STATUS OF LMFBR SAFETY TECHNOLOGY

1. FISSION GAS RELEASE FROM FUEL PINS

February 1980

Edited and published by
WESTINGHOUSE ELECTRIC CORPORATION
for the United States Department of Energy
on behalf of the OECD Nuclear Energy Agency
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OECD NUCLEAR ENERGY AGENCY
38 BOULEVARD SUCHET
F-75016 PARIS
FRANCE
The Organization for Economic Cooperation and Development (OECD) was set up under a Convention signed in Paris on 14th December 1960, which provides that the OECD shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;

- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development;

- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The Members of OECD are Australia, Austria, Belgium, Canada, Denmark, Finland, France, the Federal Republic of Germany, Greece, Iceland, Ireland, Italy, Japan, Luxembourg, the Netherlands, New Zealand, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States.

The OECD Nuclear Energy Agency (NEA) was established on 20th April 1972, replacing OECD's European Nuclear Energy Agency (ENEA) on the adhesion of Japan as a full Member.

NEA now groups all the European Member countries of OECD and Australia, Canada, Japan and the United States. The Commission of the European Communities takes part in the work of the Agency.

The objectives of NEA remain substantially those of ENEA, namely the orderly development of the uses of nuclear energy for peaceful purposes. This is achieved by:

- assessing the future role of nuclear energy as a contributor to economic progress, and encouraging cooperation between governments towards its optimum development;
- encouraging harmonization of governments' regulatory policies and practices in the nuclear field, with particular reference to health and safety, radioactive waste management and nuclear third party liability and insurance;

- forecasts of uranium resources, production and demand;

- operation of common services and encouragement of cooperation in the field of nuclear energy information;

- sponsorship of research and development undertakings jointly organized and operated by OECD countries.

In these tasks, NEA works in close collaboration with the International Atomic Energy Agency, with which it has concluded a Cooperation Agreement, as well as with other international organizations in the nuclear field.

The Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers who have responsibilities for nuclear safety research and the nuclear licensing process. The Committee was set up in 1973 to develop and coordinate the Nuclear Energy Agency's work in nuclear safety matters, replacing the former Committee on Reactor Safety Technology (CREST) with its more limited scope.

The Committee's purpose is to organize international cooperation in nuclear safety. This is done essentially by:

(i) exchanging information about progress in safety research and regulatory matters in the different countries, and maintaining banks of specific data; these arrangements are of immediate benefit to the countries concerned;

(ii) setting up working groups or task forces and arranging specialist meetings, in order to implement cooperation on specific subjects, and establishing international projects; the output of the study groups and meetings goes to enrich
the data base available to national licensing authorities and to the scientific community at large. If it reveals substantial gaps in knowledge or differences between national practices, the Committee may recommend that unified approach be adopted to the problems involved. The aim here is to minimize differences and to achieve an international consensus wherever possible.

The technical areas at present covered by these activities are as follows: particular aspects of safety research relative to water reactors, fast reactors and high-temperature gas-cooled reactors; probabilistic assessment and reliability analysis, especially with regard to rare events; siting research as concerns protection against external impacts; fuel cycle safety research; the safety of nuclear ships; various safety aspects of steel components in nuclear installations; licensing of nuclear installations and a number of specific exchanges of information.

The Committee has set up a sub-Committee on Licensing which examines a variety of nuclear regulatory problems, provides a forum for the free discussion of licensing questions and reviews the regulatory impact of the conclusions reached by CSNI.
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FOREWORD

Fission gas release from fuel pins is one of the safety topics selected to be the subject of a state-of-the-art report, by the CSNI Group of Senior Experts on LMFBR Safety R & D of the OECD Nuclear Energy Agency.

Following the OECD's request, the U.S. Department of Energy (DOE) agreed to assume the leading responsibility in preparing the report. In January 1979 the Advanced Reactors Division, Westinghouse Electric Corporation, began its preparation on behalf of DOE. Under the direction of Mr. John Graham, Manager, Nuclear Safety and Reliability Engineering, and Mr. Roger W. Tilbrook, Manager, Reactor Safety Engineering, the first draft was completed in August, 1979, and submitted to representatives of the CSNI members for review. A review meeting was held on August 19, 1979 prior to the ANS/ENS International Meeting on Fast Reactor Safety Technology in Seattle, Washington, U.S.A. under the chairmanship of Mr. Tilbrook. The attendees were Messrs. Brown (WARD), Cowking (UKAEA), Haga (PNC), Hori (PNC), Justin (CEA), Kramer (KfK), and Reynen (CEC-JRC, Ispra). Comments were also received from Drs. M. A. Groimes of the Fast Reactor Safety Technical Management Center, J. H. Scott of Los Alamos Scientific Laboratory on behalf of the U.S. Department of Energy and the Nuclear Regulatory Commission, respectively, and from Mr. L. M. McWethy of the Hanford Engineering Development Laboratory. The reviewers, in general, accepted the technical content of the report, the main modifications being to incorporate additional details from other national programs. This information was provided by:

Mr. C. B. Cowking  Risley Nuclear Power Development Laboratories
                        United Kingdom Atomic Energy Authority
                        Warrington, Cheshire, United Kingdom

Dr. K. Haga  Dr. M. Hori  Power Reactor and Nuclear Fuel Development Corporation
                        O-Arai Engineering Center
                        O-Arai, Ibaraki-Ken 311-13, Japan

Mr. F. Justin  DSN/SESPP
                        CEA - CEN de Fontenay-aux-Roses
                        B.P. No. 6 F-92260 Fontenay-aux-Roses
                        France
With this additional information incorporated, a final draft was issued for further review by the working-level representatives and authorities of various countries. Comments were also received from Drs. H. K. Fauske of the Fast Reactor Safety Technical Management Center, USDOE, R. Curtis of the Advanced Reactor Safety Research Division, Office of Nuclear Regulatory Research, USNRC, and R. T. Lancet of the Atomics International Division, Rockwell International. The assistance in co-ordinating the international aspects of the preparation of this report is rendered by Dr. M. E. Stephens of the Nuclear Safety Division, OECD Nuclear Energy Agency.

The preparation of this report represents the culmination of work by members of the staff of the Advanced Reactors Division, Westinghouse Electric Corporation. Their significant contributions are acknowledged:

Mr. R. G. Brown* - Editor and author
Mr. S. A. Chang - Editor and author
Dr. N. P. Kadambi - Author

*Current address: Science Applications, Incorporated, Palo Alto, CA.
EXECUTIVE SUMMARY

Fission gas release from random fuel pin failure is one of the specific local faults whose potential consequences on safety must be evaluated. It has been the subject of considerable research for a number of years in most nations devoted to the development of the liquid metal fast breeder reactor (LMFBR). This report presents the status of the safety technology that has evolved from the vast amount of publicly available data reporting this research.

It has been postulated that fission gas release from a single fuel pin failure during normal operation may have the potential to lead to pin-to-pin failure propagation in the fuel subassemblies of LMFBRs. This report presents the safety technology for the fission gas release phenomena in LMFBRs of current designs that employ mixed-oxide fuel and stainless steel cladding, according to the possible mechanisms by which the potential effects can influence adjacent pin performance, namely:

A. Clad temperature transients:

- Local gas blanketing, which includes:
  1. Gas jet impingement
  2. Bubble adherence
  3. Bubble entrapment in a blockage wake

- Two-phase flow effects
- Coolant flow reversal

B. Transitory mechanical loadings:

- Deformation due to bending moment
- Deformation due to contact force
- Resonance vibration

The important conclusions are that, for current reactor designs employing mixed-oxide fuel and under the normal steady-state operating conditions:
• The safety technology on fission gas release from fuel pins is well developed with the support of a vast amount of experimental data and many years of experience.

• Fission gas release by itself is not a potential cause for pin-to-pin failure propagation, and therefore, further research is deemed unnecessary.

• Fission gas release can accentuate the effects of a local blockage on fuel performance if the gas bubbles are entrapped in the wake, even to the extent of causing dryout and clad melting in the wake of an otherwise non-boiling blockage. In view of the minimal evidence for the occurrence of blockages and the evidence for the self-stabilizing characteristics of blockage induced damage, it is recommended that further research on this restricted aspect of fission gas release be an extension of local blockage studies if further consideration is deemed necessary.
I. INTRODUCTION

A. Purpose of the Report

This report presents the current status of the safety technology developed to deal with fission gas release from fuel pins in liquid metal fast breeder reactors, and how this technology relates to quantitative risk assessments supportive of safety reviews and licensing processes. This report also provides a basis for peer reviews and dialogue as to the status and direction of continuing research.

B. Objectives for the Development of the Technology

It has been postulated that fission gas release from a single fuel pin failure during normal operation may be capable of leading to the failure of additional pins. It is one of the specific local faults which must be evaluated for potential safety consequences. A local fault is generally defined to be any condition which results in a regional mismatch of power generation and coolant flow outside the bounds considered in normal design evaluations. This power/flow imbalance may either influence only a small region within the pin bundle, or it may be a subassembly-wide condition affecting the performance of every pin in the bundle. The study of fuel pin failure propagation is concerned with the occurrence of these faults and the potential for the spread of damage throughout the subassembly, which might make continued operation unsafe or lead to a loss of coolable geometry within the subassembly. Of ultimate concern is the possibility that the damage will propagate to other subassemblies before the situation is detected and the reactor shut down. Only in such extremely unlikely scenarios leading to a core disruptive accident would there be some risk to the health and safety of the public.

Current LMFBR fuel pin designs employing mixed-oxide fuel and stainless steel cladding contain margins which provide a high level of assurance that the pin will not fail during its design lifetime. However, as a large number of pins are used in the core ($\sim 10^5$ in current large plant designs), it is not possible, statistically, to
preclude pin failures with any degree of confidence. Random pin failures are anticipated and as such, are not in themselves considered to be true local faults. Because fission gas release following clad breach can lead to a transitory disturbance of local coolant flow, considerable attention has been given in the safety technology programs of all nations undertaking LMFBR development to evaluating the potential for adverse effects from this occurrence.*

C. Scope of the Report

The safety technology related to fission gas release can be categorized in accordance with the mechanisms through which the potential effects can influence the performance of adjacent pins. The postulated mechanisms for failure propagation depend upon the location and geometry of the initial pin failure and the fission gas release rate, and may be considered to be the mechanisms which would cause clad temperature transients on adjacent fuel pins, or those which would cause transitory mechanical loadings on adjacent fuel pins due to pressure pulses. Included in the former category are:

1. Local gas blanketing of the adjacent pin surface, which results from gas jet impingement, bubble adherence to some surface in the fuel subassembly, and bubble entrapment in the wake of local blockages. These phenomena are discussed in Section III. A.

2. Two-phase flow effects which include the reduction in coolant flow due to two-phase pressure losses and due to the decrease of the effective coolant density. These effects are considered in Section III. B.

3. Coolant flow reversal which results from large gas release and the resulting slug flow. This phenomenon is treated in Section III. C.

*In the U.S. Department of Energy program,¹ the issue has been identified as a major goal in establishing the first of the four lines of assurance - LOA 1, Accident Prevention. The work breakdown structure for fission gas release phenomena under this program is in general agreement with the scope of this report.
The various types of transitory mechanical loadings and failure modes which have been postulated are evaluated in Section III. D.

These discussions are concerned only with pin failures which occur under normal steady-state operations. Failures produced by other initiating events such as reactor transients are not within the scope of this document. Some consideration of the effect of fission gas entrapped in the wake of local blockages is presented in Section III. A. 3. Although this is not normal steady-state operation as blockage formation is considered to be very unlikely, it is an area of local faults where fission gas release is of concern and is therefore included in this report.

D. Summary of Current Status

Fission gas release phenomena have been the subject of numerous analytical and experimental investigations (e.g., References 2-6). As a result, there is a large quantity of data which demonstrates that fission gas release from fuel pins does not cause additional fuel failures under normal steady-state operating conditions and that further research is therefore deemed unnecessary, at least as far as mixed-oxide fuel is concerned.

If fission gas becomes entrapped in the wake of a local blockage, it can enhance the blockage effect on fuel performance. Although the extent of this enhancement has not been clearly delineated, there is reassuring evidence that the consequences remain inside the sub-assembly envelope and do not escalate rapidly. It is therefore recommended that this combined effect be studied as an extension of the research on local blockage if it is considered necessary at all.
II. FUEL PIN DESIGN AND FISSION GAS RELEASE EXPERIENCE

To provide a background for the discussion, Section II. A. summarizes the fuel pin design parameters and operating conditions of the current LMFBRs. Experience with fission gas release from testing and reactor operation is summarized in Section II. B, to demonstrate and support the validity of the technologies developed therefrom.

In establishing operating limits for the fuel, blanket and control pins, consideration must be given to a variety of factors. These include operating temperatures, types of transients expected, stress loadings, and material properties, which contribute significantly to the allowable operating conditions to which the cladding can safely be subjected. In order to achieve a balanced design, i.e., one which is both safe and also provides the characteristics needed for efficient and economic operation, it is necessary to use reasonably conservative, rather than extremely conservative, limits. The primary objective in establishing the allowable operating conditions for the fuel pins, therefore, is to define realistic limits which are based upon the capability of the actual cladding material. To this end, in order to define the performance limitations of the cladding under steady-state and transient conditions, strong emphasis has been placed on in-pile irradiation programs. Furthermore, extensive fuel pin evaluation programs are being conducted in operating LMFBRs to confirm and increase the usefulness of the data obtained from in-pile testing. For example, the Fast Flux Test Facility (FFTF) and Clinch River Breeder Reactor (CRBR) projects are being built upon the experience gained in EBR-II, GETR, and TREAT, just as Phenix upon Rapsodie and PFR upon DFR, and currently Super Phenix and CDFR upon Phenix and PFR respectively.

To supplement the in-pile testing program, out-of-pile testing on prototypic cladding material (both irradiated and unirradiated) is also used to better understand the response characteristics of the cladding to various loading conditions. These data are utilized in conjunction with available analytical techniques to establish quantitative limits which can then be used in performance evaluations. The limits thus established are "yardsticks" by which the cladding performance can be measured during analysis of steady-state and transient operations.
Some of these criteria are mentioned in the discussions of the fission gas release technologies in Section III to illustrate the margins to failure for various conditions.

A. Fuel Pin Design and Operating Conditions

The probability of pin failure during steady-state operation is expected to be extremely small because of the conservatism incorporated in the design process to assure that the limiting pins will incur no loss of integrity during transient conditions prior to reaching the goal burnup. Nevertheless, because of the large number of pins in the core, it is not possible, statistically, to preclude pin failures. Furthermore, there is a small chance that a minor manufacturing defect which could reduce the pin lifetime might pass undetected through the numerous quality control checks. Because of the severe economic penalties associated with a premature shutdown to remove failed fuel, the plant must be designed to accommodate failures. One aspect of accommodation is to demonstrate the benign consequences of fission gas release.

