No. 16586

UNITED STATES OF AMERICA
and
UNITED KINGDOM OF GREAT BRITAIN AND NORTHERN IRELAND

Implementing Agreement for the collaboration of the United States Nuclear Regulatory Commission (USNRC) and the United Kingdom Atomic Energy Authority (UKAEA) in experimental programs concerned with the safety of nuclear reactors (with appendices). Signed on 10 November and 14 December 1976

Authentic text: English.

Registered by the United States of America on 27 April 1978.

ÉTATS-UNIS D'AMÉRIQUE
et
ROYAUME-UNI DE GRANDE-BRETAGNE ET D'IRLANDE DU NORD

Accord d'application concernant la collaboration entre la United States Nuclear Regulatory Commission (USNRC) et la United Kingdom Atomic Energy Authority (UKAEA) dans le cadre des programmes expérimentaux relatifs à la sûreté des réacteurs nucléaires (avec appendices). Signé les 10 novembre et 14 décembre 1976

Texte authentique : anglais.

Enregistré par les États-Unis d'Amérique le 27 avril 1978.

IMPLEMENTING AGREEMENT[1] FOR THE COLLABORATION OF THE USNRC AND THE UKAEA IN EXPERIMENTAL PROGRAMS CONCERNED WITH THE SAFETY OF NUCLEAR REACTORS

This AGREEMENT effective as of the 14 day of December 1976 between the UNITED STATES NUCLEAR REGULATORY COMMISSION (hereinafter referred to as "USNRC") and the UNITED KINGDOM ATOMIC ENERGY AUTHORITY (hereinafter referred to as the "Authority")

WHEREAS, the Governments of the United Kingdom and United States have an Agreement for Cooperation on the Civil Uses of Atomic Energy signed on 15 June 1955,[2] and

WHEREAS, the USNRC and the Authority are considering the continuation of the cooperation in fast reactor safety technology and extension to cooperation in the safety aspects of thermal reactors by signing an Agreement in the Field of Nuclear Safety Research and Development; and

WHEREAS, the International Energy Agency (hereinafter referred to as the IEA) encourages participating countries to undertake, as a matter of priority, cooperative programs on nuclear safety; and

WHEREAS, there is an increasing number of nuclear power reactors worldwide of proven types and of advanced designs; and

WHEREAS, international collaboration in research on the safety of nuclear power reactors is recognized to be of benefit to the health and welfare of people of all countries; and

WHEREAS, the achievement of full reciprocity in the exchange of technical information in the field of reactor safety research is a common objective; and

CONSIDERING that a program of research on Aerosol Release and Transport in LMFBR plants (ART) and the Heavy Section Steel Technology (HSST) is to be conducted by the USNRC; and

CONSIDERING that the Authority is to carry out a program of research on Core Debris Control, Fracture Mechanics, and Dryout and Post-Dryout Performance on SGHWR fuel elements; and

CONSIDERING that the USNRC and the Authority each wish to have access to the aforesaid programs of research of the other on a collaborative basis:

NOW, THEREFORE, the USNRC and the Authority agree as follows:

Article I. DESIGNATION OF THE COLLABORATIVE PROJECT

1. The USNRC and the Authority shall collaborate, in accordance with the provisions of this Agreement, in a program, Technical Program A, which is summarized in Appendix A to this Agreement, and which is to be managed by the

[1] Came into force on 14 December 1976 by signature, in accordance with article VII (1).
[2] United Nations, *Treaty Series*, vol. 229, p. 73.

USNRC, and also in a program, Technical Program B, which is summarized in Appendix B to this Agreement, and which is to be managed by the Authority.

Article II. SCOPE OF AGREEMENT

A. *Scope of Responsibilities*

1. Subject to the availability of funds, the USNRC agrees to provide the necessary personnel, materials, equipment and services for the performance of Technical Program A and, subject to the availability of funds, the Authority agrees to provide the necessary personnel, materials, equipment and services for the performance of Technical Program B.

2. Responsibility for decisions concerning changes or modifications in Technical Program A shall lie with the USNRC, and responsibility for decisions concerning changes or modifications in Technical Program B shall lie with the Authority.

