No. 19461

UNITED STATES OF AMERICA, DENMARK, FINLAND and SWEDEN

Agreement on research participation and technical exchange between the United States Nuclear Regulatory Commission (USNRC) and the Nordic Group in the USNRC Power Burst Facility and Heavy Section Steel Technology Research Programs and the Nordic Group's Water Reactor Safety Research Programs covering a three-year period (with appendices). Concluded on 26 June 1979

Authentic text: English.

Registered by the United States of America on 9 December 1980.

ÉTATS-UNIS D'AMÉRIQUE, DANEMARK, FINLANDE et SUÈDE

Accord de participation aux recherches et d'échanges techniques conclu pour une durée de trois ans entre la United States Nuclear Regulatory Commission (USNRC) et le Groupe nordique dans le cadre des programmes de recherche de l'USNRC sur le dispositif de bouffée de puissance et sur la technologie des aciers à profil épais et des programmes de recherche sur la sûreté des réacteurs à eau du Groupe nordique (avec appendices). Conclu le 26 juin 1979

Texte authentique : anglais.

Enregistré par les États-Unis d'Amérique le 9 décembre 1980.

AGREEMENT[1] ON RESEARCH PARTICIPATION AND TECHNICAL EXCHANGE BETWEEN THE UNITED STATES NUCLEAR REGULATORY COMMISSION (USNRC) AND THE NORDIC GROUP (FORSOGSANLAEG RISO, DENMARK; VALTION TEKNILLINEN TUTKIMUSKESKUS, FINLAND; AND STUDSVIK ENERGITEKNIK AB, SWEDEN) IN THE USNRC POWER BURST FACILITY (PBF) AND HEAVY SECTION STEEL TECHNOLOGY (HSST) RESEARCH PROGRAMS AND THE NORDIC GROUP'S WATER REACTOR SAFETY RESEARCH PROGRAMS COVERING A THREE-YEAR PERIOD

The Contracting Parties,

Considering that the United States Nuclear Regulatory Commission (USNRC) and the Nordic Group

(*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*) Recognize that they are participants in the cooperative programs on reactor safety research of the International Energy Agency (IEA), as defined in the article IV of the Guiding Principles for Cooperation in the field of Energy Research and Development, agreed upon by the IEA Governing Board;

(*e*) Have an interest in applying the rights of the participants with respect to intellectual property consistent with article VI of the Guiding Principles for Cooperation in the Field of Energy Research and Development; and

(*f*) Have expressed their intention to participate cooperatively in the USNRC-funded Power Burst Facility (PBF) research program at the Idaho National Engineering Laboratory and Heavy Section Steel Technology (HSST) research program at the Oak Ridge National Laboratory, and in the Nordic Group's Water Reactor Safety Research Programs performed in the Nordic countries,

Have agreed as follows:

Article I. PROGRAM COOPERATION

The USNRC and the Nordic Group, 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 PBF program as described in appendix A (1), in the USNRC HSST program as described in appendix A (2), and in the Nordic Group's reactor safety research programs, including the fuel related programs described in appendix B (1), the primary pressure boundary

[1] Came into force on 28 August 1979 upon signature by all the Parties, in accordance with article V (A). The signatures were affixed as follows:

State	*Date of signature*	*State*	*Date of signature*
Denmark	28 August 1979	Sweden	13 August 1979
Finland	22 August 1979	United States of America	26 June 1979

safety programs described in appendix B (2), and the miscellaneous programs described in appendix B (3).

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 Nordic Group's water reactor safety research programs and receipt of information under this Agreement, agrees to permit the Nordic Group to participate in the PBF and HSST programs.

2. The USNRC agrees to provide the necessary personnel, materials, equipment and services for the performance of the PBF and HSST programs described in appendices A (1) and A (2), or as amended, subject to the availability of funds.

3. The USNRC agrees to permit the Nordic Group 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 Nordic Group to assign one technical expert as a consultant to each of the PBF and HSST program review groups which will periodically review the status of the present program and of future program plans.

5. The USNRC agrees to grant the Nordic Group 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 Nordic Group access to operational computer codes and data 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 programs included under this Agreement, and the transport and related costs for apparatuses and other equipment furnished by the USNRC.

