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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

NEA/CSNI/R(2006)6
Unclassified

DRAFT PILOT REPORT

APPROACHES TO THE RESOLUTION OF SAFETY ISSUES

English - Or. English

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

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), Korea (12th December 1996) and the Slovak Republic (14 December 2000). 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 27 OECD Member countries: Australia, Austria, Belgium, Canada, Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its Member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

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

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers. It was set up in 1973 to develop and co-ordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries.

CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of its programme of work. It also reviews the state of knowledge on selected topics of nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus in different projects and International Standard Problems, and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups and organisation of conferences and specialist meeting.

The greater part of CSNI's current programme of work is concerned with safety technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment and severe accidents. The Committee also studies the safety of the fuel cycle, conducts periodic surveys of reactor safety research programmes and operates an international mechanism for exchanging reports on nuclear power plant incidents.

In implementing its programme, CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

Executive Summary

The purpose of this report is to present in a concise form how some safety matters associated with currently operating light water reactors have been addressed. The issues discussed in this report are common to member countries with currently operating LWRs (PWR, BWR, VVER) and, as such, have wide interest in the nuclear safety community. Accordingly, this report can serve as a reference for researchers, regulations and others (e.g., industry) interested in understanding the approach and status of issues. This report should also be useful for knowledge transfer by documenting what has been done or is planned regarding selected safety matters and as a source for identifying reference material containing additional detail.

The issues addressed in this report should not be viewed as questioning the safety of operating reactors, which have reached very high operational safety record, but rather as areas where uncertainty in knowledge exists, where safety assessment has been based on conservative assumptions, and where regulatory decisions need, or will need to be confirmed. Thus, the development of sound technical bases through continuing research will improve the current knowledge and allow for more realistic safety assessment.

The safety issues discussed in this initial version of the report are:

- design basis accident spectrum
- severe accident issues
    - reactor pressure vessel integrity
    - hydrogen control
    - containment integrity
    - accident management
- station blackout
- high burnup fuel
- power uprates
- ECCS strainer clogging
- boron dilution.

For each issue, the scope of the issue is defined, its status discussed and planned work or research described, including schedule. This pilot version of the report is limited to input from nine countries (Belgium, Czech Republic, Finland, France, Germany, Japan, Korea, Sweden and the U.S.). An overview of this information for each issue by country is provided in the table.

This document doesn't contain a comparative analysis of the status of resolution of the safety issues in the member countries as it would require additional time and manpower. Nonetheless, it can be maintained as a living document and updated periodically as needed.

Summary Table

Issues	Belgium	Czech Rep.	Finland	France	Germany	Japan	Korea	Sweden	U.S.
Design Basis Accident Spectrum	Use NRC Reg. Guide 1.70	Use NRC Reg. Guide 1.70	Based on national YVL-guides	Based upon US practice	Guideline documents issued by safety authorities contain list of accidents which must be analyzed. New DBAs are under consideration	Nuclear Safety Commission guides used	Use NRC Standard review Plan NUREG-0800	Use NRC Reg Guide 1.70	Reg. Guide 1.70
Severe accidents: RPV integrity	Reactor Cavity Flooding	Reactor Cavity Flooding under study	VVER Auto depressurization and in-vessel retention of molten debris by external cooling BWR Auto Depressurization and molten core cooling by lower drywell flooding	Depressurization capability provided	Depressurization system for BWRs and PWRs	Auto depressurization In-vessel and ex-vessel water injection	Auto depressurization In-vessel water addition	Strengthening of existing water sources i. e. CRUD flow, and adding new water sources being studied	Reactor Cavity flooding in some plants Auto depressurization in BWRs
H_2 control	Passive auto-catylitic recombiners	Under study	VVER Passive Autocatalytic recombiners and ignators in lower part of containment BWR Innerted containment	To install passive auto-catalytic recombiners by 2007	- Inert atmosphere-BWRs (line 69) - catalytic recombiners in drywell, BWRs (line 72) - passive autocalitic recombiners - PWRs	Ignitors in Ice Condenser Containments	Under study		inert containment atmos - MKI and II ignitors - MKIII and IC
Containment integrity									
Accident Mgt.									

	Addressed via accident mgt.	Addressed via accident mgt.		Install of filtered vents	Filtered vent system	Addressed via accident mgt	Addressed via accident mgt.		Hardened vents - MK-I
Station Blackout	Coping capability required	No special provisions - rely on large water inventory of VVER	Coping capability required	Coping capability required	Additional power supplies added	Coping capability required	Coping capability required	Core and containment cooling by mobile equipment (vehicles)	Coping capability required
High Burnup Fuel	Assembly average burnup is limited to 55GWd/t. Experimental data needed to support higher burnups	Burnup increases being sought, will follow USNRC criteria	Max assembly burnup limited to 45 MWd/kgU. Higher burnup required experimental justification	Burnup increases expected. Experimental data being developed.	Currently under review Experimental supporting data needed.	Assembly max. burnup is 55 GWd/t. Confirmatory research underway	Characteristics to be added	Limited by national RIA-limits. These limits and under review	Peak rod average burnup up to 62 GWd/t approved. Experimental data being analyzed to support higher burnups.
Power Uprates	Power uprates up to 10% approved.	Power uprates of 9,5% sought. To be reviewed in steps in 2005-2012.	Power uprates approved and implemented, 25% for BWRs and 9% for VVER			No uprates requested to date		Power uprates are being sought, 20-30% for BWR and 13-18% for PWR. To be installed 2006-2010	Power uprates being approved. -BWR- 20% -PWRs- 5%

ECCS Strainer Clogging	Under review. Participating in international research programs.	Strainer modifications made. Continuing to participate in international research.	Strainers have been modified for both VVERs and BWRs	Under review	Under review	Under review. Expected to take 3-4 years		New strainers with much larger area and in-service cleaning capability installed both for PWR and BWR	Under review. Research underway.
Boron Dilution	Under review. No resolution prior to completion of PKL experimental program in 2007.	Resolved via analysis and procedural actions.	Modifications in the systems and administrative procedures have been done for avoiding an unintended boron concentration dilution of the coolant:	Under review.	Current information indicates no criticality concern. Confirmatory work underway.	Under review. Expected to be resolved in 2007		Prevention by administrative measures (operator instructions for detection and pump start up)	Under review. Resolution expected in 2004-2005.

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Foreword

One of the fundamental goals of the CSNI and CNRA is to share information regarding nuclear safety. This is done through meetings, workshops and various types of reports and communications. These interactions help ensure that a broad base of knowledge is considered in the assessment of safety issues and they also contribute to improved efficiency, effectiveness and consistency in their resolution. To help the sharing of a broad base of knowledge, the CSNI Programme Review Group (PRG) has initiated the production of a pilot reference document for researchers, regulators and others (e.g., industry) summarizing how various countries have addressed selected safety matters.

The issues addressed in this report should not be viewed as challenges to operating reactors, which have reached very high operational safety record, but rather as areas where uncertainty in knowledge exists, where safety margins are based on conservative assumptions, and where regulatory decisions need, or will need to be confirmed. Thus, the development of sound technical bases through continuing research will improve the current knowledge and allow for more realistic safety margins.

The objective of this report is to document the status of work completed or planned and the approach taken in the resolution of the selected safety matters, thus providing an overview of how the matters are being addressed and a perspective with which to judge the need for any additional work. This summary can also be useful for knowledge management and transfer by providing in one place summary information and references.

The scope of this "pilot" report is limited to safety issues associated with currently operating light water reactors (LWRs), which include PWRs, BWRs and VVERs. The safety issues addressed in this pilot version of the report are:

- design basis accident spectrum
- severe accident issues:
  - reactor pressure vessel integrity
  - hydrogen control
  - containment integrity
  - accident management
- station blackout
- high burnup fuel
- power uprates
- ECCS strainer clogging
- boron dilution.

Input to this pilot report is limited as it includes descriptions from nine countries: Belgium, Czech Republic, Finland, France, Germany, Japan, Korea, Sweden and United States.

The format chosen is that for each identified safety issue the issue is defined and each participating country describes in 1-2 pages how the issue was addressed including the schedule. However, the inputs by various countries in some cases differ significantly, reflecting the situation in a given member country.

It was not an objective of this pilot report to include a comparative analysis on the status of the safety issues in the member countries. Such an analysis, although it might be useful in understanding the differences in national approaches, would require additional time and manpower. This, if needed, can be done by the relevant CSNI working groups.

This document can be maintained as a living document. It may be updated periodically so that its content is kept current and additional issues and member country information may be added as requested by the CSNI/CNRA.

Chapter I

Issue: Design Basis Accident Spectrum

Issue description

Design basis accidents (DBAs) are those that are generally used to establish the functional and performance requirements for safety-related systems in nuclear power plants. The spectrum of DBAs covers a range from events that are expected to happen to those that are not expected but against which protection must nonetheless be provided by plant design and operation.

Analysis of these and other DBAs are required as part of the Safety Analysis Report to demonstrate that the plant can meet established regulatory limits. Examples of such limits include peak clad temperature, cladding oxidation, and release or radioactivity to the environment.

1.1 BELGIUM

1.1.1 Current regulatory approach

The initial license of each plant has been given on the basis of a deterministically defined list of accidents to be considered in the design. Given the decision to use the USNRC rules and regulations, that list was the one given in the RG 1.70 and in the Standard Review Plan available at the time of licensing. Some site specific external hazards were added to the list.
The acceptance criteria were those given in the USNRC regulatory documents, with the exception of the radiological consequences. Given the high population density in Belgium, stricter limits were imposed.
In the frame of the periodic safety reassessment required every ten years, the status of each plant is compared with the most recent safety standards in force in the USA and in the European Union. As a result, the design basis spectrum has been increased.
When the licensee performs reanalyses of design basis accidents (e.g. in the frame of a request to increase the power), more advanced (i.e. more best-estimate) codes and methods are often used. This is reviewed on a case by case basis by the regulatory body.

1.1.2 Technical basis supporting the regulatory approach

Given the decision to use the USNRC rules and regulations, the technical basis is the same as in the US.

1.1.3 Status of resolution

The definition of the design basis accident spectrum (including assumptions, models and acceptance criteria) is not an unresolved issue in Belgium, with the exception of the steam generator tube rupture (SGTR). Because of the strict limits for the radiological consequences of accidents, and because some physical phenomena (e.g. atomisation) were not considered initially, the present conservative calculations of the releases due to a SGTR seem to exceed the limits. A reassessment of the whole method to assess the SGTR is ongoing.

1.1.4 Research needed to support modifications

One of the key factors in limiting the releases in case of SGTR is the fission products retention in the separators and dryers of the steam generator. The ability of those equipment to trap particles and droplets is unknown in most conditions.

1.1.5 Schedule for completion of research and regulatory action

The study of the retention factor of the separators and dryers of steam generators is the main objective of the ARTIST project currently under way at the PSI in Switzerland. The completion of the programme is scheduled for 2007.

1.2 CZECH REPUBLIC

1.2.1 Current regulatory approach

The legal requirement for preparation and submission of Safety Analysis Reports (SARs) during the licensing process for nuclear facilities in Czech Republic was established for the first time in 1976 (Construction Act No 50) and in the subsequent legal documents - Act No 28/1984 on state supervision of nuclear safety of nuclear facilities and Act No 18/1997 on peaceful use of nuclear energy.

The licensing process for nuclear facilities in Czech Republic has three basic steps: site approval, construction permit and operational license. For each of the above steps the specific SAR (siting, preliminary, pre-operational-FSAR is required to be submitted to the regulatory body – State office for nuclear safety (SONS). The design basis accident analysis is a part of the latter two steps.

Recommendation of the regulatory body No 5/88 has served as a guidance document for preparation of SARs. The content of documents No 5/88 and RG 1.70 is similar, but RG is offering much more detailed instruction, acceptance criteria etc. Currently the regulatory body requires to follow the Regulatory Guide 1.70 of NRC.

Detailed content of particular SAR has been, as a rule, negotiated between SONS and NPP.

Since 80-ties the regulatory body has established the process for verification and validation of computer codes used for safety analysis in SARs. The established expert commission has 6 groups (neutron physics, termohydraulics, fuel behaviour, severe accident, strength of components and piping, radionuclide distribution). By means of the above process many computer codes have been approved by SONS for the use in SARs.

1.2.2 Technical basis supporting regulatory approach

The problem with preparation of SARs in 70-ties and 80-ties consisted of two reasons: the practice in the former USSR regarding preparation of SARs was rather different from Western practice and there were not domestically available verified computer codes for various types of accident analysis.

This situation has been step by step improved by using modern Soviet codes, by development of Czech codes and implementation of Western codes, including their verification and validation for WWER.

1.2.3 Status of resolution

NPP Dukovany

The FSAR was updated and revised during Periodic Safety Review (PSR) after 10 years of operation (1994-96) and the guidance document No 5/88 was used for its preparation. The Chapter 15 of FSAR was prepared by Czech organizations (NRI Rez, ŠKODA, Energoprojekt) by use of Russian, US, Czech and German computer codes. In connection with PSR after 20 years of operation (2005-6) and with the replacement of plant I/C system and fuel of new type (with Gd), the updated SAR will be prepared according to RG 1.70.

NPP Temelín.

The FSAR was elaborated according to RG 1.70 in the framework of NRC assistance project „Transfer of NRC approach to SONS". The Chapter 15 of FSAR – Safety Analysis was prepared by Westinghouse with the use of US computer codes and Czech computer codes.

In addition to the list of DBA provided in RG 1.70 the following events were analysed: rupture of bundle (more than 3) of SG pipes and SG manifold header break.

1.2.4 Research needed to support modifications

Since the computer codes for safety analysis of WWERs have not been sufficiently experimentally verified, the launched NEA/OECD project PBS – WWER, which is performed in experimental facility EREC in Elektrogorsk (Russia), is for the Czech Republic operating WWER-1000, of a great value.
Similar importance has the experimental verification of bubble condenser system (WWER-440/213) on experimental facilities in Russia, Czech Republic and Hungary by means of PHARE project.
On the basis of Agreement between SONS and US NRC on exchange of technical information and cooperation in the area of nuclear safety the Agreement on thermohydraulic code applications and maintenance (CAMP) is in force and Czech experts has participated regularly at CAMP meetings since 1995.

1.2.5 Schedule for completion of research and regulatory action

Schedule depends of the progress of the above mentioned NEA/OECD and PHARE projects.
The Czech participation in CAMP will continue.

SONS APPROVED IN 2003 RESEARCH PROJECT ON EVALUATION OF UNCERTAINTIES OF BEST ESTIMATE COMPUTER CODES FOR WWERS.

1.3 FINLAND

1.3.1 Current regulatory approach

The Design Basis Accident (DBA) Spectrum for licensing of Nuclear Power Plants in Finland is defined in Guide YVL 2.2 "Transient and accident analyses for justification of technical solutions at nuclear power plants" (26.8.2003). Acceptance criteria for design basis accidents are presented in Guide YVL 6.2 "Design bases and general design criteria for nuclear fuel" (1.11.1999).

The cases to be analysed are classified into the following groups: anticipated operational transients (probability not less than 10-2/year), postulated accidents of class 1 (probability 10-2…10-3/year) and postulated accidents of class 2 (probability less than 10-3/year). Also anticipated operational transients during whom a scram fails (the so called ATWS cases) shall be treated as postulated accidents.

Analysis of design basis accidents is done in a conservative fashion and includes consideration of a single failure of active components in the mitigating systems. Assumptions used in the analysis include the following:

- Parameters affecting the final results of the analysis which are essential for the acceptance requirements shall be selected from the edge of their likely range of variation so that the final result can be considered conservative.
- Protection systems are assumed to operate in the designed manner unless an event directly affects their operability. A reactor scram failure during ATWS analyses is an exception.
- Safety systems are assumed to operate at the designed minimum output unless an accident directly affects their operability.
- Normal operating systems can be assumed to operate in the way estimated as most probable.

The methods of calculation shall be adequately verified for the treatment of the events in question and reliability of the methods shall be justified. The experimental correlations used in the calculations shall be justified. Both numerical methods and physical models shall be verified. If sufficiently reliable calculation methods are not available, the analysis shall be justified by experiments.

The Operating Licenses are granted for a limited period of time. The periodic re-licensing allows an opportunity for a comprehensive periodic safety review when the status of each plant is compared with the most recent safety standards in force and new accident analyses are made. Periodic safety assessment is required every ten years for PWR's in Loviisa and every twenty years for BWR's in Olkiluoto. For BWR's there is also intermediate reassessment after ten years.

1.3.2 Technical basis supporting to regulatory approach

The analyses cover different types of accidents as well as possible. Accidents are divided into categories according to their probability and the most restricting with regard to the function and dimensioning of each safety system are analysed. LBLOCA is often, but not always, the most limiting DBA. The selecting of representative accidents is based on the international research programs.

1.3.3 Status of resolution

The approach in Finland is a continuous fulfilment of the criteria presented in the Articles of the Convention on Nuclear Safety. Also, the approach of a continuous improvement of safety is manifested in the Finnish nuclear legislation (VNP 395/1991).

In the case of coming Olkiluoto 3 NPP, event combinations that cannot be easily classified solely on the basis of the frequency of the initiating events will also be examined. These examinations are aimed to justify sufficient diversity of the safety functions and prove that no such threshold phenomena exist immediately outside the scope of the design basis events that would endanger the safety of the plant unit. These conditions are called design extension conditions (DEC). DEC accidents include for example SBLOCA concurrent with SBO and steam line break with SGTR.

1.3.4 Research needed to support modifications

Acceptance criteria for design basis accidents presented in Guide YVL 6.2 can be applied for fuel, the maximum bundle burnup of which does not exceed the value 40 MWd/kgU. With burnup values higher than this, additional research is needed to clarify fuel performance.

1.3.5 Schedule for completion of research and regulatory action

The regulatory guides prepared and issued by STUK are being continuously re-evaluated for updating to meet the most recent international scientific and technical safety standards. After the Decision-in-principle was made in 2002 for the new unit, STUK established a special plan to update the most relevant guides related to the design and construction of a new reactor.

1.4 FRANCE

1.4.1 Current regulatory approach

In 1970, when the preliminary safety analysis report for the first 900 MWe nuclear units constructed in France was submitted, the operating conditions were subdivided into three groups: operating transients and incidents, accidents activating safety devices (engineered safeguard systems) and Loss of coolant accidents. Both the list and the subdivision were derived from American practice which was the outcome of discussions between US plant designers and safety authorities. Acceptance criteria were not always clearly defined.

During the progressive adaptation of the reactor system to French codes and standard culminating in the 1300 MWe reactor design (CAB N°900-MZ –September the third 1979), this list was reconsidered and slightly expanded. Four categories of events were specified; their estimated frequency range and an indication of the maximum allowable radiological consequences for the design studies were provided. Operating transients and incidents were placed in the second category initiating events. The accidents were subdivided and assigned to either the third or the fourth category initiating events, without making a special case for LOCA. Acceptance criteria were defined when necessary. Requirements relating general rules of exploitations, especially periodical tests, were raised too.

The transition to N4 plant series implied a full re-examination of safety demonstration methods and consistency. One can mention the explicit inclusion of the human intervention phase during accident situations: Initial analysis was particularly focused on checking engineered safeguard and protection system design and therefore centered attention on the first phases of the accident where these system start up automatically. The design basis accident rules defined for N4 standardized units safety report explicitly include:

- The human intervention phase until the reactor can be maintained in a safe shutdown. The accident studies only consider actions explicitly provided for in the emergency procedures,
- The definition of the safe shutdown conditions: the core is sub-critical, the decay heat removal and the confinement of containment are ensured. A safe shutdown shall be maintained durably using only safety classified systems and equipments.

- The equipments and systems to be used:
  - All the equipments and systems shall be considered if their actions penalize the safety criteria,
  - All the equipments and systems can be considered in the second, the third and the fourth category of initiating events if they are safety classified,
  - Equipments and systems with a lower level of classification (safety related) can be considered in the second and the third category of initiating events if a justification can be provided,
- The definition of the single failure criterion,
- The assumption of the loss of offsite power in addition to the initiating event for the fourth category if this assumption is more penalizing than the opposite one. For the second and the third category, the consideration of the loss of offsite power in addition to the initiating event shall be dealt with acceptance criteria related to the fourth category.

With this approach, actions and items of equipment can be safety-prioritized, which is in fact one of the methods used for the safety classification of the equipment concerned. Specific requirements regarding safety classification of systems and equipment used for safety demonstrations have therefore been raised.
This practice has been extended to all French plant series in the framework of the safety demonstration for a new fuel management. The assumed delay in accident studies between the first alarm detection and the first human intervention has also evolved gradually from 10 to 20 minutes. This led, among others; to the implementation of an automatic trip of primary pump in case of small break LOCA on 900 MWe plants.
Moreover, transients not taken into account as DBA have been gradually added to safety demonstration which led to some important conception modifications (automatic and passive safety injection for transients initiated at shutdown states, protective device for dilutions) or post-accidental procedures (management of common mode failures, management of weakness of ECCS conception linked to low pressure pump characteristics for small break LOCA). Special rules for studying these transients have been defined.

Safety requirements may also evolve in the framework of periodic reassessment required every ten years. The status of each plant series is compared to more recent standards. As an example, the applicability of EPR safety requirement to 900 MWe plant series was considered in the frame of the third periodic reassessment of these plants, particularly for SGTR management for which French utility has to examine solutions in order to delay as far as possible the filling up of steam generator.

1.4.2 Spectrum of LOCA to be analyzed for safety demonstration

There is no intent in France to exclude LB LOCA from DBA for existing PWR on the basis of its low contribution to the overall plant risk.

As far as EPR is concerned, designer proposed LB LOCA exclusion from DBA accounting for its low occurrence probability, justified by special attention put in fabrication and in-service surveillance. IRSN agreed on this principle, but recommended to maintain LB LOCA study assuming realistic assumptions as far as equipment qualification and containment design are concerned.

1.5 GERMANY

1.5.1. Current regulatory approach:

The Design Basis Accident (DBA) Spectrum for licensing of Nuclear Power Plants in Germany is defined in the "Structured list of planned items for a standard Safety Report" (Gliederung mit Merkposten, 26.07.1976) and in the "Guidelines for the Assessment of the Design of PWR Nuclear Power Plants against Incidents" (Störfallleitlinien, 18.10.1983).

Since the amendment of the German Atomic Energy Act in 2002 it became obligatory to perform every ten years for each German NPP the Periodic Safety Review. In the corresponding "Guides for the Periodic Safety Review of Nuclear Power Plants" (PSÜ-Leitfäden, 1998) the following groups of DBAs, which have to be analysed, are listed:

- Transients
- Loss of coolant accidents
- Radiologically representative accidents
- Internal Impacts
- External Impacts

Currently the DBA spectra in the above mentioned guide for PWR and BWR plants are being reviewed taking into account among other aspects the following new developments (emerging issues):

1.5.1.1 ***Issue: ATWS incidents***

1.5.1.1.1 Status of resolution

The intention of the licensees to increase the burn up of the fuel elements and to intensify the use of MOX fuel elements prompted the federal regulator to authorise the examination of the safety margins in coping with ATWS incidents.

In Germany, ATWS incidents are not treated as part of the design basis but according to the requirements of the guidelines of the Reactor Safety Commission (RSK guidelines) for the reduction of the residual risk in case of loss of the reactor scram system during transients which are to be expected in the course of the lifetime of a reactor.

The control of the ATWS incidents in German PWRs is reviewed for each new reactor core loading. The corresponding review criteria are defined in the RSK guidelines. Demonstration of safety can be performed with analysis considering best estimate assumptions with availability of all other safety and operating systems.

1.5.1.1.2 Technical basis supporting the regulatory approach

For the performance of ATWS analyses, normal operation is generally chosen as initial state. Except for the system postulated to be malfunctioning, all other systems can be assumed to be operable. That means no additional single failures or a maintenance case has to be postulated. As a consequence, all safety valves are available for pressure limitation at any time and the injection through boron systems for long-term subcriticality is fully available. The requirements on permissible stresses are regarded to be fulfilled if the allowable stress according to ASME Code Section III, Division 1, NB-3224 Level C Service Limits is not exceeded.

The requirements for inherent safety are described in the safety criteria of the Federal Ministry of the Interior (BMI safety criteria and the RSK guidelines. Accordingly, fast power increases resulting from a positive reactivity insertion shall be limited in the short term by inherent retroactive effects. The control of the incident takes place in interaction with the other inherent properties of the plant and the boron systems.

The relevant requirement is the observance of the moderator density reactivity dependence which is determined by the fuel element types used and the core loading. For this reason, it is checked for each refuelling whether the moderator density reactivity dependence is more negative than the moderator density reactivity dependence determined as permissible. In case of modifications to subsystems of the reactor plant it is checked if this has an impact on the accident analyses including ATWS events.

Due to different core design, there are reactor plants with a steep moderator density reactivity dependence curve by which the retroactive effects of the reactor power are sufficiently limited and such plants with a flat moderator density reactivity dependence curve which requires switch-off of the RCP for control of the event. The actuation of RCP switch-off is performed either from the pump protection (pressure behind 1st seal stage high) or a reactor scram signal (RESA-K-Signal) generated by the reactor scram signal “RESA-Signal” and the signal for not reaching control rod end position.

At all PWR plants the reactor coolant pumps are switched off after an ATWS-signal. In any case, an injection of boric acid into the core area is important to cope with the ATWS.

1.5.1.1.3 Research needed to support modifications

According to the recent RSK recommendations of July 2005 further investigations have to be performed. The necessary additional investigations will be initiated

1.6 JAPAN

1.6.1 Current regulatory approach

The Nuclear Safety Commission (NSC) of Japan has issued basic safety evaluation guides for current Light Water Reactors (LWRs) such as (1) “Siting examination guide”, (2)”Safety design examination guide”, (3)”Safety evaluation examination guide”, and (4)”Dose objectives guide”. Sub-guides to complement the above guides are also provided as follows for the guide(2) “Seismic design examination guide”, “ Fire protection examination guide”, etc., for the guide (3) “Core thermal design evaluation guide for pressurized water reactors (PWR)”, “Reactivity Initiated Accident evaluation guide”, etc., and for the guide (4) Released radioactive material measurement guide, etc. For the purpose of supporting those guides, there are several reports of the advisory committee such as “Treatment of evaluation for high burn-up fuels in reactivity initiated accidents”, “Determination of thermal design limit for Boiling water reactors (BWRs), etc.
The objective, scope and selection of events and regulatory limits of DBA for current LWRs are discussed in the guide (3) with the basis of the deterministic approach that the results of the safety analysis should satisfy the acceptance criteria in the guide. Those guides have been periodically modified with reflecting new findings of experiments and rational analytical methods. For example, the revision of the “seismic design examination guide” is under way with improved analytical method. In addition, technical standards made by the academic organizations such as the Japan Society of Mechanical Engineers (JSME) and the Atomic Energy Society of Japan (AESJ) are prepared for use.

1.6.2 Technical basis supporting the regulatory approach

Technical bases for the regulations were established from the improvements of design, the progress of safety research and analytical codes, the accumulation of operation experience, etc. For example, the revision of "seismic design examination guide" has been in progress by examining basic earthquake ground motion, seismic classification of importance, and by developing seismic PSA methodology, etc. The regulatory limit of high burn-up fuel enthalpy in Reactivity Initiated Accident (RIA) was redefined in the report "Treatment of evaluation for high burn-up fuels in reactivity initiated accidents" with reflecting new findings of in-pile fuel failure experiments at NSRR, etc.

1.6.3 Status of resolution

Recently, in the "thermal design evaluation guide for pressurized water reactors (PWR)", the change of evaluation method of DNBR has been endorsed by the NSC from the ordinary statistical thermal design method to the generalized statistical thermal design method which combines the uncertainty of DNB correlation with those of the other plant parameters. For the regulatory limit of fuel failure (acceptance criteria of MCPR) for BWRs, the Standard for Assessment of Fuel Integrity under Anticipated Operational Occurrences has been proposed by AESJ that permits the post boiling transition (BT) under the limitation to certain peak cladding temperature and time duration. The results of NUPEC BT tests have much contributed to compile the standard.