B. *Participation and Attachments*

1. The USNRC shall have the right to nominate at its own expense a technical expert to be a Consultant Member of the Authority Program Review Group which shall meet periodically to review the status and progress of Technical Program B.

[2.] The Authority shall have the right to nominate at its own expense a technical expert to be a Consultant Member of the USNRC Program Review Group which shall meet periodically to review the status and progress of Technical Program A.

3. The USNRC at its own expense may attach one technical expert to participate in the conduct and analysis of the experiments of Technical Program B.

4. The Authority, at its own expense, may attach one technical expert to participate in the conduct and analysis of the experiments of Technical Program A.

5. The attachment of staff by one party to the other under Clauses B3 and B4 above shall be the subject of a separate agreement in respect of each person attached. A party proposing an attachment shall notify the other party of the name of the person proposed and shall provide such other information as may be required by the other party.

Each party may approve or reject any persons so proposed and may at any time, without giving any reason, revoke any approval previously given.

C. *Provision of Information*

1. The USNRC shall make available to the Authority all experimental data developed under Technical Program A during the period of this Agreement.

2. The Authority shall make available to the USNRC all experimental data developed under Technical Program B during the period of this Agreement.

3. The USNRC shall provide to the Authority access to operational computer codes, other than codes which form part of proprietary information, existing at the date of the parties entering into this Agreement or developed during the course of this Agreement and which are used by the USNRC to analyze experimental data arising from Technical Program A.

4. The Authority shall provide the USNRC access to operational computer codes, other than codes which form part of proprietary information, existing at the date of the parties entering into this Agreement or developed during the course of this Agreement and which are used by the Authority to analyze experimental data arising from Technical Program B.

5. The USNRC shall make available to the Authority the results of any analysis of information arising from Technical Program A.

6. The Authority shall make available to the USNRC the results of any analysis of information arising from Technical Program B.

Article III. EXCHANGE OF SCIENTIFIC INFORMATION AND USE OF THE RESULTS OF PROGRAMS*

1. As set forth in this Agreement:

i. "Participating Countries" means all states which participate in the International Energy Program as Participating Countries of the IEA.

ii. "Participants" shall mean the Governments of Participating Countries,

3. The USNRC and the Authority agree that the application or use of any information exchanged or transferred between them shall be the responsibility of the party receiving the information, and the transmitting party does not warrant the suitability of the information for any particular use or application.

4. The USNRC and the Authority agree that information arising from the Technical Programs may be made available by the recipient to Government authorities, licensees and utilities in the country of the recipient for their own use but shall not be available for publication otherwise without the agreement of the transmitting party.

Article IV. PATENTS

A. With respect to any invention or discovery made or conceived during the period of, or in the course of or under, this Agreement for the Authority's participation in Technical Program A, the USNRC on behalf of the United States Government, as recipient party, and the Authority as assigning party, and for USNRC participation in Technical Program B, the Authority on behalf of the United Kingdom Government, as recipient party, and the USNRC as assigning party, hereby agree that:

1. If made or conceived by personnel of one party (the assigning party) or its contractors while assigned to the other party (recipient party) or its contractors:
   (*a*) The recipient party shall acquire all right, title and interest in and to any such invention, discovery, patent application or patent in its own country and in third countries, subject to a non-exclusive, irrevocable, royalty-free license to the assigning party, with the right to grant sublicenses, under any such invention, discovery, patent application or patent for use in the production or utilization of special nuclear material or atomic energy; and

* This article is printed as it appears in the signed original agreement. (Information provided by the Government of the United States of America.)

(*b*) The assigning party shall acquire all right, title, and interest in and to any such invention, discovery, patent application, or patent in its own country, subject to a non-exclusive, irrevocable, royalty-free license to the recipient party, with the right to grant sublicenses, under any such invention, discovery, patent application or patent, for use in the production or utilization of special nuclear material or atomic energy.

