IAEA-TECDOC-1044

XA9848109

Generic safety issues for nuclear power plants with light water reactors and measures taken for their resolution

INTERNATIONAL ATOMIC ENERGY AGENCY

IAEA

The originating Section of this publication in the IAEA was:

Safety Assessment Section
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100
A-1400 Vienna, Austria

GENERIC SAFETY ISSUES FOR NUCLEAR POWER PLANTS
WITH LIGHT WATER REACTORS
AND MEASURES TAKEN FOR THEIR RESOLUTION
IAEA, VIENNA, 1998
IAEA-TECDOC-1044
ISSN 1011-4289

Printed by the IAEA in Austria
September 1998

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International Atomic Energy Agency
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FOREWORD

The IAEA Conference on "The Safety of Nuclear Power: Strategy for the Future" in 1991 was a milestone in nuclear safety. The objective of this conference was to review nuclear power safety issues for which achieving international consensus would be desirable, to address concerns on nuclear safety and to formulate recommendations for future actions by national and international authorities to advance nuclear safety to the highest level. Two of the important items addressed by this conference were ensuring and enhancing safety of operating plants and treatment of nuclear power plants built to earlier safety standards. Some of the publications related to these two items that have been issued subsequent to this conference are: A Common Basis for Judging the Safety of Nuclear Power Plants Built to Earlier Standards, INSAG-8 (1995), the IAEA Safety Guide 50-SG-O12, Periodic Safety Review of Operational Nuclear Power Plants (1994) and an IAEA publication on the Safety Evaluation of Operating Nuclear Power Plants Built to Earlier Standards - A Common Basis for Judgement (1997).

Some of the findings of the 1991 conference have not yet been fully addressed. An IAEA Symposium on Reviewing the Safety of Existing Nuclear Power Plants in 1996 showed that there is an urgent need for operating organizations and national authorities to review operating nuclear power plants which do not meet the high safety levels of the vast majority of plants and to undertake improvements with assistance from the international community if required. Safety reviews of operating nuclear power plants take on added importance in the context of the Convention on Nuclear Safety and its implementation. To perform safety reviews and to reassess the safety of operating nuclear power plants in a uniform manner, it is imperative to have an internationally accepted reference. Existing guidance needs to be complemented by a list of safety issues which have been encountered and resolved in other plants and which can be used in reassessing the safety of individual operating plants.

The compilation is based on broad international experience. This compilation is one element in the framework of IAEA activities to assist Member States in reassessing the safety of operating nuclear power plants. It is a compilation not only of the generic safety issues identified in nuclear power plants but also, in almost all cases, the measures taken to resolve these issues. The safety issues, which are generic in nature with regard to light water reactors (LWRs) and the measures for their resolution, are for use as a reference for the safety reassessment of operating plants.

The IAEA staff member responsible for this publication was G. Philip of the Division of Nuclear Installation Safety.

EDITORIAL NOTE

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1. INTRODUCTION

1.1. BACKGROUND

The IAEA Conference on The Safety of Nuclear Power: Strategy for the Future, held in September 1991, addressed the safety of operating nuclear power plants built to earlier standards. The conference, which was directed to decision makers on nuclear safety and energy policy, expressed the desire to achieve international consensus on this issue. Subsequently, the IAEA General Conference endorsed the recommendations in a resolution that urged the IAEA to develop a common basis on which an acceptable level of safety for all operating plants built to earlier standards could be judged.

Since that time, the relevant IAEA programme has focused on the development of guidance to assist Member States in the safety reassessment of operating plants. In addition, INSAG has developed a report on A Common Basis for Judging the Safety of Nuclear Power Plants Built to Earlier Standards, INSAG-8 (1995), which is supplemented by practical guidance issued by the IAEA.

The IAEA Safety Guide 50-SG-O12, Periodic Safety Review of Operational Nuclear Power Plants (1994), provides guidance on the conduct of periodic safety reviews (PSR) of operational nuclear power plants. This Safety Guide describes the review strategy, the safety factors to be reviewed, the roles and responsibilities of the parties involved and the review procedure. The Safety Guide aims at providing for the comprehensive reassessment of an operational plant, including whether the plant is safe as judged by current standards and practices and whether appropriate means are in place to maintain plant safety.

An IAEA publication on Safety Evaluation of Operating Nuclear Power Plants Built to Earlier Standards - A Common Basis for Judgement was issued in 1997 to provide details for the safety assessment and judgement process. This provides practical advice on the main judgements to be made in any review process of plant safety.

These publications are based on the knowledge and experience gathered in Member States to reassess safety of operating plants, such as the Safety Evaluation Programme (SEP) in the USA and other national programmes in Belgium, France, Spain, Sweden and the United Kingdom. The IAEA Extrabudgetary Programme on the Safety of WWER and RBMK NPPs has also been contributing to the safety reassessment of plants built to earlier standards and the associated safety improvement programmes in the central and eastern European countries and the countries of the former Soviet Union operating or constructing plants of these types.

A complementary approach to disseminating the knowledge and experience of Member States is to provide the lessons learned directly. The occurrence and recurrence of events, deviations from current international practice in design and operation, and results of PSA studies allow valuable insight into weaknesses in plant safety and into corrective measures to resolve them.

There is a generally accepted approach to characterizing the safety concerns in nuclear power plants, which need to be resolved, as 'safety issues'. When such safety issues are applicable to a generation of plants of a particular design or to a family of plants of similar design, they are termed 'generic safety issues'. Examples of generic safety issues are those related to reactor vessel embrittlement, reliability of insertion of control rods or strainer clogging.

This approach of using generic safety issues for identifying and resolving safety concerns has been practiced in the USA since the 1970s (NUREG-0410 and NUREG-0933). Other Member States such as France, Germany, Japan, Spain and Sweden also have experience in using similar approaches. More recently, the IAEA has developed lists of generic safety issues for the WWER and RBMK NPPs. These generic safety issues are used as a reference to facilitate the development of plant specific safety improvement programmes and to serve as a basis for reviewing their implementation.

1.2. OBJECTIVE

The purpose of this TECDOC is to assist Member States in the reassessment of operating plants by providing a list of generic safety issues identified in nuclear power plants together with measures taken to resolve these issues. These safety issues, which are generic in nature with regard to light water reactors (LWRs), and the measures for their resolution are for use as a reference for the safety reassessment of operating plants. Not all the issues presented here are applicable to all Member States or to all reactors within individual Member States. Operators, scientific/technical support organizations and regulators are encouraged to check the applicability of these generic safety issues for operating plants and their safety re-evaluation on a case by case basis, and to take benefit from the experience at other LWRs in order to ensure that all reasonable improvements are made to enhance plant safety.

The use of these generic safety issues for each specific plant under review cannot ensure that all the possible safety concerns would be identified, since these would also be influenced by plant specific situations. This reference list of generic safety issues for LWRs is intended only to assist operators and authorities who have sole responsibility for the safe operation and regulation of their NPPs.

The TECDOC covers issues thought to be of significance to Member States, based on a consensus process. Those issues which have been generally resolved by Member States or concluded to be of low safety significance were not included. Thus, some countries may have documents (such as the NUREG-0933) which contain generic issues not described in this TECDOC.

This TECDOC is a record of the successful recognition of the safety lessons embedded in plant operation, analysis and regulation.

1.3. SCOPE

Since safety issues are deviations from current safety practices in design and operation which are judged to be of safety significance, this TECDOC deals with those generic issues which are applicable to all LWRs, or to each particular type of BWR, PWR and WWER. Although WWERs are mentioned specifically in the text, they have been treated, for the most part, as a subclass of PWRs, for which most generic issues relevant to PWRs may have relevance.

The generic safety issues for LWRs compiled in Section 4 of this report reflect the broad experience of Member States, especially those with major nuclear programmes, in resolving safety concerns and in maintaining improvements to current practice. As can be seen from the actions taken in Member States with respect to the appropriate technical solutions, most issues have been successfully addressed.

The Section 4 compilation also includes those safety issues which are currently considered pending, i.e. where the root causes and the measures to be taken are being investigated. For pending issues, interim judgements have been made by Member States with respect to continued safe plant operation. With respect to such pending issues, this report reflects the status of knowledge and experience so far in dealing with them.

In the context of both these categories of issues, the measures taken for a particular issue are not a complete record of actions taken by all Member States but are intended as representative responses by individual Member States which should be of use to other Member States which are still in the process of evaluating the applicability and significance of the issues for their plants.

This TECDOC reflects material received to date from Member States. However, additional material from Member States on new issues or on measures taken in response to the issues described will be incorporated, as appropriate, in future editions. It should also be emphasized that the absence of a response from a Member State does not indicate that no action on the issue has been taken.

1.4. STRUCTURE

Section 2 provides an introduction to the use of the generic safety issues for LWRs compiled in this TECDOC. In particular, there are more details on the structure of the issues as presented and the associated rationale.

Section 3 presents generic observations on safety aspects identified from the issues in Section 4, grouped according to the source of the issues.

Section 4 is the main body of the report and contains about 140 generic safety issues related to design and operation, grouped according to areas.

2. USE OF THE REPORT

2.1. INTENDED USE

Several types of safety review such as routine reviews, special reviews following accidents and PSR are used in Member States to ensure safe operation of existing nuclear power plants including those built to earlier standards. These plant reviews are a key element in the safety reassessment to identify weaknesses in operating plants and to determine the corrective actions for safety improvements.

For this purpose a reference list of generic safety issues which is based on operational experience or events, deviations from current standards and practices and potential weakness identified by analysis is presented along with corrective measures proposed or implemented in different countries.

2.2. STRUCTURE OF GENERIC SAFETY ISSUES

The generic safety issues in this report are provided according to the following structure, with explanations given in parenthesis.

ISSUE TITLE

(A short title indicates the safety concern, type of NPP, and - if applicable - the restriction to anyone of the types, e.g. PWRs).

ISSUE CLARIFICATION

Description of issue
(reflects an appropriately sized issue and addresses:

- safety concern and root cause;
- systems, components and human performance involved, an explicit statement is given if the issue is not applicable to any type of LWRs).

Safety significance
(characterized by the:

- impact of the issue on defence in depth including the performance of main safety functions;
- validity of analyses carried out in the past; and
- operational conditions, transients or accident scenarios affected).

Source of issue (check as appropriate)

- _________ operational experience
- _________ deviation from current standards and practices
- _________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES

(representative examples of corrective measures taken by individual Member States are provided according to the following structure with relevant references. These are reproduced as received from Member States.):

Country

(– applicability of the generic safety issue to a specific plant type or plant;
– corrective measures applied as immediate compensatory or interim short term actions;

– permanent corrective measures in hardware/process changes/operating practices related to the root cause and their effectiveness, if available;
– information on status: proposed, implemented).

ADDITIONAL SOURCES
(Additional sources of information reproduced as received from Member States)

2.3. LIST OF GENERIC SAFETY ISSUES FOR LWR NPPs

(Countries from which contributions have been received with respect to measures taken for an issue are shown in parentheses)

DESIGN

GENERAL

GL 1 Classification of components
(Bulgaria, France, Germany, India, Japan, Republic of Korea, Russian Federation, USA)

GL 2 Qualification of equipment and structures including ageing effects
(Bulgaria, France, Germany, India, Japan, Republic of Korea, Russian Federation, USA)

GL 3 Inadequacy of reliability data
(Argentina, France, Germany, India, Japan, Republic of Korea, Russian Federation, Sweden, USA)

GL 4 Need for performance of plant specific probabilistic safety assessment (PSA)
(Argentina, France, India, Japan, Republic of Korea, Russian Federation, Spain, Sweden, USA)

REACTOR CORE

RC 1 Inadvertent boron dilutions under low power and shutdown conditions
(France, Germany, Japan, Republic of Korea, Spain, WWER countries)

RC 2 Unreliable insertion of control rods in PWRs and WWERs
(France, Germany, Japan, Republic of Korea, Spain, USA, WWER-1000 countries)

RC 3 Power oscillations in BWRs
(Germany, India, Japan, Spain, USA)

RC 4 Loss of thermal margin caused by channel box bow (BWR)
(Germany, India, Japan, USA)

RC 5 Accident response of high burnup fuel
(France, Japan, Republic of Korea, USA)

RC 6 Fuel cladding and control rod corrosion and fretting (PWR)
(Germany, Japan, Republic of Korea)

COMPONENT INTEGRITY

CI 1 Reactor pressure vessel integrity
(France, Germany, India, Japan, Republic of Korea, Spain, WWER countries, USA)

CI 2 Asymmetric blowdown loads on RPV supports and internals
(Germany, Japan, Republic of Korea, USA)

CI 3 BWR core internals cracking
(Germany, India, Japan, Spain, USA)

CI 4 Thimble tube thinning
(France, Germany, Japan, Republic of Korea, Spain, USA)

CI 5 Inconel-600 cracking
(France, Germany, Japan, Republic of Korea, Russian Federation, Spain, USA)

CI 6 Steam generator collector integrity (WWER)
(Bulgaria, Czech Republic, Russian Federation, Ukraine)

CI 7 SG tubes integrity
(France, Germany, India, Japan, Republic of Korea, Russian Federation, Spain, USA, WWER countries)

CI 8 Pipe cracks and feedwater nozzle cracking in BWRs
(Germany, India, Spain, USA)

CI 9 Bolting degradation or bolting failures in the primary circuit
(France, Germany, India, Republic of Korea, USA, WWER-440/213 countries)

CI 10 Heavy components support stability
(Germany, Japan, USA, WWER-440/230 countries)

CI 11 Cast stainless steel cracking
(France, Germany, India, Japan, Republic of Korea, Sweden, USA)

CI 12 Loads not specified in the original design
(France, Germany, Japan, Republic of Korea, USA, WWER-440/230 countries)

CI 13 Boron corrosion on reactor coolant pressure boundary
(Germany, Japan, Ukraine, USA)

CI 14 Steam and feedwater piping degradation
(France, Germany, India, Japan, Republic of Korea, Spain, USA)

CI 15 Steam generator internals damage and plate cracking
(Germany, India, Republic of Korea, USA, WWER-440/213 countries)

PRIMARY CIRCUIT AND ASSOCIATED SYSTEMS

PC 1 Overpressure protection of the primary circuit and connected systems
(Bulgaria, Czech Republic, France, Germany, Republic of Korea, Russian Federation, Ukraine, USA)

PC 2 Adequacy of the isolation of low pressure systems connected to the reactor coolant pressure boundary
(Germany, India, Republic of Korea, Spain)

PC 3 Reactor coolant pump seal failures
(Bulgaria, Czech Republic, France, Germany, India, Republic of Korea, Russian Federation, Ukraine, USA)

PC 4 Safety, relief and block valve reliability - primary system
(Germany, Japan, Russian Federation, Sweden, USA)

PC 5 Safety, relief and block valve reliability - secondary system
(Bulgaria, France, Germany, Russian Federation, Ukraine, USA)

PC 6 Spring-actuated safety and relief valve reliability
(India, Japan, USA)

PC 7 Water hammer in the feedwater line
(India, Japan, Republic of Korea, Spain)

PC 8 Steam generator overfill due to control system failure and combined primary and secondary blowdown
(Republic of Korea, USA)

SAFETY SYSTEMS

SS 1 ECCS sump screen adequacy
(Bulgaria, Czech Republic, France, Germany, India, Russian Federation, Ukraine, USA)

SS 2 ECCS water storage tank and suction line integrity (WWER)
(Bulgaria, Czech Republic, Russian Federation, Ukraine)

SS 3 ECCS heat exchanger integrity (WWER)
(Bulgaria, Czech Republic, Ukraine)

SS 4 Problems on the ECCS and containment spray switchover to recirculation
(France, Japan, Republic of Korea, USA)

SS 5 Diversion of recirculation water (holdups in containment)
(France, Republic of Korea)

SS 6 Boron crystallization in systems
(Japan, USA)

SS 7 Boron crystallization and dilution in the core in case of LOCAs
(France, Japan, USA)

SS 8 Accident management measures
(France, Germany, Japan, Slovakia)

SS 9 Containment or confinement leakage from engineered safety features (ESF) systems during an accident
(France, Japan)

SS 10 Steam generator safety valves performance at low pressure (WWER)
(Bulgaria, Ukraine)

SS 11 Thermal shock or fatigue caused by cold emergency feedwater supply to steam generators
(Bulgaria, Czech Republic, Germany, Japan, Ukraine, USA)

SS 12 Emergency feedwater system reliability
(Germany, USA)

SS 13 Need for hydrogen control measures during design basis accidents (DBA)
(Czech Republic, France, Germany, Japan, Republic of Korea, Russian Federation, Ukraine)

SS 14 Overfill into the main steam lines in BWRs
(Germany, Sweden, USA)

SS 15 Containment isolation of lines containing high activity fluids
(France)

SS 16 Reliability of the motor operated valves in safety systems
(France, Germany, Japan, Republic of Korea, USA)

SS 17 Reliability and mechanical failure of safety related check valves
(USA)

SS 18 Potential failure of the scram system due to loss of discharge volume (BWR)
(USA)

SS 19 Need for assurance of ultimate heat sink
(Bulgaria, Czech Republic, Russian Federation, Ukraine, USA)

ELECTRICAL AND OTHER SUPPORT SYSTEMS

ES 1 Reliability of off-site power supply
(Bulgaria, Czech Republic, France, Germany, India, Republic of Korea, Spain, Ukraine, USA)

ES 2 Diesel generator reliability
(Bulgaria, Czech Republic, Germany, Sweden, Ukraine, USA)

ES 3 Scope of systems supplied by emergency on-site power
(Bulgaria, Czech Republic, Germany, Japan, Ukraine, USA)

ES 4 Breaker co-ordination to protect loads
(USA)

ES 5 Vulnerability of swingbus configurations
(Japan, Sweden, USA)

ES 6 Reliability of emergency DC supplies
(Bulgaria, Czech Republic, France, Germany, India, Japan, Ukraine, USA)

ES 7 Control room habitability
(Bulgaria, Czech Republic, India, Republic of Korea, Russian Federation, Spain, Ukraine, USA)

ES 8 Reliability of instrument air systems
(Japan, USA)

ES 9 Solenoid valve reliability
(Germany, Japan, USA)

INSTRUMENTATION AND CONTROL (incl. Protection Systems)

IC 1 Physical separation of instrument sensing lines for the reactor protection system
(Russian Federation, Ukraine)

IC 2 Inadequate electrical isolation of safety from non-safety related equipment
(Republic of Korea, USA)

IC 3 Interference in I&C signals
(Germany, Sweden, USA)

IC 4 I&C component reliability
(Argentina, Bulgaria, Russian Federation, Ukraine)

IC 5 Lack of on-line testability of protection systems
(France, India, Slovakia)

IC 6 Reliability and safety basis for digital I&C conversions
(USA)

IC 7 Reliable ventilation of control room cabinets
(India, Russian Federation, Spain)

IC 8 Human engineering of control rooms
(Bulgaria, Czech Republic, Russian Federation, Spain, Ukraine, USA)

IC 9 Need for a safety parameter display system
(Bulgaria, Czech Republic, France, Russian Federation, Ukraine, USA)

IC 10 Inadequacy of diagnostic systems
(Bulgaria, Czech Republic, Russian Federation, Ukraine)

IC 11 Reactor vessel head leak monitoring system (WWER)
(Bulgaria)

IC 12 Availability and adequacy of accident monitoring instrumentation
(Bulgaria, Czech Republic, France, Republic of Korea, Russian Federation, Ukraine, USA)

IC 13 Water chemistry control and monitoring equipment (primary and secondary)
(Bulgaria, Russian Federation, Ukraine)

IC 14 Adequacy of reactor vessel level instrumentation in BWRs
(Germany, Japan, Sweden, USA)

IC 15 Improving the detection of primary/secondary leaks
(Russian Federation, Spain)

IC 16 Establishment and surveillance of setpoints in instrumentation
(USA)

CONTAINMENT AND OTHER STRUCTURES

CS 1 Assessment of WWER-440/213 containment dynamic loads
(Czech Republic, Hungary, Russian Federation, Slovakia, Ukraine)

CS 2 Assessment of BWR containment dynamic loads
(Germany, Japan, Sweden, USA)

CS 3 Need for containment and confinement integrity during severe accidents
(India, Japan, Republic of Korea, Russian Federation, USA)

INTERNAL HAZARDS

IH 1 Need for systematic fire hazards assessment
(Bulgaria, Czech Republic, France, Germany, India, Japan, Republic of Korea, Ukraine)

IH 2 Adequacy of fire prevention and fire barriers
(Bulgaria, Czech Republic, France, Germany, India, Republic of Korea, Russian Federation, Ukraine)

IH 3 Adequacy of fire detection and extinguishing
(Bulgaria, Czech Republic, France, Germany, India, Japan, Republic of Korea, Russian Federation, Ukraine)

IH 4 Adequacy of the mitigation of the secondary effects of fire and fire protection systems on plant safety
(Bulgaria, Czech Republic, France, Germany, Republic of Korea, Ukraine)

IH 5 Need for systematic internal flooding assessment including backflow through floor drains
(Bulgaria, Czech Republic, France, Germany, India, Republic of Korea, Ukraine, USA)

IH 6 Need for systematic assessment of high energy line break effects
(Bulgaria, Czech Republic, France, Republic of Korea, Russian Federation, Ukraine)

IH 7 Need for assessment of dropping heavy loads
(Bulgaria, Czech Republic, France, India, Republic of Korea, Russian Federation, Ukraine, USA)

IH 8 Refueling cavity seal failure
(India, Republic of Korea, USA)

IH 9 Need for assessment of turbine missile hazard
(France, Republic of Korea, Sweden)

EXTERNAL HAZARDS

EH 1 Need for systematic assessment of seismic effects
(Bulgaria, Czech Republic, France, India, Japan, Republic of Korea, Russian Federation, Ukraine, USA)

EH 2 Need for assessment of seismic interaction of structures or equipment on safety functions
(France, India, Japan, Republic of Korea, USA)

EH 3 Need for assessment of plant specific natural external conditions
(Bulgaria, Czech Republic, India, Japan, Republic of Korea, Sweden, Ukraine, USA)

EH 4 Need for assessment of plant specific man induced external events
(Bulgaria, Czech Republic, France, Japan, Russian Federation, Sweden, Ukraine, USA)

ACCIDENT ANALYSIS

AA 1 Adequacy of scope and methodology of design basis accident analysis
(Bulgaria, Czech Republic, France, Japan, Republic of Korea, Russian Federation, Ukraine)

AA 2 Adequacy of plant data used in accident analysis
(Bulgaria, Czech Republic, France, Japan, Republic of Korea, Ukraine)

AA 3 Computer code and plant model validation
(Bulgaria, Czech Republic, France, Japan, Russian Federation, Ukraine, USA)

AA 4 Need for analysis of accidents under low power and shutdown conditions
(Bulgaria, Czech Republic, France, Republic of Korea, Ukraine)

AA 5 Need for severe accident analysis
(Bulgaria, Czech Republic, France, Germany, India, Japan, Republic of Korea, Russian Federation, Ukraine)

AA 6 Need for analysis of ATWS
(Bulgaria, Czech Republic, France, Republic of Korea, Ukraine)

AA 7 Need for analysis of total loss of AC power
(Bulgaria, Czech Republic, France, Germany, India, Japan, Russian Federation, Spain, Ukraine, USA)

OPERATION

MANAGEMENT

MA 1 Replacement part design, procurement and assurance of quality
(USA)

MA 2 Fitness for duty
(Spain, USA)

MA 3 Adequacy of shift staffing
(USA)

MA 4 Control of outage activities to minimize risk
(France, Germany, Japan)

MA 5 Degraded and non-conforming conditions and operability determinations
(Japan, USA)

MA 6 Configuration management of modifications and temporary modifications
(Japan, USA)

MA 7 Human and organizational factors in root cause analysis
(Japan, Spain, USA)

MA 8 Impact of human factors on the safe operation of nuclear power plants
(France, USA)

MA 9 Effectiveness of quality programmes
(USA)

MA 10 Adequacy of procedures and their use
(USA)

MA 11 Adequacy of emergency operating procedures
(Spain, USA)

MA 12 Effectiveness of maintenance programmes
(France, Japan, Spain, USA)

OPERATIONS

OP 1 Intentional bypassing of automatic actuation of plant protective features
(India, Japan, USA)

OP 2 Response to loss of control room annunciators
(Japan, USA)

OP 3 Inadvertent introduction of chemicals and their effects on safety related systems
(India, Spain, USA)

OP 4 Precautions for mid-loop operation (PWR)
(France, Germany, Japan, Spain)

SURVEILLANCE AND MAINTENANCE

SM 1 Adequacy of non-destructive inspections and testing (WWER)
(Bulgaria, Hungary, Russian Federation, Slovakia, Ukraine)

SM 2 Removal of components from service during power or shutdown operations for maintenance
(USA)

SM 3 Use of freeze seals
(India, USA)

SM 4 Use of pressure injection of compounds to seal leaks
(India, Japan, USA)

SM 5 Inadequate testing of engineered safety features (ESF) actuation systems (lack of logic overlap)
(Japan, USA)

SM 6 Foreign material policy
(USA)

SM 7 Control of temporary installations
(France, USA)

SM 8 Clear identification of components and system trains
(Japan, USA)

SM 9 Response to low level equipment defects (plant material condition)
(India)

TRAINING

TR 1 Adequacy of fire brigade training
(France, Germany, Japan)

TR 2 Assessment of full scope simulator use
(France, Spain, USA)

TR 3 Training for severe (beyond design basis) accident management procedures
(France)

EMERGENCY PREPAREDNESS (including Physical Protection)

EP 1 Need for effective off-site communications during events
(France, Japan, Spain, USA)

EP 2 Contingency planning for physical security
(USA)

EP 3 Need for technical support centre
(Bulgaria, Czech Republic, India, Russian Federation, Ukraine, USA)

RADIATION PROTECTION (including Waste Management)

RP 1 Hot particle exposures
(Japan, USA)

RP 2 Radiation beams from power reactor biological shields
(France, Japan, USA)

RP 3 Measures implemented to comply with international recommendations (ICRP-60) on dose limits
(France, USA)

FUEL STORAGE

FS 1 Degradation of boron plates in fuel storage pool
(France, USA)

FS 2 Potential for fuel pool drainage
(India, Japan, USA)

FS 3 Damage to fuel during handling
(Germany, USA)

3. GENERIC OBSERVATIONS

The generic safety issues for LWRs compiled in Section 3 reflect the broad experience of Member States in resolving safety concerns and in maintaining improvements to current practice. As reflected in the "Measures taken" for these issues, most issues have been successfully addressed. The Section 3 compilation also includes those safety issues which are currently considered "pending", i.e. the root causes and their resolution are in the process of being developed. For pending issues, interim judgements have been made by Member States with respect to continued safe plant operation.

This comprehensive compilation has been found to be an appropriate basis to look for broader insights, lessons learned and trends which is the subject of this section.

The source or sources for each issue is provided in Section 3, that is, whether operational experience, a deviation from current standards and practices or analysis gave rise to the concern. Over half the issues had operational experience as one of their sources. This is to be expected after over 30 years of worldwide nuclear power generation. The source areas of deviation from current standards and practices and analysis each were noted in about one third of the issues. The use of analysis, particularly PSA, to identify potential issues of concern which have not been observed during plant operation is considered a sign of health with respect to industry and regulatory body concern for safety. Several of the "deviation" issues refer to WWER reactors which have recently been systematically measured against international practice.

The following discussion is organized by grouping generic observations on the issues by the primary source of the trend or common thread observed.

3.1. OBSERVATIONS BASED ON OPERATIONAL EXPERIENCE

The majority of safety issues for pressurized water reactors in the area of reactor core are related to the changing design and use of reactor fuel. Unreliable insertion of control rods (RC 2) was observed recently in several Member States in connection with the use of more innovative fuel designs and more aggressive operational strategies. Significant reductions in margins to fuel cladding corrosion limits are being observed (RC 6). Concerns about low enthalpy LWR irradiated fuel failures in tests at research facilities (RC 5) have prompted authorities to reconsider burnup extension applications and the applicability of current licensing criteria. These safety issues are considered to be connected with the observation that during the last decade the PWR operating conditions have been changed worldwide by more aggressive fuel management which is characterized by upratings, longer cycles, higher burnups. The effects of increased fuel duty, particularly higher values of burnup, peaking factors, and operating temperatures as well as longer residence times at more demanding coolant chemistry appear to have directly impacted fuel reliability margins. Efforts are going on in Member States to understand the complex mechanisms leading to any potential impacts on the ability to shut down the reactor and maintain it in a safe shutdown condition. The adverse operational experience should be regarded as a precursor to potentially more serious events. Additional attention appears warranted to thorough R&D and prototype testing prior to widespread use of new core designs or more demanding duty cycles.

Almost all of the issues in the management, operations, surveillance and maintenance, training, emergency preparedness and radiation protection areas are the result of operational experience. A number of issues (MA 6, MA 8, MA 9, MA 10, OP 1, SM 1, SM 6, SM 7, SM 8) appear related to the lack of application of disciplined quality programmes, including appropriate surveillances. The basic concepts and standards in these areas were put in place in most countries in the early 1970s. Other issues related to human factors (MA 2, MA 3, MA 7, TR 1, TR 2) and emergency situations (MA 10, TR 2, EP 1, EP 2) were thoroughly addressed in guidance documents of the early 1980s. That many of these issues are still of concern today is a reflection that these issues are dependent on the continued motivation (will) and availability of resources to address these issues on a continuing basis. Continued attention of plant owners and regulatory bodies is warranted in these human performance/safety culture areas, particularly where plants are under economic stress.

The third area where operational experience indicates a common thread is changing technology as it is reflected in I&C systems. This has resulted in the unavailability of identical replacement parts because of changes in industry product lines but has also, more importantly, posed new types of safety questions for plant owners and regulatory bodies. This is particularly true in the area of the introduction of digital technology. There are substantial safety advantages to the use of digital hardware/software in diagnostics, control and safety function actuation. However, the regulatory framework has lagged the introduction of new equipment.

It is unlikely that all problems can be considered in advance, as evidenced by the history of the original plant control and safety systems. A concerted effort to agree on safety standards for digital technology should be combined with close monitoring of plant experience and reflection of lessons learned into system design and regulatory practice.

3.2. OBSERVATIONS BASED ON DEVIATIONS FROM CURRENT STANDARDS AND PRACTICES

Section 3 brings out only a few issues which are linked to ageing of the plants. There is a worldwide increase in nuclear power generation with a significant percentage of overall electrical generation in some Member States. Many of the NPPs have been operating for quite some time and have started showing signs of ageing. As such, ageing management of these operating plants is an important item in their periodic safety review (IAEA Safety Guide 50-SG-O12, Periodic Safety Review of Operational Nuclear Power Plants (1994)) and in the extension of operation past the end of the original design lifetime. Ageing studies sponsored by the IAEA and Member States as well as periodic plant safety reviews may bring out generic safety issues of LWRs needing specific attention to assure safety. For example, the monitoring of some parameters, such as plant water chemistry, becomes important when considering plant history with respect to the feasibility of extended safe plant operation.

Several safety issues result from deviations from current safety standards and practices. While some of these relate to the need to upgrade WWERs, others apply in any situation where a reactor of older design has not regularly taken into account the operating experience of other plants and the resultant adjustment of international standards and practices. Attention is warranted to assure that conscious decisions are made by Member States with regard to the applicability and significance of new safety issues either as they arise or through periodic reassessments.

In this regard, the substantial participation in this effort and the improvement measures already taken by countries with WWERs or older reactors is noted as a positive sign that these countries are serious in their intent to achieve and maintain high standards of reactor safety.

3.3. OBSERVATIONS BASED ON ANALYSIS INSIGHTS

The completion of PSA analysis for a large number of plants has allowed a more comprehensive and systematic assessment of the safety of the plants. Systems and situations to which not much attention was paid before, are now considered relevant to safety (i.e. analysis of accidents in modes other than full power, anticipated transients without scram (ATWS), station blackout, spent fuel systems, specific configuration situations, maintenance rule, external hazards). Issues with an analysis source include GL 4, SS 8, CS 3, AA 4, AA 5, AA 6, AA 7, MA 11, EP 3, TR 3, FS 2.

Although plant specific PSAs will continue to identify vulnerabilities for particular plants, a large number of additional analysis issues with widespread applicability is not expected. Attention will need to continue, however, with respect to the adequacy of computer modeling of accident phenomena and the applicability of reliability data used in the PSA analyses.

There is also an opportunity, given current analytical capabilities and monitoring techniques, of significantly improving component reliability by the use of plant specific experience data to adjust maintenance and replacement programmes for active plant equipment. This is covered in GL 3, SS 12, SS 16, SS 17, ES 2, ES 6, ES 8, IC 4.

4. GENERIC SAFETY ISSUES FOR LIGHT WATER REACTOR NUCLEAR POWER PLANTS

4.1. DESIGN SAFETY ISSUES

4.1.1. General (GL)

ISSUE TITLE: Classification of components (GL 1)

ISSUE CLARIFICATION:

Description of issue

The safety related components in plants are required to be classified into seismic and quality groups in order to assure their safety functions. Once classified, the components are required to meet specific design, construction, testing, and inspection criteria.

The concern is applicable in particular to the old generation NPPs which have deficiencies with regard to completeness of classification of components as compared to the current standards.

Safety significance

Deviations from current requirements for manufacturing, testing, in-service inspection, and maintenance can lead to safety equipment not performing its intended safety functions.

Source of issue (check as appropriate)

- ________ operational experience
- ___xx___ deviation from current standards and practices
- ________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

The measures proposed in the modernization programme for Kozloduy Units 5 and 6 include:

- ranking of the NPP equipment according to its importance to safety in conformity with OPB-88 and other international standards; and
- analysis of the safety systems equipment behaviour in the event of an earthquake.

France

Safety classification is generally linked to 3 main technical safety objectives:

- ability to maintain integrity of the barriers (and principally of the primary coolant circuit boundary);
- ability to stop the reactor and maintain the core coolability;
- ability to prevent accident and in case of accident to mitigate radiological consequences.

These technical objectives can be declined in classification of safety functions, in classification of systems important to safety, or in classification of components.

There is not, up to now, an official classification of safety functions in France, there is however a proposal of classification for EPR[1] project.

F1 class is subdivided in:

- F1A: necessary functions to lead the plant towards a "controlled state."
- F1B: starting from a controlled state, necessary functions to lead the plant towards a "safe state", and ability to maintain that state at least 24 hours after an accident.

F2 class comprises the necessary functions to:

- Maintain a safe state beyond 24 hours and at least up to 72 hours after the initiator event (this time is expected sufficient to bring and put in operation on site components necessary to mitigate accidental consequences).
- (In the frame of supplementary conditions, beyond DBA conditions) satisfy to probabilistic objective for core melting.
- Mitigate or prevent risks of radioactive releases out of normal limits in case of internal aggression.

In EUR[2] project there is not a classification of systems, only functions and components are classified.

Classification of equipment is defined by the French rule: RFS no. IV.1.a.

Germany

The classification of components in German NPPs was carried out plant-specifically within the framework of the licensing procedure based on the BMI safety criteria for NPPs and the guidelines of the German Reactor Safety Commission.

The proposal for the classification for the new Franco-German EPR project is described in the French comment.

India

Indian BWRs being old generation reactors, their components are not fully classified by design.

Japan

The "Guide for Classification of Safety Function Importance in Light Water Nuclear Power Reactor Facilities" defines the relative importance of various functions necessary to ensure the safety. The basis had been established for adequate requirement to be met by design of structures, systems and component that are expected to perform such function.

Korea, Republic of

The structures, and components are classified according to their safety significance using the classification system in Reg. Guide 1.26 and ANSI/ANS 51.1.

The Seismic Category I structures, systems, and components are selected in accordance with the guidance provided by Reg. Guide 1.29.

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[1] European Pressurized Reactor, Franco-German project of nuclear island for an advanced PWR, designers are: EdF, Framatome, NPI (Nuclear Power International, subsidiary company of Framatome - Siemens created in 1990 to realize a common project for export bids), Siemens, and a grouping of 9 German utilities.

[2] European Utility Requirements, these European requirements for future light water reactors are established by 7 companies (or groups) from 7 States: Belgium (Tractebel), UK (NE), Germany (VDE), Spain (DTN), France (EdF), Italy (ENEL), Netherlands (KEMA), and, from 1996: Finland (IVO-TVO) and Sweden (Vattenfall).

Russian Federation

At present, the classification of WWER-1000 NPP components, systems and structures has been made on the basis of expert examination of their functional importance. Materials on classification have been submitted for approval to Gosatomnadzor of the Russian Federation in a set of the complied "Technical safety substantiation ..." (TOB) for the operating NPPs, however approval has not yet been received.

USA

US practice is provided in the referenced Regulatory Guides.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Safety functions and component classification for BWR, PWR and PTR, A Safety Guide, Safety Series No. 50-SG-DA, IAEA, Vienna (1979).
- Sicherheitskriterien für Kernkraftwerke. Bekanntmachungen des BMI vom 17. 5. 1979 - RS I 6 - 513 301 - 4/5c/15.
- RSK-Leitlinien für Druckwasserreaktoren (3. Ausgabe vom 14.10.81, einschließlich der Änderungen vom 15.12.82 und vom 31.03.84).
- RFS, Basic Safety Rule, France, IV.1.a: Classification of mechanical equipment, electrical systems, structures and civil work (21.12.1984).
- USNRC Regulatory Guide 1.26, Quality group classification.
- USNRC Regulatory Guide 1.29, Seismic design classification.

ISSUE TITLE: Qualification of equipment and structures including ageing effects (GL 2)

ISSUE CLARIFICATION:

Description of issue

In accordance with NUSS 50-C-D, Section 12, the qualification of equipment important to safety is required to demonstrate their ability to fulfill their intended functions. This qualification requirement applies to normal operating conditions, to accident conditions and to internal and external events. In addition, according to international practice, it should be possible for the plant operators and the regulatory body to examine the associated qualification reports.

A major concern with respect to WWER nuclear power plants, as shown by safety reviews, is that this practice of qualification of equipment is either lacking or not evident. An example of this is the qualification of electrical and I&C equipment, including cable connections, for LOCA conditions. Neither the specifications concerning the test procedures nor the test reports are available at the nuclear power plants. In addition, safety reviews have shown that the cable connections, especially inside the containment of WWER nuclear power plants, are not able to withstand extreme environmental conditions and consequently, they have a high failure potential under LOCA conditions.

A further example in WWERs is the seismic qualification of systems important to safety, especially ventilation systems which should be safety-graded but are not, and safety support systems like the service water pumps, fire water supply pumps and indication and recording instrumentation. Since these are not qualified with respect to seismic loads, their functional capability on demand in the case of an earthquake would be questionable.

As a NPP ages, its reliability and the safety margins provided by the design tend to decrease and its operation and maintenance are likely to increase unless the awareness of the need to manage age related degradation is imparted to and acted upon by the plant maintenance and operations personnel. Ageing of NPP systems, structures and components (SSCs) important to safety must be effectively managed to ensure the availability of required safety functions throughout plant service life, including any extended life. To manage ageing of SSCs important to safety effectively, plant owners/operators need to have in place effective programmes that provide for timely detection and mitigation of ageing degradation in order to ensure the required safety margins (i.e. integrity and functional capability) of the SSCs are maintained. The safety authorities are responsible for verifying that the ageing is being effectively managed and that effective programmes are in place for continued safe operation.

The concern is applicable in particular to old generation NPPs designed to standards which did not call for qualification of equipment especially with regard to accident conditions and ageing.

Safety significance

This issue was identified from safety reviews and represents a deviation from international practice and especially from NUSS 50-C-D. Insufficient or lacking qualification of equipment important to safety with respect to extreme environmental or seismic conditions would seriously affect defence in depth and the safety functions would be questionable for scenarios within the DB envelope.

Source of issue (check as appropriate)

- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

The measures proposed in the modernization programme for Kozloduy Units 5 and 6 include:

- analysis of available qualification documentation concerning safety important equipment, and
- qualification of ventilation systems, cable penetrations, fire doors, and fire alarm facilities, etc.

France

At the beginning of the eighties, it was noticed that a thermal ageing with an increasing embrittlement and a loss of resilience and toughness affected austerno-ferritic molden steel, in particular RCCS elbows and volutes of primary coolant pumps.

Important work was undertaken to assess lifetime of these parts of components. Ageing mechanisms were studied through an important programme of tests, with accelerated ageing of samples and tests on full scale specimens. Some sensitive elbows were replaced when occurred steam generators replacements [5 elbows for Dampierre-3 (3 elbows on hot legs and 2 "cold elbows" adjacent to SG)], 5 other elbows will be replaced during the next SG replacements at Gravelines-4 and Tricastin-4.

Since 1993, much progress has been made in the knowledge of mechanisms of ageing. Mechanical tests at full scale (and scale 2/3) showed that there was no risk of break due to embrittlement. Some very limited tear of defects were observed, very far from instability or break, so no replacement is mandatory for safety reasons, nevertheless metallurgical examinations and gammagraphy control of elbows are planned for the ten-yearly inspections to check if the evolution of defects is in accordance with predictions.

Germany

Requirements for the qualification of electrical and I&C equipment for LOCA conditions are contained in KTA 3403, 3502, 3504, 3505, 3506, 3507 and 3705. The procedure for the periodical demonstration of their qualification for LOCA conditions is described in KTA 3706 (draft).

The seismic design of equipment in German NPPs is based on the KTA 2201. Re-evaluation is carried out within the framework of the periodic safety reviews taking into account the revised KTA 2201.1 and KTA 2201.3/4.

The safety practice in German nuclear power plants has resulted in extensive measures to counteract impermissible effects of ageing in the systems, structures and components used, such as:

- design, fabrication and testing of systems, structures and components in accordance with accepted standards and regulations, taking state-of-the-art knowledge with regard to ageing into account
- monitoring of systems, structures and components and their operating conditions with regard to changes relevant to safety
- regular replacement of failure-prone component parts of systems, structures and components within the framework of preventive maintenance
- upgrading and exchange of systems, structures and components in case significant weak points regarding safety are detected
- optimising the technical systems and operating conditions
- validation of the design by simulating worst-case EOL conditions in large-scale experiments.

At present, no need is seen for any separate ageing management programme or regulation.

India

The qualification programme has been initiated. Cables and cable connections have been qualified for LOCA conditions.

Japan

The requirements on the functions of equipment important for safety are described in the relevant sections of "Guide for Safety Design of Light Water Nuclear Power Reactor Facilities", and their compatibility has been reviewed in course of the safety review process and, for certain equipment, demonstration test of the functions and seismicity had been carried out.

As for ageing, the government (the Ministry of International Trade and Industry) prepared "Basic Policy on Aging" and utilities are evaluating, in accordance with it, long term integrity and current maintenance of major equipment and structures of their power stations with long years of operation. Through such evaluation, matters requiring attention will be found out, and long term maintenance plans will be developed.

Korea, Republic of

To assure the required functional operability of safety related equipment under the harsh environmental conditions, environmental design and qualification equipment are performed in accordance with the requirement of IEEE 323-1974.

The seismic qualification of seismic category I mechanical and electrical equipment should meet the requirements of IEEE 344-1987. It is done by testing, analysis or combination of both, as appropriate.

Regulatory body is considering to implement 5-year PSR (Periodic Safety Review) to identify any weakness.

Russian Federation

The improvement of equipment qualification is supposed to be carried out in the following stages:

- review of existing documents in the light of their completeness both as to nomenclature and scope of the requirements involved in them;
- complete the list of qualified equipment;
- updating of generic specifications and standards available in RF for mechanical and electrical equipment;
- perform missing qualifications on some equipment including Prz PSD, SG PSD, BRU-A.

The Russian proposal plans the reconstruction of cable connections inside the containment to ensure their operability under the expected post LOCA conditions.

USA

Seismic and environmental qualification of equipment is required in US plants to assure that the equipment will perform its intended functions under design-basis loading and environmental conditions. For older plants which did not receive adequate qualification when it was designed, the NRC has required that the equipment be reevaluated.

The NRC has recognized that the effective management of equipment ageing and maintenance at nuclear power plants are essential for continued safe operation. The maintenance rule, 10CFR50.65, gives the NRC the regulatory authority to verify and ensure that licensees are monitoring the effectiveness of maintenance activities for all SSCs deemed important to plant safety.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Implementation and review of NPP ageing management programme, A Safety Practice, Safety Series No. 50, Final Draft, Vienna (1995).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Code on the safety of nuclear power plants, design, Safety Series No. 50-C-D (Rev.1), IAEA, Vienna (1988).
- Kerntechnischer Ausschuss (KTA), Germany, Nuclear Safety Standards.
- Title 10, Code of Federal Regulations, Part 50, 10CFR50.49, Environmental qualification of electrical equipment.
- Title 10, Code of Federal Regulations, Part 54, 10CFR54, License renewal.
- USNRC Unresolved Safety Issue A-46, Seismic qualification of equipment.
- USNRC Generic Letter 87-02, Verification of seismic adequacy of mechanical and electrical equipment in operating reactors.
- Recent USNRC Information Notices 92-27 Supplement 1, 97-10, 97-11, 97-13, 97-29, 97-45.

ISSUE TITLE: Inadequacy of reliability data (GL 3)

ISSUE CLARIFICATION:

Description of issue

The qualitative judgement of a nuclear power plant can be made on the basis of plant and engineering experience. To enable the quantitative evaluation (e.g. PSA) of a nuclear power plant, however, a well-organized database is a pre-requisite.

The type and degree of detail of the database is determined by the intended use of detail of the evaluation. For a general evaluation, a generic data base might be sufficient. But, for a systematic evaluation of the level of plant safety and for an optimal maintenance programme, a plant specific data base is generally required.

Safety significance

The lack of component and human specific reliability data can lead to inadequate decisions with respect to design or procedure modification and regulatory requirements when using quantitative tools (e.g. PSA) to identify weaknesses and their prioritizations.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Argentina

Several procedures have been developed to assure quality of the reliability data of NPPs. The utility have been recording reliability data on many areas of reactor plant from more than ten years, in particular on Atucha I PHWR NPP, with more than 20 years of operating experience and unique design. Such data retrieval was increased with the development of the PSA taking into account the scope and the applications of such assessment. Realistic input data for PSA models were necessary, in particular to include common cause failures (CCF) data due to its dominant risk factor on PSA's. The basis and the clear justification of safety related component data is a regulatory requirement.

France, Germany, Sweden, USA

Reliability data from safety related components in nuclear power plants have been collected since the mid-70s in France, USA and Sweden. All plant owners in the Nordic countries have joined together and established a database for component failures, which is continuously updated. A new edition of the database with updated reliability data (both plant specific and general) including analysis of failure rates, uncertainties etc. is published about every four years to assist the utilities in performing their PSAs. The same work has been performed with respect to initiating events. Recently, the 2nd edition has been issued, based on Nordic experience.

A work has also been initiated for collecting reliability data for common cause failures (CCF). The work is planned to be a collaboration between France, Germany, Sweden and USA, where CCF data have systematically been collected for a long time. The working group is called ICDE (International Common Cause Failure Data Exchange).

In France, raw data, which have been collected by the plant operators, are brought together into the experience feedback databases of the EdF, Research and Development Division. The data, which is required for developing a number of tools, are validated, processed and analysed with methods applied or developed by the group. Several databases have been developed by EdF depending on the demands, e.g. RPDF, Periodic Reliability Data Book, SRDF, Reliability Data Collection System.

In the USA, the nuclear power utilities have a common database (NPRDS), which is only a failure database. It does not include information on planned or unplanned unavailability, which is also necessary for probabilistic safety evaluations. In the US, true reliability and availability data has been collected sporadically by the NRC and the utilities to facilitate probabilistic evaluation of specific issues. The USNRC is considering rulemaking for owners of nuclear power plants to routinely collect and report reliability data that is suitable for use in PSAs.

India

The data collection is established only for some safety related systems and equipments.

Japan

In Japan, the interim report of the "Technical Committee on the Common Subjects on Nuclear Power Plants" also recommends to develop domestic data from the standpoint of more practically reflecting the favourable operating performance. Currently, the utilities and the Central Research Institute of Electric Power Industry have been accumulating data on a continuous basis, and the equipment failure rate database were developed for PSA use.

Generally, it is thought desirable, in implementing PSA evaluation, to use a plant specific database to reflect plant operating and maintenance experiences. In Japan, however, examples of troubles are evaluated and measures are applied to other plants to prevent their recurrence, and failure data are more or less consistent among plants, and it is therefore thought that there should be no problem even if a domestic common database is used. Also, as a domestic common database allows a larger sampling group, it should be more advantageous due to less uncertainties than establishing individual plant database.

Korea, Republic of

Korean nuclear power plants have had 110 reactor years operation time and about 400 reactor trips. KINS (Korea Institute of Nuclear Safety) constructed a database (NEED; Nuclear Event Evaluation Database) on reactor trips for the systematic application of operational experience feedback.

Input data were prepared by the licensee and submitted to the KINS for review. The input prepared by the licensee were reviewed by a special team at the KINS consisting of experts in this area. The special review team examined the input data and corrected misunderstandings. The basic design principle applied in preparing input was the maintenance of consistency with IAEA-IRS.

The utility and vendors have taken reliability data from INPO NPRDS "Nuclear Plant Reliability Database System" so far, but a long term project for the development of a Korean-specific reliability database system is supposed to get started in time.

Russian Federation

Plant specific database of Balakovo, Kalinin, Novovoronezh NPPs have been done.

ADDITIONAL SOURCES:

- T-book, 4th edition - Reliability data of components in Nordic nuclear power plants, TUD-Office, Vattenfall Energisystem AB, 1994.
- I-book, 2nd edition - Initiating events at Nordic nuclear power plants, SKI-Report 94:12.
- NPRDS, Nuclear plant reliability database system, INPO.

ISSUE TITLE: Need for performance of plant specific probabilistic safety assessment (PSA) (GL 4)

ISSUE CLARIFICATION:

Description of issue

Both the basic safety philosophy and the original design of safety systems of existing nuclear power plants have mainly been based on deterministic criteria. These criteria have been formulated largely on the basis of sets of pre-defined initiating events, specification of a single failure assumption for systems responding to these events and a qualitative treatment of the likelihood to determine acceptable consequence guidelines. The criteria used have been expressed as design rules (e.g. General Design Criteria), that have been applied to evaluate the acceptability of proposed plant designs.

This approach has provided adequate protection against the selected DBAs, but has also resulted in a focus on the particular safety features that are designed to cope with the selected initiating events.

The development of probabilistic techniques, especially Probabilistic Safety Analysis (PSA) for nuclear power plants, has greatly improved the possibilities to view plant safety within a uniform framework. PSAs have greatly improved the possibilities to:

- Identify safety issues (involving multiple failures or low probability events of high consequence).
- Evaluate their overall impact.
- Compare different issues.
- Prioritize among safety issues.

Furthermore, the performance of a PSA will often result in new priorities among known safety issues.

Among the most important plant specific features to be included in a PSA, are:

- Categories and frequencies of initiating events.
- Secondary effects from initiating events.
- Realistic success criteria of safety systems (system capacity, operating times, and required human interaction, possibility of recovery).
- Detailed system models, including room dependencies of both the system itself and relevant auxiliary systems.
- Plant specific component failure data.
- Plant specific analyses of human interaction.
- Plant specific evaluation of vulnerability to external events.

The need for and usefulness of PSA as a tool for integrated safety evaluation became obvious in connection with the analysis of early safety related incidents and accidents in nuclear power plants. The early performance of a large number of PSAs and intensive research activities within areas of special complexity have resulted in a gradual development and a steadily increasing maturity within the field.

Safety significance

Probabilistic safety assessment can be seen as an extension of deterministic analyses by systematically considering accident sequences and events which go beyond the design basis of a nuclear power plant. PSA applications are evolving and are useful in complementing the deterministic approach as long as the PSA used has an adequate quality for the intended applications and the staff carrying out the probabilistic assessment include experienced personnel who know the capabilities and limitations of the PSA and of the plant design and operation.

A lack of adequate PSA applications would certainly make it impossible to find out vulnerabilities or weaknesses in the safety balance of the plant for setting priorities in safety improvements.

Source of issue (check as appropriate)

- __________ operational experience
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

The level of activity world-wide has been considerable, both regarding the actual performance of PSAs and when it comes to supporting research activities.

Argentina

The main benefit PSA offers is a deep knowledge of the safety aspect through design, operation and maintenance of the facility, including the identification of the dominant probability contributors of the core damage, allowing at the same time the qualitative and quantitative comparison of design alternatives as well as operational modes of the plant. The aim of such Assessment is to evaluate the safety level of the NPP, identify the areas that might require improvements, compare the safety level with national standards and international recommendations and assist the operation of the plant.

The degree of details and scope of the PSA of the Argentinian NPP's were defined by the Regulatory Authority taking into account the above mentioned applications required. The scope of radioactive release sources, operational states, initiating events, event trees, fault trees and quantification was also defined for each plant.

The PSA is also used to support regulation basically to prioritize inspection tasks, to perform safety reviews and to identify weaknesses, backfitting and design changes. The PSA applications from the regulatory point of view are: identification and resolution of plant vulnerabilities, re-evaluation of regulatory requirements, support regulatory interaction, improve operator training on effective areas, require effective modifications.

According to our experience, the use of the PSA as a regulatory tool as applied in Atucha 1 NPP PHWR- KWU, showed high benefits to plant safety. The regulatory requirements issued on Atucha 1 NPP based on the PSA related to design changes (prioritization of plant modifications) and procedure improvements were very important to safety in terms of core damage probability and in safety itself. The PSA is being use in Atucha 1 NPP to improve both operator training programme and the plant safety culture. The backfitting policy, upgrading programme and design changes important to safety related to Embalse NPP will be analyzed using the PSA as a regulatory decision-making tool.

France

Currently EdF intend to further develop PSA models in the frame of its PWR <u>subseries</u> (900 MW, 1300 MW and 1400 MW).

India

PSA has been taken up for Tarapur Atomic Power Station (TAPS).

Japan

In Japan, probabilistic safety assessment of individual plants have been performed on internal events, and it has been done within the framework of examination of measures against severe accidents. The results of assessment were reported to the regulatory body in "Report on the Examination of Accident Management." The results of PSA have been arranged by reactor type and by containment vessel type, taking into consideration plant safety design, and proper measures against severe accidents have been identified for each type of the plant.

Korea, Republic of

There are implicit recommendations by the regulatory body to carry out PSA for all new and operating plants in Korea. The PSAs are performed to identify plant specific vulnerabilities to severe accidents and to seek cost beneficial improvements. We have already completed Level 1 PSA for four of our eleven operating plants, and Level 2 PSA in process including accident management programme for seven new plants. IPE will be applied to the remaining five plants in operation.

The PSA work scope concentrates on the initiating events during full power operation mode. However, to identify the plant's vulnerabilities during low power and shutdown operation mode, it was recognized that the future R&D plan for shutdown PSA should be prepared.

Russian Federation

For some NPPs a partial probabilistic safety analysis of the first level has been performed, in particular for Rostov and Balakovo-4 NPPs. For Zaporozhe, Kalinin and Novovoronezh NPPs it is under way.

Spain

There is a PSA framework programme issued by CSN in 1983, according to which all NPPs are carrying out PSA studies. This programme is being reviewed by CSN in 1997.

Most plants have already completed their level 1 PSA and some have completed level 2 PSAs considering external events. In a 5-year period, all plants are to complete the level 2 PSAs, including external events and all operational modes.

The licensees are considering the feasibility of maintaining models and data updated so as to keep a living PSA.

Sweden

A condensed summary of measures taken in Sweden includes the following:

- Performance of plant-specific PSAs by utilities, based on general requirements from the Swedish Nuclear Power Inspectorate (SKI)
- Common research activities within a number of areas, e.g.
  - (a) general PSA methodology;
  - (b) external events;
  - (c) common cause failures; modeling and data;
  - (d) human interactions;
  - (e) initiating events (LOCA);
  - (f) PSA uses for TechSpec evaluation.

- Extensive activities concerning common experience data bases that are continuously updated
  - (a) component failure data;
  - (b) frequencies of initiating events.

USA

A probabilistic safety analysis has been performed for all large US nuclear power plants in response to the need to perform Individual Plant Evaluations (IPEs). The NRC has issued a policy statement on the use of probabilistic analyses in the regulatory process. A recent document has been issued summarizing initial results from IPEs received and reviewed by the NRC. This document, NUREG-1560, Parts 1 and 2, also provides a discussion of the attributes that contribute to a high quality PSA.

ADDITIONAL SOURCES:

- MOHT - OTJIG - EDF generic reference programme for the modernization of WWER-1000/320, Revision 6 (February 1997).
- MOHT OTJIG RM, EDF-DE-CLI, Revision 6, February 1997.
- The safety in the Spanish nuclear power plants, Nuclear Safety Council, Spain, May 1992.
- Integrated PSA programme, Nuclear Safety Council, Spain, August 1986.
- NUREG-1560, "Individual plant examination programme perspectives on reactor safety and plant performance", Draft report for comments, November 1996.
- USNRC Policy Statement on Use of probabilistic risk assessment.
- USNRC Generic Letter GL-88-20 (IPE).

4.1.2. Reactor core (RC)

ISSUE TITLE: Inadvertent boron dilutions under low power and shutdown conditions (RC 1)

ISSUE CLARIFICATION:

Description of issue

Two types of inadvertent reactivity insertion may happen at low power or shutdown conditions:

- Low rate boron dilutions due to a leak into the primary circuit or due to human errors.
- Fast boron dilutions caused by a rapid and massive injection of pure water into the core. This could happen, for instance, when restarting a main primary pump after a period of shutdown at low residual power and therefore with no, or low level of, natural circulation and with presence of clear water in the loops due to leakages from the connected auxiliary circuits or condensation processes in the steam generator (see AA 4 and SS 7). It should be taken into account that the natural circulation is blocked when the Residual Heat Removal System (RHR) is in operation. In such conditions, in case of total loss or shutdown of the primary pumps, homogeneous conditions are not guaranteed in all the loops.

The issue and its impact on the safety of a PWR were identified within the framework of the probabilistic safety assessment of the French PWRs at the end of the 80s.

Inadvertent dilution can in principle be detected using nuclear flux measurements and boron concentration monitoring. Neutron flux monitoring is difficult during reactor shutdown or startup phase, because neutron level in the ionization chambers is very low (10^4 to 10^3 n/cm^2×s). The continuous boron concentration monitoring is based on boron meters which have limited sampling points in the RCS.

Safety significance

In-depth analyses have shown that the design of the PWR is adequate to cope with all anticipated slow dilution scenarios.

Fast injection of pure water into the core could result in prompt criticality in some reactors with large potential damages to the first barrier (fuel cladding) in a situation where the third barrier (containment) might be open.

Source of issue (check as appropriate)
- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

The following measures are being implemented in all French PWRs:

- piping modifications in the auxiliary systems to prevent any spurious dilution;
- automatic shut-off of the dilution and of the primary gas extraction functions following a loss of all main primary pumps;
- interlock preventing dilution or primary gas extraction when the RHR system is connected;
- alarm in the control room if the boron concentration of the water injected into the primary circuit is too low compared with a preset value;

- modification of all the operating procedures involving the boron dilution function in order to prevent any spurious dilution;
- alarm in the control room and interlock preventing control rod withdrawal when the flux doubling time is short.
- PSA studies carried out indicate for inadvertent boron dilutions a core melt risk of about 10^{-7}, taking in account measures as already performed, however Safety Authority has required an increasing of criticity margin for shutdown state (from -1000 pcm to -2000 pcm).

Germany

A slow boron dilution at low power or shutdown will be detected by monitoring the boron concentration and/or the neutron flux. To avoid unintentional slow boron dilutions during refueling, administrative measures have been taken. There is a check list which includes all valves of systems connected with the RCS through which pure water could flow into the RCS. During refueling the closed position of these valves must be verified.

A fast boron dilution in the RCS will be prevented by design measures. An ongoing boron dilution is stopped automatically if one of the following events occurs: (1) initiation of a fast power limitation, (2) actuation of reactor scram, and (3) stop of all reactor coolant pumps. A redundant interlock will prevent injection of demineralized water in the RCS. This is performed by an automatic closure of valves in the Chemical and Volume Control System (CVCS) and by switching off the demineralized water pumps.

Japan

Dilution of boron is managed by injecting a certain limited amount of demineralized water into the reactor coolant system. Dilution of boron exceeding the specified level could not occur as the valve on the demineralized water injection line closes automatically upon completion of injection of a certain amount of demineralized water.

It is designed so that the operator is required to go through procedures of two steps of switching to the dilution mode from the automatic supply mode, and operation of the start switch, and that no dilution occurs if one of the steps is not performed, thus the possibility of inadvertent dilution is minimized.

The chemical and volume control system is so structured so as to limit the maximum boron dilution ratio after recognizing abnormal conditions through an alarm "High Neutrol Flux during Stoppage of Reactor in Neutron Source Domain" or an alarm "Control Rod Cluster Insertion Limit." Thus the operators can have enough time to take appropriate correction measures.

Korea, Republic of

At the Ulchin NPP Units 3 and 4:

- Analysis showed that the cold shutdown mode results in the least time available for detection and termination of the event.
- The indications and/or alarms are available to alert the operators that a boron dilution event is occurring in each of the operational modes.

Spain

New operating procedures have been written on boron dilution, as well as on new surveillance procedures for calculation of shutdown margin and sub-criticality.

The technical specifications applicable to refueling mode have been modified.

Dilution pathways have been submitted to administrative controls in order to prevent them from happening inadvertently.

One plant has introduced the "variable shutdown margin" into the technical specifications, another one has re-analysed the accident of boron dilution contained in the "accident analysis" section of its Safety Analysis Report.

Redundancy and diversity have been required for those alarms that detect boron dilution accidents. These alarms shall be referred to within the procedures of "failure instructions."

Countries operating WWER NPPs

The following measures are planned or being implemented in WWER reactors:

- Modification of the operational procedures and installation of additional interlocks to prevent uncontrolled boron dilution.
- Replacement of the existing boron meters with the new NAR-12 models which have increased measuring accuracy and the ability to monitor B-10 isotope. In addition, in order to have greater sensitivity immediately after replacement of 1/3 of the core, a new neutron monitoring system that uses new detectors (AKNP-7-02) with sensitivity and improved signal processing, displays and alarms are under development.
- Performance of an analysis on the possibilities for primary coolant dilutions and their consequences.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- NUREG/CP-0158 - NEA/CSNI/R(96)3: Proceedings of the OECD/CSNI Specialist Meeting on Boron dilution reactivity transients; held in State College, Pennsylvania, USA, October 18-20, 1995.
- USNRC Information Notice 96-69, Operator actions affecting reactivity.

ISSUE TITLE: Unreliable insertion of control rods in PWRs and WWERs (RC 2)

ISSUE CLARIFICATION:

Description of issue

Increased droptime and/or jamming of single or several control rods had happened at some western PWRs and Soviet-designed WWER-1000 reactors. These abnormal events were observed during reactor scrams or control rod drop tests. The drop time exceeding the design limit, or control rod jamming at a position above the bottom and even at the top of the reactor core is considered a licensing event which violates the technical specifications of plant operation. Three root causes for this safety issue have been identified so far:

(i) The delayed insertion of all control rods at Daya Bay (1994/95) was caused by progressive wear of control rods and the control rod cluster guides above the core due to inappropriate design of the continuous guide structure.
(ii) CDRM blockings or sticking rods in the upper position as observed at some French plants is considered to be caused by degraded parts of the CDRMs affected by high frequency of maneuvers performed.
(iii) Large RCCA guide thimble deformation within deformed fuel assemblies (FA) caused by irradiation induced creep and promoted by several factors is considered to be the root cause for delayed insertion and/or jamming of control rods mainly in the dash pot area in several cases: WWER-1000 reactors at Zaporozhe, South Ukraine, Rovno, Khmelnitsky, Balakovo and Kalinin (1992 and later) during the third year of fuel cycle and Kozloduy 6 after two years fuel cycle; Ringhals 3 and 4 (1994); South Texas-1, Wolf Creek and North Anna (1995/96); Doel-4 and Tihange-3 (1996) and Almarez-2 (1997).

The factors recognized so far are such as excessive axial forces on FAs induced by holddown springs, upper internals and guide thimble tube growth, initial (FA) deformation, burnup, lateral FA stiffness as well as absorber swelling and its cladding wear, corrosion deposits on the rodlets and guide tubes, etc. may also contribute.

All Member States concerned have implemented compensatory and/or interim measures such as frequent drop tests, changes of the operational regime as well as measures to eliminate the contributing factors such as reduction of axial FA loads, increasing of FA stiffness, insertion of FAs with lower burnup on RCCA positions, etc.

Compilation of data to get an early indication of these events and to prepare proper operator actions are also recommended.

Russian WWER-1000 fuel vendor is considering a new FA design avoiding stainless steel used for guide thimbles and guides so far, higher lateral stiffness and heavier RCCAs.

These events which directly impact fuel performance due to increased fuel duty such as higher burnup, increased operating temperatures, larger fuel cycles, etc. are considered to be the consequence of insufficient validation of new fuel designs within the last decade.

Safety significance

The events of delayed insertion and/or jamming of control rods due to excessive friction of RCCAs in their guide thimbles raised concerns about the operability of control rods in high burnup FAs. Although most of the testings and analysis to date has shown that the control rods have reached the dashpot area of the guide thimbles and that adequate shutdown margin has been maintained (even if all control rods are sticking in the dashpot area) there have been indications of degraded drop times and stuck rods well above the dashpot area.

There is a broad safety concern that these events may be precursors of more significant control rod problems affecting the required shutdown margins and rod droptimes necessary to terminate certain DBAs properly. The CRDM failures of sticking control rods in the upper position have been considered even more safety significant.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

A first indication of an abnormal friction phenomena during the drop of the RCCAs has been encountered in 1995 at Nogent 1. The consequences of the friction inside the thimble guide tube were a significant increase of the drop time inside the dashpot and a disappearance of the RCCA recoil at the end of the drop, but without any impact on the drop time till the entrance of the dashpot.

Since 1995, the same phenomena were observed on several 1300 MWe plants Belleville 1 and 2, Cattenom 2, Paluel 4. On Paluel 3, during a requalification test after an exchange of the CRDM, five RCCAs stopped before their full insertion. The same problem occurred on Nogent 1 in January 1997. An in-depth investigation programme was set up and the results indicated that the fuel assembly deformation is correlated to this problem.

To manage this problem following actions are systematically undertaken:

- drop time assessment at the end of the cycle;
- measurement of the RCCA insertion and removal forces in the pool site.

The results obtained give indications on the fuel deformations and make easier to define the next core configuration in order to put the most relevant fuel below the control rod positions.

In addition the fuel vendors are requested to improve the fuel assembly design to mitigate the problem. The remedies are the following:

- to increase the lateral stiffness of the assembly;
- to reduce the axial force due to the hold down spring system;
- to reduce the gap between the assemblies at the grid level, to anticipate the contact of the assemblies
- to reinforce the stiffness of the dashpot by an increase of the metal thickness in this zone.

Germany

Regular inspections of rod cluster control assemblies (RCCAs) are performed during refueling periods by visual examination, eddy current testing and diameter profilometry. Generally, all RCCAs will be inspected within four years period.

Also regular tests of rod drop time are required.

Only minor deformation of fuel assemblies has been observed at German PWRs as a result of radiation induced creep. This caused damages at some fuel assemblies (FA) spacer grids during refueling, but did not affect movement or fast insertion of control rods. New improved FAs have been constructed to minimize bending. For this purpose, very stiff „blind“ guide thimbles have been implemented within the FAs.

Japan

In Japan, there have been no event similar to ones occurred at Wolf Creek, etc. As a practice of periodical inspection, control rod drop tests prior to reactor start-up and control rod operability tests during operation are conducted in Japan.
After learning the occurrence of the events in USA, control rod drop tests were conducted during shutdown operation at a representative plant with relatively high burnup in the reactor vessel.

Also, control rod drag test were conducted on relative high burnup fuel in a spent fuel pit. In both cases, no abnormalities were observed. Even though there are some structural differences between USA and Japanese fuel, finding and measures taken in USA is also being closely watched in Japan.

The issue on the wear of control rod has also been examined in Japan and the following measures have been taken:

- As for conventional control rods, their life has been established and they are replaced accordingly.
- An improved type with measures taken as to Cr plating on cladding tubes and reduction of neutron absorber diameters has been introduced, and its performance is being studied.

Korea, Republic of

At the Ulchin NPP Units 3 and 4:

- The ability of the CEAs to have their position changed is tested quarterly during power operation.
- At every refueling shutdown, the CEAs are stepped over their entire range of movement and are drop-tested to demonstrate their ability to drop within the required time.
- As a test result, CEA drop time was within 4 seconds and met the requirements in case of YGN 3&4 which are reference plants of Ulchin 3&4.
- The structural adequacy of the CEA guide tubes was evaluated against the mechanical loads imposed by seismic and LOCA events and was reviewed and accepted by KINS staff during the licensing safety review.

Spain

One Spanish PWR, Almaraz-2, has experienced this problem, where visual inspections with baroscope as well as dimensional and corrosion inspections based on eddy current tests and bouldering have been carried out.

Simulator training sessions simulating this scenario are being stressed, emphasizing the use of those emergency procedures which are applicable.

Rod drop times and drag forces are required for PWRs and in some plants, they have already been performed.

A complete rod insertion is required to be verified for each reactor scram and in case any rod does not fully insert, a rod drop time test and rod recoil are required.

USA

The USNRC issued a Bulletin in 1996 based on adverse experience in US plants.
In light of recent incomplete control rod insertion events, the incomplete understanding of the root causes, and the rate at which such problems appear, the ability to insert the control rods fully must be demonstrated at appropriate intervals in order to meet the current licensing basis. A proposed Bulletin Supplement is under consideration by the USNRC which would specify burnup limits as one way to resolve the problem. Should licensees choose to limit burnups to less than those stated, the problem

would be resolved. An evaluation of the safety impact of options available, such as testing, redesign of the core, or rigorous engineering analysis, is the responsibility of each licensee.

Countries operating WWER-1000 NPPs

- Control rod drop times are measured at least once every 3 months. If any control rod drop time is more than 4 seconds, the next test is carried out within a month.
- If excessive rod drop times are observed at full coolant flow rate, operation with three or two reactor coolant pumps at correspondingly reduced power is permitted, provided that the measured drop time of any rod does not exceed 4 seconds. If the transfer to operation with three or two coolant pumps is not successful, then the unit has to be shut down.
- In order to minimize the potential rod insertion problems, fuel assemblies which have been used for 2 years are not inserted into the control rod locations, but are replaced by new fuel assemblies with nearly the same physical characteristics.
- Before loading of fuel assemblies into the core, they are tested on stands for verification of free control rod movement. The deviations of lifting and lowering forces from normal values should not exceed ± 3 kg. The central instrument thimbles are measured by means of a specially designed calibre.
- Investigations of root causes are being made.
- A new design of the control rod, with approximately 30% greater weight to shorten the drop time, and a new fuel assembly design are being tested in WWER-1000 reactors.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Control rod insertion reliability for WWER-1000 NPPs, IAEA, WWER-SC-121, 1995 (report to the Steering Committee).
- CSN/IS/29/95, Report to the Congress and Senate, Nuclear Safety Council, Spain.
- USNRC Bulletin 96-01, Control rod insertion problems.
- USNRC Information Notice 96-12, Control rod insertion problems.

ISSUE TITLE: Power oscillations in BWRs (RC 3)

ISSUE CLARIFICATION:

Description of issue

Power oscillations in boiling water reactors (BWRs) occur when fluctuations in the thermal-hydraulic conditions within the reactor core cause the void fraction of the reactor coolant to travel through the fuel bundles in harmonic density waves. Divergent power oscillations result from thermal-hydraulic, neutronic generated density waves occurring within the fuel bundles. Typically, in the normal operating domain, power oscillations are stable. However, operating in the region of high power/low flow increases the probability of power instability. Consequently, monitoring of flux levels as indicated on the Average Power Range Monitors and the Local Power Range Monitors is necessary in the region of possible power oscillation.

Although methods to analyse the instability mechanism have recently become available, small variations in the factors that cause power instabilities affect the predictable performance of any given reactor for a given flow. Those factors are primarily, void fraction, fuel time constant, power level, power shape, feedwater temperature and core flow. Additionally, the design of the fuel rod and bundle can affect the formation and propagation of the void density waves. These factors affect the power/flow region at which power oscillations are probable.

Safety significance

Non-divergent power oscillations resulting from normal control systems response are not of significant concern. The primary safety concern is for divergent power oscillations and the ability of the power reactor control system to detect and suppress the oscillations before they can challenge the fuel safety limits and damage fuel cladding.

Source of issue (check as appropriate)

- ___xx_____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

India

Power oscillations not observed at TAPS.

Germany

After the event in La Salle unit 2 in 1988, two events with power instabilities occurred in German BWRs: in KKI-1 in 1991 and in KWW in 1992. In both cases the power oscillations were terminated by a reactor scram.

The measures taken in German BWRs are plant specific. In some plants a programme for monitoring core stability during power operation exists; in other plants specific stability tests are performed at the beginning and at the end of each fuel cycle. This core stability test is considered of importance with respect to the future use of MOX-fuel elements.

In all German BWRs additional I&C measures have been implemented to automatically limit reactor power in those cases when the plant parameters approach the instability region in the "power to flow map" (diagram for coolant flow versus reactor power).

Japan

Following the occurrence of an unstable phenomenon in La Salle Nuclear Power Station Unit 2 in 1988, the Ministry of International Trade and Industry of Japan re-examined nuclear thermal hydraulic stability at the Safety Evaluation Sub-Group of the Technical Advisory Group. As a result, the following measures have been identified:

- To clarify the necessary responses in case of occurring instability.
- To establish the methodology on regional stability.
- To review the conditions being applied to the stability analysis.
- To define a range for stable operation to avoid intrusion into the unstable range by the selected control rod insertion system and/or rod blocking monitor system.

Also, stability is re-evaluated when a fuel loading pattern and/or an operating plan have been established in each operating cycle.

Spain

There have been some minor power oscillations in Spanish BWRs which have been reported to IRS. Measures taken to cope with the issue have been adopted along with the BWR Owner's Group recommendations. The most relevant of these are:

- Operators training to prevent any entrance into regions of instability, for this quick recognition of power oscillation symptoms and for the management of the Fraction of Core Boiling Boundary (FCBB) concept.
- Modification of the Technical Specifications related to "thermohydraulic stability" in order to enlarge the region of forbidden operation and requirement for immediate reactor scram in case of *entrance into "region A" of operation, etc.*
- Cofrentes, a BWR-6 plant, will introduce for the next fuel cycle an automatic insertion of selected control rods, in case of instability conditions, activated by the protection system, and S.M. Garoña, a BWR-3 plant, plans to introduce some additional administrative controls.

Sweden

In Swedish BWRs there have been some experiences gained from both global and local power oscillations. The following position has been taken by the regulator:

- A predicted core stability shall be demonstrated by the utilities in their application for core loading (reloading). Startup test shall be performed and DR <0.8 shall be demonstrated.
- Tools for operator information showing actual DR during operation shall be installed.
- Partial scram (insertion of selected control rods) is accepted as an automatic control measure in case of detected high DR.

USA

The NRC issued two bulletins which required the BWR licensees ensure that adequate operating procedures and instrumentation are available and adequate operator training is provided to prevent the occurrence of uncontrolled power oscillations during all modes of BWR operation.

ADDITIONAL SOURCES:

- IAEA-IRS (Incident Reporting System); Incident report No. 1245 (Germany).
- *NUREG/CR-6003 ORNL/TM-12130*, Density wave instabilities in boiling water reactors, issued September 1992.

- NUREG/CR-5816, Wulff, W., et al., "BWR stability analysis with the BNL Engineering Plant Analyser," issued November 1991.
- USNRC Inspection Report 50-397/92-37, W.P. Ang, et al., "Special inspection," issued November 1992.
- General Electric, BWR owners group long-term stability solutions licensing methodology, NEDO-31960, May 1991 and NEDO-31960, Supplement 1, issued March 1992.
- USNRC Bulletin 88-07, Supplement 1, Power oscillations in boiling water reactors, issued December 30, 1988.

ISSUE TITLE: Loss of thermal margin caused by channel box bow (RC 4) (BWR)

ISSUE CLARIFICATION:

Description of issue

In August 1988, four failed fuel rods were found in separate fuel assemblies of a boiling water reactor (BWR). The fuel rods were heavily oxidized and the cladding was penetrated just below the top spacer grid. Secondary internal hydriding was found near the bottom of the fuel rods, resulting in the loss of fuel material. Examination of the fuel assemblies indicated that the rods failed because the channel boxes covering the fuel rods had become excessively bowed. This bowing increased the flow gap and the neutron moderation around the failed fuel rods, resulting in localized power peaking and dryout of the fuel rods. These channel boxes were in their second bundle lifetime and had been placed in the core with one side next to a fuel bundle with fresh fuel. The modeling used by the plant process computer did not consider the effects of channel box bowing and the resulting geometric variation between the reloaded and once-burned fuel assemblies. The failure to correctly model these effects resulted in non-conservative calculated values of the Minimal Critical Power Ratio (MCPR) of the core. These non-conservative values allowed the plant to be operated in steady-state conditions beyond the MCPR safety limit.

Safety significance

The degree of channel box bowing during the first bundle lifetime is typically not excessive and can be readily taken into account when calculating expected critical power ratios. However, channel box bowing during second bundle lifetimes may be much greater and have a significant effect on core thermal limits. Exceeding core thermal limits may result in fuel damage during normal operation and increased fuel damage during upset or accident conditions. Therefore, when calculating core reload configurations, it is important to use a methodology that accounts for channel box bowing or that uses a conservative bounding value for channel box bowing.

Source of issue (check as appropriate)

- ____xx___ operational experience
- _________ deviation from current standards and practices
- ____xx___ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Germany

Fuel channel box bow is a well known problem for BWRs. In principle, channel box bow is considered in the core design by neutron-kinetic and thermal-hydraulic calculations. Therefore, the degree of box bow must be supervised. Additionally, minimizing the box bow must be achieved by specific measures during design and construction of the channel boxes.

Reuse of irradiated channel boxes will be planned in the frame of a channel box management programme, which includes also the control of used channel boxes. Every irradiated channel box is checked with respect to parameters as length, bow, drill, aging, corrosion. For this purpose an automatic device for channel box check has been constructed. An irradiated channel box is only used twice if it fulfills certain criteria. On the other hand, no upper limit exists regarding the multiple use (number of fuel cycles) of channel boxes.

India

- No fuel failures directly attributable to the reason.
- Use of the channels limited to 25000 to 30000 MWd/t.
- Channels checked by go-no-go gauge for bulging prior to re-use.
- Reactor physics calculations take into account bypassing of flow.
- Ref. GE Reports.

Japan

Learning from the fuel damages (dryout) at Oskarshamn Nuclear Power Station Unit 2 in which the excessive bending of the channel box was attributed to reuse them, the channel boxes are not reused in principle in Japan.