1.6.4 Research needed to support modifications

With respect to implementation of the post BT standard into regulatory procedure, it is necessary to examine its applicability to BWRs by regulatory body.

1.7 KOREA

1.7.1 Current regulatory approach

During the past three decades, Korea has continuously constructed nuclear power plants as part of national energy plan to meet the increasing energy demand. As the present time, 18 nuclear power plants are in operation and the 8 units under construction, and 2 more units will be built by the year 2015. As the numbers of NPPs are increasing, "continued improvement of the safety of NPPs" has been set as national policy for the construction and operation of NPPs. According to this national policy, when an application for a Construction Permit (CP) or an Operating License (OL) of the NPPs is applied, the regulatory safety review is focused on the continuous design evolution and the enhancement in safety level through the reduction of core damage frequency to maintain or reduce the current total risk caused by additional NPPs.

During the safety review, regulatory approach for the design basis accidents is as follows:

For the newly designed or different type plant compared to the previously reviewed plant, special attention is paid on the confirmation of the safety adequacy of newly introduced design features, on the suitability of the safety enhancement measures, and on the safety analysis methodology. If necessary, the regulatory audit calculation is performed to confirm the comprehensive plant safety through evaluating the thermal-hydraulic response of the plant and performance of safety system following the postulated design basis accidents. For the same type plants with modification, safety review is focused on the modified portions and their related effects on the plant safety and the regulatory audit calculation is not performed.

1.7.2 Technical basis supporting the regulatory approach

In Korea, as well as in many other countries, the basic concept of nuclear safety is not only to protect the public health and safety from radiation hazards, but also to protect the environment from any potential harmful effects that should be kept as low as reasonably achievable.

This concept is underlined in the Atomic Energy Act of Korea, which provides the legal foundation for nuclear activities. The regulation and licensing of nuclear power plants are based on the provisions of the Atomic Energy Act, the Enforcement Decree and the Enforcement Regulation, and the Notice of the Minister of the Ministry of Science and Technology (MOST).

Safety Review Guideline (SRG) based on the U.S NRC Standard Review Plan (NUREG-0800) was developed to provide the technical guidance to reviewers in performing safety reviews. Safety analysis for the design basis accident should satisfy the acceptance criteria in the chapter 15 of the SRG.

1.7.3 Status of resolution

As of December 2003, 6 KSNP (Korean Standard Nuclear Plant) type units are in operation and 6 units are under safety review for the CP and the OL. When the KSNP was applied for the CP for the first time, safety review was performed to confirm the comprehensive plant safety through evaluating the postulated design basis accidents. Regulatory audit calculations for the DBAs such as LOCA, SGTR, LOFA, and FLB were performed and safety adequacy of the KSNP for the DBAs was confirmed. After the first KSNP, the KSNP has been continuously improved and enhanced in safety features to reflect the changes in major industrial codes and standards and the technical development in the industry, and also to incorporate the feedback of operating experience. Since the KSNP design is standardized the safety review for the design basis accidents was focused on the modified portion and their effects on the overall plant safety.

In spite of these efforts to continuously improve the design safety of the KSNP, the design basis accident, the steam generator tube failure, occurred at Ulchin Unit 4, which is one of the first KSNP, during the shutdown mode operation on April 5, 2002. It was found that circumferential and axial cracks were developed in the locally deformed area inside the tube from just above the tube sheet. The major cause of failure is pertaining to the longitudinal Stress Corrosion Cracks initially developed in the bulge section. The regulatory body prepared Enhanced Steam Generator Tube Integrity Program and implemented many counter measures. By following the program, the utility, Korea Hydro and Nuclear Power (KHNP), improved the radiation monitoring system and leak rate estimation procedures particularly for reactor shutdown, and took measures to enhance the tube integrity of steam generators both in manufacturing and operating.

Meanwhile the KHNP filed the application with the regulatory authority in September 2003 for the construction of Shin Kori units 3&4 which refer to APR1400, evolutionary LWRs based on the CE system 80+ design. They were under development for the past 10 years since 1992 and obtained a Standard Design Approval (SDA) from the MOST in May 2002. Some new design features include (1) direct vessel injection (DVI) of emergency core cooling (ECC) water, (2) fully digitalized I&C systems, (3) workstation based control room and (4) refueling water storage tank inside the containment (IRWST). Therefore, an intensive and in depth safety review for these design features is underway. As a result of the safety review for SDA for the APR1400, the following safety issue has been raised in the DVI among the newly introduced design features: During reflood phase of the ECC after large break loss of coolant accident, complicated thermal hydraulic phenomena can occur in downcomer: ECC water sweep out, ECC bypass, and downcomer boiling. The current Evaluation Model (EM) computer code can not model these phenomena.

1.7.4 Research needed to support regulation

KHNP will submit the uncertainty analysis results by using Best Estimate (BE) computer code to backup the EM code conservatism for the core cooling capability at late reflood phase during the safety review for the CP of Shin Kori units 3&4 which is under safety review for the CP and is scheduled for completion in 2004.

1.7.5 Schedule for completion of regulatory action

Regulatory authority will review the preliminary design safety of DVI including uncertainty analysis results during the CP safety review for Shin Kori units 3&4 and its final design safety of that during the OL safety review which is scheduled in December 2007.

1.8 SWEDEN

1.8.1 Current regulatory approach

The general content of the Safety Analyses Report (SAR) is described the Swedish regulation SKIFS 2004:1. According to these regulations the SAR shall contain a description of how the safety of the plant is achieved. The description shall reflect how the plant is built, analyzed and verified, and how existing requirements with respect to construction, function, organisation and operation, are fulfilled.

For the spectrum of accidents that shall be analyzed for the SAR, the RG 1.70 Rev 3 has been used as a basis. In addition, some specific Swedish requirements, like the 30 minute rule for operators and special acceptance criteria of reactivity initiated accidents (RIA), have been applied. The Swedish SARs also include the requirements on mitigation of consequences of core melt accidents.

1.8.2 Technical basis supporting the regulatory approach

The technical basis has traditionally been close to the practice used internationally, in particular the US rules and regulations. The general design criteria in 10CFR50 Appendix A are often used as criteria also for Swedish plants.

1.8.3 Status of resolution

The requirement to maintain the SAR as a living document reflecting the actual safety status of the plant has been in effect since the mid 1980s. More precise regulations were published in 1998. Since then the utilities have been working on reconstitution of the design basis. The requirements on the SAR were reiterated and emphasized in new regulations of 2004. In particular, the importance of having an updated SAR has been emphasized as a basis for power uprates.

1.8.4 Research needed to support modifications

Not very much research is needed to confirm the design basis currently used. Sweden is participating in international research, for instance, in order to confirm RIA-limits. Needs for optimization of reactor operation could potentially lead to use of deterministic methodologies which are more best-estimate. A more risk informed approach in licensing is also being discussed. Such methodologies would require substantial research efforts before implementation.

1.8.5 Schedule for completion of research and regulatory action

A schedule for completion of safety research can not be given. Such research would have to continue as long as we have nuclear power plants in operation.

1.9 UNITED STATES

1.9.1 Current USNRC regulatory approach

The design basis accidents for U.S. LWRs are specified in Section 15 of Regulatory Guide 1.70 "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." The design basis accident spectrum covers a range of initiating events including:

- reactivity insertion accidents
- loss of coolant accidents
- loss of forced circulation accidents
- steam line break accidents
- loss of power accidents
- fuel handling accidents
- external hazards (e.g., high winds, floods, etc.)

In general, analysis of design basis accidents is done in a conservative fashion and includes consideration of a single failure of active components in the mitigating systems. Additional details on the large break loss-of-coolant accident (LBLOCA) are provided below.

The requirements for LBLOCA analyses and criteria for their acceptability are contained in several different NRC regulations, including:

- 10 CFR 50.34 (Contents of license applications for nuclear power plants)
- 10 CFR 50.46 (Emergency core cooling system acceptance criteria)
- 10 CFR 50, Appendix A, General Design Criterion 35 (Emergency Core Cooling)
- 10 CFR 50, Appendix K, Required and Acceptable Features of LOCA Evaluation Models

Additional information can be found in other documents, such as Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," and NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants"

The current approach requires that a spectrum of LBLOCAs be analyzed, up to and including the instantaneous double-ended guillotine rupture of the largest coolant pipes in the plant. For many plants, especially pressurized water reactors, this accident establishes many of the operating limits for the plant as well as the performance criteria for emergency core cooling systems.

1.9.2 Technical basis supporting the regulatory approach

There is no specific technical basis supporting the current regulatory approach with regard to the choice of the break spectrum. The instantaneous double-ended guillotine cold leg break (DEGCLB), which is often the most limiting DBA, was chosen as a conservative bounding event to define the challenges to the emergency core cooling system. More than 40 years of research on LOCA phenomenology and plant response in numerous in-reactor and out-of-reactor test facilities have supported the development of sophisticated models for the analysis of the entire spectrum of LOCAs, up to the DEGCLB.

1.9.3 Status of resolution

The U.S. NRC is moving ahead with plans to modify requirements for design basis LOCA analysis, to allow consideration of risk and realistic system, structure, and component (SSC) behavior during a LOCA. Use of the modified regulatory requirements would be voluntary on the part of licensees. The proposed changes include:

- Use of an alternate design basis LOCA as determined by a quantitative evaluation of the LOCA contribution to overall plant risk, as defined by core damage frequency.
- Credit for leak before break, under some conditions.
- Elimination of the requirement (in GDC 35) for consideration of a coincident loss of offsite power during a LOCA.
- Performance-based acceptance criteria for ECCS performance in place of current deterministic criteria.

The proposed changes to the LBLOCA requirements are discussed in SECY-02-0057 and SECY-01-0133. The technical bases for the proposed changes are described in Attachment 1 to SECY-01-0133.

1.9.4 Research needed to support modifications

Research needed to support risk-informed and performance-based changes to LOCA-related regulations are discussed in Attachment 2 to SECY-01-0133. Areas in which research is required include:

- Post-quench ductility of zirconium-based cladding alloys
- Determination of LOCA frequencies for use in risk studies
- Improvements to LOCA analytical models
- Behavior of high burnup fuel/cladding under LOCA conditions

1.9.5 Schedule for completion of research and regulatory action

The Commission has directed the NRC staff to proceed with completion of research and development of draft regulatory changes on a parallel track. Issuance of the final revised regulations would likely require another 1-2 years, depending on Commission action and resolution of public comments on proposed rule changes.

Chapter II

Issue: Severe Accidents

Issue description

Severe accidents are generally considered to be events beyond the design basis of a nuclear power plant. These scenarios involve an initiating transient or loss-of-coolant accident, accompanied by the postulated failures of multiple safety-related systems, thus compromising the capability to maintain adequate cooling of the fuel and resulting in significant damage to the fuel (core melting), possibly leading to the release of significant amounts of radioactivity from the primary system into the containment. Under certain circumstances, the containment may also be postulated to fail or to be bypassed (e.g., through the steam generator relief valves in a PWR), resulting in a major radioactive release to the environment.

Severe accidents and related phenomena are generally not addressed explicitly in nuclear power plant regulations. Issues which are discussed in this report include:

- Reactor vessel integrity
- Hydrogen control
- Containment integrity
- Severe accident management.

Each of these issues is described briefly, below.

Reactor vessel integrity

Severe accidents may involve core damage that progresses to the stage at which core geometry is compromised, and molten core materials may relocate into the bottom of the reactor pressure vessel (RPV). The molten mass may then attack the RPV, degrading the vessel wall, vessel penetrations, or both. Maintaining an intact RPV serves a dual purpose: retention of the core inventory of radioactive materials, and mitigation of potential challenges to containment integrity that might occur with the release of molten core material into the containment (see "Containment Integrity," below). Strategies may be employed to depressurize the RPV and to provide adequate cooling of the debris, from either inside or outside the RPV, so as to protect the RPV walls and penetrations and to maintain vessel integrity.

Hydrogen control

Generation of hydrogen during an accident involving significant core damage is a concern for light water reactors (LWRs). Non-condensible gases that remain in the reactor coolant system (RCS) can interfere with coolant flow to the core, while release of the hydrogen to containment can result in the development of hydrogen concentrations sufficient to pose a threat of hydrogen deflagration or detonation. Such events could cause increases in containment pressure that could affect containment integrity for certain containment designs.

Containment integrity

The containment building is the last barrier against the release of radioactive material to the environment in the event of a severe accident. Consequently, it is essential that the containment remain intact for a substantial period during severe accidents, to prevent such releases. The containment can be challenged in several different ways, depending on its design. Examples include attack by molten core material on the concrete basemat and/or the containment structure itself; overpressurization from heating of the containment atmosphere by stored energy and decay heat; deflagration or detonation of combustible gases; or steam explosions caused by molten core material falling into water pools below the reactor. Radioactive material can also be released by bypassing the containment even if the building itself remains intact. Strategies may therefore be employed to make the containment more robust against failure modes and to identify and reduce or eliminate potential bypass pathways to the environment.

Severe accident management

Nuclear power plants include numerous non-safety-related systems that could potentially operate during severe accidents and possibly reduce or eliminate their impacts. In some cases, such systems are employed in ways for which they were not specifically designed, and operator actions may be required to reconfigure or activate them for these purposes. The term "accident management" is used to indicate a systematic approach for identifying non-safety-related systems that could be employed in a severe accident, identifying and planning any actions necessary to realign or reconfigure those systems in such an event, and determining the best ways in which to employ the systems' capabilities.

2.1 BELGIUM

2.1.1 Current regulatory approach

In September 1986, the Regulatory Body sent a letter to the NPPs, requesting to consider severe accidents, and in particular:

- reactivity transients (dilution, …)
- containment ultimate resistance
- hydrogen control
- source term and filtered containment venting

Later the implementation of severe accident management (SAM) guidance was required within the frame of the periodic reviews.

2.1.2 Technical basis supporting the regulatory approach

In addition to the US technical basis, which always serve as an important source of information, the information made available through the numerous related activities and meetings in the frame of the OECD/NEA/CSNI was extensively used.

2.1.3 Status of resolution

2.1.3.1 Initial studies

A thorough review has been performed of all initiators potentially leading to reactivity increases. Measures have been taken to prevent them as far as possible. It concerned essentially provisions to isolate potential sources of non borated water.

The ultimate resistance of the containment of each plant has been computed and the weakest element identified. For some plants, improvement measures have been taken to increase the ultimate capacity of personnel airlocks with inflatable seals.

Severe accident analyses were performed first with the STCP and later with the MELCOR code. Plant-specific models for the Belgian NPP's have been developed for severe accident simulation. These models have been used in support of several projects such as probabilistic safety analyses (PSA), development and implementation of mitigative measures and SAM guidelines, and for operators training and emergency plan exercises.

2.1.3.2 Cavity flooding

Flooding the reactor vessel cavity is a SAM measure in order to mitigate the consequences of core/concrete –interactions and, if possible, to prevent basemat melt-through. An analysis of advantages and consequences of implementing a cavity flooding procedure has indicated that cavity flooding before vessel failure should be considered as the most appropriate strategy to cool the molten core in the reactor cavity. However, the implementation of such a cavity flooding procedure requires a feasibility study which shows the benefit of cavity flooding and which addresses the technical problems to be solved in order to implement a cavity flooding procedure in a existing power plant. This study was performed for the seven Belgian nuclear power plants.

Based on this analysis, it was found that for some existing NPP's cavity flooding prior to vessel failure would be possible on condition that the containment is flooded because water supply paths have been identified. The water volume required to break up the debris stream into particles and to form a coolable particle bed, is small compared to the RWST volume. For plants where the cavity flooding is not possible (Doel NPP's), a study was carried out to investigate the possibility to modify the layout of the cavity by means of a passive system creating a connection between the sump and the cavity. This modification has been implemented and guarantees cavity flooding in case of severe accidents when water is available in the sump.

2.1.3.3 Hydrogen control

Calculations performed have shown that the production of hydrogen and carbon monoxide during a severe accident can endanger the containment or safety-graded equipment. Four hydrogen control systems were considered: ignitors, containment inertisation in normal operation, containment inertisation after an accident occurrence and finally Passive Autocatalytic Recombiners (PARs) placed inside the containment. The comparative study clearly established the advantages of the PARs compared to other systems.

Because no adverse effects were identified, it was concluded that this equipment is safety oriented and can not worsen the conditions resulting from a highly hypothetical severe accident. Hence the utility decided to install PARs in all units; the installation took place from 1995 to 1998. The sizing was based on the then existing calculations performed with STCP, considering the containment as a single volume.

Since the installation of the PARs, a catalytic model for the MELCOR code was developed in order to allow integrated calculations taking into account the atmosphere composition and the hydrogen distribution in the different compartments of the containment. These calculations have been used as confirmation for the sizing of catalytic recombiners.

Sensitivity calculations were performed in the PSA level 2 study in order to investigate the influence of various SAM actions on containment performance. The use of catalytic recombiners reduces the hydrogen concentration in the containment and prevents large hydrogen burns leading to containment failure. Therefore, early containment failure (i.e. before or at vessel failure) becomes unlikely. After vessel failure, the containment integrity is challenged by basemat melt-through and by static overpressurisation.

2.1.3.4 Containment integrity

The most recent Belgian level 2 PSAs include a probabilistic assessment of containment performance and containment failure mode by means of a large containment event tree, which is designed in such a way that the influence of accident management measures can be conveniently analyzed. The impact of some SAM actions like autocatalytic recombiners installation, cavity flooding, and RCS depressurization after core damage have been taken into account in order to assess the contribution of measures to the reduction of the containment failure probability. Containment failure refers to loss of integrity of the last fission product barrier, i.e. structural containment failure, isolation failure (failure to isolate a penetration) and containment bypass (steam generator tube rupture or interfacing system LOCA). The main conclusions of the PSA level 2 study concern the dominant containment failure modes:

- Combustion of hydrogen and carbon monoxide mixtures is a significant challenging phenomenon for the containment;
- Taking into account the installation of catalytic recombiners, the probability of containment failure before or at vessel failure is negligible;
- The containment is significantly threatened by basemat melt-through and by static overpressurization after vessel failure.

Based on these results, different preventive and mitigative means have been examined to cope with severe accident risk.

2.1.3.5 SAM guidance

This project aimed at providing the operators with procedures or guidelines enabling them to deal with complex situations not formally considered in the standard Emergency Response Guidelines, including accidents in which a significant portion of the core melts. The objective of these SAMG programs is to indicate actions that should bring the plant to a controlled stable state and, above all, mitigate any challenges to the fission product barriers. The plant personnel must use the available plant information to determine the best SAM measures. Obviously, efficient decision making is essential under these high stress conditions. Therefore, to assist nuclear power plant operators, SAMGs were developed.

The approach followed for implementing state-of-the-art; plant-specific guidelines differs. The Tihange SAMG's are new guidelines based on the generic SAMG's developed by the Westinghouse Owners Group where as for Doel, the existing utility-drafted Severe Accident Procedures were upgraded and their scope extended. A specific SAM training programme has been put in place in each plant.

2.1.3.6 Outlook

Further improvements to the SAM provisions may be considered in the frame of the next periodic safety reviews.

2.1.4 Research needed to support modifications

No additional research is required to support modification of the guidance documents.

2.1.5 Schedule for completion of research and regulatory action

Not applicable.

2.2 CZECH REPUBLIC

2.2.1 Current regulatory approach

The current legislation
- Act No 18/1997 Atomic Act on peaceful use of nuclear energy
- Regulation No 307/2002 on radiation protection
- Regulation No 318/2002 on emergency planning and preparedness
- Governmental Decision No 11/1999 on determination of emergency planning zone
- Package of crisis legislation (Acts No 239/2002 and 240/2002)

has been stipulating the requirements regarding the response to severe accident (SA).

The approach of regulatory body – SONS to SA is based on:
- the above legal framework
- obligation of Czech Republic from international treaties (IAEA Conventions)
- promotion of necessary research projects (since 1993 using computer code MELCOR)

In 1994 SÚJB issued by a letter to NPPs requirement which is based on NRC Generic Letter No 88-20 Individual Plant Examination for severe accident vulnerability's (10 CFR 50.54.f) requiring the evaluation and identification of WWER-440/213 and WWER-1000 vulnerabilities to severe accidents.

2.2.2 Technical basis supporting regulatory approach

2.2.2.1 Reactor pressure vessel integrity

For prevention RPV failure it is necessary to flood shaft of RPV by water and to cool it from outside. With the current design of WWER-440 the technical solution is to deliver the water to shaft over postament of SG confinement room. Other solution is to deliver the cooling water from the floor of this room. The benefits of these solutions are under consideration. After realisation of this modification, the other modification will be under consideration i.e. removal of the thermal shielding under RPV, which currently prevents the efficiency of outside cooling of RPV.

2.2.2.2 Hydrogen control

During the last years the methodology for evaluation of hydrogen risk has achieved a good progress. The system for hydrogen control for WWER-440/213 is sufficient for design basis accident (DBA).

According to the results of PHARE project performed for Paks NPP as a pilot study for WWER-440 the capacity of systems for hydrogen control (based on recombiners) has to be for severe accidents several timer higher.

2.2.2.3 Containment integrity

The containment system of WWER-440/213 (bubble condenser) is a pressure-suppression type of containment. It means that strength and tightness requirements are substantially lower than for full pressure containment.

The containment of WWER-1000 is standard full pressure containment. The performed calculation predicts the loss of integrity of the containment of WWER-1000 at about 1,2 MPa of internal overpressure.

2.2.2.4 Accident management

The Emergency Operating Procedures (EOPs) has been developed since 1993 as SA prevention measure. They are symptom based, safety related and task oriented unit procedures. Development of EOPs was based on the technology and know-how transfer using methodology developed by WOG and was performed jointly by Westinghouse and plant experts. Altogether about 40 EOPs were developed and 6 CSF Status Trees. The EOPs activities cover development, verification, validation, training and maintenance of EOPs. This process was on both NPP-Dukovany and Temelín fully realized.

Similarly the development of SAMGs was initiated both for control room staff and for Technical Support Centre. Basic approach is that where the EOPs are terminated, the SAMGs are entered at onset. SAMGs are separate documents from the EOPs. SAMGs have exactly defined interface with plant Emergency Response Plan (on site and off-site).

SAM strategy is to limit fission product releases, to maintain RCS integrity, and containment integrity, to establish heat sink and to preserve equipment and instrumentation.

The SAMGs were developed jointly by Temelín NPP staff a Westinghouse experts using the EPRI SAMG Technical basis report, NRC documents (NUREGs), PSA (level 1) for Temelin NPP, IPE insights from several WOG utilities, other plant PSAs and SA analyses etc.

2.2.3 Status of resolution

2.2.3.1 Accident management

At Dukovany NPP the SAMGs preparation is completed and since January 2004 the training of plant staff will be initiated. At Temelín NPP the SAMGs process is in the stage of procedures verification.

2.2.3.2 Hydrogen control

Each unit of Dukovany NPP is equipped with emergency venting of reactor and independent systems for hydrogen control. The selected system is based on passive catalytic recombines

The current system is sufficient for maximum DBA and loss of off-site power during 1500 seconds taking into account of all potential sources of hydrogen. The system is composed of 17 catalytic recombiners distributed in the hermetic boxes.

Hydrogen monitoring system is composed of 16 hydrogen sensors per unit distributed is areas of maximum hydrogen concentrations after the accidents. The measured values are displayed in the main control room. Temperature sensors are also located in the hermetic boxes. Both systems are environmentally qualified.

2.2.4 Research needed to support modifications

Nuclear Research Institute Rez plc (NRI) has been participating in NEA/OECD research projects for severe accidents (RASPLAV, MASCA) which are focussed on corium behaviour in RPV. This is important for investigation of coolability of RPV by external cooling.

A national research project is proposed, which will be focussed hydrogen problem. The project will serve as a basis for future decision making process.

It will be very beneficial to perform cost-benefit analysis for optimal solution of hydrogen elimination (probably by combination of recombiners and ignitors) during the severe accidents in WWER-440/213.

On the basis of Agreement between SONS and USNRC on exchange of technical information and cooperation in the area of nuclear safety the NRI has been participating in the USNRC program of severe accident research (computer code MELCOR and MCAP program etc.) and in the implementation of the above codes for the specific conditions WWER-1000/320 and WWER-440/213.

2.2.5 Schedule for completion of research and regulatory action

The participation of Czech Republic in USNRC program of severe accident research will continue.

2.3 FINLAND

2.3.1 Current regulatory approach

Decision of the Council of State 395/1991 gives a quantitative limit for release of radioactive materials (100 TBq Cs-137 equivalent) arising from severe accidents. The decision also explicitly refers to severe accidents as the design basis events for the containment.

Detailed technical requirements are given in the STUK Regulatory Guides (YVL Guides). The most important Guides concerning severe accidents are:

- YVL 1.0 “Safety criteria for design of nuclear power plants”,
- YVL 2.2 “Transient and accident analyses for justification of technical solutions at nuclear power plants”,
- YVL 2.4 “Primary and secondary circuit pressure control at a nuclear power plant”,
- YVL 2.8 “Probabilistic safety analysis in safety management of nuclear power plants”,
- YVL2.7, “Ensuring a nuclear power plant's safety functions in provision for failures”, and
- YVL 3.5, “Ensuring the firmness of pressure vessels of a NPP”

2.3.2 Technical basis supporting regulatory approach

The technical requirements in YVL Guides are based on the state of the art knowledge in severe accident phenomenology. The YVL Guides are regularly updated.

2.3.3 Status of resolution

2.3.3.1 Reactor pressure vessel integrity

BWRs
The Olkiluoto BWR severe accident management doesn’t rely on in-vessel retention but melt cooling within the containment.

VVERs

The Loviisa 1 and 2 VVER-440 units have been modified to ensure in-vessel retention of molten debris by external cooling. The work included:

- Installation at both units two manually operated, independent high capacity primary system depressurisation valves.
- Modification of the thermal and neutron shield support to enable lowering of the shied by operator. The modification was needed to ensure wide enough flow path for the external coolant.
- Channel modifications at the upper end of the reactor cavity to ensure that the generated steam can escape to the steam generator compartment.
- Creation of additional inlet channels for coolant from the steam generator compartment
- Installation of strainers in reactor cavity to screen out debris from the coolant.

2.3.3.2 Hydrogen control

BWRs
The Olkiluoto 1 and 2 (BWR) containment is inerted during normal operation.

VVERs

The Loviisa 1 and 2 hydrogen management has been improved by plant modifications:

- Ice condenser top and intermediate door remote opening mechanisms have been installed to enable the operator force open the doors from the control room. The modification was needed to ensure hydrogen mixing within the containment.
- Passive autocatalytic recombiners have been installed in the containment
- Ignition capability was kept in the lower compartments.

2.3.3.3 Containment integrity

BWRs

The regulating valves of the Olkiluoto depressurisation system have been modified to secure depressurisation of the primary system. The Olkiluoto containment lower drywell will be flooded by the operator prior the RPV failure to protect the lower drywell penetrations. Both units have been equipped with filtered venting systems to cope with generation of non-condensible gases.

VVERs

External containment spray systems have been installed at Loviisa 1 and 2 to ensure long-term containment cooling. Power is supplied by dedicated local diesel generators. The system is manually started and controlled.

2.3.3.4 Accident management

Both Finnish plants have introduced severe accident management (SAM) procedures. The procedures are built based on the plant specific key severe accident management actions.

BWRs

The Olkiluoto emergency operating procedures for severe accidents covers:
- Primary system depressurisation
- Flooding of the lower drywell
- Containment water filling as a long term measure
- Procedures for containment filtered venting
- Instructions to recover active core and containment cooling systems

VVERs

The Loviisa SAM comprises of:
- Depressurisation of the primary circuit by opening of the depressurisation lines
- Lowering of the neutron shield around the RPV lower head
- Forcing the ice condenser doors open
- Actuation of containment external spray
- Actuation of containment isolation signals if necessary

A criterion for the three first actions is elevated core exit temperature (450 ^{o}C). Containment external spray is actuated if containment pressure exceeds 1.7 bars.