The fuel pin design parameters and steady-state operating conditions of several current LMFBRs are summarized in Tables II-1 and II-2 respectively. These data are provided as reference material for the evaluations on fission gas release effects presented in Section III.

B. Fission Gas Release Experience

The most convincing evidence that fission gas release does not affect the safety and reliability of plant operation is the world-wide operating experience which has been acquired with sodium cooled reactors. Hundreds of thousands of mixed-oxide pins have been irradiated in power reactors and test facilities. Fuel performance under conditions prototypic of those expected for a commercial LMFBR has been excellent. The failure rates for these operating conditions are on the order of 0.1% of all pins in the core. In some instances, pins have been operated well beyond their design limits to obtain information on the margins in the design, and to understand the conditions which could produce failure and the resultant consequences.
<table>
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<th>PARAMETER</th>
<th>UNIT</th>
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<th>F. R. OF GERMANY SNR-300</th>
<th>ITALY PEC</th>
<th>JAPAN JOYO MARK 2</th>
<th>JAPAN MONJU</th>
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Note: Data taken from "General Characteristics of Fast Reactor Fuel Pins" by K. Kammerer, in Transactions of the American Nuclear Society pg. 301-304, April 21-25, 1975, and partially verified or revised by the International Review Group Members.
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<td><strong>EOL</strong> 585</td>
<td>592*</td>
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<td>610</td>
<td>610</td>
<td>650**</td>
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<td>615</td>
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<td>50.0* 24.0**</td>
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<td>5.7</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>MAXIMUM SUBASSEMBLY PRESSURE DROP</strong></td>
<td>BAR</td>
<td>4.5</td>
<td>5.0</td>
<td>2.9</td>
<td>3.3</td>
<td>5.3</td>
<td>7.3</td>
</tr>
</tbody>
</table>

**REMARKS**

* DESIGN
** PEAK NOMINAL
* WITH SYSTEMATIC FACTORS
** 2σ VALUES
HETEROGENEOUS CORE
* 2σ VALUES

**NOTE**: DATA PROVIDED BY THE INTERNATIONAL REVIEW GROUP MEMBERS.
Even under such severe conditions, there has never been any instance of rapid pin-to-pin failure propagation, nor has there been any observable effects of fission gas release on neighboring pin performance. This section summarizes this experience.

1. In-Pile Experience

There is a substantial quantity of failed fuel performance data from both in-pile tests and during normal reactor operation. In no instance has any fuel failure propagation been attributed to fission gas release. Even in the DFR MK II subassemblies in which there were numerous failures (Section III.A.2), the failures all appeared to be independent events and not attributable to failure propagation due to fission gas release.

References 7 through 15 describe the LMFBR fuel performance experience of the various European programs. The conclusions are all very similar and only the consensus is presented in this section. In general, mixed-oxide fuel performance has been good. Under irradiation conditions prototypic of normal LMFBR operation, failures have been infrequent. For example, the initial loading of fuel in Phenix achieved a burnup of 72,000 MWD/DM with only one failure. 7 The fission gas release was described as "small puffs" released over a period of several months. The quantities were so small that it was some time before the subassembly location could be determined. This behavior is corroborated by experience in other reactors. While fission gas release data have not been reported specifically, the response of the reactor cover gas monitors to fission product release indicates that release rates are low. In most instances, the failures occur near the hotter end of the pin and are intergranular cracks. Those instances where large splits in the cladding have been observed have occurred when the fuel was operated for a period after breach. In these cases, it is almost certain that the breach was originally small, and degraded as a result of fuel-sodium chemical reaction.

The mixed-oxide fuel failure data base for U.S. designs originates primarily from testing and operation in EBR-II. The Run-to-Cladding Breach (RTCB) portion of the steady-state fuel
irradiation program has demonstrated that fuel failure is a benign event and additional pin failures do not occur as a consequence of fission gas release.

Failures in the RTCB program have occurred over a wide range of pin power, cladding temperature, and burnup. Pins with internal gas pressures as high as 117 bars (1700 psi) have failed in RTCB testing. In each instance, the fission gas release has been gradual, and no detrimental effects on neighboring pins or subassembly duct have been observed. In most cases, the failed pins retained a significant portion of their fission gas inventory. In general, the cladding defects have been characterized as pin-holes or hairline cracks along the steel grain boundaries, or a porous, frit structure, rather than a large, well-defined crack. Nearly all failures have occurred in the fueled region near the hot (upper) end of the pin, where the cladding is weakest. One exception was a failure in the insulator pellet region just below the plenum which has been attributed to contamination during pin fabrication. This occurrence, in Subassembly designated X106, together with other experiments is discussed in detail in Appendix A.

Although a significant amount of data exists, few true endurance failures under prototypic conditions have been observed. Most of the breaches have been caused by some abnormality, such as bundle distortion creating local hot spots. As a consequence, failure has occurred earlier in life than expected. Yet even under these severe conditions, no deleterious effects either to the neighboring pins or to the subassembly wall have been observed. Even under conditions of pin-to-pin contact, where the distortion would worsen the effects of fission gas release, there has been no evidence of failure propagation.

2. **Out-of-Pile Fission Gas Release Experiments**

Because fission gas release rates are rarely measured in-pile, out-of-pile tests have been performed to support model development for safety analysis.
Early out-of-pile experiments were conducted in the U.S.\textsuperscript{17} and in Britain\textsuperscript{18} with gas injected onto a heated plane or curved surface submerged in water, to study the structure of a submerged gas jet impinging onto the surface and to investigate the mechanism of the associated heat transfer. Tests in sodium with a 3-pin bundle were later performed by Argonne National Laboratory.\textsuperscript{5, 19} In these tests, heated argon or xenon gas was injected through the injector onto the electrically heated pins, and the temperature responses in the impingement region were recorded. The results showed that the pin surface temperature increased because of the gas release, and the highest temperature rise was in the gas impingement area. Under the most severe conditions, with heat flux of 250 \text{ W/cm}^2 (0.793 \times 10^6 \text{ Btu/hr*ft}^2), the maximum cladding temperature rise was $\sim 240^\circ \text{C}$ ($\sim 430^\circ \text{F}$). These tests\textsuperscript{5, 17, 18, 19} are described in more detail in Section III.A.1, Gas Jet Impingement.

To evaluate the thermal and hydrodynamic effects of fission gas release, out-of-pile experiments were also conducted in Japan\textsuperscript{20} using a 37-pin bundle with a central gas injector pin, and with electrically heated pins and dummy pins. Argon gas was injected in spurts and continuously through nozzles of various sizes into the flowing sodium stream and onto the adjacent pins to simulate gas release in the fueled section of the fuel pins. The temperature response of the pin surface was very similar to that reported in References 5 and 19 mentioned above. It was also found that the measured pressure pulses were less than 1/5 of the initial gas plenum pressure in the transient release experiments, and no fuel damage due to mechanical loading was observed. A more detailed description of these experiments is presented in Appendix A.

To develop gas release models, out-of-pile tests with irradiated mixed-oxide fuel pins and with simulated pins have also been conducted in many countries. The major objectives were to establish the relationship between gas release rate and various parameters such as burn-up, size and location of cladding breach, pressure differential between gas plenum and fuel bundle, gas release time, fuel bundle condition, and number of failed pins. While some of these tests are summarized in Appendix A, others are referred to or briefly described in Section III where the basis of the safety technology is discussed.
III. STATUS OF TECHNOLOGY FOR FISSION GAS RELEASE PHENOMENA

This section presents current safety technology for fission gas release phenomena according to the postulated mechanisms by which the potential pin-to-pin failure may propagate through the fuel subassemblies of LMFBRs of current design (mixed-oxide fuel and stainless steel cladding). The applicability of this technology to a particular design can be checked by comparing the fuel pin design parameters and the operating conditions with those listed in Tables II-1 and II-2 upon which this technology is based.

A. Local Gas Blanketing

One of the postulated modes of propagation is local overheating of adjacent pins by gas blanketing. This condition may be either short term, present only during the initial period when gas is released following pin failure, or long term, as in gas bubble adherence and gas entrapment in a blockage wake. While any form of gas blanketing as a consequence of pin failure is considered to be very improbable, such concerns have been the object of several experimental and analytical assessments. This section examines all of these phenomena and summarizes the conclusions of these evaluations.

1. Gas Jet Impingement

It has been postulated that gas release following breach could be sufficient to produce a jet which impinges upon the surface of a neighboring pin. Reduced heat transfer in the region of the impinging gas would then result in local overheating and perhaps failure of the adjacent pin. Figure III-1 is a schematic representation of this phenomenon. Experimental and analytical studies on gas jet impingement have focused on defining the conditions necessary to elevate the local temperature of the neighbor pin. By comparing these results to the conditions anticipated from stochastic pin failure, the potential for additional failures can be evaluated.
Figure III-1. Schematic Diagram of the Fission Gas Jet Impingement in LMFBR Subassemblies

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The first jet impingement experiments were performed by Argonne National Laboratory (ANL) using a water-cooled, rectangular flow channel with a heated surface. The test section was designed to allow a fixed volume of gas to be discharged through an orifice and impinge on the heated wall. Thermocouples measured the resulting wall temperature transients. The test parameters included the following key variables:

Coolant velocity: 3.0, 6.1, 9.1 m/sec (10, 20, 30 ft/sec)
Orifice diameter: 0.41, 0.64, 1.30 mm (0.016, 0.025, 0.051 in.)
Plenum gas pressure: 34.5, 51.7, 68.9 bars (500, 750, 1000 psia)
Surface heat flux: 157.6, 315.2 W/cm² (0.5 x 10⁶, 1.0 x 10⁶ Btu/hr·ft²)

Tests were run with and without bypass flow around the heated channel. High-speed motion pictures of an unheated test section showed the gas jet development and structure.

The primary purpose of the study was to investigate the mechanism of heat transfer during jet impingement. It was concluded that the dominant mode of heat transfer at the point of jet impingement was spray cooling by liquid entrained by the jet as it traveled from the orifice to the heated surface. In fact, the area of the impinging jet was actually cooled below its steady-state temperature. The only instances where elevated temperatures were recorded in the impingement region were in runs where the gas release voided the channel and thus no liquid was available for entrainment. This phenomenon occurred because of the narrow, confined flow channel (equivalent in area to about two triangular LMFBR subchannels). In a reactor pin bundle, gas could expand to adjacent flow channels. It was noted that spray cooling might be less effective in sodium where convective heat transfer is not as dominant as in water. However, calculations based on the worst conditions encountered in the experiments predicted that an impinging gas jet in an LMFBR would not produce pin failure.
These findings are corroborated by a British experimental program\textsuperscript{18} which determined the structure of a submerged jet impinging on a surface. They also observed significant fluid entrainment into the jet. This work described the two-phase structure at the impingement surface in terms of three regions: a core region with a relatively small droplet concentration, an impingement zone with the bulk of the entrained fluid and a thin liquid film, and an outermost region with a thicker liquid film and gas flow parallel to the surface. This study confirmed that jet impingement does not lead to a large area where there is complete cladding dryout. It appears that spray cooling and the residual liquid film will preclude a rapid, adiabatic rise in the local fuel pin temperature.

To eliminate the major uncertainties associated with the previously described ANL water-cooled experiments, a series of tests was performed in a sodium-cooled rig which simulated the bundle flow geometry.\textsuperscript{5,19} The test section, illustrated in Figure III-2, consisted of three uniformly electrically heated pins arranged in an equilateral triangle surrounded by a triluted shroud. The pins were prototypic of FFTF and CRBR fuel designs with a 0.584 cm (0.230 in) OD. They were spaced by a 0.157 cm (0.062 in) wire wrap, wrapped helically around the heaters at a 30.5 cm (12.0 in) pitch. Each wire had seven internal thermocouples at 7.6 cm (3.0 in) intervals. The heaters were provided with internal thermocouples, spot-welded at the junction onto the inside of the heater sheath, and located axially \(\sim 18.2\) cm (7.2 in) downstream from the test-section inlet and facing the central coolant subchannel. The total flow area of the test section was 1.476 cm\(^2\) (0.229 in\(^2\)).

The coolant inlet temperature for each run was 315\(^\circ\)C (600\(^\circ\)F). Argon, heated to 510 or 720\(^\circ\)C (950 or 1328\(^\circ\)F) was released through the injection needle which penetrated the shroud between two of the heater pins. Subsequent tests were performed with xenon (at 510\(^\circ\)C with a 0.058 cm nozzle) to examine the effect of gas composition. The position of the injection needle was adjusted to \(\sim 0.142\) cm (\(\sim 0.056\) in) from the pin that was to be blanketed. This distance corresponds to the minimum spacing in the FFTF and CRBR fuel rod bundle designs. The pin subjected to the impinging gas could be

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Figure III-2. Schematic Representation of a Cross-Section of the Three-Pin Sodium Test Section at the Gas-Injection Point
moved axially so the location of the gas jet relative to the internal thermocouples could be varied, thus permitting investigation of the axial temperature profile in the impingement region. Steady-state tests were performed to eliminate the effects of flow and temperature transients.

Some 460 runs were made which studied the following ranges of parameters:

1) Gas type: argon and xenon

2) Needle I.D.: 0.033, 0.058, and 0.084 cm (0.013, 0.023, and 0.033 in)

3) Heat flux: 25 to 250 watts/cm² (0.793 x 10⁶ to 7.93 x 10⁶ Btu/hr•ft²)

4) Coolant flow rate: 140 to 690 g/sec (0.31 to 1.52 lbm/sec)

5) Gas plenum temperature: 510°C and 720°C (950°F and 1328°F)

6) Gas plenum pressure: 7 to 54 bar abs (101.5 to 783.2 psia)

7) Relative axial position of needle and internal pin thermocouples: Adjustable

The following discussion summarizes the results and conclusions of this extensive series of tests. More data are available in References 5 and 19.

For low gas pressures, the axial distance affected by the impinging gas was found to be ±1.1 cm (0.43 in). As the plenum pressure was increased, the impingement area became less clearly defined. This is illustrated by the three data sets which comprise Figure III-3. It can be seen that the pronounced temperature peak that was observed in the 7 bar gas pressure test is absent for the 14 and 54
Figure III-3. Cladding Temperature Rise, $\Delta T$, in the Impingement Area Versus $Z$ for the Operating Conditions Listed

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bar tests under the same conditions. In all cases, the skewed temperature profile resulted from the deflection of the gas jet by the flowing sodium.

The data clearly indicate that, as expected, gas jet deflection by the coolant increases for

a. increasing coolant velocity,
b. decreasing gas jet diameter (at constant gas velocity);
c. decreasing density in the gas jet.