Sweden

The current position is that a channel box is not permitted to be used for a second bundle lifetime.

Channel box bowing is taken into account in the calculation of the expected critical power ratios. The methodology used for core reload calculations has been developed by SKI in co-operation with ABB Atom and Vattenfall.

USA

The General Electric Company and the Advanced Nuclear Fuels Corporation have submitted reports to the NRC that describe methodologies for incorporating the effects of channel box bowing in critical power ratio analyses.

The NRC issued a bulletin requiring that BWR licensees determine whether any channel boxes are being reused after their first bundle lifetime and, if so, ensure that the effects of channel box bow on the critical power ratio calculation are properly taken into account.

ADDITIONAL SOURCES:

- General Electric Company Report, Effect of channel bow on margins to core thermal limits in BWRs, submitted by letter dated November 15, 1989.
- Advanced Nuclear Fuels Corporation Report, Critical power methodology for boiling water reactors - Methodology for analysis of assembly channel bowing effects, ANF-524(P), Revision 2, Supplement 1, submitted by letter dated November 30, 1989.
- USNRC Bulletin 90-02, Loss of thermal margin caused by channel box bow, issued March 20, 1990.
- USNRC Information Notice 89-69, Loss of thermal margin caused by channel box bow, issued September 29, 1989.

ISSUE TITLE: Accident response of high burnup fuel (RC 5)

ISSUE CLARIFICATION:

Description of issue

Recent experimental data suggest that high burnup fuel may be more prone to failure during design-basis transients and reactivity insertion accidents than previously thought. Tests on the relationship between fuel failure enthalpy and burnup for pressurized water reactor fuel rods indicate lower failure initiation enthalpy thresholds (measured in differential calories/gram) than was considered in the evaluation of currently approved fuel burnup limits.

Safety significance

Recent integral tests with high burnup fuel under reactivity transient conditions indicate that the bounding peak fuel enthalpy that may cause fuel cladding failure may decrease significantly with increasing fuel exposure. A study for the Electric Power Research Institute on fuel irradiated up to burnups of 60 GWD/MTU (gigawatt-days per metric ton uranium) indicates that while strength increases, ductility was found to be very low and dependent on hydrogen content and fluence, both of which increase with burnup. This contradicts a previous belief that metal annealing at operating temperatures would prevent radiation damage from accumulating after some initial period.

Although further testing is needed for conclusive determinations, present test data indicate that fuel rod damage during reactivity transients can increase significantly with high burnup. This increased fuel rod damage is of concern because of the potential for increased release of fission particles. However, fuel coolability is expected to be maintained during design-basis accidents. This expectation may be modified depending on results from future tests.

Source of issue (check as appropriate)

- ___________ operational experience
- ___________ deviation from current standards and practices
- ____xx_____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

A programme has been undertaken in 1988 to assess fatigue strength of fuel clad under irradiation and fatigue tests on sensitive fuel rod parts were realized. These tests are achieved, they have shown a reduction of Zircalloy strength, but this reduction is stabilized after the second cycle, and is not affected neither with the temperature of irradiation nor with load follow.

At the end of 1993, tests were performed, the purpose of which being:

- to assess the accident response in case of high burnup fuels (CEA CABRI loop);
- to assess margin with regard to the current embrittlement limits on fuel clad (1204 ^{0}C), oxidation rate below 17%), (TAGCIS-TAGCIR programmes[3]);
- to qualify fuel for 63000 MWd/t.

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[3] *Trempe en APRP (Accident de Perte de Réfrigérant Primaire) de Gaines de Combustible à Irradiation* Simulée / Réelle, quench during a LOCA of fuel clads undergoing simulated or real irradiation, (TAGCIS: Simulated, or TAGCIR: Real, irradiation of fuel clad), french acronyms of R&D programmes to qualify high burnup fuel to a LOCA.

As the first CABRI test, at the end of 1993, showed that some fission product dispersion could occur with a very corroded clad (corrosion thickness of 100 microns), currently French Safety Authority impose not to exceed 47000 MWd/t. Taking in account the results of other CABRI tests, EDF expect this limit could be increased up to 52000 MWd/t.

Other tests, at the Studsvik installation (experimental reactor R2), were realized with 2 fuel rods irradiated during 5 cycles to assess the limits of burnup effects and tests of 2 fuel rods irradiated during 4 cycles to assess burnup effects on load follow.

The GEMMES project (Gestion des Evolution et Modifications des Modes d'Exploitation en Sûreté: Modifications and evolutions for a safe fuel management project) started in 1993 with the objective of extending the fuel cycle from 12 months to 18, fuel enrichment is increased from 3.1% to 4% and burnup from 34000 MWd/t to 47000 MWd/t, the first refueling occurred at Cattenom-4 in June 1996. The GEMMES project for the overall 1300 MW French standard PWRs should be achieved at about mid-98.

Some specific PSA studies have been performed in the frame of assessment of high burnup fuel (based on CABRI tests).

Japan

Taking into consideration the results of NSRR and CABRI tests, the technical meeting on "the evaluation of RIA effects on high burnup fuel" had been held under the Basic Design Advisory Committee of the Ministry of International Trade and Industry.
As for the PCMI fuel failure, it is expected that safety can be confirmed by performing evaluation modifying the current threshold for PCMI.

Currently, the Nuclear Safety Commission is studying the issue, taking into consideration the results of examination by the Ministry of International Trade and Industry.
(Preliminary evaluations about the effects of additional release of FP gas on evaluation of radiation dose, effects of mechanical energy on the pressure boundary and on coolability suggest that there should be no serious safety concerns.)

Korea, Republic of

At the Ulchin NPP Units 3 and 4:

- The fuel rod design is based on an extensive experimental data base and on an extension of experimental knowledge through design application of ABB-CE fuel rod evaluation codes.
- The experimental data base includes data from ABB-CE or ABB-CE/KWU joint irradiation experiments, from ABB-CE and KWU operating commercial plant performance and from many basic experiments conducted in various research reactors which are available in the open literature.
- Evidence currently available indicates that zircaloy and UO_2 fuel performance is satisfactory to exposures in excess of 52 GWD/MTU.

USA

The NRC continues to review the data from recent tests and follow ongoing work to confirm the validity of the test results.

The NRC notified the industry of this issue through several information notices. The issue is being pursued through US industry groups. An exchange of information with other countries is ongoing.

ADDITIONAL SOURCES:

- G. P. Smith, et al., Hot cell examination of extended burnup fuel from Calvert Cliffs-1, EPRI TR-103302, November 1993.
- M. E. Cunningham, *et al.*, Development and characteristics of the rim region in high-burnup UO_2 fuel pellets, Journal of Nuclear Materials, Vol. 188, page 19, June 1992.
- P. E. MacDonald, et al., Assessment of light-water-reactor fuel damage during a reactivity-Initiated accident, Nuclear Safety, Vol. 21 No. 5, September-October 1980. USNRC Information Notice 86-58, Dropped Fuel Assembly, July 11, 1986.
- USNRC Information Notice 94-64, Reactivity insertion transients and accident limits for high burnup fuel, August 31, 1994.
- USNRC Information Notice 94-64, Supplement 1, Reactivity insertion transients and accident limits for high burnup fuel, April 6, 1995.
- USNRC Information Notice 94-13, Unanticipated and unintended movement of fuel assemblies and other components due to improper operation of refueling equipment, February 22, 1994.
- Supplement 1 to USNRC Information Notice 94-13, June 28, 1994.
- USNRC Information Notice 90-77, Inadvertent removal of fuel assemblies from the reactor core, December 12, 1990.
- Supplement 1 to USNRC Information Notice 90-77, February 4, 1991.
- USNRC Information Notice 90-08 Kr-85, Hazard from decayed fuel, February 1, 1990.

ISSUE TITLE: Fuel cladding and control rod corrosion and fretting (RC 6) (PWR)

ISSUE CLARIFICATION:

Description of issue

Within the framework of tests during refueling, damaged fuel elements were detected. The observed damage consisted of fuel cladding wear as a result of fretting induced by spacers and secondary damage like rupture of a fuel rod end-piece, increased cladding corrosion (oxidation layer up to 140 mm) and spacer-corners with high fret-wear. As a result of comprehensive inspection programmes of the reactor internals (visual inspections) and failure analysis of the major fretting problem, a local limited flow-induced vibration stress on a few fuel rods can be assumed as root cause. The initiation of oxidation is probably a problem of a special type of fuel element, which can be reduced by optimization of the application conditions (power reduction, cycle duration).

Safety significance

The loss of integrity of an important barrier against fission product releases is a challenge to protective goals. Damaged fuel rods lead to increased coolant contamination and can further cause increased concentrations of fission products in the off-gas. Loose fragments of fuel elements in the core (e.g. fuel rod end-piece) can induce further cladding damages, for instance by cooling channel closure (the rod end-piece was found in the flow-borehole of the fuel element head-plate) and by fretting.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Germany

Comprehensive inspection programmes were performed during refueling periods by visual examination, oxid-layer thickness measurement and eddy current testing.

The main measures against fretting were:

- the new fuel elements, inserted on the same positions, on which the damages occurred, were mounted with dummy-rods;
- changes of spacer-design.

For the reduction of oxidation, the fuel operation planning was optimized and as a consequence of the advanced calculation, the oxide layer is required to be limited to 100 mm for all fuel rods at the end of the fuel cycle.

Some control assemblies were replaced by improved wear resistant assemblies.

Japan

- Developments of advanced claddings are in progress to reduce water side corrosion.
- Fuel assembly design modifications have been implemented to mitigate fretting failure.

(These are reported at 1997 ANS Topical Meeting.)

Korea, Republic of

At the Ulchin NPP Units 3 and 4, the phenomenon of fretting corrosion, particularly in zircaloy clad fuel rods supported by zircaloy spacer grids, has been extensively investigated.

Since irradiation-induced stress relaxation causes a reduction in grid spring load, space grids must be designed for end-of-life conditions as well as beginning-of-life conditions to prevent fretting caused by flow induced tube vibration.

Examination of zircaloy fuel rods after six cycles of exposure at Ft. Calhoun, five cycles at calvert Cliff-1, and five cycles at ANO-2 whose fuel is the same as the Ulchin 3 &4, indicate fuel rod fretting between the fuel rod and space grid is rare.

ADDITIONAL SOURCES:

- Control rod insertion reliability for WWER-1000 NPPs, IAEA, WWER-SC-121, 1995.
- Operational experiences with German NPPs 1994.
  VGB Kraftwerkstechnik 75 (1995), Heft 4.
- Operational experiences with German NPPs 1993.
  VGB Kraftwerkstechnik 74 (1995), Heft 4.
- 1997, ANS, Topical Meeting Proceedings, pages 550-554.
- 1997, ANS, Topical Meeting Proceedings, pages 309-317.

4.1.3. Component integrity (CI)

ISSUE TITLE: Reactor pressure vessel integrity (CI 1)

ISSUE CLARIFICATION:

Description of issue

The reactor pressure vessel (RPV) integrity is ensured by a margin between its load bearing capacity affected by material properties degradation and the acting loads, which occur during operation. The neutron irradiation of the vessel material during operation causes changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors. Embrittlement caused by neutron irradiation reduces the fracture toughness of the RPV beltline materials. RPV surveillance programmes are used to monitor the changes that result from exposure of these materials to neutron irradiation at conditions similar to the RPV wall. The loads to be considered in the vessel integrity assessment are, in addition to normal operation, and pressure testing, mainly related to plant states leading to a pressurized thermal shock (PTS) event, characterized by rapid cooldown with a high level of primary system pressure. Such events depend strongly on the actual plant's configuration, systems's operation and operator's actions. Only pressurized water reactors (PWR) would experience PTS events. Qualified non-destructive examination is required to support the RPV assessment. The consideration given to all of these aspects in the original vessel integrity assessment was not adequate in some cases.

In the specific case of the RPV, it is not feasible to implement independent levels of protection and consequently, it is necessary to incorporate preventive measures into the design and operation of the vessel.

Safety significance

The RPV failure, a beyond design basis accident (BDBA) scenario, would result in the loss of cooling of the fuel and of confining the radioactive material in the RPV. Consequently, the first two barriers, fuel rod cladding and the RPV, would be lost and the containment building integrity would be threatened.

Source of issue (check as appropriate)

- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

As stated in the text the most important issue for pressure vessel integrity is embrittlement, mainly due to the modifications of the ductile-brittle transition temperature during irradiation. To improve knowledge on these modifications, specimens of metal representative of vessel shell ring and welds are installed near the internal side of vessel. After a sufficient time in the reactor, tests of toughness of these irradiated specimen are performed to confirm modelization of this phenomena.

Furthermore loading pattern of fuel allowing a reduction of neutron flux on the vessel is systematically applied for French 900 MW PWRs (standard called CPY) and will be soon extrapolated to other plants, taking in account the planned extension of fuel cycle.

Extrapolation of the current data allows to confirm a lifetime of 40 years for the vessel without any major problem.

However further studies are performed to confirm margins facing the risk of fast fracture: PSA to assess the sensitivity of main parameters which contribute to the risk and deterministic studies, taking in account fluence at the end of reactor life (reassessment with Tripoli code), exhaustive review of transients and selection of dimensioning transients, justification of defect sizes deducted from feedback experience of ISI (In-Service Inspection) and from R&D experiments for cracks under the liner. A mechanical study integrating all these parameters is in progress.

Germany

Following a requirement of the guidelines of the German Reactor Safety Commission, most PWRs in operation in Germany have limited by design the maximum fluence at the RPV wall to a value $< 10^{19}$ n/cm^2 (E>1 MeV). Besides this, the RPV beltline materials have sufficiently low copper levels. So for these plants irradiation embrittlement is low, their margins against brittle failure will be sufficient for their lifetime. The requirements for their surveillance programme have been described in KTA 3203: at least two sets of specimens from each beltline material are to be irradiated with 50% and 100% of the EOL fluence. For the other plants with higher fluences, KTA 3203 is not applicable, and their surveillance programmes are following mainly ASTM standards.

For the two oldest plants with elevated copper levels in the beltline materials, special irradiation programmes have been installed in the plants and in research reactors with dedicated materials to assess their fracture toughness. Scrap samples have been taken from internals to improve fluence determination. Measures were implemented to reduce the neutron flux (as low-leakage core and dummy fuel elements) and to mitigate the possible stresses during transients like PTS events (as preheating of emergency core cooling water (ECCW), derouting the ECCW to the hot leg, protection against overpressurization). Besides, particular efforts have been made to improve the non-destructive testing of the beltline material especially of the internal surface and the core weld. This limits the maximum crack sizes to be postulated in the analyses.

For the integrity analyses, simplified conservative methods maybe used; however, if necessary and accepted by the authorities, state-of-the-art methodology may also be applied, using best-estimate values for fracture toughness and stress intensity during the transient. With these methods sufficient safety margins could be shown for all plants for 40 years of operation. Annealing of the core weld is not required.

India

- RPV surveillance programme in force.
- Sample coupons being tested as per specified programme.
- No sign of distress as per evaluation.

Japan

In Japan, demonstration tests have been conducted since 1984 as to the evaluation of integrity of reactor vessels against PTS events. The integrity evaluation techniques for brittle fracture and the results of integrity evaluation have been approved by the committee of experts group, and the integrity of operating reactors against brittle fracture has been confirmed.

Korea, Republic of

Fracture toughness of reactor vessel should be appropriately evaluated and the surveillance programme should be suitably established. The reactor vessel beltline materials shall have sharpy upper-shelf energy, as determined from sharpy V-notch tests on unirradiated specimens of 75ft-lb. The extraction period of surveillance specimen should be in compliance with the requirements of Notice 92-20 of the Ministry of Science and Technology (MOST).

In-service inspection and material surveillance programmes are conducted during the service life of the vessel, which further ensure that vessel integrity is maintained.

Spain

The RPVs of the Spanish NPPs have monitoring programmes in accordance with Appendix H to 10CFR50 and Revision 2 of Regulatory Guide 1.99.

All the licensees have recently been requested to collect and submit the detailed data of the RPV materials to the regulatory body. This will be used to evaluate the margins for operation in the reference temperature, the absorbed energy of the RPV materials, and the effect of the residual elements on them.

No RPV thermal annealing is foreseen at the present time in the Spanish NPPs.

Countries operating WWER-440/230 NPPs

After it was recognized, that the vessel embrittlement is significantly higher than expected, a set of measures was proposed by the plant designer. The measures were addressing the material properties (flux reduction and annealing), loads (heating up of ECCS water, steamline isolation, system solutions-interlocks) and introduction of volumetric non-destructive testing for in-service inspection. Measures were also taken to improve the knowledge on the vessel material by vessel sampling. Re-evaluation of the original integrity assessment is underway at a number of plants.

Countries operating WWER-440/213 NPPs

Implementation of a supplementary surveillance programme is being considered at some plants to reduce uncertainties due to deficiencies found. In order to prevent embrittlement, flux reduction measures (low leakage loading pattern, dummy shielding assemblies) have been introduced. To reduce PTS loads due to cold water injection, heating of ECCS water has been recommended. Annealing was conducted at the Loviisa plant in 1996.

Countries operating WWER-1000 NPPs

For plants under construction, a modification of the surveillance programme is considered. Specimen containers will be located in positions representative of vessel wall conditions at Temelín NPP. Based on the fracture mechanics analysis, it was recommended to heat up the accumulator water to 55°C and to prevent injection of ECCS water with temperatures below 20°C for all the plants. The use of low leakage core loading patterns in WWER-1000 reactors would reduce RPV wall fluence by approximately 30%. It was planned to introduce partial low leakage loading patterns at some plants during 1994 (fuel assemblies with high burnup to be placed at the core periphery).

USA

GL 92-01, Revision 1. Supplement 1

The NRC issued GL 92-01, Revision 1, Supplement 1 on May 19, 1995 to require that all addressees identify, collect and report any new data pertinent to analysis of the structural integrity of their RPVs and to assess the impact of that data on their RPV integrity analyses.

Most licensees indicated that they are participating in owners group actions that will determine whether new information is available. Some licensees have provided new materials data that was not available when they responded to GL 92-01, Revision 1 in 1992. These data are being reviewed by the staff. No licensee identified a significant RPV integrity issue.

PTS Rule Revision

The PTS rule, 10CFR50.61, was revised on December 19, 1995. The revised rule (a) permits generic, values of unirradiated reference temperature different than that specified in the rule, if justification is provided, (b) requires the results from plant-specific surveillance programmes be integrated into the RT_{NDT} estimate if plant specific surveillance data has been deemed credible, (c) incorporated into the rule the credibility criteria specified in Regulatory Guide (RG) 1.99, Revision 2, and (c) required the chemistry factor and margin value be calculated using the methodology and values specified in RG 1.99, Revision 2, if credible surveillance data is used to estimate the RT_{NDT}.

Industry Database

EPRI has develop an industry database. The programme objective was: (a) to combine all available materials data into an integrated, common material database, (b) to develop special data search and retrieval capabilities for ease of use, (c) to develop tools for assisting utilities in resolving vessel material concerns, and (d) to establish a convenient mechanism to incorporate new information into the database for use in addressing future integrity issues.

Reactor Vessel Integrity Database

As a result of the NRC staff review of licensee responses to Generic Letter (GL) 92-01, Revision 1, a comprehensive database was developed to compile and record summaries of the materials properties of the reactor vessel beltline materials for each plant. This database is known as the RVID. Version 1.1 was issued in 1996.

In addition to the licensee responses to GL 92-01, the following documents were included in the review process and development of the RVID: surveillance capsule reports, documents referenced in the GL 92-01 submittals, and, as applicable, pressurized thermal shock (PTS) submittals, P/T limits reports and responses to NRC staff requests for additional information (RAI). The staff reviewed the data from these source documents and documented them in the RVID tables. Responses to the close-out letters to GL 9201 are not necessarily reflected in this version, but will be included in a future version of the RVID. Revision 1 of the RVID and the RVID user's manual will be available on the NRC's world wide web homepage at http://www.nrc.gov/. The target date for availability of the database is June 1996. To access the homepage one must have an Internet account and a web browser.

Thermal Annealing ASME Code Case, Rule, and Regulatory Guide

At the ASME Section XI meetings in Chicago in August, 1995, the Task Group on Thermal Annealing undertook development of a Code Case on Thermal Annealing of Reactor Vessels on a high priority basis. The Code Case (designated N-557) was passed by the ASME main committee on December 1, 1995. Code Case N-557 received final approval by the ASME Board of Nuclear Codes and Standards (BNCS) on March 19, 1996. The supporting technical basis document for Code Case N-557 will be published in an appropriate technical journal in 1996.

The thermal annealing rule (10CFR50.66) addresses the critical engineering and metallurgical aspects of thermal annealing. The final rule was approved by the Commission and published in the Federal Register on December 19, 1995. The regulatory guide on thermal annealing (RG 1.162) was processed in parallel with the rule package and was published on February 15, 1996. NUREG/CR-6327, which provides the supporting technical basis for irradiation embrittlement recovery from thermal annealing, was issued in March 1995. The work in this report provides the basis for the computational embrittlement recovery models in RG 1.162.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- Y. Mishima et. al. "PTS Study in Japan", Int. J. of Pres. Ver. & Piping (1994).
- Concept of safety upgrading of the operating units WWER-1000. Rosenergoatom, 1995.
- The safety in the Spanish nuclear power plants, Nuclear Safety Council, Spain, May 1992.
- Title 10, Code of Federal Regulations, Part 50, 10CFR50, Appendix A: GDC 31 and 32.
- Title 10, Code of Federal Regulations, Part 50, 10CFR50, Appendix G.
- Title 10, Code of Federal Regulations, Part 50, 10CFR50, Appendix H.
- Title 10, Code of Federal Regulations, Part 50, 10CFR50.66, "Requirements for thermal annealing of the reactor pressure vessel", December 19, 1995.
- Title 10, Code of Federal Regulations, Part 50, 10CFR50.61.
- Title 10, Code of Federal Regulations, Part 50, 10CFR50.61 (revised), "Fracture toughness requirements for protection against pressurized thermal shock events," December 19, 1995.
- USNRC Regulatory Guide 1.162, "Format and content of report for thermal annealing of reactor pressure vessels," February 1996.
- USNRC Regulatory Guide 1.154.
- USNRC Regulatory Guide 1.99 Revision 2.
- ASME Code Case N-557, "In place dry annealing of a PWR nuclear reactor vessel," March, 1996.
- USNRC Generic Letter 92-02, Revolution of Generic Issue 79 “Unanalyzed reactor vessel (PWR) thermal stress during natural convection cooling.”
- USNRC Generic Letter 92-01 Rev 1, Reactor vessel structural integrity.
- USNRC Generic Letter 92-01 Rev 1 Supplement 1, Reactor vessel structural integrity (May 19, 1995).
- USNRC Information Notice 96-32, Implementation of 10CFR50.55a(g)(6)(ii)(A), “Augmented examination of reactor vessel.”

ISSUE TITLE: Asymmetric blowdown loads on RPV supports and internals (CI 2)

ISSUE CLARIFICATION:

Description of issue

Certain transient loads that could result from a postulated reactor coolant pipe rupture adjacent to the reactor vessel had been underestimated in the design analysis. Loads are lateral loads due to jet or reaction forces, differential pressures in the annulus between the vessel and reactor cavity and differential pressures across the core barrel. Consequently, the reactor vessel supports and internals may not have the margins of safety intended.

Safety significance

Failure of RPV supports would challenge the integrity of the reactor coolant system and the functionality of safety systems. Failure of reactor internals could affect cooling of the fuel and cause stuck control rods.

Source of issue (check as appropriate)

- __________ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Germany

The RPV supports and internals are designed for a postulated 0.1 A leak. The application of the break preclusion concept eliminates the local dynamic effects of postulated guillotine pipe breaks in German NPPs.

Japan

As for the unresolved safety issue A2 (asymmetrical blow down load) issue, Japan has been solving the problem by applying the LBB concept when replacing steam generators since 1992 in the old PWR plants.

Measures with an introduction of the concept of LBB are outlined below:

- As the LBB Concept can be applicable for the primary coolant pipe, the concept was introduced when replacing steam generators.
- With an introduction of the LBB concept, leakage be detected before piping break allowing suitable time for plant shutdown.
- Asymmetric load due to break will not occur since the plant may be shutdown before pipe break.

Korea, Republic of

The application of LBB technology eliminates the local dynamic effects of postulated pipe breaks from the design basis. There are 4 piping systems in which the application of LBB is accepted. Those systems are the reactor coolant piping, the pressurizer surge line, the shutdown cooling system, and the safety injection system.

The comprehensive vibration assessment programme against steady-state and transient vibration was performed at the Yonggwang Units 3 and 4. As a result, the structural integrity of the reactor internals against steady-state and transient vibration had been confirmed.

USA

Plant assessments were received and evaluated for all licensees of PWRs in the United States.

For some important break locations, research and analysis was performed to justify a leak-before-break approach and thereby eliminate the need to consider the dynamic effects of postulated pipe breaks.

ADDITIONAL SOURCES:

- WCAP 9787 (May 1981), Tensile and toughness properties of primary piping weld metal for use in mechanistic fracture evaluation.
- WCAP 9558, Revision 2 (May 1981), Mechanistic fracture evaluation of reactor coolant pipe containing a postulated circumferential throughwall crack.
- Letter Report NS-EPR-2519, E. P. Rahe to D. G. Eisenhut (November 10, 1981), Westinghouse response to questions and comments raised by members of ACRS subcommittee on metal components during the Westinghouse presentation on September 25, 1981.
- NUREG/CP-0155, "Proceedings of the seminar on leak-before-break in reactor piping and vessels"; held in Lyon, France, October 9-11, 1995", US Nuclear Regulatory Commission, April 1997.
- NUREG-0609, Asymmetric blowdown loads on PWR primary systems, US Nuclear Regulatory Commission, January 1981.

ISSUE TITLE: BWR core internals cracking (CI 3)

ISSUE CLARIFICATION:

Description of issue

The BWR reactor internals support the core, direct the water flow and separate steam from water. In addition to the core support structure the internals comprise the feedwater spargers, the jet pumps assemblies and the steam separator and dryer assemblies. Severe degradation of the reactor internals due to intergranular stress corrosion cracking (IGSCC), irradiation assisted stress corrosion cracking (IASCC) and fatigue of BWR reactor vessel internal components has been observed at operating plants. Other degradation mechanisms are thermal ageing and irradiation embrittlement. The core shroud is the most affected component among a list of internals susceptible to IGSCC. IASCC affects sections subject to high neutron flux even at relatively low stress level (threshold), such as the top guide. Identification of cracking at the circumferential beltline region welds in several plants during 1993 led to the publication of USNRC Information Notice 93-79. Most of BWR plants have inspected their core shrouds and other internal components during planned outages, and some BWRs have identified extensive cracking.

NUREG-1544, "Status Report: Intergranular Stress Corrosion Cracking of BWR Core Shrouds and Other Internal Components," dated March 1996, summarized the status of industry initiatives to date regarding BWR core shrouds and other internals components.

Safety significance

A failure of the reactor internals could affect the control rod insertion capability, the coolant distribution and flow and cause another damage through loose parts. Safety functions controlling the power and cooling the fuel may be questioned.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Germany

In 1994, cracks caused by intergranular stress corrosion cracking (IGSCC) were observed during visual inspection on some parts of the core internals in the oldest German BWR plant in operation, manufactured of niob-stabilized stainless steel. The intended replacement of the core shroud was not performed because of the economically motivated decision to decommission the plant.

Additional in-service inspections were carried out during the normal annual outages in all German BWR plants. Up to now no further cracks were observed in core internals of any German BWR plant. Moreover, possible crack formation mechanisms such as sensitization by heat treatment or welding, influence of cold work and irradiation assisted cracking (IASCC) were investigated. Although no indications for crack formation were found, further specific inspections are carried out within the framework of in-service inspections. Fluid-dynamical and structure-mechanical analysis were performed to assess the possible consequences of assumed cracks with regard to the safety functions "controlling reactor power" and "cooling the fuel" during accidents. It was shown that the safety functions can be ensured even with postulated large cracks.

India

All accessible areas of the core are inspected regularly during refueling outages with special attention now to core shroud.

No sign of cracks observed on core shrouds of both the units in the latest refuelings in year 1995-96.

FW spargers have been replaced in TAPS-1 in 1987 as per GE recommendations. No cracks were observed.

Japan

In Japan, the measures coping with SCC related issues in vessel internal structures were examined around 1975. The mechanism of IASCC accelerated by irradiation has not been clearly defined yet, and IGSCC is considered as a major cause of degradation.

Such measures as control of reactor coolant water chemistry and application of low carbon stainless steel to the reactor internal structures have been taken.

As for some plants for which the measures had not been implemented, detailed investigation has been conducted and find some cracks due to stress corrosion. The core shroud is being replaced as a preventive measure at those plants.

Spain

Spanish BWR licensees have performed an inspection of the reactor vessel internals using visual techniques. In one of the two plants, cracking has been found in the core shroud and in one jet pump support. A restart of the plant was authorized after the evaluation of the integrity of the components and the compliance with pre-established acceptance criteria, imposing conditions for an increased surveillance of reactor operating parameters. Finally during the last outage, the core shroud was repaired by the tie-rods technique, and inspection was performed by volumetric non-destructive testing

USA

Significant circumferential cracking of core shrouds has been discovered at the Brunswick Unit 1, Dresden Unit 3, Quad Cities Unit 1, Oyster Creek, and Vermont Yankee nuclear stations. In light of the extent of cracking observed at these plants, the NRC staff evaluated potential safety concerns associated with the possibility of a 360 degree circumferential separation of the shroud following a postulated loss-of-coolant accident (LOCA). The NRC staff considered the potential for separation of the shroud during postulated accidents to either prevent full insertion of the control rods, or open a gap large enough to preclude the emergency core cooling systems from fulfilling their intended safety functions. The accident scenarios of primary concern are the main steam line break and the recirculation line break. The more serious event associated with cracks in the upper shroud welds (e.g., H2, H3) is the steam line break, since the lifting forces generated may be sufficient to elevate the top guide and potentially cause difficulties with rod insertion. The recirculation line break is the more serious event associated with cracks in the lower elevations of the core shroud. The recirculation line break is a greater concern at lower weld elevations because this type of LOCA has the potential to result in a lateral displacement of the shroud. Such a lateral displacement of the shroud could affect the ability of control room operators to insert control rods into the core and could prevent adequate core cooling.

In consideration of the consequences of a 360 degree through-wall failure of the shroud coincident with a LOCA, the NRC has conservatively estimated the risk contribution from shroud cracking and determined that it does not pose a high degree of risk at this time. However, the NRC has also determined that structural margins specified in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code could be exceeded if the cracks were sufficiently deep and continued propagating through the shroud during normal operating, transient or accident conditions, possibly resulting in the loss of a layer of the defense-in-depth strategy. Therefore, the staff has concluded that it is appropriate for BWR licensees to implement timely inspections and/or repairs of their core shrouds. To implement this position, the NRC staff issued Generic Letter (GL) 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors," dated July 25, 1994, requesting that BWR licensees inspect their core shrouds by the next refueling outage and justify continued safe operation until inspections can be completed. This position enabled the staff to verify compliance with the in-service inspection requirements of Section 50.55a of Title 10 of the Code of Federal Regulations (10CFR50.55a), and ensured that the risk associated with core shroud cracking remains low.

As of early September 1994, the NRC staff received all of the BWR licensee submittals in response to GL 94-03. The staff has completed its evaluations of the licensee responses and has transmitted the safety evaluation reports to the appropriate BWR licensees. The staff concluded that, for all cases, BWR licensees had provided sufficient justification to operate their facilities until core shroud inspections or repairs could be implemented, The staff based its conclusions on the following factors:

(1) No 360 degree through-wall core shroud cracking has been observed to date in any US BWR at which the licensee performed a shroud inspection.

(2) All analyses performed by US licensees to date indicate that, even if cracking did exist in a particular BWR core shroud, sufficient ligaments would remain in the shroud so that the structural integrity of the shroud would be ensured for the remainder of the plant's operating cycle.

(3) No US BWR has exhibited any of the symptoms (power-to-flow mismatch) that would be indicative of leakage through a 3600 through-wall shroud crack.

(4) Main steam line or recirculation line breaks are both considered to be low frequency events.

(5) There were only short durations until core shroud inspections were to be conducted or repairs were to be implemented by the individual BWR licensees.

To date, core shrouds have been repaired (modified) at the Brunswick Units 1 & 2, Hatch Units I & 2, FitzPatrick, Oyster Creek, Quad Cities Units 2, Nine Mile Point Unit 1, and Pilgrim nuclear plants. Repairs will be made at additional plants if inspection results indicate that large scale cracking of circumferential shroud welds has occurred, or may be made at the discretion of the licensee in lieu of comprehensive core shroud examinations (pre-emptive core shroud modifications). These repairs or modifications are designed to ensure the structural integrity of the core shrouds based on an assumption that the shroud circumferential welds are completely cracked, and are being reviewed by the NRC staff on a case-by-case basis.

In the spring of 1994, the industry formed a new organization, the BWR Vessel and Internals Project (BWRVIP), to address the issue of IGSCC of BWR internal components. The BWRVIP is headed by several high level utility executives to ensure that top executives in the industry are aware of its function, purpose and efforts. The BWRVIP subsequently submitted documents addressing an integrated safety assessment of the issue, guidelines on performing nondestructive examinations (NDE) of core shroud welds, guidelines on inspection scopes for BWR core shrouds, and generic guidelines and acceptance criteria in regard to performing flaw evaluations and repairs of BWR core shrouds. The NRC staff has approved the generic repair criteria document, the latest revision to the BWRVIP guidelines regarding core shroud inspection scopes and flaw evaluations, and the BWRVIP guidelines regarding core shroud NDE methods.

ADDITIONAL SOURCES:

- CSN/IS/26/94 and 30/96, Report to the Congress and Senate, Nuclear Safety Council, Spain.
- NUREG-1544.
- General Electric Sil 572 (Oct-93).
- USNRC GL-94-03, Intergranular stress corrosion cracking of core shrouds in boiling water reactors.
- Recent USNRC Information Notices 95-16, 95-17, 97-02, 97-17.
- USNRC Information Notice 94-42 and 94-42 Supplement 1.
- USNRC Information Notice 93-79.

ISSUE TITLE: Thimble tube thinning (CI 4)

ISSUE CLARIFICATION:

Description of issue

The thimble tubes for the PWR incore instrumentation, over most of their length, serve as a portion of the reactor coolant system (RCS) pressure boundary. Thus, the wear of the thimble tubes results in degradation of the RCS pressure boundary and can also create a potentially non-isolable leak of the reactor coolant. Furthermore, thimble tube thinning could result in multiple thimble tube failures beyond a facility's design basis during flux mapping operations or a transient event.

A thinning of thimble tubes occurred as a result of flow induced vibration. Thimble tube wear had generally been detected at locations associated with geometric discontinuities or area changes along the flow path (such as areas near the lower core plate, the core support forging, the lower tie plate, the upper tie plate, and the vessel penetration). The thimbles are contained and supported over most of the path from the seal table through the fuel assembly. The thimble is unsupported for a short distance from the top of the lower core support plate to the point of entry into the fuel assembly lower nozzle. This unsupported area is most often subject to flow-induced vibration wear.

Safety significance

Excessive wear of the thimble tubes results in degradation of the reactor coolant system pressure boundary and could lead to a non-isolable small loss of coolant accident. Multiple thimble tube failures could result in a beyond design basis accident situation.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

This issue appeared in France in 1985 is under control using:

- a reinforced surveillance;
- installation of anti-vibratory devices;
- replacing the thimble tubes of insufficient thickness, with new tubes of increased thickness.

Germany

The in-core instrumentation of the Siemens/KWU design operated in Germany is completely different (e.g. 6 or 8 pressure-tight vessel head penetrations). Therefore thimble tube thinning is not an issue in Germany.

Japan

In order to cope with this issue, a practice of conducting ECT examination, and method to remove or replace it depending on the extent of wearing had been established. When leakage occurs, it can be stopped by closing the isolation valves.

Korea, Republic of

Periodic inspection of the thimble tubes is supposed to perform while plant outage and to determine the extents of wear and whether replacement is necessary.

Spain

Thimble tubes are being inspected in the Spanish PWRs using eddy current techniques. The acceptance criteria for these inspections is a maximum thickness loss of 60%. The implementation of a procedure to isolate the leaking thimble tubes using automatic valves is being considered.

USA

In US PWR plants, periodic inspection of the thimble tubes is necessary to determine the extent of wear and whether replacement is necessary. The licensee for the Salem plant had laboratory tests conducted to model the wear phenomenon, and then developed a wear resistant flux thimble tube design. After three cycles of operation, there was no significant wear of the new design.

ADDITIONAL SOURCES:

- Title 10, Code of Federal Regulations, part 50, 10CFR50 Appendix A, General Design Criterion 14.
- Licensee Event Report 50-272/81-028.
- Westinghouse WCAP-12866, "BMI flux thimble wear," Jan. 1991.
- USNRC Bulletin 89-09.
- USNRC Information Notice 87-44.
- USNRC Information Notice 87-44, Supplement 1.

ISSUE TITLE: Inconel-600 cracking (CI 5)

ISSUE CLARIFICATION:

Description of issue

Primary water stress corrosion cracking (PWSCC) of Inconel-600 has been detected in the control rod drive mechanism (CRDM) penetration nozzles, pressurizer heater thermal sleeves and instrument nozzles at several plants in the USA and Europe. Examinations showed that the cracks were principally axially oriented, although in some cases there were circumferential cracks. The safety implication of an axial crack is not considered a significant threat to the structural integrity and most likely will result in a small leak. Circumferential cracks have been also observed. The difference in the cracking morphology has been attributed to the different mechanical working (rolling vs. reaming) being performed on these nozzles and thermal sleeves. Circumferential cracking poses a more serious safety concern because if it were to go undetected, it could lead to a sudden structural failure of a component rather than to a limited leak.

Safety significance

The sudden failure of a penetration nozzle could lead to a small loss of coolant accident and, in the case of an RPV head penetration, to a control rod ejection, which is a sequence within the DBA envelope.

Source of issue (check as appropriate)

- ____xx_____ operational experience
- ___________ deviation from current standards and practices
- ___________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

At the end of September 1996:

- 18 vessel heads on 54 were replaced with new CRDM penetrations in Inconel-690.
- 11 on the remaining 36 have no cracks.
- On the remaining 25 vessel heads, 85 CRDM penetrations are affected by axially oriented cracks, (3.5% of remaining CRDM penetrations in Inconel-600). Generally crack lengths are lower than 3 mm, and, during the last control, only 3 vessel heads were found with crack lengths higher than 5 mm (maximum: 8.8 mm).

Finally, 7 vessel heads were replaced in 1996 and it is expected to maintain replacements at the same annual rate.

The maintenance policy performed from 1994 has permitted to keep under control the degradation process (about constant from 1993).

For 1300 MW NPP, strategy will depend on fuel management (cycle duration) and on the treatment of CRDM functional problems.

Germany

The use of Inconel-600 alloy in German NPPs is very limited compared to other NPPs. Inconel-600 for vessel head penetration nozzles is only used in two German PWRs. The peripherical nozzles were inspected by eddy current tests and no cracks were found. This is explained with different geometric and manufacturing conditions (lower wall thickness and post-weld heat treatment) for one plant. The other plant has only been operated for a short time.

Japan

As for Inconel-600 alloy, since PWSCC has been observed in steam generator tubes, utilities have performed evaluation of possibility of occurring PWSCC on the RCS boundary material, and an inspection has been performed on reactor vessel head penetrations which were considered as the most severe conditions. No abnormalities were observed, however, considering the findings on the vessel heads in foreign countries, the possibility of it occurring in Japan cannot be denied in a long term, replacement to the vessel head with Inconel-690 penetration and/or reduction of upper head temperature are being undertaken.

Inconel-600 alloy is not used for pressurizer heater nozzles.

Korea, Republic of

Most primary system penetration nozzles of Inconel-600 intended to operate at relatively high temperature in the piping, reactor vessel, pressurizer, and SGs had been replaced by Inconel-690. The replacement of each nozzle was made by using the acceptance criteria curve based on yield strength and RCS temperature.

In case of Yonggwang Units 3 and 4, evaluation of susceptibility of the CEDM nozzles to PWSCC was performed. The yield strength of all of the CEDM nozzles is less than 41 ksi which is unlikely to crack during a 40 calendar year period using conservative assumptions if the temperature is below approximately 607^{o}F. However, the coolant temperature at the reactor vessel head remains nearly equal to the core exit temperature of 622^{o}F. Therefore, a modification of reactor internals alignment keys, which permit cool water to flow into the upper region of the vessel head, had been made, this modification reduces the highest predicted local temperature in the CEDM nozzle-head intersection region.

Russian Federation

Inconel-600 is not used in WWER NPPs, however, vessel head penetration cracking was found at prototype plant due to thermal cycling (NVV 1,2; stainless steel). New design was developed and NVV 1,2 modified accordingly. The modified design which includes flange connections was then used for all other WWER plants (see also IC 12).

Spain

PWSCC of Inconel-600 is not a new phenomenon. However, little special attention has been given in the past to the inspection for PWSCC in Inconel-600 applications other than that associated with the steam generator tubes. Licensees are reviewing their Inconel-600 applications in the primary coolant pressure boundary, and, when necessary, implementing augmented inspection programmes.

All the Spanish PWR plants with Inconel-600 adaptors in the CRDM penetrations (6 plants) have been inspected. No cracks have been detected in these components except in one of them, where a leaking axial crack was detected in a visual inspection and as a result of a complete inspection, using ECT and UT techniques, an extensive cracking was found. The root cause of this cracking was a chemical contamination of the primary system with resins of a demineralizer as a consequence of the retaining grid rupture. The cracking mechanism was not PWSCC but intergranular attack (IGA) oriented by the fabrication residual stresses in the Inconel-600 base material of the adaptor, which was highly sensitized because of the tube fabrication process. 16 spare penetrations (with no CRDM) were highly affected and most of them had cracks with a circumferential orientation following the welding line. None of these cracks was through-wall.

The reactor vessel head of this plant was temporarily repaired by plugging all the 16 spare penetrations and grinding the active penetrations, which had much smaller defects (4 penetrations), with the EDM process. Recently this RPV head has been replaced by a new one with Inconel-690 penetrations.

Other two RPV heads have also been replaced recently.

USA

An action plan was implemented by the NRC staff in 1991 to address PWSCC of Alloy 600 VHPs at all US PWRS. As explained more fully below, this action plan included a review of the PWR Owner's Groups' safety assessments, the development of VHP mock-ups by the Electric Power Research Institute (EPRI), the qualification of inspectors on the VHP mock-ups by EPRI, the review of proposed generic acceptance criteria from the Nuclear Utility Management and Resource Council (NUMARC, now the Nuclear Energy Institute (NEI)), and VHP inspections. As part of this action plan, the NRC staff met with the Westinghouse Owner's Group (WOG), the Combustion Engineering Owner's Group (CEOG), and the Babcock & Wilcox Owner's Group (B&WOG), to discuss their respective programmes for investigating PWSCC of Alloy 600 and to assess the possibility of cracking of VHPs in their respective plants since all of the plants have Alloy 600 VHPS, Subsequently, the NRC staff asked NUMARC to coordinate future industry actions since the issue was applicable to all PWRS. Each of the PWR Owner's Groups submitted safety assessments through NUMARC to the NRC on this issue. After reviewing the industry's safety assessments and examining the overseas inspection findings, the NRC staff concluded, in a safety evaluation dated November 19, 1993, that VHP cracking was not an immediate safety concern. The bases for this conclusion were that if PWSCC occurred at VHPS: (1) the cracks would be predominately axial in orientation; (2) the cracks would result in detectable leakage before catastrophic failure; and, (3) the leakage would be detected during visual examinations performed as part of surveillance walkdowns before significant damage would occur to the reactor vessel head. In addition, the NRC staff had concerns related to unnecessary occupational radiation exposures associated with eddy current or other forms of nondestructive examinations, if done manually.

The first US inspection of VHPs took place in the spring of 1994, at the Point Beach Nuclear Generating Station, where no indications were uncovered in any of the 49 CRDM penetrations. The eddy current inspection at the Oconee Nuclear Generating Station, in the fall of 1994, revealed 20 indications in one penetration. Ultrasonic testing (UT) did not reveal the depth of these indications because they were shallow. UT cannot accurately size defects that are less than one mil deep (0.03 mm). These indications may be associated with the original fabrication and may not grow; however, they will be reexamined during the next refueling outage. Palisades conducted a limited examination of eight in-core instrumentation penetrations and found no cracking. An examination of the CRDM penetrations at D. C. Cook, in the fall of 1994, revealed three clustered indications in one penetration. The indications were 46 mm, 16 mm, and 6-8 mm in length and the deepest flaw was 6.8 mm deep. The tip of the 46-mm flaw was just below the J-groove weld. Virginia Electric and Power Company (VEPCO) inspected North Anna Unit 1 during its spring 1996 refueling outage. Some high stress areas (e.g., upper and lower hillsides) were examined on each outer ring CRDM penetrations and no indications were observed using eddy current testing.

The USNRC has now issued Generic Letter 97-01 "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations", dated 03 April 1997, to address this issue. As stated in the Generic Letter, the NRC staff has concluded that vessel head penetration cracking does not pose an immediate or near-term safety concern. In the long term, however, degradation of the control rod drive mechanisms and other vessel head penetrations is an important safety consideration that warrants further evaluation. The Generic Letter asks what if any periodic inspections US licensees are are performing. It also asks the licensee's bases for concluding acceptability of plans to not perform inspections.

ADDITIONAL SOURCES:

- CSN/IS/26/94, 27/94 and 28/95, Report to the Congress and Senate, Nuclear Safety Council, Spain.
- USNRC GL 97-01, Degradation of control rod drive mechanism nozzle and other vessel closure head penetrations, 04/03/97.
- USNRC IN 96-11, Ingress of demineralized resins increased potential for stress corrosion cracking of control rod drive mechanism penetrations.
- USNRC IN-90-10, Primary water stress corrosion cracking (PWSCC) of Inconel-600.
- USNRC IN-89-65, Potential for SCC in SG tube plugs.

ISSUE TITLE: Steam generator collector integrity (CI 6) (WWER)

ISSUE CLARIFICATION:

Description of issue

WWER NPPs are equipped with horizontal steam generators (SG). Two cylindrical collectors (hot leg and cold leg collector) form part of the boundary between the primary and secondary circuit in a SG. Tubes are attached to these collectors. The WWER-440 SG collectors are made of austenitic stainless steel, the WWER-1000 SG collectors are made of clad carbon steel. There is a bolted cover on the top of the collector (primary pressure boundary) and on the shell above the collector cover (secondary side).

In addition to the collector cover bolted point failure at Rovno (caused mainly by operational practices used), cracks were revealed in the collector threaded holes at some other plants. The degradation mechanism was identified as stress corrosion cracking. The damage is possibly associated with both design (cover seals, requiring relatively high load) and operational aspects (maintenance, ISI, operation itself).

In the period from late 1986 to 1991, cracks have been revealed in the ligaments between tube holes in the collectors of 24 steam generators at 6 NPPs of WWER-1000 units. The operating time before detection of this damage has varied between 7000 and 60,000 hours for the affected steam generators. These cracks have mainly been detected in the cold collectors. Some indications have also been reported for the hot collector. The damage observed is related to deficiencies in the design, manufacturing technology, material quality and operational aspects.

The integrity of the second barrier is seriously degraded due to a combination of design, manufacturing and operational aspects. The related monitoring is not sufficient. Compensatory measures have been established, their full effectiveness has to be still demonstrated.

A minimum leakage size of 100 mm equivalent diameter has been calculated by the designer, considering all possible collector failure models.

Safety significance

A large primary to secondary leak due to the steam generator collector damage is not considered a DBA scenario, although loss of integrity of collectors has been observed at operating plants. A large primary to secondary leak can impair safety functions in case the BRU-A valve fails to close. This could lead to a bypass of containment and leakage to the environment via a BRU-A valve. The long-term cooling of the core may be endangered.

Source of issue (check as appropriate)

- ____xx_____ operational
- ___________ deviation from current standards and practices
- ___________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria, Czech Republic, Russian Federation, Ukraine

The issue is addressed by the designers and interim compensatory measures have been taken:

(1) – additional tube rolling;
– low temperature heat treatment to release the residual stresses in the collector;
– collector head release by cutting off the contact part of the flange with SG shell;

- improvement of feedwater and blowdown system to mitigate the environmental and temperature effects on the cracked collector; and
- regular inspections on the collector cracking based on optimized NDT.

(2) A new steam generator design has been developed with a collector made from stainless steel and reduced number of tubes mounted by hydraulic tube expansion.

(3) Improvement of the water chemistry and its control will be implemented in order to prevent degradation.

(4) A programme to monitor secondary circuit components degradation will be developed and implemented along with appropriate NDT methods.

(5) The issue has been discussed separately in detail. The measures proposed are summarized in detail in the IAEA report on "Steam Generator Collector Integrity of WWER-1000 Reactors" published in 1993.

(6) Emergency procedures have been developed for this type of event and this event has been incorporated into full scope simulators.

Bulgaria

Kozloduy NPP: A study of the steam generator collector rupture is proposed by the plant. The main assumptions used to perform the study are:

- The most severe case seems to be the one with loss of external power. This assumption will be considered;
- The study will be performed considering the single failure criteria; and
- The study will be performed till a safe state.

Czech Republic

Temelín NPP: The accident was analysed by Westinghouse as a part of the Temelín NPP PSAR. Two limiting analyses, a core integrity analysis and a radiological consequences analysis were performed for this accident. A break area of 100 mm equivalent diameter was assumed and function of the SDA valve was modeled. Core integrity analysis was performed with conservative low SI flowrate, conservative beginning of accident system parameters and with operator actions neglected. It was shown that the core never uncovers and 10CFR50.46 limits are met.

The radiological consequence analysis was performed for conservative mass and energy releases to the environment through SDA(BRU-A). The analysis assumes that no operator actions are taken to terminate SI flow and conservatively high total mass and activity released to the environment in 2 hours was assumed. The off-site doses calculated are within the dose acceptance criteria.

After the SG overfill has occurred, the flow out of the SDA valve is approximately equal to the failed seal ring flowrate and it is approximately equal to SI flowrate. It was proved that there is sufficient time available for operators to take action to depressurize the primary system, terminate SI and thus avoid the depletion of the sump volume.

Ukraine

Rovno NPP: The plant plans to re-evaluate the SG collector rupture (100 mm), considering that:

- the time duration of the simulation will be increased up to reaching the safe state or up to stabilization of the parameters;
- the single failure criteria will be applied (worst case to be defined);
- adherence to the acceptance criteria will have to be demonstrated;
- a possible collapse of the steam line will be considered;

- a calculation of both the core behaviour and the radiological consequences will be performed; and
- the recommendations made by Riskaudit during their review process for Unit 3 will be considered.

Additionally, in view of the results obtained, the accidental procedure will be revised.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety aspects of WWER-1000 reactors, IAEA-CT-0670, 1992.
- Steam generator collector integrity of WWER-1000 reactors, IAEA, WWER-RD-057 (SC-076), 1993.
- Analysis of WWER-1000 safety enhancement measures, IAEA, WWER-RD-080 (former WWER-SC-092), 1994.
- Safety review mission to Zaporozhe NPP, Final Report, IAEA, WWER-RD-064, 1994.
- Modernization Programme, Kozloduy NPP Units 5 and 6 January 1995, Version 0.

ISSUE TITLE: SG tubes integrity (CI 7)

ISSUE CLARIFICATION:

Description of issue

The steam generator (SG) tubes in many PWR plants of different vendors have shown several degradation phenomena during their operation, and these have weakened their function as the primary to secondary barrier. This degradation has been discovered by in-service inspection programmes and by the operational experience of many NPPs in which leaks ranging from minor leaks to complete single tube ruptures have been detected. So far no multiple tube rupture has occurred.

The main degradation mechanisms identified are stress corrosion cracking (SCC) at different locations of tubes, intergranular attack (IGA), "denting" and "fretting."

A SG tube rupture would lead to a primary to secondary leakage and could affect the cooling of the fuel and the confinement of radioactive materials.

Safety significance

A single tube rupture is within the DBA envelope. Operating experience have demonstrated the capability of the plants to control the situation, with only a minor radiological impact.

Multiple tube rupture is a BDBA scenario; it could lead to the containment bypass, loss of primary water inventory in the long term and could affect the cooling of the fuel.

Source of issue (check as appropriate)

- ____xx_____ operational experience
- ___________ deviation from current standards and practices
- ___________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

Several mechanisms can threaten steam generator tube integrity. For example, more than 1300 SG tubes were plugged in 1996 (more than 1500 in 1995), main causes are:

- for 300 pluggings: different types of corrosion (from primary and secondary circuits);
- for 250 pluggings: tube support plates degradation;
- for 540 pluggings: abnormal positioning of anti-vibration bars (preventive maintenance to avoid a potential risk of rupture).

Other causes are erosion-corrosion (Gravelines 2/3/4 at the last plate level).

Inconel-600 used for SG tubes is sensitive to stress corrosion of primary coolant (200 tubes have been plugged in 1995 and 450 in 1994 for all of 54 units) but the main concern is the corrosion by secondary circuit, which imply a very attentive control of the water chemistry.

A policy of SG replacement is in progress depending on the rate of plugging and the lifetime of the plants, SG of Gravelines-2 have been replaced in 1996, (SG replacements on 6 units have been carried out in France at the end of 1996).

Germany

Contrary to worldwide steam generator tube degradation problems, steam generators of the Siemens/KWU design operated in Germany have proved by operating experience their low sensitivity to tube corrosion or any other steam-generator-tubes-related problems.

To avoid primary water stress corrosion cracking (PWSCC), optimized Incoloy 800 is used as steam generator tubing material. The engineering design of the internals avoids concentration of impurities by avoiding crevices and flow stagnation zones as well as high stresses in the tubes and minimize tube vibrations and consequently avoids different well-known types of steam generator tube corrosion such as intergranular attack (IGA), intergranular stress corrosion cracking (IGSCC) and other phenomena, such as flow-induced vibration and wear.

In the seventies, phosphate wastage occurred in the steam generator tubes as the only degradation mechanism of significant experience. To avoid this, the German utilities replaced their condenser tubes with new ones made of stainless steel or titanium, thereby creating suitable conditions for changing the conditioning to "High" All Volatile Treatment (AVT, pH level > 9.8). Since 1985, wastage corrosion has no longer been an issue either for steam generators in German PWR plants.

India

Extensive tube failures occurred in both SGs of both Indian BWRs. The SGs have been isolated and the units are being operated in single cycle at reduced load of 160 MWe since 1984.

Japan

It has been the Japanese practice that ECT examination is applied on all steam generator tubes during each annual inspection to confirm their integrity. As for those tubes for which significant indications are observed, either plugging or sleeve repair is performed. Most of steam generators having degradation of the tubes (tube material: MA 600 alloy) were replaced with newly designed steam generators (tube material: 690TT) or are in the process of replacement. Thus, the possibility of tube break is considered to be extremely low.

In Japan, operation of the plant with primary to secondary leak is no admitted.

N-16 monitors, etc. are installed and continuous monitoring allow them to shut down the plant in the early stage of leakage, if it happens.

Korea, Republic of

SG tube design complies with the following recommendations to the extent that they are applicable to:

- Prevention and detection of loose parts.
- SG tube in-service inspection (ISI).
- Secondary water chemistry and impurity control.
- Use of Nitrogen-16 and area radiation monitors.
- Primary coolant iodine activity limit.

Recently, regulatory body requested that Inconel-690 be used in future plants (Ulchin 5 and 6) for SG tubes to provide increased resistance to corrosion.

Russian Federation

At present the devices have been developed for automatic control of volumetric activity of live steam and blow down water of SG - UDPG-04R and UDZHG-20R, respectively.

UDPG-04R is intended for measurement of volumetric activity of gamma-radiation of radionuclides in live steam when the device is located in the close vicinity to the steam line under control. Recording of gamma-quanta is performed with the use of scintillation detection unit. The device may operate in the channel of an automated information and measuring system with the following kinds of measurements:

- operative determination of volumetric activity of steam (time of averaging the information is 100s);
- determination of the moment at which the prescribed thresholds for volumetric are exceeded, and the generation of the corresponding signal.

UDZHG-20R is intended for measurement of volumetric activity of gamma-radiating nuclides in fluid and operates with the standard electronic and physical equipment. In the working position the measuring box of the device is connected to the circuit under control with help of nozzles forming the bypass line. Principle of operation of the device and possibility of its introduction into the channel of information and measurement system are similar, as a whole, to UDPG-04R.

To introduce the system of coolant leak control with the use of devices UDPG-04R and UDZHG-20R the following main measures are required:

(a) trial run of the facilities under actual conditions with possible optimization of the methods of leak monitoring including: examination of leaky SGs, determination of a character of entering and distribution of radionuclides in SG boiler water - at the first stage; development and verification of the model of distribution of radionuclides in boiler water in the course of leaky SG operation for the cases of primary occurrence of leak and operation of SG under stationary conditions - at the second stage; revision of instructions on determination of SG leak values - at the third stage;
(b) development of the project of connection of the monitoring systems and equipment;
(c) development of the covering documents for the system.

Spain

All the PWR plants with potentially affected SGs have had increased in-service inspection of the tubes, with a scope of 100% tubes and full-length inspection. Additionally, in the plants with relevant degradation, an improved primary to secondary leak detection system, using Nitrogen-16 as tracer, was installed and more restrictive limits were enforced in the Technical Specifications.

In 4 of the 7 PWR plants in Spain, it has been decided to replace the SG because of this problem, 3 of which has been actually replaced to date. The improved leak detection system will be maintained.

USA

All PWR plants are periodically inspected for steam generator tube degradation with techniques and frequencies adjusted to the degradation phenomena observed in the affected unit and in units of similar design. A number of PWRs in the USA have replaced steam generators. Regulatory efforts are now underway to develop rules and guidelines which better recognize the wide variety of degradation phenomena.

Countries operating WWER NPPs

State of the art in-service techniques are being introduced along with the development of justified plugging criteria and leakage limits. Monitoring systems are being improved.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- MOHT - OTJIG - EDF generic reference programme for the modernization of WWER-1000/320, Revision 6 (February 1997).
- *CSN/IS/29/95 and 30/96, Report to the Congress and Senate, Nuclear Safety Council*, Spain.
- NUREG-0844.
- USNRC Bulletin 89-01 (May-89).
  - Supplement 1 (Nov-90).
  - Supplement 2 (Jun-91).
- USNRC BL-88-02, Rapidly propagating cracks in steam generator tubes.
- USNRC GL-97-05, Steam generator tube inspection techniques.
- USNRC GL-95-05, Voltage based repair criteria.
- USNRC GL-95-03, Circumferential cracking of steam generator tubes.
- USNRC Generic Letter 85-02.
- Recent USNRC Information Notices 96-38, 97-26, 97-88.
- USNRC IN-95-40, Supplemental information to Generic Letter 95-03, "Circumferential cracking of steam generator tubes."
- USNRC IN-94-88, In-service inspection deficiencies result in severely degraded steam generator tubes.
- USNRC IN-94-87, Unanticipated crack in a particular heat of Alloy 600 used for Westinghouse.
- USNRC IN-94-62, Operational experience on steam generator tube leaks and tube ruptures.
- USNRC IN-94-43, Determination of primary to secondary steam generator leak rate.
- USNRC IN-94-05, Potential failure of SG with kinetically welded sleeves.
- USNRC IN-93-56, Weakness in emergency operating procedures found as a result of steam generator tube rupture.
- USNRC IN-92-80, Operation with steam generator tubes seriously degraded.
- USNRC IN-91-67, Problems with the reliable detection of intergranular attack (iga) of steam generator tubing.
- USNRC IN-91-43, Recent incidents involving rapid increases in primary to secondary leak rate.
- USNRC IN-90-49, Stress corrosion cracking in PWR steam generator tubes.
- USNRC IN-89-65, Potential SCC in SG tube plugs.
- USNRC IN-89-33, Potential failure of Westinghouse steam generator tube mechanical plugs.

ISSUE TITLE: Pipe cracks and feedwater nozzle cracking in BWRs (CI 8)

ISSUE CLARIFICATION:

Description of issue

Pipe cracking has occurred in the heat affected zones of welds in primary system piping in BWRs since mid-1960. These cracks have occurred mainly in Type 304 stainless steel which is the type used in most operating BWRs. The major problem is recognized to be IGSCC of austenitic stainless steel components that have been made susceptible to this failure by being "sensitized", either by post-weld heat treatment or by sensitization of a narrow heat affected zone near welds.

"Safe ends" (short transition pieces between vessel nozzles and the piping) that have been highly sensitized by furnace heat treatment while attached to vessels during fabrication were very early (late 1960's) found to be susceptible to IGSCC. Because of this, the US AEC took the position in 1969 that furnace-sensitized safe ends should not be used on new applications. Most of the furnace-sensitized safe ends in older plants have been removed or clad with a protective material, and there are only a few BWRs that still have furnace-sensitized safe ends in use. Most of these, however, are in smaller diameter lines.

Earlier reported cracks (prior to 1975) occurred primarily in 4-inch diameter recirculation loop bypass lines and in 10-inch diameter core spray lines. Cracking is most often detected during ISI using UT techniques. Some piping cracks have been discovered as a result of primary coolant leaks.

Inspections of operating BWRs revealed cracks in the feedwater nozzles of reactor vessels. Most of these BWRs contained 4 nozzles with diameters ranging from 10 in. to 12 in. Although most cracks ranged from 1/2 in. to 3/4 in. in depth (including cladding), one crack penetrated the cladding into the base metal for a total depth of approximately 1.5 in. It was determined that cracking was due to high-cycle fatigue caused by fluctuations in water temperature within the vessel in the nozzle region. These fluctuations occurred during periods of low feedwater temperature when flow is unsteady and intermittent. Once initiated, the cracks enlarged from high pressure and thermal cycling associated with startups and shutdowns.

Safety Significance

Cracking of piping and feedwater nozzles of the BWR threatens the integrity of the reactor pressure boundary.

Source of issue (check as appropriate)
- ___xx______ operational experience
- ___________ deviation from current standards and practices
- ___________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Germany

Within the framework of in-service inspections in 1991, cracks were found in the recirculation system (made of stabilized austenitic stainless steel) of a German BWR plant. This caused extensive in-service inspections in all other German plants. A total of about 3000 welds were inspected in BWRs by NDT and 88 defected welds were observed, especially in the pipework of the reactor water cleanup system and the water supply system for the bearings of the internal circulation pumps. The cracking mechanism was classified as intergranular stress corrosion cracking (IGSCC) as a result of partial sensitization (Cr depletion at grain boundaries) during welding and the influence of high-temperature water containing O_2 and H_2O_2 during operation.

The pump bearing systems have been modified, therefore a large part of affected piping was obsolete. Furthermore, replacement of austenitic pipework with a reduced number of welds was performed using niobium-stabilized stainless steel of optimized chemical composition (low carbon content, higher stabilization ratio) and improved (small gap) welding methods. The in-service inspection programmes were modified regarding the experience obtained.

India

Extensive cracks due to IGSC have been observed in the piping except in reactor main recirculation loop. Majority of the piping has been replaced with 316 LN piping.

FW nozzle being regularly inspected in each refueling outage and no cracks were found.

Spain

In a Spanish BWR plant started in 1972, it was detected during the 1983 refueling outage stress corrosion cracks in the recirculation lines field weldings. A temporary repair of the affected welds was made with a "weld overlay" and in the next refueling most of the recirculation lines were replaced. The original material was SS-304 and the new one is SS-316NG. As an additional measure to protect components of corrosion cracking on Hydrogen inspection system was installed thus reducing oxygen content and the electrochemical potential of the coolant. The same plant which had presented leaks through the thermal sleeve had to be replaced in 1994 for new feedwater nozzles with double thermal sleeves and safe ends of a different geometry.

USA

The issues on pipe cracks and feedwater nozzle cracking were resolved with the issuance of NUREG-0313 and NUREG-0619, respectively.

ADDITIONAL SOURCES:

- CSN/IS/5/83 and 22/92, Report to the Congress and Senate, Nuclear Safety Council, Spain.
- NUREG-0933, Safety Issues A-42, A-10, 86, 119-4.
- NUREG-0313, Rev. 2, January 1988, "Technical report on material selection and processing guidelines for BWR coolant pressure boundary piping."
- NUREG-0619, "BWR feedwater nozzle and control rod drive return line nozzle cracking", November 1980.
- USNRC GL 88-01, "NRC position on IGSCC in BWR austenitic stainless steel piping."
- USNRC GL 84-11, "Inspection of BWR stainless steel piping."

ISSUE TITLE: Bolting degradation or bolting failures in the primary circuit (CI 9)

ISSUE CLARIFICATION:

Description of issue

Various failure and degradation mechanism have been identified that adversely impact the integrity of bolts used in safety related applications and in applications important to safety. Depending on the nature of the degradation mechanism a potential for common mode failures exists for same system or redundant system components. In addition, leaks from flanged joints represent a significant part of total number of leaks in primary circuit.

Safety significance

Bolt degradation or failure could lead to the degradation or failure of safety related systems or systems important to safety and to loss of coolant accidents.

Source of issue (check as appropriate)

- ____xx_____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

In France some cracks have been observed in the bolting of vessel internals at Bugey-2, where cracks were found on 64 screws (out of 960), some cracks have also been discovered at Fessenheim (first 900 MW PWR subseries). Controls for standard PWRs, 900 and 1300 MW, are foreseen in 1997, but it seems that the better design of bolting (in particular: better cooling) prevents from having that type of degradation.

Germany

In the eighties, systematic failures were detected and investigated at outer and inner core-shroud bolts in German PWR plants. The damage can be described as pure intergranular stress corrosion cracking (IGSCC).

The repair concept involved the discontinuation of the use of Inconel X-750 for the core-shroud bolts, these being replaced by the stabilized austenitic stainless steel material, strain-hardened by light cold working. Up to now, the experience is favourable.

India

Reactor head studs and reactor recirculation pump casing studs are inspected ultrasonically every refueling outage and their threads are examined visually. No degradation has been observed.

Korea, Republic of

Proven bolting designs, materials, and fabrication techniques are employed.

Further, ISI will meet the requirements of ASME Code for the RCPB and its support bolting.

All vessel bolting material receives ultrasonic and magnetic-particle examination during the manufacturing.

USA

The US Nuclear Regulatory Commission (NRC) addressed the subject in the resolution of Generic Safety Issue (GSI) 29, "Bolting Degradation or Failure in Nuclear Power Plants." The scope of GSI-29 included all safety-related bolts, studs, embodiments, machine/cap screws, other special threaded fasteners, and all their associated nuts and washers. The NRC resolved this issue without developing any new requirements, based on licensees continuing to implement actions taken in response to previous NRC guidance and the industry's initiatives in this area. The NRC's resolution of this issue is documented in Generic Letter 91-017, "Generic Safety Issue 29, Bolting Degradation or Failure in Nuclear Power Plants", October 17, 1991. The GSI-29 resolution encompassed a number of industry initiatives and previous NRC actions. The bases for resolution are further documented in NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," which was published by NRC in June 1990. (Concerns regarding reactor vessel closure studs are being addressed under GSI-109, "Reactor Vessel Closure Failure," and were not considered under GSI-29.)

Since establishing GSI-29 in 1982, the staff has issued a number of bulletins, generic letters, and information notices on bolting events judged to be safety-significant (See NUREG-1339). The bulletins and some of the generic letters required both one-time actions and continuing programmes. These actions and continuing programmes were considered by the staff in resolving GSI-29.

Countries operating WWER-440/213 NPPs

A collector cover lift up occurred at the Rovno WWER-440/213 plant during operation, caused by stress corrosion cracking in the bolted joint. The possible causes for the failures observed were improper water chemistry, improper chemical composition of bolt lubricant used, and maintenance procedures. The in-service inspection used failed to detect the degradation in a timely manner. It is proposed to apply eddy-current control of bolt holes in the collector flanges. It is proposed to change the technology of fastening the screws on the primary collector covers in such a way as to guarantee good positioning and predetermined forces acting on each bolt. Changing the design of the SG collector head to limit the possible lift-off of the cover (being prepared for Mochovce NPP and for new NPPs). Changes of water swell level and moisture blow-ups should be limited so that they should not reach the region of bolts fastening the collector cover. This helps to prevent damage to the bolts

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- NUREG-1339, Resolution of Generic Safety Issue 29: Bolting degradation or failure in nuclear power plants, US Nuclear Regulatory Commission (June 1990).

ISSUE TITLE: Heavy components support stability (CI 10)

ISSUE CLARIFICATION:

Description of issue

During the NRC licensing review for the North Anna Units 1 and 2 PWRs, questions were raised regarding the potential for low fracture toughness of the steam generator and reactor coolant pump supports. Lamellar tearing of the support materials also was of concern. Two different steel specifications (ASTM A36 and ASTM A572) covered most of the material used for these supports. Toughness tests, not originally specified and not in the relevant ASTM specifications, were made at various temperatures. The toughness of the A36 steel was found to be adequate, but the toughness of the A572 steel was relatively low at 80 degree F. Because other PWRs use similar materials and designs, this concern regarding the supports may be applicable for other PWR plants. The NRC has studied this issue under a Task Action Plan A-12. Insufficient toughness coinciding with a low operating temperature, the presence of flaws, and non-redundancy of critical support members could result in failure of the support structure under postulated accident conditions, especially loss-of-coolant accidents and earthquakes.

Reactor pressure vessel supports are subject to neutron irradiation at low temperature during plant operation. The neutron flux is lower than that at RPV, but the low irradiation temperature could result in higher embrittlement rate. Some reactor pressure vessel supports were fabricated without special requirements on fracture and radiation resistance. Steel surveillance specimens that had been irradiated in an environment believed to be similar to that in operating reactor cavities exhibited a greater than expected shift (increase) in the nil-ductility-transition temperature. This indicated that there was a potential for excessive embrittlement of reactor vessel supports. Moreover, the RPV supports are in many cases difficult to access or inaccessible for in-service inspection.

This issue is applicable for some specific RPV support designs only.

Safety significance

The loss of integrity of the supports of heavy components could further aggravate the plant condition during a design basis accident. A failure of RPV supports would also challenge the integrity of the reactor coolant system and the functionality of safety systems.

Source of issue (check as appropriate)

- ____xx_____ operational experience
- ____xx_____ deviation from current standards and practices
- ____xx_____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Germany

The RPV in German PWRs is supported between the main piping by separate lugs. At this level, fluence is very low and no significant embrittlement is expected.

Japan

The type of the reactor vessel support adopted in Japanese PWR is "nozzle support on the concrete shield" type. Evaluation in Japan indicated that there should be no problem of embrittlement of reactor vessel support adopted in Japan.

USA

A reassessment of the surveillance data and its relevance to reactor vessel supports showed that the embrittlement of the test specimens was not excessive and that the measured gamma flux was several decades greater than that in an operating light water reactor cavity. As a result, there were no serious safety concerns.

In the case of North Anna Units 1 and 2, the licensee raised the temperature of the A572 beams in the steam generator supports to a minimum temperature of 225 degree F, prior to reactor coolant system pressurization to levels above 1,000 psig. Auxiliary electrical heat was supplied as necessary to supplement the heat derived from the rector coolant loop to obtain the required operating temperature of the support materials.

The solution to this issue was made available in October 1983 with the publication of NUREG-0577, Revision 1. This resolution contains no backfit requirements.

Countries operating WWER-440/230 NPPs

Some limited evaluation of the annular water tank, which serves as a biological shield and vessel support was performed.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- NUREG/CR-6117, Neutron spectra at different high flux isotope reactor (HFIR) pressure vessel locations, Oak Ridge National Laboratory, December, 1993.
- NUREG/CR-5644, Consequence evaluation of radiation embrittlement of Trojan reactor pressure vessel supports, Oak Ridge National Laboratory, October, 1990.
- NUREG/CR-5320, Impact of radiation embrittlement on integrity of pressure vessel supports for two PWR plants, Oak Ridge National Laboratory, January, 1989.
- NUREG-1509, Radiation effects on reactor pressure vessel supports, US Nuclear Regulatory Commission, to be issued.
- NUREG-0577, Revision 1, Potential for low fracture toughness and lamellar tearing in pwr steam generator and reactor coolant pump supports, US Nuclear Regulatory Commission (October 1983).

ISSUE TITLE: Cast stainless steel cracking (CI 11)

ISSUE CLARIFICATION:

Description of issue

Cracks have been discovered in cast primary pump bodies, valves and elbows in main circulation loops. Dye penetrant testing revealed a number of cracks. The cracks were found both on the inside and on the outside of the casings. Moreover, the duplex austenitic-ferritic structure of cast steel pipes used in some NPPs is a potential cause of an embrittlement due to the thermal ageing effect on the ferritic part. It is difficult to perform reliable volumetric non-destructive testing by ultrasonic techniques of the cast material.

Safety significance

Undetected cracks could lead to a loss of coolant accident or to a functional failure of the affected component, challenging the core cooling.

Source of issue (check as appropriate)
- ____xx____ operational experience
- __________ duration from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

A surveillance programme has been established to follow the behaviour of the pipes made of duplex cast steel alloy.

Germany

No duplex cast stainless steels, which are sensitive to thermal ageing, are used in German NPPs. Besides, the mechanical properties of steels which are used in German NPPs are investigated by some destructive testing. Time-dependent influences on the mechanical properties even after 100.000 hours of service have not been detected up to now. The properties are in most cases within a 10% margin of the original properties.

India

Inside and outside surfaces of one of the reactor recirculation pump body was examined by VT and PT in 1996 for the Indian BWRs. No cracks were observed.

Japan

Visual and other inspections (liquid penetration inspection in some case) are performed to pump casings during annual inspections. No cracks are found on the cast stainless steel casing of pumps and valves.

Korea, Republic of

Utility and industries initially proposed to fabricate the pressurizer surge line and the RCS valves from cast austenitic stainless steel (SA351 F8M). However, thermal aging of cast austenitic stainless steel at the reactor temperature reduces the fracture toughness of the materials.

In addition, it is difficult to inspect cast stainless steel using ultrasonic techniques.

Because of the thermal aging and inspection concerns, utility considered such alternatives to cast materials as wrought material. As a result, the surgeline was fabricated from wrought stainless steel product forms. Therefore, thermal aging and the difficulties associated with ultrasonic inspection of cast stainless steel will not apply to the surgeline.

In addition, regulatory body requested that the utility consider alternative materials to cast stainless steel for other components with the best materials selection for a specific application such as large-diameter valve bodies.

The standard range of permissible ferrite content specified for stainless steel castings is from 8% to 30%. Therefore, regulatory body requested to limit the ferrite content of austenitic stainless steel castings with normal operating temperature to a maximum of 20%.

Sweden

In Sweden at the nuclear power plant Oskarshamn 1, which has been in operation since 1971, cracks have been discovered during a longer shutdown period, in 1995 in cast stainless steel, inside a valve in a main circulation loop. Penetrant testing revealed a large number of cracks both in valve and pump casings. (Ultrasonic testing was used without success). The cracks were found both on the inside and on the outside of the casings. The deepest crack was up to 50% of the wall thickness, which in the valves is 65 mm.

The cracks were found in rough unmachined areas. Two different types of cracking patterns were found. One type showed a crazed pattern with shallow cracks. These cracks usually disappeared when the surface was grounded down about 7 mm. The other type showed longer and deeper cracks and did not fully disappear after shallow grinding.

All the cracks found were surface cracks. No crack growth rates have been estimated. Metallographic analysis showed three different crack modes:

- intergranular cracks that follow the phase boundary between ferrite and austenite
- hot tearing or reheat cracking in or near a weld repair area
- intergranular cracking in the pure austenitic matrix.

The ferrite content in the cracked areas was very low. It was concluded that the cracks were fabrication defects, but environmental cracking can not be excluded. The possibility is not yet excluded, that the cracks can start as fabrication defects and then propagate by environmental cracking.

USA

Recently, the NRC staff reviewed the cracked samples from Oskarshamn 1 at Sweden, and agreed with their findings and conclusions. The root cause of the cracking is due to the castings significantly of ferrite content. It is very unlikely that such castings are in service in the US nuclear power plants. The staff is not aware of any reported cracking of cast stainless steel in the US similar to that found at Oskarshamn 1.

ISSUE TITLE: Loads not specified in the original design (CI 12)

ISSUE CLARIFICATION:

Description of issue

Piping connected to RCS, including the pressurizer surge line, are subject to flow and thermal transients during normal operation of the plant. These transients involve repetitive thermal shocks, stresses and thermal stratification. It may result in not anticipated displacement of lines. Measurements performed during plant operation have indicated loads, which were not specified and taken into account in the original design. Cracks and corrosion were observed in some plants. The resulting loads may exceed fatigue and stress limits. Restriction of piping expansion could further worsen the situation. Thermal stratification itself can introduce fatigue cracks on lines connected to RCS between the RCS and the first isolation valve or beyond the first isolation valve. Lack of monitoring of unspecified loads could lead to unpredictable failures.

The standard technical specifications (TS) for newer operating licenses require licensees to keep account of the number of transient occurrences to ensure that transient limits, based on design assumptions, are not exceeded. However, some older plants for which detailed fatigue analyses were performed on pressure boundary components using the rules of American Society of Mechanical Engineers (ASME) Section III, Code Class 1, do not have TS requirements for monitoring actual transient occurrences. These transients could significantly affect the fatigue life of the reactor coolant system (RCS).

Safety significance

Unspecified loads can be a potential cause of pipe breaks due to, e.g., fatigue damages, resulting in a loss of coolant accident or in partial unavailability of a safety related system.

Source of issue (check when appropriate)

- ____xx_____ operational experience
- ___________ deviation from current standards and practices
- ___________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

For the French plants additional measurements campaign were launched on some units, and studies based on mockups and stress analysis were performed. As a result, in-service inspection was improved and surveillance procedures modified.

This issue is generally linked with the possibility of establishing of a two-phase state: air/water or steam/water in lines connected to RCS with isolation valves. That diphasic state can induce corrosion attacks in the part of lines connected to RCS between the RCS and the first isolation valve. Up to now these attacks are limited to weld junctions and disk assembly of swing check valves. The studied solution consist to avoid these diphasic states with a sufficient pressure and operational procedures of air venting.

Another issue called "Farley-Tihange" phenomena is linked with cold water leakages, due to a bad leaktightness of ECCS (Emergency Core Cooling System) valves (during tests of leaktightness of 21000 ppm tank isolating valves), and involves a thermal fatigue in the lines connected to RCS. This problem concerns only 900 MW subseries, studies are in progress.

Germany

Extensive measurements were performed at German BWR and PWR plants to determine real loads of the pipe lines. It was found that thermal stratification occurs in the horizontal part of some lines. Based on the monitoring results, the operational modes were optimized in the German BWR and PWR plants to decrease the loads due to stratification.

