No. 17741

UNITED STATES OF AMERICA
and
FEDERAL REPUBLIC OF GERMANY

Agreement on research participation and technical exchange between the United States Nuclear Regulatory Commission (USNRC) and the Federal Minister for Research and Technology of the Federal Republic of Germany (BMFT) in the USNRC Power Burst Facility (PBF) and Heavy Section Steel Technology (HSST) research programmes and the BMFT Fuel Behavior and Superheat Steam Reactor (HDR) safety programmes covering a four-year period (with administrative understandings dated 28 April and 6 May 1977 and appendices). Signed on 26 April and 6 May 1977

Authentic text: English.

Registered by the United States of America on 17 April 1979.

AGREEMENT[1] ON RESEARCH PARTICIPATION AND TECHNICAL EXCHANGE BETWEEN THE UNITED STATES NUCLEAR REGULATORY COMMISSION (USNRC) AND THE FEDERAL MINISTER FOR RESEARCH AND TECHNOLOGY OF THE FEDERAL REPUBLIC OF GERMANY (BMFT) IN THE USNRC POWER BURST FACILITY (PBF) AND HEAVY SECTION STEEL TECHNOLOGY (HSST) RESEARCH PROGRAMS AND THE BMFT FUEL BEHAVIOR AND SUPERHEAT STEAM REACTOR (HDR) SAFETY PROGRAMS COVERING A FOUR-YEAR PERIOD

The Contracting Parties,

Considering that the United States Nuclear Regulatory Commission (USNRC) and the Federal Minister for Research and Technology of the Federal Republic of Germany (BMFT):

(a) Have a mutual interest in cooperation in the field of reactor safety research,

(b) Have as a mutual objective improving and thus ensuring the safety of reactors on an international basis,

(c) Have as a mutual objective the achievement of full reciprocity in the exchange of technical information in the field of reactor safety research,

(d) Have entered into a Technical Exchange and Cooperation Arrangement in the Field of Research and Development on Reactor Safety, dated the sixth day of March 1974,[2]

(e) Recognize that their respective countries are member nations of the International Energy Agency which encourages cooperative programs on reactor safety research, and

(f) Have expressed their intention to participate cooperatively in (i) the USNRC-funded Power Burst Facility (PBF) research program at the Idaho National Engineering Laboratory, which is owned by the United States Government and operated under contractual arrangement between the EG&G, Inc., and the U.S. Energy Research and Development Administration (USERDA), (ii) the USNRC-funded Heavy Section Steel Technology (HSST) Program at the Oak Ridge National Laboratory, which operated under contractual arrangement between the Union Carbide Corporation and USERDA, (iii) the Superheat Steam Reactor (HDR) program, and (iv) the Fuel Behavior program, both operated by the Gesellschaft für Kernforschung mbH (GfK) Karlsruhe, under contract to BMFT,

Have agreed as follows:

Article I. PROGRAM COOPERATION

The USNRC and the BMFT, in accordance with the provisions of this Agreement and subject to applicable laws and regulations in force in their respective countries, will join together for cooperative research in the USNRC

[1] Came into force on 6 May 1977 by signature, in accordance with article V (*c*).
[2] United Nations, *Treaty Series*, vol. 1066, No. I-16222.

PBF program (Appendix A), the HSST program (Appendix B), the HDR program (Appendix C), and the Fuel Behavior program (Appendix D).

Article II. SCOPE OF AGREEMENT

A. *Scope of responsibility—USNRC*

1. The USNRC, in consideration of the technical benefits received by its participation in the Fuel Behavior and HDR programs and by its receipt of information under this Agreement, agrees to permit the BMFT to participate in the PBF and HSST programs.

2. Subject to the availability of funds, the USNRC agrees to provide the necessary personnel, materials, equipment, and services for the performance of the PBF and HSST programs, as described in Appendices A and B, or as amended.

3. The USNRC agrees to permit the BMFT to assign one mutually agreed upon technical expert to each of the PBF and HSST programs for participation in the conduct and analysis of program experiments.

4. In addition, the USNRC agrees to permit the BMFT to assign one technical expert as a consultant to each of the PBF and HSST program review groups which periodically review the status of the current programs and of future program plans.

5. The USNRC agrees to grant the BMFT and its assignees access to all experimental data and results of analyses generated by the PBF and HSST programs during the period of this Agreement.

6. The USNRC agrees to provide the BMFT access to operational computer codes developed to analyze experimental data generated by the PBF and HSST programs. Access to proprietary codes and data will not be provided except by written authorization of the owner.

7. The USNRC agrees to bear the total costs of transportation, living expenses and any other costs arising from its participation in the Fuel Behavior and HDR programs, and for the transport and related costs for apparatuses and other equipment furnished by the USNRC.

