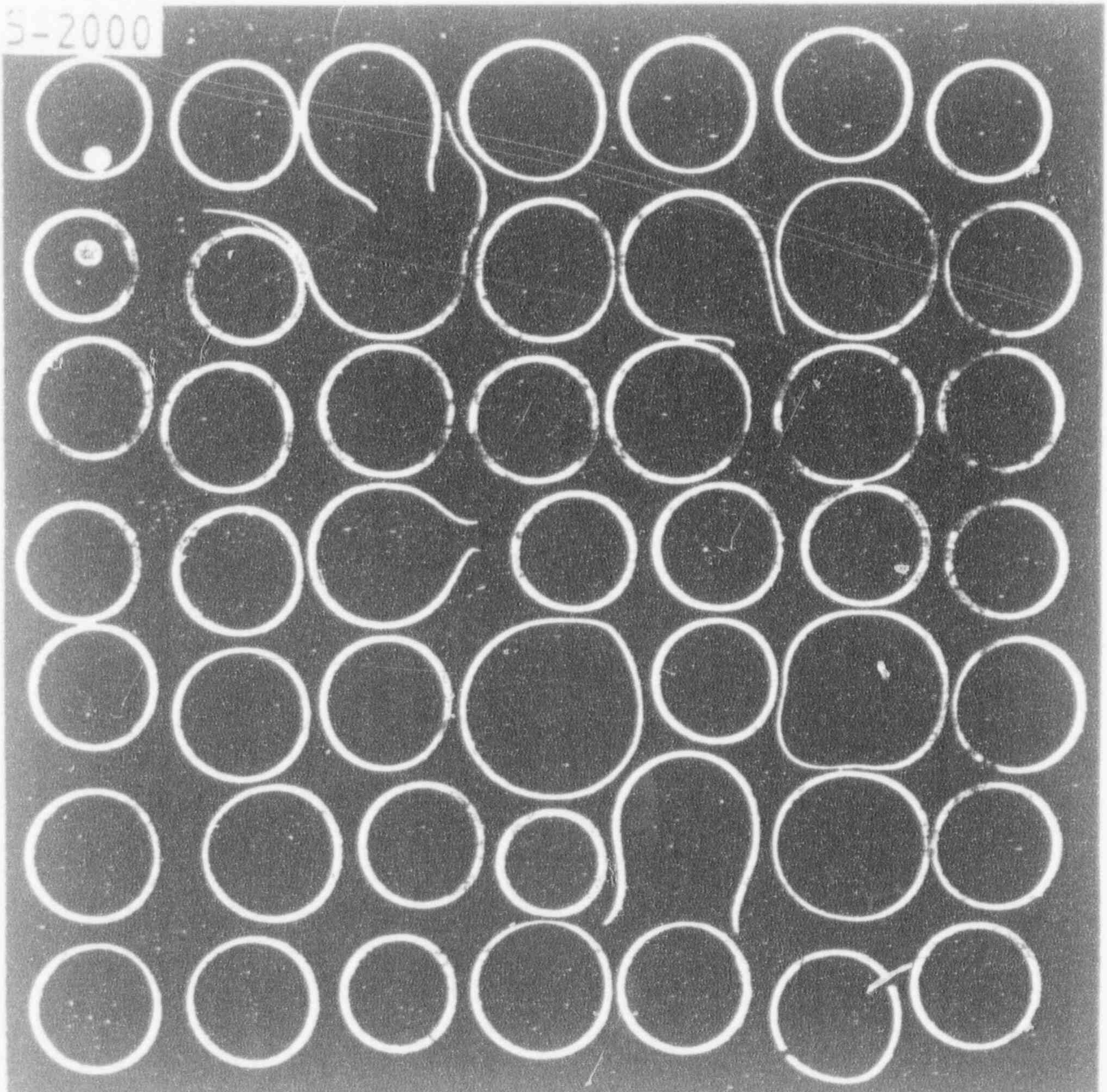


Figure 8. REBEKA 3 cross section at 2000 mm from top of bundle (elevation of maximum coplanar blockage)

