IAEA-TECDOC-1554

Generic Safety Issues for Nuclear Power Plants with Pressurized Heavy Water Reactors and Measures for their Resolution

IAEA
International Atomic Energy Agency

June 2007

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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. Publications related to these two items, that have been issued subsequent to this conference, include: 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.

In 1998, the IAEA completed IAEA-TECDOC-1044 entitled Generic Safety Issues for Nuclear Power Plants with Light Water Reactors and Measures Taken for their Resolution and established the associated LWRGSIDB database (Computer Manual Series No. 13). The present compilation, which is based on broad international experience, is an extension of this work to cover pressurized heavy water reactors (PHWRs). As in the case of LWRs, it is one element in the framework of IAEA activities to assist Member States in reassessing the safety of operating nuclear power plants. It addresses generic safety issues identified in nuclear power plants using PHWRs. In most cases, the measures taken or planned to resolve these issues are also identified.

The work on this report was initiated by the Senior Regulators of Countries Operating CANDU-Type Nuclear Power Plants at one of their annual meetings. It was carried out within the framework of the IAEA's programme on National Regulatory Infrastructure for Nuclear Installation Safety and serves to enhance regulatory effectiveness through the exchange of safety related information.

The IAEA officer responsible for this publication is G. Philip of the Department of Nuclear Safety and Security.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

The IAEA Conference on The Safety of Nuclear Power: Strategy for the Future, held in Vienna, September 1991, addressed the safety of operating nuclear power plants built to earlier standards. Participants in 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, at the IAEA General Conference, Member States endorsed the recommendations in a resolution urging 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 developed a report, entitled A Common Basis for Judging the Safety of Nuclear Power Plants Built to Earlier Standards, INSAG-8 (1995), 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. It aims at providing a 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. In the meantime, the IAEA Safety Guide 50-SG-O12 has been revised and is now available as the IAEA Safety Standards Series No. NS-G-2.10, Periodic Safety Review of Nuclear Power Plants (2003).

The IAEA Safety Series Report No. 12, Evaluation of the Safety of Operating Nuclear Power Plants Built to Earlier Standards - A Common Basis for Judgement was issued in 1998 to provide details for the safety assessment and judgement process. It provides practical advice on the main judgements to be made in any review process of plant safety.

These IAEA publications are based on the knowledge, experience and national programmes in Member States, to reassess safety of operating plants. The IAEA Extrabudgetary Programme on the Safety of WWER and RBMK NPPs has also contributed 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 insights into weaknesses in plant safety and into corrective measures to resolve them.

Experience feedback, including learning from incidents and incorporating systems, procedures and design changes is an important prerequisite for incident-free and safe operation of NPPs. The IAEA is issuing several publications in this area and this report on generic safety issues is an important complementary effort in this direction.

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 a 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.

The approach of using generic safety issues for identifying and resolving safety concerns has been practiced in many Member States since the 1970s. 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.

In 1998, the IAEA completed IAEA-TECDOC-1044 Generic Safety Issues for Nuclear Power Plants with Light Water Reactors and Measures Taken for their Resolution and established the associated LWRGSIDB database (Computer Manual Series No. 13). This TECDOC and database basically contain a list of generic safety issues, on the basis of broad international experience, could be used by the Member States as a reference in reassessing their operating plants. The applicability of these issues to each plant under review would, however, have to be checked on a case-by-case basis. It must also be pointed out that such a list of generic safety issues cannot ensure that all the possible safety concerns would be identified, since these would also be influenced by plant specific situations.

1.2. OBJECTIVE

In 1999, the CANDU senior regulators identified the need to have an analogous document addressing generic safety issues pertaining to PHWRs. They agreed to start compiling this TECDOC and to update it regularly. The first draft compilation was produced in 2000. This TECDOC is expected to assist Member States in the reassessment of operating plants by providing a list of generic safety issues identified in PHWR nuclear power plants together with measures proposed, in progress or taken to resolve these issues. 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.

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.

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.

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

1.3 SCOPE

The generic safety issues for PHWRs compiled in Section 4 of this report reflect the broad experience of Member States, 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.

Some of the generic safety issues identified in this report for PHWRs are common to other reactor types as well, and these are identified.

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 could be useful 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 through the CANDU senior regulators. 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 revisions. 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 PHWRs 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 86 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 is presented along with corrective measures proposed or implemented in different countries. This list is based on operational experience or events, deviations from current standards and practices and/or potential weakness (es) identified by analysis.

The above list of generic safety issues and the identified corrective measures, if taken, may affect the design, construction, or operation of all, several, or a class of nuclear power plants and may have the potential for safety improvements and promulgation of new or revised requirements or guidance.

2.2. STRUCTURE OF GENERIC SAFETY ISSUES

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

ISSUE TITLE
(A short title indicates the safety concern)

ISSUE CLARIFICATION

Description of issue

(Addresses: safety concern and root cause; systems, components and human 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)
(See guidance in Section 3)

- __________ 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 PHWR NPPS

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

2.3.1 DESIGN

2.3.1.1 GENERAL

GL 1 Classification of components
(Argentina, Canada, China, India, Republic of Korea, Romania)
GL 2 Environmental qualification of equipment and structures
(Canada, India, Pakistan, Romania)
GL 3 Ageing of equipment and structures
(Argentina, Canada, China, India, Republic of Korea, Pakistan, Romania)
GL 4 Inadequacy of reliability data
(Argentina, Canada, India, Republic of Korea, Romania)
GL 5 Need for performance of plant specific probabilistic safety assessment (PSA)
(Argentina, Canada, China, India, Republic of Korea, Pakistan, Romania)

2.3.1.2 REACTOR CORE

RC 1 Inadvertent dilution or precipitation of poison under low power and shutdown conditions
(Canada, India, Republic of Korea, Romania)
RC 2 Fuel cladding corrosion and fretting
(Argentina, Canada, India, Republic of Korea, Romania)

2.3.1.3 COMPONENT INTEGRITY

CI 1 Fuel channel integrity and effect on core internals
(Argentina, Canada, China, India, Republic of Korea, Pakistan, Romania)
CI 2 Deterioration of core internals
(Argentina, Canada, India, Republic of Korea, Romania)
CI 3 SG tube integrity
(Argentina, Canada, India, Republic of Korea, Pakistan, Romania)
CI 4 Loads not specified in the original design
(Argentina, Canada, India, Republic of Korea, Romania)
CI 5 Steam and feedwater piping degradation
(Canada, China, India, Republic of Korea, Romania)

2.3.1.4 PRIMARY CIRCUIT AND ASSOCIATED SYSTEMS

PC 1 Overpressure protection of the primary circuit and connected systems
(Canada, China, India, Republic of Korea, Romania)
PC 2 Safety, valve and relief valve reliability
(Canada,, India, Republic of Korea, , Romania)
PC 3 Water hammer in the feedwater line
(Canada, India, Republic of Korea)

2.3.1.5 SAFETY SYSTEMS

SS 1 ECCS sump screen adequacy
(Argentina, Canada, India, Republic of Korea, Pakistan, Romania)
SS 2 Potential problems in ECCS switchover to recirculation
(Argentina, Canada, India, Republic of Korea, Pakistan, Romania)
SS 3 Severe core damage accident management measures
(Argentina, Canada, China, India, Republic of Korea, Romania)
SS 4 Leakage from systems penetrating containment or confinement during an accident
(Canada,, India, Republic of Korea,)
SS 5 Hydrogen control measures during accidents
(Argentina, Canada, China, India, Republic of Korea, Romania)
SS 6 Reliability of motor-operated and check valves
(Argentina, Canada, India, Republic of Korea, Pakistan, Romania)
SS 7 Assurance of ultimate heat sink
(Argentina, Canada, India, Republic of Korea, Pakistan, Romania)
SS 8 Availability of the moderator as a heat sink
(Canada, India, Republic of Korea,)

2.3.1.6 ELECTRICAL AND OTHER SUPPORT SYSTEMS

ES 1 Reliability of off-site power supply
(Argentina, Canada, India, Republic of Korea,, Romania)
ES 2 Diesel generator reliability
(Argentina, Canada, China, India, Republic of Korea, Pakistan, Romania)
ES 3 Reliability of emergency DC supplies
(Argentina, Canada, India, Republic of Korea, Romania)
ES 4 Control room habitability
(Canada, China, India, Republic of Korea, Romania)
ES 5 Reliability of instrument air systems
(Argentina, Canada, India, Republic of Korea, Romania)
ES 6 Solenoid valve reliability
(Canada, India, Republic of Korea, Romania)

2.3.1.7 INSTRUMENTATION AND CONTROL (incl. Protection Systems)

IC 1 Inadequate electrical isolation of safety from non-safety related equipment
(Argentina, Canada, India, Republic of Korea, Romania)
IC 2 I&C component reliability
(Argentina, Canada, India, Republic of Korea, Pakistan, Romania)
IC 3 Lack of on-line testability of protection systems
(Argentina, Canada, India, Republic of Korea, Romania)
IC 4 Reliability and safety basis for digital I&C conversions
(Argentina, Canada, India, Republic of Korea, Romania)
IC 5 Reliable ventilation of control room cabinets
(Argentina, Canada, India, Republic of Korea, Romania)
IC 6 Need for a safety parameter display system
(Argentina, Canada, China, India, Pakistan, Romania)
IC 7 Availability and adequacy of accident monitoring instrumentation
(Canada, India, Republic of Korea, Pakistan, Romania)
IC 8 Water chemistry control and monitoring equipment (primary and secondary)
(Canada, India, Republic of Korea, Pakistan, Romania)
IC 9 Establishment and surveillance of setpoints in instrumentation
(Argentina, Canada, India, Republic of Korea, Romania)

2.3.1.8 CONTAINMENT

CS 1 Containment integrity
(Argentina, Canada, China, India, Republic of Korea, Pakistan, Romania)

2.3.1.9 INTERNAL HAZARDS

IH 1 Need for systematic fire hazards assessment
(Argentina, Canada, China, India, Republic of Korea, Romania)
IH 2 Adequacy of fire prevention and fire barriers
(Argentina, Canada, India, Republic of Korea, Pakistan, Romania)
IH 3 Adequacy of fire detection and extinguishing
(Argentina, Canada, China, India, Republic of Korea, Pakistan, Romania)
IH 4 Adequacy of the mitigation of the secondary effects of fire and fire protection systems on plant safety
(Canada, India, Republic of Korea, Romania)
IH 5 Need for systematic internal flooding assessment including backflow through floor drains
(Argentina, Canada, China, India, Republic of Korea, Romania)
IH 6 Need for systematic assessment of high energy line break effects
(Argentina, Canada, China, India, Republic of Korea, Romania)
IH 7 Need for assessment of dropping heavy loads
(Canada, India, Republic of Korea, Romania)
IH 8 Need for assessment of turbine missile hazard
(Argentina, Canada, China, India, Republic of Korea, Romania)

2.3.1.10 EXTERNAL HAZARDS

EH 1 Need for systematic assessment of seismic effects
(Argentina, Canada, China, India, Republic of Korea, Pakistan, Romania)
EH 2 Need for assessment of seismic interaction of structures or equipment on safety functions
(Argentina, Canada, India, Republic of Korea, , Romania)
EH 3 Need for assessment of plant-specific natural external conditions
(Argentina, Canada, India, Republic of Korea, Romania)
EH 4 Need for assessment of plant-specific man induced external events
(Argentina, Canada, China, India, Republic of Korea, Romania)

2.3.1.11 ACCIDENT ANALYSIS

AA 1 Adequacy of scope and methodology of design basis accident analysis
(Argentina, Canada, India, Republic of Korea,, Romania)
AA 2 Adequacy of plant data used in accident analyses
(Argentina, Canada, India, Republic of Korea, Romania)
AA 3 Computer code and plant model validation
(Argentina, Canada, China, India, Republic of Korea, Romania)
AA 4 Need for analysis of accidents under low power and shutdown conditions
(Argentina, Canada, China, India, Republic of Korea, Romania)
AA 5 Need for severe accident analysis
(Argentina, Canada,, India, Republic of Korea, Romania)
AA 6 Need for analysis of total loss of AC power
(Argentina, Canada, India, Republic of Korea, Romania)
AA 7 Analysis for pressure tube failure with consequential loss of moderator
(Argentina, Canada, India, Republic of Korea, Romania)
AA 8 Analysis for moderator temperature predictions
(Argentina, Canada, India, Republic of Korea, Romania)

AA 9 Analysis for void reactivity coefficient
(Argentina, Canada, India, Republic of Korea, Romania)

2.3.2 OPERATIONAL SAFETY ISSUES

2.3.2.1 MANAGEMENT

MA 1 Replacement part design, procurement and assurance of quality
(Argentina, Canada, India, Republic of Korea, Romania)
MA 2 Fitness for duty
(Canada, India, Republic of Korea, Romania)
MA 3 Adequacy of shift staffing
(Canada, India, Republic of Korea, Romania)
MA 4 Control of outage activities to minimize risk
(Argentina, Canada, India, Republic of Korea, Romania)
MA 5 Degraded and non-conforming conditions and operability determinations
(Argentina, Canada, India, Republic of Korea, Romania)
MA 6 Configuration management of modifications
(Argentina, Canada, India, Republic of Korea, Romania)
MA 7 Human and organizational factors in root cause analysis
(Canada, India, Republic of Korea, Romania)
MA 8 Impact of human factors on the safe operation of nuclear power plants
(Argentina, Canada, India, Republic of Korea, Romania)
MA 9 Effectiveness of quality management programmes
(Canada, India, Republic of Korea, Pakistan, Romania)
MA 10 Adequacy of procedures and their use
(Canada, India, Republic of Korea, Romania)
MA 11 Adequacy of emergency operating procedures
(Argentina, Canada, India, Republic of Korea, Romania)
MA 12 Effectiveness of maintenance programmes
(Argentina, Canada, India, Republic of Korea, Romania)
MA 13 Availability of R&D, technical and analysis capabilities for each NPP
(Canada, India, Republic of Korea, Pakistan, Romania)
MA 14 Strengthening of safety culture in organisations
(Canada, India, Republic of Korea, Pakistan, Romania)

2.3.2.2 OPERATIONS

OP 1 Operating experience feedback
(Argentina, Canada, India, Republic of Korea, Pakistan, Romania)

2.3.2.3 SURVEILLANCE AND MAINTENANCE

SM 1 Adequacy of non-destructive inspections and testing
(*Canada, India, Republic of Korea, Romania)*
SM 2 Removal of components from service during power or shutdown operations for maintenance
(Canada, India, Republic of Korea, Romania)
SM 3 Use of ice plugs
(Argentina, Canada, India, Republic of Korea, Romania)
SM 4 Control of temporary installations
(Canada, India, Republic of Korea, Romania)
SM 5 Response to low level equipment defects (plant material condition)
(Canada, India, Republic of Korea, Romania)

2.3.2.4 TRAINING

TR 1 Assessment of full scope simulator use
(Argentina, Canada, India, Republic of Korea, Romania)

TR 2 Training for severe (beyond design basis) accident management procedures
(Canada, India, Republic of Korea, Romania)

2.3.2.5 EMERGENCY PREPAREDNESS

EP 1 Need for effective off-site communications during events
(Argentina, Canada, India, Republic of Korea, Romania)

2.3.2.6 RADIATION PROTECTION

RP 1 Hot particle exposures
(Argentina, Canada, India, Republic of Korea, Pakistan, Romania)

RP 2 Management of Tritium
(Canada, India, Republic of Korea, Romania)

2.3.2.7 FUEL HANDLING

FH 1 Damage to fuel during handling
(Argentina, India, Republic of Korea, Romania)

3. GENERIC OBSERVATIONS

The generic safety issues for PHWRs compiled in Section 4 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 4 compilation also includes those safety issues which are currently considered "pending", i.e. the root causes and their resolutions are in the process of being developed. For these pending issues, interim judgments have been made by Member States with respect to continued safe plant operation. The hallmark of the Nuclear Industry is to review all aspects of the performance on a continuous basis and keep effecting improvements. These characteristics seem to be well reflected by the nature and variety of the Generic Safety Issues included in this document.

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. Many issues are not specific to PHWRs but are applicable to the whole Nuclear Industry, with a few of them being applicable to other industries as well. These are included to make this document a "stand alone" and comprehensive one. Some issues arise due to the unique nature of PHWRs and their design differences such as on-power refuelling and the use of fuel channel technology, and operational differences such as the management of heavy water and tritium. These are reflected in issues pertaining to tritium, complexity, health management of coolant channels, additional systems and equipment, on-power fuelling, etc.

The source or sources for each issue are provided in Section 3: that is, whether operational experience, a deviation from current standards and practices, or analysis gave rise to the concern. Nearly 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 were noted to be in over half 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 safety oversight.

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

3.1 OBSERVATIONS BASED ON OPERATIONAL EXPERIENCE

Almost all the issues in the management, surveillance and maintenance, training, emergency preparedness and radiation protection areas are the result of operational experience. Generally, these issues are generic to the nuclear industry and are elements or attributes of a nuclear management programme. Shortcomings in these elements can give rise to issues pertaining to the effectiveness of a nuclear management programme which, in turn, can have either direct or indirect nuclear safety implications. The elements of a nuclear management programme can be broadly classified into the following focus areas:

Operate the Plant: includes issues on elements such as conduct of operations, control of plant status and operations procedures (i.e. MA3: Adequacy of shift staffing).

Maintain the Plant: includes issues on elements such as conduct of maintenance, outage planning and management, work management, materials management and calibration (i.e. MA4: Control of outage activities to minimize risk).

Manage Risk: includes issues on elements such as radiation protection, PSA, fire protection, emergency preparedness and worker protection (i.e. TR2: Training for severe (beyond design-basis) accident management procedures).

Manage Engineering Design: includes issues on elements such as protecting the asset (e.g. fuel channels, steam generators, chemistry, NDE, in-service inspection, procurement) and engineering change control (i.e. MA1: Replacement part design, procurement and assurance of quality).

Manage Performance: includes issues on elements such as worker practices, corrective actions, independent assessments and managed systems (i.e. MA7: Human and organizational factors in root cause analysis).

Manage People: includes issues on elements such as promoting and supporting a strong safety culture, training, human performance management and workforce planning (i.e. MA2: Fitness for duty).

One area where operational experience indicates a common trend 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. The verification and validation of digital system and software are challenging tasks. 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. There are issues connected with digital technology in two other areas as well, i.e. deviations from current standards and analysis insight.

3.2 OBSERVATIONS BASED ON DEVIATIONS FROM CURRENT STANDARDS AND PRACTICES

For PHWRs the pressure boundary includes the coolant channels. A lot has been learned on various aspects of coolant channel behaviour and several steps have been taken by member states to strengthen coolant channel health management.

Section 4 brings out only a few issues which are linked to ageing of the plants. Many NPPs have been operating for many years 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), which, in the meantime, has been revised and is now available as the Safety Standards Series No. NS-G-2.10, Periodic Safety Review of Nuclear Power Plants (2003); and INSAG-8, A Common Approach to Judge NPPs Built to Earlier Standards) 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 PHWRs 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. PHWRs however are comparatively “younger” and life cycle management needs more effort as more plants age. In some cases, components unique to PHWR require special studies which cannot be obtained from work on other types of reactors.

Several safety issues result from deviations from current safety standards and practices. Some of these apply to reactors of older design which have not regularly taken into account the operating experience of other plants and the resultant adjustment to 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. These are generally in areas where substantial advances are made worldwide. Examples are fire protection, seismic qualification, internal and external hazards etc.

In this regard, the substantial participation in this effort and the improvement measures already taken by countries with older reactors is noted as a positive development 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 and during outages, station blackout, specific configuration situations, maintenance and outage management, external hazards).

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. Efforts would be required in areas of severe accident management; symptom based emergency operating procedures, taking fuller advantage of PSA especially during operational phase etc.

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.

4. GENERIC SAFETY ISSUES FOR PRESSURIZED HEAVY 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

This issue is also applicable to NPPs with LWR.

Safety-related components are required to be classified into several categories including seismic, safety, and quality groups in order to assure their safety functions. The components must also meet specific design, construction, testing, and inspection criteria.

The concern is applicable in particular to the old generation NPPs since the requirements for classification of components were not as detailed as current standards.

Safety significance

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

Equipment built to older standards is not necessarily less safe than current equipment. However, it may be more difficult to prove that its safety performance meets all safety requirements.

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:

Argentina

The Regulatory Authority reviewed component classification list checking consistency. During 1999 seismic component classification were updated. An additional classification list considering the risk impact is currently in use for regulatory purposes.

Canada

One of the pre-requisites for a plant construction license is an approved “Systems Classification List” which classifies plant systems and pressure retaining components into non-nuclear, or nuclear class 3, 2, or 1, in accordance with applicable national and international standards, which are acceptable to the regulatory body (the Canadian Nuclear Safety Commission - CNSC). These codes and standards also specify the requirements relative to subsequent testing and inspections, and quality assurance. The aspect of seismic qualification of components and systems is addressed in sections EH1 & 2 of this report.

Modifications to older plants, built to an earlier set of standards, are generally done to current standards.

China

Systems classification and Components classification of Third Qinshan Nuclear Power Plant (TQNPP, CANDU 6 type) were reviewed by the Regulatory Body (the National Nuclear Safety Administration – NNSA). The Systems classification and Components classification should meet Chinese Regulation HAF0200 and Safety Guide HAF0201. The Systems classification and Components classification list should be approved in PSAR review by NNSA.

India

AERBs Code of Practice on Design for Safety in Pressurized Heavy Water based Nuclear Power Plants (SC/D) establishes the criteria for design approaches and design requirements for PHWRs that shall be met for safe operation and in order to prevent the accident or mitigate the consequences of Design Basis Events (DBEs) which could jeopardise safety. AERB has further prepared a Design Safety Guide, AERB/SG/D-1 on "safety classification and seismic qualification for PHWRs", which aims to classify the Functions, Systems, Structures and Equipment (FSSEs) according to their importance to safety. The resulting classification determines the relevant design criteria. The design criteria are measures of quality by which the adequacy of each FSSE in relation to its importance to plant safety is ensured. This Safety Guide describes the classification procedure for the FSSEs according to their importance to safety. The procedure followed for this purpose is to identify various safety functions required to be performed in a NPP to achieve safety. These safety functions are then grouped and ranked into safety classes taking into consideration the consequences of failure of the safety function performed by the FSSE and the probability of its occurrence. Appropriate design requirement for each class is established with most stringent requirement for the highest class and so on.

This guide also covers the seismic categorisation of FSSEs as per AERB siting Code. QA requirement is determined by the Code of Practice on Quality Assurance for Safety QA, the extent of its application shall be consistent with the importance of the items to safety and shall be in conformance with the classification of items as regards to safety.

The design requirements once established has to be met by following national and international codes or standards acceptable to AERB.

In old generation NPPs the structures, systems and components (SSC) which include Items Important to Safety (IIS) of a Nuclear Power Plant (NPP) are required to function without impairment of their safety margins and reliability, as per design specifications in all operational states during service life of NPP.

The physical characteristics of SSC, change with time and use, due to ageing process. If appropriate actions are not taken the safety margins provided in the design are reduced due to ageing. Safety state (i.e. integrity and functional capability) of plant components, both passive and active, changes with use and time resulting in reduction of safety margin.

In this regard, careful assessment of the ageing characteristics of SSC, the factors influencing the ageing process, and their consequences on safety margins and reliability, is essential for planning and implementing timely actions for assuring safe operation of NPP during its life-time. Feasibility of upgrading of SSC, for complying with current safety standards will also require a planned approach for effecting necessary modifications.

A programme for all phases of Management of NPP viz. preoperational, operational, and post-operational in respect of SSC should be instituted by utility.

AERB has prepared a safety guide (AERB/NPP/SG/O-14) on ‘Life management of NPPs. This guide details the essential factors that are required for a comprehensive assessment of the ability of IIS for performing their intended functions reliably as per design specifications. Requirements for planning and implementing an effective Life Management (LM) programme for IIS in NPP are addressed. Factors that need consideration during the siting, design, construction and operation phases of NPP, for planning and implementing the LM programme are also explained in the guide. The guide covers the following aspects of the Life Management of NPP:

(i) Degradation of SSC during pre-operational and operating phase
(ii) Analysis of factors influencing ageing and assessing residual life of SSC
(iii) Measures to mitigate ageing effects
(iv) Considerations and approach for License Renewal at the end of design life
(v) Organisational aspects of LM

Elements of classification of components are addressed while reviewing items (ii), (iii) and (iv) above.

Korea, Republic of

This is one of the pre-requisites for a plant construction license. Applicable national and international standards classify plant systems and pressure retaining components into non-nuclear, or nuclear class 3, 2, or 1, which are acceptable to the regulatory body. These codes and standards also specify the requirements relative to subsequent testing and inspections, and quality assurance. See also issue EH 1, EH 2.

Romania

CNCAN (The Romanian Regulatory Authority) has to approve the "Systems Classification List" (SCL), which includes:

- requirements for the system classification as per the CANDU 600 philosophy
- requirements for the system classification as defined by the IAEA guidelines

The classification is included in a plant procedure of type "A", which means that the procedure is approved by CNCAN and it is directly referenced in the Operating License. The existence of the clarified SCL was a prerequisite for the envelope construction approval in 1993 and of the all subsequent licenses.

The initial design requirements prevail. However supplementary safety requirements are attributed to some support systems. If any conflicts exist between the two types of requirements the initial design ones were considered to prevail.

The Licensee is reviewing the SCL documents for the procurement aspects, having as a target the obsolence aspects in mind. The completion of this activity is supported under an IAEA project, too. The results will be part of the reviewed SCL to be approved by the Regulatory Body as part of the licensing process.

The overall classification resulted defines systems like:

- Special Safety Systems (ECCS, SDS1, SDS2 and Containment systems)
- Safety related Systems (EPS and EWS)
- Process Systems with safety functions (like for instance primary system, systems components of the prmary system pressure boundary etc.)

The initial concept classification including concepts on two group separation, safety categories of 1a) ..d) and 2a).c) classes are fitted into the above mentioned grouping.

Based on the safety classification list, an Item Classification list for each component of the safety systems is derived.

ADDITIONAL SOURCES:

- ARN SRNRP-1-P-8, PSA Regulatory Uses, Working Plan 1999/2000 (Argentina).
- ASME code.
- CSA standards (Canada).
- ATOMIC ENERGY REGULATORY BOARD, "Code of Practice on Design for Safety in Pressurized Heavy Water based Nuclear Power Plants" (AERB/SC/D).
- ATOMIC ENERGY REGULATORY BOARD, Design Safety Guide, "Safety Classification and Seismic Categorisation for Pressurized Heavy Water Reactors", AERB/SG/D-1.
- CNNSC, QA requirement is determined by the Code of Practice on Quality Assurance for Safety in Nuclear Power Plants - SC/QA.
- System Classification List - Cernavoda NPP Unit 1 Station Procedure.
- ATOMIC ENERGY REGULATORY BOARD, Safety Code, "Code of Practice on Quality Assurance for Safety in NPPs", AERB/SC/QA.
- ATOMIC ENERGY REGULATORY BOARD, Safety Guide, "Life Management of NPPs" AERB/NPP/SG/O-14.

ISSUE TITLE: Environmental qualification of equipment and structures (GL 2)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

The qualification of equipment important to safety is required to demonstrate its ability to fulfill its intended functions. This qualification requirement applies to normal operating conditions, to accident conditions including 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.

Safety significance

Insufficient or lacking qualification of equipment important to safety with respect to extreme environmental conditions would seriously affect defence in depth, and the safety functions, that could be questionable for scenarios within the DB envelope.

An area that needs to be addressed is the effect of aging on the environment qualification capability of equipment.

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:

Canada

There is a CNSC requirement that all equipment credited for functioning in a harsh environment be environmentally qualified. All licensees are required to identify this equipment and to environmentally qualify it. This issue is addressed in two CNSC position statements entitled “Assurance of continued nuclear safety” and “Post accident filter effectiveness” issued in 1990 and 1991 respectively. Licensees responded adequately to the first issue, which was subsequently closed. The second one, still open for some licensees, is described below.

Emergency Filtered Air Discharge System:

The emergency filtered air discharge system (EFADS) in multi-unit stations would be used to maintain containment pressure below the atmospheric level in the long term. The EFADS filters are relied upon to limit radioactive releases during the venting of containment. The single unit stations do not depend on filtered air discharge systems to ensure the effectiveness of their containment systems. It is essential, therefore, that the licensee have programmes to support the credited effectiveness of filters and to ensure prompt detection of any deficiency which could prevent the filters from performing as designed.

To achieve closure, licensees are expected to have effective programmes in place which will provide a continued assurance that under all credible post-accident conditions filter effectiveness will match or exceed that credited in safety analyses. These programmes may, as necessary, consist of laboratory and in situ tests, research and analytical efforts, adequate QA, testing, maintenance, and operational procedures and should:

- Ensure that there are appropriate tests for all filter elements.
- Demonstrate that tests are performed under representative conditions.
- Provide evidence confirming that filters are capable of performing as credited under harsh post-accident environmental conditions.
- Provide an assurance that filter degradation can be reliably detected by existing tests.
- Address consequences of hydrogen burns on filter performance.
- Confirm that there are proper quality assurance, maintenance and operation procedures in place.
- Justify frequency of tests.
- Provide an assurance that instrumentation and control equipment is such that it will allow operation of filters as designed under all representative conditions.
- Demonstrate the adequate availability of filters.
- Identify areas, if any, where continued design, experimental and/or analytical efforts are required.

India

Environmental Qualification:

The environmental qualification of instrumentation items has been carried out in a limited manner from NAPS onwards. The qualification programme consists of accelerated test, radiation test and LOCA chamber test. For example, critical instruments like solenoid valves and pressure transmitters, pressure switches, cables, etc., for containment pressure measurement have been identified for the qualification programme.

LOCA chamber test facility of a small size is functional within DAE. Instruments like PRVs, pressure transmitters, pressure switches have been qualified under Environmental Qualification for new plants. The critical instruments to be used for all future plants are to be LOCA qualified prior to their installation. Larger LOCA chamber test facility has been set up at a national research laboratory to take up this testing. The junction boxes have been qualified to be steam proof with appropriate gaskets and cable glands, and the terminal blocks have been qualified to withstand high temperature and radiation up to a dose of 100 MR. The solenoid valves in critical areas have been provided with watertight junction boxes with cable glands. A list of items to be environmentally qualified, and the extent/specification for qualification has been prepared by utility. Requirement for Environmental qualification programme for the NPPs is being addressed by AERB. Design safety guide AERB/SG/D-3 gives the requirements of protection against internally generated missiles and associated environmental conditions for PHWRs for existing plant an assessment is made on environmental qualification under the periodic safety reviews as per AERB Safety Guide AERB/SG/O-12 on ‘Renewal of authorisation for Operation of NPPs’..

An area that needs to be addressed, however, is the effect of aging on the environment qualification capability of equipment. As of now equipment is tested only initially.

Pakistan

Qualification of equipment and instrumentation of safety and safety-related systems for ability to perform their functions under expected accident conditions is considered to be of high importance. The equipment and instrumentations installed at KANUPP are of late 1960s design/manufacture and it cannot be claimed – with high degree of certainty – that these instrumentations will fulfill their task under the harsh environmental condition following an accident (LOCA or steam line break).

In order to address the E.Q issue, electrical junction boxes of safety and safety related systems have been upgraded for LOCA & MSLB qualification. The open end of cable conduits have been sealed with qualified sealant to prevent moisture ingress and its impact on the end device. The motorized valves and motors of emergency core cooling system are being replaced with qualified ones. The C&I have also been replaced with C&I complying with current standards.

Korea, Republic of

PSR rule is a major regulation not only for preserving EQ but also for continuous operation of nuclear power plant. For the continuous operation of NPP, applicant must demonstrate one of the below Time limited Ageing Analysis (TLAA) (Same as NUREG-1800 of USA)

1. The analyses remain valid for the period of extended operation.
2. The analyses have been projected to the end of extended period of operation.
3. The effects of the aging on the intended function will be adequately managed for the period of extended operation.

EQ Phase I

- EQ accident analysis if there was no HELB analysis for out side containment high energy pipe line
- Expand the equipment level of EQ master list to the subcomponent
- Field walk-down for verification of installed equipments
- Preparation of preserving EQ procedure
- Establish a qualification method for the equipments of no EQ document

EQ Phase II

- EQ type test for equipments & cables: EQ type test will accompany with refurbishment of equipment if any severe damage is anticipated with present parts and representative cable will be replaced for type test if there are no EQ test report
- Qualification by EQ analysis if material construction is similar to these EQ tested items
- Preparation of EQ Evaluation & Review paper (EQER)
- Development of web-based computer software for managing EQ system

Romania

The qualification of equipment and structures is defined by all the safety requirements applicable and it is embeded in a global document "Item Classification List", summarizing all the requirements for the special safety system, safety related system and system with safety function components. The requirements for a component are defined also in the procurement documents (EQR- Equipment Quotation Request and in the Technical Specifications attached to them), reflecting the licensing basis as formulated in the regulatory environment. Many aspects of the initial licensing basis for unit 1 (in operation) are subject to the Strategic Policy on Safety and are under review to be integrated in the first Periodical Safety Review program requirements to be issued in May 2001. The aspects of feeder thinning and their incorporation in the reviewed inspection programs, of the environmental qualification and/or many other specific requirements on detailed aspects like emergency filtered air discharge system: are part of the periodical safety review. The aspects of obsolence and equivalence for the spare parts not produced any more is a topic of highest priority and it is included in the above mentioned programs.

For Cernavoda NPP unit 2 there are from the very beginning modifications in the equipment qualifications due to the necessity to comply with the new updated licensing basis as defined by the regulatory requirements.

ADDITIONAL SOURCES

- INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Design Requirements, IAEA Safety Standards Series No. NS-R-1, IAEA, Vienna (2000).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Management System for Facilities and Activities, Safety Requirements, IAEA Safety Standards Series No. GS-R-3, IAEA, Vienna (2006).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Equipment Qualification in Operational Nuclear Power Plants: Upgrading, Preserving and Reviewing, Safety Report Series No. 3, IAEA, Vienna (1998).
- CNSC Position Statement 90G03, "Assurance of Continued Nuclear Station Safety" (Canada).
- CNSC Position Statement 91G01, "Post Accident Filter Effectiveness" (Canada).
- ATOMIC ENERGY REGULATORY BOARD, "Code of Practice on Design for Safety in Pressurized Heavy Water Based Nuclear Power Plants" (AERB/SC/D).
- ATOMIC ENERGY REGULATORY BOARD, Design Safety Guide, "Safety Classification and Seismic Categorization for Pressurized Heavy Water Reactors", AERB/SG/D-1.
- ARN/801-98 Establishment of an Aging Programme, Regulatory Requirement, 1998 (Argentina).
- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN, March 2000.
- Strategic Policy for Cernavoda NPP Unit 2 licensing process, CNCN, 1997.
- Licenses for Operation of Cernavoda NPP Unit 1, CNCAN, May 1999.
- Licenses for Unit 2 (partial works for safety systems and fuel channel installation), CNCAN 1999.
- ATOMIC ENERGY REGULATORY BOARD, Safety Guide, "Protection against Internally Generated Missiles and Associated Environmental Conditions", AERB/SG/D-3.

ISSUE TITLE: Ageing of equipment and structures (GL 3)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR. The specific cases are, of course, specific to PHWRs.

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. To manage aging of SSCs important to safety effectively, plant owners/operators need to have in place effective ageing programmes, which provide for timely detection and mitigation of aging 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 aging is being effectively managed and that effective programmes are in place for continued safe operation.

There are generic issues for the ageing phenomena in the PHWR reactors. These might be illustrated by the unexpected reduction in the wall thickness of some outlet feeders. The rate of this degradation represents a departure from the original design predictions.

Safety significance

Insufficient qualification of equipment important to safety with respect to ageing effects might reduce the effectiveness of the safety functions. Special designated programmes to cope with the ageing effects (Ageing management programmes) are expected to be performed, as part of the periodic safety review process if such a process is in place. The goal of such a programme is to minimize the risk due to ageing. Such a programme need to be dynamic to address new degradation mechanism identified during operation e.g feeder thinning.

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:

Argentina

The assessment of aging and obsolescence requires the analysis of each component that should provide both an answer and a solution to the following:

- Design specifications and requirements to be fulfilled.
- Reliability demands.
- Classification according to environmental conditions, seismicity, etc.
- Certification of test and simulations.
- Quality Assurance Programme.
- Suppliers' qualification.

The effects of obsolescence are basically evaluated when a component is to be replaced by another for corrective maintenance purposes. This is generally applied to instrumentation and control components, which are usually modern and easily replaceable items. Normally, changes introduced as a result of obsolescence are associated with corrective maintenance policies (for example: the mercury wetted relays used in safety systems). The procedure described is also applicable to electromechanical components, particularly the safety related ones.

In view of the fact that aging used to be considered only in some specific cases and not systematically, the ARN requested the Licensee to elaborate an aging management programme with the purpose of selecting the relevant components on which the impact of aging should be assessed, analysing aging mechanisms, specifying the monitoring method and, finally, suggesting the necessary actions to mitigate aging effects (maintenance, operation and design improvements). Issues raised by the Licensee are analysed by the Regulatory Authority; in fact, since these issues are aimed at improving safety conditions, the Licensee's proposals are usually accepted by the ARN. In turn, the ARN also raises issues to be discussed in advance, as in the case of the reactor pressure vessel of Atucha I Nuclear Power Plant, as well as other issues resulting from operational experience (the change of core neutron detectors at Embalse CANDU 6 NPP). In general, issues are proposed after decisive actions have been taken regarding PSA results and after the aging mechanisms of the main components have been analysed in terms of the overall plant risk. . Some ageing effect have been found during routine inspections like the feeders wall thinning less important than others PHWR´s where regulatory requirements and follow up issued by ARN to monitor the thinning rate were established.

Canada

Assurance of Continued Nuclear Station Safety:

Safety-related functions in nuclear power plants must remain effective throughout the life of the plant. Licensees are expected to have a programme in place to prevent, detect and correct significant degradation in the effectiveness of important safety-related functions.

The identification of all risk-significant systems, structures and components and their failure criteria using a systematic method, is a necessary prerequisite to a programme for the management of age-related degradation. Acceptable methods for addressing CNSC expectations are described in IAEA documents No. 338 of the Technical Report Series “Methodology and Management of Aging of Nuclear Power Plant Components Important to Safety”, and Safety Report Series No. 15 “Implementation and Review of Nuclear Power Plant Aging Management Programme”. The CNSC expects that the programme documentation include: the organization responsible for the implementation of the programme, the identification of the roles and responsibilities of the different groups directly involved with the programme, and the means for assuring adequacy and completeness of programme implementation.

The work required is expected to include the aspects of assessment, monitoring, mitigation, recording and reporting.

Feeder thinning:

The CNSC requested licensees to show that feeders are fit for service. They were also asked to show sufficient understanding of the thinning phenomenon to prevent it from threatening the integrity of the feeders.

To achieve closure, licensees are required to minimize the possibility of increasing the frequency of feeder failure beyond the failure of one feeder pipe, assumed in the safety report, taking the following into consideration:

- The problem of feeder wall thinning will be managed and there will be sufficient site-specific inspection to confirm the degradation models. Inspections should be carried out at least once every year to validate the degradation rate.
- Each station will be responsible for its own programme to resolve the problem.

- Feeder fitness for service guidelines should be established and approved for use at each station.
- Licensees will prepare and submit to the CNSC their respective strategies for management/monitoring of feeder degradation.

China

It has been observed that the feeder pipes wall thinning at outlet feeder bends in CANDU 6 reactors is higher than anticipated. According to the AECL research, the corrosion rate of feeder related the coolant pH, flow rate, and carbon steel chromium content.

NNSA required Third Qinshan Nuclear Power Company (TQNPC, Utility of TQNPP) to take effective measures to solve the Feeder Pipe Wall Thinning Issue in TQNPP PSAR review stage.

AECL analyzed the reason and commented that Cr content of 0.02% in inlet and outlet feeders was too low, so corrosion resistant performance was weak. As per AECL's opinion, increasing the CR content will improve corrosion resistant performance, and the 40-year-service life of the feeders can be ensured.

A test program to confirm that TQNPP feeders will achieve design intent is underway. AECL is committed to complete the tests and submit the TPNPP Implementation Report for further review. This issue will be closed by NNSA until the design requirements of feeders are met and confirmed by the test results.

India

India have a few NPPs like RAPS 1 $ 2 which are old and hence the need for effective (Ageing Management Programme) life cycle management (LCM) was felt both by the utility and AERB. In consultation with utility and experts AERB prepared a safety guide "Life Management of NPPs", AERB/NPP/SG/O-14. This guide addresses all the concerns brought out in the description portion of this Generic Safety Issue. This guide consists of all features which form part of an effective Ageing Management Programme brought out by IAEA and several other regulatory bodies. It includes selecting and assessment of SSCs, both active and passive, main environmental and operational stressors, determination and safety margin, SSC qualification, determination of safe residual life, follow-up actions etc. It was realised that for a healthy old age care and data collection from inception stage is essential. This starts from design stage, for eg. It satisfactory coupons are not installed initially, vessel wall condition and degradation due to various stressors cannot be determined. Similarly, healthy operation (ex: good chemistry, avoiding frequent stress cycles etc.) is essential for safe extended life and thus all these are brought out in the life management guide. The guide brings out management aspects of LCM in addtion to pre-operational and operational life management (ageing) aspects.

In additiion to AERB/NPP/SG/O-14 guide, AERB guide on "Renewal of Authorisation for Operation of NPPs", AERB/SG/O-12 also covers few issues of LCM. In consultation with utility and experts AERB has formulated the policy and methodology for application of current standards to existing NPPs, as well as judging NPPs, built to earlier standards. The O-14 guide has converted principles brought out in INSAG-8 "A common basis for judging the safety of NPPs built to earlier standards" into practiceable and easily understandable steps which can be applied to arrive at regulatory decision. It is improtant to be dynamic in our approach to enable new opeational feedbacks (ex: feeder thinning) to be absorbed and acted on. SG/O-14 on LCM forms a good tool to monitor the life management activities of the NPPs by AERB.

Environmental qualification of SSCs, after ageing, is an issue that needs further review.

Korea, Republic of

For Wolsong Unit 1, the permanent on-line monitoring system for feeder thickness measurement was installed at two critical channels. For Wolsong Units 3&4, the thickness of all the feeder pipes was measured and will be checked periodically.

KINS required KEPCO to extend to Wolsong Unit 1 the scope of the periodic inspection for feeder pipe thickness measurement, using more reliable non-destructive examination techniques. KEPCO was also required to establish a mid- and long-term research programme.

Feeder pipe thickness measurements for the Canadian nuclear power plants e.g., Point Lepreau and Gentilly-2 in 1995 and 1996 showed that the wall thinning of feeder pipes near the end fitting outlet occurred relatively faster than expected. CANDU-6 Station Information Bulletin 96-2, "Feeder Wall Thickness Measurements" was come out in 1996. According to the recommendation of the Bulletin 96-2, Wolsong Unit 1 was performed in-service inspection at the first curvature position near the End Fittings for 42 feeder pipes in 1996. Since then, the wall thickness was measured during every outage period and the number of inspected feeder pipes was increased by request of regulatory body. A total of 229 feeder pipes has been inspected in 2000 outage period. The results show that the wall thinning is gradually going in progress and some feeders have a possibility to be reached to minimum allowable wall thickness in near term. The erosion-corrosion is a main reason for wall thinning of outlet feeder pipe made by carbon steel, which has the potential to reduce the design life of some feeders. Therefore, it is necessary to periodic monitoring of the wall thickness and estimation of the erosion-corrosion rate.

Pakistan

Feeder wall thickness

Unexpected reduction in feeder wall thickness has been found in several CANDU reactors. Feeder pipe thickness measurements for the Canadian nuclear power plants i.e. Point Lepreau and Gentilly-2 in 1995 and 1996 showed that wall thinning of feeder pipe near the end-fitting outlet occurred relatively faster than expected. Presence of an active degradation mechanism (wall thinning) in feeders if left untreated can result in several feeder failures. Since then, wall thickness of all CANDU reactors is being measured during outage period and the number of feeder pipes inspected has been increased on the instruction of regulatory body.

KANUPP has been required to continue the measurement of feeders wall thickness at every opportunity and is also required to submit to PNRA a plan providing details for feeder pipe wall thinning assessment and management strategy.

Availability of Channel Activity Monitoring (AH) System:

AH system is designed to identify the channel containing the failed fuel. The system is unserviceable for the last several years and cannot be taken into service due to some undetectable leak in activity monitoring rooms. Efforts were made by KANUPP to locate these leaking points in the past but no result could generally be achieved. It was only late 2000 that some success was achieved in identifying the leaking coil during a special test performed with coolant temperature 525^0F.

A large number of leaking coils were identified. KANUPP has isolated these coils by cutting and welding the corresponding lines. The problem has not yet been resolved and further efforts are underway to arrest the issue.

The ageing problems as part of the safety requirements for the systems and components were part of the current licensing concerns, and are reflected in the license requirements for unit 1 and the generic requirements for unit 2. These requirements were also scrutinized under various projects in the framework of the national TC projects with IAEA and PHARE year I project with EU.

These aspects were reiterated in the Strategic Policy on Safety and are under review to be integrated in the first Periodical Safety Review program requirements to be issued in May 2001. The aspects of obsolence and equivalence for the spare parts not produced any more is a topic of highest priority and it is included in the above mentioned programs.

For Cernavoda NPP unit 2 there are from the very beginning modifications in the equipment qualifications and/or requalification from the moment of the restart of the licensing process, as a necessity to comply with the new updated licensing basis as defined by the regulatory requirements.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Management System for Facilities and Activities, Safety Requirements, IAEA Safety Standards Series No. GS-R-3, IAEA, Vienna (2006).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Power Plant Life Management Processes: Guidelines and Practices for Heavy Water Reactors, IAEA-TECDOC-1503, IAEA, Vienna (2006).
- CNSC Position Statement 90G03, “Assurance of Continued Nuclear Station Safety” (Canada).
- CNSC Position Statement 91G01, “Post Accident Filter Effectiveness” (Canada).
- ATOMIC ENERGY REGULATORY BOARD, “Code of Practice on Design for Safety in Pressurized Heavy Water Based Nuclear Power Plants” (AERB/SC/D).
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- ATOMIC ENERGY REGULATORY BOARD, Safety Guide, “Life Management of NPPs” AERB/NPP/SG/O-14.

ISSUE TITLE: Inadequacy of reliability data (GL 4)

ISSUE CLARIFICATION

Description of issue

This issue is also applicable to NPPs with LWR.

A well-organized component reliability database is a pre-requisite to enable the quantitative evaluation (e.g. PSA) of a nuclear power plant.

The type and degree of detail of the database is determined by the intended use 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/or for an optimal maintenance programme, a plant specific database is generally required.

Safety significance

When using quantitative tools (e.g. PSA) to identify weaknesses and their prioritizations, the lack of component- and human-specific reliability data can lead to suboptimal decisions with respect to design or procedure modifications and regulatory requirements.

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 has been recording reliability data on many areas of reactor plant for more than ten years, in the particular case of Atucha I, with more than 20 years of operating experience and unique design. Such data retrieval has increased with the development of the PSA taking into account the scope and the applications of such assessments. Realistic input data for PSA models were necessary, in particular to include data for common cause failures (CCF) due to its dominant risk factor on PSAs. The basis and the clear justification of safety-related component data is a regulatory requirement. Many reliability parameters are checked before to be used on PSA, such as: Test frequency, restoration time, average test duration, maintenance unavailability, failure rate and demand failure. Embalse NPP has also a large failure rate data bank recorded during the operating experience (17 years).

Canada

Recording reliability data is a requirement for special safety systems. Reliability data must be reported as a part of the Annual Reliability Report required by regulatory standard S-99.

Realistic and reliable data are among the criteria which would make a PSA suitable for regulatory decision-making.

PRA work in Canada utilizes CANDU component failure databases, which have been developed by the utilities, based typically on their plant-specific operating experiences. A need for a generic CANDU component database has been realized and AECL is in the process of starting a pilot project to develop such a generic database. Various CANDU utilities are being invited to participate in this pilot project by providing their plant-specific database to AECL for a selected set of components. For design PSA work, CCF analysis is being performed using the UPM (Unified Partial Method) technique developed by SRD in England.

India

The importance of having reliability data base specific to Indian PHWRs of assured quality was recognized since long. There is an additional problem faced due to nascent nature of many Indian Vendors who are not able to provide reliable failure data of equipment they have supplied. Inability to standardise equipment has further complicated the issue. In the development of a co-ordinated PSA programme initiated at 1990, the emphasis was laid to collect and create plant specific data base. To speed up the process of collecting and processing failure parameters, an Apex Committee was constituted and Expert Working Groups were formed to delineate the responsibilities and time frame for completion of assigned tasks. An in-house PSA group within the AERB has been reconstituted to support the PSA development and review PSA programme. This group besides co-ordinating with the Core group has been making a parallel effort to generate plant specific data base from the plant documents such as safety related unusual occurrences, monthly/annual plant performance reports, technical bulletins, maintenance history dockets of components. Besides component failure data, separate groups have been working on generating data on common cause failure and human reliability. The preliminary human reliability data collection, processing, modeling with reference to MAPS are in progress. Efforts are on to incorporate features in the simulator for Kaiga Plant to reliability data for human actions initially for the design basis events and then gradually expanding to all the dominant accident sequences. There is a separate cell to streamline human factor input to PSAs.

The newly introduced computerised maintenance management system will be helpful in collection and consolidation of failure data.

It is expected that the good quality reliability database will be produced progressively with first batch data on component failures available on the network by the end of 2000. Till then, generic data base as applicable and plant specific data base as available will continue to be used for PSA related studies to meet regulatory requirements. The earlier practice of reporting to AERB, observed realibility of safety system/equipment from testing/maintenance data and comparing with values assumed in safety analysis/technical specifications is revived.

Under the co-ordinated PSA programme started in 1990, efforts were made to prepare plant-specific component reliability database. The newly introduced Computerized Maintenance Management System (CMMS) was supposed to be useful in this regard. Even though, this software is installed at most of the NPPs, some fields are yet to be incorporated in the same for PSA based data collection.

A standard “generic component reliability database” is being prepared by AERB. The development of plant-specific component reliability database will be initiated by AERB as a priority task for the improving the quality of the PSA, which can be used in the Risk-Informed Decision-making.

The work for the Human Reliability data collection has been initiated by AERB with help of consultant.

Korea, Republic of

KINS has "Nuclear Event Evaluation Database" system to provide operational information for regulatory body, utilities, and public and to systematically manage and maintain information on nuclear power plant major events.

KEPCO, the utility, has several reliability database systems such as "Plant Outage Information & Tracking System" and "Power Unit Maintenance Management System/Nuclear-II." The main purposes of these systems are 1) sharing information on failures occurred to ensure the ability to cope with a failure, 2) centralized management of failure data, and 3) increasing component reliability by establishing a systematic maintenance plan.

Management system for nuclear power plant diesel generator information and maintenance history, DGIS, was developed by KAERI (Atomic Energy Research Institute) and KEPCO has been recording reliability data through this system.

Share of database developed by each organizations, agreement on components and systems for reliability data collection, proper human reliability data and common cause failure data, and use of plant specific data are current concerns about reliability database in Korea.

Romania

The reliability databases existing at unit 1 have to be reevaluated, so that to be included and used by the PSA level 1 project going on for the site. The project is a set of requirements of the use of PSA level 1 formulated by the Regulatory Body (CNCAN) since 1991 and reviewed in the Operating License issued in May 1999.

In this moment there is a set of 1 system being monitored, as part of the License provisions. On the other hand the Licensee is processing under COG and IAEA a project to develop reliability database for CANDU's.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Data Collection and Record Keeping for the Management of Nuclear Power Plant Ageing, Safety Series No. 50-P-3, IAEA, Vienna (1991).
- CNSC Regulatory Standard S-99 (Canada).
- ENREN 1772/96 Embalse PSA Regulatory Requirement, 1996 (Argentina).
- AR Standard 3.1.3, Criterion Curve (Argentina).
- Requirements for the PSA level 1 for Cernavoda NPP Unit 1, CNCAN, 1992.
- Cernavoda NPP unit 1 Operating License, CNCAN, May 1999.

ISSUE TITLE: Need for performance of plant-specific probabilistic safety assessments (PSA) (GL 5)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

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, including specification of a complete mitigating system failure assumption for systems responding to these events (dual failures), and a qualitative treatment of their likelihood to determine acceptable consequence guidelines. The criteria (generally in terms of dose limits) are used to evaluate the acceptability of proposed plant designs.

This approach has not only provided adequate protection against the selected design-basis accidents (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 PSAs 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, and
- 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 dependencies of both the system itself and relevant auxiliary systems,
- plant-specific component failure data,
- plant-specific analyses of human interaction, and
- plant-specific evaluation of vulnerability to external events.

The need for, and the usefulness of, PSA as a tool for integrated safety evaluation, became obvious in connection with analyses 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

PSAs 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 it out include experienced personnel who know the capabilities and limitations of the PSA , specific CANDU aspects and of the plant design and operation.

A lack of adequate PSA applications would make it more difficult to identify vulnerabilities or weaknesses in the safety balance of the plant, or to set priorities for 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:

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 assessments 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 detail and the scope of the PSA of the Argentinean NPP's were defined by the Regulatory Authority taking into account the applications mentioned above. 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, supporting regulatory interaction, improving operator training on effective areas, and identifying 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 the 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.

The Argentine Regulatory Authority applies a series of standards that take into consideration both deterministic and probabilistic issues. The most significant regulatory issues considered within the probabilistic philosophy are:

a) the establishment of a high-level probabilistic safety criterion (PSC),
b) PSA as a regulatory requirement for licensing, and
c) additional regulations of a probabilistic nature at different levels.

The PSC objective is limiting individual risk associated to accident situations that may occur in nuclear installations to the value accepted for their normal operation. The Regulatory Authority performs a follow-up of all the stages in the PSA as an initial screening, prior to its thorough evaluation. Considering the significance of attaining consensus during the initial stages of the process, such follow up ranges between the establishment of the models and methods to be used and the review of the assumed hypotheses and of the simplifications applied in the analysis. In addition to the probabilistic criteria, the Argentine Regulatory Authority is using PSA as a regulatory tool for developing risk-based inspection guidelines, for making decisions on backfitting, for performing incident analyses and for training purposes.

Probabilistic techniques are also applied in regulatory decisions related to operating plants based on PSA studies. For instance, it must be demonstrated that reliability targets assigned to safety systems or functions—consistent with the role of the safety system or functions in different accident sequences—are met during plant service. In this case, additional regulations exist for failure analysis of risk evaluation, providing additional data to the failure analysis itself. The fundamental goal of the regulatory philosophy for plant operation indicates that risks should be as low as reasonably achievable and that risks should remain below defined limits.

Canada

Although PSA is not yet a regulatory requirement in Canada, Regulatory Guidance document G-6 defines it as a part of the Safety Analysis, and it is requirement for refurbished plants.

One licensee has performed PSAs for two plants and is presently performing PSAs for each of its remaining plants. That licensee has begun to use probabilistic arguments in submissions to the CNSC. Regulatory documents on how to handle such a "blended approach" are presently being developed (P-151, Policy on Risk-Informed Decision-Making, G-152, Guidance for Risk-Informed decision-Making, and G-42, PSA Attributes for Regulatory Decision-Making).

As the two existing PSAs (Pickering-A Risk Assessment, known as PARA, and Bruce-B Risk Assessment, known as BBRA) have shown, the main risk contributors differ dramatically from one plant to the other. This confirms the principle that PSA results cannot be used generically, but only on a plant specific basis.

These PSAs have identified the importance of some plant features which, while known since commissioning, had not been deemed safety critical. This has resulted in regulatory actions requiring plant improvements. One example can be found in the conditions imposed on re-starting Pickering-A.

Atomic Eenergy of Canada Limited also performs PSAs at several stages for the plants it designs and markets. Early on in the design cycle a conceptual PSA is performed, to ensure that the design and safety goals have been met. This is especially important for a new design such as CANDU 9, where the PSA is used as a design tool and may lead to design changes. Later in the design, as the details are finalized, a “generic” PSA is performed. Near the time of handover of the plant to a customer (depending on the contract), a more detailed PSA is performed for that particular plant and site.

China

Although PSA is not yet a mandatory regulatory requirement in China, TQNPC will perform PSA for TQNPP. PSA report will be submitted to NNSA in middle of 2001. NNSA will review TQNPP PSA report during TQNPP FSAR review stage.

India

In AERB's current policy, the insights gained from PSA are considered together with those from other analyses in decision-making regarding the acceptability of the safety of the NPPs. PSA provides an estimation of risks; it also gives information on a balanced design by revealing interaction between engineered features and weak areas in a design. If PSA is carried out using standardized methodology, the state of the art technology and plant specific data it can help regulators in taking faster and consistent decisions

The application areas of PSA and the benefits accruing from use of PSA for prioritization of operational tasks by the utilities and regulatory functions with regard to the identifications of weaknesses, backfits/modifications and changes, improvement of operators training, Tech. Specs., optimization etc. have been well emphasised in the coordinated PSA development programme initiated by AERB. Periodic discussion meets are held to exchange ideas and accelerate the progress of PSA related studies.

Emphasis is on standardising the methodology and software used and having level-1 & 2 PSA for all the NPPs. AERB Manual No. SM/O-1 on probabilistic safety assessment for nuclear power plants provide guidelines to conduct PSA Level-1, 2 & 3 studied taking into account both internal and external events, CCF and Human reliability analysis.

PSA studies for specific applications like Living PSA / Risk Monitor, RCM, TS optimizations with regard to Allowed outage time (AOT) and Surveillance Test Interval (STI) and Ageing Management, RB-ISI, internal hazards to dynamic effects and man induced external hazards, operator training and accident management are being attempted gradually. AERB desires that applications for renewal of authorisation and periodic safety review (PSR), application for changes in surveillance requirements and any modification proposals from operating NPPs to accompany PSA based studies.

Presently for New Plants, regulatory requirements involve establishing target reliabilities of all safety and safety related systems and working out core damage frequency with internal events For operating plants PSA studies with plant specific failure data should ensure that reliability of plant systems/components was within the reliability target values. Instead of linking final result of each PSA based analysis to change in CDF it is being thought of to identify few intermediate level of goal, which may give a better understanding during regulatory decision-making.

AERB approach paper on PSA brings out regulation on PSA, highlighting establishment of probabilistic safety goals/criteria, desirable/mandatory regulatory requirements of PSA studies for licensing/authorization process, procedure of review etc. Limited PSA (Level 1) studies have been completed for plants at Narora, Kaiga and Madras. Detailed PSA Level-1 for Kakrapara has been accepted by AERB after review. Utility has also submitted PSA Level-2 for Kakrapar and Tarapur 3&4.

Risk-Monitor software has been developed by BARC, which has been subsequently revised with the state-of-the-art PSA methodology. The validation of the software is in the progress.

This PSAs studies are performed and submitted by utilities to AERB. There are no mandatory requirements on PSA. For the effective integration of the PSA into Decision-making, AERB policy on PSA will be prepared in the framework of “Risk-Informed Approach”.

Korea, Republic of

CANDU has a long history of PSA. However, the Canadian regulations and PSA practices and methodologies are different from those of Korea, where KINS requires their implementation. The methodology of level 1 PSA, as developed by AECL, could not fully satisfy KINS. For example, AECL did not consider the common cause failure of redundant components. As a result, consideration of common cause failures was an early requirement by KINS.

The regulatory guideline by KINS, which requires plant-specific safety evaluation using a PSA methodology, is also applied for CANDU plants. In that regulatory guideline, it is a requirement to perform level 2 PSA for new Nuclear Power Plants (NPPs) to verify plant safety against severe accidents.

PSA was widely used to identify systematically the safety of CANDU design and operation. The outstanding results with respect to the plant vulnerability come from phase-2 PSA which consists of internal event analysis, external event analysis, and source term analysis in the containment. The utility wants to update the works performed by AECL (phase-1 PSA), which includes the code development effort, such as severe accident analysis code, i.e., ISSAC. The ultimate objectives of the phase-2 PSA are to assure the plant safety improvement, and to provide the basis of CANDU severe accident management.

The results from the external event analysis show the relatively great importance with regard to the core damage and the containment integrity, which mainly comes from the over-conservatism in the seismic hazard analysis and the lack of mitigating procedures.

In order to reduce the significant risk of core damage, the following actions were recommended and implemented in the Wolsong NPPs.

- Change of test interval of shutdown cooling system
- Reinforcing the emergency water system structure bracing for the seismic event
- Establishment of fire protection program for the reactor building
- Instituting a specific procedure and training against the flood scenario

It is required to provide the plan for the ISSAC code verification and validation and to provide the implementation plan on severe accident management.

Pakistan

Probabilistic safety analysis (Level-1) is a very important part of the modern NPP safety regime, because it identifies those accident sequences, which dominate the core damage risk. Once the models have been developed they can be used dynamically, to pinpoint vulnerabilities requiring corrective action.

