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SAFETY RESEARCH NEEDS FOR RUSSIAN-DESIGNED REACTORS

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NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

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Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996) and the Republic of Korea (12th December 1996). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of all OECD Member countries, except New Zealand and Poland. The Commission of the European Communities takes part in the work of the Agency.

The primary objective of the NEA is to promote co-operation among the governments of its participating countries in furthering the development of nuclear power as a safe, environmentally acceptable and economic energy source.

This is achieved by:

- *encouraging harmonization of national regulatory policies and practices, with particular reference to the safety of nuclear installations, protection of man against ionising radiation and preservation of the environment, radioactive waste management, and nuclear third party liability and insurance;*
- *assessing the contribution of nuclear power to the overall energy supply by keeping under review the technical and economic aspects of nuclear power growth and forecasting demand and supply for the different phases of the nuclear fuel cycle;*
- *developing exchanges of scientific and technical information particularly through participation in common services;*
- *setting up international research and development programmes and joint undertakings.*

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.

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RECHERCHES SUR LA SÛRETÉ DES RÉACTEURS DE CONCEPTION RUSSE

ÉTAT DES BESOINS

FOREWORD

There are currently eight countries operating 60 Russian-designed nuclear power reactors with 12 in various stages of construction. There are three principal reactor types: the 440 MW VVERs (Pressurised Light-Water Reactors) including the older 230 models and the newer 213 models, the 1 000 MW VVERs and the RBMKs (fuel channel, graphite-moderated reactors). Significantly different safety features are found not only among the various reactor types, but also within the various generations of each type.

Following the break-up of the Soviet Union in 1991, major national and international organisations initiated programmes to assess and improve the safety of nuclear power plants in countries operating Russian-designed reactors. Included were bilateral and multilateral assistance programmes, co-ordinated by the G-24, among them being: the programmes of the European Commission (PHARE, TACIS), the efforts of the International Atomic Energy Agency (IAEA), the nuclear safety account of the European Bank for Reconstruction and Development, and, of course, the Co-operation and Assistance Programme of the OECD Nuclear Energy Agency (OECD/NEA).

The NEA's Steering Committee has endorsed a broad-based programme of co-operation and assistance to both the Central and Eastern European Countries (CEEC) and the New Independent States (NIS) of the former Soviet Union in planning and executing safety research programmes with a view to building up know-how and capabilities in safety technology pertaining to their nuclear power plants. The OECD Nuclear Energy Agency is carrying out this programme of co-operation under the auspices of the OECD Centre for Co-operation with Economies in Transition (CCET).

This programme is based on the traditional areas of strength of the OECD/NEA, namely nuclear safety research and regulation, and is intended to contribute to and improve the CEEC/NIS nuclear safety culture by concentrating on long-term objectives, as a complement to the near-term technical upgrades to the highest risk plants and improvements of operational safety. As part of its continuing effort, the OECD/NEA established, in 1995, an OECD Support Group on the Safety Research Needs for Russian-Designed Reactors consisting of senior Russian and Western experts, with the specific aim of identifying the safety research needs for Russian-designed reactors. This report presents their findings.

This report is published on the responsibility of the Secretary-General. The views expressed do not necessarily correspond to those of the national authorities concerned.

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EXECUTIVE SUMMARY

In June 1995, an OECD Support Group was set up to perform a broad study of the safety research needs of Russian-designed reactors. This Support Group was endorsed by the CSNI.

The Support Group, which is composed of senior experts on safety research from several OECD countries and from Russia, prepared this Report. The Group reviewed the safety research performed to support Russian-designed reactors and set down its views on future needs. The Support Group, under the chairmanship of Dr. Eric S. Beckjord, met on three occasions: in Paris, July 1995; in Moscow, May 1996; and in Paris, July 1996.

The Support Group reviewed the report of the Senior Group of Experts on Reactor Safety Research (SESAR), approved by the CSNI in 1993, and used it as a starting point for this Report. The scope of SESAR was safety research in OECD countries. Many of the issues identified in SESAR also apply to Russian-designed reactors.

At its first meeting, the Support Group decided on a structure for its review, and formed the following seven Task Groups:

- Thermal-Hydraulics/Plant Transients for VVERs.
- Integrity of Equipment and Structures for VVERs.
- Severe Accidents for VVERs.
- Operational Safety Issues.
- Thermal-Hydraulics/Plant Transients for RBMKs.
- Integrity of Equipment and Structures for RBMKs.
- Severe Accidents for RBMKs.

Because of similarities between Western LWRs and VVERs, safety research in OECD countries applies to VVERs to a considerable extent, as identified in the report. Some important elements apply as well to RBMKs.

Within the scope of this report, the intention is to identify the research needs in a general way: the detailed description of a work programme is part of the definition of future projects.

The emphasis of the study is on the VVER-type reactors in part because of the larger base of knowledge within the NEA Member countries related to LWRs. For the RBMKs, the study does not make the judgement that such reactors can be brought to acceptable levels of safety but focuses on near term efforts that can contribute to reducing the risk to the public. The need for the safety research must be evaluated in context of the lifetime of the reactors.

The principal outcome of the work of the Support Group is the identification of a number of research topics which the members believe should receive priority attention over the next several years if risk levels are to be reduced and public safety enhanced. These appear in the Conclusions and Recommendations section of the report, and are the following.

General Conclusions

- The most important near-term need for VVER and RBMK safety research is to establish a sound technical basis for the emergency operating procedures used by the plant staff to prevent or halt the progression of accidents (i.e., Accident Management) and for plant safety improvements.

- Co-operation of Western and Eastern experts should help to avoid East-West know-how gaps in the future, as safety technology continues to improve.

- Safety research in Eastern countries will make an important contribution to public safety as it has in OECD countries.

- RBMK safety research, including verification of codes, starts from a smaller base of experience than VVER, and is at an earlier stage of development.

Technical Conclusions

- Research to improve human performance and operational safety of VVER and RBMK plants is extremely important.

- VVER thermal-hydraulic and reactor physics research should focus on full validation of codes to VVER-specific features, and on extension of experimental data base.

- Methods of assessing VVER pressure boundary integrity must be verified, and material property data bases extended.

- VVER severe accident research should focus on validation of codes for accident management procedures, and on extension and qualification of an appropriate data base for materials properties and their interactions.

- RBMK thermal-hydraulic research is needed to improve the technical basis for further development of RBMK safety criteria.

- Assessment of the integrity of the RBMK primary coolant circuit, and especially the fuel channel, requires urgent research. Methods of assessing RBMK pressure boundary integrity must be verified, and material property data bases extended.

- RBMK severe accident research should focus on prevention of accidents and Accident Management for cases of loss of heat sink and Beyond Design-Basis Loss-of-Coolant Accidents. For these purposes, simple physical models and parametric codes need development and should be systematically used in plant specific analysis.

Recommendations

- A Safety Research Strategic Plan should be developed. Such a plan sets goals, defines products, and describes when and how work will be done, including determination of research priorities.

- Key players, including regulators, operators, plant designers and researchers should be involved in developing and implementing this plan and its execution and applying the results.

- International co-operation in safety research should be encouraged for purposes of improving quality, preventing technical isolation and cost sharing.

- New approaches, such as technical fora for specific technical topics, should be established to make safety research information in OECD countries available to researchers working on the safety of Russian-designed reactors.

1. INTRODUCTION

Report Background

This is the Report of the OECD Support Group on Safety Research Needs for Russian-Designed Reactors. The Report is the result of the decision of the Committee on the Safety of Nuclear Installations (CSNI) to perform a detailed study of the safety research needs for VVER and RBMK reactors. The CSNI made this decision in support of the broad-based NEA policy of assisting both the Central and Eastern European Countries (CEEC) and the Newly Independent States (NIS) of the former Soviet Union in planning and executing safety research programmes with a view to building up know-how and capabilities in safety technology pertaining to their nuclear power plants. The Support Group (Annex 1), which is comprised of experts from OECD[1] countries and from Russia, combines the experience of the participating OECD countries with the Russian knowledge of their technology and plants. The findings, therefore, have a broad base in nuclear safety and safety research. The Support Group, under the chairmanship of Dr. Eric S. Beckjord, met on three occasions: in Paris, July 1995; in Moscow, May 1996; and in Paris, July 1996.

The Report is the third study of safety research sponsored by CSNI. In 1985, CSNI reviewed safety research programmes underway in its member countries. Subsequently, as a result of changes in the safety research environment and increasing need for international co-operation, it established the Senior Group of Experts on Reactor Safety Research (SESAR) to review safety research within OECD countries and to set down their views on likely safety research needs and priorities.[1] SESAR focused on safety research within OECD countries, and did not consider programmes of countries outside OECD. Nevertheless, because of the pertinence of many of its findings to the new study, SESAR is an excellent point of departure for the this Report.

When the OECD Support Group undertook this study, the members were aware of the concurrent EU, CEEC and CIS study [2,3] aimed at defining important safety research projects for Russian-designed reactors to be funded by the European Commission. This study focuses on research needs rather than specific projects. Although the two studies have a different focus, the aim, from the beginning, was to produce a complementary Report that avoided duplication.

[1] To avoid confusion in this Report of "OECD", "Western", and "Eastern" countries, we define the following: "Western" countries means the same as "OECD" countries. The new "Eastern" countries of the OECD are not included since they were not members when this study was initially undertaken. None of the "Western" or "OECD" countries have VVER and/or RBMK reactors except Finland. All "Eastern" countries, dealt with in this report, have VVER and/or RBMK reactors.

Report Objective

The objectives of the OECD Support Group are:

- To carry out a broad study of the safety research needs for Russian-designed nuclear power plants of the VVER and RBMK types.

- To identify the safety issues where additional experimental and analytical research efforts are needed, having given adequate consideration to the applicability of safety research data available in OECD countries to limit the additional experimental needs.

- To document the findings in an appropriate report.

- To make the information available, as a reference document for future activities, to governments, appropriate funding agencies and research institutions.

The aim is to develop practical conclusions and recommendations regarding high priority safety research that will assist decision makers and managers of research funding in planning, initiating, and carrying out programmes that can improve the safety of these reactor types.

This Report will help to provide justification for research proposals that respond effectively to its findings, but in addition, endorsement of the specific research projects by the users of research is extremely important. The users of safety research are regulatory authorities, plant owners and reactor designers. Involvement of users of research in planning research projects helps to assure that successful research will be applied effectively and helps, thereby, to obtain funding.

Clearly a long term solution to the job of enhancing and maintaining nuclear safety in Eastern countries requires a process carried out by the Eastern countries themselves. International co-operation will help achieve the goal, but it cannot do so by itself indefinitely. The central actors must be the authorities in the Eastern countries who have responsibilities for nuclear safety, in plant design, construction, operation, in safety research, and in regulation. We expect organisations in Eastern countries to take the next step, and that is to respond to the research recommendations by preparing specific research proposals addressing the safety issues.

We emphasise the point that this is a report on safety research needs. The Support Group has not studied the safety of individual nuclear plants, and has made no judgement on the safety of operating Russian-designed reactors.

Report Process

The impetus for the study is the recognition that, within the context of the lifetime of these reactors, short term efforts to address most obvious weaknesses in the Russian-designed reactors should gradually give way to longer term actions with the same objectives, based on deeper understanding of the safety issues through safety research. Safety research has been the source of many improvements to nuclear plants in OECD countries, through application to plant technical and operational changes, and through training of researchers who can work on new safety issues that arise. It is reasonable to expect that the same process will occur in Eastern countries.

The intent of the study is to identify specific new research topics applicable to the VVERs that will not duplicate earlier work. It is also the intent of the study to identify needs based on the important differences that exist between VVERs and LWRs in design of components and systems, in materials of construction, and in containment.