Japan

Records are kept on the actual number of transient as necessary to fatigue evaluation. There are only few cases of unplanned shutdown, and the actual number of transients is very less than those postulated in design.

In a periodical safety review, fatigue evaluation is performed on reactor pressure vessel using actual number of transients.

Thermal stresses due to changes in flow and/or temperature during normal operation is evaluated in the design stage on the basis of technical standards.

As for thermal stratification, the following measures are being taken:

- Damages had been experienced in the RHR piping due to thermal stratification. The cause of the stratification was attributed to the valve gland leak. The measures such as proper adjustment of valve close position have been taken to avoid the fluctuation of the thermal stratification plane, and no damage have been observed since that.

- As for pipe failures resulting from valve seat leak reported in Farley Unit 1 and Tihange Unit 1, the possibility of seat leak occurring seems to be low in Japan because the double valve are installed on the upper stream of the lines in some cases and overhaul inspection of the valves is performed periodically. It has been confirmed through stress evaluation of piping systems on actual layout that the integrity will be maintained even assuming some seat leak occurs.

- As for pressurizer surgeline, such measures as minimum opening or keeping open of spray valves during starting up and shutting down are taken to prevent occurrence and disappearing of thermal stratification. In inspection (volumetric) is conducted to confirm integrity.

Korea, Republic of

Design of the pressurizer surgeline was originally based on approximate hydraulic calculations of little or no flow stratification at normal operating conditions. A re-evaluation of the surgeline design was performed for flow stratification. Accordingly, the surgeline was rerouted, support was redesigned and a thermal expansion and fatigue analysis was performed reflecting an additional loading due to the top-to-bottom temperature gradient. In addition, a two dimensional finite element model was changed to evaluate local stresses due to thermal gradient.

Countries operating WWER NPPs

At some WWER-440/230 plants one of the three pressurizer surge lines was removed since it was difficult to demonstrate its integrity also due to lack of monitoring of unspecified loads. A replacement of other lines was considered.

At some WWER plants, temperature monitoring has been implemented in lines subject to thermal stratification and thermal cycling, which is used to evaluate fatigue damages.

USA

The USNRC has issued the following bulletins which address the phenomena described in the Issue Clarification:

- USNRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems, " 1988;
- USNRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," 1988.

The USNRC studied one aspect of this issue under Generic Safety Issue (GSI)-78, "Monitoring of Fatigue Transient Limits for Reactor Coolant System." This GSI was developed to determine whether transient monitoring (cycle counting) is necessary at operating plants. The scope of the GSI also included an assessment of the risk when fatigue design limits are exceeded in pressure boundary components. The NRC staff obtained available records of transient monitoring from the licensees at seven US plants. These records, for the most part, contained the number of cycles associated with such plant operation changes as plant heatups and cooldowns. The staff concludes that licensing-basis fatigue criteria have not been exceeded at operating plants and that the fatigue failure of piping is not a significant contributor to the core-melt frequency. However, this issue will be reconsidered for plants operated beyond their original design lifetime.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- USNRC Bulletin 88-11, "Pressurizer surge line thermal stratification," 1988.
- USNRC Bulletin 88-08, "Thermal stresses in piping connected to reactor coolant systems, " 1988.

ISSUE TITLE: Boron corrosion on reactor coolant pressure boundary (CI 13)

ISSUE CLARIFICATION:

Description of issue

The primary coolant contains dissolved boric acid, which has rather strong corrosive effect on low alloy carbon steel components of the reactor pressure boundary. Reactor coolant that leaks out of the reactor coolant system loses water by evaporation, which results in the formation of highly concentrated boric acid solutions or deposits of boric acid crystals.

In several cases the leaking coolant has caused significant corrosion problems to the low alloy carbon steel components. The problems are related to situations when leaks below technical specification limits develop or when the installed leak detection system is not functioning due to various reasons. The scope of the affected components included the threaded fasteners (studs or bolts) in steam generator and pressurizer manway closures, valve bonnets, and pump flange connections installed on lines having a nominal diameter of 6 inches or greater, control rod drive flanges (RPV head) and pressurizer heater connections that do not have seal welds to provide leaktight joint.

Safety significance

The affected low alloy carbon steel components may not meet the design criteria. The degradation of the reactor coolant pressure boundary could lead to a loss of coolant accident.

Source of issue (check as appropriate)

- ____xx_____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Germany

Present practice in German NPPs does not allow operation with leaks of the reactor coolant pressure boundary. Minor leaks may be detected early by walk-downs of the containment. To date, the number of such leaks has been very limited and no indications of boron corrosion on the reactor coolant pressure boundary have been found.

Japan

When any leakage of primary coolant is observed, continuation of the operation is not allowed in Japan as a practice. Utilities have been conducting sufficient visual inspection during periodical inspection, and no corrosion leading to damages to equipment and/or bolts have been observed.

Ukraine

At Khmelnitsky NPP a vessel head had to be replaced due to a corrosion damage associated with the leaks in the CRDMs flanged joints. Improvement of the reliability, maintenance and testing of the leak monitoring system was proposed by the plant designer OKB Gidropress, Russian Federation. (See Issue I & C 8 in IAEA, EBP-WWER-05).

USA

US plants have been asked to consider the following topics:

(1) a determination of the principal locations where leaks that are smaller than the allowable technical specification limit can cause degradation of the primary pressure boundary by boric acid corrosion;
(2) procedures for locating small leaks;
(3) methods for conducting examinations and performing engineering evaluations;
(4) corrective actions to prevent recurrences of this type of corrosion.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- IE Bulletin No. 82-02, Degradation of threaded fasteners in the reactor coolant pressure boundary of PWR plants, dated June 2, 1982.
- USNRC Generic Letter No. 88-05, Boric acid corrosion of carbon steel reactor pressure boundary components in PWR plants, dated March 17, 1988.
- USNRC Information Notice No. 86-108, Degradation of reactor coolant system pressure boundary resulting from boric acid corrosion, dated December 29, 1986.
- USNRC Information Notice No. 86-108, Sup. 1, Degradation of reactor coolant system pressure boundary resulting from boric acid corrosion, dated April 20, 1995.
- USNRC Information Notice No. 86-108, Sup. 2, Degradation of reactor coolant system pressure boundary resulting from boric acid corrosion, dated November 19, 1987.
- USNRC Information Notice No. 86-108, Sup. 3, Degradation of reactor coolant system pressure boundary resulting from boric acid corrosion, dated January 5, 1995.

ISSUE TITLE: Steam and feedwater piping degradation (CI 14)

ISSUE CLARIFICATION:

Description of issue

Steam and feedwater piping is subject to degradation by corrosion, stress corrosion cracking, fatigue due to thermal stratification, water hammer and vibration. The piping material is usually carbon steel. Thermal stratification in feedwater piping can lead to significant stresses that may exceed design limits for fatigue. The problem can be linked to the piping design, the operation of the auxiliary feedwater system, and chemical characteristics of feedwater. In addition, the piping for feedwater or lines carrying two-phase fluids might be subject to erosion corrosion, causing wall thinning. In general, the piping is subject to less stringent in-service inspection as compared to primary piping and significant degradation could occur without being detected. The consequences of breaks vary depending on the break size, location (inside or outside the containment), and plant type (PWR or BWR). Severe degradation and several degradation related failures of the piping have occurred in the operating plants.

Safety significance

Failure of steam and feedwater piping would challenge safety and safety related systems, and other systems. It could lead to recriticality, RPV PTS event and affect the heat removal from the core. High energy piping breaks are of safety significance with respect to plant personnel.

Source of issue (check as appropriate)

- ____xx_____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

For French plants an inspection programme was launched and completed by mechanical analysis and expertises. Based on the results of these investigations, in-service inspection of the sensitive parts of the feedwater piping was improved and the plants are prepared to replace the piping at any time if necessary. Further studies are being performed to improve the knowledge of the phenomenon and the means of prevention and surveillance.

Germany

Some cracks have been detected during the construction of piping systems in German BWR plants. The material used for the piping including the main feedwater lines was 17MnMoV64, a precipitation hardening steel of higher strength. The nature of the cracks led to the decision to broaden the applied inspection programme for the pressure-retaining components where precipitation hardening steels of higher strength were used. In this context defects were detected in low-alloy ferritic steel piping in several BWR plants which were categorized as cracks due to strain-induced corrosion cracking (SICC). A principal decision has been taken to solve the problems in connection with the used precipitation hardening steel of higher strength by extensive replacement of this type by steels with lower strength and better weldability, such as 20 MnMoNi 5 5 or 15 MnNi 6 3. This action was performed between 1980 and 1984 in 4 operating BWR plants. In one plant the changes were already incorporated during the construction of the plant.

In consequence of the pipe rupture of the main feedwater system which occurred due to flow-assisted corrosion in unit 1 of the Finnish Loviisa NPP in May 1990 it was recommended to check the in-service inspection programmes in all German NPPs with regard to the early detection of wall thinning in the areas of high turbulences. The existing in-service inspection programmes were extended and optimized. No relevant wall thinning was observed in the main feedwater lines of German BWR plants during the additionally performed inspections. This is explained by the sufficiently high oxygen content in the feedwater and comparatively favourable hydrodynamic conditions and higher resistance material.

India

Periodic wall thickness measurement has not revealed any significant thickness reduction in the piping.

Japan

Major mechanisms of degradation in carbon steel piping are identified as thinning and/or thermal stratification.

(1) As for thinning, learning from the damage in feed water line of US Surry Plant, survey results and findings have been summarized and incorporated into the control guide to be applied to the operating plants.

    The following inspection programme is implemented in addition to the code ISI:

    - to inspect for the wall thinning at the possible locations during annual inspection;
    - to inspect those parts for which thinning cannot be denied based on past investigation;
    - to shorten inspection intervals for those parts where thinning is observed, and such measures as repair and replacement are to be taken to restore the specified minimum thickness.

    The replacement of the affected piping are implemented whenever necessary to do so.

(2) As for thermal stratification, learning from the failure of the feedwater piping resulting from the fluctuations of thermal stratification due to repeated start and stop of pumps at D.C. Cook 2, etc. in 1979, the practice to maintain continuous flow of auxiliary feedwater to minimize the fluctuation in thermal stratification is established.

Korea, Republic of

Regulatory body requested utility to identify the specific materials for the steam and feedwater system. Specifically, utility stated that carbon-manganese and chromium-molybdenum steels are to be used for the main steam and main feedwater systems.

For carbon steel piping systems, the following methods to minimize erosion/corrosion are reflected:

- the bulk velocity is limited to prevent excessive erosion of the pipe wall;
- piping design and routing will be used to reduce susceptibility of the piping to pipe wall thinning.

Spain

In Spain, there have been two pipe ruptures due to erosion corrosion: a main feedwater pipe (suction side) in a BWR in 1989 and a steam pipe break in a line connecting the turbine with the feedwater heater in a PWR in 1991.

In previous years, there have been some other ruptures of minor pipes belonging to the "balance of plant", such as drains of turbine control valves housings, etc. that led to repairs requiring power reductions.

So far, major preventive pipe replacements have been carried out in susceptible areas in certain geometries - such as turbine carry over and carry under, geometry and material of condenser outlets in some PWRs - where carbon steel has been replaced with low alloy steel.

The NPPs have an erosion/corrosion surveillance programme impact which requires the testing of an average of 300 areas every cycle, 25% of them belonging to FWS, using straight beam ultrasonic tests.

USA

The current status of this topic in the US is described in a forthcoming report: NUREG/CR-6456 (INEL-96-0089), "Assessment of Licensee Programs to Evaluate Pressurized Water Reactor Feedwater Nozzles, Piping, and Feedrings Cracking and Wall Thinning." (To be published.)

Carbon steel pipe degradation by erosion-corrosion caused several pipe ruptures in the nuclear plants in the USA. A catastrophic feedwater pipe rupture which occurred on December 1986, at Surry Nuclear Power station and caused four fatalities prompted the industry and the NRC to undertake actions aimed at preventing future erosion-corrosion caused pipe ruptures. In June 1987, the Nuclear Utilities Management and Resources Council (Nuclear Energy Institute) issued guidance for inspecting the affected piping and the Electric Power Research Institute developed a code for predicting components degradation by erosion-corrosion. This code has undergone several stages of development and currently, under the name of CHECWORKS, is used by the majority of utilities in the USA. After 1986, the NRC initiated an extensive effort of plant inspections and the results were published in NUREG-1344, "Erosion/Corrosion-Induced Pipe Wall Thinning in US Nuclear Power Plants." In addition, the NRC has issued Bulletin No, 87-01 and Generic Letter 89-08 requesting all the licensees to develop programmes for inspecting the components in their plants prone to erosion-corrosion degradation. The NRC has also incorporated inspection for erosion-corrosion into the regular plant inspection procedures and included Inspection Procedure 49001 into the NRC Inspection Manual. The NRC keeps the nuclear industry informed about erosion-corrosion failures of plant components by issuing Information Notices whenever a serious degradation of reactor components or pipe rupture occurs. Ten such Information Notices have been issued.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- NUREG-1344, "Erosion/corrosion induced pipe wall thinning in US nuclear power plants," April 1989.
- NUREG-0933, A prioritization of generic safety issues, Emrit, R., Division of Safety Issue Resolution, US Nuclear Regulatory Commission (September 1994).
- USNRC Inspection Procedure 49001.
- USNRC Bulletin 87-01
- USNRC Generic Letter 89-08.
- Recent USNRC Information Notices 95-11, 97-84.

ISSUE TITLE: Steam generator internals damage and plate cracking (CI 15)

ISSUE CLARIFICATION:

Description of issue

Problems have been experienced recently in the US with steam generator internals at three PWRs (two problems with tube support plates and one with the wrapper barrel). In the first example, pieces of a carbon steel drilled hole type tube support plate were found on the next lower support plate. It was postulated that improper placement (too close) of the cleaning nozzle caused an erosion/corrosion problem. In the second example of the same type of tube support plate but at a site where the steam generators had not been chemically cleaned, ligament cracks were found at the uppermost support plate and a portion of the periphery in one area was broken away. In this two examples, the support plate damage effectively eliminated lateral support to tubes within the affected region. Causes for the cracking and potential generic implications are under investigation. In the third example, visual inspections revealed that the bottom of the wrapper had dropped down by 20 millimeters in one steam generator and by five millimeters in another steam generator. Further investigation revealed that wrapper welds at each of six vertical supports in the first steam generator and at three of six vertical supports in the second steam generator had failed, allowing the downward displacement of the wrapper. While the cause continues to be investigated, the preliminary assessment is that unanticipated axial restraint against differential thermal expansion between the wrapper and steam generator pressure vessel shell led to significant loading of the wrapper vertical supports and the failed welds.

In a number of WWER-440/213 NPPs (Dukovany, Paks, Rovno) the feedwater distribution nozzles were found to be damaged and had to be replaced using different material (stainless steel) and a modified design. The damages to nozzles were different depending on nozzle positions, ranging from complete nozzle destruction to moderate damage.

Safety significance

The damage of the lateral support would affect the vibrational stability and the ability to sustain earthquake and LOCA loadings. The complete fall of the wrapper barrel could lead to the loss of feedwater, damage to the largest radius tube U-bends, loose parts, and tube rupture.

In case of WWER-440/213 NPPs, the concern is the damage caused by loose parts in the secondary side.

Source of issue (check as appropriate)
- ____xx_____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Germany

The tube bundle supports of Steam Generators in Germany are of lattice-type in the straight run of the tubes and made of horizontal and vertical strips in the U-band area. Limited problems have been experienced in a few older plants. The construction was upgraded and in the new plants an improved construction is applied. Today, steam generator internals damage is no more an issue in Germany.

India

The SGs are isolated on the shell side in both the units and are not in use due to excessive tube leaks. No internal inspection of the SGs are being done currently.

Korea, Republic of

The SG internal components are designed to maintain localized fluid velocities below the critical velocities that will cause excessive tube vibration.

The horizontal egg crate tube supports are designed and located to maintain the natural frequency of the tubes higher than the existing frequency induced by cross-flow in the tube bundle entrance region.

SG tube vibration test had been performed on a test model for the Yonggwang Units 3 and 4 economizer and lower tube bundle region to demonstrate a significant additional margin with 120% flow conditions and some margin for 140% flow conditions. The test showed good result of no excessive flow-induced vibration.

USA

Utilities implemented repairs and will monitor the position of the wrappers.

The NRC staff is also monitoring the experiences of the foreign units in these problem areas and evaluating the potential implications for the domestic units.

Countries operating WWER-440/213 NPPs

The designer of the steam generators, OKB Gidropress, has designed new FWD pipes made of stainless steel and provided with a modified system of FWD nozzles. These new FWD pipes were installed in all steam generators of the Rovno NPP. The measurements after replacement showed that under normal operating conditions the parameters of the steam generator have not been deteriorated. Another design prepared by Vitkovice in Czech Republic is characterized by FWD pipe location above the water level and FW distribution through long downcomers into mixing boxes situated at the level of previous FWD pipes. Such FWD pipes have been installed in 16 steam generators at Dukovany NPP. The temperature and measurements performed there have shown that there are no dangerous cold zones on steam generator bottom nor walls.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- USNRC GL-97-06, Degradation of steam generator internals.
- Recent USNRC Information Notices 96-09, 96-09 Supplement 1.

4.1.4. Primary circuit and associated systems (PC)

ISSUE TITLE: Overpressure protection of the primary circuit and connected systems (PC 1)

ISSUE CLARIFICATION:

Description of issue

There have been numerous reported incidents of pressure transients in PWRs where Technical Specification pressure and temperature limits of the RCS have been exceeded. The majority of these events occurred while the reactor was in startup or shutdown conditions and at low reactor vessel temperatures. The issue is the reliability of the cold overpressure protection system and especially the safety and relief valves situated either on the pressurizer or RHR systems. The protection systems in US plants used to mitigate and reduce the potential for these events are termed low temperature overpressure protection (LTOP) systems.

More precisely safety and relief valves, when used to perform safety functions such as mitigation of steam generator tube rupture accident or cold overpressure protection, may have to fulfill their function in water flow conditions. However, such conditions were not always anticipated at the original design stage. Moreover, using safety and relief valves not qualified for water flow to limit the RPV pressure in water solid conditions could lead to a loss of coolant accident.

All PWRs can be affected.

Safety significance

The unavailability of overpressure protection system may result in a beyond design condition of the reactor coolant system including connected systems.

Source of issue (check as appropriate)

- ____xx_____ operational experience
- ___________ deviation from current standards and practices
- ___________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

At Kozloduy NPP, at the present time, there is no device to protect the primary circuit against cold overpressure for all the causes, particularly when the primary is in the water solid state. In addition to the administrative measures being taken, only a specific automatic protection device is installed for the makeup pumps (TK system) to disconnect them at low temperature if the primary pressure increases.

Nevertheless, the study of an automatic protection device is in progress, which takes into account all the overpressure possibilities and the vessel behaviour towards the end of its service life. The Kozloduy NPP expects that the results will be available at the end of 1995.

Moreover, a specific study to generalize the protection of the primary circuit against overpressurization caused by inadvertent operation of the high pressure injection pumps (TQ n3) and high pressure boron injection pumps (TQ n4) is underway.

The pressurizer safety valves are planned to be qualified for carrying steam or a mixture of steam and water in the current qualification programme. Future requirements will be provided to the manufacturer concerning the capability of carrying water and of performing primary feed and bleed.

Czech Republic

At Temelin NPP, the pressurizer safety valve system was reconstructed. Now there are two safety valves in the design (with the capacity as of three safety valves as in the original design); each of them is controlled by a couple of control valves. A relief valve controlled by an electromagnetic valve was added. There is a motor-operated isolation valve before the relief valve which is controlled automatically or manually. Sempell-Babcock AG company is the producer of the pressurizer safety valve system. Valves of pressurizer safety valve system are sized for the water flow-rate.

Regarding the protection against cold overpressurization, the following measures have been implemented:

- Disconnection of the makeup pumps and switch off of pressurizer heaters from signal "primary coolant temperature below 100°C and system pressure above 3.5 MPa."
- Administrative and technical measures prevent water injection by high pressure injection pumps. These requirements are specified in the Temelín NPP design documentation and will be contained in procedures for normal operation.

France

On French plants, cold overpressure protection is provided by pilot operated safety valves on the pressurizer when the RHR system is isolated. When the RHR system is operating, the protection is provided by pilot operated safety valves on the discharge lines from the RHR pumps. All these valves are qualified to operate in water conditions. In addition, specific requirements have been included in the Technical Specifications.

Germany

Cold overpressurization of the RCS during shutdown states is prevented by a low temperature overpressure protection (LTOP) system.

If the pressure in the RCS exceeds a value which is given by the function of cold leg temperature versus RCS pressure, the protection system will limit the pressure in the RCS by opening the pilot operated relief valve at the pressurizer. This valve is capable of relieving steam as well as water, but its relief capacity is limited.

To avoid a rapid overpressurization through an unintentional start of the high pressure safety injection pumps, the HP safety pumps are taken out of service (electrically blocked in the switchyard) at round about 3 MPa. This is an administrative measure.

Korea, Republic of

- The low temperature overpressure (protection (LTOP) system is provided by the spring-loaded liquid relief valves in the SCS (Shutdown Cooling System) inlet lines. These valves are set at a pressure low enough to prevent violation of the 10CFR50, App. G heatup and cooldown valves should a pressure transient occur during low-temperature operations. Alarms on SCS isolation valves annunciate when the RCS temperature is reduced below the maximum temperature for LTOP and the valves are not fully open.
- For LTOP considerations, two types of events are considered as the design-basis events. These events are:
  - the mass addition transient caused by charging and safety injection flows following and inadvertent SI actuation, and
  - the heat addition transient caused by the restart of a reactor coolant pump.

As an analysis result, the relief capacity meets the required relief flow for the worst transient and all the design criteria for LTOP.

- This provides assurance that the temperature-pressure limit presented in 10CFR50 APP. G is not exceeded during any transients.

Russian Federation

In order to prevent cold overpressure of the primary equipment of WWER-1000, which can occur as a result of erroneous actions that design provides for the interlocking for disconnection of makeup pumps as per cold pressurization parameters ($P > 33$ kgf/cm^2, $t < 130^oC$), disconnection of pressurizer electrical heaters, closing of valves on ECCS tanks and other measures directed to isolation of sources of increased pressure.

In addition, administrative measures are provided, pertaining to removal of the power supply to emergency high pressure pumps, which are contained in operation manuals as well as other measures aimed at isolating the sources of high pressure.

In addition to the mentioned measures the emergency signalling by high level in the pressurizer is supposed to be performed.

It is also necessary to consider, as a supplementary measure, possible mounting of safety valves in the lines of scheduled cooldown with regard for possible single failure of the interlockings above mentioned or a single personnel error during implementation of administrative measures.

Ukraine

The modernization programme of the WWER-1000/320 NPPs in the Ukraine includes a study on the status of qualification of the pressurizer safety valves for water flow and, depending on its results, the valves will be modified or replaced.

As for the protection against cold overpressurization, the following measures have been implemented in Ukrainian plants:

- automatic disconnection of the makeup pumps with signal "primary coolant temperature below 100°C";
- administrative and technical measures to prevent water injection by ECCS pumps below 15 bars;
- administrative and technical measures during hydrotests to prevent pumps from injecting water when the primary coolant temperature is below a level which is determined according to the expired lifetime of the vessel.

At Rovno NPP, three identical sets of pulse safety valves (ISD Sempell type) were installed on the pressurizer of Unit 3. The Rovno NPP considers replacing the existing pressurizer safety valves, since there is no certification document available for the qualification of these valves. The new valves for Unit 4 are of the type UF 50024-100 PS. The certification document of this type of valve indicates that it is used for steam flow, steam-water mixture flow and water flow. But test results provided in the certification only related to air flow and steam flow.

The Rovno NPP is considering a cooperation with other European countries to qualify the Sempell type valves with steam-water mixture flow and water flow. Since the feed and bleed from the primary circuit side is considered to be used, the valves have to be qualified with water flow.

For the Rovno NPP Unit 4, two new safety valves are to be installed which are provided by the Russian manufacturer according to the national standards to protect against cold overpressure. The actuation pressure is 20 bars since the design pressure of the discharge line of the RHR system is 22 bars.

The Rovno NPP Unit 3 has already installed two non-safety graded safety valves on the discharge line of the makeup pumps as a temporary measure These valves will be replaced by safety-graded ones when they are available

At Zaporozhe NPP, protection of the primary circuit against overpressure during incidents is provided by three safety valves on the pressurizer For Units 1 to 4, only one valve can be opened intentionally by actuating pilots from the control room and the emergency control room The other two safety valves are acting as passive pilot operated safety valves For Units 5 and 6, they have a new design of valves All the pilots of the safety valves can be actuated from the control rooms The pilots can be isolated by an additional pilot in case of impulse pipe failure The qualification of safety valves for water flow is not confirmed

To prevent the risk of primary cold overpressurization, some measures have been taken so far

- creation of a nitrogen blanket in the pressurizer when the primary circuit temperature is close to the critical temperature
- control of the pressurizer level by the makeup system,
- automatic stopping of the makeup pumps in case of primary circuit pressure higher than 35 bars during shutdown,
- administrative measures to avoid inadvertent startup of high pressure safety injection pumps and boron injection pumps in shutdown

USA

Regulatory Guidelines and Operating Considerations

In US plants, regulatory guidelines require that a low temperature overpressure protection (LTOP) system be designed and installed to prevent exceeding the applicable technical specifications and Appendix G limits for the reactor coolant system while operating at low temperatures The technical specification and Appendix G limits are termed the P/T limits, and are determined in accordance with 10CFR50 Appendix G and Appendix G of Section XI Licensee's have administrative controls to ensure that LTOP systems are operable at coolant temperatures less than 200F or reactor vessel metal temperature less than RT_{NDT} + 50F, whichever is greater The LTOP systems must limit the maximum pressure in the vessel to 110% of the P/T limits curves

The US primary system safety and relief valves have been qualified by testing for the design-basis fluid inlet conditions In addition to hot steam conditions, the PORVs were tested for representative low temperature liquid flow, and the safety valves were tested at representative hot liquid flow The NRC required that the reliability of the PORVs be improved to provide better low temperature overpressure protection (LTOP) and better operation at operating temperature Better maintenance and improved Technical Specifications were implemented

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992)
- MOHT - OTJIG - EDF generic reference programme for the modernization of WWER-1000/320, Revision 6 (February 1997)
- 10CFR50. Appendix G, "Fracture toughness requirements "
- USNRC Regulatory Guide 1 99, Revision 2, "Radiation embrittlement of reactor vessel materials "
- NUREG-0737, Item II D 1 - Safety and relief valve testing

- ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, "Fracture toughness criteria for protection against failure."
- Unresolved Safety Issue 26 - Reactor vessel pressure transient protection.
- Generic Safety Issues 70 & 94 - (Generic Letter 90-06) Power operated relief valve & block valve reliability and additional LTOP for LWRs.

ISSUE TITLE: Adequacy of the isolation of low pressure systems connected to the reactor coolant pressure boundary (PC 2)

ISSUE CLARIFICATION:

Description of issue

Isolation valves between the reactor coolant system and the low pressure interfacing systems may not adequately protect against overpressurization of low pressure systems if they are leaking or inadvertently opened because of personnel errors or equipment failure.

During a startup of the Biblis A plant on December 17, 1987, the first check valves in the pipe from the reactor coolant system to the residual heat removal (RHR) system had to close for overpressure protection. The position indication and a plant computer alarm annunciated a non-closed position of the first check valve in one train of the RHR system. This was overlooked by the operator. Therefore plant startup was continued with the check valve in the open position. The leakage through the first check valve caused the actuation of a relief valve, releasing primary coolant to the chemical and volume control system.

On September 17, 1994, while the Wolf Creek reactor was in Mode 4, there was an inadvertent blowdown of reactor coolant through the residual heat removal system to the refueling water storage tank due to improper operation of the pressure isolation valves. This event occurred because of concurrent activities involving manipulation of RHR valves while cooling down to begin a refueling outage. The inadvertent blowdown of reactor coolant was terminated in about a minute by closing one of the RHR valves that was being manipulated. Continued blowdown through the RHR system would have uncovered the reactor hot leg and introduced steam into the refueling water storage tank header line which is the water supply line for the emergency core cooling system (ECCS) pumps.

Safety Significance

Reactor coolant system boundary isolation failure could result in coolant blowdown and overpressurization of the low pressure piping. This can lead to a loss of coolant accident and containment bypass that, if combined with failures in the emergency core cooling systems, would result in a core-melt accident with significant off-site radiation releases.

Source of issue (check as appropriate)

- ___xx_____ operational experience
- __________ deviation from current standards and practices
- ___xx_____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Germany

As a consequence of the incident in the NPP Biblis unit A, the German authority required for all German NPPs (PWRs and BWRs) an exhaustive investigation of the adequacy of the isolation of the low pressure systems connected to the reactor coolant system. The available isolation means (isolation valves, check valves), their actuation, the instrumentation for detection of an overpressure or overtemperature condition in these systems have been investigated in detail. All German NPPs have analysed the Biblis event for applicability to their plants. In specific cases, design improvements or administrative measures have been implemented.

India

The existing administrative control has assured no such incident of inadvertent opening of the isolation valves.

Korea, Republic of

For all interconnected systems and components, the utility changed the design pressure from 484 psia up to max. 900 psia. However, regulatory body plans to request to demonstrate for each connecting system and component that the degree and quality of isolation or reduced severity of the potential pressure challenges compensate for and justify the safety of low-pressure system or components.

Spain

Spanish plants have in place administrative measures to reduce the risk of personnel errors or malfunction of the low pressure systems isolation valves. They have analysed the Biblis A and the Wolf Creek events and incorporated them into the training programmes.

In some PWR plants some interlocks of RHR isolation valves have been eliminated.

ADDITIONAL SOURCES:

- NUREG-1463, "Regulatory analysis for the resolution of Generic Safety Issue 105: Interfacing system loss of coolant accident in light water reactors", July 1993.
- USNRC Information Notice 95-03, "Loss of reactor coolant inventory and potential loss of emergency mitigation measures while in a shutdown condition."
- USNRC Information Notice 92-36, "Intersystem LOCA outside containment."

ISSUE TITLE: Reactor coolant pump seal failures (PC 3)

ISSUE CLARIFICATION:

Description of issue

Reactor coolant pump seal failures could significantly challenge the makeup capabilities at nuclear stations In PWRs, this can notably happen during station blackout conditions

In WWERs, injection of cold water to the reactor coolant pump (RCP) seals is provided by makeup pumps which are not backed up by emergency diesel generators and is isolated if a safety injection occurs

The injection of cold water is backed up by an autonomous cooling circuit which is designed to cool the lower bearing of the primary pump and avoids the primary hot water to reach the primary pump seals The autonomous cooling circuit which is circulated by a special impeller in the RCP and backed up by an emergency pump with emergency diesel generator power supply is able to take water from the primary circuit and to inject into the bearing after its cooling when the pump stops or starts In case of loss of off-site power and a failure of the emergency pump of one primary pump, there is a possibility of damage of the primary pump seals and develop a small primary break

The makeup flow would also be lost as a consequence either of a containment isolation signal or of a random failure The possibility of cooling the RCP seals would be also lost in the case of total station blackout

Safety significance

Inadequate provisions in the design of the main coolant pump to keep the seal tight affect the first level of protection of defence in depth A loss of seal injection flow may damage the seal and lead to a SB LOCA The protection of the second barrier is affected The safety function cooling the fuel might be impaired by increased challenge of the safety systems

Source of issue (check as appropriate)

- ___xx______ operational experience
- ___________ deviation from current standards and practices
- ___xx______ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

At Kozloduy NPP, the following proposals have been made

- Backup of the makeup pumps by diesel generators, and modifications of the protection logic of some makeup valves to not close them in case of containment isolation,
- Installation of an additional diesel generator on each unit in order to supply important consumers which are not backed up by emergency diesel generators

Czech Republic

At Temelin NPP, the power supply to makeup pumps is backed by diesel generator

The reactor coolant pump seal remains tight even without a supply of sealing and cooling water for a minimum of 24 hours This is certified by Russian research centre "Energonasos CKBM" based on pump testing as well as operational experience accumulated

France

In the frame of beyond DBA analysis some specific devices have been added to improve safety as specific backup in case of blackout. A specific GUS (ultimate backup device): turbine generator (gas turbine) for 1300 and 1400 MW subseries or diesel for 900 MW subseries, (allows maintaining seal water injection to the primary coolant pump via the test pump, and necessary control for operators).

Germany

The seals of the reactor coolant pumps (RCPs) are cooled by the CVCS. Part of the make-up flow of the CVCS pumps is directed to the RCP seals. The pumps of the CVCS are supplied by the Diesel Generators in case of LOOP. Only in case of station black out (loss of all AC power) the seals are without cooling and will heat up. To avoid a small break LOCA under such conditions, a special seal at the shaft of the RCPs - which is normally not in operation - has to be closed remotely. The behaviour of this final seal has been tested in a test house and it was demonstrated that this seal is capable to withstand the primary coolant temperature.

India

RCP seal cooling water injection was provided in 1984 in addition to the original seal cooling provision. The injection is from CRD feedpump which is powered by diesels. The seal failures have since reduced significantly.

Korea, Republic of

- This issue includes improving the reliability of RCP seals by reducing the probability of seal failure during normal operations and under abnormal conditions. Specifically, acceptable resolutions to this issue include an RCP seal design that ensures the RCP seal integrity following SBO for an extended period.
- The seal is cooled through two independent and redundant seal cooling systems: the seal injection system and the component cooling water system (CCWS).
- The CCWS is a safety grade system satisfying the single failure criterion.
- The design also includes an onsite alternate AC (AAC) power source (third D/G) to power the charging pumps that supply seal injection water to cool the RCP seal during an SBO.

Russian Federation

A specific reactor coolant pump (RCP) sealing test was performed in the Russian Federation. The 3 min. running down time and actual operating parameters were not simulated. But the report on the "Instruction of RCP GTsN-195M" (195-00-0013PE, 1986) provided by the manufacturer states: "The RCP seal can operate without seal water and service cooling water during the running down time without damage" (Section 1.3.10).

Under emergency operating conditions in the event of complete blackout of NPP, loss and decrease of flow rate of cooling and sealing water during a period of time, which is above than is permissible, may occur, and RCP elements may fail.

In performance of safety analyses of beyond-design basis accidents the characteristic of GTZN-195M seal to provide tightness of the primary circuit, within the design limits, during 24 hours with loss of locking water supply to the seal, will be taken into account as well as loss of cooling media supply (intermediate circuit, service water) and inoperable pumps of independent circuit. That is to say, such accidents as blackout or loss of ultimate heat sink may proceed during the first 24 hours, as a minimum, with the primary circuit being tight.

The mentioned characteristic of GTZN-195M was confirmed by both the special tests, and operation experience.

Three special tests were performed with the actual pump. For these tests the existing test rig "Dnom 500" was used into which the actual pump is placed for performance of acceptance tests after manufacturing. Resistance of the primary circuit is ensured by the special orifice plate. The test rig was upgraded:

(1) to provide for constant parameters: temperature and pressure during long time (not less than 24 hours);
(2) to provide for actual conditions of the cooling medium with loss of heat removal from the containment.

The first two tests were performed in 1989-1990 with GTZN-195M pump. In the first test the environmental conditions were reproduced. In the second test the environmental conditions were reproduced.

The third test was performed in 1992 with GTZNA-1391 having the sealing unit similar to that of GTZN-195M.

During all tests temperature and pressure were kept stable and equal to nominal values (300^{o}C at 160 bar).

By the results of the tests the following were noted:

(1) No leaks from seals both before tests and after them.
(2) Rubber items of the seal kept their required properties securing tightness of the seal but were recognized unacceptable for further operation, that is, their replacement was needed.

There is some operation experience of WWER-1000 NPPs which confirms the high reliability of RCP seal. At least two events support this.

As to information from NPP Kozloduy, the NPP blackout took place at power at unit 6 on 22.09.1992 (with start up of DG), for a period of about 10 hours. There was no cooling water supply during this time. After restoration of power supply the examination of RCP seals was made and showed their satisfactory condition. A similar case was observed at Zaporozhe NPP where duration of blackout was 2 hours.

In performance of the analyses it is also necessary to take into account the measures proposed by the given programme; such as installation of common unit diesel generator station providing power supply to such important systems as makeup blowdown of the primary circuit and auxiliary feedwater. Connection of the primary circuit makeup blowdown system to reliable power supply does not increase the risk related to possible dilution of the primary coolant with pure condensate because:

(1) there is automatic interlocking for disconnection of pure condensate supply with RCP disconnected, or reactor scram;
(2) the accumulated volume of pure condensate in the pipelines from makeup pumps, as shown by the experiments on coolant mixing with the water supplied, cannot lead to hazardous consequences.

Connection of primary makeup blowdown system is performed after removal of prohibition (for the conditions of steamline break inside the containment) for opening of isolation valves, when pressure is decreased in the containment, due to operation of sprinkling system. The existing estimations show that this period does not exceed 5-6 hours.

Ukraine

At Rovno NPP, in Unit 3, an additional line which connects the pump auxiliary impeller with the pump seal via cooler has been added. In the case of a loss of off-site power event, assuming a single failure in the circulation pump, the circulation of water can be maintained by the auxiliary impeller during the coastdown of the RCP. This measure is a backup to the circulation pump of the autonomous cooling circuit.

Regarding the measures proposed for Unit 4 directed to the RCP seal cooling system, the same modification will be made.

In addition to the improvements related to the RCP seal cooling system for the Rovno NPP Unit 4, other improvements related to RCP to be introduced, such as thermal barrier of new design, more reliable antireversal device, wrench for group tightening of the main flange of RCP, vibration monitoring, procedure for pump casing defect identification and some additional calculation, will further improve the reliability and safe operation of RCPs.

At Zaporozhe NPP, for Units 1 to 4, in case of loss of off-site power, the makeup pumps stop because they are not backed up by diesel generator. In this situation, the cooling of the main coolant pump seal is provided by circulation of primary water cooled by an intermediate coolant circuit which itself is cooled by the essential service water (one circulating pump associated to each main coolant pump, this autonomous cooling circuit prevent the hot water from reaching the main coolant pump bearing). If one of these circulating pumps fails, the seals of the corresponding main coolant pump are not cooled.

For Units 5 and 6, one additional diesel generator for each unit backs up the makeup pumps.

USA

NRC staff studies and analyses concerning seal leakage are documented in NUREG/CR-5167 Cost benefit Analysis for Generic Issue 23. This report identifies several modes of RCP seal leakages which may be in excess of that assumed in station blackout rule (10CFR 50.63)

However, in 1995, the NRC decided against the publication of a rule to address this issue in light of the uncertainties in the analyses and the doubts on significant safety benefit through a major backfit.

Licensees were informed of the concern through the documents listed in the reference. During the IPE response, the licensees addressed this issue and have instituted procedures and training.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER- 1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- MOHT - OTJIG - EDF generic reference programme for the modernization of WWER-1000/320, Revision 6 (February 1997).
- NUREG/CR-5167, Cost/benefit analysis for Generic Issue 23, Reactor coolant pump seal failure.
- NUREG-0933, Generic Safety Issue 23.
- USNRC Generic Letter 91-07, Reactor coolant pump failures and its possible effect on station blackout.
- Recent USNRC Information Notices 96-58, 97-31.
- USNRC Information Notice 95-42, commission decision on the resolution of Generic Issue 23, Reactor coolant pump failure.

ISSUE TITLE: Safety, relief and block valve reliability - primary system (PC 4)

ISSUE CLARIFICATION:

Description of issue

Protection of the primary circuit against overpressure during incidents is provided by safety valves. If a safety valve would fail to close in connection with a pressure transient, the transient would run into a small break LOCA. The risk of safety valve failure to close is high in some NPPs where safety valves are not qualified for steam/water mixture and water flow.

In some countries, pilot operated relief valves, (PORVs) and block valves were originally designed as non-safety components intended for operation under normal conditions. The PORVs were provided for pressure control of the RCS during normal operation and transients. The block valves were installed because of expected leakage from the PORVs. The valves were not required for safe shutdown or to mitigate the consequences of accidents.

The TMI-2 accident focused attention on the reliability of the PORVs and the block valves and requirements were established regarding electrical power backup, environmentally qualified position indication and verification operation under all expected flow condition including water flow (see PC 1, Overpressure protection of the RPV at water solid conditions and at low temperatures). Analyses showed that the role of the PORVs and the block valves had changed such that the valves are now relied upon in certain safety related functions.

The safety related functions identified are:

- Mitigation of a design basis steam generator tube rupture (SGTR) accident. At a SGTR, if pressurizer spray is not available, the PORVs are manually opened to reduce the RCS and SG pressure and mitigate the consequences from release of activity. After pressure equalization the PORVs are closed.
- Plant cool down at some accident sequences and transients can be mitigated by using the PORVs and HPI pumps for cooling down the RCS until the RHR-system can be engaged. During this mode of operation, known as feed-and-bleed, the PORVs are manually opened and either remain open or experience multiple openings and closures during the cooldown phase.
- Low temperature overpressure protection of the reactor vessel during startup and shutdown. The PORVs are set to open to reduce the reactor pressure at established pressure/temperature limits. Therefore the valves and associated systems must fulfil the specification in SRP Section 5.2.2.

For operating plants to fulfil the safety related functions above, the following items would be of greatest immediate benefit for the increase of the reliability of the PORVs and block valves with associated equipment:

- Verification of the valve function during feed-and-bleed operation and prolonged operation at SGTR. The functions need to be verified regarding relevant flow condition, operation time, number of opening/closure cycles and environmental conditions.
- Valve exhaust piping shall be designed for the relevant flow loads.
- Valve position indication and power supply shall be comparable to safety grade.
- Perform In Service Testing including stroke testing of the PORVs should only be performed during Hot Standby or Hot Shutdown. The block valves should be included in the MOV test programme.
- Modify the limiting conditions of operation of PORVs and block valves in the Technical Specifications.
- Include the valves in an operational quality assurance programme.

This issue is relevant for PWRs and WWERs.

Safety significance

A lack of qualification of safety valves for water flow would seriously affect the primary circuit integrity.

PORVs are involved in the mitigation of some accidents listed above and as such their insufficient reliability might impair the fulfillment of safety functions.

Source of issue (check as appropriate)

- ____xx_____ operational experience
- ___________ deviation from current standards and practices
- ___________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Germany

The safety valves and relief valves at the pressurizer are capable of relieving steam, steam-water mixture, and water. These valves have been upgraded in the frame of the severe accident prevention and mitigation concept. One of the measures of this concept is primary feed and bleed. As a consequence, the valves and the corresponding pipes have been upgraded to cope with water loads.

Japan

Based on the lessons learned from the TMI accident, the following measures are taken for the safety designing of pressurizer relief valve:

- the power of the instrumentation and control air compressor system necessary for actuating the pressurizer relief valve are to be fed from the safety related electrical system;

In addition to the above, the following measures are taken:

- as a provision to the pressurizer relief valve failing to close, an automatically block valve is added to the original design upstream of the pressurizer relief isolation valve;
- verification tests are performed on pressurizer relief valves, taking into consideration the required conditions on safety functions.

Russian Federation

The qualification of pressurizer PSDs for steam, water and steam-water mixture is required for their application under beyond DBAs:

- ATWS;
- feed and bleed procedure;
- pressure decrease under severe accidents.

(a) NPP in operation

As to pressurizer PSD there were two suppliers: SEMPELL and PENZA valves factory (Designer of PSD - Ukrainian Department of TZKBA). Documentation of SEMPELL Supplier guarantees possible operation of PSD with steam, water and steam-water mixture, however western supervisory bodies express a doubt in validity of these guarantees. There are no documents on qualification of SEMPELL PSDs at operators disposal.

The present programme proposes to perform the review of the qualification file of SEMPELL. If necessary, a repeated qualification could be performed or the replacement of which by valves already qualified could be considered.

PENZA valves are qualified for water, steam and steam-water mixture at Kashira experimental centre and have been operated at units 5 and 6 of Zaporozhe NPP during 4 years, and during 3 years at unit 4 of Balakovo NPP.

(b) NPP under construction

The NPP should be equipped with qualified valves.

Sweden

- Measures taken in the Swedish PWRs:
- Basic closure function verified in EPRI tests.
- Improved position indication and backup by air accumulators has been implemented.
- A project for verification of the valve function during feed-and-bleed and prolonged operation has been launched.

USA

The US PORVs have been made more reliable to accomplish the necessary safety functions by qualification testing and improved maintenance and Technical Specifications.

ADDITIONAL SOURCES:

- MOHT - OTJIG - EDF generic reference programme for the modernization of WWER-1000/320, Revision 6 (February 1997).
- Ringhals Report 0346/95, Förstudierapport, Åtgärder på tryckhållarens ventiler, (In Swedish).
- NUREG-0737, TMI action plan requirements.
- NUREG-0933, Generic Issue 70: PORV and block valve reliability.
- USNRC Generic Letter 90-06, Resolution of Generic Issue 70 and Generic Issue 94.
- Recent USNRC Information Notices 95-34, 96-02, 96-42.

ISSUE TITLE: Safety, relief and block valve reliability - secondary system (PC 5)

ISSUE CLARIFICATION:

Description of issue

In the case of a primary to secondary leak, capable of overfilling the steam generator (multiple tube rupture, collector rupture, etc.), water could enter steamlines and reach the relief and safety valves. The lack of qualification of these valves to operate with water or water-steam mixture can then lead to their failure to reclose after opening.

Safety significance

If water flows through the SG safety and relief valves, they may not be able to close and cut off the stream when the pressure is lowered. In this case, the accident is worsened with an effluent release to the environment.

Source of issue (check as appropriate)

- ____xx_____ operational experience
- ___________ deviation from current standards and practices
- ____xx_____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

For the Kozloduy NPP, it is planned to study the possibilities for installing an isolating valve on the steam line upstream of the BRU-A valve.

France

For French plants a testing programme has been launched for all safety and relief valves of the SG in order to check their capability to operate (and reclose) with water flow.

Germany

The safety valves and also the relief valves at the main steam lines are capable to operate under all fluid conditions: steam, steam-water mixture and water. This has been verified by valve tests.

Russian Federation

The steam dump valve to the atmosphere upgrading programme shall be developed, in this case the measures on design changes and changes of control system shall be prepared. The programme shall cover the matters to be clarified as a result of works on qualification of equipment.

Ukraine

The modernization programme of WWER-1000/320 NPPs envisages either a modification of the steam generator safety and relief valves or their replacement by more reliable ones.

For the Rovno NPP, the steam generator safety and relief valves have so far not been qualified to carry water. The NPP plans to demonstrate a qualification of the BRU-A and to exchange the safety valves.

For the Zaporozhe NPP, in case of a primary to secondary leak, the primary circuit water can quickly fill the corresponding steam generator and the line up to the relief valve (BRU-A) The BRU-A valve is not isolatable and not qualified to be operated with water flow

The steam generator safety valves are pilot operated by the steam pressure The opening of the safety valve can be manually operated from the main control room by means of two electromagnets installed on the pilots These electromagnets are also automatically de-actuated by a high pressure signal Only one impulse line common to the two pilots is going to the safety valve The two safety valves are "safe close position" type, in the event that the impulse line fails, the valve will be kept closed and non-operable

USA

US plants use operating procedures to depressurize the primary system and control steam generator level during a SGTR event Studies and actual operating experience have shown that the risk due to overfilling a SG is not high

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992)
- MOHT - OTJIG - EDF generic reference programme for the modernization of WWER-1000/320, Revision 6 (February 1997)
- Generic Safety Issue 126, Reliability of PWR main steam safety valves (NUREG-0933)
- NUREG-0844, USNRC integrated program for unresolved safety issues A-3, A-4, A-5, regarding steam generator tube integrity, Sept 1988
- Recent USNRC Information Notices 96-03, 96-61, 97-09

ISSUE TITLE: Spring actuated safety and relief valve reliability (PC 6)

ISSUE CLARIFICATION:

Description of issue

Wrong pressure settings or inoperable pressure relief valves have been discovered in BWR and PWR plants. In some cases, the calibration has been wrong and in other cases the relief valves have been equipped with wrong springs left in an inoperable configuration or tested under the wrong environmental conditions. The pressure relief function cannot be tested at full pressure in an in-service test. So far, no damaging overpressurization of a reactor or a steam generator has occurred.

One reason for this type of degradation in the pressure relief function is deficiencies in the quality assurance programme. The quality assurance programme should ensure that pressure relief valves installed are correctly calibrated, set to the right trip point and equipped with correct springs.

Some valve designs have had a poor history of set-point drifts.

Safety significance

If spring actuated relief valves do not open at all or at a too high pressure, there is an increased risk for overpressurization of e.g. the reactor vessel or a steam generator. Similarly, valves which stick open or open at too low pressures can initiate transients.

Source of issue (check as appropriate)

- ____xx_____ operational experience
- ___________ deviation from current standards and practices
- ___________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

India

Proper procedures exist for testing the RVs. An incident of lifting of reactor RV at pressure lower than its setting was reported. Cause was identified to be a QA deficiency and corrective measures taken.

Japan

In Japan, valves are manufactured under a strict quality control system to minimize the manufacturing errors. In case of manufacturing error exists, steam tests and nitrogen gas tests are conducted in a pre-operational test to confirm the set pressure. Nitrogen gas test is conducted in a periodical inspection to confirm that the shift of the set pressure is within the allowable limit.

USA

This was evaluated as a generic safety issue and assigned a high priority.

ADDITIONAL SOURCE:

- NUREG-0933, Prioritization of Generic Safety Issues, USNRC, July 1991.
  Generic Safety Issue 165 - Spring actuated safety and relief valve reliability.

ISSUE TITLE: Water hammer in the feedwater line (PC 7)

ISSUE CLARIFICATION:

Description of issue

The issue was raised after the occurrence of various incidents of water hammer that involved steam generator feedrings and feedwater piping Incidents have been attributed to such causes as rapid condensation of steam pockets, and steam-driven slugs of water Most of the damage involved pipe hangers and restraints However, there have been several incidents which have resulted in piping and valve damage

Because of the continuing incidence of water hammer events, the number of phenomena and the potential safety significance of the systems involved, systematic review procedures should be developed to ensure that water hammer is given appropriate consideration in the design and in the review of operating reactors

Safety significance

Piping and valve damage resulting from a water hammer can lead to both a transient initiation and interference with the function of safety systems required to cope with the transient

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

India

Incident of water hammer have been observed in FW, RHR and ECC systems The corresponding valve seat leaks have been corrected Operating procedures were augmented to minimize water hammer

Japan

Excessive water hammer on the feedwater line has not been experienced so far in Japan In view of the USI A-1, however, J nozzles have been installed on the steam generator feedwater ring Operating procedures have been accordingly improved, and it has been reflected on the simulator training In addition, paying attention to the potential possibilities of water hammer of such area as the vicinity of main steam isolation valves, it is made as practice to take the water hammer load into design consideration

In BWRs, as examples, depending upon the system, layout and operating conditions. combination of the following measures are properly employed to minimize the possibility of occurring water hammer phenomena

- Proper ring arrangement to exclude negative pressure
- Installation of surge tank
- Installation of vent lines and bypass lines
- Installation of air chambers
- Venting from gas pocket prior to start up
- Warming up of lines before opening isolation valves
- Installation of steam traps

- Provide down slope in steam lines.
- Confirmation of proper opening/closing speed to avoid water hammer.
- Installation of proper support for the piping around the quick actuating valves.

As a practice, the effectiveness of above measures on preventing water hammer is confirmed through the system functional test during construction and start up test.

Korea, Republic of

- The utility addresses the issue of water hammer through a combination of design, operational, and testing considerations, such as designing for the proper routing and sloping of lines, providing adequate drainage and venting to protect against water or steam entrainment, and consideration in the design analysis of dynamic loads resulting from water hammer.
- Guideline should be provided for hot functional testing, as well as operating and maintenance procedures that require proper precautions to minimize the potential for water hammer.

Spain

Water hammer events have been considered in the design of the Spanish NPPs, and in particular in the PWR feedwater lines.

Several PWR plants have been provided with specific lines and an associated automatic logic to prevent the occurrence of water hammer in the Steam Generators feed-water lines. In other PWR plants, the operating procedures have been modified for the same purpose. New steam generators, used as replacement components in some plants, have a specific design against water hammer failures.

USA

Tables 3-1 and 3-2 in NUREG-0927, Rev. 1, identify water hammer causes and preventive measures (both design and plant operational related) for US BWR and PWR systems. Tables 1-1 and 1-2 in NUREG-0970, Rev. 1, provide a relative ranking of safety significance attached to water hammer occurrences in important systems for BWRs and PWRs. (These reports cover more than steam generator feedrings and feedwater piping). These insights identify potential weaknesses from a deterministic and operational point of view. A summary of the USNRC 1985 evaluation of the USI A-1 Water Hammer safety issue and concluding actions taken is contained in NUREG-0933, page 2.A.1-1.

ADDITIONAL SOURCES:

- NUREG-0970, Revision 1.
- NUREG-0933, A Prioritization of Generic Safety Issues, USNRC, December 1996.
- NUREG-0927, Revision 1, Evaluation of water hammer occurrences in nuclear power plants, US Nuclear Regulatory Commission (March 1984).
- NUREG-0606, Unresolved Safety Issues summary, US Nuclear Regulatory Commission (Latest edition).
- NUREG-0371, Task action plans for generic activities (Category A), US Nuclear Regulatory Commission (November 1978).
- USNRC Information Notice 91-50, Supplement 1, Water hammer events since 1991 (July 17, 1997).
- USNRC Information Notice 87-10, Supplement 1, Potential for water hammer during restart of residual heat removal pumps.

ISSUE TITLE: Steam generator overfill due to control system failure and combined primary and secondary blowdown (PC 8)

ISSUE CLARIFICATION:

Description of issue

At the beginning of the 80s, operators at NPPs became aware of the need to avoid overfilling steam generators and not operating steam systems with water accumulation.

This issue was a general potential safety problem of PWR's concerning event sequences with additional control system failures, i.e. in case of steam generator loss of coolant accidents.

Steam generator overfill and its consequences have received USNRC staff and industry attention because of the frequency and severity of overfill events and the insufficient development status of appropriate emergency operational procedures.

The NRC Office of Analysis and Evaluation of Operational Data produced a report with observations and recommendations concerning the problem of steam generator overfill and combined primary and secondary side blowdown. This report documents results of studies completed with experienced operational data and expressed safety concerns in the following areas:

- increased dead weight and potential seismic loads placed on the main steamline and its support should this line become flooded,
- the load placed on the main steamlines due to the potential for rapid collapse of steam voids resulting in water hammer,
- the potential for secondary safety valves sticking open following discharge of water or two-phase flow.
- the potential for rupture for weakened tubes in the once-through steam generator on Babcock and Wilcox-designed plants due to tensile loads caused by the rapid thermal shrinkage of the tubes relative to the steam generator shell.

Safety significance

After finishing the comprehensive measurement programme, described below, this issue was given a medium priority ranking and pursued by the staff. It was found that steam generator tube rupture and steam line overfill events pose a relatively low public risk. Comparable risk results for steam generator tube rupture were also published. Thus, this issue was resolved and no new requirements were established.

Source of issue (check as appropriate)

- ____xx_____ operational experience
- ___________ deviation from current standards and practices
- ___________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Korea, Republic of

- The KSNP has a main feedwater isolation system to protect the SGs from overfill which includes redundant remotely operated isolation valves in each main feedwater line to each SG. The valve actuation system is composed of redundant trains A and B, and each train's instrumentation and controls are physically and electrically separate from and independent of those of the other train.

- In a SBLOCA, high pressure safety injection is delivered at a pressure considerably above 1275 psia. The KSNP also incorporates a safety-grade Auxiliary Feedwater Systems (AFWS) which is automatically actuated by an auxiliary feedwater actuation system signal (AFAS) from the ESFAS. There is one AFAS for each SG, initiated by low SG water level in a 2-out-of-4 logic.
- An ESFAS high SG water level interlock will isolate AFW to preclude SG overfill. The RCS depressurization rate is manually controlled by the operator from the control room to prevent overcooling of the reactor vessel, by throttling the AFW and/or using the pressurizer sprays.
- In summary, consistent with the requirements and guidance of GL 89-19, the KSNP incorporates:
  - SG overfill protection, and
  - an automatically initiated safety-grade AFS.

Furthermore, a Tech. Spec. for verifying overfill protection availability, and emergency procedure guidelines for a SBLOCA, are established.

USA

In 1981, NRC has required all licensees of operating plants and holders of construction permits to determine which scenarios are credible for the plant and include in the overall training programme, plant-specific information stressing the importance of feedwater as well as the possible consequences of steam generator overfill. This information should be factored into the initial operator training programmes and the operators requalification programmes.

In order to provide an integrated work plan for the resolution of this issue in the following next ten years, the work scope involved the following main staff actions:

- survey the code requirements and industry practice for eddy current testing procedures and assess the capability of current methods to detect steam generator tube degradation,
- review the results and conclusions of studies on steam generator tube ruptures and propose partial specific modifications to the Standard Review Plan including tube integrity, operator action time, and offsite dose limits,
- reassess the problem concerning steam generator staff actions for potential inclusion in an integrated resolution; reassessment of radiological consequences, reevaluation of design basis steam generator tube rupture, supplemental tube inspections, integrity of steam generator tube sleeves, denting criteria, improved accident monitoring, reactor vessel inventory measurement, reactor coolant pump trip, control room design review, emergency operational procedures, organizational responses, and reactor coolant system pressure control,
- review the effects of water hammer, overfill, and water carryover on the secondary system and connecting systems and develop proposals for mitigating the consequences, consider the effects of sagging due to water weight, operability of valves, and other components when subjected to two-phase flow of liquid.

ADDITIONAL SOURCES:

- NUREG-1150, Severe accidents risks: An assessment for five US nuclear power plants", USNRC, (Volume 1) December 1990, (Volume 2) December 1990, (Volume 3) January 1991.
- NUREG-0844, USNRC integrated program for the resolution of unresolved Safety Issues A-3, A-4, A-5, regarding steam generator tube integrity, USNRC, September 1988.
- USNRC Report, AEOD (Office of Analysis and Evaluation of Operational Data), Observations and recommendations concerning the problem of steam generator overfill and combined primary and secondary side blowdown, December 17, 1980.

4.1.5. Safety systems (SS)

ISSUE TITLE: ECCS sump screen adequacy (SS 1)

ISSUE CLARIFICATION:

Description of issue

The containment is equipped with sumps to collect the water of the primary circuit and of the ECCS water storage tank after a LOCA in order to recirculate the water in the second phase of the accident. The openings of the containment bottom plate are covered with a screen which is intended to prevent debris penetration to the suction of the ECCS and containment spray system.

The thermal insulation used inside the containment and the total area of the screen above the sump/ECCS water storage tank together with dust and dirt that occur in containments form a combination that raises high safety concerns regarding the possibility of maintaining ECCS circulation after a medium or large LOCA. Operational experience based on recent events in Sweden and in the USA have demonstrated that even a relatively small amount of similar fibers can efficiently block a large screen area. Tests in Zaporozhe have demonstrated that a small amount of fibrous material can plug the ECCS heat exchangers. In addition, the sump screens must be designed and installed to ensure that the screening function is maintained (see also SM 7, Temporary installations and SM 6, Foreign material policy).

Safety significance

The insufficient design of thermal insulation of equipment and pipelines inside the containment can, under LOCA conditions, lead to a common mode failure by clogging the sump screens and/or the ECCS heat exchangers with a high risk of losing ECCS recirculation. In this situation, the function can be questionable (or disabled in extreme situations) for scenarios within the DB envelope.

The ECCS function can also be affected by inadequately screened debris.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

At the Kozloduy NPP, tests have been performed using the spray system. For these tests, particles and debris were carried by the system and reached the filtering system. The results of these tests have already led to a change in the design of the horizontal device at the entrance of the sump filtration system for avoiding their blockage.

Other measures will be taken like specific procedure to use trains of the safety injection. These measures need a complete evaluation before being installed.

Czech Republic

Temelín NPP: new meander networks with increased resistance to clogging is used. Tests have been prepared that prove insulation resistance (hardened, stitched mattress) against rupture.

France

2 events occurred in 1996: vinyl sheets used in containment works have been forgotten in the vicinity of the sump or fallen down into the sump, the consequence of which being a risk of blocking the RCCS sump.

Germany

In Germany, as a consequence of the incident in Barsebäck the potential effects of LOCA generated debris on the effectiveness of the ECCS was analysed thoroughly. As a result measures were mainly necessary in BWRs whereas in PWRs only minor changes were needed.
Measures at BWRs included:

- At some plants openings were cut into the ECCS strainers in the suppression pool and in the sump to prevent their clogging. Debris of any kind is then filtered from the water flow by suction strainers installed before the ECCS pumps outside the containment.
- In one plant the ECCS strainers in the sump were enlarged substantially.
- In most BWRs an additional close meshed grating, covering the whole cross-sectional area, was installed above the sump.

In addition, in some plants the amount of insulation material in the containment was reduced by removing insulation from small-diameter pipes as well as from pipes and components which are cold during normal operation.

India

Additional screens have been installed on the existing screen in the drywell to avoid major debris blocking the flow.

Russian Federation

At some WWER-1000 plants certain corrective measures have been taken to reduce the negative effects of damaged thermal insulation. A modified design of the filters inside the sump has been implemented at all operating plants.

Sweden

In Sweden, as a result of the incident in Barsebäck in 1992, modifications were made at all BWR plants. For the older plants, this consisted of a combination of a much bigger screen area and the capability for back-flush operation with change of insulation material. For the newer BWRs, which already had big screen areas, the capability for back-flush operation was improved.

Ukraine

Full scale operational tests on the existing primary circuit thermal insulation were performed at the Zaporozhe NPP Unit 5 and the South Ukraine NPP Unit 3, with a simulated LOCA condition of diameter 300 mm break and 8 m^3 thermal insulation material dispersed in the water at steam-water mixture discharge conditions. The tests lasted 8 hours. Based on the test results, compensatory measures will be implemented, and an emergency operating procedure has been prepared. A modified design of the filters inside the sump has been implemented at all operating plants.

For the Rovno NPP, the main points of the compensatory measures are as follows:

- Improving the screens and filters of the containment sump.
- In case of a LOCA event, after actuation of three trains of ECCS, switch one of them to a standby

mode in 15 minutes to keep it in a "clean" condition. At that point in time when the two low pressure heat exchangers are degraded or blocked by the heat insulation debris deposition on the heat transfer tube surface, the "clean" standby one is switched back into service. For the long term cooling of the core, a switchover from the low pressure heat exchanger to the spent fuel pool heat exchangers is done by valve realignment.

This compensatory measure has been implemented at Unit 3 for the two trains of ECCS, and the third one was expected to be completed in 1996 when I&C parts and cables are available.

For the Zaporozhe NPP, the 3 suction lines of the 3 trains of the low pressure injection system are connected to the containment sump. This sump is filled in normal operation with 630 m^3 of borated water. A test was performed on Units 5 and 6 during which the spray pumps and low pressure injection pumps worked 72 hours to demonstrate that the design has been well adapted. The loss of water due to trapping in certain areas of the containment after spraying situation is estimated to be about 200 m^3.However the test showed that the filters into the sump should be modified in order to reduce the number of the filter-grids damaged by loose parts and plugged by thermal insulation material.

USA

The USNRC requires that emergency core cooling systems (ECCS) be able to provide for long-term cooling capability following a LOCA. Unresolved Safety Issue A-43, "Containment Emergency Sump Performance" resulted in issuance of NUREG-0897 and Regulatory Guide 1.82, Rev. 1 in 1985. The Barsebäck-2 incident in 1992, and similar blockage occurrences in US BWRs in 1993, resulted in the USNRC issuing Generic Letter 96-03 which called for redesign and installation of much larger ECCS suction strainers. Regulatory Guide 1.82, Rev. 2, which provided specific guidance for the design of BWR ECCS suction strainers, was issued with Generic Letter 96-03. Installation of larger passive ECCS intake strainers in US BWRs is underway and is scheduled for completion in 1999. It has also become apparent that lessons learned from the BWR re-evaluations should be applied to a re-evaluation of PWR ECCS sump design, which the US has undertaken. In addition to thermal insulation debris, other materials such as aged or degraded containment surface coating materials (e.g., paints) which can result in additional particulate materials are being reviewed to assess potential impact on sump debris screens. In May 1997, the NRC published a Proposed Generic Letter for public comment entitled "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident because of Construction Deficiencies and Foreign Material in the Containment," which contains current NRC information on this subject.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- Knowledge base for emergency core cooling system recirculation reliability, NEA/CSNI/R(95)11, February 1996.
- USNRC RG 1.82, Revision 2, (May 1996) "Water sources for long term recirculation cooling following a loss-of-coolant accident."
- USNRC RG 1.82, Revision 1, (November 1985) "Water sources for long term recirculation cooling following a loss-of-coolant accident."
- NUREG-0897, Revision 1, USNRC, October 1985, "Containment emergency sump performance, Technical findings related to unresolved Safety Issues A-43.
- American Nuclear Safety Transactions, June 1997, Volume 76, p.285, "Design evolution of emergency core cooling intake systems for US light water nuclear power reactors."

- USNRC Bulletin 96-03, “Potential plugging of emergency core cooling suction strainers by debris in boiling water reactors”, May 6, 1996.
- USNRC Bulletin 95-02, Unexpected clogging of a residual heat removal (RHR) strainer while operating in a suppression pool cooling mode (October 17, 1995).
- USNRC Bulletin 93-02, Debris plugging of emergency core cooling functions strainer.
- USNRC Bulletin 93-02, Suppl. 1, Debris plugging of emergency core cooling function strainer (Feb. 18, 1994).
- Proposed Generic Letter: Potential for degradation of the emergency core cooling system and the containment spray system after a loss-of-coolant accident because of construction deficiencies and foreign material in the containment, *Federal Register*, Vol. 62, No. 92, Tuesday, May 13, 1997, pp. 26331-40.
- Recent USRNC Information Notices 95-47 Revision 1, 96-59, 97-27.

ISSUE TITLE: ECCS water storage tank and suction line integrity (SS 2) (WWER)

ISSUE CLARIFICATION:

Description of issue

The WWER-1000 plants are equipped with ECCS and containment spray systems that have a similar design basis and similar basic configuration as in western PWRs. These systems have 3 ´ 100% redundancy with the exception of the ECCS water storage tank which is common to all subsystems. The same tank serves as a containment sump. The tank is located under the containment and has open connections to the containment through the bottom plate. Each of the three safety system trains has one suction line from the ECCS water storage tank to the residual heat exchangers and further to the low pressure safety injection, high pressure safety injection and spray pumps. The suction line is equipped with one containment isolation valve.

The configuration of the ECCS water storage tank can lead to loss of coolant during an accident, if there is a passive single failure in the tank itself, or in any of the three suction lines between the tank and the containment isolation valve. Such a failure occurring during LOCA would lead to severe core damage and bypass of the containment.

Safety significance

The vulnerability of the ECCS water storage tank and suction lines to single failures affects level 3 of protection of defence in depth. In these LOCA scenarios, the safety function cooling the fuel would be questionable or even lost in extreme situations. However, the probability of an undetected leak before the accident or of a leak occurring during the recirculation phase is considered very low, since the suction lines are operated under conditions of low pressure and temperature.

Source of issue (check as appropriate)

- ___________ operational experience
- ____xx_____ deviation from current standards and practices
- ____xx_____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Czech Republic

For the Temelín NPP, double piping (pipe inside the pipe) will be installed in the line of the tank-sump up to first closing valve (included). Possible leakages at this exposed section are contained within the piping of bigger diameter located there.

Russian Federation, Ukraine and Bulgaria

Non-destructive testing of the vital piping sections are being conducted at all operating plants.

Accident analyses related to the loss of primary coolant out of the containment boundary have been included in the modernization programme of Ukraine.

For the Rovno NPP, annual non-destructive test on the integrity of the suction line has been performed for Unit 3. All welds on the suction line are inspectable. The same NDT inspection will also be implemented to Unit 4.

The Rovno NPP does not consider other potential solutions, since the suction line is at low pressure and with good material conditions.

For the Kozloduy NPP, to solve the problem of a direct bypass of the containment on the non-isolatable section of the suction line upstream of the isolation valves, the Kozloduy NPP has the intention of studying the possibility of installing a double envelope and to propose a solution for the problem related to the welding of the penetration through the containment.

ADDITIONAL SOURCE:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.

ISSUE TITLE: ECCS heat exchanger integrity (SS 3) (WWER)

ISSUE CLARIFICATION:

Description of issue

In WWERs, residual heat is removed in emergency conditions, and for some of them in shutdown conditions, through heat exchangers of the low pressure safety injection systems. These are cooled by the essential service water system which transfers the heat directly to a spray pond or to a lake, depending on the plant in question. There is no closed loop intermediate cooling system.

The spray ponds are isolated against the soil, but it is unavoidable that airborne dirt and/or biological matter enters the heat exchanger, resulting in fouling and consequent degradation of cooling. If there is a major blockage of the heat exchanger, the pressure difference over the separation plate between inlet and outlet chambers of cooling water could damage the heat exchanger.

A leak in the heat exchanger would cause non-borated water flow into the primary circuit, if the heat exchanger is used for residual heat removal or in the ECCS recirculation phase. If the primary pressure is higher than the essential service water system pressure, primary coolant could be released to the environment through the essential service water system, thus by-passing the containment.

Safety significance

The vulnerability of the ECCS heat exchangers to be damaged by several mechanisms affects defence-in-depth, furthermore diagnostic means are not available at all units to detect, at an early stage, any damage of the ECCS heat exchangers. Depending on the conceivable scenarios within the DBA envelope, each safety function can be impaired by diluting the primary coolant, by losing some primary water inventory, or by bypassing the third barrier.

Source of issue (check as appropriate)

- ____xx_____ operational experience
- ___________ deviation from current standards and practices
- ____xx_____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

For the Kozloduy NPP, the following measures have already been implemented with respect to fouling:

- measurement of pressure difference is done constantly (it has to be noted that this measurement cannot be considered a reliable measure to control the heat transfer capacity);
- control of the capability of the heat exchangers is performed during every startup of the unit when the ECCS is used for cooling the primary circuit;
- control, every month, of the capability of the heat exchangers during testing on the recirculation line to the ECCS tank.

Concerning the problem of boron dilution by the service water system in case of a leak in the heat exchanger, the following measures have been taken:

- the heat exchanger is completely isolated during normal operation of the plant. In case of a leak in the heat exchanger, the pressure on the side of the ECCS will increase suddenly. The operator is alerted in the main control room;
- a measurement of boron concentration is installed in each ECCS stream.

Concerning the possibility of having an active release in case of a leak in the heat exchanger when the ECCS is actuated, three measures have been installed on each stream and signals are sent to the main control room. They are activated when the activity level is higher than 5E-10 Ci/l. In this case the operator has to take measure to isolate the corresponding stream.

For the backup of the ECCS heat exchanger by the fuel pond cooling exchanger, an item is included in the modernization programme.

Czech Republic

For the Temelín NPP, the ECCS heat exchanger design was completely changed following negative operational experience from the other NPPs and a new design using different material will be used. The heat exchange of surface is made from double-phases steel and special tensometric measurement during start-up and initial years of operation together with well established operational testing will be used. Additional measures are presently being evaluated now. Problems with fouling are to be controlled by water chemistry control and treatment. There are special surveillance means to monitor ECCS heat exchanger operability and tightness by radiation and pressure difference measurement. Automatic locking and alarms are connected to this measurement. Modification of operational or emergency modes with higher pressure inside primary side of ECCS heat exchanger leading to containment bypass is evaluated but provisions shown above are adequate.

Ukraine

Technical and practical measures either have been implemented or are planned for the WWER-1000 NPPs. The measures include:

- introduction of ECCS heat exchanger fouling monitoring;
- introduction of a method to clean the ECCS heat exchanger from fouling;
- use of new chemical reagent to prevent fouling;
- application of reliable, high accuracy and on-line detection of the radioactivity in the service water;
- analysis of the efficiency and sufficiency of the existing boron monitor.

At Zaporozhe NPP Unit 6, there exists a connection between the ECCS exchangers and the fuel pond cooling exchangers which are located in the same place for each train. The similar characteristics of these two exchangers allow for backup. This measure increases the possibility to cool safely in case of leakage or plugging of the heat exchangers.

ADDITIONAL SOURCE:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.

ISSUE TITLE: Problems on the ECCS and containment spray switchover to recirculation (SS 4)

ISSUE CLARIFICATION:

Description of issue

ECCS and containment spray operation have two different phases in PWRs, i.e. injection and recirculation.

During the injection phase the two systems take suction from the refueling water storage tank (RWST); for the recirculation phase these two systems are realigned to take suction from the containment sumps.

Switchover from injection to recirculation phases involves realignment of several valves and may be accomplished by manual, automatic or semi-automatic operations. The logic has to be designed to deal with two contradictory objectives: avoiding spurious actuation (in case of loss of power for example) and insuring a high level of protection. All these considerations involve a complicated instrumentation and control.

Due to the complexity of the logic, a problem had been overlooked : the potential for pump damage in some conditions such as an automatic switching on with insufficient water inventory in RWST or in containment sumps.

Moreover, some of the logical sequences cannot be fully tested during commissioning and/or periodic test.

Safety significance

A common cause failure could affect all the pumps of the safety systems because of an insufficient protection in their instrumentation and control system. Besides, limitation in the scope of testing has a direct impact on the reliability of the ECCS and containment spray systems.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

For French units specific studies have been performed to analyse the consequences of control power losses and the exhaustivity of the tests (periodic or during commissioning).

Modifications have been implemented to improve the original design and, in particular, prevent pump damage in case of low level in the RWST.

Japan

The logics related to the operation of ECCS and the integrity of individual equipment (pump, valve, etc.) coping with an accident, are tested functionally during the annual inspection.

For example, the switching the intake of the pumps from RWST to sump are functionally tested through the operability of the relevant valves using sham signals and man action from the control room.

Korea, Republic of

The two modes of operation, injection and recirculation, are automatically or manually initiated by a Safety Injection Actuation Signals (SIAS) and a Recirculation Actuation Signal (RAS), respectively.

The RAS changes the operation mode of the safety injection system from injection with suction from the Refueling Water Tank (RWT) to recirculation with suction from the containment recirculation sump.

Operator action is required to close the RWT discharge valves after verifying that containment recirculation sump discharge valves open on a RAS signal.

The RAS is initiated by low RWT level. Initiation of recirculation is derived from four (4) independent RWT level sensors or manually from the control room. The automatic RAS requires a 7.9% RWT level indication from two (2) of four (4) channels.

USA

US plants are expected to have analysed this issue as part of the plant design process. This issue is also a typical area reviewed in multidiscipline design inspections which have been conducted at several plants.

ADDITIONAL SOURCE:

- USNRC Inspection Manual 93813, "Safety system functional inspections."

ISSUE TITLE: Diversion of recirculation water (holdups in containment) (SS 5)

ISSUE CLARIFICATION:

Description of issue

This concern relates to the diversion of recirculation water after an accident due to an inadequate systems configuration.

The system failures of concern in the recirculation mode are the containment spray recirculation and the emergency coolant recirculation systems. Water entrapment in the containment can happen anywhere from the spray nozzles to the recirculation sump.

These failures could be caused by the refueling canal drain valves being closed, or the fine mesh strainer left in the refueling pool during operation preventing the water spray from returning to the containment emergency sump.

This issue is linked to another generic issue dealing with temporary devices and foreign materials left in the circuit during operation (see SM 6 and SM 7).

Safety significance

If the volume of water held up in sumps, pools and canal is greater than the margin of excess water allotted in the design of recirculation systems, containment cooling and/or reactor cooling could be lost for some LOCA events.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

For French plants, an overall assessment of all the risks of water diversion or trapping following an accident has been performed. As a result, the following improvements have been or are being implemented:

- modification on the refueling pool drain valves;
- restart procedures following an outage have been improved to minimize the above risks;
- studies and tests are in progress in order to eliminate or reduce the risk of containment sump strainers clogging by debris.

Korea, Republic of

The design of the containment minimizes potential trapping of safety injection and containment spray water which would prevent the return of the water to the containment recirculation sump.

The reactor cavity and the incore instrument cavity are considered to be the only locations that trap significant quantities of water.

The maximum volume of water that could be trapped by these cavities is conservatively 133,000 gallons. The total volume of water available from the RCS, the safety injection tank, and the RWT is a minimum of 719,000 gallons. Hence, about 19% of the water is trapped and unavailable for circulation.

The drain from the refueling pool floor to the containment floor is locked open to preclude trapping of water. Plugging of the drain line is precluded by large diameter and by the fact that the entire drain line (both suction and discharge ends) is submerged during the post accident sump recirculation. Thus, a potential for both forward and reverse flow exists, ensuring that no water will be permanently contained in the refueling pool.

USA

As a result of a 1997 NRC inspection, multiple deficiencies were found at the D.C. Cook Nuclear Power Station, Units 1 and 2 (ice condenser containments) affecting the emergency core cooling and containment heat removal systems. During the recirculation phase following a LOCA, the lower containment spray nozzles would deliver water from the containment spray system to an annulus area beneath the ice condenser. The plant construction did not provide a flow path from this annulus area to the containment sump. The plant design requires a manual switchover of the emergency core cooling system pump suctions from the refueling water storage tank to the containment sump on low tank level. Because of flow bias errors and instrument uncertainties associated with the level instrument for the refueling water storage tank, switchover could have been performed before the assumed amount of water was available in the sump to support pump operation. As a result, the safety margin of these systems to perform their intended recirculation and containment heat removal safety functions following a postulated loss of coolant accident was significantly diminished.

ISSUE TITLE: Boron crystallization in systems (SS 6)

ISSUE CLARIFICATION:

Description of issue

For PWRs there have been incidents at operating reactor plants in which high boron concentration has caused boric acid to crystallize in the internals of vital safety related pumps and piping, thereby rendering those systems inoperable. The source of the high boron concentration was the boron injection tanks (BIT), which were originally incorporated into the Westinghouse designs as a means of mitigating steam line break events. To overcome the reactivity addition resulting from a rapid cooldown, a high concentration of boron in the form of boric acid was used in the injection tanks. This high concentration has led to maintenance and operational burdens to licensees because of the need to prevent boron precipitation.

Safety significance

The safety significance as mentioned above is the rendering of vital safety related pumps and pipes inoperable. This presents a common failure mode of the safety injection (SI)/charging pumps by either blockage of the piping leading to the pumps or the formation of the crystallized boric acid directly inside the pumps.

Source of issue (check as appropriate)
- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Japan

For the systems containing boron of high concentration, the following measures are taken to minimize the possibility of boron crystalization:

- installation of heat tracing;
- systems are located in the building with temperature control;
- provide continuous recirculation in the piping around boron injection tank.