8. The USNRC agrees to provide the BMFT access to all results obtained from USNRC's analyses of information and experimentation developed for the Fuel Behavior and HDR programs during the period of this Agreement, including nonproprietary computer codes used in such analyses.

B. *Scope of responsibility—BMFT*

1. The BMFT, in consideration of the technical benefits received by its participation in the PBF and HSST programs and by its receipt of information under this Agreement, agrees to permit the USNRC to participate in the Fuel Behavior and HDR programs.

2. Subject to the availability of funds, the BMFT agrees to provide the necessary personnel, materials, equipment, and services for the performance of the HDR and Fuel Behavior programs, as described in Appendices C and D, or as amended.

3. The BMFT agrees to permit the USNRC to assign one mutually agreed upon technical expert to each of the Fuel Behavior and HDR programs for participation in the conduct and analysis of program experiments.

4. In addition, the BMFT agrees to permit the USNRC to assign one technical expert to each of the Fuel Behavior and HDR program review or planning groups which periodically review the status of the current programs and of future program plans.

5. The BMFT agrees to grant the USNRC and its assignees access to all experimental data and results of the analyses generated by the Fuel Behavior and HDR programs during the period of this Agreement.

6. The BMFT agrees to provide the USNRC access to operational computer codes developed to analyze experimental data generated by the Fuel Behavior and HDR programs. Access to proprietary codes and data will not be provided except by written authorization of the owner.

7. The BMFT agrees to bear the total costs of transportation, living expenses and any other costs arising from its participation in the PBF and HSST program under this Agreement, and for the transport and related costs for apparatuses and other equipment furnished by the BMFT.

8. The BMFT agrees to provide the USNRC access to all results obtained from BMFT's analyses of information and experimentation developed for the PBF and HSST programs during the period of this Agreement, including nonproprietary computer codes used in such analyses.

Article III. PATENTS

A. With respect to any invention or discovery made or conceived during the period of, and in the course of or under, this Agreement for BMFT participation in the PBF and HSST programs, the USNRC, on behalf of the United States Government, as the recipient party, and the BMFT, as assigning party, and for USNRC participation in the HDR and Fuel Behavior programs, the BMFT, as the 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 (the 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

(*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 other than by personnel in paragraph 1 above and while in attendance at meetings or when employing information which has been

communicated under this exchange Agreement by one party or its contractors to the other party or its contractors, the party 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[ing] to the other party of a royalty-free, non-exclusive, irrevocable license, with the right to grant sublicenses in and to any such inventions, 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 invention 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 the German Labor Law *(Arbeitnehmererfindergesetz)* of July 25, 1957.

Article IV. EXCHANGE OF SCIENTIFIC INFORMATION AND USE OF RESULTS OF PROGRAM

A. Both parties agree that, pending the grant[ing] by the transmitting party of approval to publish, information developed or transmitted under this Agreement will be freely available to governmental authorities and organizations cooperating with the parties. Such information, except as noted below in paragraphs B and C, may, as required by the administrative procedure in its own country, also be made available to the public by either party through customary channels and in accordance with the normal procedures of the parties.

B. It is recognized by both parties that in the process of exchanging information, or in the process of other cooperation, the parties may provide to each other "industrial property of a proprietary nature". Such property, including trade secrets, inventions, patent information, and know-how, made available hereunder and which bears a restrictive designation shall be respected by the receiving party and shall not be used for commercial purposes or made public without the consent of the transmitting party. Such property is defined as:

(a) Of a type customarily held in confidence by commercial firms;

(b) Not generally known or publicly available from other sources;

(c) Not having been made available previously by the transmitting party or others without an agreement concerning its confidentiality; and

(d) Not already in the possession of the receiving party or its contractors.

C. Recognizing that "industrial property of a proprietary nature", as defined above, may be necessary for the conduct of a specific cooperative project or may be included in an exchange of information, such property shall be used only in the furtherance of nuclear safety programs in the receiving country. Its dissemination will, unless otherwise mutually agreed, be limited as follows:

(a) To persons within or employed by the receiving party and to other concerned government agencies of the receiving party; and

(b) To prime or subcontractors of the receiving party for use only within the country of the receiving party and within the framework of its contract(s) with the respective party engaged in work relating to the subject matter of the information so disseminated; and

(c) On an as-needed, case-by-case basis, to organizations licensed by the receiving party to construct or operate nuclear production or utilization facilities, provided that such information is used only within the terms of the license and in work relating to the subject matter of the information so disseminated; and

(d) To contractors of licensed organizations in subparagraph *(c)* receiving such information, for use only in work within the scope of the license;

provided that the information disseminated to any person under subparagraphs *(b)*, *(c)* and *(d)* above shall be pursuant to an agreement of confidentiality.

D. The application or use of any information exchanged or transferred between the parties under this Agreement 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.