The emphasis is on the VVERs, mainly because of the larger base of knowledge within the NEA Member countries relating to Light Water Reactors (LWRs). For the RBMKs, there is pressure-tube reactor technology in OECD countries but the base of experience is considerably less than it is in the case of LWR/VVER technology. Consequently, the Report reflects a significant difference in the scope and depth of research recommendations for the two types of reactor systems, and is greater in the case of the VVER. In the case of RBMK safety research, the report focuses on relatively near term efforts that have a good chance of reducing risk with reasonable efforts. Specifically, the Report calls attention to safety research that aims to provide an adequate technical basis for improved operating procedures, in both normal and emergency conditions of operation.

In this study, we have paid particular attention to safety research that can lead to tangible improvement in plant safety, *i.e.*, avoiding sequences that can lead to significant degradation of safety barriers, rather than to simply improving estimates of risk; and to research that can result in specific, practical actions in the plant, either by means of engineering fixes, or by improved operations or maintenance. Prioritisation in this way ensures that research carried out to address the needs identified will have the best chance of enhancing public safety.

Report Structure

At its first meeting, the Support Group decided on a structure for its review, and formed the following seven Task Teams:

- Thermal-Hydraulics/Plant Transients for VVERs.
- Integrity of Equipment and Structures for VVERs.
- Severe Accidents for VVERs.
- Operational Safety Issues.
- Thermal-Hydraulics/Plant Transients for RBMKs.
- Integrity of Equipment and Structures for RBMKs.
- Severe Accidents for RBMKs.

Each Task Team prepared and presented its report to the Support Group as a whole for review and approval. Consequently, the Report represents a consensus of the Support Group that outlines the arguments for the safety research needs with the focus on the main technical issues that justify the need and urgency. The written text addresses three basic questions:

- What is the safety concern?
- What are the open issues?
- What are the safety research needs?

The safety research needs as identified by the seven Task Teams, and approved by the Support Group, are reflected in the structure of the Report. The chapter on the Uses of Safety Research provides examples on how Western research has been applied to improve the safety of nuclear power plants. In addition, the chapter emphasises the need for a national safety research policy.

2. USES OF SAFETY RESEARCH

In connection with this Report on Safety Research Needs for Russian-Designed Reactors it is pertinent to point out uses and applications of nuclear safety research in OECD countries, as guidance for ways in which safety research can be applied to Russian-designed reactors.

There are three main motivations for doing safety research. Safety of the public is one of these, and governmental regulatory authorities generally take the lead in this category because they are responsible for defining and enforcing safety regulations. Development of new and improved plant, system, and component design is a second category, and reactor designer-manufacturers generally take the lead in this category for the purpose of improving products for the market. Improvement of reactor operations is a third category, and plant owner-operators generally take the lead for the purpose of improving performance and operating safety. Although responsibility for each category is clearly drawn, interests in them often overlap. As a result, research sponsors may include parties other than the party with primary responsibility in a particular category. Although examples that follow refer to particular countries, the experience with research and its application is common to all OECD countries with nuclear power plants.

Research for Safety of the Public

Research in this category is of long standing. Because of prime safety reliance on reactor pressure vessel integrity, and the knowledge that radiation affects the material properties of the pressure vessel wall, research on heavy section steels used for reactor vessels began in the U. S. in the mid 1960s, and continues today, with related programmes in OECD countries. This research has focused on radiation embrittlement of reactor vessel walls, effects on weld material properties, fracture mechanics of cracks and flaws, and thermal cycling and fatigue, to name some important topics. For channel-type reactors, research on material properties, radiation embrittlement, fracture mechanics and In-Service Inspection technology has been carried out in Canada, Japan and Russia.

Emergency Core Cooling Systems (ECCS), and thermal-hydraulic code development and validation is a second major research area of long standing interest, important to normal operation of all plants, and to prediction of response in off-normal events and accidents. This research contributed greatly to development and proof of ECCS performance for small and large loss-of-coolant accidents (LOCAs), to safety assessment of reactors and systems, and to assessment of containment capability and performance. Almost all reactor plant safety reviews depend on the analytical codes that research programmes in this area have developed and made possible. All OECD countries with operating reactors do work on thermal-hydraulic codes. A number of these research programmes were co-operative international efforts, such as, for example, LOFT, BETHSY, ROSA and the 2D/3D project. For channel-type reactors of Western design, thermal-hydraulic code development and validation has used the results from large-scale test facilities in Canada and Japan.

Early questions about ECCS reliability were the spur that led to development of probabilistic safety assessment (PSA). Although engineers understood the importance of ECCS during development, they did not have adequate methods to evaluate ECCS reliability. The development of PSA disclosed that small break LOCAs and human error could lead to serious accidents, but the findings received little attention at the time. PSA methods were later vindicated by the accident at Three Mile Island (TMI). PSA has been in wide use for safety assessment of nuclear plants ever since.

Regulatory authorities in OECD countries have shown keen interest in generic safety issues (GSIs), *i.e.*, those that apply to more than one plant. Examples include loss of availability of auxiliary feedwater systems, station blackout, loss of residual heat removal, reactor coolant pump seal failure, and motor-operated valve failure. In the U. S. GSIs required much research efforts and considerable resources over a period of 15 years of discovery and resolution.

There is an extensive record of research on seismic hazards at nuclear power plants in OECD countries and others as well, including studies of seismicity and seismic events, response of nuclear plant structures, systems, and equipment to earthquake, and testing of component scale models on shake tables. The largest of these is the Tadotsu Facility of NUPEC in Japan. The research findings apply to seismic hazard assessment, and to design or modification of structures to achieve greater safety margins.

In OECD countries, severe accident studies and experiments have been a major research activity since the TMI accident. A severe accident, by definition, involves significant fuel damage or melting. Objectives of the research included a better understanding of severe accident phenomena, identification of sequences of events, evaluation of containment integrity and safety margins, and uncovering vulnerabilities in individual plants that could lead to severe accidents. Experiments performed include hydrogen burning and detonation, molten core-concrete reactions, fuel-coolant interaction, experiments with degraded cores, and containment loading from molten core material. Notable international co-operative experiments are, for example, the CORA facility for fuel failure modes in Germany, and PHEBUS for assessing severe fuel damage and determining fission product behaviour in France. Eleven OECD Member countries participated in the TMI Vessel Investigation Programme to evaluate the condition of the reactor vessel during the accident. More recently, 14 OECD Member countries and Russia have started the RASPLAV project to improve the understanding of in-vessel molten material retention and coolability within the reactor pressure vessel. Knowledge from these research efforts have improved our understanding of associated issues and provided a better quantification of the risk.

Research has also played an important part in development of improved safety by providing information and data that have been incorporated into reactor regulations. The following are a few examples drawn from U. S. experience: standards of combustible gas control in LWR power reactors; acceptance criteria for ECCSs; fire protection; environmental qualification of electrical equipment important to safety; fracture toughness requirements for protecting against pressurised thermal shock; reduction of risk from anticipated transients without reactor trip.

Research for Development of Improved Plant, System, and Component Design

Nuclear reactor manufacturers have sponsored many research programmes for development of improved plants, systems and components. They have done this work sometimes with and sometimes without government funds. Recent examples include development of a new generation of pressurised water reactors in Europe, development of plants with passive safety features in the U.S., and development of the Advanced Boiling Water Reactor in Japan. Research findings in natural circulation thermal-hydraulics, and in component development have played major roles in these developments.

Interest in improved fuel cycle economics and extended reactor operation by longer intervals between refueling has challenged designers for assurance of reliable fuel performance at higher burn-ups, and regulators as well for assurance that risk of fuel failures in accident conditions, such as reactivity initiated events, is not excessive. Efforts are being carried out (CABRI, NSRR, IGR) to validate current licensing limits.

Research for Improved Reactor Operations

Several problems have come to light during plant operations, maintenance, and repair that have required research and development. Research efforts on the part of plant owner-operators and research organisations that serve them are contributing to improved operation and maintenance. Some important examples are summarised below.

Reactor vessel research includes demonstration of thermal annealing to restore the nil-ductility transition temperatures of vessels, and development of improved detection of cracking in vessels and reactor internals.

Steam Generator research includes development of water chemistry to reduce sludge accumulation, development of improved technology for sludge removal, and a search for corrosion inhibitors. There is work underway to develop more reliable NDE (non-destructive examination) systems for detection and sizing of tube flaws so that tubes need not be plugged unnecessarily.

Component ageing and equipment obsolescence are important problems requiring research and development. There is research underway to find ways of identifying ageing degradation and indicators that show when component replacement is needed. Electrical cable ageing and degradation represents a safety concern, and there is research underway to show when replacement is needed. The replacement of analogue controls with digital control systems is happening, and research is helping to make these conversions trouble free.

Development and application of better NDE technology and conditioning monitoring techniques are helping to improve the reliability of maintenance work. Better assessment of motor-operated valve performance is helping to improve reliability.

Research is helping to improve safety assessment by various means including development of probabilistic safety analysis for fire hazards, and development of seismic qualification requirements for replacement equipment.

Several OECD countries have done research on human performance at nuclear plants, and there have been many applications. Some of the important applications are operator qualification and training, improvement of control room instrumentation displays and the person-machine interface in control rooms to reduce or eliminate operator error, development of maintenance procedures, and of Emergency Operating Procedures.

Recently, much research in this field has been focused on the verification and validation of accident management strategies developed to prevent or cope with severe accident conditions.

Development of a National Safety Research Policy

The above discussion has given some examples of how safety research has been successfully utilised in OECD countries with nuclear power plants. Safety research is carried out best within a broad framework of a safety research policy that sets forth goals, objectives, a strategic plan for achieving them, and incorporates tools for reviewing and measuring progress toward goals. Measuring progress helps to identify timely course corrections in research programmes and to avoid waste of resources. These tasks, as well as priority setting are the overall responsibility of national safety authorities.

A strategic plan for safety research identifies problems that require research, setting specific objectives for the long term and the short term. It makes provision for funding the research programmes, for locating the best researchers and facilities for research projects, and for requiring the researchers to develop a specific plan of action for each project. In effect, a research strategic plan becomes the organising principle for the safety research programme.

Planning research should take into account a range of factors that influence success or failure. The success of a research activity in any area cannot, of course, be guaranteed. Nevertheless, factors such as the availability of qualified and experienced researchers, timeliness for completion and integration with the full process of safety enhancement are known from OECD experience to be important in the successful completion and implementation of safety research. Such factors need to be included in the planning process for Russian-designed reactors.

3. THERMAL-HYDRAULICS/PLANT TRANSIENTS FOR VVERs

Thermal-Hydraulics

Safety Concerns

Thermal-hydraulic phenomena govern most of the accidents in the design basis and beyond design basis ranges. The thermal-hydraulic safety concerns deal with:

- Safety analysis of design basis accidents (DBA) for evaluating the adequacy of the design to cope with accidental situations.
- Safety analysis of beyond design basis accidents (BDBA) for evaluating if consequences can be considered as acceptable.
- Accident management (AM) development to prevent or mitigate accident consequences.

To address these safety concerns, thermal-hydraulic codes should be sufficiently developed to describe all phenomena and sufficiently validated against relevant experimental data. In addition, accident prevention is particularly important for many VVER and RBMK plants as they have no or limited containment function.

In addition to large and small break LOCAs, there are some other important accidents which should attract particular attention due to the specific features of the VVER reactors, *e.g.* primary-to-secondary leaks due to steam generator (SG) tube ruptures or collector break and steam line breaks with primary side overcooling. For accidents such as loss of off-site power, or loss of feedwater, or, control rod withdrawal, a VVER-440 behaves quite differently than a Western PWR due to the presence of hot leg loop seals and horizontal steam generators resulting in plant specific oscillatory natural flow. Asymmetric loop behaviour must be accounted for in the analyses.