The work on PSA level-1 was initiated by KANUPP is mid nineties with strong technical support from IAEA. The main objective is to establish a base case PSA to be used to identify and remove weaknesses in a risk effective way. Another objective is to assess the level of the plant safety and compare it with existing standards. An implicit objective is to gain a better in-depth understanding of the plant. Level-1 PSA is also a re-licensing requirement for KANUPP.

Initial quantification and recovery analysis was completed during 1998 and PSA-1 was completed in 1999. KANUPP agreed to host a formal IAEA International Peer Review Services (IPERS) mission to review KANUPP PSA-1. But due to delay in updating of KANUPP Final Safety Analysis Report (KFSAR) project, instead of IPERS, a pre-IPERS mission was invited to review PSA-1 work in 1999. A full scale IPSART review was carried out in 2001. A PSA application project was also started from February 2002 covering the following:

- Evaluation of AOTs (allowed outage time)
- Design modifications
- Improvement of Human Performance
- Optimization of surveillance test intervals

The PSA report was issued in first quarter of 2002 and has been reviewed by Pakistan Nuclear Regulatory Authority (PNRA). The major contribution of <u>initiating events</u> is:

- Loss of Offsite Power 24%
- Loss of Instrument Air 19%
- Small LOCA 15%
- Forced Shutdown (FSD) 14%
- Secondary Side Event Group 9%

- Loss of Normal & Control Power Group 8%
- Loss of Primary Pressure Control Group 7%
- Intermediate & Large LOCA 2%
- Loss of Component Cooling Group 2%

The Regulatory body (PNRA) has endorsed the recommendations for implementation provided in the PSA-I report which are amongst others, as under:

i. An aggressive scam reduction program
ii. Improving the reliability of emergency boiler feed water
iii. Making starting of emergency boiler feed water an auto action rather than a field action
iv. Improve LOCA handling
v. Improving training viz-a-viz the loss of instrument air initiating event
vi. Increasing the charging tank capacity of the coolant
vii. Installing a third Diesel generator with capability to provide redundancy on running failures of existing two diesels

As part of PSA application project, KANUPP plans to carry out improvement based on the above recommendations in two phases:

- First phase, including items (i) to (iii), (v) & (vi), has been completed in the year 2003. After Phase-II, it is expected that the CDF will reduce to 5E-4 per year.

Other tasks of PSA applications project i.e. Evaluation of AOTs, Optimization of Surveillance Test Intervals and Improvement of Human Performance have also been completed. PSA applications report will be issued after approval. Fire PSA for KANUPP has been initiated and is scheduled to be completed by June 2006.

Romania

The basic design of a CANDU 600 is a combined approach of deterministic and probabilistic criteria. The requirements are defined at two levels:

- definition of the acceptable doses, which consider the impact on staff, population and environment
- definition of probabilistic criteria for special safety systems, safety related systems and process systems.

The two levels are similar to the Core damage frequency and Releases categories concepts from the PSA levels 1 to 3 methods. There were two possible uses for the PSA methods in regulation:

- risk informed regulation
- risk based regulation

By the basic initial safety licensing the CANDU 600 is closer to the use of risk informed regulation, with the comments that the risk could be derived based on the basic design in a two level approach as defined above. By trying to fully implement the PSA level 1 and further more PSA level 2 and 3 approaches the whole system classification and safety philosophy is to be deeply reviewed. This was the basic approach of the Regulatory Body in Romania when, after it required formally a PSA level 1 for the Cernavoda unit1 commissioning licensing, approved a mixed approach as temporary situation consisting on:

- review the reliability analyses as postulated by the system requirements and extend them to 11 systems, under the original designer responsibility, as licensing conditions
- review and update the Safety Design Matrices, under the original designer responsibility, as licensing conditions
- perform PSA level 1 study, under the Romanian Licensee responsibility, as a support to the Regulatory Body decisions in the licensing process.

An internal process was developed in the Regulatory Body to evaluate all the probabilistic analyses and to practically use them in the licensing decisions.

All the commissioning process was intensively scrutinized and evaluated using this internal process, including the decision to perform supplementary commissioning tests for the areas identified as needing this type of confirmation.

After the completion of commissioning the Operation license required that the PSA level 1 shall be completed as part of the relicensing process and now is part of the Periodical safety review developed for Unit 1. The development of external PSA and extent of PSA level 1 is also required.

The License is committed to perform the PSA level 1 for external events up to 2004. The PSA programs for the Licensee are under IAEA projects support, too.

In the meantime it was required that the start of the researches on PSA level 2 and clarification on methodologies and codes to be used has to have significant progress and partial results coincident with the PSA level 1 schedule.

For Cernavoda Unit 2 the requirements for PSA as conditions of the licensing process were formulated since 1997 and are now considered by the Licensee.

ADDITIONAL SOURCES:

- CNSC Regulatory Guidance Documents G-6, G-152, “Guidance for Risk-Informed decision-Making”, and G-42, “PSA Attributes for Regulatory Decision-Making” (Canada).
- CNSC Document P-151, “Policy on Risk-Informed Decision-Making” (Canada).
- Ontario Power Generation (OPG) Documents “Pickering-A Risk Assessment” and “Bruce-B Risk Assessment” (Canada).
- ENREN 1772/96 Embalse PSA Regulatory Requirement, 1996 (Argentina).
- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.
- Strategic Policy for Cernavoda NPP Unit 2 licensing process, CNCAN 1997.
- Requirements for the PSA level 1 for Cernavoda NPP Unit 1, CNCAN, 1992.
- Cernavoda NPP unit 1 Operating License, CNCAN, May 1999.
- Internal CNCAN guideline on the use of probabilistic analyses in the regulatory process, under approval (English version as communicated in the IAEA/NEA TCM in Paris, 1998 on the use of PSA in licensing decisions).
- ATOMIC ENERGY REGULATORY BOARD, safety guide, “Consenting Process for NPPs and Research Reactor” AERB/NPP/SG/G-1.

4.1.2 Reactor Core (RC)

ISSUE TITLE: Inadvertent dilution or precipitation of poison under low power and shutdown conditions (RC 1)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

In some units, during shut down conditions, poison is added to the moderator system to maintain the necessary sub-criticality. During extended shut downs it is required to have purification through ion-exchangers. It must be ensured that the poison is not removed under these circumstances.

It should also be noted that inadequate pH control in the moderator system could lead to gadolinium precipitation, and therefore an inadvertent addition of reactivity.

Safety significance

Inadvertent removal of poison would reduce the margin of sub-criticality.

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:

Canada

Gadolinium poison is usually added to the moderator to maintain the reactor in a “guaranteed shutdown” state. Strict administrative controls are maintained to ensure that inadvertent removal via the purification circuit does not occur. Gadolinium precipitation is prevented by maintaining an adequate pH level.

India

Originally the moderator purification system was kept isolated in Indian Nuclear Power Plants during shutdown conditions to prevent removal of poison (Boron is used as poison in 220 MWe Reactor and Boron/Gadolinium is used as poison in 540MWe Reactor). However, due to buildup of Cobalt and subsequent contamination of the equipment, the design modification in moderator purification system was carried out. This modification included two Boron saturated Ion Exchange columns which would be operated in boron saturated mode only during reactor shutdown conditions. Interlocks have been provided to prevent valving in of fresh Ion exchange columns during shutdown. Also, strict monitoring of purification flows and Neutron Log Power is done by the operator. In addition, the frequency of sampling for boron during shutdown is increased keeping in mind purification removal capacity (in case of inadvertent valving in of fresh ion exchangers). Make up of heavy water is avoided during reactor shutdown or done with appropriate boron addition, if absolutely necessary. For 540MWe Reactor, during shutdown, moderator purification system gets isolated. Continuous monitoring of Neutron signal with alarms set suitably above the current readings is done and on observation of any rise in the neutron signal unintentionally, addition of boron equivalent of 5 ppm is made mandatory as a part of Technical Specification. To avoid precipitation of Gadolinium in the moderator, if present, the pH control of moderator at 5.0-5.5 is always ensured.

Korea, Republic of

During the outages gadolinium poison (or boric acid) is used to maintain the reactor in shutdown state with sufficient negative reactivity. Inadvertent purification of boric acid is prevented by operating procedures which state that the valve connected to the ion exchanger and D2O supply line should be lock closed during the shutdown period. Opening of these valves requires the permission of the senior reactor operator.

Romania

The document on Guaranteed Shutdown State is a type "A" document, is directly referenced in the license and it is approved by CNCAN. Gadolinium poison is added to the moderator to maintain the reactor in a "Guaranteed Shutdown" state. There are in place, as per this procedure, very clear and rigurous administrative controls to ensure that inadvertent removal via the purification circuit does not occur. Gadolinium precipitation is prevented by maintaining an adequate pH level.

All these requirements are part of the regulatory practice to have hold points as part of the licensisng process for main activities (including restart after outages) during outages. The monitoring of the heat sinks and guaranteed shutdown states are important parts of this process.

ADDITIONAL SOURCES:

- Cernavoda NPP Unit 1 Guaranteed Shutdown State procedure, 1999.

ISSUE TITLE: Fuel cladding corrosion and fretting (RC 2)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

The condition of a certain very small number of fuel bundles irradiated in CANDU reactors has been observed to differ from that predicted and accounted for in design, operation, and safety analysis documentation. The fuel bundles in question have shown signs of more-than-expected degradation such as end plate cracking, spacer-pad wear, element bowing, sheath wear, bearing-pad wear, oxidation of defective fuel, and fission produce release. Element bowing, sheath strain and, disappearance of the CANLUB layer, are other fuel conditions whose impact on safety assessment require quantification.

Fuel bundle degradation depends on the reactor design, fuel channel, fuel design, fuel manufacturer, and operating conditions. Since theoretical models have been unable to correlate adequately the fuel condition to these factors, fuel and pressure tube inspections are necessary. Owing to the number of factors upon which the degradation depends, the inspection program must be extended beyond inspection of defective fuel to observe these changes.

Fuel bundle degradation is sometimes also accompanied by fretting and scratching of the pressure tube.

The effect(s) of some of this bundle degradation on reactor safety is (are) not known partially because of a lack of experiments and safety analysis methods. As such, the important fuel and fuel channel parameters to measure are difficult to identify.

Safety significance

Damaged fuel bundles can be a source of debris and 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 can induce further cladding damages, for instance by cooling channel blockage.

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:

Argentina

The main failure causes were problems related with plug – sheet welds (manufacture problems). In the sheets inspected due to be failure's suspected, fretting evidences were not observed. Nevertheless, in most of cases only were possible to determine the secondary fault but not the primary fault.

Canada

1.CNSC

Some fuel inspections have been conducted and the results submitted to the CNSC (Canada), some Canadian licensees do not have a formal process to ensure that the fuel and fuel channel conditions are identified and accounted for.

The CNSC has identified this issue as a "Generic Action Item", and issued a Position Statement to licensees giving the closure criteria and the required completion schedule.

To achieve closure, licensees are required to perform the following:

1. Implement an action plan to eliminate excessive fuel and channel degradation in acoustically active channels (where applicable).
2. Response to this closure criterion should include a rationale showing that the situation at all reactors is acceptable for long-term operation. If this cannot be shown, plans and schedules should be submitted, showing how this problem will be resolved.
3. Implement an effective formal and systematic process for integrating fuel design, fuel and channel inspection (in situ), fuel and channel laboratory examination, research, operating limits and safety analysis. This process must have the following features:
   (a) annual review (by the licensee) to demonstrate effective implementation and adequate corrective actions taken for deficiencies identified in the review;
   (b) sufficient resources for each participating group (design, inspection, examination, research, safety analysis, and operation) to ensure that the fuel condition is known and accounted for adequately;
   (c) clearly defined maximum allowable limits, under normal operation, on fuel condition in terms of sheath strain, element bowing, wear (spacer pad, bearing pad, end plate), pressure tube scratching and wear, burnup and residence time; design documentation and pressure tube fitness-for-service guidelines should be updated accordingly;
   (d) a determination, for the full range of the operating envelope, the power boost sheath failure threshold for CANLUB fuel and the chemistry effects of CANLUB on centerline temperature and fission product release;
   (e) assurance that the safety analysis accounts for the allowable fuel condition when combined with aging effects such as pressure tube creep, the effect of CANLUB in the fuel, and any chemistry effects on temperature and fission product release, including a calculation of the number of sheath failures resulting from a bounded loss of power control;
   (f) a surveillance program that demonstrates compliance with identified limits, e.g., detection of significant changes in fuel condition caused by changes in fuel fabrication and factors affecting acoustic resonance; and
   (g) allowance for AECB audits.

2.Industry

Several work packages to address the CNSC's generic action item on the Impact of Fuel Condition on Safety Analyses have been undertaken within the CANDU Owners Group (COG). In-reactor irradiation and post irradiation examination (PIE) have been used to quantify the impact of selected fuel parameters on fuel performance. These include:

(a) In-bay inspection techniques to measure fuel dimensions, non-destructive burnup determination, and improvements to visual inspection systems;

(b) Irradiation of fuels manufactured using (a) pore formers to assess the effect of fuel density, (b) high and low diameter clearances between fuel and clad, and (c) CANLUB/non-CANLUB elements; and

(c) PIE of power reactor fuels to assess the impact of reactor ageing (e.g. pressure tube creep and sag).

This series of experiments have found no evidence that these conditions are detrimental to fuel performance or safety related issues. This COG program is continuing and will continue to assess special manufacture fuel bundles and fuel from the ongoing fuel surveillance program.

COG has also funded the review and update of the Technical Specifications for 37-element CANDU fuel to ensure the design baseline is complete and open to audit.

Data from manufacture, irradiation and PIE are routinely entered into an Irradiation Fuels Database to ensure well characterized information is available to validate the computer codes used for safety analysis.

One of the Canadian utilities has developed a plan to achieve resolution and closure of all significant fuel performance issues by 2005. The plan involves verifying/establishing the basis for fuel limits, developing methods to check fuel condition, and developing a process to assess fuel condition and identify problem precursors.

India

Fuel cladding fretting does not appear to be a serious problem in India even though one failure has been observed due to this. The iodine and fission products in heat transport system and fuel failure rates are satisfactory and much below limits. Volumetric examination of more than 500 fuel channels in various Indian reactors have not indicated any defects of generic nature. However, in view of the issues raised in this document, it is planned to have a fresh review.

Korea, Republic of

Systematic cladding failure as a result of fretting induced by spacer or bearing pad is not occurred in Korea.

However, review is to be performed on the problem described in this document , if necessary.

Romania

The problems were subject of commissioning tests and verifications. There are no apparent problems on this topic in this moment and no events with safety significance encountered. However this is an aspect of possible future review based on the feedback from operation.

During operation the Licensee is performing mandatory test programs and preventive maintenance actions to support the maintaining of the control rods insertion characteristics, as confirmed by the commissioning results; we may notice also that there were no events related to this aspects for the operating period.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Power Plant Life Management Processes: Guidelines and Practices for Heavy Water Reactors, IAEA-TECDOC-1503, IAEA, Vienna (2006).
- CNSC Position Statement 94G02, "Impact of Fuel Bundle Condition on Reactor Safety".

4.1.3 Component Integrity (CI)

ISSUE TITLE: Fuel channel integrity and effect on core internals (CI 1)

ISSUE CLARIFICATION:

Description of issue

This issue is specific to NPPs with PHWR.

The coolant channels in CANDU reactors increase in length and diameter with the passage of time, due to creep, induced by temperature and irradiation. This requires periodic adjustments, which is a cumbersome operation.

Furthermore, several mechanisms have been identified as contributors to fuel channel degradation, which could lead to fuel channel failure. These mechanisms are the following:

(a) deuterium ingress into, and embrittlement of, the pressure tube (PT) material,
(b) rolled-joint cracking,
(c) fretting,
(d) formation of blisters,
(e) PT deformation, e.g. elongation and sagging, and
(f) material property changes, e.g. changes in material tensile properties, fracture toughness, and delayed hydride cracking velocity.

Finally, there are other concerns related to fuel channel integrity due to ageing and accidents particularly during loss-of-coolant or loss-of-heat sink (e.g. moderator) events. These are the following:

1. The fuel channel design life has not yet been demonstrated.

2. PT local creep rupture during the high pressure part of the large LOCA blowdown due to:

   (a) creep and annealing of irradiated PT,
   (b) thermal hydraulic temperature gradients,
   (c) temperature gradients from bearing pad contact with PT, and
   (d) temperature gradients from contact of fuel with PT, hence:

      i) more rapid ballooning at fuel bearing pad plane causing fuel element/PT contact, and

      ii) fuel element sagging, embrittling, or appendage braze melting causing contact between fuel element and PT.

3. Inadequate subcooling on calandria tube to prevent dryout and rupture upon initial contact between ballooned PT and calandria tube (CT):

   (a) contact conductance of irradiated pressure tubes and calandria tubes,
   (b) local moderator flows and temperature gradients, and
   (c) effect of fuel element to PT contact on subcooling requirement.

4. Miscellaneous:

   (a) axial creep and rupture of PT in the hot ring beside the garter spring,
   (b) heating and pullout of PT rolled joint,
   (c) feeder rupture
   (d) pullout of CT rolled joint,

(e) constrained axial expansion of fuel,
(f) moderator system reliability,
(g) local CT dryout beneath sagged PT,
(h) delayed ECCS repressurisation and PT ballooning, and
(i) transport of embrittled fuel from core.

Safety significance

1. Although the CANDU reactor design has the capability to withstand the consequences of a pressure tube rupture, designers and operators must strive to reduce the probability of pressure tube failure. Fuel channel failure consequences are severe, particularly when taking into consideration the potential for damage to other channels and/or core internals.
2. Radiation doses associated with fuel channel replacement adjustments are high.
3. Fuel channel replacement entails a very high economic as well as radiation dose penalty.

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:

Argentina

CNA-I reactor pressure vessel surveillance program was initiated in 1974, by introducing 30 test specimens distributed in the lower reflector, inside the moderator tank. In 1980, 10 additional capsules containing samples of the same type, but made of A508 class 3 material were introduced, aiming at obtaining a relative reference for neutron exposure. On the other hand, irradiated capsules were withdrawn and examined. No definite conclusion could be obtained about the pressure vessel material behaviour, due to the differences between the neutron spectra present on samples located inside the moderator tank and those spectra on the pressure vessel wall.

Due to the uncertainties associated with the previous results, Siemens - Kraftwerk Union AG carried out an irradiation program in the German VAK reactor in 1985. The experiments were carried out with probes made of the same pressure vessel basis material as well as with A508 class 3 material, simulating the conditions on the CNA-I pressure vessel wall.

Although there are some uncertainties due to the short irradiation time (high acceleration factor), irradiation carried out by Siemens - Kraftwerk Union AG indicated that 35ºC was the temperature for ductile-to-brittle transition at the end of CNA-I lifetime (32 full power years). On the other hand, preliminary results of the most unfavourable LOCA analysis showed that the pressure vessel would always be at a temperature higher than 35ºC at the end of its lifetime.

The Regulatory Body has carried out an assessment of the available information related to the reactor pressure vessel. It was concluded that there are uncertainties regarding the reactor pressure vessel integrity under certain accidental operational situations.

On the other hand, NASA has initiated an evaluation of those accidental scenarios having the most unfavourable stresses for the pressure vessel integrity (Pressurised Thermal Shock analysis), as well as a program of the necessary actions (i.e. improvement in the surveillance program), to be carried out in order to minimise such uncertainties contained in the studies.

In addition, it is important to emphasise that the design characteristics of the plant do not facilitate an optimal mixture of the injection of the emergency system water, due to the fact that the loop seal always contains cold water, which has a negative contribution to the effects of a thermal shock during the eventual occurrence of an accidental situation.

Therefore, with the purpose of ensuring that the reactor pressure vessel will continue preserving the appropriate safety margin, it has been required that the necessary measures should be taken in order to reach the year 2001 having heated the water contained in the high pressure accumulators of the core emergency cooling system.

The CNA 1 reactor cooling channels are degraded in way which could not be predicted in the original design. The above mentioned degradation consists fundamentally of both cooling channels external cover foils degradation and cooling channels sagging. Besides, due to the fact that to reduce the cooling channels corrosion and erosion effects, these components were covered with a non ferrous alloy containing 60% cobalt, commercially known as Stellite-6, so as a result of the material activation produced by the core neutronic flux, cobalt-60 is formed contributing additionally to the radiation field in certain places of the reactor building and consequently, to the increase of occupational doses. As a result of this, CNA-I normalised occupational dose is higher than in those nuclear power plants where Stellite-6 is not used.

Due to above mentioned Responsible Organisation was required to replace the all the original coolant channels by others without Stellite-6. This task is gradually going on at each programmed outage and up to now 152 channels have been replaced, representing more than 60 % of their total number.

The CANDU reactors cooling channels (pressure tubes) are degraded in way which could not be predicted in the original design. It is known that the degradation mechanisms are: hydrogen uptake, blister formation, rolled joint cracking, sag/deformation and fuel-bundle bearing-pad fretting. The Regulatory Authority issued a requirement regarding the degradation mechanisms and ageing concerns for Embalse NPP pressures tubes. Pressure tubes become susceptible to the formation of the hydride blisters when the hydrogen equivalent concentration exceeds a threshold called “blister formation threshold” (BFT).

The licensee was required to demonstrate before the end of 1998 that hydrogen equivalent concentration for all the pressure tubes is below the BFT based on the specific assessment model for Embalse NPP. The concentration and the uptake rate of equivalent hydrogen must be obtained from the scraping results. In addition, an appropriate sequence for garter springs repositioning (SLARing) must be defined according to the above mentioned results. To verify that the degradation is kept within admissible values and to take the corresponding remedial actions, in service inspections and tests is carried out. For that reason, an in service inspection programme was implemented and put to practice in 1986 with the objective of repositioning all garter-springs and inspection all pressure tubes.

It is important to highlight that except for failures in A14 and L12 pressure tubes caused by anomalies in the LIM RAM tool used for the repositioning of the garter springs during 1995 Embalse NPP programmed shutdown (see significant events), no other anomaly in relation with the pressure tubes was recorded.

The criteria to select the different pressure tubes that were inspected had their basis on: 1) avoiding the operation of the power station when there was contact between pressure tubes and calandria tubes and with pressure tubes predicting level of hydrogen equivalent or superior to the one during the initiation of the zirconium hydride blister formation process (blister formation threshold), and 2) considering the deformation of the pressure tubes which might prevent the use of the repositioning tool or the movement of the garter springs.

Besides, as a consequence of the failure in the A14 and L12 pressure tubes, a programme for the assessment of the material extracted from such tubes has been implemented in order to determine the level of hydrogen equivalent and the deuterium intake velocity. Apart from this, the future possibility to implement sample scraping in some pressure tubes is being analysed.

The selection criteria for the pressure tubes inspected during the last planned outage were: Priority list proposed by AECL Memo: "Embalse spacer repositioning proposal" - E. G. Price - Feb. 94, and Changes due to new predictive hydrogen intake model taking into account the operating experience

In June 1998 AECL completed a pressure tube blister susceptibility assessment to evaluate the potential for hydride blister formation in the pressure tubes (PT) of Embalse NPP (CNE). Several deuterium ingress cases were analyzed using deuterium concentration data from other CANDU 6 reactors, concluding that for each case PTs could potentially reach blister threshold formation (BTF) before the CNE October / November 1998 planned outage. Therefore, were required to complete a long term assessment of hydride blister susceptibility by monitoring deuterium uptake in PTs at Embalse.

Based on the above mentioned assessment, to avoid PT / calandria tube (CT) contact, had been improved the garter springs repositioning program schedule on going during planned outages.

Regarding the above mentioned considerations, the Nuclear Regulatory Authority (ARN) required to utility to constrain CNE operation until fulfil the following:

(1) Demonstrate that there are not PTs containing H/D over BTF using a specific CNE model to determine deuterium content and uptake. The model had been obtained from scrapping results performed during last planned outage. Besides, until fulfils its requirements into the next three months, additional measures during operation for PT failures early detection were recommended.

(2) Submit a new garter springs repositioning program for remanding PTs, according to model results

The utility fulfils the above mentioned requirement as follow:

During the October 1998 CNE planned outage had been implemented the in-core scrape sampling program, which involved a total of ten PTs that were sampled after about 109,600 EFPH equivalent full power hours (EFPH) or 119,400 hot hours. The purpose of PT scrape sampling was to monitor the uptake of deuterium. The samples were selected by the licensee from a list of recommended candidate channels issued by AECL to provide an indication of the variation in deuterium concentration along the length of the PTs.

Scrape sampling was performed using the wet scrape tooling in which a separate tool is delivered to each sampling location in sequence by the fuelling machines and cutting is performed fully immersed in the PHTS coolant under both temperature and flow corresponding to shutdown condition.

The sample analyses were performed by using two techniques. The non destructive measurement of the temperature corresponding to the Terminal Solid Solubility for hydride Dissolution (TSSD) using Differential Scanning Calorimetry (DSC) and Hot Vacuum Extraction – Mass Spectrometry (HVEMS). DSC is normally only performed to provide consistency check on the HVEMS results.

The assessment of CNE scrape sampling results, performed in October 1998, indicated lower deuterium uptake rates in comparison with previous assessment performed using deuterium concentration data from other CANDU 6 reactors. There is a difference in deuterium uptake behaviour between the two PHTS flow loops (higher in loop 1 than loop 2). The differences still are being investigated.

The assessment of CNE scrape sampling results let to perform studies with CNE specifics data determining that until reactor reach 160,800 EFPH, that means approximately October 2004, conditions to blisters formation would not be given.

Canada

Activities by the regulatory body:

(a) To address deuterium ingress, licensees are required to:

- justify operation with Hydrogen Equivalent (Heq) concentrations predicted to the end of the license period,
- periodically monitor pressure tube hydrogen levels in situ by scrape sampling, and
- periodically remove and examine surveillance tubes for deuterium and hydrogen concentration.

(b) To address rolled-joint cracking, licensees are required to:

- establish shutdown procedures that avoid fast fracture,
- establish a Leak Detection System that is active at all times during operation,
- monitor hydriding by period removal and surveillance examination of rolled joints, and
- monitor crack detection through in-service inspection (ISI).

(c) To address fretting, licensees are required to:

- determine the population of at risk by in-service inspection program,
- limit the heatup/cooldown cycles and operating times based on the fret population, and
- limit operation if the probability of initiating a crack is high.

(d) To address blister formation, licensees are required to:

- ensure that reactors are not operated with a PT which has a detected blister in, or which is in contact with, its CT, and which meets or exceeds the current threshold criterion for hydrogen concentration for blister formation under continuous operation; this is achieved by maintenance programs that consist of:
- prediction and monitoring of Heq concentrations,
- predictions and monitoring of PT-to-CT contact, and
- removal of contact by repositioning garter springs before the blister formation threshold is reached.

(e) To address PT deformation, the licensees are required to:

- measure deformation through a comprehensive in-service inspection program,
- shift fuel channels to keep them on their bearings as they elongate,
- reposition garter springs to keep sagged tubes out-of-contact, and
- replace PTs and CTs that have sagged to the point where they are difficult to refuel.

(f) To address material property changes, licensees are required to:

- remove PTs for periodic testing from CANDU lead units, and
- carry out tests to determine material tensile properties, fracture toughness, and delayed hydride cracking velocity.

Activities by the industry:

The Canadian nuclear industry has identified and is acting upon the following key elements of a "Pressure tube aging management program":

(a) understanding pressure tube aging (material properties, operating conditions, aging mechanisms, condition indicators, consequences of aging-related degradation and failures, operating experience, research and development),
(b) definition of an aging management program (coordinating activities, documentation, program optimization),
(c) managing aging mechanisms (following procedures, chemistry control of water and annulus gas),
(d) inspection, monitoring and assessment (leak rates, fitness for service assessment), and
(e) maintenance/replacement (mitigation of tube degradation, replacement).

China

Pressure Tube (PT) integrity issue has been identified by NNSA during TQNPP PSAR review. It is required by NNSA that the designer should demonstrate the efficient measures have been taken to guarantee the integrity of PT. The measures have been reviewed by NNSA.

According operation experience of Canada, CANDU fuel channels have degraded in different ways. The problems associated with them are: rolled joint cracking, hydrogen uptake (ingress), formation of blisters, and deformation.

In TQNPP PSAR review, NNSA require that the licensee set up measures to avoid fast fracture due to rolled-joint cracking. There is an Annulus Gas System (AGS) ,which is a leak detection system ,that is active at all time. It can detect any leakage from pressure tubes to meet Leak Before Break (LBB) requirement. In AGS, dry CO_2 gas is supplied to the annuli between the pressure tubes and calandria tubes. The dewpoint and rate of change of dewpoint for the recirculated gas is continuously monitored. Sampling and analyzing the gas for moisture contents provides a means for leak detection from pressure tube. The extent of hydriding can be monitored by periodic removal of rolled joints.

In the short term, hydrogen uptake could influence the formation of blisters. We do not expect hydrogen uptake to reach levels high enough to cause embrittlement during operation. NNSA require that the licensee set up measures to ensure that pressure tubes remain ductile.

In CANDU reactors, garter springs keep pressure tubes from contacting their calandria tubes. In early design of CANDU, the garter springs can move during operation. The movement of garter springs allowing the pressure tubes to sag, contact the calandria tubes, and form blisters can crack and eventually fail the tube. To maintain some element of the defense-in-depth approach, NNSA require that the licensee set up measures to avoid the contact of pressure tubes and calandria tubes. In TQNPP, the tight garter springs will be used to avoid its moving.

According to operation experience of CANDU, there are 3 types of deformation of pressure tubes:

- diametrical creep and wall thinning;
- channel sagging; and
- axial elongation.

Although a lot of researches have been done in Canada, there are still many uncertainties associated with the long-term consequences of the above degradation mechanisms. NNSA require that the licensee to increase the level of management by inspecting and testing during the operation in order to pressure tube integrity.

The material (Zr-2.5Nb) of PTs used in TQNPP was purchased from Russia. It is first time that the Russian material is used in CANDU. NNSA required the utility perform the radiation performance test of TQNPP PT material. The testing results should be provided to NNSA for review.

It's required that the hydrogen content of the material should be strictly controlled during manufacture stage.

India

India is operating three reactors with Zircaloy-2 pressure tubes and ten reactors with Zirconium- 2.5% Niobium Pressure tubes. Reactors under construction are Zr-2.5% Nb pressure tube with four tight fitting garter springs.

First Generation Reactors of RAPS-1 & 2 and MAPS-1&2 have open annulus between calandria tube and pressure tube with air flowing through the annulus. The reactor RAPS-1, RAPS-2 (before retubing and MAPS-1 & 2 have two loose fit spacers which separate the pressure tube from calandria tube. NAPS 1 & 2 and KAPS-1 have four loose fit spacers whereas from KAPS-2 onwards (including retubed RAPS-2) have four tightfit spacers. From NAPS-1 onwards all reactors have close annulus where dry CO_2 is purged continuously to help detection of any leakage through the pressure tube. RAPS # 2 has an unique combination of zirconium - niobium pressure tubes with an open annulus and thus faced some unique challenges.

The continuous purge of CO_2 also helps in avoiding accumulation of Deuterium/Hydrogen in the annulus.

History of operation:

The first PHWR commissioned in India was RAPS-1 which was made critical in 1973. This reactor operated at low power for substantial period of time due to crack in end shield and therefore has logged in only 7.2 EFPYs till now. RAPS unit-2 was taken for en-masse coolant channel replacement in 1994 at 8.5 EFPYs & has operated since 6/6/1998 after en-masse coolant channel replacement for 5.85 EFPY.

MAPS-1 operated successfully for 10.1 EFPY till 20/08/2003 and its en-masse coolant channel replacement program is in progress. MAPS-2 has been operating since 23-07-2003 after coolant channel replacement and has completed 1.68 EFPY so far.

NAPS-1 has been taken for coolant channel rehabilitation job after completion of 9.69 EFPY & NAPS-2 has completed 9.35 EFPY so far and KAPS-1&2 are operating after 8.72 and 8.52 EFPYs respectively. Kaiga-1 & 2 are operating after completion of 3.75 & 4.36 EFPYs.

RAPS-3 & 4 have completed 4.34 EFPY & 3.85 EFPY respectively as on 30.09.05.

We have not encountered any leakage from pressure tube or rolled joint so far except a very small leak in one tube in MAPS which was attributed to broken moderator inlet manifold. No abnormal axial creep or creep sag was noticed in any channel so far.

We have also not encountered any arrest of the channel on bearing sleeve/journal ring.

Coolant channel Replacement in RAPS-2:

Coolant channel replacement in RAPS-2 was completed by indigenous technology using indigenously developed tooling. The entire operation was completed in 24 months time at total man-rem consumption 836.9. There was no unusual occurrence or over exposure during the entire coolant channel replacement activity.

Coolant channel life management program:

Coolant channel life management programme is of vital importance and in India all its aspects including its manufacture, installation, operation including chemistry control, inspection, R&D,

analysis, maintenance of data etc are given due importance and all international practices are incorporated.

Coolant channel Inspection:

Extensive inspection campaigns were undertaken in RAPS-2 (before en-mass coolant channel replacement), and subsequently in MAPS reactors. So far 900 channels have been inspected. Indigenously developed inspection equipment and technology was deployed for the inspection. The acceptance criterion for ultrasonic was limited to 3% of wall thickness defect notch. Line focus was used for ultrasonic flaw detection.

Eddy current technique was deployed for detection of inner surface flaw. No surface flaw was detected in any pressure tube in Indian PHWRs contrary to debris and bearing pad fretting reported by other countries. The channel inspection tool is also used for measuring the gap profile between calandria tube and coolant channel as well as to detect position and tilt of garter springs. The Non-Intrusive Vibration Detection Technique (NIVT) based on vibration diagnostics has been developed and validated. This technique helps in short-listing the channels where ‘contact cannot be ruled out’ and thereby reduces the scope of inspection. NIVT are used in RAPS/MAPS to screen out possibility of contacting channel or otherwise.

Coolant channel scraping:

India has also developed scraping technology for dry as well as wet channel.

10 channels have been scraped in last two years in MAPS unit-1. The Hydrogen analysis technology in scrape samples has also been matured. Differential Scanning Calorometry technique is deployed for scrape sample analysis.

Garter Spring Repositioning:

India has also developed dry as well as wet channel garter spring repositioning system. 45 channels have been taken up for garter spring repositioning in three campaigns. All these channels are operating safely.

Creep Measurement:

India has indigenously developed tool for channel axial creep measurement by mounting T-MAC tool on fuelling machine.

Extensive program for channel creep analysis and channel creep adjustment is in force in all PHWRs. We have provision for arresting missile generated due to guillotine rupture of pressure tube in our design. 6mm gap for coolant channel creep can be accommodated by the location of fasteners on the studs. Thus creep adjustment is required to be undertaken in every annual shut down based on the actual creep. We have encountered recently some trouble of stud bending in KAPS-1 & 2 which is probably because of movement of fasteners in service. This issue is under investigation. Creep adjustment is however consuming lot of manrems. Larger adjustments & decrease of frequency is being assessed. Experience from other Member States would be useful.

Post Irradiation Examination (PIE) Program:

Realizing the fact that we have to operate power reactors with Zircaloy-2 pressure tubes in 6 reactors for several more years, extensive post irradiation examination program was undertaken for Zr-2 pressure tubes.

Tubes from RAPS-1&2 and MAPS-1&2 were removed at various EFPYs and full length tube was subjected to PIE examination. This included Hydrogen morphology, material properties, blister detection, changes in dimensions, oxide layer thickness and its characteristics etc. 33 pressure tubes have been subjected to PIE examination so far. Only one channel removed from RAPS-2 at 8.5 EFPYS indicated 4 blisters. Maximum size of blister was 0.4 mm. This channel had initial hydrogen 24 ppm and was contacting towards outlet end. This blister was benign, as the material adjacent to blister was ductile.

Analytical Tools:

We have computer codes, SCAPCA, HYCON, BLIST and CEAL which give analytical assessment of the channel sag profile, Hydrogen concentration in pressure tube along the length of the channel, blister assessment and assurance for LBB. The analytical codes are in a position to assess the temperature at the CT/PT contact location and are also able to assess the temperature gradient and Hydrogen diffusion rate. These codes are capable of giving assessment of channel for various EFPYs of operation.

R & D Programs:

Extensive R & D program has been undertaken for Zr-2 and Zr-Nb pressure tubes even though the existing technology is pretty matured. This includes irradiation in high flux facilities. Blister formation and detection technology, flaw detection technology, Oxide layer thickness.

Safety issues relating to coolant channel failure:

In our analysis we have assumed simultaneous rupture of one pressure tubes and calandria tubes. Integrity of neighboring channels has been assessed. Capability of regulation and protection systems to handle such incidence is demonstrated.

Hot pressurization scheme has been implemented in all PHWRs.

Procedure for handling LOCA due to coolant channel rupture exist and operators are given frequent training.

Extensive mock-up was done for 0 clearance rolled joint which has helped in maintaining integrity of rolled joint of Zr-Nb pressure tubes.

Regulatory Approach:

AERB of India has adopted a cautious approach for coolant channel life assessment. They not only monitor the safety assessment but also keeps close watch on R & D programs.

Fitness for service criteria has been evolved for Zr-2 and Zr-Nb pressure tubes. Channel having any ET or UT indication, having contact or having excessive creep are having evaluation and acceptance methodology enforce.

Zircoloy-2 pressure tube having contact can be left in service till computed 0.2mm deep blister is reached. Any such channel will be considered fit for service only if time between 0.2 and 0.65 mm blister is more than 1 EFPY.

Korea, Republic of

Operating Guideline for Pressure Tube

As a conditional item for Construction Permit (CP) of Wolsong Unit 2, KINS required in 1992 for utility to provide recommended PHTS operating guidelines based on the latest knowledge of the Delayed Hydride Cracking (DHC) and on the Leak Before Break (LBB) analysis of pressure tube.

The guidelines were submitted to KINS in May 1995 after two years of preparation by AECL and KEPCO.

These guidelines cover both normal operating procedures and those to be used in case of pressure tube leaking. Therefore, it provides technical guidelines for the response of the station operator to indication of moisture in the Annulus Gas System (AGS) and for the preparation of operating procedure. Leak detection, leak confirmation, shutdown to Zero Power Hot (ZPH), leak search, and criteria for initiation of reactor cooldown and depressurization are covered.

The guidelines were reviewed by KINS and approved. These guidelines are expected to be applied at Wolsong Units 2,3,4 as well as at Wolsong Unit 1 as a backfit. Then the potential threats to pressure tube structural integrity posed by DHC will be minimized.

Improvement of AGS abnormal procedure

In AGS the CO2 gas is supplied through 4 inlet headers which have 11 strings respectively. The 380 fuel channel annuli are connected in a series-parallel arrangement of these 44 inlet tubes of AGS. A moisture level, showing a continuous tendency to increase, indicates a leak into the system from a calandria tube or a pressure tube. As a method to locate the leak source in pressure tube, one by one elimination technique was proposed in PSAR of Wolsong Units 2, 3 & 4: Three out of four header valves are closed at one time and moisture level is observed. This will narrow the location down to one the four headers. Once the header giving high moisture reading is identified, the same procedure is applied through each of 11 inlets in turn by closing the rotameter outlet needle valves in the other ten lines.

KINS found it to be too time-consuming. Instead of this one by one elimination technique, KINS requested to apply the dichotomization process: The half of the concerned lines are closed at one time until narrowing the location down to one. This process could largely reduce the time necessary to identify leak location.

This is incorporated in the abnormal procedure of AGS in Wolsong Units 2, 3 & 4. And the application of this procedure to Wolsong Unit 1 will be also requested.

Pakistan

First Fuel Channel Integrity Assessment (FCIA) was conducted in 1993. In-service inspection of eight fuel channels and the metallurgical examination and testing of fuel channel G-12 in the FCIA-93 campaign did not detect any abnormality. It revealed that the dimensional changes in the pressure tube were small, the deuterium ingress had been low, mechanical properties were satisfactory and it was predicted that these would change little with further service. Based on these facts, it was concluded that all the pressure tubes were probably in good condition and suitable for at least 5 years service and probably a further 10 years service from 1993.

It was, however, pointed out in FCIA-93 that uncertainty in the condition and location of the garter springs and lack of enough data representative of KANUPP fuel channel material were the areas that needed to be addressed to in future FCIA's.

The 2nd Fuel Channel Integrity Assessment (FCIA-2003) was carried out on the results and recommendations of FCIA-93. FCIA-2003 addressed both the general aging mechanism associated with CANDU reactors, and specifically addressed the concerns expressed in FCIA-93. In-Service Inspection (ISI) of seven fuel channels was carried out in FCIA-2003 along with hydrogen/ deuterium sampling, full scale burst testing and material testing of removed pressure tube from fuel channel G-12. Three of the channels inspected in FCIA-93 were again inspected in FCIA-2003 to establish aging trend. An evaluation period ending at 142,236 EFPH (equivalent to 15 years of service assuming capacity factor shall remain what KANUPP has achieved in its initial 30 years of operation) was selected for the assessment of service life of KANUPP fuel channel.

The campaign of FCIA-2003 has revealed following:

(i) In the inspected channels, garter springs are believed to be intact and found to be at or close to their design locations and as such garter spring in other channels could be believed to be at their design location.

(ii) Changes in fuel channel elongation and wall thickness are not expected to be a concern for KANUPP up to about 142,000 EFPH the anticipated time of plant operating hours in the PLEX period. The F-15 channel does not appear to be elongating although no operational problem in this fuel channel (including on power fuelling) is reported.

(iii) Gap measurements between PT and CT in each channel has shown that one channel, G-09, is in contact near the outlet end of the channel. However, the blister assessment determined that the blister formation threshold (BFT) would not likely to exceed up to about 142,000 EFPH i.e. during 15 years of extended plant life.

(iv) The measured hydrogen concentrations are variable, but consistent with as-manufactured values. The deuterium concentrations follow expected trends of increasing content from inlet to outlet and increasing rates with temperature.

(v) The measured axial DHC velocities are within the 95% confidence intervals for cold-worked Zr-2.5% Nb material and that the time of 15 hrs. for a through leak to reach CCL is still a conservative figure.

(vi) The fracture toughness measured on burst specimens of the G-12 pressure tube agrees with the results from small specimens. This confirms critical stress intensity factors that are only very slightly greater than that given by the lower bound curve for cold-worked Zr-2.5% Nb materials for some conditions.

(vii) Predictions in PT diameter increase and location of in-service flaws requiring disposition being resolved in reference to future inspection which is proposed to be as early as possible but well within ten years period.

In the light of above observation of FCIA-2003, KANUPP would prepare a more detailed plan for the assessment and inspection of its fuel channel for its continued and safe operation.

Romania

The issue is of high importance for licensees and regulatory Body. It was also subject of R&D activities starting from the moment the knowledge transfer in 1978 was initiated. The pressure boundary for the primary system includes all the main and auxiliary systems related to it. The ISI program are defined and going on. The results of the first channel inspection are being evaluated and their conclusions are to be reflected in the PIPD review (program of the periodic inspection for pressurized systems). However it is premature to have extensive results and comments and it is sure that further revisions of this document will include them. The operational procedures for Annulus Gas

System, as basis for a LBB type of approach to be used in an early detection and indication on possible failures of the fuel channel is being used as part of the operational procedures.

The use of the fitness for service concept was started as part of the first pressure tubes inspection performed for unit 1 in 1999. On the other hand there are actions taken for defining the basic approaches of the regulatory body safety concerns as part of the process of ensuring the plant life-time. These actions will be integrated in the first set of requirements to be issued by the Regulatory Body for the Periodical safety review for Cernavoda NPP unit1, in May 2001.

The Licensee will complete the program related to the Aging/Life Time Management suported by an IAEA project and this topic is part of it.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: CANDU Pressure Tubes, IAEA-TECDOC-1037, IAEA, Vienna (1998).
- ASME Section XI and Section III, Subsection NB3200.
- Canadian Standards CSA-N285.4, N285.2, N285.6.1, N285.6.6, N285.6.7.
- ENREN 145/97, RQ4, Regulatory Requirement Pressure Tubes damages, 1997 (Argentina).
- ENREN 449/97, IR Pressure tubes Requirements, 1997 (Argentina).
- ARN 1685/98, RQ-CNE_12, Regulatory Requirement, Pressure Tube Blisters thresholds, 1998 (Argentina).
- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.
- Strategic Policy for Cernavoda NPP Unit 2 licensing process, CNCAN 1997.

ISSUE TITLE: Deterioration of core internals (CI 2)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

The reactor internals include the control rod guide tubes, calandria and pressure tubes/fuel channels, and moderator parts. The following ageing mechanisms have been observed:

1. Calandria tubes have the potential to contact the liquid poison injection shutdown system nozzles (LISS nozzles) and/or the horizontal flux detector units, due to creep during the design life.
2. Fretting due to flow induced vibration must be evaluated

Safety significance

The degradation mechanisms could lead, if uncorrected, to operation outside design conditions. Deterioration of core internals could affect shutdown system capability and/or coolant distribution and flow, and could cause other damage through loose parts. Such degradations are usually slow allowing time for detection and corrections.

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:

Argentina

Ageing problems were detected in the internals of Atucha 1 (see RC2 and CI1), however at present there is no evidence of internals deterioration of Embalse NPP. However, Interaction between designer (AECL) and utility specialists´ had begun to make measurements of the distance (space) between the liquid injection poison system nozzles´ and the calandria tubes. In principle, it is scheduled get then first measurements during the 2002 plant scheduled outage. Preceding from measurements performed in other CANDU plants shows spaces greater than estimated through specific calculus models´.

Canada

Canadian Utilities have instituted ageing management programmes which include inspections aimed at identifying any degradation in core internals (see also Issue CI 1).

The LISS nozzles have been repositioned slightly so that contact with LISS nozzles will not occur during the lifetime of the channel.

The sag of pressure tubes has been fully accounted for in new designs.

India

After the incident of calandria tube leak at Douglas Point Nuclear Generator Station in Canada, due to fretting caused by adjacent booster rod flow tube due to flow induced vibration, extreme care is being taken to ensure maintenance of proper flows through the flow tubes. While reactor trip is there for low flow, there are high flow annunciations. Special anti-vibration locking arrangements were made on the globe valves that are throttled to ensure they do not open or close during reactor operation.

Due to moderator inlet manifold failure at the Madras Atomic Power Station (MAPS), there was leak from adjacent calandria tube. AERB have asked the licensee to conduct periodic inspection by TV camera to guarantee continued safe operation but the units were derated to 75%FP. In addition detailed review has been done of all the safety aspects related to operation of MAPS units in the new configuration after inlet manifold failures.

Subsequently during enmasse coolant channel replacement and safety upgradation, three numbers of sparger tubes has been introduced in both the units at MAPS. These sparger tubes serve the purpose of moderator inlet flow distribution in calandria. AERB permitted both the units to operate at full power after this modification.The OPRD leak on calandria of Rajasthan Unit 1 (similar to the leak that occurred at Douglas Point Station) has been repaired by Indium sealing. For other stations, OPRD have been relocated in accessible areas.

Korea, Republic of

In Wolsong Unit 1, the measurement of gap between Calandria tubes and Liquified Injection Nozzles (LINs) has not been performed yet since initial operation in 1983. Through the licensing review for Wolsong Unit 2, 3, and 4, it was concluded that the contact of Calandria tubes with LINs is to be potential threat to reactor safety, an some counter-measure to avoid such contact should be necessary specially for the oldest CANDU reactor in KOREA such as Wolsong Unit 1. As a first step, an imminent research on the measurement and prediction of the gap was requested by the regulatory body. In response to this KEPCO proposed to do an extensive research program to cope with some anticipated safety problems.

KINS required that KEPCO provide an appropriate measures to avoid safety problem from such contact. As a first step, it was recommended that in Wolsong Unit 1, which is expected to be affected by that contact, a measurement of the gap between Calandria tubes and LINs be carried out.

KEPCO made an instrument to measure the gap between Calandria tubes and LINs through viewing ports and measured the gaps by using ultrasonic waves for Wolsong Unit 1.

KINS reviewed the results and concluded that there is a possibility to contact the Calandria tubes with LINs within design life. So KINS recommended that KEPCO measure the exact gap reducing rate, re-measure the gap for the estimation of contact time in good time and consider an some counter-measure to avoid such contact.

Romania

The aging and radiation effects on the core internals were not considered in the design basis. However many aspects of this type as defined in the initial licensing basis for unit 1 (in operation) are subject to the Strategic Policy on Safety and are under review to be integrated in the first Periodical Safety Review program requirements to be issued in May 2001. The aspects of aging and radiation effects and their incorporation in the reviewed inspection programs are part of the periodical safety review. These aspects include also in our view those ones related to obsolence and/or moral aging. The results of the programs will be included in further reviews of this document.

The Licensee will complete the program related to the Aging/Life Time Management suported by an IAEA project and this topic is part of it.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: CANDU Pressure Tubes, IAEA-TECDOC-1037, IAEA, Vienna (1998).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: CANDU Reactor Assemblies, IAEA-TECDOC-1197, IAEA, Vienna (2001).
- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.
- Strategic Policy for Cernavoda NPP Unit 2 licensing process, CNCAN 1997.

ISSUE TITLE: SG tube integrity (CI 3)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with PWR.

Concerns relating to the steam generator (SG) tube integrity stem from the fact that the SG tubing is subject to a variety of corrosion and mechanically induced degradation mechanisms that are widespread throughout power plants. These degradation mechanisms can impair tube integrity if they are not managed effectively. Recent years have seen leaks in CANDU plants due to steam generator tube degradation by mechanisms including stress corrosion cracking (SCC), intergranular attack (IGA), pitting, fretting and fatigue. In addition to tube corrosion, corrosion of internal structures such as supports, and fouling of the primary and secondary sides of the steam generator have also led to performance degradation.

A steam generator tube rupture would lead to a primary-to-secondary leakage which could affect fuel cooling and the confinement of radioactive materials, and to unplanned outages. Also, fouling of steam generator tubes results in a degradation in thermal performance which could lead to plant derating. However, there has been no case of a steam generator tube rupture in CANDU NPPs.

Safety significance

Steam generator tube ruptures cause a loss of primary coolant inventory and, potentially release of small amounts of coolant outside containment. Maintaining the integrity of SG tubes is a safety-related issue. However, failure of a single steam generator tube, or even few tubes, would not be a serious safety-related event in a CANDU reactor. The leakage from a ruptured tube is within makeup capacity of the primary heat transport system, so that as long as the operator takes the correct actions, the off-site consequences will be negligible. However, assurance that no tubes deteriorate to the point where their integrity could be seriously breached as result of potential accidents, and that any leakage caused by such an accident will be small enough to be inconsequential, can only be obtained through detailed monitoring and management of steam generator condition.

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:

Argentina

The steam generator tube material is Incoloy-800 in both Argentine nuclear power plants Atucha I NPP and Embalse NPP. The two NPPs uses eddy current inspection methods.

Atucha 1 NPP: Steam generator tube integrity is controlled through the in-service inspection program which is based on the ASME Code, Section 11. The ASME Code does not establish the inspection frequency of steam generator tubes but the in-service inspection program defines it as 100% (3947 tubes per steam generator) once every two years. Currently, 225 tubes (from 7,894 initially existing steam generator tubes) have been plugged. Successful lancing was made in Atucha I NPP improving steam generators’ performance.

Embalse NPP (CNE) steam generator tube integrity is monitored through the in-service inspection program which is based on the standard “CAN/CSA - N285.4-94 PERIODIC INSPECTION OF CANDU NUCLEAR POWER PLANT COMPONENT”. This standard requires that the number of tubes periodically inspected in a single – unit power plant must be a minimum of 10 % of the total number of tubes in one steam generator, chosen from those in the inaugural sample. Nevertheless, according were established in the CNE in service inspection program, always were inspected more tubes than specified by this standard and are inspected 100% tubes (3,542 tubes) of each steam generator once in two years. At present were plugged 32 from 14,168 initially existing steam generator tubes. The latter is considered as a good performance.

Need to perform both lancing and magnetite cleaning processes in next scheduled outages are being evaluated. It is important to highlight that neither lancing nor magnetite cleaning processes has been performed yet.

Canada

As part of safe operation, Canadian utilities are required to develop management processes to implement SG integrity programs. Steam generator tubes should be inspected according to the periodic inspection program (PIP) as per the requirements and acceptance criteria in the CSA standards. In addition, in-service-inspections (ISI) plans are developed to manage the ongoing in-service degradations discovered during PIP inspections. CSA N285.4 requires: (a) pre-service inspection of at least 25% of the tubes in each steam generator, and (b) sampling of 10% of the tubes in one steam generator at five-year intervals thereafter. To monitor known degradation, most Canadian plants are now following augmented in-service inspection programmes which are far beyond minimal requirements set by the CSA standards.

Canadian utilities have provided an action plan for continued assessment of SG tube degradation. While the action plan provides a strategy to monitor the degradation, it does not necessarily include plans for mitigating or fixing the problem. One of the key elements of action plans is development of comprehensive fitness-for-service guidelines (FFSG) and a Life Cycle Management Plans (LCMP). These guidelines provide data to determine the life of existing steam generator tubes and supports. On the other side, information generated through the LCMP’s application will provide a good base on which to assess continued fitness for service for steam generators throughout the industry in Canada. Degradation-specific management requires detailed knowledge of the specific nature and severity of the flaws present in a given steam generator. This detailed knowledge, in turn, requires that robust methods be applied or developed for survey inspections, so flaws can be reliably detected and correctly characterized.

The conventional eddy current (EC) bobbin coil is currently the principal tool used by Canadian utilities to detect flaws in steam generator tubing. Bobbin coils permit fast inspection rates and have been reasonably reliable for the types of flaws generally seen in the past. However, their effectiveness for some flaw types, such as cracks in dents, is questionable, and it is now generally recognized that they are relatively ineffective for circumferential IGSCC. Supplemental probes of varying design (rotating pancake coils, pancake arrays, ultrasonic probes, etc.) are employed to help resolve questionable bobbin-coil indications or generally improve inspections. Ontario Power Generation (OPG) and Atomic Energy of Canada Ltd. (AECL) have developed a specialized NDE probe (known as C-3 probe) for tight circumferential cracks. Improving the lower limit of detection of localized corrosion, in particular circumferential cracks, means earlier detection and more time for remedial action and implementation of a preventive maintenance strategy. Increased probe sensitivity implies more accurate measurements of crack propagation rates, and better diagnosis of the degradation. Both are essential to determine the effect of remedial actions and possibly the solution. Some Canadian utilities have been using advanced probes for years.

From the regulatory point of view, the Canadian Nuclear Safety Commission attempts to focus on the long term endpoint, on rejection criteria (i.e., conditions beyond which operation will not continue), problem mitigation and solution. Examples of such outcomes include the request in the case of Bruce 2 for a planned shutdown when SG flaws exceeded the acceptance criteria, chemical cleaning of Pickering steam generators, which appears to have eliminated initiation of pitting, and installation of antivibration bars at Bruce B which have shown reduced fretting wear.

India

In India, the steam generators (SG) used in older stations, i.e. Rajasthan and Madras Units are hairpin type and for the later units steam generator are of mushroom type. While mushroom type steam generator have good arrangement for lancing, blow down, in-service inspection, etc. hairpin type SG do not have these features. Until 1995 there were no steam generator system failures in Indian PHWRs. The chemistry control is extremely satisfactory and these are part of technical specification in India. However, after 1995 there have been failures in steam generators of older units i.e. at Madras and also the newer units i.e. at Narora.and Kakrapar Improvement in blow-down capability, sludge removal by chemical methods, replacement of tube bundles, installation of viewing windows, etc. are being implemented.

Three incidents of SG tube leak occurred in MAPS during 1995-1997. The tube leak had occurred under the sludge near the tube sheet. The chemical analysis of the sludge indicated iron and copper as major metallic constituents. The defective tube was examined using various techniques and the result indicated that under deposit pitting corrosion was main cause of failure. Following the incidents in MAPS, both units of RAPS having similar steam generators were partially inspected for assessment of their health by cutting and removing one HX from hairpin steam generators. Sludge deposits were small and sludge was non-adherent. The overall health of RAPS steam generators appears to be better than that in MAPS. This may be due to the use of lake water in RAPS as against seawater used in MAPS. MAPS#1 & 2 steam generators were replaced in the year 2005 and 2004 respectively during en-mass coolant channel replacement.

Narora and Kakrapar stations also experienced steam generator tube failure in the recent past. The cause of these failures was established after cutting and examining the failed tube. This indicated that the failure has taken place due to foreign material entering and hitting the tube. Mushroom type steam generators are subjected to ECT, as per “In service inspection” programme and results of ECT has indicated no major deterioration.

In Indian PHWRs the detection of SG tube leak is primarily done by feed water sampling for the presence of tritium. This method gives a sensitivity of less than 1 Kg / day leakage. However to identify the leaky SG, the gross beta-gamma detectors which detect the presence of N-14 and O-16 in SG blowdown water is used.

Korea, Republic of

Sludge lancing hole was not considered in early design phase, and the hole was not installed in all Wolsong Units. The sludge lancing of Wolsong Unit 1 has not been performed since initial operation in 1983. At present, Wolsong Unit 1 has approximately 4 to 6 inches height of sludge.

KINS required that the repair program include the installation of lancing holes in all Wolsong Units; KEPCO submitted the schedule of installation for each unit.

- Wolsong unit 2: October 2000
- Wolsong unit 1: 2001
- Wolsong unit 3: 2001

Pakistan

KANUPP has six steam generators with half-inch diameter Monel-400 tubes. These steam generators were manufactured by B&W, Canada. They are being inspected regularly to assess the condition of steam generator tubes and vessel for safe and continued operation of the plant. Following is the summary of condition assessment of KANUPP steam generators:

1. During 32 years of plant operation, only one incident of a tube leak was experienced and was successfully isolated and tube was plugged. Thereafter no leakage was experienced from any of the steam generators till to date.

2. The results of several inspections by eddy current examination, constriction monitoring, vessel weld inspection and visual examination (i.e. 1990, 1993, 1996-97, 1998, 1999, 2000 & 2003) under the CSA (tubes) and ASME (vessel) codes did not reveal any significant flaw/ abnormality in any of the steam generators that would impair the unit for service. After analyzing the results of above inspections, Steam Generators seem fit for operation based on the following reasons:

Tubes: The population of plugged tubes in all six steam generators was 70 nos. (0.86%) in 2000. This number has increased from 70 to 82 (1.0%) in 2003 (based on ET, denting and tube removal result). The 82 plugged tubes include 36 dented tubes and other for pitting and tube removal reasons. In year 2000, thermal hydraulic evaluation based on plotting polynomial curve showed that about 1.3% (109 tubes) up to 2003 and 6.2% (503 tubes) up to 2006 would be plugged. However in 2003 this figure was only '82' i.e. only 12 tubes instead of 39 tubes were plugged since 2000.

Vessel: The condition of vessel welds is acceptable as revealed in essential inspections performed according to In-service Inspection Program (ISIP) based on ASME section XI. The examination yielded results that showed no evidence of any unacceptable flaw indication.

Internals: Visual examination above tube sheet and at 1st tube support plate of all SG units in 2000 also did not reveal reportable abnormality on secondary side. This is again true after evaluation of one unit in 2003 outage

3. Following tube plugging criteria is being implemented at KANUPP:

3.1. For normal tubes: 40% or more tube wall loss based on eddy current

3.2. For dented tubes: Tube with opening reduced to 0.250 inch or below (called severely dented tube) using specially designed stabilizer bars according to international practice (IAEA-TECDOC-981)

4. As a result of above inspections, following are areas of concern:

- Sludge pile with maximum estimated height of 12 inches above tube sheet and tubes experiencing degradation phenomenon under sludge, pitting is dominant.

- Tubes are undergoing denting phenomenon at first support plate in hot leg region and few tubes have reduced in ID to nearly half of original ID (no other TSP had evidence of tube constriction).

- During activity of water lancing it was observed by ET that soft sludge from central region tubes was not removed and water lancing was not effective in this area. One cause was that tubes are triangular pitch and also obstructing the water jet.

5. KANUPP since 1993 has in place following activities to monitor SG safe operation:

(a) Surveillance / monitoring of degradation observed.

(b) Sampling for chemistry and radiation effects in secondary side.

(c) Material study of removed tube.

(d) In-service inspection of the SG unit.

(e) Plugging of tubes:

- With wall loss $\geq$ 40% of wall thickness (as per CSA and ASME codes) base on eddy current result. These are mostly occurring in tubes located under sludge above the tube-sheet in the hot leg region.

- Severely dented tubes with <0.250 inc. Its location is the first tube support plate in the hot leg region.

- Where doubtful signal exist and can not be characterized (in the absence of specialized techniques).

(f) Contingency plan for tube leaks, if any.

(g) Degradation control (stabilizer bar, sludge removal, chemistry improvement etc.)

6. Sludge Removal:

(a) To arrest the tube degradation, sludge lancing of secondary side of all SGs was under taken during year 2000.

(b) Access ports (hand holes) were needed for sludge lancing of SGs. Three hand holes, one above first support plate and two above tube sheet, were drilled in each of the six steam generators.

(c) The Steam Generator cleaning comprised of soft sludge removal above the tube sheet. Fixed type lancing tool through 'No Tube Lane' (NTL) was used. Most of the soft sludge above tube sheet has been removed; however, this technique was not effective for the cleaning of hard sludge and sludge on first tube support plate. Efforts are being made for sludge removal above tube sheet and cleaning of tubes crevices at first tube support plate.