2. If made or conceived by personnel other than the personnel of Paragraph 1 above, as a result of attendance at meetings or as a result of employing information which had been communicated under this exchange arrangement by one party or its contractors to the other party or its contractors, the party of such personnel making the invention shall acquire all right, title and interest in and to any such invention, discovery, patent application or patent in all countries, subject to the grant to the other party of a royalty-free non-exclusive, irrevocable license, with the right to grant sublicenses, in and to any such invention, discovery, patent application, or patent, in all countries, for use in the production or utilization of special nuclear material or atomic energy.

B. Neither party shall discriminate against citizens of the country of the other party with respect to granting any license or sublicense under any invention pursuant to subparagraphs A.1. and A.2. above.

C. Each party waives any and all claims against the other party for compensation, royalty or award as regards any such inventions or discovery, patent application, or patent, and releases the other party with respect to any and all such claims, including any claims under the provisions of the U.S. Atomic Energy Act of 1954, as amended, and appropriate U.K. laws and the Authority assumes the obligation under the U.K. Law insofar as the USNRC and its contractors are concerned.

Article V. PROGRAM CHANGE, TERMINATION AND ACCESSION

It is agreed that:

1. If either Technical Program A or B is substantially increased in scope the parties shall consider ways in which the equitable balance of the exchange may be maintained.
2. If either Technical Program A or B is substantially reduced or eliminated, work mutually agreed to be of equivalent interest may be substituted by mutual agreement.
3. Either party may withdraw from this Agreement on giving 6 months' notice to the other.
4. Other Participating Countries at all times may take part in either of the Technical Programs under this Agreement subject to the agreement of the party managing the Program.

Article VI. DISPUTES

Any dispute between the parties concerning the interpretation or application of this Agreement shall be settled by consultation and discussion.

Article VII. DURATION

1. This Agreement shall remain in force for 4 years after its effective date, which shall be the latter date of signature, and may be extended by mutual agreement.

For the United States Nuclear Regulatory Commission:		For the United Kingdom Atomic Energy Authority:	
By:	LEE V. GOSSICK	*By:*	G. H. KINCHIN
Title:	Executive Director for Operations	*Title:*	Director, SRD
Date:	14 December 1976	*Date:*	10 November 1976

APPENDIX A

1. AEROSOL RELEASE AND TRANSPORT (ART) PROGRAM

The objective of the Aerosol Release and Transport Program is to produce the experimental data and analytical techniques necessary to translate the calculated energy deposition sequences of the hypothetical core disruptive accidents (HCDA) into the "source" available for release from primary and secondary containment vessels. The characteristics of the vapors produced will be determined in terms of the radionuclides of interest, their chemical and physical states, their distributions in aerosols, and their transient behavior in confined environments at high concentrations (in excess of 30 g/M^3). Mechanisms that alter the character of, or reduce the concentration of, the fuel-clad-sodium-fission product mixture during transit through the overlying sodium will also be studied.

A capacitor discharge vaporization (CDV) technique is being perfected to produce energy depositions in LMFBR fuel at rates greater than 10^6 joules/gram-sec over periods of 2 to 3 milliseconds. In this CDV system, the fuel serves as an electrical conductor. Electrical energy which is stored in capacitor banks is discharged in a controlled manner into the fuel. The objective is to provide a non-nuclear means of experimentally studying the fuel (core) response to HCDA-like energy depositions.

The experiments involving release of plutonium will be conducted in a small vessel (CRI-III) of about 0.5 M^3 volume which is doubly contained in a hog-cell facility. The program also includes the identification and qualification of appropriate simulant aerosols (e.g., UO_2) and generation techniques that will be used for additional aerosol and bubble transport studies. These low hazard ex-hot-cell tests will be made in several vessels of different sizes which include the CRI-III vessel, an existing 5 M^3 vessel (CRI-II), a large 38 M^3 "containment size" vessel (NSPP), and an auxiliary bubble-transport vessel ($\sim$1 M^3). Other appropriate existing vessels will be used for development of the capacitor discharge vaporizer (CDV) system, for development of aerosol sampling equipment and instrumentation, and for underwater tests to study the bubble behavior.

The ART Program is directing early intensive effort toward completing the CDV development. Simultaneously, simulant aerosols generated by more continuous alternate techniques are being studied in the various vessels. The program itself is divided into seven related task areas (Tasks 2 to 7 and Task 9).

