IAEA-TECDOC-1418

Implications of power uprates on safety margins of nuclear power plants

Report of a technical meeting organized in cooperation with the OECD/NEA held in Vienna, 13–15 October 2003

IAEA

AEN NEA

IAEA

International Atomic Energy Agency

September 2004

IAEA SAFETY RELATED PUBLICATIONS

IAEA SAFETY STANDARDS

Under the terms of Article III of its Statute, the IAEA is authorized to establish or adopt standards of safety for protection of health and minimization of danger to life and property, and to provide for the application of these standards.

The publications by means of which the IAEA establishes standards are issued in the **IAEA Safety Standards Series**. This series covers nuclear safety, radiation safety, transport safety and waste safety, and also general safety (i.e. all these areas of safety). The publication categories in the series are **Safety Fundamentals, Safety Requirements** and **Safety Guides**.

Safety standards are coded according to their coverage: nuclear safety (NS), radiation safety (RS), transport safety (TS), waste safety (WS) and general safety (GS).

Information on the IAEA's safety standards programme is available at the IAEA Internet site

http://www-ns.iaea.org/standards/

The site provides the texts in English of published and draft safety standards. The texts of safety standards issued in Arabic, Chinese, French, Russian and Spanish, the IAEA Safety Glossary and a status report for safety standards under development are also available. For further information, please contact the IAEA at P.O. Box 100, A-1400 Vienna, Austria.

All users of IAEA safety standards are invited to inform the IAEA of experience in their use (e.g. as a basis for national regulations, for safety reviews and for training courses) for the purpose of ensuring that they continue to meet users' needs. Information may be provided via the IAEA Internet site or by post, as above, or by e-mail to Official.Mail@iaea.org.

OTHER SAFETY RELATED PUBLICATIONS

The IAEA provides for the application of the standards and, under the terms of Articles III and VIII.C of its Statute, makes available and fosters the exchange of information relating to peaceful nuclear activities and serves as an intermediary among its Member States for this purpose.

Reports on safety and protection in nuclear activities are issued in other publications series, in particular the **Safety Reports Series**. Safety Reports provide practical examples and detailed methods that can be used in support of the safety standards. Other IAEA series of safety related publications are the **Provision for the Application of Safety Standards Series**, the **Radiological Assessment Reports Series** and the International Nuclear Safety Group's **INSAG Series**. The IAEA also issues reports on radiological accidents and other special publications.

Safety related publications are also issued in the **Technical Reports Series**, the **IAEA-TECDOC Series**, the **Training Course Series** and the **IAEA Services Series**, and as **Practical Radiation Safety Manuals** and **Practical Radiation Technical Manuals**. Security related publications are issued in the **IAEA Nuclear Security Series**.

IAEA-TECDOC-1418

Implications of power uprates on safety margins of nuclear power plants

Report of a technical meeting organized in cooperation with the OECD/NEA held in Vienna, 13–15 October 2003

IAEA

AEN
NEA

IAEA
International Atomic Energy Agency

September 2004

The originating Section of this publication in the IAEA was:

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

IMPLICATIONS OF POWER UPRATES ON SAFETY MARGINS OF
NUCLEAR POWER PLANTS
IAEA, VIENNA, 2004
IAEA-TECDOC-1418
ISBN 92–0–112004–4
ISSN 1011–4289

© IAEA, 2004

Printed by the IAEA in Austria
September 2004

FOREWORD

The safety of nuclear power plants (NPPs) is based on the defence in depth concept, which relies on successive physical barriers (fuel matrix, cladding, primary system pressure boundary and containment) and other provisions to control radioactive materials and on multiple levels of protection against damage to these barriers. Deterministic safety analysis is an important tool for conforming the adequacy and efficiency of provisions within the defence in depth concept and is used to predict the response of an NPP in predetermined operational states. This type of safety analysis applies a specific set of rules and specific acceptance criteria. Deterministic analysis is typically focused on neutronic, thermohydraulic, radiological and structural aspects, which are often analysed with different computational tools.

The advanced computational tools developed for deterministic safety analysis are used for better establishment and utilization of licensing margins or safety margins in consideration of analysis results. At the same time, the existence of such margins ensures that NPPs operate safely in all modes of operation and at all times.

To properly assess and address the existing margins and to be able to take advantage of unnecessary conservatisms, state of the art analytical tools intended for safety assessment have been developed. Progress made in the development and application of modern codes for safety analysis and better understanding of phenomena involved in plant design and operation enable the analysts to determine safety margins in consideration of analysis results (licensing margins) with higher precision. There is a general tendency for utilities to take advantage of unnecessarily large conservatisms in safety analyses and to utilize them for reactor power uprates, better utilization of nuclear fuel, higher operational flexibility and for justification of lifetime extension.

The present publication sets forth the results of a Technical Meeting on the Implications of Power Uprates for the Safety Margins of Nuclear Power Plants, which was organized in co-operation with the Nuclear Energy Agency of the OECD and was held in Vienna, 13–15 October 2003. At this meeting, specific topics relating to the utilization of safety margins for NPP power uprates were presented and discussed.

The IAEA officer responsible for this publication was M. Dusic of the Division of Nuclear Installation Safety

EDITORIAL NOTE

The papers in these proceedings are reproduced as submitted by the authors and have not undergone rigorous editorial review by the IAEA.

The views expressed do not necessarily reflect those of the IAEA, the governments of the nominating Member States or the nominating organizations.

The use of particular designations of countries or territories does not imply any judgement by the publisher, the IAEA, as to the legal status of such countries or territories, of their authorities and institutions or of the delimitation of their boundaries.

The mention of names of specific companies or products (whether or not indicated as registered) does not imply any intention to infringe proprietary rights, nor should it be construed as an endorsement or recommendation on the part of the IAEA.

The authors are responsible for having obtained the necessary permission for the IAEA to reproduce, translate or use material from sources already protected by copyrights.

CONTENTS

1. INTRODUCTION

The general strategy for operating a nuclear power plant safely is to continuously implement improvements to plant hardware, plant processes and operation. Plant improvements are in many cases related to optimization, which usually involves the use of margins in plant design, safety analysis and plant operation. However, the use of margins should be balanced with adequate ‘margin generation’, i.e. margins that are obtained by using less conservative calculations.

The greater demand for electricity and the available capacity in safety margins in some of the operating nuclear power plants are prompting nuclear utilities to request licence modifications to enable operation at a higher power level, beyond the provisions of the original licence. Such plant modifications require an in-depth safety analysis to evaluate the possible safety impact. The analysis must consider the core characteristics and the plant behaviour, taking into account the capability of the structures, systems and components, the reactor protection and safeguard systems set points.

Although the emphasis of the safety analysis is on a deterministic evaluation that safety criteria are met, concerns relating to overall plant safety and the risk impact of power uprates also require that the safety margins be evaluated with probabilistic safety analysis (PSA) methods. This is necessary to support and supplement deterministic analysis, technical judgement and to enable risk informed decision making process.

Not only power uprates have potential impact on safety but also other design changes, on fuel, structures, systems and components may influence the same safety parameters and associated margins.

Currently a number of nuclear utilities are planning power uprates for their nuclear reactors and many of them have already gone through this modification process. Generally, smaller power uprates, up to 2% can be achieved by implementing enhanced techniques for calculating reactor power. This involves the use of more precise feedwater flow measurements, which, in turn, provide for a more accurate calculation of power. Greater power uprates, (up to 7%) usually involve changes to instrumentation set points, but still do not require major plant modifications. Extended power uprates that could go up to 20% of the nominal power may require significant modifications, to major balance-of-plant (BOP) equipment.

The maximum thermal power level of a plant is included in the licence and technical specifications for the plant. Any changes must be approved by the regulatory body and therefore the licensing analyses that demonstrate the safety of the plant must be performed when planning the power uprate. The essential part of such analyses is the demonstration that the plant structures, systems and components can support safe operation after the power uprate and/or associated plant modifications, and that the results of the safety analysis remain within regulatory limits.

In January 2003, the IAEA published Safety Margins of Operating Reactors, IAEA-TECDOC-1332, which specifically discusses safety margins and their implications for decision making at plants. The publication addresses the capabilities of thermal-hydraulic computer codes, methods for safety margin evaluation and how safety margins can be utilized in the operation of nuclear power plants and when modifications are considered for them. IAEA-TECDOC-1332 was used as a basis for a technical meeting held between 13 and 15 October 2003.

In 2003, the Nuclear Energy Agency of the OECD (OECD/NEA) established a multidisciplinary action plan in the area of safety margins. The OECD/NEA decided to undertake this task in response to questions raised by safety authorities, as the industry demands for power uprates, longer operating cycles, new fuel design and increased fuel burnup required an integrated assessment of the impact of these multiple modifications on safety margins.

The important objective of the meeting was also to provide an international forum for presentation and discussion on topics related to the impact of power uprates on plant safety margins. The meeting also provided an opportunity to exchange information on national practices and experiences in the field of design and operational improvements based on utilization of margins. The national presentations are contained in the Annex at the end of this report.

The meeting also discussed and elaborated on the following topics, which are summarized in the next three chapters:

(1) **Terms and definitions for different type of margins - (design margin, licensing margin, analytical margin, safety margin, operational margin)**

(2) **Treatment of safety margins** (margin generation and margin usage) **and role of different institutions** (NPP, regulator, designer)

(3) **Power uprates and the impact on plant safety margins**

- small uprates, based on improved calculation techniques (<2%)
- greater uprates, requiring equipment change (<7%)
- extended uprates, involving major plant modifications (up to 20%).

2. TERMS AND DEFINITIONS FOR DIFFERENT TYPES OF MARGINS

An illustration of the different types of margins is given in Figures 1 and 2. The following definitions were developed by the experts for the purposes of the Technical Meeting mentioned. They are listed, according to Fig. 1, from top to bottom.

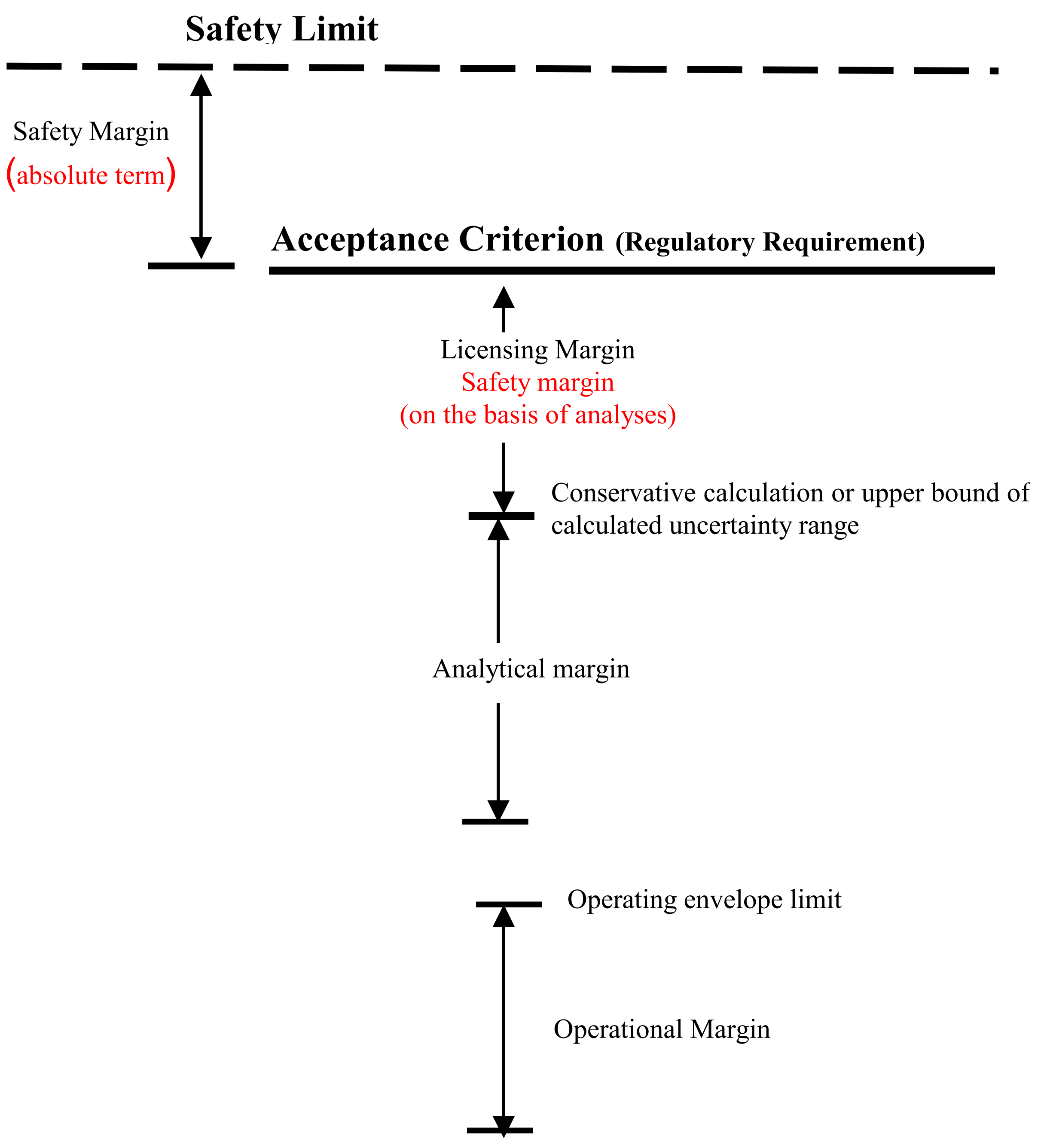


FIG. 1. The various types of margin for a nuclear power plant.

Safety limit

The safety limit is a critical value of an assigned parameter associated with the failure of a system or a component (e.g. loss of coolable core geometry).

Safety margin (absolute terms)

The safety margin is the distance between an acceptance criterion and a **safety limit**. If an **acceptance criterion** is met, the available safety margin is preserved.

Remark: The definition of "safety margin" in Refs [1] and [2] is changed to "licensing margin or safety margin only in consideration of the results of analyses" in this report. The reason is that a safety margin should be maintained, a licensing margin can be zero without reduction of safety.

Acceptance criterion

The acceptance criterion is the quantitative limitation of selected parameter or qualitative requirement set-up for the results of accident analysis. Specified bounds on the value of a functional or condition indicator used to assess the ability of a system, structure or component to perform its design function [1].

Licensing margin or Safety margin (in consideration of the results of analyses)

Licensing margin is the difference, in physical units, between a threshold that characterizes an **acceptance criterion** and the result provided by either a **best-estimate calculation** or a **conservative calculation**. In the case of a best-estimate calculation, the uncertainty band must be taken into consideration.

Remark: The definition of licensing margin is taken from Ref. [1] instead of "safety margin (in consideration of the result of analyses)". The name is changed to "licensing margin or safety margin in consideration of the results of analyses" because "safety margin" is defined here as the distance to a safety limit, e.g. loss of coolable core geometry. The licensing margin may be zero when the accuracy of a safety calculation is warranted, however, a safety margin should be maintained.

Analytical margin

An analytical margin contains an estimate of individual modelling or overall code uncertainties, representation uncertainties, numerical inadequacies, user effects, computer / compiler effects and data uncertainties on the analysis of an individual event. This shall be determined either by **conservative calculation** or by **best-estimate calculation** plus uncertainty evaluation.

Operational margin

The operational margin implies states defined by **operational limits and conditions**. This includes measurement accuracy (for example, measuring feedwater flowrate, feedwater temperature, steam quality) and controller ranges and tolerances.

Operational limits and conditions

A set of rules setting forth parameter limits that ensure the functional capability and the performance levels of equipment for safe operation of a nuclear power plant, approved by the regulatory body [1].

Design margin

The plant shall be designed to operate safely within a range of parameters (for example of pressure, temperature, power), and a minimum set of specified support features for safety systems (for example, auxiliary feedwater, capacity and emergency power supply) shall be assumed to be available. A set of design limits consistent with the key physical parameters for each structure, system or component shall be specified for **operational states** and **design basis accidents**.

Safety relevant systems and components are designed on the basis of design-basis accidents, which are in turn defined in line with the current guidelines. The design of a component (e.g. flow area of a valve, discharge rate of a pump, water supply) is done in such a way so as to fulfil the process engineering (mechanical) requirements (e.g. over pressure protection of a system, guarantee of a shutdown gradient and a minimum water level in a tank, a sufficient water supply to ensure grace time). For this, the necessary minimum value needed for reaching the process engineering (mechanical) requirements is raised by an additional **design margin**. This addition can be given in the standards and/ or be specified by the designer.

=> Design value = minimum value for the fulfilment of the process engineering (mechanical) requirement + design margin

The limits of design margin are not related to any previously defined limit and margin.

Operational states

States defined under normal operation or **anticipated operational occurrences** [1].

Anticipated operational occurrence

An operational process deviating from normal operation which is expected to occur once or several times during the operating lifetime of the power plant but which, in view of the appropriate design provisions, does not cause any significant damage to items important to safety nor lead to accident conditions [1].

Design basis accident (DBA)

Accident conditions against which a nuclear power plant is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits.

Figure 2 depicts the design margin. It is shown in a separate figure because no reference is in relation to limits and margins explained in Figure 1.

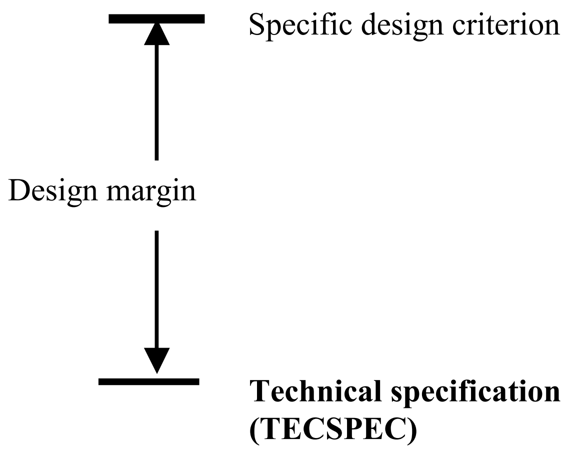


FIG. 2. Illustration of design margin.

Conservative calculation
Calculation leading to pessimistic results relative to specified acceptance criteria.

Best-estimate calculation
Calculation, which is free of deliberate pessimism regarding selected acceptance criteria. It uses best-estimate code and includes uncertainty analysis.

Power uprates
The power uprate is the operation beyond the power level originally licensed by the regulatory body.

3. TREATMENT OF LICENSING MARGINS AND ROLE OF DIFFERENT ORGANIZATIONS

Two subjects are treated in this chapter:

(1) The role of different organizations involved in the licensing process taking as example the power uprate of a reactor

(2) The generation of licensing margins or safety margins on the basis of the analyses.

3.1. ROLE OF DIFFERENT ORGANIZATIONS

Participating organizations at the meeting described the licensing process in their own countries. From those presentations, it appeared that:

- The licensing process is rather similar from one country to another, with some specific differences
- Many aspects are similar:
  - the participating organizations
  - the role of each organization that is involved
  - the interface among organizations during the licensing process
- a close and an open co-operation among all organizations is very important at each step of the process.

It was emphasized that regular meetings, progress meetings are essential to the success of the process, to assure the adequate attention is given to safety.

The involved organizations and their role in case of a power uprate were also discussed and the main conclusions are as follows:

- The utility that operates the plant is responsible for the safety of the installations. It is the responsibility of the operator to submit the application for a major modification of the plant to the Safety Authority and to demonstrate that this modification is feasible, keeping sufficient licensing margins or safety margin in consideration of the results of analyses.
- The architect/engineer can act on behalf of the Utility and/or the Nuclear Steam Supply System (NSSS) supplier can support the Utility in the safety demonstration. They establish the safety study programme, they perform a part or all of the safety analysis, they evaluate the licensing margins or safety margin on the basis of the analyses, they can propose other hardware modifications, they describe the implementation programme etc.

  They assist and help the Utility in their discussions with the Safety Authorities.
- The Safety Authority provides state supervision over the nuclear and radiation safety, specifies safety requirements, checks their fulfilment, reviews submitted documentation, and issues the licence.

  They review and comment on all the safety studies, the implementation programme, the modifications of set points, the update of Final Safety Analysis Report (FSAR) and

Technical Specification They verify the implementation phase including inspections. Finally they give or refuse the authorization.

- The Safety Authority is assisted by a technical support organization (TSO), which can be inside or outside of the regulatory organization. The TSO makes the review and comments on the safety analysis, using its own or acquired calculation tools.

In conclusion, it appears that the two main organizations in a licensing process are:

- The Utility, which has to demonstrate that the plant can be operated after the modifications with an adequate safety level (the adequate safety level may be slightly different from one country to another).
- The Safety Authority, which has to evaluate this demonstration with the help of technical support organizations and approve or reject the utility application.

3.2. GENERATION OF LICENSING MARGINS AND THEIR USAGE

The licensing margins or safety margin in consideration of the results of analyses (defined conceptually as the value of difference between regulatory acceptance and upper bound of design basis occurrences) can be increased in various ways. These are categorized as follows:

- Modification of operation

  The possible ways to increase licensing margin during operation are:

  — To reduce operational flexibility (e.g. limitation to the application of load follow operation)

  — To reduce core loading flexibility (e.g. loading pattern where new fuels are loaded outside region of the core)

  — To modify set points to decrease operation area.
- Relaxation of safety limits.
- The introduction of up to date knowledge on failure mechanism may be able to reassess the safety limits and increase the safety margin.
- Introduction of advanced methodologies.
- This includes but not limited to the use of more sophisticated tool such as 3-D calculation in reactivity initiated accidents (RIA) analysis or coupled codes), use of best-estimate (BE) analysis and statistical uncertainty estimate or reassessment of instrumentation error band.
- Introduction of advanced hardware.
- This includes but is not limited to the introduction of advanced fuel where the safety margins in absolute term are increased.

  Usually, generation of margins requires R&D efforts.

The increased licensing margin can be used for more economical fuel management, increase of plant operation flexibility, plant maintenance and surveillance optimization and compensation of ageing effects life extension and to compensate for emerging safety issues other than power increase.

4. POWER UPRATES AND THE IMPACT ON PLANT MARGINS

4.1. TERMINOLOGY

The terms associated with this subject are addressed in Section 2. For discussions in this chapter, the margins that are addressed are defined as the difference between the acceptance criterion (the regulatory requirement) and the conservative calculation on the upper bound of the calculated uncertainty, if best estimates are used. This corresponds to the "licensing margin" as defined in Chapter 2.

4.2. OBJECTIVES OF POWER UPRATINGS

The objectives for power uprating vary significantly between countries. They may be broadly categorized as follows:

- Maintaining the power level. For some reactors significant maintenance and inspection programs are required to maintain the reactor at full power and the reason could be seen as a way of counteracting ageing of the plants. An example of this is presented for older CANDU reactors (see related paper in the Annex). In this case, the pressure tube degradation caused by creep has been the limiting factor. Changes to new improved fuel will reverse the decreasing power trends for some years. Eventually replacement of the pressure tubes is necessary to support a power level near 100 %, and consider potential power uprates in these reactors. The problem of maintaining full power was worsened in some older reactors by the fact that the importance of the void reactivity coefficient had been underestimated in the development process. Implementation of low void reactivity fuel and other improvements that are planned in these reactors will restore the reactor operation to the design fuel power level and enable potential power uprates in future.
- Improvement of fuel design. Advanced fuel usually gives improved fuel economy. By modern fuel, low leakage loading patterns and coast down operational strategy, increased enrichment, and applying modern analysing techniques, the number of fuel assemblies needed for a reload can be significantly reduced thereby reducing the fuel cost and potentially the cost for storage of used fuel can be reduced.

  However the increased discharged burnup may require longer cooling time before transportation and/or intermediate storage e.g. which also should be taken into consideration.

  Improved fuel also requires to compensate the impact of increased operational margins on fuel reliability during normal operation.
- Increased reactor power, which is a general objective of the improved methodologies.

There is a need for flexibility in reactor operation. In a deregulated electricity market it is of importance to produce much electricity when the demand is high and to keep the flexibility to safe cost when the demand is low. There is also a need to make use of extra margins gained by backfitting and safety improvements done already for other purposes, like less conservative boundary conditions for computer code calculations, changes of equipments, etc. Replacement of equipments can also be required for lifetime extension and new hardware should be optimized for possible higher power level.

4.3. METHODS AND STRATEGIES FOR POWER UPRATES

The definition of power uprates is the operation beyond the power level licensed by the regulatory body.

Generally the smaller power uprates (less than approximately 2%) can be achieved through improved operational performance and analytical tools such as the improved performance of plant equipment both on primary and secondary side, protection and monitoring system, operator performance, etc and also by improved state of the art analysis codes and the technical insights, without compromising licensing margins.

The greater power uprates (less than approximately 7%) may require significant hardware changes such as refurbishment or replacement of equipment contributing considerably to power uprates without violating any regulatory acceptance criteria. A detailed cost- benefit analysis needs to be performed, considering implications on various aspects such as safety analysis both deterministic and probabilistic, handling of additional waste, spent fuel storage facility or reprocessing, environmental impact, etc.

The extended uprates, up to 20%, may be limited by critical reactor components like reactor vessel, pressurizer, primary heat transport systems, piping etc., or secondary components like turbine or main generator.

Timing of power uprates generally for small power uprates can be considered well before the plant reaches the plant life. This will not require refurbishment or replacement of very capital intensive equipment and operator readjustment to the new operating procedures consequent to the replacement. Also return on investment by the utility will not be affected.

The greater power uprates may be worth consideration when the plant is due for regular periodic safety assessment or the end of its design life, and the utility is looking for life extension. This may go well in line with cost benefit consideration.

The power uprates proposal needs to be well supported by the safety analyses. The analytical methods that could be followed are basically the following:

- Conservative codes using conservative models, and calculations using conservative initial and boundary conditions
- Best-estimate codes and conservative initial and boundary conditions
- Best-estimate codes and uncertainty analysis.

In conservative analyses, however, factors have to be considered as required by the regulatory body. This may include single failure criteria; supplementary failure considerations such as failure to scram; failure of power grid; discrediting or crediting operator actions beyond certain available time, etc.

4.4. REGULATORY REQUIREMENTS ON PRESERVATION OF MARGINS

The regulatory practice varies from country to country. To obtain a consensus between different countries is not easy. The regulatory positions may broadly be categorized under the following items:

- The current acceptance criteria and absolute safety margins should be preserved
- Current acceptance criteria should be fulfilled, licensing margins are allowed to become smaller.

There is no obvious method to obtain consensus between different countries. An integration of deterministic and probabilistic approach could provide the basis for decision making if modifications are acceptable; a permission is only granted when these goals are fulfilled:

(1) Acceptance criteria are met.

(2) Licensing margins and/ or safety analysis are acceptable.

Another approach put forward by several countries during the Technical Meeting was the acceptance of a limited risk increase in the short term, while risk in the long term is continuously decreased.

One of the reasons to address the safety and licensing margins is the effect of cumulative changes in the plants. Small changes that do not in themselves warrant an in-depth safety review may accumulate over the years and the plant conditions may prove to be outside the scope of the safety analysis report. Integral assessment of the impact of all changes is recommended.

4.5. WHERE LICENSING MARGINS CAN BE GAINED

The most profitable part is to gain margins for power uprate without having to make any hardware changes. These are termed “small” changes. Beyond the small changes, the cost-effective “hardware” changes will be required to gain margins by enhancing the plant power limiting equipment capability.

Small changes may include the reassessment of safety margin in consideration of the results (licensing margin) given by the difference between acceptance criteria and analytical results, reassessment of initial conditions for accident analysis, reflection of operating experiences to reduce the unnecessary uncertainties or trade-off of power peaking factor and operational flexibility such as load follow operation and improvement of accident analysis methods, in which the improved accident analysis methods are included. Further developments may be possible depending on the current licensing situation of the plant.

Small changes:

- Reduce uncertainties and take advantage by uprating.
- Code development, validation and assessment to remove uncertainties and perform realistic predictions.
- Combine PSA and deterministic methods to gain insight in the available plant safety margins.
- Operator training, difficult to assess and quantify margin, not accepted as a source of margins everywhere.
- Operating experience feedback to make more realistic assumptions and reduce unnecessary over-conservatisms.

- Utilize advanced diagnosis of potential abnormal unsafe events to prevent incidences.
- Continuously improve operating staff training, to improve plant performance and better handling of unsafe events.
- Trade-off operational flexibility to gain margin.
- Develop on-line capability to assess available licensing margins and utilize opportunity to increase power level.
- Apply most updated state-of-art technologies to reassess the available margins.
- Perform timely maintenance to prevent potential reduction of margins.
- Optimise fuel loading schemes to flatten the power shape during operation to gain power increase.

In general, extended changes usually require an additional investment to replace the limiting equipments. Therefore, these changes should be carried out based on in-depth evaluation of economical assessment and strategy of plant operation through the plant life. In other words, it should be a good timing of uprate when replacement of heavy equipments such as steam generators, pressure tubes and turbine rotors are implemented.

Hardware changes:

- Dependent on the reactor type.
- Replace limiting equipment to increased capacity equipments.
- Replace limiting equipments to better material and technology.
- Replace limiting equipment to new design and fabrication methodologies.
- Gain power increase by application of advanced fuel.
- Perform timely upgrade of hardware.
- Install state-of-the-art instrumentations to decrease uncertainties.
- Perform regular calibration and maintenance of instrumentations to improve measurement reliability.

4.6. ACTUAL LIMITS FOR POWER UPRATES

Limits for a safe and economic power uprate can be found in 3 areas:

(1) Technical facts;
(2) Economic facts;
(3) ‘Soft’ facts.

To find out the technical limits, the influence of the power uprate on all systems and components during normal operation and transients / accidents has to be evaluated.

Aspects to consider are for example:

- Plant operation and reactor core (e.g. operational point at the power flow map);

- Analysis of transients and accidents;
- Reactor pressure vessel (RPV) and RPV-Internals;
- Steam feedwater cycle;
- Electrical systems;
- Safety systems;
- Operational systems.

This evaluation can result for each system / component in a certain limit of the achievable power level, where modifications are required, or where no higher power is possible.

Economic limits can be:

- The demand for electrical power;
- The cost for analyses and hardware modifications;
- Required outage time for the modifications;
- Need for replacement of components or performing of analyses due to other reasons (ageing, lifetime extension, new safety requirements);
- Specific cost in comparison with other sources.

Soft facts which can be limiting are for example:

- Environmental aspects, storage of waste fuel etc.;
- Capacity and capability of the utility, vendor and regulator;
- Licensing risks;
- Public opinion.

4.7. OTHER CONSIDERATIONS

Other considerations when performing power uprates are for example:

- Communication of power uprates with the public;
- Quality assurance considerations.

Power uprates require significant re-evaluation of the plant design basis and as a consequence restoration of know-how. Involvement of plant, vendor, and regulator personnel increase (restores) overall knowledge. The total effect is a contribution to the increase of plant safety.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Accident Analysis for Nuclear Power Plants; Safety Reports Series No. 23, IAEA,Vienna (2002).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Margins of Operating Reactors, Analysis of Uncertainties and Implications for Decision Making, IAEA-TECDOC-1332, Vienna (2003).

ANNEX

PAPERS PRESENTED AT THE TECHNICAL MEETING

POWER TRANSIENTS SIMULATIONS IN ATUCHA 1 NUCLEAR POWER PLANT FOR MIXED NATURAL AND SLIGHTLY ENRICHED URANIUM CORE

M. Higa, J. Villar, R. Perez, J. Piñeyro
Nucleoeléctrica Argentina S.A.,
Buenos Aires, Argentina

O. Serra, A. Blanco
Comisión Nacional de Energía Atómica,
S.C. de Bariloche, Argentina

Abstract

The Atucha I Nuclear Power Plant, is a pressure vessel type reactor (357 MWe), moderated and cooled with heavy water, originally designed by Siemens (Germany) to use natural uranium (NU) as fuel. It was converted to slightly enriched uranium (SEU) 0.85 w% between 1995 and 2000 to increase the fuel exit burn-up and reduce the fuel cost. This change also increased the flattening of the axial power distribution with a reduction of the maximum linear power, increasing related safety margins. However, there was a concern that this increase of flattening of the power distribution might also increase the susceptibility to xenon induced power oscillations. These oscillations were considered in the original design and partial absorber steel rods were included to damp them. However, the operating experience showed that the amplitude of the oscillations was smaller than expected. With NU fuel the amplitude of oscillations (top to bottom) induced by the movement of the regulating control rods was higher during power reductions but it was not necessary to use the partial rods. In this work, several simulations were done and consisted in a reduction from 100% to 80% of full power (FP), during one hour, and staying on that level for about two days, with a subsequent recovery to FP with a 2%/h rate. The core behaviour was analysed when the number of SEU fuel elements in the core was increasing and verifying that the channel and linear powers were below the operation limits. Calculations were also done in homogeneous SEU cores. Those calculations were done using the WIMS cell code and the PUMA ·3D diffusion code developed in Argentina. The reactor model included spatially dependent Xe135 and fuel and coolant temperature. For the mixed cores calculations revealed mitigated flux oscillations, with decreasing amplitude and the damping coefficient of the amplitude oscillations decreased with the SEU fuel element number. This was only noticeable in the 80% operation step, without affecting the maximum linear power. In the case of SEU homogeneous core, the amplitude of the flux oscillations registered an increase slightly greater than mixed cores. The linear powers were always below the limit (600W/cm). At the present time the plant is operating with a full SEU core without operating problems due to xenon induced power oscillations.

1. INTRODUCTION

Atucha I (CNA1) is a nuclear power station, pressure vessel type designed by Siemens (Germany), moderated and cooled with heavy water. Fuel assemblies were originally 36 active natural uranium UO_2 rod vertical cluster (5.3 m long).

In the beginning the total thermal power was 1100 MW, at the end of 1977 this power was increase to 1179 MW.

Power regulation is made through 3 hafnium rods (usually called R2 black bank) and 3 steel rods (RG grey bank). Additional 21 hafnium rods are used for shutdown purposes.

Since 1995 up to 2000 it was gradually converted to slightly enriched uranium 0.85 w% (SEU). The main benefit we obtained using SEU is an improvement in the fuel exit burn-up (5900 MWd/tU to 11400 MWd/tU) and the fuel per energy generated unit (1,2).

It is also increased the flattening of the axial power distribution resulting in a reduction of the maximum linear power, and this is of the great benefit in relationship to the safety margins. However, this increase of flattening might also produce an increase of the susceptibility to power oscillations xenon induced. These oscillations were previously considered in the original core design and partial absorber steel rods were included to damp them, especially during power transient, for example in load cycles and start-ups. With NU fuel the amplitude of oscillations induced by the movement of the regulating control rods was higher during these power reductions but it was not necessary to use the partial rods.

In the present work the possibility that these flux oscillations occurred and in this eventual situation the channel and linear power were below the operation limits were analysed.

2. SIMULATIONS

2.1 Calculations Methods

Calculations were done using the PUMA_II 3D reactor code (3), developed in Argentina, with a full core model, axial representation of 20 segments, xenon spatial dependence, thermal hydraulic feed-back (2,3) and time steps of 5 or 6 minutes.

In the initial reactor models 10 axial segments were used (as usual in fuel management calculations), without thermal hydraulic feed-back and larger time steps of 15 or 30 minutes.

The changes introduced in the model produced significant improvements in the comparison with measured parameters in the load cycle simulations.

Besides, the importance of modelling moderator temperature variations was also seen.
The cross sections and xenon related parameters required for the PUMA calculations were obtained with the WIMS_D4 cell code (4).

The complete simulations of power transients required doing core calculations, with measured power and rods positions data during several days.

2.2 Load Cycles

Different cases of load cycles were studied, corresponding to cores with increasing numbers of SEU fuel assemblies.

The simulated cycles of the plant consisted in a reduction from full power (100%) to 80% during one hour, and staying on that level for about 1 or 2 days with a subsequent recovery to full power with a 2% /h. This value is usually considered the maximum allowed power rate in relation with the *pellet cladding interaction* (PCI) power ramp failures also called by other authors *stress corrosion cracking* (SCC).

To analyse the core behaviour with respect to the flux oscillations when the number of SEU fuel elements in the core was increased, several simulations were done.

At the first time for a natural core with 12 SEU a real load cycle was done. For this case the control rods positions were obtained from data of the plant.

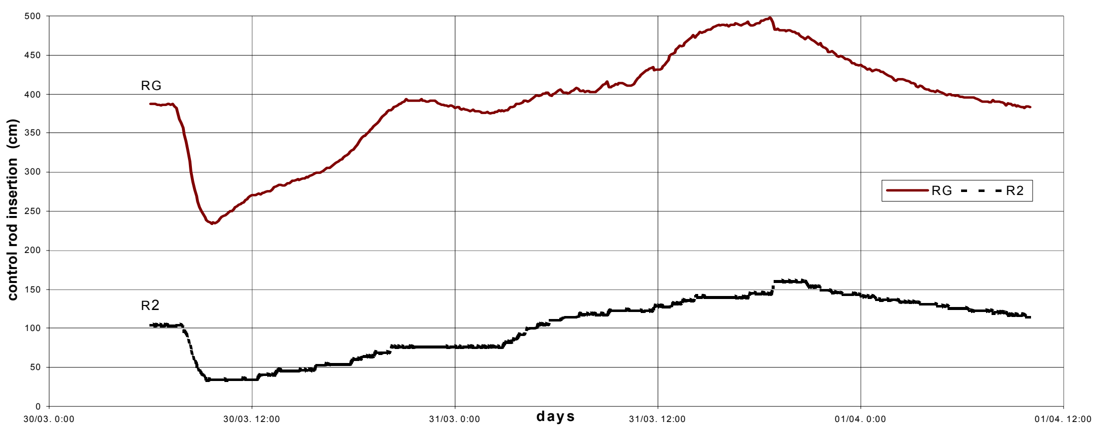
Later two load cycles with 60 SEU fuel elements and with 99 SEU fuel elements were simulated. In both cases the control rods positions were obtained by calculation using the criterion for maintaining constant the core reactivity.

Other real load cycles (using measured positions of the regulating rod banks) with 71, 120, 151, and 171 SEU fuel elements were later calculated. Finally using the measured position of the control rods during the measured load cycle with 171 SEU fuel assemblies, a full SEU core (252 SEU fuel assemblies) was also analyzed.

The simulations also verified that during these oscillations the maximum linear power did not exceed the operational limit of 600 W/cm.
A detailed version of what it is presented here can be seen in references (5,6,7).

For a typical load cycle the R2 and RG bank positions were shown in Figure 1.

Figure 1
Control Rod Positions in a typical load cycle



3. RESULTS

The main results of simulations are presented, particularly those belong to the parameters which can be measured or estimated from measured plant data during the load cycle. Some of these parameters are the maximum linear power, the thermal flux in some In-core detectors and the asymmetry axial factors (defined to each detectors chain, represent flux behaviour on the axial direction). Table 1 shows a summary of the simulated load cycles for mixed and homogeneous SEU cores.

Table 1

	Cases (number of SEU fuel elements)	Control Rod Position	Maximum Linear Power (w/cm)	Flux Axial Oscillations	Observations
a	12 SEU	from the real cycle	540	mitigated	without feed-back 8/4/95 cycle
b	60 SEU	by calculation	547	mitigated	(ρ ≈constant)
c	71 SEU	from the real cycle	544	mitigated	7/2/98 cycle
d	99 SEU	obtained from b)	547	mitigated	
e	120 SEU	from the real cycle	**576**	mitigated	21-25/10/98 cycle
f	151 SEU	obtained from c)	531	mitigated	
g	171 SEU	from the real cycle	535	mitigated	01/04/99 cycle
h	Homogeneous SEU	by calculation (ρ≈constant)	> **600**	not mitigated	no appropriate rod positions
i	Homogeneous SEU	obtained from c)	**542**	mitigated	extrapolated rod positions from 7/2/98 cycle
j	Homogeneous SEU	obtained from g)	530 - 538	mitigated	1/4/99 cycle

Figure 2 shows the results of the maximum linear power calculated and the comparison with plant estimations obtained from maximum In-core detector value during the 30/03/99 cycle (called NU-MAX). Figure 3 presents the calculated axial asymmetry factors and the average of all in-core detectors.

Figure 2
Maximum Linear Power

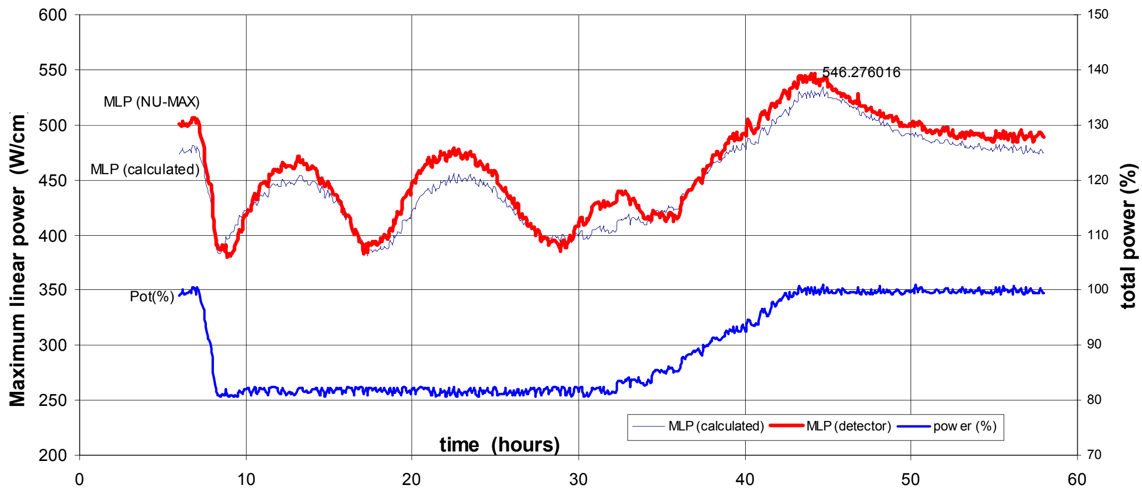
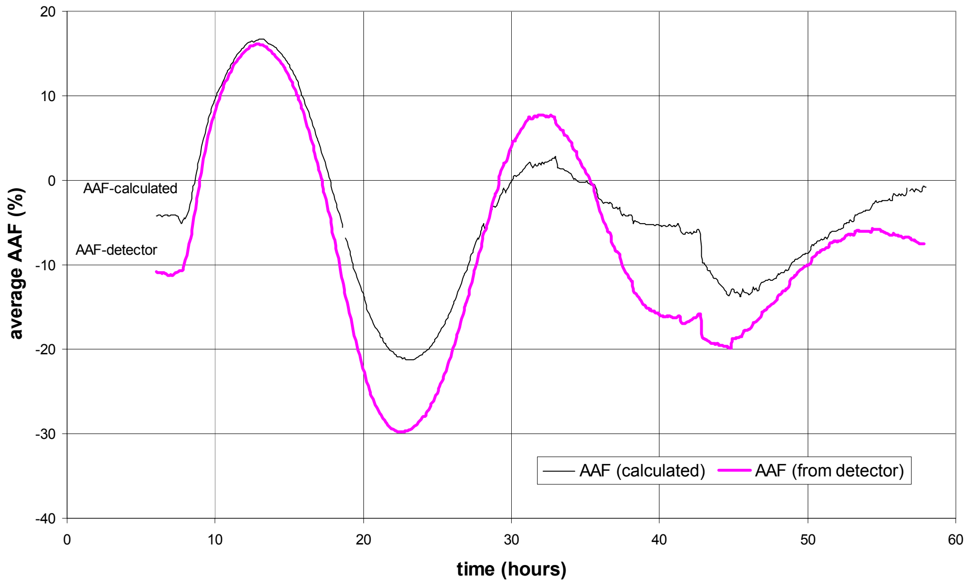


Figure 3
Axial Asymmetric Factor



The Figure 4 shows a comparison between the calculated and read thermal flux values on two different detectors for the cycle with 171 SEU fuel elements (30/03/99). Particularly it is reasonable the agreement between calculated and read flux with a small difference on the predictions of the instant when the maximum is achieved. Also, a good coincidence in the maximum values and a slight underestimation (15%) on minimum values. No satisfactory explanation of possible reasons of this observed discrepancy was found. Probably it would be associated to any limitation in the calculation model.

Figure 4
Thermal Flux on In-Core Detectors

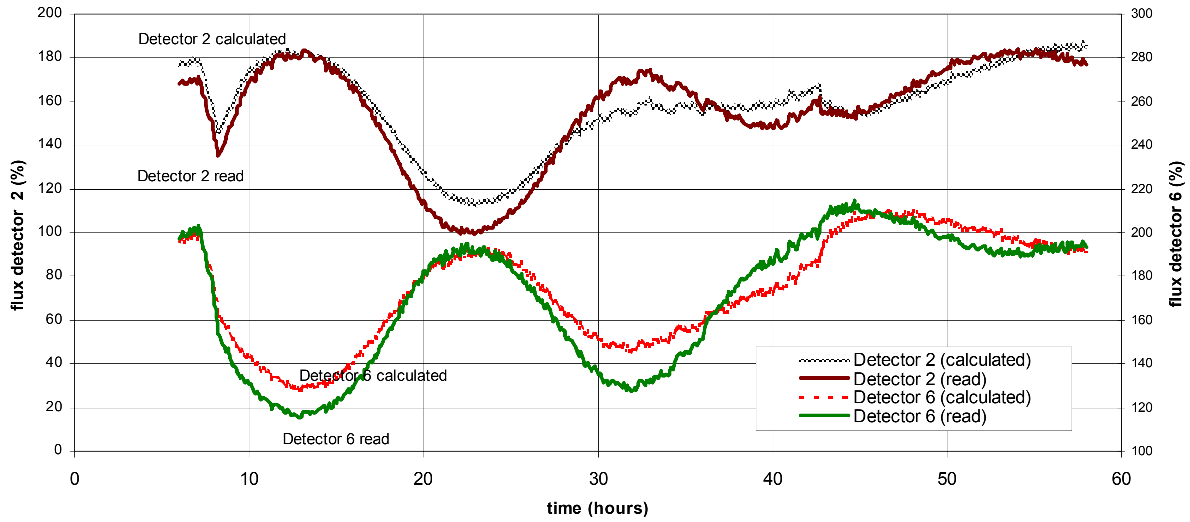
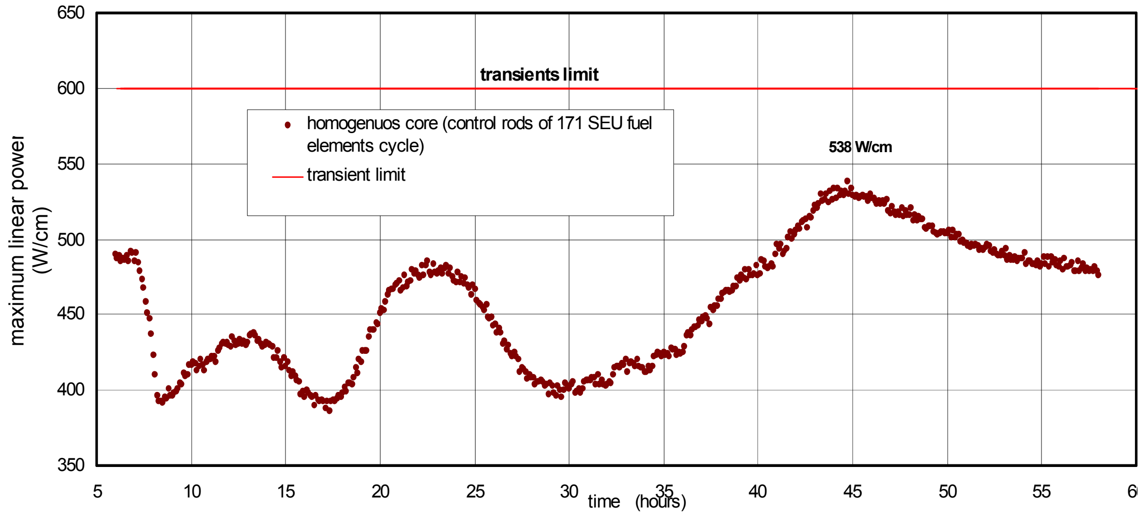


Figure 5 shows the calculated maximum linear power evolution for an homogeneous core using the control rod positions of the cycle with 171 fuel elements.

Figure 5

Maximum Linear Power in a Load Cycle for a SEU Homogeneous Core



The PCI effect was carefully analysed. In all the cases, the calculations showed that the linear power and its variation during the load cycles were below of allowed limits.

5. CONCLUSIONS

- At the 80% level the results show damped neutron flux oscillations induced by xenon axial variations of decrease amplitude of approximately 20 hours period. When the full power was again achieved the oscillations amplitude was noticeable reduced.

- A damping coefficient of 1.7 and 2.5 was found. A slight decrease in this value increasing the number of the SEU fuel elements. The described effect is only present in the 80% level without affecting the maximum linear power in the transient.

- In all the analysed cases the higher values of maximum linear power during the transient were maintained below 600 W/cm. These values were achieved during the period back to full power.

- During the simulation of the real cycle for the high number of SEU fuel elements, the maximum linear power value was 546 W/cm.

- For an homogeneous SEU core with the rod control positions extrapolated from the case of the 171 SEU fuel elements case, the maximum linear power was 538 W/cm.