USA

The potentially affected licensees were asked to reevaluate the need for maintaining high concentrations of boron in their respective boron injection tanks. This was based on improved analysis methods for calculating consequences of a steam line break which demonstrate that the negative reactivity needed to be added was lower than originally thought. Therefore the need for the highly concentrated boron injection tank has been reduced or eliminated. For the licensees that have demonstrated that there was no longer a need for the high concentrations, the staff would allow for the removal of the BIT or a reduction of the boron concentration. For the plants still required to provide high boric acid concentrations for safety injection, plant-specific procedures normally provide for flushing the SI system after every SI actuation to prevent boric acid precipitation in the piping and for periodic sampling of the SI system.

ADDITIONAL SOURCES:

- USNRC Generic Letter No. 85-16, High boron concentrations, dated August 23, 1985.
- IE Information Notice No. 86-63, Loss of safety injection capability, dated August 6, 1986.

ISSUE TITLE: Boron crystallization and dilution in the core in case of LOCAs (SS 7)

ISSUE CLARIFICATION:

Description of issue

In a LOCA, core cooling is performed by borated water injection coming from storage tanks or accumulators in the first phase. The second phase consists of recirculating the water from the containment sumps by safety injection and spray systems. The water is injected into the reactor coolant system, first into the cold legs then into the hot legs. For cold leg breaks, injecting into the cold legs may result in vaporizing the water in contact with a hot core and increasing the boron concentration in the reactor vessel. Consequently, boron could crystallize in the core, thus diluting the recirculating water.

To avoid this risk, the injection of water into the hot and cold legs has to be performed alternatively.

Switching the injection from cold legs to hot legs and vice-versa every day (the exact time has to be calculated depending on the plant characteristics) during the entire recirculation period complicates the operating procedures and lowers the reliability of the safety system.

Following a LOCA, boric acid solution is introduced into the reactor vessel by modes of injection. In the initial injection mode, borated water is provided from the accumulators from the refueling water storage tank and from the boron injection tank (Westinghouse only). After this initial period, which may last somewhere between 20-60 minutes, the Emergency Core Cooling System (ECCS) is realigned for the recirculation sump. It is recirculated from the sump to the reactor vessel and back to the sump through the break. A portion of the water introduced into the reactor vessel is converted into steam by the decay heat generated in the core. Since the steam contains virtually no impurities, the boric acid content in the water that was vaporized remains in the vessel.

Some analyses of certain events and sequences indicate that there is a potential for a rapid injection of unborated reactor coolant into a PWR reactor core. This rapid injection of unborated water may lead to fuel damage.

For a range of reactor coolant system (RCS) leak rates, the secondary system is needed to remove some fraction of core decay heat. During this period it is postulated that reflux condensation will occur in the steam generators and that unborated water will be condensed on the downflow side of the U-tubes, collecting in the reactor coolant pump (RCP) loop seals. This pure water is then available for rapid insertion into the core via subsequent RCP pump start or by natural circulation processes following refill of the RCS. The range of break sizes where this scenario is thought to be credible extends from about 0.5 to 2% of the cold leg flow area (roughly, the flow areas associated with a 2" and a 4" diameter pipe, respectively). For a 2" pipe break, it is postulated that the RCP loop seals can become filled with unborated water in about 24 minutes following the onset of reflux condensation. This allows enough time for the RCP loop seals to become filled with unborated water prior to accumulator injection which would occur 40 to 50 minutes into the event. For larger break size increases, latter reflux condensation would occur and the accumulator would inject early in the transient. For smaller sizes, the RCS will remain full and natural circulation will not be lost.

Safety significance

The boron crystallization in the core may lead to recriticality by dilution of the injected water. Repeated valve operations to cope with this risk might reduce the reliability of the system and thus question long term operation of the emergency core cooling system.

The safety significance of this event is the consequences of the continuous build (to a saturation level) of boron in a crystalline form in the reactor vessel. Unless dilution is provided, the concentration of the boric acid in the core will continue to increase. Without the dilution flow, the concentration of the boric acid will eventually reach the saturation limit, and any further increase in boric acid inventory will cause its precipitation. Boric acid deposited in the core may clog flow passages and seriously compromise the performance of the ECCS.

While the boron dilution scenario postulated is viable, an unborated water slug should circulate to the core after first being substantially mixed with the borated water that is present in the system. Research is needed to understand the complex thermal-hydraulic processes that are involved before any definitive conclusions can be made in this regard.

Source of issue (check as appropriate)

- ___________ operational experience
- ___________ deviation from current standards and practices
- ____xx_____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

Modifications are being implemented in all units to allow water injection simultaneously in hot and cold legs of the reactor coolant system. This results in the long term operation of the emergency core cooling system eliminating the risk of boron crystallization and without repeated valve operations.

In France high boron concentration tanks (21000 ppm) are installed only in the first subseries 900 MW PWR NPPs, issues have been solved by thermal insulation and by heat tracing and simultaneous injection in cold and hot legs. A new automatic boron injection system has been installed in 1300 MW and 1400 MW units, using a boron concentration of 7000 ppm, via reactor boron and water makeup system (RCV/REA).

Japan

In Japan, the PWRs are designed and operated in the following manner to ensure the safety in LOCA:

- Safety injection system are designed to connect in both hot leg and cold leg of the reactor coolant system.
- In the operational manual, it is required to prevent crystallization of boron in the core.
  (i) After long term of LOCA, it should be exchanged to operate hot leg injection from cold leg injection.
  (ii) Reactor coolant pumps should be restarted after LOCA in sub-cooled reactor coolant state from natural circulation flow.

USA

The NRC staff reviewed solicited proposals from the Vendors for preventing build up of boric acid concentration and arrived a the conclusion that the only way of assuring dilution flow through the core is to provide one of the following modes of operation for the ECCS:

- alternate injection to cold and hot legs;
- simultaneous injection to cold and hot legs;
- simultaneous cold leg injection and hot leg suction.

The safety analyses for most of US nuclear plants do not address explicitly by the above sequence of dilution events. The staff is continuing to study other possible sequences that could cause a similar unplanned injection of unborated water.

This accident scenario involves several assumptions regarding plant conditions and equipment configuration and, therefore, may not apply to a particular US plant. Training and procedures that emphasize the need at all times to ensure uniform boron concentration in the reactor coolant system at all times and the implementation of appropriate action in starting an idle reactor coolant loop can reduce the probability of occurrence of such an event.

An information notice (USNRC Information Notice 91-54) was issued September 6, 1991.

ADDITIONAL SOURCE:

- USNRC Information Notice 91-54, "Foreign experience regarding boron dilution" (September 6, 1991).

ISSUE TITLE: Accident management measures (SS 8)

ISSUE CLARIFICATION:

Description of issue

In order to cope with the design basis accidents (DBA), safety systems are installed which are reliable, redundant and, to a far-reaching extent, diverse and which perform their functions if the outside power supply fails. The efficiency and reliability of these systems is demonstrated in detail in the course of the licensing process and during subsequent operation. The concept of incident control has proven its worth.

In the course of time, e.g. following the TMI-accident, and further development of safety engineering, additional measures to control hypothetical system failures and combination of failures or to mitigate the consequences of such beyond DBAs were introduced as accident management measures. These primarily addressed the use of existing equipment in beyond DBA scenarios.

In some countries, further plant modifications resulted from enhanced regulatory requirements according to the development of the state of the art in science and technology of precautionary measures in safety related areas as e.g. fire protection, internal flooding and earthquake.

The consequences of hypothetical system failures and combinations of failures, which have not been taken into account explicitly when designing the plant, were and are also being investigated within the scope of safety studies, reactor research and risk studies. (See also AA 4, Need for severe accident analysis and MA 10, Adequacy of emergency operating procedures).

Safety significance

Accident management measures can help ensure an early control of the state of the plant and the retention of fuel and fission products in the reactor pressure vessel and in the primary circuit with a high degree of effectiveness even if the events exceed the design basis. In the absence of accident management measures, limitation of fission product release and the prevention of long-term contamination would not be achieved for certain low probability scenarios.

Source of issue (check as appropriate)
- __________ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

H: Beyond Design Basis Procedures
Those procedures are used in the frame of "supplementary operational conditions" which consider certain combination of events, the consequence of which, for similar frequencies as those of DBA, could lead to more severe radiological consequences for the environment.

H1: Procedures used in case of loss of ultimate heat sink or failure of heat transport systems designed to extract residual heat [for example SEC (essential service water system) or RRI (component cooling system) or pumping station]. If the primary circuit is closed, thermal energy is removed via SG (using the turbine-driven auxiliary feedwater pump, with suction on Auxiliary Feedwater tank and back-up with demineralized water tank), and discharge to atmosphere via the relief valves. If primary circuit is open, a make-up through normal or additional devices can be used, that procedures are designed to keep the reactor under control for about 4 days, that delay is estimated as sufficient to recover ultimate heat sink.

H2: Procedures used in case of loss of feedwater ARE-ASG (FWFCS (feed water flow control system) - AFWS (Auxiliary feedwater system). Residual heat is extracted using pressurizer relief valves and start up of safety injection and containment spray systems (feed and bleed process).

H3: Procedures used in case of black-out (total loss of external and internal electrical sources), in case of primary circuit closed, energy is extracted like for H1 procedure, a specific GUS (ultimate backup device): turbine generator (gas turbine) for 1300 and 1400 MW subseries or diesel for 900 MW subseries, (allows to maintain seal water injection to the primary coolant pump via the test pump, and necessary control for operators), in case of break on primary circuit makeup is realized using a specific motor-driven pump. (control in a safe state for about 10 hours).

H4: Mobil link EAS-ISBP (Containment spray system - Safety injection at low pressure) installed some days after a LOCA is used to insure a mutual backup. After some days back-up of exchangers and pumps can be realized through mobile equipments (see U3 procedure).

U: "Ultimate" procedures
These procedures and devices have been added to mitigate the radiological consequences of very unlikely accidents called "severe accidents."

U1: Ultimate procedure depending on the changes occurred in the core outlet temperature and the availability of SG, that procedure determine the most effective measures to be taken in connection with: SG utilization, safety injection, pressurizer relief valves and primary coolant pumps.

U2: Ultimate procedure used to containment leakage control, localization of leaks and possible subsequence reinjection of highly radioactive effluents. Automatic devices avoid to transfer these effluents towards normal treatment process when their activity is too high.

U3: This ultimate procedure consists in some pluggings of the drain network under the reactor pit to improve the basemat resistance to corium penetration (only for Cruas, due to a specific installation with seismic bearing pads).

U5: This ultimate procedure of depressurization and gaseous release filtration is used when a slow increasing of pressure occurs in the containment after a severe accident. To maintain containment integrity a prefilter with metallic medium (located in the containment) and a sand filter (external to the containment) are manually operated where necessary to mitigate radiological impact in the environment to an acceptable level.

Germany

The majority of the following listed modifications and backfitting measures (partially as accident management measures) were implemented in the past decade in German NPPs. Some of the measures are still under discussion. Especially for AM procedures, an emergency manual was prepared and implemented into the plant documents but separate from the operating manual.

For German PWRs:

The following measures have been implemented in some German PWRs:

- Improved electrical power supply, e.g. 3rd connection to offsite grid (underground cable), increased capacity of 220 V batteries, 2-4 h depletion time;
- Improved reactor protection system and control logic qualified for accident environmental conditions;
- Modifications in the reactor protection system to enable defeating interlocks or overriding protective trips

- High Pressure Safety Injection (HPSI) and Low Pressure Safety Injection (LPSI) from the containment sump into the reactor core;
- In addition to the auxiliary boration system, alternative injection with the operational volume control system from the RWST or the containment sump into the primary circuit;
- External event protected, diverse auxiliary / emergency feedwater system 4 x 100 %;
- Installation of Feed and Bleed at secondary side, a preventive AM, which could be applied e.g. to transients, SBLOCA, containment bypass sequences and event sequences with a complete loss of SG feedwater injection;
- Improved and qualified equipment on secondary side for fast cooldown functions. Break preclusion quality of the steam lines up to the steam safety valves, the steam relief station and the main steam isolation valve (MSIV);
- Installation of Feed and Bleed at primary side in addition to secondary F&B. Qualified equipment, e.g. the pressurizer safety valves and relief valves are designed for all 2-phase flow conditions. The pressurizer valves can be kept open at low coolant pressure;
- Installation of overpressure protection by equipment qualification, e.g. valves designed for anticipated transient without scram (ATWS) situations, coolant pressure control logic with separated limitation and control, addition isolation signal for the pressurizer relief valves with improved position indication;
- Improved fire protection measures regarding to computer based fire alarm, alarm identification and fire fighting in cable ducts;
- Improvements regarding containment bypass sequences (SGTR and V-sequences). In order to secure the isolation function of the primary side e.g. two redundant isolation valves are placed in lines, which are connected to primary circuit and penetrate the containment. In the event of increased pressure and activity the ventilation system is isolated. Filtered air is supplied to the control room;
- Maintaining the containment integrity by containment venting and hydrogen recombining by installation of catalytic recombiners for all German PWR-containments.

For German BWRs:

- Use of alternative RPV-Injection sources (preventive AM) and as backup for low pressure injection;
- Improved residual heat removal measures and electrical power supply;
- Improved reactor protection system and control logic;
- Verification of pressure limitation and depressurization of RPV by diversification of SRVs;
- Primary containment venting (PCV) for prevention of containment overpressure failure;
- Additional cooldown line for coping with unisolatable SLB (V-sequence) and against overfilling of main steam line;
- Additional fire protection measures, e.g. inertization of primary containment or wetwell.

Japan

The BDBA responses procedures developed reflecting the TMI-2 accident were adopted to the modification of the procedures. The results of examination performed on each plant after the Chernobyl accident were summarized by utilities into "Report on the examination of accident management." The report contains the necessary modifications of facilities as well. It is also incorporated into procedures for education and training.

Examples of Accident Management measures in Japan are as follows:

For PWR

(1) The measures to maintain reactor shutdown.
   - Emergency boron injection after the loss of reactor shutdown functions with core cooling by diversified secondary cooling system powered from modified non-safeguard buses.

(2) The measures to maintain core cooling.
   - Utilization of the turbine bypass system.

- Continuous injection by makeup of water resource and alternative recirculation.
- Cooling down and recirculation by utilizing non-safeguard facility.
- Containment cooling by natural convection.
- Alternative auxiliary component cooling.

(3) The measures to retain radioactive materials.
- Containment cooling by natural convection.
- Water injection into CV from raw water tank through spray ring header.
- Forced depressurization of reactor coolant system by means of power operated relief valves.
- Controlled hydrogen combustion.

(4) The measures to maintain the function related to safety support system
- Alternative auxiliary component cooling.
- Power supply from adjacent unit.

For BWR

(1) The measures for reactor shutdown.
- Alternative rod insertion (ARI).
- Recirculation pump trip (RPT).

(2) The measures for water injection into reactor/containment.
- Alternative water injection (the makeup water and fire protection systems).
- Automatic depressurization (automatic safety relief valve opening logic).

(3) The measures for heat removal.
- Hardened wet-well venting.
- Alternative heat removal using existing equipment.
- Recovery of failed functions of the residual heat removal system.

(4) Support system
- Emergency power utilization from adjacent unit.
- Recovery of failed functions of emergency diesel generators.

Slovakia

- Improved electrical supply
  - third connection of emergency busbars to the hydroplant
  - increased batteries capacity = 220V from 30 minutes to two hours
  - "complementary emergency source" underground cable supplied from neighbouring two units directly to the selected consumers
  - transportable DG to supply 0,4 kV consumer e.g. charging pump, or batter
  - DG cooling system independent of technical service water

- Primary heat sink
  - upgrading of pressurizer safety and relief valves system to cope with feed and bleed mode of operation

- Subcriticality
  - backup of charging pumps with additional power sources and borated water sources

- Secondary heat sink
  - external emergency feedwater system connected to the SG blow down system
  - installation of steam relief station and MSIVs on each SG steamline
  - emergency feedwater pump cooling system independent of technical service water
  - external feedwater source, e.g. fire truck

- I&C Modifications

- *Additional reactor scram signals*

- Additional ESFAS signals and logics

- Replacement of RPS and ESFAS systems detectors by seismic and environmentally qualified ones

- PAMS installation in the control room

- Emergency control room installation

- Radiation monitoring system in confinement for emergency conditions (environmental qualification and ranges)

- Teledozimetric system installed at the site and 30 km zone

- Fire protection zone
  - improvement and extension of fire detection system
  - additional automatic fire extinguishing system installed in all safety related cabinets and in control room (panels and consolas)

- Operating procedures and training
  - development of critical function status tried and function restoration procedures
  - appropriate procedures developed for every modification implemented
  - operating personnel training on usage of EOPs and equipment systems modified

ADDITIONAL SOURCES:

- E. Kersting, J. von Linden, D. Müller-Ecker, W. Werner: Safety analysis for boiling water reactors, GRS-98, July 1993.
- Deutsche Risikostudie Kernkraftwerke, (Phase B), Verlag TÜV Rheinland, Köln 1990.
- Deutsche Risikostudie Kernkraftwerke, (Phase A), Verlag TÜV Rheinland, Köln 1979.

ISSUE TITLE: Containment or confinement leakage from engineered safety features (ESF) systems during an accident (SS 9)

ISSUE CLARIFICATION:

Description of issue

Following a plant accident involving cladding failure, safety systems located outside the containment may have to function with large radioactive inventories in the fluid they process. The potential exists for leakages from systems such as the spray and the safety injection systems recirculating water for long term cooling, to the air, soil or auxiliary connected systems.

Any resultant releases are controlled (air) or collected (liquid) for transfer and clean-up through the radwaste system.

In these cases, one can consider that the third barrier is extended from the containment to the operating safety systems and their connected auxiliary systems.

During the TMI accident, releases from safety systems contaminated the premises housing the systems and led safety authorities in different countries to recommend specific measures to mitigate this issue.

Safety significance

The high activity of the potential leakages can make the treatment or filtration by the systems working in their design mode questionable ; moreover leaks through the connected auxiliary systems (air cooling or water cooling systems) can induce a release of radioactivity to the environment.

Source of issue (check as appropriate)

- ____xx____ operating experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

For French plants an overall assessment has been performed to check out the continuity of the third barrier and its potential extension. As a result two main modifications were implemented:

- To increase the leaktightness of the buildings housing the safety systems outside the containment (recirculation lines).
- To transfer highly radioactive leakages following an accident back into the reactor building.

In order to avoid contamination spreading, the air leaktightness of the rooms containing recirculating systems is controlled and safe (radiation-shielded) storage is provided. The leaktightness between the safety systems and their auxiliary systems is periodically tested.

Japan

The systems are so designed that radioactive materials are recovered into the container during a LOCA. The following design features are provided:

- The reactor containment is designed to maintain its integrity against the pressure and temperature raised by the energy of the primary coolant being discharged during a LOCA with postulating the most severe break. It is also designed that total leakage from the entire containment vessel including the airlocks and penetrations for cables and piping is kept below 0.1%/d of the inventory (PWRs) and 0.5%/d (BWRs). Thus, the integrity of the containment boundary is maintained.

- An annular clean-up system (PWR) and standby gas treatment system (BWR) are provided to remove iodine from the radioactive leakage gas through the penetrations.
- Reactor auxiliary building air cleanup system is provided for PWR to remove iodine from the radioactive leakage through ECCS pipings used during recirculating operation outside containment. For BWR such leakage is negligibly small.
- Both inside and outside the containment, automatic isolation valves or shut-off valves are provided on the following piping systems:
  - the piping being connected to the reactor coolant boundary;
  - the piping opening to the containment atmosphere;
  - piping which have the possibility of being damaged in case the primary coolant system fails.

ADDITIONAL SOURCE:

- USNRC Information Notice 96-13, Potential containment leak paths through hydrogen analyzers.

ISSUE TITLE: Steam generator safety valves performance at low pressure (SS 10) (WWER)

ISSUE CLARIFICATION:

Description of issue

At the present time, two types of valves are installed on each WWER steam generator line (2 safety valves, 1 relief valve). The safety valves on the WWER have the possibility of adjusting the secondary pressure from 84 bars to 40 bars. The relief valve can do it from 74 bars to 1 bar. The relief valves in WWER-440/213 NPPs cannot be used in all situation because they are located downstream of the main steam isolation valves (MSIV). The existing probabilistic studies have pointed out some transients for which, in case of a failure of the relief valves, it was necessary to be able to use the safety valves at low pressure to cool the primary circuit.

Safety significance

Decay heat removal by feeding the steam generators and relieving at the SG valves is the only safe procedure to cool the fuel on the secondary side. The inadequate design of the SG valves question the safety function in the long term in case of failure of all relief valves (BRU-A) which is a BDBA.

Source of issue (check as appropriate)

- ________ operational experience
- ________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

For the Kozloduy NPP, it is planned to perform studies with a view to finding other more effective types of safety valve which operates in a wider band of pressure change.

Ukraine

The modernization programme of WWER-1000/320 NPPs envisages either a modification of the steam generator safety valves or their replacement by more reliable ones.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).

ISSUE TITLE: Thermal shock or fatigue caused by cold emergency feedwater supply to steam generators (SS 11)

ISSUE CLARIFICATION:

Description of issue

The probability of injecting cold water into the steam generators in operation in the internal structures and connected pipings to prevent thermal shock or fatigue should be reduced by considerations in design.

The following events were reported:

(1) In WWER plants, cracks were found in feedwater pipings in the steam generators. These cracks were caused by thermal shock due to cold water injection into the pipe.
(2) In a couple of PWR plants in the US, cracks were found in the feedwater piping. In these cases, cold water had been accidentally injected by operator error.

If the normal feedwater is lost at a WWER-1000/320 plant, the auxiliary feedwater pumps would start from a low SG water level and the emergency feedwater pumps would start from a very low level. The auxiliary feedwater pumps do not have diesel backed power supply, and thus the loss of normal AC power supply leaves the emergency feedwater system as the only means for feedwater supply. The emergency feedwater (temperature around 20°C) is not heated, and therefore every startup of the emergency feedwater system involves a thermal shock to the SGs.

Safety significance

Damage of the internal structures might cause degradation of the steam generators and damage of connected pipings might cause coolant leak. Even if designs are made to cope with a limited number of thermal shocks, every provision should be taken to reduce or avoid them.

Source of issue (check as appropriate)

- ____xx____ event
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

At the Kozloduy NPP, measures have been taken to minimize the number of thermal shocks on the steam generators. To reduce the number of activations of the cold emergency feedwater supply to SGs, the pumps of the EFW system will be electrically backed up during startup and shutdown of the plant.

Czech Republic

At the Temelín NPP, the auxiliary feedwater pumps have the power supply backed up by diesel generators. Water inventory in the feedwater tank and turbine condenser is large (ca. 450 m^3). In addition, there is another inventory of demineralized water (ca.1000 m^3). The whole trains of demineralized water from demineralized water storage tanks (ca. 1000 m^3) into feedwater tanks, turbine condenser and emergency feedwater tanks have the power supply backed up by diesel generator. The use of the emergency feedwater system is very unlikely.

Germany

In the case of a failure of the main feedwater system, in new German PWRs the auxiliary feedwater pumps will start automatically and pump hot water from the feedwater storage tank to the steam generators at reduced reactor power. If the auxiliary feedwater system fails, the emergency system will start and feeds cold water into the steam generators. All feedwater systems are of high quality (e.g. redundant level control, auxiliary feedwater pumps supplied by emergency diesel generator), so that the probability for system failure and consequently for cold feedwater supply is low. For older plants the procedure is similar.

Japan

The auxiliary feedwater system is connected to emergency power supply in all PWR plants, and is used in upset and emergency conditions when normal makeup system could not be available due to normal power loss.

In order to minimize thermal shock, hot water from the deaerator is slowly supplied to the steam generator during plant start up or shutdown using the dedicated water filling system, normally in most plant. During abnormal operation such as loss of on-site power supply, the condensate tank is used as a water source to supply water continuously through the auxiliary water supply system avoiding intermittent flow. Thermal fatigue analysis has been perform on them to confirm that there are no problems under transient conditions through the plant life.

Ukraine

For the Rovno NPP, the number of allowed injections of cold water to the SG when the level is low has been increased to 30 times according to the design (previously 5). The corresponding logic will be modified, also considering the risk of spurious activation.

For the Zaporozhe NPP Units 5 and 6, the auxiliary feedwater pumps which suck hot water from deaerator are backed up by the additional diesel generator to avoid possible damage of steam generators by cold water from the emergency feedwater tanks in case of loss of normal AC power supply.

USA

The auxiliary feedwater and emergency feedwater systems are one and the same system in US PWR plants. Nearly all of them have an emergency diesel generator backed power supply for the event of a loss of offsite power. The auxiliary feedwater supply is connected to the main feedwater system in the proximity of the steam generators. The design analysis of the feedwater system assumes the feedwater from the condensate storage or other sources is about 40 degrees F and the main feedwater nozzle has a thermal sleeve to protect it from thermal shock. During startup from complete dry-out condition of a steam generator, operating procedures require an initial trickle feed to prevent thermal shock.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).

ISSUE TITLE: Emergency feedwater system reliability (SS 12)

ISSUE CLARIFICATION:

Description of issue

Heat removal from PWR plants following reactor trip and a loss of off-site power is accomplished by the operation of several systems, including the secondary system via the steam relief to the atmosphere. The auxiliary (emergency) feedwater system (AFW) functions as a safety system because it is the only source of makeup water to the steam generators for decay heat removal when the main feedwater systems becomes inoperable.

On June 9, 1985, Davis-Besse NPP had a partial loss of feedwater while operating at 90% power. Following a reactor trip, the loss of all feedwater occurred. Operating experience and studies performed by NRC staff and the nuclear industry indicated that AFW systems failed at a relatively high rate, for instance:

- loss of all AFW due to common-mode failure of AFW pump discharge isolation valves in closed position;
- excessive delay in recovery of AFW due to difficulty in restarting AFW pump steam driven turbines, if turbines are tripped;
- interruption of auxiliary feedwater flow due to failures in steam and feed line break accident mitigation features.

The initiating events and sequences for the total loss of feedwater are plant-specific, and thus, individual plant assessment is needed.

Some PWR plants currently in operation, including WWER plants, do not have a systematic assessment on this event, and therefore, there is no basis to judge the adequacy of emergency procedures, operator training and necessary hardware upgrading.

Safety significance

Total loss of feedwater which already happened in an operating plant implies the loss of heat removal safety function, if no other measures is available. The event significantly contributes to the core-melt frequency.

Source of issue (check as appropriate)
- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Germany

In German PWRs the emergency feedwater system gained special attention already previous to the mentioned event. Probabilistic investigations had shown the particular significance of the feedwater supply reliability to cope with transients.

Therefore most plants are equipped with a four-train independent emergency feedwater system powered by four dedicated Diesel generators. In addition, these plants dispose of an auxiliary feedwater system (start-up and shut-down system) equipped with two pumps supplied by emergency power supply. As a result there are two diverse, redundant auxiliary/emergency feedwater systems available.

In older PWRs additional independent emergency feedwater systems have been backfitted which in general consist of 2x100% trains. Moreover accident management measures have been prepared at the German PWRs to cope with the loss of all feedwater supply systems. These measures aim at re-establishing a feedwater supply by using all available measures, e.g. the content of the feedwater tank or firewater pumps.

USA

The NRC staff performed plant-specific reviews of the reliability of the AFW systems in some plants, including assessments of the system design, operating experience, and emergency procedures. From these reviews, the staff determined whether additional means of secondary decay heat removal capability was necessary. No further hardware modification was determined to be required for the 1 B&W plant and the 2 Westinghouse plants on the basis of the startup feedwater pump and AFW system sharing capability, respectively. The licensees for 3 of the 4 other plants committed to install additional means of secondary decay heat removal, thereby resolving the issue. The staff issued a plant-specific backfit analysis for the other plant requiring the addition of a third train of secondary decay heat removal.

The USNRC staff determined that all PWRs in the US should meet the reliability criterion (10^{-4} to 10^{-5} unavailability/demand) for the auxiliary (emergency) feedwater system. A reliability analysis and/or any necessary system modifications and procedural or maintenance changes should be reviewed by USNRC staff.

Most Westinghouse PWRs adopted the primary side feed and bleed cooling as a backup to the total loss *of feedwater event.*

ADDITIONAL SOURCES:

- Gesellschaft für Reaktorsicherheit (GRS) mbH: Deutsche Risikostudie Kernkraftwerke, Phase B, Verlag TÜV Rheinland, Köln 1990.
- NUREG-1275, Vol. 10, Operating experience feedback report - Reliability of steam turbine-Driven standby pumps, October 1994.
- NUREG-0933, A prioritization of Generic Issues, July 1995.
  Generic Issue 124: Auxiliary feedwater system reliability, Revision 3.
- NUREG-0800, "Standard review plan", US Nuclear Regulatory Commission.
- NUREG-0737, Clarification of TMI action plan requirements, November, 1980.
- NUREG-0635, Generic assessment of small break loss-of-coolant accidents in combustion engineering designed operating plants, January 1980.
- NUREG-0611, Generic evaluation of feedwater transients and small-break loss-of-coolant accidents in Westinghouse designed operating plants, January 1980.
- Memorandum for H. Thompson from D. Grutchfield, "Potential immediate generic actions as a result of the Davis-Besse Event of June 9, 1985, US Nuclear Regulatory Commission, 1985.
- USNRC Generic Letter 90-04, Request for information on the status of licensee implementation of Generic Safety Issues resolved with imposition of requirements or corrective actions, April 25, 1990.
- USNRC Information Notice 93-05, Repetitive overspeed tripping of turbine driven auxiliary feedwater pumps, July 9, 1993.
- USNRC Information Notice 88-09, Reduced reliability of steam driven auxiliary feedwater pumps caused by instability of Woodward PG-PL Type Governors, March 18, 1988.

ISSUE TITLE: Need for hydrogen control measures during design basis accidents (DBA) (SS 13)

ISSUE CLARIFICATION:

Description of issue

Following a LOCA in a LWR, combustible gases, principally hydrogen, may accumulate inside the reactor containment as a result of:

- metal-water reaction involving the fuel element cladding;
- the radiolytic decomposition of the water in the reactor core and the containment sump;
- the corrosion of certain construction materials by the spray solution;
- any synergistic chemical, thermal and radiolytic effects of post-accident environmental conditions on containment protective coatings and electric cable insulation.

The hydrogen generation can lead to combustible gas mixtures. Inadvertent, random ignitions occurring in such a process may cause uncontrolled deflagrations, which could lead to high flame speeds and corresponding high quasi-static pressure loads for containment structures or even to deflagration-detonation transition phenomena with corresponding dynamic loads. These phenomena may cause, e.g. damage to compartment walls, missile effects or local leakages in the containment shell. Such effects may only occur in case of more than 4 % volume of hydrogen in the atmosphere and without high steam content (steam-inertization).

In general, for typical PWRs with large containment free volume, calculations demonstrate that less than 4 % volume of hydrogen is generated during a controlled large LOCA with good air-mixing conditions.

Safety significance

Insufficient hydrogen removal systems for use during DBA scenarios may seriously affect the ability to avert damage to the containment or confinement function.

Source of issue (check as appropriate)

- __________ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Czech Republic

For the Temelín NPP, the hydrogen burning and monitoring system has already been implemented into the project. Indicators for post-accidental hydrogen monitoring system will be supplied by OLDHAM of WEC within its scope of supply.

Passive recombiners from Siemens will be used for burning; they meet requirements on usage at the Temelín NPP (for example: Reliable operation of equipment in terms of planned LOCA when hydrogen concentration is up to 10 percent by volume in the course of at least 100 days since the beginning of accident; seismic resistance equals to MPZ; life is 30 years).

France

For the whole French units 3 thermal recombiners are available to be used in case of DBA (2 for redundancy + one recombiner assumed to be unavailable for maintenance). A delay of 3 days maximum is necessary to intervene on any French site, while about a week has been calculated to reach critical hydrogen concentration levels. They are stored on a French NPP site. The problem of hydrogen for BDBA is currently assessed and, depending on the conclusions of the programme of tests carried out, could induce new capacities to mitigate risks linked with hydrogen.

Germany

Since 1981, there are safety rules in Germany which require the following measures for limiting the hydrogen concentration in PWRs in case of design basis measures:

- calculations of the hydrogen source rate taking into account certain instructions regarding net production rates for core and sump radiolyses, effective decay power of the core and sump, radiolysis in the fuel storage pool, metal-water reaction in the core and other metal water reactions,
- if the calculations reveal that the hydrogen concentration can reach values above the ignition limit in certain areas of the containment vessel, active features shall be provided which will ensure sufficient forced flow mixing of the containment vessel atmosphere,
- if the calculations of the integral hydrogen concentration reveals that a volume concentration of 4 % cannot be precluded, passive features, e.g. catalytic recombiner, with reliable availability and function shall be provided,
- installation of system for monitoring the local and temporal distribution of hydrogen in the containment atmosphere and for measuring the activity of the gas.

In typical German PWRs, two catalytic recombiners are installed and located at the top of the steam generator houses.

Japan

In Japan, the "Guide for safety design of light water nuclear power reactor facilities" requires the systems to control containment vessel atmosphere that "the flammable gas concentration control system shall be designed to be capable of controlling the concentration of hydrogen or oxygen present in the reactor containment in case of the postulated events for reactor containment design, thereby maintaining the integrity of the containment facility" (Guide 33 (2)).

In BWRs, design consideration are made such that the inert gas inside the containment vessel and installation of flammable gas treatment facilities allow to suppress hydrogen concentration not reach the flammable conditions even after a LOCA.

It has been also confirmed through a safety design evaluation that the design mentioned above does not allow hydrogen to reach the flammable conditions even after a LOCA.

At PWRs with large containment vessel, generation of hydrogen during a LOCA has been evaluated through the safety evaluation and confirmed that the hydrogen concentration is sufficiently low even after an accident.

At PWRs with ice condenser type containment vessel, design considerations are made such that the installation of flammable gas (electrical) recombiners allow to suppress hydrogen concentration not to reach the flammable condition even after a LOCA.

It has been also confirmed through a safety design evaluation that the design mentioned above does not allow hydrogen to reach the flammable condition even after a LOCA.

Korea, Republic of

- The containment hydrogen recombiner system (CHRS) consists of two fully redundant 100% capacity and independent loops each with a hydrogen recombiner and associated piping connecting to the containment atmosphere.

- The CHRS components are designed to withstand LOCA containment pressure and temperature.
- The CHRS prevents hydrogen combustion during and after a design basis accident by ensuring that the hydrogen concentration in the containment never reaches 4%, its lower flammability limit.

Russian Federation and Ukraine

The design of the hydrogen removal system has been carried out for Russian and Ukrainian plants. The hydrogen-oxygen recombiner elements are being tested.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- RSK Leitlinien für Druckwasserreaktoren, 3. Ausgabe, 14. Oktober 1981.

ISSUE TITLE: Overfill into the main steam lines in BWRs (SS 14)

ISSUE CLARIFICATION:

Description of issue

Water in the steam lines in a BWR may degrade the pressure relief function. The probability for getting water in the steam lines are relatively high for some occurrences. There are two possibilities to overcome this problem: to verify the capability to reduce the reactor pressure with existing pressure relief valves even if there is water in the steam lines or to install qualified valves designed with this capability.

This problem has not been analysed in safety reports. Recent events which could have led to situations with water in the steam lines have brought this matter to attention.

Safety significance

If the steam lines are filled with water during normal operation or during a transient and no possibility exists to handle this situation, the risk for over pressurization of primary systems is significant.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Germany and Sweden

Pressure relief functions with capability to blow water have been installed at some German plants and have been installed at all three units in Forsmark, Sweden and will be considered in the modernization of other plants in Sweden.

USA

US BWRs have level 8 trips to protect the reactor from getting water in the steamlines, into turbine-powered high-pressure coolant injection (HPCI) turbine, reactor feedwater pump turbines and the main turbine. The SRV's are not designed to relieve liquid as presently configured.

ADDITIONAL SOURCE:

- E. Kersting, J. von Linden, D. Müller-Ecker, W. Werner: Safety analysis for boiling water reactors, GRS-98, July 1993.

ISSUE TITLE: Containment isolation of lines containing high activity fluids (SS 15)

ISSUE CLARIFICATION:

Description of issue

All accidents or incidents affecting fuel rod cladding integrity without any increase of the containment pressure can lead to high activity fluids being transferred outside the containment. Accidents such as TMI-2 or hypothetical accidents, such as rod ejection, involving cladding failure before energy release into the reactor building do not trigger containment isolation immediately and lead to a transfer of contaminated fluid towards outside systems.

Safety significance

Following an accident, auxiliary systems may have to deal with high radioactive water and the potential exists for an unexpected release to the environment. In addition, waste treatment systems may have to process beyond design radioactive releases.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

On French plants the lines containing primary coolant are automatically isolated when leaving the containment, in case primary coolant radioactivity exceeding a predefined limit is detected.

ISSUE TITLE: Reliability of the motor-operated valves in safety systems (SS 16)

ISSUE CLARIFICATION:

Description of issue

The operating experience of nuclear power plants indicates that a number of valves, or valve operators, fail to operate as specified in the technical specification either under test conditions or on demand.

Gate valves, especially, are used in a variety of applications in the safety systems and may be required to open during or immediately following the postulated design basis events ; moreover, the events which most severely challenge plant safety usually involve the most rapid system cooldown and depressurization rates and therefore the largest pressure differentials in and around these valves. Accordingly, pressure locking and thermal bindings represent potential common cause failure modes on gate valves ; operating experience also indicates that routine cycling, preoperational testing and surveillance testing may not provide a reliable means of ensuring valve operability during all transient or accident conditions.

The malfunction of valves can be due also to improper switch settings or underestimating thrust/torque requirements, or overestimating motor actuator output, and sometimes two or more of these causes simultaneously.

Reactor operating experience indicates that testing under static conditions. The safety authorities require the utilities to establish programmes to ensure the operability of MOVs in safety related systems to perform their safety function. A combination of design reviews, improved surveillance/maintenance programmes and valve testing have to be used to address this issue. More generally, a test capability to represent design basis conditions and links between preoperational tests, periodic tests and factory tests have to be considered and test procedures have to be improved to take the above developments into account.

Safety significance

Malfunctioning of power operated valves could create unacceptable situations during accidents and contributes to the risk associated with postulated core-melt accident sequences.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

Regarding underestimating thrust/torque requirements, French nuclear power plants prepared procedures to ensure that correct switch settings are maintained throughout the life of the plant and adjust switch settings when necessary.

Regarding pressure locking and thermal binding, the nuclear power plants have evaluated potential for pressure locking and thermal binding of gate valves and have taken action (modifications) to ensure that these phenomena do not affect the capability of power-operated gate valves to perform their safety related functions.

Germany

The mentioned problems with motor operated valves have also been encountered in Germany. Therefore dedicated measures have been initiated. The determination of the correct thrust/torque requirements of valve actuators and the improvement of gate valve reliability is subject of an ongoing programme. The adopted concept consists of the development of a well accepted calculation method, a review of the valve construction and improved maintenance procedures. To prevent pressure locking of gate valves technical modifications such as the drilling of a small hole into one of the gate valve plates have been introduced.

Japan

Motor-operated valves are subjected to the inspection at the factory before shipping to confirm their operation though closing and opening tests under the design pressure conditions.

Also, during in service inspection, they are tested under conditions as close as possible to demand conditions, for instance, by opening the outlet valve of SI pump in shut off operation with the witness of the official inspectors.

From 1979 though 1983, flow blocking tests were conducted on the safety system motor-operated gate valve of the same design and production as the operational plants to confirm that it opens and closes as required. Today, the fundamentals of the design are still maintained.

Korea, Republic of

- Regulatory body requested that NPPs ensure the capability of MOVs in safety related system by
  - reviewing MOV design bases
  - verifying MOV switch setting
  - in-situ testing MOVs
- Commitment is reached to complete this issue within 5 to 8 years by the utility
- Also, regulatory body requested that NPPs ensure reliability of the safety related power-operated Gate Valves that are susceptible to pressure locking or thermal binding.

Sweden

The programme consists of baseline measurements and in-plant diagnostic testing of valves and actuators. The Forsmark Kraftgrupp AB valve diagnosis concept uses the active power and operating time.

Continuous valve performance is based on long-term valve quality, which is achieved through a timely comprehensive valve maintenance programme. The maintenance programme is based on both performance experience (preventive maintenance) and expected behaviour (predictive maintenance). The interval between overhauls of safety related MOVs is 4 years.

Preventive maintenance establishes maintenance activities on equipment independent of its actual condition, but based on general past experience. Predictive maintenance aims at considering the specific equipment condition (e.g. ageing or deterioration) to anticipate repair or replacement needs.

Once baseline diagnostic measurements of the valve main parameters (e.g. active power, operating time, torque, stem thrust and switch settings) have been made, it is possible to detect long-term deviations during lifetime. Additional valve parameters are also measured and compared with the baseline measurements to localize operational irregularities.

If a deviation is found for an individual valve, in-situ testing is done during a plant shutdown. The information gained from in-situ testing is analysed and then used in maintenance programmes.

The test result is automatically collected and stored in a computer based documentation system (as finger prints) during test performance. The system allows data to be compared and trends to be evaluated.

USA

On June 28, 1989, the US Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 89-10, "Safety related Motor-Operated Valve Testing and Surveillance," as a result of problems with the performance of motor-operated valves (MOVs) in US nuclear power plants. In GL 89-10, the NRC requested that nuclear power plants ensure the capability of MOVs in safety related systems by reviewing MOV design bases, verifying MOV switch settings initially and periodically, testing MOVs under design basis conditions where practicable, improving evaluations of MOV failures and necessary corrective action, and trending MOV problems. The NRC requested that plants complete the GL 89-10 programme within approximately three refueling outages or five years from the issuance of the generic letter. Most US nuclear power plants have notified the NRC of the completion of their programmes to verify the design basis capability of safety related MOVs as requested in GL 89-10. The NRC is completing its review of GL 89-10 programmes at US nuclear power plants.

On August 17, 1995, the NRC issued GL 95-07, "Pressure Locking and Thermal Binding of Safety related Power-Operated Gate Valves," requesting that US nuclear power plants ensure that safety related power-operated gate valves that are susceptible to pressure locking or thermal binding are capable of performing their safety functions. US nuclear power plants have responded to GL 95-07. The NRC is reviewing the submittals from nuclear power plants describing their response to GL 95-07.

ADDITIONAL SOURCES:

- K. Kotthof, H. Liemersdorf: “Operating experience with motor operated valves in the FRG”, Joint NEA/IAEA Specialist Meeting on Motor operated valve issues, Paris, April 1994.
- USNRC Bulletin 85-03 (November 15, 1985), "Motor operated valve common modefailures during plant transients due to improper switch settings."
- Supplement 1 to USNRC Bulletin 85-03 (April 27, 1988).
- USNRC Generic Letter 96-05, Periodic verification of design basis capability of safety related motor operated valves.
- USNRC Generic Letter 95-07 (August 17, 1995), "Pressure locking and thermal binding of safety related power operated gate valves."
- USNRC Generic Letter 89-10 (June 28, 1989), "Safety related motor-operated valve testing and surveillance."
- Supplement 1 to USNRC Generic Letter 89-10 (June 13, 1990), "Results of the public workshops."
- Supplement 2 to USNRC Generic Letter 89-10 (August 3, 1990), "Availability of program descriptions."
- Supplement 3 to USNRC Generic Letter 89-10 (October 25, 1990), "Consideration of the results of NRC-sponsored tests of motor operated valves."
- Supplement 4 to USNRC Generic Letter 89-10 (February 12, 1992), "Consideration of valve mispositioning in boiling water reactors."
- Supplement 5 to USNRC Generic Letter 89-10 (June 28, 1993), "Inaccuracy of motor operated valve diagnostic equipment."
- Supplement 6 to USNRC Generic Letter 89-10 (March 8, 1994), "Information on schedule and grouping, and staff responses to additional public questions."
- Supplement 7 to USNRC Generic Letter 89-10 (January 24, 1996), "Valve mispositioning in pressurized-water reactors."
- USNRC Information Notices on MOVS: IN 89-61, IN 89-88, IN 90-21, IN 90-37, IN 90-40, IN 90-72, IN 91-58, IN 91-61, IN 92-17, IN 92-23, IN 92-26. IN 92-41, IN 92-59, IN 92-70, IN 92-83, IN 93-01, IN 93-37, IN 93-42, IN 93-54, IN 93-74, IN 93-88, IN 93-97, IN 93-98, IN 94-10, IN 94-18, IN 94-49, IN 94-50, IN 94-67, IN 94-69, IN 94-83, IN 95-14, IN 95-18, IN 95-18 Supplement 1, IN 95-30, IN 95-31, IN 96-08, IN 96-27, IN-96-30, IN 96-48, IN 97-07, IN 97-32, IN 97-76.

ISSUE TITLE: Reliability and mechanical failure of safety related check valves (SS 17)

ISSUE CLARIFICATION:

Description of issue

Check valve operating problems have resulted in significant transients, increased costs, and decreased system availability. The problems have included excessive wear and tear of internal parts, loss of locking devices and valve discs, sticking valves, and seat leakage. Also failure of check valves have occurred from stress corrosion cracking, corrosion and erosion.

The problems with check valves have been caused by weakness in several activities relative to their performance. Check valve design has been faulty in some instances and has led to failure of internal parts. Application of the valve such as sizing, installation, and location may not be appropriate for system conditions. Because of improper categorization with respect to maintenance, some check valves important to safety have not been subject to routine testing with respect to each of their safety functions. Also, the inservice tests required by NRC regulations through reference to the ASME Boiler and Pressure Vessel Code are not fully sufficient to predict or to indicate valve degradation. Further, inadequate procedures have led to improper valve assembly.

Safety significance

Malfunctioning of safety related check valves could create unacceptable situations during accidents and contributes to the risk associated with postulated core-melt accident sequences.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

USA

As a result of the continuing problems with check valve performance, the USNRC has issued several information notices and reports.

ADDITIONAL SOURCES:

- NUREG/CR-5944, ORNL-6734, A characterization of check valve degradation and failure experience in the nuclear power industry.
- NUREG/CR-5944, ORNL-6734/V2, Vol. 2, A characterization of check valve degradation and failure experience in the nuclear power industry.
- NUREG-1352, Action plans for motor operated valves and check valves, June 1990.
- USNRC Information Notice 97-31.

ISSUE TITLE: Potential failure of the scram system due to loss of discharge volume (SS 18) (BWR)

ISSUE CLARIFICATION:

Description of issue

In 1980, following a manual scram actuation at Browns Ferry, Unit 3, 76 of 185 control rods failed to insert fully. A second manual scram was initiated and all partially inserted rods were observed to drive inward, but 59 remained partially withdrawn. A third manual scram was initiated and 47 rods remained partially withdrawn. An automatic scram occurred and all rods inserted fully. An additional manual scram would probably have produced the same result.

The BWR control rod drives (CRDs), which insert and withdraw the attached control rods are essentially water-driven hydraulic pistons. To accomplish a scram, water pressure is applied to the bottom side of the piston by opening a scram inlet valve; a scram outlet valve to relieve water pressure above the piston and the rods are rapidly driven up into the reactor core. Water discharged from 185 individual CRDs is collected in two separate headers consisting of a series of interconnected 6 inch diameter pipes called the scram discharge volume (SDV). Each SDV is designed to be continually drained during normal operation and ready to receive the scram discharge water when a scram occurs. The SDV was apparently partially full of water at the time of the event leaving insufficient room for the discharge water. Upon scram actuation, the CRDs drive the rods into the core until the pressure equalized on each side of the piston and the rods stopped inserting.

Following each scram actuation, the system was electrically reset allowing water to drain from the SDV, permitting the rods to insert further with each scram attempt. Sufficient water was finally drained from the SDV to allow the rods to insert fully on the forth scram signal. It is postulated that either the drain piping was plugged or inadequate venting of the SDV prevented the water from draining from the SDV following a previous scram.

When a BWR is not in a scrammed state, the scram valves are held closed by control air pressure and reactor coolant is retained on the upstream side of the closed valves. In this state, the scram valves perform reactor coolant boundary and primary isolation functions. During and immediately following a scram the SDV system assumes a reactor boundary function and a primary containment isolation function. It is during this fully pressurized state of the SDV system that there exist a potential safety concern associated with a break in the SDV system piping.

Safety significance

The full availability of the scram discharge volume is necessary to ensure complete insertion of the control rods on scram actuation. Its loss may jeopardize the safety function controlling the power.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

USA

On July 3, 1980, IE bulletin No. 80-17 was issued requiring all BWRs to conduct prompt and periodic inspections of the SDV; perform two reactor scrams within 20 days while monitoring pertinent parameters to further confirm operability; review emergency procedures to assure pertinent requirements are included; and conduct additional training to acquaint operating personnel with this type of problem. On July 18, 1980, supplement 1 to bulletin 80-17 was issued. This supplement required an analysis of the "as built" SDV; revised procedures on initiation of the Standby Liquid Control System (SLCS); specifying in operating procedures action to be taken if water is found in the SDV; daily monitoring of the SDV until a continuous monitor can be installed and studying of designs to improve the venting of the SDV.

ADDITIONAL SOURCES:

- Operation and maintenance instructions control rod drive system for Browns Ferry nuclear plant, GER-9585/9586, June 1971.
- Safety concerns associated with pipe breaks in the BWR scram system, Office for Analysis and Evaluation of Operational Data, USNRC.
- Report on the Browns Ferry 3 partial failure to scram event on June 28, 1980, July 30, 1980, Office for Analysis and Evaluation of Operational Data, USNRC.
- Report on the interim equipment and procedures at Browns Ferry to detect water in the scram discharge volume, September 1980, Office for Analysis and Evaluation of Operational Data, USNRC.

ISSUE TITLE: Need for assurance of ultimate heat sink (SS 19)

ISSUE CLARIFICATION:

Description of issue

Heat sink could be defined as the complex of sources of service water necessary to operate, shutdown and cool-down a nuclear plant safely. The issue is to deal with the reliability of the sources themselves such as rivers, lakes, ponds and the reliability of systems and components intended to transfer heat from safety related systems to the cold sources. The ultimate heat sink must be shown to be capable of dissipating the heat following normal or abnormal situations, under unfavorable meteorological conditions, for periods long enough to guarantee the safety of the plant.

In a comprehensive USNRC review and evaluation of operating experience related to service water systems (ESW) (NUREG-1275 Volume 3), a total of 980 operational events involving the ESW system were identified, of which, 12 resulted in complete loss of the ESW system. The causes of failure and degradation included: (1) various fouling mechanisms (sediment deposition, biofouling, corrosion and erosion, foreign material and debris intrusion); (2) ice effects; (3) single failures and other design deficiencies; (4) flooding; (5) multiple equipment failures; and (6) personnel and procedural errors.

The initiating events and sequences for the total loss of heat sink are plant-specific, and thus, individual plant assessment is needed.

Some LWR plants currently in operation, including WWER plants, do not have a systematic assessment on this event, and therefore, there is no basis to judge the adequacy of emergency procedures, operator training and necessary hardware upgrading.

At each plant, the ESW system supplies cooling water to transfer heat from various safety related and non-safety related systems and equipment to the ultimate heat sink. The ESW system is needed in every phase of plant operations and, under accident conditions, supplies adequate cooling water to systems and components that are important to safe plant shutdown or to mitigate the consequences of the accident. Under normal operating conditions, the ESW system provides component and room cooling (mainly via the component cooling water system). During shutdowns, it also ensures that the residual heat is removed from the reactor core. The ESW system may also supply makeup water to fire protection systems, cooling towers, and water treatment systems at a plant.

The design and operational characteristics of the ESW system are different for PWRs, BWRs and WWERs, and also differ significantly from plant to plant within each of these reactor types.

Safety significance

A complete loss of the ESW system could potentially lead to a core-melt accident, posing a significant risk to the public.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

The Kozloduy NPP plans to perform an accident analysis of the case of a total loss of electrical power supplies, including total loss of the service water.

Czech Republic

At Temelín NPP, the problem of loss of service water is addressed in the framework of implementation NUS Halliburton Audit recommendation - Action Item 17.B. The system will satisfy the requirements of Regulatory Guide 1.27.

Russian Federation

A guide for accident management including some beyond design basis accidents has been developed by RRC Kurchatov Institute. Based on this, some Russian utilities have written the corresponding procedures.

Ukraine

The scope of the Ukrainian modernization programme includes plans to develop recommendations to manage accidents related to the loss of heat sink.

USA

In 1981, The Calvert Cliff Unit 1 lost both redundant trains of service water when the service water system became air-bound as a result of the failure of a non-safety related instrument air compressor aftercooler (NUREG-0933, issue 36).

For the open cycle service water system, bivalves were observed at approximately 45% of all US plant sites (NUREG-0933, issue 51). Serious fouling events in open cycle SWS were reported at Arkanas Nuclear One, Unit 1 (corbicula), Brunswick 1 and 2 (oyster), Pilgrim (mussles), San Onofore 1 (barnacles), Rancho Seco (corrosion products), Sequoyah Unit 1 (asiatic clams), and some WWER plants (heat exchanger fouling) as well.

The NRC staff surveyed seven multiplant sites and found that loss of the ESW system could be a significant contributor to core damage frequency (CDF) (NUREG-0933, issues 130 and 153). The generic safety insights gained from this study supported previous perceptions that ESW system configurations at other multiplant and single plant sites may also be significant contributors to plant risk and should also be evaluated.

ADDITIONAL SOURCES:

- IAEA 50-SG-D6, Ultimate heat link and directly associated heat transport systems for NPPs.
- Modernization programme, Kozloduy NPP units 5 and 6 January 1995, Rev. 0.
- USNRC Regulatory Guide 1.27, Ultimate heat link for NPPs, (Rev. 2), January 1976.
- NUREG-1275, Operating experience feedback report, US Nuclear Regulatory Commission.
- NUREG-0933, A prioritization of Generic Safety Issues, US Nuclear Regulatory Commission, 1991.

4.1.6. Electrical and other support systems (ES)

ISSUE TITLE: Reliability of off-site power supply (ES 1)

ISSUE CLARIFICATION:

Description of issue

Offsite power supplies are required to be available to have sufficient capacity and capability to ensure that the fuel and reactor boundary are maintained within specified acceptable limits and core cooling, containment integrity, and other vital safety functions are maintained during accident conditions.
The main concerns with the offsite power supplies are as follows:

- The reliability of the offsite power system as the preferred source
- Vulnerability of safety related equipment to sustained degraded voltage
- Adequacy of design interfaces of offsite and on-site power sources.

In a multi-unit nuclear power plant, it is questionable whether one set of two startup transformers can provide enough power for normal shutdown of two or even three plant units, as would be required after a loss of power from the main offsite grid. Furthermore, there is a concern about the electrical system failures spreading simultaneously to several units through their common startup transformers.

Safety significance

Reliable offsite power supplies are essential to avoid challenging the emergency power supplies whose potential failure is one of the main contributors to the overall core damage frequency, as quantified by most of the performed PSA. Degraded voltage conditions can impair the operation of the equipment without producing a transfer to onsite power supplies.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

A project to install a new pair of startup transformers each with a capacity of 63 MVA is underway in the Kozloduy NPP. The aim is to have a separate pair of startup transformers at each unit.

Czech Republic

The Temelín NPP has a set of two startup transformers for each unit which can back each other up.

France

GDC 17 is partially applied in France, due to the general interconnected network. Redundancy is physically applied on the lines between the NPP and the switchyard linked to the general network: even if they are in the same "electrical corridor" distance between transmission towers is calculated to avoid that a tower falling down does not affect the other line.

Reliability of off site power supply is associated with the postulated duration of failure.

A study performed in 1994 indicates that:

- in case of short network failure (time of recovering: about 30 minutes, mean frequency: 5.6 10^{-2}/year);
- in case of long network failure (about 8 days for climatic reasons, mean frequency: 4. 10^{-4}/year);
- in case of switchyard failure (time of recovering: about 24 hours, mean frequency: 2. 10^{-4}/year).

Taking into account internal sources reliability, and the different reactor operational situations, the study indicates that these data induce a probability to reach an H3 situation (black-out) of 3.5 10^{-4}/year/unit. Contribution of H3 to core melt risk as the result of PSAs studies is:

- for 900 MW PWRs: probability 7.10^{-7}/unit/year (1.5% of global risk of 5. 10^{-5});
- for 1300 MW PWRs: probability 9.10^{-7}/unit/year (7% of global risk of 1.5. 10^{-5}).

Germany

At all nuclear power plants (NPPs), at least two decoupled grid connections are available for supply from the grid. In addition, most of the NPPs dispose of a further grid connection which can be used for emergency power supply in the long term.

The interconnected grids of the electrical power suppliers are remotely controlled to a far reaching extent and are monitored automatically. Faulty means of operation and faulty grid sections can be isolated in a selective approach and bypassed for power transmission purposes. In the case of an immanent grid breakdown, loads will be separated from the grid under a five- stage plan in order to be able to continue the operation of the grid in the area of the NPPs. For the restoration of the grid, a sufficient number of power plants are available which can be started up without power supply from the grid (hydro-electric power, gas turbines). The sites of these plants are geographically distributed such that power transmission to the NPPs will be possible via at least one route.

In the case of disturbances, the protection of the auxiliary power supply of the NPPs is given top priority. The necessary measures are fixed. In addition, there are specifications for the electrical power suppliers in Germany as well as in Europe for the control and limitation of grid disturbances and the restoration of the grid following a grid breakdown.

The licensees have demonstrated that their NPPs can be supplied again with auxiliary power from the grid within one to two hours following a large scale failure of the grid caused by an electrical disturbance. Regarding mechanical damage of the overhead transmission routes, there is a requirement that in case of steel tower collapse and the postulated consequential damage, at least the NPP`s further connection to the grid will remain intact in order to supply the necessary emergency power loads.

India

- Additional startup transformer has been provided to improve the reliability.
- The grid supply failure is about once a year.

Korea, Republic of

The regulatory Body requested the utility to demonstrate that effects of electrical grid disturbances on its grid are such that the limiting under frequency decay rate of 3Hz/sec is not exceeded.

The utility has conducted grid stability studies on the transmission system using a digital computer program. The study results have shown that the ROK transmission network is adequate to maintain stable system operation.

Spain

Spanish NPPs are provided with two independent off-site power supplies. A study of the electric network has been made to quantify the availability of each NPP off-site power supply and the stability of the whole network. This study has supported the decisions concerning the implementation of the station black-out rule in all the NPPs.

Degraded voltage conditions have been considered in all the NPPs and back-up protections have been implemented to initiate the transfer to on-site power supplies if a degraded voltage over loss of off-site power setpoint is maintained during a set time.

Ukraine

For NPPs in the Ukraine, this issue is under consideration or implementation.

In Rovno NPP, two 330kV/6kV startup transformers each with 63MVA is planned for implementation in May/June 1996 for Unit 4.

In Zaporozhe NPP, auxiliary power supply of all units at the Zaporozhe NPP is under normal operating conditions maintained via two station transformers providing energy to 4 auxiliary bus bars. If a major disturbance within the 750 kV system should occur, an automatic power transfer to a set of startup transformers will be carried out automatically. At the Zaporozhe NPP, there are two sets of startup transformers installed which is in compliance with Ukrainian and Russian regulations. Each set of the startup transformers consists of a pair of transformers with an identical rating compared with the station transformers. In the unlikely but not excludible event, that more than three units must be shut down simultaneously and have to draw the required auxiliary power via the startup transformers - e.g. because of a loss of the 750 kV main grid - the available capacity of the startup transformers will not be sufficient to cope with this operating mode. As a consequence any additional unit must be shut down by operating the emergency cooling systems which will receive energy from their related emergency diesel generators, because at these two units the main heat sink will not be available due to a lack of an appropriate energy supply. This finding was already discovered by the power plant staff and plans had been elaborated to install another set of startup transformers.

USA

The nuclear power plants in the US meet the requirements of GDC 17 (Part 50, Appendix A) by providing two physically independent offsite power circuits connected to 10CFR the onsite power distribution system. Each circuit has sufficient capacity and capability (assuming that the onsite power systems are not available) to assure that the specified acceptable fuel design limits and design conditions of the reactor pressure boundary will not be exceeded, and to ensure that core cooling, containment integrity, and other vital functions will be maintained in the event of postulated accidents. The NRC staff reviews to determine that the offsite power system satisfies the criteria set forth in the Standard review Plan and can reliably perform its intended design functions during normal operation of the plant, anticipated operational occurrences, and accident conditions. This requires examination of the loads required to be powered from each offsite power source for all plant operating conditions; continuous and fault ratings of breakers, transformers; loading, unloading, and transfer effects on equipment, and the power capacity available from each source.

The staff has required of each applicant to perform their electrical transmission grid stability analysis for each of the offsite power sources to the plant. The basic requirement is that the loss of the nuclear unit or the loss of the largest operating unit on the grid or the loss of the most critical transmission line will not result in the loss of grid stability. Also, the electric power systems important to safety are designed to be periodically tested including the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

In addition to the undervoltage scheme to detect the loss of offsite power at the Class 1E buses, all the nuclear power plants in the US were required to have degraded voltage protection schemes with time delay to protect the Class 1E equipment (especially the 480 Vac contactors which have a pick-up value of 80 percent or 85 percent). Branch Technical Position PSB-1 of NUREG-0800, "Adequacy of Station Electric Distribution System Voltages," and IEEE Standard 741, "Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations," discuss in detail the concepts associated with degraded voltage protection. In summary, two separate time delays are selected for the degraded voltage protection. The first time delay is of a duration to establish the existence of a sustained degraded voltage condition (i.e., something longer than a motor starting transient). The subsequent occurrence of a LOCA immediately separates the Class 1E distribution from the offsite power system and connects to the EDG. For Non-LOCA conditions, the time delay selected is of the duration so that the permanently connected Class 1E loads are not damaged. Branch Technical Position PSB-1 of NUREG-0800, "Adequacy of Station Electric Distribution System Voltages," and IEEE Standard 741, "Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations," discuss the concepts associated with degraded voltage protection.

Regarding the adequacy of design interfaces of offsite and onsite power source, the two power sources are interlocked so that they can never be paralleled except during the EDG testing. The event of concern under this mode of operation is an accident concurrent with a loss of offsite power and a single failure preventing the opening of the feeder isolation breaker through which the paralleling of the offsite power system was being accomplished. Under such a condition, the EDG under test will be feeding the whole plant (the remaining EDG(s) will not be commanded to start, as the other safety bus is still detecting a voltage) until the offsite power feeder breaker is tripped on reverse power, or the EDG under test is tripped on under frequency or voltage restrained overcurrent relay.

Also, all the nuclear power plants are required to comply with the Station Blackout Rule (loss of all offsite and onsite power supplies) required by 10CFR50.63, "Loss of All Alternating Current Power."
The nuclear power plants in the US may use either two startup transformers or use one startup transformer as one source and a unit auxiliary transformer via main stepup transformer (where the main generator circuit breaker is used) as the other offsite power source. In a multi-unit plant, there are generally two startup transformers per nuclear unit and each startup transformer is sized to simultaneously start (block loading) all the accident loads of its safety bus, in case of an accident (LOCA) if a sequencer is not used. If a sequencer is used, each startup transformer is sized for the full loads of its safety bus plus the starting of the largest load on that safety bus. In plants where there are two startup transformers for the two units, the staff has traditionally required that each startup transformer should be sized to supply the accident loads of one unit and safe shutdown of the other unit.

Several events at nuclear facilities in the United States have resulted in dual-unit transients. The causes were due primarily to electrical disturbances (usually lightning strikes) which resulted in the loss of power to reactor coolant pumps or rod control systems. In one case a plant runback occurred when air compressors were lost due to the loss of electrical control equipment in the control air system.

To address these issues, procedures are developed and personnel are trained to address multiple challenges at multiple units. However, experience has shown that situations develop during these events that are not always foreseen and prepared for. The NRC and the industry are addressing these generic issues (particularly lightning protection and other grid disturbances) by focusing on the capability of off-site systems to have sufficient voltage and power to supply safety systems following a single or dual unit trip.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety Review Mission to Zaporozhe NPP, IAEA, WWER-RD-064, 1994 (Mission report).

- Results of the safety review of NPPs in the FRG by the RSK; Recommendation by the Reactor Safety Commission (RSK); Nov. 1988.
- The safety in the Spanish nuclear power plants, Nuclear Safety Council, Spain, May 1992.
- 10CFR Part 50 Appendix A, GDC 17, "Electric power systems."
- 10CFR Part 50 Appendix A, GDC 18, "Inspection and testing of electric power systems."
- 10CFR50.63, "Station blackout."
- NUREG/CR-3992, "Collection and evaluation of complete and partial losses of offsite power at nuclear power plants."
- NUREG-0800, BTP PSB-1, "Adequacy of station electric distribution system voltages (Degraded voltage)."
- NUREG-0800, BTP ICSB-11, "Stability of offsite power systems."
- Standard Review Plan 8.1, " Electric power introduction."
- Standard Review Plan 8.2, "Offsite power system."
- IEEE Std. 741, "Protection of class 1E power systems and equipment in nuclear power generating stations."
- USNRC Generic Letter 79-036, "Adequacy of station electric distribution system voltages."
- USNRC Information Notice 95-37, "Inadequate offsite power system voltages during design basis events."
- USNRC Information Notice 91-68, "Careful planning significantly reduces the potential adverse impacts of LOOP events during shutdown."
- USNRC Information Notice 88-50, "Effect of circuit breaker capacitance on availability of emergency power."
- USNRC Information Notice 86-87, "Loss of offsite power upon an automatic bus transfer."
- USNRC Information Notice 84-38, "Problems with design, maintenance, and operation of offsite power systems."
- USNRC Information Notice 79-04, "Degradation of engineered safety features."

ISSUE TITLE: Diesel generator reliability (ES 2)

ISSUE CLARIFICATION:

Description of issue

Events which result in a loss of offsite power require reliable emergency diesel generators (EDG) to supply all necessary safety systems with power to make possible a safe shutdown of the plant. It is also shown in most PSAs that the starting up reliability of the EDGs has a high level of importance to reduce the core damage frequency.

Safety significance

Improvements of the starting reliability of onsite EDGs will reduce the probability of events which could escalate into a core melt accident.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ___xx_____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

The Kozloduy NPP plans to take the following measures:

- analysis of the diesel generator startup failures and improvement measures based on their results;
- installation of two additional diesel generators shared by two units.

Operating experience showed that there are problems of lubrication during the startup phase of the diesel generators. The solution of lubrication during the diesel startup phase is underway. Currently the lubrication pump does not work during diesel startup. For the functional tests of the diesel generators at nominal power, non-safety consumers as an additional load are used. A parallel operation of the diesel generator with the grid is not planned.

Czech Republic

At Temelín NPP, each diesel generator is located in a separate room. It has two additional common diesel generators for two units. The capacity of one common diesel generator is able to cover the need of both units. The problems of diesel generator excitation were solved in the design.

Possibilities of using off-site supplies from hydropower stations during NPP blackout are investigated.

Germany

Starting reliability of DGs: The DGs in German NPPs are automatically started by different actuation signals: either low voltage or low frequency. The second signal has been implemented because of the general requirement that diverse signals should exist for automatic actuation of safety systems.

The volume of compressed air stored in the bottles of the starting system must be sufficient to start a Diesel engine 6 times before the bottles are exhausted.

Station blackout: Beside the large on-site DGs which will supply safety systems in case of design basis accidents like LOCA with additional loss of off-site power (LOOP), which is assumed in the safety analysis, there is an additional set of small DGs which are designed for power supply of vital consumers during external events (seismic events, air plane crash). These small DGs will also start automatically in case of LOOP and additional failure of the large DGs to supply the vital emergency power buses.

Sweden

By a number of PSAs it has been verified that common cause failure (CCF) for emergency diesel generators (EDG) are a significant contributor to core damage frequency.

Therefore a high level of reliability is designed into the diesel generator units and is maintained throughout their service lifetime by surveillance testing, maintenance and operating programmes.

The four EDGs designed for each unit are provided with completely independent essential systems and the power ratings of the diesel generators are determined by calculating peak load demand under limiting conditions.

The EDG are ready for immediate startup at any time in their own design conditions, except during maintenance (limited and suitable parts of the maintenance programme is planned and performed during plant normal operation).

EDG maintenance programme, periodic and surveillance testing, are continuously improved and include test of the dynamic behaviour during simultaneous disconnections of off-site power, diesel startup, sequential loading and disconnection of full load.

Ukraine

At Ukrainian NPPs, all events of diesel generator startup failures are being investigated, root causes are being defined and the corresponding corrective measures are being developed and implemented.

At Rovno NPP, the diesel generators are physically separated. Two of them are located in one building separated by walls and the third is in another building on the other side of the unit.

Operating experience on the basis of functional tests of emergency diesel generators showed, that there were problems of startup air (corrosion products) and lubrication of the startup air valves. These problems have been solved.

In 1995, the functional tests have been performed without startup failures. The functional tests of the diesel generators are performed monthly at about 2.4 MW and yearly at nominal power (at 5.6 MW). A parallel operation of the diesel generator with the grid is not automatically planned, but it is used for functional tests. Possible causes of common cause failures are evaluated by the NPP.

The compressed air system consists of two compressed air bottles for six accelerated starts of the diesel engine. Compressed air is not automatically replaced by the compressor unit

The diesel generator start is only initiated by undervoltage actuation (25% of nominal voltage) on the 6.0 kV emergency busbar. Depending on the results of the reliability analyses, a second startup criteria "frequency low" could be proposed.

Based on the observed diesel generator failure rate and common cause failures and also on the frequency of loss of off-site power, the probability of a station blackout, exceeding a duration corresponding to the start of core damage, will be evaluated within the framework of a PSA.

At Zaporozhe NPP, three emergency diesel generators are installed within all units of the power plant. Within Units 1-4 safety and non-safety loads are connected to the engines of train 1 and 2 whereas the engine of train 3 supplies safety loads only. A strict distinction between these two consumer categories is made within Units 5 and 6. Safety loads here are strictly devoted to their redundant diesel engines, while non-safety loads are energized from a separate pair of diesel generators which are used for Units 5 and 6. The latter configuration has adequate redundancy, as one out of the two additional diesels is capable to carry the none-safety loads of Units 5 and 6 at the same time.

The criterion for diesel start is a voltage signal which is derived by monitoring the voltage level at each of the 6 kV emergency busses. When this voltage decreases to 25% of the nominal value, the startup signal is generated and causes the engine to start. In comparison to international practice the criterion of 25% of the nominal value appears to be extremely low. In western appliances it is usually in the range of 80% - 85% of the nominal voltage and is therefore within the specified voltage range for the consumers. With this philosophy a very reliable energy supply of the emergency busses can be guaranteed because diesel start will automatically be initiated if voltage does not recover to standard values within a specified delay time of a few seconds. In cases of extended periods of operation at substandard voltage it cannot be excluded that safety related motors may be tripped due to overload as a consequence. Other safety motors that shall be switched on in this phase may fail to start, because bus voltage is too low.

USA

The purpose of onsite ac power system (EDG) is to provide power to the respective Class 1E safety buses on the loss of offsite power. The EDG is started either on a LOCA or a loss of offsite power. A high level of reliability is designed into the diesel generator units and is maintained throughout their service lifetime by periodic testing, maintenance and operating programmes. SRP (NUREG-0800) Section 8.3.1 defines an acceptable basis for meeting these design criteria and refers to several guidance documents including Regulatory Guide 1.9, NUREG-0660, IEEE 387 and Diesel Engine Manufacturers Association (DEMA) Standard. In conjunction with these design criteria, the NRC staff requires all the applicants and licensees to establish and maintain a quality assurance programme to assure that it will perform its intended safety function. In addition there are surveillance requirements and Limiting Conditions for Operation (LCO) which have evolved based on past operating experience. These requirements provide a means of assuring the long term reliability and operability of the EDGs.

As part of USI A-44, Station Blackout, the staff considered new requirements to reduce the risk of core damage frequency from station blackout events. Thus, attaining and maintaining high reliability of EDGs was a necessary input to the resolution of USI A-44. The staff also issued GL 84-15 to the licensees to reduce the number of cold/fast start, and to attain and maintain a reliability goal for their EDGs.

In the US, the staff reviews that each EDG is seismically qualified and is located in a separated seismically qualified room has a reliable, redundant starting air system of adequate capacity. As a minimum, the air starting system should be capable of cranking a cold diesel engine five times without recharging the receivers.

Lubricating oil is normally delivered to the engine wearing parts by one or more engine driven pumps. During starting, an electrically driven lubricating oil pump, powered from a reliable dc power supply, is installed in the lube oil system which operates in parallel with the engine driven main pump to supply fast lube oil.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- Qualitative CCF analysis with cause defence matrices, Pilotstudy for diesel generators. ABB Atom Report RPC91-76 (in Swedish).
- Defensive measures against CCFs for diesel generators, Quantitative analysis. ABB Atom Report, RPC91-57 (in Swedish).
- Specific CCF mechanisms of diesel generators at real demands. Avaplan Oy, CCF DGRD.
- Comparison of generic CCF data for diesel generators, Avaplan Oy CCF DGGD.
- Test strategies for standby diesel generators, Avaplan Oy DG Test.
- DG Database: Latent critical faults. Swedish and TVO1/2 Plant Experience 1980-89. Avaplan Oy, NDGDB LC.
  (The above mentioned references are available at the Swedish Nuclear Power Inspectorate).
- 10CFR Part 50 Appendix A, GDC 18, "Inspection and testing of electric power systems."
- 10CFR Part 50 Appendix A, GDC 17, "Electric power systems."
- 10CFR50.65, "Requirements for monitoring the effectiveness of maintenance of nuclear power plants."
- 10CFR50.63, "Station blackout."
- USNRC Regulatory Guide 1.160, "Monitoring the effectiveness of maintenance of nuclear power plants."
- USNRC Regulatory Guide 1.155, "Station blackout."
- USNRC Regulatory Guide 1.9, "Standby power supplies."
- NUREG/CR-4347, "Emergency diesel generator operating experience."
- NUREG/CR-2989, "Reliability of emergency AC power systems at nuclear power plants."
- NUREG-0800, BTP ICSB-17, "EDG trip circuit bypass."
- NUREG/CR-0660, "Enhancement of onsite diesel generator reliability."
- NSAC-108, EPRI, "Reliability of emergency diesel generators at US nuclear power plants."
- Standard Review Plan 9.5.6, "EDG starting system."
- Standard Review Plan 9.5.5, "EDG cooling water system."
- Standard Review Plan 9.5.4, "EDG fuel oil storage and transfer system."
- Standard Review Plan 8.3.1, "AC Power System (Onsite)."
- Diesel Engine Manufacturers Association (DEMA).
- USNRC Bulletins 79-09 and 79-023.
- USNRC Circulars 77-016 and 88-011.
- USNRC Generic Letter 94-01, "Removal of accelerated testing and special reporting requirements for EDGs."
- USNRC Generic Letter 84-15, "Proposed staff action to improve and maintain diesel generator reliability."
- USNRC Generic Letter 83-041, "Fast cold start of diesel generator."
- USNRC Generic Letters 79-017 & 79-021, "Transmittal of NUREG-0660."
- USNRC Generic Safety Issue B-56, "Diesel reliability."
- Recent USNRC Information Notices 96-23, 96-67, 97-41, 91-85, Revision 1 (February 27, 1997).
- USNRC Information Notice 91-55, "Failure caused by a test link in 4.16 kV switchgear."
- USNRC Information Notice 91-29, "deficiencies identified during electrical system functional inspection."
- USNRC Information Notice 91-06, "Lockup of EDG and load sequencer circuits."
- USNRC Information Notice 90-51, "Failures of resisters in the power supply circuitry of electric governor system."
- USNRC Information Notice 85-32, "Recent engine failures of EDGs."
- USNRC Information Notice 84-69, "Operation of EDGs."

ISSUE TITLE: Scope of systems supplied by emergency on-site power (ES 3)

ISSUE CLARIFICATION:

Power supply by diesel generators is provided to safety systems that are necessary to cope with design basis accidents. In WWER reactors, the scope of systems with diesel backed power supply is limited in comparison with the common international practice, and does not cover many systems that would reduce the severe accident risk by facilitating management of anticipated incidents.

Examples of safety relevant systems without diesel backed power supply are the following:

- primary circuit makeup water system;
- auxiliary feedwater system;
- cooling system for control rod drives;
- radiation control panel;
- telephones for communication between control room and the plant;
- pumps for filling diesel generator fuel tanks (tanks have fuel for 8 hours of operation);
- DC distribution system in turbine hall.

All of the above systems would be needed for proper management of incidents that entail complete loss of off-site power supply and necessitate plant cooldown to cold shutdown state. Specifically, the makeup system of the primary circuit would be needed for depressurization and for main coolant pump seal injection (even though the seals are less vulnerable than in western PWR types and withstand without failure a loss of seal injection for at least several hours). Availability of normal makeup would also speed up boration of the primary circuit.

The operation of auxiliary feedwater system would eliminate thermal shocks to the steam generators by preventing unnecessary startup of the emergency feedwater system.

Power supply for the above mentioned systems cannot be taken from the existing diesel generators because their capacity is exhausted by the existing loads.

Safety significance

Insufficient diesel backed power supply for the management of emergencies affects the safety functions and can be questioned in beyond DBA scenarios.

Source of issue (check as appropriate)

- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

At Kozloduy NPP, a project to install 2 additional diesel generators is underway. The additional DGs will back up the makeup system and auxiliary feedwater system.

Czech Republic

At Temelín NPP, there are two additional common diesel generators for two units. The capacity of one common diesel generator is able to cover the need of both units. Common systems supply a safety related system consumption and a common personal and expensive equipment safety consumption.

Germany

All systems important for safety of a NPP must be supplied by an emergency power supply. These are especially those systems which are needed for reactor shutdown, keeping the reactor in a shutdown condition, cooling the core and prevention of radioactive releases into the environment.

The capacity of the emergency power supply (DGs) is high enough to supply also such systems as the high pressure boration system (primary circuit makeup) and auxiliary feedwater system (for plant startup and shutdown), although these systems are not classified as safety grade.

Japan

The "Guide for Safety Design of Light Water Nuclear Power Facilities" requires to take into consideration the following:

The electrical systems shall be designed to allow the structures, systems and components with safety functions of especially high importance to be fed by either off-site power or emergency on-site power when they need electric power to fulfill their safety functions. (Guide 48.1).

The electric power for the component important to safety and CRDM cooling systems are fed from the emergency bus.

As examples of loads to which power can be supplied from the emergency batteries, CVCF power supply, ECCS DC control power supply, M/C · P/C breaker operating circuit, cooling system motor when the reactor is isolated may be pointed out.

Ukraine

At Rovno NPP, each of the two additional diesel generators has a priority connection to Unit 3 or 4. A cross-switching would be possible when two additional busbars for reliable station service (3BJ and 3BK) are also installed in Unit 3. The emergency backup of some additional consumers can be provided. Two important safety related consumers are connected to the additional diesel generators:
- two makeup pumps of the primary circuit (including support equipment);
- two auxiliary feedwater pumps (including support equipment).

The makeup system of the primary circuit is needed for depressurization and for speeding up the boration of the primary circuit. The activation of the auxiliary feedwater system before the startup of the emergency feedwater system reduces thermal shocks to the steam generators .