Task 2, "Cold Proof Tests," will provide a comparison of the behavior of a standard aerosol in two vessels (CRI-II and CRI-III). The aerosol will be produced by rapid oxidation of inductively heated uranium metal. In addition, UO_2 aerosols will be generated by any of the several alternative methods under consideration which include using UO_2 as a DC arc consumable electrode, direct vaporization of UO_2 from a tantalum carbide filament heat source, and vaporization of UO_2 through inductive heating in a water-cooled, copper crucible.

Task 3, "Trace Level Experiments," will better simulate fuel aerosols by adding, to UO_2, trace amounts of radioactive elements representing plutonium and certain fission products. Baseline reference data will be obtained for later comparison with real fuel aerosols.

Task 4, "Fuel Vaporization Test Without Sodium," will use the CDV system and fuel simulant to provide the first estimate of the upper limit fuel aerosol source term as a function of the HCDA energy deposition. These tests, however, will not include sodium in the vaporization process.

Task 6, "Fuel Vaporization Tests With Sodium," will extend Task 4 by adding selected amounts of sodium to be vaporized simultaneously with the fuel.

Task 7, "Effects of Radiation," would add significant amounts of previously activated sodium to the fuel aerosol and apply an external gamma source to investigate possible effects of the reactor radiation environment on the aerosol behavior.

Task 5, "Fuel Aerosol Simulant Tests," is a parallel multipurpose effort in which the CDV technique will be optimized, fuel aerosol simulants identified and their behavior characterized, underwater and undersodium bubble dynamics and transport behavior studied, and a determination made of the attenuation of the source term due to the bubble rise and contact with the sodium and upper plenum structures.

Task 9, "Aerosol Studies in the Aerosol Test Facility (ATF)," involves the coexistent behavior of UO_2 and sodium aerosols and will be carried out in the modified Nuclear Safety Pilot Plant. The experiments will serve to determine scaling effects by comparison with results obtained in smaller equipment and also to determine the extent of interaction between uranium dioxide and sodium aerosols. The conditions will cover those resulting from a range of accident severity up to and including loss of flow (LOF) and unprotected transient overpower (TOP).

SUMMARY OF AEROSOL RELEASE AND TRANSPORT PROGRAM

Task (No.)	*Vessels*	*Concentration range (g/m^3)*	*Type of aerosol*	*Generation technique*	*Remarks*
Cold Proof Tests (2) ..	CRI-II CRI-III	1 to 30	U_3O_8	oxidation of uranium metal	calibration of system, demonstration of scale-up
Cold Proof Tests (2) ..	CRI-III	1 to 10	UO_2—SS	probably by consumable electrode	alternative generation techniques are being considered
Trace Level Experiments (3)	CRI-II	1 to 30	UO_2—SS (tracers)	probably by arc vaporization hearth furnace	generation techniques depend on availability of apparatus
Fuel Vaporization Tests Without Sodium (4) ..	CRI-III	1 to 30	UO_2—SS UO_2—PuO_2—SS	CDV	upper limit source term as function of energy deposition
Fuel Vaporization Tests With Sodium (6)	CRI-III	1 to 30	UO_2—PuO_2—SS + Na_2O	CDV	possible physical-chemical effects of addition of sodium aerosol

(continued)

Radiation Effects (7) ..	CRI-III	1 to 30	UO_2—PuO_2—SS + sodium	CDV	addition of 10 to 1000 curies of Na 24 + external gamma source
CDV Development (5) .	CRI-I Fast Vessel	>30	UO_2—SS	CDV	optimum CDV design
Fuel Aerosol Simulant Tests (5)	Fast Vessel	1 to 30	UO_2—SS simulants		bubble behavior and source attenuation mechanisms
Aerosol Test Facility (9)	NSPP	1 to 5 1 to 50	U_3O_8 U_3O—sodium oxide	consumable are sodium fire	effects of coagglomerates in secondary containments

II. THE HEAVY SECTION STEEL TECHNOLOGY (HSST) PROGRAM

The Program

The Heavy-Section Steel Technology (HSST) Program is a major Nuclear Regulatory Commission (NRC) sponsored safety engineering research activity devoted to development of a quantitative basis for assuring adequate margins of safety against fracture of the primary coolant pressure boundaries of water-cooled nuclear power reactors. The principal objects of study are the thick-walled pressure vessels of these reactor systems. All relevant aspects of the technology of the steels and weldments commonly used in reactor pressure vessels are being investigated. Another important part of the program is to establish quantitative relationships between the characteristics of materials and loading conditions under which fracture would occur in a flawed structure.