- These results together with the hypothesis that in the homogeneous core the control rod movements would not have significant changes respect to 171 SEU fuel elements (equivalent to 2/3 SEU homogeneous core) would indicate that in a real cycle for SEU homogeneous core the flux oscillations would not produce an overcome the limits.

- In the simulations of real cycles a good agreement between calculates and read in core detector values was found. This agreement significantly improved when 20 axial pieces and 5-6 min time steps were used. For 171 SEU fuel elements case the discrepancy was 2%.

- In all these cases, they were not compromised situations respect to the pellet cladding interaction (PCI or SCC).

- Considering the general results of the cycles the criterion used for getting the control rod positions does not result realistic. It might be convenient to deeply analyse this aspect of the present study and try to find an algorithm that would allow to adequately predict the control rod movement.

- At the present time the plant has an homogeneous SEU core without operating problems due to xenon induced power oscillations.

ACKNOWLEDEGEMENTS

The authors wish to thank to Jose Fink for the very important advice about the technical items and the critical review of the manuscript.

Also to Ana M. Lerner for her suggestions on WIMS code and to Carlos Grant for his contribution on several aspects on PUMA code.

Finally to the CNA1 personnel for their contributions to the present report.

REFERENCES

[1] J. Fink, M. Higa, R. Perez, J. Piñeyro, J. Sidelnik, J.A. Casario, and L. Alvarez, "Overview of the SEU Project for Extended Burnup Fuel at the Atucha-I NPP. Four years of operating experience". Paper presented at the Technical Committee Meeting on Technical and Economic Limits to Fuel Burnup Extension. Bariloche, Argentina, 15-19 November 1999.

[2] J. Fink, M. Higa, R. Perez, J. Piñeyro, J. Sidelnik, J.A. Casario, and L. Alvarez, "Main Aspects of the SEU fuel Program at the Atucha-I PHWR after five years of operating experience". Proceedings of the 21st Annual Conference of the Canadian Nuclear Society. Toronto, Ontario, Canada, 11-14, June 2000.

[3] C. Grant, "PUMA - Versión II. Descripción y uso" (PUMA code – Use and description). Internal Report CNEA-IT 1043/90, (1990).

[4] M.J. Halsall, "A Summary of WIMS-D/4 Input Options", AEEW-M1327, June 1980.

[5] A. Blanco, O. Serra, "Simulación de Ciclos de Potencia (100-80-100)% en la CNA-I para núcleos mixtos de uranio natural con 12, 60 y 99 elementos de uranio levemente enriquecido (0.85%)" (Load cycles simulation on CNA1 for natural uranium mixed cores with 12, 60 y 99 SEU fuel elements), Internal Report CNEA-CAB 47-03/98, (1998).

[6] J. Villar, M. Higa, "Informe Preliminar de las Simulaciones de Ciclos de Carga (100-80-100%) en la CNA-I para Núcleos con 71, 151 y 252 EC ULE" (Preliminary report of load cycle simulations on CNA1 for cores with 71,151 y 252 SEU fuel elements), Internal Report NASA-IT-FN-009/98.

[7] O. Serra, "Simulación de Ciclos de Potencia (100-80-100)% en la Central Nuclear Atucha I con Núcleos de Uranio Levemente Enriquecido (0.85%)" (Load cycles simulation (100-80-100)% on CNA1 for homogeneous SEU fuel elements), Internal Report CNEA- CAB 47-09/99, (1999).

BELGIAN EXPERIENCE ON POWER UPRATES

M. Vincke
Association Vinçotte Nuclear,
Brussels, Belgium

Abstract

Much experience has been gained in Belgium on power uprates of nuclear power plants. Indeed, among the seven Belgian nuclear power plants in operation, power uprates were performed for three of them (Doel 3, Tihange 1 and Tihange 2), while a power uprate is presently planned for a fourth plant (Doel 2): the related licensing process has been started. For Tihange 2, the power uprate occurred in two steps of about 5 %, a steam generator replacement occurring at the second step. For Doel 3, Tihange 1, and Doel 2 as presently planned, the single-step power uprate is also coupled with a steam generator replacement. To allow such a final uprate value of 10 %, core design evolutions leading to new key parameters (before or upon the power uprate: fuel cycle extension, increased fuel enrichment, …), major equipment modifications (steam generator replacements, addition of a seventh main steam safety valve in each steam line during the steam generator replacement of Tihange 2) and changes of instrumentation setpoints are needed. Also, new methodologies are introduced, trying to take advantage of unnecessarily large safety margins in some safety analyses (use of best-estimate codes, of statistical methods, …). The purpose of the paper is to present a global, descriptive overview of these power uprates, putting the emphasis on their main characteristics, in particular their impact on the main plant and core parameters, the main related methodological changes and the modified instrumentation setpoints.

1 INTRODUCTION

This paper gives a global, descriptive overview of the power uprates performed up to now in Belgian Nuclear Plants, from a licensing point of view and putting the emphasis on their main characteristics. A global presentation of the different power uprates is first given in section 2.

For each power uprate, all the accident studies of the Safety Analysis Report need to be performed, unless non-reanalysis can be justified. Besides accidents, other verifications concern the capacity of safety systems and auxiliary systems, the radiological consequences of accidents, the impact on regulations and protections, the reference and transition cycles, and the mechanical design of the plant. The main assumptions used in the studies are summarized in section 3 for the main plant parameters and in section 4 for the core parameters.

To be feasible, a power uprate may require methodological changes in the safety studies (see section 5), and modifications in the plant (see section 6). Also the plant Technical Specifications and the plant procedures have to be adapted and the Safety Analysis Report has to be updated; these aspects are not covered here.

Section 7 concludes.

2 GLOBAL PRESENTATION OF POWER UPRATES IN BELGIUM

Table 1 presents the power evolution with time of the four concerned Belgian plants, starting from their initial design. These are three-loop plants except Doel 2, which is a two-loop plant.

For the four plants, the power uprate (PU) was coupled with a steam generator replacement (SGR), but for one of them (Tihange 2), the power uprate occurred in two steps, the first one without SGR. For the Doel 2 two-loop plant, a PUSGR is planned for 2004: the related licensing process is now running so that Doel 2 is only mentioned in Table 1 as an indication. The rest of this paper focuses on the already performed power uprates (three-loop plants).

In Table 1, the indicated powers are the reactor nominal thermal powers (in MWth). Every PU has been worked out in the frame of a project; the project acronyms are given for further reference. The rates of the power increases are given (in %) with respect to the initial-design powers.

		Tihange 1	Doel 3	Tihange 2	Doel 2
Initial Design	Year	1974	1982	1982	1974
	Power	2660 MWth	2785 MWth	2785 MWth	1192 MWth
PU	Project			APAC T2	
	Year			1992	
	Rate			4.3 %	
	Power			2905 MWth	
PUSGR	Project	APRGV T1	APRGV D3	APRGV T2	*VSGP+ D2*
	Year	1995	1993	2001	*2004*
	Rate	8 %	10 %	10 %	*10 %*
	Power	2873 MWth	3064 MWth	3064 MWth	*1310 MWth*

Table 1 - Global overview of PU projects in Belgium

3 MAIN PLANT PARAMETERS

Table 2 gives the values of the main plant parameters used in the safety studies performed for the Belgian PU projects. In each case, a maximum Steam Generator Tube Plugging (SGTP) rate of 5 % was assumed.

	APAC T2	APRGV D3	APRGV T1	APRGV T2
Power	2905 MWth	3064 MWth	2873 MWth	3064 MWth
Nominal T_{avg}	306.0°C	303.2°C	302.7°C	303.2°C
Vessel T/H design flow	61,780 m^3/h	62,000 m^3/h	63,366 m^3/h	62,000 m^3/h
Nominal SG pressure	59.0 bar a	58.6 bar a	59.9 bar a	61.3 bar a

Table 2 - Main plant parameter values of the PU projects

4 REFERENCE CORE

4.1 Main features

The main characteristics of the reference cores defined in the Belgian PU projects are displayed in Table 3.

The fuel enrichment used at the initial design of the Belgian plants was of about 3 % typically. Before the APAC T2 project, there have been authorizations for the maximum allowed discharge burnup (55,000 MWd/t – the effective corresponding value at the initial design of Belgian plants had an order of magnitude of 30,000 MWd/t) and for the maximum fuel cycle lengths (18 months – compared to 12 months initially). For the APRGV T2 project, there were two reference cores, one with UO2 fuel only and one with UO2 + MOX fuel. The data in Table 3 concern the reference core with UO2 fuel only. For the Doel 3 plant, there has been a MOX project after the APRGV D3 project, thus for PUSGR conditions.

In Table 3, the maximum linear power density values result from the power capability studies related to Condition II transients.

	APAC T2	APRGV D3	APRGV T1	APRGV T2
Fuel enrichment	4.5 %	3.95 %	4.35 %	4.6 %
Fuel cycle lenght	18 months	12 months	18 months	18 months
Av. discharge BU	48,000 MWd/t	48,000 MWd/t	48,000 MWd/t	45,000 MWd/t
Max. linear power density	590 W/cm	590 W/cm	656 W/cm	590 W/cm
Burnable poison	---	---	Gd	Gd
MOX	---	---	---	Y

Table 3 - Main features of the reference cores in the Belgian PU project

4.2 Main key parameters

In each PU project, a reference core is defined and has to be as much bounding as possible with respect to the equilibrium cycle corresponding to the reference loading pattern of the project. A set of parameters, called “key parameters”, is calculated for this reference core. After adding the related uncertainties and a provision to take the variability in the real loading patterns into account, the values of these key parameters are then verified in the project accident analyses.

The project reference core is supposed bounding for the core reloads subsequent to the power uprate. However, the real subsequent cores are never identical to the project reference core: more or less important differences can be observed, for example depending on the fuel type used or the way the core is managed.

In order to verify the conformity of a reload without repeating all the safety studies, the reload key parameters are compared to the project ones. Table 4 lists some of these key parameters for the different PU projects, and gives the project impact for the hot spot and channel factors. The value 1,500 pcm of the shutdown margin used in some projects was not used in practice.

	APAC T2	APRGV D3	APRGV T1	APRGV T2
Hot spot factor F_Q	2.13 -> 2.30	2.13 -> 2.30	2.07 -> 2.18	2.30 (=)
Hot channel factor $F_{\Delta H}$	1.55 -> 1.65	1.55 -> 1.62	1.60 -> 1.64	1.65 (=)
Shutdown margin	1,500 pcm	1,500 pcm	1,500 pcm	1,770 pcm

Table 4 - Main key parameters of the reference cores in the Belgian PU project

It should be noted that in the APRGV T1 project, the control band around the reference axial offset and situated above 85% of the nominal power had to be reduced from ±5 % to ±3% in order to avoid xenon oscillations incompatible with the hot-spot factor limit defined by the LOCA studies.

5 MAIN METHODOLOGICAL CHANGES

To be feasible, a power uprate may require a better assessment of safety margins in order to compensate the penalties resulting from the PU. In order to take advantage of unnecessarily large safety margins, new methodologies can be introduced for some safety studies. The use of a new methodology in the frame of a project has to be accepted by AVN before its application to the concerned safety study. The main methodological changes introduced in the frame of the Belgian PU projects are presented in Table 5.

	APAC T2	APRGV D3	APRGV T1	APRGV T2
LBLOCA	Code: conservative Appendix-K models but with improvements	Code: conservative Appendix-K models -> best-estimate	Code: conservative Appendix-K models -> best-estimate	Code: conservative Appendix-K improved models -> best-estimate F_Q decrease with burnup
SBLOCA		Code: conservative Appendix-K models -> best-estimate		Code: conservative Appendix-K models -> best-estimate
DNB correlation	Related to reference fuel of the project			
DNBR criterion	Combination of uncertainties: deterministic -> statistical			
Overpressure transients, heating transients		Code: conservative -> best-estimate		
Integrity of primary components (SG excluded)		Exclusion of the large break of a main coolant pipe for mechanical design purposes (Leak Before Break)		

Table 5 - Main methodological changes introduced in the Belgian PU project

The essential trend is to introduce more realistic approaches. In the APRGV T2 project, a moderate decrease of the hot spot factor F_Q with burnup had to be taken into account in the LBLOCA study, because the fuel pellet thermal conductivity decreases with burnup.

6 MAIN PLANT MODIFICATIONS

6.1 Main equipment modifications

Besides the SGR, which is a major equipment modification, other equipment modifications were performed in the Belgian PU projects. Table 6 lists the most important ones.
As a result of the PU, the SGTP assumption and, in case of SGR, the modification of the primary and secondary volumes, the secondary overpressure study may lead to setpoint modifications for the main-steam safety valves (and also, in some cases, to instrumentation setpoint modifications). Main-steam SV setpoint modifications occurred in all Belgian PU projects. In the APRGV T2 project, a seventh SV was added for each SG in order to guarantee a sufficient relief capacity after PUSGR.

In the APAC T2 project, the need of a sufficient amount of boric acid after LOCA led to guarantee the interconnection of the three tanks of the fuel storage pool; also, as a result of the residual heat increase, the PRZR heating time before retrieving saturation conditions had to be reduced: this was achieved by increasing the PRZR heater power.

In the APRGV T1 project, “cavitating venturi” were installed in the AFW lines in order to limit the AFW flow rate lost to the break in case of a FWLB accident, and to increase the AFW flow rate to the intact SGs.

	APAC T2	APRGV D3	APRGV T1	APRGV T2
SG replacement	---	Y	Y	Y
Main steam SV setpoint modification	Y	Y	Y	Y
Addition of main steam SV	---	---	---	7th SV added for each SG
AFW system	---	---	Cavitating venturi added	---
Fuel storage pool	Tank interconnection	---	---	---
PRZR heaters	Increased power	---	---	---

Table 6 - Main equipment modifications in the Belgian PU projects

6.2 Modified instrumentation setpoints

In each PU project, the accident analyses validated the setpoints of the reactor protection system and safety systems. The main setpoint modifications performed in the Belgian PU projects are summarized in Table 7.

	APAC T2	APRGV D3	APRGV T1	APRGV T2
ΔT protections	Y	Y	Y	Y
RT on PRZR High Level	---	Y	Y	---
High (TT) and Low (RT) SG Levels	---	Y	---	---
Spray actuation	---	---	---	Delay added
SI on High Containment Pressure	---	---	---	L/L added on measured p

Table 7 - Main setpoint modifications for the protection and safety systems in the Belgian PU projects

In each project, the thermal-hydraulic core design study results in DNBR design limits, which are used to define the core protection diagram and hence the setpoints of the overpower ΔT and overtemperature ΔT protections. The penalty functions of these protections result from the power capability studies related to Condition II transients.

In the APRGV D3 and T1 projects, the secondary-overpressure studies led to lower the setpoint related to the reactor trip on High PRZR level.

In the APRGV D3 project, the replacement of transmitters and an uncertainty reevaluation resulted in modifications of the setpoints related to the turbine trip on High SG Level and to the reactor trip on Low SG Level, with more margin in the safety analyses for both signals.

In the APRGV T2 project, a delay was added to the containment spray actuation as a result of the LBLOCA analyses, and a lead/lag was added to the containment pressure measurement in order to anticipate the closure of the MFW isolation valves in case of SLB.

In each project, regulations were verified by transient analyses; some of them had to be modified.

7 CONCLUSION

This quick, descriptive overview of power uprates performed in Belgian nuclear power plants illustrates several conditions, not necessarily occurring together, for a power uprate to be practicable:

- plant modifications (including major ones like SGR);
- setpoint modifications of protection and safety signals (in particular for the ΔT protections);
- methodological changes (more realistic approaches for the safety demonstrations);
- core design evolution (fuel cycle extension, increased fuel enrichment, …).

THE MASS VELOCITY EFFECT ON THE OVERTEMPERATURE PROTECTION LIMIT IN PWR REACTORS

E. Maprelian
B.D.F. Baptista

Energetic and Nuclear Research Institute of the Brazilian Nuclear Energy Commission,
IPEN-CNEN/SP,
São Paulo, Brazil

Abstract

This paper analyzes the effect of the mass velocity on the overtemperature, ΔT, trip function in PWR reactors. The analyses were done with data from the Columbia University critical heat flux experimental test section and from the 1995 look-up tables for critical heat flux in tubes. The experimental conditions include mass velocities in the range of 500 to 4000 kg/sm^2, pressures from 12 to 16 MPa, and local quality from -40% to 30 %. The results show that, for mass velocities higher than 2500 kg/sm^2, a pressure reduction implies in a more restrictive condition in terms of DNB. For mass velocities below 1500 kg/sm^2, an inversion of this behavior is verified. As the general approach used to define the protection curves was developed for reactors designed for higher mass velocities, the observed behavior change indicates the need for a new methodology for other projects. The preliminary results showed that, for a design with low mass velocity, it would be better to divide the overtemperature curve into two first grade equations; the first one for the DNB thermal limit, and the second one to the saturation temperature limit. The suggested approach allows for the elimination of unnecessary safety margins in the reactors designs with low mass velocities.

1. INTRODUCTION

The concept of Overtemperature Protection Curves has been largely employed in the protection of PWR nuclear power plants. The operation is kept distant from the thermal limits of DNB (Departure from Nucleate Boiling) and saturation temperature in the reactor outlet and the reactor is shutdown if operation reaches these limits.

The methodology for the construction of protections curves, developed by Westinghouse [1], considers reactors with high mass velocities in the core, generally over 2,500 kg $s^{-1}m^{-2}$. In the case of reactors designed for lower mass velocities, as 1,500 kg $s^{-1}m^{-2}$, like the IRIS Advanced Reactor, or even for lower values, as those considered in several small reactors designs, few adaptations in this methodology are necessary.

The objective of this paper is to present an investigation of the effect of the mass velocity on the behavior of the thermal limit of DNB, thus on the overtemperature protection curve.

Section 2 analyzes the behavior of DNB as a function of the mass velocity. Data from the Test Section #53 of the Columbia University [3] and from the 1995 Look-up Tables [4] were used.

Section 3 shows a case study of an hypothetic small reactor to demonstrate the effect of the low mass flow velocity, of 800 kg $s^{-1}m^2$, on the behavior of the protection curves and on its construction.

Section 4 presents the main conclusions of the present study.

2. DNB AS A FUNCTION OF PRESSURE AND MASS VELOCITY

The overtemperature protection curve is designed to protect the reactor with respect to the thermal limits of DNB and saturation temperature (T_{sat}) at the reactor outlet. The last limit is necessary to keep the proportionality between Power and Temperature difference, ΔT. Equations of ΔT as a function of the mean temperature (T_{avg}) and pressure (P) are set. The temperature difference is related with the reactor thermal power and T_{avg} with the inlet and outlet temperatures.

It can be seen that, with a fixed power, then a constant ΔT, any increase in the pressure will allow an increase in the T_{avg} to reach the saturation temperature limit, T_{sat}. By other side, with the DNB limit this is not true. The critical heat flux is related with many other factors like the local conditions as temperature and pressure (the local quality), mass flow velocity and also with the heat flux.

To study the behavior of the DNB with respect to the ΔT, T_{avg} and pressure, data from critical heat flux experiments were chosen from the "*The 1995 look-up tables for critical heat flux in tubes*"[4] and from the Test Section # 53 from the Columbia University [3].

2.1. The 1995 look-up tables for critical heat flux in tubes

The 1995 look-up tables for critical heat flux in tubes gives many data based on experimental critical heat flux studies. Table 1 presents data from a set of points from this reference. The data considers only subcooled conditions. Table 1 shows the local subcooled quality as a function of pressure for six different values of mass flow velocities. The table also shows the calculated outlet temperatures as a function of quality and pressure.

Figures 1 and 2 show, respectively, the local quality and outlet temperature as a function of pressure for each mass flow velocity.

Fig. 1 shows that the tendency of the critical quality, for every test, is to reduce with the increase in the pressure. The critical quality seems to be more feasible to represent the behavior of the critical heat flux.

Table 1- Local qualities and temperatures for Critical Heat Flux as a function of Pressure [4].

Pressure	**G=500**		**G=1000**		**G=1500**		**G=2000**		**G=2500**		**G=3000**		**G=4000**	
	Q"= 3.919		**Q"=3.372**		**Q"= 3.177**		**Q"= 3.172**		**Q"= 3.225**		**Q"= 3.274**		**Q"= 3.390**	
	Xloc	**Tout**	**Xloc**	**Tout**	**Xloc**	**Tout**	**Xloc**	**Tout**	**Xloc**	**Tout**	**Xloc**	**Tout**	**Xloc**	**Tout**
12	0.0	324.6	0.0	324.6	0.0	324.6	0.0	324.6	0.0	324.6	0.0	324.6	0.0	324.6
13	-13.2	307.4	-10.0	313.6	-4.7	323.1	-3.1	325.8	-2.7	326.5	-2.1	327.5	-1.0	329.3
14	-18.2	306.8	-11.9	318.1	-7.1	326.1	-4.7	329.8	-3.7	331.3	-2.8	332.7	-2.2	333.6
15	-39.0	277.3	-13.7	323.1	-10.3	328.3	-8.3	331.2	-6.2	334.2	-4.4	336.7	-3.9	337.2
16	-	-	-22.9	317.4	-13.2	331.6	-11.3	334.2	-9.3	336.7	-7.8	338.6	-6.2	340.5

[P] - MPa; [G] - kg $s^{-1}m^2$; [**Q"**] - MW m^2**;** [T] - °C; [X] - %.

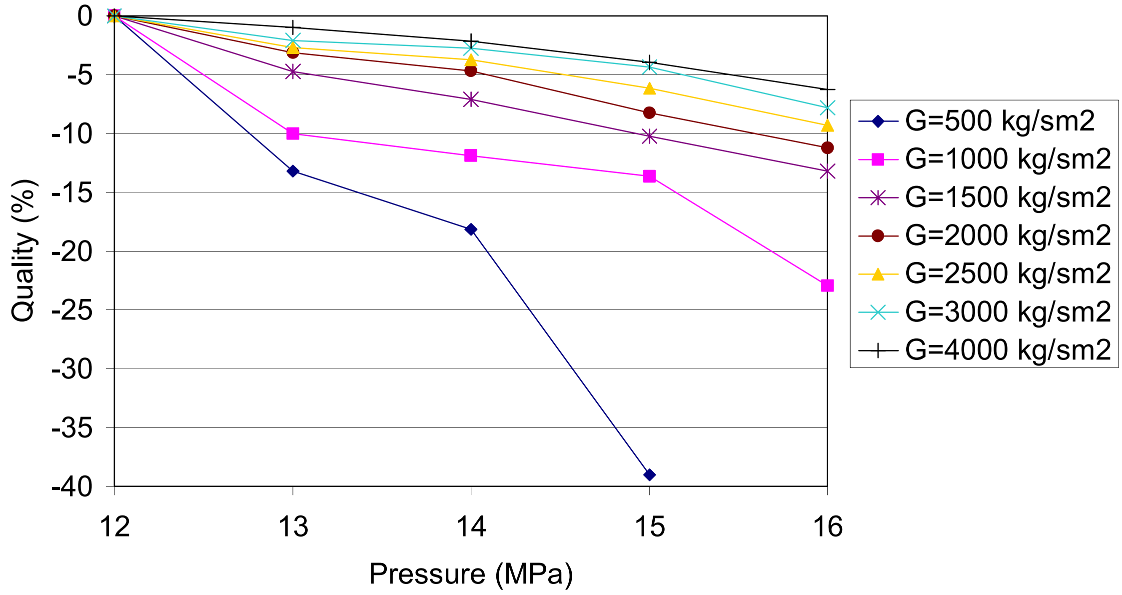


Figure 1 – Local Quality versus Pressure.

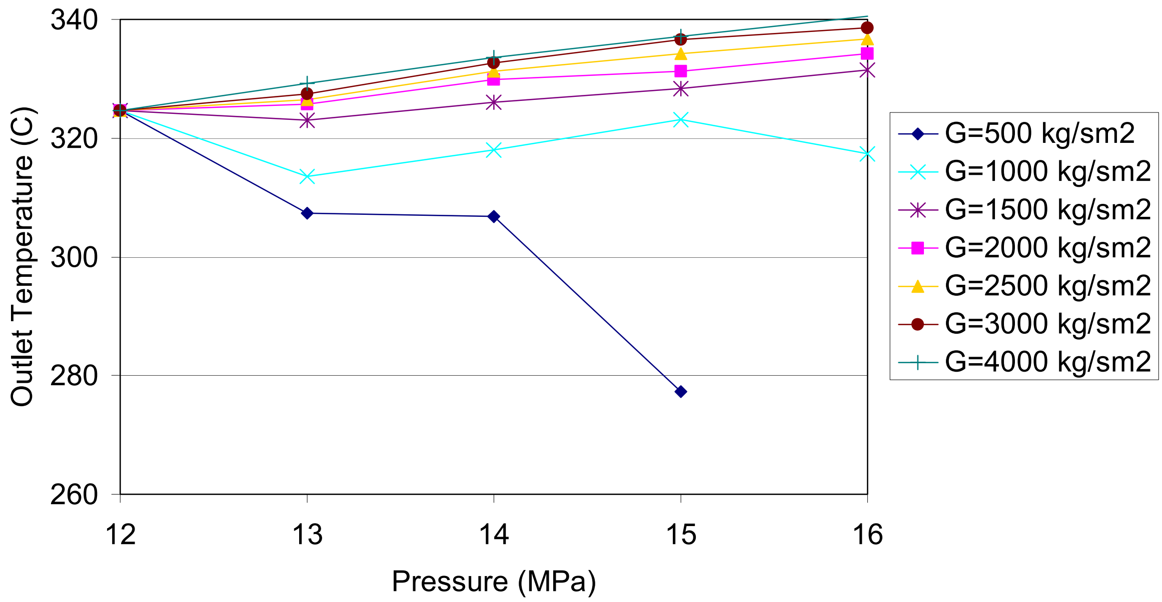


Figure 2 – Outlet Temperature versus pressure.

Taking the behavior of the outlet temperature (Fig. 2) into consideration, different tendencies are verified: for mass flow velocities higher than 1,500 $kg.s^{-1}m^2$, the outlet temperature increases with the increase in the pressure; for lower values there are regions were the behavior is quite different, the outlet temperature reduces with an increasing pressure; and, for values below 1,000 kg $s^{-1}m^2$, the tendency is the reduction in the outlet temperature with an increasing pressure for the full pressure range.

2.2. Columbia University Test Section #53 – Combustion Engineering

The experimental results obtained with the Columbia University Test Section #53 [3] confirm the observations above. Fig. 3 and 4 show the reduction of the critical quality and inlet temperature (T_{in}) with the pressure, for a G value around 1,370 kg $s^{-1}m^2$. Fig. 4 also shows the values of T_{in} calculated with COBRA3P [5] for DNBR equal unity, with the EPRI correlation [6].

EPRI correlation presented a coherent behavior in reproducing the experimental results with an increasing with pressure, but conservative, detachment from the measured values. This analysis has only numerical meaning as we always can find, for the same inlet temperature, a DNBR value below 1.3.

In Fig. 5 and 6 the conditions are equivalent to that of Fig. 3 and 4 but with a mass flow velocity of 2,670 kg $s^{-1}m^2$. Both, experimental and analytical results are in accordance with that observed with data from the Look-up tables.

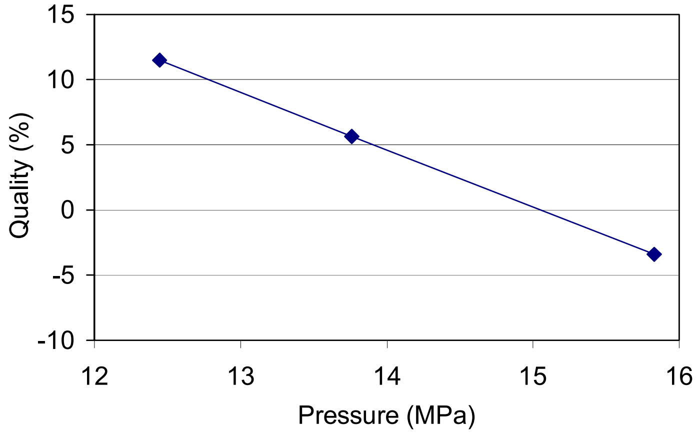


Figure 3 – Critical quality versus pressure for constant heat flux and G ≅ 1,370 kg $s^{-1}m^2$.

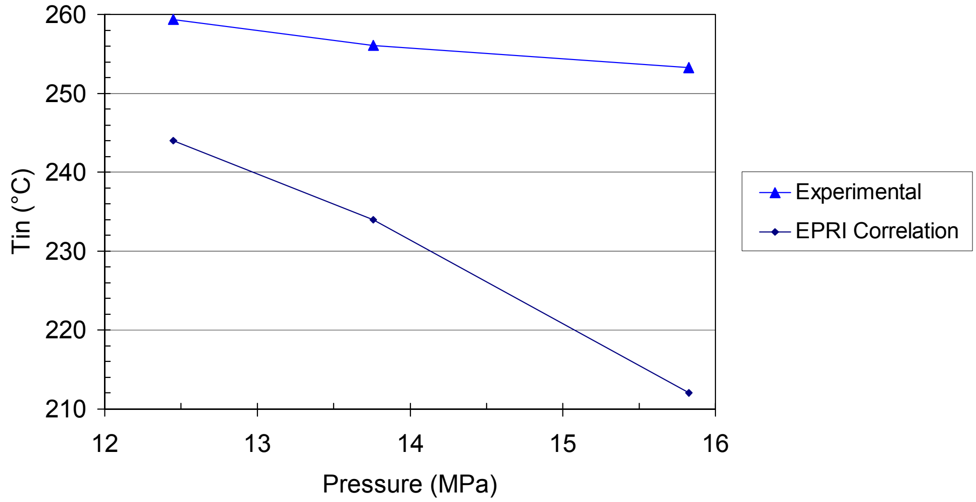


Figure 4 – Inlet temperature versus pressure for DNBR=1 and G ≅ 1,370 kg $s^{-1}m^2$.

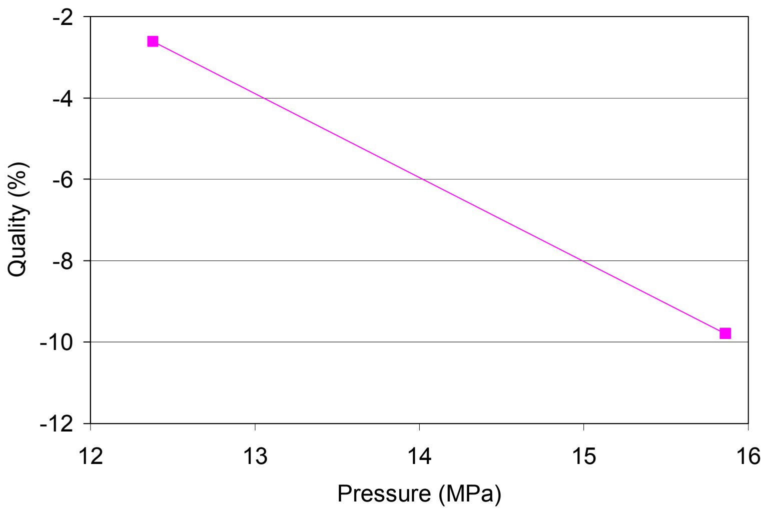


Figure 5 – Critical quality versus pressure – G ≅ 2,670 kg $s^{-1}m^2$.

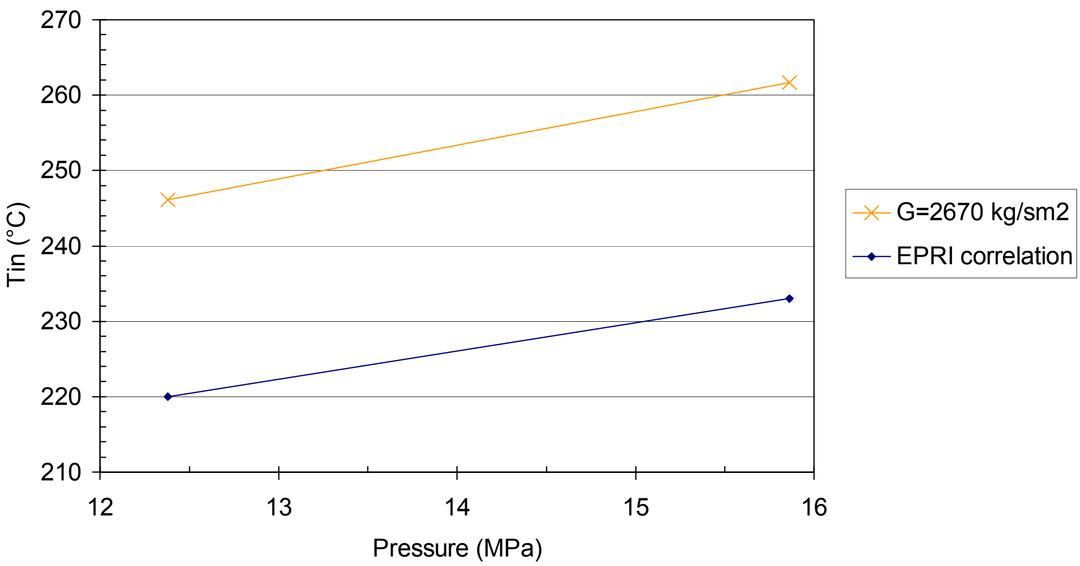


Figure 6 – Inlet temperature versus pressure, DNBR=1 and G ≅ 2,670 kg $s^{-1}m^2$.

Fig. 7 and 8 show the experimental results of critical quality and critical heat flux for 720kg $s^{-1}m^2$ with a constant inlet temperature. Fig. 9 shows the DNBR calculated with COBRA3P and the EPRI correlation for the conditions presented in Fig. 8. Observe that Fig. 7 and 8 show the same tendency of decreasing critical quality and the decreasing in the critical heat flux with the increasing pressure. Fig. 9 shows the tendency of EPRI correlation in produce more conservative results with the increase in pressure.

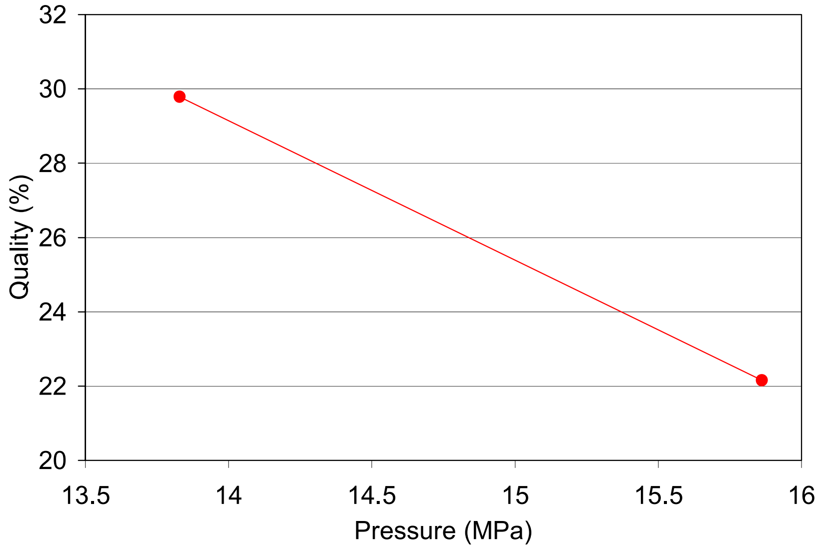


Figure 7 – Critical quality versus pressure for Constant Tin and G ≅ 720 kg $s^{-1}m^2$.

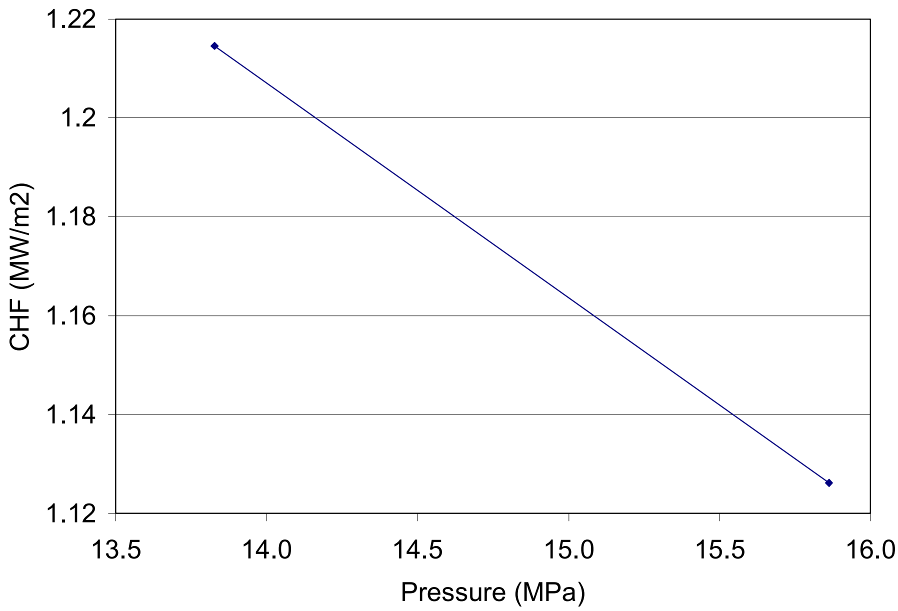


Figure 8 – Critical heat flux versus pressure for Constant Tin and G ≅ 720 kg $s^{-1}m^2$.

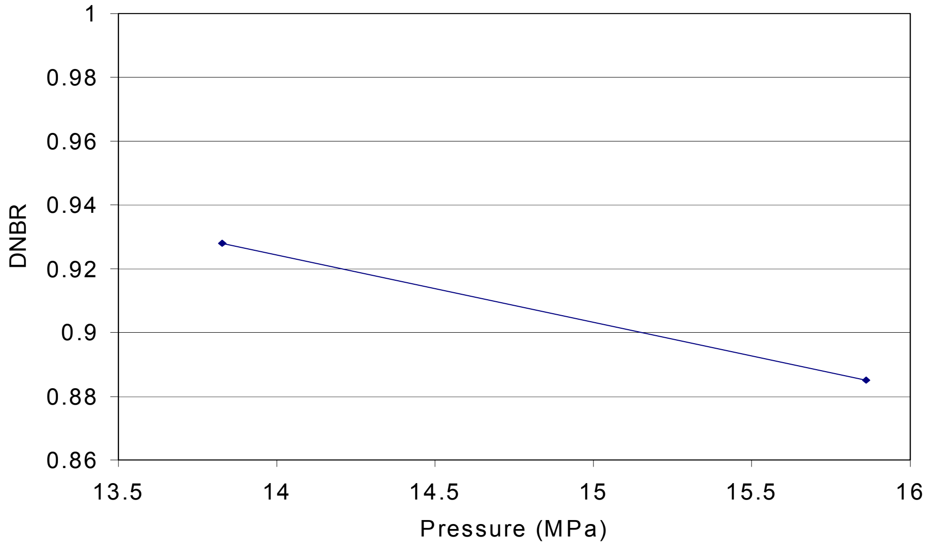


Figure 9 – DNBR (EPRI) versus pressure for Constant Tin and G ≅ 720 kg $s^{-1}m^2$.

3. CASE STUDY OF AN OVERTEMPERATURE PROTECTION CURVE

This section presents a case study based on a hypothetical small reactor of ~50MW(t), designed for a low mean mass flow velocity. The purpose is to check the effect of this low velocity on the overtemperature protection curve.

Figs. 10 to 12 show the normalized overtemperature, overpower and steam generators overpressure curves for the primary system pressures of 12, 14 and 15.5 MPa, at nominal flow conditions. The values of ΔT and T_{avg} were normalized with respect to the nominal conditions. The protection curves philosophy is based on the methodology presented in reference [1]. The analyses to find the limiting DNB were performed with COBRA IV [6]. These figures also show the proposed curves to increase the permissible operating area.

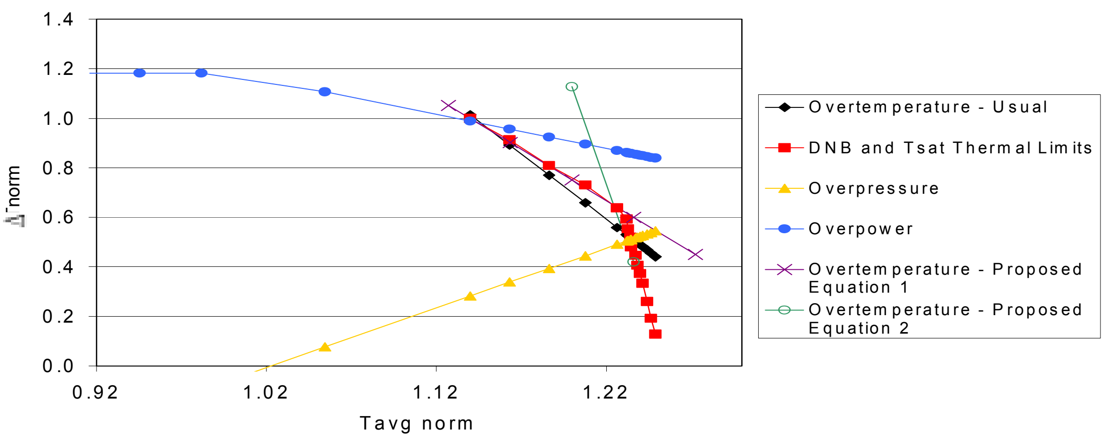


Figure 10-Normalized Protection Curves and DNB and T_{sat} Limits– Pressure 15.5 MPa

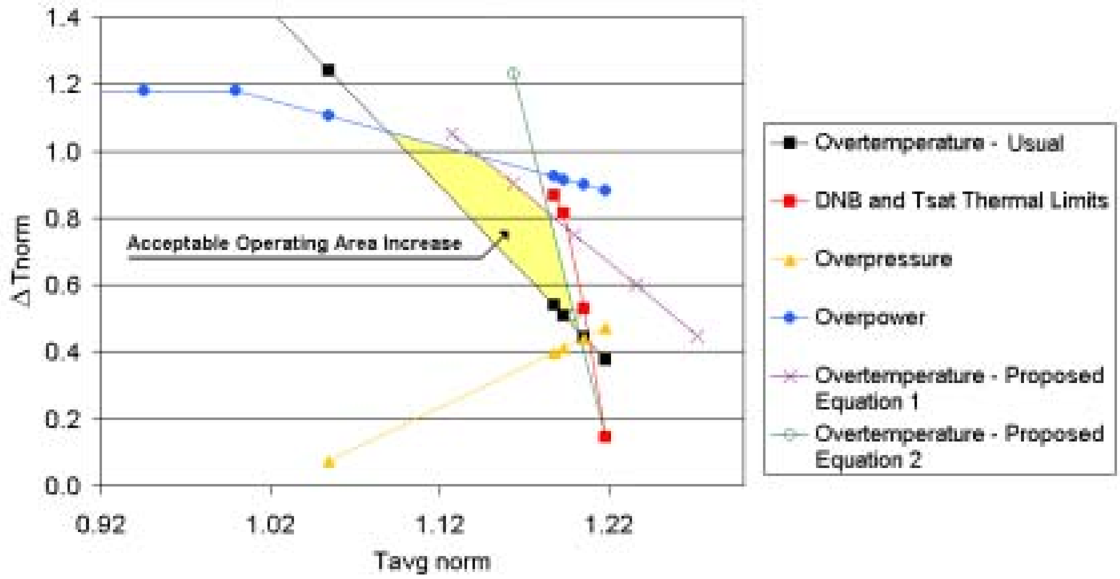
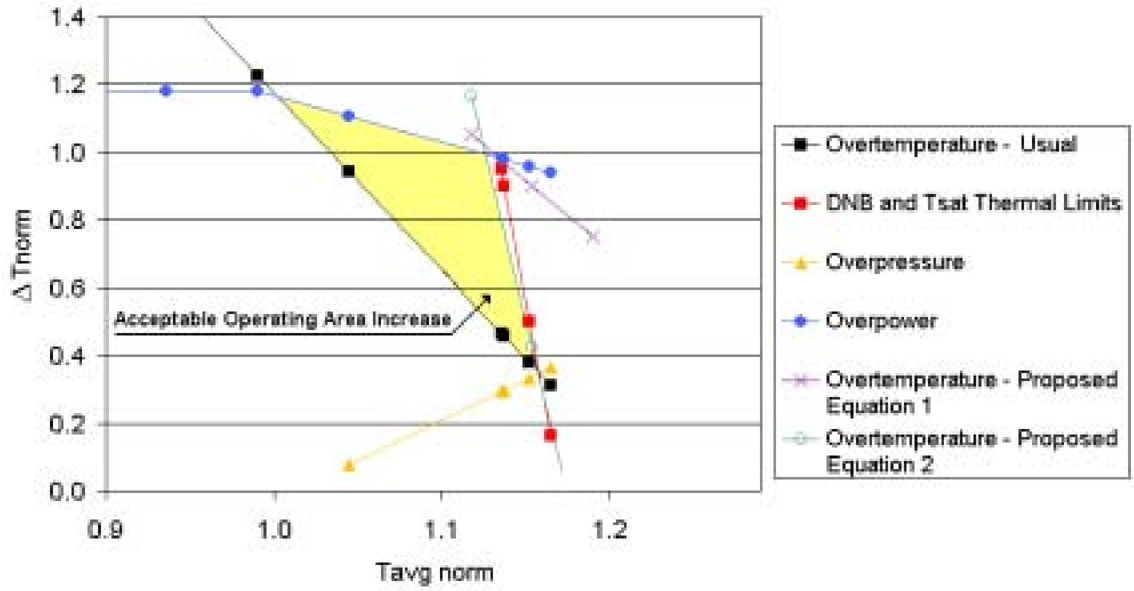


Figure 11 - Normalized Protection Curves and DNB and T_{sat} Limits – Pressure 14 MPa



T e
polynomial approximation:

$$\Delta T_{norm} = K_1 - K_2 \times (T_{avgnorm} - T_{avgnom}) + K_3 \times (P - P_{nom}) \quad (1)$$

This approximation does not allow to take into consideration the different behavior of the thermal limits, DNB and T_{sat}, bellow the value of 1,500 kg $s^{-1}m^2$.

The main results of this study case is to show how the overtemperature curve detaches from the thermal limits, producing unnecessary operating restrictions.

The overtemperature curve was done for the pressure of 15.5 MPa and a positive pressure correction coefficient, K3, was obtained. The unique transformation of such kind of correction is a single translation, but it can be seen the need for a “rotation” to improve the permissible operating area.

The suggestion is to consider two first degree curves instead of only one as proposed in [1]. “Equation 1” corresponds to the safety limits of DNB while “Equation 2” to the saturation temperature limit. The existence of a change in the behavior at pressures about 14 MPa and considering that there are no significant loss of area, we found that the adoption of a single equation set for the higher pressure of 15.5 MPa is a good option (see Figs. 10 to 12).

The adjustment for the equation 2 does not present any difficulty. Table 2 presents the format of these equations:

Table 2 – Equations for the Overtemperature curves.

Equation 1	$\Delta T_{norm} = K_4 \times T_{avgnorm} + K_5$
Equation 2	$\Delta T_{norm} = (K_6 \times P - K_7) \times T_{avgnorm} + (K_8 \times P + K_9)$

The net gain in terms of “permissible operating area,” as a result of using the above concepts, is represented by the yellow area in Figs. 10 to 12.

4. CONCLUSIONS

The main reason for this paper is to demonstrate the need of a carefully analysis of the plant operating conditions prior to the development of protection curves.

Analysis of experimental data show that the behavior of the temperature (not of the critical quality) correspondent to the DNB conditions, with pressure, changes according to the range of mass flow velocity. This change in behavior is important in the definition of the overtemperature protection curves.

The study case presented showed a possible loss of permissible operating area as a consequence of this behavior.

For the specific case of a nuclear reactor operating at low mass flow velocities, the proposed methodology allows for an increase in the operating area without loss of safety. However it is important to state that this methodology is subject to specific control characteristics of the plant in design.

It is also important to observe the need for a correct choice of DNB data and correlations to design and analyze a new PWR concept mainly in the case of lower mass flow velocities, lets say below 1,500 kg $s^{-1}m^2$.

Nomenclature

ΔT – vessel average temperature difference (°C)
CHF – Critical Heat Flux (MW/m^2)
DNB - (Departure from Nucleate Boiling)
DNBR - (Departure from Nucleate Boiling Ratio)
K - constant
P - Pressure (MPa)
T - Temperature (°C)
X – Quality (%)
G – Mass Velocity (kg s^{-1}m^2)
Q" – Heat flux (MW/m^2)

Subscripts

$_{1;\,2;\,\ldots 9}$ - constants index numbers
$_{avg}$ - average
$_{cr}$ - critical
$_{in}$ inlet
$_{loc}$ - local
$_{nom}$ – nominal
$_{norm}$ – normalized
$_{out}$ - outlet
$_{sat}$ – saturation

REFERENCES

[1] Ellenberger, S. L. et al. Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions. Westinghouse Eletric Corporation, March 1977, WCAP-8746.
[2] Groeneveld D. C. et al. The 1995 look-up table for critical heat fluxes in tubes. Nuclear Engineering and Design 163 (1996) 1-23.
[3] Fighetti C. F., Reddy, D. G. Parametric Study of CHF Data. Volume 1-3. Columbia University New York, NY, EPRI NP-2609.
[4] MASTERSON, R.E., COBRA-3P: An improved version of COBRA for full core light water reactor analysis, Nuclear Eng. and Design, 48:293-310, 1978.
[5] Weeler, C. L. et al. COBRA-IV-I: An Interim Version of COBRA for Thermal-Hydraulic Analysisof Rod Bundle Nuclear Fuel Elements and Cores. Battele Pacific Northwest Laboratories. Richland, Washington. Mar 1976, BNWL-1962.