Article V. FINAL PROVISIONS

A. Any dispute between the parties concerning the interpretation or application of this Agreement, which is not settled by negotiation or other agreed mode of settlement, shall be referred to a tribunal of three arbitrators to be chosen by the parties, and who shall also choose the chairman of [the] tribunal. Should the parties fail to agree upon the composition of the tribunal or the selection of the chairman, the President of the International Court of Justice shall, at the request of the parties, exercise those responsibilities. The tribunal shall decide any such dispute by reference to the terms of this Agreement and any applicable laws and regulations, and its decision on all questions of facts shall be final and binding on the parties. Contractors, subcontractors or consultants to the parties hereto shall be regarded as parties to this Agreement for the purpose of this paragraph.

B. This Agreement shall also apply to Land Berlin, provided that the Government of the Federal Republic of Germany has not made a contrary declaration to the Government of the United States within three months from the date of entry into force of this Agreement.

C. This Agreement shall enter into force upon signature of the parties and shall remain in force for a period of 4 years.

D. Either party may withdraw from the present Agreement after providing the other party written notice 6 months prior to its intended date of withdrawal.

E. The USNRC may at its option participate in a continuation of the BMFT Fuel Behavior and HDR programs beyond the 4-year period of this Agreement under mutually acceptable terms and conditions.

F. The BMFT may at its option participate in a continuation of the USNRC PBF and HSST programs beyond the 4-year period of this Agreement under mutually acceptable terms and conditions.

For the United States Nuclear Regulatory Commission:

By: [*Signed—Signé*][1]

Title: Executive Director for Operations

Date: April 26, 1977

For the Federal Minister for Research and Technology of the Federal Republic of Germany:

By: [*Signed—Signé*][2]

Title: Head of Energy Research and Technology Subdivision

Date: May 6, 1977

ADMINISTRATIVE UNDERSTANDINGS BETWEEN THE USNRC AND THE BMFT

An Agreement between the BMFT and USNRC on BMFT participation in the USNRC PBF and HSST programs and on USNRC participation in the BMFT Fuel Behavior and HDR programs within the framework of the US-FRG bilateral arrangement has been negotiated. This Agreement would also be within the framework of an IEA multilateral cooperative agreement for the PBF, HSST, HDR programs when negotiated.

The coordinators for the bilateral technical information exchange arrangement have arrived at the following Administrative Understandings of the details of the BMFT participation in the PBF and HSST programs and of the USNRC participation in the HDR and German Fuel Behavior programs.

1. The Gesellschaft für Kernforschung mbH, Karlsruhe (GfK), is acting as agent for the BMFT in executing this Agreement.

2. Under special circumstances BMFT and USNRC may desire to send one or more technical experts for a short period of time to review or investigate a specific technical problem related to the analysis or experiments of the respective projects. Short term visits by BMFT and USNRC technical experts may be arranged by mutual agreement on a case-by-case basis. The NRC and the BMFT will provide the technical experts making such visits data and documents (excluding proprietary information) concerning the technical problem to the best of their ability within the constraints of available manpower and minimum interference with the program.

3. The Agreement states the categories, data, documents, computer codes, etc., that are to be made available to the USNRC and the BMFT. Other information which may be withheld includes that which deals with organizational, budgetary, personnel or management related matters.

4. BMFT and USNRC will endeavor to select as technical experts for assignment to the program individuals who can contribute positively to the program. BMFT and USNRC technical experts assigned to the program for extended periods will be considered visiting scientists (non-salaried) within the project and will be expected to participate in the conduct of the analysis and experiments of the program as directed.

[1] Signed by Lee V. Gossick — Signé par Lee V. Gossick.
[2] Signed by Manfred Popp — Signé par Manfred Popp.

5. The Agreement states that each party is permitted to assign one technical expert to each of the partner's projects. Both parties agree that one other technical expert may be assigned if so desired.

BMFT and USNRC technical experts will be assigned to mutually acceptable positions within organizational structures of the respective projects.

Both parties recognize the desire of their partner to have one of their technical experts assigned to a position in the organization of each of the partner's projects where they may be able to have an overview of the technical programs. Both parties will endeavor to the best of their ability to fulfill their partner's desire in this regard.

6. Both partners will have access to all reports written by their partner's technical experts assigned to the respective projects which derive from their participation in those projects.

7. Administrative details concerning questions such as security, indemnity and liability related to the assignees will be negotiated and will appear in personnel assignment agreements between USNRC contractors and BMFT contractors.

Both partners will recommend their assigning parties these agreements to be concluded on the basis of a standard arrangement agreed upon by USNRC and BMFT.

For the United States Nuclear Regulatory Commission:

By: [*Signed—Signé*][1]

Title: Director, Office of Nuclear Regulatory Research

Date: April 28, 1977

For the Federal Minister for Research and Technology of the Federal Republic of Germany:

By: [*Signed—Signé*][2]

Title: Head, Nuclear Safety Research Section

Date: May 6, 1977

APPENDIX A

THE POWER BURST FACILITY (PBF) PROGRAM

The facility

The Power Burst Facility is a water-cooled and moderated reactor contained in an open top steel vessel. The PBF is operated for the Energy Research and Development Administration (ERDA) and the Nuclear Regulatory Commission (NRC) by the EG&G Idaho, Inc. (EG&G).