Open Issues

Thermal-hydraulic aspects of safety analysis and measures to improve safety of VVER reactors are directly connected with the capabilities of the thermal-hydraulic (TH) codes to model accidents in the nuclear steam supply system. Eastern organisations are developing domestic TH codes and applying Western best estimate TH codes. Hence, the quality of accident analysis in Eastern Countries generally depends on (a) the degree of development and verification of the domestic codes, and (b) the degree of knowledge of Western best estimate TH codes, their adaptation to VVER-specifics and their verification.

VVER thermal-hydraulic behaviour is influenced by peculiarities of the VVER-type of reactors compared to Western PWRs, e.g. for VVER-440s they include:

- Six primary loops (smaller diameters, parallel loop effects).

- Loop seals in both hot and cold legs (affects natural circulation).

- Shrouds around fuel assemblies (less cross-flow, bi-directional flows more probable).

- Fuel assembly follower of a control assembly in a high lower plenum (heat source, boiling in lower plenum).

- Horizontal steam generators (low elevations, horizontal tubes affect natural circulation modes considerably).

- Large primary and secondary side water inventory (incidents happen more slowly).

VVER TH code validation matrices have been developed. Systematic work on the improvement of the VVER code validation matrices is being performed under the auspices of the OECD. Together with the OECD validation matrices, these matrices should constitute the basis for code validation. Preliminary analysis of the matrices shows that additional experimental studies on integral test facilities and separate effect tests are needed.

The degree of verification of the domestic TH codes has been recognised as being quite high. Nevertheless, validation against additional tests is necessary and it would be desirable to make use of some experimental data from the OECD countries. For Western TH codes, their validation against OECD experimental data has been generally performed but their adaptation to VVER-specifics and the corresponding validation should be completed.

The development of symptom-oriented accident management (AM) procedures has been started in the Eastern countries and it is necessary to broaden the capabilities of TH codes for the modelling of AM scenarios. AM scenarios should be experimentally tested, additional code validation should be performed based on these tests, and analysis of the efficiency of AM measures should be carried out with the verified TH codes.

For best estimate codes, quantitative evaluation of code uncertainties is needed. This requirement is common for both Western and Eastern countries.

Safety Research Needs

Development of Symptom-Oriented AM Measures

The development of symptom-oriented AM measures is certainly a high priority for improving plant safety. It should include:

- Experimental investigations of AM scenarios on integral test facilities.
- Validation of TH codes for application to AM analysis based on AM experiments.

- Evaluation of the efficiency of AM measures with verified TH codes.

Additional Experimental Investigations

The needs for additional experimental investigations involve:

- Investigations of transients and accidents on integral test facilities.
- Investigations of separate effects.

The research needs should be defined according to the code validation matrices taking into account the safety importance of the specific VVER phenomena, and considering not only the previous experimental work done in Eastern countries but also the applicability of the large amount of work performed for PWR in the OECD countries.

Improvement of VVER Code Validation Matrix and Development of Russian Data Bank (Test Facilities and NPPs)

A basic VVER code validation matrix has been recently developed but additional work is still needed to improve and complete what should be the reference for the validation of the TH codes. This additional work includes:

- Analysis of existing experimental data needed for filling the gaps in VVER code validation matrices.
- Selection of suitable data for code validation.
- Selection of phenomena and processes which are specific for VVERs.
- Selection and analysis of transients and incidents which have occurred at VVER NPPs.

The data necessary for the validation matrices should be incorporated into a readily accessible data bank.

Additional Validation of Thermal-Hydraulic Codes

Additional validation of the TH codes should be performed, based on the VVER and CSNI validation matrices and on the additional experiments to be performed.

Reactor Physics

Safety Concerns

In accident situations, reactor physics calculations determine the power transients which are applied through the fuel rods (involving thermo-mechanics) to the coolant (involving thermal-hydraulics). The strong feedback coming from the thermal-hydraulics conditions which closely influence reactivity conditions must also be taken into account.

As a consequence, reactor physics is used to address safety concerns, especially for all accidents involving rapid reactivity/power transients such as ATWS, RIA, or boron dilution accidents. For these cases, neutron kinetics phenomena are critical, and reactor physics is, therefore, a key, high priority discipline to address these safety concerns. In addition, reactor physics plays a key role in the analysis of other scenarios which could potentially lead to recriticality in VVER cores.

The solution to these concerns should involve:

- The development and improvement of neutron kinetics codes.
- The development and improvement of the coupling between reactor physics codes and TH codes.

Open Issues

VVER core safety is influenced by the following main peculiarities of the VVER-type reactors compared to Western PWRs:

- Hexagonal form of the fuel assemblies and triangular fuel rod lattice, fuel pellets with a central hole, higher fraction of structural materials in the core.

- Absence of the axial blankets, and non-optimised design of spacers in a fuel assembly.

- A somewhat smaller core and higher specific power.

- Rather high concentration of the liquid boric acid in the coolant because there is little use of burnable absorbers and flux flattening materials.

- Presence of the control assembly followers and fuel assembly shrouds (for the VVER-440 only).

The detailed evaluation of the spatial power distribution and its evolution with time during accidents should be considered as one of the main tasks in the calculational analysis. The general approach to full-core neutron analysis is the diffusion approximation for the solution of the neutron transport equations. For most practical cases, this approximation is sufficient and is being successfully applied. But in some typical accident cases, this approximation either cannot be applied directly or requires additional evaluation or sensitivity analysis. Improved numerics with negligible numerical diffusion appears necessary to treat realistically accidents with steep gradients in coolant conditions in the core. Examples of transients where the local changes in the neutron spectrum are very strong include: low coolant density or “water-vapour” boundary in the core; fast control rod ejections at the HZP conditions; and cold pure water injection. In addition, the existing few-group libraries are insufficient for such situations in the core.

Besides these developments of libraries, verification should be performed. It appears that an important open issue is related to the lack of well-described available experimental information on transient VVER behaviour which could be used for the verification of the codes.

Although the coupling of the 3-D neutronic with thermal-hydraulic codes for the VVER accident analysis is partly available or underway, the possibility of representing the real 3-D feedback from thermal-hydraulics and thermal-mechanics needs additional evaluation. Similar activities are being undertaken by Western countries for LWR, so it may be considered as a common open issue.

The urgency of the solution of these open issues is based on the need for detailed safety analysis of VVERs in accidents where the spatial effects and time dependent coupling effects are essential.

Research Needs

Improvement of Dynamic Reactor Physics Codes for VVER Accident Modelling

These improvement actions should include:

- Extension of multi-group libraries to treat the neutron dynamics during an accident.
- Development of improved neutron-physical models.
- Development of the data base on the neutron dynamics at operating VVERs.
- Validation of the 3-D neutron kinetics codes.

Validation of the Coupled 3-D Neutron Kinetic, Thermal-Hydraulic, Thermal-Mechanic Codes

Validation, based on available NPP data on transient VVER behaviour and suitable experimental data, of the coupled codes should be undertaken.

Containment Thermal-Hydraulics (Short Term)

Safety Concern

This chapter addresses the thermal-hydraulic aspects of short term containment response to design basis accident (DBA) conditions and other beyond-DBA scenarios that may compromise the containment function. Long term questions such as those encountered during severe accidents are discussed later on this Report.

Containment thermal-hydraulic must be properly understood and modelled since some accident conditions may challenge the adequate performance of the last safety barrier. Pressure/temperatures transients, compartments behaviour, distribution of steam-gas-drop mixture, heat and mass transfer in presence of non-condensable gases and containment by-pass constitute some examples of issues to be considered.

Open Issues

The main thermal-hydraulic processes in VVER and PWR containments are similar, except for some VVER-specific problems, such as the VVER-440-213 bubble condenser containment system. In addition, the consequences of accident management in VVER-440 containments should be investigated due to the concerns about the effects of their leaks to the environment.

Codes which correctly describe the phenomena and are well validated against relevant experiments are needed to address these safety concerns. Further verification of Western and Eastern codes and their use for the VVER safety analysis is a generic issue. As for the validation of Eastern codes, there is a need for extending the validation domain and consequently for the use of the Western experimental data base (HDR, Battelle, NUPEC).

For the bubble condenser of VVER-440-213 containment, the specific open issues/research needs have already been established by the OECD Support Group on Bubble Condenser Research and a research project is being discussed in the framework of a EC programme.

Research Needs

VVER-440-213 Bubble Condenser Containment

For the bubble condenser containment, research activities should focus on the verification of system performance through the execution of the necessary integral tests and separate effect tests. This should be supported by specific model development, if needed.

Development of VVER Containment Code Validation Matrix

The action already initiated by the OECD Support Group should be pursued and specifically by:

- Analysis of existing experimental data needed for filling gaps in the matrix.
- Selection of suitable data for code validation.
- Development of a readily accessible experimental data bank.
- Identification of additional experiments which are needed in containment thermal-hydraulic.

Additional Verification of Eastern Thermal-Hydraulic Codes Against Integral and Separate Effect Tests

There is a need to carry out the verification of Eastern TH codes against integral and separate effects test such as HDR, Battelle, NUPEC. This verification should also include cross-verification with Western codes.

4. INTEGRITY OF EQUIPMENT AND STRUCTURES FOR VVERs

Safety Concerns

The integrity of the reactor coolant boundary, and the integrity and leak-tightness of containment structures are necessary elements to achieve an acceptable level of safety of light-water reactors. Therefore, the methods applied for the assessment of the integrity of components and structures must be verified for the full range of application. It is common experience that the intended safety margins used in the design of equipment and structures may not cover all influences resulting from the long-term operation of the power plant. Some of these influences are connected to design deficiencies, others are related to unanticipated materials ageing phenomena. With respect to VVER reactors, the safety concerns are related to:

- Capability and limitations of present evaluation methods for reactor pressure vessel integrity.
- Predictive capabilities for assessing the integrity of the steam generator collector of the VVER-1000.
- Integration of the leak-before-break concept into the overall safety approach.
- Capability of non-destructive testing methods to characterise defects and change of material properties.
- Verification of predictive models to demonstrate containment performance.

Open Issues

Based on a number of safety analyses that have been performed and technical discussions in various national and international safety reviews, the most urgent open issues are:

Evaluation of Reactor Pressure Vessel (RPV) Integrity:

- fracture resistance which takes into account the actual condition of vessel material under operation condition, including:
  - effect of RPV cladding on crack initiation and crack growth,
  - increase in material crack resistance as a consequence of a warm pre-stress effect,
  - verification of pressurised thermal shock (PTS) assessment methods;
- irradiation embrittlement level which takes into account the recovery heat treatment and actual flux level for VVER-440;

- irradiation embrittlement of VVER-1000 containing up to 1.9 wt.% of nickel in weld joints and possibilities of applying a recovery heat treatment.

Evaluation of the Integrity and Lifetime of a VVER-1000 Steam Generator Collector:

- effect of residual stress distribution from the expansion process;
- local secondary water chemistry condition;
- conditions for accelerated crack growth;
- leak rate prediction.

Seismic and Ageing Assessment of Equipment and Structures:

- confirmation of design basis seismic events;
- seismic fragility assessment of components;
- ageing assessment of critical VVER components.

Approach to Derive Leak Size for Safety Analysis:

- effect of conditions beyond the system technical specification;
- environmentally related ageing mechanisms;
- effect of internal/external impact;
- effect of In-Service-Inspection (ISI) and monitoring systems.

Containment Performance:

- fluid structure interaction effects for the bubble-condenser of the VVER-440-213;
- long-term integrity of pre-stressed concrete containments of the VVER-1000 type and change in leak-tightness.

Interaction With Other Fields of Research

Integrity assessments should always address the effect of loads, the materials and the defects. Therefore strong emphasis is given to the system and thermal-hydraulic analyses to identify the boundary loadings. For the integrity assessment of the reactor pressure vessel, there is a need to describe the fluid mixing and condensation phenomena under LOCA conditions, as well as the heat transfer coefficients in such a way that the loads acting on the structure are representative.