Coolant velocity was found to have little or no effect on cladding temperature for gas pressure \( P_g \) greater than three times the local pressure on the coolant channel \( P_s \). Some effect was noticed as the plenum pressure was reduced below this level.

For the most severe combination of test parameters considered (heat flux of 250 W/cm\(^2\) or 45.9 W/mm, 0.793 x 10\(^6\) Btu/hr-ft\(^2\) or 14 Kw/ft), the maximum cladding temperature rise was \( \approx 240^\circ C \) (432\(^\circ\)F). This occurred over a narrow range of gas flow rates with a plenum-to-channel pressure ratio \( \frac{P_g}{P_s} \) less than 3. Figure III-4 illustrates the test data for the 250 watts/cm\(^2\) (0.793 x 10\(^6\) Btu/hr-ft\(^2\)) heat flux. The peak temperature occurred for a plenum pressure of \( \approx 5.2 \) bar abs (75.4 psia). Extrapolation of the xenon data to this heat load yields a maximum temperature rise of \( \approx 160^\circ C \) (288\(^\circ\)F).* In general, the temperature rise was proportional to the heat flux.

The temperature of the impinging gas was found to have little effect on the cladding temperature for the values considered in the test. This was because the sodium still played the dominant role in the heat transfer process in the impingement. Distinct regions of dryout where convection to the gas jet dominates do not appear to have

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* Only a limited number of runs were performed with xenon. The maximum heat flux used in the xenon tests was 126 watts/cm\(^2\) (0.4 x 10\(^6\) Btu/hr-ft\(^2\)).
occurred in these tests. Spray cooling appeared to dominate for plenum-to-channel pressure ratios in excess of \( \sqrt{3} \). For lower gas pressures, the liquid film remains on the cladding surface and thus governs the heat removal process.

The maximum cladding temperatures as a function of plenum-to-channel pressure ratio \( (P_g/P_s) \) are indicated in the data sets presented in Figure III-5. It can be seen that the maximum cladding temperature increases were recorded for relatively small values of \( P_g/P_s \). Hence, all other factors being the same, a higher plenum pressure will cause a smaller temperature rise.

It was discovered that for the bundle geometry and test conditions used in this program, the breach aperture that yielded the maximum temperature rise was somewhere between 0.033 and 0.084 cm. The first three data sets in Figure III-5 show the variation in the results with the three injection needle diameters used in the testing. The temperature rises for the middle-sized needle (0.058 cm) exceeded the values recorded for the larger and smaller diameters, when all other parameters were held constant.

To place these test results in perspective, it must be realized that the gas flow rates examined in the experiment were orders of magnitude greater than those observed in both mixed-oxide pin failures and the GE failed-fuel simulation tests.

The gas flow rates used in the impingement tests ranged from \( \sqrt{0.14} \) to 3.2 gm/sec. By comparison with the data in Table A-3 in Appendix A, it can be seen that in most cases, the flow rates in the impingement tests were greater than the maximum values achieved in the failed pin simulation experiments (see Appendix A). A comparison with the gas release data from the B3B experiment (Appendix A), has shown the flow rates in the ANL impingement tests to be a factor of \( 10^4 \) larger.

The other fuel performance data summarized in Appendix A also demonstrate that gas release rates of failures in the fuel zone are very gradual. The increased resistance of the fuel pellets would
Figure III-5. Cladding Temperature Rise, Δ T, in the Impingement Area Versus $P_g/P_s$, for the Operating Conditions Listed

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likely preclude gas jet formation for breaches in the fueled zone. Furthermore, the breach size simulated by the injection needle is much larger than the pin-holes which are most commonly observed. An analysis of apertures smaller than the 0.033 cm used in the tests determined that the flowing sodium would deflect the escaping gas before it could reach the adjacent pin. Thus it appears that jet impingement will not occur for the failure conditions anticipated in the fueled zone. Gas jets could be formed as a result of breaches in the plenum region. However, since there is negligible heat generation in this area, the thermal transient imposed on the neighboring pin will not lead to failure.

Thus, for the operating conditions of U.S. mixed-oxide fuel designs, the failure conditions simulated by these experiments are unlikely to be encountered. The gas release behavior encompasses any condition which might be encountered during operation. Nevertheless, these experiments have formed the basis for conservative analyses to demonstrate the margins in the FFTF and CRBR fuel designs.

The results of the analysis for FFTF illustrate the margin to failure. The maximum cladding temperature for the worst-case jet impingement conditions was calculated for the most limiting fuel pin operating conditions (3σ hot channel factors were used). The maximum cladding temperature as a function of axial position (above core midplane) is shown in Figure III-6 for both beginning of cycle 5 (when maximum steady-state duct temperature occurs), and end-of-life conditions. Results are presented for initial and rated FFTF operation (316°C and 422°C, or 600°F and 792°F reactor inlet temperatures respectively). Using the updated temperature integrity limit for FFTF cladding (See Figure III-7), it can be seen that failure of even the most limiting pin subjected to the worst-case impingement conditions is unlikely. These calculations were performed for steady-state impingement conditions and thus conservatively assume an infinite supply of fission gas.

In addition to these evaluations, calculations have been performed to compare the fission gas release rates in the ANL experiments with those predicted by conservative analytical models for
Figure III-6. Cladding Midwall Temperatures Resulting from Gas Jet Impingement Assuming Hypothetical Worst Gas Jet Flow Rates and Unlimited Gas Supply (FFTF).
Figure III-7. Original and Updated FFTF Cladding Temperature Integrity Limits
breaches occurring during operation. Using the Murata gas release model\textsuperscript{24} with a 0.02 mm (0.0008 in.) effective diametral gap (which is conservative beyond 1 a/o burnup) the maximum gas flow rates as a function of axial failure position were computed. Comparing these to release rates in the ANL tests revealed that even the worst-case gas flow rates were at least an order of magnitude less than those used in the experiments. Hence, impingement is unlikely to occur for significant burnups.

At the beginning of life a relatively large fuel/clad gap may exist prior to restructuring. Hence the resistance to gas flow is reduced. However, the gas inventory in the pin is relatively low. It has been shown\textsuperscript{21} that when the failure conditions are such that impingement could occur, the duration of the transient is so brief that large temperature rises in the adjacent pin do not occur.

Stress analyses have been performed\textsuperscript{25} to determine the response of a fuel pin subjected to nonuniform temperature distributions corresponding approximately to those experimentally determined in the ANL tests. The most severe test conditions (240\textdegree C temperature rise, for 250 w/cm\textsuperscript{2} heat flux case) were analyzed. It was found that the maximum equivalent uniaxial stress (√2200 bar or √32,000 psi) was more than 25% less than the yield strength of the irradiated cladding at the elevated temperatures experienced during the impingement. Hence, failure due to thermal stresses, even for the conservative test conditions, will not occur. Creep effects for these conditions are insignificant, as the time to rupture at the maximum stress and clad temperature is of the order of hours. Due to the limited fission gas inventory, a gas jet would only persist for a few seconds or less. Hence creep rupture would not occur.

Thus, it is concluded that the formation of fission gas jets from failed pins is extremely improbable for failures in the heated zone of existing U.S. mixed-oxide fuel designs. Even if gas jet impingement is postulated, the resulting transient on the neighboring pins will not lead to additional failures. This conclusion is also corroborated by the Japanese PNC test\textsuperscript{20} as described in detail in Appendix A.
Furthermore, even if an additional pin failure did occur, its consequences would not significantly affect fuel subassembly performance. Gas release from secondary failures would be directed toward the initially failed pin and thus any propagation would be self-limiting and confined to a small region of the bundle. Hence, it can be concluded that gas jet impingement due to fission gas release is not a credible mechanism of pin-to-pin failure propagation.

2. Gas Bubble Adherence

Another form of local gas blanketing is the adherence of gas bubbles to the unwetted clad surface of fuel pins. These gas bubbles may be fission gas released from failed fuel pins or gases disentained from the coolant. This mode of localized overheating has been observed in the British fuel irradiation experiments\textsuperscript{11, 26} performed in the Dounreay Fast Reactor (DFR) with downward coolant flow. The post-irradiation examination showed pins with elliptical or "teardrop"-shaped discoloration, or "tearstains." Most of the pin failures occurred at the center of such teardrop marks which appeared in the fueled section only. These failures took the form of longitudinal cracks, 10 - 25 mm (0.4 - 1.0 inch) long, surrounded by the "tearstains" which delineated the overheated area.

All the failures seen occurred over the uppermost 130 mm (5.1 inch) of the fuel column at temperatures below 400\textdegree C (752\textdegree F) and coolant velocity below 2 m/sec (6.6 ft/sec), conditions which do not normally prevail in large power LMFBRs. Teardrop marks without failures occurred over the top 150 mm (5.9 inch) of the fuel column and also at grid pips. There were numerous tearstain/tearstain, tearstain/failure, and failure/failure paired events approximately facing one another on adjacent pins, suggesting that these events were related either because the initial cause of the hot spot was big enough to simultaneously blanket portions of both pins, or because gas emitted from the first failure enlarged the blanket to cause failure of the second pin. The gas emitted from this second failure was directed back to affect primarily the part of its neighbor which had already failed,
across the whole subassembly. The fact that tearstains did not generally extend around more than a third of the pin circumference would support this model.

This observation leads to the conclusion that gas bubble adherence will not cause pin failure propagation. Furthermore, the probability of the occurrence of gas bubble adherence in LMFBRs of current design is considered small because:

- Gas entrainment from free surface in the upper coolant plenum is less likely since the flow direction in the core is upward, rather than downward as in DFR.

- Drag and buoyancy forces on the gas bubbles are both directed upwards, rather than opposing each other as in DFR, so that continued adherence of the bubbles to the fuel pins at power is less likely.

3. Fission Gas Entrapment in Blockage Wake

An LMFBR subassembly with a local blockage may accumulate inert gas downstream of the blockage and the gas blanketing may cause the local temperature to increase further. This inert gas may originate either from fission gas release from a fuel pin with breach upstream or in the wake of the blockage, or from entrainment by the sodium at a free surface. A local blockage in the fuel zone is considered to be a very unlikely event due to the design features of current LMFBR fuel subassembly bundles, inlet nozzles and core inlet modules. However, the consequences of fission gas entrapment in the wake of local blockage in the fuel zone may cause concern.

a. U.S. Experiments

Oak Ridge National Laboratory in the U.S. conducted out-of-reactor tests\textsuperscript{27} using a test subassembly with 19 electrically-heated pins in a scalloped duct. The radial geometry of this test subassembly was prototypical of the FFTF and CRBR fuel
designs. The heated length was 53 cm (21 inches). The six central flow channels were blocked by a non-heat-generating stainless steel blockage plate. The blockage area was 12% of the total cross-sectional flow area of the test subassembly. The blockage plate was 6.4 mm (1/4 inch) thick and was brazed to the central rod 38 cm (15 inches) downstream of the start of the heated section. Spacer wire and internal thermocouples were used for temperature measurements. The tests were conducted to study the effect of fission gas entrapped in the blockage wake with quasi steady-state boiling downstream of the blockage and at single-phase steady-state condition. The quasi steady-state boiling condition was produced by decreasing the test-section flow at constant heater power just enough to initiate sodium boiling, and was maintained long enough to give consistent data without endangering the integrity of the fuel pin simulators. The on-set of boiling during the tests was determined by monitoring thermocouple temperatures and audio signals from the electro-magnetic microphone attached to the test housing and from hydrophones and lithium niobate crystals. Boiling was evident when the temperature exceeded the expected sodium saturation temperature, and when the "crackling" noise of sodium vapor bubble collapse was detected. The tests were run with and without inert gas injection. To simulate fission gas release, argon at the inlet temperature was injected into the sodium stream at the test section inlet and at various rates (all <1 cm³/sec).

The quasi steady-state boiling test results showed that:

- The argon gas injection (with void fraction $\varphi = 0.001$ and .004 at entrance) was partially responsible for the presence of vapor in the unblocked channels surrounding a 6-channel blockage wake, although boiling across the entire bundle cross-section did not occur.

- The sodium bulk temperature in the free stream adjacent to the blockage wake (calculated from the power-to-flow ratio) strongly influenced the radial extent of this boiling zone propagation because it governs the condensation capability of sodium vapor.
Increasing the void fraction of the inert gas from 0.001 to 0.004 did not essentially influence the extent of the local boiling zone.

Stable temperature conditions throughout the bundle were in evidence at the end of the quasi steady-state boiling periods with gas injection.

There was no evidence of cladding dryout (loss of liquid film) in the local boiling zone during quasi-steady-state boiling.

The single-phase steady-state test results showed that:

- From the test runs analyzed, no general correlation could be established between blockage-wake temperature increase due to gas injection and void fraction, heater power, and flow.

- A small void fraction of argon gas ($\alpha = 0.00065$) accounted for one-fourth of the combined temperature increase ($160^\circ C$ or $288^\circ F$) due to both gas injection and the six-channel central blockage ($120^\circ C$ or $216^\circ F$).

- The maximum measured temperature increase due to gas injection was ($40^\circ C$ or $72^\circ F$), and occurred in the blockage wake at 16 mm (0.63 inch) downstream of the blockage.

- The highest average temperature increase during the entire period of gas injection (61 sec) was $35^\circ C$ ($63^\circ F$), and occurred at the same location as the maximum temperature increase.
b. **FRG Experiments**

Tests under even more severe conditions were conducted in the F.R. of Germany to study the effect of fission gas entrapped in the wake of local blockages. In the early stage of the test, a test section of one-half of a 169-pin subassembly of the SNR 300 Mk1A type was used for tests with water. The blocked flow area was 21% of the total cross-sectional flow area and was located in the corner of the test bundle. The gas was injected into the main flow stream 5 cm (1.97 inches) downstream of the blockage through the duct wall, both continuously and in pulses. Gas behavior was studied by means of high-speed photography. It was observed that under certain combinations of water and gas flow rates, gas would accumulate and be trapped downstream of the blockage.

A full-scale test section with a 21% corner blockage was later used for sodium tests. Gas was also injected directly into the blockage wake 5 cm (1.97 inches) downstream of the blockage to simulate the fission gas entrapment. The gas behavior was similar to that in the water tests, as shown in Figure III-8. A noticeable effect on cooling could be realized only if the sodium velocities and gas injection rates fell into the shaded gas accumulation region in Figure III-8. It can be seen that in the normal LMFBR sodium velocity zone (≈8-8 m/sec), a continuous fission gas release rate of >>5 cm$^3$/sec (0.3 in$^3$/sec) at 5 cm (1.97 inches) downstream of a blockage of 21%, which is a very unlikely situation, would be required to cause gas entrapment in the blockage wake and increase the coolant and clad temperatures.