8. The USNRC agrees to provide the Nordic Group access to all results obtained from USNRC's analyses of information and experimentation developed for the Nordic water reactor safety research programs during the period of this Agreement, including nonproprietary computer codes used in such analyses.

B. *Scope of responsibility — Nordic Group*

1. The Nordic Group in consideration of the technical benefits received by its participation in the PBF and the HSST programs and receipt of information under this Agreement agrees to permit the USNRC to participate in the Nordic water reactor safety research programs.

2. The Nordic Group agrees to provide the necessary personnel, materials, equipment and services for the performance of the Nordic water reactor safety research programs described in appendices B (1), B (2) and B (3), or as amended, subject to the availability of funds.

3. The Nordic Group agrees to permit the USNRC to assign two mutually agreed upon technical experts to research projects within the Nordic water reactor safety research programs for participation in the conduct and analysis of program experiments.

4. In addition, the Nordic Group agrees to permit the USNRC to assign a technical expert as a consultant in regard to the development of the present and future Nordic Group's water reactor safety research program under this Agreement.

5. The Nordic Group agrees to grant the USNRC and its assignees access to all experimental data and results of analysis generated by the Nordic water reactor safety research programs during the period of this Agreement.

6. The Nordic Group agrees to provide the USNRC access to operational computer codes and data developed to analyze experimental data generated by the Nordic water reactor safety research programs. Access to proprietary codes and data will not be provided except by written authorization of the owner.

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

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

Article III. 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 Nordic Group participation in the PBF and HSST programs, the USNRC on behalf of the United States Government, as the recipient party, and the Nordic Group as assigning party, and for USNRC participation in the Nordic water reactor safety research programs, the Nordic Group 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 nonexclusive, 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 nonexclusive, 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 to the other party of a royalty-free, nonexclusive, 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 will assume the responsibility to pay awards or compensation required to be paid to its nationals according to the laws of its country.

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

A. Both parties agree that, pending the grant 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. This Agreement shall enter into force upon signature by appropriate representatives of the USNRC and of the three parties comprising the Nordic Group as listed below, and shall remain in force for a period of 3 years.

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

C. 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.

D. The USNRC may at its option participate in a continuation of the Nordic water reactor safety research programs beyond the 3-year period of this Agreement under mutually acceptable terms and conditions.

E. The Nordic Group may at its option participate in a continuation of the USNRC PBF and HSST programs beyond the 3-year period of this Agreement under mutually acceptable terms and conditions.

F. The USNRC and the Nordic Group shall each appoint within 30 days after the signing of this Agreement a coordinator to coordinate and implement the provisions of this Agreement.

G. If the PBF and/or HSST research programs are substantially reduced or eliminated, equitable work, determined by the USNRC and the Nordic Group to be of equivalent programmatic interest, will be substituted as may be mutually agreed.

H. If the Nordic water reactor safety research programs are substantially reduced or eliminated, equitable work, determined by the USNRC and the Nordic Group to be of equivalent programmatic interest, will be substituted as may be mutually agreed.

I. The Institutt for Atomenergi, Norway, might in the future find it compatible to its activities and interest to join the Nordic Group for the execution of this Agreement. The USNRC and the present parties comprising the Nordic Group therefore hereby state that the Institutt for Atomenergi is entitled to subscribe to the Agreement which otherwise remains unchanged.

J. A copy of this Agreement shall be deposited with the Executive Director of the International Energy Agency, in recognition of the Agency's interest in international cooperation in energy research and development.

For the United States Nuclear Regulatory Commission:

By: [*Signed — Signé*][1]
Title: Executive Director for Operations
Date: June 26, 1979

For the Nordic Group:

Forsøgsanlaeg Risø, Denmark:

By: [*Signed — Signé*][2]
Title: Director
Date: August 28, 1979

Valtion Teknillinen Tutkimuskeskus, Finland:

By: [*Signed — Signé*][3]
Title: Director General
Date: 1979.08.22

By: [*Signed — Signé*][4]
Title: Research Director

Studsvik Energiteknik AB Sweden:

By: [*Signed — Signé*][5]
Title: President
Date: 1979-08-13

APPENDIX A (1)

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

[1] Signed by Lee V. Gossick — Signé par Lee V. Gossick.
[2] Signed by Niels Holm — Signé par Niels Holm.
[3] Signed by Pekka Jauho — Signé par Pekka Jauho.
[4] Signed by Veikko Palva — Signé par Veikko Palva.
[5] Signed by Kjell Hakansson — Signé par Kjell Hakansson.