2.3.3.4 Research needed to support modifications

BWRs

Most of the Olkiluoto BWR modifications for severe accidents were installed already during the late 80's. Since that, confirming research has been conducted by the utility to:
- install a pH control system to remove iodine from gas phase
- study ex-vessel fuel coolant interactions at RPV failure
- study mixing and combustion of hydrogen leaking from the containment to the surrounding building
- investigate ex-vessel coolability in particle beds

VVERs

Loviisa plant is a unique combination of a VVER-440 primary circuit with ice condenser containment. Due to this, extensive research effort by the utility was needed to develop a plant specific severe accident management strategy. The main elements were:
- thermal hydraulic experiments (COPO) to determine melt heat fluxes within the RPV lower head
- thermal hydraulic experiments (ULPU) to determine critical heat fluxes on the RPV exterior
- hydrogen mixing experiments (VICTORIA) with a scaled Loviisa containment model
- large scale experiments with the German HDR containment to verify the containment external cooling design

2.3.3.5 Schedule for completion of research and regulatory action

Severe accident modifications are in the finishing stage at both Olkiluoto and Loviisa plants.

2.4 FRANCE

2.4.1 Current regulatory approach

In France, from 1983 to 1990, studies on PSA level 1 have been performed by the Technical Safety Authority Support (IRSN) on 900 MWe PWRs, and by the Utility (EDF). on 1300 MWe PWRs.

In May 1992, the French Safety Authority (DGSNR) requested the Utility (EDF) to have a deep isight on on French reactor severe accident phenomenology and to submit his work to DGSNR Technical Safety Authority Support (IRSN) for advice. The conclusions of the IRSN's analysis had to be presented for review to the expert reactor "independent" advisory group (GPR). This living process concerned all aspects of severe accidents and has been spread over several years.

Based on level 1 PSA and deterministic approaches, the DGSNR issued several recommendations relating to accident management measures and to safety relevant technical improvements, for example the implementation of Passive Auto catalytic Recombiners.

More recently, PSA level 1 studies were upgraded according to the periodic safety reviews process.
In addition, PSA level 2 studies on 900 MWe PWR were performed both by EDF and by IRSN in order to assess severe accident sequences and associated radioactive releases ; they led to propose modifications in the framework of the third periodic safety review for 900 MWe plants, in addition to deterministic classical approach.

Following the fives' first GPR meetings, the DGSNR asked for the utility to propose a referential covering severe accident aspects and defining associated requirements. The analysis of the utility's proposal is going on.

2.4.2 Technical basis supporting regulatory approach

The general technical basis is supported on the state of the art of the severe accident phenomenology as described by OECD in dedicated reports.

The specific technical basis is described in the technical reports prepared in the frame of GPR meetings. These reports are not available to OECD CSNI.

2.4.3 Status of resolution

2.4.3.1 Issue: Reactor Vessel Integrity

In order to reduce the probability of high pressure Reactor Vessel Rupture, which could lead to early loss of containment integrity (Direct Containment Heating), a strategy based on the depressurisation of the primary system (primary bleed) has been required in the severe accident management guide (GIAG). This strategy requires in all PWR-plants to open the three safety valves (SEBIM valves).

Recently, the GPR experts have recommended the utility to implement, on 900 MWe plants, a specific instrumentation allowing detecting the reactor vessel rupture during severe accident scenarios, in order to get information on the beginning of the Melt Corium Cooling Interaction (MCCI).

2.4.3.2 Issue: Hydrogen Control

Starting in 1992, limitation of hydrogen concentration during severe accidents with core damage in PWRs was a concern. Investigations have shown that a large amount of hydrogen will be generated in the case of severe accidents with core damage or complete core meltdown. The DGSNR asked investigations and developments for the early removal and reduction of hydrogen. Finally, in 2001, the utility (EDF) was required to implement Passive Auto catalytic Recombiners (PARS) on all French PWRs before the end of 2007. The first implementation of PARS on a plant was planned for August 2004.

More recently, the GPR experts have recommended the utility to study the implementation, on 900 MWe plants, of specific instrumentation allowing following the hydrogen concentration in the containment during severe accident scenarios, in order to get information on the accident evolution.

2.4.3.3 Issue: Containment Integrity

The general mechanisms leading to the loss of containment integrity have been investigated.

Mechanisms leading to early failure of containment

In addition to hydrogen combustion and DCH (see above), steam explosion and direct release pathways can lead to early high releases to the environment.

The investigation of the consequences of a steam explosion in the pit of the reactor on the internal containment structures, which is actually a major concern, is now going on. The Utility has to perform complementary studies on this issue.

Concerning direct release pathways (β mode in Rasmussen classification) the Utility has to investigate its impact in the framework of level 2 PSA studies.

Mechanisms leading to late failure of containment

In order to prevent slow over-pressurisation of the containment in the event of a severe accident with core damage, the DGSNR required the installation in all French PWRs of a filtered venting system (called U5) allowing retaining radioactive aerosols. Investigations concerning the opportunity of complementary iodine filter implementation are going on by the utility.

High uncertainties remain on MCCI which is an important issue regarding radiological consequences. Current R&D is going on especially in order to investigate the impact on water cooling and to reduce uncertainties on basemat time rupture. The strategy of out-vessel corium management is going on.

2.4.3.4 Issue: Accident Management

First, radiological Source terms were identified in order to build emergency plans:

- Source Term S1 corresponds to early containment failure (a few hours after the onset of the accident) ;
- Source term S2 corresponds to direct release to the atmosphere following loss of containment integrity one or several days after accident initiation (a delay of 24 hours is assumed in the studies) ;

- Source term S3 corresponds to indirect delayed release to the atmosphere through paths enabling a certain amount of fission products to be retained. The radiological consequences corresponding to this source term have been assessed and population protection measures examined in the light of these consequences.

In parallel with the definition of source terms, the French study programs included examination of each mechanism leading to the loss of containment integrity and defined ways of reducing the probability of the consequences by reinforcing the final containment barrier and implementing specific procedures.

The prevention of core melt risk was done first by the implementation of complementary procedures applicable in case of loss of redundant systems called H1 (total loss of heat sink), H2 (total loss of steam generator steam water supply), H3 (total loss of electrical power supplies) created to manage the total loss of system supplies, and by the Ultimate procedure U1 which in particular prescribes (under conditions) the opening of the SEBIM safety valves. More recently, specific Reactor state monitoring procedures were created.

In order to reduce the Source Term S2 nearly to S3, specific measures have been taken:

- the check of defective containment isolation function : U2 procedure(monitoring) ;
- the implementation of filtered containment venting : U5 procedure (venting / sand filter) ;
- the implementation of the U4 procedure (basemat drain filling up).

In addition, a severe accident management guide (GIAG) defining appropriate actions in order to reduce release levels have been implemented. The strategy is oriented to the containment integrity protection.

2.4.4 Research needed to support modifications

The reduction of source term uncertainties is based on experimental programs, such as PHEBUS PF.

A new program called “Source Term” is performed in France (2005-2010) with an international support with the objective to study the iodine behaviour (CHIP, PARIS, EPICUR experiments), the B4C behaviour (BECARRE experiment), the air inflow impact (MOZART experiment) and FP releases (VERDON experiment).

Concerning MCCI, code validation is underway at IRSN (ARTEMIS program, VULCANO tests) in order to validate the basemat penetration time of French PWRs.

DCH experiments on French detailed cavity pit are on going at FZK (Germany) in order to improve the French DCH RUPUICUV code in such a complete geometry.

In order to reduce high level uncertainties regarding steam explosion prediction, experimental programs are going on in France (KROTOS ...) and insights will be drawn in the frame of OECD SERENA program.

2.5 GERMANY – Accident management

2.5.1 Current regulatory approach

Against the background of the Chernobyl accident, the Reactor Safety Commission (RSK) performed a safety review of all German nuclear power plants between 1986 and 1988. One focus of this safety review was on the area of severe accidents. Based on the findings of the review RSK issued several recommendations relating to accident management measures and to safety relevant technical improvements.

There are no explicit legally binding requirements to provide hardware for beyond design base conditions to prevent core damage and mitigate core melt scenarios. Implementation of engineered features for BDBA management is assumed by the existing PSR guidelines. The legally binding requirement (according to the current Atomic Law) to perform Safety Reviews has to be based on the current PSR-Guidelines. Design Basis Accidents and Beyond-design-basis Accidents have to be considered in a Periodic Safety Review.

2.5.2 Status of resolution

→ Implemented in all NPP on voluntary basis, in line with RSK recommendations.

2.5.2.1 Issue: Reactor Vessel Integrity

2.5.2.1.1 Current regulatory approach

RVI became a significant issue in the mid eighties based on results of the German Risk Study Phase B. the study showed for PWR a dominant contribution of high pressure core melt scenarios for transients with total loss of feed water. The frequency of possible large early release paths can be reduced by secondary and primary bleed and feed measures.

Those measures are implemented in all NPP.

2.5.2.1.2 Technical basis supporting regulatory approach

PWR plants:
In all PWR plants the concepts for realizing the depressurization of the primary systems (primary bleed) are based on back fitting the pressuriser valve station:

- installation of (motor operated) pilot valves for medium controlled safety and relief valves (designed for steam, water or two phase mixture operation) with a redundant resp. diverse actuation by spring-loaded or magnetic operated control valves,
- inclusion of the measures with strict separation from the operation related instrumentation and control system.

BWR plants:
The possibility of a diverse depressurization of the reactor vessel is available in all plants. The recommendation was followed by the installation of additional diverse safety valves.

2.5.2.1.3 Status of resolution

In line with the RSK recommendations implemented in all NPP.

2.5.2.1.4 Research needed to support modifications

Additional analyses are needed to check whether the bleed and feed measures are also effective for event groups such as Small LOCAs and by pass sequences with total loss of safety functions.

2.5.2.1.5 Schedule for completion of research and regulatory action

2.5.2.2 Issue: Hydrogen Control

2.5.2.2.1 Current regulatory approach

Investigations (German Risk Study Phase B) in the eighties have shown that a large amount of hydrogen will be generated in the case of severe accidents with damage or complete core melt. The RSK has discussed this issue extensively since the 247. Session of October 18, 1989. The RSK has demanded investigations and preparations for the early removal and reduction of hydrogen.

Regulatory basis is different in the *Länder* (license or supervision).

2.5.2.2.2 Technical basis supporting the regulatory approach

Limitation of hydrogen concentration during severe accidents with core damage in PWRs:
Investigations have shown that a large amount of hydrogen will be generated in the case of severe accidents with core damage or complete core meltdown. The RSK has demanded investigations and developments for the early removal and reduction of hydrogen.

- Pending problems:

The final assessment of the possibility for unintentional ignition of hydrogen by the catalytic recombiners is pending.

2.5.2.2.3 Status of resolution

Passive autocatalytic recombiners, finally recommended by the RSK, have been implemented in all PWR plants.

Inertisation of the containments of BWRs:

To prevent hydrogen detonation during severe accidents in the BWRs of construction line 69, the containments have been inerted. This measure also completely covers the most unfavourable conditions during loss-of-coolant accidents. Concerning the BWR plants of construction line 72, their wetwells were inerted and the drywells equipped with catalytic recombiners.

2.5.2.2.4 Research needed to support modifications

With regard to the formulation of the source terms of hydrogen and steam in case of severe accidents, there are principally uncertainties regarding the release rates. Recombiners were developed to ensure continuous depletion of atmospheric hydrogen under the varying conditions of an accident. Although other ignition sources are also expected in the containment, recombiners may become igniters themselves. All pressure and temperature loads resulting from it are small, as experiments showed so far. These combustion processes will not lead to the destruction of the containment. Nevertheless, the processes for ignition at recombiners and their consequences have to be analysed systematically.

2.5.2.2.5 Schedule for completion of research and regulatory action

2.5.2.3 ***Issue: Containment Integrity***

2.5.2.3.1 Current regulatory approach

In order to prevent the overpressurisation of the containment even in the event of a severe accident with core damage, the RSK recommended the installation of several measures.

The main measures for maintaining the integrity of the containment concern:

- prevention of high-pressure failure of the reactor system by upgrading of the pressure suppression systems,
- implementation of a system of passive autocatalytic recombiners (PARs) for prevention or mitigation of the consequences of hydrogen/carbon monoxide combustions in the containment of PWR plants,
- inertisation of the atmosphere of the pressure suppression system of BWR plants of the 69 construction line by means of nitrogen during normal operation or partial inertisation (wetwell) and PAR installation in the pressure suppression system of BWR plants of the 72 construction line, and
- installation of filtered venting for preventing containment overpressurisation failure.

2.5.2.3.2 Technical basis supporting the regulatory approach

In the German Risk Study, Phase B (DRS-B, PWR), accident-induced phenomena and processes during core melt scenarios were described which may lead to an early or late failure of the containment.

An RPV failure under high pressure, direct heating of the containment atmosphere by fine fragmented melt (DCH), formation of combustible or detonable gas mixtures (H_2 problems), such phenomena and processes were presented in the DRS-B as cause for an early containment failure. In the late phase of an accident, the containment integrity is endangered by a continuous pressure increase in case of failure of residual heat removal and the potential penetration of core melt through the containment basemat.

Research concentrated on the development of accident management measures for LWR plants (PWR and BWR) in order to maintain containment integrity during an accident with core melt.

2.5.2.3.3 Status of resolution

The measures have been implemented in all NPPs.

2.5.2.3.4 Research needed to support modifications

Implementable measures for operating plants for preventing melt-through of the containment basemat in the late phase of an accident with release of molten core materials in the area below RPV are currently not available.

For new reactor concepts, so-called core catchers are currently being developed or under construction internationally. Such concepts are not implemented in existing plants.

However, research should be continued systematically on the topics

- coolability of core melt in the containment, and
- long-term molten core/concrete interaction (OECD project),

in order to be able to develop and implement measures for delaying basemat melt-through, where required.

2.5.2.3.5 Schedule for completion of research and regulatory action

*2.5.2.4 **Issue: Accident Management***

2.5.2.4.1 Current regulatory approach

For the control of design-basis accidents, safety systems are installed which are reliable, for which redundancy and diversity is given to a high degree, and which fulfil their function in case of loss of offsite power. The efficiency and reliability of these systems is verified within the frame of the licensing procedure and the following operation in detail. The concept for the control of accidents has proven itself.

In the course of time, e. g. after the Three Mile Island accident, and the advancements in the field of safety technology, further measures have been implemented for the control of hypothetic system failures and combination of failures or as accident management measures. These were mainly oriented towards the use of available equipment in scenarios for beyond-design-basis accidents.

In the rules and regulations, measures “beyond the design” are not required (these are voluntary measures of the plant operators). Within the frame of safety reviews, however, the existence of accident management measures is checked.

2.5.2.4.2 Technical basis supporting the regulatory approach

Following the recommendations of the RSK accident management measures have been implemented since the late eighties in order to detect beyond- design-basis-accidents timely and reliably, to keep them under control and to bring them to an end with as little damage as possible.

The preventive measures of accident management are to avoid severe core damage. Main goal is to maintain or restore cooling of the reactor core and to convey the nuclear power plant into a safe condition. The mitigating measures, on the other hand, are to reduce serious radiological impact on the plant site and the environment. Here, the main goal is maintaining the activity-retaining barriers still available and to ensure long-term controlled conditions of the plant for the protection of the environment.

The accident management measures are based on a flexible utilisation of available safety and operating systems, even beyond design usage with the risk of their damage, and on the utilization of external systems. Extensive technical and administrative precautions have been taken in the German nuclear power plants in order to be able to perform effective accident management measures should an event actually occur.

In the case of **pressurised water reactors** the precautions concern the preventive measures:

- secondary side bleed and feed,
- primary side bleed and feed,

  and the mitigating measures:
- assured containment isolation,
- primary side bleed,
- filtered containment venting,
- H2 countermeasures,
- supply-air filtering for the main control room.

In the case of **boiling water reactors** they concern the preventive measures:

- an independent injection system (construction line 72),

- additional possibility for injection and refilling of the reactor pressure vessel using existing systems,

  and the mitigating measures:
- assured containment isolation,
- pressure relief of the reactor pressure vessel,
- filtered containment venting,
- inertisation of the atmosphere of the containment (construction line 69) or of the pressure suppression pool air volume only, supplemented by H2 countermeasures in the dry well (construction line 72),
- supply-air filtering for the main control room.

Auxiliary measures supporting the preventive and mitigating measures in both reactor types are:
- emergency power supply from neighbouring plant unit (if existent),
- sufficient capacity of the batteries,
- possibilities for a prompt restoration of the off-site power supply,
- an additional off-site power supply (underground cable),
- sampling system in the containment,
- emergency organisation with training and emergency exercises.

2.5.2.4.3 Status of resolution

The technical development of the H_2 countermeasures for pressurised water reactors is completed, based upon the RSK recommendations. The installation of these recombiners is now completed in most NPPs. A sampling system for the control of the atmosphere in the containment has been developed and is installed at present in several plants. All of the other on-site accident management measures have been implemented.

The feasibility and effectiveness of the accident management measures has to be demonstrated on the basis of representative estimations and plausibility considerations.

The accident management procedures are described in detail in an accident management manual.

2.5.2.4.4 Research needed to support modifications

For the verification that a comprehensive accident management is available and for its completion, additional analyses are required as well as accompanying experimental research for the limitation and reduction of existing uncertainties in connection with the analyses of accident sequences. Such uncertainties are also of primary importance for maintaining competence and knowledge required for the remaining operating lives of LWR plants.

2.5.2.4.5 Schedule for completion of research and regulatory action

2.6 JAPAN

2.6.1 Issue: Reactor Pressure Vessel Integrity

2.6.1.1 Current regulatory approach

Reactor pressure vessel integrity after molten core slumping into the lower plenum of RPV is discussed in a severe accident management strategy (AM). Implementation of AM into existing LWRs was recommended in the report of the Nuclear Safety Commission (NSC) of Japan (1992). Following this recommendation, the utilities implemented AM in all operating commercial LWR plants and the review of the AM by the regulatory body was completed in 2002. The implemented AM includes both modification of automatic depressurization system (RPV) activation logic, and alternate in-vessel and ex-vessel water injection.

2.6.1.2 Technical basis supporting the regulatory approach

The conclusion of OECD TMI-VIP project said that about 19 tons of the debris in the lower plenum of RPV were cooled by water at 10 – 100 K/min. This suggests that water in the low plenum is more effective to cool the debris than previous expected. This is one of motivations of in-vessel debris retention accident management. One of possible debris cooling mechanisms in the low plenum is water penetration into the gap between the solidified debris and the RPV wall. Breakup of the molten debris in the water pool of the lower plenum also enhances the cooling of the debris.

2.6.1.3 Status of resolution

No further rule making is considered now in Japan, concerning the reactor pressure integrity.
The implementation of AM into all existing plants was completed and the regulatory body finished its review in 2002. The review report said that the implemented AM was effective to further improve the plant safety.

2.6.1.4 Research needed

Investigation of debris cooling mechanism in the lower plenum has been performed in several countries. For example, results of ALPHA program at JAERI in Japan ,in which molten oxide simulant material was poured into the simulated lower head with water, implied that a narrow gap was formed between the solidified simulant debris and lower head wall, and water migration into the gap enhanced debris cooling. The results of LHI tests at NUPEC, in which a mixture of UO2, Zr, ZrO2 and SUS was used, also implied that the gap formation between the debris and the lower head wall. However, in both tests, decay heating in the debris was not simulated. Focusing effect caused by molten metallic layer in the low plenum is one of key phenomena to challenge the RPV integrity. OECD-MASCA project is addressing stratification phenomena of the molten pool and the partitioning of fission products (FP) within the different layers of the melt. Japan is now participating on this project.

2.6.2 Issue: Hydrogen Control

2.6.2.1 Current regulatory approach

The countermeasures for hydrogen combustion during a severe accident were studied as one of SA/AM countermeasures studies in Japan as descried in the following section 2.3.5, the issue of Accident Management.

Regarding hydrogen countermeasures, the risk of containment failure by hydrogen combustion for large dry containment PWR was very low, but it was remarkably high for Ice condenser type PWR. Therefore glow-plug igniter systems were installed in ice condenser type PWRs.

Meanwhile the utilities voluntarily studied the SA/AM for future plants. The Atomic Energy Society of Japan had organized a special committee on the evaluation of the thermal hydraulic phenomena in severe accident. The committee had continued the investigation of present status of thermal hydraulic in severe

accident. Industry had completed the detailed implementation of the accident management measures, and industry had established also a self-regulatory document (SRD) mainly on phase II accident management for the containment design of the future reactors. The requirements for hydrogen control were discussed in SRD and laid down settled criteria as follows:
- Preventing detonation for non-inerted containment.
- Preventing combustion for inerted containment

2.6.2.2 Technical basis supporting the regulatory approach

NUPEC carried out a hydrogen mixing test and three types of combustion test: small-scale test, large-scale test, and high temperature test jointly funded by the U.S. NRC. By these tests, it was studied and confirmed the basic hydrogen combustion behavior, and developed a database, which could be used to assess hydrogen combustion behavior during a postulated severe accident of a nuclear plant. The high temperature test showed that the opening of compartment decreased the critical deflagration-to-detonation transition (DDT) hydrogen concentration and the large-scale hydrogen combustion test, simulating eleven sub-compartments in containment, showed that the detonation would not occur up to 15 % hydrogen concentration due to multi-diverged compartment effects.

2.6.2.3 Status of resolution

An issue that remains under study is accumulation of hydrogen gas generated by fuel coolant interaction and ex-vessel molten core concrete interaction, and radiolytic decomposition of water in late phases of a severe accident.

2.6.2.4 Research needed to support modifications

On going PHEBUS-FP Project and OECD MCCI Project are expected to improve the both uncertainty of in-vessel and ex-vessel hydrogen gas generation.

2.6.3 Issue: Containment Integrity

2.6.3.1 Current Regulatory Approach

Strategies to maintain the containment integrity in the event of a severe accident have been discussed in phase II accident management (AM). The implementation into existing LWRs was recommended in the report of the Nuclear Safety Commission (NSC) of Japan (1992). Following this recommendation, the utilities established AM strategies in all operating commercial LWR plants and they were reviewed by the regulatory body in 2002. For a new plant, the report of NSC (1992) also recommended to complete the establishment of AM strategies before fuel loading

2.6.3.2 Technical Basis Supporting the Regulatory Approach

Strategies employed in existing LWRs were based on knowledge obtained from SA researches which have been performed in past a few decades. They include in-vessel water injection to cool the molten core in RPV, ex-vessel water injection to prevent the drywell shell attack and/or mitigate MCCI, hydrogen control and alternate decay heat removal system from the containment.

2.6.3.3 Status of Resolution

Implementation of AM strategies to prevent containment failure in all existing LWR plants was completed and the regulatory body finished its review in 2002. The review report said that the implemented AM was effective to further improve the plant safety. The report also recommended that AM shall be updated if new useful knowledge is obtained.

2.6.3.4 Research Needed

The current AM strategies is based on the existing knowledge, however, effectiveness of some of strategies to maintain the containment integrity still have uncertainty. Japan is now participating on OECD-SERENA and OECD-MCCI projects regarding to ex-vessel debris coolability.

2.6.4 Issue: Accident Management

2.6.4.1 Current Regulatory Approach

In May of 1992, the Nuclear Safety Commission of Japan (NSC) strongly encouraged reactor establishers to prepare plans for effective accident management, defining accident management as means to extensively reduce the latent risk of nuclear installations in the paper of "Accident Management as a Measure for Severe Accidents at Light Water Nuclear Power Reactor Facilities". Nuclear and Industrial Safety Agency (NISA) urged reactor establishers to prepare the measures for accident management in July 1992, clarifying that it would not take any specific statutory requirements. The reactor establishers submitted to NISA study reports on accident management for each operating commercial power reactor in March 1994.
Reviewing the study reports from the reactor establishers, NISA had concluded that the PSA performed and the measures for accident management chosen by the reactor establishers were adequate for enhancing the safety of each commercial power reactor, and NISA urged reactor establishers to implement the proposed measures for accident management further and reported it to the NSC in October 1994. The measures for accident management implemented and proposed by reactor establishers are shown in Tables 1 and 2. The NSC approved those measures for accident management in December 1995 after reviewing them.

In April 2004, NISA clarified its position on accident management and published "Fundamental Recommendations for Accident Management" which include the recommendations on management organization, facility and system, knowledge base, information and communication, and education of staff, and presented it to the NSC.

2.6.4.2 Technical Basis Supporting the Regulatory Approach

The effectiveness of the accident management countermeasures had been verified in NUPEC. Containment integrity tests in Nuclear Power Engineering Corporation (NUPEC) intend to evaluate debris coolability under accident management (AM) conditions, to identify the behavior of combustible gas in a containment vessel during a severe accident, to evaluate fission product removal performance under AM conditions, and to validate the reliability of its mechanical strength, thus ascertaining the capability of the containment vessel to retain radioactive materials in an accident.

2.6.4.3 Status of Resolution

Reactor establishers have been making efforts voluntarily in implementing the measures for accident management approved by the NSC, by installing the necessary systems and components during the periodical outage and establishing the management organization, preparing procedures and making the training program. Operating commercial power reactors have completed the measures for accident management by February 2002.
The measures for accident management implemented by reactor establishers have been reported to NISA by May 2002 together with the results of PSA performed to quantitatively assess measures for accident

management to each reactor type. NISA reviewed the report, referring to the advice of specialists of the subcommittee of the Advisory Committee on Nuclear and Industrial Safety and reported the result of review to the NSC held at October 31, 2002.

2.6.4.4 Research Needed

The ability of AMs to mitigate accident propagation by utilizing an advanced simulation program will be demonstrated the existence of this margin by computer simulations which are more sophisticated than those possible with current analysis codes.

Table 1 Outline of the Measures for Accident Management (BWR)

Function	Items implemented by 2002
Reactor shutdown	i) A separate signal system from the emergency reactor shutdown system is established which causes the reactor power drop, for example by the insertion of substitute control rods. ii) Re-circulation Pump Trip (RPT) activation with the same signal above.
Injection of water into nuclear reactor and containment	i) Automation of the reactor pressure reduction ii) Alternative water injection - Injection of water by the fire protection system and the condensation feed water system into reactor and containment vessel
Heat removal from the containment	i) Substitute heat removal by the reactor core cooling and purification system, etc. ii) Pressure-proof reinforcement vent iii) Restoration of malfunctioning equipment in the residual heat removal system
Support to safety function	i) Utilization of alternative power ii) Restoration of emergency diesel generator

Table 2 Outline of the Measures for Accident Management (PWR)

Function	Items implemented by 2002
Reactor shutdown	i) Diversity of emergency secondary cooling
Cooling of reactor core	i) Utilization of the turbine bypass system ii) Continual injection by alternative supply and alternative recirculation iii) Cool down and recirculation iv) Natural circulation cooling within the containment v) Alternative auxiliary component cooling
Confinement of radioactive materials	i) Natural circulation cooling within the containment ii) Water injection within the containment iii) Forced cooling by primary system iv) Deliberate burning of hydrogen by igniters (only ice-condenser containment plant)
Support to safety function	i) Alternative auxiliary equipment cooling ii) Utilization of alternative power supply

2.7 KOREA

2.7.1 Issue: Reactor Pressure Vessel Integrity

2.7.1.1 Current Regulatory Approach

Currently specific regulatory requirements against severe accidents are absent, but Korean government has issued a "Policy Statement on Severe Accident in NPPs" in October of 2001. Even though the policy statement is not a requirement in a strict sense, still it requires the operating plants to have both an Level 2 PSA and a plant specific AMP. Also the policy required to take into consider the severe accident mitigation capability for the newly designed plants. Thus the regulatory approach in Korea is relying on this government policy.

