UNIVERSITÀ DI PISA

Facoltà di Ingegneria

Corso di Dottorato di Ricerca in
SICUREZZA NUCLEARE E INDUSTRIALE

Tesi di Dottorato di Ricerca

FRAMEWORK AND STRATEGIES FOR THE INTRODUCTION OF BEST ESTIMATE MODELS INTO THE LICENSING PROCESS

Calogero Sollima

Anno 2008

UNIVERSITÀ DI PISA

Facoltà di Ingegneria

Corso di Dottorato di Ricerca in
SICUREZZA NUCLEARE E INDUSTRIALE

Tesi di Dottorato di Ricerca

FRAMEWORK AND STRATEGIES FOR THE INTRODUCTION OF BEST ESTIMATE MODELS INTO THE LICENSING PROCESS

Autore:

Calogero Sollima ____________________

Relatori:

Prof. Francesco D'Auria ____________________

Ing. Gianni Petrangeli ____________________

Dr. Jozef Misak ____________________

Anno 2008

SOMMARIO

Gli attuali sforzi, per assicurare che le operazioni degli impianti nucleari di potenza si attuino in modo stabile, sicuro e competitivo, procedono di pari passo con gli sviluppi fatti nel campo dell'analisi degli incidenti dove l'analisi di sicurezza di tipo deterministico assume un ruolo rilevante nel confermare la corrispondenza e l'efficienza delle disposizioni in materia di sicurezza degli stessi impianti.
Le conoscenze tecniche sviluppate di recente offrono due opzioni per dimostrare che il livello di sicurezza è garantito da adeguati margini di sicurezza: l'utilizzo di codici di calcolo di tipo realistico, o "Best Estimate" (BE), associati a input o di tipo conservativo o di tipo realistico. In quest'ultimo caso si richiede la valutazione dell'incertezza dei risultati ottenuti.
Si prevede un maggiore utilizzo dell'analisi di sicurezza di tipo BE, nelle attività di progettazione e di licensing, sia per impianti di nuova realizzazione che per quelli esistenti per le attività di revisione periodiche di sicurezza, riclassificazione di sicurezza, aumento di potenza ed estensione di vita. Gli strumenti di analisi BE sono già ampiamente utilizzati dagli enti di controllo, dagli istituti di ricerca e in molti casi dall'industria, dai gestori e dai progettisti.
L'agenzia internazionale per l'energia atomica (IAEA), attraverso le pubblicazioni Safety Standards, raccomanda, come una delle opzioni per dimostrare l'adeguatezza dei margini di sicurezza, l'utilizzo dei codici BE con input realistico associato alla valutazione dell'incertezza dei risultati ottenuti.
L'obiettivo, che si propone il presente lavoro, è quello di contribuire allo sviluppo dell'applicazione dei metodi BE con valutazione dell'incertezza nei processi di licensing.
In questa tesi, maggiore enfasi è data allo studio degli "input e metodi" dei calcoli BE più che alla valutazione dell'incertezza, che è lo scopo principale di altre attività parallele di dottorato con le quali sono state mantenute le relazioni.
L'attività è stata sviluppata in due fasi. Inizialmente lo studio è stato focalizzato sull'approfondimento e analisi dei seguenti argomenti:
a) importanza e ruolo del licensing per la sicurezza degli impianti nucleari di potenza e, viceversa, dell'analisi di sicurezza, in particolare quella deterministica, e i margini di sicurezza per il processo di licensing;
b) il processo di licensing secondo gli standard IAEA;
c) lo stato dell'arte dell'utilizzo dei metodi BE nei processi di licensing in alcuni paesi.
L'attenzione è stata successivamente spostata sull'analisi della guida americana RG-1.157 e sono stati sviluppati alcuni temi non totalmente coperti dalle attività di ricerca attive in campo termoidraulico. In particolare mediante l' utilizzo dei codici RELAP5/MOD3 e Monte Carlo MNCP5, è stato analizzato l'influenza del calore di decadimento, della distribuzione dei beta e gamma di decadimento e l'analisi della modellizzazione del cross flow. Infine, le altre parti della RG-1.157 sono state approfondite mediante i risulti ottenuti in campo internazionale dalla ricerca scientifica e dalle relative applicazioni.
Le considerazioni fatte nell'analisi della guida americana sono da riferirsi principalmente ai reattori in pressione piuttosto che apparati sperimentali.
Vengono infine riportati i risultati conseguiti che sono stati condivisi e valutati anche da alcuni colleghi in ambito internazionale.

ABSTRACT

The current efforts to assure stable, safe and competitive operation of nuclear power plants go together with advances made in accident analysis domain where the deterministic safety analysis is an important instrument for confirming the adequacy and efficiency of provisions for the safety of nuclear power plants.
Recently made advances offer two acceptable options for demonstrating that the safety is ensured with sufficient margin: use of best estimate (BE) computer codes either combined with conservative input data or with realistic input data but associated with evaluation of uncertainty of results.
Main use of the BE of safety analysis is expected in applications, for design and licensing purposes, both for new reactor projects as well as for periodic safety reviews, safety up-grading, power up-rating and lifetime extension of existing nuclear power plants. The BE tools are already widely used by the regulatory organizations, research institutes and in many cases by the industry, the utilities and the vendors.
IAEA Safety Standards recommend, as one of the options for demonstration of sufficient safety margins, the use of best estimate computer codes with realistic input data in combination with evaluation of uncertainties of the calculation results.
The objective, proposed for the present work, is the implementation of the best estimate and uncertainty (BEPU) method into the licensing process.
In this thesis, more emphasis is given to the study of "input and method" of BE calculations than on "uncertainty evaluation", which is the main subject of a companion parallel doctorate work with which active contacts have been entertained.
Firstly the study is focused on: a) the importance and role of licensing for safety of nuclear power plants and of safety analysis, in particular deterministic safety analysis and safety margins for licensing process; b) the licensing process according to the IAEA standards; c) an overview of existing applications of best estimate methods in licensing practices in selected countries.
Later the work analyzes the American regulatory guide 1.157 and develops some issues not totally covered by research activities mainly in the thermal-hydraulic field. In particular, the results related to the decay heat, the gamma decay heat distribution and the core cross flow model are presented. This activity was implemented using of the RELAP5/MOD3 and Monte Carlo MNCP5 codes. The other parts of the RG-1.157 have been analyzed through the international scientific results and their applications. The considerations on the American RG have been addressed mainly to the pressurized water reactors more than to the simulation of the integral test facilities.
The results reported have been shared with colleagues from other countries and institutions.

CONTENTS

LIST OF SYMBOLS

AA	AVERAGE AMPLITUDE
ACRS	ADVISORY COMMITTEE ON REACTOR SAFEGUARD
AEC	ATOMIC ENERGY COMMISSION
AEA	UNITED KINGDOM ATOMIC ENERGY AUTHORITY
AECL	ATOMIC ENERGY OF CANADA LIMITED
AEAW	ATOMIC ENERGY AUTHORITY WINFRITH
ALARA	AS LOW AS REASONABLY ACHIEVABLE
ANS	AMERICAN NUCLEAR SOCIETY
AOO	ANTICIPATE OPERATIONAL OCCURRENCE
AP	ADVANCED PASSIVE
ASN	AUTORITÉ DE SÛRITÉ NUCLÉAIRE
ASAP	ADJOINT SENSITIVITY-ANALYSIS PROCEDURE
ASTRUM	AUTOMATED STATISTICAL TREATMENT OF UNCERTAINTY METHOD
ATG	ATOMIC ENERGY ACT
BE	BEST ESTIMATE
BEAU	BEST ESTIMATE ANALYSIS AND UNCERTAINTY
BEMUSE	BEST ESTIMATE METHODS UNCERTAINTY AND SENSITIVITY EVALUATION
BEPU	BEST ESTIMATE PLUS UNCERTAINTY
BIC	BOUNDARY INITIAL CONDITIONS
BMU	BUNDESMINISTERIUMS FÜR UMWELT, NATURSCHUTZ UND REAKTORSICHERHEIT (FEDERAL MINISTRY FOR THE ENVIRONMENT NATURE CONSERVATION AND NUCLEAR SAFETY)
BMWA	GERMAN MINISTRY FOR ECONOMY AND LABOR
CNEN	COMISSÃO NACIONAL DE ENERGIA NUCLEAR
CATHARE	CODE FOR ANALYSIS OF THERMAL HYDRAULICS DURING ACCIDENT AND FOR REACTOR SAFETY EVALUATION
CCFL	COUNTER CURRENT FLOW LIMITING
CDF	CORE DAMAGE FREQUENCY
CEA	COMMISSARIAT A L'ENERGIE ATOMIQUE
CFR	CODE OF FEDERAL REGULATIONS
CHF	CRITICAL HEAT FLUX
CIAU	CODE WITH CAPABILITY OF INTERNAL ASSESSMENT OF UNCERTAINTY
CNSC	CANADIAN NUCLEAR SAFETY COMMISSION
CPS	CONTROL PROTECTION SYSTEM
CSAU	CODE SCALING, APPLICABILITY AND UNCERTAINTY
CSNI	COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS
DAA	DATA ADJUSTMENT AND ASSIMILATION
DH	DECAY HEAT
DIMNP	DIPARTIMENTO DI INGEGNERIA MECCANICA, NUCLEARE E DELLA PRODUZIONE
DRM	DETERMINISTIC REALISTIC METHOD
ECCS	EMERGENCY CORE COOLING SYSTEMS
ENUSA	EMPRESA NACIONAL DEL URANIO, SA

EUR̃	EUROPEAN UTILITY REQUIREMENTS
EOP	EMERGENCY OPERATING PROCEDURE
EPR	EUROPEAN PRESSURIZED REACTOR
FC	FUEL CHANNEL (IN RBMK NPP)
FFT	FAST FOURIER TRANSFORM
FFTBM	FAST FOURIER TRANSFORM BASED METHOD
FSF	FUNDAMENTAL SAFETY FUNCTIONS
FSAR	FINAL SAFETY ANALYSIS REPORT
GASAP	GLOBAL ADJOINT SENSITIVITY-ANALYSIS PROCEDURE
GDH	GROUP DISTRIBUTION HEADER
GRS	GESELLSCHAFT FÜR ANLAGEN UND REAKTORISICHERHEIT
GRNSPG	NUCLEAR RESEARCH GROUP SAN PIERO A GRADO
GSUAM	GENERIC STATISTICAL UNCERTAINTY ANALYSIS METHOD
HAEA	HUNGARIAN ATOMIC ENERGY AUTHORITY
IAEA	INTERNATIONAL ATOMIC ENERGY AGENCY
IPSN	INSTITUT DE PROTECTION ET DE SÛRITÉ NUCLÉAIRE
IRSN	INSTITUT DE RADIOPROTECTION ET DE SÛRITÉ NUCLÉAIRE
ISP	INTERNATIONAL STANDARD PROBLEM
ITF	INTEGRAL TEST FACILITY
KAERI	KOREA ATOMIC ENERGY RESEARCH INSTITUTE
LERF	LARGE EARLY RELEASE FREQUENCY
LHGR	LINEAR HEAT GENERATION RATE
LOCA	LOSS OF COOLANT ACCIDENT
LOFT	LOSS OF FLUID TEST
LWR	LIGHT WATER REACTOR
MCC	MAIN CIRCULATION CIRCUIT
MCP	MAIN CIRCULATION PUMP
NAÜ	ORSZÁGOS ATOMENERGIA HIVATAL (HUNGARIAN ATOMIC ENERGY AUTHORITY)
NEA	NUCLEAR ENERGY AGENCY
NPP	NUCLEAR POWER PLANT
NRC	NUCLEAR REGULATORY COMMISSION
OECD	ORGANIZATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT
PCSAR	PRE-CONSTRUCTION SAR
PCT	PEAK CLAD TEMPERATURE
PDF	PROBABILITY DISTRIBUTION FUNCTION
PIE	POSTULATED INITIATING EVENT
PIRT	PHENOMENA IDENTIFICATION AND RANKING TABLE
POSAR	PRE-OPERATION SAR
PRA	PROBABILISTIC RISK ANALYSIS/ASSESSMENT
PSA	PROBABILISTIC SAFETY ASSESSMENT
REF	REFERENCE
ŘEŽ	NUCLEAR RESEARCH INSTITUTE PLC
RG	REGULATORY GUIDE
RLBLOCA	REALISTIC LB-LOCA
RSK	GERMAN REACTOR SAFETY COMMISSION
RTA	RELEVANT THERMAL-HYDRAULIC ASPECTS
s	SECOND

SAR	**SAFETY ANALYSIS REPORT**
SB-LOCA	**SMALL BREAK LOCA**
SBO	**STATION BLACK OUT**
SKI	**SWEDISH NUCLEAR POWER INSPECTORATE**
SARA	**SATATIO SAR**
SÚJB	**STÁTNÍ ÚŘAD PRO JADERNOU BEZPEČNOST (STATE OFFICE FOR NUCLEAR SAFETY)**
SUSA	**SYSTEM FOR UNCERTAINTY AND SENSITIVITY ANALYSIS**
TC	**TECHNICAL CO-OPERATION**
UMS	**UNCERTAINTY METHODS STUDY**
USA	**UNITED STATES OF AMERICA**
WENRA	**WESTERN EUROPEAN NUCLEAR REGULATORS' ASSOCIATION**
WF	**WEIGHTED FUNCTION**
γ	**GAMMA RAYS**

LIST OF FIGURES

LIST OF TABLES

ACKNOLEDGEMENTS

The present work is the product of a long term activity started by the professors and researchers of the Department of Nuclear, Mechanical and Production Engineering (DIMNP) of the University of Pisa and further developed within the Nuclear Research Group of San Piero a Grado (GRNSPG) in cooperation with international scientists, research institutions, industries and international organizations.
A warm acknowledgement goes to all the persons and friends, here not directly mentioned, which introduced me in such technological area. In particular it is well appreciated the help and friendship of:

- *W. Giannotti, G. Galassi and M. Cherubini, from the GRNSPG, and N. Forgione, from DIMNP, in introducing me in the thermal-hydraulic codes;*
- *V. Giusti and R. Ciolini, from the DIMNP, in supporting me in the application of the Monte Carlo code and gamma analyses;*
- , M. Bykov, A. Moskalev and A. S. Zubchenko (Gidropress, Russia), J. C. Ferreri (Regulatory body, Argentina), H. Glaeser (GRS, Germany), E. Hicken (*Forschungszentrum Jülich, Germany,)* A. Madonna (ITER Consult, Italia), B. Mavko and A. Prosek (Jozef Stefan Institute, Slovenia), R. Uddin (University of Illinois Urbana – Champaign, USA), E. Uspuras and A. Kaliatka (Lithuanian Energy Institute, Lithuania) to have reviewed all my work and supported me in this experience;
- G. Pilone, as friend, has shared this experience with me.

Finally, a special acknowledgement is for my parents, brothers, sister and sister in law for the family support.

1. INTRODUCTION

Safety is an essential part for the peaceful use of nuclear power generation.
The safety of nuclear power plants is based on the concept of defence in depth [1], Fig. 1, which indicates successive physical barriers (fuel matrix, cladding, primary system pressure boundary and containment) and other provisions to control the release of the radioactive material and on multiple levels of protection against damage to these barriers and against undue radiological impact on the plant itself and its surroundings (level 5, related to emergency plan, has been omitted). It should be noted that in Fig. 1 the events can also not move from a protection level to the next protection level, e.g. a break of the primary coolant pipe (protection level 3) may happen without any abnormal operation (level 2) before.

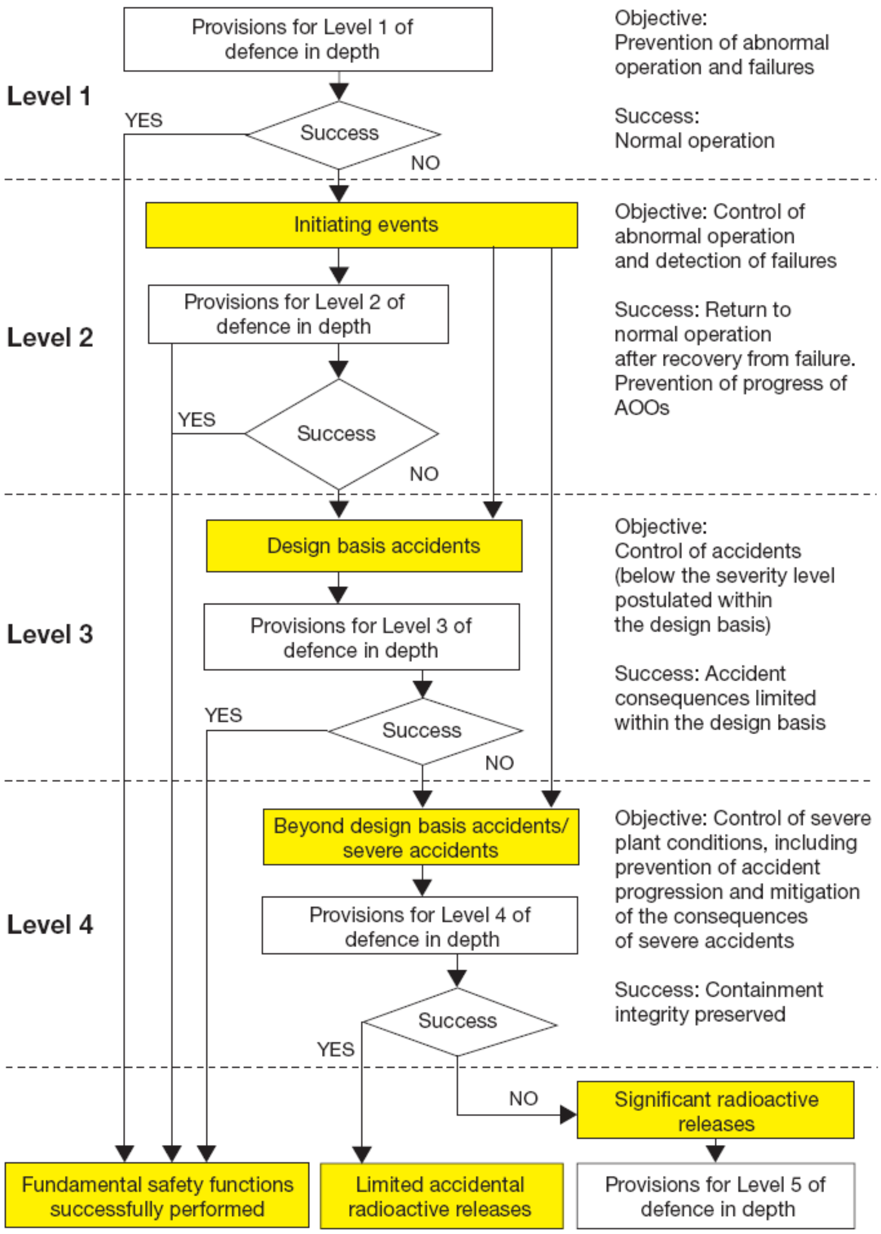


Fig. 1 – Flow chart for defence in depth [1]

The evaluation of plant safety is transformed in a set of acceptance criteria (such as maximum peak cladding temperature, maximum pressure in the primary system, etc.) to be met under a wide range of plant operating conditions to confirm the preservation of the physical barriers. The acceptance criteria are normally set forth by the national regulator. Their fulfilment is demonstrated by the results of safety analysis documented in the safety analysis report (SAR). To thoroughly evaluate the plant response to a number of unintended events such as operating errors and equipment failures (referred to as postulated initiating events) the use of computer simulation is essential.
The safety analysis is performed to the detail which is determined by the actual knowledge of physical phenomena expected to occur during the accident, by the computer power and computer code availability and experimental and analytical database. In other words the safety analysis has been an evolving process started from simple simulations with simplified models up to today's complex calculations using advanced computer codes.
In the case of licensing calculations, any insufficiency in safety analysis has to be compensated with additional assumptions. Impact of these assumptions is expected to be conservative, i.e. the results of the analysis are supposed to be worse than real plant response.
The safety analysis tools are broadly used within the framework of the design of new plants and operation of existing plants, including licensing of new NPP projects, safety upgrading programmes of existing NPPs, periodic safety reviews, renewal of operational licences, use of the safety margins for reactor power up-rating, better utilization of nuclear fuel and higher operational flexibility, for justification of lifetime extensions, development of new emergency operating procedures, analysis of operational events, and development of accident management programmes. Significantly increased capacities of new computation technology made it possible to switch over to the new generation of computer codes, with the use of best estimate codes with treatment of uncertainties, and coupling of computer codes.
The procedure to perform the evaluation uncertainty of the best estimate simulation is referred to as best estimate plus uncertainty (BEPU) methodology. Several uncertainty methods based on the concept of BEPU methodology have been developed, implemented and tested and are now in the phase of practical evaluation by the professional community. Recent conclusions state that qualified uncertainty methods are mature for practical applications [2] and [3].
The present document deals with the framework and the strategies for the introduction of BE models into the licensing process.

1.1. Scope

IAEA Safety Standards recommend, as one of the options for demonstration of sufficient safety margins, the use of best estimate computer codes with realistic input data in combination with evaluation of uncertainties of the calculation results. Best estimate analysis with evaluation of uncertainties is the only way for quantification of the existing safety margins. Its broader use in the future is therefore envisaged, even though it is not always feasible because of the difficulty of quantifying code uncertainties with sufficiently narrow ranges for every phenomenon and for each accident sequence. Since evaluation of uncertainties is a very complex issue, further work in this area is very much needed, in particular towards facilitating broader use of fully best estimate analysis with main response to the spread of the concepts "fixed" in a number of research centers and institutes. As the data and analytical methods quality improves, their application for the licensing calculations becomes eligible. Therefore the regulator has to be capable to expertly review such calculations and eventually even perform independently the peer-review calculations. The profound knowledge and experience with the methodology gives also the regulator the possibility to guide the technical community on its consistent application.