The test train

Fuel elements and fuel element assemblies to be tested, 1 to 25 fuel rods in the first core and 1 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 tubehead 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 five 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 4-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 (four at a time) and nine 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 °F)	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 mWsec sloped burst)	

APPENDIX A (2)

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 (~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 (1)

THE NORDIC GROUP'S FUEL RELATED SAFETY RESEARCH PROGRAMS

111. *Fuel performance and reliability analysis (Denmark)*

A three-dimensional fuel performance computer code, WAFER, simulates the elastic/plastic behaviour of a segment of a fuel pin throughout its power history, including calculation of local effects such as clad ridging. The code capability does not include severe accident conditions.

A new computer code for fuel reliability prediction, FRP, calculates the statistical distribution of stress, strain, temperatures, etc., in a segment of a fuel pin. Future work will concentrate on the definition of failure conditions, and verification of the code including the collection of necessary statistical data.

112. *Experience with UO_2-Zr performance in overpower tests (Denmark)*

The Danish UO_2-Zr programme comprises irradiation in the materials testing reactor DR-3 at Risø of well-characterized test fuel pins. An important part of these experiments are overpower tests, where BWR and PWR type fuel pins are irradiated to medium-to-high burnups, non-destructively characterized in the Risø hot cells and then submitted to controlled power increases beyond the previous heat load level.

Results from such tests, including pre-irradiation, pre-ramp and post-ramp measurements, could be made available for computer code verification.

113. *Provision of high-burnup UO_2-Zr test fuel pins (Denmark)*

The Danish UO_2-Zr programme comprises irradiation in the Halden Reactor (Norway) of a number of test fuel assemblies with well-characterized BWR-type fuel pins. Five of the assemblies are still in the reactor, with burnups in the range 21.000 - 39.000 MWD/te UO_2 (avg. assembly). The following material and design variables are being tested: pellet length and surface, clad alloy, clad thickness and surface treatment, fuel-clad gap and pin length. Fuel pins from one or more of these assemblies could be made available for experiments in the PBF.

121. *Probabilistic methods for predicting core-wide fuel damage and fission product release during an LWR LOCA (Finland)*

Several computer models have been developed for the estimation by probabilistic techniques the number of fuel rods puncturing in an LWR core during a LOCA. The latest fuel failure model MCRF will be integrated with the multi-purpose fission product release code ACCREL, which calculates the prior-to-accident conditions of the fuel and fission product release from the reactor core during the accident. Future efforts will include refining the physical submodels of the code, collecting relevant statistical input data, sensitivity studies, example calculations for different reactor types and comparisons with relevant experiments.

131. *Fission product volatility in intact fuel rods (Sweden)*

Fuel pins previously irradiated and cooled are reirradiated for a few days in the R2 reactor in order that interesting species of short half-life (Te, I, Ba, etc.) are present. After irradiation the fuel pins are rapidly transported to the hot cells.

In one type of experiment axial and radial distribution of fission products is measured using Ge(Li) detector γ-spectrometry complemented by other techniques such as metallography.

In the other type of experiment the fuel pins are heated to temperatures of interest in LOCA analyses (800-1200 °C) and the arrival of volatile fission products at the cooler plenum is monitored.

132. *Fission gas release from high burnup UO_2 (Sweden)*

A technique is being developed for determination of residual fission gases in high burnup fuel. Later on this technique is to be applied to the determination of residual gas in available commercial fuel.

133. *Fission product release (Sweden)*

The project consists of two parts:

1. Follow up of fission product release data from commercial power reactors.
2. Measurement of fission product release immediately following power ramping to failure in a test loop of the Studsvik R2 reactor.