RECENT EXPERIENCE IN EVALUATION OF SAFETY MARGINS OF CANADIAN NUCLEAR POWER PLANTS

A. Viktorov
Canadian Nuclear Safety Commission,
Ottawa, Canada

Abstract

In the recent past, several discoveries stemming from analytical and research work have lead to a realization that safety margins previously demonstrated in the Safety Reports of some CANDU plants may have been overestimated. The new analyses indeed had shown that to preserve the safety margins some restrictions needed to be applied to operational limits. In some cases these restrictions included limiting the maximum allowable power. In particular, Bruce B reactors have been limited in the recent years to 90 % full power operation due to concerns related to consequences of large break loss of coolant accidents. The Canadian Nuclear Safety Commission (CNSC) has requested all licensees of power reactors to implement measures to restore and improve safety margins. Those utilities that had power deratings imposed on operation of their reactors are also being driven in their margin improvement efforts by the economic incentive to have their licensed power limit restored to 100% full power. Approaches adopted by different utilities vary depending on the specific situation and may include:

- optimization of operational limits and conditions
- refinement of safety analysis tools and methods (in particular, development of a best estimate and uncertainty methodology)
- further experimental investigation to better validate tools used in accident analysis
- implementation of design changes (the most significant being the new fuel design)
- development of an integrated risk informed licensing methodology to demonstrate that the overall risk is little unaffected by the recent discoveries
- implementation of accident management strategies as a toll to address residual risks.

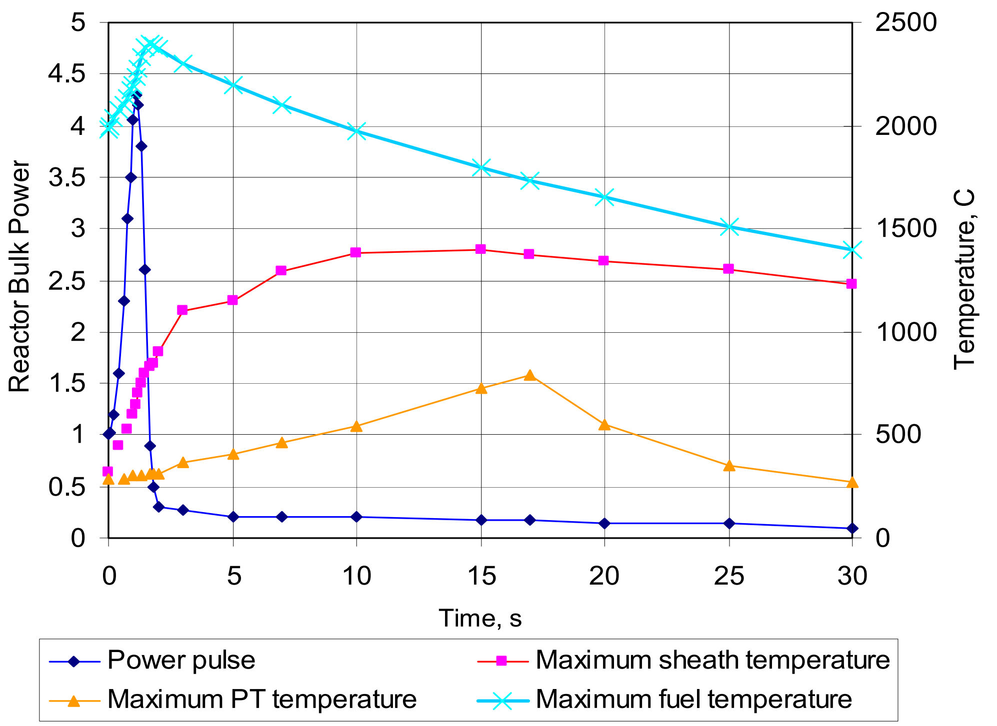
Many of these activities require significant advance effort and are novel in licensing application. The CNSC staff is involved in these efforts through identification and development of regulatory requirements and expectations.

A concise outline of major activities in the area of safety margins and power up-rates in Canada is given in this presentation. In the past, in Canada there have been more power deratings than increases in the licensed reactor power, a situation caused directly by the erosion of safety margins. Deficiencies in earlier safety analyses, new knowledge gained from experimental research, as well as ageing effects, have led to the need of restoration of margins in order to maintain high power operation. Recently, some of the licensees have embarked on ambitious programs to improve safety and economic performance of their plants; such programs include implementation of significant design changes and may lead to applications for power increase.

1. FACTORS AFFECTING SAFETY MARGINS IN CANDU REACTORS

Because of the positive void reactivity coefficient, Large Break Loss of Coolant Accident (LBLOCA) in CANDU reactors leads to a power pulse, which is terminated by fast acting shutdown systems. In CANDU reactors there are two independent shutdown systems designed to act quickly to minimize energy deposition in the fuel to safe levels. Combined with the action of the Emergency Coolant Injection System, this should prevent or, in the very least, minimize fuel damage in case of LBLOCA. Figure 1 gives a schematic presentation of the major parameters' behaviour during the initial phase of a LBLOCA in a CANDU reactor.

Fig. 1 Typical temperature transients in CANDU LBLOCA



In practice, however, demonstration of the effectiveness of shutdown systems in maintaining fuel and fuel channel integrity has met certain difficulties. In some cases these difficulties were such that reactor power derating was in order. In other words[1], the safety margins to LBLOCA acceptance criteria have been found to be small and, sometimes, inadequate.

One of the more recent difficulties of this kind was associated with the recognition of a reactivity effect associated with the rapid relocation of fuel in channels in the broken pass in case of a LBLOCA. In CANDU reactors fuelled against flow (Bruce and Darlington reactors), a break upstream of the core leads to a shift of relatively less irradiated fuel into the

[1] According to IAEA's TECDOC-1332 [1], the safety margins are defined as "the difference or ratio in physical units between the limiting value of an assigned parameter the surpassing of which leads to the failure of a system or component, and the actual value of that parameter". It then goes on to say that "in many cases, both the limiting value and the actual value are not known precisely. Therefore, for practical purposes, the safety margin is usually understood as the difference … between the regulatory acceptance criteria and the results provided by the calculation of the relevant plant parameter".

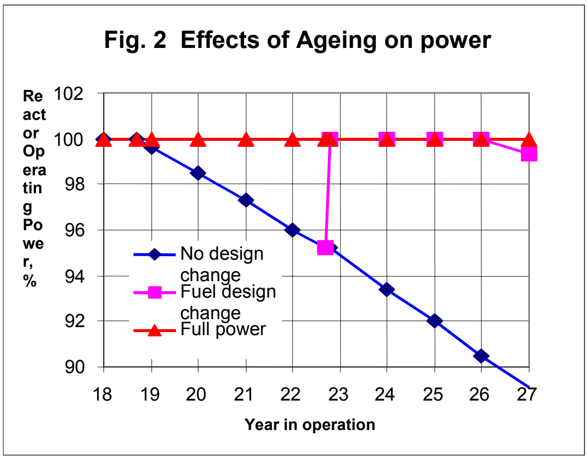
zone of a higher neutron flux. This exacerbates the positive reactivity effect caused by voiding coolant and increases the power pulse magnitude. Following the “discovery” of this effect the utility had to implement a number of compensating measures such as introduction of a new limit on relocation reactivity and imposing more restrictive limits on flux tilts, coolant and moderator isotopic purity, moderator poison concentration, and, most notably, new limits on bundle, channel and bulk reactor powers. In particular, Bruce reactors (8 units at A and B sides) were derated to 60 % full power. A new safety analysis was carried out to confirm that safety margins previously shown in the safety report were still maintained.

Following the discovery of the fuel string reactivity relocation effect and initial reactor power de-rating, the utility initiated a program of activities to return units to full power operation. These activities included further optimization of operational limits, implementation of design changes to limit the relocation reactivity effect, and development of a more systematic safety analysis methodology that relied on mechanistic models and computer codes. As a result of extensive utility efforts to improve safety margins for the LBLOCA, Bruce reactors were allowed to increase power to 90 %, with the intent to eventually return to 100 % full power operation.

However, a new challenge arose. Development and application of mechanistic models and codes, which represent all key phenomena taking place in reactor accidents in order to accurately predict accident consequences, lead to the issue of comprehensive code validation. The Canadian nuclear industry, at the request of the CNSC, implemented an industry-wide, systematic program of code validation. One of the outcomes of this program was a discovery of a substantial under-prediction, by the reactor physics codes used at that time, of the void reactivity effect, and, consequently, of the power pulse in case of a LBLOCA. This discovery necessitated LBLOCA re-analysis for all CANDU reactors and further tightening of operating limits. The CNSC requested all CANDU licensees in Canada to demonstrate the adequacy of safety margins for all design basis events affected by this finding (in particular, for LBLOCA), and, if necessary, to implement measures to improve reactor safety.

Another concern which in some cases has lead and is likely to lead in the future to power restrictions of CANDU plants is related to effects of plant ageing. The time dependent degradation of plant components affects both their functional capabilities and reliability. There are several specific mechanisms related to ageing effects in CANDU reactors; those with most significant impact on safety margins are the pressure tube and steam generator degradation.

Ageing of pressure tubes manifests itself through the irradiation induced creep (axial and diametral), sagging, change of material properties, and mechanical damage (scratching and fretting marks). Steam generators experience fouling of tubing and leaks between the primary and secondary sides. Ageing of both the pressure tubes and steam generators, among other impacts, negatively affect LBLOCA consequences and thus further highlight the issue of adequacy of safety margins.



To counter these effects the life cycle management is in effect at all stations to monitor the extent of ageing degradation and ensures that corrective measures are implemented in a timely manner. However, the life cycle management may not be fully effective in countering the ageing effects and power derating may become a necessity in order to maintain adequate margins (Figure 2 illustrates the impact of ageing effects on the reactor power).

2. APPROACHES TO DEMONSTRATE AND IMPROVE SAFETY MARGINS

Even though the most recent finding concerning underprediction of the void reactivity coefficient affected all CANDU utilities, the extent of the impact varied because of the differences in design and the previously demonstrated margins to acceptance requirements. Because of that, the compensating measures implemented by utilities also varied significantly. Nevertheless, the following general elements of the margin improvement activities can be pointed out:

- *demonstration* of margins as they currently exist by using a modern safety analysis methodology
- *improvement* of margins (where needed) by implementing design changes or adjusting operating conditions
- *redefinition* of margins in the overall safety context, using risk-informed licensing framework

CANDU-6 utilities (Pont Lepreau and Gentilly-2 stations) had comfortable safety margins shown previously in the safety reports and felt that moderate restrictions on some operating parameters would be sufficient to compensate for the underprediction of the void reactivity effect. The licensees' position is that for most likely reactor conditions LBLOCA margins have not changed significantly since the original analysis done in the early 80s. To confirm this position, a safety assessment program has been initiated which

- utilizes modern and validated codes
- accounts for ageing effects (plants are more than 20 years old)
- studies sensitivity of results to various operating and modelling parameters.

The program of activities to improve safety margins undertaken by Ontario Power Generation Inc (OPG) which operates 4 units at Darlington and 8 units at Pickering stations, includes analytical activities (conducting confirmatory conservative analysis and development of the best estimate and uncertainty methodology), evaluation of potential design changes (in particular, new fuel design) and changes to the licensing approach. The latter envisages greater emphasis on risk-informed approach to integrated safety assessment. This may involve re-definition of the role of LBLOCA in the licensing framework and putting more effort towards events with larger contribution to the overall risk.

Bruce Power (2 units at Bruce A and 4 units at Bruce B are operated by this utility) has embarked on an ambitious program of margin restoration, the most significant element of which is the implementation of a new fuel design. A major driver for such an extensive program is the intention to restore the full power operation (from the current 90%).

3. DEVELOPMENT OF SAFETY ANALYSIS METHODOLOGY

The safety analysis methodology is the tool that provides the “calculation results” needed to determine safety margins. This tool evolves with the growing knowledge and computing capabilities; however, the evolution is not necessarily continuous. Practical needs greatly influence the direction and speed of the safety analysis methodology development which, in the Canadian context, have been mostly driven by the LBLOCA analysis.

Following the discovery of the fuel relocation reactivity effect in early 90s, the so-called Limit of Operating Envelope (LOE) safety analysis methodology was developed which replaced the stylized, overly conservative, approach used previously. The centre piece of the LOE methodology is the premise that the combination of bounding assumptions about operating parameter values (set simultaneously at their allowed limits), and mechanistic, best estimate models allows to predict accident consequences that are conservative with high confidence.

The LOE safety analysis methodology essentially remains the basis for licensing of power reactors in Canada, as it is relatively straightforward and provides what is seen as a conservative prediction of accident consequences. However, in cases where safety margins are small and show a tendency to shrink further, application of this approach has encountered difficulties. The LOE approach by design assumes limiting operating conditions, neglects some of the phenomena or systems which would reduce the extent of accident consequences, and incorporates biases which make consequences worse. It is generally accepted that safety margins in reality are larger than shown by the LOE analysis. However, in the framework of the LOE methodology it is not possible to “recover” and quantify these “hidden” margins.

The practice of simultaneously setting all operating parameters at their limiting values was thought to more than compensate for uncertainties in the models and offer an assurance that the analysis results would be conservative. This still holds in most cases; however, for the LBLOCA, as we saw, it was necessary to repeatedly tighten many operating limits so that currently stations routinely operate with certain parameters (e.g. flux tilt, isotopic purity, moderator poison concentration, etc.) not so far away from the limits. This casts the claim of conservatism of the LOE application for LBLOCA into doubt if we also consider the fact that modelling uncertainties are relatively large.

The utilities which felt that the demonstrated safety margins for LBLOCA may not be sufficient to allow flexibility of operation and accommodation of any further “discoveries” have initiated development of a new safety analysis methodology. The new approach is based on the best estimate analysis with integrated accounting of uncertainty. While the primary objective of this development is to provide a basis for systematic quantification of safety margins, there are other objectives, such as better support for definition of operating limits, optimization of compliance activities, prioritization of experimental activities, and effective utilization of earlier analysis results.

The underlying idea for the best estimate analysis is that the combination of the most probable operating conditions and system performance with best estimate models of accident phenomena will result in prediction of realistic plant response and consequences of an accident. To account for variation in operating conditions and uncertainty inherent to even the best available models, a systematic uncertainty evaluation is an embedded part of the new methodology. The approach followed in Canada has many common elements with the CSAU methodology which was provided by the US NRC as an alternative to 10CRF50 Appendix K conservative rules [2]. References 3 & 4 give some further details of the Canadian best estimate and uncertainty safety analysis methodology.

The systematic development of the best estimate method in Canada started in earnest around 1998, and so far there have been several test applications, including the early phase LBLOCA analyses for Bruce, Darlington and Gentilly-2 reactors, loss of flow event for Darlington, as well as analysis of certain accidents for a MAPLE research reactor. An international panel of experts was set up in 2002 to review the methodology and its applications; the panel provided several recommendations which, among others, included:

- improvement of methods for selection of uncertainties
- integration of validation activities with the best estimate analysis
- minimization of systematic errors
- evaluation of sensitivities at more limiting operating conditions
- explicit justification of uncertainties based on comparison with data
- use of diverse representation of modelling uncertainties, etc.

At the same time as the expert panel conducted its evaluation of earlier applications of the best estimate analyses, Ontario Power Generation undertook a new project concerning the LBLOCA at Darlington reactors. Consequently, the panel recommendations were not fully acted upon in this application; however up to date this is the most complete best estimate and uncertainty safety analysis conducted in Canada. CNSC staff is carrying out a review of this analysis to determine whether the methodology is mature enough for licensing purposes. It would not be proper to discuss this analysis and its results while the review is still ongoing; however, it can be mentioned that, comparing with the traditional LOE analysis, the best estimate analysis predicted notable increase in safety margins. For example, the maximum predicted fuel centreline temperature, at the 95th percentile, decreased by about 200^{o}C, and the maximum clad temperature decreased by over 300^{o}C.

As can be seen, the best estimate and uncertainty analysis may indeed allow showing existence of more comfortable safety margins than it is possible with the traditional approach. This encourages licensees to further develop this methodology to be applicable in licensing. While the best estimate safety analysis has advanced significantly in the last few years, in Canada it is still considered premature to use it as a licensing method on its own. At the moment the best estimate analysis is regarded as a supporting licensing tool because there still are a number of difficulties to be overcome. The most significant of these difficulties lies, perhaps, in validation of codes and quantification of modelling uncertainties. The amount of data available for code validation and determination of uncertainties, especially under prototypical accident conditions (in particular, for CANDU LBLOCA) remains limited. Additional validation effort will likely be required to better qualify codes for application in best estimate safety analysis.

4. DESIGN CHANGES TO IMPROVE SAFETY MARGINS

In addition to performing more in-depth and detailed analyses to demonstrate that reactors currently operate with acceptable margins, the industry also undertook a study of design options to improve safety margins. These design changes are meant to provide an assurance against future power de-ratings due to any unexpected “discoveries” or ageing effects, and allow returning to full power operation where restrictions currently exist. In identification of potential design changes a number of factors were considered:

- the size of increase in safety margins
- risks involved with the changes
    - technical robustness (lack of susceptibility to small changes in reactor operating conditions)
    - development effort (the proposed change should not require lengthy development)
    - licensing risk (the proposed design modification should not require changes in the licensing practice)
    - impact on other design basis events
    - impact on the reliability of systems
- cost factors
    - economic cost (including the outage cost)
    - radiation dose
    - impact on operation
    - impact on maintenance

Three types of design modifications are being considered for CANDU reactors with respect to LBLOCA margins:

- changes in shutdown system instrumentation to ensure earlier trip signal
- mechanical/hydraulic changes in shutdown systems to accelerate insertion of negative reactivity
- fuel design change to reduce positive reactivity effects.

One of licensees (Bruce Power) has completed evaluation of design options and selected implementation of a new fuel design (low void reactivity CANFLEX design, Figure 3) as the solution offering the most substantial safety improvements (as well as economic benefits).

The new fuel design includes changes to the fuel bundle (assembly) geometry and introduces enriched uranium (at the moment all CANDU reactors use natural uranium) and burnable neutron absorber. Compared to the current fuel design, the new fuel offers a number of safety advantages:

- reduction of the positive void reactivity coefficient
- lower linear power rating
- lower fuel temperatures
- lower gap inventory of fission products
- increased margin to dryout.

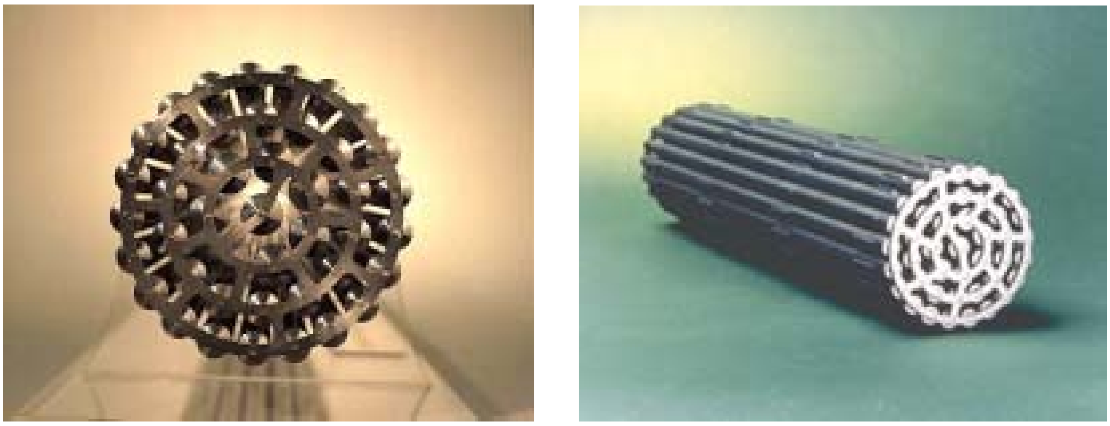


Fig. 3. CANFLEX fuel

It is expected that the new fuel design would reduce the maximum predicted fuel centreline and the maximum cladding temperature, depending on the initial reactor power, somewhere in the order of 300-400 ^{o}C. The design of new fuel is also optimized to off-set effects of the primary heat transport system ageing by increase in the critical heat flux.

Introducing a new fuel design, especially with a transition from natural to enriched uranium, is a costly undertaking. The utility fully intends to use the safety advantages offered by the new fuel in support of the request to increase the licensed reactor power and thus to recoup expenditures while still gaining improvement in safety margins.

Another utility is in the process of detailed evaluation of various options to improve safety margins; one of the considered design changes is the low void reactivity fuel design.

5. REGULATORY CONSIDERATION FOR APPROVING POWER INCREASES

The CNSC staff approach to evaluation and acceptance of proposed changes in design or operating conditions, including power increase, is based on the following general principles:

- All safety impacts of the proposed change should be evaluated in an integrated manner. If several changes are implemented, then the cumulative effect on safety margins should be considered in the decision making process. If reductions in margins are predicted for certain events, the safety benefits from the proposed change should outweigh the anticipated safety margin reductions.

- The scope of design verification and safety analysis activities in support of the proposed change should be appropriate for the nature and scope of the change, based on the as-operated and maintained plant, and should reflect operating experience at the plant. Data, methods, acceptance criteria and assessment results in support of the change must be verified, validated, documented and available for review.

- Programs of surveillance and compliance activities should be established to monitor the effect of changes on plant operation and performance of systems and equipment.

- Any new challenges to safety (such as criticality issues associated with the use of enriched fuel) caused by changes in design or operating conditions should be identified, evaluated and shown to meet all applicable requirements. While it is not a requirement for older plants to fully comply with the modern requirements, a cost-benefit assessment should be performed to identify the cost-effective safety upgrades.

As a rule, it is expected that power increases would be linked to physical plant changes which lead to improvements in safety margins such that the overall risk to the public, personnel, and environment would not increase.

6. POTENTIAL CHANGES IN THE LICENSING FRAMEWORK

The safety margin is an important concept in the modern nuclear power reactor licensing practice. It is invoked in many situations, including applications for power increase, but also in cases of significant back-fitting, refurbishment, re-licensing, etc. There is a general expectation of on-going improvement in safety margins, even though this is not a regulatory, or legally enforceable, requirement. The plants built to earlier standards may not be economical to back-fit to current requirements; a formalized benefit-cost assessment methodology has been developed and used in Canada in licensing to support the extent of design changes for older plants in order to bring them in compliance with the modern standards and improve their safety.

In should be also recognized that safety margins, as defined earlier, may not easily allow an integrated approach to safety assessment. One must keep in mind that, even if safety margins are small for a specific event, that may have little impact on the overall safety if the risk contribution from this event is negligible.

In Canada, both the regulator and industry see advantages in furthering practical application of risk-informed principles in the licensing. The integrated risk-informed approach to safety assessment would allow:

- integration of diverse safety assessment tools (such as deterministic and probabilistic safety assessments)
- strengthening of defence-in-depth by identifying dominant risks
- risk-informed decision-making on plant back-fits and refurbishment
- treatment of outstanding safety issues commensurate with their risk significance
- linking of accident management (in particular, severe accident management) with the outcomes of safety assessments.

The deterministic safety analysis (the basis for the current licensing safety evaluation), probabilistic risk assessment and severe accident management are seen as three major components of the risk-informed, integrated safety assessment. Risk-based safety goals and limits (such as the expected public dose, large release frequency, or severe core damage probability) would need to be established to provide criteria for judging the safety of a facility.

The industry has expressed their opinion that the development of an integrated risk-informed approach to evaluation of safety on nuclear reactors would help bring long-term stability, and, in particular, reduce the risk to future power deratings caused by discoveries that may have a significant impact on some of the design basis events, but only a limited effect on the overall risk. The CNSC is currently in the process of development of a regulatory position on the risk-informed approach to safety assessments.

REFERENCES

[1] Safety Margins of Operating Reactors. Analysis of uncertainties and implications for decision making. IAEA-TECDOC-1332, January 2003.

[2] Quantifying Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large Break Loss-of-Coolant Accident, NUREG/CR-5249, EGG-2659, 1989.

[3] J.C. Luxat, R.G. Huget, D.K. Lau, F. Tran, ADevelopment and Application of Ontario Power Generation=s Best Estimate Nuclear Safety Analysis Methodology≅, in Proceedings of ANS International Meeting on Best Estimate Methods in Nuclear Installations Safety Analysis≅, Washington, DC, November 2000.

[4] H.E. Sills, A. Abdul-Razzak, R.B. Duffy, N. Popov, ABest Estimate Methods for Safety Margin Assessment≅, in Proceedings of ANS International Meeting on Best Estimate Methods in Nuclear Installations Safety Analysis≅, Washington, DC, November 2000.

IMPACT OF NPP KRŠKO POWER UPRATE ON EQ PARAMETERS

T. Bajs, S. Špalj, D. Grgić
University of Zagreb,
Zagreb, Croatia

Abstract

Qualification of equipment important to safety in nuclear power plants ensures its capability to perform designated safety function on demand under postulated service conditions, including harsh accident environment (e.g. LOCA, HELB). This paper presents the brief overview of the environmental qualification (EQ) parameters identification and calculation, as well as the EQ zone maps development for Nuclear Power Plant Krško (NEK) after modernization (SG replacement and power uprate).

1 INTRODUCTION

An electrical equipment age in service and its capability to perform within its specification (especially in harsh environments) is deteriorated. Since the environment is a potential for common cause failures, the purpose of equipment qualification is to demonstrate the capability of safety-related equipment to perform its safety function in aged conditions and under extreme conditions after design bases event (DBE). EQ parameters that characterize these conditions are: temperature and pressure, relative humidity, chemical spray, submergence and radiation. In NEK specific EQ zones were developed that have similar values of relevant EQ parameters, well defined boundaries and unique identification number.

EQ zones were defined on the basis of the EQ parameter values during, normal, abnormal, accident and post-accident conditions. After steam generator replacement and power uprate at NEK (plant modernization) it was necessary to recalculate these values and re-evaluate EQ margins. In this paper update of EQ parameters and zones for NPP Krško after plant modernization is presented.
In year 1999 Faculty of Electrical Engineering and Computing (FER) finished the calculation of environmental parameters and the development of EQ Zones for NPP Krško, [1]. After NEK Modernization Project (year 2000) all calculations should have been reviewed to take into account actual plant state.

2. NEK EQ ZONES BASIS

2. 1 Scope

For the purpose of NEK EQ project FER has analyzed following parameters for normal, accident and post-accident conditions:

- temperature,
- pressure,
- relative humidity,
- chemical spray,
- submergence,
- radiation.

EQ parameters are prepared and corresponding EQ zone boundaries are established for all areas where the relevant electrical equipment is located.

EQ zones are determined on the basis of the EQ parameter values during, normal, abnormal, accident and post-accident conditions. EQ zone can be defined as a plant location with the common values of EQ parameters and typically consists of one or more compartments within the building. EQ zone is a systematic way of specifying environmental conditions needed for qualification of equipment. Each EQ zone has similar values of relevant EQ parameters, well defined boundaries and unique identification number.

2.2 Methods and Assumptions for Environmental Conditions Calculation

On the basis of NEK Updated Safety Analysis Report (USAR), [14], Layout drawings, Plant Systems Descriptions, Master Equipment Component List (MECL) and walk down the safety related equipment was identified and safety function and performance requirements for each equipment (accident duration, long-term post-accident operability, function/environment categories) were determined. Normal and accident service conditions under which equipment must perform its safety function, using all existing operating or USAR data, supplemented with additional calculation are defined.

2.2.1 Thermohydraulic data specification

Temperature and pressure are EQ parameters of primary importance.

Normal operating temperature and pressure were based on USAR, where air conditioning, heating, cooling and ventilation systems are described, and on available operating data. For Auxiliary Building initial temperature is based on monitoring of operating temperature in the rooms.

DBE limiting pressure and temperature profiles must be defined for each location containing safety related equipment. The following methodology has been applied to determine limiting pressures and temperatures for NPP Krško.

Thermal-hydraulic conditions in reactor building and annulus during design basis accidents (LOCA and MSLB) were calculated for the reactor building (containment) in the frame of NPP Krško Modernization project. The results were presented in relevant documents [2], [3]. EQ related data were additionally analyzed and pressure and temperature envelopes were collected in separate document [4]. Methodology and assumptions used in analyses are appropriate for determination of reactor building thermal-hydraulic EQ parameters.

For all other related buildings (intermediate and auxiliary building), new High Energy Line Break (HELB) calculations were performed. All high energy pipes were identified and located including following typical high energy systems: Main Steam System, Main Feedwater System, Auxiliary Feedwater System, Chemical and Volume Control System, Steam Generator Blowdown System, Process Sampling System, and following moderate energy systems: Service Water System, Residual Heat Removal System, Circulating Water System, Component Cooling System. Limiting break locations and duration were determined.

For Intermediate Building (IB) two global areas at different elevations were identified as locations of possible high energy line breaks. In the first location steam supply line break in turbine driven auxiliary feedwater pump C room, and steam generator blowdown processing line break in the steam generator blowdown heat exchanger room were detected. Second location is characterized by main steam and main feedwater line breaks.

For the first location, mass and energy release calculations were performed using both, BE models of Steam Generator Blowdown Break (SGBDB) and Auxiliary Feedwater Pump Turbine Steam Supply Line Break and conservative models from [4]. The pressure and temperature values were calculated using GOTHIC code, [18], [19], using results from both models. For the second elevation MSLB mass and energy releases were taken from [2], while RELAP5/mod3, [17], analyses were performed to define mass and energy release in the case of Main Feedwater Line Break (MFLB). Again, pressure and temperature values were calculated using GOTHIC code and the final values were compared to the results given in [4].

HELB in the CVCS system was analyzed for two specified locations in the Auxiliary Building (AB). Mass and energy release were calculated using RELAP5/mod2, [16], computer code after the evaluation of the systems operation. The following assumptions were taken into account: all lines are at the maximum operating pressure and temperature, guillotine break happened and flow from both ends is considered when appropriate. Temperature and pressure profiles were developed for each postulated HELB location using GOTHIC code, and envelope is prepared for specified rooms in the AB**.** The final values were again compared to the results from [4]. Since a HELB is not postulated to occur during a LOCA, break in the SI, RHR and CS system was not considered during recirculation.

The environmental impact of fluids that are recirculated from inside containment to accomplish long term post LOCA core cooling were calculated for Auxiliary Building (AB). Basic requirement in that case is the selection of safety systems, heat sources and heat sinks in the considered rooms. Typical systems required to function after DBE are Containment Spray (CI), Residual Heat Removal System (RHR) and Chemical and Volume Control System (CVCS) (pipes with recirculating fluid). Usual assumption is loss of normal Heat Ventilation and Air Condition System (HVAC) in post LOCA condition. Simple two-step calculation methodology has been used for temperature calculation:

- simple geometry of heat sources and sinks in the room (list pipes according to wall thickness, insulation characteristics, flow rate and water temperature),
- simple lumped parameter approach or GOTHIC code calculation of temperature increase based on containment sump water temperature.

2.2.2 Radiological data specification

Normal operating dose rates were again extracted from corresponding USAR chapter, where the plant locations are divided into five occupancy zones,
Table 1. In determining these zones, the plant is considered to be operating at normal full power with the equivalent of one percent fuel cladding defects. Those dose rates were integrated over the period of 40 years, taking the maximum value of the dose rate in zones I to IV.

For rooms in zone V (restricted access) the normal operating dose was calculated using QADUE and DIDOS computer codes (gamma dose rate calculation for different configurations of pipes with contaminated water and for different locations - point kernel theory). The necessary data and assumptions for the calculations (radiation source and geometry for the specific plant system) were taken from USAR.

The accident dose source in all NEK areas is due to LOCA, whether it is caused by the direct radioactive release to the atmosphere (containment), or it is caused by the recirculation of radioactive fluids in the pipes (including ventilation system), or by the radioactive containment atmosphere through containment walls (for other buildings).

Zone	Occupancy	Dose Rate (mSv/hr)	Access
I	No restriction on access	< 0.0025	Uncontrolled
II	Occupational access	0.0025 – 0.025	Controlled
III	Periodic access (4 hr/wk)	0.025 – 0.25	Controlled
IV	Limited access (1 hr/wk)	0.25 - 1.0	Controlled
V	Restricted access	> 1	Controlled

Table 1 NPP Krško normal operation occupancy radiation zones

The "NPP Krško Post-Accident Shielding Review" document and its update, [5], [6] were analyzed and used for gamma dose determination in the locations of the plant where applicable, i.e., in the Auxiliary Building, Intermediate Building, Component Cooling Building and Control Room. The contribution to the gamma dose, depending on the location, comes from the following sources: containment, containment penetrations, piping with recirculated primary coolant (RHR, CVCS, CI), heat exchangers (RHR, letdown, seal water), hydrogen control system, boron injection tank, volume control tank, hold-up tank, mixed bed demineralizer, gaseous waste processing system, reactor coolant and seal return filter, seal injection filter, annulus negative pressure system filter and fuel handling building ventilation filter. Since metal and concrete are very effective shields for beta particles, beta radiation is not considered to contribute to the total dose in the AB.

Containment accident doses were determined using source term calculated by ORIGEN 2.1 for the fission product inventory from the end of full power 18-month equilibrium cycle. Release fractions to containment air and primary coolant were taken according to Standard Review Plan requirements. Containment volume and/or additional volume for dilution, as well as coolant volume for dilution (including additional water sources e.g. RWST) were taken into account in the ELISA, [20], code calculation. Code calculates daughter product buildup for noble gases, decay removal, and external removal mechanisms (spray, RCFC, purging & filtration). The calculation was performed for gamma, as well as for beta doses, because the beta radiation affects only non-shielded equipment in the containment. The containment annulus accident dose was also determined with ELISA computer code, [20], taking into account the conservative containment leak defined in USAR.

The accident dose in the containment sump was calculated using QADUE computer code (gamma dose rate calculation using point kernel theory). The fraction of the iodine fission products removed by the spray (determined with ELISA calculation) was considered as the sump source.

2.2.3 Submergence

Flooding level, as a harsh environmental parameter, resulting from LOCA and HELB, is identified for the containment and for the other buildings where the break can occur. Submergence is usually based on conservative assumptions about flood volume and on specified area geometry and drain characteristics of the room, so only simple calculation is needed.

For the evaluation of flooding level in containment the inventory of the reactor coolant system, refueling water storage tank and boron injection tank is considered.

For other plant areas amount of discharged water due to analyzed HELBs is expected to stay within rooms design limits (drainage capacity and/or configuration of areas where the breaks are located), so submergence is not postulated as harsh environment parameter.

2.2.4Relative humidity

Humidity has no significant influence on the degradation of electrical equipment during DBA. In accordance with the DOR Guidelines, a humidity of 100% and condensing, as a result of exposure to a saturated steam during LOCA and HELB, should be considered as a harsh environment. Outside containment usual candidates for harsh environment due to humidity (100% RH and condensing) are rooms where break is located. For AB and IB rooms practical limiting criteria for condensing on energized electrical equipment could be 100% RH and atmosphere temperature above 50 ^{o}C.

2.2.5 Chemical spray

The effect of containment spray system should be considered during DBA since it results in a harsh environment. Chemical spray composition is specified in the Plant Technical Specification.

3. RESULTS

3.1 Reactor Building

The two limiting accidents for Reactor Building, LOCA and MSLB were taken from [4]. The containment pressure and temperature values were additionally checked for the same mass and energy inputs by independent GOTHIC calculations. The peak containment pressure as used in pressure envelope is 410 kPa,

Figure 1. The corresponding peak containment temperature is 162 ^{o}C. Significantly higher maximum containment temperature,

Figure 2, (compared to one before modernization) is determined by small, dry steam line breaks and it is limiting parameter for environmental qualification of electrical equipment in containment. The peak temperature (105^{o}C), and pressure (105 kPa) in annulus envelopes are again higher than before due to larger amount of heat released to the containment and due to higher temperatures experienced in containment atmosphere.

In addition to two design bases accidents, the CVCS line break was calculated inside RB to evaluate the possible extension of high temperature duration and to calculate the flood level in order to determine whether the CVCS isolation valves are submerged.

Table 2 presents the results of radiological calculation for Reactor Building. The differences before and after uprate are due to changed core inventory and different limiting assembly burnup assumed in the calculation.

		BETA AIR CUM. DOSE (GY)	GAMMA AIR CUM. DOSE (GY)
CONTAINMENT	Before Uprate	3.179E+06	3.282E+05
	After Uprate	3.585E+06	3.625E+05
ANNULUS	Before Uprate	2.618E+05	1.521E+04
	After Uprate	2.836E+05	1.636E+04
SUMP	Before Uprate	N/A	1.4826E+05
	After Uprate	N/A	1.6166E+05

Table 2 One-year post LOCA Dose in Reactor Building

3.2 Intermediate Building

Limiting accidents for the IB lower elevation are Steam Generator Blowdown Break (SGBDB), with break locations in three rooms, and Auxiliary Feedwater Pump Turbine Steam Supply Line Break (AFWPTB) affecting only turbine driven AF pump room (the room is separated from rest of the building through its door Δp characteristics).

In order to cover the consequences of different sizes of Steam Generator Blowdown Break two limited cases have been analyzed:

1. break size selected to delay the blowdown isolation to approximately 1800 sec when manual operator action is assumed and
2. maximum break size (double ended guillotine break) which maximizes the flow rate, but the blowdown isolation is the fastest

There is a great difference of Steam Generator Blowdown discharge between old and new NEK Steam Generators,

Figure 3. Old SG had only one blowdown exit pipe with nozzle diameter 2.54 cm, while the new one has two blowdown exit pipes with larger diameter of nozzles. Therefore the mass and energy release from the new SG during SGBDB is higher. New pressure increase,

Figure 4, what is usually not limiting, is higher and faster due to faster opening of the break, higher mass flow rates and droplets presence which improves evaporation of superheated water.

AFWPTB in turbine AF room has no influence on any other room because of door type and the blowdown panel that allow the connection to the outside air. All other rooms are assumed to be in mild environment during this break. There is no change of EQ parameters after NEK uprate (Figure 5).

MSLB, MFLB and AFWPTB are limiting transients for the IB at higher elevation. MSLB mass and energy releases were taken from [4]. MFLB mass and energy releases were calculated with RELAP5/mod3.2 The ambient conditions were determined using GOTHIC computer code for all accidents. All rooms with feed/steam lines are almost equally affected by mentioned breaks. MSLB is responsible for peak temperature.

The calculated peak temperature,

Figure 6, is lower than one before modernization due to different calculated MSLB energy releases (lower superheat in calculated mass and energy release). AFWPTB,

Figure 5, has influence in later phase of the accident as a result of constant superheated vapour mass flow and later blowdown panels opening. Consequence of this is the extension of high temperature values (~155 ^{o}C) to 1800 seconds when operator action (isolation of the AFW pump steam supply line) is assumed. Other IB rooms at that elevation, not containing feed/steam lines are assumed to be at their maximum normal operating parameters.

There are no postulated breaks and there are no significant heat sources in the IB at elevation 115.55, so the ambient for all rooms at that elevation is considered mild regarding pressure and temperature.

The accident radiation conditions in the IB are extracted from [5], taking into account contribution from containment, main steam lines, Main Control Room ventilation filter and HVAC as radiation source. The gamma dose depends on the distance from the sources and on the thickness of wall(s) between the source and the detector. No beta radiation is assumed to exist in the IB because it is shielded by the containment wall. Additionally, for the specification of doses after NEK modernization, the corrections regarding power uprate, increase in water inventory and increase in primary to secondary leak were accounted, as already described earlier.

3.3 Auxiliary Building

The only postulated HELB in the AB is Letdown line break in the CVCS (two break locations). The mass and energy release was determined using RELAP5/mod2 computer code. The possible locations are upstream of the letdown heat exchanger and after that the break is not considered to be HELB. The break also has the significant influence on ambient parameters in most AB rooms, since they are more or less interconnected and there is no identified discharge path to the environment. Using before mentioned criteria only two rooms with potential break locations have relative humidity as harsh environment trigger (100% RH in the volume and atmosphere temperature above 50 ^{o}C).

The influence of NEK uprate on mass and energy release is negligible. The differences in pressure and temperatures result from different discharged fluid characteristics (droplets presence), redefinition of flow paths and door opening criteria and introduction of leakage paths to the environment (influence on long term behavior). The example of the temperature envelope for AB is given in Figure 7 for one of the rooms with postulated breaks. As usual, when suitable discharge path to the environment is not identified conservative long term behavior is predicted.

The effects of recirculated fluids that accomplish long term post LOCA core cooling were calculated for Residual Heat Removal System (RHR) and Containment Spray (CI) piping. The rooms that contain pipes with recirculated fluids are identified and GOTHIC calculation was performed with the following assumptions: recirculation starts as defined in [3], recirculation fluid has temperature profile calculated in [3] (sump temperature) and the normal HVAC was lost. The recirculated fluid has the long-term effect (up to the 1 year) on ambient parameters, but the influence is much less severe than the HELB and it does not lead to the harsh environment conditions.

Heatup from the safety related pumps and pump motors in rooms where essential HVAC exists was not calculated, so the EQ parameters in those rooms are assumed to be at the maximum normal operating values.

Normal operating dose rates were extracted from corresponding USAR chapter. Those dose rates were integrated over the period of 40 years, taking the maximum value of the dose rate in zones I to IV. For rooms in zone V (restricted access) the normal operating dose was calculated using DIDOS4 computer codes with the assumptions (radiation source and geometry for the specific plant system) taken from USAR.

Additionally, for the specification of doses after NEK modernization, the corrections regarding power uprate and increase in water inventory were accounted, as described before.

4. CONCLUSION

NEK modernization that included power uprate and SG replacement required that all possible changes should have been taken into account in the frame of the NEK EQ project, FER has made the review and re-evaluation of all EQ parameter calculations, and, accordingly, most of the calculations were repeated taken into account new NEK parameters. Some of calculations were within the scope of the NEK Modernization project, and they were directly used as final results or as the input to FER calculations. Other calculations were performed by FER independently (i.e. SG Blowdown Break, Calculation of doses in Reactor Building etc). Overall differences resulted from NEK modernization project are small, have expected trends, as in case of containment response, and had negligible impact on final NEK EQ zoning.