Two measures generally accepted to keep the RPV intact are to rapidly depressurize the RPV and to add water. For the newly designed plant like APR1400, the capability of rapid depressurization and the in-vessel retention through ex-vessel cooling features are reviewed. The operating plants also can depressurize the RPV rapidly, but the IVR-ERVC strategy is not applicable for most operating plants. Another way is to introduce some modifications (hardware or just some procedures) in the plant based on the PSA result and thus to reduce the core melting probability itself. The last resort goes to AMP. On time application of accident management will help to keep the RPV integrity.

2.7.1.2 Technical Basis Supporting the Regulatory Approach

To cope with the government policy, the utility is performing Level 2 PSA for the operating plants and the results for Kori-1 NPP, for example, show that the Kori-1 is weak against the Loss of Instrument Air accident. But when some modifications are introduced, the analysis shows that the core melting probability can be reduced drastically. On the other hand, KINS found that the so called in-vessel retention through the ex-vessel cooling is not feasible for Kori-1, so we expect that the well managed AMP can effectively contribute to increase safety against the RPV failure

2.7.1.3 Status of Resolution

The utility will submit the PSA result and the AMP for the existing plants, and the regulatory body will evaluate the adequacy of the AMPs. Otherwise no specific rule making is considered in a near future.

2.7.1.4 Research Needed

A study for the optimization of AMPs depending on the specific design and availability of resources to prevent RPV failure is required to reduce the risk of severe accident.

2.7.2 Issue: Hydrogen Control

2.7.2.1 Current Regulatory Approach

KSRG (Korean Standard Review Guideline) is applied as a regulatory requirement against hydrogen risk. The requirements are such that under supposing 100% MWR

- containment pressure should be still less than FLC limit
- average hydrogen concentration should be less than 10%
- no local detonation should occur

2.7.2.2 Technical Basis Supporting the Regulatory Approach

The utility analyses the hydrogen behaviour in the containment using computer codes. KINS also performs an audit calculation using MELCOR code.

2.7.2.3 Status of Resolution

For the operating plants KINS will review how the AMP action can reduce the hydrogen risks. On the other hand APR1400 (advanced type of reactor) has installed PARs and Igniters in the IRWST. But for some accident scenario, SBO as an example, it was found that it is not able to exclude the possibility of high hydrogen accumulation. Thus the evaluation of hydrogen risk in IRWST remains an issue for APR1400 and it will be further discussed during the SAR licensing process.

2.7.2.4. Research Needed to Support Modifications

3D analysis for hydrogen distribution may be required to verify the result using a lumped calculation. Also, It is recommended that DDT prevention method should be developed to avoid hydrogen risk.

2.7.3 Issue: Containment Integrity

2.7.3.1 Current Regulatory Approach

EPRI URD and Secy-93-087 are generally applied in evaluating the containment integrity for the newly designed plants. Long term integrity of the containment against MCCI after 24 hours is analysed using the MELCOR code. For the operating plants the AMPs should be able to increase the containment integrity. The risk of steam explosion will be also evaluated for both the newly designed and the operating plants.

2.7.3.2 Technical Basis Supporting the Regulatory Approach

Experiences acquired through the international research programs are lying in the analysis of the containment integrity.

2.7.3.3 Status of Resolution

Some arguments remain for the long term containment integrity of the newly designed plant. The issue will be further discussed through the SAR licensing process. The analysis of containment integrity for the operating plant is going on in connection with the AMP.

2.7.3.4 Research Needed

OECD MCCI program is under progress. However, the progress of the research indicates that the debris coolability could be still a high priority unresolved issue. A methodology to reasonably evaluate a steam explosion risk using the knowledge and tools acquired until now seems to be in need. And a research on the practical core catcher concept similar to that employed in EPR may be needed.

2.7.4 Issue : Accident Management

2.7.4.1 Current Regulatory Approach

Currently specific regulatory requirements against severe accidents are absent, but Korean government has issued a "Policy Statement on Severe Accident in NPPs" in October of 2001. Even though the policy statement is not a requirement in a strict sense, still it enforces the operating plants to have both a Level 2 PSA and a plant specific AMP. The regulatory approach of Korea is thus relying on this government policy.

The utility has a plan to implement AMPs to all the operating plants by 2008. The Korean regulatory body will review 1) Technical Basis Report and Procedure, 2) Writers' Guideline, 3) Training Program, 4) V/V Program and 5) Implementation Program of AMPs for each plant. The regulatory body enforces that the plant specific features should be taken into account in the AMP of each plant and also that the AMPs should be verified and implemented for the operating plants.

2.7.4.2 Technical Basis Supporting the Regulatory Approach

Level 1&2 PSA results are important in developing the plant specific AMPs. Until now the utility has developed the AMPs based on WOG SAMG which means that the new knowledge acquired through the severe accident research since the development of the WOG SAMG are not well incorporated.

2.7.4. 3 Status of Resolution

The utility will submit the AMPs for the operating plants according to the severe accident implementation program. Level 2 PSA results will be reviewed by KINS and the way to improve safety of the plant will be determined taking into account the plant specific features.

2.7.4.4 Research Needed

The ability of operators to handle the severe accidents is vital for the safety. In this sense, the operator should be well trained against the progress of the severe accident and also for using the AMPs which means the development of simulator for severe accident seems to be in need. The interaction of Level 2 PSA and AMP should be pursued also.

2.8 SWEDEN

Issue: Severe Accidents

2.8.1 Current regulatory approach

In Sweden there are requirements that core melt accidents shall not result in excessive dose to the public or in long term land contamination, and quantitative maximum releases have been defined. The nuclear power plants have therefore been equipped with facilities for pressure relief of the containments and for filtering of the releases.

The current strategy is to allow the core melt to fall into water in the lower part of the containment and to cool it in the reactor cavity for the PWRs or in the pressure suppression pool for the BWRs. For those BWR containments that have a dry space below the reactor pressure vessel, provisions have been made to be able to flood the dry part in the case of a core melt accident. The containment spray systems have been strengthened in all containments. Severe accident management procedures have also been implemented.

Since these measures were implemented in the eighties, it was also a regulatory requirement that international research and experience shall be analyzed by the licensees to determine whether the Swedish actions are adequate. New experience indicate that the probability to retain a core melt in the pressure vessel may be higher that earlier anticipated. If this can be confirmed, there is a possibility that the current strategy will be changed in order to strengthen the pressure vessel barrier.

2.8.2 Technical basis supporting the regulatory approach

International research and experience is being analyzed to determine the applicability to Swedish reactors. In particular, the OECD-projects have been important in this respect. A part of the regulatory research on severe accident in Sweden is carried out in co-operation with the industry. Current research in Sweden is focused on coolability of the melt both in the reactor pressure vessel and at the bottom of the containment. Various measures to improve coolability are being tested. Also steam explosions attract interest because of local challenges to penetrations for some of the containments. A test program for determination of vessel failure modes is being carried out to support decisions on new accident management strategies.

2.8.3 Status of resolution

On going studies

The accident progression sequences are studied using safety computer programs such as MELCOR. The objective is to identify probabilities and consequences of core melt accidents.

2.8.3.1 Pressure vessel integrity

Recent experience, such as inspection of the TMI-2 pressure vessel, seems to indicate that the probability to retain a core melt in the reactor pressure vessel may be higher than earlier anticipated. The continued international efforts are analyzed and own experiments on pressure vessel failure modes are carried out. If a method can be identified that would significantly increase the probability to retain a core melt in the vessel, the current strategy for severe accident management would be revised. Non-failure of the pressure vessel would leave us with two barriers instead of one for prevention of radioactive releases to the environment.

The coolability of core melt in the reactor pressure vessel is being analyzed. Both the potentials of strengthening or providing additional core cooling systems and the effects of external cooling to the pressure vessel are being studied.

2.8.3.2 Hydrogen control

The BWRs in Sweden operate with inerted atmosphere. Analyses indicate that two of the PWRs could have benefit from hydrogen control. No such control measures have yet been installed.

2.8.3.3 Containment integrity

The Swedish strategy is to cool the melt in the containment by letting the melt fall into water, either in the cavity for a PWR, or in the condensation pool or a flooded area for a BWR. There are some of important issues that have to be addressed. Molten material that falls into water may cause a steam explosion. Significant research has been carried out investigate the potential for such an explosion. It has been discussed to add surfactants to reduce the magnitude of steam explosions and fine fragmentation. The current understanding indicates that the probability for a steam explosion is low. Also the probability for direct containment heating has been deemed to be low.

The question of coolability of a particle bed that is submerged into water is still being investigated. The particle size distributions, potential formation of a crust, and interaction between melt and concrete are important parameters. Measures, like installing downcomers to enhance heat transfer from a particle bed, have been tested. Such measures have not been installed yet.

Measures needed to vent the containment in the case of an accident have been installed. No further research is planned to address these installations.

2.8.3.4 Severe accident management

Procedures have been implemented for accident management. Emergency operating procedures (EOPs) are in place. For accidents that go beyond the EOPs, severe accident management guidelines have been developed. The uncertainty in determination of the accident progression for a core melt scenario is very large. The major objective is therefore to inform the operators and organize training for dealing with very complex situations. Under such circumstances there is not very much that can be done, except adding water to the system for cooling.

Research in Sweden and elsewhere has shown that the release of iodine during an accident may be significantly affected by the water chemistry in the sumps and condensation pools. Therefore active Ph-control of the water in the reactor to be used during accidents has been installed in Swedish reactors.

2.8.4 Research needed to support modifications

Further research is needed to understand the coolability phenomena so that the issue can be concluded. Efforts are needed to conclude the question of ex-vessel steam explosions. Core cooling in the vessel and vessel failure modes are being studied to determine the probability that a core melt can be retained inside the vessel

2.8.5 Schedule for completion of research and regulatory action

A schedule for completion of safety research can not be given. Such research would have to continue as long as we have nuclear power plants in operation.

2.9 UNITED STATES

In the absence of specific regulatory requirements, the NRC has traditionally relied on other methods to ensure that U.S. plants have programs and processes in place to reduce the likelihood of severe accidents and to mitigate their consequences should they occur. These include the development of policy statements, such as the Severe Accident Policy Statement and the Reactor Safety Goal Policy Statement; issuance of generic letters, which normally request voluntary action on the part of licensees; and the development of cooperative programs with the industry.

The NRC, and before it, the Atomic Energy Commission, have considered the general issue of severe accidents for over 30 years. The development of probabilistic risk assessment (PRA) techniques, most notably in the Reactor Safety Study (WASH-1400), provided a means to quantify the likelihood of severe accidents and to help determine the ways in which plants might be especially vulnerable to such events. After the accident at Three Mile Island, the NRC proceeded to study the potential public health impacts of severe accidents. This culminated in the issuance of the Commission's Policy Statement on Severe Accidents (50 *Federal Register* 32138, August 8, 1985), which stated that existing nuclear plants posed no undue risk to public health and safety, but which also indicated that systematic examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low-cost improvements. This was followed by the Reactor Safety Goal Policy Statement (51 *Federal Register* 30028, August 21, 1986), which defined the Commission's expectations concerning an acceptable level of risk associated with the operation of nuclear power plants. The NRC staff then proceeded to develop an integrated plan for closure of severe accident issues, which included six major elements: Individual Plant Examinations (IPEs), Individual Plant Examinations for External Events (IPEEEs), Severe Accident Research Program (SARP), Accident Management (AM), Containment Performance Improvement (CPI), and Improved Plant Operations (IPO). These elements comprise the foundation for both the NRC's regulatory approach and the technical basis that supports it.

The IPE and IPEEE programs were discussed in Generic Letter (GL) 88-20, issued on November 23, 1988, and five subsequent supplements. The objectives of the programs were for each licensee to perform a systematic evaluation of plant vulnerabilities to internal and external events, and to identify and implement cost-beneficial modifications that would reduce those vulnerabilities. Insights from these evaluations helped to inform the resolution of the four severe accident issues indicated above.

2.9.1 Issue: Reactor Pressure Vessel Integrity

Reactor vessel integrity became a significant NRC concern after the Three Mile Island (TMI) accident, particularly as the progression of the accident was more completely investigated and understood. Shortly after the accident, the NRC issued a series of Bulletins (e.g., 79-05 and supplements; 79-06; 79-08) directing licensees to take actions to ensure that such an event would not recur, some of which were related to reactor vessel integrity. Several additional generic letters were issued in 1979 and 1980, addressing issues and lessons learned from the TMI accident investigations. Reactor vessel integrity has also been addressed as an integral part of the Severe Accident Research Program (SARP). Research on lower head failure has included investigation of gap cooling inside the vessel, ex-vessel cooling through flooding of the reactor cavity, and high-pressure creep rupture testing of scaled reactor vessels. Specific actions to prevent reactor vessel failure during severe accidents are addressed as part of licensees' accident management programs and generally include measures to depressurize the RPV and add water. All U.S. BWRs have the capability to automatically depressurize the reactor coolant system. Same plants also include reactor cavity flooding.

2.9.2 Issue: Hydrogen Control

2.9.2.1 Current USNRC regulatory approach

In 2003, the U.S. NRC has promulgated a final rule completing efforts to risk-inform the requirements for hydrogen control in domestic light-water-cooled reactors, i.e., refer to 10 CFR 50.44, "Combustible gas control for nuclear power reactors." Specifically, this final rule eliminated the requirements for hydrogen recombiners and hydrogen purge systems in currently–licensed light water reactors and relaxes the requirements for hydrogen and oxygen monitoring equipment to make them commensurate with their risk significance. The rule retains existing requirements for ensuring a mixed atmosphere; inerting Mark I and II containments, and hydrogen control systems capable of accommodating an amount of hydrogen generated from a metal-water involving 75 percent of the fuel cladding surrounding the active fuel region in Mark III and ice condenser containments. The rule also retains the existing analysis requirements and equipment survivability requirements for Mark III and ice condenser containments.

This rule also specifies requirements for combustible gas control in future water-cooled reactors which are similar to the requirements specified for existing plants. However, a key difference is the need to accommodate an equivalent amount of hydrogen as would be generated from a 100 percent fuel clad-coolant reaction. Particularly, if a containment does not have an inerted atmosphere, it must limit hydrogen concentrations in containment during and following an accident that releases hydrogen (equivalent to 100 percent fuel-coolant reaction) when uniformly distributed to less than 10 percent (by volume); and maintain containment structural integrity and appropriate accident mitigating features.

Currently, the NRC is pursuing issues requiring the need for the hydrogen control systems on Mark III and Ice Condenser containment to operate during a station blackout event. This is discussed further below.

2.9.2.2 Technical basis supporting the regulatory approach

The technical bases for the regulations were established from experience at TMI along with bounding estimates for the amount of hydrogen likely to be generated by a severe core damage accident. Particularly, the final rule was based on risk-informed and performance based insights that would eliminate unnecessary regulatory requirements and accordingly focus on risk significant events. Mark III and ice condenser containments were considered to be more vulnerable to damage from hydrogen because of their lower design pressures and smaller containment volumes. Note these containment types are not inerted, but rely on deliberate ignition systems, if available, generally consume the combustible gas before it reaches concentrations that can be detrimental to containment integrity.

2.9.2.3 Status of resolution

The U.S. NRC has promulgated a final rule and this effort is complete. An issue that remains under study is whether plants with Mark III and ice condenser containments must provide additional assurance that power will be available to operate the hydrogen control system. This recognizes that station blackout (SBO) comprises a significant fraction of these plants' overall risk, and that without such assurance; operation of the hydrogen control system in SBO events could be compromised.

2.9.2.4 Research needed to support modifications

No further research is considered to be necessary with respect to its implementation. Evaluation is continuing as to the need for independent power supplies for hydrogen control systems in Mark III and ice condenser containments that would provide increased reliability during station blackout events.

2.9.2.5 Schedule for completion of research and regulatory action

The final risk-informed rule was issued in 2003 and this effort is complete. Recommendations for resolution of the Mark III/ice condenser issue are expected in the near future, but the schedule for final determination of this issue is not certain.

2.9.3 Issue: Containment Integrity

Containment integrity has been addressed through the Containment Performance Improvement Program and through aspects of the SARP. Licensees operating BWRs with Mark I containments were provided with the staff's recommendations for installation of a hardened wetwell vent in Generic Letter 89-16 (September 1, 1989), and recommendations for other containment designs were discussed in Generic Letter 88-20, Supplement 3 (July 6, 1990). The potential for containment bypass during severe accidents involving steam generator tube failure is being addressed through another industry program on steam generator tube integrity that has been endorsed by the NRC. Elements of the SARP that have addressed containment integrity include research programs on: direct containment heating; fuel-coolant interactions, melt coolability and concrete interaction; and steam generator tube behavior in severe accidents. Accident management programs also include actions to mitigate the likelihood of containment failure.

2.9.4 Issue: Accident Management

2.9.4.1 Current USNRC Regulatory Approach

Accident management was discussed in Supplement 2 to GL 88-20, issued on April 4, 1990. Subsequently, the industry developed a voluntary program to implement severe accident management programs, based on broad guidelines developed by the Owners Groups for each of the vendors represented in the U.S. operating fleet (Westinghouse, General Electric, Combustion Engineering, Babcock & Wilcox). The NRC accepted the industry's approach to this issue, and implementation was completed by nuclear power plant licensees around the end of 1988.

2.9.4.2 Technical basis supporting the regulatory approach

As indicated above, the NRC has conducted a wide range of research to investigate severe accident phenomena and methods for preventing or mitigating reactor vessel and containment failure. In addition to the issues addressed in the SARP, significant insights were gained as part of the staff's independent investigation of severe accident vulnerabilities, published as NUREG-1150 (December 1990) and from the NRC's review of licensee submittals from the IPE and IPEEE programs. (See NUREG-1560, October 1997 and NUREG-1742, April 2002, respectively.) A complete list of the reports issued by the NRC and its contractors on severe accident research, phenomena, and insights is too lengthy to include as part of this summary. The final update of the NRC's integrated plan for closure of severe accident issues was issued as SECY-98-131 (June 8, 1998), and contains a discussion of the NRC's efforts to that time.

2.9.4.3 Status of resolution

The severe accident issues discussed above are considered to have been resolved for the purposes of enhancing the capabilities of licensees to deal with potential severe accidents. However, new issues periodically arise that may require further investigation, e.g., the use of high-burnup and mixed-oxide fuels and advanced reactor designs that may be subject to new severe-accident challenges.

2.9.4.4 Research needed to support modifications

Research continues to investigate new issues as noted above, and to improve analytical models for severe accident codes. Additional research may be identified to support initiatives in risk-informed regulation.

2.9.4.5 Schedule for completion of research and regulatory action

Further regulatory action in these areas is not contemplated at this time.

Chapter III

Issue: Station Blackout

Issue description

Most of the safety and non-safety systems in nuclear plants that could be used to remove decay heat from the reactor in the long term require power to operate, e.g., pumps, valve operators, etc.; however, some can be steam-driven. The U.S. Nuclear Regulatory Commission's regulations (10 CFR 50.2) define station blackout (SBO) as "the complete loss of alternating current (ac) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power concurrent with turbine trip and unavailability of the onsite emergency ac power system)." SBO is not considered to include loss of power supplied by a diverse ac power source (e.g., gas turbine or diesel generator) or by the station's batteries; however, an extended blackout could cause depletion of the batteries and eventual loss of all emergency power. This issue includes addressing those features provided to ensure a source of power remains, as well as the duration for which a plant can safely remove decay heat if all safety-grade ac power is lost.

3.1 BELGIUM

3.1.1 Current regulatory approach

Belgium has 7 PWRs in operation. They began operation between 1974 and 1985. They were designed using the American rules and regulations applicable at that time. Therefore station blackout has not been taken as a design basis accident.

However, the four newest plants were required to be able to cope with external events such as aircraft crash or gas cloud explosion. Those events could cause a loss of off-site power and at the same time destroy the on-site safety diesels. Those four plants were therefore equipped from the start with systems that, amongst others, could cope with a station blackout.

The three oldest plants were required to cope with a station blackout within the frame of the first periodic safety review, and were backfitted with new systems.

3.1.2 Technical basis supporting the regulatory approach

In addition to the US technical basis, which always serve as an important source of information, it is worthwhile to quote the IAEA TECDOC-332 "Safety aspects of station blackout at nuclear power plants".

3.1.3 Status of resolution

The four newest plants (Doel 3 and 4 and Tihange 2 and 3) were designed with an emergency second level of protection in addition to and separate from the normal safety systems. These emergency systems are located in a hardened building and are able to cope with external events such as aircraft crash or gas cloud explosion. Thus, in case of a loss of offsite power with complete destruction or failure of all the safety diesels, the emergency diesels will start and will power the emergency charging system, the emergency primary pumps seal cooling system and the emergency feedwater system. These systems will maintain the plant in a safe state. Therefore, due to the diversity of the onsite power system, a complete station blackout, involving also the emergency diesels, is highly improbable. This has been confirmed by the plant specific PSA.

A similar system has been backfitted at Doel 1 and 2 in the frame of the first ten years safety revaluation performed in 1985. A new building was erected, with two diesels, an emergency primary pumps seal injection system and an emergency feedwater system. At about the same time, also in the frame of the first ten years safety revaluation, a system similar to the French LLS system has been installed at Tihange 1. A new diesel powers the emergency seal injection pump and a control panel which allow to operate the turbine driven auxiliary feedwater pump.

All Belgian plants can now cope with a total loss of offsite power together with a loss of the normal safety diesels, and the issue of station blackout is considered to be resolved for the Belgian NPPs.

3.1.4 Research needed to support modifications

No additional research is required to support modification of the guidance documents.

3.1.5 Schedule for completion of research and regulatory action

Not applicable.

3.2 CZECH REPUBLIC

3.2.1 Current regulatory approach

Due to the high redundancy of independent normal, stand-by (2) and emergency (3) power supply systems for each unit of Dukovany NPP, the station blackout is not part of initiating event list of Dukovany SAR. For Dukovany NPP "station black out" belongs among beyond design basis accidents. The analyses of this type of events are given in a separate report (outside SAR).

3.2.2 Technical basis supporting regulatory approach

Taking into account the above fact, significance of this event and IAEA recommendations the set of procedures called ECA O were elaborated in the framework of symptom oriented EOPs for Dukovany NPP. The OEPs were developed with assistance of Westinghouse experts using generic guidelines of WOG.

The NPPs with WWER-440 have very positive feature regarding station blackout due to the layout and particularly the high volume of water in primary circuit. This feature causes that the course of station blackout is slower than at Western NPPs which gives time for plant staff intervention.

3.2.3 Status of resolution

The analyses performed have been in good compliance with the course events of this nature which occurred at several WWER plants in the past.

The situation at Temelín NPP is in the principle solved in the similar way as in the Dukovany NPP.

3.2.4 Research needed to support modifications

No research or further analysis seems to be necessary.

3.2.5 Schedule for completion research and regulatory action

See above

3.3 FINLAND

3.3.1 Current regulatory approach

The requirements for emergency power supply and SBO are contained in Finnish Regulatory Guides YVL 1.0 and YVL 5.2.

In accordance with Regulatory Guide YVL 1.0, in nuclear power plant design, the possibility of the on-site and off-site power supply units being simultaneously lost shall be considered. As provision against such a situation, the plant shall have available a power supply unit which is independent of the electrical power supply units designed for operational conditions and postulated accidents. It must be possible to introduce this power supply unit into operation quickly enough and its capacity shall be sufficient to remove reactor decay heat, to ensure primary circuit integrity and to maintain reactor sub-criticality.

According to YVL 5.2, the electrical power systems of units located on the same plant site shall be designed to make possible the feeding of electrical power from one unit to the safety important systems of the other during an accident.

Storage batteries shall be dimensioned to reliably assure the operating capability of direct current and alternating current power systems important to plant safety in accordance with system-specific operating time requirements. In accordance with Guide YVL 1.0, batteries backing up the operation of electrical systems important to safety shall maintain their capability to operate at least for two hours under any circumstances.

The provisions implemented to comply with these requirements are presented in the FSAR of NPP's. Loss of offsite power is considered as anticipated operational transient. Total loss of offsite power together with a loss of the normal safety diesels is considered as Design Extension Case (DEC) and it is assessed with the corresponding rules. Acceptance criteria are presented in Regulatory Guide YVL 6.2.

3.3.2 Technical basis supporting to regulatory approach

Probabilistic safety analyses give the basis supporting the requirements given above.

3.3.3 Status of resolution

In Finland, all NPP units are provided with 4 emergency diesels (one diesel is sufficient to supply electricity for necessary safety related systems). In case of loss of offsite electric power concurrent with turbine trip and unavailability of the emergency diesels, several provisions have been made to cope with the situation.

In Loviisa, emergency power supply is ensured from the neighbouring plant unit and plant has a gas turbine connection. Loviisa is also equipped with a net connection to the Ahvenkoski hydro power station to ensure the power supply (connection can replace one emergency diesel). Loviisa plant has small SAM-diesels, which are normally connected to the CHRS-system. They can be connected to supply power also for other safety systems. Additionally, auxiliary supplies to the electrical and automation systems are backed up by battery banks.

In Olkiluoto, emergency power supply is ensured from the neighbouring plant unit and is equipped with a net connection to the hydropower station. Emergency diesel buses can be cross-connected between plant units. Gas turbine will be built to coming Olkiluoto 3 unit, and will be connected to present units. Additionally, auxiliary supplies to the electrical and automation systems are backed up by battery banks.

Electrical power to the plant unit for OL3 is available from six diverse sources: off-site 400 kV and 110 kV grid, unit-to-unit supply connections via the Olkiluoto 400 kV substation, supply from the in-house main generator, emergency diesel generators, SBO diesel generators, and the gas-turbine plant to be built on the site. Additionally, auxiliary supplies to the electrical and automation systems are backed up by battery banks.

3.3.4 Research needed to support modifications

No additional research is required.

3.3.5 Schedule for completion of research and regulatory action

Not applicable.

3.4 FRANCE

3.4.1 Current Regulatory Approach

In 1977, following the assessment of the technical safety principles for the future 900 MWe and 1300 MWe nuclear power plants, the French Safety Authority required that the consequences of the simultaneous loss of all the external and internal electrical supplies should have been be assessed, and the measures to cope with this situation defined.

In 1983, a similar request was made for the 1450 MWe nuclear power plants.

The provisions implemented to comply with these requirements are presented in the FSAR of NPPs. The corresponding situation is considered as a beyond design basis accident and it is assessed with the corresponding rules.

3.4.2 Technical Basis Supporting the Regulatory Approach

An overall probabilistic safety objective was set by the Safety Authority in 1977 as well as practical applications in terms of studies to be performed.

The main items of the studies to be undertaken were as follows:

- design of units should be such that the overall probability of the unit causing unacceptable consequences does not exceed 10E-6 per year,

- in agreement to the 10E-6 objective, a value of 10E-7 is assumed as the annual probability of occurrence of unacceptable consequences for each event family,
- “realistic” design assumptions and methods must be used to study event families, the need to account for in design is a result of the mentioned complementary approach,
- simultaneous failures of redundant trains of safety-related systems should be studied in this framework.

3.4.3 Status of Resolution

Each French NPP is provided with the necessary provisions to cope with Station Black Out. These provisions include:

- a dedicated steam-operated system ensuring the integrity of the main coolant pump seals and safety boron injection in the primary system, as well as the electrical supply for the required instrumentation and control,
- the secondary side devices enabling removal of the decay heat.

Furthermore, an onsite ultimate electrical power supply has been implemented to enable short-term supply to the safety bars.

The effectiveness and sufficiency of the SBO management countermeasures have been initially evaluated based on PSA and “realistic” deterministic calculations and are regularly checked in the framework of Periodic Safety Review.

3.4.4 Research Needed

No additional research is required.

3.4.5 Schedule for completion of research and regulatory action

Not applicable.