7. Future Plan:

Corrective measures such as sludge removal from tube sheet and first tube support plate crevices.

(a) Proactive measures such as inspection of tubes, internals and vessel welds.

(b) Controlling of condenser tube leakage, incursion of chlorides and other hazardous elements.

(c) Re-inspection of plug seal welds on the plugged tubes.

(d) Regular blow down.

Romania

The issue is of high importance for licensees and Regulatory Body. It was also subject of R&D activities starting from the moment the knowledge transfer in 1978 was initiated.

The licensing basis for Cernavoda NPP Unit 1 includes the requirement to have as a postulated steam generator tubes failure (single and multiple). The requirements for Unit 2 include this requirement, too. More recently there are under regulatory review the requirements to correlate the multiple steam generator failures with the main steam line breaks.

The steam generator tubes are also subject to inspections in operation, as part of the regulatory and company requirements. The results of the steam generator tubes inspections are being evaluated and their conclusions are to be reflected in the PIPD review (program of the periodic inspection for pressurized systems).

Steam generator tubes should be inspected according to the periodic inspection program (PIP) as per the requirements and acceptance criteria in the CSA standards. In addition, in-service-inspections (ISI) plans are developed to manage the ongoing in-service degradations discovered during PIP inspections. The Romanian regulatory Body is seriously considering in order to override the requirements of the CSA standards for periodic inspection, which specify too small and too infrequent samples to detect deterioration of steam generator tubes promptly. It is well known that the CSA N285.4 requirements are that:

- pre-service inspection of at least 25% of the tubes in each steam generator
- sampling of 10% of the tubes in one steam generator has to be done at five-year intervals thereafter.

There is an increased interest for the Regulatory Body and the Licensee to use for steam generator tubes, as well as for the fuel channel the fitness for service concept.

It is to be noted that it is premature to have extensive results and comments and it is sure that further revisions of this document will include them.

However the License has a program for the Steam Generators tube integrity inspection, which is going on in the period 2000-2001 and the long term strategy for the SG tube inspection is under company management approval.

On the other hand it is important to note that the Licensee will complete the program related to the Aging/Life Time Management suported by an IAEA project and this topic is part of it.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Steam Generators, IAEA-TECDOC-981, IAEA, Vienna (1997).
- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.
- Strategic Policy for Cernavoda NPP Unit 2 licensing process, CNCAN 1997.
- AIRS database, October 2000, IAEA/NEA.

ISSUE TITLE: Loads not specified in the original design (CI 4)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

Damage to steam generator divider plates in a large LOCA is an issue for CANDU plants. The design requirements for steam generator divider plates to withstand differential pressure loadings during LOCAs were not adequate in all cases, so that the structural integrity of the divider plates could not be assured. Additionally, pitting at support bars due to flow induced vibration in the U-tubes has been experienced in some plants.

Pipes connected to the PHT and safety systems, are subject to flow and thermal transients during normal operation of the plant. These transients involve repetitive thermal shocks, stresses, thermal stratification and water hammer. In some cases this has resulted in unanticipated displacement of lines. Measurements performed during plant operation have indicated loads, which were not specified and taken into account in the original design for some plants. Cracks and corrosion were observed in some plants.

Safety significance

Loads which are not properly accounted for in the design can increase the frequency of pipe leaks or failures, due to e.g., fatigue damage.

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:

Argentina

Some events have occurred, at the CANDU plant, related to loads not specified in the original design. In November 1986, a short circuit and subsequent fuse failure in the high pressure ECCS caused a valve (3432PV-82) to fail open. The valve was designed to fail open because it is the safe position for proper ECCS operation. Tanks 3432 TK1 & TK3 were pressurized and, due to the presence of air in a line from the reactor building to the service building, the sudden pressurization of water tanks accelerated the water in the piping section compressing the air pocket. The air pocket collapse caused water hammer damaging some supports, but the piping remained intact. Many actions were taken including changes in testing procedures, changes in system venting procedures and the installation of an additional high pressure alarm.

Two other load-related significant events occurred at the CANDU plant. These events were due to diaphragm failure in a liquid relief valve, which allowed the pressurization of the degasser and consequently, the opening of the degasser-condenser relief valves. The most serious aspect of both events was that the degasser-condenser relief valves did not work adequately at the moment when pressure change was required; this produced significant chattering. In this case, the internals of the valves and/or associated piping can be damaged. AECL recommended that for all CANDU 6 power stations, after doing studies and dynamic tests, the current RV's Crosby be replaced with new RV's

Bopp and Reuther in order to avoid the above mentioned circumstances. Assessments and reinforcements to the piping were done as required by the Regulatory Authority.

Another case was the possibility for water hammer in the LISS gas path. The AECL SAB-94-01 report recommends that this possibility be assessed for all CANDU plants. Basically, a high moderator level or an in-core LOCA could flood the helium lines, causing water hammer; this could cause failure of the system pipework. After analyzing this problem, the utility installed additional supports and re-located three valves.

Canada

The design feedback process described elsewhere in this document ensures that experience with incorrectly specified loads is communicated to CANDU owners and is corrected in new designs.

Some changes have been made in existing plants to e.g., boiler divider plates, as a result of review of design loads.

India

There were a few problems related to this issue. There were failures due to vibrations in the circuit connected with Fuelling machines Supply System, which has large reciprocating pumps. A detailed design review by an expert group resulted in several modifications towards improvements. These include improved snubbing support systems, mandatory periodical check of hangers, augmented ISI etc. Failure of moderator auxiliary system piping resulted in changes of valves, piping and support system, which has been implemented in all the plants.

Water hammer and high velocity flow due to sudden quenching of steam resulted in extensive damage to internal components of deaerators in the feed water circuit of earlier PHWRs. A combination of modifications in the mechanical equipment (i.e. spring loading of spray nozzles, tray supports, introduction of water seal in piping to prevent steam entry into spray header etc) and process instrumentation has eliminated such failures

Loading due to pipe whip on inner containment during postulated event of failure of Main Steam Line (MSL) was not properly considered at the design stage. A study has been initiated and relevant experiments are planned towards application of Leak Before Criteria for MSL. Also, In-service Inspection programme for steam lines is being upgraded.

At the Rajasthan Atomic Power Stations the process water systems operate under siphon and is at vacuum. Some valve bodies developed pin holes due to two-phase flows, air entrapment. Several steps including throttling the down stream valve of HXs, have solved the problem. In India, there were a few problems that were faced related the above issue. There were failures in the heat transport system equipment due to vibrations. These were specifically in the circuit connected with Fuelling Machine Supply System which contain big sized reciprocating pumps. A detailed design review by an expert group resulted in implementation of several recommendations including improved snubbing, support systems, mandatory periodical check of hangers, improved inservice inspection, etc. Failure of moderator auxiliary system piping resulted in changes in design of valves, piping and support system which have been implemented in all Indian PHWRs.

Water hammer and high velocity flow due to sudden quenching of steam resulted in extensive damage to internal components of dearators in the steam cycle of a few PHWRs. A combination of modifications in the mechanical equipment (i.e. spring loading of spray nozzles, tray supports, introduction of water seal in piping to prevent steam entry into spray header etc.) and process instrumentation through a modified sequence of the processes etc. have eliminated such failures.

Korea, Republic of

This is not considered to be a generic safety issue in the Republic of Korea.

Romania

There are under consideration evaluations of loads not initially considered in the design. For these loads it is typical that the rules for their combination was not completely clear in any situation. For some specific situations evaluations of some possible impact of such loads was performed as exceptions for unit 1. There were no significant safety impacts identified so far.

Both some operating events and some already started review programs mainly for Balance of Plant (BOP) -Nuclear Steam Plant (NSP) interface systems, for aspects like water hammer effects on feedwater system, performed by the Licensee are expected to reevaluate systematically the loads initially not considered by the design.

It is important to note the fact that this is the subject of the periodical safety review for unit 1 and safety report for unit 2, which are still in the beginning phases in Romania.

ADDITIONAL SOURCES:

- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.
- Strategic Policy for Cernavoda NPP Unit 2 licensing process, CNCAN 1997.

ISSUE TITLE: Steam and feedwater piping degradation (CI 5)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

Steam and feedwater piping is subject to degradation by corrosion, stress corrosion cracking, water hammer and vibration. The piping material is usually carbon steel. The issue is influenced by 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 undetected degradation could occur.

While thermal stratification of feedwater piping has caused problems in PWRs, there have been no instances of similar issues in CANDU feedwater piping, due to a variety of design differences.

Safety significance

Failure of steam and feedwater piping would challenge safety and safety-related systems, and other systems. High energy piping breaks are of safety significance with respect to plant personnel and structures.

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:

Canada

Although this phenomenon is not currently labelled as a “generic safety issue” in Canada, some aspects have been observed in CANDU plants. Most of the phenomena mentioned in the “Description” (apart from thermal stratification) have contributed to the reported incidents of degradation/failures. Actions taken include inspections, leak detection, re-routing of piping, and the installation of protective shields.

In Canada, most utilities have put in place FAC (Flow Assisted Corrosion) programs to predict susceptibility of secondary piping to this type of degradation and to provide inspection requirements into the wall thinning inspection programs.

China

Experience shows corrosion of secondary loop line and elbows is a common problem that nuclear power plants and conventional power plants are facing. TQNPP has chosen the material with excellent corrosion resistant performance for double-flow straight line and all elbows and tees.

India

After failure of steamline at Surrey Point in USA and reported failures in extraction steam lines elsewhere, a fresh review was taken of the design, operational including chemistry control and inservice inspection aspects of steam and feedwater piping. The resultant modifications have been implemented at all Indian PHWRs.

Equipment containing corrosive liquids like acids, alkali, chlorinated water, etc. need more intensive thickness measurement as there have been cases of gross failures of walls of acid tanks, pipes, etc.

Subsequent to the incident of pipe rupture at secondary side condensate system at of Mihama-3 , Japan, AERB undertook a detailed review of the surveillance / health monitoring programme of similar piping system in Indian NPPs. All NPPs were asked to examine the adequacy of surveillance programme for periodical checking of thickness and health of non-nuclear high-pressure and high-temperature piping.

Korea, Republic of

For Wolsong Unit 1, the permanent on-line monitoring system for the thickness measurement was installed at two critical channels. For Wolsong Units 3&4, the thickness of all the feeder pipes was measured and will be checked periodically.

KINS required KEPCO that the scope of the periodic inspection for the feeder pipe thickness measurement should be extended to Wolsong Unit 1 using more reliable non-destructive examination technique, and that a mid-long term research program should be established.

Romania

These aspects are included in the evaluations going on and in the requirements for inspection. However it can not be considered that this topic is a generic safety issue in Romania.This issue is connected to a certain degree to the CI6 one and all the relevant issues related to internal and external hazards, for which qualifications of the systems impose specific requirements on the supports and piping.

The License has a program for the in service mandatory and non mandatory inspection for the steam and feedwater piping degradation. The proams are related also to aspects like errosion/ corrosion, project chemical control for these systems. The material Condition monitoring program is addressing the leakage monitoring aspects, too.

However it is expected that further experience and the results of the periodical safety review for unit 1 and safety report for unit 2, which are still in the beginning phases in Romania, will bring more information on this topic.

ADDITIONAL SOURCES:

- NUREG-1344, "Erosion/corrosion induced pipe wall thinning in US nuclear power plants," April 1989.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Primary Piping in PWRs, IAEA-TECDOC-1361, IAEA, Vienna (2003).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Technical Proceedings Series, Material Degradation and Related Managerial Issues of Nuclear Power Plants, Proceedings of a Technical Meeting held in Vienna, 15–18 February 2005, IAEA, Vienna (2006).

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

This issue is also applicable to NPPs with LWR.

The pressure boundary for the primary system includes many auxiliary systems and the degasser-condenser. The setpoints for the degasser condenser relief valve are set to protect the degasser condenser against static pressure increase. In some CANDU stations, the degasser - condenser relief valves have opened during pressure transient events in the primary systems.

Safety significance

Inadequate pressure control of the primary system and connected systems may cause pressure relief devices to be unnecessarily activated. In some cases it could result in an unnecessary opening of the degasser condenser relief valves creating a net loss of inventory from the primary heat transport system.

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:

Canada

This issue, although applicable to CANDU reactors, is not considered to be a “generic safety issue” in Canada. Adequate overpressure protection in all pressure retaining components is a licensing prerequisite. Design changes have been made to plants both in Canada and to those designed for export to reduce the likelihood of an unwanted opening of the degasser-condenser relief valves, particularly following a spurious opening of the Liquid Relief Valves.

China

It’s a licensing pre-requisite that all pressure retaining Systems and Components should be adequately protected by overpressure protection device in China.

India

In Indian PHWRs a few incidents resulted in pressure transients beyond normal limits (But within safety limits of Technical Specifications). These have been analyzed and modifications have been implemented.

During normal operation, earlier in RAPS-1&2 whenever Turbine was tripping, the reactor was tripping on primary heat transport system pressure high though the system pressure remained well below the set point for opening of Instrumented Relief Valves. This problem was overcome after carrying out certain modifications. In all the other PHWRs, such type of problem was never noted. In case of failure of PHT pressure controller resulting in stuck open feed valve and closed bleed valves, the PHT pressure may shoot up and cause over pressure trip and opening of IRVs. On opening of IRVs the bleed condenser would get pressurized, however, the safety limit would never cross as bleed condenser relief valve opening would prevent it. The various measures taken to avoid over pressurization of heat transport system are as follows:

(a) Automatic tripping of the primary pressurizing pump at high bleed condenser pressure (83 Kg/cm^2) and isolate reflux flow on bleed condenser high pressure.

(b) Automatic run down of H.T. pressure controller set point on bleed condenser level high.

(c) Fast Response Instrumented Relief Valves to take care of fast operational transients.

(d) Manual crash cooling to bring down the pressure immediately.

(e) Line from bleed condenser Relief Valve is designed for same safety class as that of bleed condenser and adequate support has been provided to guard against its failure.

(f) Periodic Surveillance test for bleed condenser RVs, IRVs and other RVs. Opening time for IRV is measured during surveillance and corrected, if required.

(g) In 540 MWe design, pressurizer has been used to take care of pressure transients.

(h) When analysis indicated that one RV provided in not adequate to cope with the transient of bleed condenser box up and both pressurizing pumps operating, additional RV has been provided on the bleed condenser in older Indian PHWRs.

With the above measures, this generic safety issue is now resolved for India.

Korea, Republic of

This issue, although applicable to CANDU reactors, is not considered to be a generic safety issue the Republic of Korea. Adequate overpressure protection in all pressure retaining components is a licensing pre-requisite.

Romania

Although there is a quite well definition of the primary system overpressure protection, compliant for separate systems with the code's requirements, there are some aspects, which might be considered important to be solved from the Romanian perspective. They are related to the coordinated definition of the pressure requirements for all the systems included in the primary system pressure boundary for various possible events (including transients).

The aspects related to this issue are under consideration. Evaluations are to be continued to confirm or not if the setpoints for the Relief Valves of the Degasser - Condenser are qualified mainly to protect the Degasser-Condenser itself as a vessel than the primary system pressure boundary (which includes the auxiliary systems). This situation seems to be generated by a design concept, which would need review in order to assure by a slight setpoint modification the adequate pressure protection for the whole pressure boundary in any postulated transients.

There were commissioning events, for which actions and evaluations are still going on in Romania and for which IAEA missions were held (like the IRS single event mission) which are still under review in order to understood correctly that the corrective actions addressed the actual root causes.

There is a possibility still considered that these type of transients are at least of increased effects due to the not yet completely solved problem above mentioned, which seems to be related to this topic.

However it is expected that the final analyses and conclusions derived as part of the relicensing process of unit 1 in May 2001 will clarify this aspects.

ADDITIONAL SOURCES:

- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.
- IRS Single Event Topical Study, IAEA - CNCAN, 1996.

ISSUE TITLE: Safetyvalve and relief valve reliability (PC 2)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

Protection of the primary circuit against overpressure during incidents is provided by safety valves. Failure of a safety valve to close following a pressure transient would lead to a transfer of inventory to the degasser-condenser, pressurizer, or bleed condenser. The risk of safety valve failure to close is high in some NPPs where safety valves are not qualified for steam/water mixtures and water flow.

In some countries, instrumented/liquid relief valves (LRVs) were originally designed as non-safety components intended for operation under normal conditions. The LRVs were provided for pressure control of the PHT during normal operation and transients.

In the case of a primary-to-secondary leak, capable of overfilling the steam generators, 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 mixtures can then lead to their failure to reclose after opening. This has however not been experienced nor is it a significant safety issue for PHWRs (since the safety case in most PHWRs does not depend on prompt steam generator isolation).

The Heat Transport System and its auxiliaries, have been provided with Relief Valves etc. To ensure there are no leaks to atmosphere (due to economic and radiation considerations), the outlet lines of the relief devices are connected to special tanks or leakage collection systems. In addition, flanged joints are avoided. Hence, in situ tests or tests after removal are both difficult. However for important valves, especially instrumented ones, the designer has provided facilities for routine in situ testing. The calandria is protected by rupture disks, which are not testable in situ.

Relief devices perform important safety functions: they protect system overpressure and thus protect the reactor coolant system pressure boundary. Proper function of these can only be guaranteed by periodic testing.

Safety significance

A lack of qualification of safety valves for water flow might reduce the effectiveness of pressure relief, and could therefore affect the integrity of the primary circuit.

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:

Canada

This issue, although applicable to CANDU reactors, is not considered to be a “generic safety issue” in Canada. Reliability of safety and relief valves is under constant surveillance.

India

An OPEC (Operating Procedure under Emergency Condition), Indian Equivalent of (EOPs) procedure has been prepared to include various conditions under which steam generator/steam lines overfilling could occur, the operator actions and subsequent checks that need to be done. An EOP already exists for stuck open IRVs.

The licensees have also been asked to review various issues related to safety valve testing including codal provisions, capability to tests, safety significance, etc.

Rupture discs are replaced as recommended by vendors.

Surveillance frequency as specified in technical specification document is followed, to ensure the reliability of Relief valves.

Korea, Republic of

Action items for LRVs following LCO were enforced. Although the original tests for LRVs are applicable, the periodic and addictive tests required from ASME OM ISTC Code were imposed.

Romania

Even if this topic is not considered as a generic safety issue in Romania, it is correlated in the Regulatory Body understanding with the PC2 issue. On the other hand this issue might be considered related mainly to the operational reliability aspects for the safety, liquid and relief valves. There are both commissioning and continuous in service tests and monitoring are mandatory to be performed in order to check the fact that these components maintain their capability. All these requirements are not different from any CANDU6 basic design requirements.

ADDITIONAL SOURCES:

ISSUE TITLE: Water hammer in feedwater and steam lines (PC 3)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

Various incidents of water hammer that involved steam generator feedrings and feedwater piping have been observed. 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 resulted in piping and valve damage.

Therefore, systematic review procedures should be developed to ensure that water hammer is given appropriate consideration in the design and 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:

Canada

Several steam/water hammer events have been reported over the years in Canadian NPPs. The causes vary from valve failures to operator errors and inadequate procedures. The resulting damage also varied; in some cases extensive damage to the deaerator internals occurred. No damage in safety systems occurred. Corrective actions identified by the utilities included modifications to procedures, modifications to level control instrumentation, and operator training.

India

Please refer to the response under item CI 4 "Loads not specified in the original design"

Korea, Republic of

This issue is not considered to be a generic safety issue in Korea.

ADDITIONAL SOURCES:

INTERNATIONAL ATOMIC ENERGY AGENCY, Guidelines on Pressurized Thermal Shock Analysis for WWER Nuclear Power Plants, IAEA-EBP-WWER No. 8 (Rev.1), IAEA, Vienna (2006).

4.1.5 Safety Systems (SS)

ISSUE TITLE: ECCS sump screen adequacy (SS 1)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

The containment is equipped with sumps to collect the water lost from the primary circuit after a LOCA in order to recirculate the water in the ECC recovery phase of the accident. The sumps are covered with a screen which is intended to prevent debris penetration to the suction of the ECCS pumps.

The thermal insulation used inside the containment and the total area of the screen above the sump together with dust and dirt that occur in containments form a combination that raises 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 fibres can effectively block a large portion of the screen area. Sump screens must be designed and installed to ensure that the screening function is maintained.

Safety significance

Break-up of thermal insulation around equipment and pipes inside the containment can, under LOCA conditions, lead to an impairment of ECC recirculation by clogging the sump screens and/or the ECCS heat exchangers.

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:

Argentina

The ECCS sump was modified to avoid ingress of foreign material into the low pressure pumps suction line. This modification consists of the addition of a wire-covered structure that surrounds the orifice that connects the ECCS low pressure pumps suction pipeline to the reactor building sump. In case of ECCS actuation such a structure will screen the water before it enters in the pump suction pipeline.

Canada

Upon learning of the incident of ECC strainer blockage at Barseback, Sweden, the Canadian regulatory body and utilities took the following measures:

Regulatory body:

a) A comprehensive study was done in 1995-1996 and concluded that licensees needed to evaluate properly the quantity and characteristics of the debris that could be generated, that fine as well as large pieces should be considered, that existing strainers in some stations were inadequately sized, and that strainers may be susceptible to significant mechanical loads due to pressure differentials.

b) Licensees were asked to consider design changes if necessary.

Licensees:

a) A comprehensive program was carried out to evaluate debris generation, transport and accumulation.
b) An experimental program was initiated under the CANDU Owners Group (COG) to study the pressure drop characteristics, the type of insulation, the effect of particulates, and the long-term behaviour of the debris bed.
c) AECL developed a fine-type strainer to provide a larger surface area.
d) Methods and guidelines have been developed for assessing ECCS sump strainers for individual NPPs to fulfill the requirements of:

   - the maximum allowable pressure drop across the strainer at the expected flow rate and temperature; and
   - assessing the debris type, flow path assessment, water hold-up and quantities of debris transported.
   - New strainers (20 times greater in area) are being installed in Pickering A & B since they are the most vulnerable. Current strainers have been strengthened at Point Lepreau.

   Laboratory results reported by some Non-CANDU countries suggest that some previously unanticipated chemical reactions, which could give rise to significant strainer clogging, are a cause for concern. These later developments and their potential implications are currently being studied by the CNSC and its licensees.

India

Indian PHWRs use the large amount of water stored in Vapour Suppression Pool (VSP) for low-pressure, long-term, re-circulation phase of ECCS. However, the spilled water from primary coolant circuit also finds its way to VSP through sumps and course screens. The vent shafts, which are major connection to high enthalpy area, are provided with large screen areas that ensure that the large debris if any, do not enter the suppression pool through this path. The pump suction is through Johnson screen located at a low level sufficiently below the open surface of this large mass of stored water in the suppression pool to preclude any light/ floating debris from entering the suction of pumps. A cage of coarse mesh has also been provided around the Johnson screen to prevent it from clogging from KAPS-2 onwards. Retrofitting of course mesh around Johnson screen requires dewatering of suppression pool, however, it is also planned to be implemented in earlier built plants namely NAPS-1&2 and KAPS-1. Containment liners (epoxy based paint) are qualified for environmental conditions expected to be prevailing under accident condition for the plants, KGS onwards. Inspection of the Johnson screen using underwater camera in NAPS-1&2 did not reveal any extraneous material around the screen. The above features prevent flow of insulation material, debris, containment liners (epoxy paint) etc. to pump suction.

Korea, Republic of

The concern on the clogging of sump screens was raised during construction phase of the Wolsong unit 4 (PHWR Type). The KINS inspector issued a finding report (No. 98-4-032 ('98.5.22.) 'Erroneous use of some radiation-resistant coating materials in the reactor containment').

Some of coating materials used in the containment building of Wolsong 4 was not environmentally qualified under the condition of a hypothetical loss-of-coolant accident. Under the accident condition (high pressure and high temperature), unqualified coating materials in this area could be separated from the surface to generate debris and transported to the emergency sump.

It was investigated that unqualified coating material was used for 1700m2 of the inside surface in the containment building, and about 400m2 out of the surface could be exposed to the containment spray water.

Related Requirements:

Korean regulation requires all operating nuclear power plants shall be examined and inspected once in a 20-month period. Even if there has been no specific measures, special attentions had been given to the sump intake performances including the clogging of the sump screens during the periodic inspections.

KINS use the U.S. rules and standards, e.g. Reg. Guide 1.82 “Water sources for long-term recirculation cooling following a loss-of-coolant accident”, Reg. Guide 1.54 “Quality assurance requirements for protective coatings applied to water-cooled nuclear power plants” and NRC generic letter 98-04 “Potential for degradation of the ECCS after LOCA because of construction and protective coating deficiencies in containment”.

Corrective Action:

KINS identified a design deficiency of the emergency sump floor and required the licensee to take a corrective action. The licensee has made some design changes for the floor in the vicinity of the emergency sump to slope gradually downward away from the sump.

As a corrective action, the licensee proposed to cover the coated surface with aluminium plates to protect the coating materials from the harsh environmental conditions and the flow of spray water. The KINS evaluated the adverse effects such as additional hydrogen generation from aluminium-water reaction in a hypothetical accident condition, and approved the licensee's action to cover 400m2 of the unqualified surface with aluminium plates for Wolsong Unit 4.

Update provided at CANDU Senior Regulators Meeting in 2005

The containment is equipped with sumps to collect the water lost from the primary circuit after a LOCA in order to re-circulate the water in the ECC recovery phase of the accident. The sumps are covered with a screen which is intended to prevent debris penetration to the suction of the ECCS pumps. The thermal insulation used inside the containment and the total area of the screen above the sump together with dust and dirt that occur in containments form a combination that raises safety concerns regarding the possibility of maintaining ECCS circulation after a medium or large LOCA. Operational experienced from recent events in Sweden and in the USA have demonstrated that even a relatively small amount of similar fibers can effectively block a large portion of the screen area. Sump screens must be designed and installed to ensure that the screening function is maintained. Break-up of thermal insulation around equipment and pipes inside the containment can, under LOCA conditions, lead to an impairment of ECC recirculation by clogging the sump screens and/or the ECCS heat exchangers. The ECCS function can also be affected by inadequately screened debris.

Activities

The concern on the clogging of sump screens was raised during construction phase of the Wolsong Unit 4 (PHWR Type). The KINS inspector issued a finding report (No. 98-4-032 ('98.5.22) "*Erroneous use of some radiation-resistant coating materials in the reactor containment*". Some of coating materials used in the containment building of Wolsong Unit 4 was not environmentally qualified under the condition of a hypothetical loss-of-coolant accident. Under the accident condition (high pressure and high temperature), unqualified coating materials in this area could be separated from the surface to generate debris and transported to the emergency sump. It was investigated that unqualified coating material was used for 1700m^2 of the inside surface in the containment building, and about 400m^2 out of the surface could be exposed to the containment spray water.

To assess the effect of the integrity of fuel channel due to the inadequately ECCS screened debris, a postulated accident of the inlet header 35% break with loss of class IV power are simulated using RELAP/CANDU code. To simulate indirectly the degree of clogging by the dust, dirt or debris in a sump screen, a series of the sensitivity studies for LPSI entrance area from 10% to 100% of the injection area were evaluated. As a result of the simulation, for the broken loop, the fuel channel was cooled by the forced convection caused by ECC (Emergency Core Cooling) injection. The fuel sheath and pressure tube in the broken loop had the maximum temperatures for the case of the minimum LPSI entrance area, which means that the 90% of LPSI entrance area are clogged by pollution materials. Moreover, even under the minimum LPSI entrance area, the fuel sheath and pressure tube temperatures could be maintained below about 530K and 400K respectively. Therefore, it can be concluded that the integrity of fuel channel is guaranteed for the case of 90% clogged LPSI entrance area.

Pakistan

In the event of Loss of Coolant Accident (LOCA), insulation material of the hot water/steam lines located in the containment building may get dislodged due to impingement of the vapour jet coming out from the line break. Such insulation debris may block/choke the moderator area AD sump opening, thus the active drainage pumps will not be able to re-circulate the spilled water to moderator storage tank.

At KANUPP a detailed study has been carried out to asses the amount of insulation debris from location of plant equipment and its passage / route to ECC sump. With this information the sump opening is being modified to avoid the ingress of foreign materials and blockage of sump. The modifications include:

(a) Retention of insulation debris at different locations / levels of the floors to avoid complete blockage of sump.

(b) Design of steel sump opening structure using grating and suitable size wire mesh.

The installation of interceptor grating around the sumps has been completed in the year 2003.

Romania

The Barseback event was part of the licensing process from 1992. There was initiated a design modification. However the lack of calculations as per the requirements for a Special safety System made the approval process complex. In the meantime (since 1999), it was identified by the Licensee that the original designer has concerns about the solution adopted. At this stage we are waiting for the final solution of the problem, taken so that by this improvement for this event the design for the design basis envelope will be not endangered. We consider that the implementation is a complicated process for the completion of this modification and its final global efficiency in any postulated event and for the Barseback type one. The basic concern is related to the fact that we should solve the problem without endangering the basic assumed by design DBA envelope, by improving the plant protection to supplementary requirements.

The Licensee has a project on going with AECL, for the filters, with a target for completion 2001 and implementation in 2002. The AECL project is the result of an R&D program in the framework of COG-Canada.

ADDITIONAL SOURCES:

- IRS Report: Barseback, Sweden, 1991.

ISSUE TITLE: Potential problems in ECCS switchover to recirculation (SS 2)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

ECCS has different phases, i.e. high and intermediate pressure injection and low pressure recirculation.

Switchover to the recirculation phase involves realignment of several valves and may be accomplished by manual, automatic or semi-automatic operations.

There is a potential for pump damage in some conditions when the switching is manual and if it is not done at the right time. In this case ECCS effectiveness can be limited by human error during operation.

Moreover, some of the logical sequences cannot be fully tested during commissioning and/or periodic tests.

Safety significance

There is a potential for impairment of the ECC recirculation phase if switchover is done incorrectly.

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

The Embalse NPP ECCS consists of three stages: high, medium, and low pressure injection. High pressure injection is effected by injecting emergency coolant from pressurized tanks to cover the core, thus refilling the fuel channels and rewetting the fuel. Medium pressure injection is provided by water stored in the dousing tank (1.5 million litres). Capacity of this tank, in the event of a medium to large LOCA, enables core cooling for about 15 minutes. it is then necessary to manipulate manually the valves that connect the dousing tank to both the primary heat transport system (medium pressure stage) and the ECCS pump suction line from the containment sump (low pressure stage). The low pressure stage assures long-term cooling, by pumping a mixture of heavy and light water from the containment sump into the core. Human error in the manual operation when switching over the water supply from the dousing tank to the pump suction line from the containment sump could cause ECCS pump cavitation. Specific training to reduce the likelihood of this human error is applied.

Canada

The transition from Medium Pressure to Recovery ECC is being automated for CANDU 6s in Canada, and automation is standard on all new CANDU designs, including Wolsong 2, 3 & 4 and Qinshan.

India

Indian PHWRs (excluding RAPS) use passive vapour suppression feature. This involves separation of reactor building volume into high enthalpy volume (V1) and remaining as low enthalpy volume (V2) interconnected by vent shafts through a large mass of water kept in the basement, acting as passive suppression pool to condense the steam as it bubbles through it during pressure equilisation between V1 and V2.

The ECCS in current Indian PHWRs is composed of three stages: high pressure heavy water injection, medium pressure light water injection, long term recirculation using water from suppression pool. All these three stages sequentially come on and continue in recirculation mode in case of LOCA, without any operator action and have feature of overlapping. The pump suction for recirculation from suppression pool (except RAPS), which stores large amount of water is so located as to ensure continued availability of required suction pressure. In RAPS redundant paths have been provided so that spilled water will finally reach for long term circulation.

Integrated ECCS tests including logical sequence of operation of the three phases are conducted during light water commissioning of PHT and ECCs. Periodic tests are conducted to check operation of related valves, logics and operation of pumps with limited flow.

Korea, Republic of

Periodic tests (monthly or weekly) are conducted to ensure the proper operation of the related MOVs, logics and operation of ECC pumps with bypassed flow. Test of ECC pumps with full flow is performed during every outage.

This issue is not considered to be a generic safety issue in Korea.

Pakistan

The Emergency Injection (IJW) System at KANUPP provides low-pressure injection of the Moderator System inventory into the Primary Heat Transport System (PHT) through operation of Moderator Pumps following a Loss of Coolant Accident (LOCA).

The current Emergency Injection System has a number of deficiencies. The main deficiencies are:

(i) The System does not meet the single-failure criteria.

(ii) The IJW system components located inside the boiler room such as motorized valves, electrical cable junction boxes, moderator pumps etc. are not qualified for LOCA environmental conditions. Hence, their reliable operation under harsh environmental conditions cannot be ascertained.

(iii) AD Sump requires protection from debris accumulation. Asbestos/insulation material is very likely to choke the AD pump suction, and thereby the return flow path to the reactor.

(iv) In certain cases of pipe break in primary system in the accessible area of containment, there could be un-isolable leak. The PHT water will not be accumulated in reactor building sump.

The Regulatory Authority (PNRA) has required KANUPP to:

(a) Demonstrate that in all conceivable LOCA scenarios, prolonged and reliable core cooling is ensured.

(b) Ensure sufficient inventory of reserve D2O and capability of its transfer is required to be maintained at all times.

(c) Demonstrate operability of IJW valves under LOCA conditions.

In compliance with the regulatory requirement and as a part of safety improvements and plant life extension, following modifications have been carried out at KANUPP in the ECC system [called Emergency Injection (IJW) system] to enhance its adequacy.

i) The IJW system is a low-pressure injection system. It is effective in limiting fuel failures for intermediate and large break LOCA. However, in case of small break LOCA, IJW injection is delayed due to slow depressurization of heat transport system. To overcome this problem, an Automatic Boiler Crash Cooling (ABCC) system has been installed to rapidly depressurize the steam generators following detection of a LOCA.

ii) A portion of heat transport system piping is located outside Boiler room in the accessible area. In case of a LOCA outside Boiler room, re-circulation of the spilled inventory was not possible. To overcome this deficiency, an Emergency Sump Transfer (EST) system has been installed. This system is capable of transferring accessible area spill to boiler room, thus enabling the re-circulation for continued core cooling.

iii) In order to meet the single failure criteria requirement of the national Regulations, redundant valves and instrumentation will be installed in the year 2006.

iv) In case of leakages with in the capacity of primary charging system, a system of Heavy/ Light Water (EHWT) system has been installed. The inventory of coolant available in this system is sufficient to facilitate the operator to crash cool and depressurize the heat transport system in case of abnormal internal or external leakages with in the capacity of charging system. The system has a facility of light water injection into reactor core after depletion of heavy water reserve inventory, if IJW system has failed or there is some delay in IJW injection due to pressure stagnation in the core.

v) Solenoid Operated Valves related to IJW system had been modified for their environmental qualification under design basis accidents.

To improve functional capability and effectiveness of core cooling function at KANUPP, a once-through, medium pressure, light water emergency core cooling system will be installed in early 2006. Furthermore, this system will not be dependent on any plant system.

Romania

Cernavoda NPP Unit 1 has a design in which after the initiation of the medium stage of ECCS and its operation for a short time in case of medium and large LOCA, requires a manual connection of the dousing tank to Primary Heat Transport System and the medium stage of ECCS itself of the water from containment. The water has to be delivered to the ECCS pump suction, which in case of a human error to connect it might induce pump cavitation. The human errors are prevented by extensive training and adequate operating procedures.

The design requirements for Unit2 are to automate this action and this might be reconsidered for Unit 1 also, provided that the probabilistic analyses will prove the final improvement of the ECCS performance as a result of this modification.

ADDITIONAL SOURCES:

- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.
- FSAR Cernavoda Unit 1, 1995.
- Strategic Policy for Cernavoda NPP Unit 2 licensing process, CNCAN 1997.

ISSUE TITLE: Severe core damage accident management measures (SS 3)

ISSUE CLARIFICATION:

Description of issue:

This issue is also applicable to NPPs with LWR.

Severe core damage accident management measures are not applied consistently in terms of analysis, design modifications and procedures.

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.

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 such as fire protection, internal flooding and earthquakes.

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 5, Need for severe accident analysis and MA 11, Adequacy of emergency operating procedures).

Note that owners of most PHWRs evaluate the consequence of dual failures which use the moderator as an emergency heat sink (e.g., LOCA + LOECC) as part of their design basis, and are required to have in place emergency procedures for such events. The issue then pertains to severe core damage sequences where the moderator is for some reason unavailable as a backup heat sink.

Safety significance

Severe core damage 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 primary circuit with a high degree of effectiveness even if the events exceed the design basis. In the absence of such accident management measures, limitation of fission product release and the prevention of long-term contamination might not be achieved for certain low probability scenarios.

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

Many procedural modifications have been performed using PSA results that were not considered in the original design. The procedures for accident management are presently being elaborated. In relation with this subject, the possible states for a nuclear power plant have been grouped in five levels:

LEVEL	*STATE*	*ACTIONS*
1	*Normal operation*	*Operating Manual*
2	*Foreseen operational incident*	*Operating Manual*
3	*Design basis accident*	*Operating Manual*
4	*Accident exceeding the design basis*	*Accident Management*
5	*Significant discharge of radioactive material*	*Emergency Plan*

The first three levels are covered by operational measures specified in the Operation Manual. As regards level 4, it is additionally split into three sub-levels,

SUB – LEVEL	***STATE***	***ACTIONS***	***OBJECTIVE***
1	*Undamaged Fuel Elements*	*Ensure core subcriticality and cooling and maintain containment integrity*	*Prevent core damage*
2	*Partially Damaged Core*	*Preserve reactor core controlled*	*Preserve primary system integrity*
3	*Failure Primary Circuit*	*Preserve radioactive material inside containment*	*Reduce radiological consequences to public*

If sub-level three's goal is not achieved, a shift to level five is indicated and the emergency plan is applied. CNA-I has an emergency plan to be applied inside and outside the installation). As already mentioned there are operational procedures for emergency cases that include:

Instructions of the **E** type - they are focused on determining the failure type and taking actions in case of loss of coolant in the primary circuit (big, small or micro-loss). They also deal with steam generator tube breakage, loss of both water supply and hot steam.

Instructions of the **T** type - They indicate how to proceed in the following cases: quick reactor trigger, boron injection, turbine trigger, simultaneous failure of two pumps of the river water supply and switch to emergency current.

It should also be mentioned that according to the operation license requisites, the Responsible Organisation has a personnel re-training program containing actions to be taken during the occurrence of abnormal events and emergencies in the installations. Such actions are based on the knowledge and analysis of operation procedures.

CNE has an elaborated set of operational procedures for accidental situations known as Operational Procedures for Abnormal Events. These procedures were developed according to the safety design matrix technique and the Canadian operational experience accumulated during several years of operation, and improved using PSA analysis. CNE personnel re-training program includes practices in simulators in which abnormal events are developed and analysed according to Operational Procedures for Abnormal Events. This enables procedure updating, as required by the license. The same importance has been given in such procedures to diagnosis of events and to operation handling to lead the installation to a safe state. Level four accident management is not taken into consideration in the

Operating Manual. Besides, as this level includes accidents exceeding their design basis, it is only possible to prescribe very general measures such as:

- Primary system depressurisation in the case of accidental sequences with core melting at high pressure.
- Controlled gaseous discharges.
- Steam generator feeding with available water.
- Recovery of the electric power system after black-out.

Considering the CANDU design features and accident evolutions analysed there is enough time to face level four, the notice to the Safety Advisory Internal Committee and the Internal Committee of Emergency Control is foreseen during emergency situations in order to carry out the analysis of the situation and issue the pertinent recommendations about the accident management.

Canada

Regulatory Body:

In September 2004, the CNSC issued for public consultation, a draft regulatory guide (G-306) entitled, "Severe Accident Management (SAM) Programs for Fission Reactors". Its purpose is to assist power reactor licensees in developing and implementing a severe accident management program in accordance with the purpose of the Nuclear Safety and Control Act. The guide is available on the CNSC website www.nuclearsafety.gc.ca. The guide addresses goals and principles of severe accident management, considerations for program development, high level accident response, SAM procedures and guidelines, accessible information, personnel training, organisation responsibilities and interfaces, communication with off-site organisations, validation and review and documentation. The CNSC position on severe accident management was furthermore articulated in a paper entitled "Canadian Regulatory Position on Severe Accident Management" by A. Viktorov, A. Bujor, R. Gibb and M. Rizk, that was delivered at the IAEA/AECL/CNSC Technical Meeting on Severe Accident Management and PSA, Toronto, November 2005. The area of SAM is one, among others, which the CNSC is quite active in ensuring that international experience is factores to the maximum extent in the national approach.

Industry:

Detailed EOPs (Emergency Operating Procedures) for design basis events have been developed for and are in use at all CANDU stations. These EOPs are validated on full scope training simulators.

Severe core damage accident analysis has been performed by AECL and OPG since the 1980s, and have been communicated to CANDU purchasers: for example, the results of severe accident analyses by AECL and KAERI/KOPEC were provided to the utility, KEPCO, in Korea. Such analyses provide insights into severe accident mitigation procedures development work.

China

It is Construction Permit condition that the Licencee should consider to take measures to prevent and mitigate severe accidents.

India

AERB, based on the regulatory reviews of PHWR design and operation and reviews conducted by utility and regulatory body following TMI and Chernobyl accidents have been updating the scenarios to be considered for safety analysis. AERB guide (AERB/SG/D-5) categorises such scenarios in five categories. These categories cover the spectrum of normal operation, anticipated operational occurrences, design basis accidents involving single or double failures and beyond design basis events. Based on experience of fire incidents and flooding the regulatory requirements have been enhanced and modifications in plants have accrued. Seismicity (Operating Basis Earthquake (OBE) and Safe Shut Down Earthquake (SSE)) is the specified design basis for safety systems and safety related systems.

Plant modifications and procedural modifications have been performed based on PSA results. For examples provision of permanent and remote manually operable alternate cooling water supply to steam generators, shutdown heat exchangers, moderator heat exchanger, end shield, etc. has been made. This cooling water supply comes from existing surplus of diesel operated fire water pumps. This feature facilitates the operator use of existing equipment in certain beyond scenarios DBA.

Emergency operating procedures (see also AA5, AA11) are prepared for listed emergency conditions. The Operational experience, PSA studies or international inputs add to this list as and when a potential weakness is identified. The operator training programme and their licensing tests conducted by AERB lay special emphasis on up to date knowledge and analysis of such emergency procedures.

The AERB design code AERB-SC-D is now been revised to incorporate inclusion of BDBA in accident management measures. However this will be for future designs only.

Recently Psa Level II studies have been carried out for a reference Indian PHWR to devise suitable accident management strategies to handle severe accidents.

Korea, Republic of

External Event PSA of Wolsong Units 2, 3 & 4

1) Seismic risk analysis
   - Fragility analysis
   - – Analyze 7 structures, 62 seismic components, 69 none seismic components, & 41 relays
   - – Structures
   - In case of earthquake, EWS structure was most weak structure due to insufficient bracing
   - – Components
   - Most of non-seismically qualified (NSQ) components has more than 0.1g HCLPF (High Confidence Low probability Failure).
   - Dominant Scenarios
   - – Seismic-induced failure of group 2 system
   - – Seismic-induced failure of group 1 system with operator failure to actuate the group 2 system
   - – Seismic-induced failure of group 1 system with random failure of EPS
   - Group 2 system (EDG) reliability is lower than that of PWR (0.03 vs. 0.01)
   - Plant HCLPF value of ground motion acceleration is lower than that of PWR
   - Recommendations
   - – Development of procedure for seismic induced accidents
   - – Reinforcing EWS structure
   - – Reinforcing NSQ transformer and MCC
   - Bolting between adjacent cabinets
   - Reinforcing anchorage
   - – Enhance EDG reliability

2) Fire risk analysis
   - No automatic fire suppression system in reactor building
   - CANDU fire frequency is similar to that of PWR
   - Investigate 110 fire areas (250 rooms)

- List 700 components for safety shutdown
- 24 areas are left for detailed analysis after screening analysis
- Dominant scenarios
- – Total loss of plant process controls due to fires in the access area of reactor building
- – Loss of group 1 system due to cable fire in gap areas between S/B and T/B
- – Main control room fires
- – Loss of offsite power due to fires in transformer areas and switchyard control building
- Recommendations
- – Automatic suppression system installation and/or development of mitigating procedure
- Gap area between S/B and T/B (14% reduction)
- Reactor Building (20% reduction)

3) Flooding risk analysis
- Unlimited water source (sea water) exists in S/B
- Screening analysis for 119 flooding area
- 8 areas are left for detailed analysis
- – RCW (recirculated cooling water) Hx. Room (sea water)
- – Condenser area
- – FW & AFW pump areas
- – IA system area
- CANDU experience data was used for piping rupture probability and expansion joint rupture probability
- Dominant sequences
- – Loss of service water and loss of feedwater due to RCW expansion joint rupture (80%)
- – Loss of condenser cooling due to condenser expansion joint rupture (20%)
- Recommendations
- – Operator training for the expansion joint rupture
- – Surveillance check of flood barrier (cf., steam-tight door)

4) Conclusions

- SCA & dousing tank play very important roles in case of external events.
- Group 2 systems are useful in case of an earthquake.
- CANDU specific failure probability of EDG is higher than that of PWR
- PSA is proved very strong tool for merits and demerits from differences in design concept, plant arrangement, and their operating experience.

Romania

There is a gradual coverage of the accident management aspects for a CANDU 6, starting from extended technical features to mitigate accidents, which are based on the concept itself:

The Abnormal Plant Operating Procedures are derived for 8 specific transients and two generic ones, already implemented and used in unit 1

Severe Accident management procedures, for which a strategic requirements for this year required the start of activities and detailed requirements and significant progress is expected as part of the periodical safety program, under completion for the May 2001 relicensing

For unit 2 the SAM procedures are conditions for the Operation License from the very beginning. Significant progress is expected to generate necessity to review this information by complying with the regulatory requirements.

Emergency Planning is being performed as per the best international recommended practice and actions are taken to put in compliance the emergency planning aspects and the technical basis for it, i.e. the severe accidents management topics.

PSA level 1 results and level 2 preliminar findings, external events hazards analyses(seismic, fire) results, which are being performed by the Licensee, are also expected to improve the understanding of the severe accident aspects for Cernavoda NPP, both units.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment and Verifications for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.2, IAEA, Vienna (2001).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Implementation of Accident Management Programmes in Nuclear Power Plants, IAEA Safety Report Series No. 32, IAEA, Vienna (2004).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level 1), Safety No.50-P-4, IAEA, Vienna (1992).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Procedures for Conducting Probabilistic Safety Assessments of Nnuclear Power Plants (Level 2), Safety No.50-P-8, IAEA, Vienna (1992).
- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.
- Cernavoda Unit 1 Operating License, 1999.
- Strategic Policy for Cernavoda NPP Unit 2 licensing process, CNCAN 1997.
- CNSC Regulatory Guide, G-306, Severe Accident Management (SAM) Programs for Fission Reactors.
- “Canadian Regulatory Position on Severe Accident Management” by A. Viktorov, A. Bujor, R. Gibb and M. Rizk.
- ATOMIC ENERGY REGULATORY BOARD, Safety Code, “Code of Practice for Design for safety in PHWR based NPPs”, AERB/SC/D.

ISSUE TITLE: Leakage from systems penetrating containment or confinement during an accident (SS 4)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

Following a plant accident involving sheath failure, safety system equipment located outside the containment may have to function with large radioactive inventories in the fluid they process. The potential exists for leakage from such systems, to the air, soil or auxiliary connected systems.

Therefore any 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 leakage 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:

Canada

Despite the applicability of this scenario to some CANDU plants (e.g. potential leaks from portions of the ECC piping), it is not currently labelled as a “generic safety issue” in Canada.

The issue is recognized in design of systems outside or penetrating containment. ECC pumps and their piping are designed as an extension of the containment envelope. The equipment is housed in concrete enclosures capable of containing spills. Intermediate cooling water systems (if used) are installed between the primary coolant and the ultimate heat sink. They can act as a barrier against the dispersion of radioactivity to the environment or ingress of inappropriate chemicals into the reactor coolant system. Any type of cooling system interfacing with the ECC heat exchangers is designed to withstand water hammer loads, cope with flooding situations, and resist corrosion.

India

Some of the important process systems and safety systems are partly located outside the primary containment and appropriate design provisions (atleast two automatic containment isolation valves in series) are incorporated to eliminate/minimise the possibility of leak of contained radioactive fluids to environment under accidental conditions.

a) Reactor coolant purification system is housed in a separate Reactor Auxiliary Building. This being a heavy water system, the leakage of fluid is normally in control and the area is provided with ventilation provisions that are typical for contaminated heavy water areas. The shielding on pipes and ion exchange columns has been designed to take account of accident conditions.

b) If the accident is of a nature that ECCS has been invoked and recirculation of water from vapour suppression pool takes place, then the active spilled and recirculated water of coolant system (diluted by large inventory of suppression pool) comes to ECCS pumps and heat exchanges which are located in the annulus space between primary containment and secondary containment. The annulus is maintained at a sub-atmospheric pressure during accident condition by purging out air through combined (Absolute and carbon) filter to environment through a 100m high stack. Thus leakage of water from ECCS system is not expected to give rise to significant release of activity to the outside environment.

c) The cooling water to heat exchangers of active systems such as shutdown cooling system, moderator system, shield cooling system, ECCS, etc. is provided by closed loop process water system. The process water is in-turn cooled by tertiary water circuit using induced draft cooling towers. However, leakage from this closed loop process water system to working areas and into tertiary water system may need close monitoring and control in case of accident if activity enters into the process water system. The required design provisions and appropriate administrative controls are in-place to assess and maintain integrity of heat exchangers of these systems

d) Primary Containment Filtration and Pump Back System, having a set of Combined (HEPA and Charcoal) filters to bring down concentration of radionuclides under the accident condition, is located inside the containment. As per the present design, the containment may need to be depressurized periodically after about 48 hours of initiation of an accident. The depressurization system also contains Combined filter and it discharges air to the outside environment through 100m stack. Appropriate design features and administrative controls are instituted to prevent unwarranted starting of the containment depressurization system, which could result in large quantities of activity to the environment.

e) Equipment like D_2O dryers etc. which are located outside the containment have three isolation dampers which close automatically like on high containment pressure or high activity. Sample lines for measurement of activity during accident conditions are shielded.

Korea, Republic of

This is not considered to be a generic safety issue in Korea.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Concrete Containment Buildings, IAEA-TECDOC-1025, IAEA, Vienna (1998).
- ATOMIC ENERGY REGULATORY BOARD,Design Safety Guide, “Containment System Design”, AERB/SG/D-21.
- ATOMIC ENERGY REGULATORY BOARD, Design Safety Guide, “Vapour Supression System”, AERB/SG/D-22.

ISSUE TITLE: Hydrogen control measures during accidents (SS 5)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

Hydrogen released in PHWR Nuclear Generating Stations during certain accident sequences may produce flammable gas mixtures in some regions of containment. The large mechanical and thermal loads generated by the ignition of these gas mixtures can threaten the integrity of the containment envelope, supporting internal walls and required safety-related equipment.

Following a LOCA, 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-denotation transition phenomena with corresponding dynamic loads. These phenomena may cause 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).

Safety significance

Insufficient hydrogen mitigation during accident scenarios may impair 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:

Argentina

The CONTAIN code was used at ARN to model hydrogen behaviour, including deflagration and detonation in the Atucha I containment. Severe accident source terms and likelihood for several containment failure modes were obtained for Atucha I NPP. The hydrogen deflagration/detonation issue was identified as a major contributor to containment early failure in Atucha, and specific studies on mitigation technologies are presently being analyzed (these involve pre- and post- inertisation, use of igniters and use of recombiners).

The particular configuration of the control rods in Atucha (made of Hafnium) was identified as a potentially relevant issue in aerosol generation, and an experimental series was proposed to study the Hafnium aerosol behaviour (STATUS), which is presently under evaluation. Regarding Embalse NPP, a review of the severe accident scenarios and phenomena involved in CANDU plants is presently being carried out. A CONTAIN model to analyze the hydrogen behaviour and the need for mitigation devices is also being developed.

Canada

Regulatory Body:

Loss of coolant accidents (LOCAs) can lead to substantial hydrogen releases to containment. Radiolysis of the water in the primary heat transport system by radiation fields from intact fuel in the core is recognized as the primary source of hydrogen generation. Radiolysis of the water collected in containment by radio-nuclides from failed fuel bundles can also lead to the release of an appreciable amount of hydrogen in the long term. In addition, for LOCA scenarios where emergency core coolant (ECC) is impaired or lost (LOECC), oxidation of over-heated fuel sheaths is expected to result in short-term releases of hydrogen into containment. The more significant long-term hydrogen releases have been shown to induce flammable and potentially explosive gas mixtures covering entire containment compartments, while the short-term releases can have similar local impact in certain regions of the affected compartments. Sensitivity studies on post-blow-down steam flows through the core have indicated an escalation in hydrogen and radionuclide releases for fuel channel flow rates below 100 g/s, with a peak around 10 to 20 g/s. A significant safety issue, unless appropriate mitigation is provided, is the challenge posed to the integrity of the containment systems and the necessary or credited post-accident structures, systems and components (SSC) inside containment, by the large combustion and potentially explosive loads from possible ignition of the long-term hydrogen releases. A second significant safety issue is related to the challenge posed to the post-accident performance of containment and its necessary/credited SSCs, by inadequate *environmental qualification* (EQ) to the induced harsh radiological and potential combustion conditions. Mitigation of the long-term hydrogen releases is also needed for viable severe accident management. CNSC staff has expressed concern as to whether the licensee's adopted course of action would be sufficient to resolve this containment issue. Hydro-Québec is adopting the deterministic, dual-failure approach, whereas the other utilities are adopting an essentially probabilistic approach. Factors requiring additional consideration include: (1) the need to adopt a separate approach for refurbished units and for units approaching their end of life; (2) consistent treatment of severe accidents; and (3) consistency of proposed modifications with the licensing basis of existing reactors. CNSC staff is currently finalizing its position with regard to the path to be aken to achieve an optimum level of containment protection. CNSC staff has decided to revise its approach to closure of this GAI.

Industry:

To mitigate the effects of hydrogen, Ontario Power Generation (OPG) has installed Hydrogen Ignition Systems (HISs). Assessments of the short term behaviour of hydrogen with the 3-D code GOTHIC were then carried out to demonstrate the effectiveness of the HISs. Based on a derived safe load criterion OPG has concluded that all potential hydrogen burns in its NGSs are benign, even for impaired or ineffective HISs and Air Cooling Units.

The amount of hydrogen in CANDU 6 plants, well mixed by natural circulation, is close to the lower flammability limit. However mitigating measures (hydrogen igniters) have also been installed in Wolsong 2/3/4/ and Qinshan TQNPPs. A number of scoping analyses to identify the potentially limiting cases were performed.

The threat posed by the long term hydrogen releases from water radiolysis in multi- and single-unit NGSs, is postulated to be effectively mitigated by mixing due to natural circulation and/or by venting through the Filtered Air Discharge System.

China

Accidents can result in fuel temperature increases (such as LOCAs). There is a potential hazard to produce and release significant amounts of hydrogen to the coolant. When this occurs, the hydrogen can then be discharged to containment through the break. Ignition of the hydrogen could result in a deflagration, standing flames or even detonation that could cause significant damage to a number of safety-related systems, and might compromise the integrity of the containment. The effects depend on such things as the overall amount and concentration of hydrogen in containment, the adequacy of mixing of hydrogen in the air, the possible existence of local pockets of higher concentration gas being trapped, and the proximity of equipment to the source of the problem, etc.

NNSA raised this issue since the beginning of the application of construction permit for TQNPP. There are many questions and responses between the NNSA and the applicant on hydrogen production, mixing and distribution, combustion, monitoring, and controllability of hydrogen following LOCA.

The Chinese Regulations and AECB Regulatory Document R-7 require that provision shall be made for controlling the concentration of hydrogen and/or oxygen if there is possibility of such an explosion of deflagration.

Licensee has analyzed those events with claimed conservative assumptions, and contends that there will be no major damage if a deflagration occurs. In addition, they have stated that the hydrogen will be sufficiently mixed globally in the containment atmosphere so that the maximum concentration would not reach the detonation threshold.

New CANDU-6 plants in China will have 44 hydrogen igniters of hot surface type in the two fueling machine rooms and boiler room.

It is required that the hydrogen concentration and distribution of the three divisions in containment should be calculated to show that the location of igniters is reasonable and there is not the possibility of local accumulation in some compartments.

India

Indian Pressurized Heavy Water Reactor (PHWR) containment can be divided in two major volumes, namely, Volume-VI in which all high enthalpy systems are located and Volume – V2 which is accessible during normal operation.

Computer Code PACSR is being used to calculate the hydrogen concentration in containment. During metal water reaction phase in a severe accident, Hydrogen is injected to Volume – VI and later in Volume-V2 due to radiolytic release. Simultaneous failure of LOCA + ECCS is postulated as beyond design basis accident for assessment of hydrogen concentration. The calculation considers the effect of compressed air in leakages into the containment and also the out leakage. Primary containment filtration and pump beek system (PCFPB) will help in mixing of hydrogen between Volume-VI and Volume-V2. Following two cases are considered for the analysis:

1. Hydrogen generation considered due to metal water reaction followed by radiolytic decomposition of coolant and suppression pool water.
2. Same as Case-1 but with hydrogen release due to radiolytic decomposition of moderator also.

It is seen that in first case, peak volumetric hydrogen concentration does not cross the flammability limit (4% V/V) in Volume-VI and steep reduction is observed in hydrogen concentration due to initiation of PCFPB system.

In the second case, the hydrogen concentration behaviour is similar as in Case-1. However, due to decomposition of moderator, the concentration increases monotonically but does not cross flammability limit.

Above analysis did not consider the operation of primary containment control discharge system which may cause further dilution of hydrogen. However, the hydrogen concentration may cross flammability limit in some local areas like F/M vault and pressure relief chamber etc. in early phase. To prevent unacceptable local peaking of hydrogen concentration following additional provisions are being considered and developed:

- Interconnection between both F/M vaults to promote better convection.
- Catalytic recombiners.

Korea, Republic of

1.Background

Hydrogen generated following postulated accidents can be ignited to produce pressure spike and heat which is the potential threat to containment integrity and equipment survivability. The Enforcement Decree of the Korean Atomic Energy Act (Article 92) and AECB Regulatory Document R-7 require that provision shall be made for controlling the concentration of hydrogen and/or oxygen if there is possibility of such an explosion or deflagration. New CANDU-6 plants in Korea will have 44 hydrogen igniters of hot surface type in the two Fuelling machine room and boiler room, but no hydrogen recombiners which can control the lower H2 concentration. The AECB issued Generic Action Item 88-G-02 which required the Canadian utilities to do the followings;

a) Hydrogen Mixing and Distribution:
   - Justify the assumption of uniform mixing. Address the possibility of non-uniform H2 concentration in the F/M vault, considering break location, steam condensation, and unavailability of F/M vault cooling fans. Assess the potential for high H2 concentrations and the consequences of H2 in the unanalyzed rooms. Reassess the functional and reliability requirements on operation of vault coolers

b) Risk from Detonation and Equipment Survivability:
   - Analyze the potential for occurrence of standing flames and their possible consequences. Consider turbulence and the presence of obstacles. Conduct further research to define the limits of flame acceleration and DDT is required

c) Installation of Hydrogen Removal Systems:
   - Provide rationale for the number and locations of igniters. Justify the assumptions on the ignition criteria

d) Verification of the computer code:
   - Verify the computer code in a representative geometry

CAN3-N290.6-M82 shows that containment hydrogen concentration can be a parameter for post-accident monitoring.

KINS reviewed this issue since the application of construction permit for Wolsong-2. There are many questions and responses between the KINS and the applicant on hydrogen production, mixing and distribution, combustion, monitoring, and controllability of hydrogen following LOCA.

2.Design Status

a) Hydrogen Mixing and Distribution:

   - Multi-node analysis of hydrogen concentration distribution following a LOCA+LOECC has been done using PRESCON2 code and also GOTHIC code.

The PRESCON2 is a lumped-parameter model which was analyzed for HDR experiments to check its capability of hydrogen mixing calculation. The GOTHIC calculation was done in order to confirm that the locations for the ignitors as previously determined are appropriate. The predicted value of maximum hydrogen concentration is 15% at 750s near the broken header in the accident node. The analyses did not take into account of various hydrogen sources except metal-water reaction. They also did not include the forced circulation induced by vault coolers of which 16 coolers are LOCA qualified and the fans are seismically qualified. Reliability requirement were reassessed and considered to be satisfactory.

b) Hydrogen Removal Systems:

- A network of 44 hydrogen igniters is installed within the reactor building, in the Fuelling Machine Vaults and the Steam Generator Room. The igniters are High temperature, helical coil type. They are turned on by an appropriate accident signal (i.e., high containment pressure or activity). Once initiated, the system stays on until manually brought down by the operator. Periodic functional testing from Main Control Room confirms circuit continuity through the igniters. The system is supplied from Class III power backed up by the Emergency Power Supply.

c) Equipment Survivability:

- The igniters will be installed in a dual configuration, i.e. on odd and even channels. The distance between the odd/even igniters will be at least 1.50 m. A protective barrier will be installed in case of space constraints. Distance from the walls of igniter coil is estimated at 12 inches.

d) Installation of Hydrogen Analyzer

- Although in the original design there was no device for measuring hydrogen concentration, a sampling facility will be added for post-accident monitoring system.