1.2. Objectives

The objective of the present work is the implementation of the best estimate and uncertainty methods into the licensing process to support a development of the regulatory guides for the application of the BEPU methodology in the licensing calculations.

There are several BE computer codes [34] that are used for the assessment of the safety analysis. They can be grouped in: thermal-hydraulics, structural mechanics and neutronics. Hereafter the thermal-hydraulic ones are used to analyze some thermal-hydraulic transients.

In this thesis, more emphasis is given to the study of “input and method” of BE calculations, more than on “uncertainty evaluation”, which is the main subject of a companion, parallel doctorate work, with which active contacts have been entertained.

The activity has been developed in two parts.

Firstly the study is focused on:

- the importance and role of licensing for safety of nuclear power plants and of safety analysis, in particular deterministic safety analysis and safety margins for licensing process;
- the licensing process according to the IAEA standards;
- an overview of existing applications of best estimate methods in licensing practices in selected countries.

Later the work aims to offer an up-to-date discussion basis for the identification of a standard BE procedure to be used with a specific, widely used code (e.g. RELAP5/MOD3), having in mind the objective of a consensus by licensing bodies on a “set to be defined” of safety (uncertainty) margins to be agreed upon for calculations performed using that procedure.

As base on implementing such considerations, the American regulatory guide RG-1.157 is analyzed and some issues, not totally covered by research activities, are developed. In particular, the considerations related to the decay heat, the gamma decay heat distribution and the core cross flow model are supported by calculations using the RELAP5/MOD3 and Monte Carlo MNCP5 codes applied mainly to a generic PWR 1000 ands as preliminary investigation to the LOFT test facility. Therefore, the considerations which follow are not addressed to the simulation of integral test facilities.

The results reported have been shared with colleagues from other countries and institutions.

1.3. Structure

The structure of the report is developed in chapters, that constitute the aim of the report, supplemented by annexes to provide additional information on some related subjects which are over the scope and objective of the thesis. The content of the document is the following:

1) Chapter 2 provides a summary description of the licensing process and of the its current practices.
2) Chapter 3 deals with an up-to-date summary description of the applications of the BE methods in licensing practice either presenting the IAEA standards and giving an overview of the applications in selected countries, i.e. USA, French, Germany, Hungary, Lithuania and Sweden.
3) Chapter 4 is the core of the report. It contains the discussion basis for the identification of a standard BE procedure. It analyzes the RG-1.157 either considering the results of the international researches and developing some calculations on subject like decay heat and cross flow modelling contribution to the PCT.
4) Chapter 5 provides with a discussion on the results achieved.
5) Chapter 6 summaries the results of the thesis.
6) Annex 1 gives an overview of the BEPU methods.
7) Annex 2 gives an example of IAEA assignment of safety principles to individual levels of defence in depth.
8) Annex 3 presents some applications of BE methods in licensing in some countries.
9) Annex 4 deals with reporting some additional results on the calculations performed on the decay heat, gamma and beta distribution.
10) Annex 5 reports the comments received by some colleagues on a paper presented in ICONE 16 related to a part of the work implemented in the thesis.
11) Annex 6 gives a list of publications of the author.

2. LICENSING PROCESS

The utilization of nuclear energy started during the II World War.
For a while its use was strongly restricted to military purposes. Since 1954, the USA Congress passed new legislation that for the first time permitted the wide use of atomic energy for peaceful purposes. The measure directed the AEC "to encourage widespread participation in the development and utilization of atomic energy for peaceful purposes" [30]. At the same time, it instructed the agency to prepare regulations that would protect public health and safety from radiation hazards.
The USA was the main country to develop rules to manage the use of the atomic energy for peaceful applications. They are collected in the 10 CFR 50 [11].
In the 60-ies many countries, dealing with the development of the Nuclear Power Plants (NPPs), implemented their own regulations looking at the NRC licensing process.
In addition to the regulatory bodies, also the nuclear organizations, namely IAEA (International Atomic Energy Agency) and OECD-NEA (Organization for Economic Co-operation and Development – Nuclear Energy Agency), were founded to develop the international cooperation among its member countries nuclear energy for peaceful purposes, "as well as to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy" [40].

2.1. Importance and role of the licensing for safety of nuclear power plants

The licensing is the process that guides the life of the NPP from the conceptual design, usually starting from the site definition, up to its decommissioning. It aims to demonstrate the capability of safety systems to maintain fundamental safety functions.
The objectives of the nuclear safety consist on to ensure conditions of localization and of plant such to satisfy the confident protection principles internationally accepted [32], e.g. the IAEA objectives [31]:

– "General Nuclear Safety Objective": *to protect individuals, society and the environment from harm by establishing and maintaining in nuclear installations effective defences against radiological hazards.* It is supported by two complementary "Safety Objectives" dealing with radiation protection and technical aspects. They are interdependent: the technical aspects in conjunction with administrative and procedural measures ensure defence against hazards due to ionizing radiation:
    - *a)* "Radiation Protection Objective: *to ensure that in all operational states radiation exposure within the installation or due to any planned release of radioactive material from the installation is kept below prescribed limits and as low as reasonably achievable* (ALARA), *and to ensure mitigation of the radiological consequences of any accidents*;
    - *b)* "Technical Safety Objective: *to take all reasonably practicable measures to prevent accidents in nuclear installations and to mitigate their consequences should they occur; to ensure with a high level of confidence that, for all possible accidents taken into account in the*

design of the installation, including those of very low probability, any radiological consequences would be minor and below prescribed limits; and to ensure that the likelihood of accidents with serious radiological consequences is extremely low.

2.1.1. Safety assessment

The licensing process starts with the definition of the safety requirements by the national and political institutions. Considering also the engineering factors relevant for safety, they include general aspects, related to the radiation protection, external and internal hazard, and possible specific systems. It should be clarified that a regulatory body does not justify the introduction of a practice.
The safety requirements provide the input to develop the guidelines for the safety analysis and to perform the safety assessment, Fig. 2.

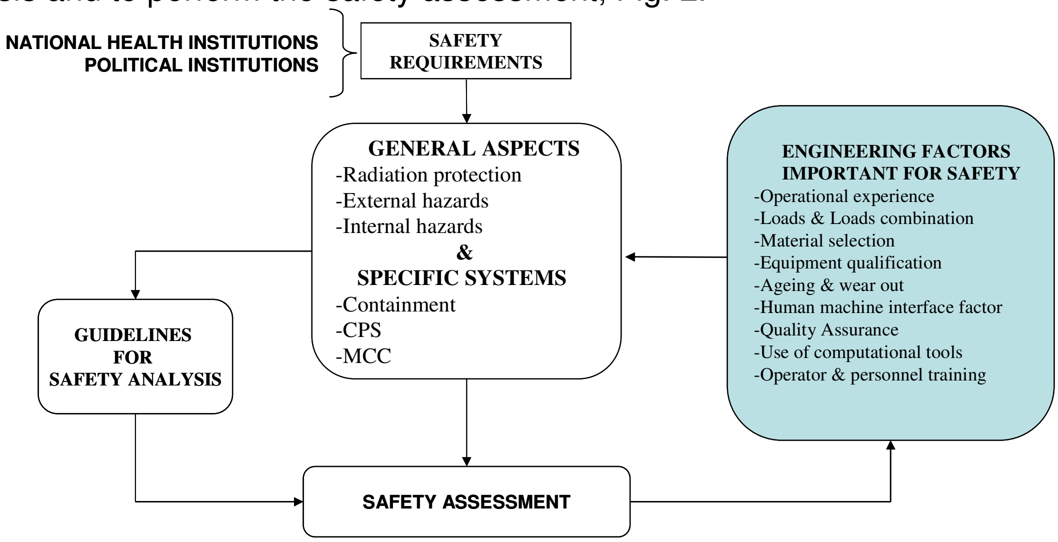


Fig. 2 –Licensing process: safety requirements

The safety assessment in licensing is the process throughout the safety analysis, deterministic and / or probabilistic, providing the safety analysis report and, hence, to the final safety analysis report, Fig. 3.

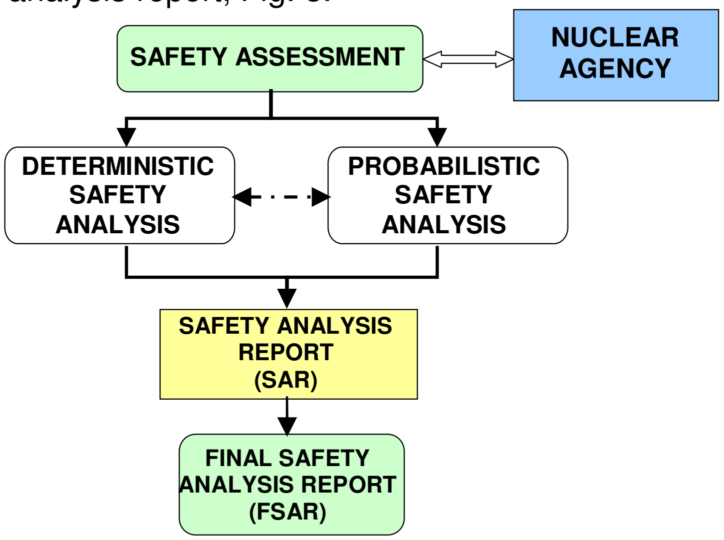


Fig. 3 – Licensing process: safety assessment

2.1.1.1. Deterministic safety analysis

The deterministic safety analysis studies the behaviour of the plant in specific operational and accidental status based on either prudential evaluation or in accordance with the specific safety requirements [32] , Fig. 4 and Fig. 5. It includes the following [33]:

1) confirmation that operational limits and conditions are in compliance with the assumptions and intent of the design for normal operation of the plant;
2) characterization of the PIEs that are appropriate for the design and site of the plant;
3) analysis and evaluation of event sequences that result from PIEs;
4) comparison of the results of the analysis with radiological acceptance criteria and design limits;
5) establishment and confirmation of the design basis; and
6) demonstration that the management of anticipated operational occurrences and design basis accidents is possible by automatic response of safety systems in combination with prescribed actions of the operator.

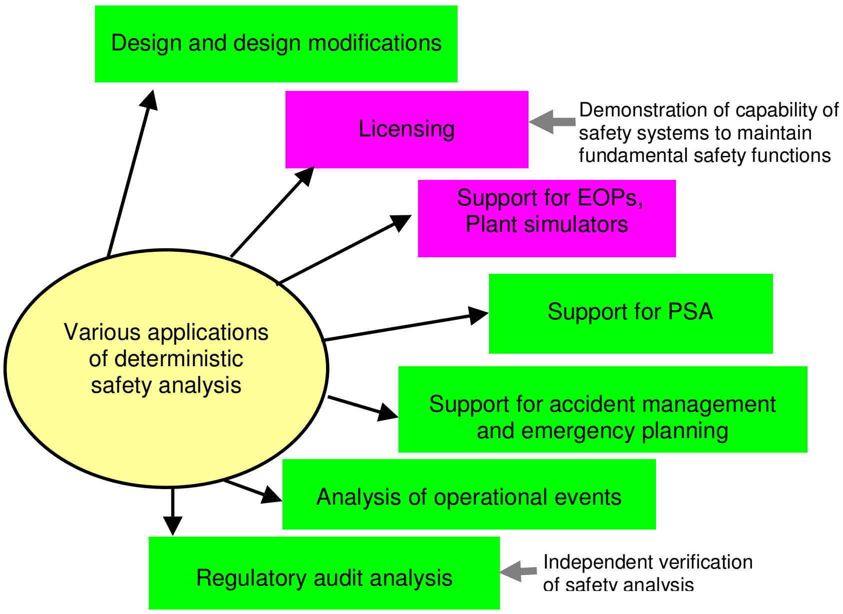


Fig. 4 – Deterministic safety analysis applications

The applicability of the analytical assumptions, methods and degree of conservatism used should be verified. The safety analysis of the plant design should be updated with regard to significant changes in plant configuration, operational experience, and advances in technical knowledge and understanding of physical phenomena, and should be consistent with the current or ‘as built’ state.

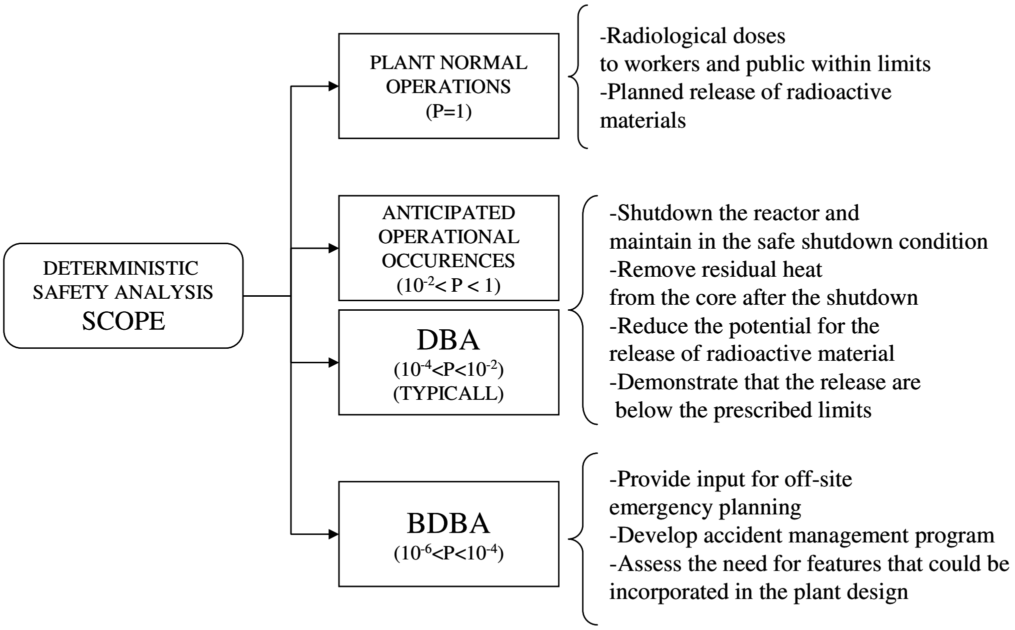


Fig. 5 – Deterministic safety analysis: scope

2.1.1.2. Safety margins

The safety assessment, as well as the safety analysis, can be carried out if the safety limits are defined.

The safety margins provide the difference, in physical units, between the critical value assigned to a parameter, associated with the failure of a system or a component or a phenomena, and the actual value of the parameter (e.g. damage of a barrier in Fig. 6). It should be noted that tipicially, real values are unknown.

In consideration of the results of the analyses, the safety margins provide the difference, in physical units, between a threshold, that characterizes an acceptance criterion, and the results provided by either a conservative calculation or a BE calculation. In later case the uncertainty band must be used when defining the safety margins [35].

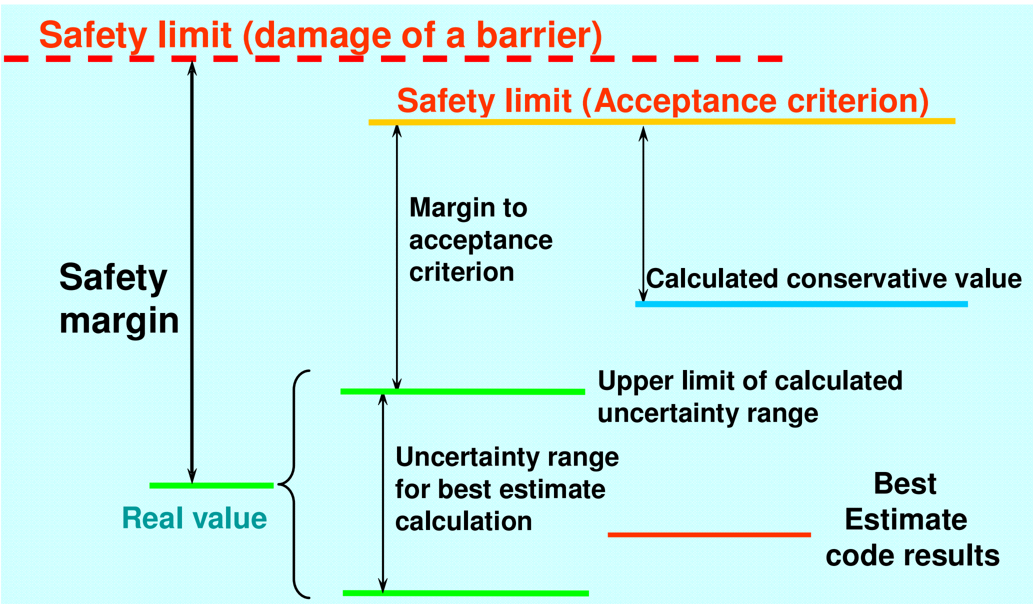


Fig. 6 – Safety margins

2.2. *Importance and role of safety analysis for licensing process*

Safety analysis represents the principal way to demonstrate the safety of a nuclear installation. It is a systematic process focused at ensuring that all relevant safety requirements are satisfied. Due to the fact that very complex phenomena might occur, sophisticated computer codes have to be used to simulate the plant response to the spectrum of the PIEs. These codes are supposed to handle a broad spectrum of complex physical phenomena including the two-phase flow thermal-hydraulic, heat transfer processes, neutron kinetics, transport of non-condensable gases and simulation of the plant control and safety systems [34].

Development of the computer codes, such as RELAPSE, the predecessor of the RELAP codes series, began in early 1960s. The first attempts leaded to very simple programs with simplified models for the thermal-hydraulics with homogenous equilibrium models of the two-phase flow processes (i.e. liquid and vapor phase having the same temperature and same velocity) and simple heat transfer models. Mediocre knowledge of the processes expected to occur during the plant accidents and transients and weak computing technology restricted the scope of the codes development. The limited knowledge of the simulated processes was compensated by introduction of large amount of conservative assumptions into the code models to assure that real plant response would not lead to more aggravate consequences.

To better understand what is happening during the accident, extensive experimental program was initiated by the technical community with majority of the experiments being performed during a period from late 1960s till early 1990s. Comprehensive summary of this research work is given in US Nuclear regulatory commission (US NRC) compendium [4].

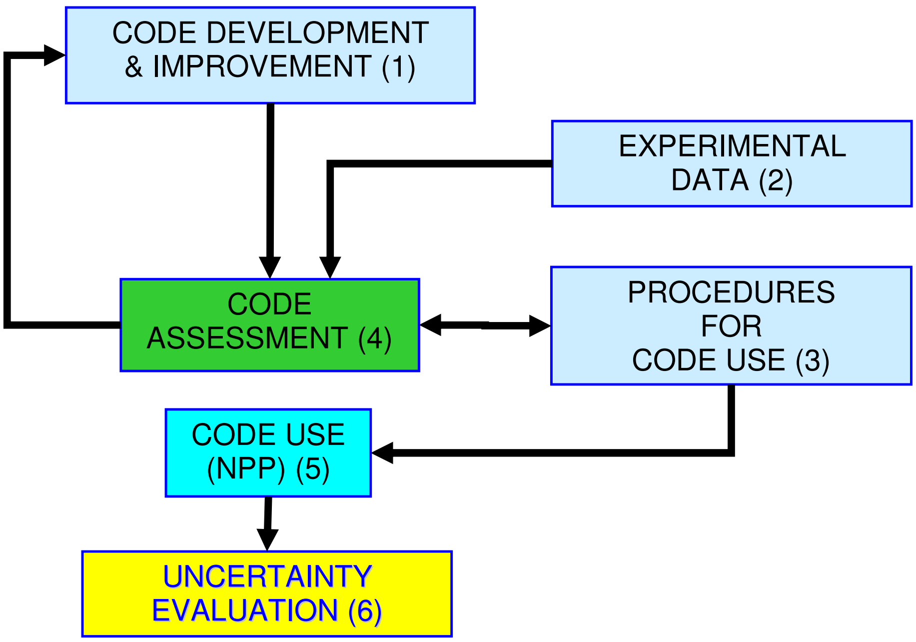


Fig. 7 – Needs for uncertainty: consistent application (development, qualification and application) of a thermo-hydraulic system code

Today enormous amount of data are available to support code development. Advanced thermal-hydraulic computer codes are programmed on a best-estimate philosophy without any intentional conservative assumptions in the code models. Results of the code simulations show the accepted accuracy when compared with the experimentally measured data, [6] and [7]. Still significant amount of code models rely on the constitutive equations which are naturally accompanied by large uncertainty. There are several simplifying assumptions and approximations, plus further data needed for the code calculations, available only with certain amount of uncertainty. This leads to a necessity of evaluating the uncertainty of the best-estimate simulations, as depicted in Fig. 7. The code development and improvement process, block 1 in Fig. 7, is conducted by "code developers" who make extensive use of assessment (block 4), typically performed by independent users of the code (i.e., group pf experts independent from those who developed the code). The consistent code assessment process implies the availability of experimental data and of robust procedures for the use of the codes, blocks 2 and 3, respectively. Once the process identified by blocks 1 and 4 is completed, a qualified code is available to the technical community, ready to be used for NPP applications (block 5). The NPP applications still require "consistent" procedures (block 3) for a qualified use of the code. The results from the calculations are, whatever the qualification level achieved by the code is, affected by errors that must be quantified through appropriate uncertainty evaluation methodology (block 6) [88].