Safety Research Needs

Considerable national and international programmes have been carried out to address these open issues. Therefore, future work has to build upon these previous results and is mainly of a complementary character. Within the scope of this report, the intention is to identify the research needs in a general way: the detailed description of a work programme is part of the definition of future projects.

Following the structure as outlined in the "Open Issues" section above, the specific safety research needs for analytical and experimental work are identified. Reference is made to relevant research projects which are in the planning stage but not yet funded.

Methods for the Evaluation of Reactor Pressure Vessel (RPV) Integrity

Fracture resistance incorporating the actual condition of the vessel material under operating conditions

In determining RPV failure resistance, conservative values of design characteristics (T_{K0} and K_{1C}) are generally used and these present the lower boundaries of the bulk of the experimental data. The given values for the nil-ductility transition temperature may be excessively conservative for a number of reactor pressure vessels for both the critical temperature and fracture toughness. To define the given reserve, it is necessary to develop and verify models for the load carrying capacity of the reactor pressure vessel in operation. The specific research has to address the validity of the calculational procedures for shallow cracks and for combined thermal and mechanical loading (pressurised thermal shock conditions). This type of research is already done in the West, but for the VVERs only limited large scale tests have been performed and not all necessary conditions have been evaluated. The experimental programme needs to be expanded to investigate the RPV cladding effect on crack initiation and crack growth. Furthermore, variations in loading conditions are necessary to investigate the difference between axi-symmetric and asymmetric loading. Of specific importance are tests which address the appearance of increased crack resistance response of a structure as a consequence of a warm pre-stress effect. To cover these topics a proposal has already been developed and is being discussed within the frame of the European Network for the Evaluation of Steel Components (NESC).

Radiation Embrittlement of Reactor Pressure Vessel (RPV) which takes account of Recovery Heat Treatment and Actual Flux Level for VVER-440s

The VVER-440 RPV re-evaluation by surveillance sample findings is complicated by the fact that the conditions of their irradiation differ with regard to the flux level. The RPVs in operation are characterised by low flux level and there is information that the time of the fluence build-up essentially affects radiation embrittlement. Apart from this there still remains the unsolved problem of RPV re-evaluation under their post-annealing operation. It is advantageous for Eastern organisations to participate in developing the IAEA data base on reactor materials radiation embrittlement.

Within the presently TACIS-sponsored programmes, trepans from the Novovoronezh Unit 2 vessel have been investigated to measure nil-ductility transition temperature based on Charpy sub-size specimens and establish through-wall material properties of an irradiated pressure vessel as well as the relationship between fracture toughness (K_{1C}) values and Charpy V-notch values. Unfortunately, the operating conditions regarding the temperature are not fully representative of the VVER-440s presently in operation. Taking these more recent investigations as a background, a complementary programme taking samples out of the reactor pressure vessels of the Greifswald Unit would be highly advisable to broaden the data base and quantify the scatter band. Within such a programme, samples which represent more typical chemical conditions of the near core weld and the operating temperature could be investigated.

Radiation Embrittlement of VVER-1000 Weld Joints containing up to 1.9 wt.% of nickel and the Possibilities for Applying a Recovery Heat Treatment

In some VVER-1000 RPV weld joints the content of nickel is much greater than 1.5 wt.% (up to 1.9 wt.%) which may significantly reduce the RPV radiation resistance. The existing data bases for VVER-1000 surveillance samples must be analysed. It is also advantageous to carry out studies to assess the possibilities of annealing the VVER-1000 RPV.

As a first step it is proposed to develop a correlation function which relates the measured properties of the surveillance specimens (which are located in a steep flux gradient and experience core outlet temperature and, hence, have a higher temperature) to the flux and temperature conditions of the near core weld. Additional experimental investigations are necessary. Further research is necessary to develop more physically-based damage functions to enhance the capability of models to predict the change of material properties with neutron irradiation. Furthermore, the relevancy of the surveillance programme for monitoring irradiation embrittlement should be analysed, and a modified surveillance programme should be developed.

Development of non-destructive Testing Methods

There is a need to develop non-destructive testing methods for characterising the change of material properties. Two items are identified which require special research. One is the development of methods to pick up the onset of micro-cracks and relate this to the fatigue damage of the material. The other item is the measurement of changes in material properties due to irradiation as well the susceptibility to sensitisation with respect to stress corrosion attack. The specific need for these methods arises because of the rather frequent application of annealing of the near core weld to reduce the embrittlement of the VVER reactor pressure vessel.

Progress in this area would provide great advantage for any kind of reactor.

Evaluation of VVER-1000 Steam Generator Collector Integrity And Lifetime

The integrity of the steam generator collectors is a key concern for the operability and life of Russian steam generators. Three to five years ago, an array of premature failures was observed for the VVER-1000 steam generators due to crack formations in the cross connections of the collector holes. As a result of process changes, the problem has been practically solved for new collectors but the procedures for evaluating the collector’s integrity and life have not been developed. The assurance of the VVER-1000 steam generator collector integrity remains one of the most important safety issues.

The effect of residual stress distribution from the expansion process have to be specifically addressed as well as the local secondary water chemistry conditions. As a result of experimental and analytical investigations it is expected that the boundary conditions can be developed which reduce accelerated crack growth. There is a need to improve the models to predict the leak rate through cracks in the steam generator collector from the primary to the secondary side.

Seismic and Ageing Assessment of Equipment and Structures

Pursuant to reviews of seismic disturbance at reactor sites in Western countries, it is appropriate to review the same in Eastern countries, and confirm the seismic design basis events for the reactor sites, using the data bases of the geological communities in the Eastern countries. There is also a need to analyse the seismic fragility of VVER components and structures, and to integrate the two sets of information into seismic margin assessments of individual plants.

Building on the base of component ageing research in Western countries, there is a need to assess ageing effects on VVER components, with focus on design and materials differences between VVER and Western reactor components, in order to contribute to development of monitoring and surveillance requirements, and effective maintenance schedules.

Methods to Determine Leak Sizes for Safety Analysis

The safety concept of the VVER 440/230 is based on tolerating very limited leak sizes for loss of coolant accidents. Safety re-evaluation have shown that the basic condition to apply a leak-before-break approach to the main recirculation piping do exist. Shortcomings have been identified and a methodology has, in principle, been developed.

It is important to note that for VVER 440/230s the leak-before-break approach, as a safety concept, will be applied in a much broader sense than in most of the Western countries. Specifically, the application of the leak-before-break approach is also used to determine if the limited ECCS and containment capacity are sufficient. This is not in line with the defence-in-depth approach but if leak-before-break sizes can be confirmed by experimental and analytical research, rigorous probabilistic assessment of piping systems may be able to provide adequate justification. In view of the overall safety approach, it is necessary to complement the methodology developed for the piping to the overall primary pressure boundary. Research is necessary to extend the present models to incorporate the effect of conditions outside system technical specification, to enhance present models to predict the effect of environmentally related ageing mechanisms on the safety margins for the pressure boundary including bolted connections, and to incorporate the effect of in-service inspections and monitoring systems on the failure rate. Furthermore, the methodology has to be expanded to include effects of internal and external impacts. The model development should include probabilistic elements. A large part of the experimental programme to verify the leak-before-break approach for the main piping has already been performed in Russia and the Czech Republic.

For the VVER-1000, there is a need for similar safety research activities since the leak-before-break approach is to be implemented in the overall safety concept.

In view of the large quantity of already-generated experimental data from numerous Western programmes, an evaluation group should be established to assure the best use of the available knowledge. Based on this review, complementary experiments can be defined to address specific questions related to the VVER materials.

Model Development to Predict Containment Performance

As an outcome of extensive discussion, two items have been identified which should be addressed by specific research programmes.

One is the fluid structure interaction effects related to the bubble condenser of the VVER 440-213 containment. In support of this work, the OECD Support Group on Bubble Condenser Containment has issued a comprehensive report on the safety research needs. A proposal is now being developed in the framework of a EC programme.

The other item relates to the long term integrity of pre-stressed concrete containment of the VVER-1000 type. Specific research is necessary to develop a model which could predict the loss of performance as a function of loss of pre-stress, caused by detrimental defects arising from operation or resulting from the construction.

5. SEVERE ACCIDENTS FOR VVERs

Safety Concerns

Safety concerns for severe accidents are well known and have been addressed in numerous reports. The most important concerns are to prevent severe accident conditions and to mitigate their consequences if they occur. For safety research, these concerns are:

- Extension of safety analysis for existing nuclear power plants (also to provide an information base for PSA).
- Development of a quantitative data base and methodology, including computer codes to support accident management procedures.
- Development of a data base and appropriate computer codes supporting reliable scientific background for enhanced safety features and safety concepts.

Open Issues

Safety analysis of NPPs requires the development of the appropriate experimental data base and computer codes capable of predicting the behaviour of system components in a wide range of accident conditions and for numerous materials. Both small- and large-scale experiments provide the necessary data base for code validation. Development and application of these codes facilitate the assessment of accident management procedures and their influence on plant behaviour. Most Western computer codes are available in Eastern countries for VVER safety assessments. In parallel, code development activities are underway in Russian institutions. The Russian codes describe, in a mechanistic manner, several phases of the severe accident: in-vessel melt progression, late phase of core degradation, molten core-concrete interactions and spreading, hydrogen combustion and detonation, and steam explosion. These codes may also be used to develop accident management procedures.

There are several high priority general research areas for supporting accident management and mitigation of consequences in the course of a severe accident [1]:

- Effectiveness of flooding for different phases of accident progression (core degradation, melt retention, in-vessel and ex-vessel cooling).
- Hydrogen production, distribution and measures to control hydrogen concentration to avoid negative consequences.
- Evaluation of possible aerosol releases pathways and avoidance of containment by-pass.
- Real-time accident progression monitoring and predictions of critical events.

Some of these issues are common to both VVERs and Western LWRs. In addition, however, there is a need for further work on specific VVER issues. High priority open issues for VVERs safety needs are:

- Validation of computer codes applied for VVER analysis.
- Development of an appropriate data base for material properties and their interactions.
- Development of advanced models and codes for common (PWR, VVER) use and VVER specific models (bubble condenser, steam generator, etc.) for their implementation into the codes, evaluation of severe accident codes adequacy and determination of additional experimental efforts if necessary.

Safety Research Needs

In-Vessel Phenomena

Early phase of core degradation

For the early phase of core degradation (before loss of core geometry) of the VVERs, priorities for research include:

- Improved understanding of the kinetics of material interactions in the course of the accident.
- Quenching of assemblies at high temperatures, including steam and steam-water mixtures, and the development of appropriate modeling methodology.
- Completion of the relevant material properties data base for the expected range of temperatures and compositions.
- Extended code validation, including uncertainties analysis.

It may be possible to make use of the information from several experimental programmes (CORA, PHEBUS-SFD, PBF, etc.) carried out in Western countries.

The material properties data base for specific VVER materials has to be extended. Models for an adequate description of VVER-specific design features such as the pressure vessel and the internal structures have to be developed and implemented in the codes.

Late phase of core degradation

The main areas of research for the late phase of core degradation are:

- Degradation of fuel assemblies and formation of debris bed.
- Core-wide degradation and melt progression.
- Relocation of corium in the lower plenum.
- Coolability of debris bed in the lower plenum.
- Molten pool behaviour in the lower plenum and its ex-vessel coolability and melt retention within the vessel.

- Determination of margins of pressure vessel integrity.
- Uncertainties analysis and accident management recommendations.

Some of these issues are of a generic nature. Other issues are very specific, for example, the fuel follower control rod design of VVER-440s may have a major effect on the relocation of corium to the lower plenum. There are several high priority international experimental programmes in progress dealing partly with the issues mentioned above, such as PHEBUS and RASPLAV. The RASPLAV project also contains supporting programmes for evaluation of material properties and code development.