The present reactor core is designed for both steady state operation (to 40 MW) and pulsed mode operation (to 1500 MWsec). A new reactor core interchangeable with the original core should be available sometime after late 1977. The new core is designed for steady state operation for testing large assemblies (clusters) of low enrichment irradiated or unirradiated fuel elements at high power densities.

Table 1 describes the general facility characteristics and compares the test capabilities of the first and second PBF cores.

The PBF currently operates on a two shift basis, but 3- or 4-shift operation during the next few years is probable. At present, reactor tests are scheduled at 7-day to 30-day

[1] Signed by Saul Levine — Signé par Saul Levine.
[2] Signed by Heinz Seipel — Signé par Heinz Seipel.

intervals, with 7 to 16 tests scheduled per 8-month operating year. Four months are allowed each year for reactor certification and maintenance.

The test train

Fuel elements and fuel element assemblies to be tested, one to 25 fuel rods in the first core and one to 64 rods in the second core are fitted into a test train, together with necessary test instrumentation. The assembled test train is then fitted into a heavy-walled vertical pressurizable cylindrical metal tube (the IPT) mounted concentric to the vertical axis of the reactor core and the containing vessel.

The in-pile tube head has six openings, permitting the active use of up to 100 pairs of instrumentation test leads. Typical test instrumentation includes inlet and/or exit flow meters (up to 5 per test), absolute and differential pressure transducers for monitoring fluid and fuel element plenum pressures, surface and internal thermocouples for monitoring fuel, clad, plenum and coolant temperatures, ultrasonic thermometers, linear variable differential transformer (deflection indicators) radiation flux monitor wires and foils and self-powered neutron detectors. Suitable instrumentation, signal conditioning equipment, and data accumulation and reduction equipment and services are available.

The program

The program for the four-year period, June 1975–June 1979, encompasses tests in each of the following areas: *(a)* Power-Cooling Mismatch (PCM), 9 reactor tests (FY76, early FY77); *(b)* irradiation effects, 14 reactor tests (FY76, FY77); *(c)* Loss of Coolant Accident (LOCA), 11 to 18 reactor tests (late FY77, 78, 79); *(d)* inlet flow blockage, 5 reactor tests (late FY77, 78); *(e)* Reactivity Initiated Accident (RIA), 7 to 18 reactor tests (FY77–79); *(f)* gap conductance and PCM parameters, 17 to 23 reactor tests (FY76–79).

This program is subject to continuous review and selective modification as test results are evaluated and further behavior demonstration and model verification needs are identified. The overall PBF test program is based on balanced support of the following Fuel Behavior Branch, RES:RSR, NRC objectives:

1. In-reactor study of fuel properties;
2. In-reactor study of fuel rod and fuel rod assembly properties;
3. In-reactor study of fuel rod and fuel rod assembly behavior under accident conditions;
4. Support of fuel element behavior model development;
5. Support of fuel element behavior model evaluation.

The several PBF test series are described in the Small Cluster Program Requirements Section of the WRSR Fuel Behavior Program Description prepared by the Systems Safety Research Division, EG&G Idaho, Inc. The test series descriptions may be summarized as follows:

(a) *Power-cooling mismatch tests.* These tests will study CHF and post-CHF fuel behavior of single rods (4 at a time) and 9 rod clusters under a variety of power and cooling conditions. Coolant flow, stored energy, and test termination temperatures will be measured.

(b) *Irradiation effects tests.* These tests will study the effects of irradiation and burnup of the thermal-mechanical properties of cladding materials and single fuel rods and the behavior of fuel rods at high power ratings. Post CHF cladding deformation will be one of the dependent test variables measured.

(c) *Loss of coolant tests.* These tests will study fuel rod behavior, e.g., clad deformation and oxidation of multiple rod assemblies, under PWR loss of coolant conditions. Results will be correlated with ex-reactor tests. Parameters to be varied include irradiation history and cold internal pressures. Test loop modifications will provide heatup and blowdown capability late in the 4-year test period.

(d) *Inlet flow blockage tests.* These tests will study fuel rod behavior, e.g., clad temperature profiles of multiple rod assemblies under inlet flow blockage conditions. Blockages of 80% and greater will be investigated. Test loop modifications will be required for these tests.

(e) *Reactivity initiated accident tests.* These tests will study irradiated and unirradiated fuel rod behavior under rod drop and rod ejection conditions. Independent rod tests, cluster tests and model development/evaluation tests will be performed. The effects of irradiation, cluster size, coolant flow, and initial power level will be studied.