However, in the tests with the gas injected as a pulse at the same injection location with an initial gas inventory of 12 cm$^3$ (0.73 in$^3$) and pressure differential of 24 bar (350 psi), the temperature increase due to the blockage and entrapped gas was found to be ≈100°C (180°F), and was reached within 7 seconds for a heater pin with a heat flux of 10 W/cm$^2$ (≈32000 Btu/hr.ft$^2$) and a sodium velocity of 3 m/sec (9.8 ft/sec). This gas pulse was chosen to simulate a fuel pin with high burnup. At a pin power corresponding to

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Figure III-8. Gas Accumulation in Blockage Wake from Continuous Gas Injection.
the full power condition of a fuel subassembly, the cladding melting temperature would have been attained about 3 seconds after the initiation of the gas pulse release.

The conclusions drawn from the results of this test with improbably large blockages are:

- Under certain combinations of coolant flow rate and fission gas release rate, the fission gas released may accumulate in the blockage wake and hinder cooling still further, and ultimately lead to local dryout. This amount of gas accumulation can be defined as the critical accumulation, and is a function of the size and location of the blockage, coolant flowrate, and the location and rate of gas release.

- The amount of fission gas released from a high burnup fuel pin is sufficient to cause the occurrence of this critical accumulation of fission gas.

- The size of the boiling zone under dryout condition is roughly proportional to the amount of the critical accumulation of fission gas. This means that if fuel meltdown occurs, the mass of fuel melted is roughly proportional to the critical accumulation of fission gas.

- Including fission gas in a "local cooling disturbance" problem will reduce the margin to boiling, and especially the safety margin between boiling inception and dryout.

\textbf{c. DFR Special Experiments}

During the final stages of normal power operation of the DFR, special experiments\textsuperscript{31} were performed with the objectives of exposing bundles of typical mixed-oxide fuel pins to coolant boiling for prolonged periods. One of these experiments was devoted to local
blockage studies with a mini fuel subassembly containing 18 fuel pins and an unheated instrument probe. This test subassembly was enclosed in a pair of concentric cluster walls sitting inside a massive steel carrier. The cluster walls confined an argon gas layer which insulated the coolant flow in the test subassembly from the comparatively cold bulk reactor flow. The test fuel pins had a nominal outside diameter of 5.8 mm (0.23 inch) and the pitch was similar to that in the Prototype Fast Reactor (PFR). The pins were spaced by honeycomb grids with axial spacing of ~100 mm. The fueled length was comparatively short, being between 286 mm (11.3 inches) and 375 mm (14.8 inches). The fuel burnups ranged from 0 to 10%, and the typical peak linear power rating was 32 kw/m (9.8 kw/ft). Because of the low inlet temperature, 230°C (446°F), and the short heated length, the axial temperature gradient under boiling conditions was between three and four times the gradient for a comparable incident in current LMFBR designs.

The local blockage was a gibbon-shaped thin steel plate mounted eccentrically, and covered approximately 70% of the available flow area. The plate was welded to the downstream edge of a honeycomb grid and sited approximately midway along the heated length. There was a narrow, internally shaped, annular gap between the blockage plate and each of the fuel pins which passed through the plate; the effect of the resulting small leakage flow upon the wake had been examined in the out-of-reactor water tests.

Thermal and acoustic sensors were used to measure temperature and to detect boiling. In addition, void detectors and electro-magnetic devices were used to detect small rapid changes in local coolant resistivity caused by temperature fluctuations or voids.

One of the local blockage experiments inadvertently entrained gas into the wake region causing fuel failure, including clad melting soon after the first rise to full power. Boiling in the test subassembly occurred after a further 10 days irradiation, and led to a further region of fuel failure downstream of the wake. Detailed post-irradiation examination showed that:
There was no evidence of any fuel melting despite the severity of the incident.

There were several examples of sintered fuel remaining intact despite removal of clad from the failed region.

There was no sign of any secondary blockage formation at the downstream grids despite the loss of complete sections of some fuel pins.

The mechanism of gas entrainment into the wake and subsequent fuel failure has been demonstrated using the heated water rig. The effects of gas in this experiment are expected to be a pessimistic analogue of a similar situation in a modern power reactor, because the DFR experiments were carried out in downward-flowing sodium where hydrodynamic effects oppose rather than assist the buoyant escape of the trapped gas.

The following conclusions can be drawn from these experiments performed in various countries:

- Fission gas entrainment in the wake of local blockage is possible and can enhance the blockage effect on fuel performance if it occurs.

- It is obvious that the increase in blockage-wake temperature due to fission gas entrainment depends on gas release rate, duration and point of release, kind of blockage (heat-generating or non-heat-generating), blockage size, geometry and location, power and coolant flow. However, no general correlation has been established.

- Fission gas entrainment in a blockage wake does not cause propagation beyond the sub-assembly due to the self-stabilizing characteristics of blockage induced damage.

- Gas entrainment phenomena may have to be considered in the design of fuel subassemblies with grid spacers, particularly at very low coolant velocity such as may occur
during reactor startup. This potential problem probably can be avoided by proper grid design. No problem exists for fuel subassemblies with wire-wrap spacers.

- As a local blockage is a complicated local fault problem and is itself highly improbable, studies of fission gas entrapment in blockage wakes should be treated as an extension of the research on local blockage if it is deemed necessary at all.

B. Two-Phase Flow Effects

One consequence of fission gas release may be a reduction in the coolant mass flow rate due to two-phase flow effects. The effect on subassembly flow depends on the rate of gas release from the failed pin. For extremely rapid gas release, a slug flow regime may be established temporarily and flow reversal be of concern. This section considers two-phase flow effects for discharge rates less than those required for flow reversal. Section III.C deals with flow reversal and its consequences.

When fission gas at low release rates enters the sodium flow, it will form a two-phase mixture which can cause an increase in coolant temperature due to the two-phase frictional effects and a reduction in effective coolant density. If these effects are significant, the temperature in neighboring pins may be affected. Evaluations have been performed to determine if additional pin failures could occur because of the elevation in the cladding temperatures of adjacent pins.

The degree of reduction in coolant mass flow rate and the resulting temperature rises depend on the radial dispersion of the gas and the cross-flow between subchannels. The largest coolant temperature rises occur when the radial dispersion and the cross-flow are at a minimum.

Whether or not the gas dispersion and cross-flow are taken into account, the worst condition (i.e. the maximum temperature rise) will occur at some intermediate position between the point of gas release and the subassembly outlet. Furthermore, without considering the gas
dispersion and cross-flow, the worst condition will occur if the point of gas release is located at the subassembly inlet or the bottom of a fuel pin because the two-phase flow extends the greatest possible length along the fuel pins. This is not necessarily true, however, if the gas dispersion and cross-flow are considered, because the temperature rise will depend on the breach location and the degree of gas dispersion and cross-flow.

Although some data on coolant cross-flow between subchannels are available such as in References 32-34, analytical treatment of two-phase flow effect due to fission gas release is hampered mainly by the lack of data on the radial dispersion rate of the gas. Experiments were performed in Argonne National Laboratory with water as coolant and argon or helium as gas, using 19-pin test sections similar to the LMFBRs of current design. It was observed that some gas bubbles progressed from the central gas-releasing pin to the periphery of an unheated 19-pin array over an axial distance of 20-30 cm (7.9-11.8 in) downstream. A heated test section with temperature sensors was also used in these tests. Gas was injected through a hole in the central pin for various hole sizes, pressure differences across the hole, and coolant flowrates. It was found that the rise of the local coolant temperatures was highest for large cladding breach sizes at relatively low pressure differences. This was due to the high rate of gas release with a low gas jet velocity at the breach, such that the initial radial gas dispersion rate was small and the gas was confined to a limited number of subchannels causing them to be more affected by the two-phase flow. Even for these worst conditions, the maximum coolant temperature increase, extrapolated to full LMFBR power, was less than 100°C (180°F) under steady-state conditions for gas release rates up to \( \sqrt{1000 \text{ cm}^3 \text{(STP)} / \text{sec}} \) (61 in\(^3\) (STP)/sec). This release rate corresponds to \( \sqrt{5.8 \text{ g/sec}} \) (.0128 lb/sec) for xenon, and is much greater than the values determined experimentally (See Section B in Appendix A). Hence, for the much lower release rates which are expected to follow pin failure, the temperature increase would be much less. In fact, for the very low release rates which have been observed for failures in the fuel zone (Section A in Appendix A), adjacent pin temperatures are probably not affected.
A detailed analysis of the two-phase flow effect on coolant temperature was performed by Freslon.\textsuperscript{36,37} Because it is difficult to assess both the resistance to the gas flow in fuel pins with any burnup and the resistance at the breach of various sizes without introducing several unknown parameters, no attempt was made to represent the actual path followed by the fission gas from the fuel plenum through the breach to the subchannel in a fuel subassembly. Rather, the minimum resistances yielding the maximum or reference gas release rate were used. The minimum resistance for the direct axial flow from the plenum to the breach was represented by the annular gap between the upper blanket or the fuel and the cladding in the cold non-irradiated fuel pin. A zero resistance was assumed to be caused by a breach having an area numerically equal to the cross-sectional area of the above gap. Various fractions of this reference rate of gas release, corresponding to a range of the resistances to the fission gas flow, were then used to determine the worst case (the maximum coolant temperature increase).

The temperature increases were predicted by solving simultaneously the two-phase flow equations with the standard transient heat transfer equations, taking into account the heat capacities of the fuel, cladding, and flowing liquid sodium, as well as the amount of heat removal by radial conduction from affected to unaffected subchannels. The analysis was performed for a fuel subassembly of the FFTF with inlet and outlet temperatures of 316\textdegree C (600\textdegree F) and 482\textdegree C (900\textdegree F) respectively, and with the following important assumptions:

1. The flow regime is annular for the void fraction, $\alpha$, greater than 0.5, and bubbly for $\alpha$ below 0.5.

2. The fission gas remains in the coolant subchannel into which it is released, which results in higher coolant temperature due to the omission of gas dispersion and flow mixing.

3. The cladding breach on the initial failed pin is at the bottom of the fuel section, which results in the longest path of the gas through the core and thus highest coolant temperature rise.
4. The initial gas plenum pressure is 69 bar abs (1000 psia).

5. The axial power distribution is uniform.

This analysis predicts that the two-phase flow of liquid sodium and fission gas switches from annular to bubbly when the fraction of the reference gas release rate, \( y \), being decreased from 1.0 to around 0.5, and for \( y \) less than 0.23, the flow pattern is bubbly from the gas release point. The results also show that, even for the above-mentioned conservative assumptions, the maximum coolant temperature increase is \(<140^\circ C (252^\circ F)\). Comparing this calculated maximum temperature increase with that extrapolated from the previously mentioned experiments of \(100^\circ C (180^\circ F)\), the conservatism in the analysis, due primarily to the omission of the gas dispersion and crossflow, is evident. However, as this analysis was performed for an average subchannel with uniform power distribution, extrapolation is required for application to the hot subchannels in CRBR and FFTF. The extrapolation is considered proper because the temperatures of interest are in the same range as that used in this analysis, and because the properties of the coolant, cladding, and fuel are approximately linearly proportional in this and the slightly extended temperature ranges. The results of this application to the hot subchannels in CRBR and FFTF show that a margin to the cladding failure threshold temperature of \(871^\circ C (1600^\circ F)\) still exists, even with all the conservative assumptions.

Freslon also considered a case with the fission gas being released to two adjacent subchannels instead of one single subchannel, and a case with mixing or distribution of the fission gas in six surrounding subchannels. The result showed that the maximum coolant temperature increase dropped from \(140^\circ C (252^\circ F)\) to \(129^\circ C (232^\circ F)\) in the first case, and to \(67^\circ C (121^\circ F)\) in the second. Therefore the safety margins predicted for CRBR and FFTF hot subchannels would be wider if cross-flow mixing were considered. Other analyses\(^{38,39}\) also show the existence of a wide margin to fuel failure caused by flow reduction associated with the slow release of fission gas.
Based on the experimental evidence and analytical predictions, coolant temperature increases due to the mixed flow of liquid sodium and fission gas are insufficient to cause fuel failure. Hence, it can be concluded that the two-phase flow effect caused by slow fission gas release does not constitute a mechanism for pin-to-pin failure propagation.

C. Coolant Flow Reversal

Coolant flow reversal is a hypothetical consequence of fission gas release. Its occurrence requires that the gas discharged from failed fuel pins be much greater than that which has been actually reported. The concern regarding flow reversal is due to the possibility of excessive coolant temperatures arising when it resides within the fueled region of the core longer than when there are no fuel failures.

The introduction of fission gas into the coolant flow stream increases the pressure drop within the two-phase section of the flow path. The magnitude of this pressure drop depends on the flow regime, which in turn depends on the release rate of gas. It is not possible for flow reversal to occur in the bubbly-flow regime. The most likely flow regime for flow reversal is one in which the phases are distinctly separated along the flow path, i.e., slug flow. When the pressure within the gas bubble rises sufficiently, its expanding boundary is constrained by the subassembly duct while it is still near the release point. If the pressure in the bubble is high enough, the expansion acts on two well-defined gas-liquid interfaces. Figure III-9 shows a schematic representation of the alteration in local pressure caused by introduction of a high pressure fission gas bubble into the coolant flow channel. The figure shows three plots; Plot A is the axial pressure profile through a normally operating high-flow fuel subassembly, and Plot B is the pressure profile as it is likely to appear very shortly after a postulated instantaneous fission gas release. In Plot B, the bubble has been shown to extend from point J to point K with constant pressure within its boundary. The shape of Plot B would change very rapidly as the flow dynamically responds to the introduction of and changes within the gas bubble. As the bubble
Figure III-9. Schematic Representation of Pressure Profiles Through Fuel Assembly
grows and the slug flow develops, the pressure profile would change to Plot C. In general, flow reversal can occur only if the pressure at J is higher than that in the inlet plenum (point E in Figure III-9).

In addition to the requirement that the bubble pressure be greater than the inlet plenum pressure, reversal of flow requires that the higher bubble pressure acts for sufficient time to overcome the inertia of the lower column of liquid. This normally rising column must be decelerated until it stops. At this point, continued higher bubble pressure could force coolant in the reverse (downward) direction. To achieve significant reverse flow velocity, the bubble pressure must overcome the frictional pressure losses through the pin bundle, shield block and orifice.

A mathematical model has been set up to express the interactive dynamics of the gas flow out of the plena and the coolant flow through the subassemblies. The model in References 40 and 41 has been substantially verified by experiments. A description of this model is given in Appendix B together with the assumptions and approximations invoked. The model does not represent the most pessimistic scenario conceivable. However, it is clear that the most pessimistic scenario is too unrealistic for consideration because a number of unlikely events have to occur in a precisely timed sequence. Such hypothetical scenarios are also discussed in Appendix B.