Fission Product (FP) release and transport

The evaluation of the source term necessitates the understanding of the physical phenomena and reliable modeling of fission product release and transport in the primary system. Several small scale experiments are available from different facilities (ORNL USA, IPSN France, NIIAR Karpov Institute Russia) to assess fission product release from irradiated fuel. The large-scale integral PHEBUS FP tests, conducted in mostly prototypic conditions, allow the evaluation of fission product release and transport for several accident scenarios. Nevertheless, there are several issues related to the VVER materials and design. Priorities for research include:

- FP release from VVER fuel and validation of models both for oxidizing and inert atmosphere.
- FP deposition in the horizontal pipes of the steam generator (SG).
- Assessments of FP release for VVER-specific scenarios through SG (tubes rupture, collector rupture, etc.).
- Evaluation of fission product releases from debris beds and molten pools for reducing and oxidizing atmosphere at high temperatures.

Ex-Vessel Phenomena

Assessments of ex-vessel steam explosion

Extensive investigations have been performed to analyze ex-vessel steam explosion consequences both experimentally and theoretically. Several codes are in the process of being developed. Large scale experiments, including the FARO tests, provide the data base for code validation. Results of the experimental investigations may be used in the analysis of VVER steam explosions.

Molten core-concrete interactions (MCCI) and debris bed coolability

Experimental programmes were conducted in the past which utilized different types of concrete as well as different corium materials, both metal and oxides. Previously developed computer models were verified and uncertainties were estimated. It is recognized that thermal-hydraulics of MCCI phenomena is sufficiently understood. With respect to FP release, the current understanding is not sufficient due to the much lower accuracy in FP predictions in comparison to thermal-hydraulic behaviour. Nevertheless, the generated data serve as the experimental data base for code validation.

Large scale experiments with VVER concrete (serpentine and ordinary) were also conducted, namely ACE-L4 and BETA 7.1. Areas of research include:

- Coolability of debris bed.
- Low volatile FP release, aerosol generation (modeling and verification).
- Spreading of the melt and its interactions with water.

To address VVER specifics, priorities for research are:

- Properties of design materials (concrete) characteristic for VVERs.
- Effect of VVER-cavity geometry.

Hydrogen transport, combustion and detonation

In the field of hydrogen behaviour in the containment, most of the open issues are the same as for PWRs [1]. Experimental data base obtained at different facilities, including those in Russia (RUT, KOPER), are used for code verification. Priorities for research include:

- Hydrogen transport and distribution in containment;
- Hydrogen burning in different regimes:
  - Implementation of deflagration to detonation (DDT) criteria to H2 distribution codes;
  - Turbulent combustion models, especially for an actual VVER-440;
  - Flame acceleration limit in H2-air-steam mixtures;
  - Mechanical response of containment structures to dynamic loads.

The specific VVER containment design must be accounted for in assessments of hydrogen safety analysis.

Material Properties and Interactions Data Base

It is widely recognized that knowledge of material properties is one of the issues important for safety evaluations. Material properties data base should include all data (thermo-dynamic, transport and mechanical properties) for the core components and mixtures at appropriate temperatures and pressure ranges. This data base includes also necessary data for kinetic processes such as material reactions and interactions. It is well known that differences in behaviour between PWRs and VVERs are associated with different design materials and as a consequence differences in mixtures and kinetic interactions. Research is needed to establish this data base.

Containment Performance, Integrity And Source Term

There is a need to assess VVER-1000 prestressed concrete containment and VVER-440 compartments to determine their response to beyond design basis accident conditions. Short term containment integrity due to overpressure, direct containment heating, hydrogen burning, steam explosions, as well as leak tightness are the most important issues. This analysis should include specific features of VVER containment like prestressing tendons, liner, penetrations, seals, etc. Margins of containment integrity should be evaluated. For such an analysis, containment failure criteria should be developed using available experimental data and finite element codes.

To evaluate the source term, analysis of aerosol behaviour in the containment including possible pathways through leakage, performance of safety systems (spray, filtration and venting systems) is required. Research needs for assessing containment performance and source term evaluations include both experimental and code development activities.

Code Development, Validation and Applications

Severe accident code validation and application for VVERs

Many Western codes, including SCDAP/RELAP5, MELCOR, CONTAIN, ICARE2, ATHLET-CD, are available in Russia for the analysis of severe accidents in VVERs. Having been developed for Western NPP, these codes reflect most generic safety issues. At the same time, they are based on the designs specific for Western NPPs and considerable modifications are necessary to implement the specific features of VVERs. In the field of applications, the still open issues are connected with the validation of codes, and with the development and implementation of VVER-specific models. The final goal of the work is to use verified codes for accident management and risk assessments. Priorities for research include:

- Review of VVER-specific experimental facilities, experimental data and associated analysis.
- Development of VVER-specific models for structures and safety systems, their qualification/verification against available tests and implementation in the severe accident codes.
- Implementation of VVER-specific material properties data base.
- Analysis of VVER-specific severe accident scenarios.

Accumulation of data relevant to VVER-specific features, plant specific safety systems, and evaluations of severe accident codes will aid in determining if additional experimental effort is necessary to verify the codes and improve the accuracy of their predictions.

Code development

It is widely recognized also that several issues need to be considered using extended modeling and analyses. For such problems, mostly mechanistic approaches have to be used. Lumped parameter codes are not always sufficient to support safety analysis and accident management. This is especially true for hydrogen distribution and combustion/detonation, melt spreading and steam explosion. To produce adequate analysis of experiments performed in the past, advanced codes should be developed for molten-core concrete interactions (such as the ACE tests) and other experiments (like RASPLAV). Using mechanistic approaches improves the predictive accuracy of the codes and they can then serve as best estimate severe accident codes.

6. OPERATIONAL SAFETY ISSUES

Safety Concerns/Open Issues

It is almost impossible to overstate the importance of human performance in improving the safety of reactor operations. The fundamental concern is with errors of commission or omission affecting a wide range of activities from plant operations and maintenance to management and the attitudes of plant staff toward safety. Analysis of operational data and other assessments have confirmed the significance of human error in initiating accident sequences. On the other hand, well-trained operators and improved man-machine interface technologies, such as symptom-based emergency procedures and safety parameter displays, can be highly effective in preventing or halting the progression of an accident sequence.

For existing reactors, particularly older designs which included less automation and human factors considerations, operational safety improvement is a high priority since there are frequently fewer physical barriers to changes in this area than there is to modifying the design of the plant. Further, it is often possible to achieve significant risk reductions through improved management, training, procedures and other operational safety enhancements in part because these reactors tend to be more demanding in terms of human performance than newer designs.

Operational safety is a vast and difficult area involving interdisciplinary research from sociological, psychological and technical areas. It encompasses the man-machine interface; the communications and procedures that control critical activities; the use of computers in the plant and the reliability of their software; the effectiveness of maintenance and safety management systems; the assurance of quality; characterising the performance of individuals and groups in modelling the total plant safety system and optimising the balance between human activity and automatic response.

Safety Research Needs

This subject was addressed in the SESAR report which described safety research in OECD countries. All the operational safety research needs identified in the SESAR report apply equally to Russian-designed reactors. The primary differences relate to the extent of existing research applicable to OECD reactors versus Russian-designed reactors and the resulting priorities for the Russian-designed reactors. For example, probabilistic safety assessment (PSA) is a maturing technology in OECD countries where research is now focused on improved modelling and new features (*e.g.* ageing and human error pattern models, software reliability, passive safety systems). The PSA technology available in OECD countries has already been transferred to the operators of Russian-designed reactors. However, its application requires extensive, credible data on equipment reliability and human performance which are not well established for these reactors. Thus, acquiring the data needed to apply the existing PSA technology is much more important at this time for Russian-designed reactors than the OECD research on extending the PSA technology. However, the designers, operators and regulators of Russian-designed reactors should participate in the development of this technology both for their potential contributions to improved methodologies and to assure that the gaps in safety infrastructure between Eastern and Western programmes are closed in a reasonable time frame.

The following sections summarise the key operational safety research needs noting those that are common to both OECD and Russian-designed reactors and emphasising the unique needs for Russian-designed reactors.

Assessment of Operational Safety

Accurate data on operating experience is essential not only for the lessons that can be applied to other plants but also for extracting quantitative, plant specific reliability and other information needed to improve operational, maintenance and management practices and to validate risk assessment models. As noted in the SESAR report there is a continuing need to improve the collection, analysis, distribution and use of these data through modern information handling technologies. This is particularly true for Russian-designed reactors where collection, analysis and reporting of operating experience is a relatively recent activity initiated following the Chernobyl accident. As a result, considerable work specific to these reactors remains to be done. Some of the highest priority activities include:

- Improvement of the methods and software used to collect, evaluate and report operating experience. This applies to both human performance and equipment reliability and includes; more consistent and comprehensive root cause analysis of operational events; incorporation of maintenance history into the equipment reliability database and continued development of performance indicators to provide an objective measure of the level of safety.
- Development and implementation of a methodology for the early identification of accident precursors based the on analysis of operating events incorporating the results of probabilistic safety assessments.
- Improvement of the methods and technology for monitoring the condition of the reactor and safety-related equipment. This includes refined methods for measuring and calculating reactivity coefficients for RBMK reactors; vibration and noise analysis, technology for monitoring the operability of motor operated valves, etc.

Human Factors

The importance of human behaviour to reactor safety is widely recognised and the subject of ongoing research in OECD countries. Some priority areas of research for Russian-designed reactors include:

- An evaluation of the applicability of success and failure response curves developed for western trained operators, maintenance and technical staff to those of Russian-designed reactors. This would include an evaluation of the effect of differences in operational philosophy, training and facility design.

- The effect of adverse environments on staff performance. This should encompass the spectrum from performing maintenance and inspection tasks under difficult conditions to operator response during emergencies. It could also include research on the effectiveness of improved man-machine interface and decision support technologies in areas such as operator response to emergencies and non-destructive examination of metal components.
- The use of simulators, training mock-ups, etc. to measure the performance of plant staff.
- The analysis of operational experience, particularly in the maintenance area, to assess the effectiveness of the normal practices, procedures and quality/management oversight. The results of these analyses provide an independent check on other sources of human performance information.

This is particularly important given the magnitude of the changes the programme is undergoing and the lack of comprehensive, computer-based configuration management systems to reduce the likelihood of errors.

Instrumentation and Control

The instrumentation and control (I&C) equipment in many Russian-designed reactors is ageing and encountering reliability problems. In addition, the degree of separation between control and protection circuits and their environmental qualification is less than required by modern Western standards. Finally, electromagnetic interference from faults in one system have disrupted other safety related systems. Since complete replacement of the I&C systems may not be practical, it is essential to be able to accurately predict and prioritise I&C components or systems requiring maintenance, modification or replacement. The development of an accurate, comprehensive failure and reliability database as discussed above is vital to this task. In addition, research aimed at improving the immunity to electromagnetic noise in control and protection systems is needed. Finally, the safety basis for the protection system set points has not been adequately verified for all design basis events. This requires developing and implementing a comprehensive, validated code suite that can accurately model plant behaviour under normal, off-normal, and accident conditions as discussed elsewhere in this report. The following is a list of near term research priorities in this area for Russian-designed reactors:

- Improved I&C equipment reliability data base including calibration and set point drift as well as component failure.
- Methods and technology to improve the immunity to electromagnetic noise in control and protection systems.
- The development and application of modern set point methodology in order to effectively incorporate the results of improved plant safety analysis.

In addition, the designers and operators of these plants should participate in the international programmes addressing the obsolescence of I&C in nuclear power plants including the application of digital control systems and the validation of their control software.