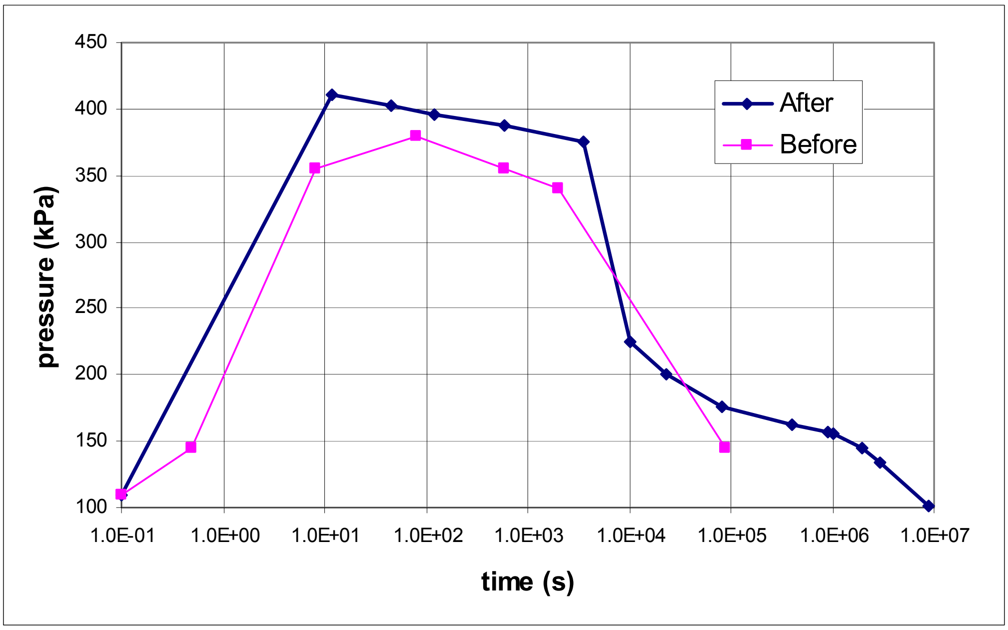


Figure 1 Containment Pressure Envelope

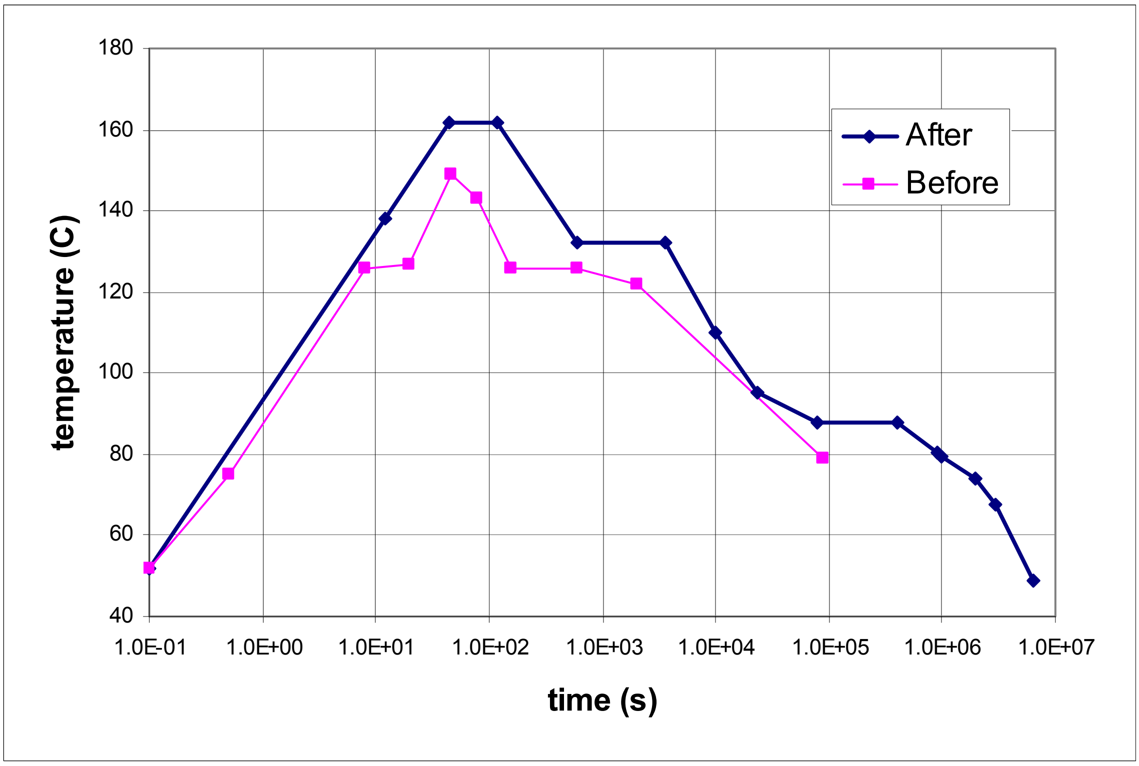


Figure 2 Containment Temperature Envelope

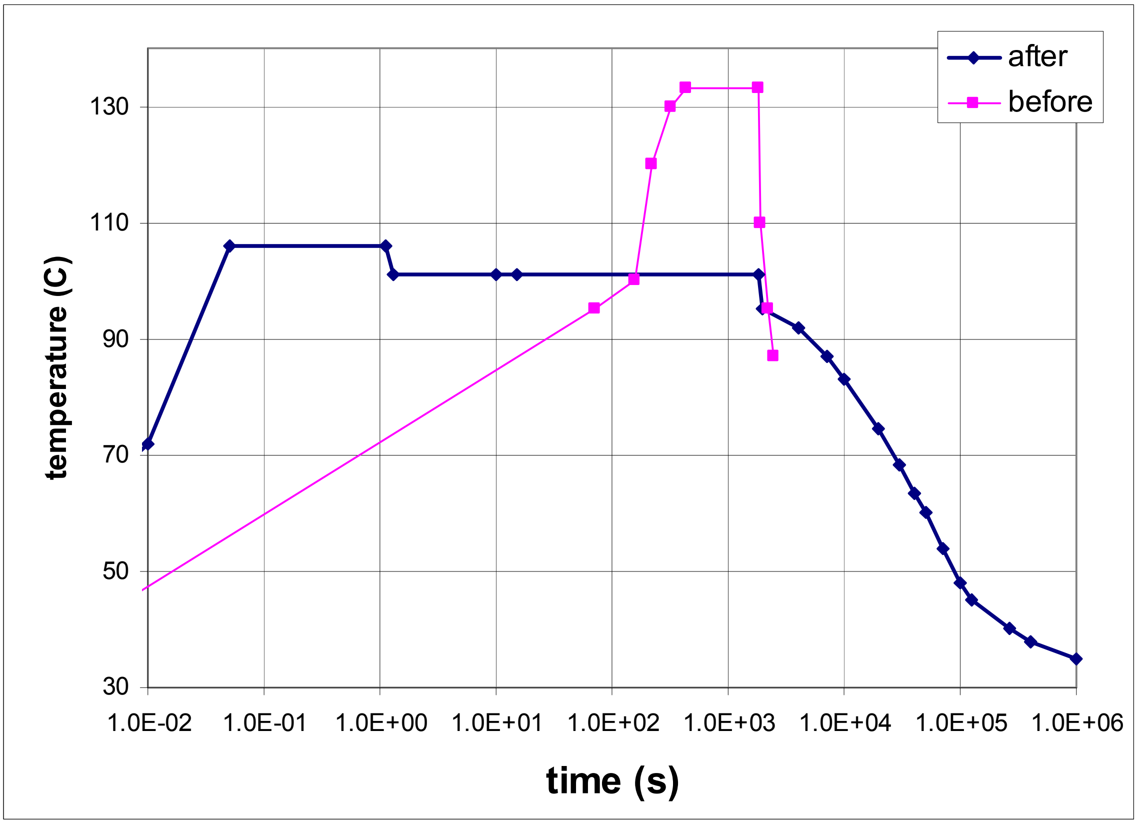


Figure 3 Temperature for Steam Generator Blowdown Break (SGBDB) in IB rooms with potential break locations

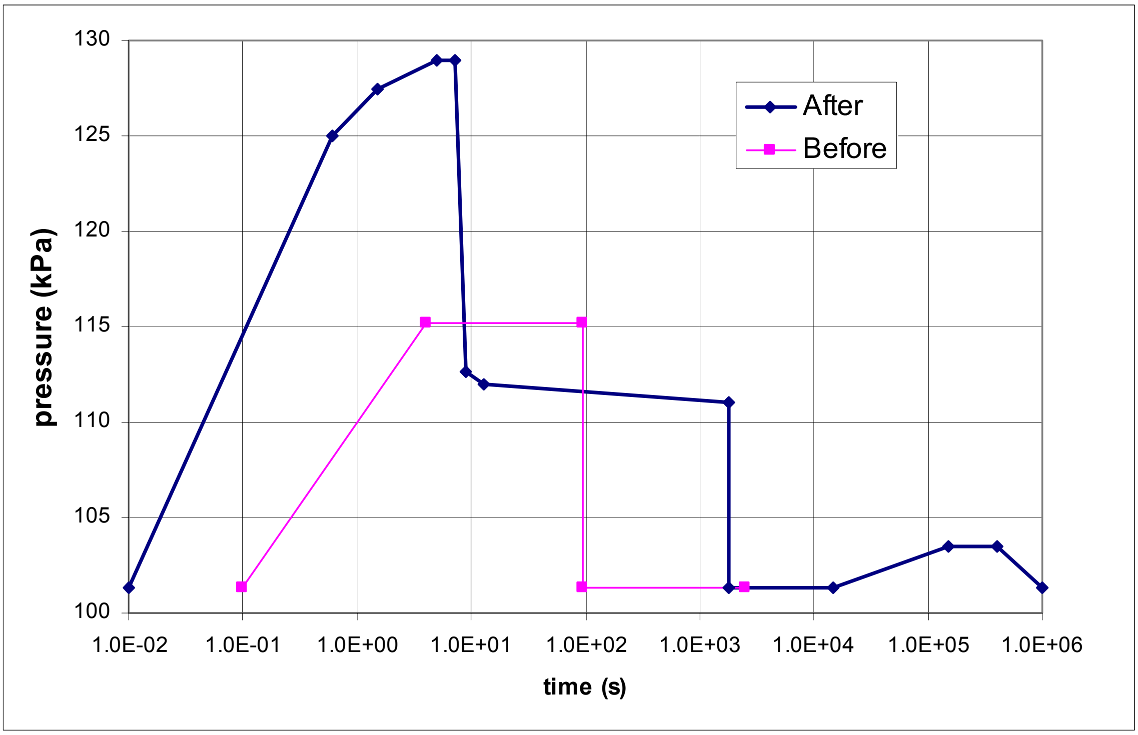


Figure 4 Pressure for Steam Generator Blowdown Break (SGBDB) in IB rooms with potential break locations

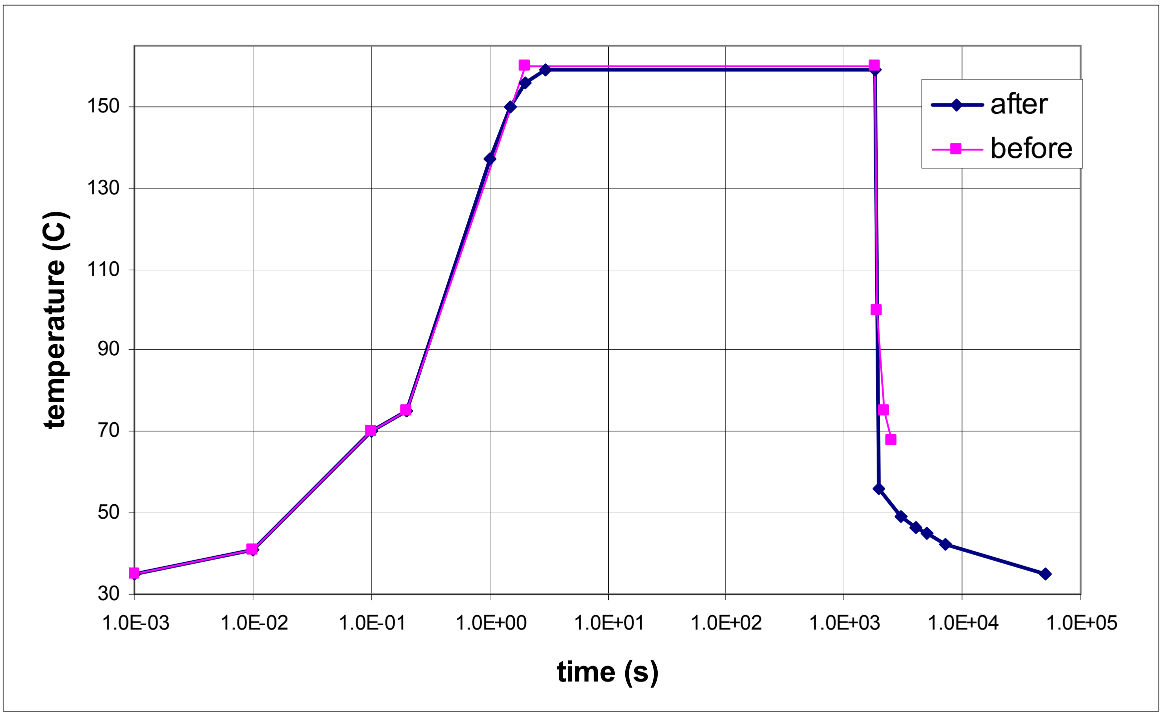


Figure 5 Auxiliary Feedwater Pump Turbine Steam Supply Line Break (AFWPTB) in the turbine driven AF pump room

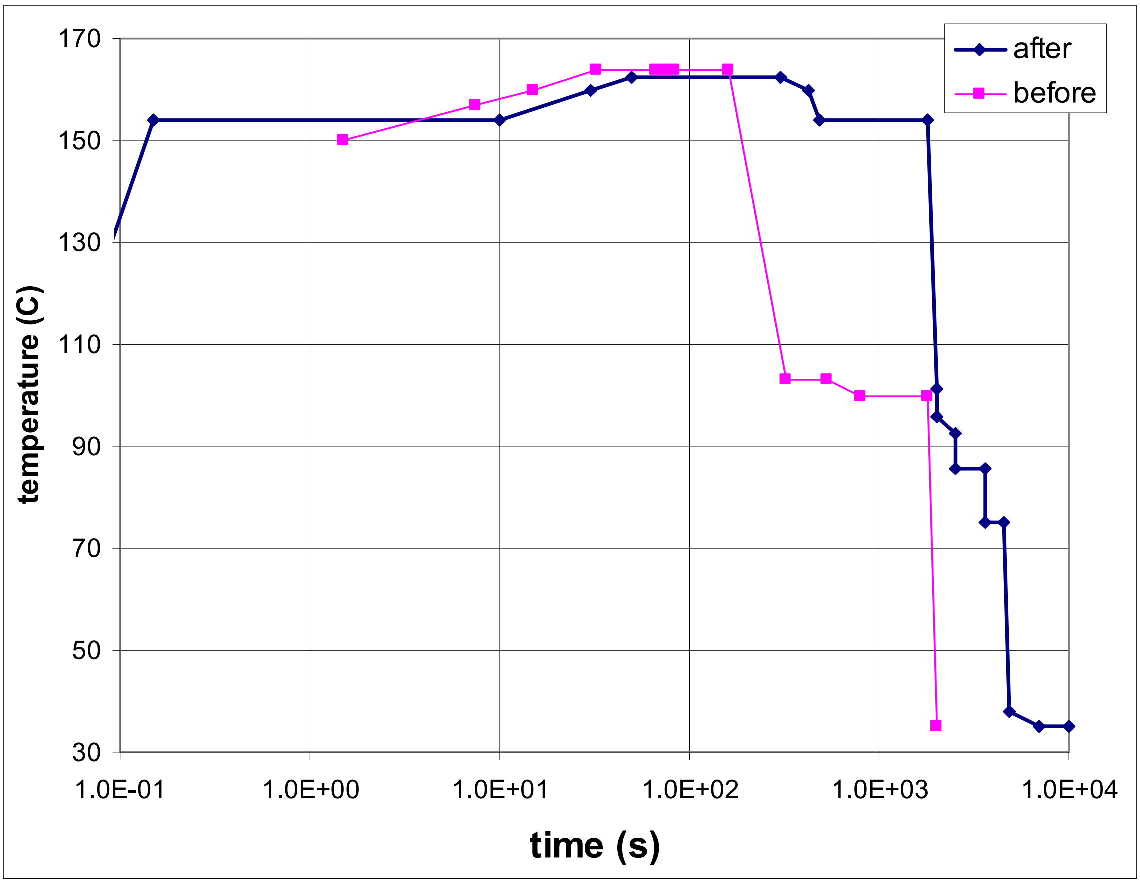


Figure 6 MSLB, MFLB and AFWPTB limiting temperature profiles for IB

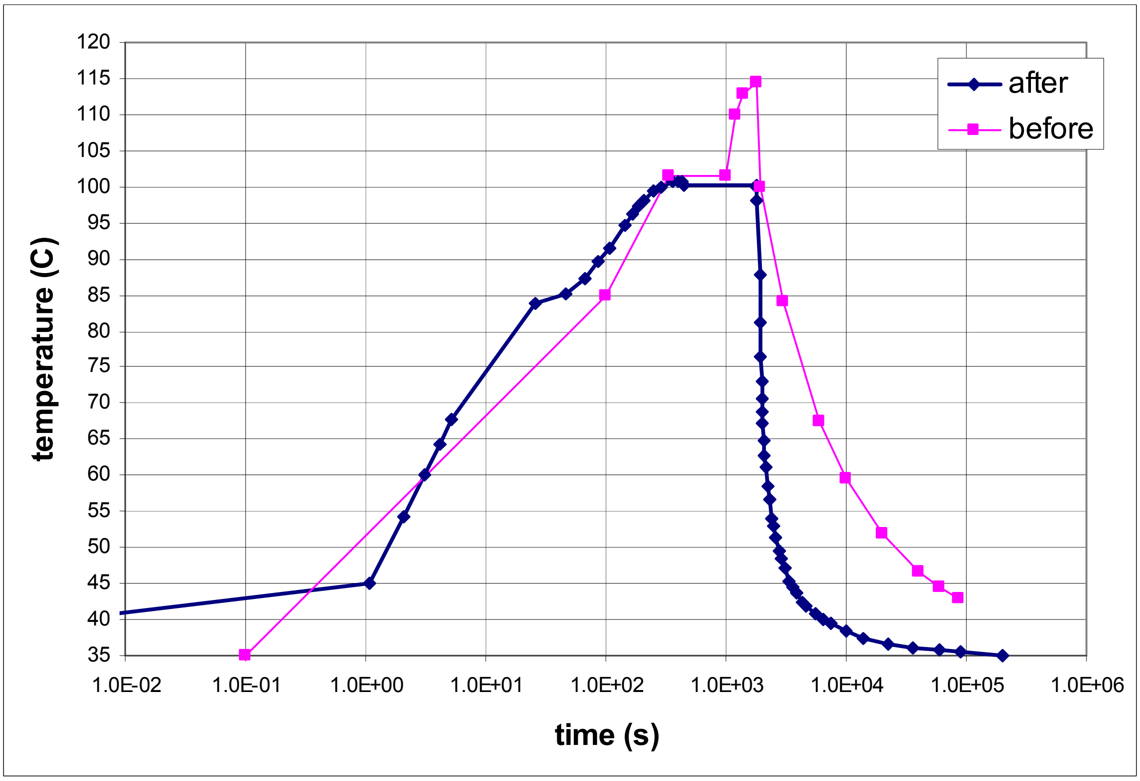


Figure 7 Typical CVCS Line Break temperature profile in room AB room with potential break location

REFERENCES

[1] Development of Environmental Qualification (EQ) Zone Maps and Conditions (Določitev EQ področij in parametrov), Rev. 1, 1999 (Report number: FER-ZVNE/SA/EQ-TR10/99-0)

[2] Containment Response to Steam Line, SSR-NEK-6.3.2, Rev. 0, Westinghouse, February 2000

[3] Containment Response to LOCA, SSR-NEK-7.8, Rev. 1, Westinghouse, April 2000

[4] Thermohydraulic Inputs for NEK EQ Program, SSR-NEK-17.0, Rev. 1, Parsons Power, March 2001

[5] WENX 94/27 "Krško Nuclear Power Plant Post-Accident Shielding Review", 1995.

[6] Addendum to the WENX 94-27, "Krško Nuclear Power Plant Post-accident Shielding Review", SSR-NEK-20.2, Rev. 1, Westinghouse February 2001.

[7] 10 CFR 50.49, EQ Rule, "Environmental qualification of electric equipment important to safety of nuclear power plants"

[8] Regulatory Guide 1.89 "Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants",

[9] Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident",

[10] NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment",

[11] IEEE Std 323 (1971,1974), "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations", "IEEE Trial Use Standard: General Guide for Qualifying Class 1E Equipment for Nuclear Power Generation Stations",

[12] IEEE Std 627-1980 - IEEE Standard for Design Qualification of Safety System Equipment Used in Nuclear Power Generating Stations

[13] Nuclear Power Plant Equipment Qualification Reference Manual, TR-100516, Project RP1707, EPRI November 1992.

[14] NEK Updated Safety Analyses Report, Rev. 7, Krško 2001.

[15] NEK Technical Specifications, rev. 80, Krško 2001.

[16] RELAP5/mod2 Users Manual, H.G. Ransom et al., NUREG/CD 4312, EGG-2396, Idaho National Laboratory, INEL, August 1985.

[17] RELAP5/MOD3.2.2Gamma Users Manual, The RELAP5 Code Development Team, Formally NUREG/CR-5535, Scientech, Rockville - Maryland, Idaho Falls - Idaho, June 1999.

[18] EPRI TR-103053-V1, GOTHIC Containment Analysis Package, Version 3.4e, Volume 1: Technical Manual, October 1993.

[19] EPRI TR-103053-V2, GOTHIC Containment Analysis Package, Version 3.4e, Volume 2: User's Manual, October 1993.

[20] J. N. Hamawi: ELISA – Technical Description, Entech Engineering, INC. April 1991.

POWER UPRATES OF PWRs IN GERMANY

H. Glaeser
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH,
Forschungsinstitute,
Garching, Germany

Abstract

Utilities intend to minimize the cost of production and to maximize outputs of the operating plants. These goals can be achieved by technical measures, such as power up-rating. Such a plant modifications require an in-depth safety analysis to evaluate the possible safety impact. The analysis has to consider all the consequences of the plant modification with respect to the margins existing. The analysis must consider the core characteristics and the plant behaviour, taking into account the capability of the systems (e.g. cooling systems, electric power, heat sinks) and the reactor protection system set-points. One key issue in improving the plant operating performance is the accurate determination of the available plant margin. In relation to thermal-hydraulic and fuel analysis margin can be characterized as the difference between calculated parameter values (e.g. peak fuel clad temperature, maximum reactor coolant system pressure, etc.) and the associated regulatory licensing limit. Their determination includes considering the examining tools and methodologies (conservative versus best estimate approach), the applicability and quality of computer codes, the prediction capability and uncertainty evaluation, the licensing criteria, and the accuracy of plant measurements. The term "safety margin" of operating reactors is often defined as the difference between a calculated parameter and the associated regulatory licensing limit [1]. Sometimes, the term is used for the distance of a calculated value to a safety limit, like loss of coolable geometry of the core. Margins assure that nuclear power plants (NPPs) operate safely in all modes of operation and at all times. Reducing such a margin to zero, i.e. to a licensing limit, is not implying a reduction of the "safety". The most important margins relate to physical barriers against release of radioactive material, such as fuel matrix and fuel clad. The situation in Germany with regard to applied and approved power uprates of pressurized water reactors (PWR) is described. The implications on evaluation of margins to licensing criteria are presented.

1. INTRODUCTION

Power uprates have been applied by several German utilities. It is claimed that sufficient margin is available by the design, especially of modern German pressurized water reactors. Relevant aspects of safety evaluation of power uprates by the assessors are presented. Satisfying conventional and nuclear licensing criteria has to be demonstrated. Means and methods will be briefly described.

2. MEASURES TO PERFORM POWER UPRATES

Firstly, as consequence of a power uprate, the water and steam loop including turbine and generator shall be capable to master the increased feed water and steam flow rates. Beside theoretical considerations pre-tests with increased power are performed to check the plant behaviour.

Different possibilities are available for power uprates of a PWR. On the primary side the core temperature difference can be increased. A decrease of the secondary steam pressure allows to increase the temperature difference between primary and secondary side. Turbine valves must be opened to enable an increased steam flow, which limits the control range of the valves. The high pressure part of the turbine must be changed for increased steam flow. Another or additional way is to increase the primary side average coolant temperature, and to increase the secondary steam pressure which improves the control range of the turbine valves. Limitations may be an increased corrosion of the fuel rod clad which is dependent on temperature. In most cases power uprates are performed by an increase of average coolant temperature. Examples of power uprates of German PWRs are given in table 1.

Table 1: Power uprates of German PWRs

Power plant	Thermal power (MW)	Uprate to (MW)	Year
Grohnde	3765	3850 3900	1990 1999
Philippsburg 2	3765	3850 3950	1992 2000
Neckarwestheim 2	3765	3850	1991
Emsland (in licensing process)	3765	3850 3950	1991 2003
Isar 2	3765	3850 3950	1991 1998
Unterweser	3733	3900	2000
Grafenrheinfeld (in licensing process)	3765	3950	2003
Brokdorf (in licensing process)	3765	3850	2003

3. EVALUATION OF SAFETY

According to §7 of the German Atomic Energy Act (AtG) the utility has to provide evidence that the protection against damage due to operation of a plant is still warranted after a power uprate, according to the state of science and technology. It has to be demonstrated that the requirements of the licensing rules are met after a power uprate.

3.1 Evaluation of operational experience, evaluation of plant pre-tests

Experience of past operation, of components, electric and control systems are evaluated. Emphasis is given to the reactor core with regard to neutron kinetics, thermal-hydraulics and structural behaviour. Another issue is radiation protection. Basis is personal inspections, as well as monthly and yearly reports of the utility. Objective is to evaluate if some aspects do not allow approval of power uprates.
An evaluation of operational pre-tests at increased power are a basis for considering parameter values under these conditions.

3.2 Effect on safety relevant components and systems

Essential areas to demonstrate the acceptability of power uprates is the effect on:

- the reactor core,
- relevant systems and components.
-

An important point is the effect of power uprates on core behaviour. All safety relevant parameters of reactor kinetics, thermal-hydraulics and thermo-mechanical design have to be checked. The safe behaviour of the actual core has to be evaluated and demonstrated for every actual core load.

With regard to systems and components two aspects are treated: The thermal-hydraulic behaviour and the structure of the components. Considered are – according to the changed parameters – the reactor coolant system with pressurizer relief system, control systems, ventilation systems, generator with own electricity supply, and the complete safety system. Due to the increase in power essential points are the release of decay heat, pressure relief of reactor cooling systems and steam generators, as well as increased water and steam flow rates. Safety systems of interest are:

- nuclear emergency and residual heat removal system,
- pressurizer relief system,
- emergency feed-water system,
- borating system,
- reactor vessel,
- steam valves,
- secondary side relief system,
- containment

Based on the existing design, all limits have to be checked, usually by new analyses of the design basis accidents.

3.3 Effect on plant behaviour during normal operation

The new power map, the plant behaviour at operational control set-point limits, the radiological inventory under normal operation due to the power uprate is investigated. The radiological increase is roughly proportional to the increase in power. Usually, radiological values are still far below the limits.

3.4 Effect on plant behaviour during operational transients and design basis accidents

The essential influence on operational transients and design basis accidents from power uprates are due to increased average coolant temperature, lower departure from nucleate boiling (DNB) ratios, and different reactivity coefficients (shut-down system must work under higher reactivity conditions). Consequently, the following transients and loss of coolant accidents (LOCA) have to be considered:

- Reactivity initiated accidents (RIA),
- operational transients, disturbances of coolant flow and heat release,
- total loss of feed-water,

- station blackout,
- steam generator tube rupture
- anticipated transients without scram (ATWS),
- LOCA,
- accidents with radiological consequences to the environment.

The DNB ratio is decreased in the case of transients, however, it is still above the design value of 1.15.

Reactivity initiated accidents are covered by the core design, also at increased power. Operational transients and LOCA need to prove that licensing limits are not exceeded. For example, during operational transients with scram actuation, the coolant pressure shall not exceed the pressurizer relief set-point. Beside these requirements from the RSK Guidelines (Reactor Safety Commission) conventional licensing requirements to limit the pressure of primary and secondary systems have to be followed.

With regard to demonstrate the performance of the emergency core cooling systems (ECCS) licensing requirements are also checked:

- maximum fuel rod clad temperature (1200°C),
- maximum clad oxidation (17% of the total clad thickness before oxidation),
- maximum hydrogen generation (1% of the hypothetical amount that would be generated if all the metal in the clad cylinders surrounding the fuel, excluding the clad surrounding the plenum volume, were to react),
- coolable core geometry,
- long-term cooling.

Calculations from the utility are required, e.g. considering loss of condenser, station blackout and turbine trip without steam dump system. Independent analyses are conducted by the assessors for confirmation. It could be demonstrated that the licensing requirements are fulfilled. Higher fuel rod clad temperatures are calculated, as well as higher fuel rod failures. According to German licensing requirements fuel rod failures shall not exceed 10% of the fuel rods during transients and LOCA. The highest number of fuel rods are calculated when a cold leg large off-set shear break of a main coolant pipe is assumed.
A demonstration to stay below 1.3 times of the design pressure during ATWS is needed. This is performed for the scenario “loss of main feed-water”. Pump trip reduces the core flow rate quickly, the coolant density decreases, and, consequently, the reactor power decreases. This requires a pump trip signal on ATWS indication as reactor control system which is available in some German reactors. A different core design (steep void curve) is needed when the pump stop is available as operation signal. The Reactor Safety Commission asks for core design that does not require a shut-down of the main coolant pumps. Investigations are underway to clarify the core behaviour under different boundary conditions (e.g. valve responses) for a loss of main heat sink ATWS.

The analyses presented by the applicants were also performed by the assessors, mostly using the thermal-hydraulic code ATHLET [2] and the fuel rod code TESPA [3] to compare with their results. ATHLET and TESPA are developed by GRS, the developments are sponsored by the German Ministry of Economy and Labour (BMWA).

The radiological consequences due to LOCA may be increased, dependent on fuel rod failures. They are still far below accident design values of §28, part 3 of the Radiological Protection Ordinance (StrSchV).

3.5 Effect on plant behaviour during beyond design basis accidents

Power uprates have some influence on emergency operating procedures during beyond design basis accidents. The question is if secondary and primary side accident management procedures are still effective. That concerns mainly the time to initiate secondary side bleed. That time is significantly reduced on plants with steam generator pre-heater chambers. The probability to perform secondary side bleed with success is reduced. An optimization of these accident management procedures is recommended.

4. METHODS OF ANALYSES

Independent analyses are conducted for confirmation by the assessors, if they deem it necessary, as part of their safety assessment. In many cases the assessors are applying different codes than the applicant. The analysis with different codes is a common German practice, not required, in order to identify the influence of codes or to assure the quality of the plant input decks. The assessors are mostly using the thermal-hydraulic code ATHLET [2] and the fuel rod behaviour code TESPA [3]. Code validation as well as various studies on evaluation of representation and plant data uncertainties and sensitivity studies help to establish confidence in the predicted NPP behaviour. It could be demonstrated that the licensing requirements are fulfilled.

Uncertainties are introduced to the calculation both through the computer code and through input data for the code. Two different categories of input data are distinguished: Data related to assumptions on availability of plant systems (normal operation systems, control systems, safety systems, i.e. single failure, loss of power supply, plus unavailability due to preventive maintenance in German licensing, and to all other NPP initial and boundary conditions. Typical examples of conservative assumptions on availability of NPP systems are non-operability of normal operation systems and control systems in accident situations, adoption of the worst single failure criterion for safety systems, and combination of an initiating event with loss of power supply in some cases.

German licensing practice is to use a best estimate code, conservative assumptions on availability of plant systems and conservative initial and boundary conditions. This has been considered as acceptable up to now (and is also suggested by the existing IAEA Safety Standards). It is still typically used at present for safety analysis in many countries, is reasonably established and its use is straight forward. In some cases just one calculation is sufficient to demonstrate safe conditions.

However, this procedure is not allowed in the USA according to the Code of Federal Regulation (CFR), for example. The 10 CFR 50.46 [4] allows either to use a best estimate code plus identification and quantification of uncertainties, or the conservative option using conservative computer code models. In many cases, conservative approach is used to avoid the cost of developing a realistic model. However, this approach provides only a rough estimate of the uncertainties, many preparatory calculations are often needed to support conservative selection of input data, and still intentional conservatism may not lead to conservative results. An example is an assumption of high power during small-break loss of coolant accident (SB LOCA), which may over-predict swell level in the core, and this leads to better core cooling, opposite to conservative requirement. Different sets of conservative assumptions are typically required for each of the acceptance criteria, and different assumptions may even be needed for different time periods of a transient.

Some world wide interest is in facilitating broader use of fully best estimate analysis, using best estimate codes, realistic input data with uncertainties, and still conservative assumptions on availability of systems. Best estimate analysis with evaluation of uncertainties is the only way to quantify the existing margins. This is especially the case for approaching licensing limits, e.g. due to power uprates, higher enrichment and higher burn-up. Broader use of best estimate analysis is therefore envisaged in the future, even though it may be difficult to quantify code uncertainties with adequate (narrow) range for potentially important phenomena. The uncertainty analysis has to be accepted by an assessor. Since assumptions regarding availability of NPP systems are still used in a conservative way, there is still conservatism in best-estimate analyses. Methods of uncertainty analyses have been developed by the vendor Framatome ANP [5] as well as by GRS, sponsored by the German Ministry of Economy and Labour (BMWA) [6].

An example of the results of an uncertainty analysis is shown in Figure 1. A double ended cold leg offset shear break design basis accident of a German PWR of 1300 MW electric power (not uprated) is investigated. Loss of off-site power at scram is assumed. ECC injection is into cold and hot legs. The accumulator system is specified to initiate coolant injection into the primary system below a pressure of 2.6 MPa. High and low pressure ECC injection is available. A single failure is assumed in the broken loop check valve, and one hot leg accumulator is unavailable due to preventive maintenance. The calculations are performed using the code ATHLET Mod 1.2, cycle D [2].

A total number of 100 ATHLET calculations was performed. Figure 1 shows at any point of time, at least 95% of the combined influence of all considered uncertainties on the calculated results is below the presented uncertainty limit (one-sided tolerance limit), at a confidence level of at least 95%. For comparison a “conservative” calculation result is shown, applying the best estimate code ATHLET with default values of the models, and conservative values for the initial and boundary conditions reactor power, decay heat, gap width of fuel rods between fuel and clad, fuel pellet thermal conductivity, and temperature of accumulator water. All these conservative values were also included in the distributions of the input parameters for the uncertainty analysis. The maximum clad temperature does not bound the 95%/ 95% one-sided tolerance limits of the uncertainty analysis over the whole transient time.

The “conservative” calculation is representative for the use of best estimate computer codes plus conservative initial and boundary conditions, like for licensing calculations, where uncertainty of code models is not taken into account. The selection of conservative initial and boundary conditions shall bound these model uncertainties. That is obviously not the case for the whole transient. The peak clad temperatures, however, are bounded due to cumulating conservative values of the highly sensitive parameters gap width and pellet thermal conductivity. An uncertainty analysis generally includes model uncertainties. Therefore, the US Code of Federal Regulation [4] requires that “uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated when a best-estimate computer code is used for the analysis.

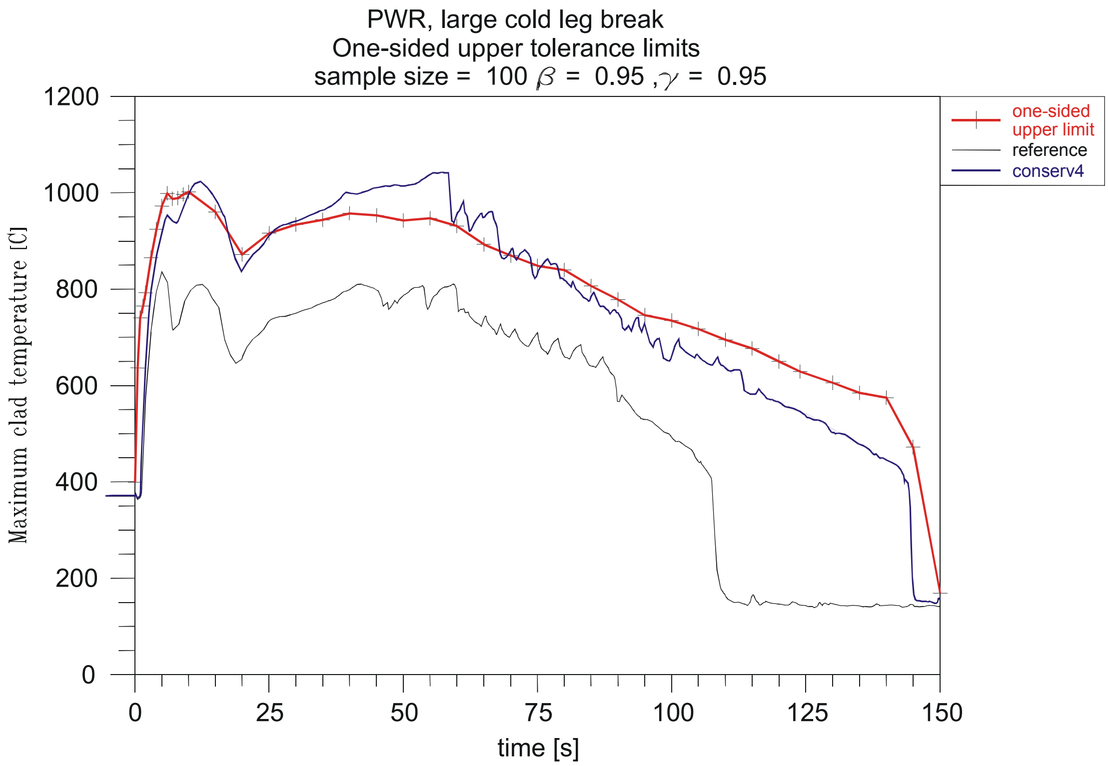


Figure 1: Calculated one-sided uncertainty limit and best estimate reference calculation compared with a “conservative” calculation of rod clad temperature for a reference reactor during a postulated double ended offset shear cold leg break.

The situation in German licensing is different to the USA. Significant differences of results are presented between conservative calculations according to the USA Code of Federal Regulation which requires to apply conservative models in conformance with the required and acceptable features of Appendix K ECCS Evaluation Models [4], and best estimate plus uncertainty evaluation. Consequently, additional margin to licensing criteria is available by changing from conservative evaluation to best estimate calculations plus uncertainty analysis in the USA. In Germany, no such margin is available. However, it is necessary to quantify model uncertainties. This is especially the case for approaching licensing limits, e.g. due to power uprates.

5. CONCLUSIONS

Power uprates of six German PWRs have been approved up to now, three applications are in the licensing process. No case of negative operation experience is reported up to now. Relevant neutron kinetic and thermal-hydraulic parameter values are within licensing limits. The effects on normal operation and transients are very small. In the case of LOCAs higher fuel rod clad temperatures and higher fuel rod failure rates are calculated. The use of design margins influence the reliability of secondary bleed emergency operations.

REFERENCES

[1] Accident Analysis for Nuclear Power Plants. Safety Report Series No. 23, International Atomic Energy Agency, Vienna, 2002.

[2] G. Lerchl and H. Austregesilo. ATHLET Mod 1.2 User's Manual. GRS-P-1 / Vol.1, Rev. 2a, November 2000.

[3] H. G. Sonnenburg, E. Hofer, M. Kloos. Methodenentwicklung zur Analyse des Brennstabverhaltens bei erhöhtem Abbrand. Abschlussbericht GRS-A-3079, November 2002.

[4] 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors", and Appendix K, "ECCS Evaluation Models", to 10 CFR Part 50, Code of Federal Regulations, 1996.

[5] F. Depisch, G. Seeberger, S. Blank: Application of Best-Estimate Methods to LOCA in a PWR. OECD/CSNI Seminar on Best Estimate Methods in Thermal-Hydraulic Safety Analysis, Ankara, Turkey, 29 June – 1 July 1998.

[6] H. Glaeser: Uncertainty evaluation of thermal-hydraulic code results. International Meeting on "Best-Estimate" Methods in Nuclear Installation Safety Analysis (BE-2000), Washington, DC, November 2000.

METHODOLOGY OF ANALYSIS FOR POWER UPRATES IN GERMAN PWR PLANTS

F. Sommer
E.ON Kernkraft GmbH,
Hannover, Germany

H. Göggelmann
Framatome ANP,
Erlangen, Germany

Abstract

Power uprates are carried out with the aim of increasing the plant's economy. This aim can either be achieved by implementing efficiency improving measures or by increasing the thermal reactor power. A gradual increase in thermal reactor power began in the German pressurized water plants of the 1300 MW_{el} series roughly twelve years ago. In this way operational experience with a power uprate of approximately 5% of the original nominal power was gathered. According to German Standards, authorization is needed if "fundamental changes are to be made, either to the nuclear power plant, or to the operation of a nuclear power plant." Among other things it is therefore necessary to prove that the "necessary precautions, issued by 'The State of the Art,' against damage by the erection and operation of a nuclear power plant are met." Proof for the admissibility of a power uprate is granted only if the systems and components affected by the power uprate fulfill the necessary requirements. The proofs of plant behaviour during operational transients and design accidents are carried out by qualified computer codes. In line with the standards, the initial and boundary conditions are specified conservatively. Safety analysis margins have not to be taken into account, since the initial and boundary conditions ensure a conservative analysis. However, the relevant influence parameters are to be examined. The proofs for beyond-design accidents (ATWS) are considered according to the realistic initial and boundary conditions laid down in the standards. Examinations and analyses are carried out to verify that the process engineering and mechanical strength requirements, along with the requirements for the design and monitoring of the reactor core are met. The aim of the proof is achieved when, even after power uprate, the considered systems and components conform to the standards. The proofs are examined by various independent experts for both completeness and compliance with the relevant standards. On account of the positive expert analysis, the relevant authority approved the power uprate. The analyses carried out following the described method show that up until now a power uprate in German pressurized water reactors of approximately 5% of the original nominal power was acceptable. The standards will also continue to be met even after the power uprate. Moreover, many years of positive operational experience with power uprate have been gathered. Neither the routine evaluation of compulsory reported events, nor the evaluation of non-compulsory reported events, combine to identify distinctive features which can be traced back to the increase in reactor power. For this reason the operational experience serves to confirm the reliability of the applied method for the verification of the power uprate.

1. GENERAL INFORMATION

The previously accepted procedure for power uprate is shown in Annex 1. As Annex 1 shows, a power uprate is achieved through efficiency improving measures and / or by an increase in reactor power. Efficiency can be improved by reducing the pressure loss in the main steam system and / or by optimising the turbine blading.

In the 1300 MW_{el} series of plants, reactor power was limited to an increase of approximately 5% of its original power at full load. In doing so the maximum steam flow of the HP-Turbine sometimes limited the possible power uprate. In these instances the operating point for full load was raised.

A higher main steam flow is produced due to an increased reactor and steam generator power. This must be compensated for by an increased feedwater mass flow to ensure a constant steam generator water level during normal operation.

2. OVERVIEW OF AFFECTED SYSTEMS AND COMPONENTS

An overview of the systems and components affected by the power uprate is given in Annex 2.
Among other things, these include:

- Reactor Core (design and monitoring)
- Reactor coolant system with over-pressure protection and pressure relief system
- Water-steam cycle including over-pressure protection on the secondary side.
- Safety systems (emergency core cooling and residual heat removal system, emergency feedwater system)
- Instrumentation and control equipment (control systems, limitation systems, reactor protection system)
- Electrical auxiliary power supply and network supply

As well as examining and proving plant behaviour at normal operation, operational transients, design basis accidents and beyond design-accidents, changes to the activity inventory and the linked radiological effects of a power uprate must also be both examined and proved to ensure that the standards are followed.

3. CRITERIA FOR A POWER UPRATE

According to the standards, authorization is needed if "fundamental changes are to be made, either to the nuclear power plant, or to the operation of a nuclear power plant". Among other things it is therefore necessary to prove that the "necessary precautions, issued by 'The State of the Art' against damage by the erection and operation of a nuclear power plant are met."

The criteria for power uprates are determined by the German Standards.

4. FEASIBILITY OF THE PLANNED POWER UPRATE

The proof for the feasibility of a planned power uprate is derived from an examination of the effects upon the plant:

- In Normal Operation
- At the postulated transients and accidents, and
- At beyond design accidents

according to the current German Standards.

5. PROOF OF THE FULFILLMENT OF THE SAFETY-RELEVANT REQUIREMENTS

5.1 Operational transients and design accidents

In the context of a power uprate the following events are to be considered:

- Increase in Heat Removal by Feedwater/Steam Cycle
  (e.g. Spurious Opening of the Main Steam Bypass)

- Decrease in Heat Removal by Feedwater/Steam Cycle
  (e.g. Loss of Main Heat Sink)

- Decrease in Reactor Coolant Flow Rate
  (e.g. Break of a RCP Shaft)

- Reactivity and Power Distribution Anomalies

- Decrease in Reactor Coolant Inventory
  (e.g. Large and small break LOCA, Steam Generator Tube Rupture)

- Radiologically relevant cases

The initial and boundary conditions are determined according to the standards and, in respect to the aim of the proof, are set conservatively (as, for example assumption of 106% reactor power at full load, 'single failure and repair')

Since, among other things, German nuclear power plants set limits on reactor power, reactor coolant pressure and coolant mass, the defined initial conditions can be guaranteed for the considered events.

For the purpose of this proof, the safety analysis margins need not be taken into account. However, the relevant influence parameters are to be examined.

5.2 Beyond design accidents

The necessary proofs for the following beyond-design accidents are performed on the basis of calculated analysis of:

- Anticipated Transients Without Scram (ATWS)
- Total failure of the steam generator feed, and
- Station black-out

In line with the standards, realistic initial and boundary conditions are to be considered for these events, i.e.:

- apart from the affected system, all systems should be available
- no consideration of 'single failure and repair'

The aim of the proof is achieved when the considered systems and components conform to the standards even after power uprate.

In the event of ‘Total failure of the steam generator feedwater supply’ and ‘Station black-out’ it is shown that even after power uprate the emergency procedures are still valid and the operator grace periods for the execution of the implant emergency procedures are sufficient.

5.3 Computer codes to prove safety-relevant requirements

Internationally recognized computer codes are used to calculate the operational transients and design-base accidents (S-RELAP 5, COCO, BETHY)

In the computer programme S-RELAP 5 the essential parts of the control and limitation systems, as well as the reactor protection system are recreated.

The computer codes show realistic models, which are validated and verified on the basis of:

- Experiments on test facilities (UPTF, LOFT, PKL, CCTF, FLECHT-SEASET)
- Plant experiments during commissioning
- Events during plant operation
- As well as various other single effect experiments and integral tests (UPTF, MARVIKEN, GE LEVEL SWELL).

6. VERIFICATION THAT THE PROCESS ENGINEERING AND MECHANICAL STRENGTH REQUIREMENTS FOR THE SYSTEMS AND COMPONENTS ARE MET.

Proof that the process engineering and mechanical strength requirements extend and comply with the following systems and components:

- Reactor core, fuel rod and fuel assembly design (including LOCA and fuel rod failure rate analysis)
- Pressure and temperature loads in the reactor coolant system
- Overpressure protection of the primary system
- Emergency core cooling and residual heat removal system and emergency feedwater system (feed rates and water supply)
- Overpressure protection of the main steam feedwater cycle

The aim of the proof is achieved when the considered systems and components conform to the standards even after power uprate.

7. SUMMARY

Examinations and analyses are carried out to verify that the process engineering and mechanical strength requirements, along with the requirements for the design and monitoring of the reactor core are met.

The aim of the proof is achieved when, even after power uprate, the considered systems and components conform to the standards.

The proofs are examined by various independent experts for both completeness and compliance with the relevant standards. On account of the positive expert analysis, the relevant authority approved the power uprate.

The analyses carried out following the described method show that up until now a power uprate in German pressurized water reactors of approximately 5% of the original nominal power was acceptable. The standards will also continue to be met even after the power uprate.

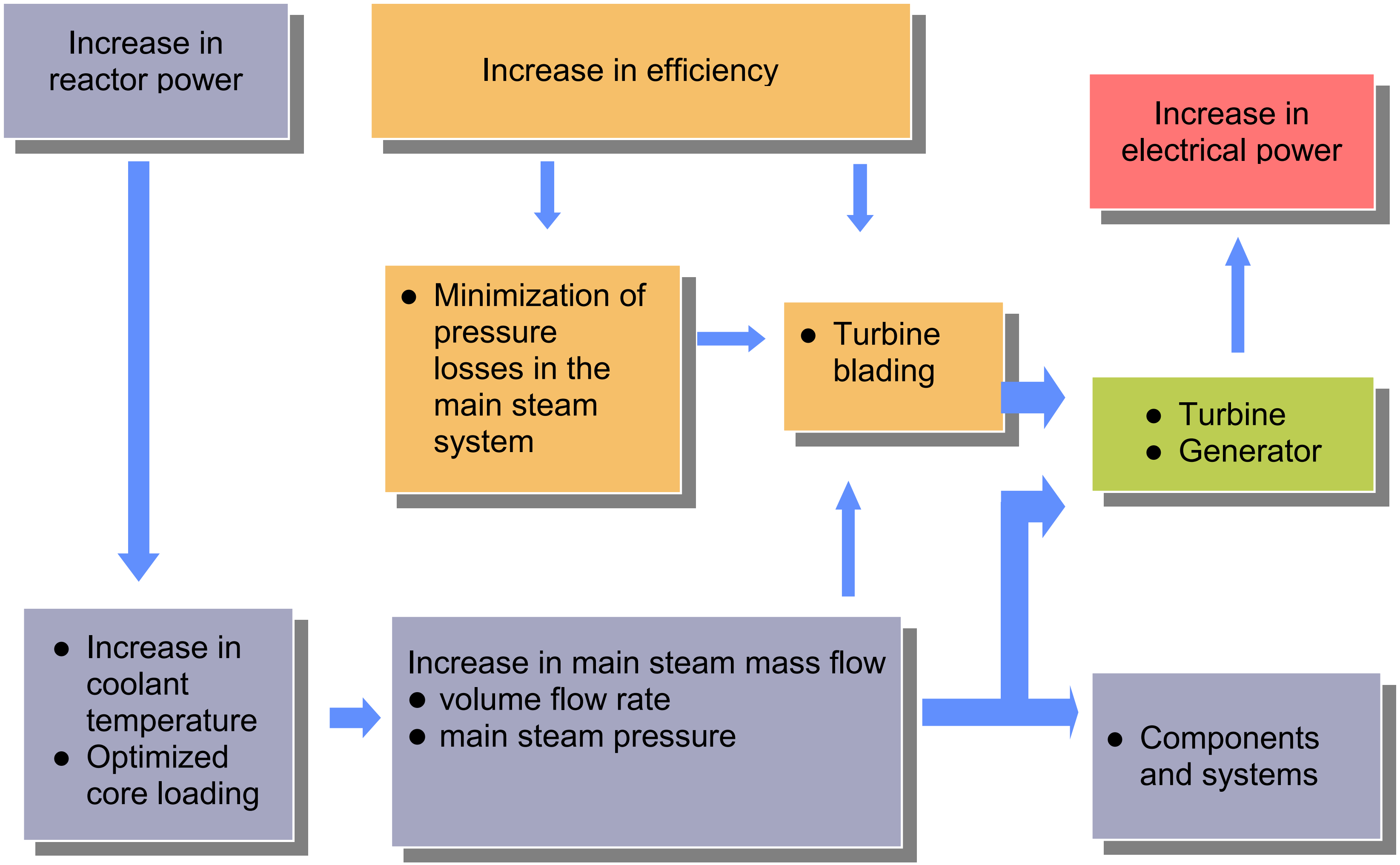
Many years of positive operational experience with power uprate have been gathered (Annex 3). Neither the routine evaluation of compulsory reported events, nor the evaluation of non-compulsory reported events, combine to identify distinctive features which can be traced back to the power uprate.

For this reason the operational experience serves to confirm the reliability of the applied method for the verification of the power uprate.

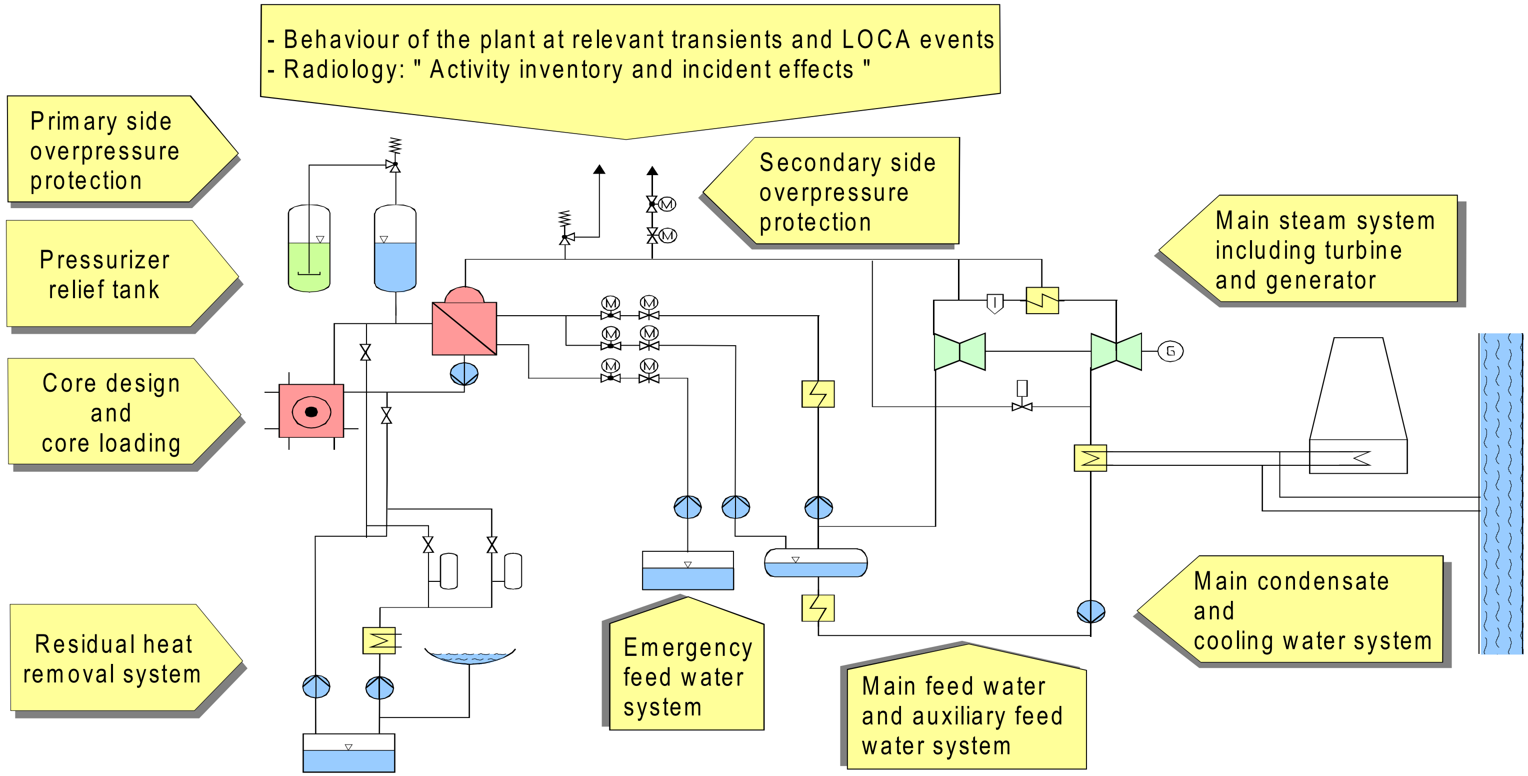
Annex 1: Possibilities for power uprates

Annex 2: Overview of affected systems and components

Annex 3: Examples of Power Uprates in German PWR Plants



Annex 1: Possibilities for power uprates



Annex 2: Overview of affected systems and components

Power Plant	Thermal Reactor Power P_{th0} at Commissioning (MW_{th})	Power uprate P_{th} (MW_{th})	Power increase P_{th}/P_{th0} (%)	Start of operation with increased power
Grohnde	3765	3850 3900	2,3 3,6	1990 1999
Philippsburg 2	3765	3850 3950	2,3 4,9	1992 2000
Neckarwestheim 2	3765	3850	2,3	1991
Emsland	3765	3850	2,3	1991
Isar 2	3765	3850 3950	2,3 4,9	1991 1998
Unterweser	3733	3900	4,5	2000
Grafenrheinfeld	3765	3950	4,9	2003 (current licensing procedure)

Annex 3: Examples of power uprates in German PWR Plants

PRESERVING SAFETY MARGINS WITH PLANNED POWER UPRATE AT PAKS NPP

J. Elter
Paks Nuclear Power Plant Ltd
Hungary

Abstract

A program has been initiated to increase thermal efficiency and uprate the Paks NPP units to a maximum reactor core power level of 1485 MWt. The engineering studies supporting the feasibility of the power uprate have been performed. The safety margins are defined as the measure of confidence that the proposed power uprate would not alter the design basis or safety limit. According to the basic principle the planned power uprate should not assume significant reduction of any safety margin, and would not require relaxation of any acceptance criteria or operational limits. Keeping the core thermal and hydraulic limitations unchanged resulted in that the effect of power uprating will be negligible in normal operation and transients. Room for power uprating is obtained by utilising advanced fuel with different core loading pattern design, by screening out excess conservatism or physical inconsistency in the design calculations and on-line core monitoring, increasing the core flow rate, and stabilisation of the primary pressure. There is an ongoing activity to revise and recalculate all transient and accident analyses utilising the latest state of the art analysis tools and experience. The approach applied is a conservative estimation. Since the maximum linear heat rate and the maximum assembly power are kept unchanged and the hydroaccumulator water content to be increased, it is expected that the hot rod cladding temperature for uprated power will have higher safety margin. The impact of the proposed power uprate on core damage frequency has not yet been quantitatively evaluated but has been addressed.