(f) *Gap conductance and PCM parameter tests.* These tests will study gap conductance and fuel rod behavior of irradiated and unirradiated rods. Parameters to be varied include irradiation history, gap size, fill gas and pressure and pellet densities. Power oscillation (transfer function technique) and integral k-dt methods will be compared.

TABLE 1. PBF TEST CAPABILITIES

	*Core 1**	*Core 2**
Test space size		
Diameter	15.5 cm	21.6 cm target 15.5 cm minimum
Active length	91 cm	91 cm (nominal)
Test coolant flow rate	0-3000 l/min	0-3000 l/min
Coolant pressure	0.3-15.6 MPa (154 atm, std)	0.3-15.6 MPa (154 atm, std)
Coolant temperature	Ambient, 343° C (650°)	Ambient, 343° C (650° F)
Test power density (max)	*a)* 18 kw/ft in a 16 rod array of highly enriched 17×17 type PWR fuel rods	*a)* 21 kw/ft in a 36 rod array irradiated (to 40,000 MWD/M) 17×17 type PWR fuel rods maximum initial enrichment 3.1 w/o ^{235}U.
	b) 18 kw/ft in a 25 rod array of highly enriched BWR-6 type fuel rods	*b)* 21 kw/ft in a 25 rod array irradiated (to 40,000 MWD/M) BWR-6 type fuel rods with maximum initial enrichment 2.0 w/o ^{235}U.
Test power rate of change		
Steady state	100%/min power increase 15%/sec power decrease	100%/min power increase 15%/sec power decrease
Pulse mode	Periods as short as 1.3 msec, natural burst (to 1500 mw sec sloped burst)	

APPENDIX B

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 investigator 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 (~241MPa)
2. Pneumatic load testing of vessels	Vessel sizes up to ~39 in. (99 cm) O.D. by 54 in. (137 cm) high

Test phase	*Capabilities*
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 IT CT specimens

APPENDIX C

THE HDR SAFETY PROGRAM

The facility

The HDR is a low MW_{th} superheated steam reactor which was designed and built in the 1960s as an experimental facility. After a short period of operation it was shut down and has since been decommissioned. At the present time the containment building and most of the important equipment are intact but the fuel and strongly activated components have been removed from the reactor. The maximum local dose rate at the inside of the pressure vessel is 80 mrem/h.

No nuclear operation is envisaged during the experiments. The test conditions will be obtained with an electrically-heated boiler. Table 1 describes the test program capabilities. The HDR plant is operated for the safety research program by the Gesellschaft für Kernforschung (GfK).

The program

The program for the four-year period, June 1975—June 1979, includes both theoretical studies as well as tests and measurements on full-scale equipment in the HDR plant: reactor building and containment structures, reactor pressure vessel, reactor pressure vessel internals, and piping systems.

The specific objectives of the program are to provide a quantitative assessment of reactor systems and components characteristics under safety design conditions. The test data will serve as a means of verifying existing analytical models and computer program.

The various sectors covered by the program are the following:

Project area non-destructive tests (EV 1,000). This project area has two objectives:

1. Evaluation of defect formation and propagation in primary circuit materials under load conditions (defect analysis);
2. Sensitivity, evidence, and reliability of non-destructive testing methods (evaluation of non-destructive testing system).

Both proven testing methods and methods at the stage of development are used prior and subsequent to the individual load tests. Testing methods include ultrasonic testing (manual and automatic), acoustic emission, acoustic holography, eddy current and potential probe method, radiography, and penetration and magnetic particle testing. Based on these non-destructive testings, defect analysis is made with the objective of evaluating the influence exerted by different load conditions on the formation and propagation of the defects.

The results derived from defect analysis with the measured data of each individual testing method are the basis of systems evaluation of the individual methods. Information on defects obtained by non-destructive testings are compared with the actual characteristic defect data—geometrical dimensions, nature, and location—which is obtained in the next phase of the program after completion of tests from a destructive testing of recognized defects at the pressure vessel and at the piping. This comparison should provide the possibility of judging the capability of detection, the defect description, and the accuracy of location of each testing method. This enables in turn conclusions to be drawn with respect to the reliability and accuracy of methods of measurements for defect recognition and delineates the optimum application of non-destructive testing methods.

Project area pressure vessel and piping investigations (EV 2,000). These investigations are intended to supply a contribution for the evaluation of the safety concepts for light water reactor pressure vessels and pipings. The most important aspect is a quantitative study of the effective safety margin of the components, especially under unfavorable material and overloading conditions. The program is comprised of the following major research areas:

Experimental stress analysis at the pressure vessel and the piping system (primary hot steam system, circulation loop) are effected for operating and maximum load conditions described in Table 1: "pressure test cold and hot", vibration tests, blowdown. From the determined strain or dislocation values, their distribution and time correlation, one may see the nature and amount of loading, especially at the highest stressed points. Depending on the results of this loading—test stress—and failure analysis on the pressure vessel are intended at aggravated conditions (thermal shock, earthquake tests at high intensity, defined weakening). These investigations presently not specified are planned in a project—phase II after the blowdown tests (1979/1980). At the present stage of project these parts are not included.