3. Review Results

a) Hydrogen Mixing and Distribution

- Forty-node PRESCON2 analysis results show that the mixture in the accident compartment and steam generator equipment room becomes flammable from 500 seconds and 1700 seconds following LOCA+ Loss of ECC. Review is being carried on the GOTHIC calculation results which show higher local concentration.
- Validation reports of the PRESCON2 code using HDR experiment have been reviewed. The prediction of the code seems to be adequate for the condition where turbulent flow is build up. But the code shows its limited predictability for natural flow condition. GOTHIC code can make up such limitation.
- Long-term hydrogen distribution was not carried out except the estimation of hydrogen production rate due to radiolysis and metal corrosion.
- Review is being carried out on the possibility of local accumulation in some compartments such as dead-end ones.

b) Hydrogen Combustion and Control

- No analysis has been done for hydrogen combustion following the design basis accident (DBA: 100% Reactor Outlet Header Break + LOECC) except a VENT code calculation on peak differential pressure and burning velocity for less severe accident (60% Reactor Outlet Header Break + LOECC) than DBA. The analyzed case and the result are not considered to be appropriate for assuring containment integrity because the differential pressure (65 kPa) exceeds design value (36 kPa)

with 6.6 % hydrogen mixture. The analysis should include long-term hydrogen transient due to hydrogen production mechanism other than metal-water reaction.
- No test results which show the sufficient hydrogen removal by the igniters were submitted.
- It is required that an emergency procedure is provided to control lean hydrogen which do not reach the flammability limit.
- The number and locations of igniters is considered to be satisfactory if the combustion analysis and test results show adequate hydrogen igniter's capability.

c) Monitoring of Hydrogen Concentration

- It is considered to be acceptable to monitor hydrogen concentration with sampling following DBA, although on-line monitoring seems to be the best way to provide information on hydrogen concentration.
- A procedure for post-accident monitoring of hydrogen concentration should be established.

d) Equipment Survivability

- The applicant believes that the results from the EPRI's Large-Scale Hydrogen Burn Experiments can be applicable to the Wolsong-2. Survivability of the safety equipment against multiple burn and diffusion flames in the vicinity of the hydrogen igniters should be evaluated further.

e) Related Actions and Research on this Issue in Canada

- Follow-up action is required on recent Canadian research to define the limits of flame acceleration and DDT including study on the effect of obstacles.

Romania

This information will be updated after significant progress will be dome as part of the Periodical Safety Review process for unit 1, by completion of the Severe Accident management procedures and the resuming of the licensing activities for unit 2, in 2001.

The Licensee will manage the performance of this project in a framework of a long term Research and Development Strategic Safety Program.

ADDITIONAL SOURCES:

- CNSC Position Statement 88G02 "Hydrogen Behaviour in CANDU Nuclear Generating Stations".
- ATOMIC ENERGY REGULATORY BOARD, Safety Guide, “Hydrogen release and mitigation systems under accident conditions”. AERB/SG/D-19.
- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.
- Strategic Policy for Cernavoda NPP Unit 2 licensing process, CNCAN 1997.

ISSUE TITLE: Reliability of motor-operated and check valves (SS 6)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

Operating experience of nuclear power plants indicates that valves, or valve operators, may 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 binding 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.

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.

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.

Safety significance

Malfunctioning power-operated valves or check valves could impair the effectiveness of mitigating systems in accidents.

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

On 1997 a motor-operated valves test program was initiated, in particular to review valve of primary system, ECCS and shutdown cooling system. The result of the program identified torque switch settings inadequacies, actuators failures and valve internals deficiencies. This program involves the

MOVs of an initial group of systems. The inspections were performed by the same specialists that actuated in Canadians CANDU 6 NPPs. On the other hand, check valve inspections are included into the preventive maintenance program.

Canada

It was found that some of the MOVs (e.g. ECC injection) may not operate under large differential pressures. All licensees have initiated programs to show that the torque generated by the valve motor can overcome the pressure differential loading. These programs include:

a) identifying the maximum pressure difference to which the valve may be subjected from hydraulic transients,
b) calculating the torque required to overcome the friction created by the pressure differential,
c) modifying MOVs as required, and
d) ensuring that adequate testing and maintenance programs are in place.

Some licensees have already taken action to improve MOV performance; these include:

a) re-calculation of the open thrust requirements, and increasing the open torque switch settings,
b) routine testing to demonstrate availability,
c) replacement of the two-rotor limit switches with four-rotor units, and
d) improvement of maintenance programs.

India

Pressure locking and thermal binding problem has been faced in a few applications in Indian PHWRs. These were in the ECCS system, steam cycle and H.T. system. In some cases a bypass valve, for pressure equalisation has solved the problem. In some other applications bigger motors were installed. In addition, discussions with the manufacturers and modifications to the torque switch settings and operating logic (bypassing over current trip for initial 10% of opening) solved the problem for ECCS motorized valves.

A few check valves needed modificaiton of the bearings of the stem and gudgeon pins. Check valves, passing in the ECCS system at Narora Station, resulted in spurious failure of rupture discs. Installation of properly sized vent and drain valves and their re-orientation solved this problem. Continuous monitoring is warranted to ensure satisfactory reliability. For engineered safety features reliability is calculated from the test, maintenance and breakdown data and ensured that actual reliability is better than both the target and value assumed in safety analysis.

Korea, Republic of

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.

Valve malfunction 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.

MOST requires the utility to establish programmes to ensure the operability of MOVs in safety-related systems to perform their safety function.

MOST requested that NPPs ensure the capability of MOVs in safety-related system by:

- reviewing MOV design bases,
- verifying MOV switch setting, and
- in situ testing MOVs.

The utility is commited to complete this issue within 5 to 8 years.

Also, the regulatory body requested that NPPs ensure reliability of the safety-related power-operated Gate Valves that are susceptible to pressure locking or thermal binding.

Pakistan

The safety Motor Operated Safety Valves (MOV's) used in the plant are of very old design. Due to obsolescence, only corrective maintenance has been carried out by KANUPP without preventive overhaul or testing on test bench. According to KANUPP, availability of spares for these MOV's or an expert to perform preventive maintenance on these MOV's, is a difficult task. However, KANUPP intends to explore the possibilities of obtaining the assistance of some expert and procurement of spares from original vendor. PNRA requires KANUPP to take necessary measures according to their plan for ageing management of MOV's that include:

(a) Overhaul and replace the aged parts in safety related actuators and MOV's.

(b) Replace the solenoid operated valves, especially those which have corroded internals.

(c) Overhaul safety related relays

Ensure enough spares before overhaul

Romania

There is no specific action forseen, initiated and/or monitored for the MOV's. Most of the critical MOV's are subject of the Mandatory Testing Program. However it is a topic under surveillance as part of the operational experience feedback program and the basic information from the AIRS database are available and considered, as for instance the calibration/testing techniques (for instance Liberty)

The Licensee is developing programs for MOV, AOV, check valves and a program on component engineering is also forseen to be started soon under COG/IAEA.

ADDITIONAL SOURCES:

- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.
- AIRS database, October 2000, IAEA/NEA.

ISSUE TITLE: Assurance of ultimate heat sink (SS 7)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

Sources of cool water are necessary to cool-down a plant safely, regardless of the state of the plant. The issue is the reliability of the sources themselves such as rivers, lakes, ponds as well as 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 unfavourable meteorological conditions, for periods long enough to guarantee the safety of the plant.

The initiating events and sequences for the total loss of heat sink are plant-specific, and thus, individual plant assessments are necessary.

Safety significance

Most PHWRs have two independent groups of systems, either of which can shut down the reactor, remove decay heat, and monitor the plant state. However a complete loss of the independent service water systems of both Group 1 and Group 2 could lead to significant core damage.

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:

Argentina

The PSA consider the possible heat sinks analysis from a probabilistic point of view. However, some doubts related with the deterministic approach appeared, as an example could be mentioned the success criteria for the different heat sinks associated with diverse scenarios and with the moderator capability to fulfil the ultimate heat sink function in all the accidental situations considered. In order to optimise the above mentioned doubts, efforts on thermohydraulics aspects by updating codes and plant models used in deterministic verifications are being made.

Canada

The “ultimate heat sink” in Canadian reactors is the moderator system. Licensees are required to demonstrate that the moderator is sufficiently sub-cooled to ensure its availability as the ultimate heat sink. The end shield cooling system is also credited as a heat sink in severe accidents. Both systems belong to different “system groups” to reduce the probability of their coincidental failures.

The presence of the moderator and shield tank water around the core slows down the progression of a severe core damage accident even if all heat sinks are unavailable. Thus failure of the calandria shell could be postponed for about a day, giving time for severe accident management.

India

AERB has issued a guide on Ultimate Heat Sink and Associated Systems (AERB/SG/D-15) which is being followed for Indian PHWRs. In general, plant design are required to demonstrate availability of service water source which is not affected by external hazards like dam break or SSE events. Such redundant source is independent of the one being used for routine purpose – river, canal etc.

This SSE qualified stored inventory of water is adequate to dissipate residual heat for a specified period (generally 7 days or more). Further, there is a provision for assuring continued cooling beyond this period by alternate arrangement. For example, readiness for creation of bore holes for additional water supply etc. Taking cooling water in such events from other units where multi-unit station exists has been provided.

Residual heat removal means from core, moderator, end shields is designed with application of redundancy and diversity as far as possible to reduce potential for common cause and to improve reliability of system. For example, feature of surplus fire water from diesel driven pumps in as a back up to all critical heat exchangers for residual heat removal is also available.

Korea, Republic of

KEPCO is required to demonstrate that the moderator is sufficiently sub-cooled to ensure its availability as the ultimate heat sink.

This is not considered to be a generic safety issue in Korea.

Pakistan

Design inadequacy of existing Boiler Feed Water (BFW) system was identified during a review of the KANUPP safety features. The secondary heat sink to remove heat from the primary coolant can be lost in the following initiating events as functional and environmental cross-links existed.

- Failures of high energy feed water and steam piping in the turbine building as well as loss of low-pressure process water.

- Complete loss of AC power (station blackout)

KANUPP reviewed the adequacy heat sink and alternatives to reduce the likelihood of loss of heat sink. An Emergency Feed Water (EFW) system has been designed to provide water to all six steam generators in the event of total loss of feed water due to rupture or break in Boiler Feed Water System or complete loss of power. The new system was required to increase the availability of heat sink to the primary coolant by eliminating functional and environmental cross-links. The EFW system is expected to meet the following as per its design features:

i) Prevent over heating to fuel
ii) High reliability
iii) Available under postulate initiating event
iv) Long term availability for the mission time
v) Functionally independent of any non-credited systems and structures
vi) Simplicity
vii) Redundancy
viii) Monitoring test and inspection provision
ix) Seismically qualified

The Emergency Feed Water (EFW) system will cater for the following postulated initiating events.

- Loss of normal and essential power, i.e. total station blackout.

- Main steam line break out side the reactor building, which may cause the failure of process water pump motors to steam environment and hence will result in failure of main boiler feed water system.
- Feed water line break in any one bank inside the reactor building.
- Feed waterline break in turbine building.
- Seismic event, which eventually lead to above postulated events.

Under PSA application project, EFW System has been automated and some hardware modifications have been made to improve the reliability of the system. The system is operational since 2003.

Romania

Cernavoda NPP has by requirements defined a series of Ultimate Heat Sinks for various possible accidents:

- For the Design Basis Accidents: they include for the normal and shutdown states systems like:
  - Steam Generators
  - Shutdown Cooling System
  - Emergency Water Supply (in case of seismic events)
- For the severe accidents the plant may be cooled by systems like:
  - End shield cooling system and
  - Moderator system

The heat Sink philosophy is well defined for CANDU 6 concept. In this respect, the technically designated features are supported by specific procedures and Abnormal Plant Operating Procedures for transients. There is also a set of procedures, out of which the Heat Sink Manual is the most important designated to define the heat sinks during shutdown periods. SAM procedures are also required to be developed in order to detail the provisions of the Safety Design Matrix in which the heat sinks are postulated for the DBA and BDBA cases. The PSA level 1 results and the researches for PSA level 2 are to support the systematic review of the postulated heat sinks.

In these last areas there are only requirements formulated and update of the information is expected by the time significant progress was made.

ADDITIONAL SOURCES:

- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.
- Cernavoda Unit 1 Operating License, 1999.
- FSAR Cernavoda Unit 1, 1995.
- Cernavoda NPP Heat Sink Manual, SNN/CNEPROD.
- Strategic Policy for Cernavoda NPP Unit 2 licensing process, CNCAN 1997.

ISSUE TITLE: Availability of the moderator as a heat sink (SS 8)

ISSUE CLARIFICATION:

Description of issue

This issue is specific to NPPs with PHWR.

During some loss-of-coolant accidents, particularly for LOCA + loss of ECCS, fuel channel integrity depends on the capability of the moderator to be the "ultimate heat sink". Fuel channel integrity is assured if the calandria tubes do not fail after contact with the pressure tube. In turn the calandria tube temperature depends on the local moderator subcooling. Analysis is performed to show that there is no prolonged film boiling on the outside of the calandria tube. Such calculations depend on a number of computer codes, which must be validated.

Safety significance

An unreliable ultimate heat sink constitutes a threat to fuel channel, and hence fuel integrity under accident conditions. Validated codes are therefore required to provide confidence in the predictions of fuel channel integrity in accidents where the moderator acts as a heat sink.

Source of issue (check as appropriate)

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

MEASURES TAKEN BY MEMBER STATES:

Canada

CNSC staff considers that moderator temperature predictions have not been validated adequately, given the tight safety margins that exist currently. Licensees were therefore required to perform three-dimensional experiments to validate the moderator temperature predictions. These experiments take place as a test facility at AECL's Chalk River Laboratories. They have been completed for the CANDU 9 configuration and the results are in agreement with the predictions of the computer codes. Tests for other designs are underway.

Generic Action Item 95G05 "Moderator Temperature Predictions" was raised by CNSC staff in 1995 to address this issue. CNSC staff also identified a potential scenario whereby a pressure tube failure could lead to loss of moderator. Summaries of the CNSC position statements addressing those issues are given below:

GAI 95G02 - Pressure Tube Failure with Consequential Loss of Moderator

Traditionally, the single and dual failure concept in safety analyses calls for analyses of initiating events, plus analyses of initiating events coupled with failure of one of the *special safety systems*. For the postulated scenario of LOCA plus LOECC, the moderator system has been credited in the analysis as a heat sink. Heat transfer to the moderator is assumed to be via PT contact with *calandria tubes* (CTs) following PT deformation due to heat-up. This mode of heat transfer has been accepted by CNSC staff, since the moderator was considered to be independent of postulated initiating events and ECC failures. However, experiments suggest it is possible for the moderator water to drain during the following postulated scenario: rupture of the PT and then end-fitting bellows, followed by CT failure, guillotine failure of the already ruptured PT, end fitting ejection and drainage of the moderator. This postulated event could result in severe damage to a large number of channels, with consequences in excess of those anticipated in the safety report. In a position statement addressing this GAI, licensees were requested to provide acceptable proposals for a course of action, including possible design changes to be implemented by the end of 2000 that would result in the mitigation of, or at least a significant reduction in, the impact of the consequences of such an event. An industry plan of action was submitted to CNSC staff in May 2000. In this plan, the industry presented its proposed evaluation criteria, including a proposed cost-benefit methodology. Subsequently, CNSC staff has modified its position statement to refer to the CNSC policy on the use of cost-benefit arguments, and to modify the closure criteria and the completion schedule to reflect recent CNSC staff and industry discussions. The industry has submitted the basis for their plans of actions in accordance with the revised position statement for this GAI, and requested closure. Assessment of this submission was on hold, but now that the guidelines for the use of cost-benefit analysis are sufficiently finalized, staff is about to review the measures proposed by the licensees to reduce the potential risk associated with this postulated event. NB Power is considering the replacement of existing seam-welded CTs by seamless (stronger) CTs as part of its refurbishment plan. NB Power has submitted documents describing the CT qualification and verification programs in 2004. CNSC staff's review of these documents led to a request to address water-hammer loading associated with the postulated PT rupture. NB Power committed to submit a report of the results of the water-hammer analysis to the CNSC by October 2004, but this has been delayed.

GAI 95G05 - Moderator Temperature Predictions

In some LLOCA events, the integrity of fuel channels depends on the capability of the moderator to act as the ultimate heat sink. As fuel channels heat up, PTs balloon radically and make contact with the CTs. Fuel channels remain intact upon contact if the moderator fluid outside the CT is cold enough to provide good heat removal capability. Channels may fail, however, if the moderator temperature is too high to prevent the outside of the CT from drying out following contact on the inside with the PT. In view of the severe consequences of channel failures, and the small safety margins that currently exist with respect to moderator temperature (or moderator subcooling) requirements, CNSC staff requested the validation of the computer code used to calculate the moderator temperature distribution against three-dimensional (3-D) integral moderator tests. An industry team representing all Canadian utilities completed the 3-D test in December 2001 to the satisfaction of CNSC staff. This was followed by the validation of the computer code MODTURC-CLAS against both separate-effect tests and the 3-D integral test. In December 2004, the industry team requested the closure of this GAI, and submitted a summary report describing all work completed. CNSC staff is currently reviewing this submission to confirm that the computer code is capable of predicting moderator temperature distribution with acceptable accuracy.

India

In earlier PHWRs (RAPS-1&2 and MAPS) moderator dumping is done to shutdown the reactor. This results in availability of moderator as heat sink in the form of sprays as long as moderator pumps are operating. NAPS onward availability of moderator is considered as heat sink during postulated accident of LOCA + ECCS. Provisions are made for at least 10% process water supply for moderator cooling, which prevents calandria tube dry out. Moderator dumping has been eliminated and fast acting Primary Shutdown System using mechanical shutoff rods has been provided for shutting down the reactor. In the absence of dumping, huge amount of moderator is available as heat sink for cooling of fuel. In case of LOCA and unavailability of ECCS heating of fuel and pressure tube will result in sagging of pressure tube and making contact with calandria tube. Thus, hot fuel will be quenched with moderator and fuel failure will be very much limited. Credit of moderator as a heat sink has been taken in the safety analysis. Therefore, availability of moderator as a heat sink has to be assured. NAPS onward, reactor is tripped on moderator low level. If level comes down further then calandria box-up logic has been provided to isolate calandria from rest of the moderator system whenever moderator level in calandria falls below a specified low level (low level setting is always above the top most channels). Apart from this, all possible sources of moderator leak have been identified. Use of mechanical seal pumps occasionally results in loss of moderator due to seal leakage. In KAPS and future 540MWE reactors canned rotor pumps are being used in place of mechanical seals pumps eliminating possibility of moderator loss from pumps. Moderator heat exchanger tube leak is also one of the source of moderator loss. To prevent moderator heat exchanger tubes failure due to flow induced vibration, mechanical stoppers have been used to restrict excess process water flow in the secondary side of heat exchanger. Good water chemistry is maintained for moderator as well as the process water to avoid corrosion of moderator pipe and calandria. Only welded joints have been used in the moderator system stainless steel piping. Failure of moderator pipe is very remote. Size of the purification lines is kept small. Also no provision has been made for moderator draining from the calandria. Overpressure Relief Devices (OPRDs) of the Calandria has been located at much higher elevation above to minimise moderator loss due to splashing in case a pressure tube and calandria tube ruptures together. Simultaneous rupture of pressure tube and calandria tube together with LOCA due to or header break will not result in moderator loss as header is above the calandria. Probability of feeder break together pressure tube and calandria tube break and failure of ECCS is very remote. There is a provision of fire water injection in moderator heat exchangers in shell side to remove heat under station blackout. Thus availability of moderator as heat sink is assured all the time.

For moderator temperature prediction and calandria tube dryout LOCA condition refer AA-8 and AA-5.

Korea, Republic of

During some loss-of-coolant accidents, particularly for LOCA + loss of emergency core cooling system (ECCS), fuel channel integrity depends on the capability of the moderator to be the “ultimate heat sink”. Fuel channel integrity is assured if the calandria tubes do not fail after contact with the pressure tube. In turn the calandria tube temperature depends on the local moderator subcooling. Analysis is performed to show that there is no prolonged film boiling on the outside of the calandria tube. Such calculations depend on a number of computer codes. Validation of these codes is therefore a safety issue.

Activities

KINS has conducted an assessments of moderator thermal-hydraulic characteristics in the normal operating condition and one transient condition for CANDU-6 reactors, using a general purpose three dimensional computational fluid dynamics code. (CFD) To investigate the thermal-hydraulic characteristics of moderator flow subject to two forces, i.e., momentum force by inlet jets and buoyancy one by heat load inside calandria vessel of CANDU-6, the analyses model has been established through the comparison with the experimental data and a series of the numerical simulation has been performed for the normal operating condition and the transient condition of 35% RIH break LOCA with LOECC. The real geometry of CANDU-6 with 380 fuel channels is simulated in this work to investigate the flow field near the fuel channels. The present model can be expected to be of great help to resolve the moderator temperature prediction item.

The comparison with the SPEL experimental data shows that the present model can reasonably predict the temperature distributions of moderator, which means that the present model has the good capability to properly analyze the fluid flow subject to the buoyancy and momentum force simultaneously. In the transient condition of 35% RIH break with LOECC, no boiling was predicted although there are a sharp peak of heat load about 1 second and the large amount of heat load due to PT/CT contact. So, since the moderator within calandria vessel has enough coolability as the ultimate heat sink, the fuel channel integrity can be maintained and assured. With the major parameters representing reasonably the flow pattern, the flow regime map has been proposed for the safe operation of CANDU reactor and can be applied to predict the flow patterns under some operational conditions.

To verify the assessment results, a 1/8 scaled calandria tank test facility (HU-KINS), to investigate the temperature distribution against three-dimensional (3-D) integral moderator is designed and constructed. Both separate-effect tests and the 3-D integral test are in progress. In the near future, KINS will evaluate the computer code is capable of predicting moderator temperature distribution with acceptable accuracy with experimental results.

Romania

The status is addressed in SS 7.

ADDITIONAL SOURCES:

- CNSC Position Statement 95G05 “Moderator Temperature Predictions”.

4.1.6 Electrical and Other Support Systems (ES)

ISSUE TITLE: Reliability of off-site power supply (ES 1)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

Off-site power supplies are required to be available and to have sufficient capacity and capability to ensure that the fuel and reactor boundaries are maintained within specified acceptable limits.

The main concerns with the off-site power supplies are as follows:

a) reliability of the off-site power system as the preferred source,
b) vulnerability of safety-related equipment to sustained degraded voltage, and
c) adequacy of design interfaces of off-site and on-site power sources.

Safety significance

Reliable off-site power supplies are essential to avoid challenging the backup 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.

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

All nuclear power plants have two decoupled grid connections available for supply from the network. Special studies to determine the independence of this grid connection and events of total loss of grid connection, led to replacement of commutation devices (e.g. switch main grid connection over auxiliary one) in one NPP and improvement of the auxiliary grid connection (by connecting a hydroelectric plant over the auxiliary grid during emergency conditions) in the other. Studies to show the actual network behaviour during dynamic response to a contingency are being performed.

Canada

A reliable off-site power supply is a licensing pre-requisite. This is achieved usually via redundant (independent) grid connections. In multi-unit stations, a local “ring bus” provides the capability of transferring power among units.

In addition, most CANDUs have two independent and redundant sets of on-site emergency diesel generators; either set will assure plant safety if off-site power is lost. For Gentilly-2 there is also an on-site gas-powered electric plant for the emergency conditions. For the other plants there is the possibility of automatic commutation from the main grid to an auxiliary grid connection.

India

Requirement of two independent off-site power supply source is stipulated as mandatory before rector criticality as per the station technical specifications. Auto transfer scheme ensures de-coupling from the grid under degraded conditions of the power grid and operation of turbo-generator on station load. Islanding schemes are introduced wherever feasible.

Over the years several modifications were done to improve the reliability of off-site power supply and capability to withstand grid disturbance/ failures. These are stated below:

a) Electrical equipment were made more rugged & tropical to suit Indian conditions, and modification were done in the areas of imbalance, underpower, under voltage, frequency dips, better insulation and capability to stand high temperature rise over ambient temperature, more frequent start ups etc.
b) Islanding schemes were introduced.
c) Electrical relay protection systems were retuned.
d) Instrumentation supplies were made less sensitive to voltage & frequency dips.
e) Diesel & UPS controls, protection schemes, Auto & Emergency transfer system were designed to withstand grid behaviour.
f) Design margins were increased.
g) Supply from alternate grid is available, in case, the grid to which station is connected, fails.
h) Nearby hydro stations with capability to operate at station service power levels, were identified as dedicated units to supply off-site power in need.i)Periodic coordination meetings are being held with regional grids to analyse problems and carry out modification
j) In a twin unit station there is provision to supply cl 4 IV from operating unit to other unit.

A Generator circuit breaker (GCB) will be provided between the main turbogenerator terminal and 16.7 kV side of the generator transformer (GT). The tap of unit transformer (UT) is taken between GCB and GT so that start up power can also be obtained from grid through GT and UT combination by keeping GCB open. This improve the reliablity of staion auxilary power supply and eliminates fast transfer of operations on Class IV buses to turbine generator trips.

Korea, Republic of

All the Nuclear Power Plants in Korea are required to install two independent offsite power sources, which designed and located so as to minimize the possibility of their simultaneous failure under operating and postulated accident and environmental conditions.

On July 1997 it was found that the offsite power for Wolsong Unit 2 did not met the requirement of two independent offsite power source during the commissioning test before the commercial operation. The remedial action was decided to install the temporary 154kV source fed from Wolsong Unit 1 switchyard by an initiation of the issue.

The 154 kV temporary source backed up as a redundant offsite power source for Wolsong Unit 2 & 3 until Feb 1998 for 7 months. The 154kV source was replaced with a bifurcated 345kV circuit for one year temporarily. A permanent 345kV grid for Wolsong Unit 2,3 & 4 Unit was completed on April 1999.

The Preliminary or Final Safety Analysis Report for Wolsong Unit2, 3 & 4 was updated to reflect the status of offsite configuration. The regulatory review was followed by the analysis of off-site power performance and testability for the electrical power system.

It was reviewed to ensure that the transient stability study was adequate to meet the Wolsong Technical Specification and FSAR for both temporary and permanent grid design configuration change with light and peak load condition when 3 phase bolted short circuit fault near transmission line. Fault study, load flow study, relay coordination study and motor starting study were completed at that time.

Romania

The off-site power supply is assured for Cernavoda NPP Unit1 by two different and independent lines. Evaluation of the reliability of the connection to the grid and the insulation studies were performed as a prerequisite of construction and commissioning phases. This study is currently required to be updated.

In the meantime however the Operating License was issued based on the assumption that the postulated event "loss of off-site power" is of higher frequency than postulated in the generic design. As a result of this situation loss of off-site power (Loss of class IV) was considered together with a LOCA accident as a Design basis Accident instead of a BDBA as in the reference design. The plant safety for this event was demonstrated by analysis in Chapter 15 of FSAR and supported by commissioning tests for various power levels, including 100% of the full power. Unit 2 is considering this postulated DBA event from the very beginning, based on the unit 1 experience. Plant behaviour during insulation period is based on the existence of the internal Standby Diesel Generators (4 x 4.4 MW) and Emergency Power Supply (2x 1 MW) - the last for seismic events, are tested and postulated by analysis, in order to cope with the insulation situation, which covers the basic needs to assure the performance of the safety functions by the safety systems.

The evaluations include both deterministic and probabilistic analyses. The probabilistic analyses performed so far are based on grid availability calculations, Safety Design Matrices and Reliability Analyses for the internal supply in case of various off-site failures, as part of the licensing process and requirements. Various versions of PSA level 1 confirmed so far their convergence with the basic deterministic-probabilistic analyses.

On the other hand it is important to note that there are actions going on to improve the national grid availability so that to comply fully with the EU requirements and plant has some specific permanently monitored electrical components, as for instance the class II invertors.

ADDITIONAL SOURCES:

- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.
- Cernavoda Unit 1 Operating License, 1999.
- FSAR Cernavoda Unit 1, 1995.
- Strategic Policy for Cernavoda NPP Unit 2 licensing process, CNCAN 1997.

ISSUE TITLE: Diesel generator reliability (ES 2)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

Events which result in a loss of off-site 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 start-up reliability of the EDGs has a high level of importance to reduce the core damage frequency.

Safety significance

Improvements in the starting reliability of on-site EDGs will reduce the probability of events which could lead to a loss of heat sink or which could escalate with 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:

Argentina

Increasing of diesel generator reliability is a constant concern. Special reliability studies as part of PSA and careful studies of events during diesel start-up, from Atucha I NPP, led to improved independence (using fire barriers), improvements in start-up logic, automatic tracking of all the start-up steps, and installation of a second independent group of diesel-generators.

Embalse NPP already has a second group of diesel generator for a selected group of accidents. The DG reliability is updated routinely. During the last years emphasis was put into reduce unavailability due to diesel Class III maintenance. For this reason, have been co-ordinated the activities of the different groups involved into a safe condition to be performed in parallel and, at the same time, all necessary works.

Canada

There are two separate groups of diesel generators for safety loads. The Group 1 diesels provide power to safety equipment following a loss of Class IV power. In addition, for emergency conditions or after a seismic event, an independent group of diesel generators (Group 2) is available. Their reliability targets are also demonstrated by periodic testing.

The design for the standby generators provides reliable auxiliary systems, such as fuel oil system, instrument air and cooling system.

Licensees are required to demonstrate, through adequate testing, that standby generators continue to satisfy their accepted reliability targets.

China

It’s the licensing requirement that diesel generator reliability should meet the defined reliability target in China.

India

Reliable starting and operation of Diesel Generators have been an issue requiring considerable attention. In the initial periods, modifications were done in the areas of avoiding low fuel oil trip, increasing of day tank capacity, modification in flashing battery circuit, etc. A circuit which monitors and indicates "diesel ready to start" condition was added and this has proved very useful. Following the common cause failure of all the diesels at Narora due to water ingress from diesel tanker, instituting of administrative controls and hydraulic separation of individual underground tanks were done at all stations. The capacities of Diesel Generators have been increased in the later stations to have more margin. All diesel are physically separated, a third diesel with physical separation, has been added in new stations while the same has been backfitted in older stations. After considerable operational feedback, it was decided that whenever a diesel is operated (example:- when another DG is under maintenance), it will be operated on isolated loads rather than synchronized to off-site source. Some changes were also made in electrical relay protection, co-ordination and tuning to increase the overall reliability. Test frequency and methodology of testing (to avoid carbonization) have been modified. Improved maintenance planning and practices, stocking of spares etc. were done. With all the above the safety margins have further improved.

Emergency Transfer (EMTR)

EMTR Scheme is designed to re-establish Class–III and class-II power supplies in case failure of Power supply to the respective buses.

Loss of Cl.-III Power Supply

On sensing under voltage on Class-III buses respective under voltage lock out relays initiate EMTR action by giving start signal to all DG sets. Once the DG set assigned to the affected bus has attained rated voltage and speed, the DG breaker is closed is closed and voltage is established to the bus automatically. In case the assigned DG to the affected bus fails to come up the standby DG gets automatically connected to the affected bus. Having established the voltage to the affected bus the loads are connected sequentially depending on the importance and at the same time without overloading DG set.

During the period when only one DG set is connected to Cl-III bus some of the load are shed and /or blocked from starting so that total Cl-III load is limited to the rating of one DG set.

EMTR & load shedding systems improve Cl.III realibility.

Loss of Cl.-II Power Supply

On sensing under voltage on class-II buses EMTR action is initiated by issuing start command to the respective DG set and close Cl.-III to Cl.-II tie circuit breaker after isolating all power supply sources connected to the bus.

Alternate cooling to DG sets

DG sets are design to operate without cooling water for short time. Cooling water is normally supplied from cooling water system of the unit. This system operates on Cl.-III power supply.

Provision has been made for supplying cooling water to DG sets from both the Units. In selecting the capacity of pumps for the cooling water system, flows through DG sets and other equipment (cooled by this system) of both the units, have been considered, to be supplied from the pumps of one unit only.

Separate Uninterrupted Power Supplies (UPS) for power and control Loads:

Separate Power UPS and Control UPS are provided for supplying Cl. II power to power and control loads in order to avoid power supply to control loads getting affected while starting heavy loads on Cl. II power supply.

Korea, Republic of

In case of the Wolsung Unit 1, the diesel generator was tripped six times while being started for periodic test. All six trips were caused by overspeed of engine. After the Woodward EGB-50C type governor was replaced with the Woodward 2301A type to resolve this problem, the reliability of start-up was remarkably improved. Recently, it was found that the excessive periodic testing of EDGs of PWR NPPs resulted in accelerated aging due to stress and wear. To mitigate the accelerated aging, the reliability program was established and appropriate reliability goal was determined according to Reg. guide 1.9(Rev.3) and Reg. Guide 1.155. And to meet the reliability goal, the performance and reliability of D/Gs are being monitored continuously through the improvement of test procedure, transition analysis of operating parameters and maintenance management. It was recommended that the CANDU type NPPs also should consider introducing the D/G reliability program apply in PWR NPPs to improve the reliability of D/G.

Pakistan

Complete loss of A.C. power (station blackout) has a potential to cause loss of cooling to fuel and Boosters rods. In KANUPP this could occur, with low probability, if off-site power is lost, followed by failures of both the available standby diesel generators. The possible consequences are however serious enough to warrant attempts to reduced the probability even further specially, as the frequency of loss of off-site power due to grid disturbances is high. On the other hand, it was difficult to achieve the reliability targets of diesel quoted in Final Safety Report due to frequent maintenance.

The study of this potential weakness in the existing design recommended the provision of a third standby diesel generator set, which was installed during 1998. Operation control logic of this diesel generator set having the same rating of that of two existing ones, allows auto start on loss of normal off-site power and can be connected to essential buses manually in case of unavailability of the other two diesel sets. The third diesel generator has been installed and the engineering work for interconnection scheme with essential power supply panel has also been completed. The third diesel will be available for operation by the end of 2005.

Romania

An off-line monitoring of the Diesel Generators availability is being performed, as part of the in service testing program, based on the results of the Class III reliability, which is being performed to demonstrate the design reliability targets for this system. Based on the need to comply with these targets the Licensee adopted the decision to change the initial Diesels in Unit1 and further commissioning tests confirmed their compliance with the imposed by design targets.. The PSA level 1 project developed as part of the Operating License and Strategic Policy requirements for unit 1 are expected to confirm generally the basic provisions in force in this moment for the testing and analysis of the Diesel Generators for unit.1. However there is one aspect for which changes are required for unit 2, correlated with the performance of risk analyses on their impact, by avoiding the synchronization of two standby diesel generators of 4.4 MW on the same bus. This is expected to be done by using two diesels of 8.8 MW as per the initial design. Feedback from operation is also expected to introduce corrections to the process.

ADDITIONAL SOURCES:

- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.
- Cernavoda Unit 1 Operating License, 1999.
- FSAR Cernavoda Unit 1, 1995.

ISSUE TITLE: Reliability of emergency DC supplies (ES 3)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

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; a high reliability and adequate capacity of this device is therefore a prime goal.

In some nuclear power plants, the designed discharge time is of 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 cope better with accident management and station black-out requirements. In case of a station black-out 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 manoeuvrable. Therefore, the reactor can be controlled and can be kept in a safe condition by performing accident management actions. 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, in some plants 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.

Over-operation of the station batteries can result in decrease of DC power to the extent that control indications and signals may become confusing, erratic or spurious.

Safety significance

Insufficient supply by batteries in emergency situations can cause a loss of safety system 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:

Argentina

At the beginning of the plant operation the batteries capability was five hours and its expected lifetime was thirty years. However, the discharge time was decreasing and during the last year only a half of the original time was reached. Therefore, during the last scheduled outage (September/October 2000) the entire batteries were replaced being recuperated the original capability.

Canada

Licensees' maintenance and test programs ensure that battery banks continue to meet their accepted reliability targets.

The design for the newer plants, as Cernavoda Unit 1, uses a longer discharge time than 30 minutes for the batteries, i.e. around 2 to 3 hours. Also some essential users for accident scenarios, such as moderator pony motors, have been added to the loads to be supplied from batteries in case of station blackout. This is also the case for Wolsong NPP in Korea and Qinshan NPP under construction in China.

The batteries are required to be available after a seismic event and therefore are seismically qualified for CANDU plants.

India

Towards assuring the reliability of emergency DC supplies, the following measures have been taken:

- Physical separation of Battery Banks associated cables and controls:

Battery Banks and associated cables and controls are physically separated to avoid common cause failures.

- Seismic qualification of Battery Banks:

Battery Banks and their supports are seismically qualified and same is retrofitted for old plants.

- Split of 100% DC busses into 2X50% buses for ease of Maintenance:

DC buses were earlier of 2x 100% capacity. Each of the 100% buses are split in to 2x50% so the only 50% of the bus may be taken for charging/maintenance. Thus 150% capacity DC bus will be available. During replacement and for newer stations the capacities have been increased to nearly 2 hours as a result of operational feedbacks. Over operation of batteries are taken care by logic interlock actuating at predetermined low voltage. Practice has been instituted to replace all batteries every 8 to 10 years even if they appear healthy.

Korea, Republic of

Generally, it has been decided to replace all batteries every 10 to 12 years.

Appropriate reliability of DC supplies is being kept through the periodic check (monthly, quarterly, annually), service test (duty cycle test) every two years, and performance test every five years.

Romania

Batteries are qualified as per the licensing requirements included in the System and item Classification Lists. Their qualifications covers also the seismicity areas, too. However the availability of class I and II supplies is also subject to reliability analyses and is being maintained by tests and checks on a periodicity as predicted by calculation and verified as part of the Mandatory Test Program during the year. The basis for this approach is similar to that from ES2. It is also to be mentioned that the feedback from operation is expected to correct this process.

ADDITIONAL SOURCES:

- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.
- Cernavoda Unit 1 Operating License, 1999.
- FSAR Cernavoda Unit 1, 1995.
- Strategic Policy for Cernavoda NPP Unit 2 licensing process, CNCAN 1997.

ISSUE TITLE: Control room habitability (ES 4)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

The control room has to be adequately designed to protect the operators from hazards such as steam, toxic gases, radioactive releases, smoke and chemical fumes. The control room should be located in such a way to avoid damage from accidents like main steam line breaks and turbine missiles unless an alternative means of managing the plant safety after such accidents is provided.

The main control rooms of some units are 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.

Safety significance

Loss of control room habitability following an accident, release of external airborne toxic or radioactive material or smoke or steam or other hazardous material can impair or cause loss of the control room operators' capability to safely control the reactor.

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:

Canada

Although not considered to be a “generic safety issue”, control room habitability is a concern, in some postulated accident scenarios, in some of the Canadian plants. Licensees are required to explore and implement mitigating measures such as adequate ventilation, additional shielding, structural strengthening, and re-routing of secondary side piping.

Most CANDU plants have a Secondary Control Area (SCA) from which all safety functions can be initiated and maintained should the MCR become inhabitable.

China

It’s a licensing pre-requisite that the adequacy measures should be taken to keep main control room habitability.

Originally, the HVAC system of the SCA was not seismically qualified. However, the SCA is an area used for maintaining the plant under safety state when the MCR becomes unavailable, especially in case of earthquake, so inhabitable conditions for staff need be ensured. Among them, the HVAC system is a critical one. It’s required that the HVAC system of SCA in TQNPP should meet seismic requirements. AECL has agreed to this design change. Hence, the system design is design basis earthquake (DBE) qualified. Passive equipment belongs to DBE Category “A”, and active equipment belongs to DBE Category “B”.

India

Main control rooms of all plants in India are equipped with survival ventilation system capable of filtering the intake air in case of radioactive releases. Use of this system is by operator action after isolating normal ventilation system. However, in the case of fire in the neighbouring turbine building in one of the units (NAPS-1 event in 1991) when smoke found its way to control equipment room, the operator could not switch off the normal ventilation as the cut off switches were not located in control room itself but near the air handling units away from control room. Suitable modifications were implemented in all plants, based on AERB recommendations on the incident. For Rajasthan units additional systems are installed to enable control room habitation in case of possible H_2S leak from adjacent heavy water plant. Supplementary control rooms and local control panels are being bacfitted in old plants. In new plants control and turbine building are separated.

Korea, Republic of

Licensees are required to explore and implement mitigating measures such as adequate ventilation, additional shielding, structural strengthening, and re-routing of secondary-side piping. For the loss of control room habitability, following an accident, secondary control room is provided in each Wolsung unit.

Romania

The control room habitability is supported by adequate ventilation and abnormal procedures support operators actions for these possible situation. Main Control Room has to be protected by improved design on Main Steam Line Breaks (MSLB), missiles, toxic gases and other hazards. APOP's for Secondary Control Room are also available in case the MCR becomes inhabitable. However organizational actions are taken to compensate this aspect.

The Licensee has on going programs for the inspection of the main steam lines, which might affect MCR as a result of their rupture. There is also a continuous Licensee interface with COG for the generic aspects of this issue.

The site license, construction, commissioning and operation are also based on evaluation and practice organizational actions to control the activities in the exclusion area and around the plant up to 35 km (impact area) in order to avoid the ingress of toxic gases from industrial activities in the Control Room. In case the Main Control Room is not habitable any more, the plant will be operated from the secondary room, as provided by the generic APOP. It is also to be mentioned that the feedback from operation is expected to correct this process. The improving process is expected to be implemented in the licensing of the unit 2 from the very beginning of resuming the activities for its commissioning. In the meantime PSA level 1 results are expected to include future corrections on the operator model.

It is also expected the systematical review under UE review project will conclude on some important actions to be taken to improve these aspects.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Control Room Systems Design for Nuclear Power Plants, IAEA-TECDOC-812, IAEA, Vienna (1995).
- Cernavoda Unit 1 Operating License, 1999.
- FSAR Cernavoda Unit 1, 1995.
- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.
- Strategic Policy for Cernavoda NPP Unit 2 licensing process, CNCAN 1997.

ISSUE TITLE: Reliability of instrument air systems (ES 5)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

Instrument air systems provide motive power for many components in a power reactor. 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. Furthermore, 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.

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:

Argentina

Two new rotating compressors belonging to the instrument air system were installed to gain more reliability. Besides, two of the original compressors system's remain connected as an alternative redundancy. Since beginning of the commercial operation, as a more significant event related with the instrument air systems, the plant had suffered a check valve failure that affected one of the air dryers equipment, causing the pressure down from 7,6 kg. / cm2 to 5,6 kg. / cm2. The pressure in the supply lines was recovered by the operators isolating the air dryer equipment under operation and starting the stand-by air dryer equipment. Besides, it is important to point out that Embalse NPP has an assured nitrogen supply to some essential valves of the pressure and inventory control system as an instrument air reserve.

Canada

This is not an issue in Canada. Key safety related components which require instrument air have redundant supplies, from Group 1 and Group 2.

India

In the initial periods the instrument air system was the cause of many station outages. Over the years several design and operational changes were made to arrive at a satisfactory safety performance and reliable operation. Main areas of improvements were better moisture removal measures, use of oil free compressors, use of more local receivers with better quality check valves, increased capacity of compressors, etc. In some stations, a dedicated diesel driven compressor with associated circuits to supply selected important devices, has been installed to meet station black out conditions. Periodic instrument air system pressure run down tests have been prescribed. Low pressure annunciation and pressure indicators in control room would assist in giving advance information and more time for operator action. In addition, one of the important tests prescribed by AERB is instrument air failure test during commissioning to prove the fail safe features of the design and the capability of the station to go to safe shut down state on instrument air failure at full power. Provision of compressed air storage tank inside Reactor Building takes care of instrument air supply after containment box up following LOCA event. Air supply to containment air lock is backed by independent air storage tank.

Korea, Republic of

Some design changes were taken to get redundancy for the worst case such as valve stuck in close position or to get bypass during valve repair.

Romania

The reliability parameters of the instrument air system were checked in a similar manner with those for diesels. It is to be noted that the instrument air lines have the necessary redundancy by design. In addition to those there was a whole process started by the off-line by comparison to licensing process results from the PSA level 1 versions which indicated that:

- overall reliability data of the instrument air is not adequate and
- data for the part of the instrument air parts which belong to safety systems are to be improved.

As a result modifications were implemented like supplementary air tanks for ECCS and important to safety components operation were implemented and also combined tests for instrument air together with service water and class I and II power failures were tested during commissioning as part of the modified commissioning tests. Procedures were also adequately changed to support operator's actions for these possible situations. It is also to be mentioned that the feedback from operation is expected to correct this process.

It is also important to note that License has in place specific maintenance programs for instrument air lines.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Instrumentation and Control Systems Important to Safety in Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-1.3, IAEA, Vienna (2002).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Management of Life cycle and Ageing at Nuclear Power Plants: Improved I&C Maintenance, IAEA-TECDOC-1402, IAEA, Vienna (2004).
- Cernavoda Unit 1 Commissioning and Operating Licenses, 1994-1995, and 1999.
- Commissioning tests and reports for Instrument air systems and safety systems, Cernavoda NPP Unit1, 1995'1996, as reflected in independently and in chapter 14 of the FSAR Phase I, 1995.
- FSAR Cernavoda Unit 1, 1995.
- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.

ISSUE TITLE: Solenoid valve reliability (ES 6)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

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 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 loss of operability 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:

Canada

The solenoid-operated valves (SOVs) are used as a pilot for AOVs (air-operated valves). Canadian NPPs have not experienced significant problems with the applications of SOV's in safety related systems. This is attributed to clean air-supply system, reliable power supply system and redundancy in safety related system applications of SOV's.

Canadian operating experience reveals that SOV's failures related typically to indirect failures, e.g. failures of SOV's due to failures of air-supply system, or power supply system. Direct SOV failures such as coil failure, passing, or internal body failures were rare.

Following considerations are given to design, selection of materials, maintenance and testing to improve availability of the SOVs:

- Periodic testing in support of system reliability targets,
- Perform regular overhaul of SOVs,
- Selection of the SOVs is consistent with following guidelines:
  - Specify high temperature coil,
  - Environmental qualification for temperature, vapour and radiation conditions
- Specify material properties suitable for normal operating and as well as accident conditions,
- Root cause evaluation of SOV failures

India

Failure of solenoid valves (SVs) of secondary shutdown systems (SSS) of NAPS and KAPS was observed due to burning of rectifiers used for conversion of AC power supply to DC for operation of the solenoid. And also, due to deposition of liquid poison crystals in the SVs, jamming of these valves resulted in common cause failure. Following steps have been taken to overcome this problem and enhance their reliability:

a) all rectifiers of the SVs have been shifted to air-conditioned room for adequate cooling and prevent their burning;
b) all SVs have been relocated at higher elevation and pipe bends are used to prevent liquid poison ingress in these valves during actuation of SSS; and
c) Surveillance of these valves is done periodically.

Some problems were also faced in a few solenoid valves used in fuel handling system and these were rectified after exhaustive trouble shooting.

Considerable drop in voltage was observed in the power supply circuit for the solenoid valves meant for opening/closing of ASDV-20 at TAPP-4. Modification has been incorporated to overcome the problem. This modification as well as all other modifications carried out to eliminate the above problem would be reviewed in detail. It will be also confirmed that such fault does not exist in the other safety related circuits and future plants.

Korea, Republic of

There are several solenoid actuated valves related to the safety in Wolsong unit1. The solenoid valves are installed at the containment isolation system. The valves are tested periodically following the ASME OM Code ISTC. The test items are stroking time, valve position, fail-safe and leakage rate.

Generally solenoid valves are used in not only safety related systems but also non-safety systems.

Solenoid valves are required high reliability because damage of solenoid coil results in failure of systems.

In general, it is not until the solenoid valves don't operate by demand signal that we know that solenoid valves have been breakdown.

Therefore, utilities measure the resistance of the coil during the overhaul.

They have replaced the solenoid to raise the reliability if there are some deviations when compared with previous value of resistance.

Romania

The reliability parameters of the solenoid valves were part of a check, similar to those for diesels. In addition to those it is to be mentioned that the tests were guided during commissioning by Key (safety) Commissioning Objectives, based on the data to be demonstrated in the FSAR. A document called Safety Analysis data List was defined for all systems and parameters for valves were also part of this system of testing in commissioning Based on the analysis results were also derived data for the Mandatory Test program defined for systems with safety functions and consequently for all components, which is applied during operation. Procedures were also adequately changed to support operators actions for these possible situation. It is also to be mentioned that the feedback from operation is expected to correct this process.

However it is important to note that no problems were encountered by the Licensee with the solenoid valves.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Management of Ageing of I&C Equipment in Nuclear Power Plants, IAEA-TECDOC-1147, IAEA, Vienna (2000).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Management of Life cycle and Ageing at Nuclear Power Plants: Improved I&C Maintenance, IAEA-TECDOC-1402, IAEA, Vienna (2004).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Incident Reporting System (IRS) Report No. 397, “Non-operating and non-closure of pilot valves due to corrosive coating”.
- Cernavoda Unit 1 Commissioning and Operating Licenses, 1994-1995, and 1999.
- Commissioning tests and reports for Systems with safety functions, Cernavoda NPP Unit1, 1995'1996, as reflected in independently and in chapter 14 of the FSAR Phase I, 1995.
- FSAR Cernavoda Unit 1, 1995.
- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.

4.1.7 Instrumentation and Control (incl. Protection Systems) (IC)

ISSUE TITLE: Inadequate electrical isolation of safety from non-safety-related equipment (IC 1)

ISSUE CLARIFICATION:

Description of issue:

This issue is also applicable to NPPs with LWR.

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 safety systems are transmitted to control or display equipment, such as the safety parameter display system (SPDS).

Electrical isolators include fibre-optic and photo-electric couplers, transformer-modulated isolators, current transformers, amplifiers, circuit breakers, and relays.

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 Safety System component which may lead to unwanted operation of other devices, while a lower level of energy could generate electrical noise that could cause the isolation device to give a false output.

Safety significance

The signal leakage through inadequate isolation devices to safety-related circuitry could damage or seriously degrade the performance of the Safety System components. In other cases, electrically-generated noise on the circuit may cause the isolation device to give a false output. All these may cause the impairment 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:

Argentina

Embalse NPP mainly utilises opto – couplings, isolation transformers and isolation amplifiers.

Canada

In all current designs, safety-monitoring systems are engineered and tested for proper isolation from automatic safety system actuation subsystems and equipment

Modifications need to be implemented in some older units to alleviate this concern. Particular care is required in computerized monitoring system retrofits to older plants.

India

Indian design of PHWRs use mainly optical isolation, isolation transformers and amplifier to isolate the signals of the safety equipment from noise and non-safety related ones. There have been some occasions which brought out lack of, as well as failures of isolation. Magnetic coupling and interference have also been observed. Considerable difficulty is experienced to track these as they occur sporadically and adequate detection technology is not available. In new plants, PLC for cotrolling the safety and non –safety related loads are separated. Programmable digital comparator system (PDCS) for safety and non-safety systems is also separted in 540 Mwe plants. In old PHWR uninterrupted power supply (UPS) systems for control systems and normal class II electrical power supply has been separated. In current designs, even the UPS supplying safety related control equipment and non-safety related equipments are segregated.

Korea, Republic of

The isolators used in Korean NPPs are qualified as Class1E components. Use of any isolators which is not satisfied with requirement of IEEE Std 323 (Qualifying Class 1E Equipment for NPPS) and IEEE 603 (Digital computers in Safety Systems of NPGS) would be a violation of FSAR which committees the use of these IEEE Stds. If any additional installation or modification of signal loop of safety system is implemented, licensee should report it. Therefore the qualification of EMI or EMC requirements should be validated in the process of reviewing the report of modification or change of safety systems.

Romania

The requirements for the electrical isolation of safety equipment is based on the Systems and Items Classification Lists, based as showed in GL1 on the Licensing bases requirements. These aspects are covered at Cernavoda NPP by design. All the safety significant items on electrical side, needing isolation have to comply with the 1E of IEEE requirements as per the EQR requirements.

Unit 2 requirements have however some modifications by comparison to the requirements for Unit1 ones related to Environmental (in the inside buildings) qualification due to the enlargement of the DBA and BDBA postulated for it.

The differences for unit 1 are under identification and are to be included in the Periodical Safety Review process from May 2001. However no major aspects impacting on safety are expected.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Management of Ageing of I&C Equipment in Nuclear Power Plants, IAEA-TECDOC-1147, IAEA, Vienna (2000).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Management of Life cycle and Ageing at Nuclear Power Plants: Improved I&C maintenance, IAEA-TECDOC-1402, IAEA, Vienna (2004).
- Strategic Policy for Cernavoda NPP Unit 2 licensing process, CNCAN 1997.
- Cernavoda Unit 1 Commissioning and Operating Licenses, 1994-1995, and 1999.
- EQR for electrical systems for Cernavoda NPP Unit1.
- FSAR Cernavoda Unit 1, 1995.
- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.
- System Classification List procedure, Cernavoda NPP.

ISSUE TITLE: I&C component reliability (IC 2)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

Safe reactor operation requires comprehensive instrumentation to actuate the reactor protection system and other I&C systems that may be necessary. The I&C equipment of NPPs are based on different technologies which can present in some cases reliability problems. Operational experience has shown that the I&C failure rate is relatively high in old NPPs due to technological obsolence and ageing.

Some I&C system designs 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 accident conditions, including the instruments used in the reactor, thermocouples, pressure transducers, flow meters etc.

Safety significance

Poor I&C reliability may result in the impairment of safety systems. The effect can worsen with age unless there is a good ageing management programme in place involving maintenance and replacement of I&C components. Also of concern is the obsolescence of digital equipment, particularly if the original equipment is no longer available.

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.

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

In order to increase the reliability of safety-significant I&C, two important activities have been performed recently. Modification of the nuclear reactor protection system of CNA1 by the addition of two new trip signals: implemented with EDM ISKAMATIC & TELEPERM C Siemens Technology instead of the old SIMATIC N and, total replacement of Vanadium & Platinum neutron flux vertical detectors & the associated wiring in CNE due to obsolescence & aging.

– Total replacement of neutron flux vertical detectors in CNA1 due to aging.

– Replacement of the mercury-wetted relays used in the safety system of CNE with doped relays due to reliability component reasons. An increase in relays failure mumber was detected and it was decided to analyse the causes. The results of this analyses indicated that mercury wetted relays type suffered mercury degradation (ageing) that provokes the sticking of the mercury with the relay electrical contacts. It means that contacts remains closed. At that time, all CNE safety systems had this types of relays as following: SS#1 and ECCS use C.P. Clare relays which are mercury wetted relays (without tin). SS#2 , Containment system and EWS use Potter & Brumfeld relays and the 40 % were doped with tin. The different relay suppliers are for diversity design requirements. To reduce relay failures it was decided to replace mercury wetted relays by mercury wetted relays doped with tin At present, 50% of the relays were already replaced according to what have been programmed.

Special considerations have to be taken into account regarding maintenance programs for solid state components and other I&C components related to safety in all NPPs.

Reactivity device's (shutoff rods, absorbers and adjusters) cables and connections from the panels to the reactivity desk were replaced. The new cables and connectors fulfil the corresponding environmental qualification requirements for the new CANDU plants.

Canada

The I&C component of safety systems and the reactor regulating system has not been a significant source of unacceptable reliability. Some components, notably mercury-wetted relays, have been replaced in order to improve I&C reliability. Obsolescence of digital equipment is a concern, particularly if the original equipment is no longer available.

India

Periodically I&C component failures/problem, both of isolated and generic nature, have been experienced in the Indian PHWRs. The older units used thermovolts for generating reactor trip contacts for some process parameters. AERB asked for a detailed review of failures, both safe and unsafe and plan of corrective action including phased replacement and these reports have been submitted. In some areas thermowell type of RTDs and thermocouples were unreliable and causing problems. These have been replaced by surface mounted RTDs. High temperature in control room and adjacent rooms used to result in common mode failure of several printed circuit boards. These were corrected by increasing the reliability and capacity of cooling systems, specifying higher ambient temperature to the manufacturers, etc. There have been a few instances of failure of junction boxes due high humidity in the pump room/boiler room and the utility has imposed an administrative limit on the steam leak in these areas after which unit is shut down to carry out rectification. Effect of instrument air on I&C reliability especially if moisture and oil is present, is dealt with in issue No. ES5.

Instrumentation systems used in older units suffer from lack of vendor support, spare parts, etc. (see item MA1). Phased replacement/upgradation of these have been instituted. An example of this is the instrumentation system upgradation in the light water dousing system of RAPS No. 1 and 2. Issues connected with upgradation by replacement with computer based systems is covered under issue No. IC 4. Some of the replaced I&C components started failing under high radiation field. Installing additional shielding/relocating as temporary measures implemented. Long term plan to sort out this issue is being followed with suppliers. To improve reliability of I&C components, impulse lines are SSE qualified.

For thermowell type and surface mounted type RTD reliability parameters are used from generic data sources i.e. IEEE and IAEA TECDOC.

I&C systems for which reliability analysis is carried out include PDCS, PLC, DPHS, RRS, ICMS and ECCS test facility.

Korea, Republic of

In order to increase the reliability of safety-significant I&C, the following activities in old plant are performed:

– replacements of neutron flux detect assemblies
– replacements of mercury-wetted relays
– upgrade of ion chamber signal processors
– SDS #1 PDC replacement plan

Pakistan

The instrumentation and control equipment installed at KANUPP was based on mid 60's technology. To overcome the problems of aging and obsolescence, it was planned to replace the existing Computers, important C&I Loops and Panel C&I devices with advanced Process Controls & Instrumentation.

The "Computers, Control and Instrumentation Back-fitting Project" was initiated by KANUPP in early 1990s. After completion of civil work, major infrastructure related activities and installation work was completed by year 2000. Remaining infrastructure related activities, installation work, pre-commissioning, testing and commissioning was completed between year 2000 and 2003. Plant is now operational since January 2004 using new I&C Systems.

Plant regulating computers (ARC and BRC) have also been replaced by new I&C systems based on PLC's. Following distinct plant functions are now being performed by these systems:

i) Reactor Power Regulation

ii) Fuel Channel Temperature Monitoring

iii) Bearing Temperature Monitoring of Pumps & Motors

iv) Failed Fuel Activity Monitoring

v) Alarm Annunciation

Twenty seven (27) measurement loops and fifteen (15) closed-control loops have been replaced in year 2003 outage. Commissioning of remaining C&I conventional and safety loops will be carried out in 2006. Safety Parameters Display System (SPDS) and Critical Parameters Display System (CPDS) have been installed and are functional.

Romania

The reliability parameters of the I&C components was part of a check similar to those for diesels. In addition to those it is to be mentioned that the tests were guided during commissioning by Key (safety) Commissioning Objectives, based on the data to be demonstrated in the FSAR and the component priority defined as per the systems safety functions. A document called Safety Analysis Data List was defined for all systems and parameters for valves were also part of this system of testing in commissioning. Based on the analysis results also derived data for the Mandatory Test program defined for systems with safety functions and consequently for all components, which is applied during operation. Procedures were also adequately changed to support operators actions for these possible situation. It is also to be mentioned that the feedback from operation is expected to correct this process.

As a result of this process Licensee decided to perform a so called 'Fix the Plant' program, to solve the problem of low reliability parameters for some plant I&C components; the program is close to completion.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Modernization of Instrumentation and Control in Nuclear Power Plants, IAEA-TECDOC-1016, IAEA, Vienna (1998).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Management of Ageing of I&C Equipment in Nuclear Power Plants, IAEA-TECDOC-1147, IAEA, Vienna (2000).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Management of Life Cycle and Ageing at Nuclear Power Plants: Improved I&C Maintenance, IAEA-TECDOC-1402, IAEA, Vienna (2004).
- Cernavoda Unit 1 Commissioning and Operating Licenses, 1994-1995, and 1999.
- Commissioning tests and reports for Instrument air systems and safety systems, Cernavoda NPP Unit1, 1995'1996, as reflected in independently and in chapter 14 of the FSAR Phase I, 1995.
- FSAR Cernavoda Unit 1, 1995.
- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.
- ATOMIC ENERGY REGULATORY BOARD, "Safety Related Instrumentation and Control for PHWR based NPPs". AERB/NPP-PHWR/SG/D-20 (2003).
- FSAR Narora Atomic Power Station 1&2.

ISSUE TITLE: Lack of on-line testability of protection systems (IC 3)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

The protection system designs of some old plants do not provide for on-line testability.

During normal operation, protection systems are in standby, therefore failures of components may not be detected. 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 should ideally 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.

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.

Testing has a direct impact on the availability of safety-related systems. Safety systems in current PHWR plants are designed for testability on-line. However, 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.

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 normal operation, 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:

Argentina

As a design requirement must to be performed periodical tests to the reactor protection systems trending to demonstrate that such systems are in an available and operative state. Tests results are included into a Safety System Annual Report that allow a performance evaluation and perform the necessary corrections and / or modifications.

Canada

On-line testability of protection systems is a regulatory requirement. Licensees routinely report (quarterly) availability figures for protection systems, on the basis of test results and failures recorded during the period. Licensees are required to demonstrate that safety system reliability targets continue to be satisfied.