No.	Computer code	Initial and boundary conditions	Availability of components and systems
1	Conservative	Conservative input data	Conservative assumptions
2	Best estimate	Conservative input data	Conservative assumptions
3	Best estimate with uncertainties	Realistic input data with uncertainties	Conservative assumptions
4	Best estimate with uncertainties	Realistic input data with uncertainties	PSA based assumptions

Tab. 1 – Summary of approaches for performing the safety analysis

2.3. Current practices in the licensing calculations

In <u>all</u> countries using nuclear energy for power production, safety analysis of prescribed set of PIEs has to be performed and documented in the safety analysis report (SAR). SAR is then reviewed and/or approved by the national regulator. Usually predefined structure and content of SAR and approved procedures and methodologies, brought out by the regulator in the form of guide, are followed.

Safety analysis in principle consists from the following key components: (a) computer code able to simulate the phenomena expected to occur during the simulating event, (b) the input data representing the simulated nuclear facility – its geometrical and material parameters (volumes, lengths, heat capacities, heat conductivities, material densities), physical parameters referred to as initial and boundary conditions (pressures, mass flow rates, temperatures), (c) models of various plant components and systems controlling its operation (valves, pumps, measurements, signals), and (d) acceptance criteria or limits. For each part of the safety analysis different approach can be used. The summary of the possible combinations can be found in Tab. 1.

2.3.1. Conservative approach

Conservative approach is based on the variation of key components of the safety analysis, listed in Tab. 1, in a way leading to pessimistic results relative to specified acceptance criteria.

Historically the conservative approach meant the use of conservative code along with the conservative assumptions about the availability of systems and components and conservative input data – the first option from Tab. 1. The results obtained by this approach are highly conservative. The high amount of conservative assumptions reflected the limited knowledge and experience to be compensated to make sure the safety of the nuclear installation can be demonstrated. Due to the excessive conservative assumptions this approach can be misleading (e.g. unrealistic behavior may be predicted by the simulation or order of events may be changed etc.) and may lead to unphysical results. In addition, the level of conservatism is unknown. Therefore, the use of this approach is not recommended by IAEA [10], however it is still mandatory in the USA according to the Appendix K of US – 10 CFR 50 [11]. In addition it should be noted that the conservative approach cannot be compared with "real" experimental results.

As more knowledge and experience about physical phenomena was accumulated, more realistic models have been programmed into the computer codes. The level of realism already reached such a level that most of the codes, available today, cannot be referred to as conservative, but they are called best-estimate. Since best-estimate does not fully equal to realistic, and moreover availability of systems and components and initial and boundary conditions are still assumed to be conservative, the second option from Tab. 1, is now typically referred to as conservative approach. This option is clearly not allowed in the USA according to the Appendix K [11], but it is widely used in many countries and is also recommended by the IAEA [10].

Below the summary of drawbacks and benefits of conservative approach is given:

- ☹ Intentional conservatisms may not always lead to conservative results. For example, high power during SB LOCA may lead to over-prediction of swell level and over-prediction of core cooling, thus lower peak cladding temperatures, which is opposite to pessimistic expectations when evaluating the peak cladding temperature acceptance criterion [12]. For such reason in "*1988, Dougall-Rohsenow was removed from the list of acceptable post-dryout correlations since it had been found to yield non conservative predictions, the only part of Appendix K that was found to be non conservative*" [121].
- ☹ Degree of conservative assumptions can change during a course of the event. Specifically selected value of the parameter can be conservative in the beginning of the event but can change to favorable value during another period of the event.

- ☹ Intentional conservative assumptions can results in misleading sequences of events and unrealistic time-scales.
- ☹ Conservative values of important parameters are typically selected based on engineering judgment (possible user effect) in combination with sensitivity calculations. Sensitivity calculations are usually limited in scope and typically do not include the investigation of the combined dependency, which means that each important parameter is tested individually without examining the possible influence when other parameters change. Moreover each of these parameters is tested for limited number of values, typically minimum and maximum is tested, so the most penalizing value can be easily omitted.
- ☹ When applying the best-estimate code in the conservative approach, the uncertainty and shortcomings of the code models are neglected assuming the intentional conservative assumptions about the availability of the systems and components and about initial and boundary conditions are sufficient to compensate for it. This compensation is never analyzed and no evidence of sufficient conservative assumptions over code models deficiencies is demonstrated.
- ☺ There is a long experience and well established procedures for conservative approach reducing the user effect.
- ☺ There is a large amount of supporting materials, i.e. various SARs, technical documents and reports with sensitivity calculations, to provide the background information.
- ☺ Simple, clear and understandable procedures to demonstrate conservative assumptions to convince regulators.

2.3.2. Best estimate plus uncertainty approach

BEPU approach (option 3 from Tab. 1) is characterized by applying the best estimate code along with nominal plant data and with best-estimate initial and boundary conditions to simulate the intended event. When performing the licensing calculations it is still expected that the availability of safety and control components and systems is defined in a conservative way, including the assumption of the single failure and loss of off-site power. However, uncertainty of the best estimate calculation has to be quantified and considered when comparing the calculated results with the applicable acceptance criteria. In the USA licensing process, this approach was formulated as an alternative to Appendix K conservative approach defined in 10 CFR 50 [11] to reflect the improved understanding of ECCS performance obtained through the extensive research [4] and the RG 1.157 [13]. Licensee is permitted to use either Appendix K features or a realistic evaluation model with consideration and qualification of uncertainty.
Development of the BEPU approach ranges almost over past three decades. The international project on the evaluation of various BEPU methods – UMS (Uncertainty Methods Study) –, conducted under the administration of the OECD/NEA [3] during 1995-1998, already concluded that 1) the methods are suitable for use under different circumstances and 2) uncertainty analysis is needed if useful conclusions are to be obtained from best-estimate codes. Similar international projects are in progress under the administration of OECD/NEA (BEMUSE – Best Estimate Methods Uncertainty and Sensitivity Evaluation [14]) and IAEA (Coordinated Research Project on Investigation of Uncertainties in Best

Estimate Accident Analyses) to evaluate the practicability, quality and reliability of BEPU methods.

BEPU approach is in detail discussed in Annex I, here only a short summary of drawbacks and benefits of this approach is given:

- ☹ Practical application can be seriously time consuming, due i.e. long and exhausting preparation of data, high number of calculations etc. This has also the impact on the requirements on the computation tools (high computer power, large data storage space).
- ☹ Selection of uncertain parameters and definition of probabilistic distribution functions can be difficult due to the lack of information. Definition of uncertain parameters is also usually based on expert judgment leading to a possible user effect.
- ☹ Extensive experimental and operational data are needed to reference applied values.
- ☺ Prediction of 'realistic' response of the plant to the PIE is given.
- ☺ Safety margins can be clearly determined.
- ☺ Statistically sound evaluation of combined influence of input parameters is performed.
- ☺ There are close links to experimental results justifying applied procedures.

3. APPLICATIONS OF BEST ESTIMATE METHODS IN LICENSING PRACTICE

Before current BE computational tools were available, only conservative licensing calculations with conservative codes were performed. Use of BE computer codes either combined with conservative input data or with realistic input data but associated with evaluation of uncertainty of results are two acceptable options, for demonstrating that the safety is ensured with sufficient margin, offered by the recently made advances [41].

There is considerable interest on use of BE tools from research organizations, utilities and regulators but with different objectives [42]:

- Regulator is interested that acceptance criteria are fulfilled with high confidence
- Utility is interested in “useful” results aiming at reduce conservatisms
- Research wishes to improve practicability of methods

Regulatory Bodies

Regulations, in most countries, permit the use of best-estimate codes or allow that the state of science and technology – “state of the art”– is applied in licensing [9].

Up to the date the BE is used in licensing in Brazil, Korea, Lithuania, Netherlands and USA. Significant activities are also performed for use in licensing in Canada, Czech Republic, France, Germany, Slovak Republic and Slovenia [44 - 52].

In particular the USA 10 CFR 50.46 allows the use of BE codes instead of conservative code models. In this case the uncertainties have to be identified and assessed so that the uncertainty in the calculated results can be estimated with a high level of probability that acceptance criteria would not be exceeded [39].

The Regulatory Guide 1.157 considers acceptable, as “High level of probability”, 95% or more [13].

Industry

From 1990s vendors started to develop BE methodologies. In particular Westinghouse has been developed a BE LOCA (Loss Of Coolant Accident) evaluation model [54]. The methodology was approved by the NRC in 1996 after an extensive review [55]. It is based on the CSAU (Code Scaling, Applicability and Uncertainty) methodology, WCOBRA/TRAC computer code and use of response surfaces to estimate PCT uncertainty distribution with the 95th percentile PCT determined from a Monte Carlo sampling and accepted as the licensing basis PCT. At the beginning it was applicable to 3- and 4-loop plants with safety injection into the cold leg [56], [57]. Subsequently, the methodology applicability was extended to 2-loop plants with upper plenum injection in 1999 and advanced passive (AP) plant such as the AP600 and AP1000 [58]. More recently, the methodology has been modified toward non-parametric methods and called ASTRUM (Automated Statistical Treatment of Uncertainty Method) in 2004. Since its approval, Westinghouse has applied the methodology to more than 30 NPPS either in the USA and abroad [59].

In 2003 Framatome ANP also licensed its realistic LB-LOCA methodology. It uses S-RELAP5 code originally developed by Siemens and follows CSAU approach but was the first to use a nonparametric order statistic method in licensing, eliminating the need for response surfaces [60]. The application of nonparametric order statistics in safety analysis was first proposed by Hofer et al. from GRS, Germany [43].

IAEA

IAEA Safety Standards recommend as one of the options for demonstration of sufficient safety margins use of BE computer codes with realistic input data in combination with evaluation of uncertainties of the calculation results [41].

Its broader use in the future is therefore envisaged, even though it is not always feasible because of the difficulty of quantifying code uncertainties with sufficiently narrow range for every phenomenon and for each accident sequence.

3.1. IAEA international standards

Typical hierarchy of safety requirements with increasing demands from top to bottom is the following:

- National legislation and binding international treaties and conventions
- IAEA safety requirements
- WENRA reference levels
- European utility requirements (EUR).

The IAEA is the world's center of cooperation in the nuclear field. It was set up as the world's "Atoms for Peace" organization in 1957 within the United Nations family. The Agency works with its Member States and multiple partners worldwide to promote safe, secure and peaceful nuclear technologies.

All IAEA safety related statutory functions and activities are linked together through an integrated safety approach (Fig. 8), for an efficient and effective delivery of the program.

This is ensured by the application of the following interdependent and complementary principles and practices:

- Assessing compliance with the IAEA Safety Standards and their application
- Providing for sustainable Education and Training
- Co-coordinated Research Programs
- Supporting Technical Co-operation (TC) projects
- Fostering Information Exchange and Networking.

The IAEA's safety standards are not legally binding on Member States but may be adopted by them, at their own discretion, for use in national regulations in respect of their own activities. The standards are binding on the IAEA in relation to its own operations and on States in relation to operations assisted by the IAEA (i.e. under technical cooperation agreements). Any State wishing to enter into an agreement with the IAEA for its assistance in connection with the siting, design, construction, commissioning, operation or decommissioning of a nuclear facility or any other activities will be required to follow those parts of the safety standards that pertain to the activities to be covered by the agreement.

However, the final decisions and legal responsibilities in any licensing procedures rest with the States.

Although the safety standards establish an essential basis for safety, the incorporation of more detailed requirements, in accordance with national practice, may also be necessary. Moreover, there will generally be special aspects that need to be assessed on a case by case basis.

Fig. 8 – IAEA integrated safety approach [124]

The attention of States is drawn to the fact that the safety standards of the IAEA, while not legally binding, are developed with the aim of ensuring that the peaceful uses of nuclear energy and of radioactive materials are undertaken in a manner that enables States to meet their obligations under generally accepted principles of international law and rules such as those relating to environmental protection, in order to do not cause damage in another State.

3.1.1. Licensing process according to the IAEA international standards

IAEA objectives [31] [§ 2.1] are implemented through the assessment of the defence in depth (Fig. 9) reported in the safety standards (Fig. 10) and other related documents (Fig. 11 and Fig. 12).

To ensure the safety of plants three fundamental safety functions (FSF) have to be performed in operational states, during and following DBAs and, to the extent practicable, in, during and following the considered plant conditions beyond the DBA [35]:

- Control of reactivity;
- Removal of heat from the fuel;
- Confinement of radioactive materials and control of operational discharges, as well as limitation of accidental releases.

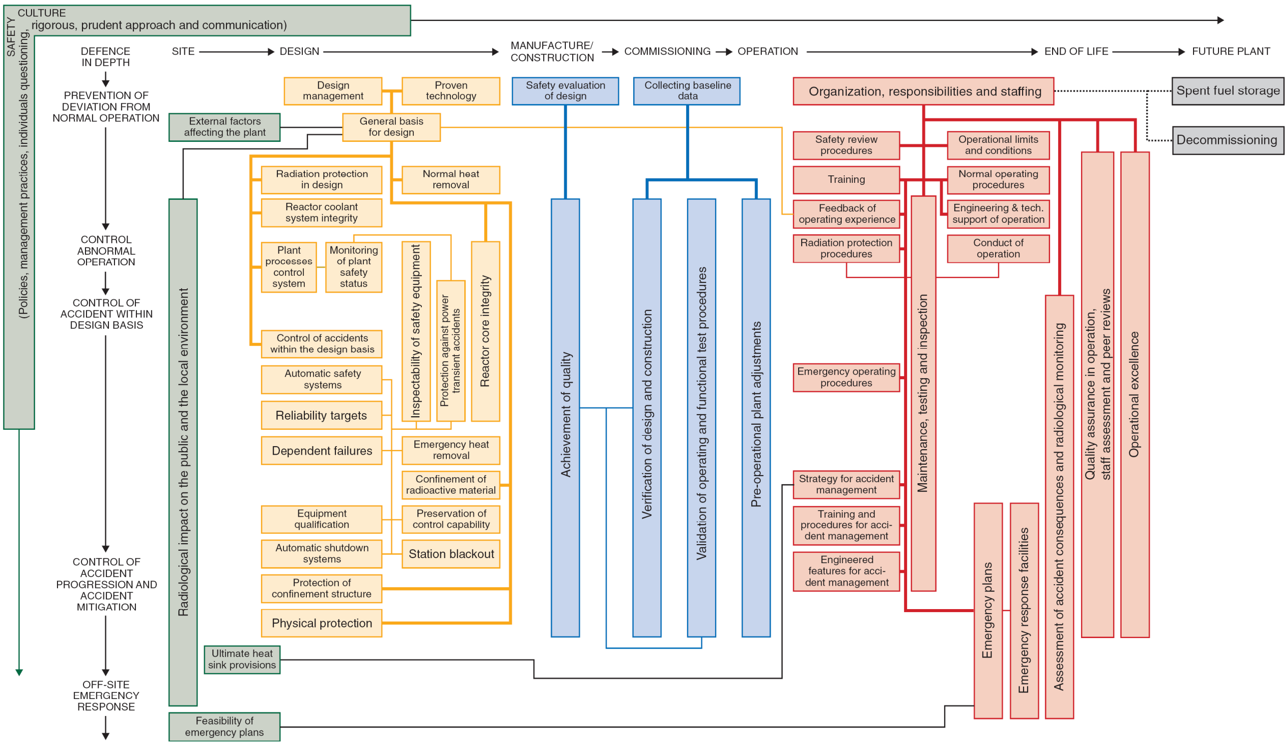


Fig. 9 – IAEA specific safety principles: coherence and interrelations [31]

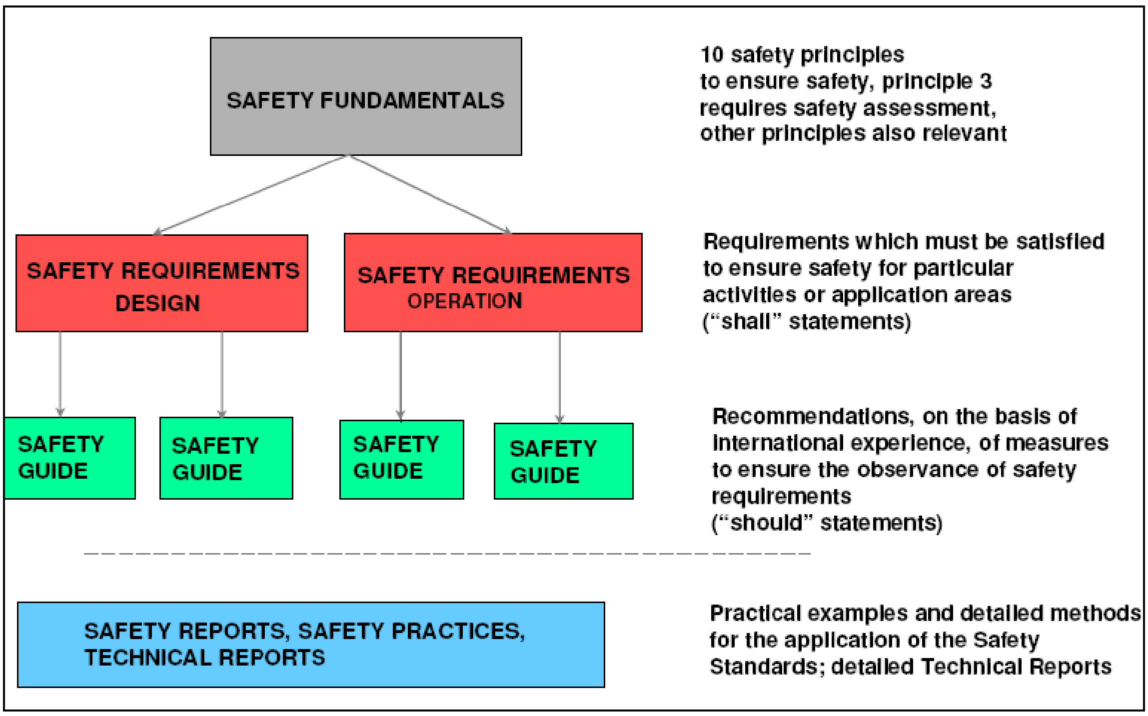


Fig. 10 – IAEA safety standards and other related documents

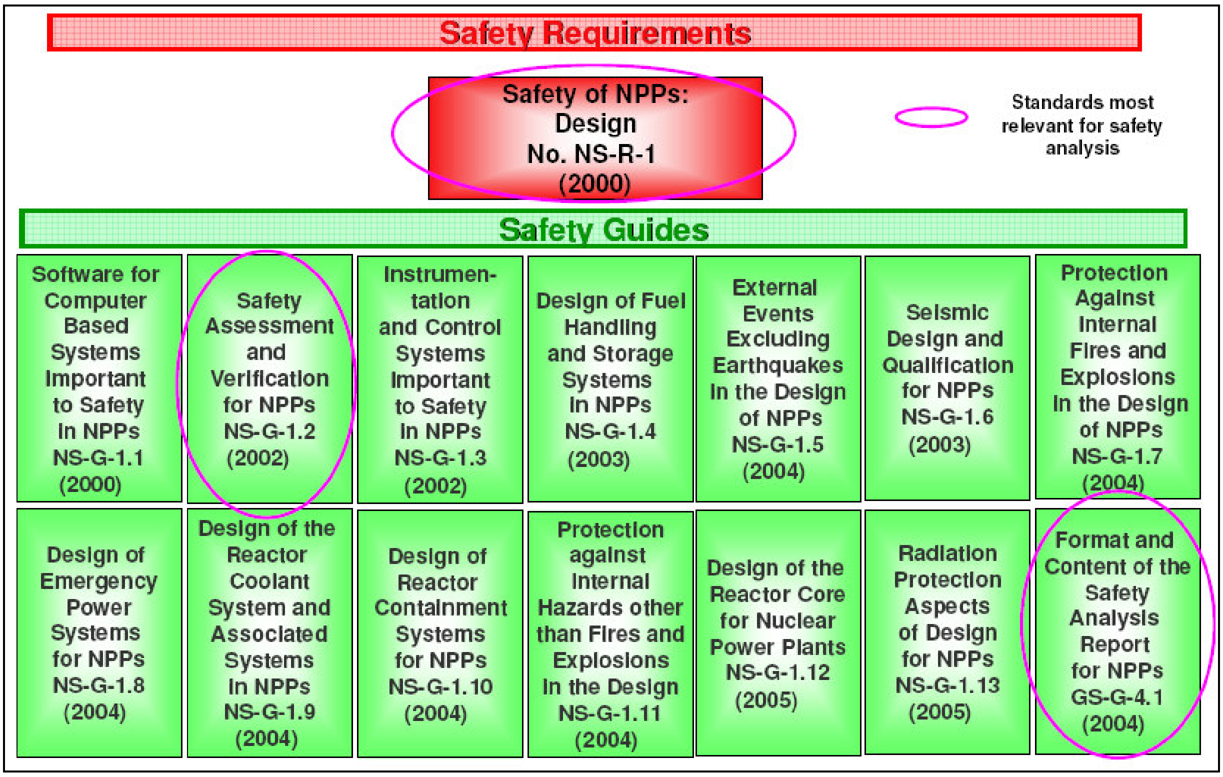


Fig. 11 – IAEA safety standards for design

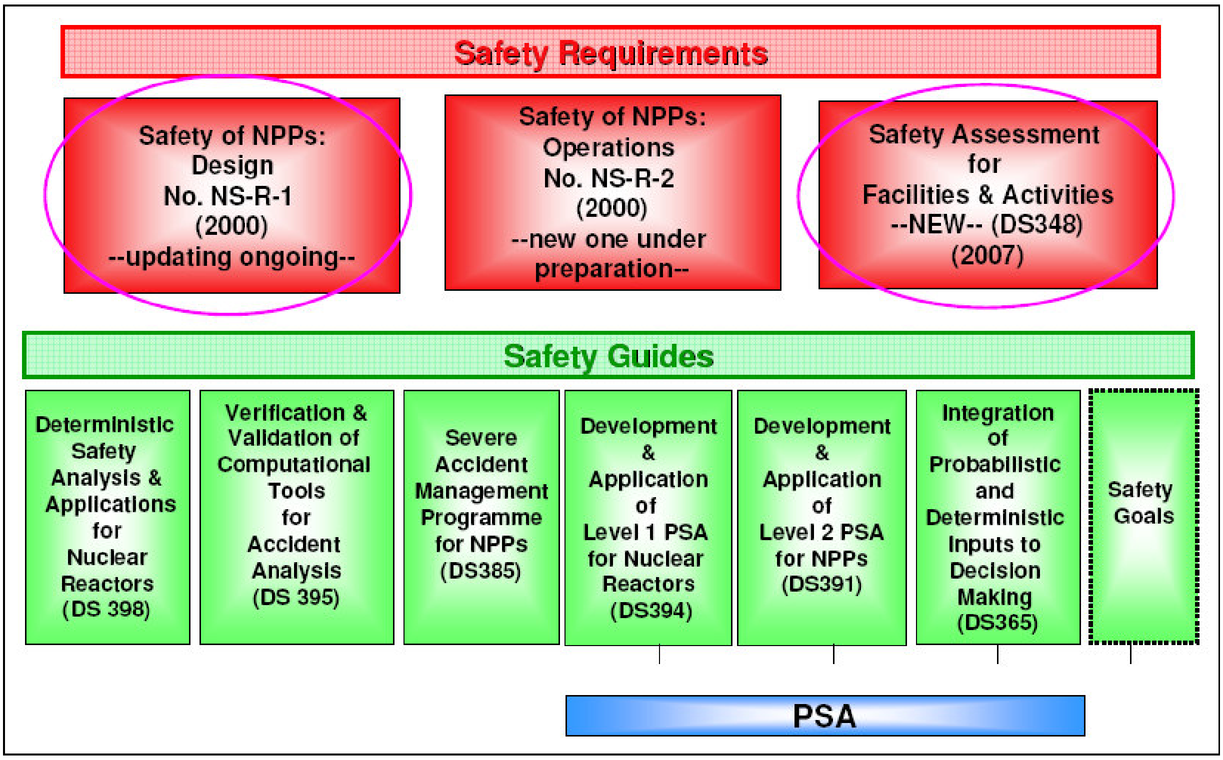


Fig. 12 – IAEA safety standards in safety assessment and accident management

3.1.1.1. *Assessment of defence in depth*

Defence in depth is an overall safety philosophy and strategy, aimed at compensating for human errors and equipment failures, which encompasses all safety activities, including the siting, design, manufacture, construction, commissioning, operation and decommissioning of nuclear power plants. It includes a more general structure of multiple physical barriers and complementary means to protect the barriers themselves, the so-called levels of defence (Fig. 1).