During the loss of off-site power, the plant computer is also connected to the additional diesel generator. Depending on the results of the plant specific study, it should be determined whether other systems need backup power after loss of off-site power supplies.

At Zaporozhe NPP, the Units 5 and 6 have one additional diesel per unit to back up the makeup pumps and other safety relevant systems. This measure is also being planned for other Ukrainian NPPs. In addition, alternative sources of electrical power, e.g. transportable diesel generators, are being investigated.

USA

Regulatory Guide 1.155, "Station Blackout" talks about the potential from severe accident sequences after a loss of all off-site and on-site power sources. The USNRC has reviewed the examples of safety relevant systems cited above and note that the following systems are already backed by the diesel power supply:
- primary circuit makeup water system;

- auxiliary feedwater system;
- cooling system for control rod drives;
- radiation control panel;
- pumps for filling diesel generators fuel tanks (tanks have fuel for 8 hours of operation).

In the US, the DC distribution system in the turbine hall is not diesel generator backed and the telephone for communication between control room and the plant is fed from the battery pack.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- 10CFR Part 50, Appendix A, GDC 17, "Electric power system."
- USNRC Regulatory Guide 1.155, "Station blackout."
- USNRC Regulatory Guide 1.9, "Standby power supplies."
- NUREG-1407, "Individual plant examination of external events for severe accident vulnerabilities."
- BTP ASB 9.5-1, (NUREG-0800), Section 4.e, "Lighting and communication."
- USNRC Generic Letter 88-20, "Individual plant examination of external events for severe accident vulnerabilities."

ISSUE TITLE: Breaker coordination to protect loads (ES 4)

ISSUE CLARIFICATION:

Description of issue

Circuit breaker coordination is a good engineering practice that can limit the degradation of safety related systems during a fault condition on a system component. Without coordination between breakers, an electrical fault on one component, such as a motor or motor-operated valve, could result in the loss of an entire motor control center or switchgear bus instead of just the single faulted component.

Electrical systems are generally designed today to ensure that the load breaker supplying a component will trip before the feeder breaker for the entire switchgear (which may be feeding other equipment). In the past (15 to 20 years ago), this coordination was not always closely scrutinized during the design of electrical systems. The NRC has issued numerous generic communications endorsing the use of breaker coordination to limit degradation of safety systems during component electrical fault conditions.

Safety significance

Safety systems at nuclear power plants are designed with redundant trains such that a failure of a single component or train will not prevent the system from performing its safety function. Thus, a lack of breaker coordination does not necessarily pose a safety significant hazard. If a faulted condition in a safety related component causes an entire motor control center or switchgear to be lost, there is generally a completely electrically independent, redundant train to perform the same function and thus the function of the safety related system is not lost.

However, the lack of breaker coordination can present more of a challenge to plant personnel trying to deal with a plant transient. It is desirable to have as much safety related equipment as possible available to plant operators to afford them more flexibility in reacting to any particular abnormal situation. With proper breaker coordination, only the faulted component is lost without affecting other components that may be fed from the same electrical bus.

Source of issue (check as appropriate)

- __________ operational experience
- ___xx_____ deviation from current standards and practices
- ___xx_____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Sweden

Circuit breaker co-ordination was addressed already in the original design. When new loads are added, a calculation or verification is always performed in order to verify that the proper co-ordination will be maintained, taking into account the new loads. This verification is part of the plant modification process.

The importance of co-ordination (selectivity) also applies to fuses. In order to maintain the selectivity, it is important to use fuses with a well known and reliable breaking characteristic. Fuses used in safety related power distribution are regarded as qualified parts and only fuses which meet the requirements are allowed to be used.

USA

The USNRC issued a generic letter which informed licensees of the various problems with electrical systems being identified with increasing frequency at commercial power reactors, including lack of proper coordination of protective devices creating the potential for an unacceptable level of equipment loss during fault conditions. In addition, several information notices were issued which alerted licensees to specific breaker coordination problems found at individual plants.

ADDITIONAL SOURCES:

- USNRC Generic Letter 88-15, Electrical power systems - Inadequate control over design processes, September 12, 1988.
- USNRC Information Notice 93-75, Spurious tripping of low voltage power circuit breakers with GE RMS-9 digital trip units, September 17, 1993.
- USNRC Information Notice 92-51, Misapplication and inadequate testing of molded case circuit breakers, July 9, 1992.
- USNRC Information Notice 92-51, Supplement 1, April 11, 1994.
- USNRC Information Notice 92-29, Potential breaker miscoordination caused by instantaneous trip circuitry, April, 17, 1992.
- USNRC Information Notice 91-29, Supplement 1, Deficiencies identified during electrical distribution system functional inspections, September 14, 1992.
- USNRC Information Notice 88-45, Problems in protective relays and circuit breaker coordination, July 7, 1988.

ISSUE TITLE: Vulnerability of swingbus configurations (ES 5)

ISSUE CLARIFICATION:

Description of issue

The use of a swingbus configuration when designing an electrical power supply system has the main goal of creating a high availability for the power supply to safety functions. This configuration makes it impossible to fulfil many requirements of physical separation between redundant trains. Therefore, single events, including human errors, can cause loss of power on entire level of affected bus. Such an event will reduce function of most redundant safety systems.

Any analyses of a power supply system against modern standards will result in deviations concerning physical separation. Analyses based on PSA-technique will very soon reveal such dependencies between redundant and diversified functions that are not easy handled within such a concept.

Safety significance

Current standards for design of power supply requires physical separation between redundant safety functions. Such requirements cannot be met in a power supply system in a swingbus configuration. CCI (common cause initiators) will have a great negative impact on safety functions as well as events caused by human errors both as an initiating event and in the managing of any PIE. The defence in depth will be significantly reduced because at least two barriers are likely to be affected, e.g. loss of residual heat removal and loss of containment integrity.

Source of issue (check as appropriate)

- __________ operational experience
- ____xx____ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Japan

Load swing design is employed in Japan and separation design is sufficiently done. There are two breakers between Load Swing design, not to degrade the physical separation because of single failure. Also, to prevent human errors, such measures as separate operation of breakers and key control are being taken.

Sweden

The Oskarshamn unit 1 is the only Swedish plant basically designed with a swingbus configuration. During construction of unit 2 the shortcomings in such a concept was realized and a separate emergency safety function was installed as a back-up to the original safety functions.

During a recently outage the swingbus configuration has been analysed by PSA-technique. As a result several minor changes have been made to eliminate unwanted weaknesses.

USA

In at least two multi-unit BWR plants, the 480 V motor controlled center (MCC) is made up of two MCCs whose buses are connected with a copper link to form one continuous bus. At the point of connection, there is a steel barrier which separates the two MCCs. This provides a dual source of power to the residual heat removal (RHR) for operation in the LPCI mode. The source breakers to the bus are interlocked such that the two EDGs are isolated from each other at all times. The tie between the two MCCs (mentioned earlier) is normally open through two breakers which are interlocked in such a manner that they can not be closed unless one of the two buses is disconnected from its normal EDG source.

It was recognized that the LPCI Swing Bus design does not totally meet the requirements of GDC 17 of Appendix A to 10CFR50. Under RG 1.6 for Class 1E power systems, a swing bus concept does not satisfy the single failure criterion. In 1976, the NRC determined that the swing bus design for BWR 3/4 plants did not satisfy GDC 17 because a single failure in the swing bus could result in the loss of the LPCI function. The design was examined and found acceptable in NUREG-0138 issued November, 1976. The Commission accepted the design principally because the emergency core cooling system (ECCS) criteria (10CFR50.46) can be met without any LPCI function (using the core spray pumps). In addition, the swing bus has loads only associated with LPCI functions, thus confining single failure effects.

The USNRC again revisited the adequacy of the LPCI swing bus design for BWR 3/4 plants for a flaw which could prevent the LPCI swing bus from transferring to its alternate ac power source with a single failure of the dc control power in one division, resulting in a failure of all LPCI pumps to inject water into the reactor vessel. The licensees made modifications which the USNRC found acceptable.

The licensees with the swing bus proposed modifications in response to GLs published in 1977 and 1979 for the degraded grid voltage conditions which the NRC staff evaluated and found to be acceptable. The NRC again revisited the swing bus issue, that there is a potential that the plants with a swing bus may not meet applicable ECCS regulations (10CFR50.46, Appendix K, or GDC 35 of Appendix A to 10CFR50) given a single failure during a degraded grid condition (with 2 out of 2) logic. That is, a single failure of the degraded voltage circuitry during a degraded grid voltage condition would prevent transfer to the EDG and would result in insufficient voltage for safety related 480 Vac equipment. The USNRC staff concluded that plants were not explicitly required to meet the single failure criterion with regard to degraded grid voltage relays. It should be noted that for sufficiently low voltage, the first level loss of voltage protection circuitry would cause transfer to the EDGs.

ADDITIONAL SOURCES:

- 10CFR50.46, "Acceptance criteria for emergency core cooling systems for LWRs."
- 10CFR50, Appendix K, "ECCS evaluation model."
- USNRC Regulatory Guide 1.6, "Independence between redundant standby (onsite) power sources and between their distribution systems."
- NUREG-0138, "Staff discussion of fifteen issues listed in attachment to November 3, 1976, Memorandum from Director, NRR to NRR Staff."
- LERs 87-045 (Fermi 2), 89-027 (Dresden 2), 89-028 (Monticello), 92-015 (Duane Arnold) and 93-005 (Monticello).
- USNRC Information Notice 86-70, "Potential failures of emergency diesel generators."
- USNRC Information Notice 88-15, "Potential problems caused by single failure of an esf swingbus."

ISSUE TITLE: Reliability of emergency DC supplies (ES 6)

ISSUE CLARIFICATION:

Description of issue

The DC power system in a nuclear power plant provides control and motive power to many components and systems (including information systems) during all phases of plant operation including abnormal shutdowns and accident situations.

Batteries are the ultimate energy source in the power plant and a high reliability and adequate capacity of this device is therefore a prime goal.

In some nuclear power plants, the designed discharge time is in the order of 30 minutes. This situation is not in compliance with modern requirements. The international trend goes towards an extension of the battery discharge time in order to better cope with accident management and station black out requirements. In case of a station blackout event, the battery is the ultimate energy source of the unit. A higher battery capacity maintains vital I & C systems in operation and illuminates the main control room. This would enable monitoring of essential plant parameters and safety significant motor operated valves would remain maneuverable. Therefore, the reactor can be controlled and can be kept in a safe condition by performing accident management actions (e.g. bleed). The extended battery discharge time leads to larger time margins for operators to decide on further actions.

A further concern for some nuclear power plants,is the lack of battery circuit monitor. Therefore, possible galvanic interruptions within the battery circuitry will not be automatically recognized, as long as the chargers are in operation. In addition, the batteries are inadequately isolated from the concrete floor and cannot withstand seismic loads. An earthquake could lead to a loss of the batteries and consequently to a loss of the non-interruptible power supply.

Batteries are designed to provide backup power increasing the assurance of DC supply reliability. When they are overoperated, DC power voltage can decrease too much making the control puzzling and leading to spurious indications or orders.

Safety significance

Insufficient supply by batteries in emergency situations can cause a loss of safety functions under accident conditions.

Source of issue (check as appropriate)
- ____xx_____ operational experience
- ____xx_____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

The Kozloduy NPP plans to replace the existing batteries with those having a higher capacity and seismic resistant capability.

Czech Republic

At Temelín NPP, the battery discharge time is at least one hour. This time is longer than the time required by accident management and station black out requirements. The batteries are seismically protected.

France

In the French plants every DC switchboard supplying power to systems important for safety is equipped with individual alarms indicating first the beginning of battery discharge to the switchboard and then the end of battery discharge (low voltage signal).

Germany

As a result of the comprehensive safety reviews which were carried out between 1986 and 1988 for all German plants the requirement on DC supply capacity was increased. To ensure measures of accident management in case of station blackout the capacity of the battery system must be sufficient to maintain DC supply at least for 2 hours instead of ½ hour before. Therefore battery systems have been extended.

Changes have also been made in the mounting support of the batteries. The objective was the improvement of behaviour during vibrations caused by external hazards.

India

Two new higher capacity battery banks with seismic resistant capability have been provided, one for each unit instead of one for the station by design.

Japan

In Japan, independent batteries as emergency power source are required to be provided depending upon the safety category grade under the Safety Design Review Guide. The seismic grade of emergency batteries shall be of Class As (functions showed be maintained against the seismic force resulting from a design base limit earthquake) under the Seismic Design Review Guide. The capacity of supplying power to the relevant loads are estimated for about five hours.

Monitoring of battery facilities are conduced under the shift patrol by checking charge/discharge status, cell external conditions and specific gravity control.

The capacity of battery facilities to supply power to continuous loads is estimated for more than five hours.

Ukraine

The Rovno NPP plans to install new batteries. The capability of the batteries to withstand earthquakes has been considered.

At Zaporozhe NPP, each reactor unit has three redundant batteries to provide energy to vital loads. Their designed discharge time is in an order of 15 to 20 minutes. The international trend goes towards an extension of the battery discharge time in order to better cope with accident management and station black out requirements. Possible galvanic interruptions within the battery circuitry will not be automatically recognized, as long as the chargers are in operation, as there is no battery circuit monitor available.

USA

The GDC 17 and GDC 18 of Appendix A to 10CFR50 requires that the on-site power systems including the batteries shall have sufficient capacity, independence, redundancy, and testability to perform their intended safety functions assuming a single failure. Spurious operations are supposed to be included in the failure analyses.

During the design stage, the batteries are sized in accordance with IEEE Std. 450 and IEEE Std. 485 including the design margins, the temperature correction factor and the aging factor. The initial battery capacity is sized 25 percent greater (aging factor) than required to take into account the battery replacement criterion of 80 percent rated capacity in accordance with IEEE Std. 450. Besides adding the aging factor, an additional 12 to 15 percent design margin is added. After considering the temperature correction factor, a battery size 50 percent greater than required may be needed.

During normal operation, the battery chargers (powered by Class 1E 480 V MCC) supply the continuous DC load of the associated divisions while maintaining a float charge on the batteries. Each battery charger is sized to recharge the battery from the design minimum charge state of 1.75 volts/cell to the fully charged state within 24 hours while supplying the steady-state loads under all modes of plant operation.

Each Class 1E 125 VDC battery is typically designed to be capable of carrying the essential loads continuously for a minimum of 2 hours in the event of a total loss of onsite and offsite ac power systems

The Class 1E 125 VDC batteries of each redundant train are located in separate, seismic Category 1 rooms. In addition to providing protection against the safe shutdown earthquake, the walls of these rooms act as fire barriers to maintain the integrity of the redundant systems. Electrical separation is also maintained to ensure that a single failure in one train does not cause failure in the redundant train.

Based on 10CFR50.63, all licensees were required to assess the capability of their plants to maintain adequate core cooling and appropriate containment integrity during a station blackout and to have procedures to cope with such an event. Regulatory Guide 1.155 describes a method acceptable to the *NRC staff for complying with the Commission regulation that requires nuclear power plants to be* capable coping with a station blackout for a specified duration. This resulted in selecting a minimum acceptable station blackout duration capability from 2 to 16 hours, depending on a comparison of the plant's characteristics with those factors which include redundancy of the onsite emergency ac onsite power system, their reliability, the frequency of loss of offsite power, and the time to restore offsite power. One coping method is ac independent, in which the plants rely on available process steam, dc power, and compressed air to operate equipment necessary to achieve safe shutdown until offsite or emergency power is restored. The batteries must provide sufficient capacity and capability for the specified duration. The battery calculations are performed in accordance with NUMARC 87-00, Section 7.2.2 to verify that the Class 1E batteries have sufficient capacity to meet SBO loads for the specified SBO duration.

The specific requirements for dc power system monitoring derive from generic requirements embodied in IEEE Std. 308 and in RG 1.47. In summary, these general requirements state that the dc system (batteries, distribution systems, and chargers) shall be monitored to the extent that it is shown to be ready to perform its intended function. Accordingly, the NRC staff has used the following guidelines for the indications and alarms to be provided in the main control room for each Class 1E dc system:

- Battery current (ammeter-charge/discharge)
- Battery charger output current (ammeter)
- DC bus voltage (voltmeter)
- Battery high discharge rate alarm
- DC bus undervoltage and overvoltage alarms
- DC bus ground alarm (for ungrounded system)
- Battery breaker or fuse open alarm
- Battery charger output breaker or fuse open alarm
- Battery charger trouble alarm (one alarm for a number of abnormal conditions which are usually indicated locally)

Monitoring cited above, augmented by the periodic testing and surveillance requirements included in the Technical Specifications, provides reasonable assurance that the Class 1E dc power system will perform its intended safety function.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- 10CFR Part 50 Appendix A, GDC 17, "Electric power systems."
- 10CFR Part 50 Appendix A, GDC 18, "Inspection and testing of electric power systems."
- 10CFR50.63, "Station blackout."
- USNRC Regulatory Guide 1.129, "Maintenance, testing, and replacement of large lead storage batteries."
- USNRC Regulatory Guide 1.47, "Bypassed and inoperable status indication."
- USNRC Regulatory Guide 1.32, "Criteria for safety related systems."
- USNRC Regulatory Guide 1.6, "Independence between redundant standby (onsite) power sources and between their distribution systems."
- NUREG-0666, "A probabilistic safety analysis of DC power supply requirements for nuclear power plants."
- NUMARC 87-00, "Station blackout."
- IEEE Std. 485, "Recommended practice for sizing large lead storage batteries."
- IEEE Std. 450, "Recommended practice for maintenance, testing, and replacement of large lead storage batteries."
- IEEE Std. 308, "Criteria for class 1E power systems."
- USNRC Information Notice 95-21, "Unexpected degradation of lead storage batteries."

ISSUE TITLE: Control room habitability (ES 7)

ISSUE CLARIFICATION:

Description of issue

Control room habitability system reviews have detected the following problems: (1) deficiencies in the maintenance and testing of engineered safety features designed to maintain control room habitability; (2) design and installation errors, including inadvertent degradation of control room leak tightness; and, (3) shortage of personnel knowledgeable about HVAC systems and nuclear air-cleaning technology.

In the original design of all WWER plants, the Main Control Room (MCR) and the Emergency Control Room (ECR) do not have their own separate independent ventilation systems which would ensure habitability in an emergency case. The main air ducts for MCR and ECR air supply are not provided with smoke detectors, the signal of which would be used to cut off the ventilation system.

The main control rooms of the units were not equipped with a ventilation system capable of filtering the intake air in case of radioactive releases. Consequently, there is a potential hazard of breathing the contaminated air in the main control rooms in case of serious accidents. The habitability of main control rooms must be ensured also in case of severe accidents.

Safety significance

Loss of control room habitability following an accident, release of external airborne toxic or radioactive material or smoke can impair or cause loss of the control room operators' capability to safely control the reactor and could lead to a core damaging accident.

Source of issue (check as appropriate)

- __________ operational experience
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

At Kozloduy NPP, till 1992, there was no separation between ventilation of the main control room and ventilation of the emergency control room. After that, improvements have been made.

For the MCR, three kinds of ventilation are installed:

- ventilation used during normal operation (2 streams with 1 ventilator with electrical backup and 1 exchanger cooled with the essential service water);
- ventilation used in case of hot temperature (1 ventilator and one exchanger using freon);
- ventilation used during accidental situations (2 streams with 1 ventilator with electrical backup, 1 exchanger cooled with the essential service water, 1 iodine filter).

These ventilation systems were designed taking into account the maximum allowable temperature of the equipment.

For the ECR, one additional stream with one ventilator with electrical backup and one exchanger cooled with the essential service water are used.

Czech Republic

At Temelín NPP, there is an emergency ventilation system of the MCR. Therefore, the ventilation system is equipped with ZFA aerosol and iodine filters. A study is elaborated for the case of emergency ventilation and habitability of the ECR. The study will be used as a basis for the implementation of possible modification.

India

The control room ventilation has been augmented with independent closed loop ventilation system which is put in service when the habitability of the control room is threatened. The system has come to help in incidents of leakage from chlorination system.

Korea, Republic of

- During normal and postulated accident conditions, the system provides
  - controlled environment for personnel comfort and equipment operability;
  - radiation shielding against airborne radioactivity release;
  - protection against the effects of high energy line rupture;
  - fire protection.
- The MCR air conditioning system is a safety related system consisting of an air condition system and emergency filtration system. The control complex ventilation system has two motor-operated isolation dampers for the outside air intakes.
- The HVAC system for the control room area is designed as seismic category I and will result in loss of system functional performance capability in the event of loss of offsite power concurrent with safety shutdown earthquake. The system ductwork is leak tested in accordance with ASME N509.

Russian Federation

The NPPs have implemented as part of their backfitting programme an independent ventilation system for both main control room (MCR) and the emergency control room (ECR) to protect personnel from air-born radionuclides and carbon dioxide. The air can be enriched with oxygen.

Spain

During the licensing of the Spanish NPPs and in later reviews to update their control room habitability systems, studies and design modifications in the following areas have been required:

- potential generation and detection of toxic gases;
- alternative air intakes to the control room HVAC systems;
- modifications in the control room HVAC filtering units.

Ukraine

At Rovno NPP, the separation of ventilation systems for MCR and ECR was done.

At Zaporozhe NPP, the main control room and emergency control room has a common ventilation circuit for Units 1 to 4 while they are separated for Units 5 and 6. This disposition cannot ensure habitability in case of fire in locations ventilated by this circuit because of the smoke spreading.

USA

The USNRC has required licensees to address the issue of control room habitability for some time. The general design criteria and regulatory guides address the issue for both post-accident radiological conditions and chemical hazards.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- 10CFR Point 50, Appendix A; General design criteria.
- USNRC Regulatory Guide 1.71.
- USNRC Regulatory Guide 1.52.

ISSUE TITLE: Reliability of instrument air systems (ES 8)

ISSUE CLARIFICATION:

Description of issue

Instrument air systems provide motive power for many components in a power reactor. The system is classified as a non-safety related system for many reactors, with safety related components assumed to fail in the safe position (on loss of air), or provided with safety related accumulators. Operating experience has revealed that the performance of air-operated safety related components may not be in accordance with their intended safety function because of inadequacies in the design, installation and maintenance of the instrument air system. Further, system recovery procedures and operator training may not be adequate to cope with loss of instrument air conditions.

Safety significance

Degradation of instrument air systems can lead to failures in safety related systems, including possible common-mode failures. Failures have occurred in several systems, including auxiliary feedwater, residual heat removal, main steam isolation, BWR scram, service water, emergency diesel generators, containment isolation, and fuel pool seals.

Source of issue (check as appropriate)

- ___xx_____ operational experience
- __________ deviation from current standards and practices
- ___xx_____ potential weakness identified by deterministic or probabilistic (PSA) results

MEASURES TAKEN BY MEMBER STATES:

Japan

In Japan, the "Design Review Guide" specifies the design requirements as below:
"Systems with safety functions of especially high importance shall be designed with redundancy or diversity and independence considering their physical make-up, working principles, assigned safety functions, etc. (Guide 9(2)).

Systems with especially highly important safety functions are defined in "Guide for Priority Groups of Safety Function of Light Water Nuclear Power Reactor Facilities" and instrumentation air source facilities come under this definition.

Instrumentation air systems are provided with redundancy and independence, and are also connected to the emergency power supplies. Meanwhile, air-operated valves for safety related systems are structured to keep the safety functions by moving into the fail-safe position, at the loss of the air supply, and are tested to ensure their functions periodically. Emergency procedures on loss of air are prepared at the nuclear power plants.

USA

Reliability of instrument air systems was tracked by the USNRC as Generic Issue 43 (in NUREG-0933). This issue was initiated in 1981.

USNRC issued a number of Information Notices to alert its licensees to problems with instrument air systems. A report, NUREG-1275, Volume 2, "Operating Experience Feedback Report-Air Systems Problems was issued in 1987, covering 29 events at several plants.

On August 8, 1988, the NRC issued a Generic Letter (88-14) to all power reactor licensees requesting them to review this NUREG and to perform a design and operations verification of the instrument air system at their plant. The verification was to include: (1) verification by test that air quality is consistent with manufacturer's recommendations for components being served,including a description of their programme for maintaining air quality (2) verification that maintenance practices, emergency procedures and training are adequate to ensure safety related equipment will function as intended on loss of air, (3) verification that the design of the entire instrument air system is in accordance with its intended function, including verification by test that air-operated components will perform as expected. This verification was to include analysis of component failure positions. Each licensee was to respond to NRC confirming completion of the above steps.

USNRC Information Notice 89-26, issued March 7, 1989, discussed some lessons-learned from responses to the Generic Letter. These related to effects on secondary containment integrity (door seals), and inadequate accumulator performance.

ADDITIONAL SOURCES:

- NUREG-0933, A Prioritization of Generic Safety Issues, Main report and supplements 1-12, July 1991.
- USNRC Generic Letter 88-14, Instrument air supply system problems affecting safety related equipment, August 8, 1988.
- USNRC IN 89-026, Instrument air supply to safety related equipment, March 7, 1989.
- USNRC IN 87-28, Air system problems at US light water reactors, forwarding case study report AEOD/C701 (March 1987), by Office of Analysis and Evaluation of Operational Data, June 22, 1987.
- USNRC IN 87-28, Supplement 1, same title, forwarding NUREG-1275, Volume 2, Operating experience feedback - Air system problems, December 28, 1987.
- USNRC IN 81-38, Potentially significant equipment failures from contamination of air operated systems, December 17, 1981.

ISSUE TITLE: Solenoid valve reliability (ES 9)

ISSUE CLARIFICATION:

Description of issue

Over the years, many failures of plant systems and components have been attributed to solenoid-operated valve (SOV) problems. Several events have occurred in which SOV failures affected redundant safety components, multiple trains of safety systems, or multiple safety systems. SOVs are in wide-spread use in each nuclear power facility. They are used in safety-related systems indirectly as pilot operators working with control system fluid (such as pneumatic or hydraulically operated isolation valves) and directly in fluid systems (such as to vent the reactor vessel head or to supply air to the starting system for emergency diesel generators). Many SOVs are also used in non-safety-related systems that can significantly affect safety systems (such as plant instrument air drier systems).

Safety significance

Because the failure of SOVs can affect multiple valve functions in safety and non-safety systems, common mode failures of these valves could contribute significantly to risk.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Germany

Based on different mechanisms, several failures of SOVs occurred over the years in German plants.

- Shrinking of magnet plunger bearing in solenoid valves which were operated in closed circuit configuration due to warming up by permanent excitation of the magnet. To prevent similar failures these bearing rings were exchanged by those made of more temperature resistant materials.
- Anchoring of valve seat and cone in solenoid valves due to growing together of oxide film on the same materials (e.g. austenitic/austenitic material) for the valve seat and cone. To clarify the basic causes extensive investigations have been carried out. Corrective measures were the exchange of material combination for the valve seat and cone.
- Non opening of SOVs caused by the existence of corrosive coating on the chrome plated cylinder liner. The corrosion coatings mainly consist of chromium-hydrate-compounds. The construction was exchanged to non chrome plated design.

During the last ten years no new generic problems with SOVs have occurred.

Japan

Through a design consideration in response to the requirements of Guide 9(2), trains will remain separated even if the solenoid fails, and thus not all system functions will be lost. Also, the system is designed to operate on the safe side even when the single solenoid valve loses its functions.

Those solenoid valves to be used in systems which are especially highly important for safety are required to be more reliable than those used in conventional facilities. Also, more careful attentions are paid in maintenance of those solenoid valves are maintained in more detail. There have been no failures of systems or equipment resulting from the failure of solenoid.

To address specific SOV failures, the USNRC has issued numerous information notices and bulletins that provide the root cause for the particular failures. Generic Letter 91-15 informed licensees of a case study report of operating experience problems with SOVs prepared by the Office for Analysis and Evaluation of Operational Data AEOD) and published as NUREG-1275, Volume 6. The case study integrates what has been learned over the past several years and provides an extensive assessment of SOV operating experience. The study describes deficiencies in design and application, manufacture, maintenance, surveillance testing and feedback of failure data, and concluded that problems with SOVs need additional attention by the industry. While the recommendations in the case study were not intended to establish regulatory requirements, many of the problems described in the report already are addressed by current environmental qualification and quality assurance rules.

As part of USNRC's ongoing regulatory activities, inspections such as Safety System Functional Inspections include the reliability of SOVs as well as other components required by safety related applications. The NRC also is providing technical advice to the Electric Power Research Institute's Nuclear Maintenance Application Center to assist in preparing an SOV maintenance guide.

ADDITIONAL SOURCES:

- IAEA-Incident Reporting System (IRS) report No. 397, "Non-operating and non-closure of pilot valves due to corrosive coating."
- K. Kotthoff, T. Riekert, A. Voswinkel, "Ergebnisse und Trends der generischen Auswertung der Betriebserfahrung", 18. GRS-Fachgespräch, Garching, 1994.
- NUREG-1275, Operating experience feedback report - Solenoid operated valve problems, Volume 6, issued February 1991.
- IE Bulletin 80-25, Operating problems with target rock safety relief valves at BWRs, issued December 19, 1980.
- IE Bulletin 80-23, Failures of solenoid valves manufactured by Valcor Engineering Corporation, issued November 14, 1980.
- IE Bulletin 79-01, Deficiencies in the environmental qualification of ASCO solenoid valves, issued June 6, 1979.
- IE Bulletin 78-14, Deterioration of Buna-N components in ASCO solenoids, issued December 19, 1978.
- IE Bulletin 75-03, Incorrect lower disc spring and clearance dimension in series 8300 and 8302 ASCO solenoid valves, issued March 14, 1975.
- IE Circular 81-14, MSIV failures to close, issued November 5, 1981.
- USNRC Generic Letter 91-15, Operating experience feedback report, solenoid operated valve problems at US reactors, issued September 23, 1991.
- USNRC Information Notices (with the most recent listed first) 96-07, 95-53, 94-71, 94-06, 90-11, 89-66, 88-51, 88-43, 87-48, 86-78, 86-72, 86-57, 85-95, 85-17 (and Supplement 1), 84-53, 84-23, 83-57, 80-39, 80-11.

4.1.7. Instrumentation and control (incl. protection systems) (IC)

ISSUE TITLE: Physical separation of instrument sensing lines for the reactor protection system (IC 1)

ISSUE CLARIFICATION:

Description of issue

Protection systems in many nuclear power plants are required to meet the design criteria of IEEE-279, "Criteria for Protection Systems for Nuclear Power Generating Stations." One of the criteria of IEEE-279 requires that .".the protection system equipment (for example, interconnecting wiring, components, modules, etc.) shall be identified distinctively as being in the protection system. This identification shall distinguish between redundant portions of the protection system."

An area which has not been considered to be within the scope of this requirement is the mechanical sensing lines of the instrumentation which feed the protection system. Since the sensing lines are essential to the reliable operation of the protection systems, identification of these lines would facilitate verification that sensing lines are appropriately separated and protected from external and internal hazards. In some cases, separation of redundant division instrument sensing lines used for the protection system has not been carried out.

In some WWER reactors, some of the primary instrumentation use a common tap to the component to which it is connected. These parameters may be used in important control systems or in the protection system. Failure of the common tap will cause failure of all instruments connected to it and may result in actuation or non-actuation of one channel of the protection system. It is a deviation from the Russian Safety rule "General safety regulations for NPPs (OPB-88)", GAN RF, Moscow, 1989.

Safety significance

The damage of the not physically separated redundant instrument sensing lines caused by an external or internal hazard, in some cases, may disable the functioning of reactor protection system.

Source of issue (check as appropriate)

- __________ operational experience
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Russian Federation, Ukraine

A proposal was made to increase the number of taps and to rearrange the actuation logic. The new arrangement of WWER-1000 pulse lines operates on a train-by-train basis for control safety system and emergency protection, for which four additional pipe sleeves are required to be made on the reactor vessel I&C nozzles and to use four available pipe sleeves (one for each hot leg loop) to measure SG pressure differential. With this arrangement, the failure of one pulse line will result only in the failure of one train out of three for the safety systems and one train out of two for the emergency protection systems (without safety system actuation and EP actuation). A practice of welding in additional pipe sleeves on reactor vessel I&C nozzles has been developed.

ADDITIONAL SOURCES:

- Report of the review of the safety improvement programme for South Ukraine NPP Units 1 and 2 to identify the safety issues of "small series" WWER-1000 NPPs, IAEA, WWER-SC-182, 1996.
- MOHT - OTJIG - EDF generic reference programme for the modernization of WWER-1000/320, Revision 6 (February 1997).
- Code of Federal Regulations Title 10 (10CFR) Part 50, Appendix A, Criteria 3, 4, 13, 22 and 23.
- USNRC Regulatory Guides (RGs): 1.11, 1.53, 1.151, 1.153.
- IEEE 279-1971, Criteria for protection systems for nuclear power generating stations (ANSI N42.7-1972), Institute of Electrical and Electronics Engineers.
- ISA-S67.02, 1980, "Nuclear safety related instrument sensing line piping and tubing standard for use in nuclear power plants."
- ISA-S67.10, 1994, "Sample line piping and tubing standard for use in nuclear power plants - ANSI/ISA."
- IEEE 603, 1980, "Standard criteria for safety systems for nuclear power generating stations."
- IEEE 384, 1977, "Criteria for independence of class 1E equipment and circuits."
- IEEE 379, 1977, "Application of the single failure criterion to nuclear power generating station class 1E systems," (Section 6.2).
- IEEE 279, 1971, "Criteria for protection systems for nuclear power generating stations."
- USNRC Generic Communications: BL 79-24, IN 81-15, IN 84-45.

ISSUE TITLE: Inadequate electrical isolation of safety from non-safety related equipment (IC 2)

ISSUE CLARIFICATION:

Description of issue

Electrical isolation devices are used to maintain electrical separation between safety and non-safety related systems in nuclear power plants (NPPs). The isolators are primarily used where signals from Class 1E safety related systems are transmitted to non-Class 1E control or display equipment, such as the safety parameter display system (SPDS).

Electrical isolators include fiber-optic and photo-electric couplers, transformer-modulated isolators, current transformers, amplifiers, circuit breakers, and relays. Isolators used in NPPs are designed to prevent the maximum credible fault in the transverse mode on the non-Class 1E side of the isolator from degrading the performance of the Class 1E side of the isolator below an acceptable level.

Observations during SPDS evaluation tests found that for electrical transients below the maximum credible level, a relatively high level of noise could pass through certain types of isolation devices and be transmitted to safety related circuitry. A high level of electrical energy passing through the isolator, could damage the Class 1E component, while a lower level of energy could generate electrical noise that could cause the isolation device to give a false output.

In the event that safety systems are affected by less than the maximum credible voltage or current faults on the non-safety side of isolation devices, the effects can range from minor degradation to complete failure of single or multiple trains of safety systems resulting in failure on demand or inadvertent operation. In one known case, a voltage transient induced by a power line fault caused a false indication that the turbine-generator output breaker had tripped, resulting in a reactor scram.

Safety significance

The signal leakage through inadequate isolation devices to safety related circuitry could damage or seriously degrade the performance of Class IE components. In other causes, electrically-generated noise on the circuit may cause the isolation device to give a false output. All these may cause the functional failure of safety systems.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Korea, Republic of

The reactor protection system and engineered safety features component control system use fiber-optic technology for isolation between protection system channels and equipment, cabinets and operator interface devices in the main control room. If an isolation error occurs, an appropriate error message is generated and diagnostic tests are then applied to isolate the cause of the error. This would include errors caused by the leakage through a fiber-optic isolator.

USA

The Generic Issue (GI) 142 of NUREG-0933 deals with staff concerns about the design of isolation devices used to ensure separation between Class 1E and non-Class 1E electrical control and instrumentation circuits. This issue was initiated in June 1987. NRC staff reviews of the implementation of the SPDS requirement indicated that some isolation devices used to provide an interface between the non-Class 1E SPDS and the Class 1E safety systems would allow signal leakage if electrically challenged. It was unknown if the amount of leakage posed a hazard to safe operation of the Class 1E system. A review of failure records does not reveal any incidents of system damage caused by isolation device challenge. However, based upon the potential design variations in future control systems resulting from application of computer technology, additional design and qualification test requirements for future plants are recommended.

ADDITIONAL SOURCES:

- NUREG-0933, A prioritization of Generic Issues, July 1995, Generic Issue 142: Leakage through electrical isolators in instrumentation circuits.
- NUREG-1453, Regulatory analysis for the resolution of Generic Issue 142: Leakage through electrical isolators in instrumentation circuits. September 1993.
- Recent USNRC Information Notices 95-13, 95-13 Supplement 1.

ISSUE TITLE: Interference in I&C signals (IC 3)

ISSUE CLARIFICATION:

Description of issue

During normal power operation of a German PWR plant, the reactor protection system (RPS) initiated a reactor and turbine trip. The cause of this event was a fault induced 2-out-of-3 signal of RPS during a switch-on phase of a feedwater pump. The signal was induced because the limit of DNB-heat flux density value in one primary loop was exceeded. But the real cause was found to be an electrical impact from a high voltage cable on the instrumentation cable during the startup phase of the pump. After readjustment of the original cable position and some tests, the plant returned to power operation again. This event has been declared as an operational occurrence and has a low safety significance. Safety significance is given by the fact that the reactor protection system was involved and during comprehensive revision tests two months later some electrical impacts of the same kind occurred spontaneously in other electrical components.

The described event was observed in a German PWR but electrical impacts from power cables or other components on instrumentation cables were also experienced in other plants (PWRs and BWRs).

The original design of the plant did not consider the effects of electromagnetic interference/radio frequency interference, heavy load switching transients and transients generated in the electrical and control systems due to external causes.

Safety significance

Malfunction of I&C systems due to electrical interferences and transients can compromise safety.

Source of issue (check as appropriate)
- ____xx____ operational event
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) results

MEASURES TAKEN BY MEMBER STATES:

Germany

This event was examined in detail and evaluated. Additional electrical protection measures (e.g. grounding connection) were introduced. In addition to that the event report was distributed to all German NPPs for further checks in this area.

Sweden

The influence from power transients was considered already in the original design and instrument cables are fully separated from power cables and run on different cable trays. The possible impact of atmospheric phenomena like lightning was also considered in the original design.

The possibility of safety significant event caused by electromagnetic interference was first addressed in the beginning of this decade. Since then the following measures have been taken:

- all equipment important to safety and the safe production of electricity have been checked regarding electromagnetic immunity. Measures have been taken where the potential for electromagnetic interference is high. The use of radio equipment is forbidden in rooms (e.g. I&C rooms) containing sensitive equipment. New equipment shall fulfil the immunity requirements of EN 50082-2 and the emission requirements of EN 50081-2, EN 55011 (CISPR 11) and EN 60555 (IEC 555).

USA

A wide variety of US industry standards and guides have been developed regarding electrical and electromagnetic interference. These as well as applicable USNRC regulations, Regulatory Guides and generic communications, are indicated in the References .

ADDITIONAL SOURCES:

- Operational experiences with German NPPs 1994. VGB Kraftwerkstechnik 75 (1995), Heft 4.
- 10CFR Part 50, Appendix A, Criteria 3, 4, 21, 22, and 23.
- USNRC Regulatory Guides (RGs): 1.75, 1.153.
- Industry Standards:
  - ISA-dTR 67.04.04, 1994, "Effects of EMI/RFI on instrumentation setpoints and indicators."
- IEEE 1050, 1989, "IEEE guide for instrumentation & control equipment grounding in generating stations."
- IEEE 519, 1992, "IEEE recommended practices and requirements for harmonic control in electric power systems."
- IEEE 518, 1982, "IEEE guide for the installation of electrical equipment to minimize electrical noise inputs to controllers from external sources."
- IEEE 472, 1971, "IEEE guide for surge withstand capability (SWC)."
- IEEE 384, 1977, "Criteria for independence of class 1E equipment and circuits."
- IEEE 142, 1991, "IEEE recommended practice for grounding of industrial and commercial power systems."
- IEEE 62.23, 1995, "IEEE standard draft application guide for surge protection of electric generating plants."
- IEEE C63.12, 1987, "American National Standard for electromagnetic compatibility limits, recommended practice."
- IEEE C62.41, 1980, "IEEE guide for surge voltages in low voltage AC power circuits."
- FIPS PUB. 94, "Guideline on electrical power for ADP installations."
- MTL-STD-461D, 1993, "Requirements for the control of electromagnetic interference emissions and susceptibility."
- MTL-STD-462D, 1993, "Measurement of electromagnetic interference characteristics."
- EPRI TR-102323, 1994, "Guidelines for electromagnetic interference testing in power plants."
- USNRC Generic Communications: CR 77-06, IN 83-83, IN 92-01, IN 92-12, IN 93-33.

ISSUE TITLE: I&C component reliability (IC 4)

ISSUE CLARIFICATION:

Description of issue

Safe reactor operation requires comprehensive instrumentation to actuate the reactor protection system and other I&C systems if necessary. The I&C equipment of NPPs are based on different technologies that is known to present in some cases reliability problems. Operational experience has shown that the I&C failure rate is relatively particularly with old NPPs I&C equipment.

Some I&C system design do not include reliability analysis of hardware and software and the reliability impact of I&C components on the whole plant safety. Such reliability analysis should include the behaviour before, during and after selected hypothetical severe accident conditions, including the instruments used in the reactor , thermocouples, pressure transducers, flow meters etc.

The I&C of some reactor types were not fully designed to provide automatic and self diagnosis of operating states and faulty conditions of such instrumentation.

Safety significance

This issue affects the design provisions and may have a direct or indirect impact on deviations from normal operation, on bringing back the installation to normal operating conditions and on the capability of engineered design features to prevent the evolution of deviations into more severe accidents. One or more safety functions can be impaired due to the insufficient reliability of the I&C system. The issue may cause initiating events during normal operation and can aggravate the abnormal conditions.

Without effective efforts in maintenance, the impact of poor reliability of I&C systems on safety will become more severe with the aging of the components.

Source of issue (check as appropriate)

- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Argentina

The failure rate of non-tin doped mercury wetted relays was found increasing significantly. An event happened at Embalse NPP, consisting on the opening of the main steam safety valves due to dormant failures of these relays. Besides, these relays are included on the safety systems of the plant. The corrective actions were to replace such relays by tin-doped relays, which prevents the jumping. Other issues related to I&C reliability components are evaluated and considered in the PSAs.

Bulgaria

Many I&C upgrades and developments of reliability based maintenance methods and "Rest Life Time" evaluation model were proposed in the Kozloduy Units 5 and 6 modernization programme [36, 37].

Russian Federation

I&C modernization programme is proposed for WWER-1000 NPPs.

I&C modernization programmes are proposed at Zaporozhe NPP.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- MOHT - OTJIG - EDF generic reference programme for the modernization of WWER-1000/320, Revision 6 (February 1997).
- 10CFR Part 50, Appendix A, Criteria 3, 4 and 21, 10CFR50.65.
- USNRC Regulatory Guides (RGs): 1.53, 1.75, 1.105, 1.152, 1.153, 1.160.
- IEEE 577, 1976, " Requirements for reliability analyses in the design and operation of safety systems for nuclear power generating stations."
- IEEE 384, 1977, "Criteria for independence of class 1E equipment and circuits."
- IEEE 379, 1977, "Application of the single failure criterion to nuclear power generating station class 1E systems," (Section 6.2).
- IEEE 352, 1975, "General principles for reliability analysis of nuclear power generating station safety systems."
- IEEE 7-4.3.2, 1993, "IEEE standard criteria for digital computers in safety systems of nuclear power generating stations."
- USNRC Generic Communications: BL 76-03, BL 76-05, BL 77-02, BL 78-01, BL 79-21, BL 79-25, BL 79-27, BL 79-28, BL 80-19, BL 80-20, BL 84-02, BL 88-03, BL 90-01, CR 76-02, CR 79-02, CR 79-20, CR 80-01, CR 80-16, CR 81-06, GL 80-70, GL 88-14, IN 79-04, IN 80-12, IN 80-13, IN 80-43, IN 80-45, IN 81-01, IN 81-38, IN 82-02, IN 82-04, IN 82-11, IN 82-13, IN 82-48, IN 82-54, IN 82-56, IN 83-08, IN 83-63, IN 84-80, IN 85-18, IN 85-50, IN 85-51, IN 85-63, IN 85-89, IN 86-47, IN 86-62, IN 87-34, IN 87-41, IN 87-61, IN 88-14, IN 88-38, IN 88-58, IN 88-69, IN 88-88, IN 88-98, IN 89-11, IN 89-42, IN 89-68, IN 90-51, IN 91-11, IN 91-45, IN 92-04, IN 92-05, IN 93-11, IN 93-85, IN 94-20, IN 95-02, IN 95-19, IN 95-22, IN 96-24, IN 96-43, IN 96-44, IN 96-44 Supplement 1, IN 96-46, IN 96-50, IN 96-62, IN 97-12, IN 97-44, IN 97-53, IN 97-69.

ISSUE TITLE: Lack of on-line testability of protection systems (IC 5)

ISSUE CLARIFICATION:

Description of issue

The protection system designs of some old plants did not provide for on-line testability.

During normal operation, protection systems are in standby, therefore failures of components may not be detected. For these old plants, periodic tests are performed to provide some confidence in the capability of protection systems to fulfil their function. These tests are designed to simulate Reactor Protection actions following an accident or incident situation and have to trigger the whole protection chain from the sensors to the actuators. Manual testing during plant operation requires time and could be the source of errors; in addition any untimely protection actuation has to be avoided during or after the test, as well as unwanted inhibition when the plant is in operation.

Testing has a direct impact on the availability of safety related systems. In some old plants, a larger portion of the protection system hardware can only be tested through the sub-group relays during outages which typically have an 18-month frequency. This could result in the risk increase at those plants.

On-line testing increases the ability to detect existing failures of the protection system and could therefore result in improved reliability of the system; hence, a reduction in plant risk.

Safety significance

During a normal mode of operation, the protection systems are in their stand-by mode. Without testing, it is not possible to demonstrate operability and the availability of the safety systems to perform their intended functions on demand.

Source of issue (check as appropriate)

- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

At French plants, automatic testers are implemented to improve the safety level of protection systems and to increase the availability of the plant in eliminating human errors and inadvertent reactor trips during periodic testing.

India

The Indian BWR reactor protection system does not have on-line testing capability. The system is, however, designed as a fail-safe system. All sensors are checked once every quarter by actually injecting signal.

Slovakia

Modification for unsafe failure detection of measurement channels have been implemented. At present tests are performed by means of initiating signals in secondary devices.

Technical specifications were modified based on PSA study Level 1 results. New test intervals were adopted.

Detailed procedures for testing of all safety systems in hot standby conditions were developed and are in use.

Design to replace existing RPS by a new one capable of self testability is ongoing.

ADDITIONAL SOURCES:

- 10CFR Part 50, Appendix A, Criteria 18 and 21.
- USNRC Regulatory Guides (RGs): 1.22, 1.58, 1.118, 1.153.
- IEEE 498, 1975, "Supplementary requirements for calibration and control of measuring and test equipment used in the construction and maintenance of nuclear power generating stations."
- IEEE 415, 1976, "Planning of pre-operational testing programs for class 1E power systems for nuclear power generating stations."
- IEEE 398, 1972, "Test procedures for photo multipliers for scintillation counting and glossary for scintillation counting field."
- IEEE-338-1977, "Criteria for periodic testing of nuclear power generating station safety systems."
- IEEE-336, 1977, "Installation, inspection, and testing requirements for instrumentation and electric equipment during the construction of nuclear power generating stations."
- ISA-dTR 67.04.07, 1994, "Use of as-found/as-left data."
- USNRC Generic Communications: BL 77-03, CR 77-13, CR 81-12, IN 83-03, IN 83-65, IN 84-37, IN 85-02, IN 85-23, IN 85-49, IN 85-75, IN 85-98, IN 88-83, IN 91-75, IN 92-23, IN 92-28, IN 92-40, IN 92-51, IN 92-65, IN 93-29, IN 93-38, IN 93-64, IN 94-08, GL 96-01.

ISSUE TITLE: Reliability and safety basis for digital I&C conversions (IC 6)

ISSUE CLARIFICATION:

Description of issue

There is a need to encourage the adoption of digital technology based on criteria reflecting the international experience in digital computers and software for safety systems of NPPs.

Since the digital technology is considerably different than analog technology, the criteria appropriate for the safety review of digital computer based system are different. Such systems are being considered for use in the reactor protection, ECCS actuation, feedwater control systems, etc.

Because of the uniqueness, a new criteria is required to gain confidence in the performance of these digital systems.

Safety significance

This issue is directly related to the performance of safety functions by protection systems and other safety systems.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

USA

The US documents listed in the References are the studies performed and the guidance issued for the use of digital technology at nuclear power plants. Each system is reviewed on a plant specific bases in the early stages until comparisons can be made with previously reviewed systems.

ADDITIONAL SOURCES:

- USNRC Regulatory Guide 1.152, Revision 1, Criteria for digital computers in safety systems of nuclear power plants. This endorses IEEE std 7-4.3.2 1993, Criteria for digital computers in safety systems of nuclear power generating stations.
- Digital instrumentation and control systems in nuclear power plants, safety and reliability issues, prepared by National Research Council.
- IEC 987, Programmed digital computers important to safety for nuclear power plants.
- IEC 880, Software for computers in the safety systems of nuclear power plants.
- ISO 9000-3, Guidelines for the application of ISO 9001 for the development, supply and maintenance of software.
- TRS 282, Manual on quality assurance for computer software related to safety of NPPs.
- TRS XXX, Verification and validation of software related to npp control and instrumentation.
- USNRC Generic Letter 95-02, Use of NUMARC/EPRI Report TR-102348 "Guideline on licensing digital upgrades."

ISSUE TITLE: Reliable ventilation of control room cabinets (IC 7)

ISSUE CLARIFICATION:

Description of issue

The function of the control room area ventilation system (HVAC) is to provide a controlled environment for the comfort and safety of the control room personnel and to ensure the operability of control room components during normal operation, anticipated operational transients and design basis accident conditions. In the event of a failure in all the redundant trains of the HVAC system, the increase of temperature in the control room area can lead in a short time to malfunctions of the electronic equipment in the control cabinets.

During a loss of both trains in the control room ventilation system in a Spanish which had a duration of 40 minutes, oscillations and failures in different control panel indications were observed after about 20 minutes with an increase in the temperature in the cabinet area of some 10°C. It was necessary to transfer several controls to manual. According to the procedures, the doors of all the control and protection cabinets were opened to enable a ventilation of the electronic components.

Failure of the redundant trains of the control room HVAC system is not considered in the design basis.

Safety significance

Malfunctions in the control room cabinets can produce a common mode failure in vital instrumentation required to control the safety systems both in the automatic and in the manual mode, and could simultaneously threaten several safety functions.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

India

Control room ventilation system has been augmented with independent closed loop ventilation system which is available for situations when the main ventilation system fails. Earlier ventilation system failures have not indicated any major failures in safety functions.

Russian Federation

To maintain the rated temperature values and relative air humidity in the main control room (MCR) and reserve control room (RCR) of WWER-1000, it is necessary to provide for independent air conditioning systems for each room, regardless of the conditions incurred in the MCR and RCR (for example, fire).

Air conditioning systems shall be developed to ensure reliable maintenance of inner air parameters in the MCR and RCR under normal and emergency conditions, seismic effects, extreme outer air parameters and also to ensure repairability of the installed equipment.

It is proposed to service the MCR using the existing systems UV02, UV06, UV55 and the RCR using the UV01 system. The UV01 system is being reconstructed in such a way, that in addition to the cooling function it will maintain the preset relative air humidity in the RCR, and deliver outside air according to sanitary standards and at the required pressure.

To this end, the existing independent air conditioners are being equipped with humidifying devices connected to the main water supply.

Spain

As a short term corrective action, it was decided that a reduction of the maximum control room temperature and an increase of the surveillance requirements should be defined in the Technical Specifications.

ADDITIONAL SOURCES:

- MOHT - OTJIG - EDF generic reference programme for the modernization of WWER-1000/320, Revision 6 (February 1997).
- NUREG-0800, Standard Review Plan, US Nuclear Regulatory Commission.

ISSUE TITLE: Human engineering of control rooms (IC 8)

ISSUE CLARIFICATION:

Description of issue

The control rooms provide the control and displays necessary for the operator to carry out actions required during normal and shutdown operations of the plant. In some NPPs, including WWER NPPs, it follows the classical division by subsystems with an "active mimic" diagram of each subsystem shown with controls for pumps and valves in their appropriate functional position on the diagram. This type of organization has led to operational problems, most notably the Three Mile Island Unit 2 accident in the USA, because the operator's attention is focused on a specific item and he tends to disregard the interactions between the subsystems. An action that is taken at one point to solve one problem may create other problems in related subsystems.

In WWER plants, there are further deficiencies related to human engineering design if a comparison is made with the most modern international practices. Indicators of different types of process measurements, for example flow and pressure, are not distinguishable, except by the engraved legend. Control switches for pumps, valves, circuit breakers, etc., all have handles with the same shape. These switches have red and green indicating lights showing the operational status of the associated component. The brilliance of these lights varies greatly, and, in some cases, it is difficult to determine which lamp is lit. There is an increased likelihood that operators will make an error in assessing the equipment status. Indicators that provide data that is important to the operator's evaluation of the safety state of the plant are not differentiated from those used for normal operations. Some of the most valuable space on the control panel, that which is directly in front of the operator, is used for infrequent activities related to plant startup and surveillance testing.

In summary, the design of the information display in some control rooms does not give the operator a rapid overview of information regarding the current state of plant and reactor safety.

Safety significance

The deficiencies in the design of the main control room and the emergency control room increase the potential incidence of human errors. These situations are possible under normal operating conditions as well as under DBA and beyond DBA conditions and could become a major contribution to the total plant risk.

Source of issue (check as appropriate)
- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

An in-depth design review of the control room is already in progress at the Kozloduy NPP under an existing programme with Gilbert-Commonwealth.

Czech Republic

The main control room of the Temelín NPP is being designed in accordance with IEC 964. Human factors considerations are also being implemented in accordance with the United States NUREG-0700. The emergency operating procedures will be developed using the Westinghouse Owners's Group (WOG) Emergency Response Guidelines (ERGs).

Russian Federation

The development and implementation of an information support system is being considered by the Russian designers.

On the basis of IAEA recommendations analysis of some engineering-psychological factors for WWER-1000 is proposed to be performed:

- panel design and arrangement of the MCR for better normal and emergency conditions;
- provision of climatic conditions at MCR under normal operation, accident and, in case of origination of external events (external impact wave, smoke-screen, appearance of toxic gases, etc.);
- illumination conditions;
- noise level;
- communications;
- other issues pertained to MCR ergonomics.

Feedback of operations personnel shall be taken into account during performance of analysis.

MCR modernization, if necessary, shall be performed with the contribution of NPP Management as per special schedule.

Spain

The requirements of USNRC Generic Letter 82-33 have been used to review the design of the Control Room and Remote Shutdown Panel and of the Safety Parameter Display System.

Within the scope of PSA studies, human reliability analyses are ongoing, including human actions during external events, operating modes different to full power and conditions after fuel damage.

Ukraine

The I&C upgrading programme for the Ukrainian NPPs includes a general design, common for all WWER-1000/320 control rooms, being carried out by HARTRON SPA in co-operation with several Ukrainian institutions. The Rovno NPP Unit 4 is considering the implementation of the upper level hardware of the I&C hierarchical system mentioned above before startup. The software developed in this first step will follow the present operational methods.

A new CIS as well as the SPDS will be installed in a second stage. New software will be then implemented and the operating methods changed.

USA

After the Three Mile Island accident, the USNRC issued interim and long term requirements for improvements in control room design with respect to human factors considerations. Discussion of this matter may be found in the References.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, "Ranking of safety issues for WWER-440 model 230 nuclear power plants," IAEA-TECDOC-640, Vienna (1992).

- MOHT - OTJIG - EDF generic reference programme for the modernization of WWER-1000/320, Revision 6 (February 1997).
- IEC 964, "Design for control rooms of nuclear power plants," 1989.
- NUREG-0737, Supplement 1, "Clarification of TMI action plan requirements," (January 1983).
- NUREG-0711, "Human factors engineering program review model," (July 1994).
- NUREG-0700, "Guidelines for control room design reviews," 1982 September 1981.
- Revision 1 to NUREG-0700 (draft), "Human system interface design review guideline," (February 1995).

ISSUE TITLE: Need for a safety parameter display system (IC 9)

ISSUE CLARIFICATION:

Description of issue

A safety parameter display system (SPDS) is part of the information system important to safety which provides information for the safe operation of the plant during normal operation, anticipated operational occurrences, and accidents.

After the TMI-2 accident, many NPPs installed SPDS, but in some plants, including all WWER plants, a similar system has not been added. An SPDS can provide the operator with the information needed to assess the critical safety function of the plant.

Safety significance

A lack of SPDS may increase the failure rates of operators, especially in the highly stressed circumstances. The failure to monitor and/or restore the critical safety functions in a timely manner could result in damage to the core, and other protective barriers.

Source of issue (check as appropriate)

- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA)

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

The Kozloduy NPP decided to install a state-of-the-art SPDS for Units 5 and 6.

Czech Republic

The Temelín NPP will install an SPDS in the main control room to provide the operator with the information needed to assess the critical safety functions of the plant.

France

Following TMI accident, a French post TMI action plan has been undertaken, in which the addition of a Safety Panel.

Each unit is equipped with two computers: one for centralized data processing the <<KIT>> system; the other for safety panel functions the <<KPS>> system. Data acquisition is ensured by a redundant set of two computers.

The KPS system is used as a post-accident conditions operating aid. Its main functions are:

- selection of the first fault;
- display of safety action confirmation (synoptic diagrams);
- diagnosis of type of thermohydraulic accident and choice of procedure;
- operating aid for safety injections;
- operating aid for cooling and depressurization;
- surveillance of safety functions;
- aid for post-accident permanent surveillance;

- synthesized display to aid operation;
- logic chart for incident instruction initiation;
- RHR system surveillance.

In addition to KIT and KPS, a desktop computer, qualified earthquake resistant, is used for calculating the margin to DNB and maximal core temperature.

Russian Federation

The development and implementation of an information support system is being considered by Russian designers.

System for presentation of critical functions and safety parameters is an automated subsystem of operator support system (OSS) of power unit.

Abroad, these same systems (OSS) have been developed and implemented subsequent to the accident at TMI NPP (USA).

Based on the operating experience of nuclear power plants in the Russian Federation one can draw a conclusion that in the course of decision-making by the operator the weak link is the stage of analyzing and understanding the situation on the basis of the received information. To help operation personnel in the main control room the concept of critical functions and safety parameters has been developed and implemented abroad at a series of Western nuclear power plants.

The essence of the concept is to present the operator with concentrated information on the state and possible breaking of safety barriers as well as to provide assistance for determining the sequence of control actions required to restore the status of the critical function in the event of its loss. System of safety parameters presentation (SSPP) is intended for identification and prevention of accidents potentially related to radioactivity release beyond the safety barriers. SSPP shall provide monitoring of power unit safety level and refer the operator to the relevant manuals in the event of serious failures in operation of power unit. Results of SSPP operation are needed at the power unit, power plant (crisis center of nuclear plants) and national level of monitoring and control of safety of nuclear plants. SSPP shall be developed and implemented at every power unit of nuclear plant and in this case the plant's displays shall be installed in the operation system of the main control room. SSPP shall operate continuously and daily under all possible operating conditions and during accidents. SSPP of all power units of nuclear plants shall be in contact with the crisis center of the whole nuclear plant. In order to receive initial information the SSPP can communicate with the standard information systems of the power unit (computer control system, in-core instrumentation system, centralized radiation monitoring information-measurement system) but preferably it will receive signals about parameters and state of valves and mechanisms directly from sensors and limit switches. Total number of analog-and-discrete input parameters can amount to 400 and 700 respectively. The number of data presentation formats on displays is 30. Information entered in the system shall be validated and the algorithm shall be verified.

Ukraine

The I&C upgrading programme for all the Ukrainian WWER-1000 NPPs includes installation of the SPDS in the main control room.

USA

After the Three Mile Island accident the USNRC issued requirements for installation of safety parameter display systems (SPDS) to ensure that control room operators had ready access to key information under accident conditions. This information is also commonly displayed in the plant technical support centre.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, "Ranking of safety issues for WWER-440 model 230 nuclear power plants," IAEA-TECDOC-640, Vienna (1992).
- MOHT - OTJIG - EDF generic reference programme for the modernization of WWER-1000/320, Revision 6 (February 1997).
- NUREG-1342, "A status report regarding industry implementation of safety parameter display systems," (April 1989).
- NUREG-0737, Supplement 1, "Clarification of TMI action plan requirements," (January 1983).

ISSUE TITLE: Inadequacy of diagnostic systems (IC10) (WWER)

ISSUE CLARIFICATION:

Description of issue

Diagnostic systems are needed to provide the operators with an early warning of a situation where the integrity of the reactor coolant pressure boundary and condition of mechanical equipment are threatened.

The original design of WWER units does not provide for adequate diagnostic systems to monitor the reactor coolant pressure boundary integrity. For example, the existing means to detect a primary to secondary leakage in steam generators is not sufficient to monitor a violation of the design limits of safe operation.

Another example is the lack of monitoring and assessing the unspecified loads (loads not specified in the original design). In penetrations, nozzles and in certain piping, high thermal loads relevant for fatigue analysis have been expected and in many cases treated with specific design. It was not possible to specify loading due to e.g. stratification at the design stage. Penetrations and nozzles are usually high stress concentration areas with specific design features (e.g. thermal sleeves, wall thickness reduction, dissimilar welds) and residual stresses. NDT and thorough integrity assessment is required but difficult to achieve in many occasions. It is common practice according to reference codes and standards to implement monitoring system to ensure the required integrity is maintained for components concerned.

The original WWER design also does not provide for adequate condition monitoring of the mechanical equipment important to safety. This monitoring should be carried out with respect to vibration, displacement, position and condition as appropriate.

Safety significance

This issue was identified as a deviation from current standards. Inadequate monitoring of the reactor coolant boundary integrity and condition of equipment can impair the safety function because of a lack of an early warning when the integrity of the reactor coolant pressure boundary is threatened. This scenario is possible during normal operational conditions with degradations of components.

Source of issue (check as appropriate)

- __________ operational experience
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria, Czech Republic, Russian Federation, Ukraine

There are plans in all countries to develop and introduce diagnostic systems for leak detection, vibration monitoring and noise monitoring.

Control and monitoring of thermal loads in relevant sections have been proposed. State of the art integrity assessment together with improved NDT have been proposed.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- ASME OM Code -1987 with 1988 Addendum, Code for operation and maintenance of NPPs.
- USNRC Bulletin 88-08, "Thermal stresses in piping connected to reactor coolant systems," 1988.

ISSUE TITLE: Reactor vessel head leak monitoring system (IC 11) (WWER)

ISSUE CLARIFICATION:

Description of issue

In the WWER reactor vessel head, the CRDMs, instrumentation, etc. are attached to the head penetrations through bolted joints (flanges). Each joint is sealed by two parallel sealing rings (Ni) and the leak detection is based on the collection of the leakage water between these two sealing rings. The leak detection system is not tested or inspected periodically. The humidity monitoring system in the upper reactor block is not sensitive enough to detect leaks in the bolted joints.

Undetected leaks could lead to a severe corrosion of the vessel head from the outside. The vessel head is covered by a steel structure filled with ceramic balls and, consequently, its outside surface is not accessible for routine inspection to detect corrosion damage.

At the Khmelnitsky NPP, a vessel head had to be replaced due to corrosion damage associated with leaks in the upper reactor block.

Safety significance

Because of the particular WWER vessel head design and monitoring equipment, a serious degradation of the head cannot be ruled out. The safety function is impaired by an increased challenge of the ECCS in case of a small LOCA. In the worst case, a control rod ejection can happen.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

In the Kozloduy Units 5 and 6, a humidity detector to enhance reliability of general leak detection in the upper vessel head has been installed, and the installation of leak detection at control rod drive mechanism and instrument flange is planned.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).

ISSUE TITLE: Availability and adequacy of accident monitoring instrumentation (IC 12)

ISSUE CLARIFICATION:

Description of issue

During and following an accident, appropriate parameters and system functions are monitored in order to enable the operator to cope with the event sequence. The operator must have sufficient information available to: (1) determine the course of an accident (2) make decisions concerning appropriate manual actions ; and (3) assist in determining what actions, if any, are needed to execute the plant emergency plan. To supplement these actions and improve the plant operation under emergency conditions it could be necessary to assist the operators with display systems making the information easy to understand and providing aids in procedures utilization.

The TMI-2 accident reinforced the need to supply the NPP operators with pressure, temperature, radiation and humidity measurements that have a measuring scale beyond the normal operating range. In case of a design basis accident or a beyond design basis accident these measurements have to provide reliable information of the conditions inside the reactor pressure vessel and the containment. The accident monitoring instrumentation in some reactors is not adequate. For example, the reactor pressure vessel level indication is currently not provided, and the level can be estimated only by indirect means. A further example is that the effluent from the ventilation duct is not properly monitored in terms of radioactivity. The information obtained on the effluent from ventilation ducts during accidents is not reliable and accurate and this can lead to an overirradiation of plant personnel and inhabitants in the vicinity.

Safety significance

Most of long term (and some short term) emergency response and recovery actions are taken by operators. A lack of appropriate information provided to the operator will increase the failure rates of operators, especially in the highly stressed circumstances. In this case, the human errors could become a major contribution to the total plant risk.

Source of issue (check as appropriate)

- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

The Kozloduy NPP proposed in the modernization programme for Units 5 and 6 to include the study and installation of a post-accident monitoring system, including reactor vessel level monitoring.

Czech Republic

A Post-Accident Monitoring System is to be installed at the Temelín NPP that meets the intent of USNRC Reg. Guide 1.97 [1] and RPV level indication. A task analysis identified the appropriate variables and established the design bases and qualification criteria for instrumention used by the operator for monitoring conditions in the primary coolant system, the secondary heat removal system, the containment,the performance of the engineered safety features, and other systems normally used to bring the plant to, and maintain it in, a safe shutdown condition, as well as the environment within and outside the containment for radiation releases. This type of information is used by the operators to monitor Temelín NPP throughout various anticipated operational occurrences and post-accident conditions. The requirements to provide post-accident monitoring capability are derived from several regulatory criteria.

France

For French plants, the French utility (EDF), after having recorded operator behaviour during tests performed on a full scale simulator, has decided to implement control room modifications.

The analysis based on a control room mockup concluded in favor of two main types of modifications:
- additional information with a computerized operator support called "safety panel" to monitor critical plant parameters and to assist the operator in his diagnosis for the choice of the right procedure;
- improvement of control board layout ergonomy by:
  - grouping some of the informations;
  - improving the labelling;
  - using active mimic diagrams.

Korea, Republic of

The Ulchin Units 3 and 4 are equipped with monitoring instrumentation to detect inadequate core cooling (ICC) and Post Accident Monitoring Instrumentation (PAMI). Examples of parameters monitored are reactor coolant system pressure, temperature, primary safety valve position, containment pressure and site radiation level, etc.

Russian Federation

A new RPV level indication system based on temperature sensors detecting the differential heat transfer from sensor to water and to steam is being developed. An effluent monitoring system for accident conditions is also planned to be installed.

Ukraine

There is a plan for installing a post-accident monitoring system (PAMS), including the RPV-level measuring system in the generic modernization programme. The Rovno NPP Unit 4 is planning to install the RPV level system. GRS from Germany is carrying out a study to define the PAMS parameters for Unit 4.

USA

Regulatory Guide 1.97, which provides acceptable instrumentation characteristics to follow the course of an accident, was issued before the Three Mile Island accident. This guidance was reinforced in a number of generic communications after the Three Mile Island accident.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- 10CFR Part 50, Appendix A, Criteria 3, 4, 13 and 22.
- USNRC Regulatory Guide 1.97, Instrumentation for light-water-cooled nuclear power plants to assess plant and environment conditions during and following an accident, US Nuclear Regulatory Commission.
- USNRC Regulatory Guides (RGs): 1.45, 1.47, 1.53, 1.97, 1.105, 1.153.
- IEEE 497, 1977, "Trial use criteria for post accident monitoring instrumentation for nuclear power generating stations."
- Draft ANS 4.5 D2, "Standard for accident monitoring instrumentation for nuclear plants."
- USNRC Generic Communications: BL 79-06, BL 79-21, BL 79-27, CR 78-19, GL 80-25, GL 80-61, GL 80-62, GL 82-28, GL 83-23, GL 88-14, GL 89-06, IN 84-70, IN 86-10, IN 91-53.

ISSUE TITLE: Water chemistry control and monitoring equipment (primary and secondary) (IC 13)

ISSUE CLARIFICATION:

Description of issue

An accurate and preferably on-line chemical monitoring system is important to enable the operator to respond in time to deviations in the primary and secondary coolant water-chemical condition indices. The specified water chemical conditions must be continuously maintained to avoid corrosion problems in the main equipment.

The chemical monitoring system presently used in some WWERs is more than 10 years old and a great deal of maintenance effort is required to ensure reliable and accurate results. It is also increasingly difficult to obtain the necessary spare parts, since the original supplier plants are located in Georgia, Armenia and Estonia.

Safety significance

The chemical monitoring system is essential to keep coolant parameters within prescribed limits. If it is not, the integrity of physical barriers can be endangered.

Source of issue (check as appropriate)
- __________ operational experience
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria, Russian Federation, Ukraine

The reconstruction of the system is planned at all WWER-1000 units.

Russian Federation

At present, most of the data on control of composition of the coolant of the primary and secondary circuits at NPP with WWER-1000 is acquired by laboratory methods. The systems in use for the chemical control of the primary and secondary circuits coolant allow only to check the compliance of water quality with standards.

The development and application of the systems for on-line automatic chemical control of the coolant of the primary and secondary circuits will allow plant personnel to:
- control coolant quality at operating parameters;
- make a timely diagnosis of the causes of water chemistry deviation during operation of the control system in on-line conditions with operator aid for elimination of the deviation cause;
- monitor automatically the corrosion condition of structural materials;
- assess the residual service life of equipment components.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- MOHT - OTJIG - EDF generic reference programme for the modernization of WWER-1000/320, Revision 6 (February 1997).

ISSUE TITLE: Adequacy of reactor vessel level instrumentation in BWRs (IC 14)

ISSUE CLARIFICATION:

Description of issue

The entry of noncondensible gases into the coolant level gauge (CLG) leads to faulty indications of the reactor coolant level in BWRs. The physical cause is radiolysis-gas generation and its penetration into the CLG as a result of fast pressure transients. Preconditions to that are existing special design features and internal and external leakages on the CLG, where the radiolysis-gas (hydrogen) can enter the reference water-leg by diffusion. Faulty indications occurred during pressure transients with decreased pressure.

Inaccurate readings were observed in reactor vessel water level instrumentation at several BWRs during controlled depressurization while entering plant outages or following reactor trips. These false readings consisted of "spiking" or "notching" of level indication. In one instance, a sustained error in level indication occurred. The root cause of these errors is the effect of non-condensible gas dissolved in the reference leg of "cold reference leg" type water level instruments. Under rapid depressurization conditions, non-condensible gases can cause significant errors in the level indication. Significant spiking may automatically actuate such systems as the primary containment isolation system. After spiking, which is of short duration, the indicated water level returns to actual level.

Bubbling of the gases may eject a significant amount of water from the reference leg, causing a false high water level reading. This occurred during a normal plant cooldown on January 21, 1993 at Unit 2 of the Washington Nuclear Power Plant, resulting in a 32-inch error in level indication that gradually recovered over a period of two hours. If the reactor is rapidly depressurized, as would occur during a design-basis loss-of-coolant accident or opening of the automatic depressurization system valves, even larger errors in the level indication could result. This is a safety significant concern. However, industry analyses indicates that significant errors would not be expected until the reactor is depressurized below approximately 450 psig.

"Cold reference leg" water level instruments measure reactor vessel water level by measuring the differential pressure of two columns of water -- the variable leg and the reference leg. The reference leg is maintained filled to a constant height of water by a condensate chamber located at the top of the reference leg. Non-condensible gases can collect in the condensate chamber and can become dissolved in the water at the top of the reference leg. These dissolved gases can be transported down the reference leg by small leaks in valves and fittings at the bottom of the reference leg.

Safety significance

BWRs use reactor level instrumentation to perform a number of functions including control functions, such as feedwater control, and protective functions, such as automatic scram and autostart of emergency core cooling system.

The CLG is part of the reactor protection system (RPS) and gives signals to actuate the RPS in case of a LOCA. Depending on special conditions, coolant filling levels can be indicated as too high and the RPS-criteria cannot be achieved. Also, depending on specific plant instrumentation configurations, there could be the potential for adverse interactions between the control systems and the protection systems. As an example, the interactions may lead to loss of reactor water level due to automatic termination of normal feedwater (control) and failure to automatically start the emergency feedwater source (protection).

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Germany

Measures taken in the German BWRs:

- Examinations of all CLGs regarding design, construction, operational performance and measurement records over a period of several years.
- Installation of catalysts into the condenser vessel of the CLG in one BWR plant, as a compensatory measure.
- The utilities of all BWRs established a working group for the coordination and performance of further investigations.

Proposed measures:

The objectives of the utilities working group are:

- installation of catalysts in further BWRs;
- installation of a diversified additional CLG.

Japan

This issue has been studied in Japan and it was confirmed that this phenomenon could not occur unless leakage from the instrumentation occurs, because non-condensable gases, dissolved in the water at the top of the reference leg, cannot be transported down the reference leg without leaks at the bottom of the leg.

Based on the operation management in Japan, which considers the highly reliable works and confirmation of soundness by periodic inspection of the water level instrumentation, leakage leading to such phenomenon is not likely to occur. Thus, significant error of water level measurement is not considered in Japanese plants.

Additionally, even if such an error occurs

- The design of water level instrumentation has redundancy.
- Safety systems, like ECCS, can be initiated by other signals than water level.
- Manuals are prepared and operators are trained for the case when water level is not identified.

After thinking about these factors mentioned above, it was concluded that no further measures are necessary.

Sweden

During transients and accidents in boiling water reactors, errors in the level measurement of the reactors pressure vessel can occur due to dissolved non-condensable gases which goes out of solution and due to boiling in the level measurement instrumentation lines.

A verification and validation study (test) of the computer code GOBLIN/WATGAS has been made by the Swedish utilities in co-operation with the Swedish Authority.
The computer code GOBLIN/WATGAS was used for the transient analysis of experiments on level measurement lines with dissolved non-condensables and with boiling effects. The test represents normal cooldown conditions and also depressurization during accidents.

The result (report) concludes that GOBLIN/WATGAS can be used to analyse the course of events in measurement lines and that conservative results will be obtained for measurement errors due to this phenomena in the instrumentation lines.

Sweden participates together with BWR owners in both Finland and Germany in a joint research programme to find and develop diversified additional coolant level gauge (CLG) measurements. In this field, a continued examination of the different CLG design proposals is on-going.

Examples of improvements implemented in the Swedish BWRs core measurement system:

- Actuation (start) of ECCS during LOCA condition is performed independent of the core level measurement signals (low level actuation).
- Installation of a new type of degassing double chamber reference vessel in Forsmark 3 and Oskarshamn 3 and an improved continuous degassing solution for the reference vessel used in Forsmark 1 & 2.

USA

The USNRC has taken several actions to address the problem. The USNRC issued Information Notice 92-54 in July 1992, Generic Letter 92-04 in August 1992, and Information Notice 93-27 in March 1993 to alert licensees and to request information concerning actions taken or planned in response to potential errors in level indication. At the staff's request, the BWR Owners Group submitted a report on May 20, 1993, discussing the impact of level errors on automatic safety system response and operator actions during transients and accidents initiated from reduced pressure conditions during plant cooldown. Based on this information, the WNP-2 event, and data from the BWROG's tests, the staff concluded that additional short-term actions needed to be taken for protection against potential events occurring during normal cooldown. On May 28, 1993, NRC Bulletin 93-03, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation," was issued, in which the staff requested each BWR licensee to implement additional short-term compensatory actions and to implement a hardware modification to resolve this issue at the next cold shutdown after July 30, 1993.

All but one licensee chose to install a backfill modification which would constantly purge the reference leg with a very low flow rate of water supplied by the control rod drive system. The constant flow of water up the reference leg would prevent dissolved gases from moving down the reference leg. One licensee has installed condensing chamber vents to prevent the buildup of gases.

On November 26, 1993, the USNRC issued Information Notice 93-89, "Potential Problems with BWR Level Instrumentation Backfill Modifications," to alert licensees to potential problems that have been identified involving hardware modifications to the reactor vessel water level instrumentation system. This information involved the potential to pressurize the reference legs of the water level instrumentation if a backfill system is installed with the injection point on the instrumentation side of the manual isolation valve in the reference leg. If that valve is closed inadvertently during backfill system operation, the closure could result in a severe plant transient. At some plants, valve closure would cause all safety relief valves to open and potentially impact ECCS response. Licensees were advised to review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems.

ADDITIONAL SOURCES:

- Operational experiences with German NPPs 1994.
- VGB Kraftwerkstechnik 75 (1995), Heft 4.
- NUREG-0933, A prioritization of Generic Safety Issues.
- AEOD/C201, Report on the safety concern associated with reactor vessel level instrumentation in BWRs, US Nuclear Regulatory Commission, January 1982.
- USNRC Bulletin 93-03, May 28, 1993.
- USNRC Generic Letter 92-04, August 1992.
- USNRC Information Notice 93-27, March 1993.
- USNRC Information Notice 92-54, July 1992.

ISSUE TITLE: Improving the detection of primary/secondary leaks (IC 15)

ISSUE CLARIFICATION:

Description of issue

As described in the issue CI 10 "SG tubes integrity", the steam generator (SG) tubes in many PWR plants of different vendors have suffered several degradation phenomena during their operation, thus weakening their function as primary to secondary barrier.

This has led to the need of improving the capability of the primary/secondary leak detection systems, to allow for a more accurate and reliable detection.

Safety significance

Inadequate detection of the potential leaks through the steam generator tubes could result in significant tube failures during operation.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Russian Federation

At present the devices have been developed for automatic control of volumetric activity of live steam and blow down water of SF-UDPG-04R and UDZHG-20R, respectively.

UDPG-04R is intended for measurement of volumetric activity of gamma-radiation of radionuclides in live steam when the device is located in the close vicinity to the steam line under control. Recording of gamma-quanta is performed with the use of a scintillation detection unit. The device may operate in the change of an automated information and measuring system with the following kinds of measurements:
- operative determination of volumetric activity of steam (time of averaging the information is 100s);
- determination of the moment at which the prescribed thresholds for volumetric are exceeded, and the generation of the corresponding signal.

UDZHG-20R is intended for measurement of volumetric activity of gamma-radiating nuclides in fluid and operates with the standard electronic and physical equipment. In the working position the measuring box of the device is connected to the circuit under control with the help of nozzles forming the bypass line. Principle of operation of the device and possibility of its introduction into the channel of information and measurement system are similar, as a whole, to UDPG-04R.