134. *Cladding behaviour in a BWR following a loss-of-coolant accident (Sweden)*

Experiments have been carried out to investigate cladding response during a relatively slow temperature transient in steam. The loading was by internal pressure. Creep of the tubes was determined. Further areas to be studied are *a*) creep of clad preoxidized to about 5, 20 and 50 µm ZrO_2, *b*) behaviour of simulated fission products during a BWR-LOCA temperature transient.

135. *Fuel rod modelling (Sweden)*

The work includes assessment and development of computer codes for fuel element behaviour and improvement of models for the various processes involved. Codes currently used are GAPCON-THERMAL-2, MOXY-EM and TOODEE-2. Development work is presently mainly on a code for fission product migration.

136. *Fission product decay heat data/Radiometric methods (Sweden)*

Spectroscopic measurements are being performed with the aim of studying the energy distribution and the total energy of beta and gamma radiation emitted 10-1500 s after thermal fission of ^{235}U.The total energy of γ emission is being determined within ± 7%. The work on β emission for U^{235} fission is completed. Measurements on Pu^{239} start shortly.

137. *Fission product release during transients (Sweden)*

A project has started for gamma spectrometric measurements of the time-dependence and composition of fission products released during transients in connection with reactor shut-down in the Oskarshamn reactor.

APPENDIX B (2)

The Nordic Group's Primary Pressure Boundary Safety Research Programs

201. *Influence of simulated service loading and temperatures on transitional and upper shelf fracture toughness of pressure vessel steel (possibly a joint Nordic program)*

This program is to be initiated with the objectives to find the influence of service load and temperature (exclusive of irradiation) simulation of materials properties. The program would contain service simulation treatment, analysis of microstructures, mechanical testing, fracturing mechanism research and dynamic fracture toughness testing.

211. *Structural reliability (Denmark)*

The scope of the work is to develop methods for evaluating the reliability of steel components (i.e. the pressure vessel) in nuclear reactors. Computer codes utilizing analytical as well as Monte Carlo methods for the calculation of the failure probability for the steel pressure vessel have been developed. Future work will concentrate on statistical models for time-dependent phenomena like crack growth, neutron embrittlement, inservice inspection, etc.

212. *Axial cracks in pipes (Denmark)*

Shallow shell equations including the effect of transverse shear show considerable differences from the classical theory with respect to the bending stress field around crack tips. Computer programs using the refined theory have been developed for an axial through crack and an exterior surface crack embedded in a Dugdale-type plastic zone. Further work would include correlation with experimental results.

221. *Effects of carbon level and heat treatment parameters on the microstructure and mechanical properties of pressure vessel steels (Finland)*

The aim of this research is to gain information on how low and medium level carbon content and varying heat treatment parameters affect the mechanical properties and microstructures of A533B and A508 cl. 2 steels. The evaluation of the microstructural parameters, which govern the mechanical properties, i.e., yield strength and crack initiation as well as propagation resistance, is carried out.

231. *Fracture toughness of pressure vessel steel (Sweden)*

The fracture toughness, J_{IC}, is being measured for A533B plate, weld and HAZ material. The program covers temperatures from room temperature to 350°C and wide range of strain rates.

232. *Residual stresses in weldments (Sweden)*

Using an X-ray diffraction camera technique and computer analysis, the residual stress field within "full thickness" weldments of A533B pressure vessel steel is determined. The influence of thermo and mechanical post weld treatment as well as repair welding on the resulting stress field is being studied.

APPENDIX B (3)

THE NORDIC GROUP'S MISCELLANEOUS WATER REACTOR SAFETY RESEARCH PROGRAMS

311. *Overload behaviour of concrete structures under 3-axial stresses (Denmark)*

8 small-scale model experiments, representing different PCRV-closure designs, have been executed in order to investigate overload behaviour, failure mode and ultimate load. The applicability of a finite element programme (P-479) has been extended by introduction of a suitable representation of inelastic material properties and failure criteria, and the program has been verified by the model tests.