Probabilistic Safety Assessment

Probabilistic Safety Assessment (PSA) is a maturing technology receiving wider and wider use. It is being employed in many new, creative ways, including risk-based regulation, operational decision support, maintenance prioritisation, and technical specification development. As noted previously, the near term priority for research in this area related to Russian-designed reactors is acquiring the data on human performance and plant specific equipment reliability needed to apply the existing PSA methodology to these reactors. Of equal importance are the validated deterministic analyses discussed elsewhere in this report that are required to establish the success criteria, core damage states and release fractions needed to obtain meaningful PSA results. While not research per se, it is also necessary to establis consensus standards or guidelines for the peer review and validation of PSA results. Support could be given to introduce Living-PSA, for both plants and regulators. Finally, more effort on joint research projects related to improving PSA methodology along the lines discussed in the SESAR report is warranted.

Accident Management Implementation

Effective, comprehensive and validated accident management strategies are essential to halt or mitigate the progression of an accident sequence or to mitigate the consequences. These accident management strategies must encompass the spectrum from response to design basis transients through severe accidents. Modern, symptom-based Emergency Operation Instructions (EOI) are being developed for most Russian-designed reactors. For many of these EOIs, detailed analysis of plant response using validated plant models is required. The results of PRA and severe accident analysis are also important inputs to this process. Finally, these strategies also need to address considerations such as control room habitability and the actions necessary to protect the public from off-site releases. Priorities for research in this area include:

- Improved simulator models capable of realistically depicting key control room signals during beyond design basis events and the proposed mitigation strategies. This is another example where Eastern programmes are upgrading to the current technology but are not yet actively participating in development of the next generation of technology under way in the West.
- Research on the use of expert systems to improve the response of plant staff to off-normal and emergency events.

7. THERMAL-HYDRAULICS/PLANT TRANSIENTS FOR RBMKS

Safety Concerns

Earlier Issues for Design Basis Accidents

The link between physics and thermal-hydraulic codes needed to be improved. The International RBMK Safety Review stated "The safety evaluation relies on the capability of the codes describing the full core behaviour. The need to improve 3D reactor core models including thermal-hydraulic feedback and to develop coupling between 3D neutronic codes and thermal-hydraulic modelling of the main coolant circuit are strongly emphasised."

Multiple fuel channel ruptures were of great concern because of the potential for lifting the reactor top plate due to the limited venting capability of the reactor cavity. (See also Chapter 8 of this Report.)

Release, transport and containment of fission products were not well modelled. In particular, the International RBMK Safety Review stated "The codes available are not adequate to describe the pellet and cladding temperature histories after fuel channel break conditions."

Today's Issues

The void effect has been significantly reduced thereby increasing the requirements for neutronic data base accuracy and for development of core physical models used in coupling neutronic and thermal-hydraulic codes. The power pulse for the largest LOCA at the Ignalina NPP, which has a void coefficient of 0.6 to 0.8ß, gives rise at most to a 10% power increase. At the Smolensk NPP where the coefficient is almost zero, the rise in power may not be detectable.

Multiple ruptures of fuel channels have diminished in significance. A better understanding of the behaviour of the RBMK under accident conditions (reduced pressure) shows that the number of simultaneous fuel channel failures which can be coped with is more than originally thought. Some units are adding reactor trips that will shut the reactor down quickly in the event of Distribution Group Header blockage thereby substantially reducing the likelihood of multiple channel ruptures.

Calculations of fission product release, transport and retention remain an important issue which should receive more attention.

Open Issues

Until the 1980's, integrated safety analyses were performed by output and input from specialised codes which generally used highly conservative models and assumptions. Physics codes used simplified links between the neutron physics and the thermal-hydraulics. There were no 3D codes (there were only 2D codes) which used coupled calculations.

Since 1989, "Best Estimate" types of codes have been introduced into RBMK safety analyses. In the current work for the Ignalina Safety Analysis Report, Western codes such as RELAP and ATHLET and Russian codes such as SADCO, MOUNT and STEPAN are being used.

The analyses of Design Basis Accidents for the RBMK is well understood today. The reduction in the void coefficient and a better understanding of alternate heat sinks result in lower fuel temperatures for some accident sequences. There are, however, several issues related to computer code development, optimisation and verification, as well as extension of the experimental data base for a better understanding of the technical basis for safety criteria.

Safety Research Needs

Technical Basis for Improvement of Safety Criteria

Improvements in the specifications and support for safety criteria are needed today in the following areas:

- Onset of film boiling.
- Fuel failure.
- Fuel channel failure.
- Post-accident hydrogen distribution.

Onset of Film Boiling

There exists today a solid base of data for steady state cases, but more information is needed for transient process in order to confirm the adequacy of the steady state correlations.

Fuel Failure

Experimental data are needed to confirm that Zr-1% Nb behaves in a similar manner as Zircaloy. In addition, further data are needed on long exposures to irradiation at moderate temperatures.

Fuel Channel Failure

There exists only a limited set of data in the range of low and medium pressure combined with high temperatures. Further data are needed on the failure of a fuel channel as a function of pressure and temperature.

Post Accident Hydrogen Distribution

The current data base on hydrogen distribution within the various compartments is very limited. A considerable amount of work is needed to undertake such calculations when the correct tools are in place. It is possible that a code like GOTHIC may be used for the complex distribution among the variety of rooms within the various types of Accident Localisation Systems associated with the different types of RBMK containment systems.

Code Improvement and Validation

Sensitivity to Modelling Schemes

While this is a generic issue not limited to the RBMK, it is particularly important for the RBMK because of the large number of fuel channels to be modelled. Further work is need in regard to determining the most appropriate nodalisation, choice of models, numerical schemes for the calculations, etc. Numerical experiments are needed to determine the sensitivity of the calculations to these items.

Further Development of Coupled Physics and Thermal-Hydraulic Codes

While considerable progress has been made in coupling the neutronic and thermal-hydraulic codes, further work is needed on making these computations more efficient. This is needed so that sensitivity calculations can be undertaken for different reactor core states in an efficient way. Currently it takes too long to produce results for a single case and this hinders the amount of sensitivity calculations which are performed.

Development of Mechanistic Fission Product Transport and Retention Codes

Current safety analysis performed for the RBMK does not contain models which mechanistically determine the amount of fission products released during an accident. Empirical factors are used in the calculations. Furthermore there are no models in use for transport and retention of fission products within the primary cooling system nor for retention in the containment. The potential exists for significant dose reductions to members of the public with more accurate modelling.

Verification of Thermal-hydraulic Codes

While there has been a considerable amount of verification already performed for RBMK codes and a verification matrix exists, this matrix needs to be continuously improved. In certain accident sequences, stable oscillatory flow regimes are set up in parallel channels. In order to verify these calculations, experimental data specific to the RBMK are needed. Research on verification of flow instabilities in parallel channels is needed for the analysis of the probability of channel tube ruptures. For some accident sequences natural circulation provides the long term heat sink and further information on counter-current flow regimes and post-dryout heat transfer are needed to support the safety case.

Verification of Physics Codes

There are currently programmes in place for validation of steady state physics calculations. The International RBMK Safety Study, performed by a consortium of countries, has shown that for the Smolensk NPP core, the code calculations performed using Russian codes gave very similar results as the Western codes when applied to the same calculation. However that same study noted that considerably more work was needed to verify transient calculations. It is very difficult to obtain actual plant data from RBMK cores for this type of work. A more likely approach will be to continue the cross-comparisons of the various libraries and computer codes used in Russia with those used in the West.

Verification of Thermal-Mechanical Codes

Fairly simple conservative criteria are currently used in determining when fuel failures occur. Additional experimental information is required on cladding behaviour under LOCA type calculations to more realistically estimate the amount of fuel which might fail. Similarly more experimental data is needed for transient processes following single channel failures in order to more realistically calculate the response of the reactor cavity to these breaks.

8. INTEGRITY OF EQUIPMENT AND STRUCTURES FOR RBMKs

Introduction

The reactor coolant system of an RBMK may be divided into the part of the circuit which flows through the fuel channels and the part of the circuit which flows from the steam separator to the steam turbine and then through the feed water system back to the steam separator. The first circuit is made of austenitic stainless steel piping and components, or of carbon steel clad with austenitic stainless steel, although there are some carbon steel valves. Each feeder pipe has a flow control valve to the channel inlet side and the flow is periodically manually adjusted. The second circuit is mainly made of carbon steel with the steam separator having more than 400 nozzle connections.

The fuel channel pressure tubes are made of Zr-2.5 Nb alloy diffusion-welded to Ti-stabilized stainless steel end-pieces. The Zr-2.5 Nb pressure tubes are 80 mm in diameter with 4-mm wall thickness located in a ~12-m diameter core. Some 1600 of the total of 2000 channels are fueled, with the remaining channels used for control rods, cooling graphite and other purposes. The Zr-2.5 Nb fuel channels are protected by autoclaving them to obtain a hard black oxide film.

Open Issues

Among the high priority RBMK generic safety issues, those related to pressure boundary integrity components are the following:

- In-Service Inspection (ISI).
- Break of Critical Components.
- Fuel Channel Integrity.
- Special Channel Integrity.
- Fuel Handling during Seismic Excitation.
- Seismic and Ageing Assessment.

Safety Research Needs

Most, if not all, of the important safety research needs for the integrity of RBMK components are related to:

- The integrity of RBMK critical components outside the core, especially the integrity of the RBMK large diameter primary circuit components, and
- The integrity of RBMK critical components inside the core, such as the integrity of the fuel channels and reactor cavity.

Further description of the safety research needs in the area of In-Service Inspection (ISI) are included under these two general research needs.

Integrity of RBMK Large Diameter Primary Circuit Components

Development of Leak-Before-Break (LBB) Technology

Leak-before-break (LBB) evaluation work is still in progress. Evaluations do not yet prove that the LBB concept applies to the RBMK pressure boundary, but future results could prove valuable in properly allocating resources. The LBB method and techniques to critical components need to be fully implemented because the guillotine break of these components, especially for the first generation of RBMK NPPs where the accident localisation systems have limited capabilities, would have major impact on the plant safety. Priorities for research include:

- Development of material data base for the application of LBB concept to piping elbows (with respect to peculiarities of geometry and technology) and equipment casings (T-joints of branches, zones of perforation, etc.).
- Development of procedures for experimental determination of the crack growth resistance characteristics of the 80-mm diameter piping butt weld materials with respect to peculiarities of assembling and repair technologies.
- Development and validation of codes for the dynamic analysis of piping systems and structures.
- Improvement of leak detection systems.
- Development of monitoring systems for assessing the condition of the metal.
- Development of integrity analysis of pressure tubes in case of water hammer due to the shut-down of the check valves.

In order to avoid duplication, adequate consideration should be given to on-going relevant research: The LBB concept is applied to drum separators of SGHWR in United Kingdom. Development of procedures for experimental testing of crack growth resistance characteristics is being carried out in Russia. Codes for dynamic analysis and piping systems and structures have been developed in USA and Japan. A leak detection system using microphones has been developed in Japan, in co-operation with Russia, to detect a small leakage of coolant.

Additional Development of Inspection Technology

There are research needs for additional development of inspection technology. Included are the development of advanced inspection methods and systems for ultrasonic inspection with capabilities for flaw sizing.

Seismic Analyses

The major safety concern is the seismic stability of the RBMK drum separators (with connected piping) and its interaction with the supporting metal structures. Therefore, a priority for safety research is the development of an analytical model for the seismic assessment of the complex "drum separator-connected piping-support structure" system.

Integrity of RBMK Fuel Channels and Reactor Cavity

Development of Inspection Technology for Fuel Channels

Inspections are performed on fuel channels at all plants using visual and ultrasonic methods. Volumetric inspections of fuel channels would be useful to analyse the development of subcritical cracks. Non-destructive examinations (NDE) of fuel channels by ultrasonic techniques or other methods is only now being implemented at all plants, and only a small number of fuel channels will be inspected when it is implemented.