For each level of defence, such strategy is shown in Fig. 13, where [33]:

- Challenges are generalized mechanisms, processes or circumstances (conditions) that may have an impact on the intended performance of safety functions
- Mechanisms are more specific processes or situations whose consequences might create challenges to the performance of safety functions
- Provisions are all the measures which can support the performance of the safety functions and prevent the mechanism from taking place.

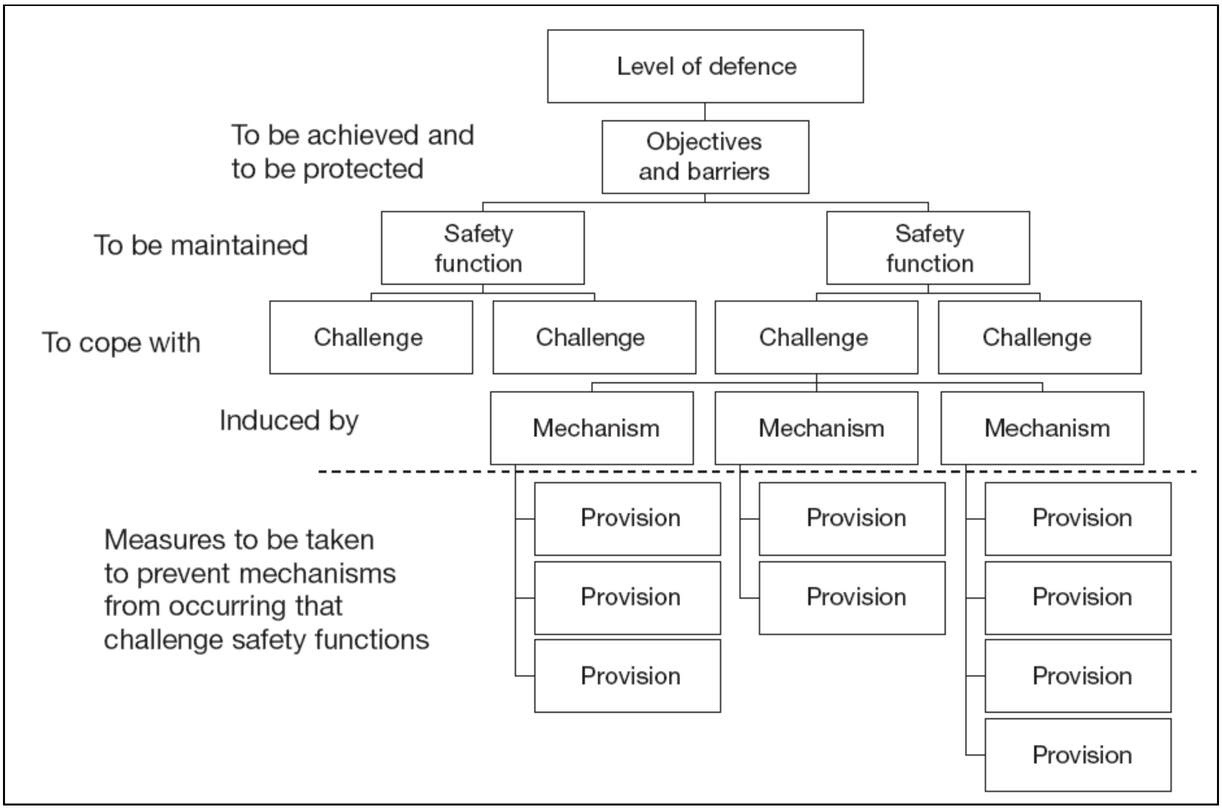


Fig. 13 – Structure for defence in depth provisions at each level of defence [33]

3.1.1.2. *Safety assessment and safety analysis*

Applying the assessment of the defence in depth to the IAEA safety objectives, Fig. 14 is developed which shows the interrelation between defence in depth, safety assessment and safety analysis.

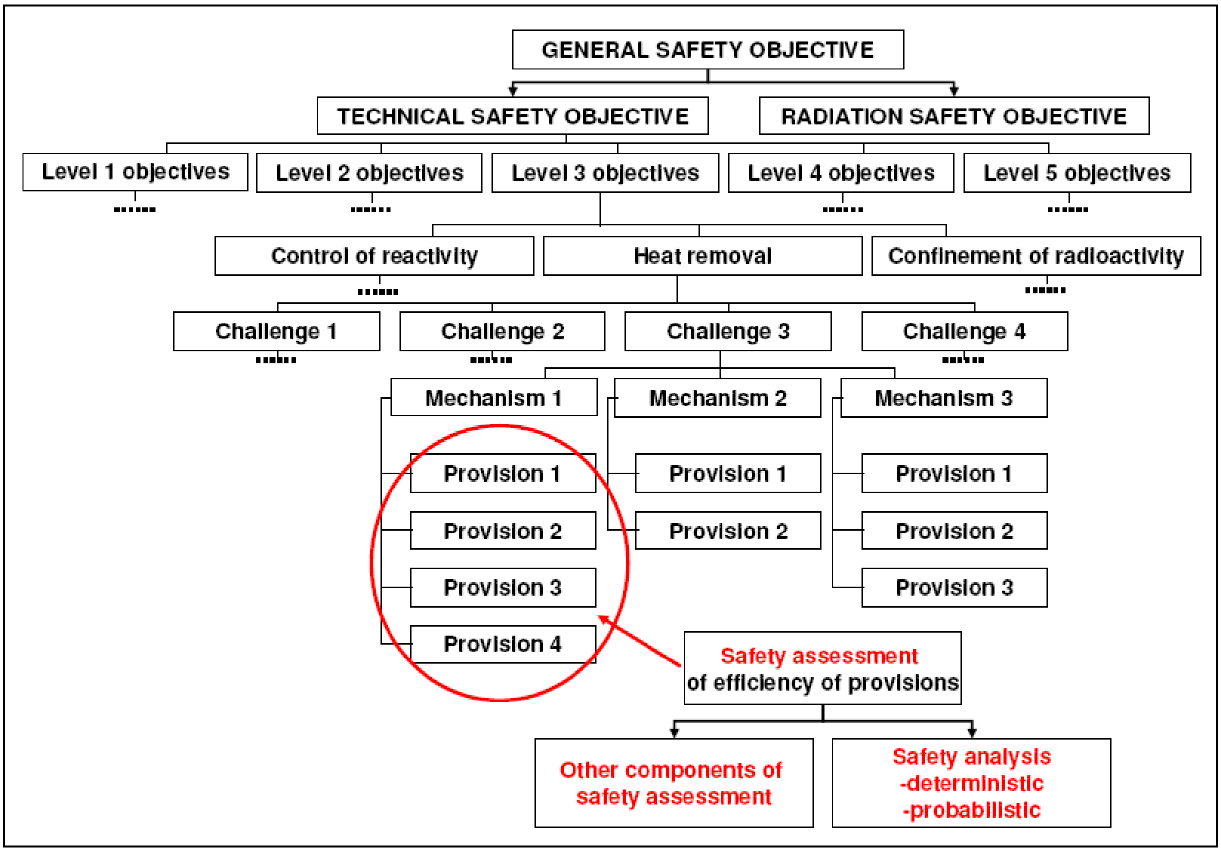


Fig. 14 – Interrelation of defence in depth and safety analysis

Safety assessment

The safety assessment is the systematic process that is carried out throughout the design process to ensure that all the relevant safety requirements are met by the proposed (or actual) design of the plant [33].

Safety analysis

The safety analysis is an analytical study by which it is demonstrated how safety requirements, such as ensuring the integrity of barriers against radioactive releases and various other requirements, are met for initiating events (both internal and external) occurring in a broad range of operating conditions and in other circumstances, such as varying availability of the plant systems. Complementary methods of safety analysis, deterministic and probabilistic, are properly balanced in evaluating the safety of an NPP.

Into the licensing process, the safety analysis is used to provide evidence to the regulatory body that the design is safe either in the design of a new plant or in the modification of the design of an existing one. Regulatory bodies may require new calculations when new evidence arises from experiments or from operational experience at the plant. Regulatory bodies may also require the use of updated computer codes which incorporate results arising from new experiments or from operational experience at the plant.

For a new plant, preliminary licensing analysis is performed prior to and as a condition of issuing the construction licence; and final licensing analysis is performed prior to and as a condition of issuing the operating license.

For modifications to an existing plant, the final licensing analysis is performed prior to connection and/or use of the equipment; for major changes, it may be a requirement to submit a preliminary licensing analysis before construction of the modification.
Licensing analysis for an existing plant may also be a requirement if its current safety assessment needs revision owing to results from research and development, from plant operating experience (e.g. results on the effects of ageing) or from the refinement of predictive models.

Deterministic safety analysis

Deterministic safety analysis predicts the response of a NPP in specific predetermined operational states to postulated initiating events. This type of safety analysis applies a specific set of rules and specific acceptance criteria. Deterministic analysis is typically focused on neutronic, thermo-hydraulic, radiological and structural aspects, which are often analysed with different computational tools.
In licensing, the objectives of the deterministic safety analysis can be summarized in:

- Demonstration of safety for all plant states
- Acceptance criteria to be specified, different for different categories of events
- Demonstration of capability of safety systems to maintain fundamental safety functions
- Acceptance criteria not to be exceeded, with margins
- Safety margins ensured by a conservative approach or by evaluation of uncertainties

Probabilistic safety analysis

Probabilistic safety analysis (PSA) combines the likelihood of an initiating event, potential scenarios in the development of the event and its consequences into an estimation of core damage frequency, source term or overall risk arising from operation of the NPP. The number of event sequences can be very large.
In licensing, the objectives of the probabilistic safety analysis can be summarized in:

- Demonstration of probabilistic design targets
    - Frequencies of PIEs
    - Reliability of control and safety systems
    - Cumulative core damage frequency
- Cumulative frequency of large radioactivity releases
- Demonstration of probabilistic acceptance criteria (if applicable)
- Demonstration of well balanced design (no single dominant contributor to the risk)

Acceptance criteria

Acceptance criteria are used to judge the acceptability of the results of safety analysis. They may:

– Set numerical limits on the values of predicted parameters;
– Set conditions for plant states during and after an accident;
– Set performance requirements on systems;
– Set requirements on the need for, and the ability to credit, actions by the operator.

In licensing analysis, the acceptance criteria are defined by the regulatory body. In some desirable situation, also the designer can propose some acceptance criteria to be accepted by the regulatory body.. In the latter case, it is the practice in some countries to reach agreement with the regulatory body on the acceptance criteria before the analysis starts.

Fig. 15 presents a qualitative interrelation between defence in depth and success criteria in function of their frequency and consequences. As for Fig. 1, it should be noted that the events can also not move from a protection level to the other protection level, e.g. a break of the primary coolant pipe (protection level 3) may happen without any abnormal operation (level 2) before.

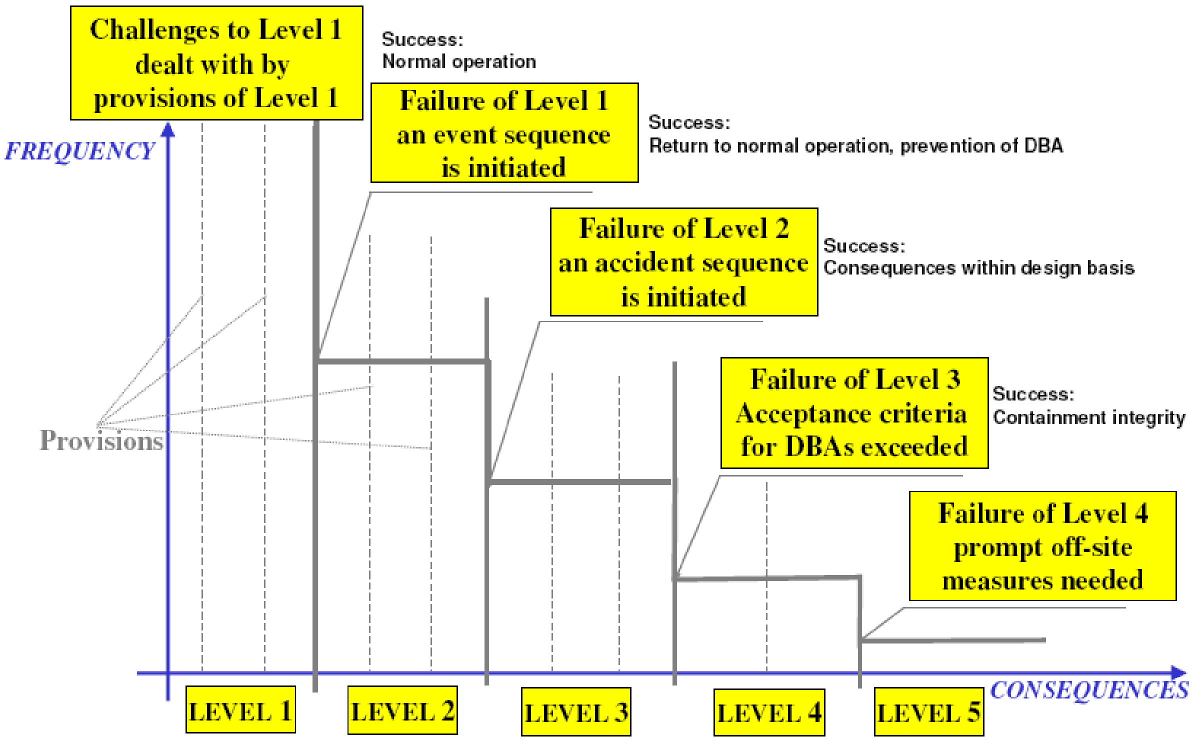


Fig. 15 – Interrelation of defence in depth and success criteria

3.1.1.3. Independent assessment and verification

The IAEA recommends the independent verifications performed separately both by the plant owner operator, who generally conducts an independent review of the design organization, and by the regulatory body [96].

The purpose of the independent safety verification is to establish that the safety assessment satisfies the applicable safety requirements. It is in addition to the quality assurance reviews carried out within the design organization.

The verification may be conveniently subdivided in phases to be performed at various significant stages of the design. Anyway, a final independent verification of the safety assessment should always be performed after the design is complete.
The independent verification can be applied both to the design and , by analogy, to other subsequent verification activities.
By definition, personnel performing such assessment are considered independent if they have not participated in any part of the design and safety assessment.
Fig. 16 gives an overview of the areas covered by the IAEA safety standards for the design of NPPs.

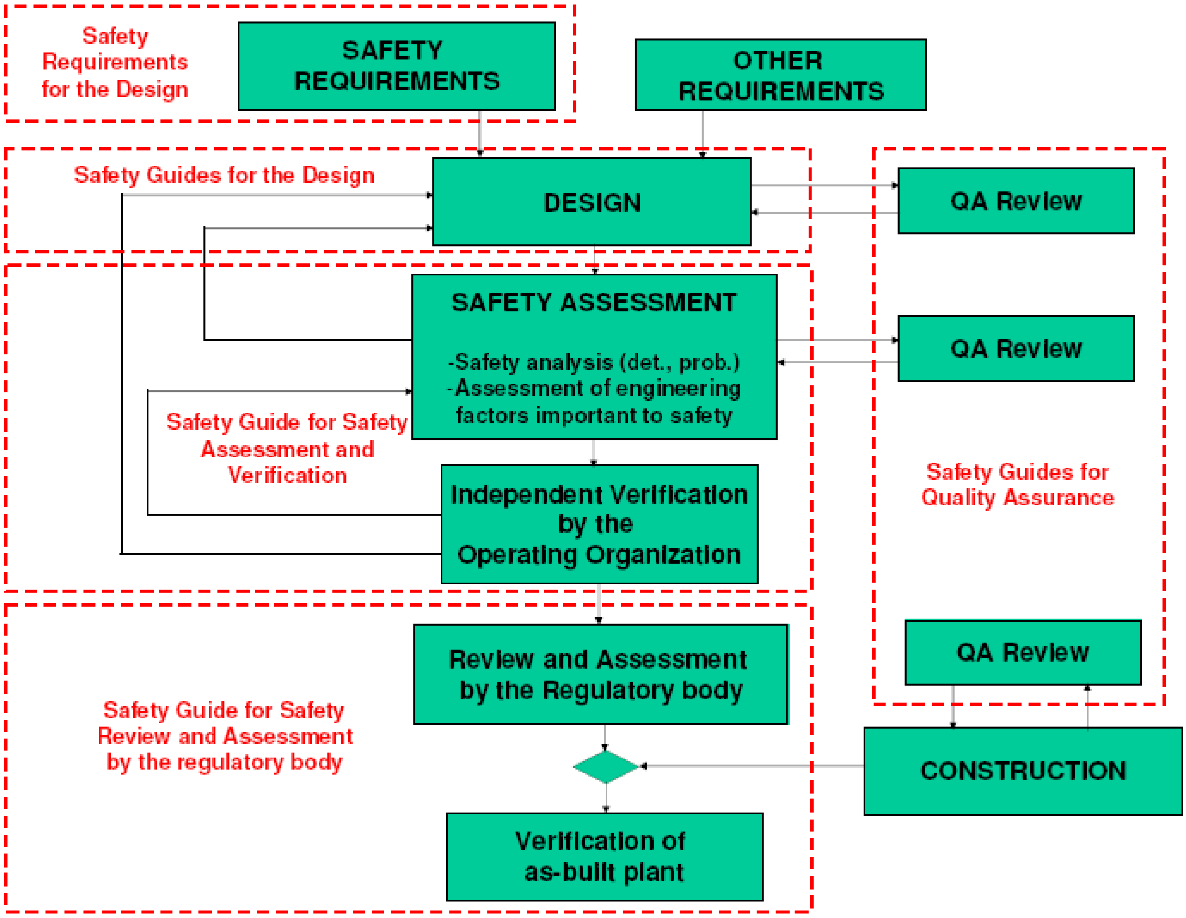


Fig. 16 – Areas covered by the IAEA Safety Standards for the design of NPPs 96]

3.1.2. Steps in licensing and role of individual partners

Despite the types and number of authorizations to be issued, in connection with a particular facility, vary between States, several points are identified, corresponding to the major stages of the authorization process, at which significant regulatory decisions are usually made and documents issued [99]:

- Approval of the site
- Authorizing construction, manufacture and installation
- Commissioning
- Operation
  - Initial routine operation
  - Routine operation
  - Return to operation after an outage

- Periodic safety review

– Modifications
– Decommissioning or closure

The process of granting an authorization is reported in Fig. 17. The license is a legal document issued by the regulatory body granting authorization to perform specified activities related to a facility or activity.