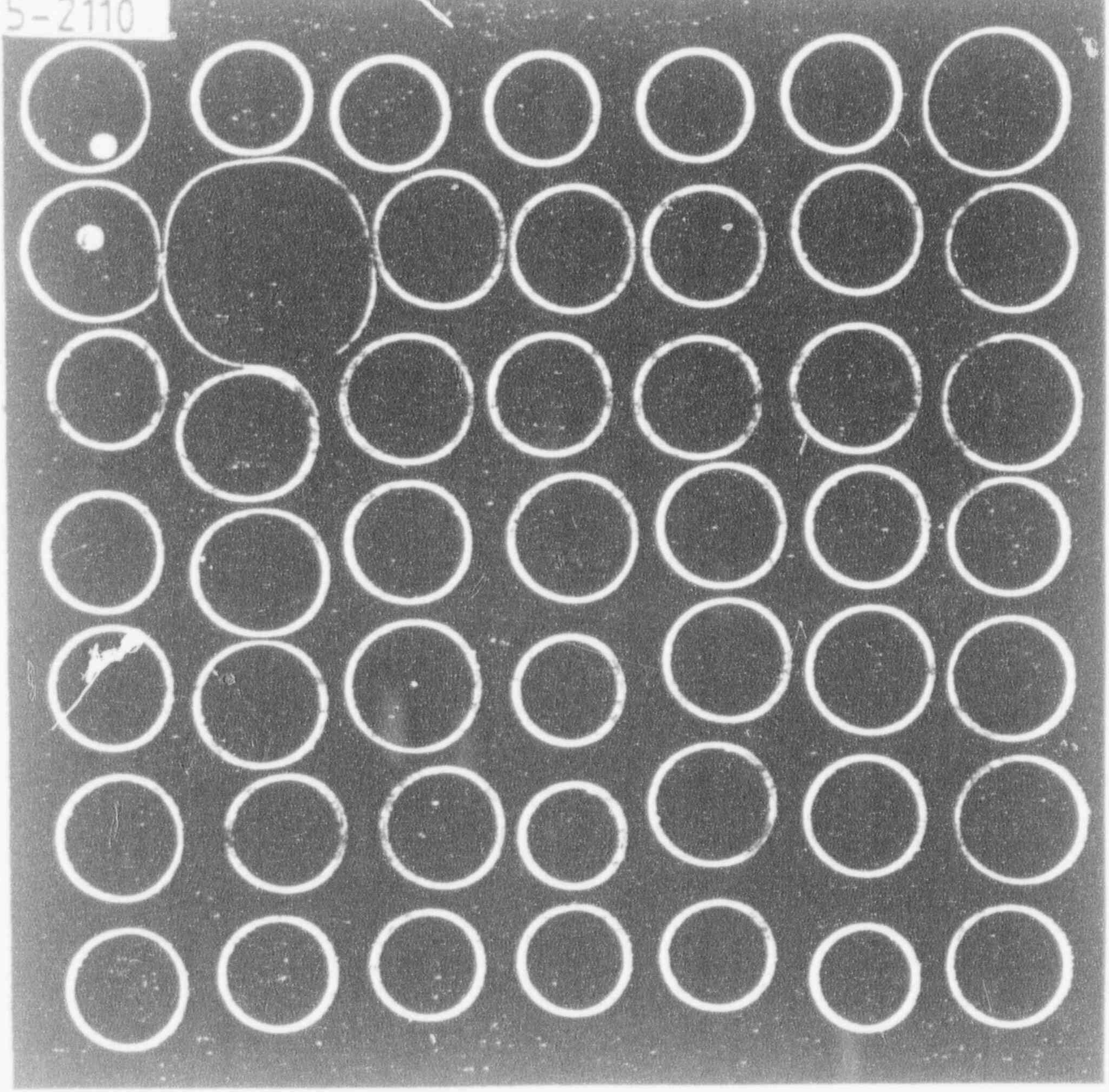


Figure 9. REBEKA 3 cross section at 2110 mm from top of bundle

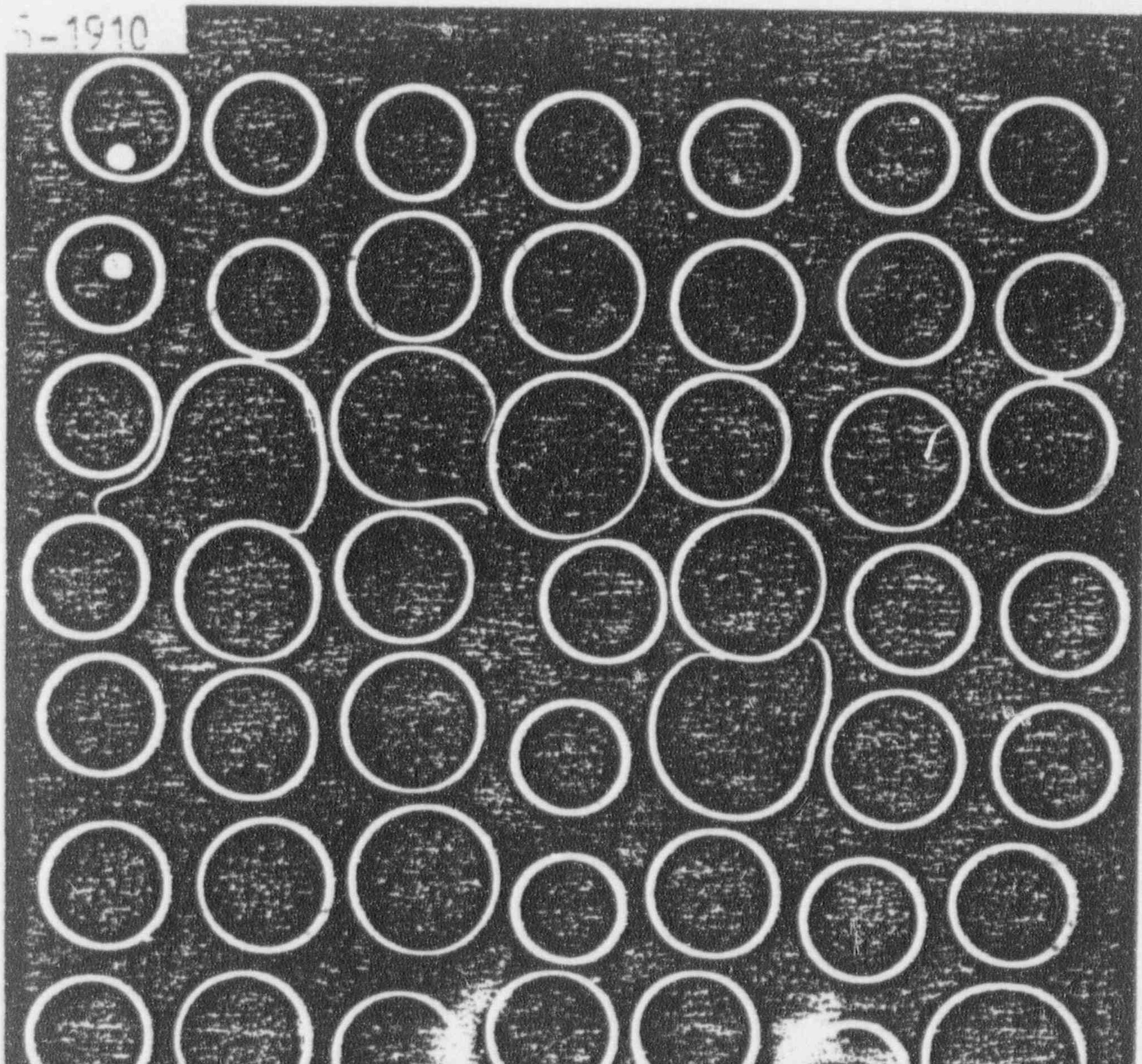