3.5 GERMANY

3.5.1 Current regulatory approach

Additionally to the emergency power supply (based on the requirements of the RSK Guidelines and KTA technical standards) the German NPPs are equipped with an extra emergency power supply for the control of external impacts (for details see the attached tables (see chapter 3.5.3) describing the power supplies of NPPs with PWR and BWR in Germany) and additional countermeasures like third grid underground connection for power supply from a neighbouring power plant etc. based on the requirements of the RSK.

3.5.2 Technical basis

3.5.3 Status of resolution

The following supplementary countermeasures are implemented in the German NPPs to avoid the occurrence of station blackout:

- Ensuring emergency power supply from the neighbouring plant unit

The possibility for supplying emergency power from the neighbouring plant unit has been implemented in all multi-unit power plants with the exception of KKI where both blocks would receive emergency power from the water power plant Niedereichbach.

- Increased capacity of batteries

All plants meet the requirement of the RSK, namely, that a direct current supply is assured for at least two hours in case of emergency power operation.

- Prompt restoration of grid supply

In all plants sufficient pressurised media are stored in pressurisers to actuate the circuit breakers required for a restoration of the grid supply in an emergency situation.

- Additional main supply via underground cable

To enhance the reliability of the grid connections, two grid connections (main and reserve grid connection) were provided for all nuclear power plants where this had not been part of the original construction already. In addition, all plants have been backfitted with a third, independent grid connection via underground cable, thereby ensuring that emergency power is supplied even in case of a very rare external event.

(For details see attached tables)

Design Characteristics	1st Design Generation	2nd Design Generation	3rd Design Generation	4th Design Generation
Number of independent off-site power supplies	At least 3			
Generator circuit breaker	Yes			
Auxiliary station supply in the case of off-site power loss	Yes, load rejection to auxiliary station supply			
Emergency power supply	2 trains with 3 diesels altogether, or 4 trains with 1 diesel each	4 trains with 1 diesel each		
Additional emergency power supply for the control of external impacts	2 trains	1 - 2 trains, unit support system at one double-unit plant	4 trains with 1 diesel each	
Uninterruptible DC power supply	2 x 2 trains	4 trains (except for 1 plant with 2 x 4 trains)	3 x 4 trains	
Protected DC power supply	2 hours			
Separation of trains	Intermeshed emergency power supply, physical separation of the emergency power supply grids	Partially intermeshed emergency power supply, physical separation of the emergency power supply grids	Largely non-intermeshed emergency power supply, physical separation of the emergency power supply grids	

Table 1: Electric Power Supply: PWR

Design Characteristics	Construction Line 69	Construction Line 72
Number of independent off-site power supplies	At least 3	
Generator circuit breaker	Yes	
Auxiliary station supply in the case of off-site power loss	Yes, load rejection to auxiliary station supply	
Emergency power supply	3 or 4 trains with 1 diesel each	5 trains with 1 diesel each
Additional emergency power supply for the control of external impacts	2 or 3 trains with 1 diesel each	1 - 3 trains with 1 diesel each
Uninterruptible DC power supply	2 x 2 trains	2 x 3 trains
Protected DC power supply	2 hours	
Separation of trains	Partially intermeshed emergency power supply, physical separation of the emergency power supply grids	Largely non-intermeshed emergency power supply, physical separation of the emergency power supply grids

Table 2: Electric Power Supply: BWR

3.5.4 Research needed

No additional research is required.

3.5.5 Schedule for completion of research and regulatory action

Not applicable.

3.6 JAPAN

3.6.1 Current regulatory approach

The requirements for the station blackout are discussed in the safety design examination guide for the light water reactors (LWRs) issued by the Nuclear Safety Commission (NSC) of Japan. Specific requirements in this guide include:
Criterion 9
- Structures, systems, and components (SSCs) important to safety should perform their safety functions, assuming a single failure and a loss of offsite power.

Criterion 27

- LWRs should cope with a short-term station blackout by ensuring plant safe shutdown and decay heat removal.

Criterion 48

- An onsite emergency electric power system and an offsite electric power system should be provided to permit functioning of SSCs important to safety.
- Electric power from a transmission network to an onsite electric distribution system shall be supplied by two or more circuits.
- The onsite electric power supplies shall have sufficient independence and redundancy or diversity to provide sufficient capacity and capability to assure:
  (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and
  (2) the core is cooled, and containment integrity and other vital functions are maintained in the event of postulated accidents, assuming a single failure.
- An electric power system related to SSCs important to safety should have capability to perform surveillance tests and in-service inspections.

Compliances with these requirements are discussed in the application submittals for reactor establishment. In the licensing process, a utility may submit additional information to the regulatory body, including the station blackout frequency and the plant capability against the station blackout.

3.6.2 Technical basis supporting the regulatory approach

The requirements for the station blackout (i.e., Criterion 27) were already incorporated in the safety design examination guide-1977. In the end of 1980s, the Japanese regulatory body addressed the station blackout issue in order to confirm the adequacy of the SBO requirements. (i.e., the SBO capability of the short-term duration). This activity was triggered by the situation that the U.S. NRC published the proposed and the final SBO rule in 1986 and 1988, respectively. The regulatory body, sharing activities with the utilities, investigated the reliability data of offsite power systems and emergency diesel generators at Japanese LWRs. Based on the results of these investigations, the NSC judged that the requirement of the short-term capability was adequate, because of the low frequency of the station blackout event.

3.6.3 Status of resolution

No further rule making is considered now in Japan for the station blackout issues. The station blackout event is, however, one of potential risk contributors to nuclear power plants (NPPs).

In 1992, the NSC strongly recommended the utilities to implement the severe accident management measures in their NPPs referring to the level 1.5 PSA perspectives.

These PSAs showed that the frequencies of the core damage and the containment failure were estimated to be considerably low values, compared with the safety target values described in the INSAG-3 report. The utilities, however, developed the accident management measures including those for the station blackout, because the core damage frequency induced by the station blackout could be considerably reduced with only minor plant modifications. These measures were already implemented by the utilities at their NPPs.

3.6.4 Research needed

The utilities evaluated the effectiveness of the accident management measures, including the SBO measures, by means of the PSA conducted in the periodic safety reviews.
The reliability data are, also successively collected by the Central Research Institute of Electric Power Industry (CRIEPI) in Japan, as to the emergency diesel generator and the offsite power supply.

3.7. KOREA

3.7.1 Current Regulatory Approach

10CFR 50.63 is the requirement to be satisfied against SBO for newly designed plants. All KSNP (Korean Standard Nuclear Plant) type plants are equipped with AAC power. Coping analysis for SBO will be performed in conjunction with PSA for the plants introduced before the KSNP design.

3.7.2 Technical Basis Supporting the Regulatory Approach

Regulatory Guide 1.155 gives guidance to licensees for compliance with the SBO rule.

3.7.3 Status of Resolution

The KSNP or later designed plants in Korea generally satisfy the SBO rule except that there is an argument for the capacity of AAC power. For the operating plants built before the KSNP, coping analysis and the AMP will show how each plant withstands against SBO.

3.7.4 Research Needed

No additional research seems to be in need.

3.8. SWEDEN

3.8.1 Current regulatory approach

The station blackout accident is the design basis for limiting the releases in accidents that are beyond design basis. The requirement is that the release shall be less than 0.1% of the core inventory of iodine and caesium for an 1800 MWth reactor in the case of a complete loss of external and internal AC-power. Research has been carried out to verify that the containment filters fulfil this requirement. It is assumed that the DC-power may be provided by batteries. The battery capacity is an important factor which determines the time available for restoring AC-power without severe plant damage.

3.8.2 Technical basis supporting the regulatory approach

The technical basis for the solutions were developed in the eighties and approved by the regulatory authority. Accident progression studies are being made in order to assess the probability and consequences of a station blackout sequence.

3.8.3 Status of resolution

The basic questions regarding effectiveness of the filters to reduce releases has been considered as resolved. However, efforts are made to strengthen the capability to add water and remove heat from the reactor systems, which is independent of the supply of AC-power.

All the power plants are equipped with mobile, diesel driven pumps that can provide external water, from fire systems or other water storages, to the containment or reactor systems. It is attempted to reduce the dependencies in the safety systems. A dedicated diesel system that was installed in one of the Swedish reactors in order to independently provide core cooling, reduced the core damage frequency by as much as a factor of 10. In some of the reactors steam driven pumps have been installed to reduce the dependence on AC-power.

Passive cooling systems which potentially can be installed in existing Swedish reactors to remove decay heat are being studied. In the oldest plant decay heat is can be removed by natural circulation through a condenser. Passive containment coolers are being considered for some of the newer plants. Such devices have not yet been installed. Potential improvements in the core cooling capability are considered in connection with major modernization projects for the reactors.

3.8.4 Research needed to support modifications

Research on passive systems is analyzed to determine the applicability to existing Swedish systems. No research of this kind is for the time being carried out in Sweden.

3.8.5 Schedule for completion of research and regulatory action

A schedule for completion of safety research can not be given. Such research would have to continue as long as we have nuclear power plants in operation.

3.9 UNITED STATES

3.9.1 Current USNRC regulatory approach

The requirements for SBO are contained in 10 CFR 50.63, "Loss of all alternating current power." All plants must be able to withstand and recover from an SBO (as defined above) of a specified duration. The duration is determined based on several factors, including:

- Redundancy and reliability of onsite emergency ac power sources;
- Expected frequency of loss of offsite power;
- Probable time needed to restore offsite power.

"Withstanding" an SBO is further defined as provision of sufficient capability to ensure core cooling and maintenance of containment integrity for the duration of the event. Licensees are required to perform a coping analysis for NRC review, including baseline assumptions, analyses, and related supporting information. The coping analysis is required to address:

- Proposed duration of SBO, justified based on the factors cited above;
- Description of procedures that will be implemented during the SBO and for recovery from the event;
- Modifications required (if any) to equipment or procedures to comply with the requirements for withstanding and recovering from an SBO (primarily for plants licensed prior to the effective date of the rule).

An alternate (diverse) ac power source, as described above, is considered to constitute acceptable capability to withstand an SBO, provided that a licensee can demonstrate by analysis the capability to deal with the event until the alternate ac source and any required shutdown equipment can be brought on-line. If a licensee can show by test that the alternate power source can be available to power shutdown equipment within 10 minutes of the onset of an SBO, a coping analysis is not required.

Additional information on implementation of these requirements using methods acceptable to the U.S. NRC are contained in various guidance documents (regulatory guides, generic communications, etc.).

3.9.2 Technical basis supporting the regulatory approach

The U.S. NRC established SBO as Unresolved Safety Issue (USI) A-44 in 1980, in part as a result of analyses in WASH-1400, "Reactor Safety Study," which indicated that SBO could be a significant contributor to overall plant risk. Studies conducted between 1980 and 1988 culminated in NUREG-1032, "Evaluation of Station Blackout at Nuclear Power Plants," which presented the NRC staff's technical findings for resolution of USI A-44. Regulatory Guide 1.155, "Station Blackout" was also developed during this period, to provide guidance to licensees for compliance with the SBO rule, particularly performance of the coping analysis.

3.9.3 Status of resolution

The SBO rule and associated technical documents were considered by the NRC to have resolved the issue of station blackout. In 2000 the NRC performed an assessment to evaluate if the SBO rule achieved its desired results. This report, "Regulatory Effectiveness of the Station Blackout Rule" (NUREG-1776, ADAMS Accession Number ML003741781), issued in August 2000, concluded that the SBO rule was effective in reducing the contribution of SBO to overall plant risk. The report also contains recommendations for modifications to the regulatory guidance and NRC inspection documents. These modifications would not impose new regulatory requirements, but would clarify and improve the consistency of existing documentation. Prior to the August 14, 2003 U.S-Canadian blackout, the NRC staff completed NUREG -1784, "Operating Experience Assessment of Nuclear Power Plant Performance" to identify changes to grid performance relative to nuclear power plants which could impact safety. After the blackout the staff completed an update of loss of offsite power frequencies and recovery probabilities. The NRC staff used this information to estimate current core damage frequency for station blackout scenarios. The results of these efforts are documented in NUREG/CR-6890, "Reevaluation of Station Blackout Risk at Nuclear Power Plants." The Advisory Committee on Reactor Safeguards (ACRS) has reviewed the staff's assessment and agrees with the staff's conclusions regarding both the effectiveness of the rule and the suggested modifications.

3.9.4 Research needed to support modifications

No additional research is required to support modification of the guidance documents.

3.9.5 Schedule for completion of research and regulatory action

Chapter IV

Issue: High Burnup Fuel

Issue Description

High-burnup fuel is defined as light-water reactor (LWR) fuel in which the average bundle burnup exceeds approximately 40 gigawatt days per metric ton of heavy metal (GWD/MT) or the rod average burnup exceeds about 45 GWD/MT. Changes in fuel pellets and cladding occur at high burnups that appear to reduce the fuel's resistance to damage; that damage could be caused by reactivity initiated accidents (RIAs), loss-of-coolant accidents (LOCAs), or other transients such as power oscillations during anticipated transients without scram (ATWS) in BWRs. Regulatory concerns associated with potential fuel damage include loss of coolable geometry as result of fuel failure and dispersal into the coolant or cladding failure due to oxidation. Potential impacts on the source terms in various accident scenarios are also of concern if high-burnup fuel should fail.

The nuclear industry began requesting increases in allowable burnup levels in the early 1980s, based on compliance with existing regulatory criteria. These requests were granted. However, studies on fuels with elevated burnups indicated that fuel behavior could potentially change as burnup increased, and that the existing criteria might require modification, and research programs were initiated to more thoroughly evaluate fuel behavior under a variety of conditions. Ultimately, the NRC identified this as Generic Safety Issue (GSI) 170, "Fuel Damage Criteria for High Burnup Fuel." A research program was defined for developing the technical basis to confirm the adequacy of the NRC's regulatory approach with regard to the use of high-burnup fuel, including advanced cladding alloys, and to look at issues associated with burnups in excess of current regulatory limits.

4.1 BELGIUM

4.1.1 Current regulatory approach

The current licensed maximum average assembly burnup for UO2 fuel amounts to 55000 MWd/t and for MOX fuel, 50000 MWd/t.

The Belgian utilities do not have the intention to significantly increase these limits in the short-term, but could be interested to be authorized to reach about 60000 MWd/t for UO2 fuel in the next five years.

The limit is written in the law (Royal Decree) so that any change requires the use of a legal process, which can be long and heavy.

4.1.2 Technical basis supporting the regulatory approach

The bases of the Belgian licensing are the USNRC rules and the associated documents. However, the fuel elements used in the power plants come from different countries so that particular rules or recommendations, suited to these countries or to the fuel providers, can also be applied.

In addition, the international experience feedback and the results of the research programs can lead to particular demands from the Belgian Regulatory Body (for example, to take into account the thermal conductivity degradation with burnup).

Finally, in case of newly designed fuel, a few test-assemblies can be allowed to be introduced into the cores before being fully licensed.

4.1.3 Status of resolution

The evolution of the fuel properties is in general a continuous function of exposure, so that it has been possible up to now to allow higher and higher burnups in small increments, backed up by test reactor experience. Today, there is however one major phenomenon regarding LWR fuel behaviour that was not considered 20 years ago, namely the so-called rim structure formation. The rim effect has an important influence on thermal-, FGR- and PCI behaviour in normal operation and transient conditions, and it is no more possible to extrapolate without more in-depth investigations.

These investigations concern mainly the following aspects: cladding integrity, fuel design limit, control rod insertion problem, criteria and analysis for reactivity accidents, criteria and analysis for LOCA, fuel rod and neutronic computer codes, and are in progress. Even if the target burnup is not very high (~60000 MWd/t), some of these aspects are relevant and must be finalised before accepting a new limit.

4.1.4 Research needed to support modifications

RIA and LOCA simulation tests using irradiated cladding are required to verify that the present criteria remain adequate for higher burnups. At the same time, new cladding materials must be tested in order to confirm their good behaviour at high burnup.

4.1.5 Schedule for completion of research and regulatory action

The schedule for action is depending on the planning of the international experimental and research programs and on the availability of the results.

4.2 CZECH REPUBLIC

4.2.1 Current regulatory approach

Dukovany NPP

Dukovany NPP has operated its four VWWER-440 (213) units since 1985 and during that period significant progress in fuel utilization has been achieved.

The original fuel cycle strategy was based on the fuel project of Russian fuel vendor (TVEL company). It was 3 years cycle with reload of one third of fuel assemblies using out – in – in reload scheme. The average fuel burn-up was 28-30 MWd/kg (maximum 32,5 MWD/kg).

Since the reloading scheme was not ideal from economic point of view and the operational experience with the fuel was very good, the operator has been preparing since 1987 the step-by-step transition to the 4 year cycle with in-in-in out reloading scheme. This loading patterns with high fuel burn-up on the edge of the core is called low leakage loading pattern. Equilibrium core of 4 year cycle was reached in 2000. Only radially profiled fuel with mean enrichment 3,82 w/o was loaded to the core and number of loaded assemblies decreased to 90. The average burn-up increased to 40-43 MWd/kg (maximum 44 MWd/kg).

Since 2003 a 5 year fuel cycle was introduced. New fuel assemblies with burnable absorbers (Gd1) with radially profiled fuel with mean enrichment (3,82 w/o and 4,38 w/o) are loaded into the core and their number decreased to 72. The average burn-up is 49-50 MWd/kg. New type of assemblies (Gd2) with unified enrichment 4,25 w/o were loaded to the core in 2005.

Currently the Dukovany NPP is planning the transition to 6 years cycle.

Temelin NPP

During the construction period the plant safety was upgraded to meet internationally practices on the basis of operationally well proven design of WWER-1000 (320). One of the major design improvement was fuel design change along with complete safety analyses, core design monitoring and operational strategy replacement.

On the basis of international tender the Westinghouse fuel – VVANTAGE-6 was selected. The technology transfer consisted of the core, hydraulic and fuel rod design packages and necessary training of plant staff at Westinghouse facilities. There are 163 VVANTAGE-6 hexagonal fuel assemblies loaded in the core including 61 rod cluster control assemblies which have neutron absorber rods. Fuel assemblies of four different enrichments are used in low leakage loading pattern. The 4 years fuel cycle is used and design burn-up is 60 MWd/kg (lead rod).

The fuel assembly pattern is optimized for each particular reload region as a part of core pattern optimisation and fuel rods with given assembly have varying uranium enrichment in both radial and axial planes.

4.2.2 Technical basis supporting regulatory approach

Safety criteria applied by the regulatory body SONS for LOCA and RIA (DNBR, cladding temperature, reactor coolant pressure) are consistent with US NRC criteria.

The comparison of fuel related safety criteria of LWR and WWER reactors was performed jointly by the NEA (SEG FSM) and the IAEA (IAEA group) and no substantial difference was found.

SÚJB has not yet introduced the safety criteria for fuel enthalpy, since fuel cladding (Zr in Temelin, Zr1Nb in Dukovany) in the primary circuit coolant chemistry do not rise the problem of thick corrosion layer in high fuel burn-up as in PWRs.

Both WWER-440 and WWER-1000 have lower outlet coolant temperature and lower linear fuel output which contribute to the facilitating of the course of transients.

4.2.3 Status of resolution

Czech Republic has been actively participating in the international cooperation where issues of very high burn-up are studied; computer codes have been applied and using this cooperation (OECD projects – CABRI, Halden project, SCIP Studsvik), cooperation with US NRC (CSARP and FRAPCON users clubs).

The domestic experimental research, sponsored by utility CEZ is rather limited.

4.2.4 Research needed to support modifications

Continuation of international cooperation is planned.

4.3 FINLAND

4.3.1 Current regulatory approach

Fuel high burnup related requirements have been given in YVL guide 6.2.

The changes affecting fuel properties, resulting from the radiation exposure (high fuel burnup), shall be taken into account in determining the limits of safe fuel usage. The criteria presented in this Guide can be applied for fuel, the maximum bundle burnup of which does not exceed the value 40 MWd/kgU. With burnup values higher than this, the acceptable limits of fuel shall be separately justified by experiments.

4.3.2. Technical basis supporting to regulatory approach

Several test programs on high burnup fuel have been conducted simulating both LOCA and reactivity-initiated accidents (RIAs), including those at the Japan Atomic Energy Research Institute (NSRR), Halden, the Kurchatov Institute, and the CABRI reactor. Justification of new fuel burnup limits are based mainly on these test results. New mechanistic fuel performance codes have been developed such as FRAPTRAN-GENFLOW for licensing purposes.

4.3.3. Status of resolution

For the operating plants in Finland a maximum fuel assembly burn-up of 45 MWd/kgU has been approved, based on experimental evidences. This limit is common for both BWR and PWR fuel designs (Zircaloy 2 , Zircaloy 4 and Zr 1% Nb (E110)).

The licence applicant has indicated that target value for fuel burnup in OL3 (EPR) is 50 MWd/kgU but no regulatory position has been taken at this stage.

4.3.4. Research needed to support modifications

Further extension of fuel burnups will require additional tests. Finland is participating ongoing test programs such as CABRI, HALDEN, ANL and NSRR.

4.3.5. Schedule for completion of research and regulatory action

Not applicable.

4.4 FRANCE

4.4.1 Status of resolution

In 1999, a first increase of fuel discharge burn up has been licensed, after performing a full revision of the plant normal operation and accidental behavior, and openly discussing and analyzing relevant safety key issues. French utility is now planning new significant increases of the discharge burn up of the fuel in the framework of the implementation of its future fuel management strategies. As regards cladding corrosion, the burn up increase implies a stronger clad resistance to internal pressure, to cope with a larger fission products build-up. In this framework, the fuel elements behavior in normal and accidental transient conditions, both in case of LOCA and RIA and regarding pellet-cladding interaction (PCI), has to be analyzed in deep. Moreover, improvements in computation methodologies and suitable qualification on experimental basis should be necessary in order to guarantee the respect of safety criteria despite the more demanding operating conditions and transients (LB LOCA among others).

4.4.2 Technical basis

The verification of fuel elements behavior is mainly based on experimental results For normal conditions, destructive analysis made on irradiated fuel rods are used for thermal-mechanical code validation. For accidental conditions, the expected "good behavior" of fuel rods is demonstrated through representative tests for RIA and PCI. Extension of LOCA criterion to high burn-up and to new fuel cladding is to be extrapolated from the results of current tests.

4.4.3 Research needed to support modifications

Some complementary CABRI tests should be necessary to analyze the fuel behavior in RIA conditions in the framework of the adoption of new clad materials and design, especially as far as MOX fuel pellets are concerned.

A suitable experimental program aimed at an in-deep analysis of the re-localization phenomena is under discussion.

4.5 GERMANY

4.5.1 Current regulatory approach

The operators of the nuclear power plants plan to further increase the target burnups for the fuel elements. The conservative accident and damage extent analyses under full consideration of high burnup effects required for safety assessments are only available in parts. In this respect, best-estimate analyses under consideration of uncertainty analyses are also taken into account.

These problems in connection with high burnup of fuel elements are currently dealt with at the RSK.

Within the frame of core loadings, the fulfilment of the boundary conditions for the new fuel used stipulated in the rules and regulations shall be proven either under conditions of normal operation and under accident conditions.

4.5.2 Technical basis supporting regulatory approach

In Germany the licensing approach on high burnup strategies – including expert judgement and appropriate safety criteria – is confirmed to be on state of the art. Positive operating experiences with pilot fuel rods confirm this evaluation. The RSK reassesses the safety margin of enthalpy fuel rod defect thresholds and recommends for RIA a limit value of 80 cal/g at a fuel rod segment burnup > 65 MWd/kg for fuel rods without oxide spallation. Additionally, further tests on transient fuel rod behaviour should be performed and the extension of the experimental database on material properties at high burnup conditions is strongly recommended for code validation.

4.5.3 Status of resolution

Concerning a further increase of the burn up of fuel elements the federal regulator authorised investigations relating to the behaviour of the fuel elements in normal operation and in accident conditions. The burn ups achieved in the reactors are recorded in a report. The licensees reported on the status of operating experience and on the existing experimental data basis relating to the behaviour of the fuel elements in case of power ramps and in case of reactivity incidents. In the analysis for the evaluation of core damage in case of LOCA the related models were adjusted for higher burn up and for cladding materials containing niobium.

4.5.4 Research needed to support modifications

Computer codes for the assessment of the behaviour of the fuel and of the fuel rods will be examined taking into consideration the effects of high burn up. The licensees participate in the CABRI-Waterloop-Programme for the completion of the experimental data basis for higher burn ups and for representative cooling conditions for the fuel elements. The codes for the assessment of the behaviour of the fuel in case of LOCA and in case of reactivity incidents will be examined and further developed.

4.6 JAPAN

4.6.1 Current Regulatory Approach

At present, the approved burnup levels are as follows;

- For the BWR 9×9 Fuel (Step-3 Fuel), assembly maximum burnup is 55 GWd/t.
- For the PWR 17×17 Fuel (Step 2 Fuel), assembly maximum burnup is 55 GWd/t.

The Nuclear Safety Commission (NSC) of Japan formulated "Evaluation Guide for Reactivity Initiated Events" in 1984. The RIA criteria, such as enthalpy limit to avoid mechanical energy generation and the failure threshold due to PCMI failure, were revised for high burnup fuels in 1998 based on the new data from the CABRI and NSRR experiments.

The revised fuel enthalpy limit is (230 - $\triangle$E) cal/g. $\triangle$E is the corresponding amount of enthalpy equivalent to the decrease of fuel melting point due to burnup increase, addition of gadolinium, plutonium etc.

The NSC formulated "ECCS evaluation guide" in 1981. The LOCA criteria are as follows;
- The peak cladding temperature shall not exceed 1,200.
- The maximum cladding oxidation shall not exceed 15% of cladding thickness.

4.6.2 Technical Basis Supporting the Regulatory Approach

The RIA criteria of Japan are based on the NSRR experiments with high burnup fuels. The LOCA criteria of Japan are based on the thermal shock experiments with non-irradiated cladding. The thermal shock experiments using pre-hydrided claddings, such as Zry-4, MDA, ZIRLO, NDA, have conducted to evaluate the burnup effects on LOCA criteria.

4.6.3 Status of Resolution

The NSC revised the RIA criteria for high burnup fuels in 1998, based on the new data from the CABRI and NSRR experiments. The JAERI and industries have conducted the thermal shock experiments using pre-hydrided claddings, such as Zry-4, MDA, ZIRLO, NDA, to evaluate the burnup effects on LOCA criteria.

4.6.4 Research Needed to Support Modifications

RIA and LOCA simulation tests using irradiated cladding will be required to confirm that the present criteria are appropriate for high burnup fuels. These tests are planned from fiscal year 2005 to 2007 at JAERI.

4.6.5 Schedule for Completion of Research and Regulatory Action

The first phase of the above tests is expected to be accomplished by fiscal year 2007.

4.7 KOREA

4.7.1 Current regulatory approach

Regulatory requirements for fuel performance include:

- To limit peak cladding temperature in LOCAs to 2200 F, total cladding oxidation to 17% of the total thickness before oxidation, and to require that the core remain in a coolable geometry.

- To require that specified acceptable fuel design limits are not to be exceeded during normal operation, including anticipated operational occurrences.

- To require that reactivity insertion accidents do not result in damage to the reactor coolant pressure boundary or significantly impair core coolability.

- To assume fuel failure when DNB occurs during a Control Rod Ejection Accident

- To limit for peak enthalpy 280 cal/g for a Control Rod Ejection Accident

The Korea's current regulatory limit on peak rod average burnup is 60 GWD/MTU.

4.7.2 Technical basis supporting the regulatory approach

The current regulatory requirements for fuel performance were based on earlier test data of fresh or low burnup fuels of less than 40 GWD/MTU. Most major nuclear countries have not changed the current regulatory requirements even if they are actively investigating the high burnup and new cladding alloy effects.

4.7.3 Status of resolution

The high burnup fuel reactor performance experiences of Korea do not show any major problems. A research project of High Burnup Fuel Safety Tests and Evaluations has started in 2002 under a joint cooperation of KAERI/KNFC/KEPRI and KINS. The purpose of this research program is to obtain performance results of Korea high burnup fuel and to develop evaluation technologies of high burnup fuel safety issues.