The material (vessel and pipings) and component (piping) loading capacity is tested in actual conditions and after defined weakening (piping: structural defects, mechanical notches, natural cracks, sensibilization). For a description of the failure under static and time-dependent loading (especially low cycle fatigue) an analysis of the strength and toughness characteristics and the fracture behavior is conducted. In connection with this it has to be clarified how the characteristic values gained from the specimen may be transferred to the components.

Theoretical investigations will be carried out for the experimentally tested components and loadings, which will supplement the design calculation. The results of the theoretical analysis will be compared with the measurements. The goal is the testing of the reliability and applicability of the model concepts usually employed for design and calculation methods.

Project area blowdown investigations (EV 3,000). The HDR-blowdown experiments are planned to provide a detailed understanding concerning the response of reactor pressure vessel internals, containment structures, and safety valves to typical loss-of-coolant conditions. The objectives are testing and further development of various fluid and structural dynamics computer codes.

Research areas foreseen include:

1. Pretest calculation and layout of the blowdown experiments, including first a description of the major factors influencing the dynamic behavior and then the capability to describe "relevant events during blowdown accidents" as:
   — Run in and reflection of a depressurization wave;
   — Subcooled single phase flow;
   — Two-phase flow;
   — Thermal stress due to temperature gradients;
2. Measurement of the various blowdown parameters and their interaction including:
   — Critical nonsteady mass flow rates;
   — Multidimensional pressure wave propagation;

— Forces on reactor pressure vessel internals;
— Deformation of internals;
— Frequency analysis of the internals;

3. Analysis of the blowdown tests loading to:
   — Comparison of computer code results and test data;
   — Test performance and further evaluation;
   — Sensitivity analysis of improved models and codes;
   — Code verification.

The reactor pressure vessel permits an investigation of asymmetrical dynamic loads which supplement the various activities at other places.

Project area earthquake investigations (EV 4,000). The earthquake investigations at HDR involve theoretical studies and measurements on soil, buildings, and containment structures, the reactor pressure vessel and piping system. These investigations are thus seen both as a means of gathering data and as a means of verifying existing analytical models and computer programs.

Calculations were performed first in the linear elastic (low-level) range of response. This task included the preparation of mathematical models using finite element and lumped mass techniques involving current state-of-the-art methods of seismic analysis. The calculation of the responses was carried out by applying forces to the mathematical models which simulated the actual forces applied to the equipment and structures during the tests.

The low-level measurements at the HDR facility were based on excitation of the containment building and equipment with mechanical vibrators, snapback techniques, and explosive charges buried in the soils near the plant.

The results of these measurements and comparisons with theoretical values provide a means for verification of the analytical models and confirmation of the theoretical methods used in the analysis.

In the next phase of the program it is intended to develop non-linear models where necessary so that the analytical methods can be used to predict the response of the structures and equipment to high-level tests.

Project area leak rate investigations (EV 5,000). In the HDR full pressure containment of nearly realistic dimensions (11,000 m^3), the following research areas will be investigated: influence of different parameters on the leakage rate in a cold plant as pressure dependence of the leak rate during and following pressure sequences, sudden change of containment tightness (critical pressure step), influence of air storage in the concrete internals during fast pressure rises, as well as influences of different parameters in a warm plant including influence of temperature laminations in the plant.

TEST PROGRAM CAPABILITIES

Test phase	*Capabilities*
1. Vibration tests at low excitation	Shaker, amplitudes up to 40,000 N, of frequency 0-40 Hz Snapback, amplitudes up to 10,000 N Buried explosives, a few 100 g up to 10 kg of dynamite
2. Loading tests for pressure vessel and piping-system	
— "Pressure test cold" (according to specification)	50-60° C, pressures from ambient to 43 bar Temperatures from 50° C up to 310° C

Test phase	*Capabilities*
— "Pressure test hot" (operating conditions)	Pressures from ambient to 110 bar
3. Leak rate measurements on the full pressure containment	
— Test on the cold plant	Pumping speeds: 0,1 bar/h and 0,5 bar/h Pressure steps: 0,25; 0,5; 0,75; 1,0 up to 5,0 bar
— Test on the hot plant (operating conditions)	Pumping speed: 0,1 bar/h Pressure steps: 0,03 and 0,5 bar
4. Blowdown tests	Operating conditions before the tests
— 4 safety valve tests	Vessel pressure: 70-90 bar Temperature: 285/300° C Diameter of blowdown section: 200 mm
— 13 reactor pressure vessel internals test	Vessel pressure: 110 bar Temp. in the innerspaces of the internals: 250-310° C Temp. stratification between annular and interspaces of the internals: difference 10-40° C Water content: 40-90% of total volume Diameter of blowdown section: 200 mm
— 7 containment tests	Vessel pressure: 70-90, 110 bar Temperature: 285-310° C Diameter of blowdown section: 200; 350; 500 mm
5. Loading-tests on test rigs (carried out on dismounted piping components at the same time as 1-4)	Temperature: 20-310° C Loading: static and variable with time; internal and external overloading up to fracture