Figure 5. REBEKA 5 cross section at 1910 mm from top of bundle

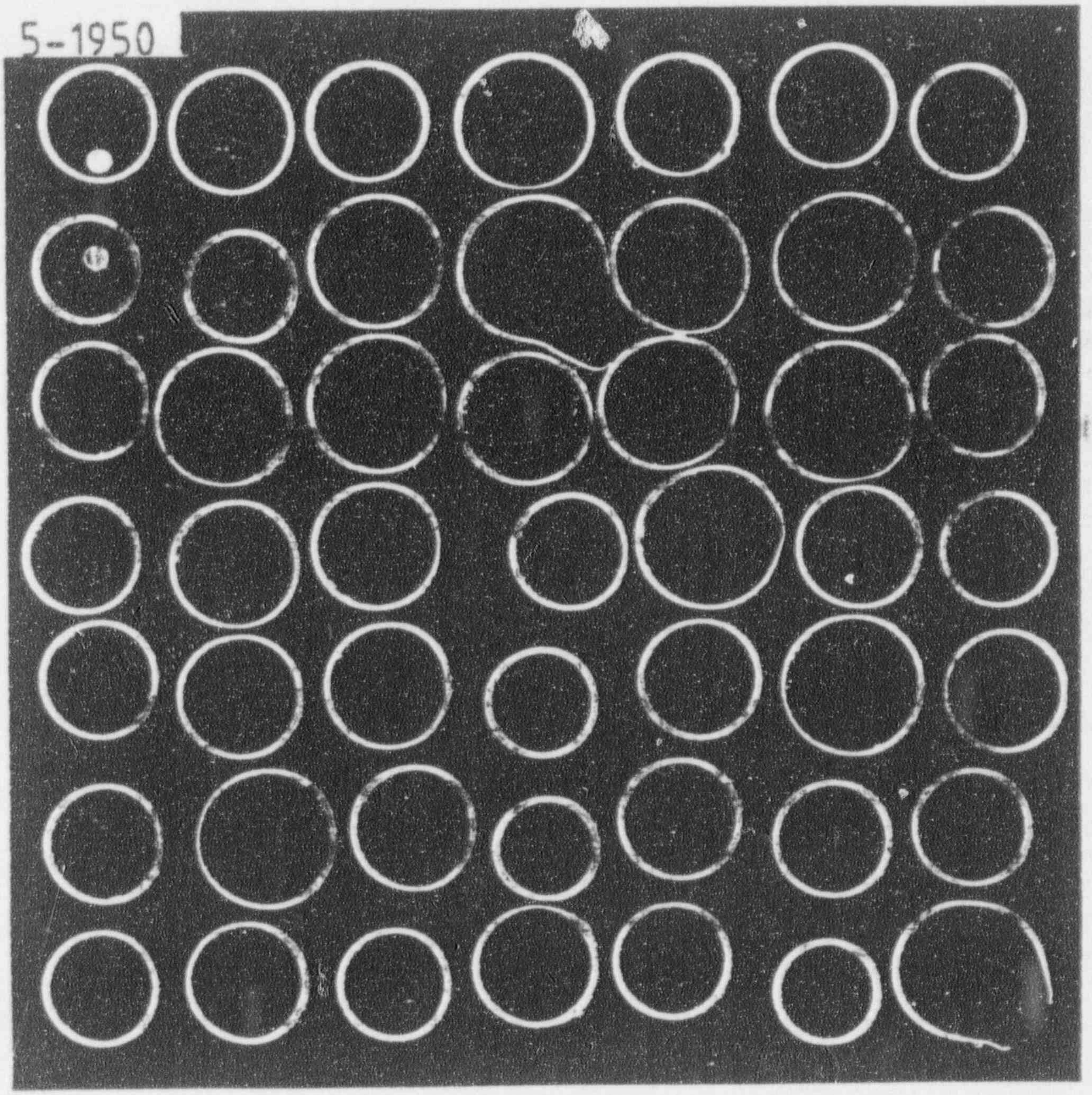


Figure 6. REBEKA 5 cross section