KINS agrees with commonly accepted international consensus that although there were technical issues requiring resolutions, these issues did not constitute immediate safety concerns. On this basis and the USNRC's preliminary conclusion that the performance of fuel with burnups up to 62 GWD/MTU would be acceptable, KINS judges the Korea's current regulatory limit on peak rod average burnup of 60 GWD/MTU does not pose safety problems. However, since KINS thinks that close monitoring and active follow-up on international activities on the issues is needed, it involves in the following activities:

- OECD/NEA CSNI Special Expert Group on Fuel Safety Margin.

- OECD/NEA CABRI Water Loop Program.

- International Conferences such as USNRC's NSRC.

4.7.4 Research needed to support modifications

As discussed above, the research required to support modifications to the current regulatory approach includes use of advanced cladding materials and high burnup fuels.

4.7.5 Schedule for completion of research and regulatory action

Korea is formally participating and contributing the OECD/NEA CABRI Water Loop Program to investigate high burnup and new cladding alloy effects for a Control Rod Ejection Accident. To resolve the high burnup fuel issues, KINS will closely monitor and actively follow major nuclear country's activities investigating the high burnup and new cladding alloy effects. KINS will closely monitor the high burnup fuel performances of Korea to strength the regulatory activities if needed. KINS judges that the Lead Test Assemblies have to be actively tested to measure their performances and several full core loadings of new design fuels are to be quantitatively monitored by the industries.

4.8 SWEDEN

4.8.1 Current regulatory approach

The requirements concerning nuclear fuel safety are given in The Swedish Nuclear Power Inspectorate's Regulations concerning Safety in Nuclear Facilities and The Swedish Nuclear Power Inspectorate's Regulations concerning the Design and Construction of Nuclear Power Reactors.

A decision from the Swedish Nuclear Power Inspectorate that gives the limit of peak enthalpy in reactivity accidents as a function of fuel pellet burnup also limits the fuel burnup in Sweden.

The current limit for fuel pellet burnup is 65 MWd/kgU.

4.8.2 Technical basis supporting the regulatory approach

The fuel performance as a function of burnup should be modelled with a computer code. The models in the computer code should be verified against experimental data. Fuel failure limits should be based on experimental data.

Several fuel test programs have been performed to simulate fuel performance and to assess fuel failure limits. The fuel performance at high burnup has been in investigated in the Halden project and in the High burnup rim project. Fuel failure limits at high burnup have been investigated in Studsvik Ramp project, in the Studsvik Cladding Integrity Project and in the CABRI project.

4.8.3 Status of resolution

The Swedish Nuclear Power Inspectorate is going to revise the limits for reactivity accidents.

4.8.4 Research needed to support modification

Further extension of fuel burnup will require additional tests. Sweden is participating in research programs at Studsvik, Halden and CABRI.

4.9 UNITED STATES

4.8.1 Current USNRC regulatory approach

Regulatory requirements for fuel performance include:

- 10 CFR 50.46, which limits peak cladding temperature in LOCAs to 2200 deg. F; total cladding oxidation to 17% of the total thickness before oxidation; and requires that the core remain in a coolable geometry.

- 10 CFR Part 50, Appendix A, General Design Criterion 10, which requires that specified acceptable fuel design limits are not to be exceeded during normal operation, including anticipated operational occurrences.

- 10 CFR Part 50, Appendix A, General Design Criterion 28, which requires that reactivity insertion accidents do not result in damage to the reactor coolant pressure boundary or significantly impair core coolability.

- Guidance to licensees on fuel performance criteria is contained in Regulatory Guide (RG) 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors."

- Staff guidance is contained in Section 4.2 of the NRC's Standard Review Plan (SRP, NUREG-0800).

The guidance in RG 1.77 and SRP, Section 4.2, provides two limits for peak enthalpy in reactivity accidents: 280 cal/g for PWR rod ejection accidents and full-power BWR rod drops, and 170 cal/g for zero- and low-power BWR rod drops. The higher value was based on prevention of fuel melting, which would be required to disperse fuel material from a failed fuel rod, while the lower value addressed local cladding overheating.

The NRC's current regulatory limit on peak rod average burnup is 62 GWD/MT.

In addition to the above requirements, 10 CFR 50.62 specifies requirements associated with the reduction of risk from ATWS events. While this bears indirectly upon fuel performance issues (see below), there are no specific fuel performance criteria in the rule.

4.8.2 Technical basis supporting the regulatory approach

The NRC has conducted an extensive confirmatory research program on high burnup fuel, beginning in 1993; much of that work has been associated with international cooperative research programs. Several test programs on high-burnup fuel have been conducted simulating both LOCA and reactivity-initiated accidents (RIAs), including those at the Japan Atomic Energy Research Institute, the OECD Halden reactor, the Russian Kurchatov Institute, and the French Cabri reactor. Data from these tests have been used to upgrade the NRC's fuel behavior and neutron transport codes to permit modeling of high-burnup fuel.

The nuclear industry has also been involved in studies associated with these issues, through the involvement of EPRI. An example of results from these studies is EPRI Report 1002865, "Topical Report on Reactivity Initiated Accidents: Bases for RIA Fuel Rod Failures and Core Coolability Criteria."

Open literature papers describing the early testing (1993-94) in the high-burnup fuel program can be found in Volume 2 of Proceedings of the Twenty-Second Water Reactor Safety Information Meeting, NUREG/CP-0140. More recent results can be found in Proceedings of the Nuclear Safety Research Conference.

As noted above, the NRC opened GSI-170 to support the development of a systematic approach to resolve technical issues associated with the use of high burnup fuel and to confirm the adequacy of the NRC's regulatory approach. Expert elicitations were conducted to identify and rank phenomena associated with LOCAs (PWRs and BWRs), rod ejection accidents (PWRs), and ATWS-related power oscillations (BWRs). The results of these exercises were reported in:

- NUREG/CR-6742, "Phenomenon Identification and Ranking Tables (PIRTs) for Rod Ejection Accidents in Pressurized Water Reactors Containing High Burnup Fuel."

- NUREG/CR-6743, "Phenomenon Identification and Ranking Tables (PIRTs) for Power Oscillations without Scram in Boiling Water Reactors Containing High Burnup Fuel."

- NUREG/CR-6744, "Phenomenon Identification and Ranking Tables (PIRTs) for Loss-of-Coolant Accidents in Pressurized and Boiling Water Reactors Containing High Burnup Fuel."

These studies were used to define additional research, beyond the testing and analysis previously conducted, needed to address the identified issues. It should be noted that some ongoing research is related to the behavior of high burnup fuel outside the reactor (e.g., dry cask storage of high-burnup spent fuel); those issues are not addressed in this summary.

4.8.3 Status of resolution

On the basis of early studies, the NRC staff reached a preliminary conclusion that the performance of fuel with burnups up to 62 GWD/MT would be acceptable. While tests indicated that cladding failure in RIAs could occur at lower peak enthalpies than those cited in NRC regulatory guidance, the staff concluded that peak enthalpies would still remain below the reduced values. It should be noted that cladding failure is used as the parameter of interest for high-burnup fuel, rather than fuel melting, as in the case of low-burnup fuel. While melting would be required to disperse fuel particles from low-burnup fuel, fission gas expansion in high-burnup fuel can disperse fuel particles through failed cladding even in the absence of melting.

The NRC completed a confirmatory assessment of operating reactors using the updated computer codes in March 2004. The results of the evaluation indicated that the use of fuel up to existing burnup limits did not pose safety problems. The staff's conclusions are documented in a Research Information Letter (RIL) entitled "An Assessment of Postulated Reactivity-Initiated Accidents (RIAs) for Operating Reactors in the U.S." Revisions to RG 1.77 and SRP Section 4.2 are now being developed.

For LOCAs, research is continuing with various cladding alloys to confirm that current fuel designs will perform acceptably during such accidents. Assessment of Zircaloy-2 and Zircaloy-4 has been completed and evaluation of advanced fuel alloys is continuing, with completion expected in 2006 for ZIRLO and in 2008 for M5. The addition of niobium to the advanced alloys has been observed to alter the behavior of the cladding under accident conditions; it may therefore be necessary to pursue changes to the oxidation criteria in 10 CFR 50.46.

Research is also continuing to investigate phenomena associated with BWR ATWS accidents. Since ATWS is an accident beyond the design basis, the regulatory criteria do not specifically require that core coolability be maintained. However, core melting is assumed to be prevented if the requirements of 10 CFR 50.62 are met and the equipment functions as designed. The most likely cause of fuel failure in such events is now believed to result from high-temperature effects on embrittled cladding, similar to that in LOCAs. Resolution of this issue is thus connected with the evaluation of Zircaloy-2 cladding, under LOCA conditions. Overall resolution of the ATWS issue is expected in 2007

Following the PIRT exercises described above, the staff concluded that, although there were technical issues that required resolution, those issues did not constitute immediate safety concerns. Based on that conclusion and the development of a research program to address the identified technical issues, GSI-170 was closed in 2001.

4.8.4 Research needed to support modifications

As discussed above, the research required to support modifications to the current regulatory approach is continuing, with completion of various experiments and associated analyses over the next 2-4 years. This research should alsol support potential future regulatory modifications, including use of advanced cladding materials, MOX fuels, and development of a performance-based option for inclusion in 10 CFR 50.46.

The NRC's position on the extension of burnups beyond current limits is that the industry must provide an adequate technical basis to support the capability of such fuel to meet applicable regulatory limits. However, as noted, the NRC is conducting additional confirmatory research to facilitate the review of expected future licensee requests for extended burnups, including the development of computer code models for burnups up to approximately 75 GWD/MT.

4.8.5 Schedule for completion of research and regulatory action

The schedule for completion of the various research programs supporting resolution of technical issues is discussed in the previous sections. Modification of regulatory guidance related to RIAs is expected to be completed in 2006. Completion of ATWS- and LOCA-related actions is expected to be accomplished over the next 3-6 years.

Chapter V

Issue: Power Uprates

Issue description

Nuclear power plants have been designed with substantial margins in operational and safety system performance. As analytical methods have been refined and existing margins have been more accurately quantified, “excess” margins have been identified, which have been used to enable significant increases in reactor power. In addition, improvements in plant instrumentation have permitted small power increases, to take advantage of lower uncertainties than those assumed in some regulations (e.g., 10 CFR 50, Appendix K). These increases, referred to as “power uprates,” are seen as means to improve plant economic performance. However, higher reactor power results in changes in other plant operating conditions, including increased hydraulic forces, corrosion, decay heat, and flow-induced vibration. Regulation of power uprates must ensure that adequate operational and safety margins are maintained.

5.1 BELGIUM

5.1.1 Current regulatory approach

Among the seven Belgian nuclear power plants in operation, power uprates were performed for four of them. For one of those, the power uprate occurred in two steps of about 5 %, a steam generator replacement occurring at the second step. For the other ones, the single-step power uprate was coupled with the steam generators replacement. To allow the final uprate value of 10 %, core design evolutions leading to new key parameters (fuel cycle extension, increased fuel enrichment, hot spot factor, hot channel factor), major equipment modifications (steam generators, main steam safety valves, auxiliary feedwater circuit, fuel storage pool, pressurizer heaters) and changes of instrumentation setpoints (overtemperature protection, high pressurizer level, high and low steam generator level) or actuation delays (containment high pressure, containment spray) were introduced.

To demonstrate the safety of the uprated units, new methodologies have been introduced taking credit of the “excess” margins evidenced in some safety analyses through the use of best-estimate codes, of statistical methods, etc. Safety margins have so been re-assessed, departing from the original conservative approach. Although the safety criteria did not change, the safety cases showing their compliances are different.

5.1.2 Technical basis supporting the regulatory approach

New methodological approaches have been validated for specific analyses: use of best estimate codes for LBLOCA analyses, use of statistical combination of uncertainties for DNBR evaluations, use of coupled TH/N codes for SLB analyses, introduction of the leak-before-break concept for mechanical design purpose.

5.1.3 Status of resolution

No specific safety issues have been identified for resolution but some incidence on former PSA results may be expected.

5.1.4 Research needed to support modifications

Additional research needs have not been identified.

5.1.5 Schedule for completion of research and regulatory action

Not applicable.

5.2 CZECH REPUBLIC

5.2.1 Current regulatory approach

The power uprating of nuclear power plants has been currently a verified practice in the world. The power has been uprated in several units with WWER-440: Loviisa NPP (Finland) – 9,1% Kola NAP (Russia) - 7% and Rovno NPP (Ukraine) - 7%. The power uprating is under preparation in Jaslovské Bohunice, V-2 (Slovakia).

In December 2003 the Czech power utility CEZ approved investment intention for power uprating of all four units of Dukovany NPP by 9,5 %. It was decided that the power will be increased step by step during 2005-2012 (LP part of turbine, HP part of turbine, reactor output).

According to the Atomic Act No 18/1997 the licensee is obliged to submit the Safety Analysis Report for any change (modification) which has an effect on nuclear safety of nuclear installation. The safety documentation has to prove that current safety level will be maintained.

The Safety Analysis Report will be prepared for the mentioned change and will be submitted to the regulatory body – SONS.

The verification of the safety and reliable operation of both units of Temelin NPP has been under way after their commissioning in 2001-2003. The power uprating of Temelin NPP is not currently planned.

5.2.2 Technical basis supporting regulatory approach

The existing analytical tools (deterministic, probalitistic) will be used for safety justification of the planned power uprating. The power uprating of WWER-440 units is facilitated by the fact, that these units have large safety margins in comparison with other PWR units.

5.2.3 Status of resolution

No safety issues have been identified.

5.2.4 Research needed to support modifications

No additional research needs have been identified.

5.3 FINLAND

5.3.1 Current regulatory approach

The main objective in power uprates are to maintain safety margins as high level as they were before power uprating. This goal required number of modifications especially in Olkiluoto where power uprating was larger (15 %). In Loviisa power was uprated 9 %.

Due to power uprating there were no changes in regulatory system. Operating license renewal was required.

5.3.2 Technical basis supporting to regulatory approach

The technical bases supporting the regulatory reviews of power uprates are the same as those that have been developed to support the original licensing review of a nuclear power plant.

5.3.3 Status of resolution

In order to fulfil the main objective Olkiluoto (BWR) the power uprating includes number of modifications all over the plant. The most important of them were

- new fuel designs
- modernisation of safety valves (two additional safety valves)
- enhancement of decay heat removal system
- new steam separators
- new moderator tank head
- modifications in turbine plant
- modernisation of pressure controller (digital automation system)
- modernisation of neutron monitoring system
- number of modifications in feedwater system
- modification of ATWS protection.

In Loviisa (VVER-440) original safety margins were larger than in Olkiluoto. The Loviisa is a LBLOCA limited plant and in order to maintain existing safety margins there were need to modify the emergency core cooling system. This modification includes new high pressure emergency pumps with higher shut off head and reduction of accumulators' opening pressure.

5.3.4 Research needed to support modifications

No additional research is required.

5.3.5 Schedule for completion of research and regulatory action

All power uprating projects have been completed.

5.4 FRANCE

5.4.1 Current regulatory approach

In France, though nuclear power plants have been designed with substantial margins in operational and safety system performance and "excess" margins have been identified through more accurate uncertainty and design analysis, those margins as well as the performance gains permitted by the adoption of a new instrumentation had never been used to achieve small power increases. Rather, existing and generated margins have been used to adopt either advanced or more economical fuel loading strategies, such as the MOX fuelling and the high burnup.

5.4.2 Technical basis supporting regulatory approach

Should such upgrading be of interest, the existing analytical tools could be advantageously used to perform the safety demonstration. The power uprating French PWRs would be eased by actual safety margins and predictable gains.

5.4.3 Status of resolution

No power uprating has been either implemented scheduled or scheduled, in French PWR nuclear power plants.

5.4.4 Research needed to support modifications

Should such power increases be decided, in principle, no additional research would be required.

5.4.5 Schedule for completion research and regulatory action

Neither specific research programs, nor regulatory actions have been undertaken been

5.5 GERMANY

5.5.1 Current regulatory approach

In the German licensing procedure there are three different levels to deal with modifications.

First, a modification may be considered as not substantial. That means, there is essentially no concern about the safety status of the plant. In this case the competent nuclear authority may require information in advance. But not official permit is needed or issued. Of course other non-nuclear requirements may apply. The respective authorities may grant a permit or demand additional measures as needed.

Second, the modification may be considered as substantial. But the safety relevant impact is negligible or only improving the safety status of the plant. In this situation the competent authority will issue a license approval but will not start the procedure for public involvement. This approach was used for modifications like MOX fuel usage in PWR's.

Third, the impact on the safety status may appear not obviously in the direction of improvement of safety, but needs more detailed examination. This normally starts the full procedure including public involvement. This process took place in the last years especially when BWRs applied for MOX fuel usage. Examples are NPP Gundremmingen and NPP Krümmel.

5.5.2 Technical basis supporting regulatory approach

The power uprates based basically on improvements at the conventional part of the plant and modification of nuclear power generation (increased enrichment, MOX insertion).

5.5.3 Status of resolution

5.5.3.1 Power increase activities at German NPP

Plant	a) type b) generation/line	Year of commiss-ioning	Power increase a) applied b) realised	elect. power (MW) a) before b) after uprating	Power increase (%)	Type of measure[1)]	technical measure comment
Biblis A (KWB-A))	a) PWR b) 2nd	1974	a) b) 1995	a) 1204 b) 1225	1,7	T	
Isar 1 (KKI 1)	a) BWR b) 69	1977	a) b) 2000 a) April 2000 b) pending	a) 907 b) 912 a) 912 b) 976	0,6 6.9	T	
Isar 2 (KKI 2)	a) BWR b)4th	1988	a) 1990 b) 1991 a) b) 1992 a) b) 1995/1996 a) 1998 b) 1998 a) b) 2002	a) 1370 b) 1400 a) 1400 b) 1410 a) 1410 b) 1440 a) 1440 b) 1475 a) 1475 b) 1500	2,3 0,7 2,1 2,,4 1,7	P T T /2 steps P T	Exchange of 2nd and 3rd stationary blade row new LP turbine Improved efficiency by measures at extraction stage and condenser
1) T: Turbine improvement (increase of conversion efficiency of steam into electric power), **P:** Increase of thermal power output							

5.5.3.1 Power increase activities at German NPP (continued)

Plant	a) type b) generation/line	Year of commiss-ioning	Power increase a) applied b) realised	elect. power (MW) a) before b) after uprating	Power increase (%)	Type of measure[1)]	technical measure comment
Unterweser (KKU)	a) PWR b) 2nd	1978	a) b) 1990 a) b) 1994 a) b) 1995 a) 1997 b) 2000	a) 1320 b) 1350 a) 1350 b) a) b) a) … b) 1410	2.3 4,4	T T T P	Exchange of LP turbine III rotor removal of main steam sieves, HP rotor mod. exchange LP turbine I/II rotor
Philippsburg 1 (KKP 1)	a) PWR b) 3rd	1984	a) April 1998 b) Aug. 2000	a) 900 b) 912 a) 912 b) 926	1,3 1,5	T T	
Philippsburg (KKP 2)	a) PWR b) 3rd	1984	a) b) 1993 a) b) 19996 a) 1998 b) 2000	a) 1390 b) 1402 a) 1402 b) 1424 a) 1424 b) 1458	0,9 1,6 2,4	T T P	
Grafenrheinfeld (KKG)	a) PWR b) 3rd	1982	a) b)1994 a) 2002 b) pending	a) 1300 b) 1345 a) 1345 b) 1410	3,5 4.9	T P	 BMU rejected draft license

[1)] **T:** Turbine improvement (increase of conversion efficiency of steam into electric power), **P:** Increase of thermal power output

5.5.3.1 Power increase activities at German NPP (continued)

Plant	a) type b) generation/line	Year of commiss-ioning	Power increase a) applied b) realised	elect. power (MW) a) before b) after uprating	Power increase (%)	Type of measure[1)]	technical measure comment
Krümmel (KKK)	a) BWR b) 2nd	1983	a) 2004 b) in process	a) 1316 b) 1383	5,1	T	exch. of HP, LP I/II turbine (2005), xch.LP III turbine (2006)
Gund-remmingen (KRB II/B)	a) BWR b) 72	1984	a) b) 1994 a) 1999 b) withdrawn 2001 a) 2001 b) pending	a) 1300 b) 1344 a) 1344 b) a) 1344 b) 1400	3,4 4.2	T	
Gund-remmingen (KRB II/C)	a) BWR b) 72	1984	a) b) 1995 a) 1999 b) withdrawn 2001 a) 2001 b) pending	a) 1300 b) 1344 a) 1344 b) a) 1344 b) 1400	3,4 4.2	T	
Grohnde (KWG)	a) PWR b) 3rd	1984	a) b) 1995	a) 1394 b) 1430	2,6	T	

1) T: Turbine improvement (increase of conversion efficiency of steam into electric power), **P:** Increase of thermal power output

5.5.3.1 Power increase activities at German NPP (continued)

Plant	a) type b) generation/line	Year of commiss-ioning	Power increase a) applied b) realised	elect. power (MW) a) before b) after uprating	Power increase (%)	Type of measure[1)]	technical measure comment
Brokdorf (KBR)	a) PWR b) 3rd	1986	a) b) 1997 a) 1997 b) rejected '99 a) 2002 b) pending	a) 1395 b) 1440 a) 1440 b) 1492 a) 1440 b)	2 3.6		 in court
Emsland (KKE)	a) PWR b) 4th	1988	a) b) 2000 a) 2002 b) pending	a) 1363 b) 1400 a) 1400 b) 1436	2,7 2.6	T P	
Neckar-westheim 2 (GKN 2)	a) PWR b) 4th	1989	a) b) 1992 a) April 2002 b) pending	a) 1325 b) 1365 a) 1365 b)	3.0		

[1)] **T:** Turbine improvement (increase of conversion efficiency of steam into electric power), **P:** Increase of thermal power output

5.5.4 Research needed to support modifications

No additional research is required.

5.5.5 Schedule for completion research and regulatory action

Not applicable.

5.6 JAPAN

5.6.1 Current Regulatory Approach

A request for a power uprate is considered to be a matter of approval of alteration in reactor establishment in Japanese regulation. However, there is no request of power uprate from licensees until now. Some licensees seem to have intention to request power uprate and the Agency for Natural Resources and Energy (ANRE) announced the budgetary assistance for promoting the industry plan. The follow-up of US and other countries leading in the regulation for a power uprate has been performed by the NISA, regulatory body, and the JNES in order to prepare for future licensing applications.

5.6.2 Technical Basis Supporting the Regulatory Approach

The technical bases for supporting the regulatory reviews of power uprates are prepared in a timely manner and effectively and consistently with those that have been developed to support the original licensing review of a nuclear power plant, since all applicable regulations must meet consistently. If improved analytical techniques are used to quantify the excess margins that are available for increased reactor power, it must be accepted by the regulatory body.

5.6.3 Status of Resolution

Currently there is no record of regulation.

5.6.4 Research Needed to Support Modifications

The JNES has been performing the development and implementation of best estimate (BE) analysis codes to evaluate properly the safety margin of the plant. In a case of power uprate it should be necessary to examine if the adequate margin is maintained even if the operating conditions change severer than those before power uprate. The task group of OECD/NEA/CSNI SMAP (Safety Margins Action Plan) focuses on the safety margin issue with the objectives of providing a methodology that enables assessment of the integrated effects of various changes in plant operating conditions, including power uprate, longer refueling intervals, and license renewal (i.e., plant aging). The JNES is participating in the task group to implement the methodology as one of the evaluation tool for safety margin.

5.6.5 Schedule for Completion of Research and Regulatory Action

There is no practical regulatory action that is considered to be required for undertaking this. However, the development and implementation of the BE code is necessary and the JNES is involved in the task.

5.7 KOREA

5.7.1 Current regulatory approach

The current climate of change in the nuclear industry has motivated the utility to enhance its economic competitiveness. To improving its economic performance, Korea Hydro & Nuclear Power Co. (KHNP) has launched the power uprate project of 950 MW Westinghouse PWRs at Kori units 3,4 and Younggwang units 1,2 by about 5% which would involve making alterations in set-points in systems and equipment, but would require no major plant modifications. The project got under way in 2002 with a feasibility study that has now been completed and come to an end in 2007.

The feasibility focused on NSSS, BOP, and turbine side system, and set a target uprate of 4.5% for each of the four PWRs. The feasibility study found that there might be moisture separation and moisture carryover problems at Kori units and the high pressure turbines in use at all four PWRs would have to be modified to handle increased steam volume flow under the targeted level of decreased steam pressure. KHNP would like to obtain the license of power uprates in near term.

While the power uprates are fairly routine in U.S. and Europe, no such program has been carried out for any of nuclear power plants in Korea. At present, the regulatory body does not have a relevant experience of safety review for the power uprates and regulatory review and approval process is not clear. But a request for power uprate would be considered to be an application for an operating licensing amendment, the overall process for which is covered by Korean Atomic Energy Act Article 21, corresponding Enforcement Decree Article 34 and corresponding Enforcement Regulation Article 17.

5.7.2 Technical basis supporting the regulatory approach

The technical bases supporting the regulatory reviews of power uprates are effectively the same as those that have been developed to support the original licensing review of a nuclear power plant, since all applicable regulations must continue to be met.

5.7.3 Status of resolution

No specific safety issues have been identified for resolution

5.7.4 Research needed to support modifications

Additional research needs have not been identified.

5.7.5 Schedule for completion research and regulatory action

Korea Institute of Nuclear Safety had initiated a research project in Apr. 2002. The purposes of this project are 1) Power uprate methodology, 2) Current regulatory status in US and foreign countries, 3) Case study of safety review and approval process, and 4) Implementation statues of open items. This project will be completed at the end of this year.

5.8 SWEDEN

5.8.1 Current regulatory approach

There have been a number of applications for power uprates in Sweden. It is the government that takes the decisions on power uprates. The role of the regulatory body is to review the applications and to give recommendations to the government. The review of the application is made in several steps; first an overall review that is the basis for the recommendation to the government, thereafter follows a detailed review of the safety analysis report and the uprate procedure.

The regulatory approach is that a positive recommendation is given to the government if the overall safety is maintained at current level or improved. Potential negative impacts on safety have to be compensated by measures that are positive for safety. In general modernisation programs that are implemented in order to extend plant lifetime, accompany the power uprates. Both conservative and best-estimate plus uncertainty methodology have been used in Sweden.

5.8.2 Technical basis supporting the regulatory approach

For BWRs the essential technical basis for power uprates is the development of new types of fuel. The development from 8X8 to 10X10-fuel has created additional margins so that more power may be produced from each fuel assembly with unchanged safety margins. Power increases in PWRs are possible because of steam generator exchanges. Smaller power uprates may also be achieved by reducing uncertainty of the measurement of the reactor power.

5.8.3 Status of resolution

The current status is that the government has approved the first power uprates and a detailed review of the Safety Analyses Report is taking place. Thereafter the procedures for the actual power uprates will be assessed. The test program of the reactors at the uprate power will be reviewed. The reviews also include assessment of the results from the tests at the higher power. The power uprates will be carried out in the period from 2006 through 2011. Preparedness to handle effects at the new power will have to be maintained during this period.

5.8.4 Research needed to support modifications

Significant research has been initiated in order to support the safety review. In particular emphasis is given to application of coupled neutron kinetic and thermal hydraulic analyses in three dimensions. The main focus will be on transients that frequently occur in Swedish nuclear power plants. The effect of increasing the power level for such incidents will be analysed. Such transients may have the potential to cause an accident if they occur together with other failures. Examples of incidents that have occurred include feed water transients, steam line closures, instability occurrences and reactivity insertion. Also some hypothetical accidents which may be associated with a significant risk will be analysed.

The research supporting the SKI review of power uprates will mostly be carried out at Swedish universities.

5.8.5 *Schedule for completion of research and regulatory action*

A schedule for completion of safety research supporting power uprates cannot be given. Such research would have to continue as long as we work with power uprates. It is also believed that there is a significant spin off of this research to other areas of importance to safety such as analyses of incidents.