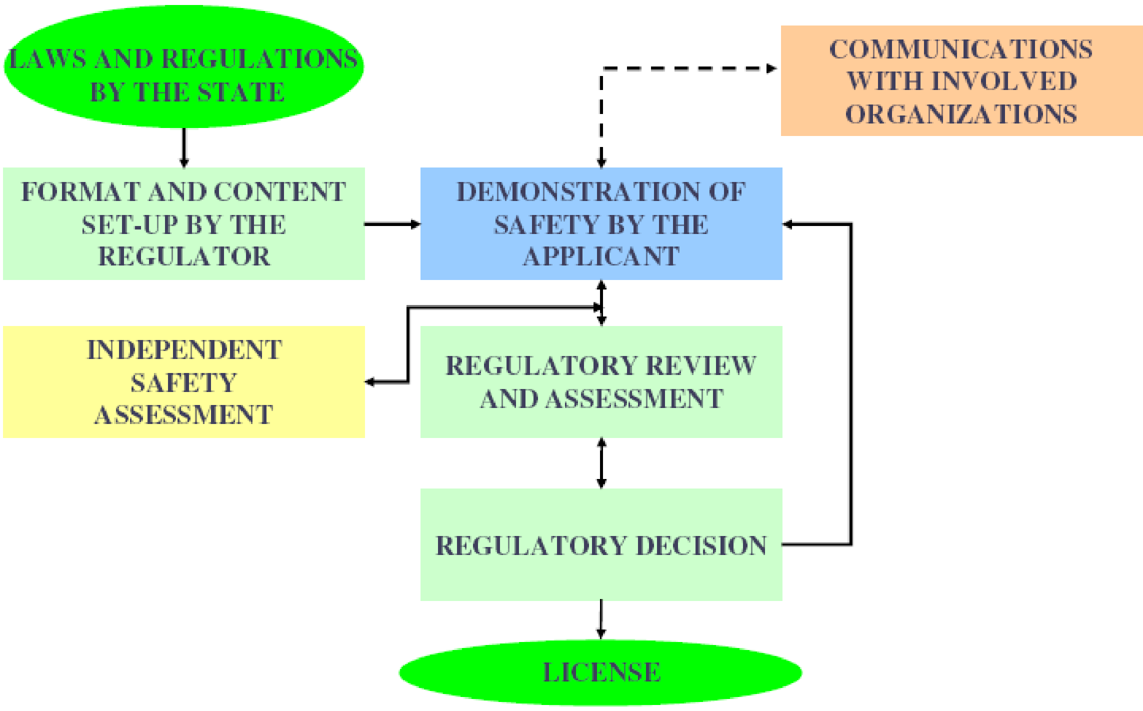


Fig. 17 – Process of granting of authorization

Government of a State

Government of a State is responsible for the adoption of legislation that assigns the prime responsibility for safety to the operating organization and establishes a regulatory body responsible for a system of licensing, for the regulatory control of nuclear activities and for enforcing the relevant regulations [31].

Regulatory body

Regulatory bodies is an authority or a system of authorities designated by the government of a State having legal authority for conducting the regulatory process, including issuing authorizations, and thereby regulating nuclear, radiation, radioactive waste and transport safety [99].
It is responsible to set safety objectives and standards, and to monitor and enforce them within the established legislative and statutory frame work.
The regulatory body must have the statutory authority, competence and resources[31]:

– to set safety standards;
– to license and inspect installations;
– to set, monitor and enforce licence conditions; and

– to ensure that corrective actions are taken wherever unsafe or potentially unsafe conditions are detected.

One key point, internationally accepted, is the independence of the regulatory body on the following aspects [102]:

– political
– legislative
– financial
– technical competences
– public information
– international.

Operating organization

The operating organization is the company or utility that is authorized by the regulatory body to operate one or more nuclear power plants. In accordance with the legal systems in the majority of States, an operating organization is the legal entity responsible for fulfilling the financial, commercial and safety obligations and any other obligations which may arise in connection with the operation of the nuclear power plants.

It shall establish [103] management functions in the following areas:

– Policy making
– Operating
– Supporting
– Reviewing

for adherence to safety requirements and procedures for safe control of the plant under all conditions, including maintenance and surveillance.

External supporting organizations

External supporting organizations (e.g. external maintenance organizations, plant vendors, research institutes and technical support organizations) are contractor personnel that may be used to perform tasks that are of a specialized or temporary nature for which it is not feasible to hire or maintain a full-time plant employee [103]. They should be trained and qualified to perform their tasks. Their roles and responsibilities should be clearly defined and understood.

3.1.3. Licensing documents

In the regulatory process there are different kinds of documentation.

Certain formal documents are required by the laws and regulations of the State or by the rules of the regulatory body. Other formal documentation are provided in response to specific requests from the regulatory body or at the initiative of the operator or other parties involved. The records of official meetings and hearings also constitute a means of formally exchanging information.

Four categories are overseen by the IAEA [99]:

– Regulatory documents
    • Legislation
    • Regulations, licences and other mandatory documents
    • Guides and other advisory documents
    • Internal guidance and procedures
– Industrial standards
– Documents produced by the regulatory body for a particular facility
    • Results of review and assessment

  - Records of inspection activities
  - Records of enforcement actions
  - Licence document
- Documents produced by the operator
  - Documents to be submitted for the authorization process
  - Reporting by the operator to the regulatory body
  - Records to be kept by the operator.

To obtain regulatory approval to build and operate a nuclear power plant, an operating organization should submit a detailed demonstration of safety, which shall be reviewed and assessed by the regulatory body in accordance with clearly defined procedures This information is presented in the form of a report, hereinafter referred to as a safety analysis report (SAR).

The requirements for SARs depend strongly on the type of regulatory regime adopted by a State, which may affect the scope and depth of the information presented.

It is common practice in many States that SARs are issued in successive and complementary parts, which may include:

1) An initial (preliminary) SAR or pre-construction SAR (PCSAR) that supports the application for authorization for siting and/or construction.
2) An updated (intermediate) SAR or pre-operation SAR (POSAR) that, in the licensing process, precedes an application for authorization to operate. It should essentially justify the finalized detailed design of the plant and presents a demonstration of its safety.
3) A finalized (final) SAR or Station SAR (SSAR) that incorporates the revisions to the intermediate report prior to the plant entering first routine operation. The final report should clearly demonstrate that the plant meets its design intent. Systematic updating of the SAR would then become a requirement for the operating organization during the remaining lifetime of the plant. This would usually be done periodically so as to reflect any feedback of operating experience, plant modifications and improvements, new regulatory requirements or changes to the licensing basis.

3.1.3.1. Format of a SAR

The SAR is suggested to have the following content:

1) Introduction
2) General plant description
3) Management of safety
4) Site evaluation
5) General design aspects
6) Plant system description and design performance
7) Safety analyses
8) Commissioning
9) Operational aspects
10) Operational limits and conditions
11) Radiation protection
12) Emergency preparedness
13) Environmental aspects
14) Radioactive waste management
15) Decommissioning and end of life aspects

3.1.4. Applicable options for demonstrating NPP safety by safety analysis, conservative and best estimate approach

The main objective of safety analysis is to demonstrate in a robust way that all safety requirements are met in order that safety margins are ensured, i.e. that sufficient margins exist between the real values of important parameters and the threshold values at which the barriers against release of radioactivity would fail (Fig. 6).

Moreover, the use of BE codes is generally recommended for deterministic safety analysis in Ref. [33]. Two options are offered to demonstrate sufficient safety margins in using BE codes (Tab. 1, Tab. 4):

1) The first option is the use of the BE codes “in combination with a reasonably conservative selection of input data and a sufficient evaluation of the uncertainties of the results.” Where, evaluation of uncertainties is meant more in the deterministic sense: code to code comparisons, code to data comparisons and expert judgements in combination with sensitivity studies are considered as typical methods for the estimation of uncertainties.
2) The second option is the use of the BE codes with realistic assumptions on initial and boundary conditions. However, for this option “an approach should be based on statistically combined uncertainties for plant conditions and code models to establish, with a specified high probability, that the calculated results do not exceed the acceptance criteria.”

Both options should be complemented by sensitivity studies, which include systematic variation of the code input variables and modelling parameters with the aim of identifying the important parameters required for the analysis and “to show that there is no abrupt change in the result of the analysis for a realistic variation of inputs (‘cliff edge’ effects).”

It could be useful to recall that sensitivity analysis must not be misinterpreted as evaluation of the uncertainties:

- Sensitivity analysis means evaluation of the effect of variation in input or modelling parameters on code results
- Uncertainty analysis means the deviation of quantitative statements on the uncertainty of computer code results from the uncertainties of the input parameters propagated through the model.

The current IAEA safety standards [33] allow for the BE selection of both categories of input data with associated evaluation of uncertainties. Thus the availability of nuclear power plant systems could also be judged based on realistic considerations. Even though such considerations are not excluded in BE analyses performed to date, it is typical to apply evaluation of uncertainties only to physical models embedded in the computer code and to nuclear power plant initial and boundary conditions, while assumptions regarding the availability of nuclear power plant systems are still used in a conservative way. Therefore, a significant conservative component still remains in present BE analyses [97].

A recent IAEA safety report [97] recommends the use of BE applications of complex thermal-hydraulic system codes, supported by uncertainty evaluation of the relevant output quantities, as a means of providing a better understanding of safety margins. It is also recommended that within the licensing process consistent procedures for the application of uncertainty methods be developed.

3.2. Relevant Regulations in Selected Countries

The BEPU methods were introduced based on the research conducted from 1974 through 1986. At that time the US-NRC initiated an effort to develop and demonstrate a licensing-acceptable BEPU method that could benefit nuclear plant operators (less conservative, consideration of uncertainties, economic gains). The code scaling, applicability, and uncertainty (CSAU) method was developed [104] and initially demonstrated for LB-LOCA in a PWR. After the pioneering CSAU, several new methods were proposed and presented at the special workshop on uncertainty analysis methods sponsored by the OECD – NEA, and the CSNI in 1994 [105]. One of the objectives of the workshop was also a preparation of the uncertainty methodology study (UMS). In the UMS (1995 to 1997) five uncertainty methods were compared [106]. The OECD-CSNI workshops in Annapolis (1996) [107], Ankara (1998) [108, 109], and Barcelona (2000) [109] also dealt with uncertainty evaluation methods. In the year 2000 the International Meeting on Updates in Best Estimate Methods in Nuclear Installations Safety Analysis (BE-2000) was held, and BE-2004 recently followed.

The demonstration and licensing applications of BEPU methods are listed in Tab. 2 Tab. 3, respectively [93]. For each application the country or organization, the transient, the plant, the number of code calculations N_{cal} (match with number of code runs with exception for USA LBLOCA and Slovenia LBLOCA), the number of input uncertain parameters varied (N_{par}), the uncertainty method used, the technique for combining of uncertainties, and the year of application / publication are shown.

The following paragraphs aim to outline the status of the application of the BE methods to the licensing process in USA, Europe, Germany, French, Lithuania, Check Republic, Canada, Brazil and Slovak Republic.

Some examples of applications in the licensing area are reported in the appendix III.

Country	Transient	Plant	Code	N_{cal} (N_{par})[a]	Method	Technique	Year
United States	LBLOCA	RESAR-3S PWR	TRAC-PF1/MOD1	184 (7)[b]	CSAU	RS (parametric)	1989
Korea	LBLOCA	Kori 3&4 NPP	RELAP5/MOD2	18 (9)	UQM	TL[c] (parametric) and RS (parametric)	1992
United States	SBLOCA	B&W PWR	RELAP5/MOD3	34 (8)	CSAU	RS (parametric)	1992
Slovenia	LBLOCA	Krsko NPP	RELAP5/MOD2	128 (7)[d]	CSAU	RS (parametric and nonparametric)	1993
Italy	SBLOCA	Krsko NPP	RELAP5/MOD2	1	UMAE	N.A.	1994
United States	LBLOCA	W[e] 4-loop PWR	TRAC-PF1/MOD2	21 (6)	CSAU	LHS-FPI	1996
Germany	SBLOCA	German PWR	ATHLET Mod 1.1	77 (45)	GRS	TL (nonparametric)	1997
Slovenia	SBLOCA	Krsko NPP	RELAP5/MOD3.2	59 (7)	CSAU	RS (nonparametric)	1999
Canada	LBLOCA	CANDU9	CATHENA 3.5b	11 (2)	CSAU[f]	RS (parametric)	1999
Japan	LOFW, turbine trip	BWR	TRACG	100 (23)	CSAU[f]	TL (parametric and nonparametric)	2000[g]
Germany	LBLOCA	German PWR	ATHLET Mod 1.2	100 (56)	GRS	TL (nonparametric)	2001
Canada	LBLOCA	CANDU6	CATHENA 3.5c/R1	64 (6)	CSAU[f]	RS (parametric)	2002
Slovakia	LBLOCA	VVER-440/213	ATHLET	93 (50)	GRS	TL (nonparametric)	2003

[a] N_{cal} = number of calculations; N_{par} = number of input uncertain parameters.
[b] Eight code runs produce 184 PCTs.
[c] TL = tolerance limits
[d] 64 runs produce 128 PCTs.
[e] W = Westinghouse.
[f] Following CSAU.
[g] Presented at BE-2000.

Tab. 2 – Demonstration application of BEPU methods [93]

Organization	Transient	Plant	Code	N_{cal} (N_{par})[a]	Method	Technique	Year
Westinghouse	LBLOCA	3- and 4-loop W[b] PWR	WCOBRA/TRAC	41 (23)	BELOCA[c]	2 RS (parametric), Monte Carlo analysis, FR[d]	1996
Westinghouse	LBLOCA	AP600	WCOBRA/TRAC	15 (3)	BELOCA[c]	RS (parametric), Monte Carlo analysis, bounding approach	1998
Siemens	LBLOCA	Angra 2 NPP	S-RELAP5	NA	GSUAM[c]	RS (parametric)	2000
Framatome-ANP and EDF	LBLOCA	Bugey 2 NPP	CATHARE GB	NA (10)	DRM	RS (parametric), deterministic evaluation model	2000
Ontario Power Generation	LBLOCA, LOFW	Bruce B NPP, Darlington NPP	NA	NA	IUA[c]	RS (parametric), functional RS	2000
University of Pisa	LBLOCA	Kozloduy 3 NPP	RELAP5/MOD3.2	1	CIAU	N.A.	2001
Framatome ANP	LBLOCA	3- and 4-loop W PWR	S-RELAP5	59 (30)	RLBLOCA[c]	TL[e] (nonparametric)	2001
KEPRI	LBLOCA	Kori 3&4 NPP	RELAP5/MOD3.1	59 (27)	KREM	TL (nonparametric)	2002
Westinghouse	LBLOCA	3- and 4-loop W PWR	WCOBRA/TRAC	59 (23)	ASTRUM[c]	TL (nonparametric)	2003
Lithuanian Energy Institute	blocking of coolant flow-rate in group distribution header	Ignalina NPP	RELAP5/MOD3.2	60 (12)	GRS	TL (nonparametric)	2003
Westinghouse	LBLOCA	AP1000	WCOBRA/TRAC	59	BELOCA ASTRUM[c]	RS (parametric), Monte Carlo, FR TL (nonparametric)	2004

[a]N_{cal} = number of calculations; N_{par} = number of input uncertain parameters.
[b]W = Westinghouse.
[c]Following CSAU.
[d]FR = Functional relationship.
[e]TL = Tolerance limits.

Tab. 3 – Licensing application of BEPU methods [93]

3.2.1. US NRC: Historical Evolution of Design Basis LOCA and 10 CFR 50.46

In the 1966-1967 time frame, research results indicated that Zircaloy cladding exposed to LOCA like conditions with peak temperatures in the vicinity of 1370 °C embrittled and ruptured, or even shattered upon cool down. This threatened the integrity of the core geometry which in turn threatened core cool-ability. Therefore a limit on the highest acceptable clad temperature was introduced as 1260 °C [U.S. Criteria 71]. Later on the AEC staff recommended that the ECCS criteria should be made more conservative to cover the areas where data were lacking or uncertainties were large, and the acceptable temperature limit for the cladding was decreased to 1204 °C [39].

The AEC staff supplemental testimony on the interim acceptance criteria emphasized that statistically based best estimate methods were then (1971-72) under development and should be applied to LOCA analysis. However, it wasn't until 1988 that 10 CFR 50.46 was revised to permit the use of best estimate analysis in lieu of more conservative Appendix K calculations. In effect a third major review of LOCA/ECCS issues was conducted to provide technical basis for this revision [39].

Regulatory Guide 1.157 [13] presents acceptable procedures and methods for realistic or best estimate ECCS evaluation models.
Regulatory Guide 1.203 [67] confirmed what outlined in the RG 1.157 and described a process that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in developing and assessing evaluation models that may be used to analyze transient and accident behaviour that is within the design basis of a nuclear power plant.
Despite of that, as it has been pointed out by ACRS in its January 11, 2001 Letter Report related to Regulatory Guide 1.157, these new regulatory guidance documents remain very qualitative and leaves considerable latitude in interpretation.
In parallel, NRC has been conducted research, together with industry, related to the acceptance criteria for ECCS. As an example, it should be mentioned that the ongoing development of a performance-based option for the embrittlement criteria in 50.46(b), and also the proposed rule for a voluntary alternative to 10 CFR 50.46, related to the definition of LOCA break sizes [110].
NRC have been licensed some methodologies:

- in 1996, the best estimate LOCA (BELOCA) methodology developed by Westinghouse, based on CSAU and the WCOBRA/TRAC thermal-hydraulic code;
- In 2003, the Realistic LB-LOCA (RLB-LOCA) methodology developed by Framatome ANP (today AREVA NP). It follows CSAU in relying on nonparametric statistical tolerance limits instead of RS; the RODEX3A fuel rod code and S-RELAP5 are used;
- In 2004 ASTRUM methodology developed by Westinghouse, It follows CSAU evaluation methodology framework. The ASTRUM uses the same code and the same uncertainty distributions as the BELOCA methodology and a revised method for combining the uncertainties. The nonparametric order statistics are used, while RS and superposition correction uncertainty is eliminated.

3.2.2. Brazilian Nuclear Regulatory Body (CNEN)

The CNEN adopts the American rule 10 CFR 50.46, September 1988 revision, that allows the use of realistic evaluation models to calculate the performance of the Emergency Core Cooling System. The LOCA analysis shall in such cases fulfil the requirement of identifying and evaluating the uncertainty in the analysis methods and inputs and this uncertainty must be considered when comparing the calculated results with the acceptance criteria[46].

3.2.3. Canadian Nuclear Safety Commission (CNSC)

Licensing of nuclear power plants in Canada has been mostly based on using a very conservative approach known as Limit of Operating Envelope (LOE) [51].
The computer codes, used by the Canadian nuclear industry for licensing safety analysis, are mostly based on best estimate models, whereas the analysis initial conditions and standard assumptions are selected as boundary values or assumptions to yield very conservative results.
Similar to practice used in the US and in many other countries, an alternative approach has been promoted for the past several years based on the best estimate and uncertainty methodology. AECL and other Canadian nuclear industry partners developed the so-called Best Estimate Analysis and Uncertainty (BEAU)

methodology for potential use in licensing applications for CANDU nuclear reactor power plants in Canada and abroad. The BEAU methodology is consistent with the CSAU developed in the early 1990s in the US.
The CNSC has stated that the conservative approach in licensing safety approach is not a licensing requirement in Canada, and that best estimate methods are optional. The CNSC made the following generic observations related to use of BEAU approaches in Canada [70], which are quoted here:

- it may provide for more operational flexibility without compromising safety, but it requires extensive justification,
- while it reduces the role of empirical ("engineering") judgment, it emphasized the role of level of knowledge and the need for high accuracy,
- it does not eliminate the need for conservative assumptions,
- it motivates a more dynamic and pro-active approach in both safety analysis area and operation; and
- it requires an extensive compliance program which should incorporate activities related to periodic measurements and verification and validation of computer codes, as well as update and/or confirmation of uncertainties and error allowances.

The CNSC has concluded [70] that BEAU-like methods have been applied in Canada for some time in support of safety and licensing calculations, particularly for the CANDU regional overpower protection (ROP) analysis, which is intended to limit the magnitude of operational fuel power transients especially during refuelling and power manoeuvring.
Recently, the CNSC conducted a research project in which guidelines for safety assessment of applications using best estimate and uncertainty methods for CANDU Nuclear Power Plants were developed [71]. It is expected that the BEAU methodology will address and meet the intent and purpose of these Guidelines in any formal application.

3.2.4. State Office for Nuclear Safety, Czech Republic (SÚJB)

Czech Republic operates 4 VVER-440 units and 2 VVER-1000 units. Their safety analyses are performed with advanced best-estimate computer codes of RELAP, ATHLET, CATHARE type which were developed for western PWRs. Up to now, these codes, while applied for licensing purposes, were used with conservative boundary and initial conditions which required a number of sensitivity analyses. The current state of uncertainty analysis methods and their incorporation into computer codes utilized for thermo hydraulic computations are best presented in the OECD study prepared following recommendations of Committee of the Safety of Nuclear Installations with the purpose to forward development of advanced thermo hydraulic computer codes [49].
Under preparation is a proposal of the methodical procedure to be applied for thermo hydraulic analyses of some selected initiating events for VVER-440/213 and VVER-1000/320 reactors, which takes into account the mentioned trends and especially OECD recommendations. Considered is, for instance, application of this method for the evaluation of such events as "leak on the secondary side", SB and LB LOCA, MSLB, using uncertainty analysis of the input data and computer code models (GRS and IRSN method). These analyses will be carried out with the objective of demonstrate safety of nuclear power plants with VVER reactors, complying with the following criteria:

– Fuel integrity preserved.
– Primary circuit integrity preserved.
– Radiological consequences evaluated.

3.2.5. French Nuclear Regulatory Body (ASN)

There is also in France a clear and increasing tendency in the licensing process, to use Best Estimate codes in the analysis of safety cases. In the overall safety evaluation which includes in France the analysis of the specific assumptions, the use of the code itself and the comparison with the safety criteria, some specific cases have been analyzed using the French Best Estimate code CATHARE. These cases have been described [69].