1. INTRODUCTION

All units of Paks Nuclear Power Plant are currently licensed for operation at reactor core power level of 1375 MWt. The plant has started a program to increase thermal efficiency and uprate the Paks NPP units to a maximum reactor core power level of 1485 MWt, approximately 8 percent increase. At the core uprate power, the general electrical output for each unit will exceed 500 MWe.

The engineering studies supporting the feasibility of the power uprate have been performed by the plant’s staff and general engineering consultant organisation (KFKI AEKI) and later reviewed by the Fortum Nuclear Services Ltd.

The overall risk associated with the increase in rated thermal power has been addressed and it is supposed that there is no impact on the calculated core damage frequency. Keeping the core thermal and hydraulic limitations unchanged resulted in that the effect of power uprating will be negligible in normal operation and transients (AOO). In order to evaluate the capacity of the ECC chain the most limiting postulated accident sequences from the Final Safety Analysis Report were reanalysed or evaluated at the uprated conditions with acceptable results. There is an ongoing activity to revise and recalculate all transient and accident analyses utilising the latest state of the art analysis tools and experience. Later it will be qualitatively justified that there is no adverse effect on success criteria and operator action failure probabilities used in the current probabilistic safety assessment model.

2. DEFINITION OF SAFETY MARGINS

In support of proposed power uprate, Paks units have to be reevaluated for operation at a rated thermal power of 1485 MWt with respect to safety analyses. Identifying the initial impact of the power uprate is the starting point of the reevaluation and to correctly account for those impacts the safety analysis report has to be modified.

The safety analysis can be considered as constituting the following obviously interdependent elements [3]:

Requirements – are all the safety objectives or goals that must be addressed to assure safety.

Evidence – is a solution or information from related studies, analyses and tests.

Argument – is showing how the evidence indicates compliance with requirements, the requirements are usually replaced by more restrictive acceptance criteria.

Context – is identifying the basis of the argument presented, these are different assumptions, justification and models.

Two strategies could be utilised to help safety arguments in justification and licensing procedure:

- Safety margins;
- Diverse evidence, argument.

Margins are usually applied to account for any unquantified uncertainty. This is why **safety margin** is created wherever an acceptance criteria not only satisfies but exceeds the requirement (do not reach safety limit) thus providing a margin of safety. The acceptance criteria are usually established by regulatory authorities and serves as the basis for licensing process. This margin accounts for the uncertainties in the limit definition itself, but this margin could not always be quantified.

The other source of safety margin is the margin to acceptance criterion. It is created when the evidence or the calculated (with conservative assumptions or with assessed uncertainty) solution is not exceeds the corresponding acceptance criterion. By doing this, confidence is further increased in the satisfaction of the requirements and there is a “margin for error” or any other uncertainties. Since the (modelling) uncertainties are not quantified and unresolved issues may also exist it is only a belief that the safety margin and the conservatism (which is an implicit margin) used in the operational parameters and system availability will cover the unquantified uncertainties.

Along with the NRC rules [5, 6] safety margins are applied to control values of parameters to account for various uncertainties and to avoid exceeding regulatory or licensing limits. The specific values that define margin are established in the plant's licensing basis. During the safety re-evaluation the safety margins that may be affected by the power uprate and review the conservatism in the evaluation and analysis methods that are used to demonstrate compliance with regulatory and licensing requirements. The safety margin for the original rated power should be compared to the margin after the power increase to determine if the uprate will reduce the margin.

Diverse argument exist wherever a number of individually sufficient solutions, information are put forward to support s particular requirement. By doing this, confidence is increased in the satisfaction of the requirement or safety limit. For increased robustness the individual arguments should ideally be based upon independent forms of evidence like: diverse forms of safety analysis and testing information, appealing to independent safety mechanisms in the design, estimated versus operational data.

In general, a safety margin can be described with two dependent parameters:

- difference between a controlling numerical value for a parameter established in the safety analyses report that represents actual load effect and the corresponding safety limit,

- the probability that a load effect would not exceed the safety limits due to physical or calculation uncertainties and errors.

Combining of these factors the safety margins are defined as the measure of confidence that the proposed power uprate would not alter the design basis or safety limit. A safety margin is preserved if that confidence could be preserved.

3. BASIC PRINCIPLES APPLIED FOR POWER UPRATE

The basic principle is that the planned power uprate should not assume significant reduction of any safety margin, and would not require relaxation of any acceptance criteria or operational limits. It has to be demonstrated that adequate safety margins will still exist after the power uprate. The plant safety shall not be reduced in any other way.

As far as achievable, the benefit shall be based on the advances in calculation and evaluation methods, and experience from more than 70 reactor years of operation units of Paks Nuclear Power Plant. To the extent possible the successful experience gained during the Loviisa modernisation and upgrading would be taken into account.

The benefit and the restoration of fuel cycle economy can be also made in future of modernised fuel designs, certain plant modification and changing of operational practice.

It is also important that the power upgrading shall not reduce the plant lifetime. The time and energy availability of the plant shall also not be reduced. The cost-benefit ratio shall be very good: the return of investment shall be only a few years.

A number of requirements are added to basic principles for the implementation planning:

- The fuel cycle will be maintained in 12 months, the negative effect on fuel cycle economy should be minimised.

- The first priority is to carry out power uprate without major plant modifications or component replacements.

- The implementation shall not extend the refuelling outage.

4. CORE DESIGN

During feasibility study of power uprate works reload design calculations has been carried out: transient and equilibrium cycles from present to uprated nominal power were analysed. The aim of this analysis was to check if existing reload design limits could be satisfied in uprated power cases. The reload design and operation practice the following parameters are limited:

- Maximal linear heat rate,
- Maximal subchanel outlet temperature,
- Assembly-, pin-, pin local (pellet) burnup,
- Efficiency of all control rods, efficiency of regulating group,
- Efficiency of one ejected rod,
- Critical boric acid concentration,
- Shutdown margin,
- Subcriticality during refuelling condition,
- Boric acid efficiency, moderator temperature efficiency, Doppler efficiency.

Examinations showed that the majority of existing limits can be satisfied in uprated power case as well - with some exceptions.

Reactivity reserve (BOC critical boron concentration)

Uprated power requires higher reactivity reserve at BOC condition in order to provide the same cycle length in calendar day unit. Critical boron concentration at BOC condition will grow. This problem can be handle if the boron concentration of safety injection tanks is increasing from present level of 12 g/kg to 13 or 13.5 g/kg. Higher concentration will be able to insert sufficient negative reactivity into the core in boron dilution case.

Local limits in general

The power uprate would cause the increase of local power and temperature values due to higher energy production. Otherwise, the basic idea was not to change local operating limits, but find the way to satisfy these in uprated power condition as well.

Subchanel outlet temperature

The hot subchannel outlet temperature has turned out to be the most limiting parameter, the goal of this limit is to keep coolant under saturation temperature during normal operation. The following measures are available to relax this limitation.

According to the calculations the proposed *advanced fuel design with expanded fuel rod pitch* is more favourable from the point of view of hot subchannel enthalpy rise. The effect is in the order of 5 %.

The gross flowrate of unit 3 and unit 4 is a few percents higher than that of for unit 1 and unit 2. The flowrate will be increased by *modifying the pump impellers.*

Coolant mixing and flow redistribution inside the fuel assembly has the tendency to smoothen the hot subchannel enthalpy rise peaking factor. It is still under discussion whether to *take into account the coolant mixing* between subchannels inside the fuel assemblies when evaluating – with the online core monitoring system – the hot subchannel enthalpy rise and DNB-margin.

Extra margins for increased reactor power would be gained by *flattening the core power distribution*. This would require a change from the full low-leakage loading pattern back towards the out-in-in loading pattern. However, the low leakage loading pattern with 4th cycle fuel at the core peripheral locations has been kept as a boundary condition to minimize the pressure vessel fluence and preserve the safety margin on the embrittlement characteristics.

Increase of nominal primary pressure has a potential of providing extra subchannel enthalpy rise margin. This was considered by the feasibility study. No basic obstacles were found but it is not initiated by now. Instead, a *stricter regime for pressure control* is proposed in order to increase the lower pressure limit.

Slightly *decreased pressure in the secondary circuit* would help to keep the core inlet temperature at lower level.

The methods listed above declared to be enough to keep the subchanel outlet temperature under saturation limit in uprated power case as well.

Maximum linear heat rate

The local linear heat rate will remain on the original upper limit of 325 W/cm originating from the assumptions made in large break LOCA analysis. The limit is decreasing with burnup mainly to prevent excess fission gas release from the fuel pellets. A conservatively high operative margin 15 W/cm will also be kept.

The margin to the allowed linear heat rate is decreasing time towards the end of cycle when control rods are being withdrawn out from the core. This is due to the power peak induced on the neighbouring fuel rods by the extra water in the control rod and fuel follower-junction region. A *new fuel follower design with Hafnium plates* is planned to damp the thermal flux peak in the junction region and providing extra margin.

5. SYSTEM AND COMPONENT MODIFICATIONS

The power uprate will cause mainly the increase of the secondary flowrates due to the higher steaming capacity of the steam generators.

The increase of the residual heat is the reason to evaluate the capacity of the ECC chain. This was done in connection of the analysing the limiting accident analyses (LOCA). Since the maximum linear heat rate and the maximum assembly power are kept unchanged and the hydroaccumulator water content to be increased, it is expected that the hot rod cladding temperature for uprated power will have higher safety margin. According to the earlier results the maximum rod cladding temperature occurred in the reflood phase. The impact is effectively dampened by the increased water accumulator inventory.

The slight increase of the primary circuit temperatures is the other main parameter changing in power increase, hot leg +3-4 °C and cold leg +1 °C. This inherently would influence the nominal pressuriser level that probably will be cut with the original level setting values. This would not cause problems concerning the low level in reactor trip transient. The temperature increase has no other remarkable influence to the primary systems.

In spite the slight temperature and power increase no remarkable changes are necessary in the reactor protection system. All the ECCS signals remain unchanged, but some of them would be actuated earlier during an accident. The reactor trip signals that are proportional with the reactor power have to be scaled up.

There are several possibilities to increase the thermal efficiency of the cycle, but these measures will be handled separately during the power increase planning and licensing phase.

6. SAFETY ANALYSIS

Deterministic accident analysis

Accident analysis for all postulated initiating events is performed to identify deterministic safety margins in case of increased thermal power.

The approach applied is conservative estimate, the initial and boundary conditions are always chosen in a conservative manner, no uncertainties calculations are foreseen.

Different aspects of an accident are usually calculated by different (if possible realistic) codes with a different set of pessimistic assumptions. The following aspects are taken into account in the safety analyses:

FC – fuel rod or cladding integrity (coolability);
PP – pressure in the primary system and in the reactor vessel;
SP – pressure in the secondary system and in the steam generators;
PTS – pressurised thermal shock;
RI – loads on reactor internals;
CC – containment internal loads;
RD – release of radioactivity, doses.

The essential assumptions regarding the operational and modelling parameters, used in the conservative estimate are the following.

Failures of safety systems should be assumed according to the single failure criterion. In order to avoid investigating all cases of possible failures, the minimum configuration of safety systems are often used. Most of the analyses are extended to the case of simultaneous loss of electricity of the plant.

It is conservatively assumed throughout the analyses that only ERP-1 signals are actuated. Functioning of control systems (e.g. power controller) is also conservatively neglected.

The uncertainties of the measured parameters are specified in the initial and boundary conditions in the direction unfavourable from the point of view of the given scenario, and the above listed analysis aspects. The axial and radial power and flux shapes are specified in the unfavourable manner that can be given on the basis of the operational experience and control calculations.

When considering reactivity induced accidents, the reactivity coefficients are set to the values corresponding to the most unfavourable operational limits, and states. The movement of the control assembly (group) with most unfavourable reactivity effectiveness (stuck rod) is assumed. The assembly with highest reactivity worth is considered in case of control assembly ejection. The modelling parameters like effective delayed neutron fraction, gap conductance, reserve reactivity, etc. are treated in a conservative manner. In analysing reactivity anomalies the operation of the reactor period protection is not assumed, though it fulfils the single failure criterion

In the LOCA accident, even if no fuel cladding failure is found, a source term corresponding to 1% fuel failure is assumed in release calculations. Iodine spiking from inhermetic fuel rods during a transient is calculated.

The containment behaviour is calculated by assuming the maximum permitted leakage rate (it is conservative from the point of view of radioactive releases and has a negligible effect on the containment pressure load).

In dose calculations and in assessing environmental effects both cases when the reactor hall ventilation is functioning (unfavourable from the point of view of the environment) and when it does not function (unfavourable from the point of view of the plant personnel working in the controlled area) are considered.

Environmental calculations are performed for persons staying at the border of the 3 km radius safety area, namely for a one-year-old child, assuming that the person consumes the food corresponding to the local contamination.

Probabilistic safety analysis

The impact of the proposed power uprate on core damage frequency has not yet been quantitatively evaluated. The main issues that are influenced [7]:

- decay heat removal success criteria;
- the time dependent operator actions.

The heat removal success criteria would not be effected, since the core thermal and hydraulic limitations remain unchanged and all the important safety margins would be preserved.

It is believed that the time frame available for successful operator response will not significantly be reduced.

The impact on large fission product release frequencies and on progression of severe accident has not been addressed yet.

Along with the assessment published in [8] it can be stated that the most significant impact results from the increased radioactive inventory and the possible time acceleration of events due to the increased decay heat level and corresponding decreased time to core uncovery. Given the large degree of uncertainty associated with some of the severe accident issues, quantitative assessment of potential impact of the relatively small increase in power level (as compared with the large intrinsic uncertainties) is very difficult, and could even be misleading. The accident sequence variabilities are far greater than the variations caused by power uprate.

REFERENCES

[1] Feasibility of the rated power increase at Paks Nuclear Power Plant, KFKI AEKI, 2001.

[2] Paks upgrading evaluation report by Fortum, Fortum Nuclear Services Ltd, Nucl-656, 2001.

[3] T. Kelly, J. McDermid: A systematic approach to safety case maintenance, University of York, YO10 5DD.

[4] Safety Margins of Operating Reactors, IAEA-TECDOC-1332, International Atomic Energy Agency, Vienna

[5] Nuclear Regulatory Commission, 10 CFR Part 50, Domestic licensing of production and utilization facilities, 1998.

[6] Nuclear Regulatory Commission, RIS-01-22, Attributes of a Proposed No Significant Hazards Consideration Determination, 2001.

[7] Safety evaluation by the office of Nuclear Reactor Regulation related to amendment No. 137, NRC, 1998.

[8] U. Schmocker, M. Khatib-Rahbar, E.G. Cazzoli, A. Kuritzky: An assessment of the risk-impact of reactor power upgrade for BWR-6 Mark-III plant, Proceedings of PSAM-3 meeting, 1997.

CONSIDERATIONS OF SAFETY MARGINS FOR DETERMINISTIC AND PROBABILISTIC ASSESSMENTS: REGULATORY PRACTICES IN INDIA

P. Hajra
Safety Analysis & Documentation Division,
Atomic Energy Regulatory Board,
Mumabi, India

Abstract

Safety analyses require evaluations of safety margins for parameters with regard to the acceptance criteria for the postulated initiating events, design modifications and changes of operational limits and conditions. These evaluations are carried out as per safety requirements with two approaches–deterministic and probabilistic. In India, for deterministic assessments, the parameters and the acceptance criteria or design basis limiting values are primarily based on international practices, national regulations, expert opinion and experience, and appear in the safety directives/documents; In some cases, these are proposed by the utility and accepted after review by the regulatory body and some are still under consideration. For 'consent' on new plant construction, commissioning or operation, parameter values are evaluated with conservative analysis approach and assessments made with regard to meeting the acceptance criteria. For existing operating Nuclear Power Plants (NPPs) although best estimate analysis with uncertainty evaluation on various sources like input data, modelling etc. is preferred, still conservative approach is accepted with sensitivity studies on identified parameters and assessments made for safety margin including minimum safety margin requirements as applicable. The paper also addresses requirements of probabilistic safety assessment for both new and operating NPPs and comprehensively discusses various PSA parameters and their target values for evaluation of margin in regulatory decision making with risk-informed approach. These parameters and their target values are established/being established, keeping in view of the state-of-the-art of PSA expertise and tools available in the country, INSAG recommendations, practices of other countries for all three levels of PSA, so as to ensure balanced design and safety to plant personnel, environment and the public.

1. INTRODUCTION

Safety Margin (SM) is the difference between the reference limiting value of any assigned parameter as accepted by the regulating body and its calculated value in safer direction. Usually limiting values will be for a set of assigned parameters generally prescribed by the regulator in safety documents. However, these can be established by the utility and should be accepted by the regulator. The evaluation of safety margin is required to demonstrate that the reactor as designed and operated can meet the challenges from Anticipated Operational Occurrences/transients (AOOs) and design basis accidents (DBAs), for better utilization of fuels, power uprates, cost reduction modifications and justification for plant life extension. These limiting values are arrived based on international practices, experience, research studies and testing for fixing design and operational limits. The regulatory acceptance value for reference for evaluation of safety margin may be further restricted as per the national practice; For example, safety limits in technical specifications for operation of a nuclear power plant (NPP) specified with respect to the integrity for reactor fuel and primary cooling system boundary could be set slightly lower than the international practices or established by the test/analysis to account for uncertainties in variables, any phenomena/process not well defined. Limiting safety system settings or reactor scram set points are always set still at lower values to account for time constant, error in instrumentation setting etc. However, the safety margin shall be always positive however small it may be for regulatory ‘consent’ to any utility proposal for construction, operation, modifications etc.

In India, there are certain parameters such as minimum subcriticality margins (following reactor scram and shutdown state), minimum time available for operator action for protection/mitigation etc. are specified for compliance. However, these values are not stringent so long as regulatory body is convinced that there is conservatism in input parameters and analysis and some margin does exist.

Safety analysis as practiced now all over the world requires evaluations with both deterministic and probabilistic assessments as a defence-in-depth principle of safety assessments. The deterministic analysis also commonly known as accident analysis (AA) has two basic approaches- conservative and best estimate, depending on analysis objectives, issues involved and stages of plant authorization. For new NPP for operation permit, utility is expected to follow conservative analysis. For reauthorization of an existing plant, although best estimate approach with uncertainty analysis (UA) is preferred, analysis with conservative approach is accepted. However, for the probabilistic safety assessment (PSA), UA is required to demonstrate that adequate probabilistic margins on the assigned parameters do exist. The Fig.1 and Fig. 2 show schematically the safety margins (SMs) for deterministic and probabilistic assessments.

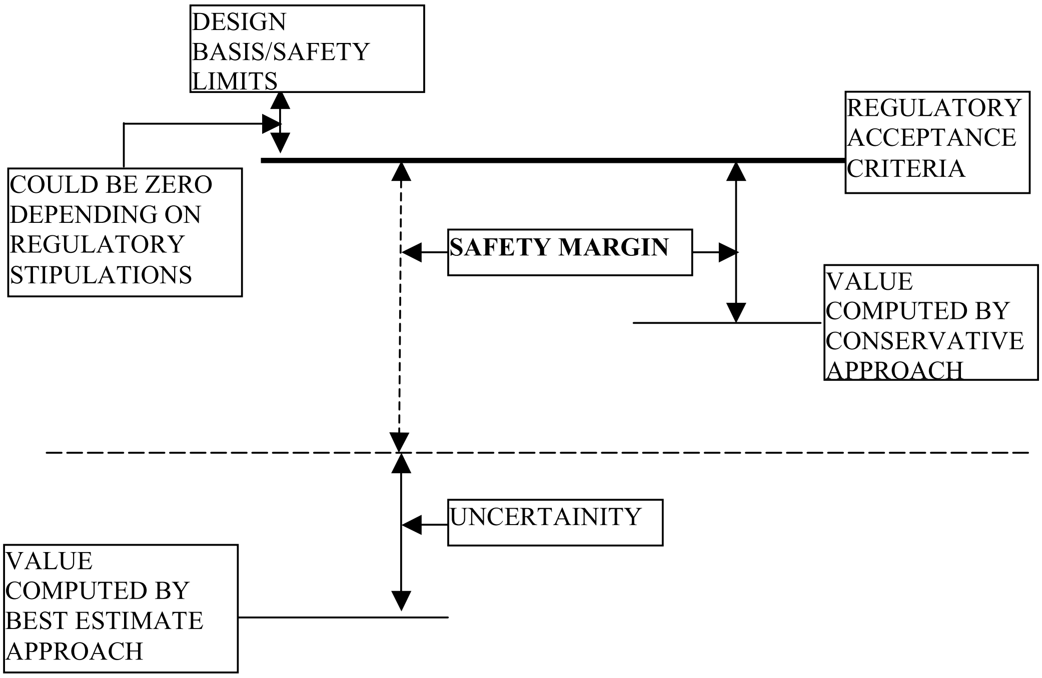


Fig. 1 Safety Margins for Deterministic (Accident) Analysis

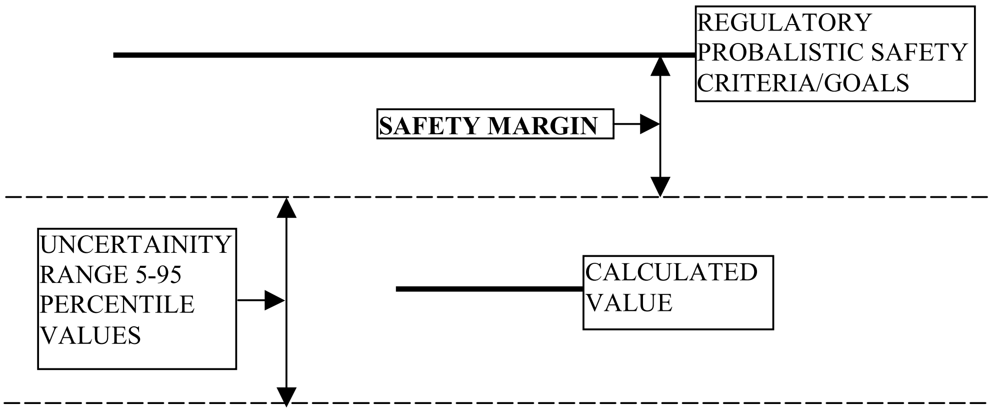


Fig.2 Probabilistic Safety Margins

The parameters considered for SMs for accident analysis vary from reactor types and national practices which besides radiation doses to public, may include departure from nucleate boiling ratio (DNBR), clad temperature, time availability for operator action etc. The parameters for PSA besides core damage frequency may include individual safety system unavailabilities, probability of large early release of radioactivity from containment, individual risk of fatality at exclusion zone etc. The limiting values for reference for evaluation of SMs henceforth called acceptance criteria/value may be set in hierarchical level; While the primary acceptance criteria are set at the highest level and accepted by the regulatory body, the secondary or even more restrictive analysis criteria which may be set by the utility/analyst need not have regulatory clearance.

The following sections elaborate the practices followed in India with regard to demonstration of SMs in AA and PSA for regulatory 'consent'. In India the word 'consent' is a general term used for regulatory clearance, approval, authorization and license depending on the facilities, activities etc.

2. DETERMINISTIC SAFETY MARGIN/ACCEPTANCE CRITERIA

2.1 Parameters

The parameters and the values set for acceptance criteria to evaluate SM for Accident Analysis (AA) are based on defence-in-depth principle having relation to integrity of fuel and cladding, Reactor Coolant System (RCS), and containment as preventive measures for ensuring nuclear (radiological) safety. These parameters are diverse and generally needed to judge the vulnerability of individual barriers and for various aspects of the accident. More stringent criteria are often applied to events with higher frequency of occurrence. Depending also on the type of reactor, these may include RCS pressure, secondary system pressure, shutdown margin, prompt criticality margin, linear heat generation rate of fuel, fuel center line temperature, fuel clad temperature, DNBR, fuel enthalpy, fuel clad strain, extent of clad oxidation, percentage of fuel failure, hydrogen generation and its concentration in containment building, containment pressure, time for manual action required to prevent/mitigate the event, radiation dose to plant personnel and the public.

The requirements of meeting these acceptance criteria are stipulated in the code and relevant guides usually give the approach/establish the values of the parameter to be taken for applicable postulated initiating events. For example, design guide on LOCA will give guidance for limiting values and methods/approach as necessary to evaluate values of parameter such as percentage of oxidation in fuel clad, clad strain/burst stress etc.

2.2 Categorization of PIEs

The values of parameters for acceptance criteria have bearing on the category of postulated initiating events (PIEs) namely AOO, Accident Condition (AC), certain identified multiple failures (severe accident) and very rare events. Many severe accidents and rare events may come under beyond design basis accidents (BDBAs) as per the practice in the country. However, some of these need to be analyzed as required by the regulatory body to ascertain ultimate capability of the plant, to explore what safety features can be engineered to prevent occurrence, what emergency procedure should be planned to meet the challenge from these PIEs and also for any cliff edge situations. Although, nomenclature may differ from country to country, the basis of such categorization is on the probability of occurrence, as mentioned below.

Category	Frequency
AOO	$1\text{-}10^{-2}$ /year
AC	$10^{-2}\text{-}10^{-4}$ /year
Multiple Failures (Severe Accidents)	$10^{-4}\text{-}10^{-6}$ /year
Very Rare Event	$< 10^{-6}$ /year

2.3 Analysis Approach

As mentioned above, there are two basic approaches in deterministic evaluations of SM: conservative and Best Estimate (BE) with UA. In India, for 'consent' of initial authorization (Licensing) usually conservative approach needs to be followed, where usually certain deterministic conservative assumptions have been firmly established to account for uncertainties in input data variation and modeling or to simplify the analysis. To mention some of these are-maximum power to be taken as rated full power (FP) + 4% FP, unavailability of shut off rod of highest worth, crediting only one Shutdown System (SDS), unavailability of one bank in secondary SDS, single failure criteria in the mitigating systems, no operator action for 30 minutes in general and for lesser time (15/20 minutes) for specific cases where operators understanding on event identification/progression is without doubt; class IV power failure occurrences anytime before, during or after the event occurrence, ignorance of first scram parameter etc.

For severe accidents/very rare events, best estimate analysis is followed taking credit of availability of normal operating systems and justifiable operator actions etc.

2.4 Acceptance Criteria

The table 1 below gives the parameters and their acceptance criteria (which may not be all inclusive)

Table 1: Parameters and their Acceptance Criteria

Reactor Type	S. No.	Description of Parameter	Acceptance Criteria
BWR	1	RCS pressure (AOO)	Below design limit
	2	Clad integrity (AC)	No melting
	3	Clad oxidation (AC)	17% original clad thickness
PHWR	**AOOs**		
	1	PHT system boundary integrity	Below design limit
	2	UO_2 centre-line temperature	No melting
	3	Critical heat flux to normal heat flux ratio	1.1
	4	Cladding strain limit	1.5%
	Accident Conditions		
	1	The maximum oxygen concentration in the least affected half thickness of clad	< 0.7% by weight.
	2	The fuel pellet radial average enthalpy of the hottest fuel element	< 200 cal/gm

	3	a) Clad stress	Below Calculated burst stress
		b) Cumulative damage fraction of fuel	< 1
	4	Coolant channel geometry	Should remain coolable
	5	Containment pressure	Peak pressure should be below design value
	6	H_2 concentration in containment	Below deflagration and detonation limits
PWR	**AOOs, ACs, Severe Accidents**		
	1	Pressure in the reactor coolant and main steam systems should be maintained below 110% of their respective design values	As stated
	2	Pressure in the reactor coolant and the main steam systems should be maintained below acceptable design limits considering brittle as well as ductile failures and fuel conditions.	Below acceptable design limit
	3	The primary reactor coolant system shall be maintained in a safe state so that short term and long term cooling is achieved.	Coolability assured
	4	The first design limit of fuel element damage should not be exceeded: 1% of fuel elements with defects of gas leakage and 0.1% of fuel elements for direct contact of coolant and fuel.	As mentioned
	5	For DNBR there should be a 95% probability at the 95% confidence level, that no fuel rod in the core experiences a DNB conditions (DNBR = 1.0).	As mentioned
	6	The maximum clad temperature shall not exceed 355°C anywhere in the core, except for a period of few minutes, when the clad temperature may rise up to maximum of 800°C.	As mentioned
	7	The maximum fuel temperature anywhere in the core	< melting
	8	The clad strain considering normal operating cycles (including load following and other manoeuvering and AOOs).	< prescribed limits
	9	Fuel damage (clad perforation) shall be assumed if DNBR falls below acceptable limit (criterion 5). If fuel damage is calculated to occur, core geometry should be intact to assure core-cooling capability.	As mentioned
	10	In case the limits of clad strain (criterion 8) and rate of increase of fuel power are exceeded anywhere in the core, fuel damage shall be assumed.	As mentioned
	11	a) The radially averaged fuel enthalpy b) On reactivity excursions a radially averaged fuel enthalpy	<586 J/g <830 J/g

<table>
<tr><td></td><td>12</td><td>a) The calculated maximum fuel element cladding temperature
b) The calculated total oxidation of the cladding shall not exceed <18% of original cladding thickness.
c) The calculated total amount of hydrogen from the chemical reaction of the cladding with water and steam shall not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
d) Calculated changes in core geometry are such that the core remains amenable to cooling
e) After any calculated successful initial operation of the Emergency Core Cooling System, the core cooling can be maintained for the extended period as required.
f) Permissible values of linear heat rate (LHR) in fuel over the core height divided conventionally into ten sections are
<table>
<tr><td>Section no. Over the height</td><td>1 to 5</td><td>6</td><td>7</td><td>8</td><td>9</td><td>10</td></tr>
<tr><td>Permissible LHR, W/cm</td><td>448</td><td>428</td><td>392</td><td>360</td><td>338</td><td>316</td></tr>
</table></td><td><1200°C.
As stated

As stated

As stated

As stated</td></tr>
<tr><td></td><td>13</td><td>Time period for the operator for identification of unscheduled boron dilution in the primary circuit till minimum subcriticality is reached under reactor state refueling shutdown</td><td>≥30 minutes</td></tr>
<tr><td></td><td>14</td><td>During power operation, hot standby, cold shutdown and reactor startup, time for the operator for identification of unscheduled boron dilution in the primary circuit till criticality is reached or till reactivity margin has not been lost as required for reactor shutdown.</td><td>≥15 minutes</td></tr>
<tr><td></td><td>15</td><td>The plant should have instrumentations to detect potential fuel loading errors after refuelling operations.</td><td>As stated</td></tr>
<tr><td></td><td>16</td><td>The sensitivity of detection procedures should be such that the loading errors not detected shall not cause DNBR limit violation.</td><td>As stated</td></tr>
<tr><td></td><td>17</td><td>Calculation of radiological consequences for potential cask drop distance of less than 9m may not be required provided appropriate impact limiting devices are employed during the cask movement. Damage to other systems due to cask drop shall be demonstrated to be acceptable or it shall be proved that such drop is not credible.</td><td>As stated</td></tr>
<tr><td></td><td>18</td><td>Tanks and associated components containing radioactive liquids outside the containment are acceptable if their failure does not result in radionuclides concentration at nearest potable water supplies source, in excess of the dose limits</td><td>As stated</td></tr>
</table>

	19	Doses for normal operation shall not exceed permissible limits.	20 mrem whole body; 60 mrem thyroid gland; 120 mrem any separate organ
	20	a) The calculated doses at the exclusion area boundary shall be less than five percent (5%) of the dose limits for accident conditions b) The calculated doses at the exclusion area boundary shall be less than twenty five percent (25%) of the dose limit for accident conditions Dose limits for accident conditions.	As stated As stated 10 rem for whole body. 50 rem for child thyroid
	21	The calculated doses at the exclusion area boundary should be less than the dose limits for accident conditions	As stated
		Very Rare Events (Hypothetical Accidents)	
	1.	Fuel enthalpy shall be limited to a value resulting in severe fuel - coolant interaction	As stated
	2	Prevention of steam explosion that could result in containment failure.	As stated
	3	Prevention of recriticality of the partial or complete core during core degradation or melting.	As stated
	4	Prevention of hydrogen explosion that could result in containment failure.	As stated
	5	Prevention of breaks in the containment from internally - generated missiles.	As stated
	6	Prevention of breaks in the containment from slow/sudden rise of pressure.	As stated
	7	Sufficient cooling of core debris for their retention under the containment.	As stated
	8	Prevention of containment failure from radiation heating caused by core debris.	As stated
	9	Maximum calculated dose for permanent residents during 1 year after accidents at distance of 25 Km from NPP shall be limited to:	10 rem for external exposure. 50 rem for thyroid glands of critical group

FBR	**AOOs**		
	1	Coolant: No coolant boiling	As stated
	2	Clad temperature below the prescribed limit	800 ^{0}C
	3	Fuel temperature	<Melting point
	Accident Conditions		
	1	Coolant: No coolant boiling	As stated
	2	Clad temperature below the prescribed limit	900 ^{0}C
	3	Fuel temperature	<Melting point
	Multiple Failures		
	1	Coolant: No coolant boiling	As stated
	2	Clad temperature below the prescribed limit	1200 ^{0}C
	3	Fuel temperature	<50% melting in max rated pellet
	Very Rare Events*		
	1	Core melt configuration should not cause re-criticality and ensure long term cooling	As stated
	2	In core disrupted accident, there should be no recriticality and energy release should be below containment design considerations	As stated
All Plants	1	Radiological dose limit	For AOO: 100 mR For DBAs: 10 Rem whole body 50 Rem child thyroid

* Events between 10^{-06} to 10^{-07} /ry occurrence frequencies are taken care by intrinsic safety and design features. Events of very low probability $<10^{-07}$/ry called residual risk are BDBAs which lead local or whole core melting.

3. PROBABILISTIC SAFETY MARGINS AND CRITERIA/GOALS

3.1 Risk-Based/Informed Approach

Although in India presently deterministic evaluation of SM has the front seat in making regulatory decision, PSA insights are being increasingly used in line with international practice, to support and supplement deterministic evaluations as risk-informed approach as against risk- based approach practiced by many countries. In risk-based approach, reference values of parameters of probabilistic analysis, as established and accepted by the regulatory body assumes the yardstick for decision making irrespective of deterministic evaluation of safety margin which is used here as complementary insights. The reference values are called probabilistic safety criteria (PSC); in risk-informed approach, these reference values are considered as goals i.e. probabilistic safety goals (PSGs) which utility should try to achieve in plant design and operation. The PSC or PSG for any parameter is established based on international practices or engineering judgment and generally be the same. However, they may differ due to regulating practices followed in the country. In India, PSGs for risk-informed approach for some parameters are established from INSAG recommendations, some are established by the utility and accepted by the regulatory body and many are being evolved with views from utility and experts. The guidelines being followed in establishing PSGs are to ensure that the plant design is balanced, and risk in operation of a plant is acceptably low, the application of PSA methodology to optimize technical specifications of plant operations for limiting conditions of operation and surveillance testing of safety significant items and to prioritize operational tasks from risk significance. These require setting PSG comprehensively at all three levels of PSA. In PSA methodology, uncertainty arises mainly due to incompleteness of identification/assessment of all the possible scenarios that can lead to undesirable consequences, modeling (conceptual/mathematical) inadequacy, numerical approximations, coding error, computations limitation and input parameter variation. Estimates of these uncertainties are necessary to assure confidence level in analysis results.

3.2 PSG

The following gives some PSGs established/under considerations for risk-informed approach:

(a) General:
  - i) No particular feature of the design makes a disproportionately large contribution to risk.
  - ii) No group of initiating events makes a disproportionately large contribution.
  - iii) The contributors which enable achievement of very low level of risk should not have significant uncertainties.

(b) Identification of critical components based on their importance in terms of certain risk increase value such as 0.1 % increase in CDF or 1 % increase in system unavailability.

(c) Safety and safety related systems unavailability targets:
  - (i) Shutdown System (SDS):
    - SDS1: < 1E-3/demand
    - SDS2: < 1E-3/demand
    - Overall: ≤1E-6/demand
    - Single System: ≤2E-5/demand
  - (ii) Engineered Safety Features:
    - ECCS: < 1E-3 yr/yr
    - Containment System: < 1E-3 yr/yr
    - Containment Isolation: <2 E-4 yr/yr
    - Decay Heat Removal System: <1E-6 yr/yr
  - (iii) Class III Emergency Power Supply: ≤ 1 E-3/Demand
  - (iv) Fire Fighting Water System: ≤1E-3/Demand
  - (v) Loss of Moderator Heat Sink (PHWR): ≤ 5 E-2 yr/yr
  - (vi) Shared Safety System: ≤1E-03 yr/yr
  - (vii) Reactor Regulating System: ≤0.3 Failure/yr

(d) Adequacy of design and operational framework (including modifications/back fits/upgrades) can be based on: (i) CDF: for operating NPP: ≤ 1E-4 per reactor year and for New NPP: ≤ 1E-5 per reactor year and (ii) limiting contribution to CDF from any dominant accident sequence < 25 %.

(e) As a part of sensitivity studies, increase in risk should not exceed 10 % at system unavailability level, 1 % at CDF level and 0.1 % at release consequence level respectively.

(f) Risk based Allowed Outage Time (AOT) should not cause exceedance in 0.1 % increase in CDF (Δ CDF/CDF) or, 1 % increase in system unavailability.

(g) Risk based Surveillance Test Interval (STI) should be arrived such that any change in STI would not result in increase in system unavailability by more than 1 % or increase in CDF (Δ CDF/CDF) by more than 0.1 %.

(h) Overall frequency of large radioactivity release beyond acceptable levels from BDBAs is less than a target value ≤ 1E-6 per reactor year (r-y) (<1E-05/r-y for existing plant).
(i) Individual risk of early fatality from radiation exposure in severe accident is less than the target value (5E-7 per reactor year).
(j) The estimated frequency of emergency radioactivity release equalling or exceeding action level of evacuation of personnel living beyond exclusion zone should not exceed 1E-07 per r-y

In Risk-Informed Performance based evaluation, performance parameters not quantifiable e.g. Safety culture, effectiveness of training etc. are also assessed. In India, Risk-Informed decision implies Risk-Informed Performance based evaluation.

4. QUALITY ASSURANCE

As brought out earlier, assumptions and uncertainties are involved in evaluation of SM be it, a deterministic method with conservative/best estimate approach, or probabilistic method. Even analysis done with the different state-of-the-art codes or even with same code with experience analysts, may produce significantly different results. Therefore, it is important to make detailed quality assurance programme and implementation, to assure evaluations of SM done by either of the two methods - deterministic and probabilistic, are reliable.

UA done either for deterministic BE approach or PSA may be worthwhile to standardize. Also, it may be possible to standardize failure postulation, boundary conditions, allowances on initial parameters of general nature in deterministic conservative analysis. Proper organization, training, qualification of analysts, detailed procedure for performance and verification, disposition of non-conformance and documentation are important elements of quality assurance (QA). The topical QA programme for SM evaluation should be appropriately addressed to the overall QA programme of the utility. The report on evaluations of SMs evaluation before submission to the regulatory body should be peer reviewed.

5. ASPECTS OF SAFETY MARGIN

There are various issues where assessment of SM is done for making regulatory decision.

The list below includes the following:

a) To demonstrate that margins exist in the analysis results for the PIEs considered in the design.
b) To show that adequate safety margins exist in the proposed safety related modification in the plant structures, systems or components.
c) To propose cost reduction oriented modifications under the cover of existing or reduced but still acceptable SM.
d) Reevaluation/improvement of SM by screening out extra conservatism in input parameters, using latest state-of-the-art code, new knowledge about a sensitive parameter allowing more realistic but still conservative value etc.
e) Improved fuelling scheme for better fuel utilization to use up some of the existing SM.

5.1 Power Uprates

Besides above, in plant proposal for significant power uprates which is the topic for this technical meeting requires reassessment of SM. The parameters to be considered for evaluation of SM may include DNBR, linear heat rate of fuel pin/channel, clad temperature, differential condensor cooling water temperatures at the outfall etc.

In India, there are no cases where utility has proposed operation at power level higher than the original rated power. However, there are few instances where reactor was derated by the regulatory body to limit any further degradation of safety significant components and later on, stage-wise proposals were made by the utility to increase power back to FP level. Two such cases are given in table 2 below:

Table2: Power derating and increase

	De-rated Power	Current Power	Reasons
RAPS UNIT 1	50 % FP	75 % FP	End-shield crack
MAPS UNIT 2	75 % FP	100 % FP	Problem in Calandria Inlet manifold

These proposals were cleared by the regulatory body after ascertaining the SMs existed for design basis initiating events, and operational limits and the capability of the safety function of the concerned component and/or related issue.

6. CONCLUSION

In conclusion, the following may be noted:

a) Evaluation of SM both by deterministic and probabilistic methods is necessary for risk-informed regulatory decision as practiced in India.

b) The practices followed in evaluating SM by deterministic method may vary with issues from country to country. Also, there may be requirement of minimum safety margin for certain parameters.

c) The parameters and their values as acceptance criteria may be different depending on reactor type and design and country’s policy for the deterministic analysis. However, for probabilistic approach, the parameters and their acceptance values will not vary with reactor design and, type, except country’s policy and hence a common set of probabilistic safety criteria/goals can be evolved.
d) It’s necessary to set standard approaches on making assumptions and failure postulations for conservative analysis. It may also be worthwhile to standardize methodology to evaluate uncertainties analysis.
e) A detailed QA programme having interface with overall (corporate) QA programme, should be implemented to assure confidence in analysis results. The SM evaluation before submission to the regulatory body should be peer reviewed.

ACKNOWLEDGEMENT

I would like to thank my colleagues especially Mr. R. Srinivasa Rao for helping me in preparation of this paper.

REFERENCES

[1] International Atomic Energy Agency, Safety Margins of Operating Reactors, January 2003. IAEA-TECDOC-1332

[2] 75-INSAG-3: Basic Safety Principles for Nuclear Power Plants, IAEA, 1999.

[3] Safety Assessment and Verification for Nuclear Power Plants; Safety Guide no. NS-G-1.2, IAEA, 2001

[4] Technical Specifications (Part B: Basis), for Standard Indian PHWR.

[5] Design basis events for PHWR: AERB Safety Guide, AERB/SG/D-5, 2000.

[6] Accident Analysis of Nuclear Power Plant, IAEA.

POWER UPRATE FOR OPERATING PLANTS IN KOREA

G.S. Auh
Korea Institute of Nuclear Safety,
Daejon, Republic of Korea

C.H. Ban
KEPCO Nuclear Fuel Company,
Daejon, Republic of Korea

Abstract

In this paper, the present status of Nuclear Power Plants in Korea is first introduced. To present the plan of Power Uprate in Korea, the types of Power Uprating per international experiences are summarized. The contents of Power Uprate for Operating Plants in Korea are background, project goal including methods to attain the goal, target plants, schedule, important technical conditions and project organization for the Power Uprate.

1. STATUS OF NUCLEAR POWER PLANTS IN KOREA

Korea has achieved rapid growth in nuclear power since 1978, when the commercial operation of the first nuclear power plant, Kori unit 1, was started. Korea now has 18 operating nuclear units consisting of 14 PWRs and 4 PHWRs. Two PWRs (Ulchin 5, 6) are now under final stage of construction, and 8 additional units are planned: 4 KSNP+s (Shin-Kori 1,2 & Shin-Wolsong 1,2), 2 APR-1400s (Shin-Kori 3,4) and 2 more APR-1400's. Out of the 8 additional units, the projects of 6 units were already started and the project of 2 more APR-1400 units is under a planning stage.

The following 2 Tables show detailed information of Nuclear Power Plants in Korea.

Status of Nuclear Power Plants

Plant		Reactor Type	Capacity (MW)	Project Management	NSSS Supplier	Plant A/E	Commercial Operation
Kori	#1	PWR	587	W/H	W/H	Gilbert	Apr. 1978
	#2	PWR	650	W/H	W/H	Gilbert	Jul. 1983
	#3	PWR	950	KHNP	W/H	Bechtel/KOPEC	Sep. 1985
	#4	PWR	950	KHNP	W/H	Bechtel/KOPEC	Apr. 1986
Wolsong	#1	PHWR	678.7	AECL	AECL	AECL	Apr. 1983
	#2	PHWR	700	KHNP	AECL/DOOSAN	AECL/KOPEC	Jul. 1997
	#3	PHWR	700	KHNP	AECL/DOOSAN	AECL/KOPEC	Jul. 1998
	#4	PHWR	700	KHNP	AECL/DOOSAN	AECL/KOPEC	Oct. 1999
Yonggwang	#1	PWR	950	KHNP	W/H	Bechtel/KOPEC	Aug. 1986
	#2	PWR	950	KHNP	W/H	Bechtel/KOPEC	Jun. 1987
	#3	PWR	1,000	KHNP	DOOSAN/CE	KOPEC/S&L	Mar. 1995
	#4	PWR	1,000	KHNP	DOOSAN/CE	KOPEC/S&L	Jan. 1996
Ulchin	#1	PWR	950	KHNP	Framatome	Framatome	Sep. 1988
	#2	PWR	950	KHNP	Framatome	Framatome	Sep. 1989
	#3	PWR	1000	KHNP	DOOSAN/CE	KOPEC/S&L	Aug. 1998
	#4	PWR	1000	KHNP	DOOSAN/CE	KOPEC/S&L	Dec. 1999

Status of Nuclear Power Plants

Plant		Reactor Type	Capacity (MW)	Project Management	NSSS Supplier	Plant A/E	Commercial Operation
Yonggwang	#5 #6	PWR PWR	1,000 1,000	KHNP KHNP	DOOSANWH DOOSANWH	KOPEC KOPEC	May. 2002 Dec. 2002
Ulchin	#5 #6	PWR PWR	1,000 1,000	KHNP KHNP	DOOSANWH DOOSANWH	KOPEC KOPEC	Jun. 2004 Jun. 2005

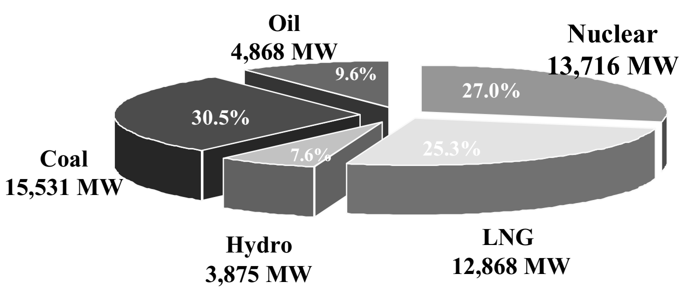
Plant		Reactor Type	Capacity (MW)	Commercial Operation	Remarks
Shin-Kori	#1 #2 #3 #4	PWR PWR PWR PWR	1,000 1,000 1,400 1,400	Sep. 2008 Sep. 2009 Sep. 2010 Sep. 2011	$KSNP^+$ $KSNP^+$ APR1400 APR1400
Shin-Wolsong	#1 #2	PWR PWR	1,000 1,000	Sep. 2009 Sep. 2010	$KSNP^+$ $KSNP^+$
New Project	N#1 N#2	PWR PWR	1,400 1,400	Jun. 2013 Jun. 2014	APR1400 APR1400

$KSNP^+$: Improved KSNP

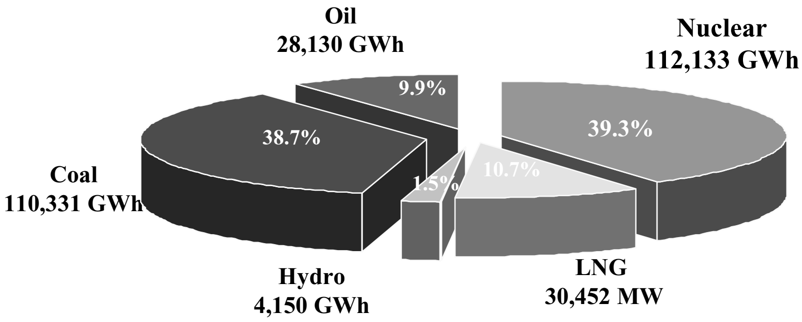
The following 2 Figures show the recent electricity generating capacities and production results of Korea from various sources including coal and nuclear. As can be seen in the figures, even if nuclear has around 27% capacity, it produces about 40% of the electricity of Korea.