To introduce the system of coolant leak control with the use of devices UDPG-04R and UDZHG-20R the following main measures are required:

(a) trial run of the facilities under actual conditions with possible optimization of the methods of leak monitoring including: examination of leaky SGs, determination of a character of entering and distribution of radionuclides in SG boiler water - at the first stage; development and verification of the model of distribution of radionuclides in boiler water in the course of leaky SG operation for the cases of primary occurrence of leak and operation of SG under stationary conditions - at the second stage; revision of instructions on determination of SG leak values - at the third stage.
(b) development of the project of connection of the monitoring system and equipment
(c) development of the covering documents for the system.

Spain

Spanish NPPs with extensive degradation phenomena in the steam generators have in operation a detection system, based on the monitoring of the short-lived N-16, installed in the steam lines. This system has been introduced into the Technical Specifications. The previous method of leak measurements, based on the monitoring of the steam generators blowdown, is used to confirm the N-16 readings.

After the ongoing replacement of the steam generator in these plants, the N-16 detection system will be maintained and the Technical Specifications leak limits will be relaxed.

ADDITIONAL SOURCES:

- MOHT - OTJIG - EDF generic reference programme for the modernization of WWER-1000/320, Revision 6 (February 1997).
- The safety in the Spanish nuclear power plants, Nuclear Safety Council, Spain, May 1992.
- USNRC Bulletin 89-01 (May-89).
- Supplement 1 (Nov-90).

ISSUE TITLE: Establishment and surveillance of setpoints in instrumentation (IC 16)

ISSUE CLARIFICATION:

Description of issue

The instrumentation and control systems important to safety fall into the following categories: reactor protection systems, engineered safety features control systems, safe shutdown control systems; and other information systems, control systems, and essential auxiliary systems important to safety. Appropriate surveillance procedures and setpoint methodology for instrumentation and control systems important to safety are required to ensure the system operability.

Events on the drift in the setpoints of instrumentation important to safety beyond Technical Specification (TS) limits have been reported in US NPPs, as well as NPPs in other countries. An unplanned change in the setpoint of an instrument will alter the actual value of the measured parameter at which a particular action is to occur. If improper surveillance procedures and/or inadequate setpoint methodology are used, the operability of the aforementioned systems cannot be relied upon to perform the desired safety function.

Safety significance

The errors in the setpoint of an instrument important to safety could result in the delay or defeat the initiation of a safety function.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

USA

In order to address the establishment and maintenance of setpoints for individual safety related instrument channels, the Instrument Society of America (ISA) issued Draft F to ISA S67.04 [1]. This standard was adopted by the NRC and was incorporated in the Proposed Revision 2 to Regulatory Guide 1.105.

ADDITIONAL SOURCES:

- Draft Regulatory Guide and Value/Impact Statement, TASK IC 010-5, Proposed Revision 2 to Regulatory Guide 1.105, Instrument setpoints, US Nuclear Regulatory Commission, December 1981.
- ISA S67.04 (ANSI N719), Draft F, Setpoints for nuclear safety related instrumentation used in nuclear power plants, Instrument Society of America, May 22, 1979.
- Recent USNRC Information Notices 95-05, 95-20, 96-22, 96-41, 96-56, 97-25, 97-33, 98-03.

4.1.8. Containment and other structures (CS)

ISSUE TITLE: Assessment of WWER-440/213 containment dynamic loads (CS 1) (WWER)

ISSUE CLARIFICATION:

Description of issue

The safety concerns about the bubbler condenser containment behaviour are related to two phases of the thermal hydraulic processes following a LOCA: (a) the initial pressure difference acting on the walls of the bubbler condenser system immediately after LOCA, and (b) the long term phenomena accompanying steam condensation in the water trays of the bubbler condenser system. In this context, there are two main issues not fully known and documented since its original design: (1) the thermodynamic behaviour of the bubbler condenser under accident conditions and its effectiveness to fulfill the assigned safety functions; and (2) the structural capacity to cope with the corresponding thermal-hydraulic loads acting on structural elements and components in accordance with acceptance criteria established for specific site and plant conditions.

Preliminary calculations conducted under conservative assumptions by the original Russian designer resulted in a value of 30 kPa for the differential pressure during the initial phase of the accident after an instantaneous double ended guillotine break of the largest pipe in the reactor coolant system, which may be considered as an upper limit until more data and investigations are available. Preliminary and conservative calculations performed on the basis of that value for Mochovce, Dukovany and Bohunice-V2 plants indicated that the bubbler condenser structure design has weak points and more detailed investigations are required before any strengthening measures are decided.

The thermal hydraulic parameters of bubbler condenser long term operation after LOCA have been verified in reduced scale tests. No large scale tests have been performed. Such tests are required to assure that there are no unexpected pressure oscillations with significantly high pressure pulses and fluid-structure interactions dangerous to the integrity and functioning of the bubbler condenser. No such loads have been taken into account for the structural design of the bubbler condenser. Experimental investigations are to be carried out within the framework of CEC/TACIS 95 project on a prototype configuration.

Safety significance

Effectiveness of the bubbler condenser is essential for fulfilling the safety function relating to limiting the maximum pressure after a DBA occurrence and to preventing the release of radioactive products to the environment. To fulfill this objective, efficient condensation of steam in water trays need to be assured without excessive bypassing of trays.

If the bubbler condenser structure (walls and caps) were to fail in the initial moment of the postulated accident, then the water would flow out of its shelves into the bubbler condenser tower early after the accident. This would provide sudden steam condensation and pressure drop inside the containment, but the water would be lost from the bubbler condenser shelves and in the subsequent stages of the accident process the bubbler condenser would not be able to fulfill its safety functions.
The bubbler condenser structural failure can question its safety function and this in turn can lead to the damage of the containment, that is, question the third barrier against release of radioactive products.

Source of issue (check as appropriate)

- __________ operational experience
- __________ *deviation from current standards and practices*
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Czech Republic

Analysis of bubbler condenser containment in NPP Dukovany is conducted within PHARE-93 programme, point 4.2.8. An IAEA mission took place in May 1996 to conduct a plant specific review like in Bohunice NPP.

Hungary

No work has been done so far. The conclusions of AGNES project states, that it would be very desirable to validate the containment thermal hydraulic models on the basis of a full scale experiment and continue the containment strength analyses. As a result the thermal-hydraulic functions of the containment and the bubbler condenser system could be assessed and the interaction between the flowing medium and the structural elements within the containment could be tested (AGNES Summary, page 36).

Russian Federation

The position of the authorities has not been fully declared. The representatives of the organization operating Kola NPP have stated that the existing SAR includes the following calculations:

- mass flow rates of steam-air mixture at the inlet into the bubbler condenser;
- determination of required quantity of water in the bubbler condenser trays;
- hydrodynamic analysis of tray behaviour;
- strength analysis of bubbler condenser metallic structure, without taking into account the dynamic phenomena in the initial period of the accident. There has been so far no direct discussion of the necessity of any further analyses or strengthening of the bubbler condenser structure.

Slovakia

An IAEA mission has reviewed the plant specific status in Bohunice and Mochovce NPPs and proposed the ways of bubbler condenser structure strengthening. An experimental study of the problem has been completed. The necessary modifications have been performed at the Mochovce NPP.

Ukraine

No decisions made as yet. Ukrainian specialists see the problem as the need for individual reinforcements of elements of the bubbler condenser.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- Guidelines for WWER-440/213 containment evaluation, IAEA, WWER-SC-170, 1995.
- Evaluation Guidelines for bubbler condenser metallic structure in WWER-440/213 NPPs containments, IAEA, WWER-SC-095, 1994.
- Review of bubbler condenser structural integrity calculations, IAEA, WWER-SC-142, 1995 (report to the Steering Committee).
- Review of bubbler condenser structural integrity in Mochovce and Bohunice V2 NPPs, IAEA, WWER-SC-159, 1995, (report to the Steering Committee).

ISSUE TITLE: Assessment of BWR containment dynamic loads (CS 2)

ISSUE CLARIFICATION:

Description of issue

The pressure and the level of water in the suppression pool of a BWR containment may oscillate in the process of condensing steam in the pool during the actuation of a safety/relief valve and a LOCA. Several kinds of dynamic loads induced by those oscillations may be applied to the wall and other structures in the pool. Those loads should be assessed quantitatively to verify that the containment integrity is maintained during the actuation of safety/relief valve and a LOCA. The assessment should be based on experimental results other types of quasi-static and dynamic loads induced by events during the actuation of a safety/relief valve and a LOCA such as the pressure increase, jet impingement, and an earthquake should be taken into account in the assessment.

Destruction of submerged pipes has occurred in a Mark II BWR in Germany due to chugging.

The MARK-I and MARK-II owners groups in the US were formed to conduct R&D programmes with General Electric. The USNRC reviewed the results of those programmes and announced the need for improvement of the suppression pool design in 1983.

Safety significance

The safety function of BWR containment to contain radioactive materials and to cool the residual heat in the core would be degraded unless the integrity of the suppression is maintained. Accordingly, the containment should be designed carefully to withstand loads during the actuation of a safety/relief valve and a LOCA and to maintain the integrity during normal and abnormal conditions.

Source of issue (check as appropriate)

- ____x____ operational experience
- _________ deviation from current standards and practices
- _________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Germany, Sweden, USA

The owners groups responded to the NRC's request and improved their plant design base after performing assessment of each plant. Those actions were followed by utilities in Sweden,Germany, and Japan based on R&D results for their respective designs of suppression pools.

Japan

In Japan, "Evaluation Guide of Dynamic Loads on Pressure Suppression System of BWR (Mark I Containment)" and "Evaluation Guide of Dynamic Loads on Pressure Suppression System of BWR (Mark II Containment)" have been established, and loads on containment pressure suppression systems are specified based on those Guides for structural design.

ADDITIONAL SOURCES:

- NUREG-0661, Mark-I containment long term programme safety evaluation report, US Nuclear Regulatory Commission, 1980.
- NUREG-0487, Mark-II containment lead plant program, Load evaluation and acceptance criteria, US Nuclear Regulatory Commission, 1981.
- Safety/relief valve quencher load test evaluation for BWR Mark II and III containments, US Nuclear Regulatory Commission, 1982.
- Evaluation guide of dynamic loads on pressure suppression system of BWR (Mark I containment) was prepared with reference to EPRI NP-906, NEDO-23545, etc. adapting the results of domestic experiments. Evaluation guide of dynamic loads on pressure suppression system of BWR (Mark II containment) was prepared with reference to NEDO-10320, NUREG-0487, etc., adapting the results of domestic experiments.
- EPRI NP-906.
- NEDO-23545.
- NEDO-10320.

ISSUE TITLE: Containment and confinement integrity during severe accidents (CS 3)

ISSUE CLARIFICATION:

Description of issue

In case of a severe accident resulting from a multiple failure of reactor components and safety systems, the pressure and temperature loading of the containment would exceed the design limits and the containment could lose its integrity. The dominant accident phenomena which can lead to overpressurization failure of the containment and which contribute to the main risks can be controlled by severe accident mitigation strategies such as:
- containment venting;
- hydrogen control measures.

The basic idea for containment venting is to open a controlled flow path to the external environment to relieve the pressure that is generated inside the containment due to various processes during the accident.

For example, the main sources of hydrogen generation which can lead to combustible gas mixtures are:

- metal-water reaction involving the fuel element cladding;
- core melt-concrete-interaction;
- radiolytic decomposition of the water in the reactor core and the containment sump.

Inadvertent, random ignitions occurring in such a process may cause uncontrolled deflagrations, which could lead to high flame speeds and corresponding high quasi-static pressure loads for containment structures or even to deflagration-detonation transition phenomena with corresponding dynamic loads. These phenomena may cause e.g. damages to compartment walls, missile effects or local leakages in the containment shell.

The effects of hydrogen burns and the possibility of containment overpressurization failure were confirmed by the TMI-2 accident and by extensive research programmes.

By implementing containment venting and hydrogen control measures, it may be possible to delay or prevent gross structural failure of the containment. This in turn would provide some additional time to mitigate the accident or to reduce the off-site-consequences of the accident compared to those produced by gross containment failure.

Safety significance

Severe accidents can generate phenomena, such as the one described above, which can lead to overpressurization of the containment beyond the design basis and, consequently, may produce the containment failure.

Source of issue (check when appropriate)
- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES

India

Provision for hydrogen monitoring for accident conditions has been introduced in the primary containment.

Japan

In Japan, as the result of an examination of severe accidents, it has been indicated that the possibility of a severe damage of the containment vessel is extremely low, and utilities are planning to implement accident management measures to reduce severe loads leading to the damages of the containment vessel.

For PWRs, it has been decided to install hydrogen combustion devices to cope with severe accidents for the plants with an ice condenser type containment vessel since the risk of damages of the containment vessel from hydrogen combustion is judged to be rather high.

In order to prevent overpressure damage to the containment vessel due to the failure of decay heat removal, such device as the containment vessel internal recirculation air conditioning device is used as a measure against severe accidents to remove heat from the containment vessel. In this way, the integrity of the containment vessel is maintained.

For BWRs, additional measures for hydrogen combustion are not necessary because inert gas is inserted in its containment vessel. In order to prevent over pressure damages, a containment vessel vent system is adopted for BWRs.

Korea, Republic of

To assure containment integrity, the hydrogen mitigation system (HMS) will keep the containment hydrogen concentration below 10% to preclude detonations during and after a severe accident.

The regulatory body requested that the hydrogen igniters be placed so as to achieve controlled hydrogen burning. Local areas of potential high hydrogen concentrations will have some hydrogen burning facilities.

Russian Federation

To reach the probability objectives in management of beyond design basis accidents of the level prescribed in the Russian Federation regulations, the analysis of feasibility to introduce a number of systems is supposed to be performed at the WWER-1000 units under construction:

- system of passive heat removal from the reactor (SPOT);
- system of hydrotanks to prevent the core melting under accidents with primary leaks and loss of active ECCS systems (hydrotanks of "II stage");
- modernization of under reactor space in order to create more favourable conditions for keeping the melted core under severe accidents and to provide for additional inventory of boron solution in the containment to flood the under reactor space;
- introduction of the system for excessive pressure release from the containment with installation of filters;
- system of passive hydrogen recombination under beyond design basis accidents;
- organization of collection of leaks through untightness in the containment under accidents.

Designs of the mentioned systems (except for organization of collection of leaks through untightness in the containment) are developed for Russian NPPs of new generation (for example, AES-92).

USA

Depending on the nature of the accident, venting may involve the release of radioactive fission products from the containment. However, there may be removal mechanisms in the release pathway that reduce the release to the environment. Examples of these mechanisms include:

- scrubbing through a water pool in the BWR containments;
- scrubbing through a bed of ice in the PWR ice condenser containment;
- deposition on surfaces in the reactor building or auxiliary building;
- scrubbing through an external filter.

These removal mechanisms vary not only among containments and reactor types, but also among the individual US plants with a given containment and reactor type. Furthermore, the risk profiles differ from plant to plant and, therefore, selection of venting system and installation and the effects of venting on risk were only determined on a plant specific basis.

In 1981, a detailed action plan for resolving the hydrogen issue was initiated and was limited to near-term rulemaking efforts which included:

- the BWR Mark I and Mark II containments hydrogen inerting rule;
- the ice condenser / Mark III containment hydrogen control rule;
- the near-term construction permit/manufacturing license rule with requirements on inerting of containment atmosphere.

The BWR Mark I and Mark II containments have operated for a number of years with an inerted atmosphere (by addition of an inert gas, such as nitrogen) which effectively precludes combustion of any hydrogen generated.

The rule for BWRs with Mark III containments and PWRs with ice condenser containments was published in 1985 and required that the affected plants be provided with a means for controlling the quantity of hydrogen produced by a 75 % fuel-cladding metal-water reaction, but did not specify the control method.

Large dry PWR containments were excluded from rulemaking because they have a greater ability to accommodate the large quantities of hydrogen associated with a recoverable degraded core accident than the smaller Mark I, II, III and ice condenser containments. Analyses which were performed to determine the pressure in a dry containment resulting from the combustion of hydrogen corresponding to a 75 % metal-water reaction, following onset of a degraded core accident and while the containment was still near its peak pressure, indicated that the peak total containment pressure was below the failure pressure.

ADDITIONAL SOURCES:

- MOHT - OTJIG - EDF generic reference programme for the modernization of WWER-1000/320, Revision 6 (February 1997).
- NUREG-0933, A prioritization of Generic Safety Issues, Item A-48, Emrit, R. Division of Safety Issue Resolution, US Nuclear Regulatory Commission, September 1994.

4.1.9. Internal hazards (IH)

ISSUE TITLE: Need for systematic fire hazards assessment (IH 1)

ISSUE CLARIFICATION:

Description of issue

In order to ensure safety, an adequate degree of fire protection should be provided in nuclear power plants. A fire hazards analysis including consequences of fire fighting should be performed before initial fuel loading and be updated during operation to verify that the main safety functions to shut down the reactor, to remove residual heat, and to contain radioactive material are maintained against the consequences of a fire.

The analysis has six separate purposes:
(1) identification of important items to safety and their locations in fire compartments;
(2) analysis of the anticipated fire growth and the consequences of the fire and fire fighting activities with respect to items important to safety;
(3) determination of the required resistance of fire barriers;
(4) determination of the type of fire detection and protection means to be provided;
(5) identification of cases where additional separation or fire protection is needed, especially for common mode failures, to ensure that items important to safety will remain functional during and following a credible fire;
(6) verification that the safety systems to shut down the reactor, to remove residual heat and to contain radioactive material are designed against the consequences of a fire.

The threat of fire to the safety of a nuclear power plant has been recognized after the incident at Browns Ferry Unit 2 in 1970 and has resulted in safety rules and guides for fire protection.

Safety significance

Previous systematic fire hazards analyses have shown, for particular plants, that accidents could be initiated and safety functions impaired as a consequence of a fire.

Source of issue (check as appropriate)
- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Most of the member states with a nuclear power programme have issued rules and safety guides for fire protection in nuclear power plants. The IAEA NUSS Programme also has documents related to this topic.

Bulgaria

A fire hazards analysis has been started within the framework of the PSA for Kozloduy NPP Units 5 and 6.

Czech Republic

For the Temelín NPP, a systematic fire hazards analysis is currently being carried out by the Designer (EGP Prague). In addition, the probabilistic fire hazards analysis, included in the scope of NPP Temelín PSA to investigate the potential for contribution of accident sequences induced by in-plant fires to the overall CDF, is nearly finished. The models are developed in a systematic manner accounting for the recommendations contained in various accepted procedure guides for performing probabilistic fire analyses (NUREGs 4840, 5042, 2815, USNRC Generic Letter 88-20, Supplement 4 and NUREG-1407). The independent review of this PSA task is to be carried out in the framework of a 2nd IAEA IPERS (Level 1 - external initiating events) in August/September 1995.

France

Rules of fire protection are defined in the French standard called RCC-I (Règles de Conception et de Construction des centrales nucléaires REP applicables à la protection Incendie: Rules of Design and Construction in PWRs NPP applied to fire protection). That standard has been modified in 1987 and 1992, taking in account experience feedback. Safety Authority asked EdF to apply these new standards to existing NPP. Consequently, EdF has undertaken an important action plan for fire protection, for all French PWRs, called PAI (Plan d'Action Incendie), PAI programme should be achieved in 2002.

Germany

Fire protection rules and requirements are defined in the German KTA (Kerntechnischer Ausschuß) Standards 2101 "Fire Protection in Nuclear Power Plants." These standards are in principle applied to the existing German NPPs. These standards demand a fire protection concept which includes several of the items mentioned in the description of the issue. Several safety reviews have been carried out for the German NPPs. In the frame of these reviews also the fire protection level of the operating plants in Germany was analysed focusing on the identification of deviations from recent standards and guidelines. In general, the adequacy of the design and plant layout with respect to fire protection means could be confirmed by these reviews. Nevertheless, various improvements and upgrading measures were implemented in the different plants for optimization of the fire safety.

Fire hazard analyses have been carried out by GRS as well as by the utilities themselves for PWR as well as for BWR reactors. The available risk studies for German NPPs are based on these fire hazard analyses.

India

Indian BWRs were designed to meet the fire standards of 1960s. Special reviews have been carried out and measures initiated to carry out feasible corrections for deficiencies from the current standards.

Japan

In Japan, there are "Review Guide for Fire Protection of Light Water Reactor Power Facilities" and "Guide for Fire Protection of Nuclear Power Plants (JEAG4607)" based on the Review Guide. They clarify measures to be taken during fire accidents. To be specific, they stipulate to shut down the reactor and to secure the functions of decay heat removal during fire accidents. The requirements of the Review Guides are as follows:

(1) Fire prevention measures composed of control of combustible materials, leakage protection of flammable materials, electrical fire protection due to over current, and prevention of natural event induced fire.
(2) Fire detection and suppression measures shall be provided in order to protect the safe shutdown function in fire.
(3) Fire mitigation measure like fire resistance wall, fire barrier, distance, and sor forth which mitigate the fire effect to safe shutdown functions.

The adequacy of the design had been confirmed in the Periodical Safety Review.

Korea, Republic of

At the Ulchin NPP Units 3 and 4, a fire protection programme is established to meet the BTP CMEB 9.5-1 of the SRP, NFPA code, and Korean fire protection laws. Fire barriers are designed in compliance with the BTP CMEB 9.5-1, NEPA code, and provided to separate redundant trains of safety related systems from any potential fires in non-safety related areas and with a minimum fire resistance rating of 3 hours, which meets the requirements of BTP CMEB 9.5-1.

The fire suppression system design is based on the evaluation of potential fire hazards throughout the plant and the effects of the postulated fires on the ability to achieve and maintain a safe shutdown condition, the ability to minimize and control the release of radioactivity to the environment, and the ability to protect plant assets and personnel, which comply with the requirements of BTP CMEB 9.5-1.

Fire detection systems are provided for all areas that contain or present a fire exposure to safety related equipment and comply with the requirements of 10CFR50, GDC, NFPA and Korean fire protection laws.

The fire hazard analysis demonstrates that the plant will perform safe shutdown functions and minimize radioactive release to the environment in the event of fire.

Sweden

In Sweden, the regulatory body requires that a fire hazards analysis and a flooding assessment be a part of the second round of the periodic safety reviews, the ASAR 90 programme. These where performed by the utilities in the 90s.

Ukraine

A programme is presently underway for a systematic analysis in order to detect possible deficiencies and to correct them. The analysis is based on the relevant national standards as well as NUSS 50-SG-D2, using a deterministic approach.

A fire hazards analysis has been carried out by Germanischer Lloyd for Rovno NPP Unit 3. The resulting recommendations will be used as the basis for the analysis of Unit 4 (under construction). This analysis is expected to be carried out prior to the commissioning of Unit 4. A PSA, which also includes the fire hazards, is planned to be carried out at a later date.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Fire protection in nuclear power plants, A Safety Guide, Safety Series No. 50-SG-D2 (Rev.1), IAEA, Vienna (1992).
- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Fire safety in the operation of nuclear power plants (Draft Report).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Evaluation of fire hazard analyses for nuclear power plants (IAEA Safety Series No. 50-P-9, 1995).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment of the overall fire safety arrangements at nuclear power plants (IAEA Safety Series No. 50-P-11).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Preparation of fire hazard analyses for nuclear power plants, Safety Reports Series No. 8, IAEA, Vienna (1998).

- INTERNATIONAL ATOMIC ENERGY AGENCY, Treatment of internal fires in probabilistic safety assessment for nuclear power plants, Safety Reports Series No. 10, IAEA, Vienna (1998).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Fire hazard analysis for WWER nuclear power plants (IAEA-TECDOC-778, 1994).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Organization and conduct of IAEA fire safety reviews at nuclear power plants, IAEA Services Series No. 2, January 1998).
- RCC-I, France, Plant design and construction rules for fire protection, 1992.
- RFS, Basic Safety Rule, France, V.2.j: General rules for fire protection, 28 December 1982.
- Safety analysis for boiling water reactors (Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, GRS-98, ISBN 3-923875-48-7, September 1985).
- Kerntechnischer Ausschuss (KTA), Germany, Nuclear Safety Standards, KTA 2101: Fire protection in nuclear power plants.
- Optimierung von Brandschutzmaßnahmen und Qualitätskontrollen in Kernkraftwerken (Gesellschaft für Reaktorsicherheit (GRS) mbH, GRS-62, ISBN 3-923875-10-X, September 1985).
- USNRC Generic Letter 86-10, Implementation of fire protection requirements.
- USNRC Generic Letter 86-10, Supplement 1, Fire barrier endurance test criteria.
- Recent USNRC Information Notices 96-33, 97-01, 97-73.

ISSUE TITLE: Adequacy of fire prevention and fire barriers (IH 2)

ISSUE CLARIFICATION:

Description of issue

In order to ensure safety, an adequate degree of fire protection should be provided in nuclear power plants. This should be achieved by a defence in depth concept including the prevention of fires from starting. In designing the plant, the amount of combustible materials and the fire load should be kept to a reasonably achievable minimum and operation or failure of any plant system should not cause fire. The on-site use and storage of combustible materials in areas adjacent to or containing items important to safety should be controlled.

In order to maintain the function of the safety systems to shutdown the reactor, to remove residual heat, and to contain radioactive material, they should be protected against the consequences of a fire. The redundant parts of safety systems should be sufficiently segregated from each other so that a fire affecting one redundancy will not prevent the safety systems from performing the required safety functions. Fire barriers between redundant systems should be qualified to fulfil their protective functions considering the time specified with the fire hazards analyses and the presence of automatic fire suppression system.

The threat of fire to the safety of a nuclear power plant has been recognized after the incident at Browns Ferry Unit 2 in 1970 and has resulted in safety rules and guides for fire protection.

Safety significance

Insufficient protection against common mode failures due to fire would seriously affect defence in depth. Safety functions could then be questionable, depending on the loss of redundant trains during DBA scenarios.

Source of issue (check as appropriate)

- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Most of the member states with a nuclear power programme have issued rules and safety guides for fire protection in nuclear power plants. The IAEA NUSS Programme also has documents related to this topic.

Bulgaria

The fire protection measures for the Kozloduy NPP were designed according to the standards applicable at the time of the original design of the plant. Some modifications have subsequently been made on the basis of VSN-01-87. Overlayers for covering cables are in place. The performance of these overlayers have been demonstrated in Bulgaria and in Moscow. Experiments have also been carried out for cables, for fire protection doors and for fire resistant coatings. The requirement related to the location of oil tanks outside the containment in separate rooms could not be fulfilled. To compensate for this, additional measures for fire detection and extinguishing have been implemented in agreement with the fire protection authorities. The separation between fire areas was designed in accordance with the original design standards applicable to the plant. Based on the results of the fire hazards analysis, further work is planned to be carried out.

Czech Republic

For the Temelín NPP, fire barriers, non-inflammable cables and flaps in the HVAC ducts have been considered. The oil lubrication of the RCPs has been retained. Additional fire fighting equipment has been installed on the RCPs. The monitoring is provided by TV cameras enabling the operator to switch the fire fighting system on manually. The 6 kV main distribution boards and 0.4 kV are arranged on separate floors and, in addition to that, each 6 kV board is located in a separate room. The fire protection zones have been rearranged. All distributions in the electrical distribution building are provided with carbon dioxide fire fighting equipment.

France

See IH 1.

Germany

In addition to IH 1, several studies to confirm the adequacy of the existing fire barriers and other fire prevention measures have been carried out.

India

Indian BWRs were designed to meet the fire standards of 1960s. Special reviews have been carried out and measures initiated to carry out feasible corrections for deficiencies from the current standards.

Fire resistant coatings have been provided on the cables. Fire barriers are being introduced.

Korea, Republic of

See IH 1.

Russian Federation and Ukraine

The use of qualified fire doors are presently being implemented. In addition, there are activities to replace oil lubrication by water lubrication using modified reactor coolant pumps, and non-inflammable liquids are planned to be used.

The fire hazards analysis conducted for the Rovno NPP Unit 3 (under construction) identified several weaknesses. The results are also planned to be used for making in Unit 4. These relate to the replacement of fire doors, overlayers for cables, penetrations and connections between fire areas, the spatial separation of the two control rooms and the use of non-combustible fluids for the lubrication of the main coolant pumps.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Inspection of fire protection measures and fire fighting capability at nuclear power plants (IAEA Safety Series No. 50-P-6, 1994).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment of the overall fire safety arrangements at nuclear power plants (IAEA Safety Series No. 50-P-11).

- INTERNATIONAL ATOMIC ENERGY AGENCY, Organization and conduct of IAEA fire safety reviews at nuclear power plants, IAEA Services Series No. 2 (1998).
- Kerntechnischer Ausschuss (KTA), Germany, Nuclear Safety Standards, KTA 2101: Fire protection in nuclear power plants.
- USNRC Bulletins 92-01 and 92-01 Supplement 1, Failure of thermo-lag 330 fire barrier system to perform its specific fire endurance functions.
- USNRC Generic Letter 92-08, thermo-lag 330-1 fire barriers.
- Recent USNRC Information Notices 95-27, 95-32, 95-49, 95-49 Supplement 1, 95-52, 97-48, 97-59, 97-70, 97-72.

ISSUE TITLE: Adequacy of fire detection and extinguishing (IH 3)

ISSUE CLARIFICATION:

Description of issue

In order to ensure safety, an adequate degree of fire protection should be provided in nuclear power plants. This should be achieved by a defence in depth concept including fire detection and extinguishing. A nuclear power plant should have a sustained capability for early detection and effective extinguishing of a fire to protect items important to safety.

Fire protection and extinguishing systems of appropriate capacity, capability, and qualification should be provided to give timely alarm and or actuation so as to enable speedy extinguishing of the fire and to minimize the adverse effects of fires on personnel and on items important to safety.

In order to maintain the function of the safety systems to shutdown the reactor, to remove residual heat, and to contain radioactive material, they should be protected against the consequences of a fire. The redundant parts of safety systems should be protected such that a fire affecting one redundancy will not prevent the safety systems from performing the required safety functions.

The threat of fire to the safety of a nuclear power plant has been recognized after the incident at Browns Ferry Unit 2 in 1970 and has resulted in safety rules and guides for fire protection.

Safety significance

If the fire detection and extinguishing equipment fail to operate, accidents could be initiated and the safety functions could be impaired.

Source of issue (check as appropriate)

- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Most of the member states with a nuclear power programme have issued rules and safety guides for fire protection in nuclear power plants. The IAEA NUSS Programme also has documents related to this topic.

Bulgaria

The Kozloduy NPP plans to modify the gas fire extinguishing system, to improve the startup speed of diesel fire pump and to seismically qualify the fire alarm facilities.

Czech Republic

At the Temelín NPP, all the parts of water fixed sprinkler devices (piping systems and also water supply) protecting cable areas and cable uptakes of the reactor islands are capable of resisting earthquakes. In addition, to increase the degree of fire protection (reactor hall), it is planned to also build high pressure fire water distribution (including water source) for manual extinguishing which is capable of resisting earthquakes. Water supply of fire extinguishing systems within the containment is provided by safety related service water system.

France

See IH 1.

Germany

In addition to IH 1, reliabilities of fire detection and extinguishing features have been plant specifically and generically determined for two reference plants by GRS.

India

Indian BWRs were designed to fire standards of 1960s. Hydrogen detection has been provided in primary containment and main DC battery rooms. Many more sensors have been introduced at locations such as RB, TOT, etc.

Japan

In Japan, there are "Review Guide for Fire Protection of LWR" and "Guide for Fire Protection of Nuclear Power Stations (JEAG4607)" based on the Review Guide. They clarify the following matters to be taken into consideration in designing firre alarms and fire fighting facilities:

(1) Installation of fire detectors suitable for the effects and characteristics of fire and the environmental conditions.
(2) Measures of maintaining the functions of fire detectors in case of loss of normal power supplies.
(3) Reliability and capacity of fire fighting water supply system sources.
(4) Redundancy and diversity of fire fighting pumps.
(5) Measures of maintaining the functions of fire fighting pump system in case of loss of normal power supplies.

In addition, other laws, including the Fire Act, stipulate what should be taken into consideration in designing as to the performance of fire detectors and the water sources of fire fighting water supply systems. Redundancy and diversity of the fire fighting pump system and compliance to the law on the location and numbers are confirmed in the course of PSR, as well.

Korea, Republic of

See IH 1.

Russian Federation

The improvements of fire detection and extinguishing systems have been taken into account in the Russian reconstruction programme SM-90-WWER. The design features for the actuation of the fire water supply system are also planned to be improved.

Ukraine

Measures for upgrading the fire detection and alarm system are envisaged within the programme of fire safety improvement for Ukrainian NPPs.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Fire protection in nuclear power plants, A Safety Guide, Safety Series No. 50-SG-D2 (Rev.1), IAEA, Vienna (1992).
- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.

- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Inspection of fire protection measures and fire fighting capability at nuclear power plants (IAEA Safety Series No. 50-P-6, 1994).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment of the overall fire safety arrangements at nuclear power plants (IAEA Safety Series No. 50-P-11).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Organization and conduct of IAEA fire safety reviews at nuclear power plants, IAEA Services Series No. 2, January 1998.
- Ermittlung von Zuverlässigkeitskenngrößen für Brandschutzeinrichtungen in deutschen Kernkraftwerken (Röwekamp, M. et al, BMU-Schriftenreihe BMU-1997-486, in press 1997).
- Kerntechnischer Ausschuss (KTA), Germany, Nuclear Safety Standards, KTA 2101: Fire protection in nuclear power plants.

ISSUE TITLE: Adequacy of the mitigation of the secondary effects of fire and fire protection systems on plant safety (IH 4)

ISSUE CLARIFICATION:

Description of issue

The secondary effects of fire are related to the interactions with the necessary means to fight the fire and to operate the plant. Inadvertent or intended activation of the fire protection systems (FPS) should not detrimentally affect the other systems.

All types of automatic and manual fire suppression systems (e.g., water, halon, CO2) have to be considered as potential sources of damage:

- water spray can make safety related systems inoperable; this can happen during a fire or by inadvertant actuation of FPS;
- gas release could be harmful for the operators working in the area;
- water release can dilute borated water contained in the pools;
- ventilation could be rendered inoperable by spurious shutting of fire dampers.

Smoke can impact plant safety in several ways:

- reducing manual fire-fighting effectiveness;
- damaging electronic equipment;
- hampering an operator's ability to safely shutdown the plant;
- initiating automatic fire protection systems in areas not affected by the fire.

Safety significance

An FPS actuation which results in an adverse interaction with the safety systems of the plant or the secondary effects of a fire could reduce the availability of systems needed to achieve and keep safe plant shutdown conditions or to mitigate accident consequences.

Source of issue (check as appropriate)
- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

At the Kozloduy NPP, there are provisions to close the dampers and automatically shut down the associated ventilation system in order to prevent a propagation of the fire.

Czech Republic

At the Temelín NPP, the rooms with a potential fire danger and evacuation corridors have been redesigned so as to comply with the standards.

France

For the French plants, an overall assessment has been performed to update the fire policy, taking into account the potential adverse secondary effects of a fire and the fire protection systems. Based on the results of this assessment, modifications and improvements have been carried out.

Germany

For the German NPPs, an overall analysis has been carried out to assess the potentially adverse effects of a fire or the operation of active fire protection equipment on safety related systems and equipment. Based on this analysis, improvements and/or modifications have been carried out where necessary.

Due to these potential secondary effects of fire and fire protection on plant safety in particular plant areas, the fire extinguishing systems are not actuated automatically.

Korea, Republic of

See IH 1.

Ukraine

Measures for removing the smoke from rooms and corridors are proposed within the programme of fire safety improvement for Ukrainian NPPs.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- Main findings from reviews of the fire protection measures in older German NPPs (Röwekamp, M., Riekert, T.; Proceedings of the OECD Specialist meeting on Fires and fire protection systems in nuclear power plants, December 1993).
- Kerntechnischer Ausschuss (KTA), Germany, Nuclear Safety Standards, KTA 2101: Fire protection in nuclear power plants.
- Recent USNRC Information Notices 95-33, 95-36, 97-82.

ISSUE TITLE: Need for systematic internal flooding assessment including backflow through floor drains (IH 5)

ISSUE CLARIFICATION:

Description of issue

Direct flooding from liquid releases as well as backflow flooding of safety systems through drains is a potential problem with implications that are broad in nature. This problem applies only to older plants (see also IH4, Adequacy of the mitigation of the secondary effects of fire and fire protection systems on plant safety).

In a WWER NPP, an inadvertent activation of the fire extinguishing water system led to a flooding of the auxiliary control room.

In addition, in WWER NPPs, emergency power distribution boards for the 6 kV, the 0.4 kV and for DC are located within the circumferential building next to the reactor. Fire extinguishing systems located in the floors above the switchgear rooms can lead to water ingress into the electrical rooms underneath when activated. Metal roof structures have been mounted to protect the switchgear cabinets from the water. Water may flood these rooms because no drainage system is installed. The emergency power supply to the ECCS systems and residual heat removal system may be lost in case of flooding in the rooms of emergency electric power distribution boards.

Safety significance

If the safety and support systems are not adequately protected against internal flooding hazards, the safe shutdown of the plant can be impaired or an accident initiated.

Source of issue (check as appropriate)
- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

The Kozloduy NPP plans to perform an analysis of internal flooding.

Czech Republic

For the Temelín NPP, a systematic flooding analysis is currently being carried out by the plant's designer (EGP Prague) in accordance with IAEA recommendations. In addition, the internal flooding analysis was included in the scope of the Temelín NPP PSA in order to investigate potential contribution of flooding induced accident sequences to the overall CDF. The independent review of this PSA task was carried out in the framework of the 2nd IAEA IPERS (Level 1- external initiating events) in January 1996.

France

A leakage of water from any circuit can be an initiator of the loss of safety function, particularly if flooding can affect two redundant trains.

Several solutions are used to avoid safety function to be threatened:
– installation of redundant trains in physically separate rooms;

- installation of drain devices in rooms which cannot be submitted to possible flooding;
- raising of equipment;
- retention pits around tanks;
- sump pumps triggered by detection of a quick variation of level in sumps;
- detection of abnormal low level in tanks, etc.

Germany

For German NPPs, a uniform protection approach regarding internal flooding does not exist. For older plants, special investigations on the effects of internal flooding by pipe rupture or failures during maintenance were included in the comprehensive safety reviews which were carried out between 1986 and 1988. As a result, in some cases, plant specific backfitting measures were necessary, e.g. leak detection system, water barriers on floor, isolating devices in systems with the potential of large water quantity release.

For the latest generation of German PWRs and BWRs, a physical separation of the safety trains is applied. To prevent effects of internal flooding on more than one redundancy, a specific detection of leaks followed by isolating the affected train is implemented.

India

Specific review of internal flooding aspect carried out for few such report incidents and corrective measures taken.

Korea, Republic of

The flood protection measures for seismic category I structures, systems and components are designed in accordance with Reg. Guide 1.102.

The safety related structures are designed to maintain a dry environment during all floods by incorporating the following safeguards into their construction:

- No exterior access openings will be lower than 1 foot above plant grade elevation.
- The finished yard grade adjacent to the safety related structures will be maintained at least 1 foot below the ground floor elevation.
- Waterstops are used in all horizontal and vertical construction joints in all exterior walls up to flood level elevation.
- Water seals are provided for all penetrations in exterior walls up to flood level elevation.
- Water proofing of walls subject to flooding is provided.

A permanent safety grade dewatering system is installed.

Ukraine

A complete analysis of an internal flood in the reactor building and the turbine hall is planned.

USA

The solution to this problem is having check valves in the floor drain system as a permanent fix or inflatable drain plugs as a temporary fix. This issue was treated with high priority because of its risk significance.

Generic Safety Issue 77 was folded into A-17. The IPE programme already included the evaluation of internal flooding and the insights from USI-A-17 was included in IPE guidance documents.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- USNRC Regulatory Guide 1.46, Protection against pipe whip inside containment.
- NUREG-0933, Generic Safety Issue 77.
- USNRC Generic Letter 89-18, Resolution of unresolved Safety Issue A-17, Systems interaction in nuclear power plants.
- USNRC Information Notice 89-63, Possible submergence of electrical circuits located above the flood level because of water intrusion and lack of drainage.
- USNRC Information Notice 87-49, Deficiencies in outside containment flooding protection.
- USNRC Information Notice 83-44, Potential damage to redundant safety equipment as a result of backflow through the equipment and floor drain system.

ISSUE TITLE: Need for systematic assessment of high energy line break effects (IH 6)

ISSUE CLARIFICATION:

Description of issue

This issue is related to the dynamic and environmental effects of high energy piping breaks inside and outside the containment and the consequences on plant safety.

High energy piping are those containing fluids with operating conditions exceeding certain limits of pressure or temperature, producing an energy release in the case of a break (or crack) affecting safety systems or structures.

The possible consequences are:

- dynamic effects caused by the whipping of the ruptured pipe;
- effects resulting from fluid flow, jet impingement, irradiation and contamination;
- variation in local ambient conditions (pressure, temperature, humidity, floods).

Safety significance

Pipe ruptures may lead to safety systems, equipments, structures and containment being damaged and/or the accident mitigation being jeopardized.

Source of issue (check as appropriate)

- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

At the Kozloduy NPP, mechanical justification of supports for safety related piping is planned.

There are no restraints on the secondary piping either inside or outside the containment. The plant proposes to perform an analysis of the risk of secondary pipe whip and, depending on the results, to consider installing pipe restraints. A further consideration is the mechanical substantiation of the safety important pipings and their supports, also with respect to seismic qualification.

Czech Republic

The Temelín NPP applied the LBB concept for the large diameter piping of the primary circuit. The plant general designer and the plant itself are now reviewing the plant system interaction and separation.

France

Systematic assessment of high energy line break comprises different types of assessment depending on effects to study:

- effects on containment atmosphere pressure and on pressure in the bunkers of different components: pressurizer, primary coolant pumps, SG, reactor pit, depending on the break localization;
- overspeed of primary coolant pumps;
- effects on vessel internals;

- effects on GV structure and internals;
- consequences on fuel (thermo-hydraulic studies);
- mechanical loads on primary coolant circuit supporting elements;
- mechanical loads on the different primary circuit loops;
- consequences on ECCS and other safety related electrical and mechanical equipment dimensioning.

Furthermore other parameters as ageing, stretch out, modifications of fuel characteristics must be taken in account when assessing these effects.

Localization, size (double-ended break or limited break), types of ruptures: longitudinal or circumferential, have to be considered depending on the type of assessment (for example a limited break can be dimensioning for the mechanical loads on primary circuit and supporting elements, as the result of a higher pressure in the circuit, whereas a double-ended break has to be considered for containment pressure assessments).

Different classical types of devices are used to avoid pipe whips: snubbers, whip restraints, jet deflectors, etc.

In addition there are French studies on a leak before break (LBB) approach, similar to the American ones (GDC 4). This concept LBB is used in the German-French project EPR for the main primary coolant loops and for the short section of main secondary loop between the external side of containment and the main steam isolation valves.

Korea, Republic of

The fluid system piping is designed in accordance with the high energy pipe criteria of App. B of BTP ASB 3-1 with respect to protection against pipe breaks outside the containment.

The plant design accommodates the effects of postulated pipe breaks by means of physical separation, enclosure indubitably designed structures or compartment, drainage systems, pipe whip restraints, and equipment shields.

Separation between redundant safety systems with their auxiliary supporting features is the basic protective measure.

There are not high energy lines in the vicinity of the control room. Therefore, there are no effects upon the habitability of the control room by pipe break, either from pipe whip, jet impingement, or transport of steam.

By means of design features, such as separation, barriers, and pipe whip restraints and jet impingement shields, the effects of a pipe break will not damage essential systems to an extent that would impair their safety function.

Russian Federation

Gidropress is working on the LBB concept application to the main circulating line (MCL) (∅850 mm) at Balakovo NPP. The connecting pipes inside the hermetic zone have not been included.

Ukraine

The studies to apply the LBB concept to primary circuit piping have been initiated on a generic basis. For the Rovno NPP Unit 4 and the Khmelnitsky NPP Unit 2, the programme has not yet been fully finalized. The intention is to perform LBB concept application on MCL and on some safety pipings connected to the MCL inside the hermetic zone.

With respect to the dynamic effects of main steam and feedwater line breaks, the following measures have been proposed:

- development and implementation of a system for pipeline dynamic fastening (SPDF) aimed at absorbing loads due to pipeline failure;
- development of the LBB concept for secondary circuit pipelines;
- assessment of the operating reliability of electrical system components, monitoring and control devices for possible mechanical and temperature effects caused by pipeline breaks.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- NUREG/CP-0155, "Proceedings of the seminar on leak-before-break in reactor piping and vessels" held in Lyon, France, October 9-11, 1995, US Nuclear Regulatory Commission, April 1997.

ISSUE TITLE: Need for assessment of dropping heavy loads (IH 7)

ISSUE CLARIFICATION:

Description of issue

In nuclear power plants heavy loads may be handled in several plant areas. If these loads were to drop in certain locations in the plant, they may impact spent fuel, fuel in the core, or equipment that may be required to achieve safe shutdown and continue decay heat removal.

In the case of the WWER NPPs, the polar crane in the reactor building lacks adequate interlocks. Interlocking is required to prevent the simultaneous transport of heavy loads over the reactor and spent fuel pool and a decoupling of the crane forks and hooks.

Safety significance

The drop of a heavy load on to the reactor or spent fuel pool could lead to a damage of the spent fuel or to a loss of the cooling capabilities and to a consequential release of radioactive materials. Additionally, if fuel is of sufficient enrichment, the normal boron concentrations that are maintained may not be sufficient to prevent a load drop from causing the fuel configuration to be crushed and result in criticality.

Source of issue (check as appropriate)

- __________ event
- ____xx____ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

At the Kozloduy NPP, the transport routes have been established such that unnecessary transport over the reactor and the spent fuel pool are avoided.

Czech Republic

At the Temelín NPP, disengaging of the suspension or crane hook is avoided by electric interlock of the crane fork pin motion. The engine of the pin drive mechanism is of such dimension that when the pin is loaded, it will not be set into motion. Transport of loads above the reactor and the spent fuel pond is interlocked. Handling is carried out in polar coordinates outside of this restricted area.

France

An example of protection against dropping of heavy loads is the heavy plate to protect the vessel head.

India

Established transport routes avoid unnecessary movement over reactor and spent fuel pool.

Korea, Republic of

The fuel building overhead crane with a cask handling hoist and a fuel handling hoist is mounted on the rail that will extend the entire length of the fuel building.

During construction, the overhead crane will travel the entire rail without any provisions for restrictions: however, provisions will be installed permanently once fuel assemblies are onsite to restrict movement of the crane over the spent fuel pool area safe heavy load handling.

These provisions will be mechanical, electrical, or a combination of mechanical and electrical, including the automatic stop of bridge, trolley and hoist movement, automatic control of bridge and hoist speed, and automatic cutoff of heavy load limit and lifting height.

In addition, procedure and/or administrative cotnrols will be provided to ensure safe operation for the fuel building overhead crane and spent fuel handling machine, and to control the safe load path and safe lifting practice.

The containment polar crane will have a main hoist and an auxiliary hoist to handle the various loads during an outge. Provisions will be provided to ensure safe heavy load handling in the containment. These provisions will include automatic control of bridge and hoist, automatic control of heavy load limits and lifting height, and a load handling path to prevent any fuel damage from a heavy load drop.

Russian Federation and Ukraine

In Russian and Ukrainian NPPs, there are special organizational measures to ensure that loads are transported outside the zones of essential equipment, and there are plans to provide cranes with an interlocking system. At Rovno NPP Unit 3, the following measures have been implemented:

- The crane can only be operated by simultaneously depressing two separate push-button controls. Both these push-button controls are within one and the same control cabin. However, they are about 1 m apart from each other.

- There are technical measures to prevent a transport over the reactor.

- The heavy loads are transported on two hooks and two chains (redundancy).

The same technical measures will be implemented in Unit 4.

USA

- Assure that there is a well designed handling system.

- Provide sufficient operator training, load handling instructions, and equipment inspection to assure reliable operation of the handling system.

- Define safe load travel paths and procedures and operator training to assure to the extent practical that heavy loads are not carried over or near irradiated fuel or safe shutdown equipment.

- Provide mechanical stops or electrical interlocks to prevent movement of heavy loads over irradiated fuel or in proximity to equipment associated with redundant shutdown paths.

- Where mechanical stops or electrical interlocks cannot be provided, provide a single-failure-proof crane or perform load drop analyses to demonstrate that unacceptable consequences will not result.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- NUREG-0800, Standard Review Plan, Section 9.1.5, Overhead heavy load handling systems, July 1981.
- NUREG-0612, Control of Heavy Loads at Nuclear Power Plants, January 1980.
- USNRC Bulletin 96-02, Movement of heavy loads over spent fuel, over fuel in the reactor core, or over safety related equipment.
- USNRC Information Notice 96-26, Recent problems with overhead cranes.

ISSUE TITLE: Refueling cavity seal failure (IH 8)

ISSUE CLARIFICATION:

Description of issue

On August 21, 1984, the Haddam Neck plant experienced failure of a refueling cavity seal during preparations for refueling. The failure of the seal caused 200,000 gallons of water to drain from the refueling cavity into the lower levels of the containment building in 20 minutes. No fuel was being transferred at the time. If a similar seal failure were to occur at a plant during fuel transfer, fuel elements could be uncovered and could result in high radiation exposure to plant personnel, possible fuel cladding failure, and release of radioactive material. Also, because the refueling cavity is connected to the spent fuel storage pool, the potential exists for this seal failure to initiate drainage of the spent fuel pool, if the fuel transfer canal were open at the time.

Pneumatic rubber seals similar to the one at Haddam Neck (used mainly at PWRs) are most vulnerable to failure. They are susceptible to misalignment, improper inflation, puncture, and rupture. Other types of seals, such as the permanent steel bellows on most BWRs, have been more reliable than pneumatic seals.

A refueling cavity seal failure is itself considered to be an initiating event for an accident sequence. The immediate result of a refueling cavity seal failure during fuel transfer is the loss of water from the refueling cavity. It is important to perform a systematic analysis of the potential consequences of such failures including the procedural measures which should be put in place to respond to such an event.

Safety significance

The possible safety consequences are as follows: (1) high radiation levels in the containment due to uncovering of spent fuel in transfer; (2) radioactive material release in the containment building due to rupture of fuel pins (by self-heating after uncovering); (3) high radiation levels in the spent fuel building due to uncovering of stored spent fuel; and (4) radioactive material release outside the containment building due to rupture of fuel pins in the storage pool. The consequences involving the spent fuel pool are based on the assumption that the fuel transfer canal connecting the refueling cavity to the spent fuel pool is open at the time of the initiating seal failure and that the canal cannot be closed.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

India

Two such failures took place at the BWR plants in India. Operating procedures have been established to handle the situation. The cavity seals are replaced with new ones prior to each refueling.

Korea, Republic of

A watertight seal is installed between the reactor vessel flange and the floor of the refueling pool during the refueling operation. The seal is removed before reactor operation.

Provisions are made to test the seal after installation and before flooding the refueling pool.

USA

In the United States, the proposed resolution to this issue is actually made up of several resolutions which apply to different plants according to each individual plant's refueling cavity configuration, seal design, and operating procedures. The various aspects of the proposed resolution are based on assessments by the Haddam Neck staff and the NRC. All of the proposed actions are aimed at bringing the affected plants into conformance with the features employed at the nuclear plants that are least vulnerable to a refueling cavity seal failure and its consequences.

The overall proposed resolution includes both mitigative and preventive measures. Proposed mitigative measures include: (1) temporary reinforcement of existing seals until permanent corrective measures are implemented; and (2) implementation of procedures to assure prompt operator response to a leak and to gross seal failure (completed at Haddam Neck). Preventive measures include: (1) installation of improved-design seals at plants with single inflatable seals; (2) replacement of double inflatable seals with permanent steel seals (completed at Haddam Neck); and (3) installation of a coffer dam to prevent spent fuel pool draining through the refueling cavity at plants where this is possible (completed at Haddam Neck).

ADDITIONAL SOURCES:

- NUREG-0933 Generic Issue 137, Refueling cavity seal failure.
- USNRC Bulletin 84-03, Refueling cavity water seal, issued August 24, 1984.
- USNRC Inspection Manual Temporary Instruction 2515/066-04, Inspection Requirements for IE Bulletin 84-03, issued December 17, 1984.
- *USNRC IN 84-93, Potential loss of water from the refueling cavity, issued December 17, 1984.*

ISSUE TITLE: Need for assessment of turbine missile hazard (IH 9)

ISSUE CLARIFICATION:

Description of issue

A turbine missile could be released from the turbine or even from generator of mainly two reasons. The first failure mode can occur during normal operating speed because of e.g. fatigue crack growth and the second could occur due to sudden unloading of the of the generator, the turbine overspeed protection system fails etc. causing the turbine to reach the destructive overspeed.

Safety significance

If a turbine disc were to fail and if a large portion of the disc were to be ejected from the turbine casing, it would be possible for the turbine missile to strike and cause damage to components or systems with safety functions. Depending on how the turbine is situated, a missile could also cause damage to the control room.

Source of issue (check when appropriate)

- _____xx___ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

Protection against turbine missile hazards can be achieved either from wall reinforcement or using different types of layout turbine buildings to minimize turbine missile risks on one site from any turbine to any reactor. Some types of layout can be used: parallel units, fan-shaped units (ex: Cattenom), head to tail units, etc.

Korea, Republic of

The SRP requires that the ratio of the K_{IC} of the disk material to the maximum tangential stress at speed from normal to designed overspeed should be at least two at minimum operating temperature. The applicant commits to meet this acceptance criteria at 115% overspeed, which is considered acceptable.

The turbine assembly is designed such that the maximum tangential stress in wheels and rotors resulting from centrifugal forces, interference fit, and thermal gradient does not exeed 0.75 of yield strength at 115% of rated speed, which is acceptable.

In addition to the various turbine trips, two independent and redundant overspeed trip systems, the mechanical overspeed trip (OST) system and the electronic overspeed system (EOS), are provided to trip turbines at 110% and 111% of rated speed, respectively. Therefore, this system is considered to meet the regulatory requirements.

Sweden

In Sweden, at Barsebäck 2 an accident occurred on 13 April 1979. A generator missile caused great damages but nothing that influenced the reactor safety.

After the accident an investigation was performed for the Barsebäck plant to show if any strengthening of walls were necessary to ensure the safety of the plant. A similar investigation was made for the Oskarshamn 1 plant in 1994.

ADDITIONAL SOURCES:

- Consequences of a generator missile at Barsebäck 2, S. Granström, H.Sundquist, Tyréns AB - SKI ref. a.2.a-1111/79 (in Swedish).
- Potential Risk to damage safety related equipment caused by turbine or generator missiles. Håkan Persson, Tyréns Byggkonsult. - OKG AB reg nr. 94-07609 (in Swedish).

4.1.10. External hazards (EH)

ISSUE TITLE: Need for systematic assessment of seismic effects (EH 1)

ISSUE CLARIFICATION:

Description of issue

To identify the vulnerabilities of the design of a nuclear power plant as built and modified over the plant lifetime with respect to seismic effects, a sytematic assessment needs to be carried out using a combination of deterministic and PSA techniques.

The following steps should be taken in the assessment:

(1) Seismic input evaluation: Design basis ground motion at the plant site should be re-evaluated. For seismic PSA, the hazard curve should be defined.
(2) Analysis of seismic response: Response of buildings, structures and components should be calculated, including the effects of soil-structure interaction for the seismic input calculated in step 1.
(3) Evaluation of fragility: The fragility of structures and components should be evaluated at each level of the intensity of ground motion. A plant walkdown would be useful to understand the installation of systems and components in a plant.
(4) Assessment of core damage probability (for seismic PSA): Initiating events and accident sequences leading to core damage should be identified using the event tree and fault tree methodologies. The conditional probability of each accident sequence should be evaluated using the fragility data of systems and components for each intensity of ground motion.

The main objective of the assessment is to compare the seismic demand (input) with the capacity of the plant and to decide on the need for upgrades.

A seismic PSA was performed in the Individual Plant Examination of External Event (IPEEE) programme in USA and in some other member states. It was shown in these programmes that the probability of core damage caused by an earthquake is of a comparable order as the core damage probability caused by internal events.

For nuclear power plants built according to earlier standards and for many of the WWER NPPs, the seismic design basis (i.e. seismic input parameters) is generally not in accordance with current international practice. The as-built status of NPP structures, components and distribution systems have to be checked against the site specific seismic loads in the event of an earthquake of the design basis level.

Safety significance

Earthquakes may possibly affect the integrity of many systems and components simultaneously and it is especially important to assess the accidents caused by multiple failure of components. A lack of systematic assessment would lead to systems requiring seismic upgrading not being identified.

Source of issue (check as appropriate)

- __________ operational experience
- ____xx____ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

The Kozloduy NPP Units 5 and 6 were originally designated for 0.1 g acceleration. The new value for the site has been evaluated to be 0.2 g. An assessment is being performed using seismic PSA.

Czech Republic

The Temelín NPP had an original seismic design basis value 0.06 g. In order to follow IAEA recommendation 1, the recommendation of the IAEA expert mission to Temelín in 1992, and the IAEA Safety Guides 50-SG-S1 and 50-SG-D15, the safe shutdown earthquake (SL2) and operational basis earthquake (SL1) values have been re-evaluated as follows:

SL2 (SSE): 7 MSK-64 PGAHOR = 0.1g, PGAUSER = 0.07g
SL1 (OBE): 6 MSK-64 PGAHOR = 0.05g, PGAUSER = 0.035g.

Currently, the assessment of seismic capacity is being carried out to comply with the relevant IAEA recommendations. In addition, a probabilistic seismic hazard analysis is included in the Temelín PSA Project scope in order to address the contribution from earthquake induced accident sequences to the overall CDF of the plant. The seismic hazard curves for the Temelín site has been developed and seismic fragility analysis has been performed for the structures and components. Based on the preliminary results (annual frequency of 0.1g PGA (SSE) earthquake is 1E-6/year), it is expected that the contribution of seismic events and the consequential accident sequences to the overall CDF will be negligible (i.e. less then 1% of overall CDF). The independent review of this PSA task is to be carried out in the framework of the 2nd IAEA IPERS (Level 1 - external initiating events) in August/September 1995.

The principles of addressing the seismic issue by the design have been developed in accordance with western standards. A design amendment has been developed by the General Designer Energoprojekt including the specifications and list of structures, equipment and distribution systems with required seismic resistance. The seismic response spectra of the structures are being calculated and interactions between seismic resistant and non-resistant structures and equipment are being analysed.

France

Safety reassessment has shown, for certain French sites, an underestimation of the level of SMHV equivalent to the maximum historical earthquake recorded (or estimated from historical data) in the "seismotectonic province" of the site. Furthermore, two types of earthquakes are considered:
- local earthquakes, with violent but short moves, they do not affect the mechanical resistance of building structures, (supplement studies are however foreseen in 1997);
- distant earthquakes, considered as dimensioning for structures and components.

In France, legislation impose:

- that safety classified mechanical components shall be designed to withstand to a standard earthquake of level SMHV + 1, called Spectre De Dimensionnement (SDD), and must keep their integrity;
- that safety components used to shut down the reactor or necessary for core coolability, and for coolability of spent fuel in the spent fuel pit, must remain functional.

Furthermore legislation asks to check that during an earthquake of 1/2 SDD level, called DSD (Demi Spectre de Dimensionnement) safety classified components remain functional and that their ability to withstand to a following postulated SDD is not affected.

It will be checked, that the margin used in the current design is sufficient.

India

Systematic assessment of seismic effects has been initiated in 1996.

Japan

As for this issue, findings obtained from past records of earthquake and survey of faults through the survey during construction have been studied in the light of "Guide for Seismic Design of LWR Facilities" and plants are designed and constructed on the basis of the results of survey. In this way, the integrity of nuclear power stations has been sufficiently confirmed.

Currently, the application of seismic PSA techniques to the plant is being studied.

Korea, Republic of

At the Ulchin NPP Units 3 and 4, the design response spectra for two horizontal directions are Reg. Guide 1.60 generic spectra. Although for vertical direction the design response spectrum is slightly different from Reg. Guide 1.60 spectrum at high frequency range, the response spectra are considered acceptable from past experiences and recent study results.

Since the design time histories meet response spectrum enveloping requirement and power spectral density function requirement and the components of design time histories in each direction are statistically independent, the deisgn time histories are acceptable. The critical damping values are consistent with Reg. Guide 1.61 and ASME Code Case N-411-1.

Since the evaluations based on related technical standard and recent study results etc. result in wholly appropriate applications, seismic design methods of Category I structures, systems, and components are considered acceptable. However, for some structures, appropriateness of fixed base analyses were not evaluated, so the acceptability of fixed base analyses should be identified.

Russian Federation and Ukraine

The following information was recently provided by Russian and Ukrainian experts:

All NPPs in the Russian Federation and Ukraine were designed considering seismic effects. The seismicity of sites was defined during NPP construction by the special organizations of the USSR in accordance with the regulations valid at that time, but the seismicity of the sites was not more than magnitude 6 as per MSK scale (SSE). At the same time, the decisions were used from the unified project designed for SSE of magnitude 7 as per MSK scale. After the new regulatory documents were issued, an analysis was performed to verify compliance with the regulations, taking into account an additional study of survey results available. It is now felt that there are reasons to not change the seismicity of the site. The performance of additional studies is required for legal compliance with the requirements of the regulatory documents. For all seismic category I structures, the strength and stability are ensured for seismic effects. The strength of the main equipment and pipelines is also ensured. The specification of seismic resistance is required only for some kinds of equipment. Work other than those outlined are considered to be not reasonable. In the improvements of WWER-1000 NPP safety as proposed by the international users group, it is suggested that anti-seismic shock-absorbers be installed on the pipelines important to safety.

Sweden

In Sweden, the regulatory body requires that seismic effects shall be a part of the design criteria for plant modifications at Swedish plants.

The USNRC requested that each licensee perform an individual plant examination of external events (IPEEE) to identify vulnerabilities, if any, to severe accidents and report the results together with any licensee-determined improvements and corrective actions.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, NUSS 50-SG-S1, rev. 1, 1991.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Probabilistic safety assessment for seismic events, IAEA-TECDOC-724, Vienna (1993).
- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- D.L. Bernreuter, et al., Seismic hazard characterization of 69 nuclear plant sites east of the Rocky Mountains, Lawrence Livermore Laboratory, NUREG/CR-5250, January 1989.
- R.J. Budnitz, et al., "An approach to the quantification of seismic margins in NPPs", NUREG/CR-4334, August 1985.
- Sobel, P., "Revised Livermore Seismic hazard estimates for sixty-nine nuclear power plants sites east of the Rocky Mountains," NUREG-1488, April 1994.
- L.E. Cover, et al., Handbook of nuclear power plant seismic fragilities, Lawrence Livermore National Laboratory, NUREG/CR-3558, December 1985.
- K.K. Bandyopadhyay, et al., "Seismic fragility of nuclear power plant components [PhaseII] (A fragility handbook on eighteen components)", Brookhaven National Laboratory, NUREG/CR-4659, Vol.4, June 1991.
- M.P. Bohn, et al., "Procedures for the external event core damage frequency analysis for NUREG-1150", Sandia National Laboratories, NUREG/CR-4840, November 1990.
- M.P. Bohn, et al., "Analysis of core damage frequency: Surry Power Station, Unit 1 external events", Sandia National Laboratories, NUREG/CR-4550, Vol 3, Rev.1, Pt.3, Dec.1990.
- J.A. Lambright, et al., Analysis of core damage frequency: Peach Bottom, Unit 2 external events, Sandia National Laboratories, NUREG/CR-4550, Vol. 4, Rev. 1, Pt. 3, December 1990.
- A methodology for assessment of nuclear plant seismic margin, EPRINP-6041, October 1988.
- Seismic hazard methodology for the Central and Eastern United States, EPRI NP-4726, July 1986.

ISSUE TITLE: Need for assessment of seismic interaction of structures or equipment on safety functions (EH 2)

ISSUE CLARIFICATION:

Description of issue

It was found in earlier plant walkdowns and seismic PSA studies that failure of non-seismically designed structures and equipments could affect the safety function to a larger extent. Typical failures would involve items which may have been overlooked in the original design (such as false ceilings in rooms containing Cat. I item) or those which have been added during operation for other architectural reasons (e.g. masonry walls). Furthermore, it is important to check unanchored (non-safety) items such as cabinets and panels for their potential impact on safety items.

Safety significance

If the intensity of the seismic ground motions exceed a certain level, there is a possibility that non-seismically designed structures and components may fail. The failure of such structures or components may result in the unavailability of safety systems and this could lead to a core damage.

Source of issue (check as appropriate)
- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

Within the framework of the reassessment plant for the 900 MW units, a programme has been launched to identify the potential interactions between not seismically designed structures and components and those which are safety related. Depending on the results, corrective modificatons will be carried out. The analysis is based on plant walkdowns by groups of experts.

India

Systematic assessment of seismic effects has been initiated in 1996.

Japan

With respect to this issue, design of reactor facilities are classified in terms of seismic importance from the standpoint of radiation effects on the environment which may occur as a result of earthquake, and facilities are seismically designed in accordance with such classification.

Currently, the application of seismic PSA techniques is being studied.

Korea, Republic of

At the Ulchin NPP Units 3 and 4, when safety related and non-safety related structure are integrally connected, the non-safety related structure is included in the model when determining the factors on safety related structures.

Such non-safety related structures (as well as non-safety related structures adjacent to safety related structures) are designed so that their failure under SSE conditions will not cause the failure of the safety related.

Measures have been taken in some USplants on the Pacific coast to strengthen non-seismically designed structures and equipments including anchor bolts against the ground motions during an earthquake.

As part of the Individual Plant Examination for External Events, all nuclear power plants in the USA are undergoing walkdowns to, among other things, identify potential interactions between designed structures, systems, and components that are not seismicaly designed and those that are seismically designed.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Probabilistic safety assessment for seismic events, IAEA-TECDOC-724, Vienna (1993).
- R.J. Budnitz, et al., An approach to the quantification of seismic margins in nuclear power plants, NUREG/CR-4334, August 1985.
- EPRI, A methodology for assessment of nuclear plant seismic margin, EPRI NP-6041, October 1988.
- M.P. Bohn, et al., Procedures for the external event core damage frequency analysis for NUREG-1150, Sandia National Laboratories, NUREG/CR-4840, November 1990.
- P.D. Smith, et al., Seismic Safety Margins Research Program Phase C, Lawrence Livermore National Laboratory, NUREG/CR-2015, Vol.1 -Vol.10, April 1981.
- M.P. Bohn, et al., Application of the SSMRP methodology to the seismic risk at Zion nuclear power plant, Lawrence Livermore National Laboratory, NUREG/CR-3428, November 1981.

ISSUE TITLE: Need for assessment of plant specific natural external conditions (EH 3)

ISSUE CLARIFICATION:

Description of issue

Analyses have shown that external events are major contributors to core damage frequencies. But experience with screening analyses of essentially all external events to which the plant could conceivably be exposed shows that only a very few of these are significant contributors to risk at US plants. Whether these events can be screened out will depend on the site characteristics and the plant design on a case by case basis.

Light structures and outside storage tanks and pipes which are safety related are generally the most vulnerable items in relation to natural external events.

Safety significance

Natural external events can cause loss of off-site power (without early recovery) leading to several serious scenarios, e.g. if the diesel generators are affected by the external events or loss of cooling water.

The lack of an adequate investigation in accordance with NUSS 50-C-S of the nuclear power plant site with respect to natural events is a deviation from current international practice.

Source of issue (check as appropriate)

- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

The Kozloduy NPP experts consider that extreme cold conditions for extended period of time, as well as tornados can be excluded from the site.

Czech Republic

The amendment of the Temelín PSAR will follow USNRC RG 1.70 contents and philosophy. Therefore, the PSAR chapters 2 and 3 will address plant specific natural phenomena issues. The Temelín NPP Designer (EGP Prague) is currently carrying out analyses of potential impacts of these phenomena (i.e. regional climatology, floods, ice effects, flooding, wind and tornado loadings, etc.) to the plant safety. In addition, the Temelín PSA Project involves assessment of those external events in order to evaluate their potential for contribution as initiators to the overall plant CDF. The independent review of this PSA task is to be carried out in the framework of the 2nd IAEA IPERS (Level 1 - external events).

India

Systematic external assessment of external flooding has been carried out in 1995. Systematic assessment for establishing seismic parameters for review has been initiated.

Japan

In Japan, in determining a site for a nuclear power station, site conditions are, as a rule, such that there have been no events which could lead to major accidents in the past, that such accidents are unlikely to occur in the future, and that events are rare to expand damages.

In "Guide for Safety Design of LWR Facilities", matters to be taken into consideration in designing with respect to natural phenomena are specified (Guide 2), and it is required to design individual facilities, taking into consideration earthquake and natural phenomena postulated. Especially as for earthquake, in view of relatively high frequency of earthquake in Japan, the Nuclear Safety Commission has established "Guide for Seismic Design of Light Water Nuclear Power Reactor Facilities" and the Japan Electric Association has established "Technical Guidelines for Seismic Design of Nuclear Power Plants" (JEAG-4601). On the basis of those guides, seismic design is evaluated for individual plants.

Korea, Republic of

At the Ulchin NPP Units 3 and 4, maximum wind speed as recurrence interval of 100 years is adequately determined using site specific data for the design of seismic Category I structures.

Determination of design win speed, gust factor, and wind pressure coefficient is based on ANSI A 58.1. Application of ASCE Paper No. 3269 for the calculation of wind pressure coefficient of which procedure is not described in ANSI A 58.1 is acceptable.

Whether tornado loadings is considered in the design or not is a conditioninal item for site permit, which is under investigation and analysis by applicant.

External PSA is the most extensive method for analysing external hazard risk contribution to core damage and was performed by the utility in accordance with NUREG-4840 and NUREG-1150.

Sweden

The total PSA research efforts in Sweden lately have been concentrated to the procedures related to analysis of external events, since it has been found out that the contribution from external initiating events could be up to 50% to the total core damage frequency. External event analyses have been proven useful and have resulted in several important plant modifications that have increased safety.

Ukraine

At the Rovno NPP, the natural phenomena like temperature, snow and wind have been considered in the design. It is assumed that the related severe conditions occur once in 10,000 years. The natural phenomenon tornado also needs to be considered for the Rovno NPP site. However, the plant has no special design against a tornado. Instead, a pressure wave with a value of 30 kPa has been considered for the design of the safety related buildings. The possible loss of water from the spray pond as the result of a tornado was also considered. To counter this, the spray pond will be sub-divided using vertical walls, so that the total inventory of water cannot be lost at any one time. These design requirements are a pre-requisite for the commissioning of Unit 4.

USA

The most extensive method for analysing external hazard risk contribution to core damage is PSA. The methods have been used since the very beginning as in NUREG-1150/ref 1/. These methods are presented in NUREG-4840/ref 2/, which provides a relatively straightforward and, in some cases, simplified set of techniques for the analysis of the full range of external events. However, the methods are continuously further developed in many projects world wide. The USNRC Standard Review plan provides additional licensing guidance and references.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Series No. 50-C-D (Rev.1), Vienna (1990).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Series No. 50-C-S (Rev. 1), Code on the Safety of Nuclear Power Plants: Siting (1988).
- NUREG/CR-4840, SAND88-3102, Procedures for the external event core damage frequency analyses for NUREG-1150, M. P. Bohn and J. A. Lambright, Sandia National Laboratories, Albuquerque, NM, November 1990.
- NUREG-1150, US Nuclear Regulatory Commission, Severe accident risks: An assessment for five US nuclear power plants, June 1989.
- NUREG-933, Task Action Plan, Item B-32, USNRC, 1990.
- NUREG-800, Standard Review Plan, Sec. 2.4.7 Review of safety analysis, USNRC 1988.
- Recent USNRC Information Notices 96-06, 96-36, 98-02.

COMMENTS:

The IAEA recommends that the operating organizations of WWER NPPs review their operation and maintenance regarding this issue. The USNRC has the same concern with respect to LWR safety and has included this issue in their task plan.

ISSUE TITLE: Need for assessment of plant specific man induced external events (EH 4)

ISSUE CLARIFICATION:

Description of issue

Analyses have shown that external events are major contributors to core damage frequencies. But experience with screening analyses of essentially all external events to which the plant could conceivably be exposed shows that only a very few of these are significant contributors to risk at US plants. Although man-induced external conditions are generally not as significant to risk as earthquakes and fire, they should systematically be evaluated for individual plant sites, both during construction and when conditions change during the plant lifetime.

It is important to assess the potential for such loading to the NPP through identification of sources in the site vicinity (e.g. airports, arsenals, pipelines, transportation routes, petrochemical facilities, etc.). The lack of such an assessment represents a deviation from NUSS 50-SG-D5. Plants without a structural containment such as WWER-440 NPPs, may be vulnerable to external man-induced events which generate extreme "blast" and "impact" type loading.

Safety significance

Accidents could be initiated and the safety functions could be impaired by man induced external events.

Source of issue (check as appropriate)

- ________ operational experience
- ____xx____ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

A site specific assessment for man-induced external events has been carried out at the Kozloduy NPP. The work was mainly done by Energoproekt, and the results are available to western experts. The assessment took the following factors into account:

- traffic on the Danube River;
- military and commercial air traffic in the vicinity of the nuclear power plant;
- railway traffic in the vicinity of the nuclear power plant;
- storages of hazardous substances in the vicinity of the nuclear power plant.

The over-flying of the Kozloduy NPP is administratively prohibited. Nevertheless, the containment can withstand the impact of a small aircraft of 10 t at a speed of 200 m/sec. This type of site assessment is carried out periodically and the last one was completed in early 1994.

There seems to be a potential for an appreciable increase in the air, river and road traffic near Kozloduy (see Site Safety Review Mission report of October 1994). Administrative measures are planned to control this increase within acceptable levels.

Czech Republic

The amendment of the Temelín PSAR will follow USNRC RG 1.70. Therefore, the PSAR Sections 2 and 3 will address plant specific man-induced external events. The Temelín NPP Designer (EGP Prague) is currently carrying out analyses of potential impacts of these events (e.g. nearby industrial, transportation and military facilities, missile protection, pipelines, etc.) to the plant safety. In addition, the Temelín PSA Project involves assessment of such external events in order to evaluate their potential for contribution as initiators to the overall plant CDF. The independent review of this PSA task is to be carried out in the framework of the 2nd IAEA IPERS (Level 1 - external events).

Based on the recommendations of the IAEA mission held in April 1990 at Temelín, the following measures concerning a liquified natural gas (LNG) pipeline are being implemented:
(a) The emergency plan is being developed by the operator of LNG pipeline (TRANSGAS) and this (including corresponding Safety Report Section) will be completed by 31.3.1997.
(b) The pipelines (LNG pipeline) are constantly being inspected from inside and the defects discovered are continuously being repaired.
(c) In the vicinity of the NPP, the gas pipelines were partitioned into smaller segments.
(d) At the beginning of 1996, an automatic gas leakage detection and localization system for LNG pipeline permanent monitoring will be installed.
(e) A new "Technical Conditions" and Safety instructions of the other companies working within the TRANSGAS protection zone have been prepared in April 1994.
(f) A final protection barrier as well as probes and protectors will be installed.

France

Man induced external conditions issue covers in fact a category of issues, each of them being very specific and generally independent from others, they are analysed in the Safety Guide 50-SG-D5 (Rev. 1, 1996), among these issues, generally the following are taken in account in France:

– Aircraft crashes

Two types of aircraft are taken in account: Lear jet 23 (twin-jet of 5.7 tons) and Cessna 210 (single-engined aircraft of 1500 kg). Generally NPP sites are outside vicinity of airports where fall frequency is higher, furthermore civil or military reglementation forbid flights at a low altitude. Solutions consist in reinforcement of civil structures and separation of safety trains to avoid common mode failures.

– Explosions

Some NPPs can be at a relatively short distance of industrial installations or coastal/inland navigation which can present a potential risk of explosion, that is the case of fuel/methane storages and/or methane tankers, in that case specific measures are implemented.

– Fire

(see Issue IH 1)

– Toxic gases

These types of external man-induced events are taken in account, like others, on a case by case basis, when the event probability per year is about 1.10^{-6} or higher.

Japan

In "Guide for Safety Design of Light Water Nuclear Power Facilities", the following matters are required to be considered in plant design:

"Structure, systems and components with safety functions shall be so designed that the safety of the nuclear reactor facilities will not be impaired by postulated extenal man-induced events." (Guide 3.1)
External postulated man-induced events are said to refer to airplane crashes, collapse of dams, explosions, etc. In Japan, nuclear power plants are not located in the vicinity of dams nor in areas where

explosion-induced events occur, and the probability of aircraft crashing onto such facilities is extremely low.

Russian Federation and Ukraine

The following information was recently provided by Russian and Ukrainian experts:

Designs of NPPs with WWER-1000/320 were developed taking into account the resistance against external man-induced events, including:

- all structures where safety systems are located were designed for the effect of an impact wave a pressure of 30 kPa;
- the containment should provide for functional stability for the effects of an airplane crash (of weight up to 10 t).

However, national regulations do not require such effects to be considered.

At the NPP site there are no sources, the effect from which may reach 30 kPa. For most NPPs in the region of possible hazards, there are no sources for an impact wave exceeding these values.

For some NPPs, an additional check of possible effects is required (for example, for the Rovno NPP with respect to existing rail traffic). Such work is planned within the framework of the programme on safety upgrading.

In case of large consequences and high probability of occurrence of the event, organizational measures will be considered.

The Rovno NPP site has been assessed with respect to man-induced external events. There are no gas lines, pipe lines or chemical or industrial plants near the plant site which could pose a potential hazard. However, there is a railway line about 500 m from the site. A detailed analysis of this will be conducted at a later date. For the present, a pressure wave with a value of 30 kPa will be considered in the design of the reactor building. If the analysis to be conducted at a later date shows that there are hazards resulting in loads higher than those resulting from this value of 30 kPa, then appropriate administrative measures will have to be enforced.

The consideration of an aircraft impact in the design of the NPP is not required according to Ukrainian regulations. An analysis has nevertheless been conducted for the containments of both Units 3 and 4 of the Rovno NPP, assuming the impact of an aircraft of 10t at a speed of 700 km/h. This is considered adequate for the containments of WWER-1000/320 NPPs. If the probability value obtained from the assessment planned in the modernization programme exceeds a generally accepted value, administrative measures will have to be enforced.

Sweden

The total PSA research efforts in Sweden lately have been concentrated to the procedures related to analysis of external events, since it has been found out that the contribution from external initiating events could be up to 50% to the total core damage frequency. External event analyses have been proven useful and have resulted in several important plant modifications that have increased safety.