at 1950 mm from top of bundle

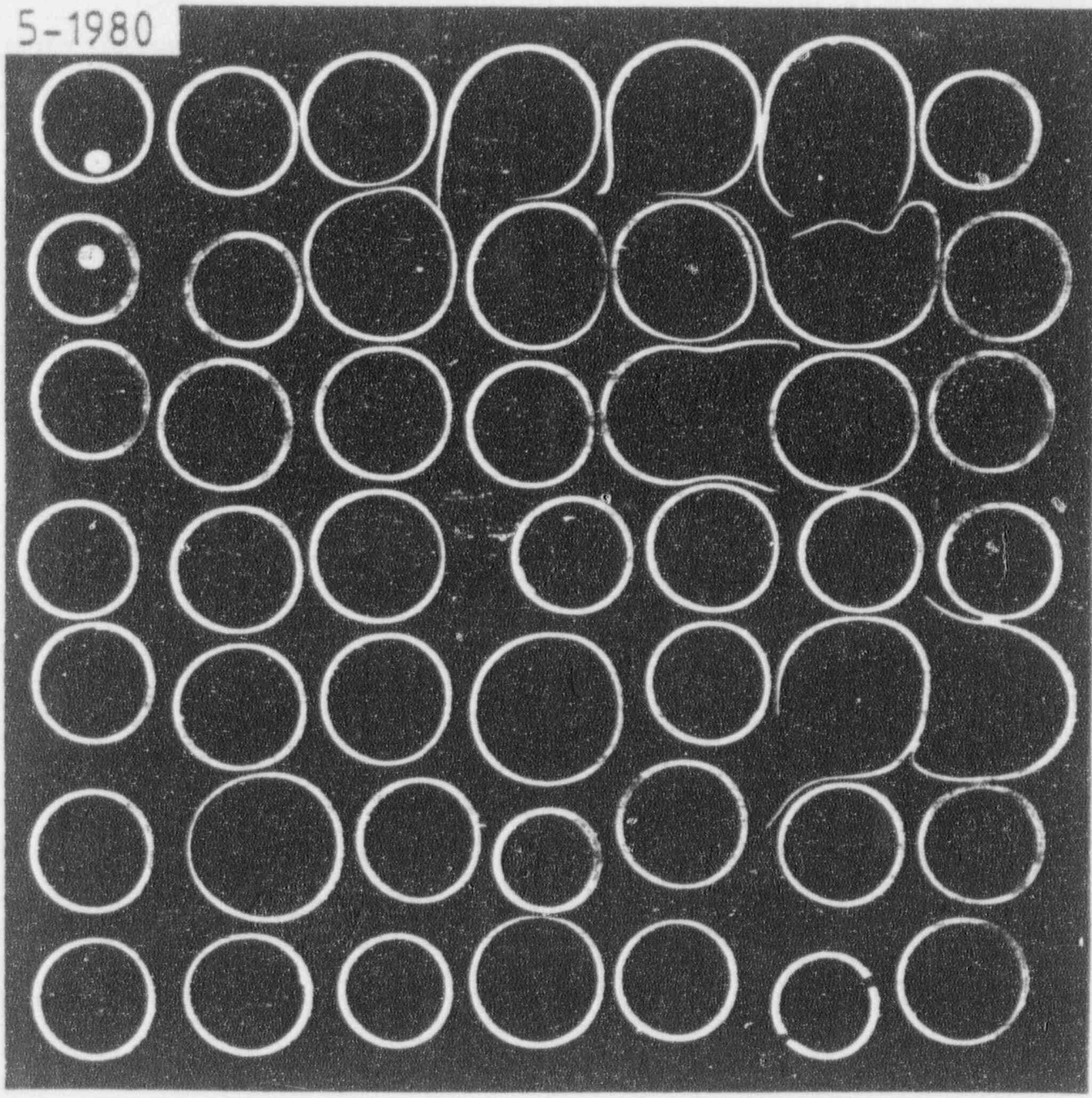


Figure 7. REBEKA 5 cross section at 1980 mm from top of bundle