Status of Nuclear Power Plants

Electricity Generating Capacity



Electricity Production Results



2. POWER UPRATING PER INTERNATIONAL EXPERIENCES

Power Uprate methods can be categorized into 3 groups per international experiences, namely, Measurement Uncertainty Recapture, Stretch and Extended Power-Upratings.

- **Measurement Uncertainty Recapture (MUR)** Power Uprating is power uprating around 1.4 % by updating Power Measurement of Ultrasonic Measurement of the Feedwater Flow Rate

- **Stretch** Power Uprating is power increase by about 5 % with Minimized Component Changes

- **Extended** Power Uprating is power uprate up to 20 % accompanied by replacement or repair of major components such as S/G, HP TBN, Generator, Transformers, etc.

The following Figure shows International Experiences of power uprating as of May 2002. As can be seen in the Figure, power upratings for 79 NPPs were completed in the USA and 20 NPPs in Europe.

International Experiences (as for 2002, May)

- USA : 79 NPPs Completed

Status		PWR	BWR
Completed	MUR	15	3
	Stretch	30	24
	Extended	1	6
Sub-Total		46	33
On-going		4	3
Total		50	36

- Europe : 20 NPPs

3. POWER UPRATE FOR OPERATING PLANTS IN KOREA

3.1. Background

Restructured electric industry in Korea favors power uprate as are the cases for most countries in the world. But Korea does not have project experience and nor safety review experience of power uprate. The KINS (Korea Institute of Nuclear Safety) has started Basic Regulatory Research in 2002. A power uprate represents changes in the operating license and needs a priori regulatory approval. It can be said that the requirements of the licensing rules have to be met after a power uprate. The followings are current industry plan and concept for power uprate.

3.2. Project Goal

The present target value of power uprating is by about 4.5 % with the Stretch method, which may be optionally accompanied by MUR method by about 1.4 %. This means that the Power Uprating will be between 4.5 % and 5.9 % on current 2775 MWt plants.

3.3. Target Plants

The Target Plants are Kori 3,4 and YGN 1,2, which are Westinghouse 3-loop type with 2775 MWt.

3.4. Schedule

The following is current schedule of the project.
Preliminary Design: September 2002 – November 2003
Detail Design: December 2003 – 2005
Licensing and Plant Application: 2005 – 2007.

3.5. How to Get the Goal

The followings are methods to attain the project goal of around 5% power uprating.

- Utilize Initial Design Margin
- Design Methodology Improvement
  - ✓ T/H Design: RTDP (Revised Thermal Design Procedure) from current RTDP (Improved Thermal Design Procedure)
  - ✓ LBLOCA: BE methodology of KREM (KEPRI Realistic EM) from current W-EM methodology
- Component Improvement
  - ✓ High Pressure Turbine blades will be changed
  - ✓ Fuel will be changed to RFA (Robust Fuel Assembly) from the present rotated V5H.

To Increase Power, the primary side power is to be increased by increasing Core Delta T.

$Q = m \cdot Cp \cdot \text{Delta T}$

Also, the present fixed value Tavg control will be changed to “Tavg Window (± 3F)” Control. The secondary side power is to be increased by increasing Steam Flow Rate

$Q = m \cdot \text{Delta h}$

The Turbine Regulator Control Program will also be changed.

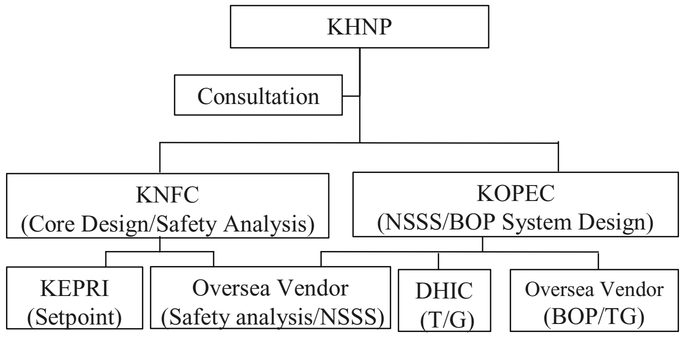
3.6. Important Technical Conditions

The followings are important technical conditions for the power uprating.

- SG Tube Plugging for Safety Analysis: 7%
- Thermal Design Flow is reduced by 1.5% to compensate for degraded RCS flow rate.
- T_cold is reduced by 4F by keeping T_hot the same
- Feedwater Temperature is increased by 6^{o}F
- Steam Pressure is decreased by 5%
- Steam Flow Rate is increased by 5%
- LBB (Leak Before Break) will be applied for the Structural Analysis

3.7. Project Organization

The following figure shows the project organization. As can be seen, the project is leaded by the KHNP (Korea Hydro and Nuclear Power), which is the nuclear power generating company in Korea. Two major overseas vendors that have many power uprating experiences will assist the primary contractors of domestic companies.


KHNP
Consultation
KNFC
(Core Design/Safety Analysis)
KOPEC
(NSSS/BOP System Design)
KEPRI
(Setpoint)
Oversea Vendor
(Safety analysis/NSSS)
DHIC
(T/G)
Oversea Vendor
(BOP/TG)

EVALUATION OF UNCERTAINTIES FOR SAFETY MARGINS DETERMINATION AT THE ANALYSIS OF MAXIMUM DESIGN BASIS ACCIDENT IN RBMK-1500

A. Kaliatka, R. Urbonas, M. Vaisnoras
Laboratory of Nuclear Installation Safety, Lithuanian Energy Institute,
Kaunas, Lithuania

Abstract

During the accident analyses for the determination of margins between physical or licensing limitations and the calculated plant parameters the "conservative", "partially-conservative" or "best-estimate" approaches were used. This paper presents the results of analysis of the worst case of maximum design basis accident at RBMK-1500 - break of main circulation pump pressure header with failure to close check valve of one group distribution header. Performed analysis allowed to estimate number of fuel channels, where fuel rods cladding could be affected. This information further can be used for the analysis of radiological consequences. Two types of analysis were performed: "best-estimate" and "partially-conservative". The comparison of calculations show that the peak fuel cladding temperatures for "best-estimate" calculation taking into account the uncertainty and sensitivity analysis is slightly lower in comparison with "partially-conservative" calculation. That enables to draw a conclusion that in the most cases both approaches – either "best-estimate" or "partially-conservative" can be applied. The latter approach looks tempting, since in this case only one calculation is sufficient; while in the case of "best-estimate" approach at least 59 calculations are required. Thus, "partially-conservative" approach requires considerably less computational time. However, when with this methodology obtained results do not meet acceptance criteria, the complete analysis by employing "best-estimate" approach is necessary.

1. INTRODUCTION

In the case of Loss of Coolant Accident (LOCA) the integrity of primary circuit is violated and coolant is discharged into compartments, which surround the pipelines and equipment of primary circuit. The amount of discharged fission products will depend on the number of fuel elements, which cladding is damaged (integrity is violated). Therefore analysis of LOCA type accidents is very important for safety evaluation of nuclear power plants.

The accident analysis performed in the frame of safety evaluation of Ignalina NPP [1] shows - from the consequences of all LOCA type accidents, taking into account the rupture size and peak fuel cladding temperature, the worst consequences for RBMK-1500 occur in the case of Main Circulation Pump pressure header break with failure to close check valve of one Group Distribution Header. This case is called Maximum Design Basis Accident. Such analysis was performed using two approaches: "best-estimate" and "partially-conservative". In the firs case the RELAP5 Ignalina Nuclear Power Plant (NPP) model with realistic boundary and initial conditions of RBMK-1500 was used and the main contributors to the uncertainty of the results were identified. However this sensitivity analysis requires a certain number of calculations to perform. The use of second "partially-conservative" approach leads to minimise the number of calculations. In this case the best estimate code RELAP5 with conservative boundary and initial plant conditions were employed.

2. MAXIMUM DESIGN BASIS ACCIDENT ANALYSIS

Thermal hydraulic analysis of Main Circulation Pump (MCP) pressure header break with failure to close check valve of one Group Distribution Header (GDH) was performed by employing best estimate thermal-hydraulic RELAP5 Mod3.2 code Ignalina NPP model. The detailed description of this model is presented in paper [2].

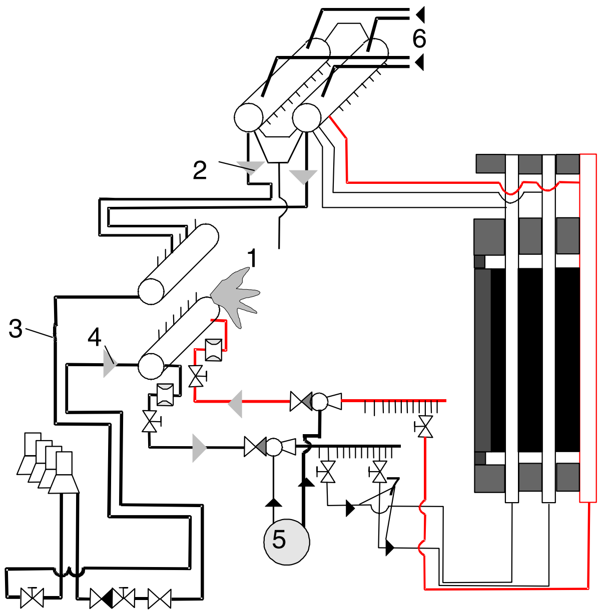


Fig. 1. MCP pressure header break with failure to close check valve of one GDH. Structure of coolant flows: 1 – break of PH; 2 – overflow of coolant from DS; 3 – MCP suction piping; 4 – coolant discharge from MCP pressure piping; 5 – ECCS water supply; 6 – steam supply from intact loop of the MCC; 7 – coolant supply into reactor core

Straight after the MCP pressure header break the water from MCP piping and Drum Separators (DS) is discharged through the break (Fig. 1). The Fast Acting Scram system is activated due to pressure increase in compartments, there the coolant is discharged. GDH check valves, which are located downstream to the break, close and prevent the loss of coolant from Fuel Channels (FC) of the affected Main Circulation Circuit (MCC) loop. The coolant flow stops in the affected MCC loop. However, within approximately two seconds cold water from the Emergency Core Cooling System (ECCS) fast-acting subsystem is supplied to these channels. That enables reliable cooling of these channels. Channels connected to the GDH with failed to close check valve are cooled by reverse coolant flow from DS. At the beginning of the accident, these FC are cooled by saturated water flow, however later (after DS gets empty) only by saturated steam. Due to worsened cooling conditions, fuel cladding temperatures in channels connected to GDH with failed to close check valve increases higher than in other channels of the affected MCC loop. It should be noted that the first fuel cladding temperature increase assert only at the very beginning of the accident and takes a very short time – no more than 10 seconds. Another fuel cladding temperature increase starts approximately 200 seconds after the beginning of the accident and it is caused by the decrease of the reversed coolant flow, which in turn is due to pressure decrease in DS of MCC affected loop (Fig. 2). Considerable temperature increase is possible only in case of operator non-intervention. Operator has a possibility to reduce coolant discharge through the break by closing the repair-valves. In modelling it was assumed that within 10 minutes after the beginning of the accident operator will intervene in the accident process and close repair-valves at the suction and pressure header of the disconnected MCP of the affected MCC loop. As it is shown in Fig. 2, decrease of temperatures starts straight after closure of repair-valves and ECCS water supply regulation (600 s after beginning of the accident). This phenomenon occurs due to the fact that pressure in DS is stabilized after closure of repair-valves. When the water level in Drum Separators of the affected MCC loop is reached the level of steam eater piping connection (~ 1400 s after beginning of the accident) to the channels connected to the affected GDH steam-water mixture from the DS start to flow. Up to that time FC were cooled by saturated steam. Such change of cooling conditions in the channels connected to the GDH with failed to close check valve leads to increase of heat transfer coefficient and fuel cladding temperature decreases down to approximately 200 ^{0}C. That corresponds to the MCC coolant saturation temperature. The fuel channels of intact MCC loop is reliably cooled with water supplied by MCP and ECCS long-term cooling subsystem.

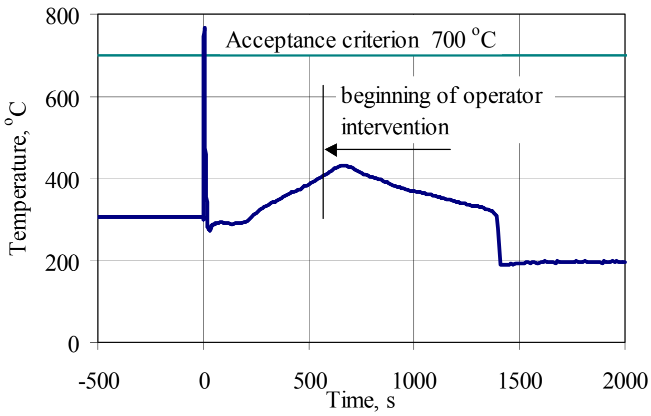


Fig. 2. Peak temperature of fuel cladding in maximum loaded FC connected to GDH with fail to close check valve

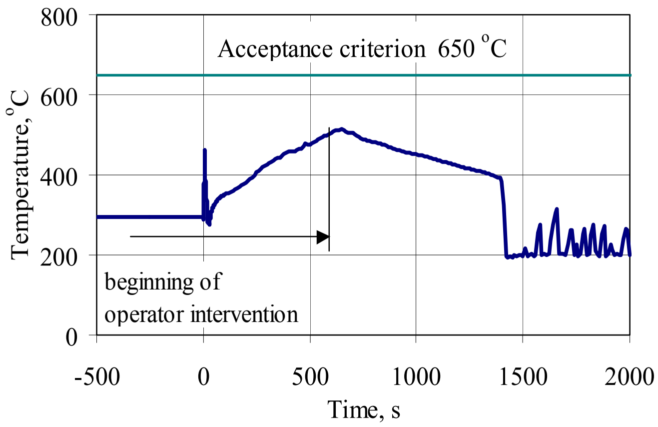


Fig. 3. Peak temperature of fuel channel wall in maximum loaded FC connected to GDH with fail to close check valve

Thus, in case of MCP pressure header break, the only dangerous fuel cladding temperature increase in Fuel Channels connected to GDH with failed check valve, which occur within the first seconds after beginning of accident, should be taken into account. At the same time, maximum fuel channel wall temperatures are considerably below the acceptance criterion of 650 ^{0}C (Fig. 3). That shows that fuel channel integrity will not be violated in any of the channels.

As it is seen in Fig. 4, the fuel cladding temperature in channels of affected GDH with power level higher than 3.0 MW exceeds the acceptance criterion of 700 ^{0}C at the initial stage of the accident. Thus, to perform uncertainty and sensitivity analysis for these channels is not necessary. The fuel cladding temperature for 3.0 MW power FC is very close to the acceptance criterion. To estimate the damage probability of this fuel channel the uncertainty and sensitivity analysis should be performed. The important parameter is fuel cladding temperature in this case. For other parameters, significant for safety (pressure, FC wall temperature) the uncertainty analysis will not performed because the considerable margin of these parameters to the acceptance criteria.

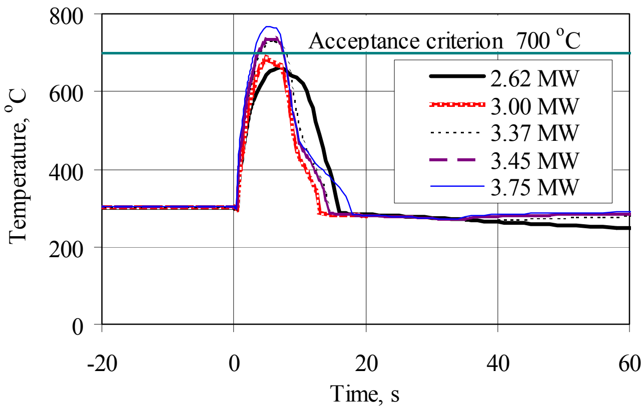


Fig. 4. Peak fuel cladding temperature in the FC connected to GDH with fail to close check valve

3. UNCERTAINTY AND SENSITIVITY ANALYSIS

During the accident analyses for the determination of margins between physical or licensing limitations and the calculated plant parameters the “conservative”, “partially-conservative” or “best-estimate” approaches were used. In the case of “conservative” accident analysis approach the use of conservative codes is combined with conservative boundary and initial plant conditions. To properly assess and address the existing safety margins, required for safety analysis, and to take advantage of unnecessarily conservatisms the “best-estimate” accident analysis approach is used. This approach consists of deterministic analysis using best-estimate codes with realistic boundary and initial plant conditions plus the uncertainty analysis [3]. There are few methods for calculating the uncertainty using best estimate thermal-hydraulic codes. The Pisa method is based on extrapolation from integral experiments. GRS, IPSN and ENUSA methods use subjective probability distributions. AEAT method performs bounding analysis [4].

The thermal hydraulic best-estimate code RELAP5/MOD3 and GRS methodology with the developed package SUSA 3.2 [5] for uncertainty analysis are used in safety analyses performed at Lithuanian Energy Institute. The GRS method considers the effect of uncertainty of input parameters, application specific input data and solution algorithms on the results of calculations. This method is based on statistical tools and provides information in a form useful to decision makers. In the guidelines on selection of uncertainty analysis methods presented in [4], the GRS company’s method is recommended for the purpose to improve the knowledge of the predicted quantity most effectively and to form an understanding of the interactions between the important processes.

The parameters, which may impact the calculation results uncertainty, can be divided into two main groups:

- initial and boundary conditions (coolant pressure, flow rate, feed water temperature, amount of steam for the in-house needs, reactor power, flow energy loss in different MCC components). These values may be impacted by measurement errors;
- RELAP5 code models, assumptions and correlations.

For the analysis the following parameters, initial values of whose may have the greatest impact to the simulation results, on the basis of the knowledge from earlier performed benchmarking calculations are selected:

- pressure in the drum separator;
- coolant flow rate through the MCPs;
- feed water temperature;
- amount of steam for in-house needs;
- reactor thermal power;
- delay time for Fast Acting Scram initiation.

From Ignalina NPP documentation the deviation values are known for these parameters. The deviation values vary from 1.5 – 2%.

For the analysis additional RELAP5 code parameters and models are selected, such as water packing scheme, vertical stratification model, counter current flow limit model, thermal front tracking model, mixture level tracking model and others. It was assumed that selected RELAP5 code parameters are varied in the area where mainly two-phase flow conditions might occur: in the vertical section before the heated channels, in the heated channels, above the heated channels, steam water communications modelling elements, break location, in the area of failed check valve and in the ECCS model. Other areas especially with single-phase conditions are excluded due to the fact that these parameters do not have impact to the results in such regions. In the basic case of calculations some of the code models were disabled. It was due to the fact that they did not have impact to the results. However, in the sensitivity and uncertainty analysis none of the potential contributors to the uncertainty of the results can be excluded.

Before the uncertainty analysis from the many best estimate code output quantities only few important results should be selected (usually – peak fuel cladding and fuel channel wall surface peak temperature, pressure inside the Fuel Channels and Drum Separator pressure), which can be compared with the acceptance criteria and for which uncertainties are evaluated. As it was mentioned previously, the fuel cladding temperature in 3.0 MW power channel of the affected GDH is closest to the acceptance criterion in the case of the postulated MCP pressure header guillotine rupture with failure to close of one GDH check valve. Therefore this code output quantity was selected for the sensitivity and uncertainty analysis. The aim of analysis is to evaluate the number of channels with affected fuel elements. Due to the fact that for the selected case only the upper limit technological parameter's is of importance, in the sensitivity and uncertainty analysis only one-sided tolerance limit is used. For the uncertainty and sensitivity analysis and according to Wilk's formula [4], one-sided tolerance limit (with 0.95 of probability and 0.95 confidence) requires at least 59 runs to be performed. In this case 60 runs were performed.

The behaviour of the calculated fuel cladding temperature in 3.0 MW power FC for all 60 calculation runs is presented in Fig. 5. As it is seen in Fig. 5, fuel cladding temperatures band does not exceed the acceptance criterion of 700 ^{0}C. The largest impact to the temperature has the delay time for Fast Acting Scram initiation and the reactor thermal power. The parameters such as mixture level tracking model in the core region initiation, Counter Current Flow Limitation initiation in the failed check valve model and ECCS resistance coefficients have considerably smaller impact to the calculated results.

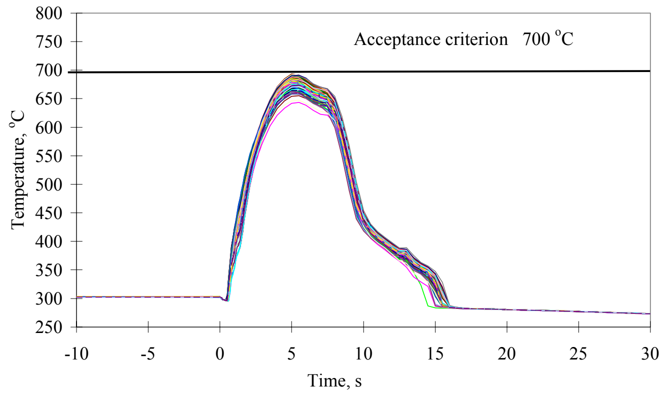


Fig. 5. Fuel cladding temperatures at the location of 2.75 m from the core bottom obtained using SUSA generated runs from RELAP5 calculations

4. COMPARISON OF "BEST ESTIMATE" AND "PARTIALLY CONSERVATIVE" CALCULATION RESULTS

As it was mentioned before, in the many countries the accident analysis is performed by using "partially-conservative" approach (best estimate codes, but the boundary & initial conditions and assumptions remains conservative). The conservative initial conditions are assumed as the worst possible initial conditions and increased (or decreased – depending on what value give more conservative results) by possible measurement errors. According this methodology, the calculation results should be more conservative, in comparison with the results of "best-estimate" approach (using realistic boundary & initial conditions plus uncertainty and sensitivity analysis). This section presents the comparison of results obtained using "best-estimate" approach, described earlier, and calculation obtained by employing "partially-conservative" approach. These two methods can be compared by comparing the margin to the acceptance criterion (see Fig. 6). Analysed accident situations consequences can be acceptable if calculated parameters' values are below the acceptance criteria. This comparison enables to verify accuracy of selected uncertainty and sensitivity calculation.

In the "partially-conservative" approach the initial operating conditions of the plant were set at their bounding limits (the conservative boundary & initial conditions were used):

- The pressure in DS is equal 6.95 MPa. It is the maximum possible pressure in the DS. This pressure is bounded by activation of equipment, protected the MCC from the overpressure (the lowest set point of activation of this equipment is equal 6.96 MPa.
- The coolant flow rate through each MCP is assumed equal 6500 m^3/h. This coolant flow rate is minimum possible and is limited by the position of throttling regulating valves.
- The feed water temperature is assumed pessimistically high and equal 467.78 K. This value is equal to the maximum possible temperature of feedwater 463.15 K, taking into account of 1 % of measurement error.
- The reactor thermal power is assumed equal to the maximum allowable reactor thermal power level 4200 MW increased by 1.06 times (3 % of measurement error and plus 3 % due to the first active control system interaction).

For the “partially-conservative” calculations, the RELAP5 code modelling parameters, which had been recommended by user manuals and were established during the RELAP5 model validation procedure, are used.

The comparison of “partially-conservative” calculation and upper boundary of “best-estimate” results (with realistic boundary & initial conditions plus uncertainty and sensitivity analysis) is presented in the Fig. 7.

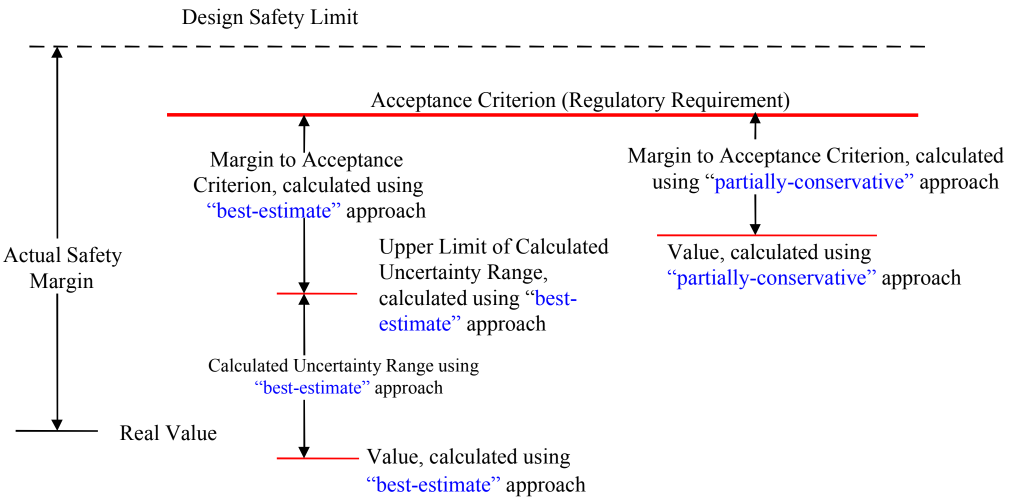


Fig. 6. Illustration of the margin to the acceptance criterion

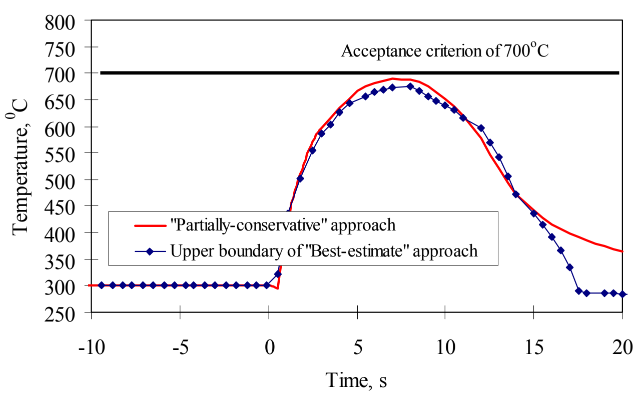


Fig. 7. Comparison of the peak fuel cladding temperature in the fuel channel of 2.62 MW, connected to GDH with failure to close check valve in case of “partially-conservative” calculation and “best-estimate” calculation with uncertainty and sensitivity analysis

Only the peak fuel cladding temperatures in the fuel channel of 2.62 MW, connected to GDH with failure to close check valve are presented. The results shows that “partially-conservative” approach can by used in the accident analysis. In this case the maximum temperatures is 10 – 15 ^{o}C higher as the upper boundary of results using “best-estimate” approach with uncertainty and sensitivity evaluation. That enables to draw a conclusion that in the most cases both approaches – either “best-estimate” or “partially-conservative” can be applied. The latter approach looks tempting, since in this case only one calculation is sufficient; while in the case of “best-estimate” approach at least 59 calculations are required. Thus, “partially-conservative” approach requires considerably less computational time.

5. CONCLUSIONS

The analysis of MCP pressure header break with failure to close of one Group Distribution Header check valve was performed using two approaches: "best-estimate" and "partially-conservative". In the "best-estimate" approach the uncertainty and sensitivity analysis of calculation results was performed using GRS company's methodology.

The obtained results showed that fuel cladding temperature in 3.0 MW power FC, connected to GDH with failed to close check valve, and estimating uncertainty and sensitivity analysis of calculation results is very close but does not exceed the acceptance criterion of 700 ^{0}C. Acceptance criterion is exceeded in the fuel channels with power higher than 3.0 MW and, thus, fuel cladding integrity in these FC can be violated. For evaluation of number of affected Fuel Channels, the real distribution of FC power in the GDH according Ignalina NPP data was analysed. Fig. 8 shows a histogram of the reference channel power distribution at the maximum permissible thermal operating power (i.e., 4200 MW) for GDH with maximum power FC based on the Ignalina NPP data. As it is seen in Fig. 8, there is a group of 12 fuel channels, which power exceeds the 3.0 MW, therefore the integrity of fuel claddings in these FC can be violated with 95% of probability and 95% of confidence. This information about the number of FC with possibly affected fuel claddings further can be used for the analysis of radiological consequences.

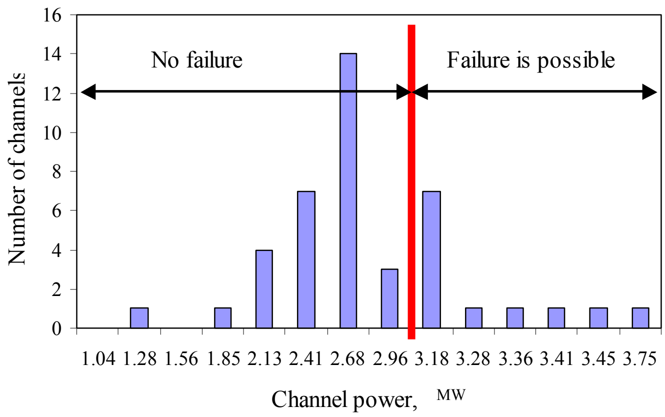


Fig. 8. Real distribution of FC power in the most loaded GDH at 4200 MW power level

The comparison of calculations, performed using both approaches, show that the peak fuel cladding temperatures for "best-estimate" calculation taking into account the uncertainty and sensitivity analysis is slightly lower in comparison with "partially-conservative" calculation. That enables to draw a conclusion that in the most cases both approaches – either "best-estimate" or "partially-conservative" can be applied. The latter approach looks tempting, since in this case only one calculation is sufficient; while in the case of "best-estimate" approach at least 59 calculations are required. Thus, "partially-conservative" approach requires considerably less computational time. However, when with this methodology obtained results do not meet acceptance criteria, the complete analysis by employing "best-estimate" approach is necessary.

REFERENCES

[1] Almenas K., Kaliatka A., Uspuras E.: Ignalina RBMK-1500. A Source Book. Extended and updated version, Lithuanian Energy Institute, Kaunas, 1998.
[2] Kaliatka A., Uspuras E. Thermal-hydraulic analysis of accidents leading to local coolant flow rate decrease in the main circulation circuit of RBMK-1500, Nuclear Engineering and Design, 2002.
[3] International Atomic Energy Agency, Accident and transient analysis of RBMK nuclear power plants, Appendix C, Vienna, 2000
[4] Glaeser H. G. Uncertainty Evaluation of Thermal-Hydraulic Code Results, Int. Meeting on "Best-Estimate" Methods in Nuclear Installation Safety Analysis (BE-2000), Washington DC, USA, 2000.
[5] Kloos M., Hofer E. SUSA Version 3.2. User's Guide and Tutorial, 1999.

ABBREVIATIONS

DS	Drum Separator,
ECCS	Emergency Core Cooling System,
FC	Fuel Channel,
GDH	Group Distribution Header,
GRS	Gesellschaft fur Anlagen- und Reaktorsicherheit mbH
LOCA	Loss of Coolant Accidents,
MCC	Main Circulation Circuit,
MCP	Main Circulation Pump,
MDBA	Maximum Design Basis Accident,
NPP	Nuclear Power Plant,
RBMK	Russian abbreviation for Channelled Large Power Reactor
SUSA	System of uncertainty and sensitivity analysis,

DESIGN CHANGES AT NPP BORSSELE 1997-2005

R. Jansen
Minsitry of Housing, Spatial Planning and Environment,
Nuclear Safety Inspectorate,
Netherlands

Abstract

Not only power uprates have potential impact on safety margins. Also other design changes, like change of core buildup, enrichment and fuel might influence the same safety parameters. In this paper the recently realised and proposed design changes from the Dutch NPP Borssele are presented. A view is given of the most imported impact of those changes and how this is coped with.

1. INTRODUCTION

The only commercial nuclear power plant still existing in the Netherlands is of PWR-type. The NPP Borssele commenced operation in 1973, is a two-loop plant with a thermal power of about 1370 MW. In chapter 2 first some plant information is given.

In 1993 the first periodic safety review was finished and resulted in major refurbishments of the plant over the period 1994-1997 (the so-called Modification Project). After this the plant would be based on a modern safety concept, with a renewed safety report. The safety report would contain modern safety analyses. Except from a core loading change in 1985, till 1997 no major changes in the fuel or core design were made. In chapter 3 a short information is given about the core loading changes.

Parallel to the preparations for the implementation of the Modification Project in 1995 the power plant embarked on a project to change the fuel design and the enrichment at the same time. The implementation started in 1997 and was complete in 2001.This is described in chapter 4.

Now the new fuel and higher enrichment are realised, the power plant has started a follow-up project to change the fuel design first (2004) and then again the enrichment (2005). This is mentioned in chapter 5.

Finally in chapter 6 conclusions are drawn.

2. POWERPLANT AND CORE CHARACTERISTICS

The most important operating parameters of the plant are the system pressure of 155 bar, system average temperature of 305 °C and core heatup of 25 °C. The core consists of 121 fuel bundles of 15x15-20 design (205 fuel rods per bundle) and 28 AgInCd control/shutdown rods. The average linear heat rate is 203 W/cm. The enrichment is 3.3 w/o and the average burnup 33 MWd/kgHM.

The core is directly controlled by the reactor protection system (automatic shutdown) of which the maximum power limit 110% and the minimum pressure distance to boiling is 15 bar can be mentioned. Furthermore the core design limits are $F_q < 2.8$ (ratio of maximum to average linear heat rate), $F\Delta H < 1.7$ (ratio of maximum to average enthalpie rise in the cooling channel) and DNBR > 1.45. Except from DNBR the values are measured discontinuously by the so-called aeroball system. Strings of metal balls are pumped to the core and subsequently read out with respect to the radiative decay. Supported by computer software programs it is possible to obtain a 3D-powerdistribution of the core including the safety parameters. For operation there are some safety margins in place. The first one is called a fabrication and measurement tolerance factor (Ft = 4%). The second one is only applied to Fq and is called margin for rodmanouvering (Fr = 21%).

3. CHANGE OF CORE LOADING SCHEME

In 1984 the power plant decided to change the loading scheme of the core. It was changed from out-in-in to in-in-out (low leakage). This change saves neutrons. It has positive effect on the reactor vessel fluence and at the same time saved 10% of fuel. As for the core margins of FΔH and Fq were used. The calculated and measured values increased, but stayed below the limits. Also safety margins in the fuelrod design were used. This project was not structurally handled as a modification. That is why some years later it was discovered that the reactor protection setpoint of the core power measurements of middle range was not adapted. This setpoint should act at a reactor power of 20%, but due to the reduced neutron density it appeared to act at values higher than 110%.

4. FIRST FUEL AND ENRICHMENT CHANGE

- Irradiation effects (fuel growth, corrosion)
- Transition core restrictions (a.o. design of old fuel !)
- Reactivity effects (MTC, FTC, control rod and boron worth)
- Hydrogen amount at severe accidents
- Fuelhandling + equipment
- Control rod drop times
- Dry and wet storage design
- Fuel cask design

The specific results of these changes were:

Since the first core of 1973 till 1997 no major change of fuel was carried through. In 1997 four fuel bundles of a new design were introduced together with an enrichment change. The reason for the fuel design change was standardisation by the vendor. The following changes were introduced at the same time:

- Enrichment 3.3 → 4.0 w/o (burnup 34 → 50 MWd/kgHM)
- Steel tubes/inconel spacers → zircalloy (effective enrichment increase 0.15 w/o)
- Corrosion resistant cladding (outside ASTM)
- DNBR limit 1.45 → 1,3
- FΔH limit 1.7 → 1.8
- Introduction of coupled neutronic/TH analysis
- Safety margins
    - Ft value 4 → 3%
    - Fr value 21 → 17%

For these kind of changes one can think of a number of subjects that may be investigated:

- Thermohydraulic compatibility (pressure loss, fuel lift forces, core bypass flow..)
- Operation closer to (partly released) limits (Fq, FΔH, DNB)
- Some startup restrictions (amount of boron available)
- Higher boron concentration in all systems
- High burnup restriction: rod limit of 60 MWd/kgHM (RIA, LOCA..)
- Temporary change of wet storage racks, replacement
- Change of handling equipment including setpoints reloading machine

In 2002 the whole core consisted of new design. No problems have been detected so far with the new fuel.

5. SECOND CHANGE OF FUEL AND ENRICHMENT

From 2002 the power plant started a project to change the fuel once again. Again standardisation is the main reason. Furthermore the power plant wants to increase enrichment from 4 to 4.4 w/o. The new fuel has the following changes:

- New spacer with reduced susceptability for flow induced vibrations and improved cooling, reduced irradiation growth
- New tube material (outside of ASTM) with reduced irradiation growth
- New cladding material with reduced corrosion (M5)
- Integrated debris filter
- Introduction of statistical fuel rod design method

The new fuel design is now under assessment by the authorities. Issues to be finalized are the thermohydraulic compatibility (transition phase), statistical design method and LOCA particle clogging.

For the higher enrichment the power plant still has to do the application. Based on the experience with the first project it is expected that the following issues may be affected:

- Dry and wet fuel storage design
- Boron concentration, change to higher than natural B10/B11 ratio
- Setpoints of reactor protection/control
- Burnup limit
- Validity of safety analysis

6. CONCLUSIONS

In the last years and the years to come the power plant has and will stepwise increase fuel use and make use of sometimes consevative margins in order to save money. At the same time the plant has and will introduce state-of-the-art fuel design which leads to improvements in safety margins. The changes lead to a limited number of plant changes like boron concentration, wet storage racks and fuel handling equipment. Also some new design methods were introduced like statistical design of fuel rods and coupled neutronic/TH safety analysis. So far there were no problems experienced. But there are many aspects to assessed and care should be taken that nothing is overlooked.

IMPLICATION OF PROBABILITY ASPECTS IN DETERMINISTIC SAFETY MARGINS FOR TYPICAL WWER DESIGN BASIS ACCIDENTS

K. Korotaev, L. Borisov, S. Borisov, N. Bukin, T. Kuznetsova, A. Podshibiakin, N. Fil, V. Shchekoldin

OKB "Gidropress",
Podolsk, Russian Federation

Abstract

This paper deals with influence of the initial and boundary conditions on the safety margins for WWER-1000 and WWER-1500 reactors during loss of coolant flow and large break LOCA. An attempt is made to take into account the probabilistic character of some initial and boundary conditions used in the deterministic safety analyses. It is shown that this could increase the calculated safety margins with the assigned degree of conservatism in the deterministic analyses remained unchanged. In world practice, a concept of safety margin [1] is frequently used for evaluation of safety level at nuclear stations concerning a number of station characteristics, in particular, those that define integrity of physical barriers on the path of radioactivity release. The concept of safety margin is not used in WWER design documentation, however, this concept is convenient for the purposes of this paper.

1. INTRODUCTION

Regulatory documents on NPP safety [2, 3] require to ensure the conservatism of safety analysis. Conservatism of the analysis is expressed, in particular, by the fact that possible uncertainties of all initial and boundary conditions shall be considered so that to obtain the conservative values of parameters to be compared with the acceptance criteria. However, the excessive conservatism can result in unjustified restrictions in operation of the reactor plant or to overcomplication of the protective systems. Therefore, it seems reasonable to study influence of uncertainty of the important input parameters of safety analysis and to estimate, on this basis, more precisely meeting the acceptance criteria with a required degree of conservatism.

The WWER-1000 and WWER-1500 reactor plants are based on similar design solutions, so the main parameters influencing the safety margins are identical. This paper deals with influence of the initial and boundary conditions on the most important safety margins related to the conditions of fuel rod cooling in the core. In particular, the paper considers how these conditions influence the safety margins such as minimum DNBR and maximum temperature of fuel rod cladding.

Deterministic safety analysis relates to consideration of uncertainty (deviation) of the input parameters, and the various deviations (in plus and in minus) can be conservative for the same parameter at different stages of accident. Consideration of influence of these deviations on the safety margin can considerably increase the number of calculations to be performed for justification of the reactor plant safety.

For example, the safety margins considered in the paper are influenced by the parameters such as:

- reactor power;
- "hot" fuel rod power including the engineering safety factor;
- gas gap conductance;
- thermal conductivity of fuel;
- fuel heat capacity;
- thermal conductivity and heat capacity of cladding;
- heat transfer coefficient to the coolant;
- axial power distribution in the fuel rod;
- coolant temperature at the core inlet;
- coolant flowrate at the core inlet;
- difference in flowrate of the various fuel assemblies because of difference in pressure loss coefficient (PLC).

Thus, the simple enumeration of possible values of parameters would result in performance of a plenty of calculations even if not to consider possible single failures, deviations in the setpoint values, etc.

Therefore, it is expedient to use the statistical approach allowing to account for uncertainty of the input parameters with reasonable restriction of the number of calculated versions.

2. SAFETY MARGINS OF WWER-1500 DURING LOSS OF COOLANT FLOW

This section of paper presents the sensitivity analysis of calculated DNBR concerning some characteristics of the core and certain parameters during loss of coolant flow. Possibility for reduction of calculation conservatism is estimated when absence of DNB is ensured with a required probability.

DNBR sensitivity as regards some governing parameters and characteristics of the core for the chosen conditions is analyzed for the WWER-1500/V-448 reactor plant using the computer code DINAMIKA-97.

Analysis of the minimum DNBR is made on an example of conditions with loss of coolant flow in the WWER - 1500/V-448 reactor plant:

- de-energization of all RCPs;
- instant seizure of one RCP concurrent with loss of NPP power.

The calculation results show that various power shapes could be conservative for different conditions from the viewpoint of DNBR. In practice, three typical power shapes given in Figure 1 are considered.

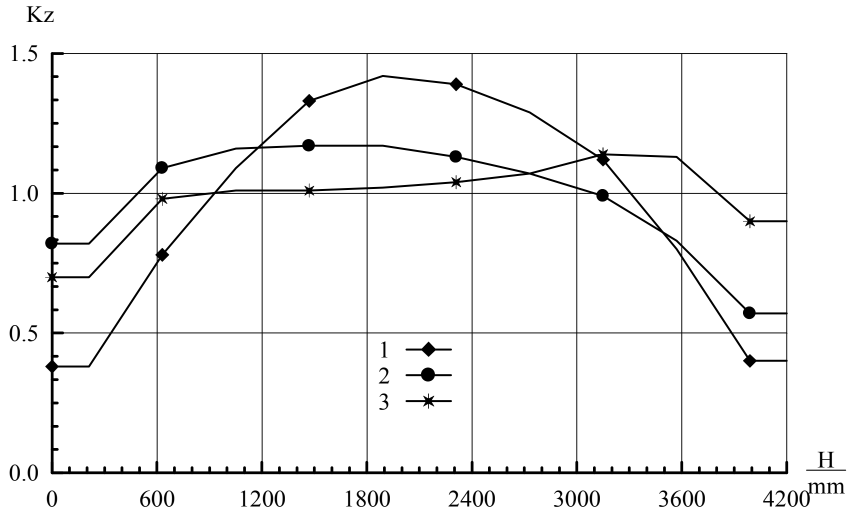


Figure 1 – Power shapes

Distribution (1) corresponds to the beginning of fuel cycle and it realizes maximum-permissible linear heat rate in the core hot spot. Distribution (2) corresponds to the beginning of fuel cycle and it realizes limiting-permissible power of FA. Distribution (3) corresponds to the end of fuel cycle.

Influence of power shape upon DNBR in the hot channel is analyzed for the above conditions with other states being equal. The results of analysis are given in Figures 2 and 3.

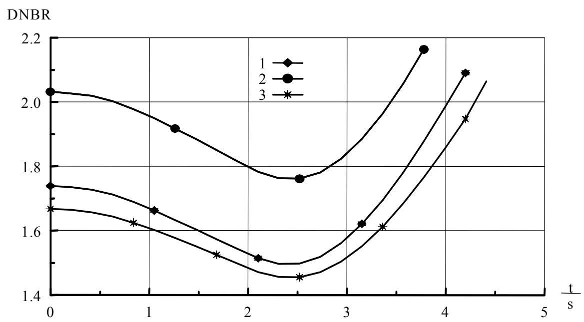


Figure 2 – Influence of power shapes during de-energization of all RCPs

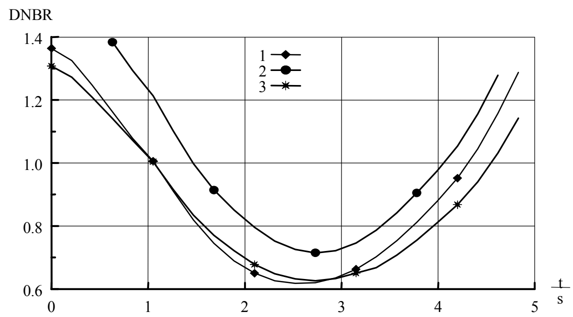


Figure 3 - Influence of power shapes during seizure of one RCP

The hot channel power factor (K_M) is obtained by multiplication of the maximum relative power of fuel rod (Kr) to be obtained in physical calculation by the engineering safety factor. The maximum relative power of fuel rod in WWER-1500 design is determined as Kr=1,50. The engineering safety factor (K_{eng}) accounts for various uncertainties such as error in the neutronic calculations, technological tolerances for fuel rod geometry, for external diameter of fuel pellet, for fuel enrichment, for fuel rod pitch, for fuel assembly dimensions, etc. The engineering safety factor is calculated statistically and its numerical value for WWER-1500 design is determined as 1,12 for the range of $3 \cdot \sigma$.

Figures 4 and 5 show the analysis results for condition with de-enegization of all RCPs from the viewpoint of influence of the engineering factor on DNBR. Some calculation versions were made for the condition with various engineering factors from 1,0 up to 1,16. It is seen from Figures 4 and 5 that influence of this factor is significant, however, there is a significant DNBR.

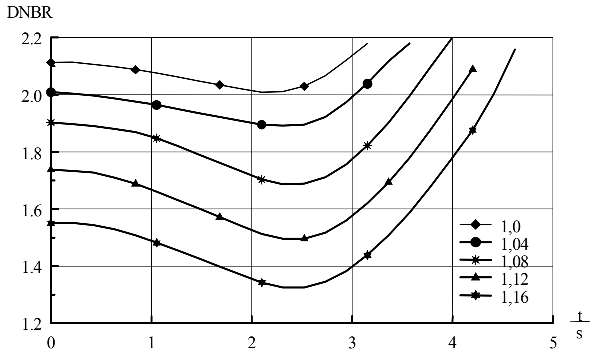


Figure 4 - Influence of K_{eng}

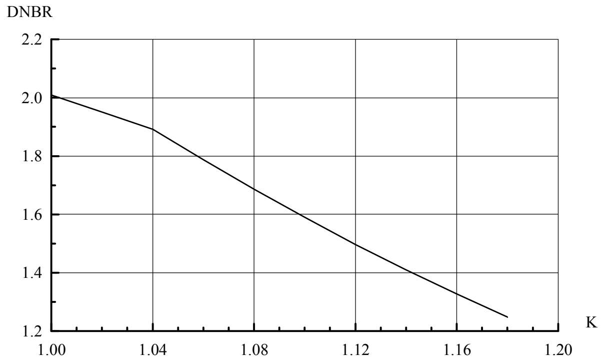


Figure 5 - Minimum DNBR versus K_{eng}

Figures 6 and 7 show influence of the engineering factor on DNBR for condition with instant seizure of one RCP. The calculations are made for the engineering factors from 1,0 up to 1,12. There is no DNB when the engineering factor is less than 1,07.

If to choose the engineering factor with conservatism which corresponds to probability 95 %, then this value will be equal to 1,066. As it is seen in Figure 7, there is no DNB with such value of engineering factor.