Reference 40 describes the experiments and gives the data which form the basis of the proposed flow reversal model. A 19-pin bundle with wire-wrap spacers and water as the simulating coolant were used. High pressure gas was injected into the coolant stream from the central pin, and the evolution of the gas bubble was photographed.

The observations from the experiments supported the slug flow model for massive fission gas release. These observations also indicate that the gas bubble began axial expansion only after it reached full radial expansion. For the gas discharge used, no flow reversal occurred. The time-dependent locations of the upper and lower gas-liquid interfaces were predicted in Reference 40 using the mathematical model and found to be in good agreement with the observations. All the data shown have
both gas-liquid interfaces above the gas release point. This evidence argues in favor of the assumption in the model that the gas flows directly into the bubble irrespective of the position of the interfaces. The conservatism of the assumption of sodium incompressibility was demonstrated since predicted pressures were significantly higher than those measured. Thus, the experimental evidence corroborates the basic soundness of the model developed in Reference 40.

A literature search showed that there are experimental data on fission gas release from a fuel pin plenum. Reference 44 reports experiments performed to study sudden gas release due to plenum rupture, as well as gas release as a jet from a small hole in the cladding. In the plenum release case the gas pressure was assumed to arise from accumulation in the plenum of all the fission gas generated within the fuel. Upper and lower plena were considered. The gas release at various rates was from one pin in a bundle with 64 pins in a square array. Water simulated the sodium coolant. For the analytical part of the study, the gas flow in the bundle was modeled as two-phase bubbly flow. Measured cladding temperatures were compared with those predicted by a simplified thermal-hydraulic model for a fuel pin.

The observations in Reference 44 also support the assumption of radial expansion of the bubble to the full extent of the channel cross section before axial expansion. Although bubbly flow was said to have been observed, inlet flowmeter readings indicated flow reversal in cases where high pressure gas was injected into the channel with low coolant velocity. The two-phase section of flow was reported not to have entered the heated section of the pin bundle. Reference 45 indicates that flow reversal was not considered to be of much significance.

Safety analyses for a normally operating reactor were performed with an adequate conservatism by considering the implications of instantaneous and simultaneous fission gas release from a large number of fuel pins. Appendix B presents the analyses for simultaneous rupture of 217 pins at two different axial locations. The dynamics of the upper and lower columns of sodium coolant are studied separately.
The predicted coolant flow is used as input into a thermal model to
determine the changes in temperature due to the flow perturbation
caused by the fission gas release. It was found that flow reversal
caued by rupture of all pins in a fuel subassembly results in coolant
temperature increase about an order of magnitude lower than the margin
to saturation. Hence, coolant flow reversal, although an extremely
unlikely occurrence in itself, will not affect assembly coolability.

D. Mechanical Loadings

It has been postulated that a rapid release of fission gas from a
failed fuel pin may generate a pressure pulse, which in turn acts as a
mechanical loading on the adjacent fuel pins and causes additional pin
failure. This mode of pin-to-pin failure propagation is of most
concern at the end of life when the fission gas pressure inside the
fuel pin is high and clad ductility is low. Depending on the location
of the breach on the initially failed fuel pin, the rate of the gas
release and its consequences are different. As discussed in Section
II. B, cladding failures in the active core and blanket regions result
in low fission gas release rates due to the high flow resistance
between the gas plenum and the breach. Pin depressurization upon
rupture in these regions is usually slow and thus no dynamic loadings
occur. Even in conservative puncture tests (Section B.1 in Appendix A)
and out-of-pile tests simulating in-core failures, the measured
pressure pulses are small. These results demonstrate that for such
breach conditions, there will be no detrimental effect on the adjacent
fuel pins due to mechanical loading even when the ductility of their
cladding is low.

If failure occurs in the gas plenum section of the fuel pin, the
gas release rate, the pressure pulse, and the impact on the adjacent
fuel pins may be much higher. Hence, there is then a greater potential
for additional fuel pin failures. This is considered to be the
limiting case, although the cladding ductility is high relative to that
in the active fuel region. There is no evidence (from post-irradiation
examination or instrument response during operation) that a breach
occurred which produced a large dynamic load. Furthermore, there is no
evidence of pin-to-pin failure propagation attributable to this
phenomenon.
There are three possible mechanisms by which the pressure pulse from fission gas release may mechanically cause damage to neighboring fuel pins, namely:

1. Deformation due to bending moment.
2. Deformation due to contact forces between fuel pins and wire wraps.
3. Damage due to resonance vibration of fuel pins.

The assessment of each of these mechanisms, their consequences, and the potential for fuel failure propagation will be presented in the following sections.

1. **Bending Moment Deformation**

Localized deformation due to bending moment has been analyzed by Hibbeler and Crawford.\(^45\) It was assumed that the fuel pin plenum section was a thin-walled hollow cylinder with simple supports, and that when it was subjected to lateral blast load, it buckles in the collapse-hinge manner.

These assumptions are conservative as there are always other wire wraps on other sides of the fuel pin, and the resistances to the loading offered by the supports at 60°-240° and 120°-300° are neglected. Bending moments and deflections caused by a certain loading on cylinders with more supports on the ends are much smaller than those on cylinders with simple supports.

The bending strength of the thin hollow cylinder decreases as the centerline deformation of the cylinder increases. In effect, the cylinder will fail if a critical bending moment is reached. It has been found\(^46\) that for a hollow cylinder having a length-to-radius ratio greater than four, the buckled shape of the cylinder caused by a side-on pressure pulse is identical to that formed by a static beam bending moment. The static critical moment for hollow cylinders developed by Wood,\(^47\) which modifies the conventional beam moment-curvature relationship to accommodate the cylinder ovalization
effect, is therefore applicable to this case. This critical moment is independent of the characteristics of the blast, and depends only on geometry and material properties.

Hibbeler and Crawford performed a dynamic analysis using an energy approach, and reached an expression for the maximum moment responding to a pressure pulse defined by the generally assumed and accepted equation:

\[ P(\theta, t) = P_0 \cos \theta e^{-\alpha t} \quad \text{for} \quad -\frac{\pi}{2} \leq \theta \leq \frac{\pi}{2} \]

\[ P(\theta, t) = 0 \quad \text{for} \quad \frac{\pi}{2} \leq \theta \leq \frac{3\pi}{2} \]

where \( P(\theta, t) \) is the side-on pressure as function of angle \( \theta \) and time \( t \); \( P_0 \) is the initial or maximum pressure at the breach; and \( \alpha \) is the decay constant. The maximum moment is developed at the mid-point on the cylinder between the two supports. Failure of the cylinder is said to occur if the load is sufficient to cause the maximum moment to equal the critical moment. The analytical expression for the maximum moment has been verified by comparison with experimental data. They further calculated the following relationship for the CRBR-type pin design at the critical moment at 816°F (1500°F):

\[ P_0 = \alpha/10 \]

for \( \alpha \geq 4500 \text{ sec}^{-1} \). This expression conservatively neglects the internal pressure in the neighboring pin.

Existing EBR-II out-of-pile experimental data showed that values of the time decay constant \( \alpha \) is always \( >4500 \text{ sec}^{-1} \) for plenum pressure greater than 69 bars (1000 psi). Therefore the critical moment can be reached only if the maximum side-on blast load is greater than 31 bars (450 psi) which is highly improbable for fuel pins of current design that have a maximum plenum pressure less than 69 bars (1000 psi). References 49 and 50 showed that the pressure pulses are on the order of 1/5 or less of the plenum pressure.
The above $P_o-\alpha$ relationship and the fission gas release model developed by Kazimi$^{51}$ (and verified with EBR-II out-of-core test data,$^{49,50}$) were used to predict conservatively the conditions required to attain the critical bending moment for fuel pins of the CRBR-design operating at 649$^0$C (1200$^0$F) and an initial plenum pressure of 103.4 bars (1500 psi). It was found that the breach area on the initially failed fuel pin has to be equivalent to the entire longitudinal pin-half for a length of 152.4 mm (6") (the approximate length between two consecutive wrapper wire segments along the pin), and to open in a time period of $<$0.07 msec in order to attain the critical value. This is obviously a highly improbable occurrence.

It has also been pointed out in Kazimi's study$^{51}$ that:

a. The pressure pulse is directly proportional to the internal pin pressure upon rupture, and the initial internal pressure is the major factor in determining the magnitude of the pressure pulse. The pressure pulse is less sensitive to the rupture area and compressed gas volume.

b. The ratio of the peak pressure pulse to the initial internal pressure upon rupture decreases as the internal pin pressure increases.

c. This ratio increases slightly as the rupture area decreases, but the energy associated with the pulse decreases.

d. This ratio increases slightly as the compressed gas volume is increased.

As the above prediction used a high initial plenum pressure (103.4 bars or 1500 psi) and the actual plenum volume, the prediction is conservative for lower initial plenum pressure according to observation item a above. In this prediction, the resulting breach
area is the maximum which could possibly happen for the pin geometry. For cases with a smaller breach area, the energy associated with the pressure pulse will be smaller, according to observation item c.

Although there are analytical and experimental results showing that damage occurred on two concentric rings of fuel pins around the source of the pressure pulse, these results were obtained using explosives simulating fuel-coolant interaction with pressure pulses of 69 bars (1000 psi) or higher, together with other conservative assumptions. As the pressure pulse generated by fission gas release is expected to be much milder than explosives, damage on adjacent fuel pins is very unlikely. Therefore, it can be concluded that fuel pin failure is not expected due to bending moment caused by the fission gas release from an adjacent fuel pin.

2. Contact Forces

The pressure pulse generated by the sudden rupture of a fuel pin may create contact forces between the adjacent fuel pins and the wire wraps behind them. The stress interactions occurring at these contact points may cause clad deformation and possible failure. Using elastic and dynamic-plastic analyses, Hibbeler investigated the stress and deformation effects developed at the wire-wrap point of contact. Again in this analysis, the plenum section of the fuel pin was of primary concern as the inertia characteristics of the plenum were low and the gas release rate from the adjacent fuel pin plenum was high. It was also assumed that the pressure pulse is exerted directly over the point of contact where maximum mechanical interaction stresses would occur. As the resistance to this pressure loading was largely dependent upon both hoop tangential and bending stresses, a thin ring represents a conservative model of the fuel pin plenum section. Choosing an upper bound to the loading, which is represented by a rupture length of 12.7 mm (1/2 in.), the analysis indicated that the localized strain energy developed at the wire wrap was about one-third of that required to produce failure of the material at the end-of-life conditions. This conclusion was based upon the conservative assumptions, and indicated conclusively that failure propagation will not occur.
3. Resonance Vibration

The pressure pulse generated by a ruptured fuel pin may cause deflection, resonance vibration, and eventually failure of the adjacent fuel pins. However, as reported by van Erp et al., for a fuel pin to be damaged by this cause, a pressure differential must be maintained across the fuel pin diameter for a time period \( t_u \) of the same order of magnitude as the time period \( t_r \) of the fundamental mode of the transversal resonance vibration of the fuel pin. Using the analytical solutions for pressure distributions on a rigid cylindrical surface due to a step pulse, Sette and Friedlander showed that the pressures on the front and back sides of the cylinder will have equalized after a time \( t_u \), counted from the moment pressure pulse reaches the front side, where \( t_u = 2d/c \). Using the sonic velocity in liquid sodium at 540°F (100°F), 2.295 \times 10^5 \text{ cm/sec} (7530 \text{ ft/sec}), and the current LMFBR design of fuel pin diameter, 0.584 cm (0.23 in); \( t_u = 5.09 \mu \text{sec} \).

Reference 58 gives the approximate fundamental resonance vibration frequency of a pin:

\[
\nu_r^2 = \left( \beta_0^4 \frac{EI}{4\pi^2 M \lambda^4} \right)
\]

where \( \beta_0 = \pi \) for simply-supported end conditions, or 4.73 for fixed-end conditions. Assuming \( E = 1.62 \times 10^{12} \text{ dyne/cm}^2 \; (2.35 \times 10^7 \text{ psi}) \) and \( \lambda = 30.48 \text{ cm (12 in)} \) for simply supported end conditions, \( \nu_r = 26.3 \text{ sec}^{-1} \) or \( t_r = 38 \text{ msec} \). From the above approximate evaluation, it is seen that \( t_r \) is much greater than \( t_u \). Although \( t_u \) would be higher if the smaller sonic velocity in the gas-sodium mixture were considered, \( t_r \) would still be much higher than \( t_u \) even if consideration were given to the effect of the mass of the fluid associated with the pin and \( \lambda = 15.24 \text{ cm (the distance between two consecutive segments of wire wrap along a fuel pin)} \). Therefore, it can be concluded that \( t_r \) is indeed much greater than \( t_u \), and that the pressure pulse cannot deflect adjacent pins sufficiently to cause vibration and failure propagation.
area is the maximum which could possibly happen for the pin geometry. For cases with a smaller breach area, the energy associated with the pressure pulse will be smaller, according to observation item c.

Although there are analytical and experimental results\textsuperscript{48,52} showing that damage occurred on two concentric rings of fuel pins around the source of the pressure pulse, these results were obtained using explosives simulating fuel-coolant interaction with pressure pulses of 69 bars (1000 psi) or higher, together with other conservative assumptions. As the pressure pulse generated by fission gas release is expected to be much milder than explosives, damage on adjacent fuel pins is very unlikely. Therefore, it can be concluded that fuel pin failure is not expected due to bending moment caused by the fission gas release from an adjacent fuel pin.

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Reference 58 gives the approximate fundamental resonance vibration frequency of a pin:

$$\nu_r = (\beta_0^4 EI)/(4\pi^2 ML^4)$$

where $\beta_0 = \pi$ for simply-supported end conditions, or 4.73 for fixed-end conditions. Assuming $E = 1.62 \times 10^{12} \text{ dyne/cm}^2 (2.35 \times 10^7 \text{ psi})$ and $L = 30.48 \text{ cm} (12 \text{ in})$ for simply supported end conditions, $\nu_r = 26.3 \text{ sec}^{-1}$ or $t_r = 38 \text{ msec}$.

From the above approximate evaluation, it is seen that $t_r$ is much greater than $t_u$. Although $t_u$ would be higher if the smaller sonic velocity in the gas-sodium mixture were considered, $t_r$ would still be much higher than $t_u$ even if consideration were given to the effect of the mass of the fluid associated with the pin and $L = 15.24 \text{ cm}$ (the distance between two consecutive segments of wire wrap along a fuel pin). Therefore, it can be concluded that $t_r$ is indeed much greater than $t_u$, and that the pressure pulse cannot deflect adjacent pins sufficiently to cause vibration and failure propagation.
4. Conclusion

Pin-to-pin failure propagation due to mechanical loading caused by fission gas release is highly improbable because it has been shown that sufficient margin exists to prevent fuel pin failure by any of the three possible transmission mechanisms.
IV. CONCLUSION

The safety technology for evaluating the effects of fission gas release from fuel pins on fuel performance and reactor safety is well developed with the support of a vast amount of experimental data and many years of operational experience. Great attention has been paid to this subject, and extensive efforts made to study its consequences by all the nations devoted to LMFBR development. Based upon all the analytical and experimental results, it can be concluded that rapid and extensive pin-to-pin failure propagation in currently employed LMFBR fuel subassemblies, due to fission gas release following random failure of fuel pins, is highly unlikely at operating conditions with power-to-flow ratios typical of plant design operating conditions. However, failures caused by other initiating events such as reactor transients are not within the scope of this document.