Continued inspection of fuel channels should focus on the development of inspection techniques for the diffusion bonded joint between the zirconium pressure tube and the Ti-stabilised stainless steel end-pieces. These inspections should be able to monitor stress corrosion cracking of the stainless steel portion of this joint.

Additional work is needed to improve the accuracy and function of the fuel channel inspection. One should be able to evaluate in-situ the changes of material properties and deformation of the fuel channel under reactor operating conditions. The development of the corresponding NDE methods, including the appropriate software, remains a high priority issue for the fuel channel.

To avoid duplication, adequate consideration should be given to on-going relevant research, like the fuel channel inspection technologies developed in Japan, Canada and Russia.

Integrity of Reactor Cavity under Emergency Conditions

Accident scenarios which could potentially lead to multiple fuel channel ruptures have been recognised as a major safety issue for the RBMK. This is because any multiple rupture of the fuel channels which exceeds the venting capacity of the reactor cavity over-pressure protection system poses a major impact on plant safety. The following research is necessary to assess the performance of the suppression system and the resistance capability of the reactor cavity to multiple rupture of the fuel channels:

- Experimental study on the performance of suppression system and integrity of reactor cavity under multiple fuel channel rupture.
- Development and verification of thermal-mechanical codes for multiple fuel channel ruptures.

- Development and verification of analytical models for loading of the cavity due to rupture of the fuel channels.
- Development and verification of analytical models for fuel channel loading and fuel handling during seismic excitation.

Development of Evaluation Methods for Ageing of Components, Piping and Fuel Channels handling during seismic excitation.

Evaluation of ageing is important for plant operation, especially for the short term evaluation of pressure boundaries in old plants. Priorities for research include:

- Development of evaluation methods for the ageing of critical RBMK components, such as degradation of fuel channel material or deformation of fuel channel.
- Development of metal condition monitoring in destructive and non-destructive examination.

In order to avoid duplication adequate consideration should be given to on-going relevant research, like the material tests on fuel channels conducted in Japan, Canada and Russia.

9. SEVERE ACCIDENTS FOR RBMKS

Safety Concerns

The design of RBMK reactors does not include a large, strong containment surrounding the whole nuclear steam supply system. Thus, compared with western PWRs for example, RBMK reactors do not benefit from the very large retention of fission products during most severe accidents that is provided by such a passive system. The strategy for minimising fission product release from a severe accident must, therefore, rely on active management of the accident to reduce its severity and to bring the reactor to a cooled, stable state as soon as possible. Some accident management procedures have been developed but a sound technical basis for severe accident management has not yet been developed. Safety research should aim at providing this sound technical basis for severe accident management.

It is important to note that for many accident sequences there is a long time period between going beyond the design basis and the onset of severe fuel damage. During this period, the plant undergoes what is essentially a severe thermal-hydraulics transient. Active management by plant operators may terminate the accident before severe fuel damage occurs. The safety research issues raised are thermal hydraulic and thus covered by Chapter 5 as an extension of design basis thermal-hydraulics research issues. In the present Chapter, the issues considered are those raised from the onset of severe fuel damage.

Open Issues

The open issues derive from the need to manage severe accidents and from the recognition that the opportunity for accident management depends on the type of accident. Research is only justified if there are significant potential benefits in terms of application to plant, in particular to severe accident management. The greatest benefits will come from operator actions that terminate core degradation. Thus, safety research on degraded core issues has priority. The potential for significant fission product retention in pipework or the reactor building also warrants research. However, there is little that can be done to enhance retention by operator action during an accident so research should have less priority.

Whole core severe accidents in RBMK reactors can be divided into three classes:

Firstly, reactivity insertion events that develop fairly quickly and can lead to large fission product releases on a short time scale. There is little chance of managing these accidents once the severe accident phase has been reached. The correct approach is prevention. Considerable work has been and is being carried out to address this.

The second class of severe accidents are characterised by loss of heat sink and progress much more slowly. This is because of the large heat sink provided by the graphite moderator which slows down the heatup of the fuel (heatup is much slower than in pressure vessel-type reactors such as VVER). There is therefore a relatively long time available to intervene in the accident after the onset of severe fuel damage. It is therefore important to ensure that all reasonable and beneficial actions are known and understood and that potentially harmful actions are known and avoided.

The third class is characterised by beyond design basis loss of coolant accidents involving, for instance, a pipe rupture with further multiple failures. This is essentially a thermal hydraulics problem which raises questions such as the adequacy of emergency cooling systems. If cooling systems are not adequate and fuel heats up and degrades, the scenario becomes similar to the second class of accidents (loss of heat sink).

Severe accident research should, therefore, focus mainly on the second class of (slowly developing) whole core accidents, with the emphasis on providing the technical basis for the development of practical severe accident management actions. This has the potential to reduce significantly the risk from severe accidents in RBMK reactors.

While the main qualitative features of severe fuel damage progression are likely to be common to most whole core accidents (other than rapid reactivity insertion events), there may be important quantitative differences due to the different initiating events. It is therefore desirable to establish the relative frequencies of the various types of initiators and their contribution to overall risk so that research is focused on the most risk significant.

In addition to whole core severe accidents, it is possible for severe fuel damage to occur in a single channel due to blockage of coolant flow in the channel caused by channel failure. Fuel damage will start fairly soon after the blockage, but the existing Accident Localisation System (ALS) will operate to localise the consequences of the accident. Three cases of channel tube rupture have occurred during RBMK operation and damage did not propagate to neighbouring channels. Nevertheless, there may be circumstances in which propagation to neighbouring channels would occur. Further research is needed to establish whether this is possible, what the consequences are for fuel damage and how best to manage such accidents.

In all classes of accident, fission products may be retained in the pipework and reactor buildings. The main open issue is whether this retention is significant (reducing release to the environment by, say, a factor of 10). Fission product retention would be much reduced if the reactor compartment fails. The threat posed by overpressurisation during a severe accident warrants research.

A small amount of severe accident research is underway in Russia. However, this is at a very early stage and considerably more work is needed to provide a sound technical basis for severe accident management. There is no research on RBMK severe accidents in Western countries.

Safety Research Needs

The question here is to identify in broad terms where research is needed and what it might be.

Existing research on fuel heatup and degradation for pressure vessel-type reactors (VVER, PWR, BWR) has some relevance in that the early stages involve heatup of oxide fuel in zirconium-based clad in an essentially rod-like geometry. However, the long, thin RBMK fuel bundles in individual pressure tubes with surrounding graphite will subsequently degrade very differently from the shorter, close packed fuel bundles in the "open channel" designs typical of pressure vessel reactors. Furthermore, the graphite moderator will have a great influence on core degradation, making the course of a severe accident very different from that in a pressure vessel-type reactor. There is thus a big uncertainty in even the basic phenomenology of severe accidents in RBMK reactors. This translates into an even bigger uncertainty in how to manage a severe accident.

Research should aim to build on the existing Russian work along with relevant pressure vessel reactor research and Western work on pressure tube and graphite moderated reactors. The first, and most important step, is to identify and understand the main phenomena. Then it is possible to address key phenomenological questions such as:

- How does an RBMK fuel channel (fuel + pressure tube + graphite) degrade?
- Is a total, uncoolable channel blockage due to degraded fuel possible or likely?
- How do chemical and physical interactions with the graphite affect degradation of the fuel channel?
- Can severe fuel damage in one channel propagate to other channels?
- What happens when water is reintroduced into a degrading channel?
- Can core debris be cooled if it falls onto the concrete below the core?
- What are the dominant processes for fission product trapping in reactor pipework and buildings?

Preconceived ideas based on the established phenomenology of severe accidents in pressure vessel-type reactors may be misleading. Thus, an important aspect of this research should be to identify significant differences and their impact on accident progression.

Existing Western and Eastern experimental data on fundamental processes provides a good starting point for addressing the phenomenological questions. These range from simple materials properties and interactions tests to fairly complex fuel bundle experiments carried out in pressure tubes. Some RBMK-specific separate effects experiments are likely to be needed for novel features of the RBMK reactor (*e.g.* fuel channel/graphite behaviour). For fission product retention, generic experiments already carried out (*e.g.* FALCON) or planned (*e.g.* PHEBUS-FP) in the West are likely to be adequate for RBMK. It is recommended that a review of existing experimental data for degraded core and fission products be carried out with the aim of identifying data that are useful for RBMK.

When phenomenology is adequately understood, plant related research questions concerned with accident management can be addressed. These include:

- What are the time scales to key events such as channel failure or propagation of a single channel accident to adjacent channels?
- What can be learned from existing instruments about the progress of a severe accident?
- Can a degrading core always be cooled by restoring coolant water flow?

- How can we know that a severely damaged core is cooled and stable?
- Can a severely damaged core become critical again?
- Will the reactor cavity fail and, if so, after how long?
- What fraction of fission products is retained in pipework or the reactor building?
- What is the impact of graphite on the long term phase of a severe accident, particularly the long term fission product release?

As with phenomenology, preconceived answers to plant related questions derived from experience with pressure vessel reactors may be misleading. For instance, adding water to a degrading core may not be desirable in all circumstances since reaction with the graphite moderator might make the accident worse. Care should be taken to focus on where there is greatest potential benefit and not go into unnecessary detail.

Answers to some of these questions, particularly those providing information for accident management decisions, will need the development of models suitable for plant calculations. Model development should focus on producing a simple, flexible tool (or set of tools) for addressing plant related questions, particularly for accident management. This implies the development of simple, parametric codes. These have the greatest potential for providing answers to plant related questions as they can be used to provide reasonable bounds for important plant parameters such as time scales or temperatures. More complex “best estimate” codes should be reserved for detailed calculations to benchmark simpler codes and for whole-core thermal hydraulics where multi-channel behaviour must be modelled. Both types of code can be based on existing pressure vessel reactor codes. Parametric codes should be validated for consistency with the experimental database, benchmarked against detailed calculations with “best estimate” codes and assessed for reasonable extrapolation of models to plant scale.

Confidence in results will come mainly from the quality of the individual phenomenological models built into the codes. However, at least a few integral experiments involving overheating fuel bundles are desirable to ensure that interactions between basic processes are adequately represented in the codes. Integral experiments are likely to require a new or modified rig. The scope and scale of the rig would depend on the results of the phenomenological research. Key parameters such as the maximum temperature needed must be determined by phenomenological research. For instance, if major core damage occurs mainly through fuel liquefaction, higher temperatures will be needed than if it occurs mainly through solid fragmentation and fuel bundle collapse. Such parameters will therefore have a significant impact on cost and technical feasibility.

10. CONCLUSIONS AND RECOMMENDATIONS

The OECD Support Group on Safety Research Needs for Russian-Designed Reactors, composed of senior experts from OECD countries and Russia, have agreed on the following conclusions.

General Conclusions

Importance of VVER and RBMK Safety Research

The aim of nuclear safety research is to provide information to plant designers, operators and regulators in support of the resolution of safety issues, and also to anticipate problems of potential significance. Better understanding of the phenomena that have an influence on reactor safety has been one of the major contributors to the improved assurance of nuclear safety.

Within the scope of this report, the intention is to identify the research needs in a general way: the detailed description of a work programme is part of the definition of future projects.

The emphasis of the study is on the VVER-type reactors in part because of the larger base of knowledge within the NEA Member countries related to LWRs. For the RBMKs, the study does not make the judgement that such reactors can be brought to acceptable levels of safety, but focuses on near term efforts that can contribute to reducing the risk to the public. The need for the safety research must be evaluated in the context of the expected lifetime of the reactors.

The most important near-term application of safety research for both VVER and RBMK reactors is to establish the technical basis for symptom-based accident management procedures that can prevent or halt progression of accidents, and to plan for future safety improvements. For RBMK, there should be special emphasis on safety research that contributes to this technical basis in order to realise the benefits of the results in a shorter time frame.