The specific objectives of the program are to provide a thorough quantitative assessment of heavy-section reactor vessel steel fracture characteristics including a realistic assessment of fracture potential and development of fracture prevention criteria. The program will include the effects of irradiation, flaw growth mechanisms, and the effects of thermal shock, with crack propagation and arrest characteristics under both stress and toughness gradients.

Table 1 describes the general test program capabilities.

The program has been underway since 1967 and over 70 technical reports or progress reports have been produced. The program is extending into studies of thermal shock, weld heat affected zones and failure under pneumatic loads.

Research Areas

The HSST Program is comprised of the seven major research areas listed below:

— Elastic-Plastic Fracture Analysis Development and Evaluation: This part of the program has been set up to develop new methods of elastic-plastic fracture analysis and to evaluate existing methods. The required fracture toughness testing is performed in this area. Also this research area provides the analytical support for the thermal shock and the pneumatically loaded intermediate test vessel (ITV) programs.

— Fatigue Crack Growth and LWR Crack Growth Analyses: In this research area, the investigators are to continue to develop fatigue crack growth rate data including the effects of material, LWR water chemistry, temperature, R-ratio, cyclic rate, hold time, loading rate, etc., and to determine a realistic upper bound relationship between da/dN and delta K. From these data, the investigators will update the crack growth analyses for LWR pressure vessels.

— Irradiation Effects: The purpose of this research area is to determine the static and dynamic toughness of irradiated reactor vessel materials. Included among the FY 1975 tasks are completion of a 4T—CT program, and performance of a study of a method utilizing a "plug" of irradiated material surrounding the crack tip in an otherwise standard CT specimen. An irradiation program, using different heats of A533 B1, A508—2 and weld material "plugs", is being performed to characterize thoroughly the static and dynamic fracture toughness of reactor vessel steels.

— Intermediate Vessel Testing: The ITV tests were completed and a report on all ITV tests prepared. Currently a weld defect in ITV—9 is being characterized and ITV—7 is being prepared for pneumatic testing.

— Thermal Shock: The aim of this research area is to verify the method of analysis that is used to predict crack propagation in a reactor vessel subjected to emergency core cooling system (ECCS) operation following a postulated loss-of-coolant accident (LOCA). Thermal shock tests on 21-inch OD test cylinders will be completed and initial tests started on 39-inch OD cylinders.

— Pneumatic ITV Testing: Investigators in this area are to develop both an analytical predictive capability and experimental data on fracture behavior under pneumatic loading. The test parameters will be set to evaluate the "leak-before-break" probability under pneumatic loading.

— Heat Affected Zone Cracking: The purpose of this research is to determine the defects caused by reheat cracking in heat affected zones.

Table 1. HEAVY SECTION STEEL TEST PROGRAM CAPABILITIES

Test Phase	*Capabilities*
1. Intermediate Test Vessel (ITV) Testing . .	Temperatures from ambient to ~200 °F (~93 °C) Pressures from ambient to ~35 ksi (~241 MPa)
2. Pneumatic load testing of vessels	Vessel sizes up to ~39 in. (99 cm) O.D. by 54 in. (137 cm) high
3. Thermal Shock Testing	Temperatures from − 10 °F (− 23 °C) to 550 °F (288 °C) Ambient pressure Specimen sizes: straight cylinders 21 in (53 cm) O.D. and 39 in (99 cm) O.D.
4. Irradiation Effects	Hot cells for studying highly irradiated Charpy, tensile and 1T CT specimens.

APPENDIX B

I. CORE DEBRIS CONTROL STUDIES

Long-Term Objectives

It is essential after a hypothetical meltdown of part or all of the core, and the resulting reactor shutdown, that the decay heat of the core debris be removed in a manner which will ensure that the location of the core debris remains under control. The overall objective of this program is to provide the basic theoretical analysis and experimental data on heat transfer necessary for the effective design of reactor infra-structure or sub-structure to achieve debris control. A variety of reactor types is being considered.