As fuel pin failure and pin-to-pin failure propagation are statistical phenomena, this conclusion does not mean that a random failure of a fuel pin will never be followed by occasional failures of one or a few neighboring fuel pins. However, even with very conservative assumptions regarding release rate of fission gas, location of cladding breach, and other parameters, sufficient margin exists within the range of nominal operating conditions to assure that the probability that additional pin failures will occur is very small.

While fission gas release by itself is not a potential cause for pin-to-pin failure propagation as concluded above, it can enhance the deleterious effects of a blockage on fuel performance if fission gas is entrapped in the wake, even to the extent of causing dryout and clad melting in the wake region of an otherwise non-boiling blockage. Various design features of different reactor systems may strengthen different safety aspects for this rare event. This special circumstance is not a cause for concern because of the evidence for the self-stabilizing characteristics of blockage induced damage and their infrequent occurrence.

Extending the above conclusion, it is evident that fission gas release does not constitute an event of concern to the health and safety of the general public. With acceptable limits on operation with failed fuel, as is already the case for light water reactors, the fuel subassemblies
in LMFBRs can be safely and economically operated until either their designed goal burnups are reached, or they are replaced at the next scheduled end-of-cycle shutdown or at an earlier date determined by programs investigating operation with failed fuel.

Furthermore, it can be concluded that further research on fission gas release for mixed-oxide fuel as used in current reactor designs is considered unnecessary. It is also recommended that more detailed and exact effects of fission gas release in conjunction with local blockage be investigated as an extension of the research on local blockages if it is considered to be necessary at all.
V. REFERENCES


26. Information from UKAEA member of review panel (C.B. Cowking)


APPENDIX A

FISSION GAS RELEASE EXPERIENCE
A. In-Pile Experience

1. EBR-II Subassembly X106

Seven pins containing UO₂ - 20 w/o PuO₂ were irradiated in Subassembly X106 in EBR-II until a defect developed in one pin at a burnup of 7 a/o. The pins comprising the test bundle were 7.37 mm (0.290 in.) O.D. with 0.51 mm (0.020 in.) annealed 304L SS cladding. The ratio of the plenum volume to the fuel volume in the pins was about half that normally used in design. Hence, there was a more rapid buildup of fission gas pressure with burnup (√80 bars or √1160 psi at failure). At the time of failure, the peak linear power was calculated to be 46.9 W/mm (14.3 kW/ft) and the maximum cladding temperature 512°C (953°F). The loss of cladding integrity was detected by a gradual increase in the EBR-II cover gas activity over a period of several weeks. The release rate was so low that Subassembly X106 was not identified as the "failed assembly" until over a month after the initial detection of fission gas release. During the period of reactor operation following failure, the concentration of fission products in the cover gas never exceeded 1% of the level expected for a complete release of the plenum inventory. Post-irradiation testing and analysis determined that the fission gas flow rate through the breach was less than 1 ml/hr.

Identification of the failed pin and the location of the breach were not easily accomplished. The failed pin was determined by performing gamma scans for 133Xe in the plenum region, and measuring the changes in weight, and balance point of each of the seven pins in Subassembly X106. The defect could not be located by standard visual examination employing magnification up to 25x. The location of the breach on the failed pin could only be discovered by internally pressurizing the element with helium while submerged in an alcohol bath. Metallographic examination showed the defect to be an intergranular crack with a grain boundary separation of √5 μm. The failure was located about 12.7 mm (0.5 inch) above the top of the fuel column, adjacent to an insulator pellet. Examination of the other unfailed pins in Subassembly X106 showed no evidence of being affected by the failed pin. Despite the fission gas pressure at the breach
location, the thermal and mechanical stresses were much lower than those near the top of the fueled region. Subsequent electron microprobe examination and a review of the previous fabrication history of the pin indicated that it may have been contaminated by chlorine at that location producing a local weak spot in the steel cladding. A-1

It is important to note that even though the breach was close to the gas plenum, the release rate was orders of magnitude less than the values required to provide the effects discussed in Section III.

2. HEDL PNL-5A (X114)*

The PNL-5A subassembly contained 19 mixed-oxide, solution-annealed, 304 stainless steel clad pins. The design parameters and operating conditions are:

- Pin Diameter: 6.35 mm (0.250 in.)
- Cladding Material: 304 SS, solution-annealed
- Cladding Thickness: 0.41 mm (0.016 in.)
- Fuel Composition: 75% UO₂/25% PuO₂
- Maximum Cladding Temperature: 494°C (920°F)
- Maximum Pin Power: 39.4 W/mm (12 Kw/ft)

A cladding breach occurred at 128,000 MWD/MT (13.1 a/o). The cover gas activity increase was so small, that reactor shutdown (due to cover gas activity limits) did not occur until three days later. A-3 Three subsequent reactor startups were required to identify the failure source.

Post-irradiation examination revealed that several pins were twisted, thus distorting the bundle geometry. A-4 There was evidence of pin-to-pin contact which may have existed for 376 efpd (7.9 a/o) before the failure occurred. This produced local hot spots which, through accelerated creep, led to the pin failure. Visual techniques failed to locate the breach. The technique used for the Subassembly

* The designation PNL-5A is the Hanford Engineering Development Laboratory (HEDL) identification for the subject assembly. The label X114 is used to identify the element during its irradiation in EBR-II.
X106 failure location established that the failure was 8.9 cm (3.5 in.) below the top of the 35.6 cm (14 inch) fuel column. Microstructural examination of the cladding showed significant intergranular cracking and porosity (as a result of creep). The fission gas release occurred when this porosity became interconnected. A continuous crack through the cladding wall was not observed. Examination of adjacent, unfailed fuel pins showed no effects from the failure, even though they were distorted, and may have been more susceptible to any effects of fission gas release because of local hot spots, and their close proximity to the breached pin.

3. **HEDL PNL-10 (X193)**

This failure occurred as a result of excessive wear from a loose bundle construction.\(^{A-3}\) The design parameters and operating conditions are:

- **Pin Diameter**: 5.84 mm (0.230 in.)
- **Cladding Material**: 316 SS, 20% cold worked
- **Cladding Thickness**: 0.38 mm (0.015 in.)
- **Fuel Composition**: 75% UO\(_2\)/25% PuO\(_2\)
- **Maximum Cladding Temperature**: 566°C (1050°F)
- **Maximum Pin Power**: 32.8 W/mm (10 Kw/ft)
- **Burnup at Failure**: 63,800 MWD/MTM

At the location of the breach, 9.65 mm (0.38 in.) below the top of the fuel column, the cladding had been thinned locally from 0.38 mm (0.015 in.) to 0.13 mm (0.005 in.). As with most other RTCB tests, the breach could only be located by internal pin pressurization. The pin-hole defect was only 0.20 mm (0.008 in.) wide. Response of the cover gas monitoring system indicated that the depressurization was gradual, and an examination of neighboring pins showed that the failure had no effect on the neighboring pins.

4. **EBR-II Subassembly X084**

This failure was one of the first oxide pins run to failure in the EBR-II test program. The breach occurred in a 16.5 a/o burnup fuel
element (in Subassembly X084) with a plenum pressure of \( \sqrt{117} \) bars (\( \sqrt{1700} \) psi). A-5 Only \( \sqrt{140} \) cm\(^3\) (8.5 in\(^3\)) of a total fission gas inventory of \( \sqrt{270} \) cm\(^3\) (16.5 in\(^3\)) was released to the coolant. There was no evidence of propagation.

5. B3B Capsule Experiment in GETR

In the B3B experiment, a mixed-oxide, helium bonded fuel pin was irradiated in the General Electric Test Reactor (GETR) in a natural circulation capsule. The important design parameters are:

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pin Diameter</td>
<td>6.35 mm (0.250 in.)</td>
</tr>
<tr>
<td>Cladding Material</td>
<td>347 SS</td>
</tr>
<tr>
<td>Cladding Thickness</td>
<td>0.38 mm (0.015 in.)</td>
</tr>
<tr>
<td>Diametral Gap</td>
<td>0.07 mm (0.0028 in.)</td>
</tr>
<tr>
<td>Fuel Composition</td>
<td>75% U(_2)O(_2)/25% PuO(_2)</td>
</tr>
<tr>
<td>Upper Blanket Length</td>
<td>228.1 mm (8.98 in.)</td>
</tr>
<tr>
<td>Fuel Column Length</td>
<td>581.7 mm (22.9 in.)</td>
</tr>
</tbody>
</table>

The irradiation conditions of 72 W/mm (22 kW/ft) and 732\(^\circ\)C (1350\(^\circ\)F) maximum clad temperature were well beyond LMFBR design limits. For this reason, the pin failed relatively early at \( \sqrt{18,000} \) MWD/MT. The B3B pin was equipped with a pressure sensor in the fission gas plenum to measure the variation in this parameter before and after clad breach.

The volume of gas in the pin at the time of failure was calculated to be \( \sqrt{33} \) cc (\( \sqrt{2} \) in\(^3\))(STP) and the pressure was 5.2 bars (76 psia). Four hours after breach were required for the pin pressure to decay to ambient conditions. The volume of gas released was \( \sqrt{14} \) cc (\( \sqrt{0.9} \) in\(^3\))(STP) with most of this release occurring during the first 45 minutes after failure. The maximum gas release rate was calculated to be \( \sqrt{0.2} \) cc/min. Closure of the as-fabricated 0.07 mm (0.0028 in) fuel/cladding gap severely restricted the flow of gas from the plenum to the breach location.
6. **SCARABEE Tests**

An international evaluation of the in-pile SCARABEE tests was conducted by a working group of experts from CEA, KfK, and UKAEA. In one experiment, the central pin in the 7-pin bundle failed due to an increase in reactor power. The results suggested that this pin, at a pressure of 30 bars (435 psi) cold, lost its gas rapidly without causing failure of other pins. Experiments with reduced internal pressure of pin did not show a noticeable importance of this parameter as failure propagation is concerned. However, these results were obtained with non-irradiated clad material.

B. **Out-of-Pile Experiments**

1. **F3A Series Experiments**

The F3A series of irradiations in EBR-II Subassembly X40 were the first unencapsulated, mixed-oxide fuel pins irradiated to a significant burnup in the U.S. LMFBR program. Sixteen pins were irradiated to an average burnup of \( \sim 46,600 \) MWD/MT with no failures. From this group, six rods (designated F3B-1 through -5 and -7) were selected to measure the rate of fission gas release. The important design parameters and operating conditions for these rods are given in Table A-1. These pins were punctured at various axial locations along the pellet column and in the gas plenum region to determine the effect of breach location on gas release rate. The puncture location for each pin is also presented in Table A-1.

Figure A-1 shows the volume of gas (at STP) as a function of time following puncture. The half-lives for gas release (i.e., the time for one-half of the gas to escape) are noted in Table A-2. The gas release from pins F3B-5 and -7, punctured at the upper insulator and in the gas plenum, respectively, was quite rapid. Within one minute all of the gas (at pressure) had been released from the plenum failure (F3B-7) and \( \sim 95\% \) from the F3B-5 pin. It was observed that the gas release rate decreased as the distance between the breach and the plenum increased. In the fueled region the gas release rates were much slower. The F3B-3 data gave a maximum release rate of 1.9 cc/sec (0.12 in\(^3\)/sec). These
<table>
<thead>
<tr>
<th>Description</th>
<th>F3B-1</th>
<th>F3B-2</th>
<th>F3B-3</th>
<th>F3B-4</th>
<th>F3B-5</th>
<th>F3B-7</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pin Diameter (in.)</td>
<td>0.290</td>
<td>0.290</td>
<td>0.290</td>
<td>0.290</td>
<td>0.290</td>
<td>0.290</td>
</tr>
<tr>
<td>Cladding Material (SS)</td>
<td>316</td>
<td>316</td>
<td>316</td>
<td>316</td>
<td>316</td>
<td>304L</td>
</tr>
<tr>
<td>Cladding Thickness (in.)</td>
<td>0.020</td>
<td>0.020</td>
<td>0.020</td>
<td>0.020</td>
<td>0.020</td>
<td>0.020</td>
</tr>
<tr>
<td>Diometral Gap (mils)(^1)</td>
<td>4.0-5.7</td>
<td>3.9-5.0</td>
<td>3.7-4.6</td>
<td>3.8-6.4</td>
<td>4.0-5.5</td>
<td>4.8-8.3</td>
</tr>
<tr>
<td>Fuel Composition</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fuel Pellet Type</td>
<td>Annular</td>
<td>Solid</td>
<td>Annular</td>
<td>Solid</td>
<td>Solid</td>
<td>Annular</td>
</tr>
<tr>
<td>Fuel Column Length (in.)</td>
<td>13.5</td>
<td>13.5</td>
<td>13.5</td>
<td>13.5</td>
<td>13.5</td>
<td>13.5</td>
</tr>
<tr>
<td>Upper Insulator Length (in.)</td>
<td>2.5</td>
<td>2.5</td>
<td>2.5</td>
<td>2.5</td>
<td>2.5</td>
<td>2.5</td>
</tr>
<tr>
<td>Maximum Clad Temperature ((^{\circ})F)(^2)</td>
<td>994</td>
<td>1099</td>
<td>1130</td>
<td>1084</td>
<td>1011</td>
<td>981</td>
</tr>
<tr>
<td>Maximum Power (kw/ft)(^2)</td>
<td>15.9</td>
<td>16.3</td>
<td>17.0</td>
<td>16.3</td>
<td>15.0</td>
<td>16.5</td>
</tr>
<tr>
<td>Burnup (MWd/MT)</td>
<td>48,400</td>
<td>50,000</td>
<td>52,000</td>
<td>49,700</td>
<td>48,800</td>
<td>50,100</td>
</tr>
<tr>
<td>Puncture Location (in. below gas plenum)</td>
<td>16.5</td>
<td>16.5</td>
<td>5.1</td>
<td>5.1</td>
<td>1.5</td>
<td>0</td>
</tr>
</tbody>
</table>

**NOTES:**

1) As-fabricated range.