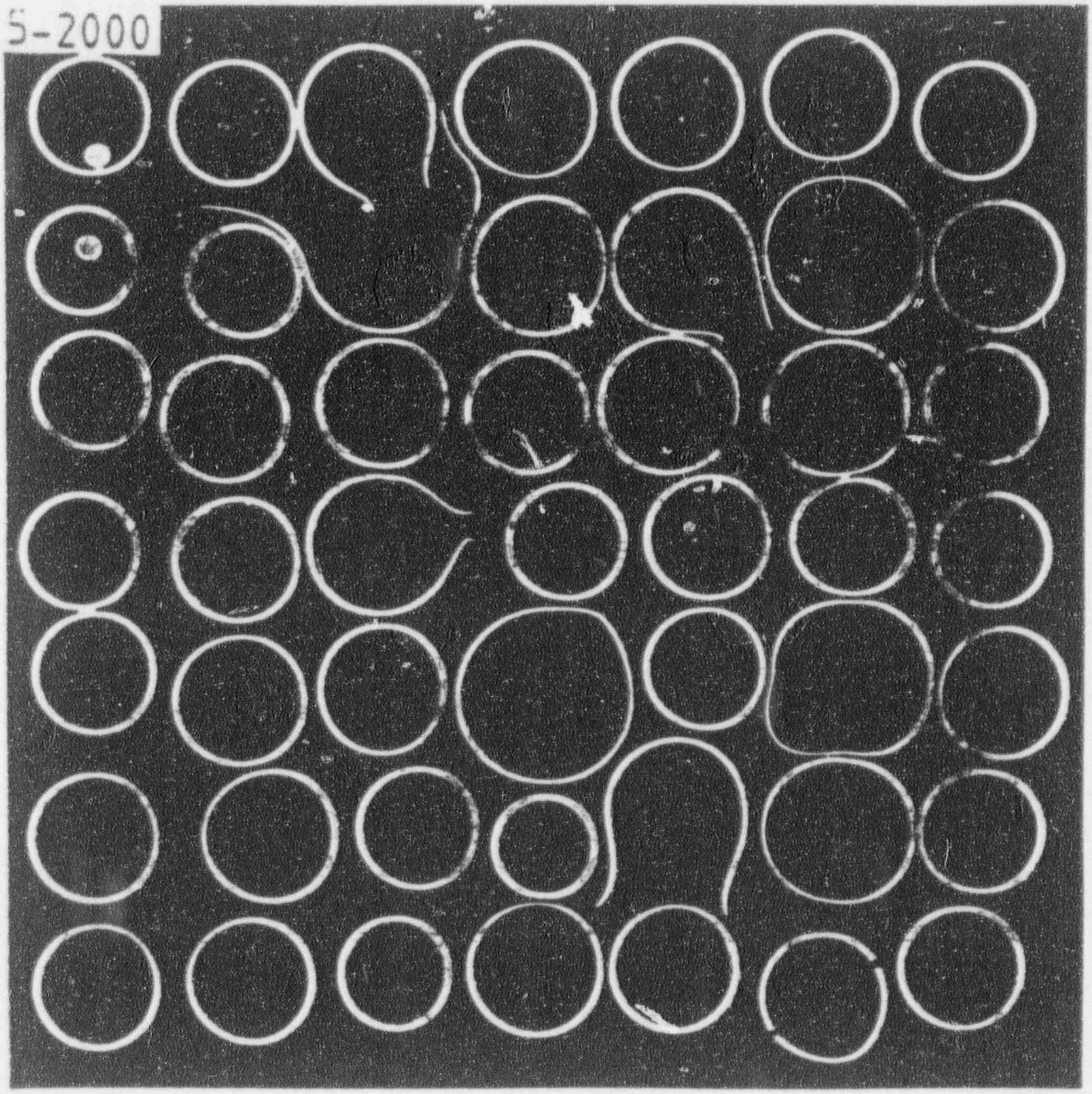


Figure 8. REBEKA 5 cross section at 2000 mm from top of bundle
(elevation of maximum coplanar blockage)