USA

The most extensive method for analysing external hazard risk contribution to core damage is PSA. The methods have been used since the very beginning as in NUREG-1150. These methods are presented in NUREG-4840, which provides a relatively straightforward and, in some cases, simplified set of techniques for the analysis of the full range of external events. However, the methods are continuously further developed in many projects world wide. USNRC Regulatory Guide 1.70 provides additional licensing guidance and references.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- USNRC Regulatory Guide 1.70.
- NUREG/CR-4840, SAND88-3102, Procedures for the external event core damage frequency analyses for NUREG-1150, M. P. Bohn and J. A. Lambright, Sandia National Laboratories, Albuquerque, NM, November 1990.
- NUREG-1150, US Nuclear Regulatory Commission, Severe accident risks: An assessment for five US nuclear power plants, June 1989.

4.1.11. Accident analysis (AA)

ISSUE TITLE: Adequacy of scope and methodology of design basis accident analysis (AA 1)

ISSUE CLARIFICATION:

Description of issue

According to the IAEA NUSS code on design, accident analysis shall be performed to ensure that the overall plant design is capable of meeting prescribed and acceptable limits for radiation doses and releases set by the regulatory body for each plant condition category. The operating organization needs additional analyses for personnel training to cope with accidents, for the preparation of emergency operation procedures, and for protection and signal settings. General guidance on scope and methodology to be used for accident analysis is provided by the NUSS guides 50-SG-D11 and 50-SG-G2 (under revision), which describe the procedures for conducting accident analysis and reviewing the results, respectively.

Design basis events are chosen in the deterministic method of the safety assessment to encompass a range of related possible initiating events which could challenge the safety of the plant. These events form the basis for sizing and selecting safety systems. Analysis is made to show that the response of the plant and its safety systems to abnormal transients and accidents considering single equipment or human failure satisfies predetermined specifications both in damage to the barriers and in doses to the population. This approach is often used in conjunction with generic or plant specific PSA to make regulatory decisions.

In addition to the national practice there are international activities to develop guidance for accident analysis including areas which have not been consistently addressed in the past, such as PTS analysis for RVP integrity assessment.

Safety significance

The lack of a comprehensive set of full scope analysis of design basis accidents and properly used methodologies increases the likelihood that transients and accidents could progress in severity and result in significant radiological releases.

Source of issue (check as appropriate)
- __________ operational experience
- ___xx_____ deviation from current standards and practices
- ___xx_____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

In most of the member countries, the safety assessment of the anticipated transients and the accidents to assess the adequacy of the safety system design has been required. The applicants are requested to conduct the safety analysis on the plant performance in accordance with those rules responding each of requests in the rules in order to maintain the transparency and the conservativeness of the assessment.

Bulgaria, Czech Republic, Russian Federation, Ukraine

Regulatory guidelines on the requirements for safety analysis are under development in the Russian Federation and Ukraine which include requirements on the scope and methodology for DBA and BDBA analysis .

Bulgaria follows the guidance given by the IAEA and Czech Republic applies USNRC Regulatory Guide 1.70., e.g. for Temelin NPP.

France

In an orientation letter CAB No. 1121-MZ, (October 6, 1983) related to French 1400 MW NPPs and addressed to EDF general manager, the Safety Authority precise (Section 1, general dispositions, §1.1. general basis for design) that accidents to consider in the design basis are the following:

- fuel handling accident;
- large break on secondary circuit (water or gaseous phase);
- blocked rotor on one primary pump;
- control rod assembly ejection;
- LOCA;
- total rupture of two SG tubes.

Japan

The scope of design basis accident analysis and acceptance criteria have been examined by the Nuclear Safety Commission and included in the guide. Nuclear power plants are subjected to safety review through a deterministic evaluation based on the guide and confirmed that the radioactive release to the environment is less than the limit. The appropriateness of analysis techniques and the scope of analysis are confirmed during safety review.

The new analysis technique is reviewed by the Ministry of International Trade and Industry and the Nuclear Safety Commission, and upon having been reflected in the Guide according to necessity, is adopted for safety evaluation.

As for procedures for operator accidents and emergencies, response training is being done, using simulators.

Korea, Republic of

At the Ulchin NPP Units 3 and 4, each postulated initiating event has been assigned to one of the following categories:

- increased heat removal by secondary system;
- decreased heat removal by secondary system;
- decreased reactor coolant flow;
- reactivity and power distribution anomalies;
- increase in RCS inventory;
- decrease in RCS inventory;
- radioactive release from a subsystem or component.

For event combinations which require consideration of a single failure, the limiting failure is selected. In sequence of events and system operation selection provides, for each limiting event, the step by step sequence of events from event initiation to the final stabilized condition. The radiological evaluation is to confirm that the calculated doses from postulated accidents lie within the limits described in 10CFR100.11 and/or applicable NUREG-0800 sections. Dose are dependent, in part, upon the meteorological characteristics assumed to determine the radiological atmosphere dilution, X/Q.

For the design basis events, resulting in a violation of the DNBR SAFDL limit, all of the fuel rods experiencing DNB are assumed to fail. The number of failed rods is calculated by the statistical convolution method. This statistical convolution technique involves the summation over the reactor core of the number of rods with a specific DNBR times the probability of DNB at DNBR.

ADDITIONAL SOURCES:

- Guidelines on pressurized thermal shock analysis for WWER nuclear power plants, IAEA, EBP-WWER-08 (former WWER-SC-157), 1997.
- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- Guidelines for accident analysis of WWER nuclear power plants, IAEA, EBP-WWER-01, 1995.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Periodic safety review of operational nuclear power plants: A Safety Guide, Safety Series No. 50-SG-O12, IAEA, Vienna (1994).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Code on safety of nuclear power plants: design, Safety Series No. 50-C-D, Vienna (1988).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Series No. 50-SG-D11 General design safety principles for NPPs (1986).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Information to be submitted in support of licensing applications for nuclear power plants, Safety Series No. 50-SG-G2, Vienna (1979).
- Guideline for evaluation of PWR design during accident, §23, Abs.3, Public Notice, Federal Radiation Protection Ordinance No.245a, GRS, Dec. 1983, FRG.
- USNRC Regulatory Guide 1.109, USNRC, 1977.
- USNRC Regulatory Guide 1.70.
- USNRC Generic Letter 97-04, Assurance of sufficient net positive suction head for emergency core cooling and containment heat removal pumps.
- USNRC Generic Letter 96-06 and Generic Letter 96-06 Supplement 1, Assurance of equipment operability and containment integrity under design basis accident conditions.
- Recent USNRC Information Notices 96-01, 96-45, 96-49, 96-55, 97-40, 97-79, 97-81.

ISSUE TITLE: Adequacy of plant data used in accident analysis (AA 2)

ISSUE CLARIFICATION:

Description of issue

All accident analyses are based on a valid data base such as: geometrical data, material properties, physical and thermohydraulic data including boundary conditions of plant operational status. Every accident analysis needs a plant model which must be constructed on the basis of valid data. This data base is subject to quality assurance programme. It is essential that plant owners obtain reliable and verifiable data on the plant as-built and modified over the plant lifetime and validate them appropriately.

Safety significance

The lack of accurate and current plant information on which to base accident analyses can lead to erroneous conclusions as to accident sequence and consequences. This could result in impairment of the preventive and mitigative capabilities of the plant.

Source of issue (check as appropriate)
- __________ operational experience
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

There are standards related to the assessment of the anticipated transients and the accidents and the validation of plant data used in the assessment in most of the member states. The member states, which do not have such standards, may adopt the IAEA safety guides for the safety assessment.

Bulgaria, Czech Republic, Ukraine

Bulgaria has an extensive QA programme related to accident analysis covering among others plant data preparation, input data preparation, cross checking procedures, input check preparation, etc.

Appropriate QA procedures and programmes are also in place in Czech Republic and Ukraine including data preparation, checking of procedures. Meetings with the Russian design organizations were performed whenever necessary to confirm the validity of data.

France

Quality of plant data are currently periodically reassessed, following a French safety rule (Decree 90-78 of January 19, 1990, art. 4) according to which Safety Authorities can, at any time, ask the utility to proceed to a safety reassessment. Currently, this reassessment is performed about every 10 years and is realized in 3 steps:

(1) Definition of a reference state.
(2) Checking that all plants of the subseries are in compliance with the reference state, (test programmes validate the proposed modifications).
(3) Assessment of the reference by the safety authority (with possible supplements required).

Japan

As for analysis model and parameters to be used in accident analysis, the "Guide for Safety Evaluation of Light Water Nuclear Power Facilities" requires the following, and nuclear power stations have gone through safety review in accordance with the Guide.

- Computer programs to be used in the analysis of postulated events, the appropriateness of its use has to be confirmed.
- Models and parameters to be used in analysis have to be so selected as to obtain serve side evaluation results.

Korea, Republic of

At the Ulchin NPP Units 3 and 4, at the PSAR stage, some of initial and boundary conditions could not be properly obtained. For such area, more attention will be given at the FSAR stage.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- Guidelines for best estimate approach to accident analysis of WWER NPPs, IAEA, WWER-SC-133, 1995.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- INTERNATIONAL ATOMIC ENERGY AGENCY, General design safety principles for NPPs, Safety Series No. 50-SG-D11 (1986).
- Decree 90-78 of January 19, 1990, France.
- Guideline for evaluation of PWR design during accident, §23, Abs.3, Public Notice, Federal Radiation Protection Ordinance No.245a, GRS, Dec. 1983, FRG.
- USNRC Regulatory Guide 1.109, USNRC, 1977.
- Recent USNRC Information Notices 96-31, 96-39, 97-27.

ISSUE TITLE: Computer code and plant model validation (AA 3)

ISSUE CLARIFICATION:

Description of issue

The predictability of a computer code and the adequacy of modeling of a plant used in the assessment of the anticipated transients and the accidents to assess the adequacy of the safety systems design must be validated through comparisons between predictions and the startup test data or system effect experiments. The adequacy of each model incorporated in a code to describe each of events included in the transients and the accidents must be validated based on information provided by separate effect experiments. Conservative assumptions may be made whenever those information to validate a code and modeling of a plant are not available.

There is no formally established consensus on what constitutes "validation" of codes or at least consensus on the validation process. However, there are general principles commonly used for code validation which reflect an agreed international practices. Code verification is required to review and accept the code source coding relative to the code description provided by supplier. In general, industry sponsored codes have been subjected to stringent verification procedures as a consequence of the regulatory licensing process.

A computer code is based on a model of the reality the code should describe. A comparison of the code predictions with experimental data gives an indication to which extent the code is validated, i.e. the model sufficiently reflects the reality. The general principles are given in the NUSS Guide 50-SG-D11 (under revision) and others and more recently for WWER-NPPs.

Particular attention has to be paid to the application of computer codes on plant models which were not used for the code development. In addition, revisions to vendor codes performed by plant operator may introduce modeling and validation problems.

Safety significance

Without careful validation of computer codes for the purpose, i.e. specific plant, they are intended to be applied, the code predictions cannot be considered reliable. This could result in impairment of the prevention and mitigative capabilities of the plant.

Source of issue (check as appropriate)

- __________ operational experience
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

There are standards related to the assessment of the anticipated transients and the accidents and the validation of plant data used in the assessment in most of the member states. The member states, which do not have such standards, may adopt the IAEA safety guides for the safety assessment.

Bulgaria, Czech Republic, Russian Federation, Ukraine

Russian design codes were validated against scaled and unscaled experimental facilities. Use of western codes in the countries operating WWER-plants is based on an ongoing validation process using different and appropriate means to demonstrate western code application to WWER plants.

France

EdF uses MAAP (Modular Accident Analysis Programme) an American computer code developed on behalf of EPRI, this code is already fully validated by international experiments, however when adaptations for French NPPs are necessary, new validation is performed, based on research and development, for example PHEBUS experimentation at Cadarache.

Japan

Analysis codes and analysis models to be used in safety evaluation are verified on the basis of operating data and test results.

Characteristics obtained in the actual facilities such as pump characteristics are to be reflected in safety evaluation. Otherwise, evaluation should be made under conservative assumptions.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- Guidelines for accident analysis of WWER nuclear power plants, IAEA, EBP-WWER-01, 1995.
- Guidelines for best estimate approach to accident analysis of WWER NPPs, IAEA, WWER-SC-133, 1995.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- INTERNATIONAL ATOMIC ENERGY AGENCY, General design safety principles for NPPs, Safety Series No. 50-SG-D11, IAEA (1986).
- Guideline for evaluation of PWR design during accident, §23, Abs.3, Public Notice, Federal Radiation Protection Ordinance No.245a, GRS, Dec. 1983, FRG.
- USNRC Regulatory Guide 1.109, USNRC, 1977.

ISSUE TITLE: Need for analysis of accidents under low power and shutdown conditions (AA 4)

ISSUE CLARIFICATION:

Description of issue

Low power and shutdown conditions have not been extensively analysed until recently.

When a reactor is shutdown for maintenance and refueling, some safety systems are switched off or isolated. A great number of operator actions are required under this situation for different purposes.

Therefore, accidents which take place during low power and shutdown conditions (LPS) have been under extensive study all over the world for several years. Results have shown that the risk of an accident initiation during the shutdown and refueling phase is high. Important contributors to risk are boron dilution, loss of residual heat removal with the reactor cooling system in reduced inventory conditions, loss of primary coolant, loss of off-site power, fires and human errors.

Plants with horizontal steam generators (such WWERs) may not need to decrease the water level in the RPV to the loop level during maintenance as compared with most other PWR's.

Safety significance

In shutdown conditions, there are less barriers and levels of protection available to prevent an event from developing into an accident. This is somewhat offset by the lower decay heat rate in the core which can allow longer times for operators to take corrective actions. All main safety functions can be affected as seen from generic observations of PSA studies made for different plant types.

Source of issue (check when appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBERS STATES:

Bulgaria, Czech Republic, Ukraine

All member states operating WWERs intend to or being in the process of carefully studying accidents during low power and shutdown conditions.

France

For the French PWRs, the scenarios of total loss of core cooling and reactivity insertion by dilution have been assessed in detail and several preventive and mitigating measures are being implemented to reduce the associated risks (see issues RC 1: Inadvertent boron dilution under low power and shutdown conditions and OP 4: Precautions for mid-loop operation (PWR)).

Korea, Republic of

At the Ulchin NPP Units 3 and 4, the regulatory body requested the utility to prepare and develop the PSA methodology and to perform the Level II PSA under low power and shutdown operational modes.

On the basis of the risk analysis of operating experience in NUREG-1449 including the accident sequence precursor analysis, following dominant events were identified:

- loss of all AC power;
- loss of RCS inventory;
- loss of reactor vessel level control.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- Accidents during shutdown conditions for WWER-1000 nuclear power plants, IAEA, WWER-SC-153, 1995.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- Modernization programme, Kozloduy NPP Units 5 and 6 January 1995, Version 0.
- Modernization programme for the Ukrainian NPPs with WWER-1000/320 reactors, August 1995, Revision 0.
- Recent USNRC Information Notices 95-57, 96-05.

ISSUE TITLE: Need for severe accidents analysis (AA 5)

ISSUE CLARIFICATION:

Description of issue

Current practice is to perform the analysis of accidents of very low likelihood but more severe than those considered explicitly in the scope of DBA and even beyond DBA. Severe accidents may cause such plant deterioration that proper core cooling cannot be maintained and fuel damage occurs. These severe accidents have a potential for major radiological consequences if radioactivity released from the fuel was not adequately confined.

The analyses of severe accidents are used to identify shortcomings in the provisions for prevention and mitigation in plant's defence in depth in case the engineered safety features fail to control accidents to be coped within the design basis envelope. Accident management measures are considered the main tools to monitor the plant status, to ensure long-term cooling to prevent recriticality and to confine radioactivity as far as possible.

Through investigations on the operational experience of power reactors including the TMI-II accident it was recognized as being possible to develop and to implement accident management. (See also SS 8, Accident management measures).

Safety significance

Inadequate severe accident analyses would affect appropriate actions to be taken during the course of an accident and may result in radioactive releases which could have been avoided.

Source of issue (check as appropriate)

- ________ operational experience
- ________ deviation from current standards and practices
- __xx__ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria, Czech Republic, Russian Federation, Ukraine

The Russian Federation, Ukraine and Bulgaria have prepared lists of initiating events resulting in severe accidents which need to be evaluated. For this purpose the Russian Federation has developed guidelines. In case of Temelin NPP, severe accidents were addressed within level 1, 2 PSAs.

France

Allowance for hypothetical severe accidents, which are still called “operating conditions not deemed to be plausible” (i.e. highly improbable events) in the French regulations are covered by what are designated “ultimate procedures”, requiring either special operating procedures or measures involving special means or a concept adopted at the design stage taking into account the risks of a hypothetical severe accident (core meltdown involving rupture of the reactor vessel).

A situation of this type can only occur as a result of the failure of all the preventive measures in force, adopted both at the design and operating stages.

Ultimate “U” procedures have been considered and described in item SS 8.

Germany

Analysis of severe accident sequences are e.g. used to investigate the safety potential of plants and to define the requirements for accidents management measures, which shall be taken in order to prevent core damage or to mitigate the consequences of core melt. In the beginning analysis has been concentrated on event sequences with a total loss of feedwater. At sequences, at which a complete loss of feedwater and the availability of designed power supply are assumed, core melt can be avoided by feed and bleed (F&B) of the secondary side and also by F&B of the primary side. Now the efficiency of selected accident management measures to prevent core melt has to be analysed in a wider spectrum of loss of coolant accidents and transients. The important phenomena after core melt, which need to be analysed, are the recombination of hydrogen by catalytic devices or ignitors, the energetic molten material - water interactions (up to steam explosions) and the heat removal from the containment together with the limitation of the containment pressure.

India

Analysis for some severe accidents have been carried out as post Chernobyl review.

Japan

Severe accident management measures have been reported to the regulatory body by the individual utilities in "Report on the Examination of Accidents Management." Their reports include measures to prevent severe accidents and to mitigate the effect of the events.

Korea, Republic of

At the Ulchin NPP Units 3 and 4, the plant design shall be designed not to cause any adverse impact on licensing process due to the requirements of NUREG-1070. Following requirements are delineated:

- Demonstration of compliance with any technically relevant portions of the TMI requirements set forth in 10CFR50.34(f).
- Technical resolution of USIs and medium-and-high priority GSIs which are identified in NUREG-0933.
- Design specific Level II PSA.

In compliance with the above requirements:

- Designed to demonstrate the compliance with the TMI requirements.
- Designed as much as practical to protect against or mitigate severe accidents by adopting the advanced design features.
- Designed considering the proposed technical resolutions of USIs, and GSIs.
- Plant specific Level II PSA.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- A probabilistic safety assessment of the standard French 900 MWe PWR, EPR900, IPSN., Apr.1990.
- German Risk Study on NPPs phase B, GRS., June 1989.
- OPB-88, General Provisions for Enhancing the Safety of NPPs, PNAE-G-1-011-89, Moscow (1989).

- NUREG/CR-6144, Evaluation of potential severe accident during low power and shutdown, July 1994.
- NUREG/CR-4551, Evaluation of severe accident risks.
- NUREG/CR-4550, Analysis of core damage frequency.
- NUREG-1335, Individual plant examination: Submittal guide, USNRC, Aug. 1989.
- NUREG-1150, Severe accident risks, USNRC, Dec. 1990.
- USNRC Generic Letter 88-20 and Supplements 1 through 5.

ISSUE TITLE: Need for analysis of ATWS (AA 6)

ISSUE CLARIFICATION:

Description of issue

Anticipated Transients without Scram (ATWS) are defined as accidents initiated by anticipated transients, which are assumed to proceed without scram. If the automatic reactor trip fails during these transients, it could have adverse effects on the integrity of physical barriers in a reactor.

According to current practice, ATWS are analysed for PWRs in order to demonstrate their capabilities to cope with these transients.

International practice considers the analysis of ATWS for a variety of initiating events such as loss of feedwater, loss of load, turbine trip, loss of condenser vacuum, loss of off-site power, closure of main steamline isolation valves, uncontrolled boron dilution, inadvertent control rod withdrawal, etc. ATWS analyses are performed in general by using best-estimate tools to determine the preventive (e.g. a diverse scram system) or mitigative measures (e.g. initiation of turbine trip and emergency feedwater supply) which need to be implemented for strengthening plants' defence in depth.

It was also recognized from operating experience in PWR plants, such as Salem Unit 2 where there was a failure to scram, that ATWS are possible. It was also shown in recent PSA studies on a BWR plant that ATWS can provide the largest contribution to the core melt probability among other initiating events.

Safety significance

Incomplete or lacking ATWS analysis or their unavailability at plants would it make impossible to understand which primary safety function(s) would be affected and which corrective measures need to be implemented to cope with ATWS. This issue affects control of accidents within the design basis (level 3 of protection of plant's defence in depth).

Source of issue (check as appropriate)

- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Several actions were taken either by the licensees or the regulatory bodies in Member States. Those actions taken by the licensees include establishment of review programmes, review of RPS and other safety-related components and relative post-maintenance, testing, review of modification, preventive maintenance and surveillance programmes, etc. Actions taken by regulatory bodies include review of licensee's programmes and activities, development and issuance of certain requirements and relative documents, etc.

Bulgaria, Ukraine

There are plans to perform ATWS analyses covering an appropriate range of anticipated transients based on a realistic basis, e.g. realistic initial conditions and assumptions, credit for non-safety systems, and without additional failures.

Czech Republic

ATWS analysis were carried out in a broad range for all credible anticipated operational occurrences in order to tune the functional design of a Diverse Protection System (DPS) which is being implemented at Temelin NPP (WWER-1000 NPP). The ATWS analyses and evaluations were performed in accordance with NUREG-0460 guidelines, using a best-estimate approach.

France

ATWS - Failure of the control rods to fail on actuation

In the event of category 2 transients (incidents of frequency > 10 E-2/unit/year), the triggering of a forced outage, which involves the dropping of all the control assemblies into the core, prevents various parameters reaching unacceptable values for French PWRs.

If in the event of such a transient not all the control rods drop into the core, the transient will continue, based solely on the intrinsic reactions in the reactor core which continue to evolve freely (ATWS). Studies performed since the early stages of the 900 MW series have demonstrated that the loss of feedwater without falling of the control assemblies is the transient with the most restricting effect on category 2 operating conditions, in relation to the first two barriers (the clad and the primary system).

Calculations have shown that once the ASG system and the turbine driven pump have come into action, in conjunction with the insertion of the regulating rods as a result of the increase in temperature (assumed to be available, since they are completely independent of the failed emergency shutdown systems), the significant parameters remain within admissible limits.

Furthermore many studies and improvements on fuel and cladding to limit risks of blockings have been performed to limit buckling (mechanical axial clearance, severe restrictions on moisture in UO2 pellets, etc.) and irradiation induced swelling (increasing of oxide densification, etc.).

Complementary means

It is therefore necessary to guarantee the complete independence of the control rod and emergency shutdown rod systems. I has also been decided to make use of an ATWS-mitigating system independent of the existing protection system, which is capable of generating the following two signals:

- a signal starting the ASG system;
- a signal triggering the turbine.

The instruments used for generating the signals in the palliative system make use of technologies and production processes different from those used in the normal protection system.

Korea, Republic of

At the Ulchin NPP Units 3 and 4, the Diverse Protection System (DPS) provides a diverse method to trip the reactor to safety concerns relative to Anticipated Transient Without Scram (ATWS) (10CFR50.62) issues.

The DPS uses input parameters and trip logic that are separate and diverse from the PPS to provide a simple, reliable control grade mechanism to initiate a reactor trip.

The DPS is a two channel grade system provided with sensors and circuitry which are diverse from those of the PPS. Pressurizer pressure is monitored by a bistable comparator to generate a trip signal to each channel whenever pressure exceeds a predetermined set point.

ADDITIONAL SOURCES:

- Anticipated transients without scram for WWER-1000 reactors, IAEA, WWER-SC-186, 1996.
- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- NUREG-0933, "A prioritization of Generic Safety Issues," US Nuclear Regulatory Commission, Dec. 1983.
- NUREG-0460, "Anticipated transients without scram for light water reactors," US Nuclear Regulatory Commission, March 1980.
- WASH-1270, "Anticipated transients without scram for water-cooled reactors, US Nuclear Regulatory Commission, September 1973.
- "Public service electric and gas company corrective action program related to reactor trip breaker failures on February 22 and 25, 1983 Unit No.1 Salem Generating Station", Docket No. 50-272, Public Service Electric and Gas Company (PSE&G), April 7, 1983.

ISSUE TITLE: Need for analysis of total loss of AC power (AA 7)

ISSUE CLARIFICATION:

Description of issue

The complete loss of AC electrical power to the essential and nonessential switchgear buses in a nuclear power plant is referred to as a « Station Blackout ». Because many safety systems required for reactor core decay heat removal are dependent on AC power, the consequences of a station blackout could be a severe core damage accident. The technical issue involves the likelihood and duration of the loss of all AC power and the potential for severe core damage after a loss of all AC power.

The origin of a station blackout can be the loss of off-site power with the on-site emergency AC power not available. In the United States, there had been numerous reports of emergency diesel generators failing to start and run in operating plants. In addition, a number of operating plants experienced a total loss of off-site electric power. In almost every one of these loss of off-site power events, the on-site emergency AC power supplies were available to supply the power needed by vital safety equipment. However, in some instances, one of the redundant AC emergency power supplies failed. In a few cases, there was a complete loss of AC power, but during these events AC power was restored in a short time without any serious consequences.

Safety significance

Three main safety concerns have to be considered in case of total loss of power:

- cooling the core (residual heat removal);
- injecting water into the primary pumps for cooling the seals;
- injecting water into the primary vessel to compensate for water evaporation when the vessel head is open.

Many plant-specific PSAs show the loss of AC power as a main contributor to core damage frequency. Some PWR plants currently in operation, including WWER plants, do not have a systematic assessment on this event, and therefore, there is no basis to judge the adequacy of emergency procedures, operator training and necessary hardware upgrading.

Source of issue (check as appropriate)
- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

The Kozloduy NPP plans to perform an accident analysis of the case of a total loss of electrical power supplies, including total loss of the service water.

One measure to recover electrical power is available and tested. It concerns the possibility to connect one diesel generator of the not affected neighbouring unit. The plant considers that this operation can be realized in two hours.

Czech Republic

The Temelín NPP has two additional common diesel generators for two units. The capacity of one common diesel generator is able to cover the needs of both units.

Possibilities of using off-site supplies from hydro-power stations during station blackout are being investigated as well as a possibility of using a common diesel generator instead of an emergency diesel generator.

Total loss of electrical power accident has been studied in the Temelín Accident Analyses in the part 15.2.6 - Coincident Loss of On-site and External AC Power to the Station. This accident is classified as an event of ANSI class II. From the performed analyses, the safety limits corresponding to this class will not be violated. Resolution of this transient will be included in EOP.

France

H3: Procedures used in case of black-out (total loss of external and internal electrical sources), in case of primary circuit closed, energy is extracted like for H1 procedure, a specific GUS (ultimate backup device): turbine generator (gas turbine) for 1300 and 1400 MW subseries or diesel for 900 MW subseries, (allows to maintain seal water injection to the primary coolant pump via the test pump, and necessary control for operators), in case of break on primary circuit makeup is realized using a specific motor-driven pump (control in a safe state for about 10 hours).

Germany

For German plants, 'Station Blackout' has been defined as the loss of all non battery powered energy supplies, because DC from batteries is also changed to AC to operate some equipment for at least two hours. As a result of the analysis, feed and bleed (F&B) of the secondary side has been regarded as an appropriate measure to prevent core melt. Due to the lack of active safety pumps, F&B of the primary side is only a measure to gain time for recovery actions. After core melt has occurred, a failure of the reactor pressure vessel at high coolant pressure is prevented by primary bleed. The recombination of hydrogen by catalytic devices or ignitors, internal or external steam explosions and the heat removal from the containment together with the limitation of the containment pressure are areas which still need to be analysed, also in case of 'Station Blackout'.

India

The incident is always analysed thoroughly. Additional full capacity DG is being provided to take care of loss of AC power.

Japan

Complying to the "Safety Design Review Guide" (2), the plant is designed so that it enables safe shutdown the reactor and secures cooling following shutdown in case of loss of total power source for a short duration.

For PWRs, practically, two systems of diesel generators are provided as emergency on-site power systems in addition to the external power system which is firmly connected to the power distribution grid. Therefore, a change of loss of power source is considered extremely limited even for a short duration. Even if total loss of power sources occurs for a short duration, the reactor could be brought to a safe shutdown through actuation of the safety protection systems and the control rod clusters.

The decay heat and residual heat could be cooled for about 30 minutes through the natural circulation of primary coolant in the primary system, and through the operation of turbine operation auxiliary water supply pump and the main steam safety valve. Necessary power for the safety protection systems and the turbine-driven auxiliary feedwater systems is supplied from highly reliable batteries to secure the safety of reactor even during the total loss of power.

For BWRs, the basic concept is almost the same as for PWRs. Reactor can be shutdown and cooled safely even after total loss of power sources for a short duration.

Russian Federation

A guide for accident management including some beyond design basis accidents has been developed by RRC Kurchatov Institute. Based on this, some Russian utilities have written the corresponding procedures.

Spain

Most plants have developed procedures to connect to hydraulic power stations located in the neighbourhood. Consequently, automatic controls of those hydraulic stations have been subjected to some modifications.

A one site licensee with two twin NPPs has incorporated an additional diesel generator able to be connected to either unit. Another licensee owning a one unit site, that counted on two emergency diesel generators plus a non emergency one, has implemented modifications and qualified the non-emergency diesel in order to allow its connection to either emergency AC bus.

Emergency procedures have been reviewed to include operations for securing operability of safety equipment in case of loss of power and manoeuvres to quick AC power recovery.

Ukraine

Two studies will be performed at the Rovno NPP: station blackout; and station blackout and failure of MCP sealing.

Depending on the results of this investigation, compensatory measures and/or modification may be introduced.

A total station blackout for the Zaporozhe NPP is presently being calculated and analysed within the framework of a PSA. These will be used to develop compensatory measures.

USA

Compliance with the SBO rule requirements is verified by review and evaluation of the licensee's submittal and audit review of the supporting documents as necessary. Follow up NRC inspections assure that the licensee has implemented the necessary changes as required to meet the SBO rule.

For the SBO coping capability, the licensee's submittal is reviewed to assess the availability, adequacy and capability of the plant systems and components needed to achieve and maintain a safe shutdown condition and recover from an SBO of acceptable duration. The review process follows the guidelines given in RG 1.155, Section 3.2, to assure:

(a) availability of sufficient condensate inventory for decay heat removal;
(b) adequacy of the Class 1E battery capacity to support safe shutdown;
(c) availability of adequate compressed air for air-operated valves necessary for safe shutdown;
(d) adequacy of the ventilation systems in areas that have equipment necessary for safe shutdown of the plant;
(e) ability to provide containment integrity; and
(f) ability of the plant to maintain adequate reactor coolant system inventory to ensure core cooling for the required coping duration.

The licensee's submittal for any proposed modifications to emergency ac sources, battery capacity, condensate inventory, compressed air capacity, ventilation systems, containment isolation valves, and primary coolant make-up capability is reviewed. The information available in the Final Safety Analysis Report is also reviewed.

ADDITIONAL SOURCES:

- Safety issues and their ranking for WWER-1000 model 320 nuclear power plants, IAEA, EBP-WWER-05, 1996.
- Safety issues and their ranking for WWER-440 model 213 nuclear power plants, IAEA, EBP-WWER-03, 1996.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, Vienna (1992).
- Review of the modernization programme of the Kozloduy NPP Units 5 and 6, IAEA, WWER-SC-143, 1995.
- Modernization programme, Kozloduy NPP Units 5 and 6 January 1995, Revision 0.
- Modernization programme for the Ukrainian NPPs with WWER-1000/320 reactors, August 1995, Revision 0.
- The safety in the Spanish nuclear power plants, Nuclear Safety Council, Spain, May 1992.
- 10CFR50.63, "Loss of all alternating current power."
- 10CFR Part 50 Appendix A, GDC 17, "Electric power systems."
- 10CFR Part 50 Appendix A, GDC 18, "Inspection and testing of electric power systems."
- Federal Register Notice 53FR23203, "10CFR50, Station blackout", June 21, 1988.
- USNRC Regulatory Guide 1.155, "Station blackout", US Nuclear Regulatory Commission, June 1988.
- NUREG-1109, "Regulatory/backfit analysis for the resolution of unresolved Safety Issue A-44, Station blackout", US Nuclear Regulatory Commission, June 1988.
- NUREG-1032, "Evaluation of station blackout accidents at nuclear power plants", US Nuclear Regulatory Commission, June 1988.
- NUREG-1032, "Evaluation of station blackout accidents at nuclear power plants - Technical findings related to USI A-44."
- NUREG-0933, "A prioritization of Generic Safety Issues, US Nuclear Regulatory Commission 1991.
- WASH-1400 (NUREG-75/014), "Reactor safety study, An assessment of accident risks in US commercial nuclear power plants", US Nuclear Regulatory Commission, October 1975.
- NUMARC 87-00, "Guidelines and technical basis for NUMARC initiatives addressing station blackout at LWRs."
- Analyses of feed and bleed measures; K. Neu, G. Herbold, B. Pütter; IAEA: RCM on severe accidents; Berlin; June 1993.
- Alternative primary feed during severe accidents in PWR's; K. Neu, G. Herbold, B. Pütter; NURETH-5; Salt Lake City; September 1992.
- USNRC Information Notice 97-21, "Availability of alternate ac power source designed for station blackout event."

4.2. OPERATIONAL SAFETY ISSUES

4.2.1. Management (MA)

ISSUE TITLE: Replacement part design, procurement, and assurance of quality (MA 1)

ISSUE CLARIFICATION:

Description of issue

There are three scenarios for part replacement:

(1) The replacement part is available from either the original nuclear grade QA programme - approved vendor or another such vendor.

If the replacement part is exactly like the original part and is available from the same vendor or another vendor who has a certified nuclear grade QA programme and can provide a like-for-like replacement part, the replacement part can be procured and installed without extensive design, evaluation, and testing.

(2) The replacement part is no longer available from a nuclear grade QA programme - approved supplier or the supplier no longer maintains such a programme.

Some manufacturers have discontinued their nuclear grade QA programme - approved QA programme but may have the replacement part available as a commercial-grade item. In this situation, the important characteristics of the part must be verified for safety related applications.

(3) The replacement part is no longer available and a commercial-grade equivalent is not available.

In this situation, the certification of the new part and the process used to certify the part must be in accordance with all nuclear grade QA programme requirements.

Safety significance

The safety related use of products not certified for safety related applications pose risks to the health and safety of the public by calling into question the availability, reliability, and operability of safety related components, equipment, and systems relied upon to maintain the nuclear plant in a safe condition.

Source of issue (check as appropriate)

- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

USA

Because of a decrease in the number of qualified nuclear-grade vendors, there has been a change in the procurement practices of the nuclear industry. In the past, licensees procured equipment from approved vendors who maintained quality assurance programmes in accordance with 10CFR Part 50, Appendix B (additional guidance is provided in Regulatory Guides 1.28, 1.33, and 1.123). However, due to the reduction in the number of qualified nuclear-grade vendors, licensees are increasing the number of commercial-grade replacement parts that are procured and dedicated for use in safety-grade applications. Therefore, there has been an increased emphasis to maintain procurement and dedication programmes that meet the requirements of Appendix B and thus assure the quality of items used in safety related applications.

Because the dedication of commercial-grade parts has increased in importance, NRC staff has placed greater emphasis on monitoring this process. The NRC staff identified common programmatic deficiencies in licensees' control of procurement and dedication of commercial-grade items used for safety-grade applications, as well as installed equipment of indeterminate quality, during inspections conducted from 1986 to 1989. As a result, the staff issued Generic Letter 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marked Products," and Generic Letter 91-05, "Licensee Commercial-Grade Procurement and Dedication Programmes." Together, these GLs clarified the staff position regarding adequate procurement and dedication programmes for commercial-grade equipment to be used for safety related applications.

In response to the GLs mentioned above, the industry has in place part dedication and certification programmes which meet Appendix B requirements. The staff has performed several assessments and inspections to verify that the programmes are in place. An inspection module (IP 38703) was issued in November 1993 for use by regional inspectors.

ADDITIONAL SOURCES:

- USNRC Inspection Procedure 38703.
- USNRC Generic Letter 91-05.
- USNRC Generic Letter 89-02.
- Recent USNRC Information Notices 96-40, 96-40 Supplement 1.

COMMENTS:

IAEA/OSART results relating to the design and quality of replacement parts are almost exclusively concerned with the adequacy of quality assurance programmes, receipt inspection, and storage and handling practices for parts following receipt by the utility. There have been no findings or recommendations relating to the use of commercial-grade parts in safety related applications. However, this issue may not have received the prominence in some other countries that it has received in the USA. Few outside the US appear to have developed commercial part dedication programmes.

ISSUE TITLE: Fitness for duty (MA 2)

ISSUE CLARIFICATION:

Description of issue

Fitness-for-duty programmes must provide reasonable assurance that nuclear power plant personnel will perform their tasks in a reliable and trustworthy manner and are not under the influence of any substance, legal or illegal, or mentally or physically impaired from any cause, which in any way diminishes their ability to safely and competently perform their duties.

Safety significance

The influence of drugs or other substances which can affect the reaction capability of the responsible personnel of a nuclear power plant, and in particular the control room operators or supervisors, can create critical situations to the plant, challenging its safety.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Spain

There are on-going discussions on the establishment of fitness for duty programmes in the Spanish nuclear power plants. No decision has been taken to date by the regulatory body.

USA

Regulations are in place in the US which require drug and alcohol testing prior to the granting of unescorted access to nuclear power plants. Random tests of the workforce are then required at a rate of 50% of the workforce per year. A behavioural observation programme is also required. Sanctions are specified for violation of fitness for duty requirements. It is believed that this programme has had a significant deterrent effect. Current positive tests are of the rate of from 1 to 3 per 1000 workers tested.

ADDITIONAL SOURCES:

- CSN Safety Guide 7.4.
- Title 10, Code of Federal Regulations, Part 26.
- USNRC Generic Letter 91-16.
- USNRC Generic Letter 05/09/91 - Summary of FFD experience.

COMMENTS:

OSART missions identified weaknesses in fitness-for-duty programmes at six different nuclear power plants in diverse locations during 1993, 1994, and 1995. In all of these plants, weaknesses in drug abuse programmes were evident. Some weaknesses in alcohol abuse programmes were also noted. However, the potential for impairment from use of alcohol appears to be more widely recognized than it is for legal and illegal drugs. Monitoring of personnel for potential impairment is often accomplished as a routine part of supervision, rather than through a structured programme that includes training of supervisors and other personnel in recognizing and dealing with fitness for duty problems. Though fitness-for-duty programmes are becoming more widely used, many plants outside the US still appear to believe that fitness for duty is not a significant problem.

ISSUE TITLE: Adequacy of shift staffing (MA 3)

ISSUE CLARIFICATION:

Description of issue

Shift staffing should be adequate for normal operations, planned evolutions, and unexpected events. Reviews and experience have shown that shift staffing is sometimes insufficient to accomplish all actions that need to be taken in response to unexpected events before additional staff can be made available to assist.

The demands for response to unexpected events before off-crew assistance is available can be significant. Extra demands that need attention from shift crews during unexpected events include:

- notification of off-site plant and utility personnel, local authorities;
- notification of local and national authorities;
- fire fighting (especially where no permanent fire brigade is available);
- execution of emergency operating procedures and emergency plan procedures;
- rescue and first aid for injured personnel.

Safety significance

Licensees are responsible, not only for meeting applicable regulatory requirements for shift staffing, but for ensuring that shift staffing and task allocations are adequate for performing all necessary functions during normal operations and the initial stages of unexpected events. Additional personnel can be called to assist with unexpected events, but during the time required for them to respond, insufficient shift staffing could result in inadequate response to emergency conditions. Insufficient protection of plant equipment from damage or malfunction could result.

Transients which affect multiple units at a plant site create unusual challenges for both systems and operations personnel. In particular, when the units are lost as a result of electrical disturbances, many safety and non safety systems are adversely affected and create particularly challenging scenarios which are not always foreseen by analyses and tests.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

USA

10CFR50.54(m) defines minimum staffing levels for licensed personnel. In addition, licensees are required to provide engineering expertise on shift. NPP licensees may provide the required engineering expertise on shift by having either a dedicated Shift Technical Advisor (STA) or one of the Senior Reactor Operators (SROs) specifically trained and qualified to perform that function. No comparable minimum staffing level requirements have been established for non-licensed operators or for other types of job classifications.
The NRC issued USNRC IN91-77 to alert licensees to the problems that could result from inadequate controls to ensure that shift staffing is sufficient to accomplish all functions required by an event. USNRC IN91-77 called attention to the practice of many licensees of assigning operating staff personnel multiple responsibilities which could impact their ability to perform all of the actions specified in the licensee's administrative controls and required by an event. However, subsequent

operating events and inspection results continued to show problems in this area. Consequently, the NRC proceeded with further research on shift staffing levels and task allocation which encompassed all licensee staff actions initially needed during an event. This research included an NRC project to address the adequacy of minimum shift staffing levels through a shift staffing study. The results of this study are described in USNRC IN91-77.

On December 15, 1992, both units of Sequoyah NPP ran back (Unit 1 ran back to 72% power and Unit 2 ran back to 67% power) due to the loss of electrical control components of the station air system (Morning Report 2-02-0155). On December 31, 1992, both units tripped due to an electrical fault in switchyard. The event was further complicated by too few operators on shift (one control room is shared by both units at Sequoyah. Only one reactor operator was available to operate Unit 2 during this event) (IN93-44).

ADDITIONAL SOURCES:

- USNRC policy, 50FR43621, issued October 28, 1985.
- NUREG-0737, "TMI action plan," issued November 30, 1980.
- NUREG-0578, "TMI-2 Lessons learned Task Force status report & short term recommendations," issued July 31, 1979.
- USNRC Generic Letter 86-04, "Policy statement on engineering expertise on shift," issued February 13, 1986.
- USNRC IN 95-48, "Results of shift staffing study."
- USNRC IN 95-23, "Control room staffing below minimum regulatory requirements."
- USNRC IN 91-77, "Shift staffing at nuclear power plants," issued November 26, 1991.

COMMENTS:

Regulatory requirements for shift staffing may vary widely among countries. OSART missions have resulted in one proposal to adjust shift staffing and in two proposals to ensure that shift personnel are adequately trained for their tasks.

ISSUE TITLE: Control of outage activities to minimize risk (MA 4)

ISSUE CLARIFICATION:

Description of issue

Plant operating experience has shown that pressurized water reactors and boiling water reactors are susceptible to a variety of abnormal events during shutdown conditions.

Traditionally, outage planning has been primarily concerned with productivity and efficient use of resources during outage periods - co-ordination of large amounts of work in diverse locations. Technical specifications and operator's judgement were relied upon to ensure that adequate safety systems and equipment were available for operation. In some cases, operators had to make frequent judgements on taking equipment out of service, with no integrated, analysed plan governing removal of systems from service. Planning, which takes into account the safety milestones and contingencies for mitigating events occurring during an outage, can minimize shutdown risks.

Some plants have used PSA or deterministic methods for configuration management to minimize risks during outages.

The technical specifications also play an essential role in maintaining the required level of safety by providing the operators with rules, covering all the states of the plant, which permits compliance with the original design.

Improper work planning and co-ordination among the several areas of a plant, e.g. operation and maintenance, can lead to incomplete or erroneous activities and to unnecessary safety equipment unavailability. Deficiencies such as lack of plant short term working plan, lack of authority and involvement of control room staff, e.g. shift supervisor when authorizing and determining priority to work to be done are examples. This issue becomes worse during outages, when the number of tasks to be co-ordinated generally increases drastically. In addition, the number of contractors in the plant also increases.

Safety significance

The main risks of safety being detrimentally affected are summarized below:

- lack of co-ordination between operation and maintenance;
- work on safety components required to be operable by the technical specifications;
- plant status changeover with unavailable components or systems required;
- requalifications missing, non acceptable test.

Source of issue (check as appropriate)
- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

For the French PWRs, several measures have been taken to monitor and reduce the shutdown risks.

The development of technical specifications for shutdown conditions is a difficult challenge because it is possible to have multiple configurations of the plant. Furthermore, the unavailability of components and systems for maintenance or repair increases the difficulty in defining fall back states. To overcome these difficulties, it would be useful to define operating domains, grouping various reactor states, taking into account the main end result for the operation and the same safety objectives. For instance, shutdown for refueling, shutdown for maintenance, shutdown with RHR operating and shutdown with S.G. operating could be subjects of separate sections, each of them regrouping different reactor configurations such as "cold shutdown for maintenance with reactor vessel slightly open."

Two aspects have to be developed:

- clear separation between the fallback state and the state of the unit for which the unavailable equipment is no longer required;
- the correlation between the different states.

Germany

The shutdown state can be subdivided into several plant operating states. For each state, plant specific safety requirements have been defined. The purpose of these requirements is to ensure the safety functions and that a sufficient degree of redundancy is available in all states to enable the safety function performance. Regarding maintenance work on specific trains of safety systems, a redundancy plan has been developed. In this plan, it is defined how many trains of the core cooling systems must be available in the different plant states and in which combinations they have to operate, thus excluding combinations where the trains in operation do no longer meet the safety requirements (consideration of single failure at one train plus unavailability of another train due to maintenance work).

The safety requirements also include the necessary restrictions regarding maintenance work on the reactor protection system and on vital I&C during the different plant operating states.

Japan

It is required that "at the operation of reactor facilities, in case that some of multiple functions of safety protection systems and engineering safety facilities are excluded" necessary functions shall be secured according to the established procedures.

On the basis of the above, utilities have established concrete safety standards for shutdown and observe them to secure shutdown safety. They are paying attention to the cases of abnormal events experienced in other countries.

There have been no reports of such cases as threatening the integrity of core during plant shutdown in Japan. They are collecting information on trend of the world.

ADDITIONAL SOURCE:

- Recent USNRC Information Notices 95-03, 95-35, 95-54, 95-57, 96-37, 96-65, 97-83.

COMMENTS:

Within the past three to five years, many utilities have begun safety analysis of outage schedules as a routine part of outage planning. For some utilities, the outage schedule not only includes the equipment that is to be removed from service for repairs or modification, but also the specific equipment that must remain in operation at any given time to provide the desired level of protection for the reactor core and fuel. The full extent to which such safety planning is used is unknown. There have been no OSART findings related to inadequate safety planning of outages.

Only two issues from the OSART findings refer to technical specifications for shutdown conditions. One concerns the establishment of limits and conditions for the shutdown state and the other concerns the provision of checklists so that the shift crew can determine that all technical specifications requirements are met during shutdown.

Examples of good practices identified by OSART missions include an outage structure organization where the co-ordination for both the preparation and the performance of work during the outage is emphasized. The responsibilities for individuals are clearly defined and understood. The procedures for preparation, scheduling and performance of the outage are structured to provide a high level of safety. The outage schedule highlights plant condition versus safety equipment/safety train availability.

ISSUE TITLE: Degraded and non-conforming conditions and operability determinations (MA 5)

ISSUE CLARIFICATION:

Description of issue

Design bases for plant systems are identified in safety analyses reports. In most countries, technical specifications control the operation of the plant and the degree to which departures from the design bases are acceptable. For example the design bases may specify that safety systems operate assuming a single failure, while the technical specifications allow limited periods of operation when a safety system cannot withstand a single failure (e.g. one of two safety trains is not operable).

Without any information to the contrary, once a system is established as operable, it is reasonable to assume that it will continue to remain operable. However, whenever the ability of a system of structure to perform its specified safety function is called into question, operability must be determined from a detailed examination of the degraded or non-conforming conditions.

When a loss of functional capability or quality is identified, the system or structure is said to be degraded. For example a concrete wall or support, the failure of which could affect a safety system, is cracked. A non-conforming condition results from failure to meet requirements. For example:

(1) There is failure to conform to a code or standard specified in the safety analysis report.
(2) As-built equipment does not meet safety analysis report design requirement.
(3) Operating experience or design review demonstrate a design inadequacy with respect to the safety analysis report.
(4) Required documentation demonstrating the qualification of equipment in accident environments is not available or deficient.

Once a degraded or non-conforming condition of a system, structure, or component (hereinafter "systems") important to safety has been identified, an operability determination should be made as soon as possible, consistent with the safety importance of the system. For systems identified in the Technical Specifications (T/S), the Allowed Outage Times contained in the T/S generally provide reasonable guidelines for the safety significance of the system and therefore for the promptness of the operability determination. For systems outside the T/S, engineering judgement must be relied upon to determine safety significance. The operability decision may be based on analyses, a test or partial test, experience with operating events, engineering judgement, or a combination of these factors taking into consideration equipment functional requirements.

Safety significance

Degraded and non-conforming conditions that impact the operability of systems, structures, or components important to safety must be timely identified and corrected so that those systems can continue to be relied upon during and following design basis events to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (3) the capability to prevent or mitigate the consequences of an accident that could potentially result in an off-site release.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Japan

The conditions and the limitations for operation of the system had been established in the safety rules on individual facilities, and operability of the system is judged by qualified engineers with sufficient operating experiences. The systems are periodically inspected, and in case any degraded or non-conforming conditions were found, operability of the system is immediately subjected to their judgement.

USA

The NRC issued a generic letter to inform licensees of two recently issued sections of Part 9900, Technical Guidance, of the NRC Inspection Manual. The first was, "Resolution of Degraded and Non-conforming Conditions." The second is, "Operable/Operability: Ensuring the Functional Capability of a System or Component."

ADDITIONAL SOURCES:

- USNRC Generic Letter 91-18, Revision 1, October 8, 1997.
- USNRC Generic Letter 91-18, "Information to licensees regarding two NRC inspection manual sections on resolution of degraded and non-conforming conditions and on operability", November 7, 1991.
- Recent USNRC Information Notices 96-20, 97-60, 97-80.

COMMENTS:

This has not been identified as a problem during OSART missions. However, some plants have not yet adapted technical specifications that clearly state allowed equipment outage times, so the issue could be more significant than expressed above.

ISSUE TITLE: Configuration management of modifications and temporary modifications (MA 6)

ISSUE CLARIFICATION:

Description of issue

Modification of nuclear power plants has been, and still is, extensively practiced by the utilities in order to update the design of their plants and to take new regulatory requirements, technology developments and their own operating experience into account.

When checking plant temporary or permanent modifications, it is essential to first ensure that the changes comply with the design bases of the affected structure, system or component, and secondly, that they are correctly incorporated into the plant documents (including the operational documents) which must, at all times, be consistent with one another.

Safety could be affected, following modifications, in three aspects:

- Historic:
  The documents must accurately reflect the physical and functional characteristics of the plant at all times, especially for the future.

- Consistency of modifications:
  The documentation must take simultaneous modifications into account and link them.

- Impact on operation:
  The configuration management must help in updating the operational documents which are affected by the modifications.

For older plants, the design bases for the systems may not have been documented or maintained. A number of plants have undertaken efforts to reconstitute their design basis.

Safety significance

Changes to plant systems maintenance and operating procedures which are made without conforming to the original design bases of the system can cause the system not to be able to perform its safety function when called on in an off-normal condition. Because safety systems cannot be tested under accident conditions during the plant lifetime, a non-functional system state can go undetected until called on in an accident.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Japan

Under "The Electric Utilities Industry Law", utility companies are obliged to obtain the approval of work plan by MITI or to notify MITI when they perform modification of the existing nuclear power plants in Japan.

Depending upon the importance, amendment of the licensing shall be applied under "The Law for the Regulations of Nuclear Source Material, Nuclear Fuel Material and Reactors."

Japan Electric Association's Guide "Guide for Quality Assurance of Nuclear Power Plant" (JEAG4101) provides the guide for design control. The modification work is also covered by this guide. By complying these law and guides, structure, systems and equipment being affected by modifications are confirmed to be consistent with the design basis. The contents of the modification are precisely recorded and reflected to the relevant document.

USA

Since the Davis Besse event in 1985, which resulted in part from problems introduced by modifications to the original design which were not adequately considered from a safety standpoint, the USNRC has placed increased emphasis on the importance of design control and configuration management. Multi-discipline team inspections such as Safety System Functional Inspection (SSFIs) were developed and conducted to determine whether plant modifications had been conducted with full consideration of the design basis. In some cases, plant modifications were found to have been performed which may have prevented successful system operation in the event that a design basis accident occurred. For example, in some cases additional electrical loads which resulted from plant modifications could have resulted in exceeding the design capability of the emergency diesel generators.

As a result of these inspections, many older US plants initiated design basis reconstitution programmes. However, due to decreased regulatory attention and competition within plants for scarce resources, some reconstitution programmes were not carried to completion. Recent findings regarding inadequate design and configuration control at some plants has now resulted in additional NRC emphasis in this area. Specifically, multidiscipline design inspections have been increased and letters have been issued to all operating plants (October 9, 1996) which required plant executives to certify that plant design and configuration are being adequately controlled and the basis for their conclusions. This has resulted in increased plant self-assessments and design document reconstitution efforts.

ADDITIONAL SOURCES:

- 10CFR50.2 "Design basis."
- USNRC Inspection Manual, Part 93813, "Safety system functional inspections."
- USNRC Letters to all operating power plants dated October 9, 1996 requiring a response under 10CFR50.54(f).
- Recent USNRC Information Notices 95-09, 96-64, 91-29 Supplement 3.

COMMENTS:

A review of OSART results does not show a significant weakness in the area of modification documentation. Several findings relate to weaknesses in the structure and formal requirements of modification programmes. One finding relates to slow updating of documents following design changes -a problem that may well exist, today, in many plants. Another finding relates to weaknesses in documentation and tracking of temporary modifications, but does not associate the problem with the permanent modification programme. During one OSART, use of a computer system was recommended to identify all documents associated with a particular modification, to ensure they are adequately updated. However, this appeared to be an efficiency or simplification item, as no problems with timely document updates were noted.

ISSUE TITLE: Human and organizational factors in root cause analysis (MA 7)

ISSUE CLARIFICATION:

Description of issue

Learning from operational experience is an essential element of any operational safety programme of a NPP. Identifying the right causes of one event requires a rigorous root cause analysis.

There are multiple tools for the analysis of root causes of events. The analysis of root causes of hardware failure is well defined, the analysis of the root causes of human performance is well developed in general, but there are still some limitations in the tools to analyse management and organizational factors. In addition, sometimes the root cause analysis is not performed to the level of detail required to identify management and organizational problems.

Safety significance

Performance of rigorous root cause analyses of the events is an essential element for maintaining and improving the operational safety of the plants, by detecting and correcting hidden weaknesses in the design or the operation practices.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Japan

Reportable incidents or troubles defined by law or ministerial order are to be promptly reported to MITI and findings of causes and plans of the measures are also reported to MITI in a timely manner.

The information is shared by other utilities as applicable to other plants, for preventive measures. Of accidents and troubles, those resulting from human errors are analysed at the Central Research Institute of Electric Power Industry using the technique of J-HPES (a Human Performance Enhancement System of Japanese version developed by the joint study). The results of analysis are provided to utilities, and they utilize them.

Spain

In Spain, the utilities must have personnel trained in the use of a widely accepted root cause analysis method and normally this is the Human Performance Enhancement System (HPES). Such analyses are expected for any significant event occurring at the plants. Some plants carry out detailed root cause analyses of most events.

The Regulatory Body also has certified investigators, who use the Management Oversight and Risk Tree (MORT) to review some of the root cause analysis performed by the utilities and, in specific cases, carry out fully independent root cause analyses of the same events.
One recurring concern is that in the investigation of root causes, the search for lessons to be learned and improvements to be implemented could be considered as being intended for putting the blame on persons.

In the United States, events involving human performance are usually investigated by the licensee using the Human Performance Enhancement System (HPES),developed by the Institute for Nuclear Power Operations (INPO) and modified by the licensee to fit plant specific needs. Similarly, INPO uses HPES as described in Program Description INPO 90-005, "Human Performance Enhancement System" and INPO Good Practice OE-907, "Root Cause Analysis." The USNRC generally uses the Human Performance Investigation Process (HPIP) as described in NUREG/CR-5455, "Development of the Human Performance Investigation Process (HPIP)." Some licensees are considering using HPIP. In addition, several companion documents are used for specific human performance issues. These include: NUREG-1545 Draft, "Communications Corrective Action Plans: Review Criteria," NUREG-0700, Revision 1, Draft, "Human-System Interface Design Review Guideline," NUREG-1220, Revision 1, "Training Review Criteria and Procedures," and "Control Room Observation Protocol (in development)." The USNRC conducts periodic reviews of licensee controls in this area in accordance with NRC Inspection Manual Inspection Procedure 40500, "Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems."

These would include :

- lowering the reporting threshold for events;
- trending and analysis of events;
- dissemination of operating experience;
- improving training for event and root cause analysis.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Systems for reporting unusual events in nuclear power plants. IAEA Safety Guides No. 93 (1989).
- Incident reporting system coding manual. OECD Nuclear Energy Agency. Working document 16. 20.07.88.
- NUREG/CR-5538 (1991) Influence of organizational factors in performance reliability. Overview and detailed methodological development.
- NUREG/CR-5455, "Development of the human performance investigation process (HPIP)."
- NUREG-1545, Draft, "Communications corrective action plans: Review Criteria."
- NUREG-1220, Revision 1, "Training review criteria and procedures."
- NUREG-0700, Revision 1, Draft, "Human system interface design review guideline."
- MORT User's Manual. DE-76-45/4. SSDC-4. Systems Safety Development Center. EG&G. Idaho.
- INPO Good Practice OE-907. Root cause analysis.
- Guidelines for the diagnostic evaluation. Rev. 2. May 1991. Memo from E.L. Jordan of May 17, 1991, USNRC.
- Program Description INPO 90-005, "Human performance enhancement system."
- USNRC Inspection Manual, Inspection Procedure 40500, "Effectiveness of licensee controls in identifying, resolving, and preventing problems."

COMMENTS:

There are other recurring issues indicated by OSART experience which are related to this general subject area.

ISSUE TITLE: Impact of human factors in the safe operation of nuclear power plants (MA 8)

ISSUE CLARIFICATION:

Description of issue

Human factors are significant in many of the operational events at operating nuclear power plants. For example, for about 6 to 7 years, human factors have been implicated in 70 to 80% of the French events classified "significant for safety." From 1988, the number of significant events is about constant: 6 to 8 significant events/unit/year (in 1986-1987 it was from 10 to 12), furthermore 27% of these events are consecutive to a defect of communication).

Sensitive points are:

- insufficient individual and collective preparation, in particular for routine and simple actions (difficulties are underestimated);
- insufficient preliminary consideration of risks, difficulties, and replies to any events;
- self control and independent control badly used;
- inadequate operational communications and co-ordination (management of interactive activities).

Examples of the most frequent incidents are:

- on multi-unit sites, confusion of units, (not only on 900 MW twin-units);
- confusion of equipment;
- defective alignment of circuit components;
- lack of preparation for certain instructions;
- lack of precision of operational requirements.

Safety significance

Inadequate consideration of human factors in regular plant activities, training and documentation and insufficient analysis of man-machine interfaces, notably contribute to increase the number of safety significant events.

Source of issue (check as appropriate)
- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

In France an action programme foresees in particular:
- to improve man-machine interfaces, documents, organization (but analyses show that it is not the main origin of HF incidents);
- to develop the individual's questioning attitude as stated in INSAG-4 (§ 59: questioning attitude + cautious approach + communication), either at local operational management level or at higher level of hierarchy;
- to set up local systems of improving competences in NPPs;
- to reassess the operational documents quality.

A HF national team has implemented experimental actions in 1996 on about 1/3 of French NPP sites. Furthermore the experience feedback of new interfaces man-machine of Chooz B1-B2 (computerized interface of the control room) will be analysed.

As one of the aftermath of the OSART inspection at Dampierre (4 x 900 MW PWRs) in 1996, EdF general inspector of safety recalled, in a report on nuclear safety in 1996, the main recommendations of OSART experts:

- to increase the landmarks on the site to valorize interventions of team leaders and managers;
- to improve operator retraining, with an independent control of certification, to be sure not to act by routine, and to develop training of auxiliary operators and technicians;
- to formalize the changing of shifts in a written document;
- to improve identification and labelling of components;
- for maintenance, to precise: means and criteria for the detection of deviations, ranking of priorities for pending work, and improving site work in fuel handling areas.

USA

The NRC is concerned with such characteristics as professional responsibility/accountability, personal dedication/performance, safety conscious environment, questioning attitude, commitment to excellence, sound procedures and effective communication. In September 1992, the Commission issued a "Policy Statement on the Conduct of Nuclear Power Plant Operations. This policy statement provides the Commission's expectations in the following areas:

- Professional conduct within CR.
- Formal performance of control room activities.
- Control room secure from intrusion.
- Operator at controls and immediate supervisor continuously alert to plant conditions and activities affecting plant operations.
- Control room activities limited to those necessary for safe operations.
- Activities outside control room with potential to affect plant operations fully co-ordinated with control room.
- Written records of plant operations carefully prepared and maintained.
- Working environment maintained to minimize distractions to operators.
- Foreign objects and materials restricted from area "at the controls."

The USNRC assesses licensee performance through onsite inspections and event investigations. Moving up the evaluation process hierarchy, the USNRC conducts Plant Performance Reviews, and utilizes the Integrated Performance Assessment Process and Systematic Assessment of Licensee Performance process. Finally, at the highest level the Senior Management Meeting process develops the plant "watch list" and identifies those plants that have demonstrated superior performance. The aspects examined in the above activities are safety focus, problem identification and resolution, quality of operations and work, programs and procedures related to safety, and safety assessment and corrective actions.

ADDITIONAL SOURCES:

- USNRC "Policy statement on the conduct of nuclear power plant operations," September 1992.
- USNRC SECY-96-093, "Guidance for senior management meeting and plant evaluation processes," May 1, 1996.
- USNRC Inspection Manual Inspection Procedure 93808, "Integrated performance assessment process (IPAP)," September 6, 1995.

ISSUE TITLE: Effectiveness of quality programmes (MA 9)

ISSUE CLARIFICATION:

Description of issue

Effective quality programmes are needed in all activities bearing on safety at a nuclear power plant to ensure with high confidence that all items delivered and services and tasks performed meet specified requirements.

In some cases, the formal quality assurance programmes which have primarily, in the past, monitored the formal approval processes for engineering and operational activities, have not been effective in preventing events or assuring that safety systems achieve and retain their intended capabilities to fulfil their safety functions if called on in off-normal conditions. In addition, the formal processes which were originally put in place to assure safety, have become inefficient and cumbersome to the point of not being thoroughly implemented by plant staffs. Many plants are re-engineering their administrative, operation and engineering processes to achieve more efficient production. This is sometimes motivated by competitive economic pressures.

To assure that a high confidence is maintained that all safety tasks are rigorously carried out, adequate quality programmes must be implemented for all safety activities. These quality programmes are increasing directed to emphasizing self-assessment by the line organization of the effectiveness of safety process by testing the adequacy of work products. Substantive evaluation, as opposed to process evaluations by the formal quality assurance organization and the regulatory body are also being emphasized.

Safety significance

The failure to rigorously assure that all safety activities at a nuclear power plant meet specified requirements can result in unnecessary challenges to plant safety systems and personnel or the inability of safety systems to perform their intended functions when called on.

Source of issue (check as appropriate)
- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

USA

The USNRC has required formal quality assurance programmes for some time. As indicated in the text above, plant experience has indicated that self assessments by the line organization are an important component of achieving both quality and efficiency. Oversight by a formal quality assurance organization is still required, and substantive evaluation as well procedural compliance is being emphasized.

ADDITIONAL SOURCES:
- INTERNATIONAL ATOMIC ENERGY AGENCY, NUSS 50-C-QA.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Series No. 75-INSAG-3.
- CSN Safety Guides Series 10.
- 10CFR Part 50, Appendix B.
- USNRC Regulatory Guide 1.33 Revision 2, *Quality assurance program requirements (Operation)*, 1978.
- USNRC Information Notice 96-28.

ISSUE TITLE: Adequacy of procedures and their use (MA 10)

ISSUE CLARIFICATION:

Description of issue

Several plants were identified, during OSART missions, where the control of documents needed improvements. Some of these deficiencies are as follows:

- lack of administrative process (procedure) to produce, validate, approve, review and control procedures, including temporary changes. This leads to problems such as: obsolete or not approved procedures in use in the control room and in the field; procedures in use with a lot of temporary changes; no means to collect, file and incorporate proposals made by operators and plant personnel to improve the operating procedures; temporary procedures with less levels of approval than the permanent procedures; and temporary procedures in force for long periods (e.g., more than a year);
- lack of adherence to approved administrative procedures to control the plant documentation, leading to deficiencies similar to the above ones;
- Procedures in use that do not reflect the last plant components / systems changes.

Safety significance

The proper use of procedures is important to safety. Procedures are based on safety analyses, and subsequent safety assessments are based on procedures. However, it is also necessary to understand the purposes and limitations of the procedures, and be able to determine whether they are applicable to any given situation.

The lack of control of procedures could lead to human errors, mislead operators and plant staff when operating the plant during normal and emergency conditions. These incorrect actions by plant operators, mainly during emergencies, could challenge the safety functions.

Source of issue (check as appropriate)

- __________ operational experience
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

USA

The USNRC expects licensees to adhere to procedures and to have established policies that effectively control the development and use of procedures and the procedural change process. Furthermore, the combination of an individual's training and the proper use of written procedures consistent with the licensee's procedural adherence policy is expected to be sufficient to provide for the successful completion of the associated task.

The licensee's self critical attitude is essential to the proper use of procedures and to the identification of procedural deficiencies. If procedural deficiencies are identified, the USNRC expects appropriate changes to the procedure will be effected prior to continuing with the procedure. However, should an emergency situation arise when the time for a proper safety response does not permit changing a deficient procedure, then deviation from the procedure in the interest of plant safety is considered appropriate with proper levels of approval.

Inadequate control of procedures continues to lead to procedural deficiencies and human errors in US plants. Although most US plants maintain relatively stringent controls over Emergency Operating

Procedures (EOPs) (See issue MA11.), they exercise significantly fewer procedural controls that have less detailed guidance regarding non-emergency procedures. The USNRC requires Procedure Generation Packages (PGPs), detailing a plant's procedure development and maintenance processes, only for EOPs. The USNRC requires each licensee to provide clear, written guidance on its procedural adherence policy. The USNRC conducts periodic inspections of non-EOP plant procedures in accordance with NRC Inspection Manual Inspection Procedure 42700, "Plant Procedures." Many licensees are modifying their normal operating, abnormal operating, and system procedures to more closely reflect the format of their EOPs.

ADDITIONAL SOURCES:

- Code of Federal Regulations, "Instructions, Procedures and Drawings," Criterion V and VI of Appendix B of Part 50 of Title 10.
- Code of Federal Regulations, "Conditions of Licenses," Parts 50.54(x) and 50.54(y) of Title 10.
- USNRC Regulatory Guide 1.33, Revision 2, "Quality assurance programme requirements (Operation)," 1978.
- USNRC Inspection Manual Inspection Procedure 42700, "Plant procedures."
- USNRC SECY-90-337, "Procedural adherence requirements."
- NUREG/CR-4613, "Evaluation of nuclear power plant operating procedures classifications and interfaces."
- NUREG-1358, "Lessons learned from the special inspection program for emergency operating procedures, conducted March-October 1988."
- NUREG-1358, Supplement 1, "Lessons learned from the special inspection program for emergency operating procedures, conducted October 1988-September 1991."
- ANSI-N18.7-1976/ANS-3.2, "Administrative controls and quality assurance for the operational phase of nuclear power plants," Sections 5.2 and 5.3.
- ANSI/ANS-3.2-1982, "Administrative controls and quality assurance for the operational phase of nuclear power plants," Section 5.2.2.
- ANSI/ANS-3.2-1988, "Administrative controls and quality assurance for the operational phase of nuclear power plants," Section 5.2.2.
- ANSI/ANS-3.2-1994, "Administrative controls and quality assurance for the operational phase of nuclear power plants," Section 5.2.2.

COMMENTS:

Several plants have established clear and comprehensive procedures to produce, validate, approve, review and control procedures, including temporary changes. Some of the measures taken by these plants are the following:

- the number of temporary procedures is low, and they are well identified and controlled;
- the plant staff are stimulated to provide comments to improve procedures. These proposals are given priority adequately;
- the permanent procedures, when partially modified by a temporary procedure, have this change referred to in the user's copy;
- periodic reviews, in same cases even included in the plant surveillance programme, are carried out by plant staff to verify that the plant procedure management is in accordance with the rules in force;
- several plants have indicators on temporary procedures (e.g., number and period in force).

ISSUE TITLE: Adequacy of emergency operating procedures (MA 11)

ISSUE CLARIFICATION:

Description of issue

Preparedness of plant staff to deal with emergencies needs improvement. The degree of improvement required varies considerably from plant to plant. Elements of this process needing attention include quality and completeness of procedures, usability of procedures, training on emergency operating procedures, availability of full scope plant specific simulators, integration of emergency management procedures for various types of emergencies and man/machine interfaces. Additionally, some emergency operating procedures are not designed to deal with "beyond design basis" accidents. (see AA5, "Need for severe accident analysis" and SS 8 "Accident management measures").

Safety significance

Capability to effectively and promptly deal with emerging emergency conditions is essential to limiting their consequences and protecting the plant from unnecessary damage and the public and the operators from the potential harmful effects of releases of radiation to the environment.

Source of issue (check as appropriate)

- ___xx_____ operational experience
- ___xx_____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Spain

New emergency procedures have been elaborated, based on NUREG-0737, Sup. 1, and GL 82-33. Control room operators are performing drills on the use of these procedures within the scope of their training schedule.

The "Severe Accident Management Guides" are under adaptation following the recommendations of USNRC Generic Letter 88-20, App. 3. This task is planned to be completed by the year 2001.

The "Accident Management Guides", a concept applicable to a Spanish plant of German design, is under development and is also planned to be completed by the year 2001.

USA

The USNRC issued guidance documents on emergency operating procedures after the Three Mile Island accident. Each US nuclear steam supply vendor developed model emergency operating procedures, from which plant owners developed plant specific procedures. These procedures were tested on plant simulators and are used in training of plant operators for off-normal conditions.

The USNRC issued requirements in this area in Three Mile Island (TMI) Action Plan Items I.C.1 (NUREG-0660, NUREG-0737, and Supplement 1 to NUREG-0737) and I.C.9. These requirements included licensee submission of "Procedure Generation Packages" (PGPs) that contained plant-specific technical guidelines, a writers guide, an EOP validation program description, and a description of the training program for upgraded EOPs. In August 1992, the USNRC issued NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures." The USNRC initiated its Emergency Operating Procedures (EOPs) Inspection Program to determine if licensees were meeting these requirements. This program focused on the (1) technical guidelines associated with EOP development,

(2) actual production of EOPs from these guidelines, (3) verification and validation (V&V) and maintenance processes associated with EOPs, (4) operator training on the EOPs, and (5) management involvement in the EOP development process. The USNRC issued NUREG-1358, "Lessons Learned From the Special Inspection Program for Emergency Operating Procedures Conducted March-October 1988" and NUREG-1358, Supplement 1, "Lessons Learned from the Special Inspection Program for Emergency Operating Procedures Conducted October 1988-September 1991" to discuss weaknesses found during these inspections and provide guidance to licensees in analysis, development, implementation, and maintenance of EOPs. The Nuclear Management and Resources Council (NUMARC) and the four nuclear steam system suppliers (NSSS) owner's groups (Babcock & Wilcox, Combustion Engineering, General Electric, and Westinghouse) sponsored and the USNRC participated in a series of EOP workshops that conveyed EOP inspection results and provided a forum for participants to share information on EOP activities. Currently, US plants integrate the use of certified, full scope, plant specific simulators in EOP development and maintenance. The USNRC monitors events for EOP involvement and conducts EOP follow-up inspections in accordance with NRC Inspection Manual Inspection Procedure 42001, "Emergency Operating Procedures," as appropriate.

ADDITIONAL SOURCES:

- USNRC Regulatory Guide 1.33, "Quality assurance program requirements (Operation)."
- USNRC Inspection Manual Inspection Procedure 42001, "Emergency operating procedures."
- NUREG/CR-5228, "Techniques for preparing flowchart-format emergency operating procedures."
- NUREG/CR-4613, "Evaluation of nuclear power plant operating procedures classifications and interfaces."
- NUREG-3632, "Methods for implementing revisions to emergency operating procedures."
- NUREG/CR-3177, Vols. 1, 2, and 3, "Methods for review and evaluation of emergency operating procedure guidelines."
- NUREG/CR-2005, "Checklist for evaluation emergency operating procedures used in nuclear power plants."
- NUREG-1977, "Guidelines for preparing emergency operating procedures for nuclear power plants."
- NUREG-1358, "Lessons learned from the special inspection program for emergency operating procedures, conducted March-October 1988."
- NUREG-1358, Supplement 1, "Lessons learned from the special inspection program for emergency operating procedures, conducted October 1988-September 1991."
- NUREG-0899, "Guidelines for the preparation of emergency operating procedures."
- NUREG-0737, "Clarification of TMI Action Plan requirements."
- Supplement 1 to NUREG-0737, "Requirements for emergency response capability."
- USNRC Generic Letter 88-20.
- USNRC Generic Letter 82-33.
- Recent USNRC Information Notices 97-06, 97-16, 97-18, 97-78.

COMMENTS:

All member states visited are aware of the improvements needed in this area.

ISSUE TITLE: Effectiveness of maintenance programmes (MA 12)

ISSUE CLASSIFICATION:

Description of issue

NPPs preventive maintenance programmes used to be based in the equipment supplier recommendations, without taking into account the results of the operation and the importance of the equipment to safety. The effectiveness of the maintenance programmes and the overall safety of the plant should be improved by considering:

- measured results of equipment and systems availability;
- contribution of equipment and systems to the risk as determined by the plant specific PSA.

Some plants have decided to implement reliability centered maintenance programmes.

Safety significance

A maintenance programme without measurable targets of equipment and system availability can lead to unacceptable reductions in plant safety.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

The issue was identified in test and/or maintenance during power operation.

MEASURES TAKEN BY MEMBER STATES:

France

French maintenance programme is currently based on results of PSA studies. This permit optimization of the programme by balancing preventive and corrective maintenance. PSA studies indicate what parameters are sensitive to affect the risk factor. This approach is currently implemented in all French sites and the main consequences consist in an increase of surveillance or new maintenance tasks on sensitive components or systems.

Japan

It is a practice in Japan that maintenance and examination of the equipment related to the reactor safety are conducted systematically during outage. At power maintenance is performed on only limited parts of normal use ventilating systems (component cooling system, normal-use systems, spent fuel pit clean-up system and waste disposal systems).

Use of on-line evaluation of core damage frequency is proposed as an extended application of PSA in the future, and is now under investigation.

Spain

All the Spanish plants are developing new maintenance programmes in accordance with the US Maintenance Rule (10CFR 50.65) and the USNRC RG-1.160, with an objective of having them implemented in December 1997.

USA

The USNRC issued a regulation which requires monitoring of the experience of equipment which affects plant safety and adjustment of maintenance programmes based on actual experience. Inspections to examine plant implementation of the maintenance rule are being carried out.

10CFR50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," (the maintenance rule) was issued on July 10, 1991. Licensees were given a five-year period to put maintenance programmes in place in accordance with the rule requirements. The rule became effective on July 10, 1996.

The rule requires that licensees (1) monitor the performance or condition of structures, systems, and components (SSCs) within the scope of the rule, (2) set goals commensurate with safety that take into account industry wide operating experience, where practical, (3) take appropriate corrective action if goals are not met, (4) perform periodic assessments of performance and condition monitoring and preventive maintenance activities, (5) balance the objective of maximizing reliability with the objective of minimizing unavailability, and (6) assess overall plant safety prior to performing monitoring and preventive maintenance activities. The scope of the maintenance rule includes both safety related and certain non-safety related SSCs that are important to safety. The non-safety related SSCs which are covered by the rule include (1) those that are relied upon to mitigate accidents or transients or are used in plant emergency operating procedures, (2) those whose failure could prevent safety related SSCs from fulfilling their safety function, or (3) those whose failure could cause a reactor scram or safety system actuation.

ADDITIONAL SOURCES:

- 10CFR50.65, Requirements for monitoring the effectiveness of maintenance at nuclear power plants (Maintenance Rule).
- USNRC R.G. 1.160, "Monitoring the effectiveness of maintenance at nuclear power plants." June 1993.
- USNRC Inspection Manual. Temporary Instruction 2515/126. "Evaluation of on-line maintenance, SALP Functional Area: Maintenance" (October 27, 1994).
- Recent USNRC Information Notices 96-15, 97-90.

COMMENTS:

No issues were identified in the OSART findings directly related to this subject. It is noted that it can take a great deal of time and money to set up a comprehensive RCM programme.

4.2.2. Operations (OP)

ISSUE TITLE: Intentional bypassing of automatic actuation of plant protective features (OP 1)

ISSUE CLARIFICATION:

Description of issue

Nuclear power plants include engineered safety features (ESF) that initiate automatically to perform required safety functions. These safety systems allow also for operator intervention and control during operating events. Once the operator intervenes and takes manual control, many ESFs will not automatically re-initiate until the ESF is restored to a standby configuration with the actuation logic reset or rearmed.

Safety significance

Inappropriate defeat or bypass of ESFs during events is of interest because the operator may not be able to recover from such events in complex, stressful situations. Rapid recovery from ESF defeats may be necessary in order to prevent serious degradation of plant conditions that could lead to serious safety consequences.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ *deviation from current standards and practices*
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

India

This aspect is emphasized in operator training. A special register kept in the control room. Needs approval of the station management for bypassing protective features.

Japan

In Japan, safety protection systems and ESF are required, redundancy, testability and functions dealing with transient conditions, by the safety Design Review Guides. Concrete method of the design is based on JEAG4604 "Guide for Design of Safety Protection Systems for Nuclear Power Plants." Systems are designed so that, for example, bypassing of safety protection systems or their removal from service will be annunciated and continuously displayed on the main control room display, and so that the necessary safety protection functions will not be lost.

USA

The NRC issued an information notice to alert licensees to the importance of having formal criteria and training regarding limitations on bypassing plant protective features.

ADDITIONAL SOURCES:

- USNRC Engineering Evaluation, Operating events with inappropriate bypass or defeat of engineered safety features, AEOD/E95-01, July 1995.
- USNRC Information Notice 92-47, Intentional bypassing of automatic actuation of plant protective features, June 29, 1992.

ISSUE TITLE: Response to loss of control room annunciators (OP 2)

ISSUE CLARIFICATION:

Description of issue

Control room annunciators are used by operators to monitor the condition of plant equipment. Several events have occurred in which control room annunciators were lost due to various reasons.