India

Most of the parameters of reactor protection system and engineered safety features have in-built on line test facility. For some parameters on- line test facility is susceptible to human error although in safe direction. A few cases do exist where on-line/on -power testing is not envisaged in initial design, as the test frequency is comparatively less. Testing these parameters need reactor shutdown. This problem was not appreciated earlier as plants were not operating continuosly for long periods. Presently Indian NPPs operate more than a year without shutdown. Therefore current injection test method has been retrofiited in some plants and being incorporated in rest of the plants where such deficiencies exist. Newer reactors have finite impulse test systems to continuosly check healthiness of electronic circuits and equipment. The AERB code on design makes this mandatory for new stations. Apart from on-line testing of the electronic parts of protection systems, the on-line testing of shutdown rods clutch release for a limited drop is demonstrated in 540 Mwe PHWRs.

Non-availability of some systems during testing (eg.:- dousing system in RAPS No. 1 and 2 where two channels are blocked during testing) has been factored into engineered safety feature non-availability calculations/target used for safety analysis.

Continuous efforts have been put to avoid errors during manual testing. These include better check lists, different colour check lists for each channel, introducing testing by electronic injection circuits upto detector and the whole circuit during outages, etc.

Korea, Republic of

On-line testability of protection systems is demonstrated "Operable" by the performance of the surveillance test periodically, which is described on the technical specification to satisfy the safety reliability targets. Licensee should report that the failure related to shutdown system occurred.

Romania

This operability is being checked as part of the self assessment program to review the operational feedback system for all the plant systems/components, including the protection systems. The necessity of testability of the systems (i.e the list of these systems) was based on the requirements from Safety Classification Lists and on the results from the probabilistic analyses (review of reliability analyses, safety design matrices and PSA level 1 various versions). It is expected to get new confirmations as part of the new PSA level 1 developed this time for unit 1 as licensing conditions, in the Periodical Safety Review Program.

However there is no event encountered so far by Licensee and / or emerging problem related to this topic to justify its definition a Generic Safety Issue.

ADDITIONAL SOURCES:

- PSA level 1 Cernavoda NPP unit 1, results as included in the reports to IPERS mission in 1995.
- FSAR Cernavoda Unit 1, 1995.
- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.
- Cernavoda Unit 1 Operating License, 1999.
- Strategic Policy for Cernavoda NPP Unit 2 licensing process, CNCAN 1997.
- ATOMIC ENERGY REGULATORY BOARD, Design Safety Guide, “Safety Related I&C Systems”, AERB/SG/D-20.

ISSUE TITLE: Reliability and safety basis for digital I&C conversions (IC 4)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

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. Digital systems have been used for decades in the control of PHWRs, as well as in the shutdown systems (first used in Darlington).

Since digital technology is considerably different from analogue technology, the criteria appropriate for the safety review of digital computer-based systems are different.

It is essential that cost-effective review methods be developed for digital safety 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:

Argentina

Non consistent measures at the input analog system in one of the control computers (DCCX) in CNE has been detected due to a printed circuit board (PCB) malfunction. A PCB containing the analog digital converters for analogue input signals corresponding to computer DCCX has been replaced. Maintenance procedures were improved to consider failures on Digital I&C conversions.

Canada

Canada has for some years been in the forefront of developing and deploying sound criteria for the reliable implementation of digital computers in safety systems, most recently in Qinshan, W2/3/4, and Darlington projects. It is not a significant technical issue for new plants but the costs associated with review can be high. Care is needed in retrofits.

India

AERB has been extremely conservative in permitting use of digital technology, computer based and software systems. Problem of verification and validation is one of the main reasons for this conservatism. Use of such a technology has been permitted, initially only for display systems to gain experience and assess reliability for each Postulated Initiated Event Atleast one hardwire signal is available for reactor protection. AERB is setting up a centre for V&V of these equipment and systems. AERB Guide on computer based system is in the final stage of preparation.

Korea, Republic of

Some operating safety systems in PWR have been upgraded using digitalized signal processors instead of analog type processors. There is no digital upgrade in CANDU special safety systems in Korea. However SDS #1 programmable digital comparators (PDC) are planned to be replaced as with new model 200. Regulators are now reviewing application documents using the evaluation guideline which was developed recently.

Romania

Even if there are no problems encountered yet for this topic, there are clear intentions of the Licensee to change under a special licensing program for Unit 1 the Marconi and Nuovo Pignone systems, since they might affect some safety support functions. For unit 2 the change will be done from the very beginning of the commissioning phase.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Specification of Requirements for Upgrades Using Digital Instrument and Control Systems, IAEA-TECDOC-1066, IAEA, Vienna (1999).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Verification and Validation of Software Related to Nuclear Power Plant Control and Instrumentation, Technical Reports Series No. 384, IAEA, Vienna (1999).
- ATOMIC ENERGY REGULATORY BOARD, Design Safety Guide, “Computer Based Safety Systems”, AERB/SG/D-25.

ISSUE TITLE: Reliable ventilation of control room cabinets (IC 5)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

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 to malfunctions of the electronic equipment in the control cabinets.

Safety significance

While most PHWRs have two independent control areas from either of which all the safety functions can be carried out, malfunctions of instrumentation in the main control room cabinets might produce a common mode failure in control and (some) 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:

Argentina

An additional conditioned air system was assembled in the control computers room to improve the ventilation system reliability.

Canada

Although a few incidents related to instrumentation malfunction, over the past 30 years, have been attributed to problems with cabinet ventilation in the control equipment room of some units, this is not considered to be a generic safety issue.

In addition most CANDUs have a secondary control area, with independent ventilation, which could be used to perform all the required safety functions should the equipment in the MCR become inoperable for any reason.

India

High temperature in control room and adjacent channel rooms caused a few cases of common mode failure of several control circuits.

Under-rating and improper design of Control Area Ventilation System has caused such failures. Increasing capacities of chillers, rerouting of some ducts, augmentation with some local packaged cooling units, increased leak tightness etc. have been implemented towards eliminating the problem. Air-conditioning systems have been provided with adequate redundancy.

Korea, Republic of

This is not considered to be a generic safety issue.

Romania

There is a maintenance program on going. Based on the Licensee experience got so far we may consider for the time being that this issue is not a generic safety one. However it is expected that the results of the actions foreseen for the issue ES4 might bring new information on this issue, too.

ADDITIONAL SOURCES:

ISSUE TITLE: Need for a safety parameter display system (IC 6)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

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. 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 extenuating circumstances.

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:

Argentina

A safety parameter display system (SPDS) was required to be installed in both NPPs CNA1 and CNE. CNA1 installed an SPDS (VISUAL DATA) and CNE has a program to install also another one.

Two monitoring equipment for the safety systems were assembled. These monitoring equipment were installed to both the shutdown system # 1 and the one equipment was assembled in the shutdown system # 2. The Two monitoring equipment are available in the main control room and they shows the safety systems parameters state. Besides, they give an alarm when any parameter become near its set point.

Canada

“SPDS” systems are considered optional due to the use of specially identified and environmentally qualified post accident monitoring (PAM) instrumentation, which is supplemented with computerized monitoring features that are an integral part of the plant design. Current CANDU designs (e.g. Qinshan) include advanced computerized monitoring features which equal or surpass those of “SPDS” systems.

China

In TQNPP, Main Control Room (MCR) has been upgraded as follows:

1. To provide two displays with big screen of 100 inches each and the attached plant display system, which are used for displaying important plant parameters so that operators are able to know the plant status timely and completely.
2. To optimize CRT alarm annunciation system and intellectualize the alarm signal.

3. To improve the MCR artistic design, including color, light, floor and noise elimination as well.

The above three design changes should keep all originally designed functions unchanged. AECL promised that even if the two displays with big screens and alarm annunciation systems fail, plant operation will not be affected.

Critical Safety Parameter Monitoring System (CSPM) is not required for AECB licensing in Canada, however important safety information under accident conditions remains the post-accident monitoring system (PAM). This is the present CANDU design status.

CSPM system shall be established in compliance with Chinese regulations GB-13624-92 and IEC (1988) Function Design Criteria for Critical Safety Parameter Monitoring System". The practice for PWR at present is to select parameters as minimal as possible. This is convenient for operators to evaluate the plant safety status under normal, transient and accident conditions, and is helpful for them to make expedient and correct judgment. It has to be acknowledged that this is a good practice, and it is a design concept which is independent of reactor type. AECL, at the request of the Owner, agreed to add CSPM system and provide a special CRT as CSPM display. The detailed parameter will be selected from the plant computer system can be used for the MCR.

24 Hour Historical Data Storage (HDS) System is added in TQNPP. As per the original CANDU design, historical data storage time depends on sampling frequency of a certain parameter. Parameters with high sampling frequency can only be stored for a few minutes. TQNPC has requested 24 hours storage so that the operators and technical support center can analyze cause of the event. AECL has agreed to make the change. Consequently, HDS server is added and connected to plant computer system; however, the original computer function remains unchanged. There will only be a need to transfer data, which is to be stored, to the HDS server, then the data will be able to be shared by CSPM, Technical Support center (TSC), and emergency response center (ERC).

Technical Support Center (TSC), which will be functional under accident conditions, is designed in TQNPP. Technical support staff will assess the plant status and provide support via operators. However, they will not excessively interfere with the control room activities. The room number of the MCR is S-326. Room S-324, which is next to the MCR, functions as the control room under normal operation, from where the staff can coordinate and approve normal operation and maintenance activities. Under accident conditions, the room will be changed to TSC. Adequate information on plant state and technical documents can be obtained in Room S-324. Room S-232A under the MCR is used for technical discussion associated with decision-making during and after emergency situation. It is also part of the TSC and is equipped with a telephone and a fax machine.

India

Old Indian NPPs were not fitted with Computerised safety parameter display system. In the current design NPPs, computerized Operator Information System (COIS) gives information on all operational parameters including safety parameters, which helps the operator to take necessary action. One of the Visual Display Unit (VDU) of COIS on the main control panel is kept dedicated to safety parameters administratively. The older units are currently retrofitted with computerized visual display units. Full fledged safety parameter Display System is installed in 540 MWe NPPs. It would be advantageous for the operator if processing of the inputs is done by artificial intelligence units to indicate summarised display like core cooling healthiness, confinement healthiness, etc.

Pakistan

The Safety Parameter Display System (SPDS) and Critical Parameter Display System (CPDS) are part of KANUPP safety upgrades implemented during the C&I back-fitting. Primary objective of SPDS is to provide concise and integrated information to assess the plant safety status continuously under all modes of operation and to guide the operator to take corrective actions when so required, to bring and maintain the plant in a safe state. SPDS provides the related process information on mimics and trends. The state of Critical Functions is determined by a fixed set of algorithms developed specially for KANUPP. Computation of these algorithms is performed continuously and results display on the SPDS stations. CPDS monitors critical plant parameters. The information from these two systems is provided to the operators and technical management through display stations in the Main Control Room and Emergency Control Center.

Romania

The APOP's control from the procedural point of view very well this aspect, i.e. critical parameters, governing conditions etc. On the other hand there is an existing on going program SCADA, but which is not intended to be used for "safety display". There is also implemented in the plant a data acquisition system for SDS1 and SDS2 to monitor "margin to trip". Finally it may be concluded that for long term actions it is foreseen in this moment that there will be the need for a safety parameter display system.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Control Room Systems Design for Nuclear Power Plants, IAEA-TECDOC-812, IAEA, Vienna (1995).

ISSUE TITLE: Availability and adequacy of accident monitoring instrumentation (IC 7)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

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 primary heat transport system and the containment. The information obtained on the effluent from ventilation ducts during accidents should also be reliable and accurate, to predict expected doses accurately.

Safety significance

Most long-term (and some short-term) emergency response and recovery actions are taken by operators. A lack of appropriate information provided to the operator could result in suboptimal accident management, especially in highly-stressed circumstances. In this case, human errors could become major contributors 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:

Canada

Qualified post accident monitoring instrumentation (PAM) provides the necessary information for adequately dealing with the full range of conditions. In the current designs (Qinshan) computerised information displays are tailored for compatibility with the emergency operating procedures.

India

India commenced preparation of “Operation Procedure Under Emergency Conditions (OPEC equivalent to EOPs) in 1977. During the preparation of these, it emerged that a better operator support system would be required for (a) knowing status after an accident (b) advise on corrective actions in an anticipatory manner. In addition special Task Forces appointed after TMI-2 & Chernobyl accidents and Narora fire incidence to arrive at lessons & suggest corrective measures for Indian NPPs also recommended better display systems. During preparation of OPECS & deliberation of the two task forces, it emerged that several features need to be added to keep track of accident progression (remotely as approach may not be possible). All those have been backfitted in old reactors and included in the design of new reactors. Examples are additional shielded sampling systems, consolidating all parameters to indicate core healthiness etc. The experience gained in all above rewiew was used as input to arrive at operator response time in AERB Safety Guide.

Korea, Republic of

Post accident monitoring instrumentation (PAMI) variables are analysed by final safety analysis report and described in technical specification such as identification of PAMI instrumentations, limiting condition and operation (LCO), surveillance requirements. There are 33 PAMI parameters in technical specification. Surveillance requirements for PAMI parameters are specified channel check, channel calibration and channel functional testing. Also Indicators in MCB are designed by using different identification methods (nameplate, color, etc,) compared with those of normal operating indicators.

Therefore PAMI variables in MCR board are identified by using different identification methods as well as maintained availability by surveillance tests.

Pakistan

After the accident at TMI-2 it became a regulatory requirement to install additional instrumentation, qualified for a post-accident environment, which could provide the essential information in main control room to indicate whether or not the plant safety functions are being accomplished after an accident.

The Accident Monitoring Instrumentation (AMI) available at KANUPP has been reviewed and where the existing equipment fulfills the requirements of AMI, it has been selected as an AMI parameter. However the existing instrumentation of KANUPP does not cover the full scope. In order to meet the modern design criteria for AMI, some new instruments have been identified that would be installed to meet the prevailing design standards as far as practically possible.

Romania

It is part of the BDBA and SAM process as part of the Periodical Safety Review. After 2001 it is expected to have details on the topic to be included in this document relevant information is also included in AA5.

In this moment these aspects are being considered at the organizational level as defined in the Emergency Plans.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Management of Life Cycle and Ageing at Nuclear Power Plants: Improved I&C Maintenance, IAEA-TECDOC-1402, IAEA, Vienna (2004).

ISSUE TITLE: Water chemistry control and monitoring equipment (primary and secondary) (IC 8)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

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.

Safety significance

The chemical monitoring system is essential to keep coolant parameters within prescribed limits. If it is not, piping may degrade more rapidly than expected and 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:

Canada

Although not labelled as a "generic safety issue", lack of adequate chemistry control in the primary and secondary sides is a concern, and has been the cause of material problems.

India

Chemical Control Systems & Procedures have been streamlined to a great extent in Indian PHWRs. The parameters that need to be controlled as well as their limits are part of the technical specifications. Round the clock coverage is given by the Chemical Control Laboratory and it has been found to be satisfactory. However, need for on-line monitoring has been felt in some vital areas and a satisfactory solution is yet to be put in place even though limited success has been achieved. These areas are as follows: -

a) Deuterium in helium cover gas moderator system (This has been achieved for RAPS-1)
b) Moderator system conductivity (this has been achieved)
c) On-line tritium detector for quick detection of Heat exchanger tube leaks especially quick detection of steam generator and moderator heat exchanger leaks.

For detection of leaks in Steam Generators and Moderator Heat Exchangers, N-16 monitors are being used online. Tritium in water monitors are used for detecting leaks in D_2O heat exchangers while reactor is in shutdown state.

Korea, Republic of

Adequate chemistry control in the primary and secondary sides is important to avoid corrosion problems in main equipment. The chemical conditions and their limits which are specified in the technical specifications and Design Manual are checked periodically by manual operation or continuously by the on-line chemical monitoring system (OCMS).

The OCMS can evaluate the secondary water chemistry parameters, for example, conductivity, pH and the concentration of sodium, dissolved oxygen and hydrazine. Ion-chromatography, Ultra-Violet Spectrometer or Atomic Absorption Spectrometer are used to evaluate the chemistry parameters which can not be obtained by the OCMS. It is essential that this OCMS will be improved in order to evaluate more chemistry parameters.

There are a lot of equipments monitoring water (primary and secondary) which affects the safety as well as corrosion to the facilities. Some parameters to be controlled related to the primary heat transport are described on the technical specification.

Conditions to be monitored for water through the on-line are as follows:

- Primary
  - D2O in H2O
  - Gas Chromatography(moderator cover gas, liquid zone controller cover gas, and heat transport storage tank gas)
  - Conductivity in Moderator.
- Secondary
  - Condenser: Sodium, Conductivity, PH
  - Feedwater: PH, Dissolved Oxygen
  - Steam Generator: Conductivity
  - Deaerator: Dissolved Oxygen

Pakistan

The long-term performance of process equipment depends entirely on strict control of chemical parameters of the process fluids, which are the main cause of ageing phenomena. If the chemical parameters deviate from their normal tolerances, even for short periods, the ageing phenomena can accelerate tremendously, usually with irreversible effects, which show up later, multiplying the possibilities of catastrophic failures later in plant operating life.

Presently KANUPP does the necessary routine monitoring by manual sampling for analysis in chemical control laboratory. This method is cumbersome, inaccurate and above all, limited in frequency. Most chemical parameters cannot be closely monitored more than once per shift.

KANUPP intended to improve the monitoring of its chemical parameters and accordingly decided to install on-line chemical instrumentations. The required on-line chemical instrumentation has been arranged, and installation work is in progress. An integrated system with individual instruments connected to a centralized and computerized sampling, display and recording systems has been established. This will cover on-line monitoring of the following parameters:

a) H_2, O_2 and N_2 monitors in cover gas

b) Sodium

c) Silica

e) Dissolved oxygen in feed water and primary coolant

f) Dissolved Hydrogen / Deuterium

g) pH and conductivity

The on-line chemical instrumentation for boiler (SG) feed water system has been installed during plant long shut down of year 2003.

Romania

Chemistry parameters monitoring is part of the plant procedures system and they are monitored and periodically reported. No significant problems were encountered so far. A special attention is also devoted to some chemistry aspects like heavy water to light water detection systems, chemistry of the systems with safety functions and radiochemistry aspects.

During the licensing process there were some improvements implemented for the chemical control of the plant and some others related to the secondary side are also to be done.

ADDITIONAL SOURCES:

ISSUE TITLE: Establishment and surveillance of setpoints in instrumentation (IC 9)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

Appropriate surveillance procedures and setpoint methodology for instrumentation and control systems important to safety are required to ensure the system operability.

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 impairment of 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:

Argentina

Decalibration on set points of the modules comparators (Siemens; technology SIMATIC N) in CNA1 belonged to signal neutron flux rise time has been detected. Calibration must be performed with the module inside the rack by documented procedures. Calibration have been done with the module outside the rack without following the corresponding procedures. Improvements on the procedures and surveillance were required.

Canada

The licensing envelope is defined in several documents such as the plant operating license, the safety report, the plant operating policies and principles, and the plant operating procedures. Limits for the safety system initiating parameters are derived such that the safety goals are achieved for all design basis events. These limits are corrected for random, systematic and other errors inherent in the process of periodic testing and calibration; subsequently, the limits are used to derive parameter impairment levels and setpoint tolerances. The following are the requirements for safety parameter tolerances:

1. In keeping with the defence-in-depth safety philosophy, nuclear power plants shall be operated with as large a margin as reasonably achievable between the normal operating values and the safety limits.
2. For all parameters affecting the safety performance of the plant, operating and maintenance procedures shall be in place to ensure plant operation continues within the analyzed safe operating envelope.
3. A permissible operating range shall be specified for process parameters to which the effectiveness of the special safety systems may be sensitive.

4. The application of tolerances to parameters affecting the safety performance of the plant shall not result in:
   - operation of the plant outside the analyzed safe operating envelope, or
   - contravention of the requirements of the regulatory documents for special safety systems (e.g. R-7, 8, 9 and 10) applicable to the CANDU NPP under consideration.
5. Where probabilistic or statistical techniques/arguments are used in support of special safety system initiation setpoint tolerances, the residual probability of failure to trip before exceeding the safety limit shall be included as a contribution to the predicted unavailability of the system.
6. The analysis in support of tolerances shall identify, quantify and justify all possible uncertainties when determining tolerances.
7. Tolerances may be changed from the values permitted by the reference safety analysis by a) redoing the reference safety analysis, or by b) doing additional analysis to justify the change in tolerance for one or more parameters. For a) the requirements given in above shall apply; for b) the following shall apply in addition to those above:
   - A change to a tolerance shall not be considered unless it has been shown that the application of the change results in a net safety benefit to the plant.
   - All applicable operating states and accident cases addressed in the reference safety analysis shall be considered in the determination to changes to tolerances.
   - All applicable plant safety limits shall be demonstrated to be met. Where a deterministic analytical approach is used, all parameters to which a special safety system initiating setpoint is sensitive shall be moved to their worst allowable values. This shall be done for all accidents for which the initiating setpoint is credited.
   - It shall be demonstrated that the changes in tolerance will not alter the effectiveness of the special safety system as claimed in the reference safety analysis.

The CNSC is currently revising its position and requirements on the basis of actions already taken by some licensees

India

The set point surveillance and calibration check requirements as per Technical Specifications for vital instruments is once a year in Indian PHWRs. Also for parameters which have displays of set points, the Technical Unit/Senior Technical Audit Engineer (STAE) makes a monthly report on the set points to bring out deviations and initiate corrective actions. In addition if during tests, where actual pressurisation/depressurisation is done during tests (ex: H.T. high & low pressure trips) and if trips occur at set points different from specified value as read in the calibrated gauge then calibration is taken up immediately.

In the new reactor designs computer based systems are used in plant controls. Set point integrity checks are incorporated as part of on-line diagnostics. The diagnostic messages in the form of alarm is available on the display units.

While copies of R8 & R10 are not available, an exercise has been conducted by designers to analyse whether two diverse trip parameters are incorporated for SDS#1 & SDS#2 for postulated initiating events. Analysis indicates this has been done.

In Indian PHWR, containment high pressure is used as an indication for accidents like LOCA. One of the safety system actuation based on containment pressure is ECCS accumulator pressurization. The setpoint for such actuation is kept as low as possible to be effective. However, the setpoint is limited by the normal operational consequences. The peak pressure in containment depends on the size and location of break, containment volume surrounding the break, cooling performance, etc. There is possibility, in some cases of Small LOCA, the discharge may not be enough to raise the containment pressure to the setpoint but exceeds the capability of make up system. Such scenario is called Blind LOCA. Under such cases, it is mandatory to have enough time for the operator to take manual action such that no adverse consequences are encountered during such accidents.

Korea, Republic of

In Korea, the technical specifications for CANDU NPPs specify calibration frequencies for special safety systems of 3 - 4 years depending on the number of channels or loops of the primary heat transfer system. Wolsong unit 1 is an exception.

In a periodic inspection of Wolsong unit 2 in 1999, KINS staff found an unexpected drift problem which is out of allowable values in SDS 2 steam generator level trip transmitters (SGLTs) after only two years of operation. Not only did half the as-found values of the twelve SGLTs meet the requirement of TS, but also some of those were out of the analysis values. The specified calibration intervals are no longer valid due to the unexpected drift problem. In an extended inspection, the licensee determined that as-found data showed that that drift problem prevailed in Wolsong units 2, 3 and 4. The worst case was Wolsong unit 4 SGLTs, which were out of allowable values when the weekly on-power spread check was performed for the first time after a few months' operation.

Additional trip parameters could be considered to cover all potential accident scenarios.

Background:

There were several potential accident scenarios that were found not to satisfy the trip effectiveness requirements of CNSC (Canada) documents R-8 and R-10:

- SBLOCA
- Pressure Tube Rupture
- Loss of Flow Accident
- Primary Heat Transport Pump Seizure
- Loss of Reactivity Control
- Loss of Inventory Control
- Loss of Pressure Control
- Feedwater Line Break

It was stated in section 15.4.1 of WS-2 PSAR that exceptions to this requirement are considered to be acceptable for limited configurations; namely if providing two parameter coverage is either:

a) not practicable (that is either technically not feasible, or if the only means of detection would be a setpoint within normal operating conditions), or
b) counter-productive to public safety, or
c) a crushing economic burden.

Regulatory Requirements:

R-10: Where practicable, two diverse trip parameters shall be incorporated into the sensing and control logic of each protective shutdown system for each of the serious process failures requiring shutdown action.

R-8: For each event specified (in Table 1 and 2) for which action by a shutdown system is required, there shall be at least two diverse parameters on each shutdown system, each designed to detect the need for and automatically initiate shutdown action such that all requirements for effectiveness are met.

Exceptions to this requirement may be permitted only if it can be shown to the satisfaction of the AECB that incorporation of a second parameter for problem against an event is: impracticable, or detrimental to safety.

Remarks:

The following should be included in the reports that will state the validity of the single trip coverage region:

- Evaluation of the changes in public risk if the single trip coverage can be removed by adjusting the setpoints of the existing trip parameters
- Analysis results which describe that the new parameter which may be conditioned by power is considered to remove the single trip coverage region
- Systems to mitigate the accident progress for the case that the accident occurs during operation in the single trip coverage region

The various remedial actions or trial attempts to resolve the problem on a short term basis have been formalized in a regulatory body, utility, NSSS designers and transmitters vendors to date such that;

1. weekly spread checks between SDS 1 and SDS 2 SGLTs ,
2. adjustment of the existing calibration frequency in TS,
3. re-calculation of instrumentation uncertainty components and allowable values,
4. confirmation of the uncertainty calculation in SDS Design Manual,
5. discussion of the definition of Minimum Allowable Performance Standards for SG level trip,
6. vendor's re-evaluation of overpressure effect or other effects in transmitter's drift
7. draft analysis of safety impact on SG trip coverage with proposed new analysis values and
8. trial modification of in situ calibration procedures to evaluate static pressure effect unique in SDS differential transmitter.

According to Canadian regulatory requirements R-8 and R-10 which are applied to CANDU-type reactor in Korea, there shall be at least two diverse parameters on each shutdown system unless the exceptions are not permitted by KINS.

1. Exceptions may be permitted if the single trip coverage regions can not be removed with the new trip parameter instead of the existing ones. In this case, it may be justified that incorporation of a second parameter protection against an event is 'impracticable' after the analysis results for the new trip parameters considered shall be reviewed.
2. If the public risk and dose are increased due to the second trip parameter, it may be justified that the second trip parameter is detrimental to safety. The analysis report shall be reviewed.
3. There is no statement in the regulatory documents that exceptions may be permitted because of economic burden. Especially, economic burden due to the extension of maintenance time is not a proper reason.

Romania

The problems related to the trip setppoints like mainly for instance those related to the Steam Generator level are subject of the current strategic policy for relicensing of unit 1 in May 2001.

However the envelope is already defined by OP&P, OM and OMT and mandatory Test program is going on to check that the requirements are met.

It is expected that after review of the actions taken information on this issue will be updated and detailed, as part of the periodical safety review process and mainly the instrumentation precision and their impact on safety will be systematically reviewed in the deterministic analyses to suport PSA assumptions, which are now being performed.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Specification of Requirements for Upgrades Using Digital Instrument and Control Systems, IAEA-TECDOC-1066, IAEA, Vienna (1999).
- CNSC (Canada) documents:
  - C-6 “Requirements for the safety analyses of CANDU nuclear power plants”.
  - R-7 “Requirements for containment systems for CANDU nuclear power plants”.
  - R-8 “Requirements for shutdown systems for CANDU nuclear power plants”.
  - R-9 “Requirements for emergency core cooling systems for CANDU nuclear power plants”.
  - R-10 “The use of two shutdown systems in reactors”.
- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.
- Cernavoda Unit 1 Operating License, 1999.

4.1.8 Containment (CS)

ISSUE TITLE: Containment integrity (CS 1)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

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 might eventually exceed the design limits and the containment could leak significantly. The challenge to containment integrity due to overpressurization can be mitigated by accident mitigation strategies such as:

- containment venting;
- hydrogen control measures;
- containment heat removal

The basic idea for containment venting is to open a controlled and filtered flow path to the external environment to relieve the pressure that is generated inside the containment due to various processes during the accident.

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.

The effects of hydrogen burners and the possibility of containment failure due to over-pressurization were confirmed by the TMI-2 accident and by extensive research.

By implementing containment venting and/or 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.

After a steam (D_2O or H_2O) leak in containment the pressure would rise to some value less than the design. The containment cooling system would take the enthalpy out and bring the pressure down to a level from which controlled depressurisation can start. Continued inleakage of instrument air could delay the depressurization or could cause a pressure increase. However current practice for most plants is to shutoff the instrument air after about 2 hours.

Safety significance

Severe accidents and DBAs like steam line break with loss of dousing can result in phenomena, such as the ones described above, which could result in containment failure. However, severe accidents in PHWRs are very slow due to the presence of the moderator and shield tank as heat sinks. This means the rise in containment pressure is also slow, giving time for severe accident mitigation and management.

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:

Argentina

The leaktightness of the Embalse containment is monitored within the surveillance containment program every five years. An important contributor to the containment leakage is the leak rate of room R001 (refuelling machine). Due to leakage detected during repetitive tests performed in Embalse NPP room R001, a specific procedure (TP-21080-02) to identify leakage points was created. In essence this procedure indicates to pressurize (reduced pressure) at both 0,210 Kg / cm^2 and 100 gr. / cm2 pressure values and to inspect inner room R001 surfaces by an adequate technique. Three tests were performed, the first two tests objective were to locate leakage points and the third one test, performed after corrective actions (fix) were taken, served to verify such corrective actions efficiency. The efficiency to detect leakage pressurizing to 100 gr. / cm^2 was demonstrated and this pressure value was incorporated to the procedure. It was important because of pressurize to 0,210 Kg / cm2 implicates evacuate water from fuel storage pools. The mentioned demonstrate that both procedure application and the corrective action taken are useful to improve containment performance. It was recommended to perform the repetitive test once in a year.

Canada

See SS 4, “Leakage from systems penetrating containment or confinement during an accident”, and
See SS 5, “Hydrogen control measures during accidents”.

China

The design pressure of TQNPP containment is 124 KPa. The measuring range of containment pressure measuring instrumentation was originally designed as –20~150KPa(g), which acted as a parameter of PAM and was displayed in MCR and SCA. The basic design concept of CANDU containment only considers peak pressure generated from large LOCA not exceeding 124KPa(g). As a parameter of PAM, an accident resulting from the main steam tube rupture inside the containment shall be considered. AECL’s calculation indicated that peak pressures of the main steam tube rupture are 179KPa(g) (available for dousing system) and 400KPa(g) (unavailable for dousing system) respectively.

It is required by NNSA to change the measuring range of the containment pressure instrumentation to –20~400KPa(g).

There are 35 local air coolers inside the containment of TQNPP. 16 of them are safety-related. The 16 air coolers will be put into service when site design earthquake (SDE) occurs, 24 hours after LOCA. The air coolers have two functions, i.e. agitating air to prevent excessively high H2 concentration in local areas and functioning as a heat sink to take out the accumulated heat inside the containment. It is required in safety review that the air coolers are needed to be in service after an earthquake. So power supply for the coolers shall also meet seismic requirements. However, the original design is not seismically qualified. The design change has been made to meet above requirement. The electricity distribution system from emergency power electricity distribution room to the local air coolers is designed to meet seismic requirements. This requirement also applies to the control system from the second control area to the local air coolers.

Many vessels of CANDU design are lined with epoxy resin, such as containment, spent resin bay and waste resin storage bay. It is required to use stainless steel liner instead of in TQNPP. Considering that the containment is a dry vessel and its inner surface is accessible to repair the liner if broken. However, the spent fuel bay is a radioactive water bay and stores waste with high radioactivity. Once the epoxy liner breaks, it is very difficult to repair the bay. Consequently, the design changes have been completed that the spent fuel bay and the spent resin storage bay epoxy liners are instead of by stainless steel liner in TQNPP.

India

The reactor building of Indian PHWR consists of a primary containment (PC), designed for peak pressure following design basis accident, and a secondary containment (SC) which completely envelopes the primary containment. Both the containments are usually painted from both sides to reduce the leakage rate in case pressurization of PC following an accident. The Technical Specifications postulate the depressurisation of containment in Indian NPP depending upon design. For older units depressurisation is envisaged from 1.2 to 6 hours after LOCA Annulus part formed in between PC and SC as kept at negative pressure during normal operation as well as during accident condition to prevent the ground level release of activity. To minimise repressurisation of RB, provision exists to reduce instrument air supply to RB.

The proof testing of primary containment is done at design pressure before the operation of reactor. Main steam line break (MSLB) accident is governing event for primary containment design. Unless there is steam generator tube leak MSLBA does not have radiological consequences, the ILRT of primary containment is carried out at LOCA peak pressure which is below the design pressure of PC.

Indian PHWR Containment is qualified even for severe accidents like LOCA with ECC failure. In such accidents, moderator acts as a backup heat sink. Thermo-mechanical behaviour of coolant channel under such accident enhances the heat transfer path to moderator. Thereby, the fuel temperature is maintained at lower value leading to lesser hydrogen generation. It is also planned to qualify the containment under the severe accident like moderator cooling failure during LOCA + ECCS failure. Under such cases, fuel melts, coolant channel and calandria tubes fail and debris can be held in calandria by providing cooling through calandria vault water. Under such accident, hydrogen generation is of concern. This is addressed through the hydrogen management measures.

Ultimate load capacity of the Inner containment structure of a few Indian NPPs was calculated which about twice the design capacity.

Korea, Republic of

The inside surfaces of containment buildings and spent fuel storage pool structures in PHWR plants are lined with non-metallic, not steel, liner, so the maintenance of the non-metallic liner is very important to ensure leak tightness.

In 1998, Wolsung 1 NPP discovered some degradations of the non-metallic liner, such as cracks and swelling, in the spent fuel storage pool structures.

The spent fuel storage bay (SFSB) is surrounded by water-proofing membrane which prevents the release of leaked cooling water from SFSB to the ground. The inflow of water to the sump located between SFSB and the membrane indicates the leakage of cooling water from SFSB or the loss of function of the membrane.

As the leakage of cooling water from SFSB causes direct release of radioactive material to environment, the collaboration of CANDU-6 owners is required to resolve this matter with consideration of detailed inspection, necessity of repair and future design change.

The leaktightness of the CANDU-6 plant containment is ensured by prestressed concrete structure and rigid type non-metallic liner applied on inside concrete surface. Since the inherent nature of concrete has permeability and no practical capability for preventing the occurrence of cracking, leaktight integrity of containment is compromised by the cracking of concrete surface which directly causes the tearing of liner and leads to bringing down the leaktightness function.

The lessons learned from the failure of ILRT requirement of CANDU-6 plants which had been appropriately designed and constructed according to related codes and specification require the improvement of liner material and related technical standards including the strengthening of qualification test provisions and periodic inspection requirements for containment.

In addition to the integrity on LOCA pressure which can be confirmed by ILRT, when the integrity on ultimate pressure (dual failure condition) is concerned, the design change of liner material to steel which has ductile nature to large deformation and leak resistance has to be considered.

The reactor containment buildings of the Wolsong nuclear power plants are made of prestressed concrete of the bonded type. As time goes by the prestress level is reduced due to concrete creep and shrinkage, along with tendon relaxation. Therefore, in-service inspection of the prestressing system is required to verify whether the effective prestress provided in the structure has reduced the safety margins used in the design.

In-service inspections of CANDU nuclear power plants, such as the Wolsong units, aren't performed in the containment building directly. The loss of effective prestress of the prestressing system in containment is estimated by comparing theory predictions applied to the design and the test results from experimentation of the test beam. The test beams are constructed of the same materials and methods as for the containment structure and stored at a site subjected to similar environmental conditions.

The test beam is very different from the size, shape and bonding state of the prestressing system in containment, so it is impossible to evaluate quantitively the variation of effective prestress in the containment building by the test beam approach.

For Wolsong stations, in order to complement the indirect characteristics of the above method, additional strain gauges method, in which strain gauges are embeded in the containment structure, is used. But, it is difficult to assess the structural integrity because this measured data is small and the strain gage is very sensitive to change of temperature after which the analysis result isn't reliable.

To ensure the leak tightness function of the spent fuel storage pool, the following activities were performed.

- inspection for degradations (every 3 month)
- measurement of the inflow water in the sump below the spent fuel storage pool (once a month)
- analysis of the radionuclide of water in the sump (once a month)
- establishment of a research program for the integrity of non-metallic liner

Pakistan

The containment of KANUPP was tested for leak rate at the design pressure of 27 psig at the time of commissioning in 1971. Since then, the reactor building was tested after every two year at two (2) psig and the leak rate at the design pressure of 27 psig was predicted by extrapolation. According to the revised KANUPP Final Safety Report (KFSAR), the containment building pressure will not rise above 13.4 psig in case of a critical break (where maximum probability of activity release exist) in PHT system headers. Therefore, testing of Containment Building at 13.4 psig will ensure the intended level safety. This has revealed the increased safety margin of the containment barrier.

KANUPP containment building was tested at 5 psig in 1993. The leakage rate was found to be within limits. The feasibility of testing at higher-pressure i.e. 10 psig and 15 psig has been carried out. For execution of these tests at KANUPP, installation of a remote monitoring system is required in containment building. The test at 10 psig will be carried out in 2006 after the installation of these measuring instruments.

Romania

Containment integrity requirements are defined as part of the Licensing Basis and they are reflected in the safety reports. In addition to the safety envelope the operational envelope is defined for containment systems, as part of the Operation Policy and Principles. These envelopes are defined for the DBA postulated. In order to check the compliance of the containment integrity with the requirements commissioning tests were performed and there are tests included in the in service inspection procedures. The containment integrity is monitored also based on the results of the support documentation on the containment behaviour for a long term after a LOCA. The leak rate tests, which are going on in this program have some points of interest, as for instance the definition of some of the refuelling machine area as part of the containment for these tests and the evaluation of the results These requirements and results are currently under evaluation as part of the strategic policy for relicensing unit 1.

On the other hand there are requirements for the SAM procedures from which evaluations for containment monitoring procedures and necessity for hardware improvements are expected as part of the Periodical Safety Review for Unit 1. The review of the passive parts of containment systems (as for instance the pre stressing cables) is also part of these programs.

It is expected that in 2001 more details and updates are to be included in this document.

The Licensee will manage the performance of this project in a framework of a long term Research and Development Strategic Safety Program.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Concrete Containment Buildings, IAEA-TECDOC-1025, IAEA, Vienna (1998).
- Strategic Policy for Cernavoda NPP Unit 2 licensing process, CNCAN 1997.
- FSAR Cernavoda Unit 1, 1995.
- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.

4.1.9 Internal Hazards (IH)

ISSUE TITLE: Need for systematic fire hazards assessment (IH 1)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

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 fire hazards analysis has six separate purposes:

- identification of items important to safety and their locations in fire compartments;
- analysis of the anticipated fire growth and the consequences of the fire and fire fighting activities with respect to items important to safety;
- determination of the required resistance of fire barriers;
- determination of the type of fire detection and protection means to be provided;
- 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;
- 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.

Fires in nuclear power plants have demonstrated that fire can be a major risk contributor to the overall plant safety. Licensees are required to ensure that their plant operations are such that risk from fire is minimal.

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:

Argentina

Standard AR. 3.2.3 establishes the safety criteria against fire (or events generated by it) and explosions resulting from fire, that may affect a nuclear power plant radiological or nuclear safety. These criteria include the stages of design, commissioning and operation of the installations. The fulfilment of the criteria contained in the above mentioned standard is verified through inspections carried out by inspectors and analysts of the Regulatory Body.

On the other hand, CNA-I and CNE have specific procedures of fire protection. Such procedures contain the description of fire compartments; the composition, responsibilities and functions of the fire brigade; the detection and alarm systems; the extinction systems; the fire-fighting drills and other aspects related to the fight against fire of permanent application in the nuclear power plants.

The PSA of CNA-I and CNE foresees the core damage evaluation due to an eventual fire that may be initiated at a set of areas of the installation known as fire compartments. The analysis methodology chosen enables the calculation of a core damage probability associated with each of the fire compartments before mentioned, in the case such fire occurs. This methodology comprises a set of tasks such as:

- Establishment of a general procedure for the fire risk analysis.
- Determination of fire compartments, fire barriers and fire propagation routes
- Calculation of fire propagation probability to adjacent fire compartments.
- Preparation of a list of affected components at each fire compartment and calculation of the corresponding failure probability rate due to fire.
- Calculation of the core damage probability associated with each fire compartment where a fire occurs.

The following software tools are used for the execution:

- Fire Database NUREG/CR 4586.
- COMPBURN III A computer code for modelling compartment fires - NUREG/CR 4566.
- Database of Fire Compartments

The analysis of the results will enable the determination of the highest risk fire compartments for the installation, and the improvements to be carried out either in components or compartments, in order to reduce their failure probabilities in case of fire.

This methodology implies an identification of fire-induced failures of components as well as of the several fire propagation routes; therefore it advantageously substitutes the classic failure analysis of common cause due to fire.

Canada

Regulatory Body:

The CNSC expects that the nuclear industry meet the relevant requirements set out in CSA Standard N293-95, “Fire Protection for CANDU Nuclear Power Plants”.

CNSC staff position can be summarized as follows:

- Fire protection is a key element in nuclear safety.
- Relevant codes and standards are applicable and their implementation contributes to overall safety.
- Fire hazards assessments, as living documents, are a sure way of ensuring the continued integrity of fire protection defence-in-depth.
- To maintain the designed benefit of defence-in-depth, the CANDU separation philosophy must be maintained and monitored.
- The effectiveness of emergency response provisions and preventive programs must be regularly and continually verified.
- Implementation of backfits at operating plants is essential to nuclear and occupant safety.

CNSC staff intends to produce a position statement addressing this generic action item. Licensees are expected to satisfy the closure criteria given in that position statement.

Industry:

CANDU 6 plants are designed to satisfy the CSA Standard CAN/CSA-N293-M87, "Fire Protection of CANDU Nuclear Power Plants". It requires applying defence-in-depth in fire protection design. The concept consists of (1) fire prevention, (2) fire detection and suppression, and (3) mitigation of the effects of fires. For the mitigation of the effects of fire, separation criteria are established and applied in the cable routing.

A fire hazard analysis is now performed during the design stage to confirm that the fire protection concept is adequately applied.

Safe shutdown is assured by ensuring that either two independent and separate groups of systems can perform all necessary safety functions. Group 2 systems are functionally and physically separated from Group 1 systems and the use of either Group 1 or Group 2 systems alone can safely shutdown the plant and maintain it so. The cables for Group 2 systems are physically separated from those of the Group 1 systems including those for power, control and instrumentation signals.

China

It's licensing requirement to apply systematic fire hazards assessment in TQNPP safety review.

India

After the Narora fire incident extensive review was instituted for all the Indian NPPs. The basic objective was to ensure that fire, as a common cause, would not prevent achieving, maintaining and monitoring of

a) shutting down of the reactor
b) core cooling
c) confining activity.

This detailed study resulted in recommendation in the areas of design provisions, operational (including maintenance) practices organisational arrangements, quality assurance, etc. AERB standard on fire protection was issued and this incorporated – current international standards. While all new stations incorporate the provisions, the older stations have been asked to come up with an action plan.

.Standard "Fire Protection System of Nuclear Facilities" and Safety Guide "Fire Protection in Pressurized Heavy Water Reactor based Nuclear Power Plants" published by AERB are being referred for matters connected with Fire Safety. Standard is applicable to all nuclear facilities including power plants and is mandatory in nature specifying requirements during design and operation. The Safety Guide addresses the design issues only with respect to PHWRs and fire hazard analysis is the key element to arrive at fire safety requirements. For new projects preparation of Fire hazard Analysis Report is essential. However, for old projects Fire hazard Analysis is being carried out and systems are being upgraded by retrofitting. Based on the Fire Hazard Analysis report following jobs have been completed. Power and control cables have been segregated. Fire retardant coating on the critical cables has been applied and fire seals to all critical cables ensured. Installation of fire dampers has been completed. Fire barriers between seal oil units, main oil tank, turbine oil tank, and diesel generator, class-II and class-III switch gears and fire water pumps have been installed. Supplementary control room has been set up where not existing and where not possible it has been ensured that essential equipment can be started from local control points/ breaker compartment. Additional DG firewater pump and sprinkler system to cover seal oil unit and oil piping has been installed. Augmentation of detection and suppression system has been completed based on Fire hazard analysis. Relocation of non-automatic sprinkler system valves for turbine building areas has been carried out for improved access. Based on the Fire Hazard Analysis report installation of fire dampers in Turbine building ventilation system and upgradation of fire alarm system at MAPS has been completed. Cable route segregation was done at KAPS.

Korea, Republic of

See SS 3

Romania

Cernavoda NPP unit 1 is designed in accordance with the basic design requirements on fire protection, i.e. the use of the safety systems separation on groups as a mean for protection to common cause failures due to fire. There is also, in addition to these aspects, a requirement for a fire hazard review of the plant to be done in accordance with the IAEA documents. The study is being developed under EU project with the information of the regulatory body. A future inclusion of the results of this study is required after the PSA level 1 completion by the end of 2002.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Protection against Internal Fires and Explosions in the Design of Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-1.7, IAEA, Vienna (2004).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Fire Safety in the Operation of Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-2.1, IAEA, Vienna (2000).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Evaluation of Fire Hazard Analyses for Nuclear Power Plants: A Safety Practice, Safety Series No. 50-P-9, IAEA, Vienna (1995).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment of the Overall Fire Safety Arrangements at Nuclear Power Plants: A Safety Practice, Safety Series No. 50-P-11, IAEA, Vienna (1996).
- 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, Organization and Conduct of IAEA Fire Safety Reviews at Nuclear Power Plants, IAEA Services Series No. 2, IAEA, Vienna (1998).
- Canadian Standard CSA N293-95 “Fire Protection for CANDU Nuclear Power Plants”.
- Standard NFPA N-805 (proposed).
- National Canadian Building and Fire Codes.
- Strategic Policy for Cernavoda NPP Unit 2 licensing process, CNCAN 1997.
- FSAR Cernavoda Unit 1, 1995.
- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.
- AERB, Standard for Fire ProtectionSystems of Nuclear Facilities, AERB/S/IRSD-1(1996).
- AERB, Fire Protection in Pressurised in Pressurised Heavy Water ReactorBased Nuclear Power Plants, AERB/SG/D-4(1999).

ISSUE TITLE: Adequacy of fire prevention and fire barriers (IH 2)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

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.

Safety significance

Insufficient protection against common mode failures due to fire would impair 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:

Argentina

See also IH 1.

Canada

The fire barriers between fire zones are designed based on the results of the fire hazards analysis. The fire barriers are tested according to the CSA Standard or an equivalent standard. Some of the concerns on fire propagation due to transient combustible materials are explicitly addressed in the fire hazard analysis and are transferred to operating personnel as inputs for developing the plant-specific fire protection programs.

See IH1.

India

Detailed fire hazard analysis is done in critical area to estimate fire load. Maximum rise in temperature in case of imaginary fire and duration of fire is estimated. Based on the information designing of fire barriers are taken up. This Fire prevention measures include use of High ignition temperature lubricating oil, use of Fire resistance Low smoke cable and special type of insulation and house keeping. Grass should be cut before it becomes dry and cut grass removed. Areas of challenging decision making is whether to provide a manual big sized pipe to quickly empty turbine lubricating system main oil tank to reduce amount of combustible in case of adjacent fire as inadvertent actuation can be disastrous to the turbine bearing. Sharing of position of member states on this issue would be useful. Fire Barriers have been provided on the cable penetration on the wall and on the floor to prevent spread of fire through cable penetrations. At RAPS-1 during upgradation of Fire Safety system 211nos of fire barriers were identified. (Reactor bldg: 53,Turbine bldg: 115,Service bldg: 40, Auxiliary bldg: two and DG-6: 01 number). Job has already been completed. This coating has been applied on all power and control cables both at the equipment end and at the source end. For power cables this coating is applied for 2 meters length at every 6 meters intervals in horizontal run of the cable trays while this coating is applied in entire length of the cable in vertical run of cable trays. Fire retardant paint coatings (Fire breaks) are subjected to chimney test. During test, it is to be confirmed that fire will not propagate beyond the portion of the cable coating for duration of half an hour. Fire doors conforming to IS-3614 are provided as means to confine fire to the room where it is originated. During up gradation of fire safety system eleven fire doors of three hours rating have been installed in various locations to segregate group A & B areas in RAPS-1. Proper design, selection and testing of fire barriers are important but their maintenance during operation, especially after modification is equally vital. The reason for spreading of fire to some vital areas during Narora fire incident was due to lack of quality assurance resulting in inadequate restoration of a few fire barriers after modification was carried out.

After detailed studies instituted at each NPPs some equipment were relocated. Example; hydrogen addition station for the turbine generator.

Korea, Republic of

Wolsung units 3 and 4, a fire protection program is established to meet the CAN/CSA-N293-M87(Fire Protection for CANDU NPP) and Korea fire protection laws. In accordance with that fire protection program, KEPCO are controlling the storage of combustible materials and handling the electric or hot work equipments. And, Wolsung units 3 and 4 are designed to use non-combustible materials, as far as practical. In case of fire, to minimize the release of the radioactive material and reduce the consequences of fire, they maintain the necessary safety function to shutdown the reactor in accordance with operation procedure. Fire area and fire barriers between redundant systems are installed in accordance with the fire hazard analyses and the fire suppression systems.

Pakistan

In accordance with NUSS Safety Guide and international practice, an adequate degree of fire protection is to be provided in NPPs. The objective is to prevent fires from starting i.e. fire prevention. One of the main concerns at KANUPP is that most of the redundant equipment, components and cable trains of safety important systems are not physically separated and not protected against fire spreading. Other concern is related to inadequate protection against oil fires.

Although, a lot of work has been undertaken by KANUPP in the past few years to improve fire detection and protection capabilities; however, PNRA observed that some deficiencies still exist, which has serious consequences on the plant safety. For example:

1.Redundant equipment, component and cable trains of safety and safety related systems are located without sufficient physical separation and are not protected against fire spreading. A fire could thus lead to loss of more than one redundancy of safety important systems.

2.Protections against oil fires are far below the satisfactory level.

KANUPP has been required to implement a comprehensive fire prevention and control program and carry out a fire PSA. Install fire barriers wherever required to:

- Segregate safety related cables as practical as possible.

- Improve habitability of control room or establish an Emergency Control Room (ECR).

Improve fire detection and suppression capability on potential fire hazard areas, especially in Turbine hall and around Primary Pumps in reactor building.

Romania

Similar to IH 1

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Protection against Internal Fires and Explosions in the Design of Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-1.7, IAEA, Vienna (2004).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Fire Safety in the Operation of Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-2.1, IAEA, Vienna (2000).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Inspection of Fire Protection Measures and Fire Fighting Capability at Nuclear Power Plants: A Safety Practice, Safety Series No. 50-P-6, IAEA, Vienna (1994).

ISSUE TITLE: Adequacy of fire detection and extinguishing (IH 3)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

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.

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:

Argentina

See IH 1

Canada

The CSA standard lists requirements for fire detection and suppression systems including the selection and location of fire detectors, design of fire signalling systems, design of fixed fire extinguishing systems, manual fire fighting capability, and so on. The adequacy of the design is addressed in the fire hazard analysis. The fire detectors and fire suppression equipment are qualified and tested regularly.

See also IH 1.

China

Fire Water Supply System of the Reactor Building (R/B) Shall Meet Seismic Requirements. The design of the Reference Plant takes portable fire extinguisher as the primary method of extinguishing fires in the R/B. According to requirements in GB J84-85 and GB J16-87 as well as basic design concepts for PWR design, a large portion of the equipment installed in the R/B needs to be functional after an earthquake. The fire water system of TQNPP, as one of its requirements to maintain its function, shall meet seismic requirements too.

AECL has agreed to make such a design change in TQNPP's design. The whole fire water system from the emergency water supply (EWS) reservoir to the R/B is designed based on DBE, and is designed to meet GB requirements.

India

Quick detection of fire help in mitigating the damage largely. Detailed area wise survey indicated in adequacy in the number, actual location as well as type of fire detectors. Following types of detectors are available: Smoke detectors, Flame detectors and Heat detectors. Smoke detectors (Ionisation type & photoelectric type), and Heat detectors (rate of rise detectors and fixed temperature detector) are in use.

Type of detectors have been used as per the existing environmental conditions e.g. in high radiation areas alone ionization type detector has not been used.

Detector location takes into account the ventilation pattern in the concerned areas to ensure smoke reaches the detector with imaginary fire in any location of the area. During installation of detectors, wherever feasible cross zoning system has been followed.

Number of detectors is limited by area covered by different detectors.

In each area along with the primary suppression system, backup suppression system has also been provided.

Periodic Inspection testing and maintenance schedule is followed strictly. Detector and associated deluge systems for transformers, oil tanks, turbine bearings, pipelines etc. requires special attention. Provision water spray for cable tray has been provided.

Quick entry into any fire location with conflicting requirement posed by physical protection and security system, radiation zoning etc. needed a lot of review and working of satisfactory methodology. Since initial period after a fire starts are vital, station staff are also trained and retrained on fire fighting. It is ensured that in each shift at least two trained people are available. Routine inspection and maintenance of portable fire extinguishers are very important. Quarterly fire drills are rotated in different areas, as one of the objectives is to verify that fire-fighting facilities are adequate in each area

Once a year Regulatory Inspection in Industrial safety is carried out. During Regulatory Inspection fire safety system is thoroughly checked.*Korea, Republic of*

Wolsung units 3 and 4, a fire protection program is established to meet the CAN/CSA-N293-M87(Fire Protection for CANDU NPP) and Korea fire protection laws. Defence in depth concept is used to minimize and mitigate the possibility and effects of a fire by the provision of suitable fire prevention, fire detection and fire suppression capabilities. In all cases, at least one back-up means of fire suppression is provided for all areas. Early detection and effective suppression of fire can help in eliminating the adverse effects of fire for the personnel and plant safety systems. For these purpose, semi-annual instruction and training for the fire extinguishing are performed for fire brigade. In addition, periodic checking and maintenances for the various fire detector and suppression facilities are performed by the plant operator and regulatory inspector in accordance with the fire protection program and Korea fire protection laws.

Pakistan

In KANUPP, fire extinguishing system falls into following categories:

i) Fire water system, comprising external hydrants, internal hose stations and fire water hose reels.
ii) Deluge system for the transformers.
iii) Heat actuated sprays for the oil circuit breakers and turbine lube oil system.
iv) Gaseous extinguishing system comprising of CO2 and halogen type extinguishers, dry chemical type, chemical foam type, and mechanical foam apparatus have been provided and are manually operated and located at necessary points throughout the plant.
v) Fire tender
vi) Gaseous flooding type automatic/manual fire suppression system for new Electronics and UPS/ Battery rooms

The relay-based Fire Alarm System installed originally in KANUPP had become obsolete and difficult to maintain. Moreover, it was covering only 20 zones of the plant while several new buildings have been constructed with in KNPC. On account of these facts, the old system has been replaced with a microprocessor-based, analog addressable fire alarm system. The new alarm system now covers all main buildings of the plant viz-a-viz Reactor Building, Switchyard, Diesel/ Distribution Room, Service Building, Turbine Building, Auxiliary Service Building, Warehouses.

Some salient features of the new Fire Alarms System include:

- Continuous self-checking
- More accurate pinpointing and display of the fire location
- Smoke detectors and rate-of-rise detectors provide much better sensitivity.

Initially there was only one (1) thermal fire detector in whole Boiler room. By the introduction of new Fire Alarm System, Boiler room is now equipped with twenty five (25) smoke type, one (1) beam detector and five (5) alarm sounders at different elevations.

To overcome the lack of fire suppression capability in Boiler room, sufficient quantity of large capacity (trolley mounted) Dry Chemical type fire extinguishers are made available in addition to the existing small size portable fire extinguishers.

Fire suppression capability in Turbine hall has been improved substantially by the provision of large capacity Dry Chemical type fire extinguishers and the introduction of Fire Tender, equipped with foam proportioning system capable of producing foam up to a capability of twenty (20) cubic meters and gives suppression against oil fires.

Prevention and fire fighting in KANUPP is achieved by good house keeping and by prompt and correct use of fire-fighting equipment. To check the alertness of station fire fighting crew and to evaluate the adequacy of emergency protection procedures, fire drills are performed periodically.

Romania

Similar to IH 1

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Protection against Internal Fires and Explosions in the Design of Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-1.7, IAEA, Vienna (2004).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Fire Safety in the Operation of Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-2.1, IAEA, Vienna (2000).

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

This issue is also applicable to NPPs with LWR.

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 other systems.

All types of automatic and manual fire suppression systems (e.g., water, halon, CO_2) 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 inadvertent actuation of FPS;
- gas release could be harmful for the operators working in the area;
- 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

Any FPS actuation which results in an adverse interaction with the safety systems of the plant or the secondary effects of a fire could initiate an additional event or impair mitigating systems required for the event.

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:

Canada

CANDU 6s do not have an automatic halon nor CO_2 fire suppression system. There are specific procedures and training specifying that before halon or CO_2 fire suppression is actuated manually, operators in the area should be evacuated.

To prevent the propagation of smoke, the fire dampers are designed to close automatically where necessary, e.g. the ducts going to the MCR. For some areas with the potential for heavy smoke, high-capacity smoke exhaust fans are installed to remove the smoke to the exhaust stacks, helping manual fire fighting. SCBAs (Self Contained Breathing Apparatus) are provided in the plants for use by manual fire fighters. The MCR is also equipped with SCBAs for fire induced smoke or toxic gases. The secondary control area is functionally and physically separated from the MCR for use in case of a fire in the MCR which requires evacuation.

The spurious actuation of the firewater system cannot affect Group 1 systems and Group 2 systems simultaneously since the Group 2 systems are functionally and physically separated from the Group 1 systems. Also, the smoke generated in the Group 1 system area cannot propagate to the Group 2 system area since there are no ducts common to Group 1 and Group 2 systems, and Group 2 systems are physically separated from the Group 1 systems.

A fire PSA for a generic CANDU 6 design has been performed.

See also IH 1.

India

Water system consisting of sprinkler and other spray system, fire hydrant or standpipe hose system, Fire water supply system fire hydrant and sprinkler, Gaseous carbon dioxide suppression system and portable fire extinguisher are in use. Halon also being used as fire extinguishing medium but it is being phased out. Water has been used as fire suppression medium where unintended starting of fire suppression system will not be detrimental to other system

CO2 as suppression medium is being used for diesel generator room. Total flooding system has been kept on auto mode. CO2 flooding system is also employed in turbine oil tank and main oil tank. Where automatic carbon di oxide systems are used, they are equipped with a pre discharge alarm system and a discharge delay to permit personnel evacuation. Provision of locally disarming automatic carbon di oxide extinguishers shall be key locked under strict administrative control

Cable fire can result in huge amount of dense smoke. This requires specially designed system for evacuation of smoke to facilitate entry. Due to this reason cables with Fire Resistant Low Smoke sheath are used for power and control cables. In case of fire dense smoke generated in the cable gallery is exhausted out with the use of smoke exhauster. Person entering the building at that moment is using self-contained breathing apparatus. In cable gallery where cables density is maximum very few electronic instrument are available. Hence damage to electronic equipments due to smoke is not applicable.

Unintended functioning of water suppression system, gas suppression system or unintended stopping of ventilation system is all connected with spurious signaling by a detector. Hence, detection system is inspected, maintained and tested strictly as per schedule.

Korea, Republic of

Wolsung units 3 and 4 are designed and installed in accordance with CAN/CSA -N293-M87 and Safety Design Guide (SDG-005). Because water spray can interfere the operation of safety related systems, the gaseous fire extinguishing system is designed to operate manually for that area instead of water-based fire suppression system. The gaseous fire extinguishing systems are operated with pre-discharge alarms for evacuation of all occupants prior to gas discharge. Since smoke can reduce manual fire-fighting effectiveness and damage electronic equipment, effective ventilation system will be utilized for quick evacuation of smoke to reduce secondary effects and facilitate fire-fighting.

Romania

Similar to IH 1

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Protection against Internal Fires and Explosions in the Design of Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-1.7, IAEA, Vienna (2004).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Fire Safety in the Operation of Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-2.1, IAEA, Vienna (2000).

ISSUE TITLE: Need for systematic internal flooding assessment including backflow through floor drains (IH 5)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

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).

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:

Argentina

Flooding evaluations are ongoing. They were required by the Nuclear Regulatory Authority to be completed during year 2001.

Canada

This issue is not considered currently to be a generic safety issue. Systematic assessment of internal flooding is a licensing pre-requisite.

Some flooding assessments such as for condenser cooling water line breaks were done in Safety Design Matrix analysis and in the internal events PSA.

A calculation for Reactor Building flooding was done to show that the flooding sources in the Reactor Building do not affect the safety functions.

Inherent flooding protection is provided by the two Group design. Flooding in a Group1 system area does not affect the safety function of the Group 2 systems. Also, the steam barriers (provided for steam-line breaks) can function as flooding barriers, and separate the areas containing safety systems from the areas of potential flooding due to CCW line breaks.