APPENDIX D

THE BMFT FUEL BEHAVIOR PROGRAM

The objective

The overall objective of the BMFT Fuel Behavior program is the development of verified analytical models for the response of fuel rods and rod bundles to Loss of Coolant Accidents (LOCA) and Power Cooling Mismatch (PCM) Conditions and the reliable description of failure mechanisms and their feedback to the Emergency Core Cooling System.

The detailed quantitative understanding incorporated in the fuel behavior code must be verified by representative experiments and delivers the basis for further improvements in the design against all types of LOCA and PCM conditions.

The program is carried out by the Projekt Nukleare Sicherheit (PNS) of the Gesellschaft für Kernforschung mbH (GfK) and by the Kraftwerk Union AG (KWU).

The program

The BMFT Fuel Behavior program is divided into the following 4 major areas and 12 specific tasks (see Figure 1):

1. *Material properties and behavior of Zry cladding and UO_2 during LOCA and PCM transients.* The main objective in this area is the determination of a verified "equation of state" of Zry at high temperatures, containing all the parameters ϕ_j which have an essential influence on the plastic strain ε.

The following five tasks are of primary importance in this area:

1.1. *Internal burst test with pressurized Zry-tubes to set up an improved empirical correlation for the bursting and ballooning behavior, respectively.*

1.2. *Investigation of the plastic behavior of Zry cladding during temperature and stress transients as well as environmental conditions of typical LOCA and PCM situations.*

1.3. *Investigation of high temperatures steam oxidation of Zry cladding.* Both the kinetics and extent of oxide formation and penetration as well as the oxygen solubility of the β-phase influence heavily the mechanical properties and therefore the plastic behavior.

1.4. *Investigation of the chemical interaction between oxide fuel and Zry cladding.* Even low fission product concentration in the fuel rod can change significantly the mechanical properties of Zry (ductility and strength at elevated temperatures as a result of stress corrosion cracking). But, in addition, since the fission products bind less oxygen than released in the fission process, the "fictive" O/M ratio of the oxide fuel is increased in irradiation fuel, which implies an increase in the oxygen potential inside the fuel rod. This results in an oxidation layer on the inner side of the Zry cladding in the course of normal operation which is of importance during LOCA and PCM transients.

1.5. *Investigation of the behavior of Zry cladding during simultaneous transient mechanical loading and chemical attack.*

2. *Behavior of fuel rods and rod bundles during LOCA and PCM transients.* The basic philosophy of the experimental program in this area is to obtain a detailed physical understanding of rod failure mechanisms in the different phases of a LOCA. For that purpose extensive out-of-pile experiments are carried out. In these out-of-pile experiments the main parameters influencing the rod behavior will be varied in a systematic way, whereas in specific in-pile tests, which will be carried out in parallel in the Karlsruhe FR2-reactor, the influence of the most important nuclear parameters is investigated.

The following four tasks are of primary interest in this area:

2.1. *Mechanical and thermal behavior of fuel rods during the blowdown phase.* These experiments are carried out with shortened PWR fuel rod simulators, indirectly electrically heated, with Zry cladding, internal pressure and thermal properties very similar to real fuel rods.

The objective is to develop a verified physical model of possible ballooning or burst mechanism resulting from the interaction between strongly changing heat transfer conditions and the mechanical response of the cladding during the blowdown phase.

A further objective is the determination of the initial thermodynamic conditions of fuel rods for the following heatup phase in the course of a LOCA.

Most important for this experiment was the development of means to control the thermohydraulic blowdown conditions.

2.2. *Mechanical and thermal behavior of single fuel rods and rod bundles in the heatup, the refill and reflood region of a LOCA.* The objective of these experiments is to verify physical models for the behavior of ballooning and bursting fuel rods during the heatup process and the interaction with the thermohydraulic conditions of the refill and reflood process. Therefore, representative hydraulic and thermodynamic conditions of the refill and reflood phase of a LOCA must be simulated (i.e., in particular the application of rods and rod bundles at full length):

— Enthalpy distribution in the rods;

— Heating rates;
— Internal gas pressure;
— System pressure;
— Flooding rates;
— Inlet temperature.

The indirectly electrically heated rod simulators have full length (to simulate the thermohydraulic conditions of the reflood process). Single rod tests as well as cluster (25 rods) tests will be carried out.