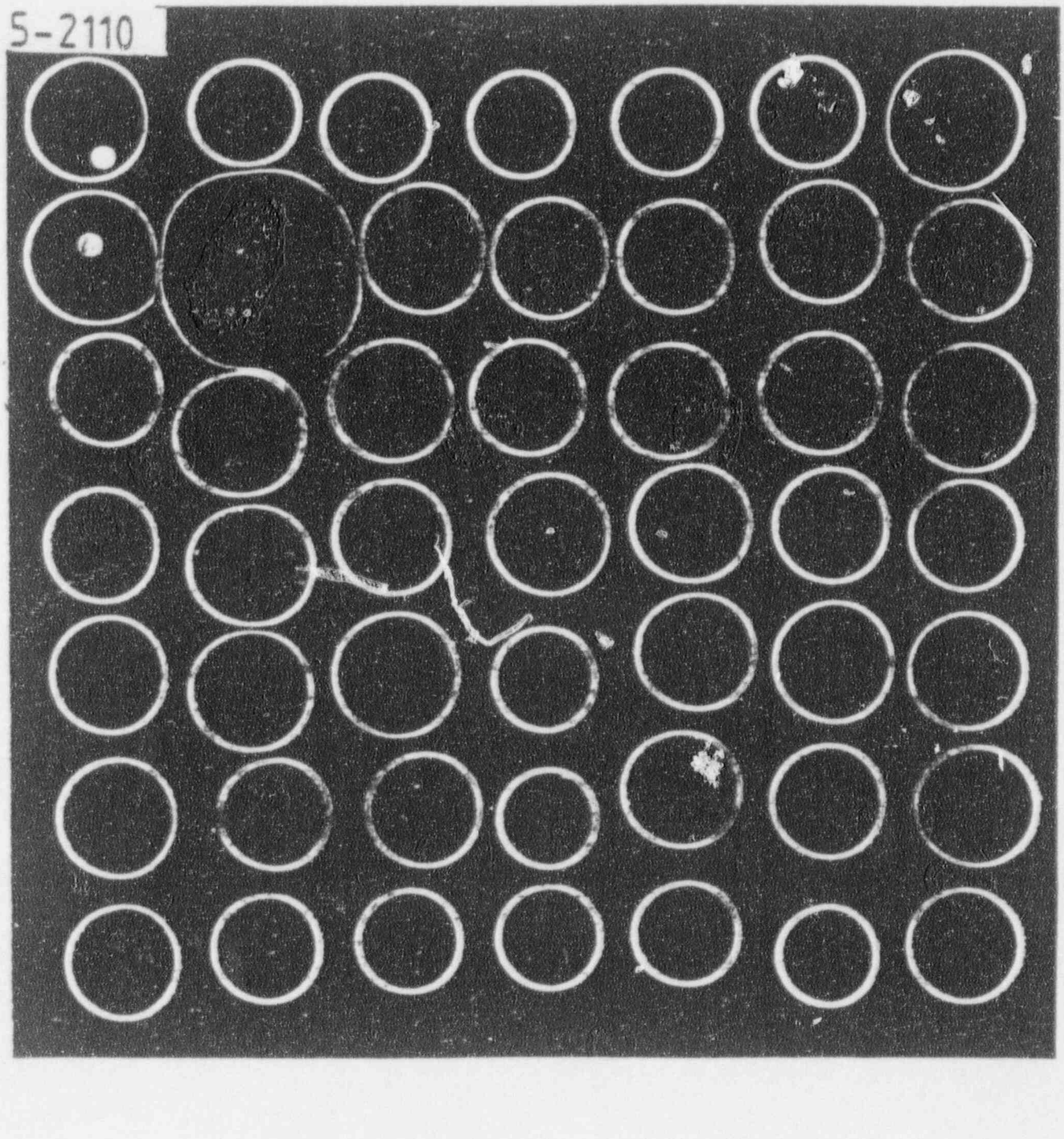