Backflow through floor drains is not a big concern since there are no significant safety systems in the lowest elevation of each building; thus it can accommodate quite large water volumes, and the backflow rate through floor drains is quite limited.

A flooding PSA for a generic CANDU 6 design has been performed.

China

It’s the licensing requirement in China.

India

Internal flooding has been covered as design basis of systems in plants from NAPS onwards. Review for RAPS-2 was also taken up during its recent shut down and relicensing after retubing and refurbishment exercise.

Kakrapar Unit had a flooding incident due to heavy rains and breakage of a bund in adjacent lake. Few areas in turbine building got flooded due to some hatches being kept open. An "ASSET" review was carried out by a specially appointed committee. The committee reviewed all aspects including internal flooding and came up with several short and long term recommendations which were implemented. AERB asked for similar reviews for all NPPs which are operating as well as those under construction. This resulted in several corrective actions being taken at each of these NPPs after which the assurance of their being able to face both internal and external flooding has greatly enhanced. No serious problems due to back flow from drains have been observed in Indian NPPs.

Special provisions have been implemented to take care of floods in some of the older units. Emergency Operating Procedures (EOP) have also been written to handle systematically the internal flooding or due to external factors like heavy precipitation or upstream dam failure. Internal flooding is possible, if failures of equipment like expansion joint, etc. takes place in the condenser cooling water system. At RAPS Unit-1, inadvertent dousing took place and the moderator room got flooded. This equipment behaved as designed but as part of lessons learnt a few design changes were determined to be required and the same were implemented.

Korea, Republic of

See SS 3

Romania

The basic design includes the evaluation of the plant behaviour as a result of a flooding (from external and internal Condenser Cooling System failure). The group philosophy provides a technical basis for the protection at this hazard. Actions based on these evaluations are included in the plant procedures. There is also the requirements in the Strategic Policy for unit 1 relicensing to integrate the results from the flooding hazard analysis review into the PSA level 1 after its completion, by 2002, similar to fire and other hazards (seism etc) in order to review in more depth the flooding events as possible common cause failures.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Protection against Internal Hazards other than Fire and Explosions in the Design of Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-1.11, IAEA, Vienna (2004).
- Strategic Policy for Cernavoda NPP Unit 2 licensing process, CNCAN 1997.
- FSAR Cernavoda Unit 1, 1995.
- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.

ISSUE TITLE: Need for systematic assessment of high energy line break effects (IH 6)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

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 pipes 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) which if unmitigated could damage 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, equipment, 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:

Argentina

Considerations of breaks affecting primary and secondary system pipelines are included into the Abnormal Events Operating Procedures (POEA) which are specific for each abnormal event. Besides, during the operator's training in full scope simulator, it is considered equipment failures due to flooding and high temperature and high moisture conditions.

Canada

This is not currently considered to be a "generic safety issue". Effects of high-energy primary- and secondary-side breaks are considered in the design and in the accident analyses.

China

It's the licensing requirement in China.

India

Based on operating experience of MAPS where release of high energy steam and water close to a Motor Control Centre incapacitated certain electrical equipment, attention has been paid to eliminate such possibilities in recent design. AERB places due emphasis on this aspect during review process of design and later by walkdown of as built design. However, in the earlier plants such possibilities do exist and AERB intends to cover this as part of safety evaluation of operating units done at every five years interval as well as by asking each NPPs to conduct a review and institute corrective actions. Such a review would also include possible flow of leakage from floor and ceilings on electrical or instrumentation equipment as there have been few instances of these. ISI of high energy lines/equipment in such problem areas has been instituted and being followed.

Korea, Republic of

This is not considered to be a generic safety issue. Effects of high-energy primary- and secondary-side breaks are considered in the accident analyses.

Romania

These aspects are embedded in the accident analysis and reflected in the whole design and operating procedure system. The two group safety systems philosophy is a technical protection provided for the plant to cope with the effects of high energy breaks. The design for the pressure retaining components as part of the ASME implementation process is one component of this system (systems are protected by design to the resulted effects and by in service inspection actions there status is monitored). There is however a potential for improvement mainly if the DBA is being enlarged, so that new more restrictive environment qualification requirements are needed This is part of the new requirements related to the issue for unit 2 and of the Periodical safety review for unit 1.

Even if the design is made in such a manner to cope with these aspects, it is quite possible however that the PSA level 1 will bring more clarification and some possible new initiating events.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Protection against Internal Hazards other than Fire and Explosions in the Design of Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-1.11, IAEA, Vienna (2004).
- Strategic Policy for Cernavoda NPP Unit 2 licensing process, CNCAN 1997.
- FSAR Cernavoda Unit 1, 1995.
- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.

ISSUE TITLE: Need for assessment of dropping heavy loads (IH 7)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

In nuclear power plants heavy loads may be handled in several plant areas. If these loads were dropped in certain locations in the plant, they may impact spent fuel or equipment that may be required to achieve safe shutdown and continue decay heat removal.

Safety significance

Dropping a heavy load onto the reactor or spent fuel pool could lead to damage of the spent fuel or to a loss of the cooling capabilities and to a consequential release of radioactivity.

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:

Canada

This assessment is a licensing pre-requisite. The design documents contain procedural limitations that require the reactor to be in a secure shutdown state before heavy loads can be lifted above reactor areas of concern. The design documents also contain recommendations to park the cranes away from critical areas during operation and not to lift equipment higher than required for clearance. Cask handling around the spent fuel bay is required when spent fuel is to be taken away for examination or during cobalt production. Procedures for these operations prohibit lifting the cask over the stored spent fuel. On stations with stainless steel adjusters there are no routine maintenance operations involving handling of heavy loads above the critical areas, notably above the reactor and spent fuel bay. Work plans for any such operations that might be required would have the heavy load path identified and risk assessment of such operations would be carried out before the operation is undertaken. The load would be moved at the lowest practical height. Stations with cobalt adjusters service them every one to two years. Again, work plans for this operation have the heavy load path identified and a completed risk assessment before the operation is first undertaken. Certain stations with cobalt adjusters are equipped with flask positioners to achieve a redundancy in flask handling above the reactor and impact pads to protect the spent fuel bay.

India

Utility has identified the zones where heavy loads being handled if dropped could incapacitate core cooling or reactor shutdown. A major design modification of limiting such movements over such locations has not been found possible. However, the frequency of such handling is quite low and is primarily to take out certain equipment for maintenance which is done during reactor shutdown and cool down condition only. All cranes are parked away from critical equipment when not in use. Good industrial practices followed during any heavy load lifting event, additional precautions (like raising the load being moved to minimum required height to limit potential energy etc.) are covered in such procedures.

In the spent fuel pool the access of heavy duty crane provided for bringing in and taking out spent fuel casks is restricted by design so that these heavy loads never travel over spent fuel stored in the spent fuel bay. However, the smaller crane provided to carry trays of spent fuel to respective positions may cause drop of such a limited load over stored fuel. AERB has asked utility to assess consequential damage to fuel if such a drop takes place. Preliminary assessment indicates that the radioactive materials released in such an event could be limited and manageable without any undue off-site impact. Future designs (and backfits to the extent possible for existing NPPs) would incorporate two breaks, two wire ropes, etc. to ensure that single failure does not cause problems.

Korea, Republic of

This assessment is a licensing pre-requisite at the design phase of the plant.

Romania

It is already taken into account in the design and it is managed as part of the operating procedures.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Protection against Internal Hazards other than Fire and Explosions in the Design of Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-1.11, IAEA, Vienna (2004).
- ATOMIC ENERGY REGULATORY BOARD, “Design of Fuel Handling and Storage Systems for Pressurised Heavy Water Reactors”, AERB/SG/D-24 (2003).

ISSUE TITLE: Need for assessment of turbine missile hazard (IH 8)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

A missile could be ejected from the turbine/generator set due to two main 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 generator, a failure in the turbine overspeed protection system, 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 might 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:

Argentina

The plant lay-out of Argentinean NPPs fulfil the regulatory requierements. Therefore, damages in the main control due to turbine missile are very unlikely because of lay out matter. The turbine is located axially to the main control room.

Canada

This assessment is a licensing pre-requisite. The probability of a turbine break-up causing a loss of safety function must be shown to be negligible.

China

It's the licensing requirement in China

India

The current nuclear power plants in India have a lay out for turbine axis with respect to Reactor Building and other safety related structures (i.e. Control Building housing safety related services and main control room etc.) in such a manner that the path of postulated trajectory of the turbine missile does not intersect these buildings. In plants prior to KAPS, turbine missile if generated could strike some components or systems with safety functions. A detailed analysis of the possible damage caused by such a postulated missile for these earlier plants is yet to be carried out. However, the

Supplementary Control Room in these plants is so located that the safe shutdown of the plant should be feasible. In addition the turbine is tripped on any signs of off-normal vibrations indicated at any of its bearings, so as to limit the consequences of failure in rotating components of turbine at an incipient level.

Korea, Republic of

This assessment is a licensing pre-requisite at the design phase of the plant.

Romania

These aspects are embedded in the accident analysis and reflected in the whole design and operating procedure system. The two group safety systems philosophy is a technical protection provided for the plant to cope with the effects of turbine missile hazards. These results are included in the safety reports and were prerequisites for the construction, commissioning and operation licenses.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Protection against Internal Hazards other than Fire and Explosions in the Design of Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-1.11, IAEA, Vienna (2004).
- FSAR Cernavoda Unit 1, 1995.
- Commissioning and Operation Licenses Cernavoda NPP Unit1.

4.1.10 External hazards (EH)

ISSUE TITLE: Need for systematic assessment of seismic effects (EH 1)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

To identify the design vulnerabilities due to seismic events, a systematic assessment needs to be carried out using a combination of deterministic and/or seismic margin, and/or PSA techniques.

The following steps should be taken in the assessment:

- Seismic input evaluation: Design basis ground motion at the plant site should be re-evaluated for seismic PSA, and the hazard curve should be defined. For seismic margin analysis a review level earthquake intensity is chosen.
- 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.
- Evaluation of fragility: The fragility of structures and components should be evaluated at each level of the intensity of ground motion. A plant walk-down would be useful to understand the installation of systems and components in a plant. For a seismic margin analysis the ability of the equipment to function for an earthquake beyond the design basis (the review level earthquake) is assessed.
- 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 evaluate the seismic robustness in comparison with plant site characteristic and to decide on the need for upgrades.

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:

Argentina

The selection and evaluation process of the sites at Atucha and Embalse for the Nuclear Power Plants with Pressurized Heavy Water Reactors was done during the 60´s and beginning of the 70’s, respectively, and the design basis parameters for the protection against external events – from natural and man-induced origins - were defined in accordance with the information and criteria valid at the time.

Thus, the design parameters corresponding to earthquakes, extreme meteorological phenomena (i.e. tornadoes) and man-induced events were determined according to the region and site specific conditions.

Regarding earthquakes, the Atucha site is located in a low seismicity region and the SL-2 earthquake level according to IAEA Safety Guide 50-SG-S1 corresponds to a minimum value of 0.10g as established by the seismic hazard study carried out in 1981 for the Atucha II Unit. The original design bases for Atucha I Unit (in operation since 1974) and the Atucha II Unit (presently under construction) did not consider explicitly the consideration of seismic loads.

The Embalse site is located in a moderate seismicity region and the original design bases for the NPP considered a value of 0.15g for the SL-2 level earthquake. Before start the operation, in 1982, the plant was reevaluated to seismic loads corresponding to a SL-2 value of 0.26g and, consequently, several upgrades were implemented.

After a period of approximately 25-15 years of operation for the units in both sites, a number of reasons justifies the necessity for reevaluating those original design parameters, including their impact in the present safety situation.

Taking all these considerations into account, as well as the fact – on the other hand and as a good performance record - that the plants safety has not been affected for any external event during their operational life, the Nuclear Regulatory Authority of Argentina decided in 1998 to formulate and carry out a programme for a gradual reevaluation of the site related (external events) design basis parameters of the Embalse and Atucha PHWR Nuclear Power Plants, which should be integrated with other programmes under development, mainly that of the probabilistic assessment of the plant safety.

As first step, and taking into account the external events against which the plants were originally protected and the significance of each of them on the plant safety, a *list of external events* for reevaluation was prepared and they were prioritized in accordance with their impact in the original design and the operating experience, as follows:

For Embalse NPP:

- Earthquakes
- Extreme meteorological phenomena (tornadoes and severe storms)
- Man-induced events (mainly, explosions and fires, external to the plant site).

For Atucha site:

- Flooding
- Tornadoes y severe storms
- Man induced external events (explosions)
- Earthquakes

For the Embalse NPP two regulatory requirements were issued by the Nuclear Regulatory Authority of Argentina for re-evaluating the seismic safety within a framework of an integrated, systematic and updated safety re-evaluation programme. In response to those requirements, the Utility has prepared a detailed work plan with definition of four phases.

Firstly, immediate actions were taken regarding the installation of new seismic instrumentation (to be operational in May 2000) and the preparation of operating procedures (currently under review) to assess actual physical damage and plant operational situation after an earthquake occurrence.

Secondly, on the basis of a regulatory requirement issued in 1996 in relation to a Probabilistic Safety Assessment (PSA) of the Plant, instructions were given to conduct the "Second Phase of the PSA" (the so called "*Seismic PSA"*), in which the seismic external event should be considered as an initiating event. As a first step in such programme, the system analysis for seismic safety will be adapted from the systems analysis already available for the "First Phase of the PSA" (i.e. for internal events). The seismic hazard curve at the site will be defined in accordance with the current Argentine regulatory norms and international practice.

For the Atucha I NPP, the re-evaluation of the original design basis against earthquake was considered with lower priority with respect to other external events as indicated in the above list. No specific actions were initiated yet, although the PSA for internal events is finished and its results could be used, partially, in the future seismic safety re-evaluation.

Canada

Assessment of seismic effects is part of the license pre-requisites at the design phase of the plant. Re-assessments are being performed for some plants.

CANDU plants are designed for two levels of earthquake, design basis earthquake (DBE, similar to IAEA SL-2) and site design earthquake (SDE, similar to IAEA SL-1). Group 2 systems including the containment system are designed for the DBE. The plant control and monitoring after an earthquake is designed to be performed in the SCA which is DBE-qualified. Design assessments for seismic-induced events are performed to confirm how the plants can be brought into safe shutdown and can be so maintained, using Group 2 systems. The heat transport system is seismically qualified so that a DBE does not induce a LOCA.

A seismic PSA for a generic CANDU 6 design has been performed. However, a seismic margin approach was used for Pickering A and is likely to be used in future for design assessment as it appears to be a more practical tool.

China

It's the licensing requirement in China

India

AERB has issued, as part of its siting guide series, a guide on Seismic Studies and Design Basis. This guide defines methodology to work out (AERB-SG-S11) Ground Motion seismic input ground motion corresponding to S1 (OBE) and S2 (SSE) levels of earthquake for NPP Sites. All the safety structures in the current PHWR designs (Narora Atomic Power Station onwards) are designed to the seismicity levels applicable to the specific site locations. AERB has Expert Groups to review the input parameters, building designs, floor response spectra, results of analytical / experimental qualification done by utility for these plants.

The older plants RAPS at Kota, Rajasthan and MAPS at Kalpakkam are in low seismic zones of the country. A programme of seismic evaluation of these plants has been taken up. Under this programme, AERB has constituted a Committee of national experts to define earthquake ground motion for review of designs of these plants. In parallel utility has identified minimum set of structures, systems and equipment which need to be qualified for old plants to ensure safety. The qualification criteria for old plants permit use of less conservative damping values and stress indices. This methodology is different than the normal seismic design of a new plant, appropriate typical analyses are being made. The evaluation of the old plants is proposed to be done by a combination of analytic and walk down exercise. Typical analysis being made are being factored by the members of the proposed walk down team to prepare a format / checklist for identifying important architectural parameters, anchoring etc. to be booked into during actual walk down.

Special reviews consisting of inspection and performance evaluation was carried out for three stations in region where mafor earthquake took place.

Korea, Republic of

The original seismic monitor, installed on the base slab of the auxiliary building, could measure only 3-axis acceleration. That does not satisfy the regulatory criteria of ANSI/ANS 2.2-88, CSA N289.5.

Therefore the seismic monitor was improved in accordance with international standards so that response spectrum can be readily available.

This assessment is a licensing pre-requisite at the design phase of the plant.

The original seismic monitor, installed on the base slab of the auxiliary building, could measure only 3-axis acceleration. That does not satisfy the regulatory criteria of ANSI/ANS 2.2-88, CSA N289.5.

Therefore the seismic monitor was improved in accordance with international standards so that response spectrum can be readily available

Pakistan

The original design of KANUPP is for 0.1 g and structures & components designed on this basis were dynamically analyzed. However, due to considerable evolution in methodologies for prediction, analysis and design, reassessment of seismic and other hazard was initiated. The intent was to determine the adequacy of anchorage and supports of components of safety and safety related systems.

In order to undertake assessment of seismic effects systematically an IAEA Seismic Safety Review Mission was invited which carried out detailed review by walk down of the plant and made short term and long term recommendation.

Follow-up of the recommendation was started. Under the long-term recommendations, geo-technical work was started and reviewed by IAEA mission in 1998. The mission recommended some additional data collection work for more precise estimation of 'g' value i.e. about 0.23 g instead of 0.2g.

Under the short-term recommendation pertaining to seismic upgrading of safety and safety related plant equipment, design work was completed. This was reviewed by IAEA experts team. Modifications, suggested by the experts, have been incorporated in the design. Implementation of most of the recommended was completed in the year 2003.

Romania

Cernavoda NPP is designed to be protected against seismic hazards. The initial requirements of using DBE and SDE earthquakes are similar for all CANDU 6 plants. However for the Cernavoda case, which is located in a country with significant seismicity, this was a very important topic from the very beginning. A lot of supplementary analyses and expertises were added so far to evaluate the seismic aspects and requirements for Cernavoda NPP and the Safety Design Matrix was reviewed. Specific supplementary activities for seismic qualification might be mentioned as the following:

- seismic walkdown was performed early during commissioning phases to review interdependencies between systems, not forseen by the existing analyses
- a seismic monitoring system, which was designed and implemented and is under current regulatory review as part of the Strategic Policy for Unit 1 relicensing in May 2001
- an evaluation of the seismic input, which was performed using also independent expertise was done

In addition to all these actions the Regulatory Body developed with an external consultant requirements for performing Seismic Safety Margin Analysis and Seismic PSA in order to require compliance of the further review of the seismic margins.

It is expected that this process will be done first as part of the seismic margin analysis and further more as part of the external PSA level 1 review, as required by the Strategic Policy for the unit 1 relicensing and periodical safety review for unit 1.

These requirements were under an IAEA mission review process and based on the general confirmation on the methodologies and scheduling there is a porces going on to implement the detailed recommendations in the updated licensing requirements.

For unit 2 there will be a need to comply with the existing requirements to fulfill these analyses from the very beginning in the licensing process.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Seismic Design and Qualification for Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-1.6, IAEA, Vienna (2003).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Evaluation of Seismic Hazards for Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-3.3, IAEA, Vienna (2002).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Probabilistic Safety Assessment for Seismic Events, IAEA-TECDOC-724, IAEA, Vienna (1993).
- Strategic Policy for Cernavoda NPP Unit 2 licensing process, CNCAN 1997.
- FSAR Cernavoda Unit 1, 1995.
- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.
- ATOMIC ENERGY REGULATORY BOARD, Safety Guide for “Seismic Studies and Design Basis Ground Motion for Nuclear Power Plant Sites”, AERB/SG/S-11/1990.

ISSUE TITLE: Need for assessment of seismic interaction of structures or equipment on safety functions (EH 2)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

It was found in earlier plant walk-downs and seismic PSA studies that failure of non-seismically designed structures and equipments could affect the safety function. Typical failures would involve items which may have been overlooked in the original design (such as false ceilings in rooms containing safety related components) 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 leading, in term, to an increase in core damage probability.

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

This issue is covered within the scope of the seismic safety re-evaluation programme presently being planned and underway in its first steps for the Embalse and Atucha I Nuclear Power Plants as mentioned in the Issue EH1. Plant seismic walkdowns by group of experts for identifying interactions between non-seismically designed structures and equipment and safety items will be part of this programme.

Canada

Seismic interactions of structures or equipment are one of the design considerations for the seismic design. The structures/equipment that do not perform a safety function, but can affect the safety related structures/equipment if they fail due to the earthquake, are designed to maintain their structural integrity (Seismic Design Category A). Also by walkdowns, on-site review of the seismically-qualified systems and components are carried out during the commissioning phase to confirm the functionality of the systems/components, including spatial interaction concerns.

India

In India, except for older Units at RAPS and MAPS, all plants have safety and safety related structures and equipment as seismically qualified to SSE/OBE level earthquake as per categorization given by AERB. Plant layout at design stage attempts to club seismically qualified structures, systems and equipment in identified seismically qualified buildings. During the design / commissioning review of safety and safety related systems Working Groups constituted by AERB along with designers also conduct walk downs to identify any potential hazard posed to the seismically qualified structures from neighbouring non-seismic structures for appropriate corrections.

The old plants RAPS and MAPS, which are in low seismic zone, are proposed to be evaluated for seismic event by a combination of analytical and walk down exercise (EH-1). This walk down would also include assessment of seismic interaction of structures or equipment on safety functions also including review whether adjoining non seismically designed structures can impair safety functions..

Korea, Republic of

Plant wide seismic walkdowns have been performed for all the Wolsung nuclear power plant units by the owner and designers at an advanced stage of construction and system installation (substantially completed). The seismic walkdown has been intended to be an additional verification process of the seismic design and qualification. The regulatory body also has carried out the seismic walkdown for all units, independently. The assessment of seismic interaction between seismically qualified equipment, components, and systems and non-seismically qualified ones in the vicinity has been part of the seismic walkdown.

Romania

Similar to EH 1

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Seismic Design and Qualification for Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-1.6, IAEA, Vienna (2003).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Probabilistic Safety Assessment for Seismic Events, IAEA-TECDOC-724, IAEA, Vienna (1993).

ISSUE TITLE: Need for assessment of plant-specific natural external conditions (EH 3)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

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. 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. This could lead to serious scenarios, such as the case of the emergency diesel generator being adversely affected and leading to a loss of cooling water. It is a requirement (e.g. see NUSS 50-C-S) to assess a plant site with respect to external events to ensure that common cause failures are adequately addressed in the design.

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:

Argentina

As said previously for issue EH1, the selection and evaluation process of the sites at Atucha and Embalse for the Nuclear Power Plants with Pressurized Heavy Water Reactors was done during the 60´s and beginning of the 70's, respectively, and the design basis parameters for the protection against external events – from natural and man-induced origins - were defined in accordance with the information and criteria valid at the time.

Thus, the design parameters corresponding to earthquakes, extreme meteorological phenomena (i.e. tornadoes) and man-induced events were determined according to the region and site specific conditions.

After a period of approximately 25-15 years of operation for the units in both sites, a number of reasons justifies the necessity for re-evaluating those original design parameters, including their impact in the present safety situation.

Taking all these considerations into account, as well as the fact – on the other hand and as a good performance record - that the plants safety has not been affected for any external event during their operational life, the Nuclear Regulatory Authority of Argentina decided in 1998 to formulate and carry out a programme for a gradual re-evaluation of the site related (external events) design basis parameters of the Embalse and Atucha PHWR Nuclear Power Plants, which should be integrated with other programmes under development - as that of the probabilistic assessment of the plant safety.

As first step, and taking into account the external events against which the plants were originally protected and the significance of each of them on the plant safety, a *list of external events* for re-evaluation was prepared and they were prioritized in accordance with their impact in the original design and the operating experience, as follows:

For Embalse NPP:

- Earthquakes
- Extreme meteorological phenomena (tornadoes and severe storms)
- Man-induced events (mainly, explosions and fires, external to the plant site).

For Atucha site:

- Flooding
- Tornadoes and severe storms
- Man induced external events (explosions)
- Earthquakes

Therefore, in relation to natural external events other than earthquakes (already discussed in Issue EH1), a complete re-evaluation of the tornadoes and severe storms hazard with more data and updated criteria was done for the Atucha and Embalse sites in 1999. This assessment included also the electric transmission lines and the results confirm the original design bases for this natural event.

It is planned to conduct the assessment for flooding at the Atucha site as part of the mentioned programme.

Canada

The evaluation for a site is done before the project design. External events evaluated include natural external events and man-induced external events. Natural external events include extreme meteorological conditions (such as high/low temperature, ice, snow, extreme winds), flooding from streams and rivers, dam failures, storm surge and seiche flooding, and earthquake. Based on the evaluation, the design basis parameters are determined considering the maximum potential worst conditions of external events. The structures, systems, or equipment that perform safety functions are designed to perform these functions during or after the maximum potential worst conditions.

See EH 1.

India

As covered for issues EH-1 and EH-2 seismic event has been used as a natural external condition for design of Indian PHWRs. Similarly extreme meteorological phenomena (like rain, wind and atmospheric temperature) determined for specific site have been considered in design. Regulatory review has also been done in a recent re-evaluation program on possible flooding or loss of ultimate heat sink in scenarios assuming upstream / downstream dams break (coupled with high rain/ seismic event) for river side plants.

It was seen that new plants, as KGS and RAPP-3&4, are designed above such a postulated flood level and have a captive seismically qualified ultimate heat sink of sufficient capacity. Older plant at RAPS has prepared an emergency operating procedure for such an event to ensure safe shutdown and cool-down during the warning period before the postulated flood reaches critical level. New 540 MWe units at TAPP-3&4 are located above postulated flood level, calculated with appropriate combination of phenomena associated with site situated at sea shore.

After flooding at KAPS, the incident was reviewed as per ASSET methodology. Recommendations arising out of this review, both design related and administrative were implemented at KAPS and other projects. For older stations, flood review has resulted in some back fits e.g. DG set at higher elevation at RAPS-1&2 and Additional Utility Building (ADUB) having one DG set and diesel driven compressor at higher elevation at MAPS. Similarly, after December 2004 tsunami due to Sumatra earthquake experienced at eastern coast of India, all the stations located at sea shores were reviewed and emergency operating procedures for external flood were updated and certain modifications were suggested.. AERB Guide on Design Basis Flood For Nuclear Power Plants at Coastal (AERB/SG/S-6B) requires considering higher of wave heights generated due to cyclonic storm or tsunami for arriving at the maximum water level at the site.

Korea, Republic of

This assessment is a licensing pre-requisite at the design phase of the plant.

Romania

The Cernavoda NPP assessment of the qualification to external hazards was a prerequisite of all the licensing phases: sitting, construction, commissioning and operation. The safety reports fulfilled for these phases (Initial Safety Analysis Report, Preliminary Safety Analysis Report, Final Safety Analysis Report) are based on detailed analyses for the evaluation of external hazards on the plant:

- seismic events
- extreme meteorological phenomena
- possible effects of human activities in the area
- flooding

The medium term Licensee PSA level 1 strategy takes into account to include the external events, too.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Design Safety Requirements, IAEA Safety Standards Series No. NS-R-1, IAEA, Vienna (2000).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Site Evaluation for Nuclear Installations Safety Requirements, IAEA Safety Standards Series No. NS-R-3, IAEA, Vienna (2003).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Meteorological Events in Site Evaluation for Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-3.4, IAEA, Vienna (2003).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Flood Hazard for Nuclear Power Plants on River and Coastal Sites Safety Guide, IAEA Safety Standards Series No. NS-G-3.5, IAEA, Vienna (2003).
- Strategic Policy for Cernavoda NPP Unit 2 licensing process, CNCAN 1997.
- FSAR Cernavoda Unit 1, 1995.
- PSAR for Unit 2, 1985.
- Strategic Policy for Cernavoda NPP Unit 1 relicensing in May 2001, CNCAN March 2000.

ISSUE TITLE: Need for assessment of plant-specific man induced external events (EH 4)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

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. 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.

It is recognised that considerable work has been done and is being done in the area of Design Basis Threats (DBTs) especially since the events of September 11, 2001. These developments have not been considered in this version of the TECDOC and the issue would need to be correspondingly updated.

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:

Argentina

As said previously for issue EH1, the selection and evaluation process of the sites at Atucha and Embalse for the Nuclear Power Plants with Pressurized Heavy Water Reactors was done during the 60´s and beginning of the 70's, respectively, and the design basis parameters for the protection against external events – from natural and man-induced origins - were defined in accordance with the information and criteria valid at the time.

Thus, the design parameters corresponding to earthquakes, extreme meteorological phenomena (i.e. tornadoes) and man-induced events were determined according to the region and site specific conditions.

After a period of approximately 25-15 years of operation for the units in both sites, a number of reasons justifies the necessity for re-evaluating those original design parameters, including their impact in the present safety situation.

Taking all these considerations into account, as well as the fact – on the other hand and as a good performance record - that the plants safety has not been affected for any external event during their operational life, the Nuclear Regulatory Authority of Argentina decided in 1998 to formulate and carry out a programme for a gradual re-evaluation of the site related (external events) design basis parameters of the Embalse and Atucha PHWR Nuclear Power Plants, which should be integrated with other programmes under development, mainly that of the probabilistic assessment of the plant safety.

As first step, and taking into account the external events against which the plants were originally protected and the significance of each of them on the plant safety, a *list of external events* for re-evaluation was prepared and they were prioritized in accordance with their impact in the original design and the operating experience, as follows:

For Embalse NPP:

- Earthquakes
- Extreme meteorological phenomena (tornadoes and severe storms)
- Man-induced events (mainly, explosions and fires, external to the plant site).

For Atucha site:

- Flooding
- Tornadoes and severe storms
- Man induced external events (explosions)
- Earthquakes

Therefore, in relation to man induced external events a re-evaluation of the original design bases is presently under way for both sites, i.e. Atucha and Embalse. The objectives are to confirm in the case of Atucha site, the parameters for a potential explosion of a gas cloud as defined by the German KTA Standard which were used for the original design of the plant, and to confirm in the case of the Embalse site that no potential for such events are expected.

A work plan for conducting the man induced external events assessment for Atucha and Embalse sites was prepared in 1999 and its implementation will start in the current year 2000.

Canada

Man-induced events evaluated include commercial or military aircraft crash, explosion, toxic chemicals, fires, collision with intake structures, and liquid spills from industrial facilities nearby, storage facilities, or from transportation. The design basis parameters are determined considering the potential maximum worst conditions from man-induced events. The structures, systems, or equipment that perform safety functions are designed to perform these functions during or after the maximum potential worst conditions.

See also EH 1.

China

It’s the licensing requirement in China

India

The present sites for plants in India are at such locations that they are significantly away from population or industrial centres. At present any new activity within 5 km radius of plant which may qualify as new external event to be considered for evaluation of safety of plant are not permitted. The site selection for Indian Nuclear Power Plants is governed by AERB code of practice on Safety in Nuclear Power Plant Siting (AERB/SC/S). For man induced external events (like aircraft crash, chemical explosion and toxic gas release, blasting operations etc.) screening value distances as specified in AERB code are followed. Thus AERB has in general not found any addition to the list of man induced external events than those considered in original design basis of plants. However, RAPS an older PHWR has a Heavy Water Plant based on hydrogen sulphide exchange process. Leakage of hydrogen sulphide in this plant can affect the nearby NPP. Action plan to deal such situation exists and exercises/drills are conducted periodically to check the readiness of the mitigatory systems. Design aspects cover measures required to cater for H_2S leaks.

Korea, Republic of

See EH 1, EH 3

Romania

Similar to EH 3

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, External Human Induced Events in Site Evaluation for Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-3.1, IAEA, Vienna (2002).
- INTERNATIONAL ATOMIC ENERGY AGENCY, External Events excluding Earthquakes in the Design of Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-1.5, IAEA, Vienna (2003).
- ATOMIC ENERGY REGULATORY BOARD, “Code of Practice on Safety in Nuclear Power Plant Siting”, AERB/SC/S (19191).

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

This issue is also applicable to NPPs with LWR.

Accident analyses 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.

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 including dual failures satisfies regulatory requirements for protection of barriers and dose to the public. This approach is often used in conjunction with generic or plant specific PSA to make regulatory decisions.

Safety significance

An incomplete set of accident analysis means that some sequences might not be fully mitigated. The lack of a comprehensive set of full scope analyses 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:

Argentina

Safety Analysis Report and other Mandatory Documents are required to be updated by the plant each 5 years according the Periodic Safety Review plan used in Argentina. In particular, Chapter 15 “Accident Analysis” and PSA evaluations are the most relevant and must be updated using the available new techniques and corresponding to the state of the art.

Canada

In general safety reports for the operating stations are updated periodically and safety reports for new stations use the most recent tools and lists of accidents. For the specific example of flow blockage, closure of this generic action item depends to a large extent on the results, and interpretation thereof, of a series of tests that the licensees will be carrying out on a specially designed test rig. The purpose of the tests is to develop a clearer understanding of the kind of interaction that would occur in the event of molten fuel ejection into the moderator.

In 1995 CNSC staff raised a generic action to address the hypothetical scenario of severe channel blockage resulting in molten fuel-moderator interaction. A summary is given below of the position statement describing that issue:

A severe flow blockage in a fuel channel, or an inlet *feeder* stagnation break, could potentially lead to fuel melting, channel rupture and ejection of molten fuel into the moderator. It is uncertain as to whether the resulting molten fuel/moderator interaction could damage the shutoff rod guide tubes and prevent shutdown system 1 (SDS #1) from functioning properly. It could also damage other fuel channels, or the calandria vessel itself. There has been a long-standing difference of opinion between CNSC staff and licensees and their respective consultants on the severity of the molten fuel/moderator interaction. Starting the first quarter of 2000, however, licensees initiated an experimental program to resolve this matter. A panel of three independent fuel-coolant interaction experts was set up to review the experimental program and the resolution criteria proposed by industry. CNSC staff accepted the panel's final recommendations and the industry's proposed closure criteria. CNSC staff also accepted the licensees' proposed experimental program schedule, which plans to conclude the experimental program by the third quarter of 2005. Although some delays have been encountered due to unexpected technical challenges and problems in obtaining the classification approval for the test facility, the first of the planned four tests was performed successfully in December, 2004. CNSC staff expects an update from the licensees on the schedule of the remaining tests in the first quarter of 2005.

India

Design basis events are chosen in deterministic manner for the safety assessment. AERB-SG-D-5 gives a comprehensive list of initiating postulating events which the licensees should cover in safety report. AERB decided to bring out a safety guide on safety analysis. The objective of safety analysis review during Periodic Safety Review is to check the extent of validity of the existing safety analysis taking into account the actual plant status, expected degradation till the next renewal of authorization or the end of predicted life and current analytical methods, safety standards and knowledge.

Korea, Republic of

KINS is reviewing the GSIs for CANDU plant. What measures should be taken will be decided when this review is over. Moreover, new research programs on CANDU safety issues are being launched.

Romania

The requirements for the Design Basis Accidents for Cernavoda NPP unit 1 were reviewed before the commissioning of the plant started and during the commissioning phases up to the moment of issuing the Test Operation License. The methods for review were based on:

- the feedback from other projects, more updated
- the systematic review using PSA level 1 analyses in parallel with the licensing process
- the use of external independent expertize for those topics for which independent review of the evaluations was needed.

It was also reviewed the trip coverage as defined in the FSAR report, which was considered of concern mainly for partial and low power states. During this process the actual status of the researches was considered, like (to refer to some of them: the experiments on molten fuel -moderator interaction, review of the analyses for flow blockage).

It was also considered the actual starting point of the CANDU 6 that the CANDU 6 for Cernavoda NPP has already by basic safety design and licensing requirements embedded in it the coverage of a broader area of BDBA events, as for instance the LOCA plus Loss of ECCS event.

The results of the review lead to a situation when some events usually in the category of BDBA are to be included in the DBA category for Cernavoda unit 1. This result remained applicable for unit 2, also. Examples of such events are:

- the loss of off-site power and LOCA accidents, for which both analyses in FSAR chapter 15 and tests in commissioning at various power levels were done.
- transients not postulated previously in the list of mandatory tests in commissioning, as the series of tests for the BOP-NSP interface systems (their coincident failure: instrument air and service water and/or instrument air and/or class i/ii failures as for instance)

The review of the licensing requirements for unit 2 lead to postulating more events from the BDBA category, some of them involving the requalification (EQ for instance) by comparison with unit.1. For unit 1 the differences are to be addressed in the Periodical Safety Review process from 2001 on. There are also going on evaluations of the Cernavoda safety evaluations, including the DBA definition as part of the PHARE program to support the unit 2 licensing process, which is a project, which is already started.

The Licensee will addres the issue of the systematical review of safety correlated also with the periodical safety review process, included in the long term research and development program and in cooperation with COG..

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment and Verifications for Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-1.2, IAEA, Vienna (2001).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Accident Analysis for Nuclear Power Plants, Safety Reports Series No.23, IAEA, Vienna (2002).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Accident Analysis for Pressurized Heavy Water Reactors, Safety Reports Series No.29, IAEA, Vienna (2003).
- CNSC (Canada) Position Statement 95G01 "Molten Fuel-Moderator Interaction".
- ATOMIC ENERGY REGULATORY BOARD, Safety Guide, “Renewal of Authorisation for Operation of Nuclear Power Plants”, AERB/SG/O-12 (2000).

ISSUE TITLE: Adequacy of plant data used in accident analyses (AA 2)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

Compliance with Bundle and Channel Power Limits:

The limiting values for bundle and channel powers are specified in the Operating License for each station.

Licensees ensure compliance with these limits by following operating procedures, which are based on analyses. However, current validation of the channel and bundle power analyses method is such that the errors associated with their calculations are not well-defined. If larger allowances for uncertainties were needed, channel or bundle power may become more limiting than bulk power.

Operation with a Flux Tilt:

The adequacy of Regional Overpower Protection (ROP) or Neutron Overpower Protection (NOP) trips for reactor operation with a flux tilt is demonstrated by analyses, which take into account different plant states for which continued operation is permitted.

ROP/NOP system design is based on information derived from simulations of certain reference and perturbed flux shapes in the reactor core. Trip setpoints are established from these simulations to prevent any channel reaching its critical power limit in case of a bulk loss of regulation. One key component in the analysis is the relationship between flux values at detector locations and channel powers for various flux shapes. The analyses assume that the ratios of changes in fluxes and channel powers due to perturbation, called simulation ratios, are invariant with respect to the reference flux shape. On this basis a limited number of combinations of credible perturbed flux shapes and an untilted reference flux shape are analysed to derive trip setpoint values. Furthermore, these trip setpoint values provide coverage for different plant states. Differences in the reference flux shape used in the analyses and actual flux shapes are accounted for by regular detector calibration.

At issue is the adequacy of error allowances used to derive the trip setpoint values, since these can be sensitive to the accuracy of reference and perturbed flux shapes and associated simulation ratios in simulations. While tilted-flux operations were not considered as initial conditions in ROP/NOP analyses, the reactors are permitted to operate with relatively large flux tilts. For initial flux shapes with relatively large tilts the invariance of simulation ratios needs to be demonstrated to support the adequacy of error allowances used to derive the trip setpoint values for all plant states for which continued operation is permitted. In order to demonstrate the adequacy of ROP/NOP trip setpoints, licensees should:

a) determine the maximum tilt permitted by the current operating procedures for prolonged operation with a flux tilt, prior to any operator action,
b) generate a steady state flux distribution, corresponding to the maximum tilt permitted by the current operating procedures, and design-basis and abnormal perturbation flux shapes, corresponding to this steady state shape, and
c) assess simulation ratios (ratios of changes in fluxes and channel powers due to perturbations) for the above flux shapes, and assess the ROP/NOP trip coverage by determining whether the ratios are invariant within any postulated error allowance.

Quality Assurance in Safety Analyses:

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. Accident analyses need a plant model which must be constructed on the basis of valid data. This data base is subject to a 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.

It is also important that licensees perform safety analyses in a systematic quality assured manner so that high confidence can be attributed to the definitions of the licensing basis and safe operating envelope for each of the licensees' stations.

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.

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:

Argentina

The program of accident analysis related to anticipated transients and loss of coolant accidents includes plant data preparation, input data preparation, cross checking procedures, input check preparation, is underway. These activities are made on the basis of reliable and verifiable data on the plant as-built and modified over the plant lifetime. In addition to, these activities are made on the basis of operation plant procedure in agreement with actualized Plant Operation Manual and PSA.

Canada

Compliance with Bundle and Channel Power Limits:

To achieve closure of the generic aspects of this issue, licensees are required to complete their related code validation program and perform additional analyses to address the identified issues related to methodology, models and computer codes. Specifically, the following should be examined:

a) the adequacy of single values of channel and bundle power limits to ensure meeting acceptance criteria for all design basis accidents; limits of the operating bundle and channel power envelops should be established and specific limiting bundle and channel power distributions defined as required;
b) the adequacy of error allowances and level of confidence, to cover the various sources of uncertainty in codes predictions, plant data measurements and methodology; the allowances should, at 98% confidence level, account for:
    - error in total reactor power normalization,
    - error in code methodology and modelling,
    - error in measurements (FINCHs and flux detector/mapping), and
    - xenon transients initiated by fuelling;

c) the adequacy of the validation of computer codes used for core-tracking and compliance with licence power limits, consistent with validation plans for other safety and licensing codes; and
d) the acceptability of compliance procedures; the following issues should be adequately addressed:
   - consistency;
   - actions when limits exceeded;
   - assurance of compliance during periods of time at which core-tracking runs are made.

Operation with a Flux Tilt:

To achieve closure, licensees are required to perform the following:

a) determine the adequacy of various error allowances and assumptions, such as the invariance of simulation ratios used in the analyses, to cover the actual maximum tilt permitted by the revised and improved operating procedures - there should be a specific error allowance for simulation ratio variation due to flux tilts in ROP/NOP error analysis, its magnitude should be sufficient to cover variations found in analyses that have been performed, as well as other cases which are not analyzed;
b) determine the sensitivity of ROP/NOP analysis results to flux tilt definition; at issue is the fact that while the RRS flux tilt definition is based upon bulk regional averaged parameters, the ROP/NOP requirement is to prevent the critical power in any individual channel being reached, and
c) determine the potential impact of factors not fully covered in current analyses on the effectiveness of the ROP systems; the main factors not covered are: the effect of transient xenon, boiling in a region of channels, and replacement of a failed detector.

Quality Assurance of Safety Analysis:

To achieve closure, licensees are required to:

a) have a QA program that includes requirements for safety analysis activities, and which meets applicable QA standards;
b) undertake a QA program assessment in accordance with N286.0 to determine the effectiveness of the licensees' current QA programs specifically with respect to safety analysis activities; this assessment shall include a formal program review and audits; the program review shall determine the extent to which the QA program meets applicable QA standards (for example, references 1-3) and the AECB staff's expectations in this position statement;
c) submit to the AECB a report of the assessment identifying program deficiencies and a plan and schedule for their correction;
d) implement the corrections to the QA program;
e) submit to the AECB sufficiently detailed information to demonstrate that the QA program meets the requirements of applicable QA standards (for example, references 1-3) and this position statement; and
f) provide six-monthly progress reports on resolution of this GAI.

India

India is currently operating 220 MWe nuclear reactors without any boiling in the channel. These small reactor cores are having channel temperature monitoring at the outlet, which is an indirect measure of the channel power. These small reactor cores do not have large flux tilts due to xenon oscillation in the axial or radial directions. Few channels in the reactor are completely instrumented in terms of flow and inlet temperature also. The thermal power estimated from the channels is compared routinely with the results available from physics fuel management codes. Thus the accuracy of the estimation of the channel power by physics codes is indirectly monitored. In-core flux monitoring at selected positions in the centre of the reactor has also been introduced in the recent design with a view to validate the bundle power calculated by fuel management codes. The 540 MWe design which has gone through comprehensive regulatory review will continue to have no boiling and channel outlet temperature will be monitored as in 220 MWe units. The 540 MWe units uses 37 element fuel bundle with bundle and channel power considerably lower than CANDU 600 MWe design. Flux monitoring in this reactor is however, as in the current CANDU design. Zone controllers have also been provided in the 14 zones to control the flux tilt. The reactor protection will be provided by a core over-power protection system as well as by channel differential temperature monitoring over large number of channels. This provisions are currently being reviewed by AERB and a cautious approach will be taken for 540 MWe units to ensure that sufficient margin in bundle and channel power are available while operating with the flux tilt.

Korea, Republic of

This issue is under review by KINS

Romania

The performance of all the calculations for the FSAR and support documents to it are to be performed based on codes which are in V&V (verification and validation) process and/or already validated. It is also required by CNCAN that the data are checked against the site specific data. There are also of highest priority the requirements that the Licensee has the capability to perform by itself all the calculations as required in the licensing basis. This is one of the basic issues of the re-licensing process for unit 1 in May 2001 and of the resuming licensing activities for unit 2.

The Licensee will address the issue of the systematical review of safety correlated also with the periodical safety review process, included in the long term research and development program and in cooperation with COG..

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment and Verifications for Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-1.2, IAEA, Vienna (2001).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Accident Analysis for Nuclear Power Plants, Safety Report Series No.23, IAEA, Vienna (2002).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Accident Analysis for Pressurized Heavy Water Reactors, Safety Report Series No.29, IAEA, Vienna (2003).
- CNSC Position Statement 91G02 "Operation with a Flux Tilt".
- CNSC Position Statement 95G03 "Compliance with Bundle and Channel Power Limits".
- CNSC Position Statement 99G01 "Quality Assurance of Safety Analysis".

ISSUE TITLE: Computer code and plant model validation (AA 3)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

Qualification of the Safety Analysis Activities:

Safety analysis includes generation of certain safety-related information about the design, operation, and expected behaviour of the reactor and the safety systems under various conditions including normal operation, operational transients, and postulated accidents such as LOCAs to ensure the safety of the plant as well as the public. Such information is a primary definer of the station's licensing basis and the bounds of the safe operating envelope. This also includes demonstrating that the station is being operated within the conditions of the operating license.

The credibility of this safety-related information depends on the degree of conservatism incorporated into the safety analysis and on the qualification of the safety analysis activities such as the analysis methodology, assumptions made, tools used, and input preparation. Safety analysis qualification, especially verification and validation of the tools such as computer programs against test or experimental data, and their appropriate application is of increasing importance among the regulatory bodies and the licensees. Some examples of issues are lack of the soundness of the methodology, poor documentation of the computer code validation, inappropriate application of the experimental data for code validation such as inadequate dimensional scaling. To improve the credibility and confidence of the safety analysis, it is desirable for the licensee to perform the safety analysis using a certified procedure for generation of the plant data, assumptions, using adequately validated computer codes, and well documented input and output decks. For this end, it is recommended for the licensee to set up a plan how to qualify their safety analysis activities, including proving the soundness of their methodology, validation of the tools for analysis, conservatism of the assumptions made, and the input preparation for the computer codes.

Recent experimental data in reactor physics area identified several shortcomings of the major analysis tools in cell codes such as POWDERPUFS-V. The most important shortcomings found are: inaccurate predictions of key parameters for accident conditions, lack of proper validation data for important phenomena and range of conditions, and a significant gap between the state of knowledge reflected in the licensees' computer codes and the current state of knowledge in this area.

PHT Pump Operation Under Two-Phase Flow Conditions:

The operation of Primary Heat Transport (PHT) pumps under low suction pressure and significant void can be detrimental to the integrity of the PHT system piping due to the generation of large-amplitude pressure pulsations and excessive pump set vibration. In the past, the PHT piping fatigue analysis was done using a limiting forcing function (harmonic excitation) obtained from laboratory tests of full-scale PHT pumps. Given the underlying assumptions, especially the amplitude and frequency of excitation, this approach was very sensitive to interpretation of the test data and their application to the PHT system. Consequently, the assessment of the piping fatigue life may not have been conservative.

Further work was therefore required to develop a mechanistic pump model from the available data base and apply it to the PHT system piping configuration. Compared to the use of some arbitrarily assumed limiting forcing function, this additional work was expected to give a more realistic representation of the behaviour of the PHT pump and piping under various accident conditions.

Safety significance

Without careful validation of computer codes for the purpose, i.e. specific plant and particular application/ scenario, to which they are applied, confidence in the results of accident analysis is reduced. 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
- ____xx____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Argentina

Validation analysis codes used in thermal hydraulic safety assessments (RELAP codes) are verified through CAMP project by contracted with international test results from scaled and semiscaled experimental facilities. Otherwise, validation analysis of plant models are made under conservative assumptions and on basis of operational data of nuclear power plant. On the Embalse NPP several validated models have been implemented using FIREBIRD III. CATHENA code supplied by AECL is also considered in the accident sequences modelling and it is included within the validation program.

Canada

PHT Pump Operation Under Two-Phase Flow Conditions:

To achieve closure of this action item, for stations other than Darlington, licensees are required to perform the following:

a) re-assess the PHT pump behaviour under two-phase flow conditions with emphasis on the unsteady feedback from the PHT system piping (consider assessment of the latest full-scale tests involving the Darlington PHT pump); and
b) update the fatigue analysis of the PHT system piping.

Validation of Computer Programs Used in Safety Analysis of Power Reactors:

Regulatory Body:

To achieve closure, licensees are required to:

a) undertake a program of program validation that will meet AECB staff expectations;
b) provide bi-annual summary reports describing the overall progress and the major milestones achieved; and
c) submit the information identified below; the extent of information provided shall be sufficient to demonstrate that the expectations in this position statement have been met; for items 4) to 9) it is at the discretion of the licensee, the way in which the information is organised into individual reports, and the extent of information needed for a particular computer program and application:

1) A list of the computer programs and corresponding applications to which the licensee considers this GAI to be applicable (September 1999).
2) Description of the key elements of the validation process (September 1999). (For the industry's proposed validation process, the purpose of each of the documents: technical basis document, validation matrices, validation plans and validation plans should be

defined, their overall content described and the inter-relationships between the documents defined.)

3) A report for each computer program which summarises the range of conditions over which the program is intended to be used and compares these with the ranges of conditions in the experimental database. The report will identify "gaps" (see 2.3 b)) in the validation experimental database and either the plans for their closure or a justification for leaving them open (March 2000).

For each event or group of events:

4) identify the inter-relationships between the type of safety analysis, the important safety limits to be met, the technical disciplines and the overall goals of the validation;
5) in each of the technical disciplines, identify all phenomena for which validation is potentially required, and justification for any ranking used;
6) identify experimental facilities which can be used to validate the programs for the required phenomenon; indicate any gross physical distortions that may make the facility and the way in which it behaves atypical of the reactor: such distortions include geometric differences (including scaling), differences in material properties, differences in fluid physical properties;
7) identify specific experimental tests that will be used in the validation exercises; indicate the ranges of relevant conditions in comparison with those typical of the equivalent reactor case;
8) produce validation plans for each of the programs: the plans should include sufficient information (together with other documents) to demonstrate that once the validation is complete the programs will be adequately validated; the validation plans should include the rationale for the extent of validation;
9) produce validation reports for each of the programs; the reports should include sufficient information to demonstrate the claimed accuracy of the program for the given application.

Industry:

A formal and extensive process has been undertaken by the industry in Canada covering validation and verification of all major safety analysis codes. While such codes have been validated in the past, the process provides an auditable trail of V&V and assures completeness. The output will be the accuracy of, and uncertainty in, code predictions for fundamental safety parameters. It is scheduled for completion in 2001.

To ensure efficient use of resources for this major endeavour, the industry has converged (with the exception of system thermohydraulics) on a single common set of codes – the Industry Standard Toolset.

Replacement of Reactor Physics Computer Codes Used in Safety Analysis of CANDU Reactors:

Regulatory Body:

To achieve closure, licensees are required to undertake a structured program of replacement of reactor physics computer codes. The program should include the following:

a) implementation of a structured approach of specific code replacement, including firm schedules and dates of replacement;
b) replacement of all the relevant reactor physics computer codes used in safety analysis and operation, including PPV, MULTICELL, PPV-based RFSP, and SMOKIN; PPV should also be retired from use for fuel management and core tracking simulations;
c) ensuring that validation of new codes is in accordance with requirements in GAI 98G02, Regulatory Guide C-149, and CSA Standard N286.7, and addresses the issues given in 2.2 (a) above;

d) assessment of the impact of code replacement on current safety margins and identify the limiting design-basis accidents whose safety margins might be significantly affected;
e) assessment of the impact on safety report updates' programs and identify ongoing and future activities that might be affected by the reactor physics codes' replacement;
f) definition of adequate interim allowances for use with key reactor physics parameters provided by PPV for safety analysis, such as void reactivity, delayed neutron fraction, fuel temperature reactivity, prompt neutron lifetime, until replacement of PPV; and
g) adequate coverage for any interim period, where a computer code has been replaced, during which period the full impact on all safety analyses may not have been fully addressed.

Industry:

The industry is in the process of replacing PPV with WIMS/RFSP. All new designs are analyzed using the new toolset, at least for critical cases. Utilities are phasing in the tools for their own plants.

China

In China, the validation and verification (V&V) for computer codes follows the international practice, i.e. the owner of the NPPs should provide documents comparing the results of their own codes or with the experimental data. In addition, when the NPPs are imported from foreign countries, the documents for approval on these codes from the Authorities of exported countries, if any, also can be the basis for NNSA's judgement. It is required that the documents of V&V of the CANDU computer code should be provided to NNSA.

India

Computer codes for dynamic analysis of Indian PHWRs:

As part of safety assessment, a number of operational transients need to be analysed to ensure that the reactor control system and the other design features provided are adequate to maintain the reactor under safe conditions. Computer codes for systems dynamic analysis are developed to carry out the required safety analysis. In view of the differences between the 220 Mwe and the 540 Mwe PHWRs, separate computer codes have been developed for the required analyses for the two reactor designs. Development of these codes involves integration of the individual models for the various plant components/equipment/systems, their interactions and feed back effects. The models to be incorporated include those for simulating the reactor core and PHT system, steam generator and the related controls for steam generator level, steam generator pressure, PHT system pressure, reactivity, reactor power setback as well as reactor regulating system details. The conservative equations are solved along with the constitutive relations for parameters like heat transfer coefficient. The core heat transfer model considers conduction in fuel, cladding as well as the effects of fuel-clad gap. The code employs a modular structure. The modules developed for individual equipment/systems are tested against experimental data. Apart from this, the overall plant dynamics model is also tested against data generated on the NPP during commissioning/testing. The code developed for the dynamic analysis of the 540 Mwe PHWR has been utilised, among other things, for assessing the performance of the control system and also optimising its design.

PHT pump operation under two phase flow condition:

As per the safety principles boiling is never allowed in PHT in Indian PHWRs. Hence PHT pumps never operate under two-phase condition except under few abnormal conditions. During abnormal condition the situation is avoided by tripping of pumps on overload or pressurizing pumps trip (and on low storage tank level in some older units). PHT is maintained solid by pumping inventory through pressurizing pumps. Thus whenever pressurizing pumps trip the PHT circulating pumps (PCPs) will also trip to avoid operation of the pump under two phase condition.

Validation of Computer programs used in safety analysis of power reactors:

Licensees are required to submit the list of Computer Codes used and their applicability in safety analysis.

Models and correlation used in each of the Code goes through vigorous assessment and modification based on the degree of conservatism by the Committee of experts. Individual models and correlation coded should yield satisfactory results with those of experiments conducted to establish the correlations. Validation of integrated model is a specific requirement by AERB. The validation of the code could be comparison with the results of experiments or international codes, which are well validated.

Safety guide (AERB/SG/D-18) has been prepared to guide the licensees in adopting the correlation, model and input parameter to assess the LOCA condition.

The codes used by utility like ATMIKA for LOCA analysis has been validated against discharge and fuel heat up during blow down period. Further, validation work has been planned to assess the thermal hydraulic behaviour during ECC injection. Similarly Transient analysis code ATMIKA.T has been well validated against the plant transients encountered in actual plant operation. Other codes such as ‘Contact’ used for severe accident, do not have experimental base for validation, the conservatism adopted in the code and the model used is assessed by the Committee of experts for best estimate approach.

Replacement of Physics Computer Codes used in Safety Analysis:

The lattice cell code CLIMAX which was used for the analysis purpose is replaced with the more conservative code CLUB mentioned in AA-9. The TRIVENI Code which is the fuel management code for the regular reactor follow-up has been modified to TRIXEN Code to account for cell xenon, based on the actual power. These codes have been validated from the data on the channel power and channel outlet temperature monitored in the operating power plants.

The importance of development of simulation models for physics analysis was recognised right from the beginning. These models are very essential to simulate the feed-back effects between core thermal hydraulics and physics during a transient. Initially some computer codes were obtained from abroad and adapted. However it was soon realised that such codes need to be developed indigenously, especially considering available computer environment. To start with simple models were developed. Subsequently, such models were improved with the availability of more powerful computing systems. Validation of these models and codes were done with available experimental data from zero energy facilities and commissioning experiments of power reactors as well as with international calculational benchmarks, through participation in the various co-ordinated research programmes, specialists meeting etc., organised by the IAEA. Computer code CLUB was developed to predict the variation of LOCA reactivity as function of burnup. A two group 3-D improved quasistatic neutronic transient analysis code 3-FAST has been developed to simulate LORA and LOCA for a large PHWR. Results of both the codes are in good agreement with IAEA TECDOC and Argentinean code PUMA-C and AECL code employed by Romania.

Computer code for containment performance analysis:

A comprehensive code system has been developed for analysis of various aspects of containment behaviour. These include: CONTRAN for containment transient analysis (used for predicting pressure temperature transients following a LOCA), HYRECAT for the analysis of hydrogen mitigation using catalytic recombiners, NAUA/MOD5 for analysing the aerosol behaviour and SPARC for analysing the aerosol scrubbing behaviour in the suppression pool. All these codes have undergone extensive validation using experimental data. A large number of tests were carried out to study the influence of various governing parameters on containment behaviour and validation of the code CONTRAN has also been carried out using test data from the Battelle Frankfurt Containment (BFC) test facility Germany. Computer code HYRECAT for the analysis of hydrogen mitigation phenomena in the containment using catalytic recombiners has been validated using data from two of the several laboratory tests performed in a 22 litre stainless steel vessel with initial hydrogen concentration of 5.1%(v/v). Further to this, large scale experiments are in progress on a 22 m^3. vessel.

Computer codes NAUA/MOD5 is an advanced multi-compartment aerosol behaviour analysis code, which models the various aerosol phenomena occurring in the nuclear reactor compartment. SPARC has been adapted and suitably modified for analysing aerosol scrubbing behaviour of the suppression pool. The code models particle captures by condensation of steam, impaction, sedimentation, centrifugal deposition and diffusional deposition. Validation of this code was done by conducting an experiment.

Computer code for coolant channel:

The computer code SCAPCA has been developed for prediction of creep-growth contact life of a pressure tube with calandria tube. The computer code HYCON has been developed to predict hydrogen concentration in pressure tube. The code BLIST-1&2 are used for predicting formation of growth of hydrogen blisters in pressure tubes. All these codes have been validated using data from RAPS and MAPS coolant channel.

Korea, Republic of

KINS is reviewing the GSIs for CANDU plant. What measures should be taken will be decided when this review is over.

Romania

This is an important requirement under implementation in the sense that the Licensee has yet to take actions to develop and maintain the capability for full independent from the original designer calculations . The progress on this issue will be updated as a result of the licensing process for unit 1 and unit 2 following the milestones setup in 2001.

The Licensee will address the issue of the systematical review of safety correlated also with the periodical safety review process, included in the long term research and development program and in cooperation with COG..

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment and Verifications for Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-1.2, IAEA, Vienna (2001).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Accident Analysis for Nuclear Power Plants, Safety Report Series No.23, IAEA, Vienna (2002).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Accident Analysis for Pressurized Heavy Water Reactors, Safety Report Series No.29, IAEA, Vienna (2003).
- CNSC Position Statement 98G01 "PHT Pump Operation Under Two-phase Flow Conditions”.
- CNSC Position Statement 98G02 "Validation of Computer Programs Used in Safety Analysis of Power Reactors".
- CNSC Position Statement 99G02 "Replacement of Reactor Physics Computer Codes used in Safety Analysis of CANDU Reactors".

ISSUE TITLE: Need for analysis of accidents under low power and shutdown conditions (AA 4)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

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 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 (for PWRs) the refueling phase is high. Important contributors to risk are boron dilution (for PWRs), 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.

Safety significance

In shutdown conditions, there are fewer 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:

Argentina

PSA for low power and shutdown conditions is a formal requirement. PSAs for such modes of operation can contribute to the core damage frequency at a level comparable with the power PSA. To perform the PSA of such operation modes it is necessary to reevaluate the accident analysis to delineate accident sequences specifically for low power and shutdown modes.

The shutdown PSA is in progress and is expected to be finished by the end of 2005. At present, there are some still pending tasks. Shutdown PSA includes the following operational states:

- Low power and hot shutdown,
- Power decrease,
- Cold shutdown with open Primary Heat Transport System (PHTS) (Steam Generators -SG - open and SLARETTE equipment operating in the pressure tubes -PTs-),
- Cold shutdown with closed PHTS (SGs closed),
- Cold shutdown with closed PHTS and available service water system (at the last stage, when the Emergency Water Supply –EWS- is connected to the service water system).

-Power start up.

Canada

Some licensees will re-visit these scenarios as part of the safety report update process.

AECL has recently undertaken PSAs of low power and shutdown states for its product line.

China

It's the licensing requirement in China.