Safety significance

Although plant annunciators are not considered safety related, they are important for the safe operation of a nuclear power plant. Further, an unrecognized loss of annunciators may increase the difficulty of diagnosing problems in plant operations and equipment. In order for plant operators to recognize and respond properly to a loss of annunciators, it is important to have clear procedures, appropriate training, and effective communications between operators and plant support personnel.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Japan

In Japan, it is required by Design Review Guide that the instrumentation and control systems shall be designed to enable monitoring and recording as required, of the parameters necessary to recognize the status of an accident and take countermeasures by adequate method over a sufficient range in case of an accident. The systems shall also be designed to enable monitoring and estimation of the status of reactor shutdown and core cooling in particular by use of two or more kinds of parameters. Based on this guide, the power supply to alarm systems are designed to have redundant emergency batteries or uninterrupted AC power to avoid loss of alarm functions.

Moreover, the training of the operators to deal with such malfunctions as loss of alarm functions are carried out under the training programme by utility companies.

USA

Loss of annunciator events are covered by emergency classification procedures of licensees, premised upon guidance of Appendix 1 of NUREG-0654/FEMA-REP-1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," (November 1980). The NRC recently issued Revision 3 to Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors" (August 1992). That revision endorsed NUMARC/NESP-007, Revision 2, "Methodology for Development of Emergency Action Levels," (January 1992), as an acceptable alternative. With respect to loss of annunciators, the NUMARC/NESP-007 guidance provides an alternative delineation of thresholds for declaring an Unusual Event, Alert or Site Area Emergency.

ADDITIONAL SOURCES:

- USNRC Regulatory Guide 1.47, "Bypassed and inoperable status indication for nuclear power plant safety systems," issued May 1973.
- NUREG/CR-3217, "Near -term improvements for nuclear power plant control room annunciator systems," issued April 1983.
- Three Mile Island Action Plan, Item I.D.(5), "Operator process communication."
- USNRC IN 94-24, "Inadequate maintenance of uninterruptible power supplies and inverters," issued March 24, 1994.
- USNRC IN 93-49, "Improper integration of software into operating practices," issued July 8, 1993.
- USNRC IN 93-47, "Unrecognized loss of control room annunciators," issued June 18, 1993.
- USNRC IN 91-64, "Site area emergency resulting from loss of non-class 1E uninterruptible power supplies," issued October 9, 1991.
- USNRC IN 91-64, Supplement 1, issued October 7, 1992.
- USNRC IN 88-05, "Fire in annunciator control cabinets," issued February 12, 1988.
- USNRC IN 80-10, "partial loss of non-nuclear instrument system power supply during operation," issued March 7, 1980.

ISSUE TITLE: Inadvertent introduction of chemicals and their effects on safety related systems (OP 3)

ISSUE CLARIFICATION:

Description of issue

The inadvertent introduction of chemical products in the primary coolant and other safety systems has caused a rapid corrosive attack of some materials. In several cases, these products were either sulphur species present in the resins of the demineralizers, released because of a mechanical failure and transported by the system flow, or the reactives used in the demineralizers generation process. Sulphur species have proven to be very aggressive against certain materials like the Inconel-600 of the steam generator tubes or other pressure boundary components.

Safety significance

Events of this nature can pose a threat to integrity of the reactor coolant pressure boundary or other safety components.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

India

Indian BWRs have seen a few incidents of this kind; both release of resins as well as chemical break through due to resin saturation. The later is mostly associated with turbine condenser tube failure. Procedure for handling such incidents including post-incident checks have being laid down and followed by the station.

Spain

Spanish NPPs have had several events in which chemical products have been introduced into the primary or the secondary systems, which, in one case, lead to extensive damage. An increase of the surveillance requirements included in the Technical Specifications has been made in the affected plants.

USA

There were a number of events in which resin bed demineralizers failed in nuclear plants in the USA. Different types of failures have occurred and were due either to equipment failures or human error. Some of them produced degradation of the demineralizer performance due to resin saturation and some resulted in damage to the resin causing its release into the stream of fluid passing through the demineralizer. This last type of failure was of a special concern because the released resin could clog pump strainers and affect some of the systems required for safe plant operation. Also, the release of corrosive or radioactive materials would increase corrosion and radiation levels in the plant. The NRC recognized the importance of the problem, because some of these events may not be bounded by the licensing basis for the current nuclear power plants. The NRC reviewed the problem under Generic Issue 71, "Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety,' and made several recommendations. They included installing filters on the outlet of demineralizers and evaluating and improving current operating and training procedures to enhance operators' capabilities and reduce the chances of human error. The NRC also continues to follow the events related to demineralizer failures and to keep the industry informed through issuing Information Notices.

ADDITIONAL SOURCES:

- CSN/IS/26/94, 27/94 and 28/95, Report to the Congress and Senate, Nuclear Safety Council, Spain.
- USNRC IN-82-14.
- USNRC IN-82-32.
- USNRC IN-83-49.
- USNRC IN-96-11.

ISSUE TITLE: Precautions for mid-loop operation (OP 4) (PWR)

ISSUE CLARIFICATION:

Description of issue

During cold shutdown periods, for instance, a steam generator tube inspection may require that the primary coolant level in the primary circuit be lowered to a minimum which is also called "low working conditions of the Residual Heat Removal System (RHR)" or mid-loop operation conditions.

Experience feedback from French and American plants has shown that the operators had difficulties in controlling the primary coolant level in such conditions in order to prevent RHR pump cavitation.

The issue was revealed by operational experience and assessed in detail in the frame of the PSA of the French PWRs.

In a few cases, this resulted in a complete loss of the RHR system. In some plants in USA, this scenario even led to boiling of the primary coolant and high radioactivity within the reactor building. The probabilistic safety assessments of the French 900 MW and 1300 MW series published in 1990 showed that a loss of RHR scenarios due to a vortex was a significant contribution to the global risk of core melt.

Safety significance

A complete loss of the RHR system could lead to a fuel damage during periods when the primary circuit and containment are open for maintenance.

Such scenarios have therefore a high safety significance.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ___xx_____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

In the French PWRs, the operator is provided with the possibility of continuously monitoring the level of the primary coolant from the top of the pressurizer down to the lower parts of the primary loops with a precise and reliable measurement.

In addition, there is a strong safety policy on mid-loop operations. All mid-loop operation must be reviewed and approved in advance by DSIN. Mid-loop operations before refueling are only approved for special situations. In Flamanville, comprehensive training in mid-loop operation has been established based on an excellent teaching file. Control room operators are regularly trained in understanding the physical background for potential risks, such as residual heat removal degradation during reactor drainage and reactor cold shutdown operations. The operating procedure for mid-loop operations also provides the control room operator with an explanation of instrumentation and controls used during this phase, particularly during a reactor drainage sequence. Such a practice improves the ability of control room operators to better carry out all actions with a full understanding of the background of the physical process during this phase.

Germany

In German PWRs, precautions for mid-loop operation can be subdivided in safety requirements and design measures.

Restrictive safety requirements are implemented in the plant manual for operation. It is required that during mid-loop operation not all available trains of the RHRS are allowed to operate in RHR mode. One of the four trains must be in a stand-by mode and must be aligned to the RWST. In case of loss of the operating RHRS pumps the train in stand-by mode can easily be used to pump water from the RWST in the RCS to increase the water volume and delay reactor coolant heatup. In addition, as long as the RCS is closed, at least one of the four SG must be operable including its dedicated emergency feedwater supply and PORV to enable core cooling via the secondary side in case of a complete loss of the RHRS.

The design measures to preclude loss of RHR during mid-loop operation imply redundant mid-loop level monitors as well as interlocks to prevent overdraining of the RCS. The operator is supported by a special monitor for the level in the RCS loops with alarms at low level in the loops. In addition, interlocks will stop draining of the RCS in case of too low level in the loops, to prevent cavitation of the RHRS pumps.

Japan

In Japan, the improvement of mid-loop operation is taken as follows:

- Technical Specification and emergency operation instruction for shutdown condition include mid-loop operation are studied and established. It is studied that during mid-loop operation, the removal measure of the decay heat in addition to RHRS is provided.
- As the improvement of the level indicator of the primary coolant loop, several kinds of the level indicator are provided.

Spain

The Technical Specifications for mid loop operation have been modified, requiring more restrictive conditions regarding the operability of safety systems and equipment involved in this condition of operation.

When planning refueling activities, the guide NUMARC 91-06 "Guidelines for Industry actions to assess Shutdown and Low Power Operations for Nuclear Power Reactors", Rev. October 1994 is followed.

The operations personnel have been trained on mid loop operation drills.

ADDITIONAL SOURCE:

- OECD NEA: CSNI Report No. 168 (Restricted) "Generic study of events involving loss of residual heat removal function"; prepared by experts of PWG 1; Nov. 1989.

4.2.3. Surveillance and maintenance (SM)

ISSUE TITLE: Adequacy of non-destructive inspections and testing (SM 1) (WWER)

ISSUE CLARIFICATION:

Description of issue

The non-destructive testing (NDT) for reactor coolant system in-service inspection for WWERs is carried out according to the individual Member States' requirements, which are in principle based on former Soviet Union Standards. Defect-reject manufacturing approach is used rather than defect-follow approach, which is capable of a timely detection of the degradation, if applicable. Different techniques and tools are used. Some deficiencies have been revealed, related to vessel inspection from outside, testing of underclad area, and testing of steam generator collectors and tubing. There is also restricted accessibility of some vessel welds, vessel head, vessel head penetrations, piping welds, steam generator shell welds, and specific piping nozzles. Furthermore, recent results of a programme similar to PISC indicate insufficient reliability of NDT methods, tools and personnel, as in the case of the PISC programme results. There are no qualification requirements for methods, personnel and equipment established at present.

Safety significance

The design of some sections of WWERs does not allow for inspection by NDT methods. The approach adopted is not adequate to detect degradation in time. Reliable in-service inspection is a key provision required to preserve the integrity of the second barrier. It cannot be considered sufficient without adequate qualification requirements. Undetected defects can result in primary circuit failures, which significantly increase the challenge of the safety function cooling the fuel. In case of a vessel break, the safety functions "cooling the fuel" and "confining the radioactive material" are both lost, leading to unacceptable consequences.

Source of issue (check as appropriate)

- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria

Design of ultrasonic and eddy current inspection equipment for SG and primary circuit, and design of visual and TV equipment inspection facilities were proposed for the Kozloduy NPP Units 5 and 6.

Czech Republic, Hungary, Slovakia

The development of NDT qualification requirements has been requested by the regulatory bodies.

Russian Federation, Ukraine

The development of improved testing has been proposed.

ADDITIONAL SOURCE:

- Safety Issues and their Ranking for WWER-1000 Model 320 Nuclear Power Plants, IAEA, EBP-WWER-05, 1996.

ISSUE TITLE: Removal of components from service during power or shutdown operations for maintenance (SM 2)

ISSUE CLARIFICATION:

Description of issue

The focus of some licensees on reducing cost by reducing the length of outages has increased both the amount and frequency of maintenance performed during power operation. Some licensees have limited the planned equipment outages to a single train of a system while others would allow multiple equipment in other systems within a single train to be out of service as long as it did not violate Technical Specifications. (See MA4 for a discussion of control of activities during outages).

In some cases, licensees have expanded the on-line maintenance concept without thoroughly considering the safety (risk) aspects. The on-line maintenance concept appears to extend the use of Allowed Outage Times (AOT) stated in the Technical Specifications beyond the random single failure in a system and a judgement of a reasonable time to effect repairs upon which the AOTs were based. The capability to withstand a single failure in fluid and electrical systems is a plant specific design requirement that is contained in the general design criteria in 10CFR50, Appendix A. Compliance with this criteria is demonstrated during plant licensing by assuming a worst case single failure which often results in multiple equipment failures. This does not imply that it is prudent to voluntarily remove equipment from service to perform on-line Maintenance on the assumption that such actions are bounded by the worst case single failure. Sole reliance caused be placed on Technical Specifications to prevent operation in risk significant configurations.

Safety significance

Safety functions can be impaired if equipment outages are not carefully controlled.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

USA

The NRC issued a letter in 1994 to the nuclear industry reemphasizing NRC guidance on voluntary on-line maintenance. Specifically, the letter stated that: 1)the practice should represent a net safety benefit and be warranted by operational necessity, not by convenience, 2) the practice should not be abused by repeated entry into and exit from the limited condition of operation, 3) the removal from service of safety systems and important non-safety equipment should be minimized, 4) Any component testing or maintenance that increases the likelihood of a plant transient should be avoided; plant operation should be stable, and 5) Configuration control and system/train unavailability must be effectively managed to minimize risk during maintenance activities.

Long term resolution to this issue will be achieved through the maintenance rule which requires: 1) assessment of overall effect of out-of-service equipment on performance of safety functions and 2) balancing of component monitoring or preventive maintenance against system unavailability.

ADDITIONAL SOURCES:

- OSMIR database.
- 10CFR50.65, Requirements for monitoring the effectiveness of maintenance at nuclear power plants (Maintenance Rule).
- USNRC Regulatory Guide 1.160, Monitoring the effectiveness of maintenance at nuclear power plants, dated June 1993.
- USNRC Inspection Manual Temporary Instruction 2515/126, Evaluation of on-line maintenance, dated 10/27/94.
- USNRC Inspection Manual Part 9900: Maintenance - Voluntary entry into limiting conditions for operation action statements to perform preventive maintenance dated 4/18/91.

COMMENTS:

Three practices of interest were identified from OSART findings:

(1) In one plant, revised technical specifications, which were to be approved, include unit fallback times for combinations of unavailability of vital equipment. The fallback times are based on results of probabilistic safety assessments. In the event of a simultaneous unavailability of vital equipment, a decision tree guides the operators to easily determine the unit fallback time that results from the combination of two or more items of equipment being unavailable. This approach maintains the defence in-depth principle by prescribing more conservative limits for multiple unavailabilities.

(2) In one plant, a computer is used to analyse unavailabilities. It provides the basic data for nuclear safety assessment by three independent and complementary groups:

- <u>Safety and Quality Team</u>: Engineers prepare weekly nuclear safety report that provides staff of the various departments and site management, with general comments concerning site nuclear safety level; indicators related to the management of equipment unavailability; and a list of event reports that are issued or will be issued.
- <u>Maintenance</u>: A computerized file is used by technical management staff members to develop statistical unavailability analyses, which are shown in graphic report.
- <u>Operations</u>: Operations managers provide an overview of unit nuclear safety through the weekly operations report. The elements developed in these reports are discussed in the weekly department meeting.

These processes have resulted in improved co-operation between the Safety Quality Team, Operations and Maintenance, and have reduced the incidence of unavailabilities.

(3) A plant has developed a process to control equipment inoperability supported by a flow chart to identify the required steps in this process. This flow chart is an important tool for the control room personnel to follow the correct process and to identify the required steps such as: definition of inoperability; check of redundant train inoperability; estimation of duration shorter than the fallback time; requalification of the equipment and restoration to normal operation.

ISSUE TITLE: Use of freeze seals (SM 3)

ISSUE CLARIFICATION:

Description of issue

Freeze seals are used to isolate components (such as inboard isolation valves) for maintenance in locations that cannot otherwise be isolated. The seal is created and maintained by applying a cooling agent such as liquid nitrogen to the exterior of the pipe. The cooling agent freezes the water within the pipe section, thus sealing the pipe. When used in the reactor coolant system (RCS) pressure boundary, these freeze seals become a temporary part of the pressure boundary.

Some licensees have used piping mockups to thoroughly evaluate freeze seal applications prior to their use on reactor system piping. Important considerations include examining training, procedures, and contingency plans associated with the use of freeze seals, and evaluating the need for and availability of additional water makeup systems and their associated support systems.

Safety significance

If a freeze seal fails, it can result in an immediate loss of primary coolant. Of particular concern would be a failure of a freeze seal in a system connecting to the vessel's lower plenum region, such as the reactor water cleanup (RWCU) system at boiling water reactor (BWR) facilities. The staff has estimated that the reactor core could be uncovered in less than 1 hour if the freeze seal failed completely in the RWCU system. Freeze seal failures in secondary systems can also be significant because of the potential for consequential failures, such as the loss of decay heat removal. The thermal stresses on the pressure boundary resulting from the freeze seal may affect the pipe metal integrity.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

India

There was an incident of failure of freeze seal in a reactor recirculation bypass 6" (line). The incident occurred after six months shutdown. Core cooling did not pose any problem in the Emergency Core Cooling recirculation mode. Procedures were revised and successful re-freezing maintenance could be carried out subsequently.

USA

An information notice was issued to alert addressees to the potential consequences associated with failure of freeze seals used to perform maintenance in piping systems.

ADDITIONAL SOURCE:

- USNRC Information Notice No. 91-41, Potential problems with the use of freeze seals, issued June 27, 1991.

COMMENTS:

As an example of measures taken, a plant developed a procedure which gives limits based on the pipe metal (for carbon steel pipe the pressure has to be less than 20 % of rated pressure) to permit a freeze seal to be installed. In addition, special post maintenance tests are required: stainless steal pipe requires only post maintenance non destructive test; and carbon steel pipe requires non destructive test before and after and, in addition, a volumetric test after the freeze seal is removed.

ISSUE TITLE: Use of pressure injection of compounds to seal leaks (SM 4)

ISSUE CLARIFICATION:

Description of issue

Improperly performed injection of leak-sealant compounds into components that are not isolable from the reactor coolant system has resulted in weakening the component's resistance to gross failure, thereby increasing the risk of a non-isolable loss-of-coolant accident. Leaking gasketed joints require carefully-controlled application of leak sealant by methods that avoid damaging the structural strength of the joint and prevent corrosive system fluid from contacting the joint fasteners. Valve body-to-bonnet joints and bolted manway joints are examples of components for which leak sealant injection has been used.

Safety significance

Failure of the intended repair can cause catastrophic failure of component parts, leading to an unisolable leak, increasing the potential for core fuel damage.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

India

Pressure injection compounds to seal leaks has occasionally been used in the feedwater system (BWR). Halogen free components are used.

Japan

The use of filling compound for repair of reactor coolant pressure boundary component (Class 1) is not allowed in Japan. The use of compound is allowed in auxiliary boiler and steam turbine auxiliary systems but the conditions, design, repair work and inspection, etc. are strictly limited by Thermal and Nuclear Power Generation Technical Association Guide (TNS-G2808).

USA

The NRC issued an information notice to alert licensees to an application of an on-line leak sealing process which substantially degraded the integrity of the reactor coolant pressure boundary and has worked with the industry on the proper application of on-line leak sealants.

ADDITIONAL SOURCES:

- USNRC Inspection Manual Part 9900, Assessing on-line leak sealing of ASME Code class 1 & 2, October 19, 1994.
- On-Line leak sealing: A guide for nuclear power plant maintenance personnel, July 1989, Electric Power Research Institute
- Recent USNRC Information Notices 97-73, 97-74.
- USNRC Information Notice 93-90, Unisolatable reactor coolant system leak following repeated applications of leak sealant, December 1, 1993

COMMENTS:

The use of such a technique is quite common in NPPs, and incidents involving such practices have been reported internationally.

ISSUE TITLE: Inadequate testing of Engineered Safety Features (ESF) actuation systems (lack of logic overlap) (SM 5)

ISSUE CLARIFICATION:

Description of issue

Tests are the major, and sometimes the only, means to verify the compliance of the "as-built" systems and components with the design before operation of the plant. They are the means for periodically checking the ability of the systems and components to perform their function.

ESF are never actuated during the normal life of the plant but they have to correctly perform their function in case of an accident. The difficulty is to test these systems in a representative configuration. The tests performed during commissioning (or requalification after modification) have to be sufficiently complete to take into account the physical parameters obtained during accidents, the interaction between systems operating simultaneously and the non-operability of some redundant components (power, I&C, mechanical). For periodic tests, which are more selective (limited range) the main concern is to be consistent with the start up tests.

Safety significance

- Testing must not lead to spurious actuation of safety systems.
- Tests have to insure that all the configurations during an accident are checked.
- Tests have to guarantee that performances are maintained throughout the plant life.

Source of issue

- __________ event
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Japan

The safety related components are to be designed as capable of being tested or inspected to verify the integrity and capability by adequate methods consistent with the importance of their safety functions during reactor operation or shutdown.

The conditions which are not able to be tested during operation are tested during the pre-operational test. Under the safety rule by the individual utilities, maintenance standards of ESF, RPS and Emergency Cooling System have been established. In case that any non-conformity to the standard is recognized by a surveillance, the plant will be forced to shut down depending on its significance.

USA

The USNRC, because of concerns that surveillance tests did not adequately show that the complete logic train was being tested, issued Generic Letter 96-01. This GL requested the following actions from the nuclear power plant licensees:

(1) Compare electrical schematic drawings and logic diagrams for the reactor protection system, EDG load shedding and sequencing, and actuation logic for the engineered safety features systems against plant surveillance test procedures to ensure that all portions of the logic circuitry, including the parallel logic, interlocks, bypasses and inhibit circuits, are adequately covered in the surveillance procedures to fulfil the Technical Specification (TS) requirements. This review

should also include relay contacts, control switches, and other relevant electrical components within these systems, utilized in the logic circuits performing a safety function.

(2) Modify the surveillance procedures as necessary for complete testing to comply with the TS. Additionally, the licensee may request an amendment to the TS if relief from certain testing requirements can be justified.

ADDITIONAL SOURCES:

- USNRC Generic Letter 96-01, "Testing of safety related logic circuits," dated January 10, 1996.
- USNRC IN 95-15, "Inadequate logic testing of safety related circuits," dated March 7, 1995.
- USNRC IN 93-38, "Inadequate testing of engineered safety features actuation systems," dated May 24, 1993.
- USNRC IN 93-15, "Failure to verify the continuity of shunt trip attachment contacts in manual safety injection and reactor trip switches," dated February 18, 1993.
- USNRC IN 92-40, "Inadequate testing of emergency bus undervoltage logic circuitry," dated May 27, 1992.
- USNRC IN 91-13, "Inadequate testing of emergency diesel generators (EDGs)," dated March 4, 1991,
- USNRC IN 88-83, "Inadequate testing of relay contacts in safety related logic circuits," dated October 19, 1988.

ISSUE TITLE: Foreign material policy (SM 6)

ISSUE CLARIFICATION:

Description of issue

Abnormal situations have been discovered in the past in NPPs, some leading to reported incidents, due to the loss of foreign materials or temporary devices in systems or components after start up of the plant. There are two kinds of unsuitable devices:

- Those that are not intentionally installed in the circuits and are not in response to functional needs, such as tape, helmets and welding rods. This is the result of inadequate cleaning procedures and quality assurance programmes by the persons involved on the site.
- Devices used during commissioning or periodic tests (flanges, diaphragm, straps and strainers). When left during operation, these can impede correct functioning of the systems. Furthermore, these items are sometimes not suitably labelled and it is difficult to keep track of them.

OSART team missions noted that several plants did not have foreign material exclusion (FME) policy, or it was not reflected in the working procedures or its practice was not well established. In addition, lack of adequate maintenance personnel training in the FME practices was also indicated as a deficiency. As a consequence, foreign material could access the systems and cause chemical and / or mechanical anomalies, challenging the equipment integrity. Some reports have been disseminated in the nuclear industry that shows how serious the consequences of this issue can be (e.g. fuel integrity and Steam Generator tube integrity).

Safety significance

The presence of foreign material in the safety systems and in the reactor coolant system represents a risk for the integrity of the systems, and in particular to the steam generator tubes and the fuel rods integrity. In addition, foreign material left in safety related systems can prevent the system from fulfilling their functions.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

USA

US licensees are expected to have implemented Foreign Materials Exclusion programmes adequate to deal with the concerns identified in this issue. In May 1997, the NRC published a Proposed Generic Letter for public comment entitled "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident because of Construction Deficiencies and Foreign Material in the Containment," which contains current NRC information on this subject.

ADDITIONAL SOURCES:

- USNRC Bulletin 96-03, Potential plugging of emergency core cooling suction strainers by debris in boiling water reactors.
- USNRC Bulletin 95-02, Unexpected clogging of a residual heat removal (RHR) strainer while operating in a suppression pool cooling mode (October 17, 1995).

- USNRC Bulletin 93-02, Debris plugging of emergency core cooling functions strainer.
- USNRC Bulletin 93-02, Suppl. 1, Debris plugging of emergency core cooling function strainer (Feb. 18, 1994).
- Proposed Generic Letter; Potential for degradation of the emergency core cooling system and the containment spray system after a loss-of-coolant accident because of construction deficiencies and foreign material in the containment, Federal Register, Vol. 62, No. 92, Tuesday, May 13, 1997, pp. 26331-40.
- Recent USNRC Information Notices 95-06, 95-47 Revision 1, 96-10, 96-59.

COMMENTS:

Several plants have established clear and comprehensive procedures on FME practices, which include administrative and specific job procedures. Specific training is given to the plant staff, including maintenance workers and contractors. The presence of supervisors and managers in the field reinforcing the FME policy is essential in the process. Very specific instructions and devices (e.g. barriers around refueling pools, covers for equipment left open and in-out controls) are applied during outages, when the large amount of equipment work, refueling activities and presence of contractors represent a major challenge for the FME principles.

ISSUE TITLE: Control of temporary installations (SM 7)

ISSUE CLARIFICATION:

Description of issue

Several plants were noted, during OSART missions, to have poor control of temporary installations. Some of the aspects identified as needing improvement are the following:

- clear definition of what is a temporary installation;
- responsibility for analysing safety aspects of the installation;
- responsibility for the different steps of approval and installation;
- procedure and responsibility for identification and controlling the installation;
- limit for number and period of installation; and
- traceability of documentation.

Safety significance

The lack of control of temporary installation in safety systems can lead to the degradation of the design characteristic of the safety equipment.

Source of issue (check as appropriate)

- ___xx_____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

Has defined an applied standard of cleanliness and management of temporary work.

USA

US licensees are expected to have implemented Foreign Materials Exclusion programmes adequate to deal with the concerns identified in this issue. In May 1997, the NRC published a Proposed Generic Letter for public comment entitled "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident because of Construction Deficiencies and Foreign Material in the Containment," which contains current NRC information on this subject.

ADDITIONAL SOURCES:

- USNRC Bulletin 95-02, Unexpected Clogging of a Residual Heat Removal (RHR) Strainer while Operating in a Suppression Pool Cooling Mode (October 17, 1995).
- USNRC Bulletin 93-02, Debris Plugging of Emergency Core Cooling Functions Strainer.
- USNRC Bulletin 93-02, Suppl. 1, Debris Plugging of Emergency Core Cooling Function Strainer (Feb. 18, 1994).
- Proposed Generic Letter; Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident because of Construction Deficiencies and Foreign Material in the Containment, Federal Register, Vol. 62, No. 92, Tuesday, May 13, 1997, pp. 26331-40.

COMMENTS:

Several plants have established clear and comprehensive procedures on temporary installations. The operation shift supervisor is normally responsible for the final approval of the installation, and has in the control room an updated status of the temporary installations in the plant. The number of temporary installations is low, and they are well identified in the field. Very specific control and instructions are set up during outages, when the number of temporary installations normally increases and there are more contractors in the plant. Several plants have indicators on temporary installations.

ISSUE TITLE: Clear identification of components and system trains (SM 8)

ISSUE CLARIFICATION:

Description of issue

Several plants reviewed by the OSART were identified with deficiencies in identification of components. Either lack of official plant component identification policy/procedure or failure in adherence to such a policy/procedure were identified as the cause of such deficiency. In units with common buildings, including control room, this aspect becomes even more critical.

Safety significance

The lack of clear identification of components and system trains could lead to human errors, particularly during accident situations. This could delay actions related to the integrity of the safety barriers.

Source of issue (check as appropriate)

- ___xx_____ operational experience
- ___xx_____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Japan

The entire plant equipment is to be identified under the Japan Electric Association Guide "Guide for Quality Assurance of Nuclear Power Plants" (JEAG4101).

USA

The USNRC and the nuclear industry in the United States have a continuing major concern with component and system train identification. In April 1986, the USNRC completed a study of 35 wrong unit/wrong train events and published results in NUREG-1192, "An Investigation of the Contributors to Wrong Unit or Wrong Train Events." This study found inadequate labeling of plant equipment, components, and areas was the leading contributor to wrong unit/wrong train errors when both the primary and secondary contribution is considered. Currently, there is no specific USNRC regulation for the labeling of controls, valves, electrical and other equipment which is located, identified or manipulated by plant personnel. US Code of Federal Regulations Title 10 Part 50.34(f)(2) does address the control room design that reflects state-of-the-art human factor principles required by Three Mile Island (TMI) Action Plan Item I.D.1. NUREG-0700, Rev.1, Draft, "Human System Interface Design Review Guidelines," issued in February 1995, provides information which includes guidance regarding component and system identification. In 1996, the USNRC completed its final Detailed Control Room Design Review. However, US plants continue to experience and identify labeling problems.

ADDITIONAL SOURCES:

- Code of Federal Regulations Title 10 Part 50.34(f)(2).
- NUREG-0700, Rev. 1, Draft, "Human system interface design review guidelines."
- NUREG-1192, "An investigation of the contributors to wrong unit or wrong train events."

COMMENTS:

During OSART missions, several plants had a clear plant components identification policy/procedure. In addition, well developed systems for reporting deficiencies in components identification were identified, which included the immediate assignment of an authorized temporary identification (tag).

In Beznau NPP, the operating group has the responsibility for the control of labelling and identifications of equipment. This responsibility is shared between the shift crews. Every crew is consequently responsible for a given number of systems. This results in an excellent status of labelling and identification of equipment throughout the plant.

ISSUE TITLE: Response to low level equipment defects (plant material condition) (SM 9)

ISSUE CLARIFICATION:

Description of issue

OSART missions have identified several plants where a large number of lower level defects in the plant are not identified and repaired. These plants were noted as having high threshold for reporting deficiencies. In addition, near misses reporting policy was not in place. This indicates that plant senior management does not have or is not communicating the expectation of a lower threshold for reporting deficiencies. In these plants, the presence of managers and supervisors in the work place was generally indicated as being an area needing improvement. As a consequence, these plants also have problems with housekeeping, including industrial safety hazards.

Safety significance

The cumulative effect of these defects could impact plant safety. It could lead to the degradation and consequent unavailability of safety equipment.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

India

Training of operators to follow-up low level defects is carried out.

COMMENTS:

Several plants have developed good policy and processes for identifying and correcting low level defects. This includes clear procedures, well-defined job description and responsibilities, and training on correlated subjects. The management commitment with such policy is well communicated through the establishment of indicators, including distribution and discussion of results and trending, and presence of supervisors and management in the field, reinforcing their expectations. Some plants have established a deficiency tagging system, which helps the field operators and plant staff in general to identify immediately, in the field, if a deficiency has been reported or not. This tagging system also helps supervisors and managers to verify, during their tours, the level of deficiencies reported and not repaired in the plant.

4.2.4. Training (TR)

ISSUE TITLE: Adequacy of fire brigade training (TR 1)

ISSUE CLARIFICATION:

Description of issue

Each operating organization of a nuclear power plant should have its own programme to train the immediate fire fighting team, the plant fire brigade, and the outside assistance from local authority to give a balanced approach between fire fighting operations and maintaining nuclear power plant safety. Leaders of the immediate action fire fighting or plant fire brigade should be not only fully conversant with plant safety features but also trained in fire fighting techniques so that a balanced approach in directing fire fighting operations and maintaining nuclear plant safety can be achieved. Members of the immediate fire fighting team and the plant fire brigade should be given regular practice and training in fire fighting.

Fire fighting training provided to fire brigade members does not always provide training under actual fire fighting conditions. In some cases no training is provided fighting large, full scale fires either due to inadequate facilities or prohibition of large fires due to state regulations. Without providing training under actual conditions, fire brigade members do not have the opportunity to function as a team to identify and correct individual performance deficiencies. This lack of real time training under actual conditions is also important to qualify an individual's mental and physical ability to function, with the use of a respirator, without the fear of sudden incapacitation.

Safety significance

Fire brigade members must be trained and qualified to respond to large fires that might occur in the plant in order to protect the reliability of safety related equipment under accident conditions.

Source of issue (check as appropriate)
- ___xx_____ operational experience
- ___xx_____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

Fire fighting training is organized on each NPP site: 3 levels of training are practiced:

1. First level

It is mandatory, for anyone working in a NPP site, to follow that first level training which consists in:
- basic notions on fire causes and fire development;
- notions of risk prevention;
- exercises on fire extinguishers.

2. Second level

The second level is mandatory for people working in technical areas (all areas except offices).

Training consists in:
- theory of sectorization;
- exercises with fire extinguishers and fire hydrants;
- *exercises using self contained breathing apparatus;*
- presentation of specific fire aspects of controlled areas;
- exercises in a confined area.

3. Third level
Main objective of this training is to learn fire fighting for members of intervention teams. Training is practicized in situations representative of ones which could occur in a NPP. That training is given in a specialized national centre and consists in:
- organization of interventions;
- learning the art of fire fighting.

Recycling of these trainings is foreseen with the following frequencies: first level every year, second level every 2 years, third level every 3 years.

For example: 12 exercises of fire fighting have been performed at Saint-Laurent-des-eaux NPP centre in 1995.

Germany

For each German NPP a professional fire brigade on the plant site is required. The members of the plant fire brigade as well as the members of the intervention team are well trained on a periodic basis. These training courses are carried out on the plant site and in special facilities for fire fighting experience. The scheduled fire drills take place regularly.

Japan

By means of training each fire fighting team is exposed to their responsibilities as determined in the fire fighting plan and where possible to take the necessary measures for the lack of training choices under real conditions.

ADDITIONAL SOURCE:

- Measures for fire protection at Neckar Nuclear Power Plant (Wiechers, R., in Fire & Safety '94, Nuclear Engineering International, ISBN 0617005583, December 1994).

COMMENTS:

OSART findings include several issues on the lack of real time fire fighting training and the lack of necessary fire fighting protective equipment for personnel.

ISSUE TITLE: Assessment of full scope simulator use (TR 2)

ISSUE CLARIFICATION:

Description of issue

The importance of simulation in the training of NPP personnel cannot be overemphasized. The most common example of a full-scope simulator (those consistent with IAEA-TECDOC-685 "Simulators for Training Nuclear Power Plant Personnel"), is the full-scope control room simulator which usually allows for the simulation of a full range of operations that can be performed from the main control room. They are plant referenced and replicate as many systems as possible, including communications as well as the actual control room environment. Simulator training for control room operators includes exercises related to normal, abnormal and emergency plant operating conditions.

For the full-scope simulator to be considered as an appropriate training tool, an assessment of the minimum configuration and performance of such a unit (i.e, simulator capabilities, environment and design control) should be done. In addition, the simulator functional and physical fidelity to the reference plant and the configuration problems adequately compensated for should be verified.

Safety significance

The main concern related to this issue is the possible negative impact on training control room personnel by using an inappropriate training tool which could result in loss of safety functions in an accident scenario.

Source of issue (check as appropriate)

- __________ operational experience
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

EdF has used simulators for years:

The first generation began in the seventies, to simulate some crucial functions as chemical and volume control system, reactor control and turbine generator control, they have been installed and operated at Bugey, Paluel and Caen.

Full scale simulators began at Bugey in 1977 to simulate normal reactor operation and postulated initiating events (primary coolant in a monophasic state), break < 2 inches. 7 other simulators were installed and used to simulate some hundreds of potential breakdowns.

To prepare operators for the new generation of computerized control room of the N4 subseries (PWRs, 1400 MW), a new simulator has been installed at Bugey in 1987. The simulator allows, in addition, to optimize the design, ergonomy and man-machine interfaces and has helped designers in anticipating the resolution of human factor issues.

At the end of the eighties, a new generation of simulators appeared to simulate some predictable types of accidents: SGTR (Steam Generator Tube Rupture), then multi-ruptures, cumulating various situations: normal transients (for example house load operation, in French "ilotage").

From 1991 a new simulator, called SIPA (Simulateur Post-Accidentel), is operating at Lyon (SEPTEN). That advanced simulator allows a real time analysis of safety in case of nuclear crisis: safety experts, control room operators, plant managers, can be exercised to react to different crisis situations, and can prepare training and emergency procedures. SIPA allows to study different types of issues in particular:

- visualization of thermo-hydraulic evolving of the safety related circuits during a LOCA;
- assessment of the main contributors of risk in PSA studies;
- validation of operational procedures with the APE approach;
- sensitivity of parameters on component dimensioning.

SIPA has been designed in the frame of CEA and EdF co-operation.

Furthermore, in 1992, EdF (operating branch) has decided:

- to reinforce training programmes on simulators, increasing from one to 2 weeks the annual time of operator training;
- to install 4 new simulators full-scale at Gravelines (subseries CPY: 900 MW), Cattenom (subseries 1300 MW), Fessenheim (first subseries 900 MW), and Chooz (subseries N4: 1400 MW);
- to improve and modernize current simulators.

All that programme should be achieved in 2000.

Spain

In Spain, simulator training is mainly done by a training company (TECNATOM), using both a PWR full scope and a BWR full-scope simulator.

Because of the wide variety of designs in Spain (PWR-Westinghouse 3 loops, PWR-1 Loop Westinghouse, BWR-GE Mark I, BWR-GE Mark III and PWR-KWU 3 Loops), the simulators currently used do not replicate each plant. The full-scope BWR simulator is a replica of one BWR plant. The full-scope PWR simulator reproduces mainly the dynamic capability of a PWR plant.

Additionally, an "Interactive Graphic Simulator" (SGI) , which uses several computer screens instead of control panels is extensively used.

The Spanish regulatory body (CSN) required the utilities to perform an assessment of the requirements of the full scope simulators. The utilities presented the assessment and the CSN finally has required for four units to have full scope simulators, for three units to perform a new analysis for the adequacy of their training simulators, and for two old plants to develop an integrated graphic simulators (SG1) and to have access to an adequate simulator.

USA

In the US, all large operating nuclear power plants have a full scope simulator which is plant specific. Configuration management processes in the plants include appropriate changes to the simulator following plant modifications to assure simulator fidelity. Simulators are extensively used for operator initial and refresher training and operator examinations and are also used to develop and verify new operating procedures. Use of simulators is also encouraged during emergency exercises.

The USNRC also trains inspectors and other staff on simulators of each US nuclear system supplier at its training centre in Chattanooga, Tennessee. The objective of this training is familiarity with the operating characteristics of various US designs, reinforcement of technical training concepts and understanding of control panel configurations which indicate abnormal conditions.

ADDITIONAL SOURCES:

- CSN Safety Guide GS-1.1.
- Title 10, Code of Federal Regulations, Part 55.45(b)(1) "Operators Licenses."
- Title 10, Code of Federal Regulations, Part 50, Appendix E, IV.F.2.
- Title 10, Code of Federal Regulations, Part 50.47(b)(14).
- American National Standard "Nuclear power plant simulators for use in operator training" ANS/A.NS-3.5 1985.

COMMENTS:

OSART missions have identified the following points:

- Numerous issues on configuration control problems resulting in several cases of negative training.
- Several issues of no plant specific simulator training available, only generic training on a standard referenced nuclear power plant simulator, combined in some cases with a plant specific compact simulator.
- Several issues of no simulator training provided before going on watch, after extended time off watch (3 to 6 months).

ISSUE TITLE: Training for severe (beyond design) accident management procedures (TR 3)

ISSUE CLARIFICATION:

Description of issue

Severe (beyond design) accident management procedures and symptom based procedures are not yet included in all NPPs in the regular set of the operating procedures. All new and changed procedures are treated as modifications and should be incorporated in training programmes and plans. Periodical refreshment of emergency operating procedures and other but seldom needed procedures should be on the simulator programmes. Training for emergencies should also include the use of those procedures. (see also SS 8, Accident management measures", MA 10 "Adequacy of emergency operating procedures and AA4, Need for severe accident analysis).

Safety significance

Severe accident procedures are generally not used, but in case of necessity their use is significant for safety in the plant and off site. Inadequate implementation of accident management procedures due to lack of training can increase the potential for radioactive releases. These procedures are important to prevent the progression of accidents as well as to mitigate the effects of any releases.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

As described in item TR 2, a specific simulator SIPA has been designed for operator training facing to all types of situations in case of severe accident.

In addition, different types of exercises are organized to clarify, improve and test the management system implemented by the DSIN, the nuclear operators and the other ministerial departments involved.

DSIN alerting system

The DSIN regularly carries out tests to check the effectiveness of its system for alerting its own staff and that of the DRIREs. This system is also activated during the exercises described below.

Different exercises performed indicate that more than 50% of the total number of staff are contacted within ten minutes of giving the alert and that the DSIN Headquarters in the emergency centre of the ministry in charge of industry in Paris can be operating within one hour.

Technical nuclear safety exercises

Exercises are performed to test the emergency management system implemented to ensure technical safety in liaison with an operator. They are based on the planning of a technical scenario which is as detailed as possible and can extend to severe accident situations.

These involve:

- a nuclear operator (EDF, Cogema, CEA, etc.) with its local and national branches;
- the DSIN and its technical support, the IPSN;
- the DRIRE for the area involved;
- if possible, simulation of the prefecture involved and the interministerial information unit of the ministry in charge of industry.

Such exercises performed on pressurized water reactor nuclear power plants with EDF have confirmed the importance of the national emergency teams having access to the information supplied by the nuclear reactor safety panels. In such exercises, information is supplied by the EDF training centre operating simulators using the actual actions of an operating team confronted with a simulated accident situation.

Interministerial exercises

The DSIN also participates in interministerial exercises organized by the general secretariat of the interministerial committee for nuclear safety or by the prefects, to test co-ordination of the governmental authorities in the event of an incident or accident in a nuclear installation.

4.2.5. Emergency preparedness (incl. physical protection) (EP)

ISSUE TITLE: Need for effective off-site communications during events (EP 1)

ISSUE CLARIFICATION:

Description of issue

A reliable communication system is definitely needed between a nuclear power plant and the relevant offices, including the regulatory body located outside the plant site during and after an accident. Special consideration has to be given in maintaining the system function for internal and external events such as fire and earthquake. The preparation of appropriate procedures and the training of the plant staff for communication in the emergency is necessary.

As a result of the TMI-2 accident on March 28, 1979, the NRC and others recognized a need to substantially improve the ability to acquire data on plant conditions during emergency. Typically, the regulatory body role in the event of an emergency is one of monitoring the licensee to ensure that the appropriate recommendations are made with respect of off-site protective actions. Other aspects of this role include supporting off-site authorities and keeping other agencies and entities informed of the status of the incident. To fulfil this role, accurate timely data is needed, including:

- core and coolant system conditions;
- conditions inside the containment building;
- radioactivity release rates;
- the data from the meteorological conditions at the plant.

Safety significance

The loss of off-site communication during and after an accident can prevent a systematic approach to technical assistance in an emergency from relevant organizations outside the plant and appropriate response actions by public authorities. It also impairs communication flow from a plant to the public and sometimes distorts information on progression and consequences of an accident.

Experience with voice and emergency communications links has demonstrated that excessive amounts of time are needed for the routine transmission of data and that verification or correction of data appear questionable. Incorrect data may cause a response to offsite officials with inaccurate or outdated advice that results in inappropriate actions.

Source of issue (check as appropriate)
- ___xx_____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

The setting up and the effectiveness of such a management system depends on rapid mobilization of the teams involved and multiple telecommunication resources.

The alerting system

The alert post of the DSIN (Direction de la Súreté des Installations Nucleaires) emergency centre can rapidly mobilize the DSIN teams and its nuclear departments. This post simultaneously performs two functions:

- calling of staff carrying Eurosignal receivers; the call is remotely triggered by the operators of the nuclear installations using simple procedures;
- a telephone voice-mail system for the operator to record an alert message and to broadcast it to the staff alerted, and then to record their acknowledgement messages.

Telecommunication networks

The transmission of information to the DSIN emergency centre takes place via several separate telecommunication networks to which its equipment is connected:
- public networks: the public switch telephone network, the telex network;
- private networks: specialized links (ex: EdF safety network, Transpac).

Some networks allow only the transmission of written messages (telex), others oral and/or written communication. The Transpac package switch network gives access to information originating directly from the control room of the reactor where the accident has occurred, making it possible for instance to obtain data pictures of the safety panel.

The special links operating are all permanent point-to-point links.

By mid 1990, a new telecommunication network was set up for multiconference use. This closed network interconnects the national emergency centres and the main EdF, CEA and COGEMA nuclear sites.

Japan

When an accident has occurred at a nuclear power station, it is notified to the national and local governments through telephone and/or facsimile. Main plant data are available at the site emergency centre through an emergency on line system for timely transmission. In addition, having learnt from the Hanshin-Awaji Disaster, studies are being done to secure communication using sophisticated technologies.

A technical officer from the regulatory authority is stationed at each power station to provide oversight of the management of operating conditions and to give some guidance on direct finding of conditions and investigation of causes.

In addition, an accident response manual is provided, and training on emergency communication conducted periodically.

Spain

In Spain, one automatic transmission data system covering a set of parameters consistent with those mentioned in the description of issue is required for all nuclear power plants. This system is in operation in the Emergency Center of the Regulatory Body. The number of parameters to be sent and updated each 30 seconds is variable (between 30 or 50) depending on each individual plant.

USA

The USNRC headquarters Operations Centre has developed a system which can receive and display plant post-accident information from power plant computers. Each commercial nuclear power plant is required to have an Emergency Response Data System (ERDS) which is a direct electronic transmission of selected plant parameters from the licensee's onsite computer system to the NRC Operations Center in the event of an emergency. Other system users include NRC regional offices and States within the plume exposure pathway emergency planning zone upon the request of the States. The number of parameters for each site is variable but must include data points from four types of plant conditions: reactor core and coolant system conditions; reactor containment conditions; radioactivity release rates;

and plant meteorological tower data. In addition, the system must be capable of transmitting all available ERDS parameters at intervals of not less than 15 seconds or more than 60 seconds. The transmission capability is routinely tested. Transmission during an accident must be initiated by the power plant operator.
This capability has provided efficient transmission of data during certain emergency exercises. This information is also sent to the near site Emergency Operations Facilities which are jointly staffed by utility. NRC and State and local government personnel during a significant event.

The NRC published lessons learned from the effects of Hurricane Andrew on the Turkey Point nuclear power plant during August, 1992. The NRC also performed inspections of off-site communication systems at several plants to determine their diversity and vulnerability to severe natural events.

The NRC incident response organization acquired several portable telephone units which use satellite relay of the transmissions. These units are sent to plants which are threatened or have just experienced a natural event such as a hurricane or tornado. The portable telephones can be used to assist in the transmission of emergency information to off-site authorities by the nuclear plant.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, 50-C-D (Rev.1) IAEA, Vienna, (1988).
- INTERNATIONAL ATOMIC ENERGY AGENCY, 50-SG-06, Vienna, (1982).
- IAEA Safety Series No. 50-C-D (Rev. 1) #364, #365, #366, IAEA, Vienna, (1988).
- The safety in the Spanish nuclear power plants, Nuclear Safety Council, Spain, May 1992.
- NUREG-1474, "Effect of hurricane Andrew on the Turkey point nuclear generating station from August 20-30, 1992 USNRC and INPO, Washington, DC, USA, 1993.
- NUREG-1394 Rev.l, "Emergency response data system (ERDS) implementation."
- NUREG-0654.
- USNRC Generic Letter No-89-15, "Emergency response data system."
- USNRC Information Notice 97-05, Offsite notification capabilities.
- USNRC Information Notice 93-53, Effect of hurricane Andrew on Turkey point nuclear generating station and lessons learned, USNRC, Rockville, MD, USA, 1993.

ISSUE TITLE: Contingency planning for physical security (EP 2)

ISSUE CLARIFICATION:

Description of issue

Physical protection of nuclear facilities from radiological sabotage involves a number of elements, such as physical barriers, intrusion detection, control of access, response capabilities, and contingency plans. A safeguards contingency plan is a documented plan to give guidance to licensees in order to accomplish specific objectives in the event of threats, or events such as radiological sabotage relating to nuclear power reactors. The goals are to identify reasonably possible threats and events, plan for the necessary resources, and organize the response effort at the licensee level, and ensure the licensee's response is integrated with the response of other entities such as local law enforcement, fire and radiological emergency response, State and Federal agencies, etc. Each plan identifies and defines the perceived dangers and incidents that it covers and the general way in which they should be handled.

Safety significance

Acts of radiological sabotage could result in the release of radiation and endanger public health and safety.

Source of issue (check as appropriate)

- __________ operational experience
- __________ deviation from current standards and practices
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

USA

The NRC issued regulations in 1978 for physical protection of plants in which special nuclear material is used. Among other requirements, each nuclear power reactor, as a condition of its license, is required to have a safeguards contingency plan. A Regulatory Guide (RG 5.54) was issued to provide guidance on the format and contents of such a plan.

The NRC, in co-ordination with other government agencies, monitors the threat environment world-wide, and, when conditions so warrant, advises licensees of a change in threat level.

In 1989, a Generic Letter was issued to power reactor licensees requiring inclusion of a land vehicle bomb as a threat to be considered for safeguards contingency planning. In particular, upon notification by the NRC of a change in threat level, licensees were to be prepared to implement measures to protect against unauthorized vehicle access. In August 1994, the NRC modified its design basis threat characterized in 10CFR 73.1 to add a four-wheel drive land vehicle used to transport personnel and their hand-carried equipment to the proximity of vital areas, or as a bomb. When sure to cope with the malevolent use of vehicles, the previous 1989 request to implement contingency measures is superseded.

The NRC issued a rule, 10CFR Part 73, Physical Protection of Plants and Materials, in 1978. Among other requirements for power reactors, the rule specifies in 73.55(d)(1) that "identification and search of all individuals...must be made... for detection of firearms, explosives, and incendiary devices...through the use of both firearms and explosive detection equipment capable of detecting those devices." The rule further requires that packages and vehicles be so searched prior to admittance to the protected area.

ADDITIONAL SOURCES:

- 10CFR73 Appendix C, Licensee safeguards contingency plans (originally issued March 23, 1978).
- Part 73 of Title 10CFR (Code of Federal Regulations), Physical protection of plants and materials.
- USNRC Regulatory Guide 5.54, Standard format and content of safeguards contingency plan (for Power Reactors) March 1978.
- USNRC Regulatory Guide 5.7, Revision 1, May 1980, Entry/exit control for protected areas, Vital areas, and material access areas.
- NUREG/CR-6190, Protection against malevolent use of vehicles at nuclear power plants, Volumes 1 and 2, November 1994.
- NUREG-0416 Security plan evaluation report workbook (April 1978), containing Review Guidelines #20," Searching for explosives."
- USNRC Generic Letter 89-07, Power reactor safeguards contingency planning for surface vehicle bombs April 28, 1989 and Supplement 1 August 21, 1989.

ISSUE TITLE: Need for technical support centre (EP 3)

ISSUE CLARIFICATION:

Description of issue

Recent international practice has been to design NPPs with a room where current plant data and status are compiled for display to enable technical experts to support the operators during the management of an event or accident. This room is separate from the control room. Technical support centre is a part of the emergency response facilities. Most WWER plants do not have technical support centres.

It is to be evaluated whether centres

- are suitably located;
- are habitable under emergency conditions;
- are appropriately organized for carrying out the functions of the staff assigned to them;
- have appropriate communications systems (including backups) to all required points as identified in the emergency plans;
- have regularly updated copies of *all* emergency plans, procedures and engineering material (such as plant layouts, schematics and safety system drawings);
- have systems and procedures for accident consequence assessment and methodology, including those in the final safety analysis report;
- are adequately staffed with trained personnel;
- have appropriate emergency equipment for their own protection;
- have appropriate capability for data handling and processing;
- have display facilities to aid decision making (maps, charts, status boards, safety parameters display system, etc.);
- have record keeping methods and support material.

Safety significance

The lack of a technical support centre can affect the control of abnormal operation and management of accidents. The safety functions can be impaired or even questioned for scenarios within the DB envelope and beyond.

Source of issue (check as appropriate)

- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Bulgaria, Czech Republic, Russian Federation, Ukraine

The establishment of a technical support centre with displays of critical plant information and up to date plant design documentation is proposed in all countries.

France

The Local Technical Centre (LTC) evaluates the situation of the incident and the possible development that might be foreseen. The LTC is located near the control room. This centre has ventilation systems protected by aerosol and iodine filters. It is also equipped with emergency power supply and communication means are available for all communication channels necessary with redundant backup. There is an emergency communication system based on the INMARSAT satellite communication system in case of earthquakes. The LTC is equipped with a Safety Parameter Display System (SPDS) and record keeping functions.

The LTC is provided with necessary documentation, protective equipment and adequate staff. Operations personnel are always included.

India

The existing emergency response plans require technical support centre (Emergency Control Centre) to be established with expertise technically qualified locally available at the station. No special on-line display is provided.

USA

Commercial nuclear power plants in the United States are required to have a Technical Support Center (TSC). In the event of an accident, the TSC provides plant management and technical support to plant operations personnel, and relieves the reactor operators of duties and communications not directly related to reactor operations. The TSC also performs the functions of the Emergency Operations Facility (EOF, another required emergency response facility in the US) until the EOF is functional. The TSC is staffed by sufficient technical, engineering, and senior designated licensee officials to provide needed support, and is expected to be fully operational within about one hour following the declaration of an emergency at the Alert or higher level classification.

ADDITIONAL SOURCES:

- NUREG-0737, Supplement No. 1, "Classification of TMI action plan requirements (Requirements for emergency response capability)."
- NUREG-0696, "Functional criteria for emergency response facilities."

4.2.6. Radiation protection (RP)

ISSUE TITLE: Hot particle exposures (RP 1)

ISSUE CLARIFICATION:

Description of issue

Hot particles consist of tiny fragments of materials activated in the core of a reactor or tiny fuel fragments from leaking fuel which may not be visible to the naked eye. Licensees are expending significant resources to control these particles and to prevent and mitigate worker exposures from such particles. Most Member State regulations include a limit on radiation dose to the skin; however, the limit is based on the prevention of radiation effects associated with radiation exposure to relatively large areas of the skin.

Safety significance

Hot particles doses can result in visible skin disturbances such as erythema, blistering, and ulceration. It is generally recognized that although hot particles can produce very large doses to small amounts of tissue, these doses do not pose the same level of risk as similar doses to relatively large areas of the skin.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Japan

The following measures have been taken to minimize workers exposure to radioactive particles.

(1) Production of leak tight fuel.
(2) Monitoring of fuel leakage by coolant iodine monitoring.
(3) Installation of coolant purifying system facilities (filters, resin).
(4) Continuous monitoring of density of radioactive particles in work areas.

Work areas are to be ventilated, and protectors are to be worn whenever necessary to do so.

USA

An enforcement policy recognizing this difference in risk for hot particle exposure situations was published in the Federal Register July 31, 1990, and as an enclosure to Information Notice 90-48. However, a rule is needed to codify the results of health effects research in this area and to establish a firm basis for licensees' radiation protection programmes. The NRC is establishing a technical basis for the rule making through research being performed by a National Laboratory and the NCRP.

ADDITIONAL SOURCES:

- Title 10, Code of Federal Regulations, Part 20.
- US Nuclear Regulatory Commission. "VARSKIN MOD2 and SADDE MOD2: Computer codes for assessing skin dose from skin contamination." NUREG/CR-5873; 1992.
- US Nuclear Regulatory Commission. "Dose calculation for contamination of the skin using the computer code VARSKIN." NUREG/CR-4418; 1987.

- US Nuclear Regulatory Commission. "Progress report on hot particle studies." NUREG/CR-5725; 1992.
- National Council on Radiation Protection and Measurements. "Limit for exposure to 'hot particles' on the skin." NCRP Report No. 106; 1989.
- The International Commission on Radiological Protection. "The biological basis for dose limitation in the skin." ICRP Publication 59; 1991.
- The International Commission on Radiological Protection. "1990 recommendations of the International Commission on Radiological Protection." ICRP Publication 60; 1990.
- US Nuclear Regulatory Commission. Information Notices 86-23 ("Excessive skin exposures due to contamination with hot particles"), 87-39 ("Control of hot particle contamination at nuclear power plants"), and 90-48 ("Enforcement policy for hot particle exposures").

COMMENTS:

The control of hot particles or hot particle irradiation has not been identified as a problem during any OSART mission. However, it has been a recognized problem at several plants, not only because hot particles are often difficult to detect, but because regulatory penalties (enforcement actions) for plants experiencing hot particle personnel contaminations were widely believed to be inconsistent with the actual or potential health effect. More information on the health effects of hot particle contamination could be useful in determining the most appropriate controls for hot particles and hot particle contaminations.

ISSUE TITLE: Radiation beams from power reactor biological shields (RP 2)

ISSUE CLARIFICATION:

Description of issue

On July 8 and 9, 1992, with the unit at no greater than 10% power, work crews at Limerick Unit 1 were performing troubleshooting on a main steamline sample flow isolation valve inside the drywell. The crews were unaware that a narrow, intense beam of radiation passed from a reactor water level instrumentation penetration to the inner drywell wall near the work area. The beam measured ~6" at the penetration to 1'-2' at the drywell wall. The beam was discovered when the dosimeter of one worker alarmed.

Safety significance

Narrow, intense beams of radiation can stream into accessible areas through penetrations in the biological shield, potentially causing personnel exposures above regulatory limits and exposing environmentally-qualified (EQ) equipment to high levels of radiation.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

In French PWRs access in the containment is forbidden during operation.

Japan

Biological shields are designed and constructed upon step by step confirmation, and are inspected to confirm their performance during initial power operation.

Radiation doses are periodically monitored (continuously monitored by area monitors for special areas) to confirm that there are no abnormalities.

Following the beginning of the work, an environmental survey is conducted to confirm radiation dose.

USA

The NRC issued an information notice to alert addressees to narrow, intense beams of radiation that can stream into accessible areas of a drywell through penetrations in the biological shield of a boiling-water reactor (BWR), potentially causing personnel exposures above regulatory limits and exposing environmentally qualified (EQ) equipment located in a drywell to high levels of radiation.

ADDITIONAL SOURCES:

- ORNL/RSIC-43 (ANS/SD-79/16), "Radiation streaming in power reactors."
- USNRC Information Notice 93-39, "Radiation beams from power reactor biological shields," issued May 25, 1993.

ISSUE TITLE: Measures implemented to comply with international recommendations (ICRP-60), on dose limits (RP 3)

ISSUE CLARIFICATION:

Description of issue

At the beginning of the NPPs' operations and for about 10 years annual collective dose in French NPPs were generally considered as satisfactory with annual collective doses under 2 man. Sv/unit but with the first 10 yearly controls, replacement of SGs and reactor vessel heads, that situation began to deteriorate. Furthermore, more restrictive legislation and international recommendations will even make more challenging the compliance with individual and collective radiation doses.

Safety significance

If appropriate measures to reduce doses are not implemented well in advance, the new limits established by regulations or international organizations (ICRP) will be difficult or impossible to meet.

Source of issue (check as appropriate)
- ____xx____ operational experience
- ____xx____ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

A voluntary ALARA approach has been decided to improve radioprotection in operation, main actions undertaken are:

(1) Associate subcontractors to the whole ALARA approach

During shutdown, many services are performed by subcontractors (about 80% of shutdown works) consequently they are fully implied with ALARA (definition of common objectives, training).

(2) Improving of dose recording using new analysis and tools
- refining of dose analysis for each intervention;
- developing an ambient gamma cartography;
- establishing reference radioexposure data for each type of work, particularly for main works of maintenance (ex: SG replacement);
- using of common tools for dose recording (ex: subcontractors can intervene on behalf of different operators: EdF, CEA, COGEMA, etc.). EdF uses the DOSINAT programme for the occupational exposure on its nuclear sites, a new programme DOSIMO, compatible with DOSINAT (data of DOSINAT are poured out in DOSIMO) is currently experimented to take in account exposures on the whole of French nuclear sites.

(3) Optimization of works with prioritization for those where exposures are important

Example: SG replacement, opening and closing of head vessel, valve maintenance, non-destructive controls, decontamination of components, etc.

(4) Making workers and staff responsible

Any NPP commits itself not to exceed, annual limits of dosimetry, depending on predictable shutdowns or 10 yearly controls, maintenance operations, etc.

For the whole of French NPPs the mean annual collective dose per unit has regularly decreased from 2.44 man. Sievert in 1991 to 1.59 man. Sievert in 1996. The objective for the next years is to continue this decrease to 1.2 man. Sievert in the year 2000.

(5) Reduction of some particularly costly sources in terms of exposure

Example: Case of stellites and cleaning procedures.

(6) Improving the organization and experience feedback
- use of computer applications;
- preparation of works.

Example: A centre to experiment and validate intervention techniques on main components of primary circuit (CETIC: located at Chalon-sur- Saone) is used to prepare and optimize SG replacements. Scale 1 models of main components allow not only to qualify tools and process but also to optimize radioprotection during these interventions.

USA

Outage planning in the United States plants has benefited from assigning a dedicated, full-time outage planner (usually low-to-mid management level professional) as leader of the overall outage. Additionally, each major craft group (mechanical, electrical, radwaste, etc.) will be assigned a senior radiation protection technician (ALARA specialist) months before the outage start to help focus that craft on radiological challenges, and to generate work planning packages that effectively incorporate good ALARA principles into that crafts scheduled work.

ADDITIONAL SOURCES:

- NUREG/CR-5236, SEA 87-253-08-A:1, "radiation related impacts for nuclear plant physical modifications" (Final Report).
- NUREG/CR-4409, Vol. 1-5, "Data base on dose reduction research projects for nuclear power plants."
- NUREG/CR-3469, Vol. 1-8, "Occupational dose reduction at nuclear power plants: Annotated bibliography of selected readings in radiation protection and ALARA."

4.2.7. Fuel storage (FS)

ISSUE TITLE: Degradation of boron plates in fuel storage pool (FS 1)

ISSUE CLARIFICATION:

Description of issue

The potential exists for a gradual release of silica and boron carbide from certain types of boron plates following gamma radiation and long-term exposure to the spent fuel pool environment.

Safety significance

Degradation of the boron plates in fuel storage racks could reduce the subcriticality margin in the spent fuel pool. This is a safety concern since excessive degradation could result in inadvertent criticality in the spent fuel pool.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

France

Degradation of boron did not lead to criticity problems in France, however for future fuel enriched up to 4.5% Safety Authority requires a quincunx disposal of fuel in storage racks. However French boron plates do not use Boraflex (46% silica, 4% polydimethyl siloxane polymer, 50% B4C) like in US but Boral (50% B4C + 50% Al, Boral does not contain silica like Boraflex), these plates are subject to swelling due to an hydrogen production (reaction: H_2O + Al), consequently swelling can stick fuel and impede fuel handling. To remedy to that issue, dimensional controls are performed every 6 months to check cell internal dimensions, when they are not found satisfactory, cells are blocked up. When the number of cells becomes insufficient, a new spent fuel storage rack is installed in the pool (63 cells/storage rack), this is an example of a design issue solved by operational practices, but as it is not very satisfactory, EdF has decided to proceed to neutron flux distribution measurements to check margins of criticity in pure water (5%).

USA

Four NRC information notices (INs) have been issued (listed in the references section) to alert licensees to degradation of boron plates in spent fuel pools. The NRC has also issued a generic letter (GL) on the issue. The actions requested by the GL would assure the maintenance of current regulatory requirements for subcriticality margin in spent fuel pools.

ADDITIONAL SOURCES:

- USNRC Generic Letter 96-04, “Boraflex degradation in spent fuel pool storage racks.”
- USNRC IN 95-38, "Degradation of boraflex neutron absorber in spent fuel storage racks," issued September 8, 1995.
- USNRC IN 93-70, "Degradation of boraflex neutron absorber coupons," issued September 10, 1993.
- USNRC IN 87-43, "Gaps in neutron absorbing material in high density spent fuel storage racks," issued September 8, 1987.
- USNRC IN 83-29, "Fuel binding caused by fuel rack deformation," issued May 6, 1983.

ISSUE TITLE: Potential for fuel pool drainage (FS 2)

ISSUE CLARIFICATION:

Description of issue

Inadequate maintenance practices, RHR misalignment, earthquake, fuel pool stainless steel liner punctured, failure of piping systems, siphoning from permanent and temporary systems, inadequate design features, and refueling cavity water seal failure have resulted or could result in partial drainage of the fuel pool.

Safety significance

- This condition could occur while plant is in operation, startup, or shutdown condition; it is most severe during refueling.
- Potential loss of fuel pool cooling, resulting boiling and airborne activity.
- Potential uncovering and damaging the fuel assemblies resulting in high local radiation level.
- Potential release of radioactive water.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

India

There have been 2 incidents of reactor cavity seal failure resulting in part from draining of the spent fuel pool. Procedure have been made to handle such situation that will leave at least 12 feet of water above the spent fuel.

Japan

The spent fuel storage system are to be designed so that prevention of excessive decrease of cooling water inventory in the storage system and proper leakage detection shall be possible.

Practically, the following measures are taken into considerations:

- Even when water flows out from the spent fuel pit into the path connected to other pit, the top of active fuel is at least below the flooding level.
- The inlet piping into the fuel pit is provided with a check valve.
- Leakage detecting systems and level alarm devices are installed.

USA

The NRC evaluated the potential for and consequences of a refueling cavity water seal failure as indicated in the references. In addition, the following were implemented for permanently shutdown reactors:

- Verified that the structures and systems required for containing, cooling, cleaning, level monitoring and makeup of water in the spent fuel pool (SFP) were operable and adequate, consistent with the licensing basis, to preclude high levels of radionuclides in the pool water and adverse effects on stored fuel, SFP, fuel transfer components, and related equipment.

- Ensured that the systems for essential area heating and ventilation were adequate and appropriately maintained so that potential freezing failures could cause loss of SFP water were precluded.
- Ensured that piping or hoses in or attached to SFP could not serve as siphon or drainage paths in the event of piping or hose degradation or failure or the mispositioning of system valves.
- Ensured that operating procedures address conditions and observations that could indicate changes in SFP level and address appropriate maintenance, calibration and surveillance of available monitoring equipment.

ADDITIONAL SOURCES:

- NUREG-0933 Generic Issue 137, "Refueling Cavity Seal Failure."
- USNRC Inspection Manual Temporary Instruction 2515/066-04, "Inspection requirements for IE Bulletin 84-03," issued December 17, 1984.
- USNRC Bulletin 84-03, "Refueling cavity water seal," issued August 24, 1984.
- IE Bulletin 94-01, Potential fuel pool draindown caused by inadequate maintenance practices at Dresden Unit 1. (Dresden Unit 1).
- IE Information Notice 86-74, Reduction of reactor coolant inventory because of misalignment of RHR valves. (Multiple plants).
- IE Information Notice 84-81, Inadvertent reduction in primary coolant inventory in boiling water reactors during shutdown and startup. (Multiple Plants).
- USNRC Information Notice 94-38, Results of a special NRC inspection at Dresden nuclear power station Unit 1 following a rupture of service water inside containment. (Dresden Unit 1).
- USNRC Information Notice 93-83 Supplement 1, Potential loss of spent fuel pool cooling after a loss-of-coolant-accident or a loss of offsite power.
- USNRC IN 84-93, "Potential loss of water from the refueling cavity," issued December 17, 1984.

ISSUE TITLE: Damage to fuel during handling (FS 3)

ISSUE CLARIFICATION:

Description of issue

Individual fuel assemblies must be moved within a nuclear power plant several times, for placement in new fuel storage racks, loading into the reactor vessel, removal from reactor vessel (to spent fuel pool) after use, as well as other possible moves (reracking, move to dry storage, etc.). Measures are employed (procedures, training, equipment) to minimize damage to fuel that might arise should assemblies fall or come in contact with other objects.

Operating experience shows that mechanical failures of grapples and lifting devices or improper operation of such equipment has led to situations where assemblies were dropped, or struck structures or other assemblies. However, actual damage of the fuel (or release of radioactivity) has been very limited. (see also IH 7, Need for assessment of dropping heavy loads).

Safety significance

If the cladding of irradiated fuel is damaged during fuel handling, fission products may be released and pose a radiological hazard to workers, and, if not contained, to the public.

Source of issue (check as appropriate)

- ____xx____ operational experience
- __________ deviation from current standards and practices
- __________ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Germany

Due to the different tasks the refueling machine is used for and based on the complex movements carried out by the refueling machine, its interlock logic has deficiencies with regard to certain procedures. The operating experience therefore reveals that events during fuel handling occur repeatedly. In most cases, the cause of the event is a combination of operator error and either a complete lack of, or poor reliability of, interlocking device. Because the fuel handling equipment mainly is plant specific also the measures derived from the events are different, e.g. redundancy of interlocking device, improvement of mechanical design quality, change of service instruction.

USA

Fuel handling accidents are part of the licensing review. Monitoring and ventilation systems in the fuel handling areas assist in detection and mitigation of releases should they occur as a result of damage to fuel. The potential for a fuel handling accident is considered as part of the design review of each nuclear power plant. Design features and operational limits are included to limit radiological consequences from a fuel handling accident to well within 10CFR Part 100 dose limits. In response to specific fuel handling events, NRC has issued information notices to alert licensees of circumstances of the events. US vendors have also issued letters with recommended design and procedure improvements as a result of specific problems related to equipment they supplied.

ADDITIONAL SOURCES:

- 10CFR50 Appendix A, General design criteria (GDC) 61 fuel storage and handling and radioactivity control.
- 10CFR Part 100 Reactor site criteria.
- USNRC Regulatory Guide 1.25, Assumptions used for evaluating the potential radiological consequences of a fuel handling accident in the fuel handling and storage facility for boiling and pressurized water reactors, March 1972.
- USNRC Regulatory Guide 1.13, Revision 1, Spent fuel storage facility design basis, December 1975.
- Standard Review Plan (NUREG-0800) Section 9.1.1, New fuel storage; Section 9.1.2 Spent fuel storage; Section 9.4.2 Spent fuel pool area ventilation; Section 15.7.4 Radiological consequences of fuel handling accidents.
- IE Circular 77-12, Dropped fuel assemblies at BWR facilities, September 15, 1977.
- USNRC Information Notice 94-13, Unanticipated and unintended movement of fuel assemblies and other components due to improper operation of refueling equipment, February 22, 1994.
- Supplement 1 to USNRC Information Notice 94-13, June 28, 1994.
- Supplement 2 to USNRC Information Notice 94-13, November 28, 1995.
- USNRC Information Notice 90-77, Inadvertent removal of fuel assemblies from the reactor core, December 12, 1990.
- Supplement 1 to USNRC Information Notice 90-77, February 4, 1991.
- USNRC Information Notice 90-08, Kr-85 hazard from decayed fuel, February 1, 1990.
- USNRC Information Notice 86-58, Dropped fuel assembly, July 11, 1986.
- USNRC Information Notice 85-12, Recent fuel handling events, February 11, 1985 .
- USNRC Information Notice 81-23, Fuel assembly damaged, August 4, 1981.
- USNRC Information Notice 80-01, Fuel handling events, January 4, 1980.

ABBREVIATIONS

ABB	Asea Brown Boveri AG
AC	alternating current
AEOD	Office for Analysis and Evaluation of Operational Data (USNRC)
AF	auxiliary feedwater
AFAS	auxiliary feedwater actuation system
AFW	auxiliary feedwater
AFWS	auxiliary feedwater system
AGNES	Advanced General and New Evaluation of Safety of Paks NPP (Hungary)
ALARA	as low as reasonably achievable
AM	accident management
AOT	allowed outage times
ASCOT	Assessment of Safety Culture in Operation Teams (IAEA)
ASME	American Society of Mechanical Engineers
ASSET	Assessment of Safety Significant Events Team (IAEA)
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
B&WOG	Babcox & Wilcox Owner's Group
BDBA	beyond design basis accident
BIT	boron injection tank
BNCS	ASME Board of Nuclear Codes and Standards
BRU-A	steam dump valve to the atmosphere (WWER)
BRU-K	steam dump valve to turbine condenser (WWER)
BRU-TK	steam dump valve to technological (process) condenser (WWER)
BWR	boiling water reactor
BWRVIP	BWR Vessel and Internals Project
BWST	borated water storage tank
CCI	common cause initiators
CCF	common cause failure
CCWS	component cooling water system
CDF	core damage frequency
CEOG	Combustion Engineering Owner's Group
CFR	Code of Federal Regulations (USA)
CHRS	containment hydrogen recombiner system
CIS	Commonwealth of Independent States
CLG	coolant level gauge
CR	control rod
CRD	control rod drive
CRDM	control rod drive mechanism
CSN	Nuclear Safety Council (Spain)
CVCS	chemical and volume control system
DB	design basis
DBA	design basis accident
DC	direct current
DEGB	double ended guillotine break
DG	diesel generator
DNB	departure from nucleate boiling
DPS	diverse protection system
DSIN	Direction de la Sûreté des Installations Nucleaires (France)
EC	European Commission
ECCS	emergency core cooling system
ECR	emergency control room
ECT	eddy current testing

EDF	Electricité de France
EDG	emergency diesel generator
EFW	emergency feedwater
EFWS	emergency feedwater system
EOP	emergency operating procedure
EOS	electronic overspeed system
EPR	European Pressurized Reactor
EPRI	Electric Power Research Institute (USA)
EQ	environmentally qualified
ERDS	emergency response data system
ERG	emergency response guidelines
ESF	engineered safety features
ESFAS	emergency safety feature actuation system
ESW	essential service water
ESWS	essential service water system
EFWS	emergency feedwater system
FA	fuel assembly
FCBB	fraction of core boiling boundary
FME	foreign material exclusion
FW	feedwater
FWD	feedwater distribution
FWFCS	feedwater flow control system
FWS	feedwater system
GAN	Russian Nuclear Regulatory Authority
GANU	Nuclear Regulatory Authority of Ukraine
GDC	general design criteria (USA)
GE	General Electric (USA)
GL	Generic Letter (USNRC)
GRS	Gesellschaft für Anlagen und Reaktorsicherheit (GRS) mbH (Germany)
GSI	generic safety issue
HMS	hydrogen mitigation system
HPCI	high pressure core injection
HPES	human performance enhancement system
HPI	high pressure injection
HPIP	human performance investigation process
HPIS	high pressure injection system
HVAC	heating, ventilation and air conditioning
IASCC	irradiation assisted stress corrosion cracking
ICCS	intermediate component cooling system
I&C	instrumentation and control
ICDE	International Common Cause Failure Data Exchange
ICRP	International Commission on Radiological Protection
IEEE	Institute of Electrical and Electronics Engineers
IFBA	in-fuel burnable absorber
IGA	intergranular attack
IGSCC	intergranular stress corrosion cracking
IN	Information Notice (USNRC)
INPO	Institute for Nuclear Power Operations (USA)
INSAG	International Nuclear Safety Advisory Group (IAEA)
IPE	individual plant evaluation
IPEEE	individual plant examination of external event
IPERS	International Peer Review Service (IAEA)
IPSN	Institut de Protection et de Sûreté Nucléaire (France)
IRS	Incident Reporting System (IAEA)
ISA	Instrument Society of America

ISI	in-service inspection
KINS	Korea Institute of Nuclear Safety
KWU	Kraftwerk Union (Germany)
LBB	leak before break
LB LOCA	large break LOCA
LOCA	loss of coolant accident
LOFA	loss of flow accident
LPCI	low pressure core injection
LPI	low pressure injection
LPIS	low pressure injection system
LPS	low power and shutdown conditions
LPSC	loss of power conversion system
LTC	local technical centre
LTOP	low temperature overpressure protection
LWR	light water reactor
MAAP	modular accident analysis programme
MCC	motor controlled center
MCL	main circulating line
MCPR	minimal critical power ratio
MCP	main coolant pump
MCR	main control room
MFWC	main feedwater collector
MFWS	main feedwater system
MITI	Ministry of International Trade and Industry (Japan)
MIV	main isolation valve
MORT	management oversight and risk tree
MOST	Ministry of Science and Technology (Korea, Republic of)
MOV	motor operated valve
MSIV	main steam isolation valve
MSK	Medvedev Sponheuer Karnik (scale of seismic intensity)
MSLB	main steamline break
MSLIV	main steamline isolation valve
NDT	non-destructive testing
NDE	non-destructive examination
NEED	Nuclear Event Evaluation Database
NEI	Nuclear Energy Institute (USA)
NPP	nuclear power plant
NRC	Nuclear Regulatory Commission (USA)
NRPDS	Nuclear Plant Reliability Database System
NUMARC	Nuclear Utility Management and research Council (now NEI)
NUSS	Nuclear Safety Standards (IAEA)
OBE	Operation Basis Earthquake
OECD	Organisation for Economic Co-operation and Development
OL&C	operational limits and conditions
OST	overspeed trip
OSART	Operational Safety Review Teams (IAEA)
PAMI	post accident monitoring instrumentation
PAMS	post accident monitoring system
PIE	postulated initiating event
PCV	primary containment venting
PGP	procedure generation package
PISC	Programme for inspection of steel components
PORV	power operated relief valve
PRISE	primary to secondary system leakage
PSA	probabilistic safety analysis

PSAR	preliminary safety analysis report
PSR	periodic safety review
PTS	pressurized thermal shock
PWR	pressurized water reactor
PWSCC	primary water stress corrosion cracking
QA	quality assurance
RAI	requests for additional information (USNRC)
RAS	recirculation actuation signal
RBMK	light water cooled graphite moderated channel type reactor (Soviet design)
RCCA	rod cluster control assembly
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCR	reserve control room
RCS	reactor coolant system
R&D	research and development
RG	Regulatory Guide (USNRC)
RHR	residual heat removal
RIA	reactivity initiated accident
RPS	reactor protection system
RPV	reactor pressure vessel
RTS	reactor trip system
RVID	Reactor Vessel Integrity Database (USNRC)
RWCU	reactor water cleanup
RWST	refueling water storage tank
RWT	refueling water tank
SAR	safety analysis report
SB LOCA	small break LOCA (also SLOCA)
SBO	station blackout
SC	safety class
SCC	stress corrosion cracking
SDHR	secondary decay heat removal
SDHRS	secondary decay heat removal system
SDV	scram discharge volume
SEP	Safety Evaluation Program (USA)
SFP	spent fuel pool
SG	steam generator
SGI	interactive graphic simulator
SGTR	steam generator tube rupture
SI	safety injection
SIAS	safety injection actuation signal
SKI	Swedish Nuclear Power Inspectorate
SLB	steam line break
SLCS	standby liquid control system
SOV	solenoid-operated valve
SPDS	safety parameter display system
SRO	senior reactor operator
SRP	Standard Review Plan (USNRC)
SSC	systems, structures and components
SSE	safe shutdown earthquake
SSFI	safety system functional inspection
STA	shift technical advisor
TACIS	Technical Assistance to the Commonwealth of Independent States
TAPS	Tarapur Atomic Power Station (India)
TS	technical specifications
TOB	technical safety substantiation (Russian equivalent of PSAR)

TMI	Three Mile Island (USA)
TSC	technical support center
USNRC	United States Nuclear Regulatory Commission
UT	ultrasonic testing
VHP	vessel head penetration
WANO	World Association of Nuclear Operators
WOG	Westinghouse Owner's Group
WWER	light water cooled, water moderated pressurized reactor (Soviet design)

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