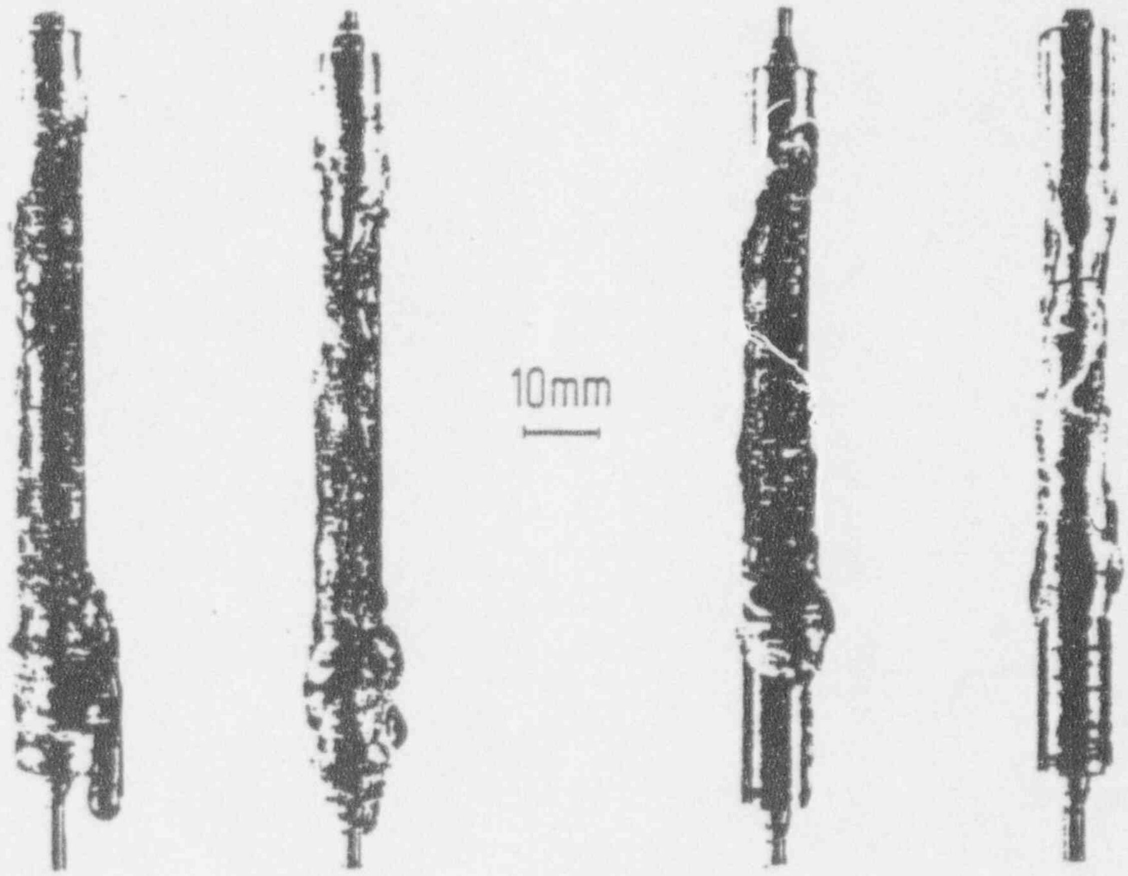