India

Low power and shutdown conditions have been considered for identifying possible accident situations. Following Chernobyl accident special attention was drawn towards operation at low power and possible scenarios resulting in neutronic disturbances during shutdown. Loss of regulation (due to malfunction of control devices, boron dilution etc.) during shutdown condition considered as a design basis event.

Precautions and availability of alternate features during identified maintenance activities on reactor coolant system, moderator system are covered in Technical Specifications for operation. Any procedure for other unusual maintenance activities on shutdown systems, coolant system, moderator system or service water system are reviewed by regulatory body before implementation.

Korea, Republic of

KINS is reviewing the GSIs for CANDU plant. What measures should be taken will be decided when this review is over. Moreover, new research programs on CANDU safety issues are being launched.

Romania

The need for analysis of the accidents at various power levels different from the nominal one is one of another basic priority issue in the licensing processes of unit 1 and unit 2. They are being addressed as follows:

- For the partial powers there are going on activities to comply with the requirements related to the Trip coverage aspects and their support documentation.
- For the shutdown states the main requirements are related to:
    - assure calculations of the "recall times' for all the scheduled activities and the available heat sinks during the outages, as prerequisites for CNCAN approval to perform specific activities, declared "witness points" or "hold points" Heat Sink Manual is a type "A " procedure (i.e approved by CNCAN)
    - monitor as per the approved procedures the Guaranteed Shutdown State (GSS), "GSS manual " is also a type "A" procedure.

The Licensee will address the issue of the systematical review of safety correlated also with the periodical safety review process, included in the long term research and development program and in cooperation with COG.

ADDITIONAL SOURCES:

ISSUE TITLE: Need for severe accident analysis (AA 5)

ISSUE CLARIFICATION:

This issue is also applicable to NPPs with LWR.

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 maintain subcriticality 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 3, Severe Core Damage Accident Management Measures).

Safety significance

Inadequate severe accident analyses would result in suboptimal accident management procedures.

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:

Argentina

The research in the severe accident area has had an increasing effort in recent years, and in this particular area the international cooperation has been essential. The issue involves training activities, research activities, international agreements.

The research activities in the severe accident area involved the modeling and analysis of severe accident phenomena in Atucha I NPP using STCP and presently MELCOR, SCDAP/RELAP and ICARE2 codes. A FUMO containment model for Atucha I was developed at CNEA and used to model LOCA sequences. The CONTAIN code was used at ARN to model hydrogen behavior, including deflagration and detonation in the Atucha I containment.

Severe accident source terms and likelihood for several containment failure modes were obtained for Atucha I NPP. The hydrogen deflagration/detonation issue was identified as a major contributor to containment early failure in Atucha, and specific studies on mitigation technologies are presently being analyzed (these involve pre and post inertisation, use of igniters and use of recombiners).

The particular configuration of the control rods in Atucha (made of Hafnium) was identified as a potentially relevant issue in aerosol generation, and an experimental series was proposed to study the Hafnium aerosol behavior (STATUS), which is presently under evaluation.

Regarding Embalse NPP, a review on the severe accident scenarios and phenomena involved in CANDU plants is presently being carried out. A CONTAIN model to analyze the hydrogen behavior and the need for mitigation devices is also being developed.

For the CAREM-25 project, several design modifications were suggested from the severe accident and containment response analysis. As part of the design modifications, the use of a totally innovative metallic in-vessel core catcher was suggested and thermal and mechanically analyzed. Besides, a specific project to develop an expert system based on fuzzy logic for Severe Accident Management in CAREM-25 is currently being carried out.

The need for international cooperation was highlighted in the field of nuclear safety research. This is particularly true for countries with a small nuclear program, like Argentina. An important international agreement was signed between the ARN and the US-NRC. This agreement started with CAMP and it was later extended to the Cooperative Severe Accident Research Program (CSARP) and Cooperative Probabilistic Risk Assessment (COOPRA).

From the experimental point of view, several small-scale experiments are proposed to be carried out in Argentina, in material interactions, aerosol generation and behavior, and passive safety features for innovative plants (including severe accidents). Besides, a joint proposal with Sandia National Lab has been presented for DOE evaluation, on an innovative metallic core catcher concept, with large-scale experiments to be performed at Sandia for the CAREM-25 geometry. In summary, the safety research for NPPs is envisaged as a need in Argentina, and the effort should increase rather than decrease in the coming years.

Canada

Although this is currently not labelled as a “generic safety issue” in Canada, the need for severe core damage accident analysis criteria is presently being addressed by CNSC and licensee staff.

The design basis analysis of the CANDU reactor includes the scenario of LOCA plus loss of emergency core cooling that may lead to some core damage, and hence regarded as some type of severe accidents.

Beyond-design-basis severe accident analysis (severe core damage analysis) has been performed for CANDUs since the 1980s by both OPG and AECL, even though it was not a regulatory requirement.

An experimental programme on the behaviour of the CANDU core under severe core damage accidents has been underway at the Whiteshell Laboratories for several years.

A severe core damage accident analysis code – MAAP-4 CANDU – has been adapted for use in PHWRs by OPG and is now being used as an Industry Standard Tool.

India

Multiple failures and rare events scenarios as per (AERB/SG/D-5) have been included in safety reports of current designs. The design code of practice requires such analysis to be done by best estimate approach. Hydrogen behaviour and coolant channel deformation and heat transfer through moderator are important phenomena covered. As is covered in CS-1 and AA-8 specific experimental studies to validate the calculations have not been carried out so far. The calculations are thus based on physics of the phenomena and any published information available. There is a need for exchange of information in this area of research carried by other countries.

Korea, Republic of

This is not a generic safety issue in Korea, the need for severe accident analysis criteria is presently being reviewed by KINS.

Romania

The requirements developed as part of the Strategic Policies for units 1 and 2 is to perform SAM evaluation and to include them:

- for unit 1 in the Periodical safety review process (to start from 2001) after relicensing for operation in May
- for unit 2 as part of the licensing flow already defined.

This topic is considered of highest priority and actions are to be taken so that significant progress is being made.

The Licensee will address the issue of the systematical review of safety correlated also with the periodical safety review process, included in the long term research and development program and in cooperation with COG.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Design Safety Requirements, IAEA Safety Standards Series No. NS-R-1, IAEA, Vienna (2000).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Operation Safety Requirements, IAEA Safety Standards Series No. NS-R-2, IAEA, Vienna (2000).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Implementation of Accident Management Programmes in Nuclear Power Plants, Safety Report Series No. 32, IAEA, Vienna (2004).
- ATOMIC ENERGY REGULATORY BOARD, Safety Guide, “Design Basis Events for Pressurised Heavy Water Reactors”, AERB/SG/D-5 (2000).

ISSUE TITLE: Need for analysis of total loss of AC power (AA 6)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

The complete loss of AC electrical power to the essential and nonessential switchgear buses in a nuclear power plant is usually referred to as a « Station Blackout ». 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. World-wide and for all reactors, there have 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.

There is an issue on two-phase thermosyphoning in PHWRs as follows. Since the primary heat transport pumps will cease to operate in a loss of Class IV power, removal of residual heat relies on natural circulation of the coolant. Although natural circulation with the primary heat transport system full was shown in actual plant tests to be effective, some partial inventory natural circulation experiments done at AECL's Whiteshell RD-14M test facility have shown degraded cooling in some channels. The issue is whether the system thermohydraulic codes used to predict two-phase thermosyphoning behaviour are adequately validated.

Safety significance

The main safety concerns to be considered in case of total loss of power are:

- cooling the core (residual heat removal);
- adequacy of thermosyphoning; and
- injecting water into the PHT system to compensate for leakage

Many plant-specific PSAs show the loss of AC power is an important contributor to core damage frequency.

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:

Argentina

This issue is under revision as a part of PSA activities.

Canada

CNSC staff issued a position statement addressing the generic issue of "Core cooling in the absence of forced flow". The closure criteria for that issue require licensees to demonstrate, that the RD-14M results (referred to in the "description" above) do not invalidate current safety analyses that credit core cooling without forced flow. Alternatively, licensees may review the safety analyses with the new knowledge acquired from the RD-14M experiments and, where necessary, implement adequate design modifications.

This generic action item has been closed for all licensees. One utility implemented design modifications to maintain heavy water feed to the intact loop (the primary heat transport system in some CANDU reactors incorporates two isolable loops) following a loss-of-coolant accident (LOCA). Other utilities have justified the appropriateness of the system thermohydraulic code predictions in this regime.

India

PSA studies conducted in the initial days of inception of nuclear power in India indicated that failures of grid power supply could a dominant event. Therefore very often station will have to depend on its own AC sources. These Station Blackout possibility has been practically considered as design basis event in India. With this in view on-site emergency AC power has been strengthened to increase redundancy, diversity of the location of the diesel etc. Inspite of this arrangement, back up water supply (from plant sources capacity available in diesel driven fire water pumps) has been engineered in all important heat removal paths during Station Blackout. An emergency operating procedure calls for Station Blackout handling procedure details out all sequential actions to be taken by operator if diesel generators do not start for six minutes after the loss of off-site power. The experience of operating plants has however shown that station blackout has happened only once and that was during a severe fire in NAPP-1 Unit. Emergency operating procedures as well as built in back up fire water supply mentioned above is quite handy to successfully provide core cooling in NPPs black out occurred five minutes after reactor trip, under trip condition, NAPS could cope up with about 18 hours of black-out.

Korea, Republic of

KINS is reviewing the GSIs for CANDU plant. What measures should be taken will be decided when this review is over. Moreover, new research programs on CANDU safety issues are being launched.

Romania

The details for this issue are included in ES1 and AA1.

ADDITIONAL SOURCES:

- CNSC Position Statement, 90G02, "Core Cooling in the Absence of Forced Flow (CCAFF)".

ISSUE TITLE: Analysis for pressure tube failure with consequential loss of moderator (AA 7)

ISSUE CLARIFICATION:

Description of issue

This issue is specific to NPPs with PHWR.

In an operating PHWR reactor a rupture of a pressure tube could result in one or more of the following:

a) a LOCA inside or outside the reactor core;
b) a breach of the moderator boundary leading to a loss of moderator heavy water (LOM);
c) damage to reactor systems, structures and components, including adjacent fuel channels, reactivity control mechanisms, the calandria, and ejection of fuel bundles into the calandria and/or the reactor vault.

Tests and analysis have shown that channel failures will not propagate and that damage to in-core components will not prevent either shutdown system #1 nor shutdown system #2 from performing its function. Analyses of such events are presented in the Safety Reports for each plant. However, tests have shown that in circumstances where the calandria tube fails after a pressure tube break, there is a possibility of ejecting the end fitting leading to a drain of moderator. Current Safety Reports do not include scenarios involving a LOCA and a loss of moderator. The issue is relevant only to the dual failure in-core LOCA + LOECC since the moderator is credited as the ultimate heat sink for the reactor.

The absence of the moderator as a backup heat sink, for the failure in-core LOCA + LOECC could lead to a severe core damage accident. Furthermore, the results of fuel channel burst tests conducted by the industry suggest that pressure tube rupture events leading to a large loss of moderator are more probable than previously assumed.

Safety significance

This event is one of the few that that might lead to a severe accident following a simultaneous LOECC.

Source of issue (check as appropriate)

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

MEASURES TAKEN BY MEMBER STATES:

Argentina

This issue is under revision as a part of PSA activities.

Canada

See Issue SS 8

India

In Indian PHWR the impact of the consequences of such event is reduced by inherent process system features existing to mitigate the consequences.

In the scenario where LOCA happens outside the core or pressure tube breaks and calandria tube remains intact, core cooling will be carried out primarily by ECC System. In an adverse situation where ECC does not actuate the moderator acts as heat sink. To ensure cooling of moderator, redundancy is provided in moderator cooling circuit by designing two independent sub-circuits. Further cooling of moderator is backed up by fire water. In case of a breach of the moderator boundary leading to a loss of moderator heavy water a procedure to box up the moderator system is provided.

In case of pressure tube and calandria tube break, the moderator system gets pressurized. A passive over-pressure relief line with rupture disc will relieve the pressure to avoid loading on the core components. Under this situation, it is possible that whole moderator inventory is lost through boiling and core without ECC may overheat and meltdown. However, molten debris is cooled within calandria by large quantity of calandria vault water for couple of days. In a very worst scenario suppression pool is the ultimate back up. The double containment provided can limit the radioactive release under such worst scenario within acceptable doses.

Korea, Republic of

KINS is reviewing the GSIs for CANDU plant. What measures should be taken will be decided when this review is over. Moreover, new research programs on CANDU safety issues are being launched.

Romania

The coverage of this aspect is being considered by CNCAN as to be covered by Licensee in the Periodical safety review process as part of the severe accident analysis and research and development national issues.

The Licensee will address the issue of the systematical review of safety correlated also with the periodical safety review process, included in the long term research and development program and in cooperation with COG..

ADDITIONAL SOURCES:

- CNSC Position Statement 95G02, “Pressure Tube Failure with Consequential Loss of Moderator”.

ISSUE TITLE: Analysis for moderator temperature predictions (AA 8)

ISSUE CLARIFICATION:

Description of issue

This issue is specific to NPPs with PHWR.

During some LOCAs, the integrity of fuel channels depends on the capability of the moderator to be the ultimate heat sink. A channel will likely fail if dry-out on the calandria tube surface occurs. Calculations done to show that pressure tube integrity will be maintained depend on several computer codes. CNSC staff believes that moderator temperatures predicted have not been adequately validated, given the tight safety margins that exist currently.

Safety significance

Insufficient moderator sub-cooling increases the likelihood of channel failure, following a LOCA, due to calandria surface dryout.

Source of issue (check as appropriate)

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

MEASURES TAKEN BY MEMBER STATES:

Argentina

This issue is under revision as a part of PSA activities.

Canada

Three-dimensional experiments are currently being conducted in a special test rig for the purpose of validating the computer codes in question. See also SS 8.

India

See also SS 8.

A research project has been initiated on (slumping og fuel pin and) molten metal falling in the liquid. There is a need for exchange of information in this area of research carried out by other countries.

Korea, Republic of

See SS 7

Romania

Information included in SS8 and modality for coverage similar to AA7.

ADDITIONAL SOURCES:

ISSUE TITLE: Analysis for void reactivity coefficient (AA 9)

ISSUE CLARIFICATION:

Description of issue

This issue is specific to NPPs with PHWR.

CANDU reactors have a positive void coefficient of reactivity. In the postulated event of a large LOCA there is an increase in core power, due to positive void reactivity feedback. CANDU reactors have specific engineered features designed to limit the voiding rate in the core and mitigate the power pulse. Two automatic independent shutdown systems have been designed to quickly insert negative reactivity to offset the positive void reactivity. This engineered design solution is based upon an inherent feature of CANDU heavy water-natural uranium lattice: a long prompt neutron life time. The timing and rundown characteristics of each shutdown system are expected to limit the magnitude and duration of the power pulse, and to ensure that the energy deposition in the fuel will not jeopardize the fuel and fuel channel integrity.

The safety analyses in support of the acceptability of the safety systems' performance to ensure meeting the fuel and fuel channel integrity acceptance criteria are based, to a large extent, on numerical simulations of the power pulse. It is therefore important that safety analyses account for the positive void coefficient of reactivity in a conservative manner. This requires the assessment of the accuracy in determination of this coefficient. For fresh fuel at cold conditions a significant amount of data is available from experiments performed in ZED-2 reactor at Chalk River Laboratory. However, the current validation of the theoretical models and computer codes used by the CANDU industry are such that errors associated with void reactivity calculations are not well defined due to a lack of specific experimental data at in-reactor operating conditions and fuel burnups. Although an allowance for uncertainty is included in the safety analysis, the adequacy of this allowance for power reactor conditions is not fully demonstrated, due to the lack of specific experimental data.

Initially, CNSC staff was dissatisfied with the adequacy of the allowance and requested the licensees to increase the uncertainty allowance and to provide more information on relevant research. The licensees initiated an industry-wide experimental program carried out in the ZED-2 reactor at Chalk River Laboratories.

Subsequent developments, including new experimental results from the experimental program indicated that the interim value of void reactivity error allowance (VREA), which is applied to predictions of the design and licensing code POWDERPUFS-V (PPV), is not appropriate. Furthermore, it has been recognized that PPV significantly under-predicts the void reactivity effect for conditions specific to power reactor and typical average fuel burnup.

There are several areas where, in the view of CNSC staff, specific actions are needed to ensure a high confidence level of results of large LOCA analyses. These areas are:

- the accuracy and validation of current reactor physics licensing methods and computer codes used for power pulse analyses;
- the suitability of the experimental program to support the validation of reactor physics codes and data for conditions specific to power reactors and anticipated accident conditions; and
- the acceptability of results of power pulse calculations performed with more accurate and validated methods, and adequate allowances, in support of safety system performance.

Safety significance

An inadequate uncertainty allowance on void reactivity may lead to higher fuel temperatures, more extensive fuel damage than predicted in the Safety Reports, and more radioactivity releases inside containment.

Source of issue (check as appropriate)

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

MEASURES TAKEN BY MEMBER STATES:

Argentina

This issue is under revision as a part of PSA activities

Canada

To achieve closure, licensees are required to complete a suitable experimental program and related analyses based upon more accurate methods and adequate allowances, and undertake adequate interim measures. The following specific closure criteria should be met:

a) Perform a systematic and comprehensive review and assessment of the various uncertainties and sources of error involved in the large LOCA methodology regarding void reactivity;

b) Provide further supporting evidence for the predictions of the void reactivity coefficient and details of the experimental program and analytical benchmarks. Ensure that the experimental program and analytical benchmarks address the following issues:

    - the effect of operating conditions, such as burnup, coolant purity, moderator poison, moderator purity, fuel temperature, and pressure tube creep;
    - the effect of uncertainties in nuclear data;
    - the effect of uncertainties related to limitations of diffusion approximation, core voiding pattern during a LOCA accident, localized absorbers representation, and fuel burnup distribution;

c) Revise the large LOCA analyses by using more accurate and validated reactor physics methods regarding void reactivity and experimentally-based allowances.

India

For calculating the void reactivity in Indian PHWRs, the lattice code DUMLAC was used in late 1970s. Subsequently the lattice cell/codes CLIMAX and CLUB an integral transport theory code have been developed which used 27 group cross section library collapsed from 69 group WIMS library have been used. These codes have been validated of the experiment conducted in ZED-2 reactor at Chalk River Laboratories in Canada. These codes were used for analysing in-core fuel management benchmarks for PHWRs by International Atomic Energy Agency which have been documented in IAEA-TECDOC-887, June 1996.

The CLUB code is being used presently for the analysis purpose, as it was found to predict the reactivity which is more conservative in the higher burn up region. AERB has reviewed the two codes CLIMAX and CLUB which predict higher void reactivity in high burn up zone.

Korea, Republic of

The uncertainty of the void reactivity coefficient in CANDU reactor is also a problem in Korea. As a Korean regulator, KINS fully agree with the CNSC's position on this matter. But KINS do not have the results of the recent experiments on void reactivity coefficient which were carried out by COG, thus KINS strongly needs the international cooperation on this issue.

The void reactivity coefficient of the CANDU reactor varies from initial fresh core to equilibrium burnt core and also varies with moderator poison and many other variables. Measurements of void reactivity coefficient at various operating conditions and accident conditions are difficult. For those cases in which experiment on the void reactivity coefficient is impossible, use of physics code based on accurate methodology could be considered. Transport method and other general method with multi-group condensation process could be compared to the experiments and to the results from the CANDU specific methodology. Because many variable such as space and time discretization, calculation of the delayed neutron parameters, thermal-neutronic feedback, etc are involved, the detailed review on the accuracy of the power pulse analysis is important.

Romania

Similar to AA 7.

ADDITIONAL SOURCES:

- CNSC Position Statement 95G04, "Positive Void Reactivity — Treatment in Large LOCA Analysis".

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

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to managing engineering design and procurement as an element of an effective nuclear management program in which safety is the highest priority.

There are three scenarios for part replacement:

A) 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.

B) 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.

C) 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:

Argentina

Both CNA-I and CNE nuclear power plants began their commercial operation with a complete spare parts stock. This gave enough time for both nuclear power plants to organise stock strategies and to determine the minimum stock value for critical spare parts. The priority criteria and other factors taken into account for this task were as follow:

Priority criteria:

- Spare parts belonging to safety systems have priority. Although the systems are redundant, the failure of any of the redundancies must be repaired in some hours time, as established in the Policies and Principles Manual. When the expiry date is reached, the plant must be shutdown and driven and maintained in a safe state.
- In second place in order of priority, there are those non redundant spare parts or components the failure of which leads to an out of service state, for instance main pump seals, spare parts of the refuelling system, turbine, etc.
- Other factors:
- Replacement frequency. It is usually based in the lifetime value given by the manufacturer or in the historic value based on the plant experience or that of any other equivalent installation (for instance any other CANDU 600).
- Time required for the arrangement of prevision. It depends on the place where the spare part is manufactured (domestic or foreign), time of transport and customs management.
- Time of delivery. It is determined by the manufacturer according to the number of spare parts, particularly if the supplier has no stock.

Depending on all these data, a minimum stock value is established taking also into account unforeseen delays during the whole process.

Another way of buying spare parts is determined by the programmed outages, for which a specific purchase is carried out of those spare parts which may be required during preventive maintenance of components and systems.

In most of the cases, the purchase of spare parts is done through the nuclear power plant suppliers: Siemens - Kraftwerk Union AG for CNA-I, Atomic Energy of Canada Limited and Societa Italiani Impianti P.A. for CNE. These make the purchase management easier, because these companies have dealers in the country, which is convenient to solve exceptional situations requiring urgent provision of a particular spare part.

In the case of CNE, a new possibility is open through Candu Owners Group, one of its services being the purchase of spare parts for CANDU plants simultaneously. This modality enables an adequate provision of spare parts.

A special treatment is required for those spare parts which have been discontinued in the fabrication of any of their components. In such cases, either a great quantity of them must be bought, or a substitution process must be initiated after analysing the technical specifications and seeking for a new qualified supplier.

It should be emphasised that since the beginning of the commercial operation of both nuclear power plants, no outages or delays in starting the plants up due to lack of spare parts were produced.

Canada

This issue is recognized and actively addressed by Canadian utilities. Programs are in place to manage the supply chain for replacement parts to ensure that acceptable, qualified supply exists for obsolete parts. In some instances, major programs to upgrade obsolete equipment have been undertaken.

India

After mid seventies, when international suppliers refused to execute orders from India, Indegeneous groups were formed both at station and central design office to qualify vendors to meet nuclear grade quality assurance (QA) programme as approved by Department of Atomic Energy. Presently major Indian suppliers for nuclear power programme have their own QA Programme for spares. Moreover the corporate QA group instituted at NPCIL takes care of the QA requirement for spares management Thus in general, to obtain replacement parts for existing units manufacturing facilities with adequate nuclear grade QA program exist in India.

Korea, Republic of

The certification of the new/spare part and the process used to certify the part must be in accordance with all nuclear grade QA programme requirements. The qualify assurance audits are conducted to verify whether all activities affecting quality at every stage of the design, manufacturing, construction, and operation of the nuclear installation are being performed in conformity with quality assurance programme approved by the regulatory body. It is conducted periodically for operating nuclear installations.

Romania

This issue is part of the Strategic Policy for unit 1. The Licensee is assuring the replacement part design, procurement and quality assurance for them, as part of the subprogram of maintaining the spare parts within the original design requirements, which is included in the Management Configuration Control program. The Licensee also has an on-going program for procurement engineering to solve problems related to the obsolence, commercial grade dedication etc. The analysis/replacement process for some components is also going on in this moment.

This issue is intended to be included also in the Periodical Safety Review since 2001. Progress and results will be reported for new revisions of this document.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for Facilities and Activities Safety Requirements, IAEA Safety Standards Series No. GS-R-3, IAEA, Vienna (2006).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Application of the Management System for Facilities and Activities Safety Guide, IAEA Safety Standards Series No. GS-G-3.1, IAEA, Vienna (2006).

ISSUE TITLE: Fitness for duty (MA 2)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to managing people and procedural compliance as elements of an effective nuclear management program in which safety is the highest priority.

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. In additon to medical aspects, psychological aspects like stress management, alerteness, etc. need also to be considered for licensed operators.

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:

Canada

Fitness for duty is an enforced requirement at Canadian nuclear power stations. At one utility the fitness for duty standards have been extended throughout the entire nuclear organization.

India

The recruitment, training, qualification/requalification procedure for control room personnel has been carefully drawn up and standardised. The licensing/ relicensing of control room personnel is done by the regulatory body. For control panel operation graduate engineers (Control engineers) are working in shift operation. Medical examination prior to the recruitment is done. To take up the position of Assistant shift charge engineer and shift charge engineer, persons must work oncontrol panels as control engineer for sufficient number of years. This has enhanced the quality and expertise of man power for control room operation and decision making ability.

Annual medical examinations ensure that basic requirements for operating staff like alertness, capacity to withstand stress, eyesight (especially colour blindness), general physical & mental health etc. are maintained at a satisfactory level. Medical examinations are done at higher frequency for personnel above the age of 45 years.

Instructions exists which restrict any persons performing critical activities to be retained for more than 16 hours and no body will be detained for overtime after night shift. This ensure that their mental and physical capacity is not effected . Any licence persons on leave or absence for more than three months is require to undergo a standard familiarisation programme.

Dedicated transport arrangement and housing near plant area with preferential allotment has been provided for shift personnel.

Korea, Republic of

This is not considered to be a generic safety issue in Korea.

Romania

Specific qualifications are required (including psychological) for the operator, shift supervisors licenced by regulatory body for operation in Cernavoda NPP. A possible approach on trying to license the shift, as a whole to confirm their compatibility is also being considered for the near future by CNCAN.

The Licensee defines specific requirements to maintain the fitness for duty of the minimum and normal complement for the operator personnel, consisting on:

- Psychological and medical examination in accordance with the national labor legislation
- Controls in the plant to monitor the operating staff from this point of view,too.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for Facilities and Activities Safety Requirements, Safety Standards Series No. GS-R-3, IAEA, Vienna (2006).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Application of the Management System for Facilities and Activities Safety Guide, Safety Standards Series No. GS-G-3.1, IAEA, Vienna (2006).

ISSUE TITLE: Adequacy of shift staffing (MA 3)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to the conduct of operations as an element of an effective nuclear management program in which safety is the highest priority.

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:

Canada

The minimum shift complement is one of the license conditions for each station in Canada.

India

The minimum staff compliance for shift operation is structured to handle normal as well as unexpected events. All field operators are rotated to different areas of the plant and hence the operators are well trained to operate all the systems of the station. This helps in manning specific areas by withdrawing operates from other areas in case of need.

Procedures made for shift turn over ensure retention of adequate staff. The fifth crew discusses all events in all shifts on their supernumerary day. Also operational feedback experience forms an essential input for requalification/relicencing of operators by AERB which is done every three years in India. If somebody joins after a lay-off greater than three months, he must go through familiarisation before taking responsibility. It is also ensured that there are at least two persons with familiarisation in a) Fire fighting, b) First aid, c) Emergency preparedness plans and procedures and one person with some basic familiarity to Fuel handling systems and its interfaces to other plant systems in shift crew. The Township of all the Nuclear Power Stations are located near by the plant site and additional staff could reach the plant site within 30 minutes, whenever needed.

Off site emergency procedure has been standardised and off site emergency manual is prepared jointly by utility and local civil authorities. This has helped in establishing better and clear understanding between utilities and local authority for Off site emergency. Regular emergency drills are conducted and some time local peoples are also involved for better understanding of the emergency procedures and preparedness.

- Standard procedure exists for notification during ON Site and OFF emergency condition. Round the clock coverage of telecom operator was instituted.
- Elaborate EOP are available.

Korea, Republic of

This is not considered to be a generic safety issue in Korea. The minimum shift complement is one of the license conditions for each station.

Romania

The shift is defined by a document of type “A” on minimum shift complement. Due to the CANDU specific this minimal shift is not defined by the bigger (by risk impact) incidents/accidents, but by those requiring more manual actions, i.e. loss of instrument air. Any change to this complement minimal leads to automatic plant shutdown. It is a license limit.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Human Resource Issues Related to an Expanding Nuclear Power Programme, IAEA-TECDOC-1501, IAEA, Vienna (2006).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Human Performance Improvement in Organizations: Potential Application for the Nuclear Industry, IAEA-TECDOC-1479, IAEA, Vienna (2005).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Managing Human Resources in the Nuclear Power Industry: Lessons Learned, IAEA-TECDOC-1364, IAEA, Vienna (2003).

ISSUE TITLE: Control of outage activities to minimize risk (MA 4)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to outage management as an element of an effective nuclear management program in which safety is the highest priority.

Plant operating experience has shown that reactors are susceptible to a variety of abnormal events during shutdown conditions.

Specific attention is required in PHWR's to controlling the guaranteed Shutdown State (GSS) to maintain the reactor critical during an outage. Additionally, attention to outage heat sink management is required since fuel is not removed from the reactor during normal outages.

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.

Many utilities perform safety assessments of outage schedules as a routine part of outage planning. 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.

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;
- Inappropriate returning to service and specially check procedures;
- Spread of contamination;
- Unplanned exposures.

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:

Argentina

The NPPs should have, by operating procedures, at all times the reactor sub critically and at least two redundant ways for core heat removal, the procedures include power supply and minimal configuration of other systems. A PSA for shutdown state is being performed for one NPP and scheduled for the other. Therefore it is envisaged a cuantitative control risk of the plant during outage schedules at planing level will be established. Additional aspects are considered related to regulatory considerations, outage plant management and organization, maintenance and operations activities and ALARA plans.

Canada

Control of risk during outages is an integral part of outage management. This is addressed through a combination of detailed outage planning, which includes identification of potential nuclear and conventional safety hazards, identification of acceptable shutdown operating states, and control of activities, including strict procedural compliance, during the outage. The basic elements of nuclear safety involving control, cool and contain are adhered to in outages , just as they are adhered to during power operation.

India

Indian PHR stations has detailed Technical Specification clauses for Reactor shutdown state to ensure that adequate sub-criticality margin and core cooling is maintained at all time during outage activities. Shutdown system (SD#1) is so designed that all the shut off rods remains fully IN in the Core during shutdown state with moderator system injected with sufficient quantity of boron poison to ensure enough sub-critical margine. Shutdown system # 2 remains in poised state. For taking maintenance activity in any of the PSS rod, one rod is only permitted to be withdrawn at a time. Similarly for SD#2 system, one out of 3 channels is only allowed to be taken for maintenance. The control absorber rods (Cobalt Rods) and shim rods (SS rods) are interlocked such that, one at a time only can be taken for maintenance. These measures, although causing some constrains in outage management ensures guaranteed shut down state. The operating logic is so built that on Reactor trip, the boron removal capacity by ion exchange column is taken out of service and only boron saturated IX columns (only 2 Nos.) can be put in to service. Routine boron sampling even during Reactor in shutdown state ensure boron concentration in Moderator system. Guaranted shutdown state (GSS) is spelt out for PHWRs and included in AERB safety guide for physics AERB/SG/D-7 "core reactivity control".

Heat Transfer System or Moderator System shutdown, if needed for maintenance purpose can only be taken after specified elapsed time after reactor shutdown. The various conditions for such shutdown are defined in Technical Specification. Also Station Instruction exists which does not allow to take up such maintenance jobs during moderator system shutdown in station's where part of ECCS is from

Moderator System. Detailed PSA has been carried out for the unlikely situation when both the shutdown cooling system become unavailable and analysis proved that under such scenario, Reactor Core Cooling shall be maintained by thermosyphening. Special procedure was prepared for such situation giving sequence of isolation and time duration of shutdown. On very long shutdowns header draining is some times resorted to and specially approved AERB procedure is followed.

New Technical Specification is under preparation for units under large scale replacement of Coolant Channels as it is much different from normal outages.

To take care of large number of tasks to be co-ordinated during shutdown maintenance, special maintenance shift including maintenance engineers of various decipline comes in round the clock shift in addition to normal shift crew. This helps in carrying out planned maintenance and avoid incomplete or erroneous activities and unnecessary safety equipment unavailability.

Standard procedures and OTOs (Order to Operate) have been prepared and being followed to carryout specific jobs such as Header Level Control, Isolation and Outages of electrical systems, enhancing safety system availability by “gagging” providing temporary electrical, air or water supply etc. These procedures and OTOs are continuously updated based on operation & maintenance experience. Special regulatory inspections are conducted by AERB inspector during plant maintenance shut down.

Korea, Republic of

This is not considered to be a generic safety issue in Korea.

Romania

Information already included in AA4 and RC1.

The Licensee programs for outages have some important supplementary, not mentioned above, features, which will be mentioned bellow:

- There are organizational controlled programs for outages
- The activities are being performed based on the work plan evaluated by interdiciplinary teams (System Engineers, Engineering, Operation, Nuclear Safety etc.)
- Level 1 flowcharts for outages, which include the configuration of the heat sinks during the period are approved by the Nuclear safety department and the Station Manager
- The daily heat sink configurations are controlled by procedures

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Management Strategies for Nuclear Power Plant Outages, Technical Reports Series No. 449, IAEA, Vienna (2007).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Indicators for Management of Planned Outages in Nuclear Power Plants, IAEA-TECDOC-1490, IAEA, Vienna (2006).

ISSUE TITLE: Degraded and non-conforming conditions and operability determinations (MA 5)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to the conduct of engineering and operability evaluation as elements of an effective nuclear management program in which safety is the highest priority.

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 Operating Policies and Principles (OP&P) or Technical Specifications (T/S), the Allowed Outage Times contained in the OP&P or 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 OP&P or 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.

Operability tests should be drawn-up comprehensively and with adequate frequency. Good return-to-service procedures are important. In many occasions some safety systems remained unavailable, unnoticed in a latent manner.

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:

Argentina

The Polices & Principles Manual includes some generic allowed outage times (AOTs) which were not determined by using a probabilistic methodology. Therefore, the PSA will be used to optimise the AOTs values.

Canada

Some Canadian utilities have initiated a project labelled "Safe Operating Envelope (SOE)" in order to rectify several perceived deficiencies related to:

a) fragmentation of the safety analysis basis on which the stations were licensed,
b) demonstration that all safety analysis limits, credits and assumptions have been systematically extracted from the safety analysis and documented on a system-by-system basis,
c) proper incorporation of safety analysis limits and assumptions into the operating documentation, and
d) ensuring that a change control process for keeping the SOE up to date with respect to the analysis-of-record exists.

They also instituted a formal process for Technical Operability Evaluation (TOE) which is embodied in a procedure. This requires timely evaluation and justification of continued operation when an adverse condition is identified as having a potential operability concern. If continued operation cannot be justified then conservative decision-making requires that the unit or units be shutdown.

Additionally, all utilities are currently updating their stations' respective safety reports.

India

Based on routine test results, reliability analysis of each train of safety systems and equipment are carried out regularly and are reported in station monthly performance report. Procedure exists and are followed to change the test frequency based on these reliability analyses in Indian PHWRs after obtaining AERB's approval.

PSA and reliability study of systems and componenets are updated periodically by the utilities. AERB reviews these study results while granting permission for operation of Unit once in 5 years (Application for Renewal of Authorization , ARA Permission for operating a station after review of ARA is subjected to reliability values of plant systems and components meeting the design intents and safety analysis requirements. Living PSA programme in each of the Utilities is planned for implementation by the utilities.

Non-compliance of any of the clause of Technical. Specifications. Clauses is classified as SER (Significant Event Report) and continuation of the unit operation is permitted only after review and analysis by AERB. Each of the SERs are reviewed by AERB and stipulations are given based on analysis, test results, operating experience, engineering judgement or a combination of these factors taking into consideration equipment functional requirements.

Operability tests need to be comprehensive, all encompassing with proper "alignment" (as would be when called for), with adequate periodicity etc. to ensure timely uncovering of latent non-availability. A well drawn out walk through, both in the field and control room can uncover many non-conformities. Design provisions like hand switch error, ready to start indications for safety equipment like Diesels, Emergency Boiler feed pump etc. have been incorporated.

Korea, Republic of

In order to maintain a consistency in operational requirements imposed on different reactor types, Korean regulatory authority required the utility (KEPCO) to comply with the technical specifications format stipulated by MOST notice 83-3.

Now PWR type of technical specifications format is to be applied to CANDU reactors in Korea. So Wolsong units 2,3 & 4 have already complied with this requirement and Wolsong unit 1 is implementing the replacement of its OP&P with PWR type technical specifications.

KEPCO provided a revised technical specifications of Wolsong units 2,3 & 4 at the end of last year to make clear and reflect operating experiences.

Romania

The Regulatory Body licenses not only the safety envelope defined by the FSAR, but also the operating envelope. The operating envelope is defined by Operating Policies and Principles for Cernavoad NPP and it includes:

- extended appendices defining detailed requirements for a list of systems considered of safety significance as per the System Classification List
- generic provisions in the main text, making reference to a series of other type "A" documents, which are also part of the operating envelope definition (like for instance the Imparement manual)
- it operates also in a given procedural environment as part of the overall QA program, approved by the Regulatory Body

Any derogations, deviations to these documents are subject to prior CNCAN approval before implementation and internal License procedures are in place to check the continuous compliance with the operating envelope requirements. There is an existing system for approving deviations, which includes request for Station Manager (and CNCAN) Approval - RSMA, jumper record system etc. dealing with the systematic control of the deviations.

CNCAN required and Licensee also developed Technical Specifications for Cernavoda NPP, for which the phase of checking against the commissioning results and as-built status is a pending issue. They were however used only for information during the licensing process in the last commissioning and test operation phases and the decision on their use for unit 1 and/or 2 is to be taken after their review.

ADDITIONAL SOURCES:

ISSUE TITLE: Configuration management of modifications (MA 6)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to managing engineering design and change control as elements of an effective nuclear management program in which safety is the highest priority.

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 some safety systems cannot be tested in totality under simulated accident conditions, 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:

Argentina

The Operating licence requires that "All modification that affect safety must be approved by the regulatory authority ". More over, all plants has procedures that imply that all major design change, o a change that affect safety, the original designer of the plant is consulting about it, in some way, to assure that the original design bases of the plant are improved or at list not spoil.

Canada

The importance of maintaining configuration management throughout the Life cycle of a station is recognized. There have been some instances where configuration management has been lost and the consequent extreme impact on the utilities has significantly raised awareness of the importance within the organizations. Significant effort is being expended at one utility to restore configuration management and ensure that programs for design management, engineering change control and procurement engineering are well integrated to meet engineering policy regarding control of design activities.

India

Head quarter Instruction (HQI) on procedure to carry out modifications exist in PHWR stations. These HQI defines the procedure separately for design changes during construction and operating stages. All modifications that affect safety related system must be approved by the regulatory authority. HQI defines the various steps required to process a design change. This includes review by system design engineer, design intents as specified in safety analysis report are met, approval and implementation procedure, updating of design, operating manuals, flow sheets and EDs, safety reports etc. One set of marked up master copy of flow sheets, EDs and operating instruction showing change incorporated, is kept in main control room till all the changes in all the concerned documentations are implemented. Periodic checking by regulatory authority is done to ensure that the changes are incorporated in the various documents within stipulated time.

Procedure for temporary modification and "Jumpering" of logic exists in all utilities. In most of the cases temporary modifications/ Jumpering is done during shutdown to fulfill conditions to carryout tests etc. Procedure exists to put a Jumper, and its removal. These jumpers are checked by station management prior to unit start up and reactor criticality to ensure no jumper is left which may result in an unsafe condition.

"Technical bulletins" are issued after modification and these ensure configuration control.

Korea, Republic of

Korea Atomic Energy Law requires that all modifications which affect safety must be approved by the regulatory authority.

Romania

The plant configuration control is connected in the licensing process for Cernavoda with the fulfilling of the Strategic policy requirements. As part of this process a trial period up to next year was approved for a new set of type "A" procedures for the plant configuration control. These issues are correlated in Romanian situation with the clear definition of the Design Authority after the AAC (consortium from AECL-Ansaldo) left the country and of the design reference. These requirements are similar for unit 2, too.

The Licensee developed a management configuration control program at the level of INPO/IAEA requirements in the field. The procedures passed successfully an IAEA review. Based on this system the modifications in the plant are discouraged as a policy and the safety significant changes are to be approved by CNCAN.

The progress on these issues will be reported after the results from the actions going on will be reviewed.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Configuration Management in Nuclear Power Plants, IAEA-TECDOC-1335, IAEA, Vienna (2003).
- Nuclear Power Corporation of India Limited, Head Quarter Instructions No. 5012 & 304.10 on procedures to carry out Engineering changes during construction and operation stages (India).

ISSUE TITLE: Human and organizational factors in root cause analysis (MA 7)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to managing performance through a corrective action program as an element of an effective nuclear management program in which safety is the highest priority.

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 details required to identify management and organizational problems.

It is also important to have a blame-free culture in the organization to ensure that all facts and observations are available and are input into the root case analysis.

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:

Canada

Human and organizational factors are systematically taken into consideration by licensees and the regulatory body when analyzing significant operating events. Dedicated “Human factors” units exist within each organization. Additionally, one utility has established a Corrective Action Program with associated procedural controls on identifying, classifying and reporting adverse conditions with associated formal processes to review the reports, develop plans and assign tracked corrective actions. Root cause analysis is an integral element of this program, staff are trained to perform it and it is routinely conducted.

India

During early operation of PHWR station, it was realised that root cause to identify human error is a most important and difficult job. While operators were willing to share the cause of error, the fear of punishment was holding them back often. While this problem is still not fully solved, a good sanction and reward system can improve things greatly. Following additional measures were taken to identify root cause of human error related events and reduce human errors.

It was observed that chance of human error is often there when some non-relating operation is being done during end of the shift. Understanding was established not to take up any non-routine jobs during end of the shift operation to the extent possible.

Punishing and fault finding attitude was discarded. Confident was built up that punishment or blame is not the aim but it is important to identify the root cause to prevent re-occurrences. A blame-free culture is encouraged and mistakes are seen as a stepping stone for drawing lessons and avoiding events.

The root cause was carried out by calling the crew, in whose shift the incident has occurred, in their off-time, so that the analysis could be carried out before some facts are forgotten and the hidden weeknesses missed.

Several courses have been held on root cause analysis methodology and on "ASSET" review through IAEA. The detail to which root cause need to be analysed is to be dictated by details of corrective action proposed.

AERB is separately conducting analysis of all SRUORs (reportable events) to draw generic lessons, rating of each unit, etc.

On few occasions human error has contributed due to design inadequacy. In one of the PHWR stations, control room operator, tripped the reactor by pushing the ganged trip push button while intending to raise reactor power, a similar ganged switch. The root cause was due to mirror image design philosophy for control room H.Ss and the control room operator was working in other unit's control panel (twin unit station having same control room) before he was shifted to the affected unit.

Training on safety culture, STAR (STOP, THINK, ACT and REVIEW) were made integral part for licensing qualification.

Questioning attitude, rigorous and prudent approach in operation and maintenance group was encouraged.

Use of simulator for training and licensing of operation personnel has been made compulsory. Simulator for RAPS-1 type reactor was built and commissioned in 1985. For standardised Indian PHWR design (220 MWe) simulator is installed and commissioned in KGS-1&2, RAPS-3&4 and TAPS-3&4 (540 MWe). Similar simulators in other stations are also planned. This will help in minimising human errors and improve quality of operation staff. For new proposed units, AERB has stipulated that installation and commissioning of simulator is mandatory prior to criticality. The system of information exchange between similar stations was established. Operation and maintenance unit personnels visit other stations periodically and share the experience of each other. This has resulted in positive improvement in root cause analysis, safe and reliable operations and avoid repetition of similar incidents.

Lesson learnt and feed back from other similar stations was included in the requalification programme for licensed staff.

IRS and other international reports are discussed in each station and appropriate actions were taken.

Korea, Republic of

1) The major issues in a view of human factors engineering

CANDU has used human engineering related requirements, which are different from or in conflict with practice in PWR's. Therefore, there is a need to review carefully the application of the related requirements and to take action to correct when KINS inspects the Wolsong site.

1. Inappropriate Character Organization on Label
   To improve readability by the operator, character organization on labels on control panels shal be classified into system, subsystem, component, instrument ID, etc, and shall have a hierarchical organization and size configuration, per each designation. CANDU, however, does not incorporate such requirements. Although character arrangement shall have a hierarchy organization, this requirement is ignored but instrument ID has priority.

2. Deficiency of Consistency in Label Sized and Contents
   The label applied in Wolsong 2 has been installed without a fundamental rule on its size and only in consideration of space on a control panel.

3. Label Position and Deficiency of Consistency
   Labels shall be positioned above corresponding instruments in order to always show the meaning of an instrument. In Wolsong 2, because labels are positioned below corresponding instruments, they are hidden by the operator's arm or hand when the operator manipulates various controllers. The label used for a system title is positioned above the instrument group. As previously mentioned, labels have been positioned on the above or the below of instruments. Especially, there may be confusion because labels are arranged without consistency.

4. Inappropriate Configuration of Instruments
   Instruments with round type and transparent glass have been used in the past. The instruments, however, are not used nowadays because of this defect in viewing according to human engineering. The instruments have a problem that a part of their round type glass always makes a reflection by lighting on ceiling. However, instruments with round type glass have been used in a matrix arrangement in Wolsong 2.

   Nowadays, the slope of a control panel is decided in a view of human engineering and in consideration of the reflection of various kinds of instruments.

   NRC of USA had already pointed out the above for PWR. PWR in Korea had replaced round type instruments with digital instruments to improve the above problem.

5. The Inappropriate Thickness of Mimic Flow Line
   Mimic flow lines are used to help operators, showing the flow of system and the relation of instruments. However, mimic flow lines on CANDU control panels are inappropriate thick and occupy much space. It may cause confusion in recognizing a label with similar thickness or size.

6. The Inappropriate Consideration of Parallax
   Although important and frequently used instruments shall be positioned within the range of operators' vision, many instruments are installed on the upper board of control panels in Wolsong 2. Therefore, it is expected that parallax problems will often occur when operators read instruments.

7. The Inappropriate Division of Instrument Scale

Although instrument scale makes it a principle to divide into 0, 1, 2, 3, , 0, 2, 4, 6, , 0, 5, 10, 15, 0, 10, 100, 1000, , etc., the scale of some instruments has been divided into 0, 3, 6, 9, without following the principle of human engineering.

8. The Abuse of Colour Coding
   The purpose of colour coding is for operators to identify easily. If similar colours are abused, it may obstruct the above purpose and cause confusion. The number of colours is confined within eleven(11) in the principle of human engineering.

9. The Inappropriate Configuration of Selection Switches
   Some selection switches have inappropriate configurations that run counter to the rules of human engineering. Some switches are unsuitably arranged and cannot identify what the selection switches imply.

10. The Inappropriate Arrangement of Status Lightings
    Status lightings shall be put on the upper right of related instruments. However, some instruments did not follow the principle. In addition, instruments are close together or the arrangement of instruments is inappropriate and so there is a high probability to cause confusion if the status lightings of instruments are too close. It is easy for Operators to think that a status light is generally positioned near its related instrument.

Romania

CNCAN developed as part of the new regulatory pyramid, agreed in its main features and regulations by EU representatives as part of the PHARE Phase I project, a guideline for the requirements implementing the IAEA recommendations on the Self Assessment, including the human factors aspects. Details are expected to evolve however in a different CNCAN guideline. These documents are under internal CNCAN review, but their principles are part of the Strategic Policy for unit 1 relevant activities and to be included in the Periodical Safety Review process. The root causes analyses are however already implemented as part of a set of requirements from the license and the Strategic Policy for Unit 1, in type “A” procedure of the plant and their extension to the low-level events was made based on the recommendations of IAEA and WANO missions.. The Licensee is using specific human factors analysis tools, too. The procedures are compliant with the CNCAN reporting requirements and define, together with the

CNCAN internal process for event reviews (including CNCAN specific database), the national feedback from operation process. This process was subject of an IAEA mission of OSEF experience in 2000. The recommendations and suggestions are under implementations and results will be reported, too.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Legal and Governmental Infrastructure for Nuclear, Radiation, Radioactive Waste and Transport Safety Requirements, IAEA Safety Standards Series No. GS-R-1, IAEA, Vienna (2000).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Operation Requirements, IAEA Safety Standards Series No. NS-R-2, IAEA, Vienna (2000).
- INTERNATIONAL ATOMIC ENERGY AGENCY, A System for the Feedback of Experience from Events in Nuclear Installations Safety Guide, IAEA Safety Standards Series No. NS-G-2.11, IAEA, Vienna (2006).
- IAEA/NEA Incident Reporting System (IRS), Reporting Guidelines (1998).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Human Resource Issues Related to an Expanding Nuclear Power Programme, IAEA-TECDOC-1501 (2006).

- INTERNATIONAL ATOMIC ENERGY AGENCY, Human Performance Improvement in Organizations: Potential Application for the Nuclear Industry, IAEA-TECDOC-1479, IAEA, Vienna (2005).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Managing Human Resources in the Nuclear Power Industry: Lessons Learned, IAEA-TECDOC-1364, IAEA, Vienna (2003).
- INPO Good Practice OE-907. Root cause analysis.
- Program Description INPO 90-005, "Human performance enhancement system".

ISSUE TITLE: Impact of human factors on the safe operation of nuclear power plants (MA 8)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to work management in maintenance and plant status control in the conduct of operations elements of an effective nuclear management program in which safety is the highest priority.

Human factors are significant in many of the operational events at operating nuclear power plants.

Examples of 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;
– 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
- ____xx_____ potential weakness identified by deterministic or probabilistic (PSA) analyses

MEASURES TAKEN BY MEMBER STATES:

Argentina

The detection, correction and prevention of human errors are carried out through two clearly distinguished processes: the incident analysis and the global and systematic study of the installation safety. In the first case, the process acts on the abnormal or unexpected events corresponding to actual situations that happen in the installations (operational experience). Such events are unique opportunities to detect and correct human errors, identifying the imperfections regarding organisation, persons, materials and practices. In this case the key elements are the quality of the report on the event occurred, the rigour in the investigation about its deepest causes and the corrective actions carried out.

In the second case the probabilistic safety assessment technique is used, part of which consists in the identification of human actions and the evaluation of their relative importance on the installation safety. The errors can be classified in pre-accidental (errors occurred during periodic tests or maintenance) and post -accidental (errors occurred in accidental situations).

These pre-accidental and post-accidental errors are analysed in the same way as the behaviour of components, equipment and systems, but using human reliability analysis techniques. Those evaluations are part of the probabilistic safety analysis and their results enable the definition of those areas requiring improvements on both the operation procedures and the man-machine interface, as well as the identification of cases in which the operators' training and retraining shall be intensified.

It should be pointed out that in both nuclear power plants, the periodic training of the operation personnel in full scope simulators and other simulators existing at the installations, constitutes an additional means to detect, correct and prevent human errors in accidental situations, contributing, in addition, to the improvement of operation procedures.

On the other hand, standards AR 3.2.1 and AR 3.4.1 establish the information the operator should count with in order to take safety related decisions, the prohibition of interventions during the period immediately after the occurrence of accident initiating events and the characteristics of the man-machine interaction related to the design of the reactor instrumentation and protection systems .

The proper policies and management of the operating organisation are the basic support to obtain the expected results regarding the anticipation of undesirable events that may happen.

Once such events have happened, the installation Primary Responsible, supported by the Responsible Organisation, determines the responsibility degree, if any, of persons who may have incurred in errors and applies the corrective measures and, if corresponds, the applicable sanctions.

On the other hand, having the Regulatory Body analysed the event, issues requirements and, if it is the case, applies the corresponding sanctions to the Responsible Organisation, the Primary Responsible and the involved personnel.

During the safety inspection and evaluation process of the nuclear power plant, the Regulatory Body pays special attention to find early signals and trends such as:

- Weaknesses in the safety policies.
- Weaknesses in accident analyses.
- Procedure violation.
- Operator errors.
- Deficient training.
- Deficiencies in the use of operational experience.
- Weaknesses in emergency planning.

The aim of the human reliability assessment is to improve the plant global safety, identifying deficiencies in the operator actions and providing whatever needed to analyse and perform possible corrective actions.

The probabilistic safety assessments showed, through human reliability analysis application, that it was necessary to carry out modifications to the installation enabling the operator to be more reliable to take countermeasures, to make improvements regarding abnormal operating procedures and re-training the operating personnel on certain analysed accidental sequences, where the human actions play an important role related to safety.

The data used in human reliability models depend explicitly on the applied model and come from operational experience, generic data and practices in foreign simulators of compatible plants, since in the country there are no nuclear power plant full scope simulators. Specifically, the human reliability analysis carried out for CNA-I probabilistic safety assessment was based on generic data for the human error failure rate, from factors, recovery and uncertainty factors. CNA-I operational experience provided task execution times, actuation frequency for components and equipment, and equipment recovery times.

For the case of CNE probabilistic safety assessment, the feasibility of incorporating information coming from periodic practices CNE operating staff is carrying out at Gentily's full scope simulator, was considered.

Canada

Human factors issues associated with plant operation are addressed through development of clear governance within a program that defines the conduct of operations. Standards are defined for conduct of operations which define the expectations on staff. Procedures have developed for operator activities in both normal and abnormal events, with specific evaluation of human factors and their verification through simulator-based training.

India

Considerable efforts have been put to systematically analyse human performance, ergonomics, etc. and effect improvements. Initial design of PHWR station, mainly RAPS#1, was such that requirement of human intervention was large. Human actions had resulted in events and unit outages and also given unique opportunities to detect and correct human error and identification of design inadequacy. The events were systematically investigated and corrective actions were taken. For example in the early PHWR design, to curtail Heavy Water inventory, the boilers draining was an essential operator's action on Reactor trip, to maintain system inventory for Core Cooling. Similarly boilers filling was an essential part of operator's action during system heating up.

The need for boiler draining and filling was eliminated in subsequent units by providing additional D2O inventory of primary Heat Transport System (PHT). Many design changes were incorporated in subsequent reactors to achieve the intents of no operator's intervention for initial 30 minutes following an event or accident as given in AERB guide on operation.

With the introduction of PLC and computer based control, safety and data acquisition systems in the standard design plant, introduction of graduate engineers to operate control panel was adopted. This has resulted in better man-machine interface.

Importance of effective training, qualification and licensing to reduce the impact of human factor on safe operation of Nuclear Power Plant was effectively demonstrated during Fire incident in NAPS unit # 1. EOP for station black out was used and reactor was tripped manually and manual crash cool down was initiated by Control Room Staff. This has resulted in excellent handling of station black out (SBO). SBO handling procedure was modified based on NAPS experience.

Use of probabilistic safety assessment technique for human reliability analysis using human reliability models was long felt. AERB constituted a Working Group where performance of Human related issues were analysed using this method. The above methodology is being used regularly and their findings are utilised for areas needed improvements on both the operation procedures, design and man-machine interface. Also it is used regularly to identify cases in which the operators' training and retraining needs enhancement.

During the AERB regulatory inspection and review of incidents, special emphasis is given to find areas of

- Weakness in the safety policies
- Weakness in accident analysis
- Procedure violation
- Operator error
- Deficient in training
- Deficiency in the case of procedures and experience
- Weakness in emergency planning
- Assessment of safety culture in organisation and individuals

Korea, Republic of

See issue MA 7

Romania

Information is already included in and correlated with that one presented in MA7

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Human Resource Issues Related to an Expanding Nuclear Power Programme, IAEA-TECDOC-1501, IAEA, Vienna (2006).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Human Performance Improvement in Organizations: Potential Application for the Nuclear Industry, IAEA-TECDOC-1479, IAEA, Vienna (2005).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Managing Human Resources in the Nuclear Power Industry: Lessons Learned, IAEA-TECDOC-1364, IAEA, Vienna (2003).

ISSUE TITLE: Effectiveness of quality management programmes (MA 9)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to imbedding quality within an effective nuclear management program in which safety is the highest priority.

Effective quality management 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 management programmes must be implemented for all safety activities. These programmes are increasingly 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:

Canada

Quality assurance programs are required for all nuclear operating organizations in Canada to meet requirements of the relevant standards produced by the Candian Standards Assocation (CSA) and to meet regulatory requirements. These quality assurance programs requirements cover all aspects relating to nuclear safety. To ensure that quality assurance is effective, the operating organizations have internal performance assurance functions that are responsible for evaluating, assessing and auditing work. Furthermore, the regulator performs independent evaluations and audits of licensees to ensure that quality programs are effective. Deficiencies found as a result of either internal or external evaluations, assessments and audits result in actions being raised with specific time periods for completion of the actions, and full and open tracking of the actions.

India

Effectiveness of quality assurance and total Quality Management (one of its principles says doing things right the first time) are essential inputs for safe and reliable performance. A topical QA document issued as Headquarters Instructions (HQI) serves as a parent document to individual station QA documents. Indian experience indicates that Quality Assurance needs to be built into business practices, systems, procedures, work practices, people rather than "doctored" through a large Quality Assurance Section (No body likes to be told how he should do his job!). Quality culture needs to be a corporate value and AERB encourages the above approach and the same is observed to have yielded good results. On the basis of above approach, development of (a) organisations both at Headquarters, at each NPPs and also at AERB, (b) procedures, (c) systems, (d) inspections, (e) training, (f) motivations for adherence etc. have been completed and put into practice. Priciples being brought out in IAEA guide in management system are being incorporated in NPCIL and AERB documents.

Korea, Republic of

See issue MA 1

Pakistan

QA represents a management control system that the nuclear plant should establish and use in attaining safety objectives for nuclear installations. Like any management control system, QA require an organizational structure with defined responsibilities and functions, documented programs, established goals and objectives and prescribed procedures for performance evaluation. Failure of management to establish and implement QA effectively as a management control system may cause major problems in the quality of design, manufacture, construction projects and operation of nuclear power plants.

At KANUPP, an Operational Quality Assurance program was established during mid eighties and a program manual was prepared in line with Canadian Standards on “Operations Quality Assurance for Nuclear Power Plant” national regulations and guidelines. The Quality Assurance Division conducts Comprehensive QA audits of Operation, Engineering Support, Chemistry Control, Health Physics, Maintenance, Procurement, Material Management and Training. Routine inspections of all the field activities related to areas mentioned in QA manual are also conducted. The work to be done by the work groups is reviewed from QA point of view before it is started, and during and after completion of work, to verify that QA requirements in this regard are fulfilled.

A non-conformance reporting program was also established and implemented. QA verification of all important plant documents such as Operation and Maintenance Procedures, Station Instructions (SIs), Change Approvals (CAs), etc. is also carried out. The Quality Assurance team ensures that the workgroups clearly understand management expectations to establish and maintain quality culture at KANUPP.

The Quality Assurance Division essentially makes sure that the work groups clearly understand and meet management expectation to establish and maintain quality culture at KANUPP. Operations QA program has been established. In line with PNRA regulations, the program is being developed to cover other important areas of the plant.

Romania

The generic framework of the Licensee polices are in accordance with the Quality Assurance Manual and the License has as a target to implement a self assessment program to supplement the already in place system defined by those policies and practices (audits etc.). The corrective actions are being monitored as part of the feedback from operation process.

There is also defined in internal draft a regulatory guideline for the self assessment, which is scheduled to be issued to the Licensee in 2001, as part of the detailed definition of the main Periodical Safety Review process regulatory requirements.

This issue is part of the Strategic Policy for unit 1 and to be included in the Periodical safety review since 2001. Progress and results will be reported.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for Facilities and Activities Safety Requirements, IAEA Safety Standards Series No. GS-R-3, IAEA, Vienna (2006).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Application of the Management System for Facilities and Activities Safety Guide, IAEA Safety Standards Series No. GS-G-3.1, IAEA, Vienna (2006).
- INTERNATIONAL NUCLEAR SAFETY ADVISORY GROUP, Basic Safety Principles for Nuclear Power plants, 75-INSAG-3 Rev. 1(1999).

ISSUE TITLE: Adequacy of procedures and their use (MA 10)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to managing performance of worker practices as an element of an effective nuclear management program in which safety is the highest priority.

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.

Several plants have established clear and comprehensive procedures to produce, validate, approve, review and control procedures.

Safety significance

The proper use of procedures is important to safety. 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:

Canada

Procedural compliance is a requirement in Canadian nuclear stations and is rigidly enforced. However, it is also recognized that procedures may at times be deficient and therefore staff are required to apply both a questioning attitude and conservative decision making in execution of procedures. In situations where procedures appear inadequate or confusing staff are required to stop, report the situation to their supervisor and only proceed once the situation has been satisfactorily resolved. The adequacy of procedures is routinely reviewed and a number of the procedures are routinely tested during operator training.

India

During initial days of PHWR operations, while recognising and practicing the use of procedures, expertise and high skill of operation and maintenance personnel were considered for specific jobs. System of authorisation to carry out special jobs by operation was constituted over and above qualification and licensing procedures. Standard Order To Operate (OTO), check-lists were prepared and used for safety related systems and equipment.