Background

The core debris produced by melt-out may be contained within the reactor vessel itself through the use of internal catchers in some configuration, or a sacrificial secondary containment bed within a prepared receptacle below the reactor may provide the final lodging place for the core melt. The validity of either of these containment concepts depends on the power rating, the bed rock, the method cooling and type of coolant. It also depends on the form of the core debris, whether primarily in large lumps or in fragmented fines.

At present in the UK the information relating to both of these concepts is rather fragmentary. Two ongoing projects are proposed within the UK program which will also be of value to others who may wish to collaborate.

Project 1:	Sacrificial Secondary Containment beneath the Reactor.
Object:	To develop a detailed framework within which different methods of secondary containment can be evaluated, and the consequences of variations in the choice of particular materials and geometries examined speedily. For this a computer program is an essential tool. General features rather than specific physics is emphasized in this project.
Project 2:	The Development of Core Catchers within a Reactor.
Object:	To develop a detailed framework within which the deployment of different types of internal core catchers can be examined and the overall characteristics evaluated. For this a computer program is an essential tool also. The physics of specific core catcher configurations will be fed into this and gross consequences assessed.

It is intended that these two projects will go ahead with a small team of people and it is suggested that it would be beneficial to other countries to attach members to these projects for a period.

These projects will highlight in a systematic way where data is inadequate for making soundly-based designs for core debris control structures.

Work on some specific areas of physics is in progress both in the UK and in other countries where core debris control research is being carried out. Two areas where we are contributing are:

(*a*) In the theoretical analysis and computer modelling of
- (i) Convection and heat transfer of specific geometries for internal catchers, and
- (ii) The growth of the region molten core debris/rock mixture in sacrificial beds as the fission products decay.

(*b*) Analogue experiments studying convection, involving heat generation.

Facilities Available at Culham

(i) For Computer Modelling, the OLYMPUS system has been developed to a state in which intelligible computer codes can be programmed quickly.

(ii) A small-scale heat transfer laboratory in which lasers and laser holography enable diagnostic measurements of heat flux to be made. The presence of fusion work at Culham provides additional laser expertise on site.

II. DRYOUT AND POST-DRYOUT PERFORMANCE TEST ON 60-PIN FUEL ELEMENTS FOR SGHW REACTORS

Objective

1. The establishment of the dryout and post-dryout performance of 60-pin fuel elements (with a pin diameter of 12.2 mm) forms an important part of the work associated with the

development of an advanced fuel element for SGHW reactors. The performance endorsement program involves out-reactor dryout tests to be carried out in the electrically heated 9 MW rig heat transfer facility at AEE Winfrith in 1975, followed by in-reactor dryout and post-dryout tests to be carried out in the Winfrith SGHWR power station in 1977.

9 MW Rig Dryout Tests

2. The 9 MW rig at AEE Winfrith is a major heat transfer facility which enables dryout tests to be carried out on multi-rod electrically heated test sections which simulate full-scale SGHWR fuel elements. The 60-pin test section will be contained within a vertical channel of SGHW reactor pressure tube dimensions and will be subjected to widely ranging conditions of power, flow, pressure and inlet sub-cooling in order to determine the map of onset-of-dryout over a wide range of variables. The 9 MW rig tests will provide detailed information over a wide range of parameters and will also provide advance information on the evaluation and location of the onset of dryout and hence give some guidance on the location of the dryout detectors in the in-reactor test fuel element.

Tests

3. Dryout powers will be measured over the following ranges of variables:

Flow	2	16 10 kg/s
Pressure	30	70 bar
Sub-cooling at inlet	20	80 kj/kg

Tests will be carried out with and without the outer six sparge tubes. Some tests will be carried out by reducing flow, keeping power, etc. constant.

Heat Flux Profile

4. The specification of the heat flux profile is as follows:

Axial form		:			Heated length 3.66 mm		
Radial	100	:	70	:	55	:	55

The radial distribution represents start-of-life conditions for a uniformly enriched fuel element. DC electrical heating will be used. Test pins diameter — 12.2 mm.