2) Beginning-of-Life.
results confirm what has been inferred from failures during normal operation; despite the fact that the puncture tests were performed at room temperature, and the defect size was a large 3.18 mm (0.125 in) hole - both factors which would tend to increase the gas flow rate. Hence, the fission gas release rates resulting from the F3A series puncture tests are considered to be a conservative basis for use in fission gas release analyses.

2. GE Out-of-Pile Simulation Tests

In addition, some out-of-pile tests with simulated fuel pin geometry have been performed to acquire data for model development. Experiments performed by General Electric simulated the flow of gas through the fuel/clad gap using alumina pellets in steel tubing. The geometry was representative of an idealized zero-burnup condition. The purpose was to determine the gas release rate as a function of gap thickness and length between the breach and the fission gas plenum. Pressure differentials from 1.72 to 95.15 bars (25 to 1380 psi) were studied. The flow rate data are summarized in Table A-3.

Calculations were performed to compare the flow rates with those observed in the F3A series tests (pins 3 and 4 in Table A-1). It was determined that the release rates in the out-of-pile simulation tests were approximately a factor of 3 to 10 greater than those observed in the puncture tests, which as noted previously were conservative values for estimating release rates for failures under prototypic LMFBR operating conditions. Furthermore, it should be recognized that the conditions in the test bear little resemblance to those in a fuel pin. During the initial rise to power, prior to any significant fission gas generation, pellet cracking and thermal expansion will affect the gap geometry. As burnup increases, the gap eventually closes. When plenum pressures become large, the fuel/clad gap is essentially closed, thus providing an increased gas flow resistance which is orders of magnitude larger than these tests. Hence, the higher gas pressures tested were inconsistent with the zero-burnup geometry simulation.
### TABLE A-2
GAS RELEASE HALF-LIVES FOR F3A SERIES PIN PUNCTURE TEST

<table>
<thead>
<tr>
<th>Pin Number</th>
<th>Location of Puncture</th>
<th>Gas Release Half-Life</th>
</tr>
</thead>
<tbody>
<tr>
<td>F3B-1</td>
<td>Bottom insulator (16.5)</td>
<td>( \sim 1.2 \text{ min.} )</td>
</tr>
<tr>
<td>F3B-2</td>
<td>Bottom insulator (16.5)</td>
<td>( \sim 15.0 \text{ min.} )</td>
</tr>
<tr>
<td>F3B-3</td>
<td>Top portion of fuel column (5.1)</td>
<td>( \sim 25 \text{ sec.} )</td>
</tr>
<tr>
<td>F3B-4</td>
<td>Top portion of fuel column (5.1)</td>
<td>( \sim 40 \text{ sec.} )</td>
</tr>
<tr>
<td>F3B-5</td>
<td>Upper insulator (1.5)</td>
<td>( \ll \sim 25 \text{ sec.} )</td>
</tr>
<tr>
<td>F3B-7</td>
<td>Plenum</td>
<td>( \ll \sim 25 \text{ sec.} )</td>
</tr>
</tbody>
</table>

*Could not be measured accurately by instrumentation.

### TABLE A-3
GE OUT-OF-PILE FISSION GAS RELEASE SIMULATION EXPERIMENTS

<table>
<thead>
<tr>
<th>Simulated Fuel-Cladding Diametral Gap (mils)</th>
<th>Fuel Column Length (in.)</th>
<th>Differential Pressure Range (Δp-psi)</th>
<th>Range of Flow Rates (gm/sec)</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.7</td>
<td>31.25</td>
<td>( \sim 300 \text{ to } 1280 )</td>
<td>0.00287 to 0.0548</td>
</tr>
<tr>
<td></td>
<td>16</td>
<td>( \sim 140 \text{ to } 785 )</td>
<td>0.00174 to 0.0542</td>
</tr>
<tr>
<td>2.8 to 3.1</td>
<td>30</td>
<td>( \sim 25 \text{ to } 1380 )</td>
<td>0.00234 to 0.731</td>
</tr>
<tr>
<td></td>
<td>15</td>
<td>( \sim 200 \text{ to } 1265 )</td>
<td>0.202 to 0.827</td>
</tr>
<tr>
<td>4.8</td>
<td>30.5</td>
<td>( \sim 100 \text{ to } 1280 )</td>
<td>0.192 to 1.56</td>
</tr>
<tr>
<td></td>
<td>15</td>
<td>( \sim 100 \text{ to } 1090 )</td>
<td>0.214 to 1.53</td>
</tr>
</tbody>
</table>
3. PNC Out-of-Pile Experiments

In addition to the other experiments which are referred to in Section III, out-of-pile experiments were also conducted by the Power Reactor and Nuclear Fuel Development Corporation (PNC) of Japan to evaluate the thermal and hydrodynamic effects due to fission gas release. A-10 The test section used in these experiments was a 37-pin bundle consisting of a central gas injector pin, seven electrically heated pins and the rest dummy pins. All the pins were wrapped with spacer wire except the central pin such that a gas injector pin with injector nozzles of various diameters (0.3, 0.5, and 0.8 mm, or 0.012, 0.020, and 0.031 in.) and axial locations could be inserted into the test section. The pins were 151.5 cm (59.65 in.) long with an outside diameter of 6.5 mm (0.256 in.) and were arranged with a triangular pitch of 7.9 mm (.311 in.). The heated pins had a heated section of 45 cm (17.72 in.) and had a uniform heat flux. Heated argon gas was injected into the flowing sodium stream and onto the adjacent heated pins to simulate fission gas release in the fueled section of fuel pins. The gas injection point was either 18.5 cm (7.28 in.) or 22.5 cm (8.86 in.) downstream of the beginning of the heated section, but was 12.0 cm (4.72 in.) upstream for the two-phase flow heat transfer tests. Thermocouples, voidmeter, pressure transducer, flowmeters, and accelerometers were used to record the various responses due to the simulated gas release phenomena. Figure A-2 shows the simplified test section arrangement and Figure A-3 shows the schematic diagram of the gas injector pin. The test conditions are tabulated in Table A-4.

Based upon the measurements of temperature, pressure, flow velocity and acoustic noise fluctuation, the following major conclusions were drawn. However, it was pointed out that when these test results are applied to the actual reactor conditions, attention should be paid to the fact that the actual fission gas is composed mainly of xenon while these tests were performed with argon which has a density about 30% of that of xenon.

- The pin surface temperature increases due to the gas injection. The highest temperature increase occurs in the gas impingement region. The heat transfer coefficient in
Figure A-2. Test Section with a 37-Pin Bundle (PNC-FS-345).
Figure A.3. Schematic Diagram of Gas-Injector Pin (PG-1S-2886).
<table>
<thead>
<tr>
<th>Test Case</th>
<th>Gas Injection Nozzle Diameter, mm</th>
<th>Gas Plenum Volume, cm³</th>
<th>Gas Plenum Pressure, bars</th>
<th>Gas Release Rate, g/sec</th>
<th>Gas Release Rate, cm³ (STP/sec)</th>
<th>Quality</th>
<th>Heater Pin Heat Flux, w/cm²</th>
<th>Sodium Inlet Temperature, °C</th>
<th>Sodium Inlet Velocity, m/sec</th>
</tr>
</thead>
<tbody>
<tr>
<td>Transitory Release</td>
<td>70, 1240</td>
<td>15.9, 29.6 - 75.9</td>
<td>6 - 64</td>
<td>0.3 - 4.17 (Initial)</td>
<td>354 - 2340 (Initial)</td>
<td>34 - 1930</td>
<td>36.4, 77.3 - 91.6</td>
<td>228 - 276</td>
<td>1.95, 4.79 - 5.03</td>
</tr>
<tr>
<td>Continuous Release</td>
<td>0.3, 0.5, 0.8</td>
<td>0.3, 0.5, 0.8</td>
<td>0.29 - 2.05</td>
<td>0.06 - 3.44</td>
<td>162 - 1150</td>
<td>7.1 x 10⁻⁵ - 4.5 x 10⁻³</td>
<td>9.1 - 93.8</td>
<td>221 - 279</td>
<td>0.5 - 5.0</td>
</tr>
<tr>
<td>Two-Phase Flow</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
this region decreases to 1/20 to 3/20 of the sodium single-phase value. This two-phase heat transfer coefficient is similar to that obtained in the continuous release experiments. Extrapolating the results to actual reactor conditions with heat flux of 200 w/cm² (0.6344 x 10⁶ Btu/hr*ft²) and sodium velocity of 5 m/sec (16.4 ft/sec), the increase of the pin surface temperature due to gas impingement is calculated to be less than 240°C (432°F).

- The pressure pulses introduced by the transient gas injection are less than 1/5 of the initial gas plenum pressure. Neither damage nor deformation was observed in the pin bundle after a total of 14 transient experiments. A mechanical loading effect of fission gas release in the fuel region is therefore not a possible cause of fuel pin failure.

- In the transient release experiments, the sodium inlet velocity increases due to gas injection, and the amount of this increase is proportional to the gas plenum pressure. However, the experimental values are considerably smaller than the theoretical values obtained by the one-dimensional slug-ejection model.

- The outlet velocity fluctuation and the acoustic noise increase with the increase in gas release rate.

C. References


A-7. "One and Seven-Pin Loss of Coolant Experiments in SCARABEE" by a working group of experts from CEA, KfK, and UKAEA, to be published in Nuclear Engineering and Design.


APPENDIX B

MATHEMATICAL MODELS
FOR
COOLANT FLOW REVERSAL
A. Introduction

Stochastic failures of fuel pin cladding are expected in spite of the best efforts at quality control and provision of design margins. All operational, experimental and analytical data indicate that stochastic failure of one or a few fuel pins will not propagate to other undamaged pins in the vicinity. The immediate consequence of cladding failure is the release of fission gas contained in the plenum of each pin. Since cladding failure is usually a pin hole crack, the fission gas would be released as bubbles into the coolant stream. The depressurization would be gradual with the only cause for concern being decreased heat removal capability of the two-phase mixture of gas bubbles and sodium coolant. This appendix considers the relatively remote possibility that the failures were much larger than pin-hole size and that enough pins are involved to cause the discharge of sufficient fission gas to reverse coolant flow.


References B-1 and B-2 describe a one-dimensional model which explicitly considers flow reversal. The model considers coolant and gas dynamics but does not consider heat transfer. Fission gas release is assumed to occur from breaches in cladding of any given number of pins. The cladding breaches are treated as orifices for gas flow, which are assumed to appear simultaneously, instantaneously, and at the same axial elevation in all the pins. The gas flowing out of the breaches is assumed to form a bubble which covers the entire cross section of the subassembly. The gas outside the plena is assumed to remain within one bubble throughout the transient. The gas flow into the bubble is determined by the pressure difference between the gas plenum and bubble, assuming that these pressures are uniformly distributed within their boundaries. The transient is assumed to begin at the time of pins failure and end when the lower gas-liquid interface reaches the subassembly outlet.
A schematic representation of the system used for modelling in Reference 50 is shown in Figure B-1. By applying the law of conservation of momentum and of continuity, Reference B-1 derives the following expressions for the motion of the upper and lower gas-liquid interfaces:

\[
\frac{\rho}{g_c} \left[ (L-X_u) + \frac{A_c}{A_{TR}} H_L \right] \frac{d^2 x_u}{dt^2} = (P_D - P_E) - \rho \frac{g}{g_c} (L - X_u + H_L)
\]

\[
+ \rho \frac{U_c |U_c|}{2g} \left[ 1 - \left( \frac{A_c}{A_{TR}} \right)^2 - \left( \frac{A_c}{A_{TR}} \right)^2 \left( \text{Friction factors for transitions} \right) \right]
\]

\[- \left( \text{Friction loss thru bundle over length L-X_u} \right) \]

\[ (1) \]

\[
\frac{\rho}{g_c} \left[ x_L + L_{IV} \frac{A_c}{A_I} \right] \frac{d^2 x_L}{dt^2} = (P_D - P_b) - \rho \frac{g}{g_c} (L_{IV} + X_L)
\]

\[
+ \rho \frac{U_L |U_L|}{2g_c} \left[ \frac{A_c^2}{A_I} - 1 - \frac{A_c^2}{A_I} \left( \text{Friction factors for transitions} \right) \right]
\]

\[- \left( \text{Friction loss thru bundle over length X_L} \right) \]

\[ (2) \]

where,

\[ U_c = \frac{dx_u}{dt} \text{ and } U_L = \frac{dx_L}{dt} \]

\[ \rho = \text{sodium density} \]

\[ g_c = \text{dimensional conversion factor} \]

and other terms are defined in Figure B-1.
Figure B-1. Schematic Representation of Model
The left side of the equations contains the inertial lengths for the liquid columns and the acceleration, and the right side contains the terms for driving pressure, static head and frictional losses. The velocity-dependent terms are normalized to that in the pin bundle section. The sodium is considered to be incompressible, which is a conservative assumption because the predicted pressures would tend to be higher than those actually experienced.

The dynamics of the gas phase were modelled assuming an isothermal expansion of the bubble; the gas temperature was taken as being equal to that of the sodium at the point of breach. This assumption is justified because any reduction in temperature of the bubble due to expansion is likely to be compensated by heat input from the sodium and structures which surround the bubble. The constant temperature assumption is likely to be conservative because the volume of the bubble is over-predicted. Conservation of mass in the bubble requires that

$$\frac{d}{dt} (V_b \rho_b) = Q$$

(3)

where $V_b$, $\rho_b$, and $Q$ are respectively the volume, density and mass flow rate of gas from plena to the bubble. The subscript "b" denotes the bubble. From Equation (3) and the perfect-gas law,

$$\frac{d\rho_b}{dt} = \frac{1}{R T_b V_b} \left[ Q R T_b - \rho_b \frac{dV_b}{dt} \right]$$

(4)

The density of gas in the plena of the breached pins is obtained from the mass conservation relationship,

Mass of gas in the plena = Original mass - Mass in bubble
+ Initial mass in bubble,

which translates to:

$$\rho_p = \frac{p_{po}}{R T_{po}} - \frac{V_b}{V n} \rho_b + \frac{m_0}{V n}$$

(5)
where \( v \), \( n \), and \( m_0 \) respectively are the plenum volume per pin, number of breached pins and an arbitrarily small mass of gas to initiate calculation. The subscript "\( p \)" applies to the plenum and "\( o \)" denotes start of transient at zero time. The expansion of gas from the plenum is assumed to be adiabatic.