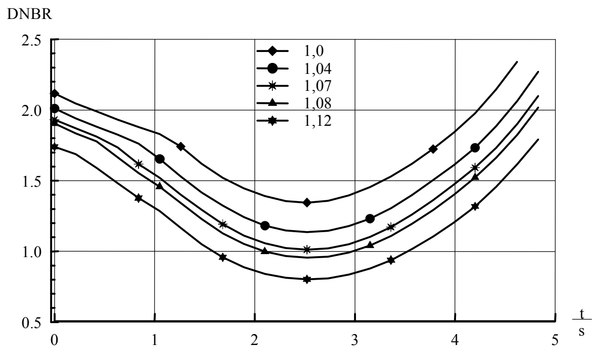


Figure 6 - Influence of K_{eng}

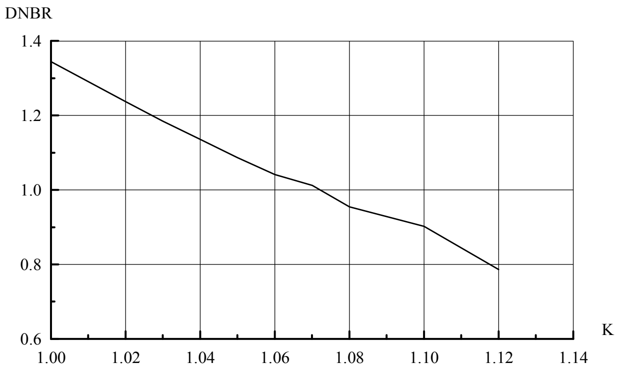


Figure 7 - Minimum DNBR versus K_{eng}

3. SAFETY MARGINS OF WWER-1000 DURING LOSS OF COOLANT ACCIDENT

This part of the paper deals with sensitivity of the results from analysis of large break LOCA for WWER-1000 as regards some typical boundary and initial conditions. For some initial conditions, whose numerical values depend on probability of realization, their influence on the calculated safety margins as regard maximum temperature and depth of local oxidation of fuel rod cladding in the core hot spot is considered.

The analysis was performed by the computer code TETCH-M-97.

Figure 8 shows influence of the maximum linear heat rate on temperature condition in the core hot spot. The calculations were performed for two values of initial linear heat: 448 W/cm and 417 W/cm. The last value corresponds to numerical value of engineering factor being conservative with probability not less than 95 %. Such confidence level is considered usually acceptable in technical applications. In particular, the most important acceptance criterion in the safety analyses for anticipated operational occurrences is absence of DNB in the core hot spot with probability not less than 95 % (although such incidents have much more frequency of occurrence). It is seen from Figure 8 that maximum linear heat rate has significant influence not only on temperature maximum in the first peak, but also on temperature condition in the core hot spot at the subsequent stage of accident (and, hence, on local oxidation).

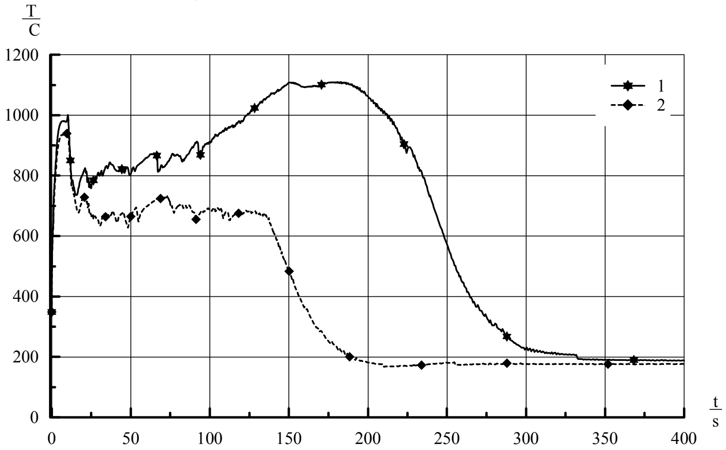


Figure 8 - Influence of linear heat rate
1 – for interval 3σ
2 – for probability 95%

The gas gap conductance between fuel pellet and fuel rod cladding determines the initial temperature of fuel. Since cladding-to-coolant heat removal at the moment of accident practically stops, the cladding is heated due to fuel-accumulated heat. Therefore, maximum cladding temperature in the first peak depends on the reference fuel temperature and, hence, on the gas gap conductance.

Figure 9 shows influence of the gas gap conductance on temperature condition in the core hot spot. The calculations were performed for two values of gas gap conductance: (1) - according to the approach assumed currently for the design safety analyses, and (2) - conservative one with probability not less than 95 %. It is seen, that the gas gap conductance exerts influence not only on temperature maximum in the first peak, but also on temperature condition in the core hot spot at the subsequent stage of accident (and, hence, on local oxidation).

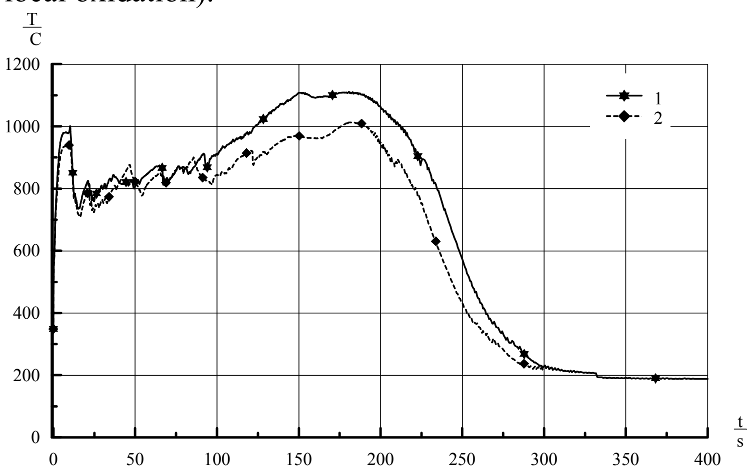


Figure 9 - Influence of gas gap conductance
1 – conductance according to design approach
2 – conductance for probability 95 %

Residual heat significantly influences the core temperature condition in the period after the first peak, when the core is uncovered and heat removal from the fuel rods is limited to low coefficient of heat transfer to the superheated steam. In design analysis of large break LOCA the residual heat is assigned as function of time using the standard correlations (for example, ANSI-79) for infinite time of reactor operation at full power. With this the error of correlation is taken into account conservatively for interval 3σ, i.e. for probability not less than 99,86%. Figure 10 shows influence of residual heat on temperature condition in the core hot spot. The calculations were performed for two residual heat values: (1) - according to the approach assumed currently for the design safety analyses, and (2) - conservative one, with probability not less than 95 %. It is seen, that residual heat exerts very strong influence on cladding temperature in the period after the first peak (on the reflooding stage) and, hence, on depth of local oxidation of the cladding. In particular, for case (2) one should not expect significant oxidation because the cladding temperature practically throughout the whole accident remains below the threshold of steam-zirconium reaction.

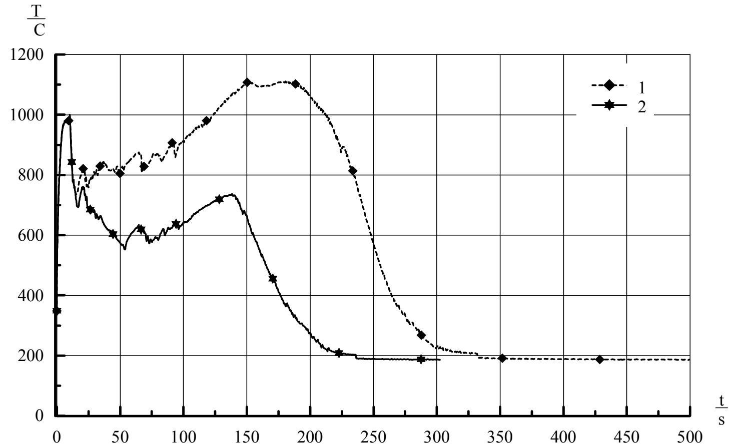


Figure 10 – Influence of residual heat
1 – for interval 3σ
2 - for probability 95 %

4. EFFECT OF STATISTICAL APPROACH TO DETERMINATION OF DNBR

For WWER-1500 reactor plant the minimum DNBR was calculated with account for statistical scatter of some parameters such as:

- engineering safety factor by fuel rod power;
- initial reactor power;
- error in ratio for critical heat flux.

Normal distribution of probability with average value equal to 1,0 and standard deviation σ equal to 0,04 was accepted for the engineering safety factor by fuel rod power. Normal probability distribution of correction coefficient for critical heat flux with average value equal to 1,0 and standard deviation σ equal to 0,131 (OKB "Gidropress" formula for critical heat flux) was accepted for error in determination of critical heat flux. Uniform distribution of probability in a range of possible power deviation ±4 % of the nominal one was accepted for the initial power level

A value was selected for each parameter at random according to the assigned law of distribution. To obtain the sets of values in such a way (the examples are given in Table 1) the condition of loss of NPP power has been calculated. In all, 300 alternative calculations have been performed. A histogram of DNBR distribution given in Figure 11 was constructed by the obtained data.

Table 1

No. of calculation	Kr·K_{eng}	Coefficient in the formula for calculation of critical heat flux	RP initial power	DNBR
1	1,474	1,153	4,26E+03	2,439
2	1,403	0,902	4,24E+03	1,59
3	1,481	0,839	4,14E+03	1,902
4	1,449	0,995	4,29E+03	2,774
5	1,398	0,879	4,12E+03	2,462
295	1,424	0,978	4,19E+03	2,262
296	1,522	1,089	4,18E+03	1,898
297	1,45	1,212	4,24E+03	2,145
298	1,555	1,068	4,39E+03	1,411
299	1,511	0,798	4,09E+03	2,352
300	1,515	0,996	4,40E+03	2,446

The average value of minimum DNBR amounts to 2,03. The results of obtained statistics by the value of minimum DNBR are approximated by normal distribution with standard deviation σ=0,42. The calculation results show that the minimum DNBR amounts to not less than 1,347 with probability not less than 95 %, i.e. there is a significant safety margin. The DNBR value obtained by the usual deterministic approach is not less than 1,0 with probability not less than 95 %.

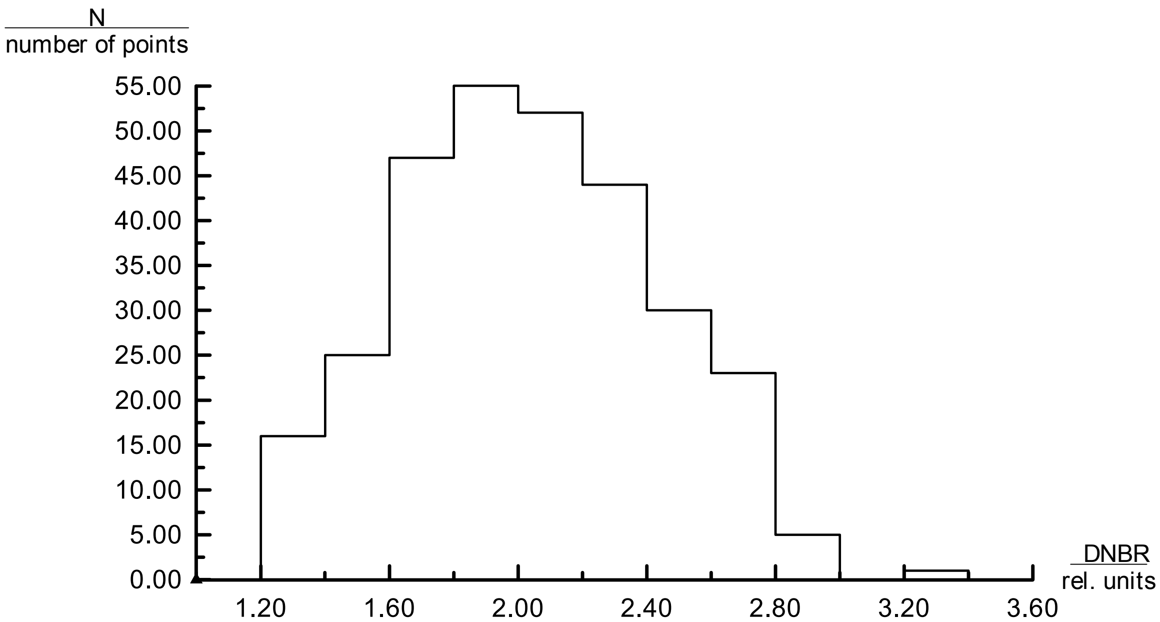


Figure 11 – Histogram of distribution of minimum DNBR

For WWER-1000 reactor plant, the minimum DNBR was calculated with account for possible statistical scatter of some parameters. A set of varied parameters was assumed as for WWER-1500 calculation.

Normal distribution of probability with average value equal to 1,0 and standard deviation σ equal to 0,0533 was accepted for the engineering safety factor by fuel rod power. Normal distribution of probability of correction coefficient for critical heat flux with average value equal to 1,0 and standard deviation σ equal to 0,131 (OKB "Gidropress" formula for critical heat flux) was accepted for error in determination of critical heat flux. Uniform distribution of probability in a range of possible power deviation ±4 % of the nominal one was accepted for the initial power level.

A value was selected for each parameter at random according to the assigned law of distribution. To obtain the sets of values in such a way the condition with loss of NPP power has been calculated. In all, 100 alternative calculations have been made. A histogram of DNBR distribution given in Figure 12 was constructed by the obtained data.

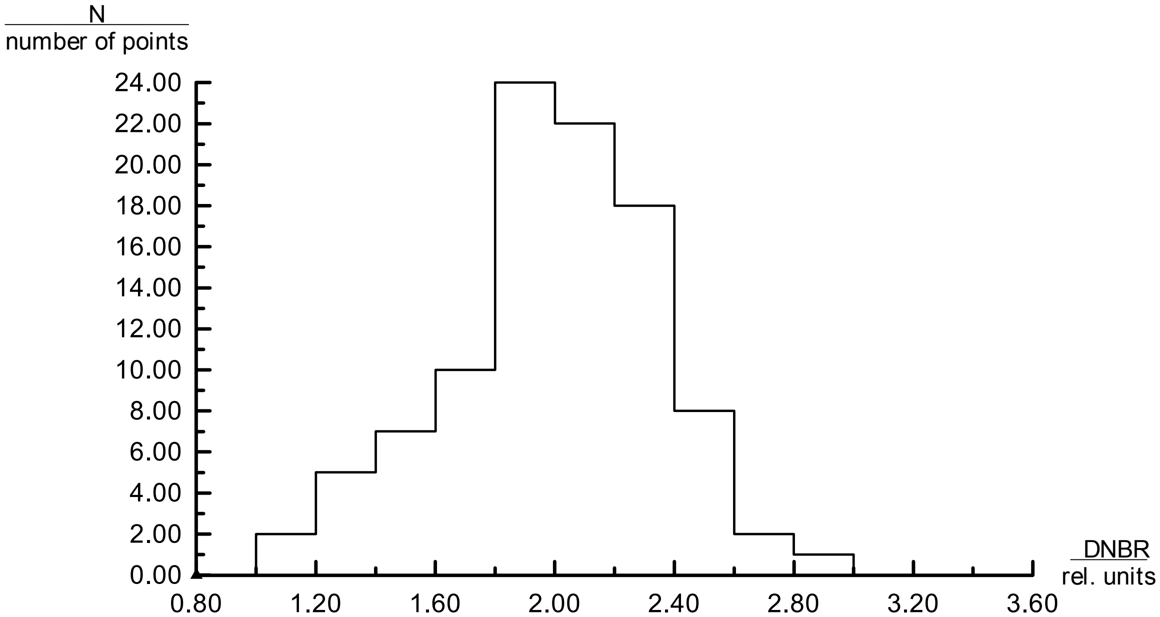


Figure 12 - Histogram of distribution of minimum DNBR

The average value of minimum DNBR amounts to 2. The statistics results obtained for the value of minimum DNBR are approximated by normal distribution with standard deviation σ=0,355. The calculation results show that the minimum DNBR amounts to not less than 1,416, with probability not less than 95 %, i.e. there is a significant safety margin. The DNBR value obtained by the usual deterministic approach is not less than 1,0 with probability not less than 95 %.

5. CONCLUSION

The results of analysis show on an example of calculation of conditions with loss of coolant flow through the core that for WWER-1500 plant there is a significant safety margin by the minimum DNBR.

Calculation of accident conditions with seizure of one RCP concurrent with loss of power shows significant increase of the calculated margin by temperature of fuel rod cladding when calculating with less conservative engineering safety factors by fuel rod power. In particular, preservation of integrity of the hottest fuel rod cladding can be shown.

In analysis of condition with large break LOCA for WWER-1000 the high level of conservatism is shown when estimating the main acceptance criteria for this accident - maximum temperature and depths of local oxidation of fuel rod cladding. The obtained results have shown that at technically reasonable conservatism as regards consideration of possible uncertainty of even some initial conditions the calculated consequences of design basis accident with large break LOCA could be significantly mitigated.

The statistical analysis of DNBR for WWER-1500 reactor plant has shown that the criterion on absence of DNB is met, namely, that DNBR is not less than 1,347 with probability not less than 95 %.

The statistical analysis of DNBR for WWER-1000 reactor plant has shown that the criterion on absence of DNB is met, namely, that DNBR is not less than 1,416 with probability not less than 95 %.

The analysis results show that the proposed approach can be the basis for analyses of the safety margins. However, for each specific safety margin it is necessary to define technically acceptable probability with which this margin shall not be violated, as it was done for DNBR.

REFERENCES

[1] Safety margins of operating reactors and implications for decision-making including considerations of uncertainties of analyses. IAEA Technical Committee Meeting, Vienna, October 15-19, 2001.

[2] Общие положения обеспечения безопасности атомных станций (ОПБ-88/97). ПНАЭ Г-01-011-97, 1997. (General provisions for ensuring safety of nuclear power plants (OPB-88/97). PNAE G-01-011-97, 1997)

[3] Правила ядерной безопасности реакторных установок атомных станций, ПБЯ РУ АС-89, ПНАЭ Г-1-024-90, Москва, 1990. (Nuclear safety regulations for reactor units of NPP, PBYa RU AS-89, PNAE G-1-024-90, Moscow, 1990.)

SAFETY MARGINS AND IMPROVED PLANT PERFORMANCE

J. Husarcek
Nuclear Regulatory Authority of the Slovak Republic (UJD SR),
Bratislava, Slovakia

Abstract

The Slovak utility has started a program for modernisation and power up-rating of the Bohunice V-2 NPP. This has included a gradual increase of the thermal reactor power, plant renovation, changes in set points of safety and control systems, and increase of the fuel burn-up. This paper presents a concept of safety margins in the light of safety limits, acceptance criteria, and values computed by conservative and best estimate calculations. Approaches used for the safety margin estimations are shortly described. Bohunice V-2 power up-rating program is introduced, and application of safety margins to NPP modifications, improved NPP performance and operational flexibility is given. The Bohunice V-2 Upgrading Program is complex. It includes safety improvement and cost reduction oriented modifications. There are indications that safety margin of key parameters may not decrease, or some of them could even increase in comparison to the status before the realisation of the power up-rate. An explanation can be found in NPP safety upgrading, improved unit efficiency, and decrease of data uncertainties used in the safety analyses.

1. INTRODUCTION

Currently, a number of nuclear utilities are planning the power up-rate of their units and many of them have already gone through this modification process. The maximum power level of a nuclear power plant (hereinafter NPP) is included in the technical specifications for the NPP. The regulatory body must approve changes. Safety analyses are means of demonstrating that adequate safety will still exist after power up-rate and associated plant modifications.

This paper presents a concept of safety margins in the light of safety limits, acceptance criteria, and values computed by conservative or best estimate calculations. Approaches used for the safety margin evaluation are shortly described. The Bohunice V-2 power up-rating program is introduced, and application of safety margins to NPP modifications, improved NPP performance and operational flexibility is given. The paper is intended to contribute to the discussion how power up-rate influences the NPP safety margins and to share the experience in that area as well.

To prepare this paper, the IAEA document (TECDOC-1332 /1/) was used.

2. SAFETY MARGINS

The safety of NPP is based on the defence-in-depth concept and adequate protection, which relies on successive physical barriers to control radioactive material and multiple level protections against damage of these barriers. Safety analyses are means of demonstrating that there is undue risk caused by plant operation. Acceptability of overall safety and evaluation of safety margin of a NPP is usually performed and confirmed by appropriately balanced deterministic and probabilistic safety analyses.

Safety margins are the differences in physical units between the established safety limits of assigned parameters associated with failure of changes of a systems or components or with phenomena under consideration, and the calculated values of those parameters. Safety limits are limiting values established for safe operation of the NPP or determined in the design of the NPP. Safety limits for the safe NPP operation are specified in the Technical Specification (TS) for a NPP, and shall not be exceeded during normal plant operation including some anticipated operational transients. These safety limits are excluded from further considerations. Acceptance criteria are generally associated with the assigned parameters for the design basis accidents (DBA) and for some beyond design basis accidents (BDBA). The values of acceptance criteria are fixed as per international standards and accepted by regulatory body as well. They are more restrictive than what the plant is designed for. For practical purposes, the safety margin is understood as the difference in physical units between the stipulated acceptance criteria (regulatory requirements) of assigned parameters associated with failure of changes of systems or components or with phenomena under consideration, and the calculated values of those parameters. Consequently, reducing the safety margin to zero does not necessarily mean that the safety limit is reached. An illustration of the safety margin concept is shown in Figure 1 and commented in the subsequent sections.

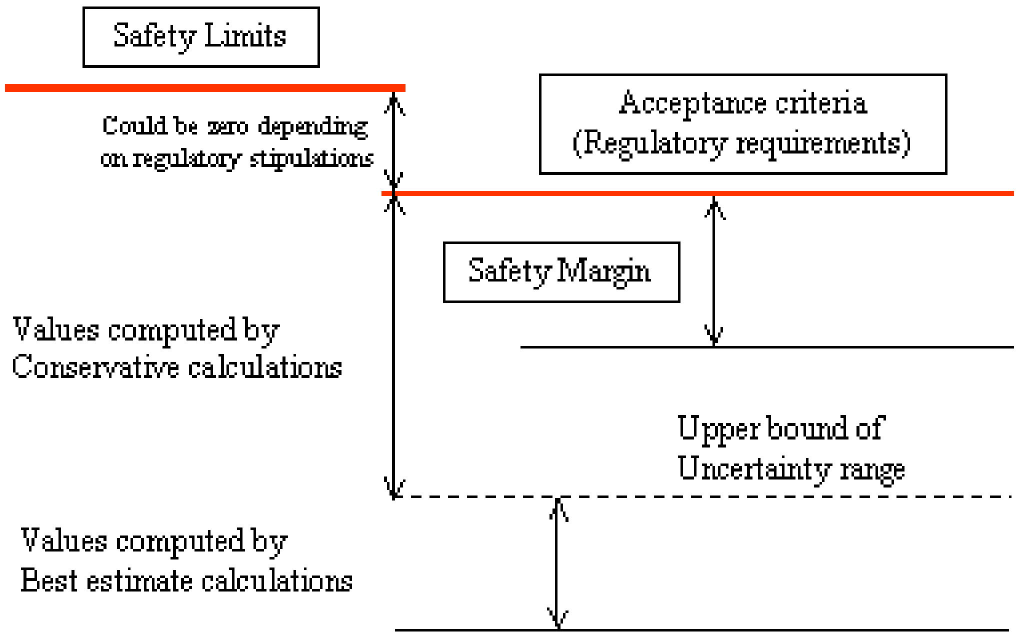


Figure 1: Safety margins illustration

2.1 Deterministic Acceptance Criteria

Acceptance criteria are mostly aimed at preventing damage to barriers against the uncontrolled release of radioactivity and to prevent the unacceptable radiological impact of accidents by means of various measures, including accident management and emergency procedures. Typically, criteria are expressed in terms of a numerical band of a number of calculated parameters, although some qualitative requirements are also used. The acceptance criteria /2/ reflect safety requirements such as:

a) Preventing inadvertent core criticality and excessive power increase;
b) Preventing or reducing the possibility of fuel cladding damage;
c) Limiting damage to the nuclear fuel, including structural damage;
d) Preventing loss of leak tightness or damage to the integrity of the primary circuit;
e) Preventing damage to the integrity of the containment;
f) Limiting radiological impact of the accident within a prescribed period under given conditions;
g) Providing sufficient time for accident management or for emergency response.

The licensee has to provide analytical and experimental evidences to the regulatory body that all qualitative and quantitative criteria are fulfilled.

2.2 Probabilistic Safety Targets

Although emphasis is more focused on deterministic evaluation of safety margins, current international trend requires evaluate the safety margins with probabilistic safety analysis (PSA) as well, to support and supplement deterministic analyses, technical judgement and experience to arrive at the risk informed decision.

The probabilistic safety criteria are viewed as aspiratory targets. Therefore, risk reduction well below these targets could impose great economic burdens, including large capital and operating expenditures. On the other hand, exceeding these criteria significantly could have large economic and social consequences as results of nuclear accidents with constrains that an adequate level of safety must be assured without regards to cost. However, beyond this level of safety, cost and social implications must be considered in dealing with safety improvements.

The top-level probabilistic safety criteria, that should be maintained, include the core damage frequency (hereinafter CDF) and the large early release frequency (hereinafter LERF). The CDF goal of 1×10^{-4} per reactor year and the LEFR goal of 1×10^{-5} per reactor year can be satisfied by various means, therefore the licensee should demonstrate that the overall CDF and LERF criteria are fulfilled (internal events, external events, and all NPP operating states), and plant design is well balanced. For demonstrating this balance, the failure probability of any safety system should not exceed 1×10^{-3} per demand; in case of reactor protection system, the failure probability should not be higher than 1×10^{-5} per demand /3/.

3. ASSESSMENT OF SAFETY MARGINS

Calculations by complex computer codes are used to assess the values of safety margins. For this purpose best estimate and conservative calculations are used. While arriving at the safety margins due considerations should be given for conservatism or the uncertainties in calculation depending on the methodology adopted for computation to assure adequate confidence level either quantitatively or qualitatively as acceptable to the regulatory body.

The methodology to be followed requires a use of the state-of-the-art technology and assurance of the quality in the evaluation of safety margins. The parameter and acceptance criteria, on the values of these parameters considered for assessment of safety margins, are governed by the type and characterisation of the failures (events), phenomena, and changes in the tests or procedures considered. Safety analyses should assure that the safety margins are identified and evaluated for each applicable acceptance criterion.

Fulfilment of deterministic acceptance criteria and evaluation of safety margins (deterministic and probabilistic) is documented in safety documentation submitted to regulatory body for licensing purposes /4/.

For some failure or phenomenon or change in tests or procedures of safety significance under consideration, it may not be possible to calculate the safety margin with the state-of-the-art technology available. Demonstrating either qualitatively or quantitatively, that those situations are adequately covered by the set of design basis transients and that they do not produce an acceptable increase in the usual risk indicators, usually solves this problem. In those cases where the exclusive use of qualitative arguments demonstrates that the safety margin exists, the calculations may be avoided.

3.1 Conservative Estimation

Conservative analysis provides pessimistic estimation of the process behaviour relative to the acceptance criteria under consideration and has to be performed in accordance with prescribed methodology. Best-estimate code in combination with pessimistic assumptions is usually used for conservative analysis. Each step in the conservative analysis, starting from the selection of initiating events, should assure safety margins. Separate set of input parameters and separate accident scenarios should be defined conservatively for each acceptance criterion. Consequently, it may happen that the same initiating event can be analysed with different initial and boundary conditions (failure assumption, accident scenario, etc.) depending on the acceptance criterion, which is under consideration. Supplementary failures in redundancies in mitigating systems are assumed in the analysis beyond single failure criteria if failure probabilities are considerable or required by regulatory body.

3.2 Best Estimate Estimation with Uncertainty Analysis

The best estimate with uncertainty analyses use modelling techniques to realistically describes the physical processes occurring in NPP. Uncertainty analyses are provided to determine the confidence interval of calculated results. There are several techniques how to perform the best estimate calculations with uncertainties. They use the best estimate codes and the best estimate models.

The conservative approach does not give any real indication of the actual safety margin between the acceptance criteria and allowance NPP response. By contrast, the best estimate analyses allow an elimination of unnecessary conservatism in the analyses, and may allow the regulatory body and NPP operator to establish a more consistent picture about actual safety margin.

3.3 Quality Assurance

Assurance of the quality in the evaluation done by two different methods (approaches) namely the conservative analysis and best estimate with uncertainty analyses is essential. This requires that the choice of initial parameters, boundary parameters, and their values, assumptions, and models is judicious. The used computer codes have to be adequately validated. Only well know modelling and internationally accepted techniques can be applied. The analysts have to be properly trained and qualified. The records and documentation to the analyses have to be detailed enough to understand and reproduce the analysis. The variations in the results with the use of different codes and analysts performing the task could be significant. The analyses before submission to the regulatory body should be peer reviewed.

4. UTILISING OF SAFETY MARGINS IN OPERATION AND MODIFICATION OF NPP

There are two basic types of NPP modifications with respect to their purpose: safety significant modifications and cost reduction oriented modifications within the acceptance criteria. Next chapters of the paper deal with second case, i.e., use of safety margin in the NPP power up-rate.

4.1 Bohunice V-2 Upgrading Program

After implementation and completion of the Bohunice V-1 NPP and the Mochovce NPP safety improvement programs, the most important long-term program is the “Bohunice V-2 NPP units upgrading and safety improvement program” (hereinafter “Upgrading Program”) /5/. The concept of “Upgrading Program” was approved in 1997 and should be completed in 2008 year. Main objectives of the “Upgrading Program” are:

a) Achievement of the required units operation safety through reaching the probability targets according to the IAEA recommendations (INSAG 3) for the NPP in operation;
b) Extension of the Bohunice V-2 NPP lifetime to a minimum of 40 years, in accordance with the development plan of SE, a.s.’s production and the technological base;
c) Increase the unit output to the level 102% in second halve of 2004 year, and 104% in 2006 of the current nominal power.

The power up-rate is planned achieved by implementing new highly enriched uranium resulted in the higher fuel burn-up (4-year fuel cycle), NPP reserves, and improved unit efficiency (mainly secondary circuit). This involves more precise measurement of physical parameters in NPP, and decrease of uncertainties, e.g., in the determination of reactor power (change from ±4% to ±3%N_{nom}).

4.2 Utilisation of Safety Margins

The starting point for utilising safety margin is that the current safety margins and weak points are known and well identified. The basis for this exists in licensing analyses.

In each case of modification in the NPP, it is necessary to analyse in details steady state and dynamic characteristics of the NPP including neutron-physical and thermal-hydraulic aspects, behaviour of materials of individual components and their operability, and functional reliability. The analyses should take into account of appropriate values of input parameters, required settings of protective and control systems and interlocks, instrumentation with their sensitivities, acceptance criteria including limits and conditions for the safe operation, and relevant operating procedures, etc. The results of analyses have to be documented in new revised safety documentation (safety analysis report). Regulatory body can accept highly profitable modifications like power up-rate if affected margins stay at acceptable level.

The Bohunice V-2 Upgrading Program is complex. It includes safety improvement and cost reduction oriented modifications. There are indications that safety margin of key parameters may not decrease, or some of them could even increase in comparison to the status before the realisation of the power up-rate. An explanation can be found in NPP safety upgrading, improved unit efficiency, and decrease of data uncertainties used in the safety analyses. The results have to be documented in new (revised) safety documentation (safety analysis report).

5. CONCLUSIONS

The Bohunice V-2 NPP has started the power up-rating to increase the unit output to the level 102% in the middle of 2004 year, and 104% in 2006 of the current nominal power. The power up-rate is planned achieved by implementing new highly enriched uranium resulted in the higher fuel burn-up, NPP reserves, and improved unit efficiency (mainly secondary circuit). This involves also more precise measurement of physical parameters in the NPP, and decrease of uncertainties, e.g., in determination of the reactor power. There are indications that safety margin of key parameters may not decrease, or some of them could even increase in comparison to the status before the realisation of the power up-rate. An explanation can be found in NPP safety upgrading, improved unit efficiency, and decrease of data uncertainties used in the safety analyses.

REFERENCES

[1] Safety Margins of Operating Reactors, Analysis of Uncertainties and Implication for Decision Making, IAEA, IAEA-TECDOC-1332, Vienna, January 2003

[2] Requirements for the Performance of Safety Analyses, UJD SR Safety Guide, BNS I.11.1/1995, Bratislava, 1999 (in Slovak)

[3] Use of PSA Methodology in the Regulatory Body Performance, UJD SR Safety Guide, BNS I.4.2/1999, Bratislava, 2000 (in Slovak)

[4] Regulation on Safety Documentation No.318/2002, UJD SR, Bratislava, 2002

[5] National Report of the Slovak Republic, Compiled in Terms of the Convention on Nuclear Safety, September, 2001, Chapter 2, Pages 31-33

MANAGEMENT OF SAFETY MARGINS AT NPP KRSKO

J. Spiler
NPP Krsko,
Slovenia

A. Janezic
SNSA,
Slovenia

Abstract

NPP Krško (NEK) strategy related to safety of plant operation is to continuously implement improvements to plant hardware, plant processes and operation. The general policy is that plant safety shall not be degraded due to plant aging and plant changes. Plant improvements are in many cases related to optimization which usually involves use of margins in plant design, safety analyses and plant operating parameters. To maintain original plant safety level and to achieve safety enhancement based on new safety standards and operating experience use of margins is necessary to optimize and improve plant operation and performance. However use of margins shall be balanced with adequate margin generation. Margin generation can be achieved through margin retrieval associated with original design and safety analysis, use of improved and more detailed modeling in design and safety analysis, use of modern safety standards which do not contain unnecessary conservatism and through installation of new hardware. In this paper three different projects are presented which involved management of safety margins on a plant level, system design and equipment operating parameters:

1) SGR and power uprate-deterministic safety analysis
2) SGR and power uprate PSA assessment
3) Rx vessel head parameters evaluation

1. INTRODUCTION

Within the modernization program, the set of consistent, comprehensive safety analyses was performed, to demonstrate that the plant could operate safely under new conditions. Methodology selected in performing these studies, have numerous references in US, as well as in Europe. To document and license the increase of Krško Nuclear Power Plant (NPP) power to 2000 MWt with replaced steam generators (RSG) , an exhaustive program of analyses has been conducted. It also covers operation in a range of reactor coolant temperatures, SG tube plugging level of up to 5% and cycles duration of 12 to 18 months.

As far as feasible, the same methodologies/licensing basis, as existed before, have been followed. R.G. 1.70 rev.3 was observed for the accident analyses. A few changes in methodologies/licensing basis have however been introduced. They include Leak-Before-Break studies to exclude large break LOCA dynamic effects. Throughout the work, emphasis has consistently been placed on limiting plant modifications as far as feasible.

In addition to the required set of licensing analyses, NEK decided to perform the integrated safety assessment (ISA) of all plant modifications/changes, with the available plant PSA model and methodology. The starting point and extensive review of the NEK design modification/change data base, and implementation of the reviewed changes/modifications into the PSA Level 1/Level 2 analysis model (originally finished 1992 and 1995 respectively), developed within IPE/IPEEE project (Individual Plant Examination for internal and external events) was completed.

2. ANALYSIS OF SG REPLACEMENT AND POWER UPRATING

2.1. Methodologies/changes to licensing basis

The analyses have been conducted as far as feasible using the same methodologies as exists in the current licensing basis. There has, however, been a limited number of changes to replace obsolete methods, to address new issues and to regain margin. Major changes are described below. All the safety analyses have been done in compliance with R.G. 1.70 Rev. 3. In some instances, maximalist interpretation of the number of cases has been used. Acceptance criteria have generally been taken from the ANS standard criteria except that, in some cases, a more restrictive Westinghouse internal criterion is used often to simplify the analyses.

The program addresses possibility of operation in a range of reactor coolant temperatures (the operating window).

Except for the LOCA Appendix K analyses, the ANS 79 plus 2 sigma decay heat has been used.

For the DNB analyses, the Revised Thermal Design Procedure has been used (instead of the Improved Thermal Design Procedure). This universally accepted methodology statistically combines measurement and correlation uncertainties.

For the steam line break (SLB) core analysis, the shutdown margin had to be reduced.

The Rod Withdrawal from Subcritical Conditions analyses have been made using the TWINKLE code instead of the obsolete WIT6 code.

The overpressure protection has been analyzed taking into account pressurizer safety valves loop seal clearing time, unlike the existing USAR studies, which are non conservative in this respect.

For Large Break (LB) LOCA, the BASH methodology has been used, i.e. the same methodology as in the current USAR. However, as was done for the cycle 16 studies, besides chopped cosine power distribution, skewed to top power distributions have been considered.

The Small Break (SB) LOCA analyses have been made using the NOTRUMP computer code instead of the obsolete codes chain used in the existing USAR.

The analyses of containment response to LB LOCA and to SLB have used the latest Westinghouse methodology for the mass and energy releases and CONTEMPT for the calculation of the containment pressure and temperature time histories, instead of the obsolete methods of the current USAR. In the case of SLB, the onset and amount of water entrainment have been calculated by the RSG supplier.
The calculations of radiological consequences have used new meteorological data and have taken into account the removal of the NaOH tank in the containment spray system.

In order to eliminate the need to consider LB LOCA dynamic effects in the mechanical analyses, as it is allowed and even encouraged by GDC-4, a Leak-Before-Break study has been made for the reactor coolant loop and for the class 1 auxiliary lines greater than 6 inches, using the USNRC approved methodology.

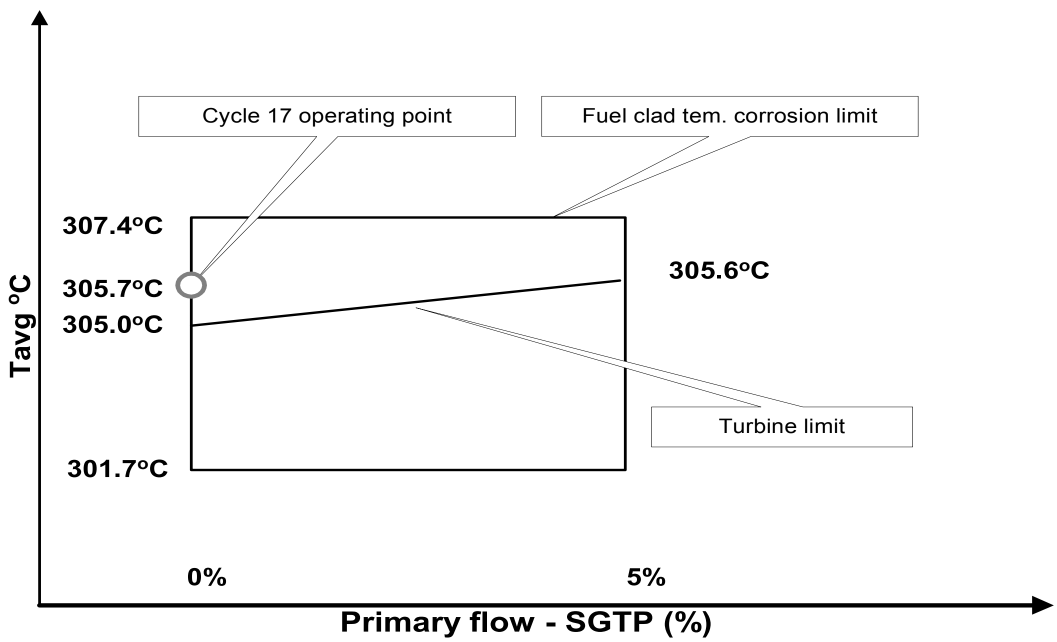
For the reactor coolant loop seismic reanalysis, the more accurate time-history method has been applied and utilized a 3D shock as an input, instead of the 2D shock of the current USAR.

For the reactor internals re-evaluation, a 3D time-history analysis of the reactor vessel/ internals/vessel support/fuel system was also performed for the seismic and pipe break cases, using the WECAN code and employing standard methodology, approved by USNRC.

The reactor internals heat generation rate and the reactor vessel fast neutron fluence calculations use new code version and updated cross-section library and scattering cross-section matrices.

2.2 Operating window analysis and evaluations

The operating parameters to be considered in the analyses have been calculated for a range of average reactor coolant temperatures of 301.7°C to 307.4°C and for 0% and 5% of SG tube plugging. However, with the turbine as it exists, the window available for operation at 2000 MWt is currently lower limited to about 305°C (calculated value). For cycle 17, the temperature has been selected at the maximum value for corrosion of remaining fuel with Zirc-4 cladding, i.e. 305.7°C. Based on observed turbine valves position at 2000 MWt, it is obvious that temperatures lower than 305°C are achievable at this power. When all fuel will be with ZIRLO^{TM} cladding, available temperature window will extend from the actual minimum achievable value to 307.4°C. A later turbine modification could make the full analyzed window available. The window approach provides NEK with the flexibility to select the optimum operating point for each fuel cycle and to plug SG tubes if required, without reanalyses up to 5 %.



Analyses highlights

A very comprehensive program has been conducted, encompassing interface information for the design of the RSG, fuel analyses, accident analyses, systems evaluations including functional definition of the required plant modifications, control studies and mechanical analyses. Some aspects are briefly presented hereafter.

Accident Analyses

A full spectrum of accident analyses has been conducted. This goes beyond what is done for other similar programs in which a number of events are addressed by evaluation. All of them have provided results which fulfill the respective criteria. A few areas are summarized in the following paragraphs.

DNB Analyses

The DNB analyses use the Revised Thermal Design procedure (RTDP). This universally accepted methodology, new for Krško, statistically combines measurement and correlation uncertainties. As input to this statistical combination, a RTDP uncertainty analysis has been conducted for the associated measurements. The RTDP has not only contributed to accommodate the power increase, but has additionally permitted to incrase the enthalpy rise hot channel factor (FDNH) to 1.62 (including uncertainties). The acceptability of the new operating conditions from the standpoint of DNB, have been validated by the most critical DNB analyses, i.e. the Rod Withdrawal at Power and the Complete Loss of Flow, and has been confirmed by the performance of the other DNB analyses. This includes the Feedwater System Malfunction that results in a Decrease in Feedwater Temperature, which has been analysed for a scenario much more severe than in the previous analysis.

LOCA Analyses

The Large Break (LB) LOCA analyses have been conducted using the BASH methodology, as for the previous plugging analyses. It was possible to mantain the peaking factor unmodified at 2.34 and yet fulfill the criteria of the analyses, i.e. maximum peak clad temperature not exceeding 1204°C and percentage of zirconium - water reaction not exceeding 17 %. These results have been obtained despite the consideration of skewed to top power distributions.

For the Small Break (SB) LOCA, besides the fact that an updated methodology has been used for the typical analysis, a sensitivity study to the time of reactor coolant pump trip, has been included.

Overpressure Protection Analyses

The overpressure protection analyses consist in analyses of a Loss of External Electrical Load or Turbine Trip without steam dump actuation, in order to confirm that the pressurizer and steam generator safety valves are adequately sized to prevent overpressurization of the reactor coolant system and of the steam generator steam side. With respect to previous analyses, a major change is the consideration of the pressurizer safety valves loop seal clearing time and of the pressure drops down to the steam generator safety valves. Despite the significant resulting penalty, the criteria of the analyses are fulfilled and the safety valves are confirmed to have adequate sizing for the new operating condition.

Containment Analyses

The containment analyses involve the generation of mass and energy releases in case of LB LOCA and in case of SLB, and the calculation of the containment pressure and time histories following those accidents. In the case of SLB, analysis was first performed assuming dry steam but it rapidly turned out that acceptable results could not be obtained with this excess conservatism. It was thus necessary to take into account water entrainment and to have the RSG supplier determine thesholds and amounts for the water entrainment. The containment analyses criteria are met, even when considering an uncertainty applied to the containment volume.

Systems Evaluations

The capabilities of the NSSS systems to accommodate the new operating conditions have been verified for the safety injection system, residual heat removal sytem, chemical and volume control system, reactor make-up water system, spent ful pit cooling system and auxiliary feedwater system. To be noted that boron dilution is now administratively excluded in mode 4 without reactor coolant pump in operation and in mode 5 in order to avoid too severe boron requirements. For the safety injection system, the time for switchover to hot leg recirculation after LOCA has been determined. The NSSS instrumentation recalibration requirements have been defined. The setpoints of the NSSS control sytems have been determined and shown to enable the plant to accommodate load variations without plant trip. It is to be noted that a small number of setpoints become a function of the selected point in the operating window. The BOP systems have also been verified to have the capability to accommodate the new operating conditions with, however, a small number of modifications which are discussed in chapter 5. Evaluations of the turbine-generator have also shown acceptable results. The Digital Electro-Hydraulic System was renormalized.

Mechanical Analyses

The Leak-Before-Break analysis of the primary loop piping was performed in accordance with the criteria set forth in SRP 3.6.3. Specifically, the operating experience was reviewed, the material condition was assessed, with particular attention to thermal aging degradation, leak rate calculations were performed, and a crack stability analysis was carried out to demonstrate that the required minimum safety margins were satisfied. It was found that the NPP Krško primary loop piping exhibits excellent material toughness properties and that both the operating experience and leak detection capability at the plant, as well as the piping loads and stress levels are very typical of other Westinghouse plants for which leak-before-break has been successfully demonstrated. Margins of safety on loads, crack size and leak rates have been calculated per the USNRC criteria. Based on this plant specific evaluation, it was concluded that dynamic effects of reactor coolant loop pipe breaks need not to be considered any more in the structural design basis of NPP Krško.

A similar analysis has been performed to demonstrate that pressurizer surge line breaks, RHR line breaks and accumulator line breaks can also be eliminated from the structural design basis of Krško NPP.

Primary Components Mechanical Evaluation

A review of primary component (Reactor Vessel, Reactor Coolant Pump, Pressurizer and CRDM pressure housings) stress reports has been made with the purpose of verifying that each primary component pressure boundary will continue to meet its structural integrity criteria in the operating conditions resulting from new thermal and pressure transients. Addenda to original equipment specifications and to original stress analyses have been issued which respectively document the changes in component loadings and the recalculated stresses obtained by rationing the original analysis stress results. Analyses have also been made to verify acceptability for the reactor vessel internals, including impact of increased gamma heating rates and response to seismic and LOCA excitations. The reactor vessel fluence has been evaluated and the new heatup and cooldown curves have been determined.

2.3 SNSA APPROVAL OF ANALISES FOR SG REPLACEMENT AND POWER UPRATING

According to Slovenian regulations the nuclear power plant safety status must be documented in a (Final) Safety Analysis Report. The minimum information required to be included and a format are established by regulatory guidelines. The Krško NPP Safety Analysis Report therefore contains information that describes the facility, presents the design basis and the limits on its operation and presents analyses of structures, systems and components and postulated accident analyses of the facility.

The replacement of the steam generators and the power uprating have affected the current primary operating parameters. In addition, the new steam generators have different geometry, material properties and different hydraulic characteristics. All changes and modifications have had an impact on the original and current licensing and design basis documentation; therefore, new safety analyses and assessments have been required to prove that the plant will be able to operate safely. The safety reassessment and analyses cover thermal-hydraulic (TH), mechanical and structural aspect of modifications introduced by the modernization project.

Comprehensive analyses were started and performed by Westinghouse in 1997 to demonstrate plant safety performance and to confirm the mechanical integrity and life time of systems and components.

The analyses needed to prove that all transients and accident conditions remain within the limits and acceptance criteria for the operating window. The original analyses were performed for one operating condition only, while the new analyses covered an operating window. The concept of the operating window provided more flexibility in the plant operation than the currently licensed operating point. The analyses verified the plant maneurability for a selected operating window and safe operation with new steam generators at an uprated power. The analyses were proceeded in four major phases:

- Phase 1, Estabilshment of new operating conditions (operating window),
- Phase 2, Verification of new operating conditions,
- Phase 3, Plant operating justification,
- Phase 4, Plant Documentation.

The analyses supporting the operating window were consistent with American and European practice.

All of the above analyses are documented in Work reports, Summary report and a revised Updated Safety Analyses Report (USAR) including a revised Krško NPP Technical Specification.

These documents represented the documentation submitted to the regulatory body (SNSA) for approval.

Each of work reports was reviewed in parallel by the Krško NPP, the Technical support organizations (TSO's) and the SNSA. Those reviewers resulted in a list of comments and required changes. After the clarification and resolution of all comments TSO's according to the Slovenian licensing legislation prepared Independent evaluation report(s), which were submitted together with other licensing documentation (Work reports, ect.) to the SNSA for final review and their approval.

3. KRSKO MODERNIZATION PROJECT PROBABILISTIC SAFETY ANALYSIS AND A SUMMARY OF THE MOST IMPORTANT FINDINGS

Within modernization program, the set of consistent, comprehensive safety analyses was performed, to demonstrate that the plant could safely operate under new conditions.

Methodology selected in performing these studies, have numerous references in US, as well as in Europe. In addition to the required set of licensing analyses, NEK decided to perform the integrated safety assessment (ISA) of all plant modifications/changes, with the available plant PSA model and methodology. The starting point was complete and extensive review of the NEK design modification/change data base, and implementation of the reviewed changes/modifications into the PSA Level 1/Level 2 analysis model (originally finished 1992 and 1995 respectively), developed within IPE/IPEEE project (Individual Plant Examination for internal and external events).

The overall project was done in two stages:

- evaluation of plant modifications conducted from the beginning of 1993 up to the outage of 1999 inclusively, referred to as Stage 1 of ISA Project,
- evaluation of modifications that were underway or planned for implementation in the rest of 1999 and in 2000, including those involving power uprate and steam generator replacement (Stage 2 of ISA Project).