With the number of human error incidents, AERB and utilities gave top priority for preparations of operation and maintenance procedures and adherence to the procedures specially for safety related systems. The procedures were made available on station computer local area network (LAN) system. Approval of design engineers and regulatory authorities were incorporated for preparation of procedures. To have strict control on change in procedure if required, the safety related procedures, such as reactor protective system testing forms were allowed to be made available by printing only. Different colour coding for each of the three protective system channelswas adopted for OFS form to carry out safety system testing to avoid human error during testing. Top management and AERB are keeping special attention for non-adherence to procedures which leads to an event. Safety culture was incorporated and encouraged to follow procedures by operation and maintenance personnel while there is tendency not to use procedures in frequently carried out operations due to complacence/overconfidence, it is also important to use them for non-frequent operations. Detailed plans with brainstorming can avoid accidents like Chernobyl. Checklists where parameter values need to be filled rather than just ticking off, are useful to ensure their use. OFS forms are revised to indicate normal values of each parameter.

Continuous surveillance and perusal by AERB and utilities, all the required procedures for operation and maintenance safety system equipment and important equipment are being followed. New procedure sand updating of procedures based on in-house and international experiences are being done. For some special jobs, detailed procedures are made. (such as common system shutdown-effecting both PHT and moderator system) taking into account safety analysis and assessment and approval of AERB is obtained prior to carry out such jobs.

It is very advantageous to have the procedures prepared and check them out during commissioning for validation to get vital feedback and improving them. AERB inspection covers a complete check on all aspects of procedure usage and documentation system.

Korea, Republic of

This is not considered to be a “generic safety issue” in Korea. KEPCO is conducting the periodic review which are carried by plant staff to verify that the plant procedure management is in accordance with the rules in force.

Romania

It could be mentioned that the plant procedures are being under surveillance and review process, but no major problems were encountered so far. On the other hand it is important to be noted that controls and other actions are being considered by the License and required by the Regulatory Body so that the adherence to the procedure is increased. This is however a continuous process.

This is a topic of further evaluations and reports after significant results of the operating plant will exist and they will be evaluated as part of the self assessment and operating experience feedback process.

ADDITIONAL SOURCES:

ISSUE TITLE: Adequacy of emergency operating procedures (MA 11)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to managing risk as an element of an effective nuclear management program in which safety is the highest priority.

Necessary features of Emergency Operating Procedures (EOP) 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 AA 5, "Need for severe accident analysis" and SS 3 "Severe core damage accident management measures").

Safety significance

The capability to deal effectively and promptly with 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:

Argentina

Operating and Emergency procedures (POEAs) are constantly improved and modified considering the operating experience and the findings of PSA when human actions performed during all accidental sequences. The procedures proposal to be applied to face both an abnormal event or an emergency occurrence are as following: 1. To identify through direct analysis of indications that appears in the main/auxiliary control room that an emergency occurred. 2. To guide in carrying out the operations required to get the plant on safety shutdown condition. 3. Give operator the direct sequence of the indications to facilitate the event/emergency diagnosis. This symptoms based on diagnosis methodology is used in all the cases. Personnel involved with plant PSA activities has the responsibility to incorporate information contained in the PSA into the POEAs. To make compatible this information amount with the required clearness, the relevant issues and priorities are discussed between both personnel from operations and PSA personnel.

Canada

Procedures are in place for normal and abnormal events. As well, emergency operating procedures (EOP) are established for handling events of potential escalating severity. These procedures are developed in large measure using information from both asfety analysis and probabilistic risk assessments. Simulator-based operator training and operating experience feedback are used, as appropriate, to regularly assess adequacy of procedures.

India

EOP (Emergency Operating Procedures) to handle emergency and off normal operation is being prepared and used since early days of PHWR stations. First set of EOPs was prepared in 1977 two years before TMI-2 accident. A dedicated group is constituted in Headquarters consisting of operation, maintenance, design and AERB personnel to prepare the EOP for new stations. On advice of AERB, EOP was prepared on event sequence and time base and considering the Control Room operator's ability to understand and follow the steps. This has made the EOP manual "user friendly."

With experience of Steam Generator tube leak incident, it was realised that to handle small leak (about 2 kg/day) EOP were not available. EOP were prepared on such incidents. Updating of EOP based on experience, PSA analysis and operational feed back is being done.

Special set of EOP has been prepared for fuel handling systems. All EOP are constantly updated. Seventy per cent syllabus of requalifications done once in three years for operating licence, is from incidents and EOP. Such training stood in good stead as the NAPS fire incident was handled by the operator following almost entirely, the EOP for station black out (SBO). During the initial preparation of EOPs several design changes and operator support systems were recommended and incorporated. India has not prepared EOP for majority of BDBAs and this effort will now be intensified. Plant specific simulators and other training aids are used for continuous training and familirization of EOP. The progress of preparation of symptom based EOP is slow.

Korea, Republic of

AOMs (Abnormal Operating Manuals), a part of PGP (Procedure Generation Package) for establishment of EOPs (Emergency Operating Procedures), were developed for the operating licence of Wolsong units 2, 3 and 4. The AOMs include 11 event-oriented AOMs and 2 symptom-oriented AOMs. Each event-oriented AOM covers specific event such as LOCA, SLOCA, MSLB, and so on. One symptom-oriented AOM is for the diagnosis of the events and the other one is for the events not specified in event-oriented AOMs or multiple events. Wolsong units 2, 3, and 4 developed EOPs according to the developed PGP including the AOMs.

Wolsong unit 1, which already started commercial operation before TMI, developed plant specific AOMs and EOPs referring those of Wolsong units 2, 3, and 4 in 1999.

Also, Wolsong units 1, 2, 3, 4 are developing the AOPs (Abnormal Operating Procedures) for compensation of the shortcomings of the OP&P in the failure of combined systems such as instrument air system. This project will be completed in 2000.

Romania

The existing system is based on the generic experience of the CANDU6 plants. The plant is operated both using "event oriented" procedures and "symptom oriented" procedures The Abnormal Plant Operating Procedures (APOP's) are part of the documents defining the operating envelope approved by the Regulatory Body. Information for this issue is to be connected with that provided under issues SS3 and SS7.

This is a topic of further evaluations and reports after significant results of the operating plant will exist and they will be evaluated as part of the self assessment and operating experience feedback process.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-2.2, IAEA, Vienna (2000).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Development and Review of Plant Specific Operating Procedures, Safety Reports Series No. 48, IAEA, Vienna (2006).

ISSUE TITLE: Effectiveness of maintenance programmes (MA 12)

ISSUE CLASSIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to maintaining the plant as an element of an effective nuclear management program in which safety is the highest priority.

The overall objective of maintenance programs is to ensure that structures, systems and components will continue to function at the required level of reliability and effectiveness throughout the operational life of the plant. To achieve this objective it is necessary that installed equipment meet performance requirements and operate when needed, and that equipment malfunctions and deficiencies be identified and corrected in a timely manner.

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.

The issue was identified in test and/or maintenance during power operation.

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

MEASURES TAKEN BY MEMBER STATES:

Argentina

Argentine regulatory philosophy is non-prescriptive, as a consequence regulatory policies implementation related with nuclear power plants maintenance activities is performance – based, that means is based on safety goals fulfillment. Regulatory control safety goals referred to nuclear power plant the maintenance activities is to verify that such activities assures the safe operation of components, structures and systems according were established into the Operation License and the rest of mandatory documents.

To reach the above mentioned goals, was established a regulatory criteria set that are applied to nuclear power plant maintenance evaluation. Such criteria are divided into the following interest issues: safety culture; management; organization and functions; knowledge and personnel skills; facilities and equipment; preventive maintenance; predictive maintenance; corrective maintenance;

procedures, records and histories; conduct and control of maintenance activities; material condition; in service inspection; stores and warehouses and outage management.

For each one of the above issues, the corresponding objective is defined associated to the criteria set application established to evaluate the maintenance effectiveness and to perform the regulatory control required. This implies a definition of the maintenance aspects that must be affected to regulatory control (objectives) and how must be performed such control (criteria).

Maintenance tasks are divided into preventive and corrective. For preventive maintenance tasks, a program is available in which frequency and scope of each equipment or component maintenance is indicated. For early detection of failures maintenance is complemented with techniques such as vibration analysis, ultrasound, eddy currents, infrared analyses, etc.

Preventive maintenance is carried out being the plant in operation and during programmed outages. During these last ones, the big components that must be checked only during shutdown are controlled, such as the main pumps of the primary circuit, moderator pumps, etc. Corrective maintenance is carried out every day, according to the priority assigned to the solution of failures.

Every task is supported by a working plan containing indications for its performance, including ALARA principle considerations.

For the case of maintenance tasks never carried out before, it is required that a working plan be written and approved before its performance, which must be done in collaboration with Engineering, Operation, Maintenance and Radiological Protection sections.

Each equipment has its historical report containing information related with its time of use, failures and preventive and corrective maintenance actions performed on it. This equipment historical report is used as a basis for the improvement of both preventive and corrective maintenance, being also used as database for the determination of spare parts stock and failure frequency needed for the probabilistic safety assessment.

The programmed outages frequency for nuclear power plants maintenance was determined as a function of the following permanent criteria:

- Fulfilment of in-service inspection program.
- Revision of big components according to the manufacturer recommendations.
- Periodic tests of safety systems, which require that the plant be out of service for their execution.
- Steam generators inspection.
- Optimisation of occupational exposure.
- Optimisation of contractors services.

Since the beginning of the nineties, the following tasks were added to those already considered in the permanent criteria during the programmed outages:

Atucha I Nuclear Power Plant

- Replacement of fuel channels covered with Stellite-6 alloy by others fuel channels containing an LC-1c alloy without cobalt
- Backfitting: Second heat sink and emergency power supply system modification.

Embalse Nuclear Power Plant

- Pressure tube inspection program.
- Feeders inspection as a part of the in service inspection program.

Canada

Effective maintenance programs are a key element to ensuring both the ongoing health of plant systems and the integrity of safe operation. Canadian utilities have implemented maintenance governance which address programs for conduct of maintenance, managed of predefined maintenance work, maintenance work management and material management, amongst other aspects.

India

Analysis of outage allocation, events, non-availability etc. indicated that maintenance is an area that would require more intensified efforts. This was done in a concerted manner with rich dividends. Quality assurance has improved considerably in maintenance. Maintainability has been an important input to the designers. Efforts have been put in improving maintainability, better maintenance aids and tools, use of computer in history preparation, condition monitoring and reliability centered maintenance, training with mock-ups, blown up and cut sections of completed equipment etc, improved maintenance procedures, spare parts management, resource planning, work planning, future profiling, effective organisation, duties, responsibility and role clarity, inter-sectional horizontal communication, improved coordination, better outage planning etc. Computerised maintenance management system (CMMS) programme initially installed at Kakrapar Atomic Power Station (KAPS) and later incorporated for all stations. AERB is contemplating on instituting performance indicators, including in maintenance and these could include procedure maintenance backlog, deficiency reports more than, 1, 2, 3 months old, failures maintenance, MTBF, percentage completion of planned maintenance activities, number of people trained and qualified etc. Maintenance is an important area included in the periodic regulatory inspection.

Status and availability of spare parts has been made on computer network and it is interlinked for all stations to quickly identify the locations.

Korea, Republic of

See issue GL 2, SS6, CS 1

Romania

The Licensee maintenance program is under review based on the plant own experience and on the regulatory requirements. Licensee has also some specific topics under consideration as part of this process, as for instance the component engineering actions to assure the maintenance planning for 13 weeks ahead in any moment.

This is a topic of further evaluations and reports after significant results of the operating plant will exist and they will be evaluated as part of the self assessment and operating experience feedback process.

It is an important issue of the Strategic Polic\y for relicensing unit 1 and part of the forseen periodical safety review.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Management Strategies for Nuclear Power Plant Outages, Technical Reports Series No. 449, IAEA, Vienna (2007).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Indicators for Management of Planned Outages in Nuclear Power Plants, IAEA-TECDOC-1490, IAEA, Vienna (2006).

ISSUE TITLE: Availability of R&D, technical and analysis capabilities for each NPP (MA 13)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to maintaining staff capability as an element of an effective nuclear management program in which safety is the highest priority.

Effective R&D to support a nuclear power program needs to be assessed periodically to ensure that facilities and research skills in key technology areas remain available when required. Similarly, design knowledge and capability must be maintained within functions and organizations that have been designated “design authority” roles.

Adequate technical and analysis capability is an important input to the efforts of operational and maintenance staff in the safe and efficient operation of nuclear power stations. Such capabilities could be essential inpputs to root cause analysis, reliability centered maintenance condition monitoring, transient analysis, safety evaluation of systems , continuous trending of safety system performance etc. In addition, it is important to have adequate instrumentation to provide the necessary inputs.

Safety significance

The above contributes to improved and sustained good performance of safety 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:

Canada

This constitutes an industry-wide concern. Steps are currently being taken by the utilities to ensure that an adequate R&D base continues to be available to address safety-related issues. However, all utilities are involved in planning and monitoring progress of joint R&D work and evaluating the products of this work to ensure that their needs are met. To enable them to perform these functions the utilities have staff with the requisite knowledge and skills in the various technical disciplines pertaining to nuclear safety.

India

The Indian PHWRs are manned by graduate or postgraduate engineers. The selection procedure and standard for fresh engineers for design, R&D and operating units is identical. The initial basic orientation training is also identical with operations training imparted subsequently. Thus generally adequate technical and analysis capability exists in the staff and the selected groups are specially trained for root cause analysis, reliability assessment from operating and surveillance records, condition monitoring and continuous assessment and trend monitoring of safety systems. Where some R&D is required like development of special plug for end shield cracks and sealing arrangement for OPRD at Rajasthan Unit No. 1, R&D work was done at NPP site itself. For complicated analysis, R&D, in depth reviews etc. NPPs obtain support from Headquarters, BARC and other R&D

organizations, etc. In NPCIL headquarters a separate R&D group has been created. In many cases, for specific tasks special committees are set up with experts from Bhabha Atomic Research Centre, Safety Research Institute of AERB and premier academic and R&D organizations in country.. AERB has found the above arrangement satisfactory. The design group at utility headquarters is a big organisation. Dedicated groups are designated for each NPP to provide design, analysis & field engineering support. Thus each NPP has a " design authority". Loss of corporate memory & knowledge is fortunately, not a problem presently in India.

Korea, Republic of

In each nuclear power station the root cause analysis, reliability assessment on the operating and surveillance records are carried out by RETs (Resident Engineers Teams) from KOPEC in Korea. Thus general adequate technical and analysis capability exist in that staff. And also, KEPRI supports the KEPCO and it carried out safety related R&D.

Pakistan

At KANUPP, in-house capability for reliability and risk assessment (PSA & its application) exists on site. Similarly a dedicated accident analysis group also exists on-site. Establishment of facility such as Design & Development (Mechanical), Control & Instrumentation Application Lab (CIAL), Computers & Development Division (CDD), KANUPP Institute of Nuclear Power & Engineering (KINPOE) etc. provide R&D and some analytical facilities to KANUPP.

Romania

This is a topic of further evaluations and reports after significant results of the operating plant will exist and they will be evaluated as part of the self assessment and operating experience feedback process.

It is an important issue of the Strategic Policy for relicensing unit 1 and part of the forseen periodical safety review.

Licensee will consider as a method of implementation of these requirements long term R&D projects.

ADDITIONAL SOURCES:

ISSUE TITLE: Strengthening of safety culture in organisations (MA 14)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to the attitudes, mechanisms and processes that are established to promote and support safety culture within an effective nuclear management program in which safety is the highest priority.

A good design and satisfactory operational safety are required to ensure adequate safety levels throughout the life of NPPs. To ensure good operational safety, it is important that there should be adequate framework (systems, procedures, documents) as well as good safety culture. Organisations should have business plans to strengthen safety culture to ensure good communication, rigorous and prudent approach and questioning attitude among the operating staff.

Safety significance

Good safety culture in individuals and organisations is an important ingredient for enhanced safety levels.

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:

Canada

Strengthening the safety culture and maintaining it is an on-going activity in the stations. National and international practices are always taken into consideration. Feedback from operating experience is a continuing effort. Specific governance has been introduced to ensure elements of "safety culture" are recognized and and actively pursued. These include definition of behavioural expectations in standards for the organization, such as promoting reporting of all adverse condtions, active coorective action programs to address deficiencies, application of conservative decision making, promotion of questioning attitudes, and clear definitions of roles and responsibilities, amongst others.

India

Although safety culture cannot be mandated, AERB encourages NPPs to have business plans to strengthen safety culture. Such business plans are under various stages of implementation at Indian NPPs. Internal assessment following IAEA ASCOT guidelines have been carried out at all NPPs, weak areas recognised and corrective actions are being implemented. AERB has ascertained that there is top and middle management commitment to safety culture at all NPPs. AERB and utility recognise that maintaining satisfactory performance and moving towards excellence is a continuous journey and not a destination and efforts need to be continued.

Korea, Republic of

Safety culture is very important not only to the operating staff in the stations but also to the regulatory side. Good safety culture in individuals and organizations of nuclear family is the most important factor to level up the safety.

Pakistan

KANUPP has implemented the following programs to strengthen safety culture within the plant.

1. Self-Assessment Program at KANUPP

After development of self-assessment program a station instruction has been issued and the program has been implemented since first quarter of March 2004. Following activities have been initiated for the promotion of an effective and continuous self-assessment.

(a) Use of Drop Boxes: In order to facilitate the easy reporting of the plant problems (minor or major) a user friendly reporting system has been introduced. Any plant deficiency / problem may be reported anonymously through drop boxes on a prescribed form. Problem reporting forms are made available with the drop boxes, which are simple in comprehension so as to make them usable by the lower staffs as well. These boxes have been installed at different convenient locations of plant within easy approach of plant personnel.

(b)Hoarding Boards: A safety promotion campaign has been started by displaying "Hoarding Boards". These display board are printed with safety related slogans. These boards have been installed at different locations of the plant.

(c) Open Lectures: Open lectures are being developed and will be delivered to plant personnel. The goal of these lectures is to promote the understanding and implementation of self-assessment and safety culture.

(d)Self-Assessment at Divisional / Section Level: It is the responsibility of respective divisions to develop and implement program to assess their performance continuously and improve at all level. WANO Performance Objective and Criteria, PROSPER guidelines can be used as a reference documents in promotion of self-assessment culture.

(e) Independent Evaluation of Self Assessment Program / Activities: An evaluation of the self-assessment activities in different divisions of plant will be conducted twice per year to uncover the weaknesses/ deficiencies and achievements. Any significant shortcomings that are noted during assessment will be included in bi-annual report which will be presented to management.

A comprehensive independent evaluation of the effectiveness of the self-assessment activities / program in the various division/ unit/ section of plant will be conducted on the basis of the WANO Performance Objective and Criteria and PROSPER guidelines. The frequency of this assessment will be once every 2 years. The duration of this assessment will be two weeks.

Follow-up of findings and recommendations for further improvement will also be conducted to ensure that the weaknesses have been removed. The follow up team will consist of a leader and two to three experts. The duration of follow-up will be one week.

2.Equipment Performance Monitoring Program

Equipment Performance Monitoring Plans (EPMPs) of safety, safety related and safety support systems have been developed. These plans will be implemented in three phases. In the first phase, EPMPs of Emergency Core Cooling System, Containment System and

Essential Power System have been implemented in January 2005. Degradations of critical equipment of these systems are being monitored, trended and compared against the acceptance band/ criteria. If any degradation will be found exceeding the acceptance band/ criteria, necessary corrective action will be initiated.

3. Corrective Action Program

In order to improve safety and reliability of the plant, KANUPP has developed a Corrective Action Program (CAP) called Event & Issue Identification and Corrective Action (EIICA). The EIICA (CAP) program has been formally started from second quarter of 2004. The program is to be implemented in three phases. In the first phase of the program reporting, coding, regulator interface, screening, evaluation and implementation of corrective actions is underway. In the second phase, trending of event using event and cause codes will be done to identify generic issues and vulnerabilities to allow corrective actions to be taken before significant problem results. While in the last phase, a plan will also be prepared and implemented to share the lessons learnt with other utilities within country.

4. Regulatory Evaluation for the Assessment of Safety Culture

In December 2004, a Pilot Regulatory Inspection of KANUPP in the area of Safety Culture was conducted by the Regulatory body. INSAG-04 was used as a reference document for this inspection. Meetings, interviews and discussions were held between KANUPP management and the regulatory team. It was decided that at this stage no regulatory action may be initiated based on the findings. A formal inspection on safety culture of KANUPP will be conducted by PNRA in the third quarter of 2005.

Romania

Information is included in MA7.

It is also to be noted that this is an issue for which even if a lot of actions were taken at the plant by the Licensee (training courses for supervisors, use of the Abnormal Condition Report system, generic plant personnel training, implementation of STAR in the training programmes etc.) initiation of cooperation actions is considered very useful by the Licensee.

ADDITIONAL SOURCES:

4.2.2 Operations (OP)

ISSUE TITLE: Operating experience feedback (OP 1)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to incorporating operating experience feedback into a corrective action program as an element of an effective nuclear management program in which safety is the highest priority.

It is important to draw lessons continuously from national and international experiences to improve the safety performance. It is necessary to set up a structured system to facilitate the above to enable drawing lessons from IRS, WANO & COG, national experience, etc.

Safety significance

Drawing lessons and effecting corrective actions can improve safety performance.

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:

Argentina

The Regulatory Authority required the utilities an "Operating Experience Management Program" to be implemented by each NPP considering as follows:

1. Use of available data bases such as IRS, IAEA, WANO, COG, INPO, etc., as a part of normal activities.
2. To select the event that could be applied to argentine NPP´s.
3. To perform both an investigation and root cause analysis corresponding to each event selected.
4. To perform both an identification and implementation of corrective actions to avoid future event re-occurrences.
5. To divulge information to both own plant personnel and foreign plant personnel.
6. To perform analysis, trend determination, control verification and plant performance safety review.
7. Use of performance indicators.
8. Lesson learned.

Besides, the utilities have been required to send the Regulatory Authority a quarterly advance report of the program related activities.

Canada

Within each licensee organization there exists an event reporting and corrective action follow-up system. At the national level, a section of the CNSC is dedicated to handling “significant event reports”, following-up on corrective actions and initiating investigations if necessary. Canada also participates in the international event reporting system (IRS) jointly operated by the IAEA and the OECD/NEA. Canadian licensees are also members of WANO and INPO.

India

AERB is issuing a safety guide (AERB/NPP/SG/O-13) on Operational Safety Experience Feedback on NPPs. Utility reports the incidents to AERB in accordance with the reporting criterion. AERB reviews these reports at various levels in its multi-tier review system. Its findings and observations are communicated to all the Indian NPPs. The utility has set up an operating experience review committee (OERC) at all its stations and IRS, WANO, COG and incidents of all Indian NPPs reports are discussed and necessary improvements are incorporated in a continuous manner. Senior operation engineer is involved in the review and shares his experience with operating crew personnel. Dedicated cells for operating experience feedback exists in all NPPs, utility head quarter and AERB. Emphasis is given to review all low-level events at each NPPs and review feed back is made available to all concerned agencies. Also learning from inputs other than incidents like reveal of deficiencies during operation or maintenance, information exchange with other national and international bodies, technological developments in related field, etc have been adopted to the extent feasible.

Korea, Republic of

Utility level operating experience feed back has already been implemented by operator in Korean NPPs. However, establishment of National level OEFB system, which is required by the Article 19 of the Convention on Nuclear Safety, is under going. Some of the parts of the system are already completed and in effective. Others are under development.

Completed parts and related activities are as follows;

1. Reporting Criteria

   Reportable events and reporting procedures are prescribed in the Notice of the Ministry of Science and Technology 96-25 'Incidents and Accidents Reporting in Nuclear Installations' which has been effective since 1993.

   About 20 to 30 events are reported per year according to the reporting criteria.

2. Event Database system: NEED (Nuclear Event Evaluation Database)

   NEED system was developed as an event database for Korean NPPs in 1995.

   All the reported events with coded watchlist, same as IAEA AIRS form, and full text are stored in the NEED system. Event search and statistical analysis are available with the NEED system. Currently about 500 events are stored in the database.

3. International Information Exchange

   Information exchange with international organizations or other countries is one of the important activities of operating experience feed back. AIRS data from IRS/NEA are analyzed and applicable ones are selected to get lessons and feedback. Every year, several events on Korean NPPs are selected and reported to IRS.

   Information exchange with other countries, regular or event based, is implemented by bilateral agreements.

4. Performance Indicators

Performance Indicators (PIs) for Korean PWR plants are developed in 1997 and the trend of each plant is monitored and analyzed. The PIs are composed of 10 indicators in 8 areas and annual report is published. Unusual change or decline of performance will be identified and increased regulatory efforts and resources are assigned to the area. PIs for CANDU are under development and will be completed in 2000.

Under going parts are;

1. Implementing Program

   Implementing program is under development and draft will be completed in the end of 2000. This program includes review of legislative provisions, event selection guidelines and method, determination of corrective actions and implementing procedures and so on. Final determination will be made after trial application of the draft program.

2. Development of NPA

   Utilization of an analyzing tool with other measures would be helpful for effective event analysis. A Nuclear Plant Analyzer is under development to be used as an event analyzing tool to provide an realistic simulation and quantitative features of events. The development of an analyzer for Korean Standard Nuclear Plants (KSNPs) will be complete in 2000 and analyzers for other reactor types including CANDU and Westinghouse type reactors will be followed.

3. Expansion of feedback area

   Not only the event experience but other regulatory experience including former licensing issues will be included in the OEFB system. Those issues are identified and under review to determine applicability. This will make the OEFB system an integrated regulatory experience feedback system.

Pakistan

At KANUPP, Operating Experience Feed Back system exists for exchange of plant operational information. Methods of using operating experience are structured to provide applicable information to the concerned persons. When Operating Experience Feedback personnel analyze the causes of recurring event, it determines as to why a lesson was not effectively learned.

Operating Experience Feedback (OEF) program of KANUPP consists of (a) internal OEF and (b) external OEF.

(a) INTERNAL OEF

The objective of this program is to identify/report all problems (i.e. event/issue) whether Significant, Low Level or Near Miss and take appropriate corrective actions to avoid the recurrence of events through two program.

- Corrective Actions Program (CAP) program

- Event identification / reporting programEXTERNAL OEF

The station’s goal for external operating experience is to use lessons learned from industry and station operating experience effectively and efficiently to improve plant safety and reliability. Learning and applying the lessons from operating experience is an integral part of station culture and is always encouraged. When Operating Experience Feedback personnel analyze the causes of recurring event, it determines as to why a lesson was not effectively learned. Station personnel regard operating experience as helpful and important to them, they use this information at every opportunity. Methods of using operating experience are structured to provide applicable information to the right personnel in time to make a decision.

Main sources of information:

vi) *COG*: COG Bulletin, COG CANDU Performance Indicator, GOOD Practice, JUST-IN-TIME Briefings, OPEX Database, Summaries of OPEX Weekly Screening Meeting, Monthly newsletter, Cognizant

vii) *WANO*: SERs, SOERs, JIT, OE Database, EARs, MERs, ENRs, ETRs, Good Practices, Annual Reports

viii) IAEA/ IRS/ IRT

ix) All in-house reports including Station Outage Reports, Unusual Occurrence Report, Monthly Technical Report, In-service Report, Special Technical Report.

x) Information Exchange visits to other NPPs.

Review and Implement of Operating Experience in KANUPP

A team of engineers from all concerned divisions is responsible to review messages from COG OPEX and WANO/INPO. The messages either retrieved through the Nuclear Networks or received in the form of reports are scrutinized. Screened material sent to relevant personnel for review and relevance with KANUPP. If the information is found applicable to the plant are incorporated through changes in procedures, systems or equipment, etc. in order to prevent recurrence of industry event at KANUPP. KANUPP also raises specific queries regarding its own problem where it is felt that COG or WANO could be of help. The nuclear networks have been of great assistance in resolution of some of the technical problems faced by KANUPP. KANUPP also respond to the queries raised by other NPPs in the areas where it has sufficient experience and expertise. A number of changes in plant system and procedures have been carried out on the basis of operating experiences.

Romania

Information is included in MA7

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Management Strategies for Nuclear Power Plant Outages, Technical Reports Series No. 449, IAEA, Vienna (2007).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Indicators for Management of Planned Outages in Nuclear Power Plants, IAEA-TECDOC-1490, IAEA, Vienna (2006).
- ATOMIC ENERGY REGULATORY BOARD, “Operational safety feedback on nuclear power plants“, AERB/NPP/SG/O-13.

4.2.3 Surveillance and maintenance (SM)

ISSUE TITLE: Adequacy of non-destructive inspections and testing (SM 1)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to the conduct of engineering to protect the assets as an element of an effective nuclear management program in which safety is the highest priority.

The non-destructive testing (NDT) for reactor coolant system in-service inspection is carried out according to the In-Service Inspection programme and it is essential to give assurance of safe operation of equipments and components between inspection periods. A well drawn out inspection programme and its implementation is necessary. Current techniques and tools must be updated according to the state of the art. Reliable in-service inspection is a key provision required to preserve the integrity . Some deficiencies have been revealed, related to inspection from outside, testing areas, and testing of steam generator collectors and tubing. There is also restricted accessibility of some piping parts, penetrations, piping welds, steam generator shell welds, and specific piping nozzles. Qualification requirements are also necessary

Safety significance

Timely detection of degradation is essential to maintain reliability of plant systems. Undetected defects can challenge the safety functions.

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:

Canada

All Canadian nuclear operators have active in-service inspection programs both to meet regulatory requirements and to ensure protection of their assets. Additionally, there is ongoing R&D work aimed at improving inspection techniques and devices as well as attention to planning and execution of in-service inspection work as part of integrated outage management programs.

India

In-service inspection (ISI) including non-destructive testing (NDT) have gone through a gradual evolution in India. While designers have not factored ISI initially, this is now an essential input. The scope and the extent of ISI need to be optimised to achieve a balance between man-rem and time spent and benefits arising from it. Increasing inspection consumes a good chunk of station manrems and efforts are required to reduce this and balance it with returns. Special ISI tools need to be developed for MAPS calandria inspection, RAPS end-shield and OPRD inspection etc. Updating is also being done on inspection of heat exchanger tube, piping, equipment internals etc. Specialised crews with equipment and procedures need to be set up at each NPP. Modification would require tests like hydrostatic, helium leak tests etc. similar to initial construction tests and need to be equally rigorous. Management needs to maintain the independence of the inspection crew and factor this into organisational aspects. Indian NPPs are able to derive advantages from a good Headquarters support as well as from R&D institutions of DAE who have a strong NDT groups. AERB safety guide on in-service inspection of nuclear power plants (AERB/SG/O-2) gives guidelines on ISI of NPPs.

As per the current regulatory practices systematic Pre Service Inspection (PSI) programme has been instituted for safety significant equipment/component such as feeders, heavy water heat exchangers, and steam generators tubes etc. to establish the efficacy of NDT technique and to collect base line data.

After Annual Shut Down of a unit the unit is started back only after regulatory review to ensure satisfactory completion of all the important jobs including In-Service Inspection (ISI).

Korea, Republic of

The MOST Notice 95-1 (revised 98-12), "Regulation on the in-service inspection and test on the facilities of nuclear power plant" requires the in-service inspection and test of pumps and valves of nuclear power plants. This Notice states that ASME OMc code IST should be applied to the In-Service Tests of the safety-related pumps and valves. ASME OMc code IST was developed for PWR and BWR. Due to the differences of the design characteristics between PWR and CANDU plants, some difficulties occurred in the application of ASME IST code to pumps and valves for the In-Service Test of CANDU.

The primary purpose of the test in CANDU is to maintain the system reliability while that of PWR is focused on the component functionality.

The followings are current issues to be considered.

1. Selection of the components to be tested
2. Classification of the components to be tested
3. Duplication of the original CANDU safety tests and ASME in-service tests
4. Measurement of the performance of pumps on which any flow meter or any pressure meter is not installed

KINS concluded that the safety-related components, pumps and valves, should be tested and required the licensee to submit the In-Service Test plan on the basis of the ASME code requirements.

KINS staff has encountered a number of pump related issues such as ultrasonic flow meter uncertainties and relief requests for test that are determined by the KEPCO to be impractical.

The relief request had been submitted stating it was impractical to test each pump because flow meters and/or pressure gages on some pumps were not installed

KEPCO is required to demonstrate that the safety related pumps remain capable of performing their intended safety functions

Romania

The License has as part of its valid licenses conditions specific programs for the mandatory inspections, for which the hold points, methods, results analysis methods and the reporting requirements are defined as per the regulatory requirements. Supplementary non-mandatory inspections are being performed by the Licensee. There are no significant aspects to be reported so far on this issue

For this issue, which is also part of the Strategic Policy and further periodical safety Review process, the results on the programs will be reported further on in case they will be relevant to define it as a generic safety issue.

ADDITIONAL SOURCES:

- ATOMIC ENERGY REGULATORY BOARD, safety guide, "In-service inspection of nuclear power plants", AERB/SG/O-2.

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

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to the outage planning and management as an element of an effective nuclear management program in which safety is the highest priority.

A focus on reducing cost by reducing the length of outages can increase both the amount and frequency of maintenance performed during power operation. Planned equipment outages are potentially limited by avoiding violations of either Operating Policies and Principles (OP&Ps) or Technical Specifications. (See MA4 for a discussion of control of activities during outages).

Expansion of the on-line maintenance concept requires thorough consideration of the safety (risk) aspects. The on-line maintenance concept appears to extend the use of Allowed Outage Times (AOT) stated in the Technical Specifications, where they are in use, beyond the random single failure in a system and a judgement of a reasonable time to effect repairs upon which the AOTs were based.

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:

Canada

Removal of components from service is performed under strict procedural controls with assessment of the potential adverse impact of the equipment removal. Increasingly, probabilistic risk assessment is being applied as a tool to identify and assess change in risk associated with such activities, thereby providing a means for supporting risk informed decision making into activities resulting in changing system configuration, as well as supporting optimization of outage work.

India

Initial designs did not provide adequate arrangements for performing on-power maintenance in Indian NPPs. Selection of improper isolation valves, lack of venting and draining facility, spacing of electrical cells in MCCs and buses etc., non-segregation of control power supplies etc. brought in severe restrictions in on- power maintenance. Thus a large amount of jobs are piled up in major outages and quality can become a casualty. These are being corrected by efforts in many areas.

The Allowable Outage Time (AOT) in technical specifications has been drawn up based on technical judgement. It is planned to uses PSA to make these more scientific. In addition AERB has asked NPPs to reduce downtime of equipment figuring in technical specifications with view to not only exceed the AOTs but reduce the AOTs themselves in technical specifications. The utilities are planning to do this by stocking adequate spares, using tested, assembled spares, better tooling, training, working round the clock etc. On-power maintenance is done as per approved maintenance plans to ensure that adequate defence in depth is maintained.

Korea, Republic of

This is not considered to be a “generic safety issue” in Korea.

Romania

Information is included in MA5 and AA4. Operating documents are strictly controlling the process.

ADDITIONAL SOURCES:

- IAEA OSMIR database.

ISSUE TITLE: Use of ice plugs (SM 3)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to the maintenance management and procedural controls as elements of an effective nuclear management program in which safety is the highest priority.

Ice plugs are used to isolate components (such as inboard isolation valves) for maintenance in locations that cannot otherwise be isolated. In PHWR's ice plugs are used in heat transport system feeder pipes to isolate fuel channels for maintenance. 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 (PHTS) pressure boundary, these ice plugs become a temporary part of the pressure boundary.

Some licensees have used piping mockups to thoroughly evaluate ice plug applications prior to their use on reactor system piping. Important considerations include examining training, procedures, and contingency plans associated with the use of ice plugs, and evaluating the need for and availability of additional water makeup systems and their associated support systems.

Safety significance

If an ice plug fails, it can result in an immediate loss of primary coolant. Ice plug 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 ice plug 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:

Argentina

Instructions and considerations to perform ice plugs in pipelines were updated including the requirement to utilise volumetric tests. Such Instructions and considerations are includes in the Operating Manual.

Canada

Ice plugs are used as a means of isolating certain components which do not have direct isolation provisions, for example the feeder pipes in the reactor heat transport system. This is done under strict procedural controls according to standard procedures that have been established using many reactor-years of experience.

India

Use of Ice-plug is adopted to isolate components for maintenance which cannot be isolated otherwise. Liquid Nitrogen (N2) is used for freezing the water within the pipe section. They became temporary part of pressure boundary.

Adequate liquid N2 stock is made available. Monitoring of the ice-plug integrity by temperature and using proper devices for formation of ice-plug is done. Contingency plans are prepared to take care of eventualities in case of failure of ice-plug. Care is taken to see that ice-plugs are formed away from weld joints. ISI of the lines is carried out later.

Korea, Republic of

This is not considered to be a “generic safety issue” in Korea.

Romania

This aspect is controlled by the existing normal plant procedures, including contingency actions forseen by the work plan system. No problems were encountered so far on this issue.

On the other hand this issue is part of the programs reported on MA12 and any further results will be reported further on.

ADDITIONAL SOURCES:

ISSUE TITLE: Control of temporary installations (SM 4)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to the engineering change control as an element of an effective nuclear management program in which safety is the highest priority.

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:

Canada

Temporary changes are controlled as part of an engineering change control program with specific procedures to limit and control temporary modifications or installations and provide assurance that the necessary approvals have been obtained at the appropriate pahses of such work.

India

AERB has taken a stand that no temporary installations are permitted in safety systems. However, there have been cases where temporary equipment were used in process systems or safety support systems. These are well documented with proper review and approvals and are kept trackof.

Korea, Republic of

This is not considered to be a “generic safety issue” in Korea. All the temporary installations which are related in safety systems should be approved by regulatory body before installation.

Romania

This is an issue correlated with MA5 and MA6

ADDITIONAL SOURCES:

ISSUE TITLE: Response to low level equipment defects (plant material condition) (SM 5)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to reporting plant material condition and corrective action as elements of an effective nuclear management program in which safety is the highest priority.

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.

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:

Canada

Attention to low level equipment defects has been given greater emphasis in recent years. At one utility this is part of the corrective action program and involves reporting such conditions, correcting the condition immediately, where possible, or establishing tracked actions to correct the problem within an assigned time period.

India

A system of detecting/reporting low level events/defects/near misses, analysing the same and taking corrective actions can go a long way in incident/ failure free performance. AERB has taken a stand that it would like a low level reporting system to be instituted at each NPP. But, however, there is no need to change the reporting criteria. AERB has expressed that a summary to indicate that the low level system is being well run would be adequate. NPPs are finding a problem to decide on correct lower threshold for the system to be meaningful but not having too many reports being generated. It is advisable to have a special cell at each NPP. Introduction of computerised maintenance management system in some of the plants helped in managing low-level deficiencies. This system also helps supervisors and managers to verify the level of deficiencies reported and not repaired in the plant. AERB Safety Guide "Operational Experience Feedback for Nuclear Power" (AERB/SG/O-13) also gives guidelines for low level event reporting system.

Korea, Republic of

This is not considered to be a “generic safety issue” in Korea.

Romania

The deriving of low level thresholds for early indications on the possible trends affecting safety, as defined by the Abnormal Condition Report and Material Condition report systems, is one of the targets of the new framework of the operating experience feedback, as described in MA7

ADDITIONAL SOURCES:

- ATOMIC ENERGY REGULATORY BOARD, Safety Guide, "Operational Safety Experience Feedback for Nuclear Power" AERB/NPP/SG/O-13.

4.2.4 Training (TR)

ISSUE TITLE: Assessment of full scope simulator use (TR 1)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to managing human performance through training as an element of an effective nuclear management program in which safety is the highest priority.

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 inadequate or incorrect operator response in the plant control room.

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:

Argentina

An annual training is performed at Gentilly II (G2) full scope simulator (CANADA) by operation personnel belonging to the main control room. Besides, G2 is similar to Embalse and the full scope simulator is adapted to Embalse requirements. The training is based on consider scenarios that requires the POEAs use. Besides, the operators are trained in both conventional operation and relevant events occurred in the plant. The training / re-training evaluation is performed considering the results of both the actuation in the simulator and the written examinations.

Canada

Although this issue and the next one (TR 2) are not labelled as “generic safety issues”, an entire CNSC division is dedicated to the assessment of licensees training programs, and licensee personnel competence, via, among other things, written examinations, audits, and full-scope simulator examinations. Every CANDU plant in Canada has a full-scope simulator for operator training.

India

During 1989, training simulator based on RAPS-1 was commissioned and licensed operator training and retraining was started on this simulator. . PC based simulation for training was made available in MAPS. RAPS-1 simulator has been upgraded for RAPS-2 and MAPS type stations. The standarised design full scope simulator has been now commissioned at Kaiga 1 & 2 and RAPS-3&4. In TAPP-3&4 (540 MWe) PHWR, a training simulator based on its design has been installed, commissioned and is being used for training and licensing of operation personnel.Also installation of similar simulator is being taken up in Rajasthan. Putting a full scope simulator in each new site has been made mandatory by AERB.*Korea, Republic of*

Wolsong plants have a full scope simulator for operator training. KINS division is dedicated to the assessment of licensee personnel competence, written exams, audits and full scope simulator exams.

Romania

The new regulatory guidelines and the operators actions are aimed at:

- defining the framework and limits of the use of simulator for operator training
- define the necessary actions to approve a simulator for the use as part of the training and licensing activities
- define the use of simulator to check assumptions on possible causes and remedies for events under investigation

There are specific results on this issue. It is considered a generic safety issue, but reports are expected after the unit 1 relicensing will take place in May 2001 and the licensing activities at unit 2 will be restarted.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Authorization of Nuclear Power Plant Control Room Personnel: Methods and Practices with Emphasis on the Use of Simulators, IAEA-TECDOC-1502, IAEA, Vienna (2006).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Guidelines for Upgrade and Modernization of Nuclear Power Plant Training Simulators, IAEA-TECDOC-1500, IAEA, Vienna (2006).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Use of Control Room Simulators for Training of Nuclear Power Plant Personnel, IAEA-TECDOC-1411, IAEA, Vienna (2004).

ISSUE TITLE: Training for severe (beyond design-basis) accident management procedures (TR 2)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to managing risk, in part through training, as an element of an effective nuclear management program in which safety is the highest priority.

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. Periodic re-training in 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 4, Accident management measures", MA 11 "Adequacy of emergency operating procedures and AA 5, Need for severe accident analysis).

Safety significance

Inadequate implementation of severe 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:

Canada

See TR 1.

India

The process of analyzing Postulated Initiating Events related to Beyond Design Basis has been started. This also includes review of international practices to handle severe accidents and setting up of modalities to exchange information at international level. This analysis also forms as input to PSA Level-2. Further, the analysis would help in devising suitable accident management strategies to handle severe accidents in designing overall Severe Accident Management Procedures (SAMP). The work on devising symptom based procedure to handle severe events has also been initiated. Once, such procedures are reviewed and established this will be added in training curriculum and would form a regular feature of Training and Licensing of operators of Indian Pressurised Heavy Water Reactors.

Korea, Republic of

See issue MA 11.

Romania

Training for SAM procedures is considered part of the implementation of SAM approach and is included in the topic related to it from the Strategic safety Policy and Periodical safety Review. Report is expected to detail information after significant results will be available and evaluated.

ADDITIONAL SOURCES:

4.2.5 Emergency preparedness (EP)

ISSUE TITLE: Need for effective off-site communications (EP 1)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

This issue is generic to the nuclear industry and relates to managing risk as an element of an effective nuclear management program in which safety is the highest priority.

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 during 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, regulators in many countries recognized the need to substantially improve the ability to acquire data on plant conditions during emergencies. Typically, the regulatory body's role in the event of an emergency is one of monitoring the licensee's activities to ensure that the appropriate recommendations are made with respect to 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 fulfill 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:

Argentina

When the licensee operating a NPP declares an emergency situation in Argentine, among other things, resident inspector and Regulatory Authority headquarters must be informed immediately according with the procedures. Resident inspectors (working at or living in the vicinity of the plants) go to the On-site emergency control room. A Senior Technical officer of the plant must be fully dedicated to give Regulatory Authority Headquarters general and detailed information of plant conditions and On-site and Off-site emergency arrangements taking place. A team of Off-site emergency response experts is going to be immediately dispatched to the Off-Site emergency Centre while at Regulatory Authority Headquarters a Centre for support and organisation is established with nuclear safety and radiological protection teams for the assessment. The reliability of communication and transmission of information is based on multiple, different and redundant systems, on the public, private and national security domains.

For the initial communication "Beepers" (telephone and then radio text broadcast) and mobile telephone are the alternatives to regular telephone lines. A back up is the HF and VHF system of the a National Safety Force Quarters (National Gendarmery) located at each NPP sites, a very reliable system used for national internal security which allows communication from and to NPP to Regulatory Headquarters under extreme conditions. The Off-site teams could have communication with NPP and Off-Site emergency Centres using VHF "handies", mobile and satellite phones. Facsimile and e-mails are also alternatives. The system is tested on a regular basis. In particular for the Y2K testing communication system (when communications were saturated because New Year salutations) and the results where excellent. A system for transmission of real-time of plant data is in the final stage of development based on satellite communications.

Canada

The adequacy of such communications is routinely evaluated via exercises involving CNSC staff, licensee personnel and provincial and local authorities.

India

Considerable efforts have been made to provide a number of necessary off-site emergency communication systems at Indian NPPs Following communication systems are available at Indian NPPs:

Telephones facilities with trunk dialing facilities

Intercom

Cordless telephones/Walkies-Talkies

VHF wireless radio sets

Hot line/ Satellite Communication System

Telex/Fax

E-mail/ Internet

Computer Local Area Network

Power line carrier communication

Radio/Television/Cable network.

Emergency Control Centres, Environmental Survey and Meteorological Laboratory (ESML) have also been provided with VHF wireless sets and telephone for communication. Linking of all the units of

DAE with dedicated satellite data communication system, an emergency real time data transmission system(ERDS), is under commissioning. Need for this was felt to enable Headquarters support for Narora fire incident.

Necessary data required for monitoring/ handling emergencies can directly be obtained through E-mail from control room or ESML.

There exists an emergency preparedness manual at each site having important telephone/fax numbers and e-mail addresses. As a ready reckoner, a sheet of paper having these numbers is available with all concerned state agencies and kept in the control room.

To ensure the availability of communication systems, necessary checks are carried out daily from the control room by making contact with concerned off-site agencies (i.e.; DAE emergency control room Mumbai, Off-site emergency Directors office, NPC Headquarters Mumbai etc.)

During AERB regulatory inspection random communication checks are being carried out as a part of audit.

Walkie-talkies/ mobile are also useful for internal and external communication, a challenge in this is that their unsatisfactory performance inside the containment building. Inspite of all the efforts, communication remains a weak link and sustained efforts are required to maintain it at a satisfactory level. The fact that a top official of the government communication organisation is a permanent member of the crisis management group of the Indian Department of Atomic Energy indicates the importance that subject of communication demands.

Korea, Republic of

Radiological emergency plan in Korea is based on the National Civil Defense Law and the Atomic Energy Law. The latter regulates the requirements of emergency planning and preparedness for nuclear power plants. The Ministry of Science and Technology (MOST), which is the nuclear regulatory body in Korea, has upgraded its emergency preparedness regulations to ensure that the protective measures are taken by nuclear power plant licensee to protect the health and safety of surrounding general public. One facet of the upgraded regulations is the requirement that the licensee submit revised radiological emergency plans to the MOST for evaluation and review according to the recently published MOST Notice 96-4," Criteria for Preparation and Evaluation of Radiological Emergency Plans ". The Licensee is also required to submit the procedures used to implement their emergency response plan.

The National Civil Defense Law stipulates to establish the radiological emergency plans concerning response for protection of civil and properties on a national level in case of nuclear emergency. The MOST has to establish a basic plan for radiation protection every five years according to the civil defense law, and the organizations concerned including the local government have to prepare its own implementation plans every one year in accordance with the basic plan. The "Civil Defense Implementing Plan" (ROK Civil Defense Law, Article 10, and its Implementing Regulation, Article 11) and MOST Notice 96-4 require the licensee to conduct an exercise to evaluate the major portion of emergency response capabilities. This exercise is normally held each year requires the licensee to conduct an exercise to evaluate the major portion of emergency response capabilities. This exercise is normally held every 3 year for each site, and involves not only the licensee, but also various response groups from the concerned province, county and local agencies.

Romania

This is considered a very important issue resulting from many exercises and reviews of the emergency plans. However reports of details will be delivered if significant aspects and difficulties will be encountered in implementing it.

ADDITIONAL SOURCES:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Design Safety Requirements, IAEA Safety Standards Series No. NS-R-1, IAEA, Vienna (2000).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Preparedness and Response for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GS-R-2, IAEA, Vienna (2002).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Arrangements for Preparedness for Nuclear or Radiological Emergencies Safety Guide, IAEA Safety Standards Series No. GS-G-2.1, IAEA, Vienna (2007).
- ATOMIC ENERGY REGULATORY BOARD, "Preparedness of Operating Organisation for handling Emergencies at Nuclear Power Plants" AERB/SG/O-6 (2000).

4.2.6 Radiation protection (RP)

ISSUE TITLE: Hot particle exposures (RP 1)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

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. PHWR's have experienced infrequent incidences of hot 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 can produce very large doses to small amounts of tissue, It is generally recognized that 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:

Argentina

The following measures have been taken to minimise workers exposure:

- Improvement of control during fuel production (QA programme).
- Monitoring of the fuel before get it into the reactor (visual inspection).
- Coolant-purifying systems are installed since the beginning of operation.
- Detection on-line during operation (early alert system, daily control of primary system D2O).
- Continuous monitoring into working areas includes ventilation systems.
- Use of adequate equipment (clothes, masks, filters) to perform works that involve possible risk.

Also Regulatory measures have been taken to prevent personnel contamination due to hot particles. Those measures have been specially concentrated in the control of the source (preventive more than reactive measures).

Canada

This issue is addressed and controlled through on-site radiation protection procedures, and through the maintenance of a rigourous safety culture.

India

One of the important sources of hot particle exposure is fuel, either due to its failure or during fuel handling. Multifaceted efforts in the areas of design improvements, better fuel management, chemistry control, operational controls etc. have resulted in satisfactory fuel performance. Fuel handling system incidents have also been reduced by systematic efforts.

The problem of hot particle spots by deposition of radioactive particles is experienced in Indian PHWR stations. Hot spots are mainly observed on the dead ends of the piping of moderator and PHT system.. In some of the NPPs Cobalt ingress in moderator system from cobalt based component of regulating rods has been experienced. During initial days of operation the release of cobalt from these components was not realised and purification columns were valved out during shutdown. Presently removal of cobalt by Ion-Exchange columns during shutdown continues. Logics and procedures were modified so that boron saturated IX-columns are only valved IN during unit shutdown to ensure removal of activity. This has reduced the Cobalt activity buildup and hot-spot. Adequate monitors are installed to map hot spots in shutdown accessible areas.

Decontamination of moderator system was carried out in one of the stations and activities were brought down. This procedure is now well established and will be used whenever required. Decontamination does not remove hot spots in some cases. . Replacement of piping is donein such cases AERB monitors hot spots very closely. Special shielding and precautions are taken to minimise exposures due to hot spots specially during maintenance and ISI activities.

Decontamination of PHT system is carried out prior to take up any major jobs, such as ISI, major maintenance, enmasse coolant channel replacement etc..

In new plants layout of piping, equipment and shielding requirement has been modified to reduce the exposures or high radiation field due to hot spots.

Korea, Republic of

The hot particle contamination to workers has not been experienced in PHWRs of Wolsong site as well as in PWRs in Korea. Currently a Regulatory Guideline has not been provided specifically for the hot particle control. However, the relevant procedures are in existence and are implemented by the NPPs. The plant radiation protection procedure for personal contamination control and decontamination contains the definition of hot particles, the method to detect the contaminated spots and level of contamination, and refers to calculational program for skin dose due to hot particle contamination if found.

Pakistan

Problem of significant increase in radiation fields in Primary Heat Transport (PHT) system was studied in depth and was also discussed with IAEA experts. It revealed that the problem occurred due to presence of some fuel bundle(s), with minor defect (porosity), in core for long duration. The bundle(s) could not be located since porosity was too small to detect, and due to unavailability of failed fuel detection system. The presence of defective fuel bundle in the core causes the gradual increase in radiation field in the system. Later these bundles were discharged from the core as per fuel management program and decreasing trend in radiation field was observed. Now the fuel failure detection system has been refurbished and it will be able to identify the defective fuel bundle well in time preventing such happening in future. Also chemical decontamination of one of the steam generator has been performed on experimental basis. Now the radiation field in the heat transport system has returned to the normal.

Romania

These activities are being controlled by the existing radiation protection procedures, by the generic work safety procedures and based on the existing results and experience this issue is considered not to be a generic safety one from our perspective.

ADDITIONAL SOURCES:

ISSUE TITLE: Management of Tritium (RP 2)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

Management of tritium is generic to PHWR's. Although tritium is produced in much greater yield in PHWR's than in other reactor types, effective methods of control have been developed. These methods include source containment and, for tritiated water, the main source, collection on dryers. Care and attention needs to be applied to measures for tritium control, barriers and work processes for handling tritium.

Tritium emissions are also probably the major source of public critical group exposure around PHWR facilities, but the doses are small, typically less than 1 % of the regulatory limit of 1 mSv per year. Worker tritium doses are routinely less than 10% of the occupational dose limit. Although plants do suffer source control failures from time to time, from a radiation protection perspective management of the potential safety problems associated with tritium has been very successful.

Safety Significance

Large sources of tritium are generated in operating PHWR reactors, and present in the environments of the reactors, and associated waste facilities. The sources are not always present in the well-understood form of tritiated water. If these sources are not managed well, significant whole body exposure of workers, and the public as a result of emissions, may occur.

Source of issue

- ___xx_____ operational experience
- __________ deviation from current standards and practices
- __________ potential weaknesses identified by deterministic or PSA

MEASURES TAKEN BY MEMBER STATES:

Canada

At the Darlington site, OPG has been operating a tritium removal facility successfully for many years. It is designed for a steady state extraction rate of not less than 8.0 x 10^6 Curies of tritium in 7000 hours of operation per year, from a feed of 3.4 Curies per kilogram D_2O. At steady state operation, the return stream has 0.28 Curies per kilogram D_2O; the extraction rate is therefore 1143 Curies per hour at a feed rate of 366.3 kilograms per hour.

India

De-tritiation has been done at one of the units at MAPS and heavy water swapping (substituting highly tritiated heavy water by less tritiated heavy water) in one of RAPS Units.. For some time the regulatory body became prescriptive and imposed restrictions on DAC level in shutdown accessibleareas to reduce shutdown uptake. A heavy water task force was set up and implementation of its recommendations resulted in considerable improvement. Stations have special plans/ checks/checklists to reduce DACs during operation as well as prior to starting outage jobs. AERB has asked all plants to prepare plans for reduction of tritium in PHT water on priority and Moderator System later , which are being implemented.

Korea, Republic of

KEPCO decided to design and construct a Tritium Removal Facility (TRF) and submit a plan for the tritium separation and process technique. This facility is scheduled to operate in 2006. The design target of the (TRF) is to keep a tritium concentration in the moderator of 10Ci/kg-D2O, and in the coolant of 0.5Ci/kg-D2O. The processing capacity of the TRF would be about 120kg-D2O/hr.

Detailed design, construction and pre-operation will be conducted from 2001 until 2005.

Romania

There are researches going on this topic and evaluations are being performed in extending their scale and connecting to the plant needs. However the licensing process is in a beginning phase and results will be reported by the time significant conclusions will be reached.

ADDITIONAL SOURCES:

4.2.7 Fuel handling (FH)

ISSUE TITLE: Damage to fuel during handling (FH 1)

ISSUE CLARIFICATION:

Description of issue

This issue is also applicable to NPPs with LWR.

Fuel bundles are loaded in PHWR's on a frequent basis by on-line fueling. On-line fueling is controlled remotely by computer-based fuel handling systems. Similarly, transfer operations of discharged irradiated fuel bundles to the spent fuel storage bays is performed by remotely controlled transfer mechanisms in which the bundles remain flooded by water and are not placed in significantly elevated positions during the transfer operations. The fuel handling systems have proved to be generally reliable. However, there have been instances involving damage to fuel.

Safety significance

If the cladding of fuel is damaged, fission products may be released and pose a radiological hazard to workers.

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:

Argentina

Some damages that affected fresh fuel occurred due to falls or crashes during inspection activities. Nevertheless, there were no damages to fuel sheets. In the case of spent fuel, during handling activities within the pool, some fuel sheet damages due to the handling effect were observed when they were being arranged into the baskets of the dry storage system. Nevertheless; were not observed sheet breaks and fuel structural compounds breaks.

Canada

See Issue RC 2

China

India

Operating experience shows that mechanical failures of spent fuel does happen once in a while due to improper operation in fuel transfer system or handling in bay. Failures which have released radioactivity in fuel transfer system enclosures or in storage pool have been very limited. A standing task force of experts reviews each off-normal operation of fuel handling to improve design or operating procedures to avoid re-occurrence of incident at any station. Checks on fresh fuel are being carried out before loading in core.

Korea, Republic of

See issue IH 7

Romania

This is a typical CANDU aspect considered already in the basic design, i.e the management of the fuel inside the plant, and for this issue with a focus on the fuel damages. Even if the fuel defect rate is very low and well under the required by design and licensing basis limits, there is a permanent concern to assure compliance with a well defined barrier in fuel handling in the plant, both while in the reactor and during refuelling activities. There are actins going on under the Strategic Policy for relicensing unit 1, which are related to this topic. However by the time evaluations of the results will indicate significant aspects they will be reported for the update of this document.

ADDITIONAL SOURCES:

ABBREVIATIONS

AC	alternating current
AF	auxiliary feedwater
AFAS	auxiliary feedwater actuation system
AFI	area for improvement
AFW	auxiliary feedwater
AFWS	auxiliary feedwater system
ALARA	as low as reasonably achievable
AM	accident management
AOM	abnormal opearting manual
AOT	allowed outage times
APOI	abnormal plant opearting procedure
ARA	application for renewal of authorisation
ASCOT	assessment of safety culture in operation teams (IAEA)
ASDV	atmospheric steam discharge valve
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
BDBA	beyond design basis accident
BIT	boron injection tank
BNCS	ASME Board of Nuclear Codes and Standards
BWST	borated water storage tank
CCI	common cause initiators
CCF	common cause failure
CCWS	component cooling water system
CDF	core damage frequency
CFR	Code of Federal Regulations (USA)
CHRS	containment hydrogen recombiner system
CIS	Commonwealth of Independent States
CLG	coolant level gauge
CMMS	computerised maintenance management system
COG	CANDU owners group
COIS	computerised operator information system
CR	control rod
CRD	control rod drive
CRDM	control rod drive mechanism
CVCS	chemical and volume control system
DAC	derived air concentration
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
ECCS	emergency core cooling system
ECR	emergency control room
ECT	eddy current testing
EDG	emergency diesel generator
EFW	emergency feedwater
EFWS	emergency feedwater system
EMTR	emergency transfer

EOP	emergency operating procedure
EOS	electronic overspeed system
EPR	European pressurized reactor
EQ	environmentally qualified
ERC	emergency response centre
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
FME	foreign material exclusion
FW	feedwater
FWD	feedwater distribution
FWFCS	feedwater flow control system
FWS	feedwater system
GCB	generator circuit breaker
GDC	general design criteria (USA)
GSI	generic safety issue
GSS	guaranteed shutdown state
GT	generator transformer
HDS	historical data storage
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
IFBA	in-fuel burnable absorber
IGA	intergranular attack
IGSCC	intergranular stress corrosion cracking
IN	information notice (USNRC)
INSAG	International Nuclear Safety Advisory Group (IAEA)
IPE	individual plant evaluation
IPEEE	individual plant examination of external event
IPERS	international peer review service (IAEA)
IRS	incident reporting system (IAEA)
ISI	in-service inspection
KSNP	Korean standard nuclear plant
LBB	leak before break
LB LOCA	large break LOCA
LCO	limiting conditions of operation
LOCA	loss of coolant 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
MIV	main isolation valve
MORT	management oversight and risk tree
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
NOP	neutron overpower protection
NPP	nuclear power plant
NRPDS	Nuclear Plant Reliability Database System
NUSS	nuclear safety standards (IAEA)
OBE	operation basis earthquake
OL&C	operational limits and conditions
OPEC	emergency operating procedure (India)
OPG	Ontario Power Geneartion
OST	overspeed trip
OSART	Operational Safety Review Teams (IAEA)
OTO	order to operate
PAMI	post accident monitoring instrumentation
PAMS	post accident monitoring system
PDCS	programmable digital comparator system
PIE	postulated initiating event
PCV	primary containment venting
PGP	procedure generation package
PHWR	pressurised heavy water eeactor
PHT	primary heat transport
PISC	programme for inspection of steel components
PLC	programmable logic control
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 (Russian 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
ROP	regional overpoewr protection
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
SCA	secondary control area
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
SLB	steam line break
SLCS	standby liquid control system
SOE	safe operating envelope
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
SSS	secondary shutdown state
STA	shift technical advisor
TACIS	technical assistance to the Commonwealth of Independent States
TS	technical specifications
TMI	Three Mile Island (USA)
TOE	technical operability evaluation
TRF	tritium removal facility
TSC	technical support center
UPS	uninterrupted power supply
UT	ultrasonic testing

VDU	video display unit
VHP	vessel head penetration
WANO	World Association of Nuclear Operators

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