They cover mainly some small and intermediate break studies and analyses of emergency procedures. The best estimate approach will certainly be generalized for the future EPR reactor.

A more systematic approach for licensing will have then to be defined. In support to these trends in the safety analysis area, several research programs have been started in France:

– uncertainties evaluation NEA/CSNI/R(99)10 methods (IRSN approach)
– the development of statistical tools
– the evaluation of the uncertainties of elementary individual physical models.

Electricité de France and Framatome have developed an accident analysis method based on the use of realistic computer codes called the "Deterministic Realistic Method" (DRM). Its principle is based on quantification of the calculation uncertainty, which is taken into account deterministically when the results (target parameters) are compared to the acceptance criteria. To ensure that the value of a target parameter is conservative, a penalization mode is introduced into the realistic model. The penalties are chosen to preserve a realistic response of the code. The DRM was first applied (1997) to LBLOCA for a French three-loop pressurized water reactor. In France and Belgium the principles were approved in the year 2000.

3.2.6. German Ministry of Environment and Reactor Safety (BMU)

German regulation allowed that the state of science and technology is applied in licensing, i.e. the increase of experimental evidence and progress in code development during time could be used. There was no requirement to apply a pure evaluation methodology with licensed assumptions and frozen codes. The thermal-hydraulic system codes became more and more BE based on comprehensive validation. This development was and is possible because the rules and guidelines provide the necessary latitude to consider further development of safety technology [42].

BE codes are allowed to be used in licensing in combination with conservative initial and boundary conditions. However, uncertainty quantification is not required up to now. Since some of the initial and boundary conditions are more conservative compared with those internationally used (e.g. 106% reactor power instead of 102%, a single failure plus a non-availability due to preventive maintenance is assumed, etc.) it has been claimed but not demonstrated that the uncertainties of code models are covered. Since many utilities apply for power increase, calculation results come closer to some licensing criteria.

Efforts are underway in Germany to include uncertainty evaluation in licensing. The German Reactor Safety Commission issued a recommendation to perform uncertainty analysis in loss of coolant accident safety analyses, recently. A more general requirement is included in a draft revision of the German nuclear regulation which is an activity of the BMU.
According to the recommendation of the German reactor safety commission (RSK) to perform safety analyses in licensing, the following deterministic requirements have still to be applied:

- Most unfavourable single failure
- Unavailability due to preventive maintenance
- Break location
- Break size and break type
- Double ended break, 100% through 200%
- Large, medium and small break
- Loss of off-site power
- Core power (at accident initiation the most unfavourable conditions and values have to be assumed which may occur under normal operation taking into account the set-points of integral power and power density control; measurement and calibration errors can be considered statistically)
- Time of fuel cycle.

No limit of safety margins is specified, a licensing limit is sufficient. It has to be assured that a given plant will not exceed the licensing limits. The purpose of using a best BE code and performing an uncertainty evaluation is to provide assurance that the licensing limits will not be exceeded with a probability of 95% or more.

3.2.7. Hungarian Atomic Energy Authority (HAEA)

The US-NRC and IAEA Guidelines for BEPU evaluation are applied. However, no test data are available for large breaks in VVER reactors. An uncertainty analysis will be performed using the ATHLET code and the GRS uncertainty method for a pressurizer surge line break on the PMK test facility [69].
In the frame of the AGNES project different initiating events are considered: DBA, PTS, ATWS. Pessimistic assumptions are applied to bound uncertainties from code model imperfections. Pessimistic moderator density reactivity coefficients were assumed. In addition, a loss of AC power at the occurrence of high cladding temperature was considered. After a review of these conservatism, the reactivity feedback was replaced by 3D reactor physics, the cladding gap conductance, and the engineering factor for the hot assembly was revised.

3.2.8. Lithuanian State Nuclear Power Safety Inspectorate (VATESI)

The Lithuanian regulatory adopts the American rule 10 CFR 50.46, September 1988 revision, that allows the use of realistic evaluation models to calculate the performance of the Emergency Core Cooling System.
In Lithuania the BE approach is successfully applied not only for LOCA but also for reactor transients, reactivity initiated accidents and confinement system response analyses[112]. For the licensing of Ignalina NPP, VATESI requires the use of BEPU and sensitivity analysis, Tab. 4 option 3.
Recently, comparison between BEPU results obtained by GRS method application and partially conservative results using RELAP50MOD3.2 code was done for three cases, which cover the worst postulated RBMK-1500 accident and transients. For

each calculation 60 runs were performed to determine upper bound fuel clad temperature. In general, partially conservative results provide higher peak temperatures. Nevertheless, the partially conservative approach requires proper boundary and initial conditions, which for different phenomena during the accidents can differ. The uncertainty analysis was performed using on the statistical tools based the GRS developed methodology (package SUSA) [93].

3.2.9. Swedish Nuclear Power Inspectorate (SKI)

The SKI traditionally adopts the American rule 10 CFR 50.46, Appendix K using conservative codes with conservative initial and boundary conditions including conservative assumptions [52].

The trend in Sweden is to use more best estimate methods however without taking the effort to perform an uncertainty analysis. Best estimate codes are allowed to be used in licensing in combination with conservative initial and boundary conditions.

Utilities are being encouraged to develop generic and plant specific best estimate methodologies for ASEA-Atom reactors.

3.3. Main problems and limitations in use of best estimate methods for licensing; advantages and disadvantages of the present licensing practice

Recently made advances offer two acceptable options for demonstrating that safety is ensured with sufficient margin: the use of best estimate computer codes either combined with conservative input data or with realistic input data associated with evaluation of uncertainty of results. Before current BE computational tools were available, only conservative licensing calculations with conservative codes were performed.

Pure BE calculations may not give sufficient information about the behaviour of a NPP. The role of uncertainty in BE analysis is fundamental. There are several reasons for the necessity to perform an uncertainty evaluation, among them:

- The code assessment process depends upon data usually measured in small scale facilities and not in full power reactors;
- The models and the solution methods in the codes are approximate;
- Boundary and initial condition are not precisely known;
- User effect can lead to uncertainties in computer models and calculated scenarios.

Consequently best estimate predictions of nuclear power plant scenarios must be supplemented by proper uncertainty evaluations in order to be complete.

At the present time BE methodologies with uncertainty analysis are most widely used for safety analyses performed in the context of transients and design basis accidents (DBAs), where the associated uncertainties related to the understanding and modelling of physical phenomena are rather well defined and limited. The majority of BE applications in licensing are related to LB-LOCA. Beyond design basis accident (BDBA) analyses are currently not part of the licensing processes, however BE analysis for this type of condition are envisaged with the main concern related to core melt prevention. Their application to the BDBAs is still rather limited due to large uncertainties associated with the calculated results and predictions. There are also a larger number of uncertain input parameters than in DBA analysis.

The motivation to use best estimate systems codes and to calculate the uncertainty of the final results is convincing: they can be used for licensing purposes in cases of plant systems modifications and improvements, power up-ratings, core optimizations, maintenance and repairs planning, core cycle extension, etc. The plant operators may therefore fully exploit a large number of techniques to maximize plant operational efficiencies, power output, and plant operational cycles. These capabilities, in turn, enable a utility to operate a plant with minimum cost and downtime.

There is a lack of an established set of specific regulatory requirements applied to the acceptance of a realistic evaluation model used to analyze the LBLOCA. The ACRS of US NRC emphasizes in an ACRS Letter Report [30, 111], *"We perceive a need for the staff to be more specific about what are acceptable methods of deriving and expressing the uncertainties in codes and how these methods are to be used in the regulatory context ... . The staff should revaluate the design specifications for the outputs of codes and their relationship to present and anticipated regulatory requirements"*.

It is also recognized that in several countries there are initiatives in developing these requirements and in redefining the LBLOCA [46].

Some specific points related to the application of BE in the licensing process are:

- The safety limits are not going to be changed. Rather, the use of BE codes will change in a direction of establishing the limits and "measure" the margins,
- In the U.S.A., the "old" criteria are still used by the applicants, but BE methods are allowed supported by a PRA evaluation,
- A "common approach" of France and Germany differs in details, e.g., in France one code is used by both regulator and utilities, while in Germany different codes are used to have a better chance to uncover common cause failure,
- Public scrutiny of regulators forces to keep conservative criteria, but BE evaluation may be used ,
- Assessment of codes against validation matrices helps to estimate uncertainties,
- So far, the performed comparisons between CFD and 1-D codes lead to no conclusions.
- With strong emphasis on physics, the use of computer codes may give some valuable input and support to decision making and licensing, in addition to the use of the available research results in a distilled form,
- Nodalization is equally important as modelling. A coarse nodalization may give a correct average, but leads to incorrect estimate of local conditions (classic examples: void and two phase flow regimes) [69]. Anyway, a fine nodalization does not assure by itself an improved meaning of results.
- They in general require a significant initial generic investment and therefore may be considered as more complicated and costly
- The benefit may be for specific plant
- Perception of licensing risks, very large investments at stake
- Some competence structure not favourable
- The uncertainty in the realistic evaluation model is to be quantified and considered when comparing the results of the calculations with the applicable limits, so that there is a high probability that the criteria will not be exceeded.

Some benefits are considered to be:

- Better safety assessment with more focus on real risks
- Identify risk dominating sequences, possible to use engineering judgment
- Improve national competence
- Licensing codes can be assessed against plant data and experimental data from integral test facilities
- Licensing codes can be used to improve understanding of plant behavior and help to identify potential or latent failures
- The BE analyses provide a good view of existing margins or limits on nuclear power plant operation, consequently offering a basis for possible parameter optimizations [37].

3.4. Available future alternatives, ways for more regular use of best estimate methods in licensing

There is a very strong need to use BE codes instead of conservative codes in licensing. Presently, some countries prefer bounding approach by using best-estimate codes with conservative parameter values, conservative boundary and initial condition assumptions to evaluate the margin to licensing limits. This is done without the use of a detailed uncertainty evaluation analysis, due to cost effectiveness considerations and time requirements for the necessary analysis. In this sense, existing uncertainty methods are found to be useful, but there is a strong need to apply to different transient types. Since the uncertainty methods which have been developed are very rigorous, the challenge is to come up with a simplified method that can be used with confidence in the licensing process. A full best-estimate approach with uncertainty analysis may be considered in the future. In addition, there is a transfer of uncertainties when computer codes are coupled from different disciplines, e.g., thermal-hydraulic, neutron, and containment codes. These "transfer of uncertainties" are known presently, but the problem is not addressed in detail.

The trend in accident analysis continues to move in the direction of best estimate approaches rather than conservative analysis; however, conservative analyses are still used in many cases.

The reasons for moving from a conservative approach to best estimate plus uncertainties are related to:

- Increasing the margin for operation: to up-rate nominal power, to allow different reload options leading to higher burn-up, etc.
- Obtaining a more accurate prediction (including timing of events) of the actual behaviour of the plant in hypothetical accident scenarios. This certainly allows the possibility of optimizing emergency operating procedures.

There are already a number of IAEA guidance documents dealing with the qualified use of advanced computational tools for safety analysis.

Various options exist for combining computer code types and input data for safety analysis, as reported in § 2.2 in Tab. 1 here repeated as Tab. 4.

Option	Computer code	Availability of systems	Initial and boundary conditions
1	Conservative	Conservative assumptions	Conservative input data

2	Best estimate	Conservative assumptions	Conservative input data
3	Best estimate	Conservative assumptions	Realistic input data with uncertainties[#]
4	Best estimate	PSA based assumptions[*]	Realistic input data with Uncertainties[#]
[#] Realistic input data is used only if the uncertainties or their probabilistic distributions are known. For those parameters whose uncertainties are not quantifiable with high confidence, conservative values are to be used. [*] In lieu of PSA based assumptions, reliability based calculations may also be employed for quantifying the availability of systems.			

Tab. 4 - Various options for combination of a computer code and input data

The Option 1 approach is often called very conservative or Appendix K (of 10 CFR 50, USA) analysis in the case of a LOCA. Many regulatory bodies prescribe the conservative models / correlations to be used for safety analysis. In the case of Appendix K, code models are prescribed, for example, the Baker-Just correlation for fuel clad oxidation, etc.

The Option 2 approach is similar to the Option 1 approach except for the fact that a best estimate computer code is used instead of a conservative code. However, it must be noted that in some countries Option 2 is referred to as conservative analysis. In USA the CFR does not permit the use of this option. In this document, conservative analysis always implies the use of a conservative code.

In the Option 3 approach, the initial and boundary conditions of parameters are taken as realistic with consideration of their uncertainties. In other respects, i.e. use of computer codes and assumptions regarding availability of systems, it is the same as Option 2.

Option 4 is not yet a part of current licensing practices. This option is connected to future developments in risk informed regulations.

The topics that are needed to complete a licensing process and require the application of a computational tool can be distinguished various groups of relevant topics:

- Initial or boundary values,
- Physical phenomena and modelling,
- System availability,
- Other licensing requirements.

Within the first two groups it is necessary to make a distinction between:

a) parameters for which the uncertainty can be calculated and,

b) parameters for which the evaluation of uncertainty is difficult.

As a general rule for case a), the nominal value of the parameter may be taken for the reference calculation provided that the related uncertainty is justified, while for case b) conservative values should be adopted. In some particular cases a regulatory body may still specify the use of conservative values for case a) parameters.

The following items may be considered as examples of the aforementioned groups:

GROUP 1

1) Initial power.

2) Decay power.
3) Linear power (i.e. maximum linear heat generation rate (LHGR)).

GROUP 2

4) Gap model.
5) Hydrogen production and clad embrittlement.
6) Clad burst and fission product release, transport and deposition.
7) Ballooning including cool-ability after ballooning.
8) Radiation heat transfer.
9) Modelling of fuel (i.e. pellet deformation and cracking) including the consideration of high burn-up and MOX fuel.

GROUP 3

10) Consideration of single failure criterion.

GROUP 4

11) Mechanical loads.
12) Long term cooling including the debris effect upon core cooling and cavitation of ECCS pumps.

4. NEED FOR AN AGREED UPON SPECIFICATION OF BEST ESTIMATE APPROACHES IN VIEW OF A WIDER ACCEPTANCE OF THE BE APPROACH BY LICENSING AUTHORITIES

The topics, considered in the previous chapters give evidence of the efforts applied by scientific and technical community to implement the BE methods in the licensing process. The analysis has shown that

- the industry is more and more focusing on development of new licensing BEPU;
- the research is focusing on the codes with internal assessment of the uncertainty.

Notwithstanding the guidance offered by USNRC Regulatory Guide 1.157 [13], the subjective choices available to the analyst in performing a BE calculation are still many. This is true even if the computer program used is univocally identified. As a consequence, it has to be expected that many authorities require an ad-hoc uncertainty analysis as a necessary complement of any BE calculation. If the BE procedure for any computer program were more precisely defined, probably it could be expected that the requirement of an individual uncertainty analysis could be removed in many cases. In fact, uncertainty (safety) margins could probably be defined for any BE calculation performed using a specific computer code in a qualified environment (e.g. user qualification, computer).

The aim of this chapter is to offer an up-to-date discussion basis for the identification of a standard BE procedure to be used with a specific, widely used code (e.g. RELAP5/MOD3), having in mind the objective of a consensus by licensing bodies on a “jet to define” set of safety (uncertainty) margins to be agreed upon for calculations performed using that procedure.

4.1. General remarks

The term BE can be used in various ways in nuclear safety analysis. To apply the BE analysis into the licensing process some procedures have to be discussed by the international scientific community. Such process could give a better definition of the meaning of BE analysis and help to overpass the resistances of who does not completely believe in the usefulness of such methodology.

The term BE is usually used as a substitute for “realistic”, indicating that the calculation attempts to predict realistic plant response applying current knowledge and modeling of physical phenomena involved.

It is useful to correlate the definition of BE with some concepts in probabilistic evaluations.

The concepts of MEDIAN, MEAN and MODE of a probability distribution of a variable are particularly useful.

The meaning of the three terms and their definition is illustrated by the Fig. 18 and the following mathematical definitions.

The MEDIAN (MD) value of v is defined as the value for which half of the values lay to the left and half of the values lay to the right ("central" value); MD can be identified by the following identity :

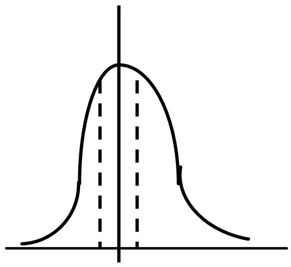


Fig. 18 – Probability distribution

$$MD = \int_{0}^{V_{MD}} p(v)dv = \int_{V_{MD}}^{\infty} p(v)dv \qquad (10)$$

The MEAN (ME) value of v is simply the mean (average) value of the distribution; it can be identified by:

$$ME = \int_{-\infty}^{\infty} vp(v)dv \qquad (11)$$

The MODE (MO) value of v is simply the value corresponding to the maximum probability of the distribution.
It is here proposed, with reference to the concepts of MO, ME and of MD, to identify a *BE evaluation with the MO of the distribution of all possible evaluations which could be made*. The evaluation of the uncertainty of the distribution takes care of the possible difference of results from the most probable one.
The term BE can be either applied to a simple calculation, as, for example, to a system code calculation of a reactor transient or accident, or, more in general, to the overall evaluation of a plant situation of interest, as, for example, the evaluation of the PTS (Pressurized Thermal Shock) danger on a plant.
In this document, the term BE is intended to be applied to the overall evaluations of situations of interest and not only to transient/accident calculations. With this assumptions, issues as the Single Failure Criterion fall within the scope of the work.

4.2. Examples of technical issues to be studied and first proposals for a discussion on best estimate specifications.

Hereafter, in order to start discussion an agreed upon procedure for BE analyses, some examples of main technical issues to be studied are presented.
Since this discussion tends to be practically applicable, some boundary assumptions have to be made in order to limit the field of study.
Here it is assumed:
1) the basis for the specification of BE analyses is the USNRC Regulatory Guide1.157 [13];
2) consideration of computer codes for system analysis are, moreover, confined to the widely diffused RELAP5/MOD3.3 code;
3) a PWR is here taken into consideration, unless otherwise indicated. Therefore, the considerations which follow are not addressed to the simulation of Integral Test Facilities.

4.2.1. Initial stored energy of the fuel

According to the NRC guide [13], BE fuel models is considered acceptable provided the models include essential phenomena that affect the evaluation of the thermal conductivity of the pellet.

The values for thermal conductivity, for fuel and for its heat capacity here suggested, are the most recent INSC values (see Fig. 19 and Fig. 20, downloadable from the site www.insc.anl.gov/matprop/). This parameter is very important for the evaluation of peak cladding temperature [42].

The figures consider also the porosity effect of the fuel.