Avoiding Isolation and Technology Gaps

To the extent that technical isolation of safety experts contributed to the current situation, co-operation between Western and Eastern safety experts will be advantageous. Western safety technology, although at a high level, continues to improve. Therefore, sufficient resources must be devoted to bringing the designers, operators and regulators of Russian-designed reactors into the development of the next generation of safety technologies to assure that any significant gaps are closed in a reasonable time period.

Safety Research will Make a Difference

There is every expectation that the safety research recommended in this report will make a difference to public safety if it is carried out successfully and applied. The precedent for this conclusion is found in the record of accomplishments of safety research in OECD countries. This record is described in the section of the report on Uses of Safety Research. The research recommended runs parallel to work that has been successfully applied in OECD countries.

RBMK Safety Research at an Earlier Stage of Development

Because of the similarity of Western PWR and Russian VVER technology, and widespread use across the world, there is far more experience with PWR/VVER technology than there is with RBMK technology. This common experience explains the greater level of detail in the research needs identified for the VVERs compared to the RBMKs. RBMK research needs start from a smaller base and are, therefore, at an earlier stage of development, although some experience of Western pressure tube type reactors is applicable to RBMK. Thus while the broad areas of research needs are similar for both reactor types, specific detailed needs may be more advanced for the VVER.

Technical Conclusions

Operational Safety Assessment

It is almost impossible to overstate the importance of human performance to improving the safety of reactor operations. For existing reactors, operational safety research is a high priority since there are usually fewer barriers to improvements in this area than there are to modifying the design of the plant. Further, it is often possible to achieve significant risk reductions because these existing reactors tend to be more demanding in terms of human performance than newer designs.

Research for improving operational safety is extremely important for VVERs and RBMKs alike. It includes work for the purpose of human error reduction, development of PSA data bases for Russian-designed plants, including equipment reliability and human performance, development of accident precursor methodology, improving condition monitoring of safety related equipment, improving the I&C reliability data base and set point methods, and developing technical bases for emergency operating procedures for Russian-designed plants.

VVER Design-Basis Thermal-Hydraulics and Reactor Physics

Thermal-hydraulic phenomena and reactor kinetics govern normal operation, Design-Basis Accidents, and Beyond Design-Basis Accidents of VVERs. Safety improvement, including development of Accident Management procedures, depends in turn on quality of safety analysis, and validated thermal-hydraulics and reactor kinetics codes. Research needs in this area should focus on full validation of codes for specific VVER features. Extension of these codes to Accident Management procedures is a key area for improving reactor safety. Experiments should be done as necessary to validate codes used for VVER safety evaluation.

For much the same reasons, research is needed to validate containment codes. Specific research (integral and separate effect tests) is needed to verify the bubble condenser system performance.

Integrity of VVER Equipment and Structures

Integrity of reactor coolant boundary and leak-tightness of containment/confinement are both necessary, and methods of assessing integrity and leak-tightness must be verified. To ensure integrity of the pressure vessel it is necessary to extend the data base on material properties of material in and near the weld zones subject to irradiation for VVER-440s and VVER-1000s, and to address specifically the re-embrittlement condition of VVER-440 pressure vessels after one or more irradiation/annealing cycles.

Improved models of pressure boundary components and containment structures are needed for assessment purposes, and experimental data are needed for their verification. In addition, it is necessary to develop improved NDE methods for monitoring condition of actual material properties of components, in order to ensure safe assessment of remaining life of these components.

VVER Severe Accidents

The most important task is to improve the safety assessment of operating plants. For this purpose research is necessary to improve codes, to validate Accident Management procedures, and to quantify safety margins. In spite of the large efforts spent in the OECD countries during the last decades there is still a need for well-focused research on specific issues. Generally, the research needs for Russian-designed reactors are coincident with these issues, but work related to unique VVER design features is also needed : to develop data bases for VVER materials properties and interactions, work on Beyond Design-Basis Accident containment response, and work on VVER source term modelling and hydrogen safety.

RBMK Design-Basis Thermal-Hydraulics

As with the VVER, safety improvements depend on quality of safety analysis, and validated thermal-hydraulic and reactor kinetics codes. To achieve these ends, safety research is necessary to improve the technical basis employed to develop safety criteria relative to fuel and fuel channel failures and post-accident hydrogen distribution. On the analytical side, research is needed to validate integral best-estimate thermal-hydraulic codes, to improve the accuracy of the neutronic data base, and to couple neutronic and best-estimate thermal-hydraulic codes. In the case of containment thermal-hydraulics and the reactor cavity, the research need is improvement and validation of best-estimate codes that describe performance in the respective case.

Integrity of RBMK Equipment and Structures

Integrity of the primary coolant circuit, and especially the fuel channel, is a major safety issue for the RBMK. Methods of assessing integrity must be verified. There are urgent safety research needs to address in this issue. They include the following: development of improved In-Service Inspection technology; development of improved monitoring and assessment of fuel channel integrity for effects of ageing; development of analytical models for rupture of fuel channels; and development of analytical models for response of fuel channels to loading due to fuel channel rupture, and to seismic loading. Experimental data are needed for materials properties, for assessing integrity, and for verification of analytical models.

RBMK Severe Accidents

Prevention of accidents is the best way to reduce RBMK risk. In severe accident conditions it is necessary to rely on Accident Management to reduce the severity, because there is no containment surrounding the whole reactor system. In the case of severe fuel damage caused by fast reactivity insertion, there is insufficient time for Accident Management. This underlines the importance of preventing or limiting Reactivity-Initiated-Accidents. In the cases of loss of heat sink and Beyond Design-Basis Loss-of-Coolant accidents, there is time for Accident Management to be effective in reducing accident severity. For these purposes, development of simple, physical models and parametric codes and their systematic use in plant specific analysis are necessary. These models and codes should be based on OECD and Russian data bases and, where necessary, additional separate effects and integral experiments specific to RBMKs, *e.g.,* for fuel overheating, and zirconium-graphite interaction.

Recommendations For Safety Research

Steps to Make It Happen

We recommend preparation of a Safety Research Strategic Plan that sets goals, defines products required of safety research, and describes how and when work will be implemented. Such a plan also establishes research priorities based on the needs of the users of safety research results. In effect the plan becomes the organising principle for the safety research programme, and it provides answers to the questions that funding authorities can be expected to ask when they decide how to allocate their funds.

There is more to making safety research happen than preparing a list of topics or writing proposals. This report on Safety Research Needs can serve as a starting point for preparation of an appropriate strategic plan.

Steps to Make It Effective

We recommend that key players in the Eastern nuclear community be involved in the planning process, the execution of research, and the application of results to improving safety. The key players are officials in government for energy and safety regulation, nuclear plant owner/operators, reactor design and construction organisations, and safety research organisations. Involvement of key players is essential to planning research, getting work underway, carrying it through, and effectively applying results.

The Role of International Co-operation

Nuclear safety is typically an international issue. International collaboration takes many forms, including sharing information, experience and resources, and has long been an important feature of nuclear safety research. The pressure on research budgets and the commonality of objectives and interest on research results are increasing this need. Co-operation with the Eastern countries has substantially improved in the last few years, especially in the field of reactor safety and safety research. The report has identified considerable scope for international co-operation in the research needs for Russian-designed reactors. Therefore collaboration should be increased if we are to preclude the technical isolation of the current situation. It must be stressed, however, that reliance on international research cannot be a substitute for healthy national programmes. There is a level of effort under which national programmes become ineffective even if they are invigorated by international collaboration.

We recommend international co-operation in reactor safety research. It brings about sharing of knowledge and technical experts as well as funding. It offers the possibility of bringing together the best means to work on problems world-wide rather than solely on national bases. This can only improve quality of work and results.

Safety Technology Transfer

In view of large amount of research information available in OECD countries and the potential applicability for addressing safety concerns of Russian-designed reactors, efforts should be dedicated to finding and implementing new approaches to share the information and assess its application to Eastern reactors. Such approaches, e.g., a forum for a specific technical topic, should enhance the efficiency of the information transfer and reduce the potential for duplication.

ANNEX 1

MEMBERS OF THE OECD SUPPORT GROUP ON SAFETY RESEARCH NEEDS FOR RUSSIAN-DESIGNED REACTORS

Dr. Eric S. Beckjord, Chairman, Consultant, United States of America
Dr. R. Allan Brown, AECL, Canada
Dr. Heikki Holmstrom, VTT Energy, Finland
Dr. Michel Reocreux, IPSN, CEA, France
Mr. Helmut Schulz, GRS, Germany
Mr. Klaus Liesch, GRS, Germany
Ing. Giampiero Santarossa, ENEA, Italy
Dr. Yoshitaka Hayamizu, PNC, Japan
Dr. Vladimir Asmolov, RRC Kurchatov, Russia
Prof. Leonid Bolshov, IBRAE, Russian Academy of Sciences, Russia
Dr. Sergei Bougaenko, RDIPE/NIKIET, Russia
Dr. Yuri N. Nikitin, RDIPE/NIKIET, Russia
Dr. Alexandre Potapov, ENTEK/RDIPE, Russia
Dr. Vladimir Proklov, NSI Kurchatov, Russia
Dr. Valerii Strizhov, IBRAE, Russian Academy of Sciences, Russia
Dr. Leonid. M. Voronin, VNIIAES, Russia
Dr. Stephen R. Kinnersly, AEA Technology, United Kingdom
Dr. John R. Honekamp, PNL, United States of America

Secretariat:
Dr. Gianni M. Frescura, Nuclear Safety Division, OECD/NEA
Mr. Nobuo Maki, Nuclear Safety Division, OECD/NEA
Mr. Javier Reig, Nuclear Safety Division, OECD/NEA
Dr. Herbert E. Rosinger, Consultant

ANNEX 2

TASK LEADERS AND ADDITIONAL CONTRIBUTORS TO THE CHAPTERS IN THIS REPORT

3. Thermal-Hydraulics/Plant Transients for VVERs. **Task Leaders**: Dr. Michel Reocreux, France, and Dr. Vladimir Proklov, Russia. **Additional Contributors**: Dr. A. Suslov, Dr. I. Elkin, Dr. A. Devkin, Dr. P. Alekseev, Dr. M. Lizorkin, Dr. A. Ephanov and Dr. L. Yegorova; Russia.

4. Integrity of Equipment and Structures for VVERs. **Task Leaders:** Dr. Sergei Bougaenko, Russia, and Mr. Helmut Schulz, Germany. **Additional Contributors**: Dr. Getman, Dr. Dragunov, and Dr. Vasiliev, Russia.

5. Severe Accidents for VVERs. **Task Leaders:** Mr. Klaus Liesch, Germany, and Dr. Valerii Strizhov, Russia. **Additional Contributors**: Dr. A. Efanov, Dr. S. Dorofeev, Dr. G. Taranov, and Dr. M. Veshchunov, Russia.

6. Operational Safety Issues. **Task Leaders:** Dr. John R. Honekamp, United States of America, and Dr. Leonid. M. Voronin, Russia. **Additional Contributors**: Dr. M. Zentner and Mr. V.M. Vitkov, Russia.

7. Thermal-Hydraulics/Plant Transients for RBMKs. **Task Leaders:** Dr. R. Allan Brown, Canada, and Dr. Yuri N. Nikitin, Russia. **Additional Contributor**: Mr. C. Blahnik, Canada.

8. Integrity of Equipment and Structures for RBMKs. **Task Leaders:** Dr. Sergei Bougaenko, Russia, and Dr. Yoshitaka Hayamizu, Japan. **Additional Contributor**: Mr. A. Arjaev, Russia.

9. Severe Accidents Research Requirements for RBMKs. **Task Leaders:** Dr. Stephen R. Kinnersly, United Kingdom and Dr. Yuri N. Nikitin, Russia.

ANNEX 3

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