Figure 9. REBEKA 5 cross section at 2110 mm from top of bundle



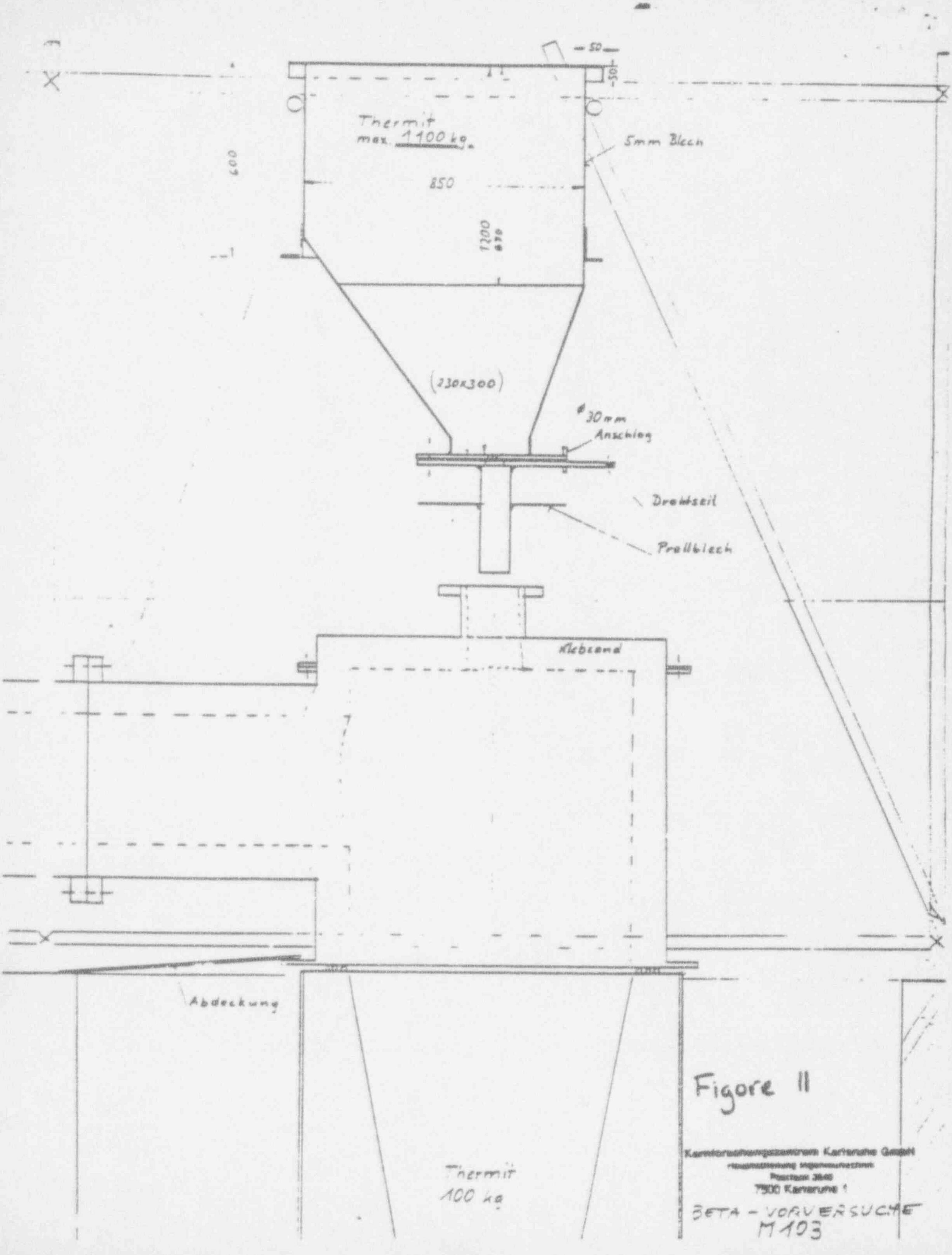
dT/dt = 5 K/s dT/dt = 10K/s

T_{max} = 2000°C , ΔP = 60 bar

KfK IMF I

Transient UO_2 / Zry-4 interaction experiments in argon

Figure 10.


50
50
Thermit
max. 1100 kg.
5mm Blech
600
850
1200
870
(230x300)
ø 30mm
Anschlag
Drahtseil
Prallblech
Klebsand
Abdeckung
Thermit
100 kg
Kernforschungszentrum Karlsruhe GmbH
Postfach 3640
7500 Karlsruhe 1
BETA - VORVERSUCHE
M103

Figore II

Table 2. BETA Experiment Matrix - Status 9/17/81

TEST	MELT	TEMP	TEST OBJECTIVES
H 1	300 kg steel	1700°C	Heat transfer and chemical reactions in metallic phase at 1700°C
H 2	300 kg steel + 300 kg Al_2O_3	1700°C	Heat transfer and chemical reactions in metallic and oxide phases at 1700°C
H 3	300 kg steel + 300 kg Al_2O_3	>2[illegible]0°C	Behavior of the melt at maximum temperature
H 4	300 kg steel + 300 kg Al_2O_3	>2000°C	Same as H 3 to test for reproducibility
H 5	300 kg steel	1500°C	Behavior of the metallic phase just before the beginning of solidification
N 1	300 kg steel	1400°C	Solidification events in the metallic melt
N 2	300 kg steel (solid cylinder)	1400°C	Behavior of a completely solidified metallic clump
N 3	300 kg Al_2O_3 + x kg SiO_2 + 100 kg steel (liquid)	1600°C	Solidification events in the oxide phase with gas formation and flow
N 4	300 kg Al_2O_3 + x kg SiO_2 + 100 kg iron (solid cylinder)	1500°C	Solidification events in the oxide phase without gas formation and flow
S 1- S 3	300 kg Al_2O_3 + x kg SiO_2	transient	Heat transfer and solidification events with variation of the material properties of the oxide phase

ENCLOSURE 4

CSNI

WG 1 Operational Experience and Human Factors

Lead - C. J. Heltemes, AEOD
Backup - G. Lainas, NRR

WG 2 Transients and Breaks

Lead - T. Speis, NRR
Backup - L. Shotkin, RES

WG 3 Primary Circuit Integrity

Lead - C. Serpan, RES
Backup - W. Johnston, NRR

WG 4 Source Term & Environmental Consequences

Lead - W. Houston, NRR
Backup - M. Silberberg, RES

WG 5 Risk Assessment

Lead - R. Blond, RES
Backup - A. Thadani, NRR