5.9 UNITED STATES

5.9.1 *Current USNRC regulatory approach*

The NRC categorizes power uprates into three categories. "Measurement uncertainty recapture" uprates take advantage of improved reactor power measurement techniques, and are generally limited to no more than 2%. "Stretch" power uprates, up to about 7%, are increases considered to be within the design capacity of the plant. The percentage is plant- specific and depends on operating margins incorporated in the plant design. Changes to instrument setpoints are made, but major plant modifications are generally not required. "Extended" uprates are also plant-specific, and may be up to approximately 20%. In these cases, significant modifications are generally required to plant equipment, usually on the secondary side, to allow the turbine and associated systems to handle higher steam/water flows.

A request for a power uprate is considered to be an application for an operating license amendment, the overall process for which is covered by NRC Regulations 10 CFR 50.90, 50.91, and 50.92. Power uprate requests are reviewed by the Office of Nuclear Reactor Regulation. The overall process for uprate reviews depends on the category into which the uprate falls.

Reactor vendors have developed generic guidance for licensees who apply for power uprates. The guidance includes analytical methods that the NRC has reviewed and has determined to be acceptable. In addition, review guidance for the NRC staff has been developed for measurement uncertainty recapture and extended power uprates. In general, the review guidance refers back to the NRC's Standard Review Plant (NUREG-0800), and is intended to provide for a comprehensive evaluation of potential impacts on operational and safety system performance to ensure that the NRC's regulations and associated limits and acceptance criteria continue to be met.

5.9.2 *Technical basis supporting the regulatory approach*

The technical bases supporting the regulatory reviews of power uprates are effectively the same as those that have been developed to support the original licensing review of a nuclear power plant, since all applicable regulations must continue to be met. Improved analytical techniques that are used to quantify the excess margins that are available for increased reactor power must meet applicable NRC regulatory requirements.

5.9.3 *Status of resolution*

The NRC has approved over 100 power uprates since 1977. Ten applications are currently under review, and more than 20 additional applications are expected in the next several years. The uprate percentage has ranged from less than 1% to 20%. In general, few issues have arisen as a result of increased reactor power. However, some BWRs have experienced degradation and/or failure of the steam dryer in the upper part of the reactor vessel, which has been attributed to the higher steam flows associated with the uprate. The steam dryer is not safety-related; nonetheless, its failure could potentially have a significant impact on safety system performance.

Although there are no specific safety issues that have been identified for resolution, assessment of the impact of power uprates on plant operational and safety performance is incorporated into various Office of Research (RES) programs.

5.9.4 Research needed to support modifications

RES conducts assessments of operating experience to ensure that impacts such as steam dryer failures are appropriately evaluated. As part of this effort, a Safety Margins Research Project has been initiated, with the objective of providing a methodology that can assess the integrated effects of various changes in plant operating conditions, including power uprates, longer refueling intervals, and license renewal (i.e., plant aging). The impacts of power uprates in areas such as ECCS performance (10 CFR 50.46), time available for operator actions, and severe accident behavior are also being evaluated as part of RES programs in those areas.

5.9.5 Schedule for completion of research and regulatory action

As indicated above, there is no specific regulatory action that is considered to be required to undertaken for this issue. The schedules for research associated with power uprates is tied to the individual issues into which this consideration is included. The safety margins methodology is in progress, and is also associated with a CSNI-sponsored effort. A NUREG report on the safety margins methodology is scheduled for publication in June 2006.

Chapter VI

Issue: ECCS Strainer Clogging

Issue description

During loss-of-coolant accident (LOCA) scenarios, long-term emergency core cooling and containment cooling capability is provided by recirculation from either the containment sump (PWRs) or the suppression pool (BWRs). However, debris may accumulate in these areas, either through poor "housekeeping" practices or as a result of the LOCA itself, whereby the impact of effluent from the break causes material (e.g., insulation, paint) to become dislodged from structures and components and accumulate in the sump or suppression pool. This debris can then clog the strainers or screens in these areas and sufficiently degrade or prevent recirculatory cooling, thus leading to inadequate heat removal from the fuel and, potentially, severe core damage.

6.1 BELGIUM

6.1.1 Current regulatory approach

Belgium has 7 PWRs in operation. They began operation between 1974 and 1985. They were designed using the American rules and regulations applicable at that time. Therefore the sump screens were designed according to the USNRC requirements.

The four most recent units (Doel 3 & 4 and Tihange 2 & 3) took RG 1.82 rev. 0 into account for the recirculation sump design. Several improvements were made in the framework of the first PSR to the oldest units (Doel 1&2 and Tihange 1) such as enlargement of the sumps strainer in 1985, taking into account the requirements of RG 1.82 rev. 0.

The problem of recirculation sump screen clogging due to the accumulation of insulation debris after an accident was also identified as a safety issue in the first PSR for Doel 3&4 and Tihange 2&3 and the second PSR for Doel 1&2 and Tihange 1, that started in the nineties. The licensee reviewed the characteristics of the sump strainers according to the revision 1 of the RG 1.82.

Later on, the Barsebäck incident and the events leading to the NRC IN 93-34 showed that the analyses performed in accordance with the RG 1.82 could underestimate the potential for loss of NPSH of the ECCS and CSS pumps. After issuance of the results from the parametric evaluation for PWR recirculation sump performance (GSI-191) performed by LANL under NRC contract (NUREG/CR-6762) as well as other international information, AVN asked the Belgian licensee in June 2002 to increase its efforts to investigate and to solve this issue.

6.1.2 Technical basis supporting the regulatory approach

Same as in the US

6.1.3 Status of resolution

The action plan proposed by the licensee, and approved by AVN, covers the following topics:

- walk-downs in the reactor building of the different Belgian units;
- support to the operators related to the potential loss of recirculation;
- evaluation of modifications performed in foreign countries;
- evaluation of the applicability of the technical assessment performed by LANL to the Belgian NPPs;
- follow-up of research activities and experimental work related to the strainer clogging issue.

The methodology used to perform the walk-downs was mainly based on the document issued by NEI. The main objective of these walk-downs is to collect information allowing to perform further evaluations on the potential clogging of sump strainers. The items looked at are for example: the sumps and their strainers, the ECCS components, the containment compartments, the paints and other coatings, all materials that could significantly impair recirculation.

The utility also examined the possibility to increase the strainer area and has started a programme to increase it as far as reasonably possible in all units, without calculating the minimum required surface at this stage. The structural integrity of the strainers will also be improved. Later, an evaluation will have to be performed to demonstrate that the issue is resolved. If it is found that the new sumps are still not adequate, further modifications will have to be considered.

In addition, studies are ongoing to develop tools and guidelines in order to:

- assist the operator in identifying the clogging of the strainers;
- propose alternatives or corrective measures to be taken when it occurs;
- examine possible modification of existing procedures in order to delay the switch over in recirculation mode;

6.1.4 Research needed to support modifications

No specific research is planned in Belgium, but the utility plans to join the research programmes of AREVA (Framatome ANP) and WOG EPRI (long term chemical effects).

6.1.5 Schedule for completion of research and regulatory action

No schedule has been set.

6.2 CZECH REPUBLIC

6.2.1 Current regulatory approach

As a response to Barseback event in 1992 and in the framework of the 1st Periodic Safety Review (PSR) after 10 years of operation of Dukovany NPP, the regulatory body – SONS issued in 1995 the operational licence for the next 10 years . This licence contained among others conditions regarding the sump screen clogging which required to assess the vulnerability of sump screen clogging and to realise necessary corrective actions.

6.2.2 Technical basis supporting regulatory approach

Dukovany NPP

In 1996 -1998 a number of analyses and tests were performed:

Estimates of the quantity of dislodged insulation were made in accordance with the 7L/D double cone criterion for LB LOCA with guillotine break of the main circulation pipe located at the connection of the cold leg to the SG collector.

Jet impingement tests and studies of insulation fragmentation were performed on the dynamic experimental facility of GOSNICAES. Kashira, Russia.

Characteristics of the insulation materials and debris types, and rate of debris transport to the screen strainers were investigated at the Research Institute of Energetic Installations (VUEZ), Levice, Slovakia. The same organisation performed a series of strainer clogging tests on the mock-ups of the original and modified screens.

Analyses of the pressure and water level during various LOCA accidents, for assessing and setting the acceptable values for these parameters.

It was shown that screens of the original design were not capable to fulfil the required safety function in the whole expected range, and that corrective modifications were necessary. The final and most important modifications of the screens were:

The original multi-row arrangement of internal strainers was replaced by module arrangement and the inlet strainer with 10 x 10 mm mesh was replaced by perforated metal sheet with 10 mm diameter holes. This arrangement has also a good gravitational self-cleaning effect.

The total area of internal strainers was increased about four times.

Finer wire with square mesh was used for inner strainers.

Temelin NPP

Analyses of the strainers clogging risk in Temelin NPP were performed, in compliance with request from the SONS, within the framework of the plant modernisation during its construction. The range of the risk was evaluated by in a similar approach as for NPP Dukovany, in accordance with US NRC RG 1.82, Rev. 1, and for the quantification of dislodged insulation the 7L/D criterion was used. The reference accident used in the dislodged insulation quantification was LB LOCA with double-ended cold leg break under the steam generator.

The range of analyses and tests was essentially the same as for NPP Dukovany, and experimental tests were performed at the same organisations. The first stage of analyses and tests has indicated that the strainer capacity is adequate. For the final experimental verification, VUEZ has built a mock-up width-scaled 1:6 and height-scaled 1:4. All six strainers were modelled within 10 separate tests, which verified the efficiency of the strainer elements.

6.2.3 Status of resolution

Dukovany NPP

The original area and volume parameters of screens were unchanged, similarly as the total water flow through the screens. The screen inlet is protected by a cover grid structure. The water flow was reduced from the original 6 cm/s to 1,4 cm/s. Each sump is equipped with water level measuring device working with a binary signal and providing indications and alarms to both the main and the emergency control rooms. The mentioned modifications were fully implemented at all units.

Temelin NPP

The results proved that the strainer functionality is sufficient and that no hardware modification is needed. In the present design, the water level is measured in front of the coarse inlet strainer and after the last fine one, by three independent measuring channels (qualified for LOCA conditions). The water level is indicated in the main and the emergency control rooms. The EOPs were appropriately revised and give the possibility to operator to switch off surplus pumps of ECCS with the aim to decrease carry away of the insulation debris to the screens.

6.2.4 Research needed to support modifications

Czech organisations intent to participate at new proposed joint research NEA project on strainer clogging.

6.3 FINLAND

6.3.1 Current regulatory approach

Finnish YVL-requirements provides basis for further demands concerning ECCS strainer performance. Licensees have to demonstrate, that long-term emergency core cooling capability is ensured by recirculation from the sump. This issue was "opened" in the early 1990's, and the technical approach was to perform test to estimate strainer pressure drop, learn basic phenomenology and then develop robust filter designs and assess efficiency of countermeasures. Also alternative insulation materials were tested.

The old design criteria were reassessed and a new set was developed:

- debris generation: criteria for PWR is 7 L/D 100% destruction, BWR double cone to limiting walls.
- 100 % debris transportation
- head loss from experimental correlations for prototypic sump/strainer geometries
- require comfortable margin to NPSH at worst load (factor of 1.5 to 2)
- require backflushing for long-term

6.3.2 Technical basis supporting to regulatory approach

During 1990's, number of tests were performed by domestic utility Fortum to estimate the debris generation, transportation, clogging of the sump screens and backflushing performance. The designs of new sump screens were based on these tests.

Additional justification for the design was received from international test programs which are presented in NUREG reports and others.

6.3.3 Status of resolution

In Loviisa plants, the emergency sump structures of the containment have been completely re-designed after a foreign operational event indicated that the original design had essential deficiencies. The new strainer structures of the sumps are designed to collect the largest possible amount of damaged insulators without disturbing the emergency core cooling function. In addition, the new sump strainers have been equipped with backflushing system to ensure the long-term function.

As a result of the sump modification, a need has also been noted to evaluate more closely the functioning of the high pressure safety injection pumps during a sump circulation. That is why the functioning of the pumps have been examined with water including insulator-impurities.

In Olkiluoto 1 & 2 plants, strainers have initially large strainers area and needed no modification. However, after the evaluation of the performance in a case of LBLOCA, the backflushing system was much improved and insulation materials were partly replaced.

6.3.4 Research needed to support modifications

There are no specific research needs for present NPP units. Strainer validation tests for coming Olkiluoto 3 unit were performed at the end of 2004.

Finland will be participating international research programmes.

6.3.5 Schedule for completion of research and regulatory action

Not applicable.

6.4 FRANCE

6.4.1 Debris impact on emergency coolant recirculation

IRSN considered that, in case of both primary coolant circuit and secondary circuit rupture, the risk of reactor building sumps clogging can impact simultaneously the emergency core cooling system (ECSS) and emergency containment spray system (ECSS) availability. Accounting for this potential risk, the standing group of experts stated that this phenomenon had to be studied in priority and for all the French PWRs. It has also been stated that the debris impact on emergency coolant recirculation assessment must take into account all the different sizes of primary or secondary pipe breaks inside containment. All the studies were aimed at evaluating the risk of loosing SIS and ECSS pumps due either to head loss or air ingestion, as well as the consequences on the SIS and ECSS equipment and on the reactor core.

6.5 GERMANY

6.5.1. Current regulatory approach

The regulatory requirements on residual heat removal in case of LOCAs are included in the BMI safety criteria, in the incident guidelines, in the RSK guidelines and the KTA safety standards. Special attention has been given to the sump clogging issue in the RSK recommendations of the 320th and 374th meeting.

The BMI safety criteria require a reliable emergency core cooling system. Specified limits should not be exceeded in case of a single failure and in case of maintenance for fuel elements, for core internals and for the containment initiated by breaks of different sizes, break locations and at different operating conditions. In addition, the residual-heat removal system has to be designed in such way that the pressure and temperature in the containment will be reduced in the long term.

In the RSK statement of the 320th meeting, it was required that

- a 0.1 leak can to be assumed for pipes with break preclusion,
- dislodging of insulation material shall be determined with the cone model of the US-NRC,
- the retention in the containment is 50 %, and

In 2004 at the 374th meeting, a revision of the RSK-statement of 1998 has been released:

Basic principles are:

- Recommendations can be applied to PWRs and partially to BWRs
- A fully analytical treatment is not possible
- Postulates take into account existing uncertainties
- Experimental demonstrations for e.g. transportation rate in sump water and head loss across sump strainer are needed
- No accident management measures should be applied at design basis accident conditions,
- Accident management measures have to be planned to assure sufficient core cooling in case of sump clogging,
- The sump strainers have to be designed in such a way, that debris will deposit preferably at the sump screen

Recommendations are e.g.

- 60% to 80% of the insulation material sediments in the sump,
- The emergency core cooling system has to be designed in such a way that there is no steam flow out of the core in the sump recirculation mode,
- For the evaluation of the available suction head over pressure in the containment can be applied under conservative assumptions in case of pressure losses in layers of mobilised insulation and latent debris deposited on the screens.

On the basis of the RSK statement of the 320th and 374th meeting, the design of the sump strainer has to be checked for all German nuclear power plants whether the safety margins of the sump strainer design are sufficient to ensure core cooling under the boundary conditions specified by the RSK.

6.5.2. Technical basis supporting regulatory approach

After the Barsebäck event in July 1992, GRS prepared Information Notice 14/92 in October 1992 in which it was recommended for all BWR plants to check the functioning of residual heat removal under the conditions of the incident. In addition the RSK has proposed that corresponding investigations should also be performed for PWR plants, including experiments on dislodging and transportation of insulation material.

Detailed analytical and experimental investigations e.g. on dislodging of insulation material (Kaefer tests) and on the transport of insulation material (ABB container tests) have been performed based upon small scale tests in the years before 1998.

In 1996, the OECD report “Knowledge Base for Emergency Core Cooling System Recirculation Reliability“ was published with the support of GRS.

In 1998, the RSK passed a statement on the 320th meeting in which the actual knowledge was summarised in a comprehensive manner.

Results of US-NRC analytical and experimental investigations concerning the GSI 191: ‘PWR sump performance’ indicated in 2001 that some aspects of the strainer clogging issue may have been underestimated. The resolution of GSI 191 is scheduled by the US-NRC at the end of 2007.

In order to verify small scale test results from years before 1998 the German licences have performed large scale tests in 2001:
These tests cover the following aspects:

- Selection of representative insulation material
- Debris generation and size distribution
- Transport in the sump
- Sump strainer head loss
- Sump strainer penetration
- Clogging of the fuel elements by penetrated debris.

In 2004, the RSK published recommendations on the 374th meeting taking into account the new insights. Plant and material specific validation tests have been required by the RSK, too. In addition phenomena like the thin bed effect or corrosion of structures in the sump have to be investigated more in detail.

6.5.3. Status of resolution and Research needed to support modifications:

Following the recommendations from the 374th RSK-meeting AREVA performs experiments on the transport of insulation material in the sump and on deposition of insulation material at sump strainers by means of plant-specific combinations of dislodged insulation material, other materials and plant-specific strainer configurations.

According to the statement of the RSK in the 374th meeting there are still some aspects in the sump clogging issue, which need further improvements in order to reduce uncertainties. Long term experiments should be performed to investigate the influence of the water chemistry and corrosion products. A major aspect of concern is the thin bed effect of fine fibrous material and of mixtures of fibrous with micro porous material.

As a consequence the German federal government (BMWA) is supporting experiments in German research centres. These tests, which are currently performed, cover e.g. the following topics:

- Size, form and distribution of dislodged insulation particles,
- Sink behaviour of insulation material in a static water column,
- Flow behaviour for the transport of insulation material in horizontal coolant flow for determining sedimentation and resuspension,
- Experiments on deposition and penetration behaviour at the sump strainers,
- Integral experiments on the penetration of the break mass flow into the sump (jet, plume, shower) and its influence on the flow behaviour in the sump,
- Development of single effect models for CFD simulation of a particle/water two-phase flow, their validation by means of single effect experiments and their applicability for the description of all transport phenomena in the sump.

On behalf of the BMU, GRS adapts its computer codes for the cooling circuit and the containment to the requirements on LOCAs with dislodging of insulation material for determining the thermal hydraulic boundary conditions for verifying sufficient core cooling, load transfer at the sump strainers and on the cavitation margin at the emergency cooling pumps with sufficient accuracy.

In addition, strategies and procedures have to be developed in the long-term (approximately 10 h) to prevent high head loss across the sump screens as well as related accident management measures.

6.5.4 Schedule for completion research and regulatory action

On the basis of the RSK statement of the 320th and 374th meeting, the design of the sump strainer has to be checked for all German nuclear power plants whether the safety margins of the sump strainer design are sufficient to ensure core cooling under the boundary conditions specified by the RSK.

There are some investigations, which are not finished. It can't be foreseen in which way the results of these national tests and international investigations like the US-NRC activities on GSI 191 may influence regulatory actions

6.6 JAPAN

6.6.1 Current Regulatory Approach

Japanese regulations require that the capability for long-term emergency cooling be maintained after a LOCA. The requirements for this are reflected in the following regulations:

- Acceptance criteria in the ECCS evaluation guide
- Technical standards on nuclear power generation equipment Article 17 (ministerial order 62)

During 1994-95, ANRE (the former agency corresponding to NISA), then Japanese regulatory authority, judged that the regulated NPPs retain sufficient safety margins against these issues. The major bases for judgments are:
- Foreign materials are adequately managed by Japanese licensees.
- Materials of thermal insulators in CVs at Japanese NPPs are different from those of troubled NPPs, and robust for massive generation of debris.

Then ANRE concluded that the LOCA- generated debris would not plug up strainers and jeopardize the ECCS cooling capabilities.

During October 2003 – June 2004, examination results of several BWR plants, reported by utility to NISA in response to the NISA' re-examination orders, revealed the existence of variety of foreign materials including fragments of mending tapes, pieces of cloth and tissues, tools etc.
This triggered the present re-assessment and review action of NISA. NISA had an attention that the US had analyzed and assessed the issues by means of newly developed conservative models. These acclivities culminated in licensees' action to provide larger scale strainers to resolve the BWR problems by the end of fiscal year 2007.

6.6.2 Technical Basis Supporting the Regulatory Approach

Implementation of evaluation method for ECCS strainer clogging issue based on the NRC developed BLOCKAGE code has been conducted. The JNES has been in charge of performing the necessary modification and application for the representative BWR plants and the evaluation of the effectiveness on the current type BWR strainers were completed. The modification is under way for the evaluation of a larger scale new type BWR strainers and furthermore for PWR plants for the next step.

6.6.3 Status of Resolution

In the period of June 2004 – March 2005, utilities for representative BWRs (3 CV types) started the examination of plant insulators, the assessment of strainer effectiveness, the analyses of clogging effects, etc. Then some of the utilities reported the results to NISA that under the conservative evaluation conditions the strainer effectiveness were not secured for almost all of their plants and provisional action would be taken.

The NISA checked utilities' methodologies, results of examination and analysis, and remedial actions with the help of quantitative analyses performed by JNES. Then, in the safety evaluation workgroup, the interim draft was issued concerning the validity of strainer effectiveness evaluation method in BWR, the validity of the measures on equipments, the contents of a technical standard, etc. For BWR utilities, implementation of the measure was directed by NISA on the equipment which reflects the evaluation of the effectiveness of the strainer. For PWR utilities, presentation of a report was directed on the effectiveness and the validity of the evaluation method of a sump screen.

6.6.4 Research Needed to Support Modifications

The following researches are considered to be needed to support modifications;
- experiments to obtain reliable head loss correlations for combinations of materials in Japanese plant,
- experiments for generation of clogging material for PWR,
- experiments for chemical effects and downstream effects,
- qualification of the BLOCKAGE code to evaluate the BWR strainer clogging issue,
- modification of the BLOCKAGE code to evaluate PWR sump clogging issue
- methodology using PSA.

6.5.5 Schedule for Completion of Research and Regulatory Action

BWR: Measures will be fully implemented by the end of fiscal year 2007.
PWR: Direction will be issued after completion of the assessment of evaluation method and the improvement of examination guide.

6.7 KOREA

6.7.1 Current regulatory approach

The regulatory rule applicable to this issue is specified in

1) No. 3 of Article 30, (Emergency Core Cooling System), No 16, Regulations on Technical Standards for Nuclear Reactor Facilities, ETC, Enacted by the Ordinance of the Ministry of Science and Technology, (amended by July 2001)
2) No. 5 of Article 3 (Acceptance Criteria), MOST Notice 2001-39 "Performance Standards on Emergency Core Cooling System in Pressurized Water Reactors". Dec. 2001.

Assurance of Long Term Cooling following LOCA was required in these requirements, as similar to the 10 CFR 50.46 of USA.

USNRC RG 1.82 (Rev.1, 2, and 3) have been regarded as detailed technical standard for sump design and have been applied to Korean NPP. For the old plants built before this issue, the full compliance with the associated rules and guides may not be expected.

The plan and method to resolve this issue has been requested during the Periodic Safety Review of Kori Unit 1 and Unit 2. Also, the regulatory inspection of containment cleanness has been conducted in depth level for the most Korean NPP.

KINS will evaluate the necessity of additional requirements for resolving the sump clogging issue considering the result of the assessment of the Korean NPP and foreign reference plants and environments.

6.7.2 Technical basis supporting the regulatory approach

KINS staff is reviewing the USNRC regulatory documents including Information Notices, Bulletins, NUREGs, GL's, RG's, and the related reference materials including NEI method. The technical contents will be used for the regulation on this issue.

6.7.3 Status of resolution

Eight Korean Standard Nuclear Power Plants (KSNP) have an identical containment layout and sump design, Also the same insulation material has been commonly used. The regulatory position in RG 1.82 Rev.1 (1982) was already applied and the compliance of the requirements was required for those plants. However, the compliance with the additional requirements in RG 1.82 Rev.3 (2003) has not been evaluated. Since the clogging issue has never been evaluated for the existing plants before KSNP a certain vulnerability to the clogging issue may be existed.

Currently, KINS has continued to collect the plant information including amount of insulation material needed to assess the issue. KINS plans to evaluate the debris impact in plant specific way for the some selected NPP. After this evaluation, the appropriate resolution can be sought.

6.7.4 Research needed to support modifications

No specific research is fixed yet, however the participation of the international joint research program, e.g. OECD, and the required domestic research program are being considered in the future national research plan.

6.7.5 Schedule for completion research and regulatory action

No schedule has been set.

6.8 SWEDEN

6.8.1 Current regulatory approach

Background information is given in [1]. On July 28, 1982, a steam line loss-of coolant accident occurred when a safety relief valve inadvertently opened in the Barsebäck-2 boiling water reactor. The steam jet stripped fibrous insulation from adjacent pipework. A part of this insulation was transported to the wetwell pool and clogged the intake strainers for the drywell spray system after about one hour. This was significantly shorter than previously anticipated. Although the incident in itself was not serious, it revealed a weakness which under other circumstances could have led to the emergency core cooling failing to provide water to the core. It was decided that 5 of the reactors were especially vulnerable and that those would remain closed until the ECCS strainer clogging issue had been resolved.

The Swedish regulatory approach for dealing with LOCA generated debris was based on the robust design approach. It consists of a few very simple assumptions that aim to make the strainer head loss evaluation very conservative. For the robust design approach several processes were reviewed in order to quantify the amount of debris generated after a LOCA that would clog the strainers.

For BWRs the following assumptions were used in order to provide the overall robust solution:

- Amount of material dislodged was estimated using a conservative model
- All dislodged material was transported to the wetwell pool
- All this material would settle on the strainers of one train
- Conservative data for pressure drops.
- Net positive suction head (NPSH) for the pumps shall exist.
- Capability to maintain strainer flow during operation and instrumentation to monitor the pressure drops should be provided.

These assumptions led to strainers with very large areas. Cleaning capability through backflushing was provided to all strainers.

One of the PWRs has fibrous insulation on the steam generators. Two PWRs had essentially reflective metallic insulation. Similar assumptions as for BWRs were provided for the PWRs:

- All insulation material on one loop inside the missile protection will be dislodged
- 2/3 of this material would accumulate on the strainers for one train

The rest of the assumptions were the same as for BWRs. This led also for the PWRs to strainers with very large areas. A part of a PWR strainer is sacrificial and a part of the strainer area may be cleaned during operation.

6.8.2 Technical basis supporting the regulatory approach

The technical basis supporting the robust regulatory approach was basically experimental. Most data were produced by the Swedish utilities. The experiments revealed a number of uncertainties that had not earlier been taken into account, which contributed to the selection of robust solutions. These uncertainties made it difficult to determine upper bound for certain phenomena. The major uncertainties are listed below.

Dislodged material and drywell transport
It was found that the uncertainties in the amount of dislodged material could be very high. Some of the separate effects showed much higher retention in the drywell than observed in Barsebäck. It was therefore difficult to obtain confidence in such data.

Type of insulation tested
The new experimental information was obtained using insulation material that had been used in reactors. Contrary to new insulation material, old material would tend to stay suspended in the water for an extended time and thus be more available for strainer clogging. Earlier tested new insulation material tended first to float on the water surface and thereafter quickly sink to the bottom.

Wetwell transportation
The wetwell transportation uncertainties are dependent on sedimentation and resuspension. Parameters like turbulence under post-LOCA conditions are of importance. It was also observed that larger particles may settle and provide enrichment of finer particles as they approach the strainers.

PWR recirculation
Testing of a mock-up of the power part of a PWR containment revealed that recirculation could result in waterfalls that could increase the turbulence and further fragmentation of the debris.

Composition of beds on strainer
It was also observed that preparation of the samples would affect the pressure drops. In particular, steam blown material gave higher pressure drop. Also mixtures of particles and fibres could give much higher pressure drops that pure fibre beds. The final decision to require cleaning capabilities even for large strainers was based on some long term experiments, carried out in a couple of weeks time frame, that showed that the pressure drops could continue to increase by bed compaction, chemical effects, or some other effects.