2.3. *Inpile behavior of PWR fuel rods during the heatup and reflood phase of a LOCA up to high burnups.* The main objectives of the inpile tests in the steam contamination test loop in the Karlsruhe FR2-reactor is to develop a detailed understanding of fuel failure mechanisms depending upon nuclear parameters which cannot adequately simulate in the out-of-pile part of the program. Depending upon the degree of irradiation and burnup, the following phenomena are of main influence on the inpile fuel behavior:

— Internal pressure distribution during LOCA- and PCM-transients;
— Fission products present due to steady state irradiation;
— Additional transient fission gas release;
— Mechanical and thermal behavior of the irradiated fuel and at the true heat generation in the fuel;
— Chemical interaction between fission products and increasing oxygen contents with the inner side of the cladding;
— Mechanical and thermal behavior of irradiated Zry cladding (as an integral parameter of these experiments).

A further incentive for extensive testing is to investigate whether there are failure mechanisms which are not yet known.

The test series were started with unirradiated PWR pins at the end of 1975 and continued with preirradiated pins having different degrees of burnup up to 35.000MWd/t Uran. Preirradiation is performed in the FR2-reactor itself.

The results of the in-pile tests with irradiated rods request the comparison with non-nuclear, i.e., electrically heated rods. Therefore, reference experiments are planned in the same in-pile test loop in which electrically heated fuel rod simulators are used instead of real fuel rods under identical thermal hydraulic conditions.

3. *Effects of ballooning blockages on the reflood process and the efficiency of emergency core cooling systems.* In this area of the BMFT Fuel Behavior program the following two synchronized experiments are carried out and planned, respectively:

3.1. *Separate effects investigation of the influence of shape and size of possible blockage configurations upon reflood rate and distribution and ECCS efficiency.* The experiments are started with 5 fuel rod simulators in a row at full PWR-length followed by cluster tests with 25 rods. The results of these experiments serve as input for the following integral blockage tests; see item 3.2.

3.2. *Reflood process and behavior of parallel coolant channels in a partly blocked 340-rod configuration.* This extensive experiment is in the planning stage. Results of the above separate effects blockage program must be available to optimize the experimental approach.

4. *Development of the fuel behavior code SSYST.* The development of SSYST is performed in three steps:

a) Modeling the single rod behavior;

b) Modeling the behavior of rod bundles including possible failure propagation phenomena;

c) Development of analytical models to evaluate blockage effects.

The BMFT Fuel Behavior program and the several test series are described in (1), (2), (3).

This program is subject to continuous review and selective modification as test results are evaluated and further behavior demonstration and model verification needs are identified.

(1) *PNS-Arbeitsbericht Nr. 34/74, Sept. 1974. Statusbericht unter die theoretischen und experimentellen Untersuchungen des PNS zum Brennstabverhalten.*

(2) *PNS-Memorandum Nr. 62/75, May 1975.* PNS-4230, Program Description, presented at the first American-German Experts meeting on "Fuel Behavior Programs" (especially PBF-program), May 18-30, 1975, in USA.

(3) *RS 107: Verhalten von Zry-Hüllrohren unter den bei Kühlmittelverluststörfallen auftretenden Beanspruchungen, IRS-Vierteljahresberichte.*

FIGURE 1. THEORETICAL AND EXPERIMENTAL INVESTIGATIONS ON LWR FUEL ROD BEHAVIOR IN THE FRG

Exp. investigations of fuel rod behavior	*Exp. investigations of material behavior*
Out-of-pile with electr. heated fuel rod simulators (Zry-cladding, inner pressure); in-pile with PWR-typical fuel rods	LWR-typical Zircaloy material (tube and sheet specimen)
I. Blowdown-phase	
Mechanical-thermal behavior of LWR-fuel rods (PNS 4236) out-of-pile, single rods	Burst tests with pressurized Zry-tubes (KWU-Erl., RS 107)
	Mechanical behavior of Zry-4-cladding; empirical material law (PNS 4235.1)
II. Refill and reflood phase Ballooning experiments with reactor typical cooling conditions (PNS 4238), out-of-pile single rods and bundles (25 rods)	High temperature steam oxidation of Zry (PNS 4235.2)
In-pile-experiments in the DK-loop of the FR-2 reactor (PNS 4237.1), single rods, adiabatic heatup phase	Interaction fuel/Zry-cladding (PNS 4235.3)
Reference experiments to 4237.1 with el. simulators	Behavior of Zry clads during combined mechanical/chemical load (in PNS 4235.1-3)
Effects of blockages (PNS 4239), rod row (5 rods) bundle (25 rods)	
Effects of blockages, partially blocked 340-rod-bundle (KWU-Erl.)	

Theory (models, modeling laws, code-development)

I. System analysis codes (thermal-hydraulic codes): LRA, IRS, KWU	II. Fuel rod behavior code: GfK/PNS and IKE/Stuttgart (PNS 4231)