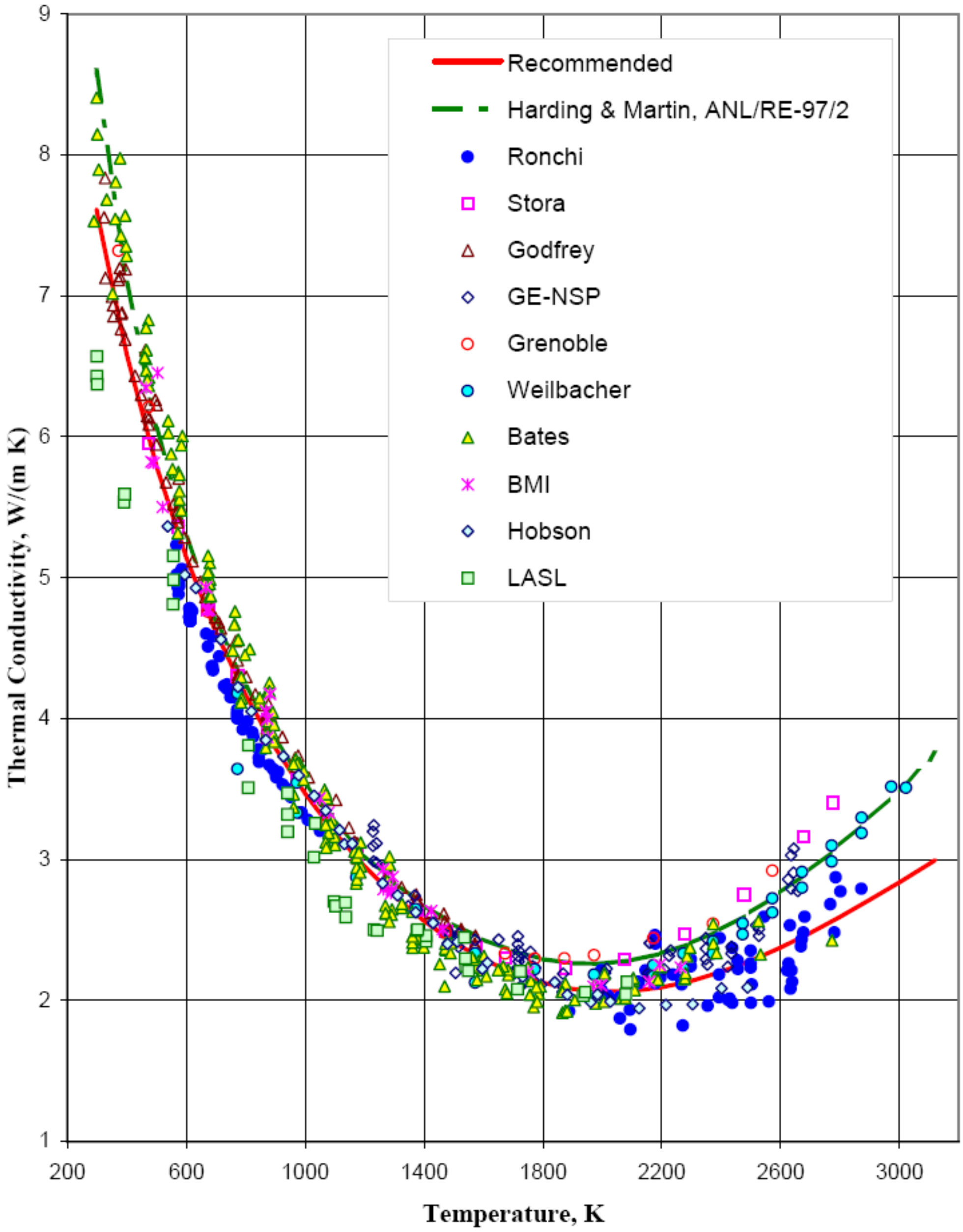


Fig. 19 – Heat conductivity: comparison of recommended equation with previous recommendation and data for 95% dense UO_2

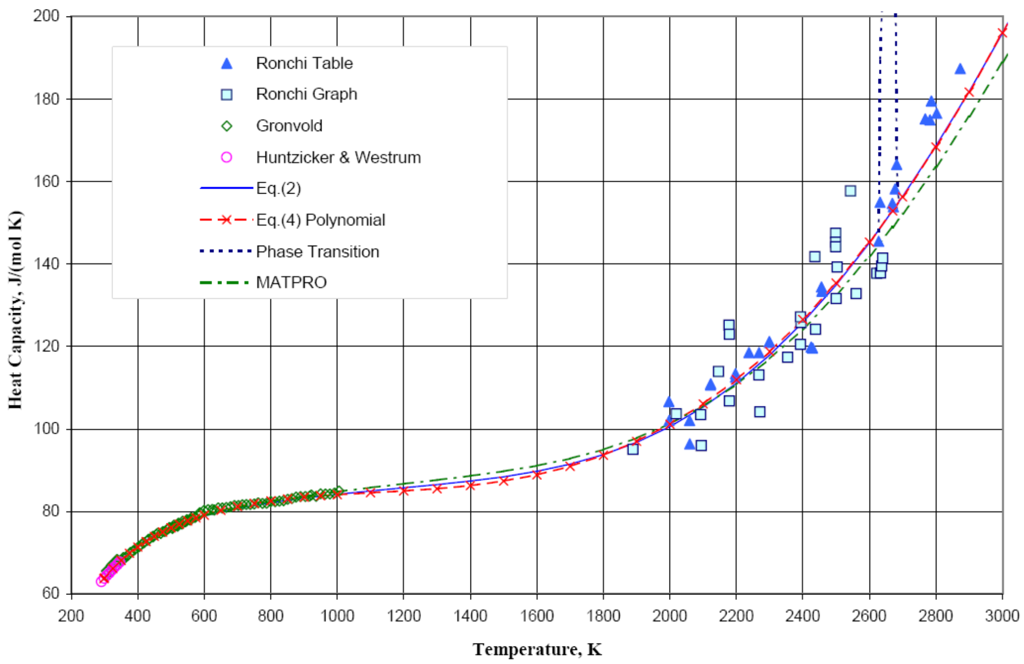


Fig. 20 – UO_2 Heat capacity: comparison of recommended equation and data with MATPRO equation

4.2.2. Fuel – cladding gap conductance

RG. 1.157 recommends the use of Lanning Model [83]. This model is substantially incorporated in RELAP5/MOD3.3 and it can be chosen by selecting the proper code options (see RELAP Manual Vol. II App. A, Section a 10.3 Gap Data[7]).

The model incorporated in RELAP has the following three assumptions/simplifications:

– the fuel-to-cladding radiation heat transfer, which only contributes significantly to the gap conductivity under conditions of clad ballooning, is neglected unless the cladding deformation model is activated ([7] Section 4-14 of Manual Vol. I);
– the minimum gap size is limited such that the maximum effective gap conductivity is about the same order as that of metals;
– the direct contact of the fuel pellet and the cladding is not explicitly considered.

The volumetric heat capacity and thermal conductivity of the fuel rod materials, except for the thermal conductivity of the gap gas, must be supplied by the user.

For the computation of the gas thermal conductivity, the user is required to provide the gas composition in terms of the mole fractions of seven common gases included in the model. The properties for determining material thermal expansion and elastic deformation are calculated from permanent data within the code and no user input is needed. The user, however, should be aware that these properties are computed under the assumption that the fuel material is uranium oxide and the cladding material is zircaloy. The properties of UO_2 and zircaloy along with gas conductivity are taken from MATPRO-11 (Rev. 2) [86].

As far as gap conductance is concerned, which is very important for PCT calculations, it is here suggested that the RELAP model is used. A comparison with other models (e.g. [83] and [85]) suggests a range of error [14], in PCT for a large LOCA, of the order of tens of K according to the specific model used, that is completely reasonable and acceptable.

The difference on the analysis performed gives a variation of 10% that corresponds to a variation of 10 – 20 K of the BEMUSE results for the PCT [14].

4.2.3. Cladding thermal conductivity and heat capacity

The values for cladding thermal conductivity and for its heat capacity, here suggested, are the most recent INSC suggested values (available at the website www.insc.anl.gov/matprop/).

This parameter is less influent using RELAP5 3.2 because the code does not consider the change of phase of the fuel.

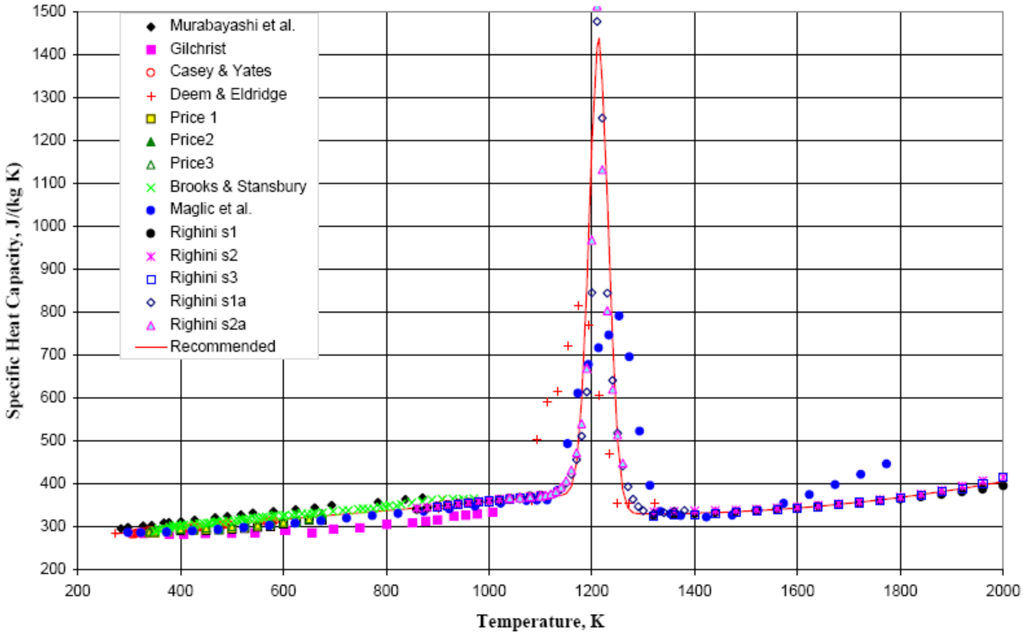


Fig. 21 – Recommended fit to zircaloy-2 heat capacity data

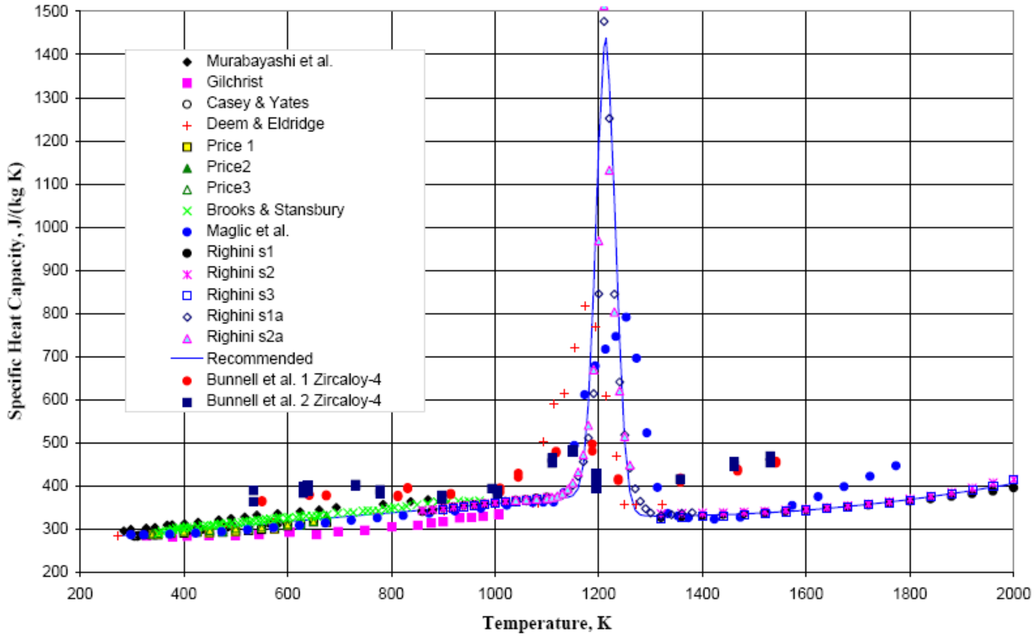


Fig. 22 – Comparison zircaloy-2 and zircaloy-4 heat capacities

4.2.4. Sources of Heat during a LOCA accident and other accidents

In this section, the decay heat power and the gamma distribution of the decay power is considered.

Decay heat is the heat produced by the decay of radioactive fission products after a nuclear reactor has been shut down. It is the principal reason of safety concern in LWR. It is the source of 60% of radioactive release risk worldwide.

In a nuclear reactor, the fission of heavy atoms, such as isotopes of uranium and plutonium, results in the formation of highly radioactive fission products. These fission products radioactively decay at a rate determined by the amount and the type of radioactive nuclides present.

The amount of radioactive materials present in the reactor at the time of shutdown is dependent on the power levels at which the reactor operated and the amount of time spent at those power levels.

Typically, the amount of decay heat that will be present in the reactor immediately following shutdown will be roughly 7% of the power level that the reactor operated at prior shutdown. A reactor operating at 3600 MW will produce 252 MW of decay heat immediately after shutdown; this demonstrates the importance of decay heat if no cooling is present.

Decay heat decreases to about 2% of the pre-shutdown power level within the first hour after shutdown and to 1% within the first day. Decay heat will continue to decrease, but it will decrease at a much slower rate. Decay heat will be significant weeks and even months after the reactor is shut down. Failing to cool the reactor after shutdown results in core heat-up and possibly core meltdown (i.e. Three Mile Island 2).

Moreover, radioactive isotopes will eventually decay to stable material. Some isotopes decay in hours or even minutes, but other decay very slowly.

The decay heat power comes mainly from five sources:

- Unstable fission products, which decay via α, β^-, β^+ and γ ray emission to stable isotopes.
- Unstable actinides that are formed by successive neutron capture reactions in the uranium and plutonium isotopes present in the fuel.
- Fissions induced by delayed neutrons.
- Reactions induced by spontaneous fission neutrons.
- Structural and cladding materials in the reactor that may have become radioactive.

Heat production due to delayed neutron induced fission or spontaneous fission is usually neglected. Activation of light elements in structural materials plays a role only in special circumstances, and is usually excluded from decay heat analyses.

To summarize, after the shutdown of a nuclear reactor, the radioactive decay of fission products, actinides and activation products produces heat that have be removed from the system. Its removal is achieved by Emergency Core Cooling Systems and heat exchangers. Those highly redundant systems are designed to provide sufficient amount of makeup water for several days without operator intervention.

4.2.4.1. Decay heat power

The RG-1.157 considers the decay heat power in the paragraph 3.2.4 Fission Product Decay Heat**:**

- The heat generation rates from radioactive decay of fission products, including the effects of neutron capture, should be included in the calculation and should be calculated in a best-estimate manner.
- The energy release per fission (Q value) should also be calculated in a best-estimate manner.
- BE methods will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.
- The model in Reference [89] is considered acceptable for calculating fission product decay heat.
- The values of mean energy per fission (Q) and the models for actinide decay heat should be checked against a set of relevant data.

4.2.4.1.1. *<u>Value of the ANS standards: decay heat curves</u>*

Data from several experiments were examined in the 1960s to provide an accurate basis for predicting fission product decay heat power. In 1971, the results were adopted by the American Nuclear Society (ANS) to assemble the first decay heat standard (ANS-5.1/N18.6). The ANS-5.1/N18.6 draft later, updated to the ANS-5.1-1973 draft standard, contained a single curve to represent all uranium-fuelled reactors.

New measurements in the seventies led to a new compilation that resulted in the adopted standard in 1979 (ANS-5.1-1979). The standard was developed to fulfil a need for evaluations of fission reactor performance dependent upon knowledge of decay-heat power in the fuel elements. Since the approval of the standard, new measurements of decay heat have been published. In addition, improved nuclear data bases have resulted in more precise summation calculations of decay heat. ANS-5.1-1979 was updated by ANS-5.1-1994. The latest version of the standard is ANS-5.1-2005.

The ANS-5.1 standard provides bases for determining the shutdown decay heat power and its uncertainty following shutdown of Light Water Reactors (LWR). The information in this standard can be used in the design, performance evaluation, and assessment of the safety of LWRs. It can be used as the basis of determining fission product decay heat power.

The ANS-5.1 standard for decay heat generation in nuclear power plants provides a simplified mean of estimating nuclear fuel cooling requirements that can be readily programmed into computer codes used to predict plant performance. The ANS-5.1 models the energy release from the fission products of ^{235}U, ^{238}U and ^{239}Pu using a summation of exponential terms with empirical constants. Corrections are provided to account for energy release from the decay of ^{239}U and ^{239}Np and for the neutron activation of stable fission products. Although the empirical constants are built into the standards, certain data inputs are left to the discretion of the user. These options permit accounting for differences in power history, initial fuel enrichment and neutron flux level.

Decay heat power from other actinides and activation products in structural materials and fission power from delayed neutron-induced fission are not included in this standard.

The main options offered by RELAP for core decay power input are the following ones:

- choice between ANS Standards 1973, 1979-1, 1979-3, 1994-4
- power history before shut-down

– an overall multiplication factor (equal to 1.2 for licensing calculations according to Appendix K to CFR 50-46[8]), intended to cover uncertainties.

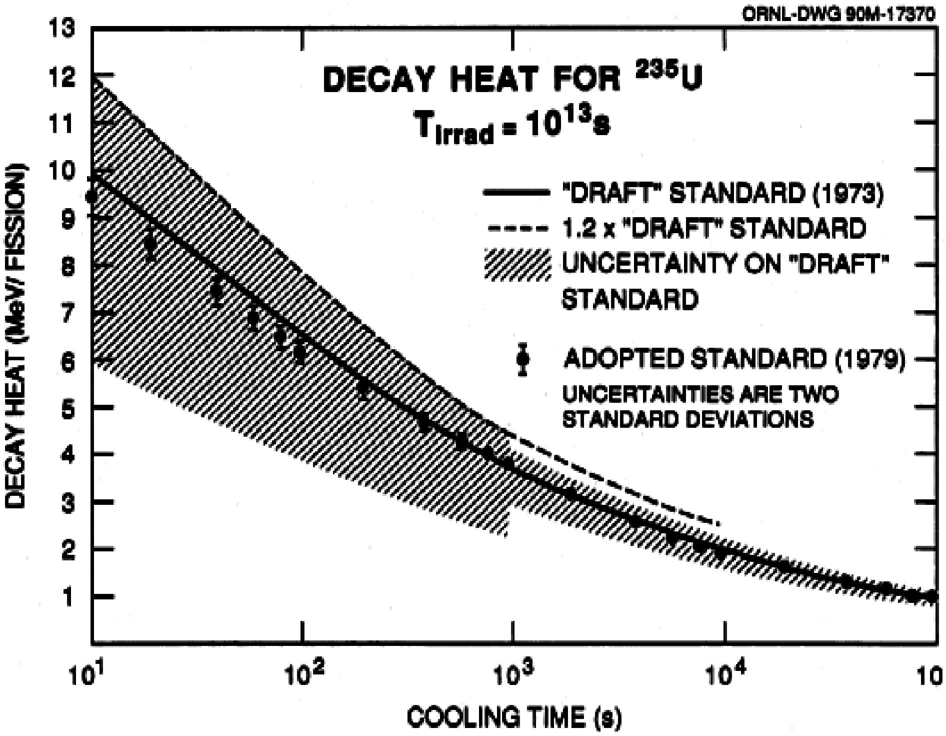


Fig. 23- Comparison of revised standard F(t,∞) for ^{235}U (1979) with 1973 standard[90]

4.2.4.1.2. *Application to the SBO and SBLOCA*

The analysis of the decay heat is applied to a generic VVER 1000, for two accident conditions: SBO and SBLOCA for a time of 11.000 s. The scheme and the description of the nodalization are presented in Annex IV.

CASE	Transient 1	Transient 2	DH POWER
1	**SBO 1**	**SBL 1**	ANS79-1 x 1.2 ; 11000 s
2	**SBO 2**	**SBL 2**	ANS79-1 x 1.0 ; 11000 s
3	**SBO 3**	**SBL 3**	As 2 but with neutrons delayed (%)/100; 11000 s
4	**SBO 4**	**SBL 4**	As 3 but with ANS79-3
5	**SBO 5**	**SBL 5**	As 3 but with ANS94-4
6	**SBO 6**	**SBL 6**	As 5 but with ANS73
7	**SBO 7**	**SBL 7**	As 5 but with a specific power history (4m90%, 1m20%, 4m80%, 4m80%, 4m70%, 3m90%, 2m60%)
8	**SBO 8**	**SBL 8**	As 5 but with a different distribution of power among the fissiles
9	**SBO 9**	**SBL 9**	As 7 but with constant neutrons delayed x100 and neutrons delayed (%) usual (148,4); 11000 s

Tab. 5 - Decay heat transient cases analysed

The cases analyzed are indicated in the Tab. 5:

- Case 1 uses the ANS79-1 with multiplication factor 1.2 (conservative calculation for actual licensing)
- Case 2 considers the real power (multiplication factor 1)
- Case 3 aim to see the effect of the neutrons delayed
- Case 4 to case 6 evaluate the effect of the decay heat ANS standard curves
- Case 7, using ANS-94-4, analyzes the effect of the power history
- Case 8, using ANS-94-4, analyzes the effect of the power distribution among the fissiles
- Case 9, using ANS-94-4, analyzes the effect of the delayed neutrons.

The calculations are performed using RELAP5 MOD 3.3

4.2.4.1.3. *Results of the analysis*

The analysis of the results is focused on the DH power produced and the hot rod temperature. Additional results are reported in Annex IV.

4.2.4.1.4. *Decay heat power*

No effect has the choice of the decay heat ANS standard on the total heat power for the SBO, as from Fig. *24* and Fig. *29*. It becomes sensitive when the scram is due to a phenomena internal to the reactor, i.e. SB-LOCA, Fig. *25* and *Fig. 27*.

When the effect of the decay heat on the choice of the standard or of the decay heat curve is analyzed and compared, the influence become evident as from the following figures (Fig. *29* ÷ Fig. 38).