312. *System reliability (Denmark)*

Methods for the evaluation of the reliability of systems are being developed as a means of improving the design and safety analyses of complex nuclear and conventional systems. Computer codes - primarily of the Monte Carlo type - were developed for this purpose. Future work will concentrate on further refinement of these techniques in order to obtain greater flexibility in the modelling, closer control of the models, and higher computational speed. The techniques will be tested on complex industrial systems.

313. *Doses from releases of radioactive material to the atmosphere (Denmark)*

A group of computer codes for calculation of downwind radiation doses from releases of radioactive isotopes have been developed. The codes are still being improved according to the state of the art.

Contamination levels and external gamma doses as well as internal doses due to inhalation of radioactive material are calculated. Release of 58 different isotopes can be treated. The Gaussian plume model is used taking into account limitations of vertical dispersion and horizontal dispersion. Fallout, wash-out, variation of wind-direction, radioactive decay and daughter products are taken into account.

314. *Study of the relation between different stability indices used in the determination of atmospheric dispersion conditions (Denmark)*

Two problems in connection with site evaluation for nuclear power plants are studied:

- Use of the meteorological statistics from one site as being representative for another site.
- Use of the relation between dispersion parameters and the temperature gradient recommended in Safety Guide 23.

The problems are studied by means of simultaneously measured micro meteorological data on more than ten 20 to 120 m high masts.

321. *Assessment of environmental impact of reactor accidents and normal operation (Finland)*

A comprehensive computer model ARANO has been developed and used extensively for the assessment of risks due to the possible health effects and economic damages caused by reactor accidents and normal operation. In the latest version of this code, the assessment has been made more realistic by changing the calculation of vertical concentration distribution to use [a]diffusion model instead of the earlier assumption of Gaussian distribution. In addition, the changes in meteorological conditions during the release or dispersion period are now considered with increased accuracy to predict the consequences and probabilities of the most severe accident situations as realistically as possible. A current specific application is the probabilistic risk/benefit evaluation of different possible sites for nuclear power plants producing both electricity and urban heat.

322. *Structural analysis pipe breaks (Finland)*

Both from safety and economical points of view, it is important to reach adequate accuracy in analyzing consequences of postulated pipe breaks. For this purpose a computer program PIPEBREAK has been developed. The analytical capabilities of the code are supplemented with experimental results concerning the complicated behaviour of the pipe wall at collision restraints. The pressure and mass flow transient in the piping system are calculated separately by suitable thermohydraulic codes.

323. *Reliability calculation methods and computer programs (Finland)*

Several well-known Monte Carlo simulation programs, in more or less modified form, are routinely used for numerical solution of fault trees. In the case of high reliability systems, however, the simulation method is ineffective and analytical methods have growing importance.

Hence, an integrated code package based on analytical method is under development. It will include the following parts: MOCUS algorithm to find minimal cut sets, IMPORTANCE code for the investigation of different important measures, and KITT code for the analytical calculation of reliability and availability.

324. *PWR ECC heat transfer experiments (Finland)*

An experimental program including hundreds of reflood tests with a single-pin forced feed facility is under way. Selected experiments are filmed to allow detailed examination of the rewetting phenomenon. [The] following parameters have been studied: rod power and axial power distribution, rod length and diameter, reflooding rate and, in particular, the amount of N_2 dissolved in the injected water.

Experiments with a multirod array are being planned. These tests would address pool boiling heat transfer, alternate ECCS and certain design-dependent special effects.

331. *Parallel channel transient CHF experiments (Sweden)*

The main objective of the investigation was to measure the time to onset of dryout for a large variety of LOCA transients. Specially the influence of possible superimposed flow instabilities caused by the interaction of parallel channels was to be investigated. The post dryout heat transfer regime and the onset of Emergency Core Cooling Systems (ECCS) were beyond the scope of this study.

332. *Indication of steam leakages in reactor and turbine containments (Sweden)*

The aim of the work is to make functional tests on switches of those types used for the indication of steam leakage in nuclear power stations.

The experiments simulate steampipe rupture conditions, and the switches monitor temperature, pressure derivative and water level in draining gutters. An extensive monitoring of environmental parameters also allows for a comparison with the results obtained by means of computer models and an improvement of these models.