Thus, the plenum gas temperature is given by:

\[
\frac{T_p}{T_{po}} = \left(\frac{\rho_p}{\rho_{po}}\right)^{\gamma - 1}
\]  

(6)

the pressure in the plenum by:

\[
P_p = R \rho_p T_p
\]  

(7)

The calculation of the gas flow from the plenum, through the breach and into the bubble, for the case when breach location is not adjacent to the plenum, is based on a model shown schematically in Figure B-2 and derived from Reference B-3. The model accounts for gas flow through the annulus between the pellets and the cladding. Since irradiation-induced dimensional changes are different for the axial blanket pellets as compared with fuel, the model permits consideration of two annular regions between the plenum and the breach. Some designs incorporate a gas plenum below the pellet stack; hence, the model is capable of computing separately the gas discharge from the bottom plenum. For flow through the annulus, the model assumes that the gas in the fission gas plenum remains at a constant temperature. The justification for this assumption is that, due to the expected slower rate of depressurization, heat transfer into the plenum gas would occur and tend to compensate for the decreasing temperature.

The model provides for three possibilities for the area of the orifice between the gas plenum and the bubble. The first is called the infinite breach which is equal to the cross-sectional area of the fission gas plenum. It is considered infinite because for the elongated plenum configuration, there would be no significant increase in the discharge for larger orifice areas. The second is called the finite breach; it is calculated from the

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input value for breach diameter. The third is applied to cases where the gas flow is in the annulus between the pellets and cladding; in these cases, the actual size of the breach itself is not considered because the annulus is assumed to be the controlling factor.

Thus, the gas discharge into the bubble is calculated using the assumption of adiabatic expansion if the breach is postulated at a location where gas from the plenum flows directly to the breach (top-of-blanket elevation) and is based on isothermal expansion for gas flow through the pellet-clad gap (breach location in fuel region of pin). In both cases, a quasi steady-state model is used. Sonic and subsonic flow for adiabatic expansion are considered separately. The gas is always assumed to flow directly into the bubble from the breach, irrespective of where the gas-liquid interfaces are located. The expressions used for computation of gas discharge in the adiabatic and isothermal cases are shown below:

\[
Q(\text{Subsonic}) = C_d \ A \ \rho_p \ \left(\frac{P_b}{P_p}\right)^{\frac{1}{\gamma}} \ \frac{2\gamma}{\gamma - 1} \ \left[1 - \left(\frac{P_b}{P_p}\right)^{\frac{\gamma - 1}{\gamma}}\right] \ R \ T_p \ g_c \ \frac{1}{2}
\]

\[
Q(\text{Sonic}) = C_d \ A \ \rho_p \ \left(\frac{2}{\gamma + 1}\right)^{\frac{1}{\gamma - 1}} \ \frac{2\gamma}{\gamma + 1} \ \left[1 - \left(\frac{2\gamma}{\gamma + 1}\right) \ R \ T_p \ g_c \right]^{1/2}
\]

\[
Q(\text{Flow Through Annulus}) = \frac{\pi}{4} \ D \ \frac{g_c}{f \ R \ T} \ \frac{P_p^2 - P_{b}^2}{\frac{\xi_1}{D_{H1}} + \frac{\xi_2}{D_{H2}}} \ \frac{1}{2}
\]

where, in addition to the terms defined before, \( C_d \) is the coefficient of discharge applicable to the breach, \( A \) is the area of the breach, \( \gamma \) is the ratio of specific heat at constant pressure to specific heat at constant volume for the fission gas, \( D \) is the diameter of the gas plenum, \( f \) is the turbulent friction factor for the annular gas flow, and \( \xi \) and \( D_h \) are as shown in Figure B-2.
The above-described model is applied until the upper gas-liquid interface reaches the subassembly exit. Subsequently, the gas bubble dynamics includes expulsion into the outlet sodium plenum. The expulsion is assumed to occur as a series of single spherical bubbles in an "infinite sea" environment. A schematic representation of the expulsion model is shown in Figure B-3, which was obtained from Reference B-2. Every bubble in the outlet is assumed to be spherical at all times with formation of a new bubble beginning when the height of the center of the first bubble above the subassembly equals the radius of the bubble plus the radius of the subassembly outlet. The model is also subject to the following assumptions:

a. Inertia and viscosity effects of the gas are neglected because they are relatively small.

b. The temperature, pressure, and hence, the density within the gas bubble are uniform. The bubble expansion is again assumed to be isothermal.

c. The free surface of the sodium is far away from the subassembly outlet and is stationary.

d. The effect of viscosity of the coolant surrounding the spherical gas bubble is considered negligible.

Subject to these assumptions, Reference B-2 derives the following expression for the radial expansion of the bubble:

\[
\frac{da^2}{dt^2} = \frac{g_c}{\rho a} \left\{ (p_b - p_a) - \frac{\rho g}{g_c} (H - \frac{S+a}{2}) - \frac{3}{2} \frac{\rho}{g_c} (\frac{da}{dt})^2 \right. \right. \\
- \frac{\rho}{g_c} \frac{(a-S)}{4a} \left( a \frac{dU_b}{dt} + 3 \frac{da}{dt} U_b \right) \\
+ \frac{\rho U_b^2}{4g_c(S+a)} \left[ 1 + \frac{5}{2} \frac{(S)}{a^2} - \frac{3}{2} \frac{(S)}{a^3} \right] \left( \frac{da}{dt} \right)^2 \}
\]

(11)
Figure B-3. Schematic Representation of Expansion of Fission Gas in Coolant Plenum Exit
The initial conditions for this equation are that the bubble radius equals the subassembly outlet radius and the bubble-wall velocity equals the velocity of the upper gas-liquid interface as obtained from the first stage computations when both interfaces were within the subassembly. Bubbles formed subsequent to the first are assumed to begin with a stationary bubble wall. The nomenclature for Equation (11) is evident from Figure B-3 with \( U_b \) given by:

\[
\frac{dU_b}{dt} = \frac{d^2S}{dt^2} = \frac{(\rho - \rho_b) g}{\rho_b + \frac{11}{16} \rho} - \frac{U_b}{V_b} \cdot \frac{dV_b}{dt} - \frac{U_b}{(\rho_b + \frac{11}{16} \rho)} \frac{d\rho_b}{dt}
\]  

(12)

The analytical model discussed above is valid only when the gas bubble forms a single slug and remains so throughout the transient. Reference B-1 gives an expression which quantitatively describes the criterion for applicability of the analytical model. This expression is based on the assumptions that (a) if the model is applicable at the beginning of the transient then it is applicable throughout, and (b) the condition for applicability at the beginning of the transient is that the volumetric discharge of gas through the breaches is greater than the volumetric flow rate of the coolant. This criterion has been shown to reduce to the following:

\[
\text{Number of breached pins} \geq \frac{B}{A} \cdot \frac{P_{bo}}{p_0} \cdot \frac{U_o A_c}{(R T_{bo} g_c)^{1/2}}
\]  

(13)

where,

\[
\frac{1}{B} = \left( \frac{2}{y+1} \right) \frac{1}{y-1} \left( \frac{2y}{y+1} \right)^{1/2}
\]  

(14)

and other terms are as defined before with the subscript "o" standing for the initial condition. If this criterion is not fulfilled, the gas flow is likely to be in the bubbly flow regime rather than the slug flow regime.
The fission gas release model has been coded using the Continuous Systems Modelling Program of IBM Corporation. The code is available at the Argonne National Laboratories. A sample analysis was performed to illustrate the use of the model and is described herein. The design parameters of the Clinch River Breeder Reactor were chosen as the basis for the analysis. The gas plenum pressure was taken as 67.9 bars (985 psia), which is representative of end-of-life for the initial core loading; the temperature was taken as 603°C (1118°F) which is representative of the hot channel coolant. The sodium velocity corresponded to the mass flow through one of the highest power subassemblies (6.5 m/sec or 21.3 ft/sec). Since the model assumes all ruptures at the same axial elevation, two cases were examined; one with ruptures at top-of-core and the other at top-of-blanket. All 217 pins were assumed to fail simultaneously while the reactor operated normally and at full power. This assumption simplifies the analysis; however, no mechanism is known which can cause it. The top-of-blanket ruptures were assumed to be of effectively infinite area.

Figure B-4 shows the results of the analysis as displacements of the upper and lower gas-liquid interfaces. It is seen that flow reversal does not occur for top-of-core ruptures. In this case, the gas slug retards the lower coolant column, but the direction remains upward throughout the transient. For the top-of-blanket ruptures, flow reversal occurs and carries the lower gas-liquid interface into the heated region of the core. The gas exerts sufficient upward force on the upper interface to expel it significantly earlier than the upper interface in the top-of-core case. However, the difference in time between the expulsions of the lower interface is relatively insignificant in spite of the substantial difference in flow effect. The main reason is that flow recovery occurs at a faster rate when the interface is pushed down further. The thermal effects of these transients are discussed later.

C. Limitations of the Mathematical Model

The mathematical model for flow reversal has limitations in the accuracy of prediction and the range of applicability with respect to analysis of hypothetical massive fission gas release. The limitations
are caused by the necessary simplifying assumptions. However, the limitations do not detract from the essential utility of the model because the predictions are in conformity with key experimental observations.

The accuracy of the model is affected by the assumptions that the gas flow is quasi steady-state and that all the gas phase occurs in a single bubble. With these constraints on the model, the plenum pressure does not achieve a practical equilibrium value at the end of the transient. For the sample analysis mentioned in Section 8 of this appendix, the top-of-blanket rupture showed the plenum pressure being sub-atmospheric at the end of the transient, whereas for the top-of-core rupture, the plenum pressure was still significantly high (∼34.5 bars or ∼500 psia) at the same time. In general, when high flow resistance is simulated between the gas plenum and the bubble, the final plenum pressure remains high, but the duration of the transient is relatively unaffected. The effect on the flow perturbation is consistent with the expected trends; that is, a low resistance causes a greater perturbation with a lower final plenum pressure.

Certain hypothetical possibilities can be identified which cannot be treated with this model. For example, all failures are constrained to occur simultaneously and at the same axial elevation. A failure sequence could be postulated in which the failures occur sequentially, are of a nature to satisfy the flow reversal criterion, and are axially placed to maximize flow reversal. However, in view of the very low probability of achieving flow reversal and multiple failures in the first place, such a hypothetical event need not be considered. The aforementioned studies conservatively encompass any reasonable failure conditions. Hence, the capability to analyze the more general problem is not required.

D. Thermal Effects of Flow Reversal

The perturbation to the coolant flow due to fission gas release will have thermal consequences if the event occurs with the reactor operating at power. With flow reversal, some coolant resides in the
heated section of the fuel subassembly for significantly longer duration than normal, causing it to overheat. In addition, cladding in the heated section will also overheat, especially if it is isolated from the coolant. Based on a conservative analysis, Reference B-2 has estimated that the cladding temperature could increase by 100°C (180°F) for every 0.1 second it is isolated from the coolant. The minimum coolant saturation temperature is approximately 890°C (1640°F), giving a margin to coolant boiling of >220°C (>400°F) under normally operating conditions. The simplified analysis in Reference B-2 would indicate that the margin to coolant boiling may be seriously depleted if the flow reversal lasts for a few tenths of a second. To show that such a conclusion is overly pessimistic, a more detailed thermal analysis was performed and is described below.

A model of a fuel pin was set up to perform coupled thermal and hydraulic analyses using the computer code TRUMP. B-4. The model and one of its applications are described in Reference B-5. An analysis similar to the one in Reference B-5 was performed for normal operating conditions using CRBR design parameters. This analysis is described here to illustrate that the actual margin to coolant boiling is available.

The thermal model used a nodal representation with sixteen fuel nodes over the 91.4 cm (36 inch) active length. Each fuel node was associated with a cladding and coolant node. For simplicity, constant material properties and constant heat transfer coefficients in the fuel-clad gap and in the sodium film were assumed. The model is capable of considering temporal and spatial variations in heat generation and coolant flow rate. Thermal conductivity is used as the parameter controlling heat flux into each coolant node; gas blanketing was simulated by varying thermal conductivity of the coolant to be consistent with the predicted flow reversal.

The sample analysis described previously was used as the basis of the thermal analysis. The thermal effects are determined by the movement of the lower gas-liquid interface because the upper interface is outside the heated region throughout the transient.
Beginning-of-life heat generation rates were used for conservatism although the sample analysis used end-of-life fission gas parameters. Figures B-5 and B-6 show the results of the analyses for top-of-core and top-of-blanket ruptures respectively. The movement of the lower interface is reproduced in each figure along with the maximum temperatures of cladding and coolant.

Figure B-5 shows that the increased coolant residence time in the core affects the cladding and coolant to a relatively minor extent. Since there is no flow reversal in this case, the heated region has coolant flow throughout, and the residence time is the only factor which influences the coolant temperature. The maximum coolant temperature increases by $8^\circ$C ($14^\circ$F) and the maximum cladding temperature by $12^\circ$C ($22^\circ$F) through the transient. The temperature increase is a factor of 28 less than the approximately $220^\circ$C ($400^\circ$F) margin available to coolant boiling. This is a sufficiently high margin that the high pressure in the gas plenum remaining after the transient computation is not a potential problem. This is because the pressure decreased by approximately 40% in the first transient and further bubble generation will rapidly depressurize the plenum; also, the thermal effect on the coolant would diminish after the first transient.

A similar event postulated at the top-of-blanket elevation (which is directly connected to the fission gas plenum) causes a much more significant effect on the coolant flow, as shown in Figure B-6. It was found that flow reversal does occur in this case, with the lower gas-liquid interface moving about 43 cm (17 inches) below the rupture elevation. The CRBR fuel pins have 35.56 cm (14 inches) of axial blanket material on each end of the 91.44 cm (36 inches) of fuel. Hence, the flow reversal uncovers 7.6 cm (3 inches) of cladding in the heated region, depriving it of coolant for approximately 0.04 seconds. The procedure for simulating exposed cladding has been described in Reference B-5. The results of the computations, as shown in Figure B-6, indicate that the maximum coolant temperature increases by $26^\circ$C ($47^\circ$F) and the maximum cladding temperature increases by $30^\circ$C ($54^\circ$F). This shows that flow reversal substantially increases the thermal effect of the transient, but it is still about a factor of

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Figure B.6. Maximum Coolant and Clad Temperature and Lower Interface Position for Top of Blanket Rupture Location at End-of-Life Conditions.
eight less than the margin to coolant boiling. The simplified
treatment of Reference B-2 would indicate considerably higher
temperature increases for the transient lasting up to 0.3 seconds.

Figure B-7 shows the axial temperature profiles of the coolant at
two instants during the transient. At the bottom of the flow reversal,
which occurs 0.058 seconds into the transient, the temperature profile
is considerably flattened by the hotter coolant flowing back into the
core. As normal coolant flow is re-established, the maximum
temperature does not change significantly, but the location of the
maximum moves back to the top-of-core elevation. Thus, a detailed
thermal model indicates that the margin to coolant boiling, under
normal operating conditions, will not be seriously depleted by failure
of a set of fuel elements injecting fission gas into the coolant stream.
Figure B-7. Axial Coolant Temperature Profiles Through Flow Reversal Transient – Top of Blanket Rupture Location at End-of-Life Conditions
E. References