6.8.3 Status of resolution

The strainers have been replaced in all Swedish reactors. The research showed that there were large uncertainties which necessitated choice of robust solutions with large conservatisms. The research has provided resolution of an important safety issue.

6.8.4 Research needed to support modifications

Further research may be needed to understand importance of, for instance, chemical effects for the differential pressure drops over the strainers. When changes are made in the containment chemistry it may be necessary also to consider the effects on strainer clogging.

6.8.5 Schedule for completion of research and regulatory action

The research on strainer clogging is essentially completed.

Reference [1]. Knowledge Base for Emergency Core Cooling System Recirculation Reliability. NEA/CSNI/R (95)11

6.9 UNITED STATES

6.9.1 Current USNRC regulatory approach

NRC regulations require that the capability for long-term emergency cooling be maintained after a LOCA. These requirements are reflected in the following regulations:

- 10 CFR 50.46(b)(5) [Emergency Core Cooling System Acceptance Criteria]
- 10 CFR Part 50, Appendix A, General Design Criterion 35 [Emergency Core Cooling]
- 10 CFR Part 50, Appendix A, General Design Criterion 38 [Containment Heat Removal]

Notwithstanding these requirements, however, the NRC has studied the issue of sump performance and sump clogging for approximately 25 years. Unresolved Safety Issue (USI) A-43 was established in 1979 to examine the issue of PWR recirculation sump designs. The resolution of this issue was documented in Generic Letter (GL) 85-22, “Potential for Loss of Post-LOCA Recirculatory Capability Due to Insulation Debris Blockage.” The technical findings from this work were published in NUREG-0897 (see “Technical Basis,” below), and guidance reflecting these finds was published in Regulatory Guide (RG) 1.82, “Sumps for Emergency Core Cooling and Containment Spray Systems,” and in the NRC’s Standard Review Plan (NUREG-0800). RG 1.82 has been updated further since 1985, as a result of new work associated with

Generic Safety Issue (GSI) 191, with Revision 3 having been issued in November 1983. GSI-191 is described further, below.

After the resolution of USI A-43, several events involving containment/ECCS strainer clogging occurred at BWRs. As a result, additional guidance was issued to BWR licensees, including Bulletin 93-02, Supplement 1, “Debris Plugging of Emergency Core Cooling Suction Strainers,” Bulletin 95-02, “Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode,” and Bulletin 96-03, “Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors.” These bulletins requested that BWR licensees implement appropriate procedural measures, maintenance practices, and plant modifications to minimize the potential for debris clogging in BWRs, and the NRC staff has concluded that all BWR licensees have addressed these issues adequately.

However, research findings from the BWR strainer clogging work again raised questions concerning the adequacy of PWR sump designs, primarily as a result of the amount and physical characteristics of the debris that could be generated during a high-energy line break (HELB). Consequently, GSI-191, “Assessment of Debris Accumulation on PWR Sump Performance” was opened by the NRC staff. The initial technical assessment of GSI-191 was completed in 2003, and Bulletin 2003-01, “Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors,” was issued in June 2003, requesting that licensees demonstrate compliance with regulatory requirements related to recirculatory cooling capability, or alternatively, to implement compensatory measures to reduce the risk from sump-clogging scenarios. Additional guidance has recently been issued in GL 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors,” requesting additional evaluations of the potential impact of sump clogging as a result of technical findings from GSI-191 research. Additional research is continuing on these issues.

6.9.2 Technical basis supporting the regulatory approach

Since 1979, the NRC has issued numerous reports, letters, notices, and bulletins related to the subject of sump clogging and long-term ECCS performance. Some of the key documents are listed below.

The technical basis for the regulatory guidance provided in the resolution of USI A-43 is contained in NUREG-0897, “Containment Emergency Sump Performance,” October 1985. Findings related to the BWR strainer clogging issue are reported in NUREG/CR-6224, “Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA-Generated Debris,” October 1995.

In addition to the generic communications and RGs noted above, the following documents report the results of technical assessments and research related to GSI-191:

- NUREG/CR-6770, “GSI-191: Thermal-Hydraulic Response of PWR Reactor Coolant System and Containments to Selected Accident Sequences”
- NUREG/CR-6771, “GSI-191: The Impact of Debris Induced Loss of ECCS Recirculation on PWR Core Damage Frequency
- NUREG/CR-6772, Vol. 1, “GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance”
- NUREG/CR-6772, Vol. 2, “GSI-191 Technical Assessment: Summary and Analysis of U.S. Pressurized Water Reactor Industry Survey Responses and Responses to GL 97-04"

- NUREG/CR-6762, Vol. 3, "GSI-191Technical Assessment: Development of Debris Generation Quantities In Support of the Parametric Evaluation"

- NUREG/CR-6762, Vol. 4, GSI-191 Technical Assessment: Development of Debris Transport Fractions In Support of the Parametric Evaluation Aug 2002

- NUREG/CR-6808, "Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump Performance"

- NUREG/CR-6868, Small-Scale Experiments: Effects of Chemical Reactions on Debris-Bed Head Loss, March 2005

- NUREG/CR-6874, GSI-191: Experimental Studies of Loss-of-Coolant-Accident-Generated Debris Accumulation and Head Loss with Emphasis on the Effects of Calcium Silicate Insulation, May 2005

- NUREG/CR-6877, Characterization and Head-Loss Testing of Latent Debris from Pressurized-Water-Reactor Containment Buildings, July 2005

- NUREG/CR-6885, Screen Penetration Test Report, October 2005.

The above list represents only a sample of the extensive compendium of technical information, regulatory guidance, and information provided by the NRC to licensees. The NRC has developed a website providing information on the subject of PWR sump performance with a complete listing of the documentation developed over the past 25 years. The address for the website is:

http://www.nrc.gov/reactors/operating/ops-experience/pwr-sump-performance.html

6.9.3 Status of resolution

Research is continuing with the objective of further developing the technical basis for resolution of this generic issue. In addition, the NRC will review licensee responses to GL 2004-02, conduct inspections at operating PWRs, and audit plant analyses and/or modifications performed in connection with the Generic Letter. The ongoing testing and analyses, as detailed in the next section, is investigating several technical issues associated with NRC regulatory concerns.

6.9.4 Research needed to support modifications

Past research included small-scale chemical effects tests, calcium silicate insulation debris tests, latent debris characterization and head loss tests, and screen penetration tests. More recently, downstream effects tests were performed to examine the effects of debris that flowed through the sump screens on throttle valves.

The integrated chemical effects testing (ICET) program was completed in Fall 2005. This major research effort involved a five-test series of 30-day tests designed to determine the extent to which post-LOCA sump pool environments generate chemical byproducts which could contribute to sump clogging. Results will be published as NUREG/CR reports in Spring 2006, but are currently summarized online at http://www.nrc.gov/reactors/operating/ops-experience/pwr-sump-performance/tech-references.html.

Research that is ongoing at this time is described below. Additional research may be conducted as needed to support the resolution of technical issues which may arise from the results of testing.

Chemical effects head loss This research effort continues the investigation of how chemical byproducts may cause recirculation pump head loss. With the ICET results as a basis, this work is testing the head loss contribution of chemical byproducts formed in post-LOCA environments. Initial scoping tests will investigate the maximum head loss severity of observed products. Follow-on testing will be conducted as necessary to determine how variables associated with containment pool environments may affect the head loss.

Debris head loss Currently, correlations are based on fibrous and metallic debris types based on work stemming from BWR sump resolution activities. Initial scoping tests indicated the need to conduct a more rigorous evaluation of the effects of calcium silicate in concert with other debris types, including coatings. This work will evaluate head loss associated with standard PWR containment debris materials to provide data for the development of improved analytical head loss correlations.

Coatings transport The objective of this work is to evaluate the transportability of coatings debris, in particular larger debris chips, to the containment pool sump screen. The effect of coatings debris variables, such as debris size, shape, density, thickness, and sump flow rate on transportability will be examined in a test flume.

6.9.5 Schedule for completion of research and regulatory action

Resolution of all issues associated with GSI-191 is scheduled for December 2007.

Chapter VII

Issue: Boron Dilution
Issue description

During certain small-break loss-of-coolant accident (SBLOCA) scenarios in pressurized water reactors (PWRs), cessation of coolant circulation accompanied by steam generation in the core and condensation in the steam generators may cause the accumulation of a substantial amount of deborated water in the reactor coolant system. If coolant circulation then begins, through either the re-establishment of natural convection or start-up of reactor coolant pumps, the deborated water may be pushed into the core, resulting in an increase of reactivity and, ultimately, recriticality. If the reactivity excursion is of sufficient magnitude, it may result in significant heat generation and fuel failure. Suppression of such scenarios is considered desirable. This issue is also connected, in part, to the issue of high-burnup fuel, because of the concern that, while boron-dilution-related reactivity excursions might not cause sufficient energy deposition to damage fuel with low or moderate burnups, high-burnup fuel could be more susceptible to damage in such cases.

7.1 BELGIUM

7.1.1 Current regulatory approach

The regulatory approach requires that during such accidents a coolable geometry and sufficient long term cooling should be maintained. A safe and stable condition must be reached with a sufficient level of subcriticality.

7.1.2 Technical basis supporting the regulatory approach

There is no boron related specific technical basis that supports the regulatory approach. The regulatory approach strategy is to provide adequate protection of the public health and safety after an accident. Research is currently in progress to estimate the recriticality occurence.

7.1.3 Status of resolution

To solve this issue it is necessary to perform experiments to investigate the dynamics of the deborated slug. Current experiments performed in the PKL facility in the context of the OECD SETH and OECD PKL projects are investigating this issue. The initial results have shown that after refill, during natural circulation, the deborated slug reaches the vessel inlet.

The deborated slug may also reach the vessel inlet by intermittent flow (without refill). Thermocouples downstream of the injection already revealed incomplete mixing of the hot deborated slug with the cold safety injection (and therefore of the deborated/borated water).

7.1.4 Research needed to support modifications

Since a deborated slug is expected to reach the vessel, further research should be focussed on the behaviour of the hot deborated slug in the downcomer and the lower plenum of the vessel.

7.1.5 Schedule for completion of research and regulatory action

The OECD SETH project will be terminated by 2004 while the OECD PKL project will be terminated by 2007 but no research action is foreseen that investigates the behaviour of the slug in the vessel.

No regulatory action is foreseen before the end of the PKL project.

7.2 CZECH REPUBLIC

7.2.1 Current regulatory approach

This issue was identified in the framework of the IAEA Extrabudgetary Program for safety enhancement of NPPs with WWER reactors in the "Issue books" for WWER-440/213 and WWER-1000/320 and ranked 1 as lowest safety importance (deviation from international practice).

The reason was that these specific scenarios were not found in SAR of WWER-440/213 and WWER-1000/320. Two main scenarios were identified small break (SB) LOCA and steam generator tube rupture (SGTR).

7.2.2 Technical basis supporting regulatory approach

Dukovany NPP

As a response to this issue the PHARE project Prevention of inadvertent primary circuit dilution was performed for WWERs – 440(213). The project selected the possible scenarios (by means of PSA), performed thermohydraulic analyses of the scenarios and evaluated increase to the CDF of the above scenarios and proposed corrective measures.

As a result of the project it was stated, that contribution of boron dilution scenarios to CDF is so low that corrective measures are not necessary. In spite of this fact the borometers were installed in the primary circuit and some improvements into abnormal operational procedures were elaborated.

Safety analyses for new fuel with gadolinium also analysed the events with potencial born dilution and /or densification in the primary circuit.

Temelin NPP

Temelin emergency operating procedure were developed based on proven and validated methodology of the Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERGs). All know safety issues are covered by the Temelin EOPs at the same level as they are addressed in the generic guidelines in line with U.S./Westinghouse practice.

For both scenarios, i.e., SB LOCA and SGTR accidents, well defined operator recovery actions exist within appropriate procedures to deal with possible boron-free water pocket or primary circuit rapid boron dilution.

The operator recovery actions following a SGTR or SGIMF are to isolate the ruptured steam generator, reduce pressure in the intact steam generators to cool down the primary circuit, and then reduce pressure in the RCS to equalize the ruptured steam generator primary and secondary pressures (stopping the leak).

A specific feature of the WWER-1000 design is the possibility to letdown the RCS directly from the SG collectors. This path can be used to letdown diluted RCS coolant before it can reach the core.

Another task specified by the Temelin EOPs is to monitor shutdown margin during unit cooling down, especially after boron injection flow from the safety systems is terminated. This shut down margin monitoring prevents a decrease of shutdown margin caused by a decrease of primary coolant boron concentration.

Monitoring of the shutdown margin during RCS cooling down after SBLOCA is also a part of the operators´ response, included in the Temelin procedure for SBLOCA. This action prevents an uncontrolled reactivity increase due to boron dilution.

Temelin is less susceptible than many other PWRs to this potential mechanism because of this relatively high capacity high pressure injection system. Temelin´s high capacity injection system reduces the range of possible size and location for these breaks.

7.2.3 Status of resolution

The safety issue has been solved.

7.2.4 Research needed to support modifications

No additional research needs have been identified.

7.3 FINLAND

7.3.1 Current regulatory approach

There are number of requirements in YVL guides related to recriticality of a reactor following an SBLOCA. These include:

- Decision of the Council of State (395/91), section 15: During postulated accidents, the rate of fuel failures shall remain low and fuel coolability shall not be endangered.
- Guide YVL 1.0: Coolable geometry and sufficient long-term cooling can be maintained.
- Guide YVL 1.0: The reactor and related systems must not contain specific features which could cause a significant reactivity increase in connection with an anticipated operational transient or postulated accident and thus aggravate the consequences of an event.

7.3.2. Technical basis supporting to regulatory approach

Results of PKL and UPTF- experiments and several technical reports have been used for technical basis supporting to regulatory approach.

7.3.3. Status of resolution

In the Loviisa plant, the following modifications in the systems and operation mode have been done for avoiding an unintended boron concentration dilution of the coolant:

- In the beginning of the fuel cycle borated water from a dedicated tank is used to dilute primary coolant. The boron content of the water is such that a possible boron dilution transient does not result in a reactivity accident.
- The dilution of the primary coolant will be interrupted in case of primary coolant pump stopping.
- Borating of the primary coolant will be started automatically in case of stopping of 4 or more primary coolant pumps.
- Before starting of a primary coolant pump the loop will be flushed with the counter-current flow through the loop.

The risk of the boron concentration dilution arising from external reasons has been reduced to an acceptable level with these measures. The safety significance of the inner boron dilution during some accident situations has been considered small based on Fortum's extensive assessments.

7.3.4. Research needed to support modifications

Finland is participating international research projects such us PKL.

7.3.5. Schedule for completion of research and regulatory action

Not applicable.

7.4 FRANCE

7.4.1 Inherent boron dilution in case of SB LOCA

Current methodology used by French Utility for inherent boron dilution studies adopts mixing assumptions based on UPTF test. The evaluation performed on this topic by IRSN in 2003 showed that the UPTF tests are not representative enough of French PWR as far as mixing phenomena are concerned. Thus, the proposed methodology hasn't been licensed by IRSN. In IRSN opinion, studies presented by French Utility do not permit completely excluding a core return to critically during SB LOCA. Therefore, it was asked to French Utility to restart and complete its analysis. In this framework, French utility is developing a new study methodology including a clarification of mixing phenomena in downcomer and lower plenum based on 3D calculations and a new quantification of plug volumes based on PKL tests, performed in 2004 and 2005.

7.5 GERMANY

Issue: Maintaining a sufficient boron concentration in case of small break LOCA

7.5.1. Current regulatory approach

There are new insights concerning safety relevant problems relating to the control of certain design basis LOCA. Newer thermal hydraulic calculations have shown that in case of the considered small break LOCA with heat transfer in reflux-condenser-mode the necessary concentration of boron is not continuously guaranteed and thus subcriticality may not be maintained.

For preventing inadvertent deboration during plant shutdown, technical and administrative measures are provided:

- A deboration during shutdown and existing recirculation flow can be prevented by monitoring the boron concentration and the neutron flux.
- An injection of deionised water into the primary circuit is prevented automatically after:
  - actuation of reactor scram,
  - switch-off of all reactor coolant pumps.
- Inadvertent boron dilution during refuelling is prevented by administrative measures. A list contains all valves in such systems which are connected with the primary coolant circuit and through which deionised water may enter the circuit. During refuelling, the “closed” position of these valves has to be checked.

Automatic and administrative measures and the monitoring measures cannot reliably preclude the injection of deionised water or the generation of low-borated condensate. Low-borated coolant may enter the primary circuit as a result of the following incidents:

- loss of residual heat removal during mid-loop operation
- leakages from neighbouring systems (steam generator, aftercooler)
- wrong boron concentration in neighbouring systems (RHR system)
- injection of deionised water from connected systems despite interlocks and administrative measures.

It has to be ensured by monitoring measures that the entry and accumulation of low-borated coolant is limited to such a degree that the inherent mixture properties in the reactor vessel increase the boron concentration of the low-borated coolant during start of coolant convection to such a degree that recriticality is prevented.

7.5.2 Technical basis

The technical basis will be given by the results of the corresponding experiments and thermal hydraulic analysis.

These are, among other things, experimental investigations performed at the Bora-Bora facility of EdF with a transparent model of a Framatome PWR with three reactor coolant loops at a scale of 1:5, the Vattenfall test facility in Sweden with a 1:5 model for a 3-loop Westinghouse PWR plant and at the German ROCOM test facility modelling a KONVOI plant. Further experiments serving as technical basis are investigations on internal deboration and mixing problems of demineralised water plugs at the German Upper Plenum Test Facility (UPTF) and PKL tests as well as the investigations at the test facility of the University of Maryland (UMCP). Further, thermal hydraulics analyses were and are performed with ATHLET and CFD codes.

7.5.3 Status of resolution

The federal authority forwarded to the licensees a comprehensive questionnaire concerning the above mentioned problem and in the meantime the answers of the licensees were evaluated. The licensees have presented to the technical experts results of experiments (PKL and ROCOM experiments) and technical reports concerning the transfer of the experimental data to the reactor conditions. These results have been supportive of clarifying the problem of the mixing phenomena in the downcomer and the lower plenum. Assessments have shown, that the sub criticality in the current core configurations is guaranteed.

7.5.4 Research needed to support modifications

Experiments for supporting the further development of the analytical methods to determine the amount of produced and accumulated condensate, the transport of the condensate to the reactor core and the mixing of the condensate with the high borated coolant in the reflux-condenser-mode will be continued with the aim to minimise the current uncertainties and to guarantee with high safety margins the sub criticality of the core. The influence of new fuel elements has to be considered.

7.6 JAPAN

7.6.1 Current Regulatory Approach

A LOCA is assumed as a typical accident to ensure the core coolability, integrity of containment, , etc. They are adequately maintained by the design. However, there are no regulations that specifically prohibit recriticality of a reactor following an SBLOCA and the boron dilution issue due to SBLOCA had not been concerned as a safety point of view. One of the reasons of this is low occurrence probability of LOCA with subsequent scenario of reactivity increase. The regulatory requirements that could bear upon the potential effects of a reactivity excursion are described in the following documents:

- Safety design examination guide
- Safety evaluation examination guide
- ECCS evaluation guide
- Reactivity initiated accident evaluation guide

7.6.2 Technical Basis Supporting the Regulatory Approach

Although there had been no explicit concern about the boron dilution after an SBLOCA, related recriticality scenarios due to boron dilution were investigated in1997~1998. The main scenario is that in the startup when boron is diluted through charging non borated water, the process continues even if RCPs stop because of loss of external power supply and finally reactivity increases. A research was conducted to examine a recriticality potential with the help of natural convection analysis using system hydraulic codes and a CFD code for three- dimensional boron mixing phenomena in the cold leg pipes and in the lower plenum of the reactor vessel. The results showed that recriticarity potentials seemed to be very low.

7.6.3 Status of Resolution

The resolution of this issue requires to demonstrate that the probability of occurrence of this type of event is very low, along with three-dimensional reactor kinetics analyses to show that even if such an event occurred, the resulting enthalpy deposition in the fuel would not pose a safety concern. The JNES participated in the OECD/NEA ISP-43 in which typical coolant mixing experiments have been analyzed with three- dimensional CFD codes. In addition to this ISP the JNES is participating in the PKL/SETH and PKL projects of OECD program which simulate directly the scenario of this issue since 2000.

7.6.4 Research Needed to Support Modifications

The JNES will continue to perform the code validation with the information obtained by the PKL project.

7.6.5 Schedule for Completion of Research and Regulatory Action

The JNES expects to complete the work mentioned above and to close the issue by approximately the end of fiscal year 2007.

7.7 KOREA

7.7.1 Current regulatory approach

The regulatory rule in Korea requires that a coolable geometry and sufficient long term cooling should be maintained following LOCA. A safe and stable condition must be reached with a sufficient level of subcriticality.

7.7.2 Technical basis supporting the regulatory approach

There is no specific technical basis that supports the regulatory approach. The regulatory strategy is to reflect the operating experience and safety issues in foreign countries in order to enhance the plant operational safety.

7.7.3 Status of resolution

On the issue the licensee proposed a conservative methodology using RELAP5 code for predicting the amount of unborated water at suction pipe and CFD code for predicting the unborated water slug transport to core. The method was reviewed for the OL for the new KSNP.

Based on the conservative assumption and boundary condition, it was accepted that the unborated water slug was completely mixed before entering the core by restarting the pumps and the additional reactivity insertion could be neglected. However, further investigation related to the slug mixing and transport is being requested.

For the relatively old plants, the issue have not been considered yet. During the periodic safety review process, KINS requested that the Licensee provide the information how to resolve the issue. After reviewing the requested information, the resolution of this issue can be more clearly defined.

7.7.4 Research needed to support modifications

No specific need to research is not identified.

7.7.5 Schedule for completion of research and regulatory action

No schedule has been set.

7.8 SWEDEN

7.8.1 Current regulatory approach

The traditional analyses of boron dilution accidents in pressurized water reactors are based upon the assumption that the boron field remains homogenous in the reactor coolant system during the dilution transient. However, a number of physical mechanisms may cause heterogeneities in the boron field with significant volumes of water with low boron concentration. The safety concern is that a large amount of water with low boron concentration may quickly reach the core and cause a power excursion.

The potential for local boron dilution became evident during commissioning one of the Ringhals PWRs in April 1982 in Sweden. It was determined that the amount of clean water of the volume of the loop seal in a stagnant loop could cause a significant power excursion if it were allowed to quickly reach the reactor core. The current regulatory approach is to prevent start-up of the pump in a loop that may contain unborated water and instructions have been implemented to avoid such a situation. The reactor operators have been informed about the safety concern and were also trained to identify situations that could lead to accumulation of boron free water

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There are two broad categories of events that potentially could lead to local depletion of boron. One is caused by inadvertent addition of clean water to a stagnant loop. Such events may be caused by malfunction of the reactor make up system resulting in an uncontrolled dilution, addition clean water for instance from the component cooling system or for instance during recovery after steam generator tube rupture using backfill procedure.

The other broad category may occur during heat removal during recovery after an accident. Under such circumstances heat may be removed by steam production in the core and condensation in the steam generators. Since the steam would contain a very low concentration of boron, the condensate may constitute water volumes which are nearly depleted from boron. The conclusions have been that this kind of scenarios is considered less probable and have so far not resulted in regulatory actions. However, this issue is being studied.

7.8.2 Technical basis supporting the regulatory approach

Boron dilution initiated reactivity accidents are a potential safety risk. However, there exists a large uncertainty with respect to the magnitude of the risk. To assess the risk associated with boron dilution events, the sources of diluted coolant have to be investigated, and how the plant behaves during various postulated scenarios. The technical basis for the regulatory approach is basically experimental and theoretical work.

In Sweden, local boron dilution event caused by pump-restart of a diluted loop has been considered to be of most safety significance. Experiments on transport of the deleted plug because of pump start-up have been carried out in Sweden and abroad. The length scale in these experiments is typically1:5 and the experiments are performed at atmospheric pressure. The main focus has been on mixing phenomena associated with the transport process.

The theoretical efforts encompass assessment of advanced computational methodology by comparison with experimental data and application of such methodology for analyses of hypothetical local boron dilution events in our reactors. The results of the experimental and theoretical efforts show that there still are uncertainties associated with the local born dilution process that can not be neglected. The conclusion is that a local boron dilution transient because of pump start-up should be avoided and that the administrative procedures and operator training to achieve this goal should be maintained.

Major facilities abroad have been used to obtain data on boron-free condensate during recovery after accidents or transients. In particular phenomena associated with the onset of natural circulation have been studied as a potential transport mechanism for a diluted plug to the core. Although several processes have been shown to be able to produce large volumes of condensate, the experiments and analyses have not been able to identify specific regulatory actions needed and such actions have not been considered as justified.

7.8.3 Status of resolution

The assessment of risks associated with local boron dilution has still significant uncertainty. The most significant safety importance is probably associated with transport of the diluted slug towards the core which is initiated by operator actions such as first restarting the pump in the diluted loop.

7.8.4 Research needed to support modifications

Many studies have been carried out on nuclear power plants in order to define potential scenarios and to determine the safety importance. These studies showed the need for further research programs in order to improve the knowledge and prediction capabilities of phenomena which occur during such events. The key phenomenon is the behaviour of the slug while it is transported by the flow to the core. Diffusion of boron and hydrodynamic mixing will be the phenomena which will modify the boron concentration distribution inside the slug. The experimental work on local boron dilution in Sweden has been terminated. There are still ongoing efforts to assess the calculation methodology by comparisons with experiments.

7.8.5 Schedule for completion of research and regulatory action

A schedule for completion of research on local boron dilution can not be definitely given. Such research would have to continue as long as the importance of the safety issue is not resolved.

7.9 UNITED STATES

7.9.1 Current USNRC regulatory approach

There are no regulations that specifically prohibit recriticality of a reactor following an SBLOCA. However, there are regulatory requirements that could bear upon the potential effects of a reactivity excursion under such circumstances. These include:

- 10 CFR 50.46: Requires that a coolable geometry and sufficient long-term cooling be maintained.
- 10 CFR 50.54: Requires that in the event of an accident, a licensee must bring the reactor to a safe and stable condition, including shutdown of the reactor and maintenance of subcriticality.
- 10 CFR Part 50, Appendix A, General Design Criteria 27, 28, 34, 35 require that reactivity control systems can protect against postulated reactivity accidents and that adequate residual heat removal and emergency core cooling be provided to protect against fuel damage that could affect core coolability.

However, the indirect nature of these requirements with respect to the specific scenarios identified for recriticality following an SBLOCA led the NRC to evaluate the potential for these event to occur and the possible consequences.

7.9.2 Technical basis supporting the regulatory approach

There are no specific technical bases supporting the regulatory approach beyond those used to develop the regulatory requirements noted above. The guiding philosophy underlying those requirements is that maintenance of reactor in safe and stable condition after an accident, with assurance of adequate heat removal and minimization of core damage, would likely provide adequate protection of public health and safety. Research has been conducted over the past several years to characterize the scenario for the three U.S. PWR designs (B&W, CE, Westinghouse) and to determine what, if any, additional regulatory guidance is required to ensure that the risk of recriticality following an SBLOCA is acceptably low.

7.9.3 Status of resolution

The resolution of this issue involves demonstrating that the probability of occurrence of this type of event is very low, along with three-dimensional reactor kinetics analyses to show that even if such an event occurred, the resulting enthalpy deposition in the fuel would not pose a safety concern. The NRC has supported scaled testing at the University of Maryland and has performed extensive analyses on both scenarios (resumption of natural circulation, reactor coolant pump restart) in the various reactor designs. Testing has been completed, and the analytical work is scheduled to be completed in the near future.

7.9.4 Research needed to support modifications

The NRC has not identified any further research that is necessary to support the resolution of this issue.

7.9.5 Schedule for completion of research and regulatory action

The NRC completed work on this issue (GSI-185) in September 2005.