The tool used for the purpose of this integrated assessment was PSA model originally developed within the frame of Individual Plant Examination for internal events (IPE) and external events (IPEEE) for NEK. Plant-specific IPE/IPEEE studies were completed during the period 1994 - 96. Their results were used to evaluate the existing plant design and operating practice from the standpoint of risk. Based on that, numerous improvements were defined, such as, for example, plant modifications related to NEK Fire Protection Action Plan. The PSA models (internal events, seismic events, internal fire, internal flood and other external events) resulting from NEK IPE / IPEEE studies essentially represented the plant risk profile by the beginning of 1993, which was defined as a “freeze date”.

To enable this, the model had first to be updated with all changes at the plant that took place since the beginning of 1993, i.e. IPE / IPEEE “freeze date”, so that it would in a realistic manner represent initial pre-modernization risk profile. This was performed through the before-mentioned Stage 1 of ISA project. The product of Stage 1 study was the updated NEK

PSA model "NEK98", which represented NEK plant by the end of Outage of 1999. This model represented a basis for Stage 2 study. It was updated and modified through the Stage 2 of ISA project into the model referred to as "NEK2000". The latter represented projected status of the plant upon the completion of NEK modernization.

The assessment of plant modifications and procedure changes conducted within the project included the following tasks:

- the evaluation of issues with regard to their safety impact,
- the identification of required changes in the PSA model and their implementation,
- the quantification of the updated PSA model and interpretation of results.

The evaluation of issues was performed in three basic steps: the initial screening, the qualitative evaluation and the detailed evaluation. There were 1391 issues evaluated in Stage 1 and 255 issues evaluated in Stage 2.

3.1 "NEK98" PSA model Results and Their Interpretation

The modifications addressed in Stage 1 have been analyzed to assess their risk impacts and to understand their implications on plant operation. There was total of 144 modifications with potential PSA impact that were evaluated in detail. They were grouped in clusters. Among others, the clusters of issues evaluated through the Stage 1 include:

- AMSAC Modification,
- Reduction of BIT Boron Concentration,
- Hot Leg Recirculation Switchover Time,
- Replacement of 125 V Class 1E Batteries,
- Four hours control gas supply to AF and MS valves,
- Replacement of inverters
- Modification of Essential Service Water system,
- Fire Protection Action Plan Modification related to Fire Area CB-3A
- EOP 3.5 Changes, etc.

A comparison was made with the results of the initial integrated IPE / IPEEE model, which represented NEK by the beginning of 1993, to understand implications to the risk profile.
The total CDF (including both internal and external events) in the case of "NEK98" is slightly lower than the IPE / IPEEE estimate. The new estimate is 2.20E-04/yr (which shows approximately 4% reduction from the IPE / IPEEE estimate of 2.30E-04/yr. The internal event CDF contribution had decreased by 2%, the seismic contribution has decreased by 6%, and the internal fire contribution has decreased by 5%. The relative contribution of the different initiator types to the total CDF was unchanged. Internal fire remained the largest contributor followed by seismic and internal events.

In general, the conclusion is that the net effect of modifications and (or) procedure changes through the period 1993 - 1998 inclusively is directed toward improving the reliability of the equipment and (or) the human actions.

Comparison of the Level 2 PSA results from "NEK98" updated model to those from IPE shows that the changes in the Level 2 risk profile are minor. The resulting change in any individual release category (RC) frequency does not exceed 10%.

3.2 “NEK2000 PSA model” Results and Their Interpretation

The modifications under the scope of Stage 2 of ISA project were those that were underway or that were planned for implementation in 1999/2000, including those involving power uprate and steam generator replacement, as well as others from NEK modernization package. Their impact on the NEK PSA model “NEK98” was evaluated and implemented. New model was named “NEK2000”. The results were generated and compared to those of “NEK98” and the implications of plant modifications on risk profile were analyzed.

There were 24 modifications with potential PSA impact that were evaluated in detail throughout the Stage 2. The purpose of evaluation was to identify items that have direct or indirect impact on relevant PSA aspects and to define the required changes in the PSA model.

Among others, the following issues were addressed and reflected in model "NEK2000":

- Fire Protection Action Plan Modifications related to Fire Areas AB-9, SW, and CB-1,
- SG Replacement with modifications of associated systems,
- Power Uprate modifications impacts,
- Spray Additive Tank removal,
- Containment Wet Cavity Design modification,
- Latest EOP-3.5 changes (Rev.8) with Inadequate Core Cooling Monitoring System (ICCMS) installation and implementation, etc.

The total CDF (including both internal and external events) obtained by "NEK2000" is significantly lower than the previous estimate (i.e. "NEK98"). The new estimate is 1.28E-04 /yr., which shows a 42% reduction from the previous estimate of 2.20E-04/yr. Considering both Stage 1 and 2 updates, the reduction is approximately 44%. This is achieved through significant reductions in internal fire and internal events CDF. Internal fire is no longer the dominant contributor to the total CDF.

The CDF for internal events obtained from “NEK2000” was analyzed in detail and compared against the “NEK98” case. The changes were observed and related to the plant modifications made. The reduction in internal event CDF is largely the result of reduction of SGTR accident sequences. The re-configurations of secondary-side systems, performed as part of the SG replacement contribute to the reduction of CDF due to various transient scenarios including Loss of Offsite Power.

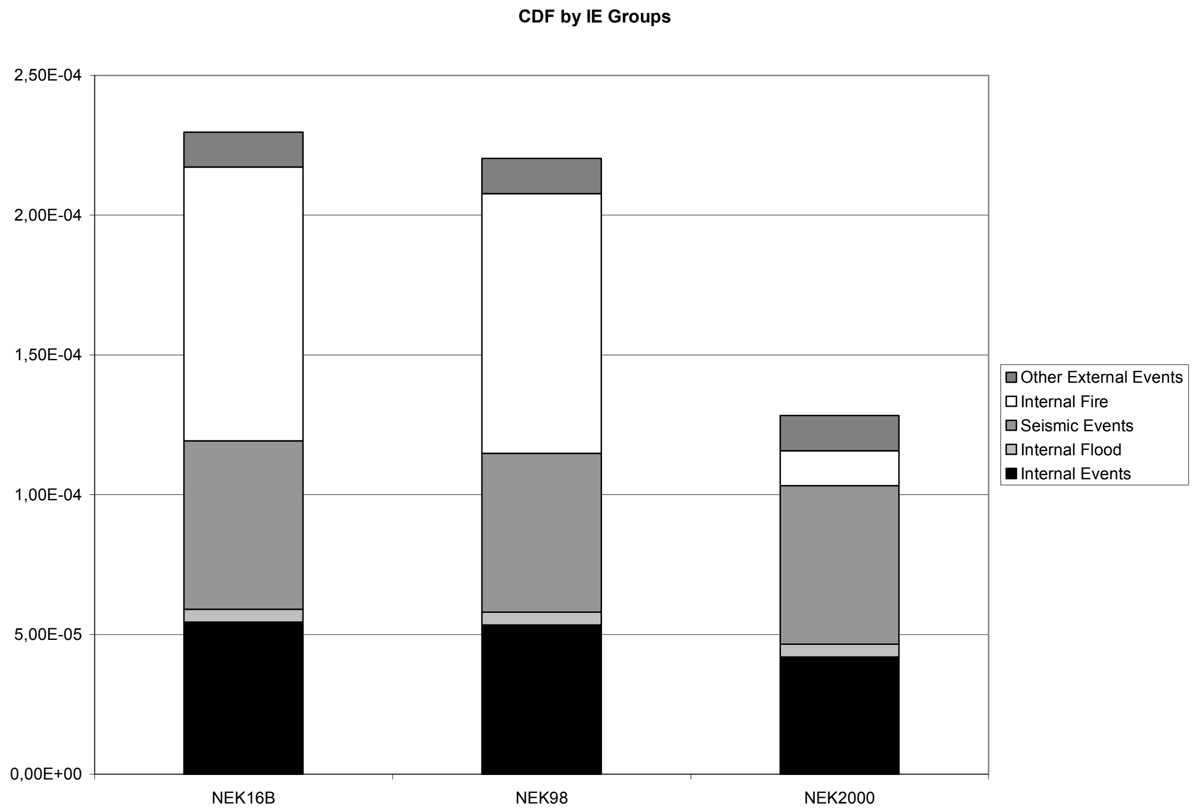
When considering the profile of the total CDF based on "NEK2000", it could be seen that the relative contribution of the various initiator types has changed with respect to "NEK98" model. The CDF due to internal fire events is significantly reduced as a result of Fire Protection Action Plan (FPAP) modifications. The CDF contributions of internal flooding events and seismic events are affected only marginally.

In addition to the reduction in plant CDF, the Stage 2 modification lead to beneficial shift in Release Categories (RC) frequency profiles, as demonstrated by Level 2 results. The RCs are grouped into "Very Small Releases" (RC1, RC2 and RC4), "Small Releases" (RC3A, RC3B, RC5A and RC5B) and "Large Releases" (RC6, RC7A, RC7B, RC8A and RC8B).

SG replacement significantly reduced RC frequencies relating to bypass failure modes (RC8A and RC8B) and, hence, "Large Releases". The "Wet Cavity" design lead to re-distribution among the "Small" and "Very Small" releases. The shift toward "Very Small" and "Small" releases achieved through modifications is highly desirable and contributes toward overall risk reduction at the plant site.

Summary and results of ISA analysis

Conducting of safety improvements is plant policy and a permanent process at the NEK. An integrated safety assessment was made of the impact of the modifications conducted from 1993 to the outage in 1999 (Stage 1) on plant safety, as well as those related to plant modernization, scheduled at 1999/2000 (Stage 2). Evaluation of results of Stage 1 and Stage 2 of integrated assessment of plant modifications clearly demonstrated that modifications lead to improvements in plant design and operation and thus contributed to the overall reduction of the plant risk.



CDF for Various Types of Initiating Events

4. RX VESSEL HEAD PARAMETERS EVALUATION

The Krško plant is currently categorised as a “Thot” plant, that is, the upper head fluid temperature is close to Thot. However, the upper internals package design at Krško is such that the field work perform a Tcold conversion is rather simple.

Recent plant operating experience with Nickel based alloys, i.e. ALLOY 600, indicates this material is in general susceptible to corrosion cracking when exposed to operating temperatures in excess of 500 deg. F (260 deg. C). ALLOY 600 has been used typically for pressure boundary components because of its thermal compatibility with carbon steel, superior resistance to chloride attack, and higher strength that the austenitic stainless steels. Time for crack initiation varies depending on the specific heat of material, operational temperatures, operational and residual stresses. Figure 1 shows a typical crack growth prediction for surface flaws in the vessel head penetrations for a range of temperatures.

The vessel head penetrations form an integral part of the reactor pressure boundary. penetration cracks and subsequent leakage provide a significant challenge to plant availability and personnel exposure limits.

In order to increase the margin against crack initiation and reduce the rate of crack propagation, one mitigative action is to reduce the bulk fluid temperature of the coolant in the reactor vessel head region. Figure 2 shows the various flow paths in the upper head region for a typical plant. The temperature on the fluid in the upper head region is determined by the relative contributions of the core exit flow which flows up through the various control rod guide tubes and the head cooling flow which is diverted directly from the reactor inlet flow. Thus, conceptually, the temperature of the fluid in the upper head region can vary between the core inlet temperature and the core exit temperature.

By increasing the fraction of reactor coolant that flows to the upper head, this temperature can be reduced. The minimum upper head fluid temperature that can be achieved is the core inlet temperature. For Krško, this capability can be achieved by removing the spray nozzle plugs, or part of them.

Conversion to upper head Tcold is also expected to provide a benefit for the large break LOCA PCT. This is due to the fact that:

- Delayed flashing of fluid in upper head improves blowdown heat transfer;
- Increased spray nozzle flow area improves steam venting during reflood. A higher flooding rate is thus expected.

APPROACH TO REGULATORY ASSESSMENT OF POWER UPRATES AND SAFETY MARGINS

W. van Doesburg, U. Schmocker
Swiss Federal Nuclear Safety Inspectorate,
Villigen-HSK, Switzerland

M. Khatib-Rahbar
Energy Research Inc.
Rockville, Maryland, United States of America

Abstract

This paper discusses both the deterministic and the probabilistic risk assessment (PSA) elements of the Swiss regulatory practices for the review and assessment of plant power uprate applications and implications on the relevant safety margins, based on the experience with the Leibstadt nuclear power plant. The deterministic element of the regulatory assessment process consists of the assessment of safety margins through a traditional (Standard Review Plan - type) transient and accident analysis, including fuel and safety system performance studies, review of technical specifications, and dose rate assessments. The PSA-based element of the regulatory assessment process consists of the assessment of risk impact of power uprate, including an analysis of the impact of the uncertainties associated with the relevant severe accident issues on risk, using plant-specific regulatory PSA models.

1. INTRODUCTION

1.1 BACKGROUND AND OBJECTIVE

The greater demand for electricity and the available capacity and safety margins in some of the operating nuclear power plants are prompting nuclear utilities to request license modification to enable operation at a higher power level, beyond the original license provisions.

In Switzerland, three utilities have requested and received regulatory authorization for power uprates. These include the Gösgen (a Siemens/KWU Pressurized Water Reactor [PWR] with a large dry containment), the Mühleberg (a General Electric [GE] Boiling Water Reactor [BWR]/Model 4 with Mark I containment) and Leibstadt (a GE BWR/Model 6 with MARK III containment).

The Gösgen plant was permitted to undergo a power uprate in 1985 from 2,808 MW(t) to 3,002 MW(t) (i.e., 6.9%). In 1992 the Mühleberg power plant also received the permission for a power uprate from 997 MW(t) to 1,097 MW(t) (i.e., 10%). On the other hand, the Leibstadt power plant twice requested and received permission to uprate. This included an uprate of 4.2% in 1985 from the original power level of 3,012 MW(t) to 3,138 MW(t); subsequently in 1998 the plant was permitted to uprate by an additional 14.7% from 3,138 MW(t) to the current power level of 3,600 MW(t) [1].

This paper discusses both the deterministic and probabilistic elements of the Swiss regulatory process for review and assessment of power uprates and safety margins based on the experience with the Leibstadt plant power uprate.

1.2 SPECIFIC ASPECTS OF THE LEIBSTADT NPP (KKL)

The standard BWR/6-238 was designed for a reactor power of about 3,600 MW(t), i.e., even after the recent power uprate KKL remains within the general design basis of the BWR/6. However, in comparison with the standard BWR/6 the KKL reactor core is smaller: it comprises 648 instead of 748 fuel assemblies. This implies a higher power density. In fact, after the recent power uprate, the power density of KKL is the highest among all BWRs (62.8 MW/m^3).

KKL differs further from the standard BWR/6 in some important aspects. First of all, a two train Special Emergency Heat Removal System (SEHR) was implemented per HSK requirement. Secondly, KKL was equipped with a filtered containment venting system. Thirdly, as a consequence of the Barsebäck event in 1992, in 1993 the suction capability of the ECCS strainers was increased considerably (i.e., from 6 x 2 m^2 to 6 x 15 m^2).

Therefore, a number of risk reducing measures had already been implemented prior to the 1998 power uprate.

2. ELEMENTS OF THE REGULATORY ASSESSMENT PROCESS

The Swiss Federal Nuclear Safety Inspectorate (HSK) has followed a regulatory safety assessment approach that includes the traditional deterministic safety analysis covering the range of normal operation to design basis accidents, and probabilistic safety assessment (PSA) for severe accidents [2].

Figure 1 shows the regulatory process for assessment of power uprate applications by various licensees. It consists of the traditional deterministic evaluation of licensing basis issues that are affected by power uprate, augmented by probabilistic analysis of various risk-related issues. In this approach, if the uprated plant cannot meet the basic deterministic requirements that are associated with the original license (e.g., defense-in-depth, redundancy/diversity of safety systems, etc.) the application for power uprate is expected to be denied. Also, safety margins that are expected to be reduced at the higher power level must still be sufficient to meet all existing safety limits and settings. Given the reduction in safety margins, in order for the application to be considered, the results of plant-specific risk analyses should also show that the expected increase in risk of severe accidents does not lead to unacceptably high risk levels and are not significant when compared with typical uncertainties in the risk of severe accidents for the same plant. Of course, the probabilistic assessment of safety margins considers systemic, human factor and success criteria issues as well as those relevant issues that can influence severe accident behavior, containment vulnerabilities and potential releases to the environment. Therefore, for the power uprate application to be granted by the authority, the uprated plant conditions must meet deterministic licensing requirements and also be acceptable from quantitative severe accident risk perspectives. A discussion of this two-tier approach follows.

2.1 Deterministic Assessment

The deterministic element of the regulatory assessment process consisted of the assessment of safety margins through a traditional (Standard Review Plan - type) transient and accident analysis, including fuel and safety system performance studies, review of technical specifications, and dose rate assessments.

For the 1985 power uprate, no additional safety analysis had to be made since all such analysis had initially been performed at 105% steam flow, which corresponds to a 4.2% power increase from the original 3012 MW(t) to 3138 MW(t). For the recent power uprate, which is described here, new safety analyses were mandatory in support of the change in operating license; in the following, these analyses will be described.

The 114.7 % power uprate request was the result of a feasibility analysis by the reactor supplier, performed on the basis that only changes on the secondary (turbine) side would be made.

2.1.1 Concerns for Deterministic Review

The following section lists the various regulatory concerns that were assessed as part of the deterministic review of normal, transient and design basis accident conditions for KKL power uprate [1].

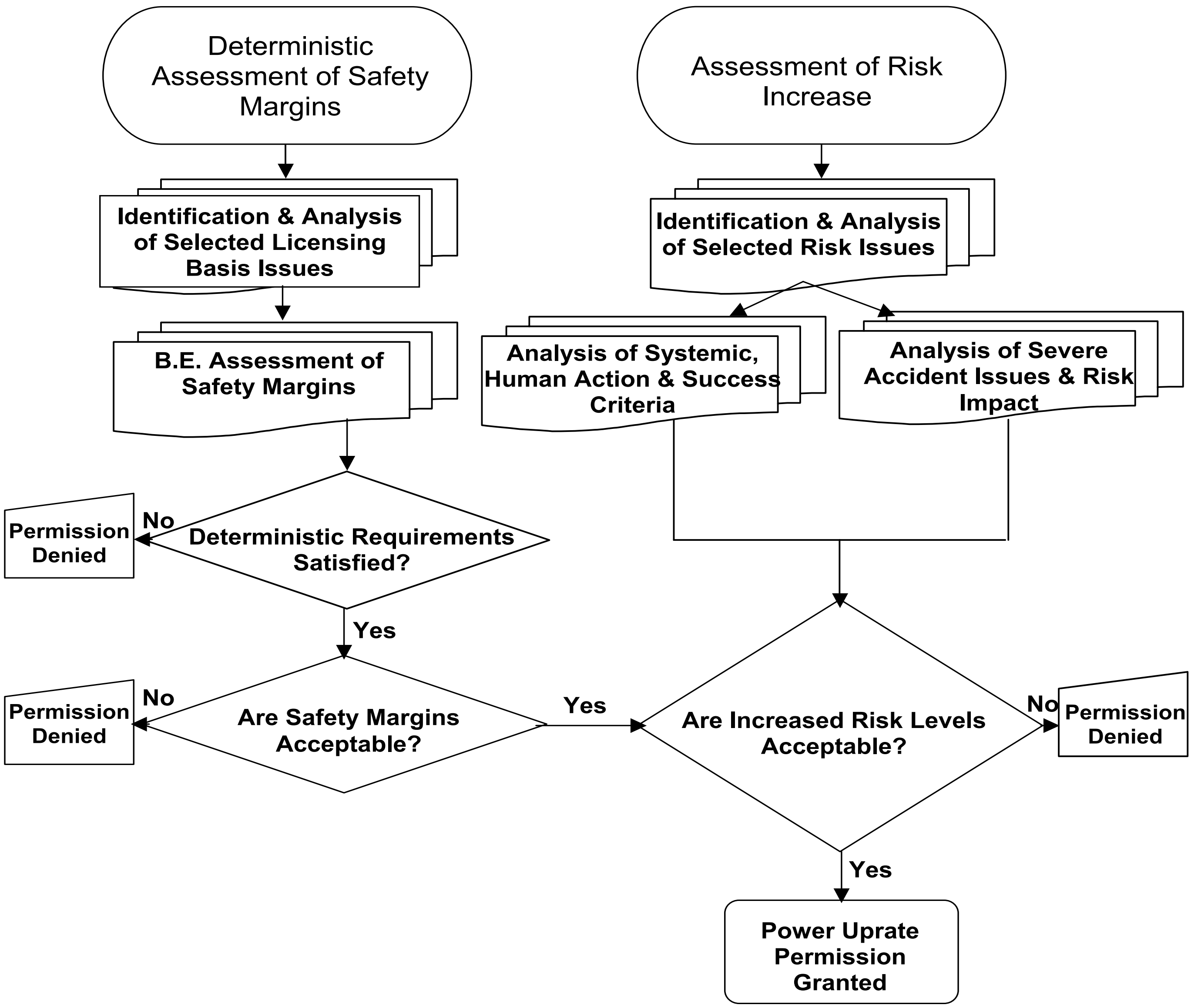


Figure 1 Flow chart depicting regulatory process for plant power uprate assessment

<u>Normal operation</u>

- Uprating the reactor power of a BWR leads to stronger boiling of the coolant. The higher void content in the core affects the reactivity feedback, which is important for the transient plant behavior.

- The recirculation flow characteristics remain unchanged in KKL. However, the feedwater and hence the steam mass flow rates had to be increased. This is not expected to lead to unacceptable vibrations and/or erosion-induced corrosion of the feedwater and the steam lines.

- Power uprate also implies a higher average neutron flux in the core. The fast neutron flux, which is responsible for irradiation embrittlement of the reactor vessel wall, is higher. This could affect the lifetime of the vessel. The issue is not critical for KKL due to the larger water gap between core and vessel wall and due to modern low neutron-leakage core loading patterns.

- The average power density in the core is not of direct safety relevance. The power uprate is achieved by adjusting (flattening) the radial power distribution in the core (by appropriate core loading): the number of fuel assemblies with higher power increases, but the maximum allowable power rating for a fuel assembly or a fuel rod remains unchanged. This proved to be possible with the usage of modern (10x10) fuel, which has a much lower power rating compared to the (8x8) fuel used previously.

- The neutronic noise during power operation was required not to lead to potential disturbances, e.g., unnecessary scrams.

Transients and design basis accidents

- The classical criteria for the fuel (linear heat rate, Critical Power Ratio [CPR], cladding oxidation, shutdown margin) and the containment (temperature and pressure after LOCA) must be fulfilled. The fuel related criteria could usually be fulfilled by a proper core loading.

- The core inventory of radioactive nuclides changes: the inventory of short-lived fission products is proportional to the reactor power whereas the inventory of nuclides with longer half-life is proportional to the fuel burnup of the core. A higher activity inventory results in a higher decay power after scram, which is important for decay heat removal following design basis accidents (DBAs). The higher activity inventory can also result in higher radiological source terms and doses to the public. This issue is explained in more detail in Section 2.2 below.

Beyond design basis accidents:

- Adequate mitigation of Anticipated Transients Without Scram (ATWS) was required to be demonstrated. Also, the risk due to ATWS should not increase significantly.

- The quantitative assessment of severe accidents was also required as part of the KKL probabilistic safety assessment (PSA). More details are provided in Section 2.2.2 below.

2.1.2 Review of Analyses

Deterministic analyses were reviewed on the following basis [1]:

- Safety limits/criteria that are defined for normal operation, transients (anticipated operational occurrences) and design basis accidents (DBA) is not changed. This means that the “boundary of what is considered safe” remains unchanged.

- There are no requirements for a minimum distance or margin to the safety limits/criteria.

The safety case submitted basically followed the existing Final Safety Analysis Report (FSAR), the scope of the actual transient/accident analysis being reduced to the limiting events. All analyses at the higher power level showed that the above-mentioned condition, that existing safety limits/criteria could be fulfilled. However, the effect from power uprate on the margins to these limits/criteria are difficult to assess because modern (realistic) calculational methods were used for the analyses at uprated power; thus, the change in margin is a combined effect of the higher power level and modern analysis methods. As an example, the Peak Cladding Temperature (PCT) values during LOCA generally decreased; here, the effect of modern analysis methods combined with the usage of modern fuels was larger than the effect of the power increase. A few examples of the deterministic analysis results follow.

Example: LOCA within Drywell

The well-known criteria of cladding temperature (<1204 °C) and the cladding reduction due to oxidation (<17 %) are maintained to specify the boundary of what is considered safe.

Due to the high capacity of the KKL Emergency Core Cooling Systems (ECCS) the above criteria are met with large margins even after power uprate. The maximum cladding temperature for the worst-case scenario, which includes single equipment failure as well as additional equipment out of service due to e.g., maintenance (N-2 criterion) is about 650 °C. This value was calculated with realistic methods and includes a bias for uncertainties based on the 10CFR50.46 Appendix K criteria. At such relatively low temperatures the oxidation criterion is also fulfilled with large margin.

The previous calculations at the power level of 3138 MW(t) resulted in a higher maximum cladding temperature of about 700 °C due to a more conservative model for the low pressure ECCS.

Example: ATWS

For KKL, the limiting ATWS includes the simultaneous closure of all main steam isolation valves (MSIV). Under such conditions (loss of main heat sink) the steam relief valves open and the reactor power must be removed by the residual heat removal (RHR) systems of the pressure suppression pool. The safety criterion is to maintain the heat removal capacity of this pool by limiting the water pool temperature to 85 °C, thereby limiting the pressure in the primary containment to a value below 2.03 bar.

In order to mitigate the consequences of this ATWS, plant operator actions are required, as early as possible, to reduce the reactor vessel water level to lower reactor power generation and then to initiate the boron injection system (Standby Liquid Control System [SLCS]) for shutting down the reactor.

As reactor power increases the time to perform the above mentioned actions decreases. Lowering the reactor water level and especially initiation of the boron injection system are crucial decisions of the operating personnel. Sufficient time needs to be provided for such decisions. The 3 minutes originally assumed in the safety analyses by the utility were considered to be too short.

HSK then proposed a plant modification to relax the requirements on the operators: an automatic feedwater pump runback on an ATWS signal was recommended to initiate lowering of the reactor water level. This provides additional time for the most critical decision to initiate the SLCS. The analyses showed that with this plant modification the plant personnel should have about 10 minutes to initiate the SLCS following an ATWS signal.

Example: Stability

Power uprate generally reduces the margin to instability. Therefore, several actions were taken to avoid the occurrence of instabilities:

- Extension of the exclusion regions in the power flow map to maintain the existing margin to instability.
- Manual selected rod insertion (SRI) to rapidly reduce power
- Automatic SRI in case of special fast transients with the potential of leading to instabilities.

The safety criterion is to limit the decay ratio (DR) to less than 1. In practice, due to uncertainties in evaluating the DR, the DR must be limited to below 0.8.

New exclusion regions were determined to maintain the margin to instability. This led to increasing the exclusion regions in the upper left corner of the power-flow map. The new exclusion regions were validated by actual stability measurements.

2.2 Probabilistic Safety Assessment

The probabilistic element of the of the regulatory assessment process consisted of the assessment of risk impact of power uprate, including an analysis of the impact of uncertainties associated with the relevant severe accident issues on the estimated risk, using a plant-specific regulatory PSA model [2].

2.2.1 Impact on Core Damage Frequency

The impact of the power uprate on core damage frequency (i.e., results of the level-1 PSA study) was not quantitatively assessed. The main level-1 PSA issues that could be impacted by power uprate include:

(1) The decay heat removal success criteria,
(2) The dynamic operator actions, and
(3) The reduced design safety margins for the important mitigating systems.

The effect of power level on decay heat removal “success criteria” was minimized by the HSK requirements of maintaining the pre-power uprate decay heat removal success criteria at the power uprate condition at Leibstadt.

It was recognized by HSK that there is some influence on the success probability of the dynamic human actions at the uprated power conditions, because one major factor that affects the probability of operator errors is the time available to respond to an event. However, a review of all important operator actions in all the regulatory PSA level-1 results showed that those actions would not be substantially affected by the expected reduction in the available operator response time, due to the increase in the reactor power level.

As it was already discussed in Section 2.1, the system design safety margin for the important mitigating systems, particularly High Pressure Core Spray (HPCS) and Reactor Core Isolation Cooling (RCIC) systems, provide sufficient margins that the increased power level would not affect the overall decay heat removal capabilities.

Therefore, it was concluded by HSK that no discernable impacts resulting from the power uprate on the internal events[1] mean core damage frequency (based on the 1995 plant conditions) of 4.4×10^{-6} per reactor year would be expected. The HSK calculated uncertainties in the estimated core damage frequency ranged from about 7×10^{-8} (5 percentile) to about 1.5×10^{-5} (95 percentile) per reactor year.

2.2.2 Impact on Progression of Severe Accidents

The impact of the reactor power uprate on the progression of severe accidents, release of fission products, and challenges to containment integrity as applicable to the Leibstadt nuclear power plant is described in this section. Table 1 lists the issues that are expected to be impacted by power uprate, including a qualitative ranking of their intrinsic uncertainties.

The assignment of low, medium and high ranks to various uncertainty issues is intended to guide the degree by which the impact of the power upgrade and fuel design changes can be characterized and quantitatively assessed. Specifically:

Low Uncertainties - The intrinsic uncertainties are small relative to the expected changes resulting from the reactor power level and fuel design. Therefore, the expected impact of the power and fuel modifications on the characterization of the issue can be quantified with confidence, as guided by relatively good knowledge of the governing physical phenomena associated with the issue.

Medium Uncertainties - The intrinsic uncertainties are not small relative to the expected changes resulting from the reactor power level and fuel design. Therefore, only trends

---

[1] At the time of the regulatory evaluation process, the external and area events PSA models were not completed; therefore, the probabilistic assessment focused on internal events only.

associated with the impact of the planned changes can be quantified with confidence, as guided by relatively incomplete knowledge of the governing physical phenomena associated with the issue.

High Uncertainties - The intrinsic uncertainties are large relative to the expected changes resulting from the reactor power level and the fuel design. Therefore, assessment of the expected trends of the impact of the changes on the characterization of the uncertain severe accident issue is difficult under all conditions of interest.

Table 1 Intrinsic uncertainties for the issues impacted by reactor power

Issues Impacted		Intrinsic Uncertainty
1. Core Radiological (Isotopic) Inventory		Medium
2. Decay Heat		Low
3. Time of Core Uncovery		Low
4. Core and Structural Heat up Rates		Low
5. Metal Oxidation/Hydrogen Generation		Medium
6. Fuel Damage and Melt Relocation		High
7. Time of Reactor Pressure Vessel Failure		High
8. Extent of Core-Concrete-Interaction and Non-condensable Gas Generation		Medium
9. In-Vessel Fission Product Release		High
10. In-Vessel Retention of Fission Products		High
11. Fission Product Retention in Pressure Suppression Pool		High
12. Ex-Vessel Fission Product Release		High
13. Fission Product Retention in Drywell and Wetwell Compartments		Medium
14. Time of Containment Failure/Containment Filtered Vent		Medium
15. Early Containment Loads	Combustion	Medium
	Direct Containment Heating	High
	Ex-Vessel Steam Explosions	High
16. Late Containment Loads	Combustion	Low
	Slow Pressurization	Low
	Basemat Penetration	Low

The qualitative analysis of various severe accidents issues demonstrated that the most significant impact of the power uprate can result from the increased radioactive inventory, and the time acceleration of events due to the increased decay heat level at the uprated power conditions. The issues of medium uncertainty were assessed and it was concluded that even though some trends could be established in terms of the influence of reactor power changes, nevertheless, the overall impact of power was not significant. The following discussion focuses on those aspects listed in Table 1 for which a significant impact due to power uprate is expected especially in comparing the expected impact relative to the intrinsic uncertainties.

Core Radiological Inventory

Table 2 lists the ORIGEN2-based core inventory for Leibstadt corresponding to the power level of 3,138 MW(t) and 3,600 MW(t). These inventories are listed for the ten representative source term groups used in the PSA. Note that for source term estimations, a ±30% uncertainty in ORIGEN2 predictions of isotopic inventories is typical.

The core inventory of short-lived radionuclides (which quickly reach their equilibrium concentration in the core) is proportional to the reactor power level. However, for the stable and long-lived nuclides, the core inventory is proportional to the power level and the burn-up (i.e., the length of time fuel remains in the core undergoing fission). The short-lived, fast decaying radionuclides dominate the total core activity in the relatively short time frame after reactor shutdown. In contrast, more than 95% of the fission product mass inventory is due to stable and long-lived nuclides. Table 2 shows that the change in the radioactive inventory of the Leibstadt core, considering all the fission products, is proportional to the total reactor power level.

Table 2 Mass and radioactive inventory as a function of core power level

Fission Product Group	Radioactive Inventory (Bq)		**Change (%)**
	3138 MW(t)	3600 MW(t)	
Xe	4.32×10^{19}	4.99×10^{19}	15.4
CsI	4.56×10^{19}	5.24×10^{19}	14.9
CsOH	4.57×10^{19}	5.30×10^{19}	15.9
Te	4.04×10^{19}	4.64×10^{19}	14.9
Sr	2.99×10^{19}	3.48×10^{19}	16.4
Ba	3.19×10^{19}	3.67×10^{19}	15.0
Ru	9.22×10^{19}	1.03×10^{20}	11.7
La	1.82×10^{20}	2.09×10^{20}	14.8
Ce	8.92×10^{19}	9.74×10^{19}	9.2
All FPs	1.97×10^{20}	2.26×10^{20}	14.7

Decay Heat

The decay heat in the reactor is a strong function of the reactor power history, and time after reactor shutdown. In the time frame of interest to severe accidents (i.e., up to 2-3 days after reactor shutdown), the decay heat is essentially governed by the decay of relatively short-lived radionuclides and is therefore, proportional to the reactor power level. The decay heat is expected to influence the progression of events in two ways:

(1) the increase in energy production rate following reactor shutdown due to 14.7% increase in reactor power, and

(2) faster progression of events resulting from the uprated power leads to a higher decay energy generation rate at a given time since reactor shutdown.

The increase in the decay heat is one of the most important factors affecting the risk of severe accidents at Leibstadt.

Assume that the decay heat is governed by the following simple polynomial:

$$Q_d(t) = a \cdot Q(0) \cdot t^{-b} \tag{1}$$

Where $Q_d(t)$ is the decay heat in MW(t), $Q(0)$ is the initial power level in MW(t), t is the time since reactor shutdown in seconds. For a typical power reactor, at 100 seconds after shutdown, a~0.2, and b~0.3 [2].

The time of core uncovery during severe accidents is related to the integral of decay heat generation rate over time since reactor shutdown, and can be derived as [2]:

$$\tau_{uc} = \left[(1-b)\frac{m_{taf} h_{fg}}{aQ(o)} + t_0^{(1-b)}\right]^{1/(1-b)} = \left[3.5\frac{m_{taf} h_{fg}}{Q(0)} + t_0^{0.7}\right]^{1.43} \tag{2}$$

Here m_{taf} is the mass of water above the top of active fuel that is required to be boiled-off to start core uncovery. h_{fg} is the latent heat of vaporization at the system pressure.

Equation (2) clearly demonstrates that the time to core uncovery is inversely proportional to reactor operating power. This inverse relationship is seen to be weakly non-linear (i.e., $\tau_{uc} \sim Q(0)^{-1.43}$). The time to core uncovery is shorter by about 20% at the 14.7% power uprate condition (assuming the water inventory is the same, even though the reactor water inventory is slightly smaller at the uprated power condition due to the increased void formation).

The effect of the decay heat is to shift the time for occurrence of major severe accident events in roughly the same proportion as the core uncovery periods. In the early phase (uncovery and heat up period), the decay heat dominates accident progression; however, the influence of decay heat becomes less pronounced as core temperatures approach 1800K, where chemical reaction energy generation due to rapid oxidation of Zr cladding and channel boxes by steam becomes increasingly dominant.

A similar approach can also be used to assess the impact of power uprate on core and structural heatup rate, and the time of containment over-pressurization failure, that show a similar power-dependence as shown by Equation (2). However, since the decay heat plays a much less impact after the start of metal oxidation (that governs core meltdown, fission product release, and subsequent progression of events), the overall impact of power on these processes is only due to the time acceleration of events as exemplified by Equation (2).

In general, fission product release and transport behavior are not directly impacted by reactor power, especially, considering the typically large uncertainties associated with these processes [2]. On the other hand, plant-specific severe accident analyses that were performed as part of the regulatory evaluation process [2] have shown an acceleration in late containment failure time that varies from about 15% to 25% as a result of the 14.7% power uprate at Leibstadt. This impact was seen to be significant in terms of the reduction in the fission product retention within the reactor containment with consequential impact on the overall risk of severe accidents at the uprated power level.

2.2.3 Impact on Risk

The level-2 PSA model included the quantitative impact of uprated power on “low” uncertainty issues; while, the model also included the quantitative impact of the trends associated with severe accident progression issues of “medium” uncertainty issues. However, the model assumed that the impact of power could not be quantified for those issues that are classified with high relative uncertainties (See Table 1).

The likelihood (and frequency) of containment failure, especially for those failure modes that contribute to large early releases were not found to be impacted by reactor power level, since these failure modes are predominately caused by systemic failures (e.g., containment bypass, containment isolation failures, etc.) and energetic events (e.g., direct heating, steam explosions, etc.) with medium and high uncertainties (see Table 1), which were not assessed to be impacted by the magnitude of the power uprate. Similarly, even though the time of late containment filter venting and/or containment failure (due to slow over-pressurization) is impacted by power level; nonetheless, the likelihood of containment failure was not considered affected by the reactor power level [2].

The results of the analysis show [2] that the accident source terms are not significantly affected by power level for very early and early releases; however, noticeable increase in the accident source terms were calculated for late containment failure modes at higher power level, which is due to the time acceleration effects discussed earlier.

The magnitude of the source term associated with the various release classes can be converted into “release activity”, where activity is defined as disintegration per second per gram or Becquerel per gram of a particular isotope as:

$$Activity[Bq/gr] = \lambda \bullet C = \frac{0.6931}{T_{1/2}} \bullet \frac{N}{A} \tag{3}$$

Where, λ is the radioactive decay constant (per second), C is the nuclide concentration, $T_{1/2}$ is the half-life $(\equiv \ln 2/\lambda = 0.6931/\lambda)$ in seconds, N is the Avagodro number (= 0.6025×10^{24}), A is the atomic weight in gram.

Since within the confines of a level-2 PSA, it is not possible to define an appropriate risk measure that reflects both the frequency of core damage, and the consequences. Therefore, as part of this evaluation, risk was defined as the product of the "release activity" and the release class frequency (i.e., activity per reactor year), integrated over all possible release classes.

Table 3 shows the comparison of risk of activity of release (excluding noble gases) for the 3,138 MW(t) power level and the current uprated power of 3,600 MW(t). It should be noted that noble gases decay in a relatively short timeframe, and their contribution to offsite consequences is not as significant (i.e., they can contribute only to the immersion and inhalation dose). It is seen that the mean risk of activity of release increases by about 30% due to the 14.7% increase in reactor power.

Table 3 Impact of power level on the estimated mean risk of activity of release

Power Level, MW (t)	Risk of release activity (Bq/yr)	Risk Increase Relative
3138	6.27×10^{11}	NA
3600	8.14×10^{11}	30%

The over proportionality in risk of release of aerosols is partly due to the linear dependence of core radiological inventory on reactor power level, and the non-linear (over proportionality) dependence of time acceleration of events and containment failure time on power. When combined, these result in higher quantities of radioactive fission products to the environment (i.e., lower retention of fission product aerosols in containment), with some minor contributions from decay of shorter lived fission products that contribute to the risk of activity of release.

Figure 2 shows that the uncertainties in the estimated risk of activity of release at the uprated power as compared with those at 3138 MW(t). It is seen that these uncertainties are comparable and much greater than the 30% increase in the mean risk as shown in Figure 3.2 and Table 3.

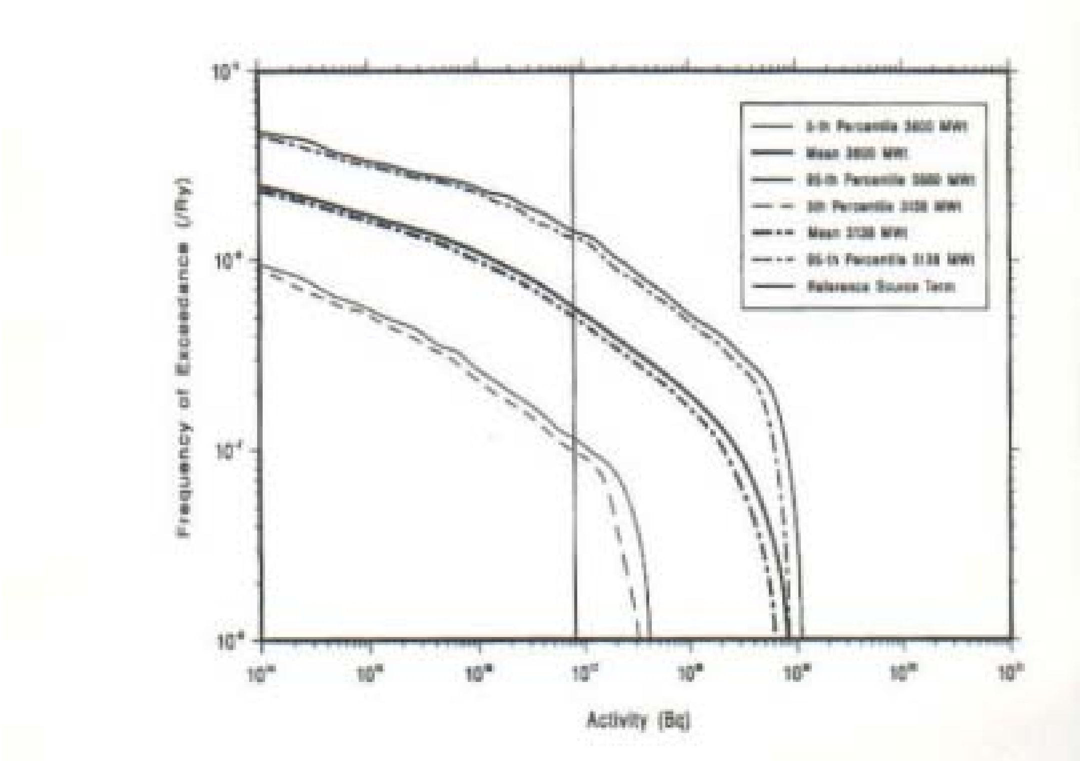


Figure 2 Uncertainties in the exceedance frequency of release activity for the uprated power conditions at Leibstadt

3. POWER UPRATE LICENSING AND IMPLEMENTATION

The first safety analyses were submitted by the utility to HSK by September 1992; more analyses, in particular the risk analyses, were submitted in subsequent years. The HSK completed the Safety Evaluation Review (SER) in March 1996; this constituted the technical basis for the federal administration to issue the new plant permit. However, this permit only materialized in 1998, due to the fact that in 1997 a severe fuel corrosion problem was discovered at KKL, caused by an unfavorable water chemistry situation (with Zn-injection). The permit was not granted until this fuel damage problem was satisfactorily resolved by appropriately changing the water chemistry settings.

Per HSK requirement, the power uprate was implemented in 4 discrete steps, each step corresponding to a higher power level. At each step/power level a test and monitoring program was required. The various tests at each step (including non-safety related testing) are shown in Table 4.

Table 4 Incremental power uprate process at KKL

System/Test	Test Goal	1996	1998	1999	2000	2002
		100%	106%	109%	112%	114.7%
Separator/Dryer	Performance				X	
Pressure controller	Performance, backup controller	X	X	X	X	X
Feedwater controller	Level control	X		X		X
Feedwater pump trip	No scram	X		X		
Feedwater runout	Maximum capacity	X	X			
Turbine control valves	no bypass	X		X		X
Level controller	Level control	X	X	X	X	X
Turbine trip	No scram			X		X
Load reject	No scram	X	X			
Recirculation control	Performance	X		X		
Trip of one recirculation pump	No scram	X				
Trip of both recirculation pumps	Partial scram, stability performance	X		X		
(In)stability	Verification of exclusion regions					X

Also, prior to the next step, satisfactory plant operational (including fuel) performance during at least 6 months was required by HSK.

In general, the process of power uprate implementation has been satisfactory; the testing was successfully completed, and no safety related operational problems were encountered during any of the uprate phases.

4. INSIGHTS AND CONCLUSIONS

The Leibstadt power uprate study demonstrated that:

- All existing deterministic safety criteria and limits could be met at the higher power level.

- The (conservatively established) margins to these safety criteria/limits remained acceptable; reductions, if at all, were minor compared with the power level increase.

- The risk of core damage (i.e., core damage frequency) was not assessed to be significant as compared with the uncertainties in the estimated core damage frequency.

- The time acceleration of events (e.g., core uncovery, start of damage, containment failure, etc.) can result in earlier and larger releases of radioisotopes to the environment. This shows a slightly over-proportional effect on risk relative to the magnitude of power uprate.

- The likelihood of containment failure was not assessed to be significantly impacted by the magnitude of power uprate. The same observation is also applicable to the frequency of large early releases.

The frequency and magnitude of large early releases was not impacted by the magnitude of the power uprate, as any impact due to 14.7% power uprate is expected to be masked by the much larger uncertainty in estimating the likelihood of early containment failure, and the magnitude of the resulting radiological releases.

Overall, the 14.7% increase in power was estimated to result in about 30% increase in risk of activity of release (used as a measure for the HSK regulatory assessment).

Given the decrease in risk due to the implementation of additional safety systems prior to the power uprate, and with the increase in the estimated risk from power uprate being much smaller than the estimated uncertainties in the risk estimates, the power uprate related risk increase was considered acceptable from a regulatory perspective.

4. REFERENCES

1. "GUTACHTEN zum Gesuch des Kernkraftwerks Leibstadt um Leistungserhöhung auf 3600 MW_{th}," HSK 12/420, KSA 12/210, Swiss Federal Nuclear Safety Inspectorate, Würenlingen, März 1996.

2. M. Khatib-Rahbar, et al, "Regulatory Evaluation of Leibstadt Probabilistic Safety Assessment," Energy Research, Inc. Report Prepared for the Swiss Federal Nuclear Safety Inspectorate, ERI/HSK 95-303, HSK 12/144, Volumes I & II, 1996.

CONTRIBUTORS TO DRAFTING AND REVIEW

Auh, G. -S.	Korea Institute of Nuclear Safety –KINS, Republic of Korea
Bajs, T.	University of Zagreb, Croatia
Ban, Ch.-H.	Korea Nuclear Fuel Company, Republic of Korea
Bazso, Z.	Nuclear Regulatory Authority (UJD SR), Slovakia
Borak, J.	NPP Bohunice, Slovakia
Charlier, A.	BELGATOM, Belgium
D'Auria, F.	University of Pisa, Italy
Dusic, M.	International Atomic Energy Agency
Eltawila, F.	US Nuclear Regulatory Commission (NRC), United States of America
Elter, J.	Paks NPP, Hungary
Furuta, T.	Toyo University, Japan
Gacs, A.	KFKI Atomic Energy Research Institute, Hungary
Garis, N.	Swedish Nuclear Power Inspectorate (SKI), Sweden
Ghias, B.	Karachi Nuclear Power Complex, Pakistan
Glaeser, H	Gesellschaft fuer Anlagen- und Reaktorsicherheit (GRS) mbH, Germany
Goeggelmann, H.	Framatome ANP GmbH, Germany
Hajra, P.	Atomic Energy Regulatory Board, India
Husarcek, J.	Nuclear Regulatory Authority (UJD SR), Slovakia
Janezic, A.	Slovenian Nuclear Safety Administration, Slovenia
Janosi, T.	Hungarian Atomic Energy Authority, Hungary
Jansen, R.	Ministry of Housing, Spatial Planning and the Environment, Netherlands
Kaliatka, A.	Lithuanian Energy Institute, Lithuania
Kasahara, F.	Japan Nuclear Energy Safety Organization, Japan

Khatib-Rahbar, M.	Energy Research Inc., United States of America
Korotaev, K.	OKB Gidropress, Russian Federation
Krajmer, I.	NPP Bohunice, Slovakia
Lee, S.	International Atomic Energy Agency
Makihara, Y.	International Atomic Energy Agency
Maprelian, E.	Instituto de Pesquisas Energeticas e Nucleares - IPEN, Brazil
Narang, B.B.	Nuclear Power Corporation of India, India
Nemes, I.	Paks NPP, Hungary
Ortwein, J.	International Atomic Energy Agency
Royen, J.	OECD / NEA, France
Sandervag, O.	Swedish Nuclear Power Inspectorate (SKI), Sweden
Serra, O. H	Comisión Nacional de Energía Atómica, Argentina
Sommer, F.	EON Kernkraft GmbH, Germany
Spiler, J.	Krsko NPP, Slovenia
Uhlir, M.	NPP Dukovany, Czech Republic
Usmani, S.A.	Atomic Energy of Canada Limited (AECL), Canada
Vela Buendía, E.	Consejo de Seguridad Nuclear –CSN, Spain
Versteeg, J.	International Atomic Energy Agency
Viktorov, A.	Canadian Nuclear Safety Commission, Canada
Vincke, M.	Association Vincotte Nucleaire, Belgium
Voswinkel, A.	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS)mbH, Germany
Wendenkampf, E.	Framatome ANP GmbH, Germany