Instrumentation

5. Dryout detection will be by 3 radiation-sensitive foil thermo-couples located at the top-end of each heated rod.

Timescale

6. The test section has been built and is awaiting installation in the 9 MW rig. Tests are expected to commence in Autumn 1975, depending on the duration of other programs on the 9 MW rig.

In-Reactor Dryout and Post-Dryout Tests

7. The Winfrith SGHWR in-reactor dryout and post-dryout test program will be carried out on a full size 60-pin element located in one of the reactor's central zone pressure tubes which is connected to a separate circuit known as the cluster loop. The cluster loop provides independent control of channel flow rate, channel inlet sub-cooling and circuit pressure. The power developed by the test fuel element is a function of its enrichment and the power level and power distribution of the reactor as a whole.

8. The in-reactor experimental objectives are:

(*a*) To investigate the onset of dryout for various combinations of channel power, inlet sub-cooling and inlet pressure;

(*b*) To investigate fuel pin cladding surface temperatures and distributions under stable post-dryout conditions and subsequent rewetting;

(*c*) To investigate fuel pin cladding surface temperature response to rapid flow reductions into stable post-dryout conditions, followed by rewetting.

9. It is anticipated that the test program will follow substantially the methods used in the first series of Winfrith SGHWR in-reactor dryout tests which are reported in the January 1974 issue of the *Journal of the British Nuclear Energy Society.*

Tests

10. Tests will normally be carried out by reducing flow and keeping power, pressure, and inlet sub-cooling constant. It is expected that tests will be performed within the following ranges of conditions:

Power	3→	6MW
Peak pellet linear rating	250→	500W/cm
Peak pellet average heat flux	65→	130W/cm^2
Sub-cooling	20→	80kJ/kg
Pressure	20→	70bar

It is expected that channel flows down to 2.0 kg/s will be involved. The tests would also involve some continuous-ECW flow conditions. Investigations of rewetting behavior during a return to normal conditions, following an excursion into stable post-dryout conditions, will also be made.

Instrumentation

11. Dryout detection and post-dryout temperatures will be measured by means of about 50 thermocouples attached to the fuel cladding outer surfaces at selected locations.

Timescale

12. Tests are currently planned to commence in mid 1977.

Means of Co-operation

13. The test programs will be completely at the discretion of the UKAEA. Staff may be attached during 9 MW rig and SGHWR cluster loop experiments.

III. Fracture Mechanics

A. *REML*

The development of fracture mechanics techniques, particularly for elasto-plastic situations in ferritic and austenitic steels. This embraces the testing of defected model vessels up to 3 in. thickness and correlation of failure conditions with those derived analytically from data from small scale toughness tests. Both COD and J integral techniques are used with the development of techniques for establishing the onset of slow crack growth. The interrelationship of data derived from quality control tests (Charpy V, tensile) with fracture toughness is being studied on a series of simulated welded structures.

Limited acoustic emission trials and laser holography tests are being applied to the tests on defected cylinders and flat welded plates containing both natural and artificial defects.

B. *RFL*

Program 1: Thermal Shock

The dominant cyclic stressing agents in reactor circuit components, including pressure vessels, are thermal transients. These result in component surface thermal shock where fatigue cracks can be initiated or pre-existing defects grow by fatigue. As such cracks grow into the material, the stress-strain field controlling growth is still that induced by the thermal transients. The work program examines experimentally and theoretically the growth of cracks in a thermally induced stress-strain field. Tests are carried out on type 304 stainless steel to examine the behavior of water reactor pressure vessel cladding and type 316 stainless steel with reference to fast reactor circuit components.

Program 2: Corrosion Fatigue

Since the discovery recently of the acceleration of fatigue crack growth at low frequencies in pressure vessel steels in a reactor water environment, a safety concern has been to know the conditions under which maximum acceleration occurs. The program examines some of the bounds of behavior by tests in simulated reactor water at ambient temperature and pressure. It includes an examination of the role of stress ratio and attempts particularly to correlate behavior with changes in crack growth mechanism.