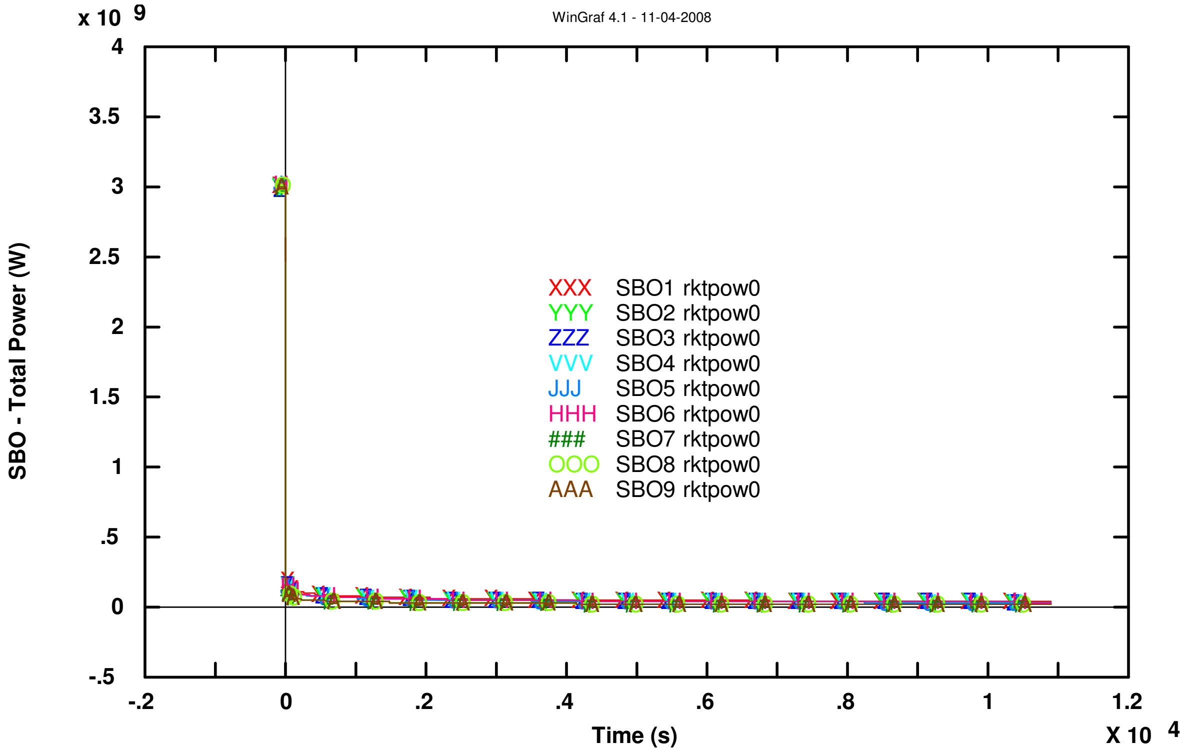


Fig. 24- Total heat power produced in the transient 1 (up to 11000 s)

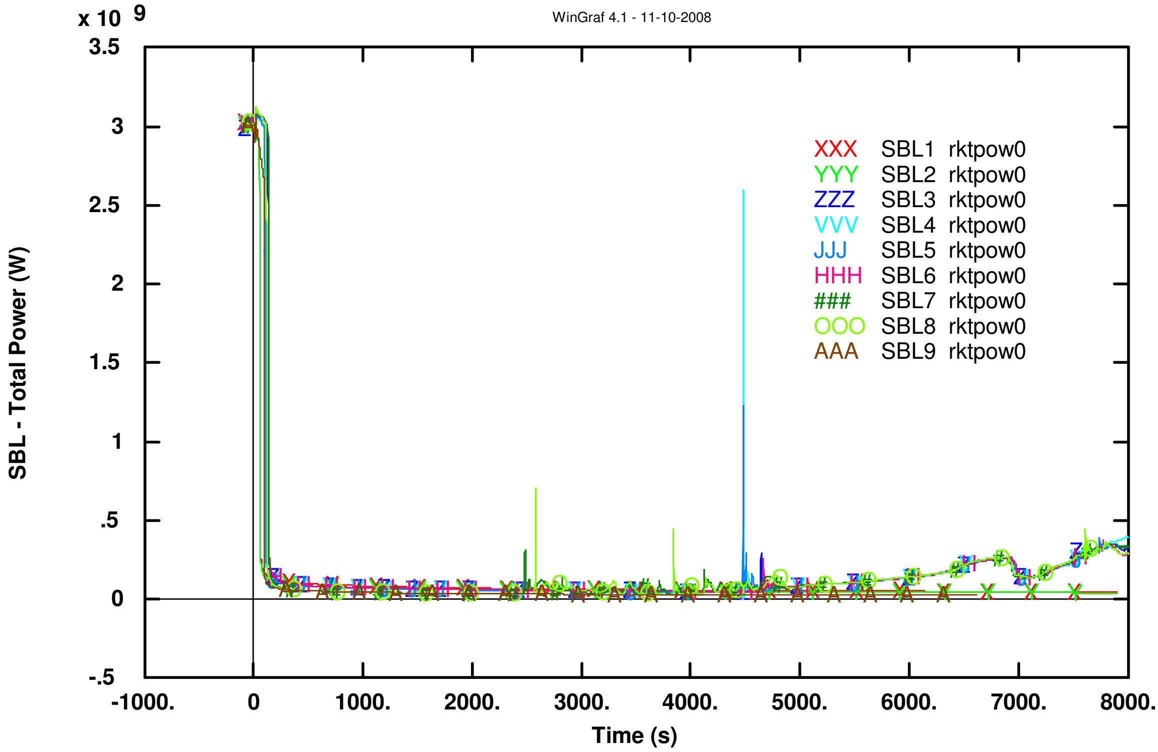


Fig. 25- Total heat power produced in the transient 2 (up to 8000 s)

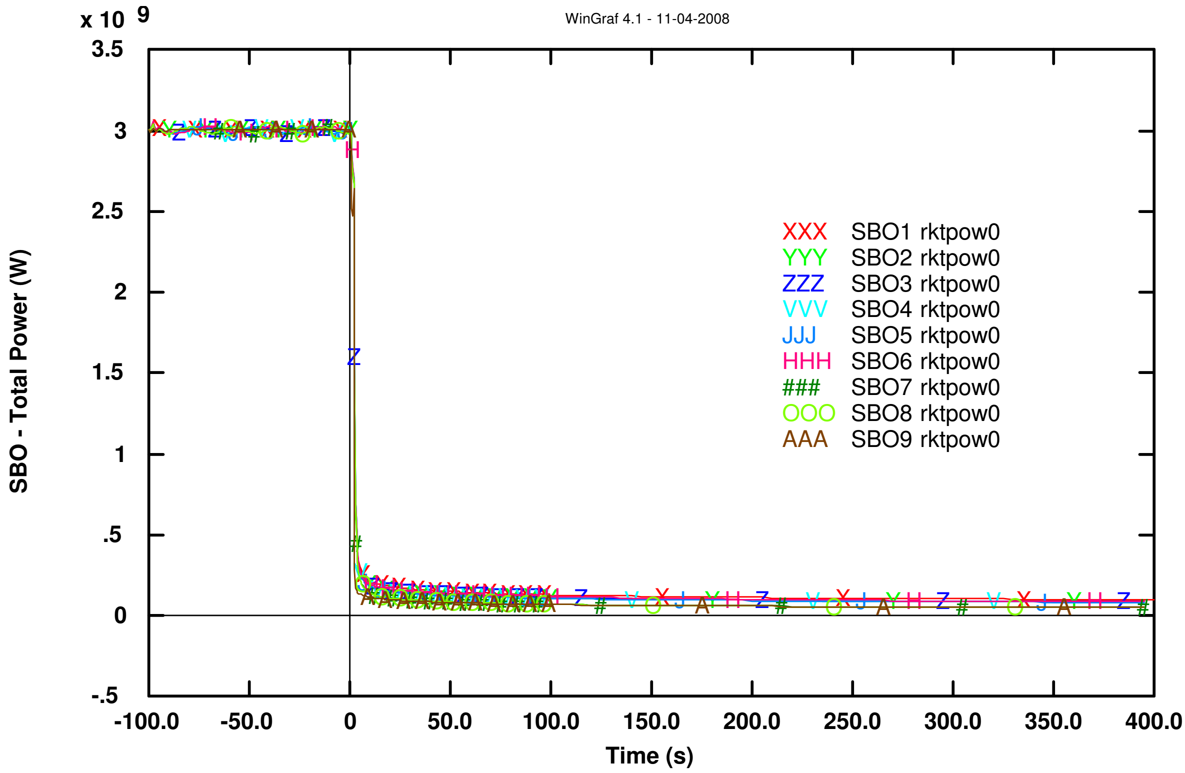


Fig. 26- Total heat power produced in the transient 1 (up to 400 s)

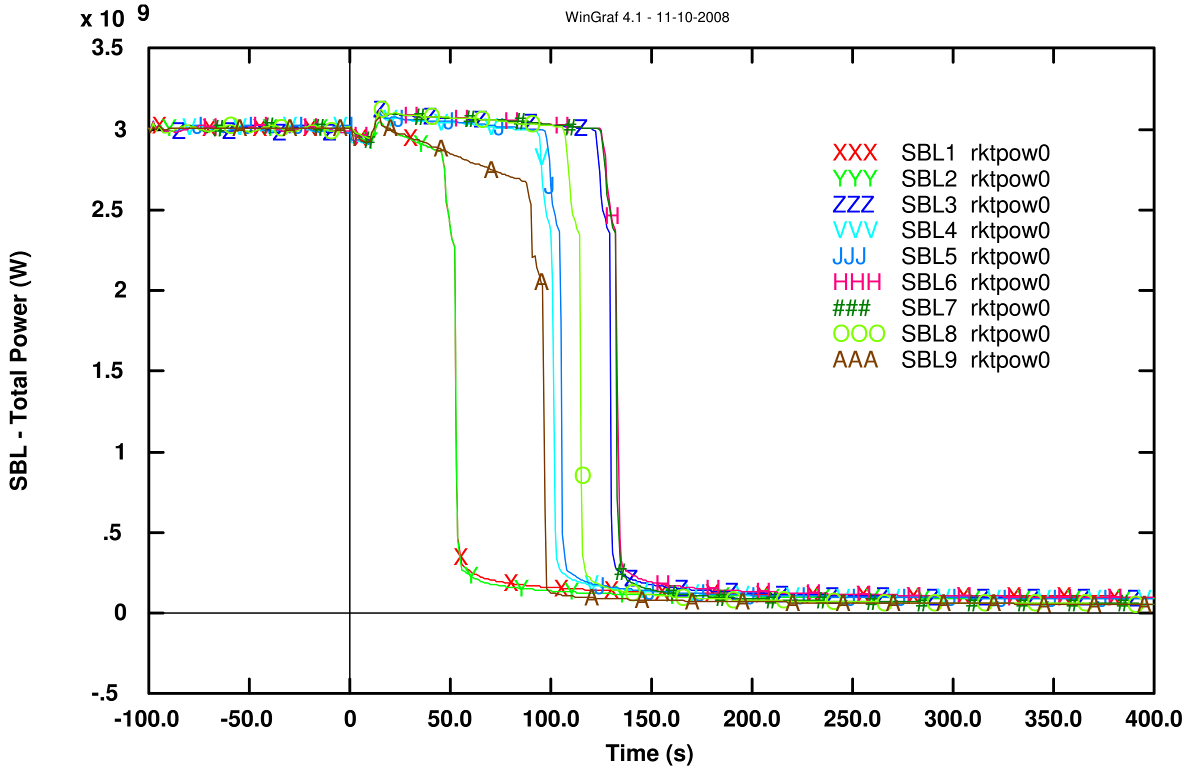


Fig. 27- Total heat power produced in the transient 2 (up to 400 s)

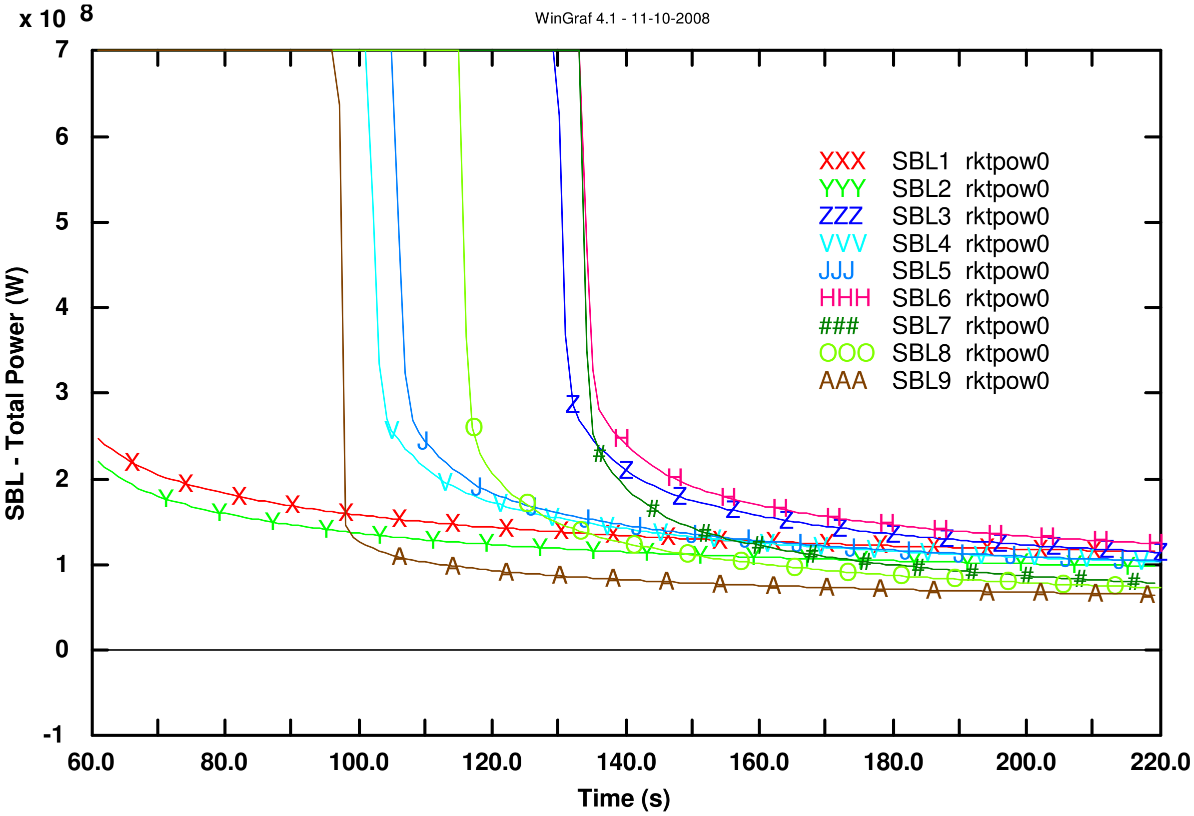


Fig. 28- Total heat power produced in the transient 2 (up to 220 s)

From *Fig. 27* and *Fig. 28* it is possible to see that there is a difference of 100s among the cases analyzed. In particular, between ANS 79 , SBL1, licensing and ANS 94-4, SBL5, there is 60s.

Fig. *29* shows the comparison of the total decay heat power for the station black out transient.

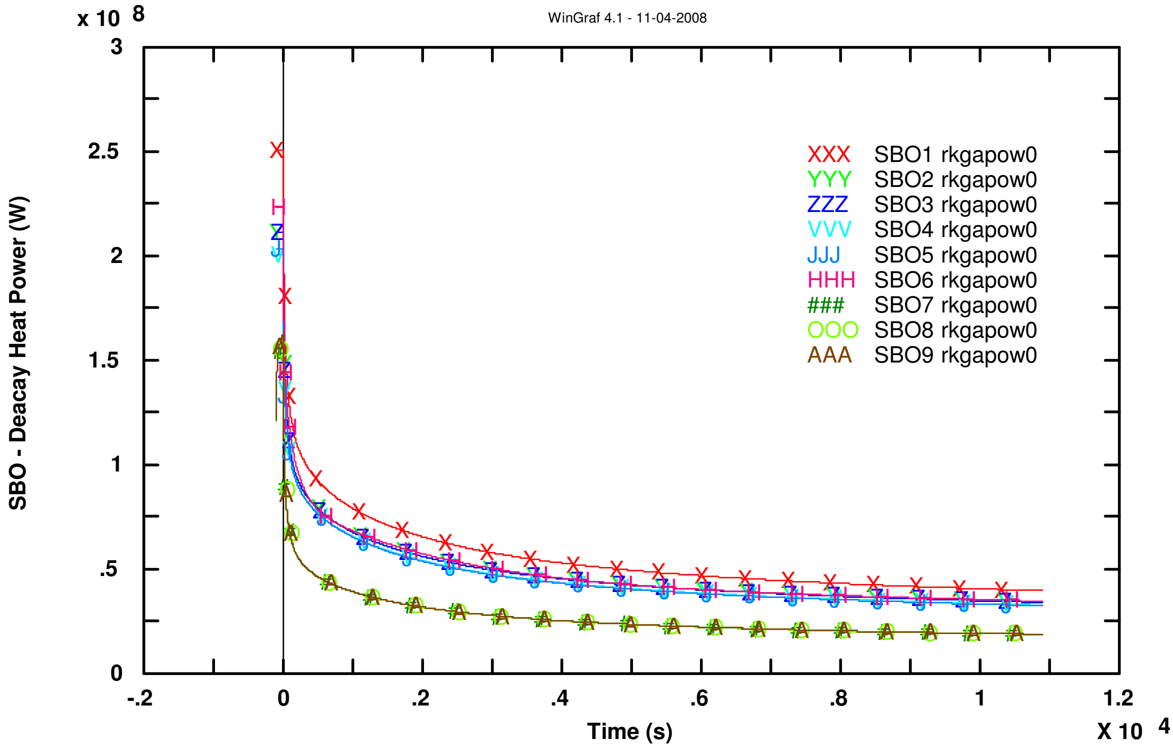


Fig. 29- Comparison of the total decay heat power produced in the transient 1 (up to 11000 s)

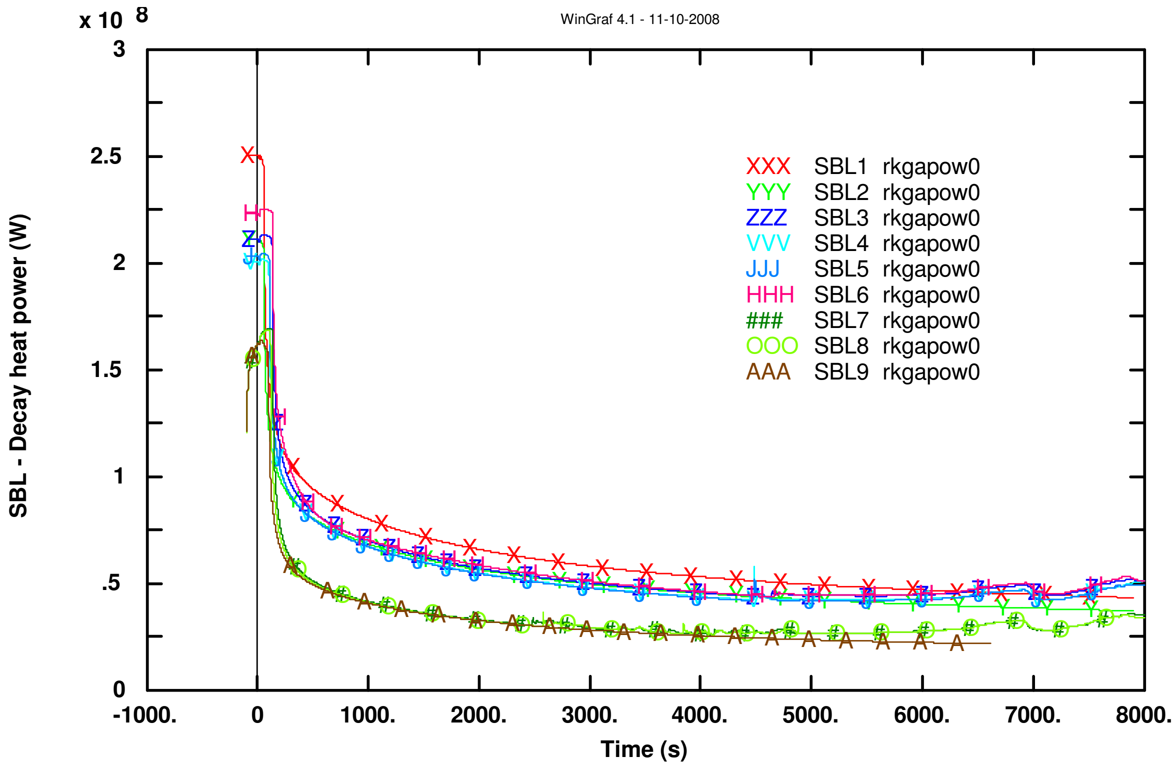


Fig. 30- Comparison of the total decay heat power produced in the transient 2 (up to 8000 s)

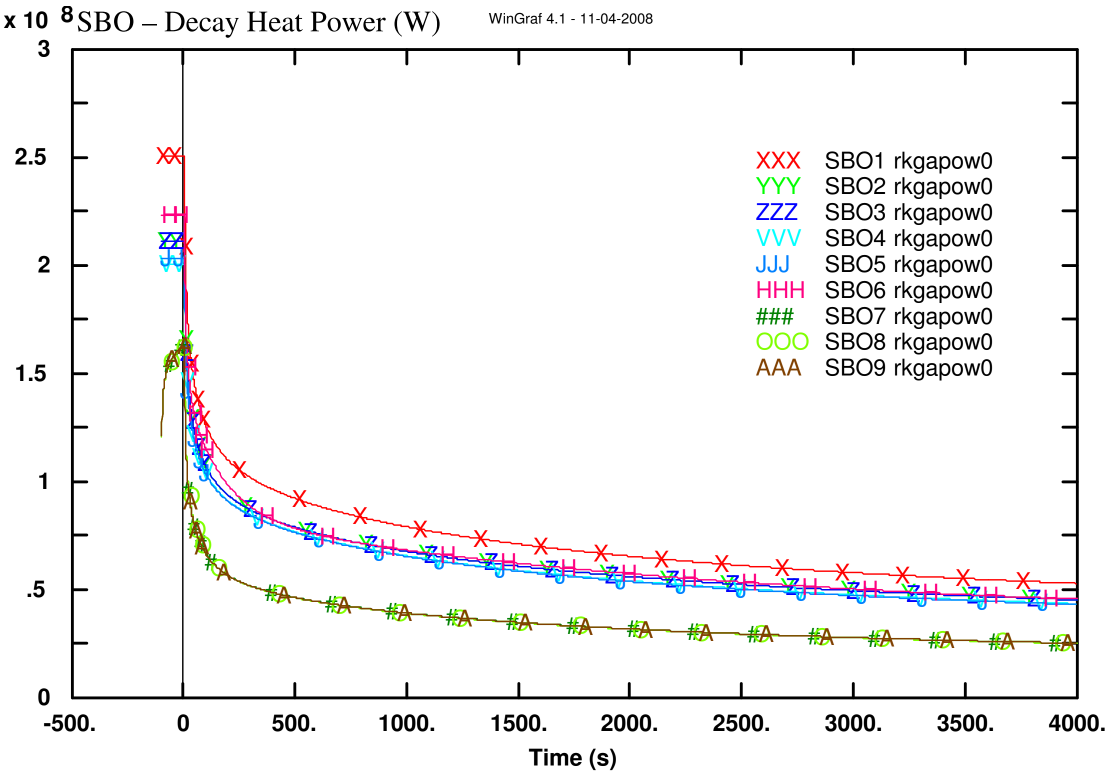


Fig. 31- Comparison of the total decay heat power produced in the transient 1 (up to 4000 s)

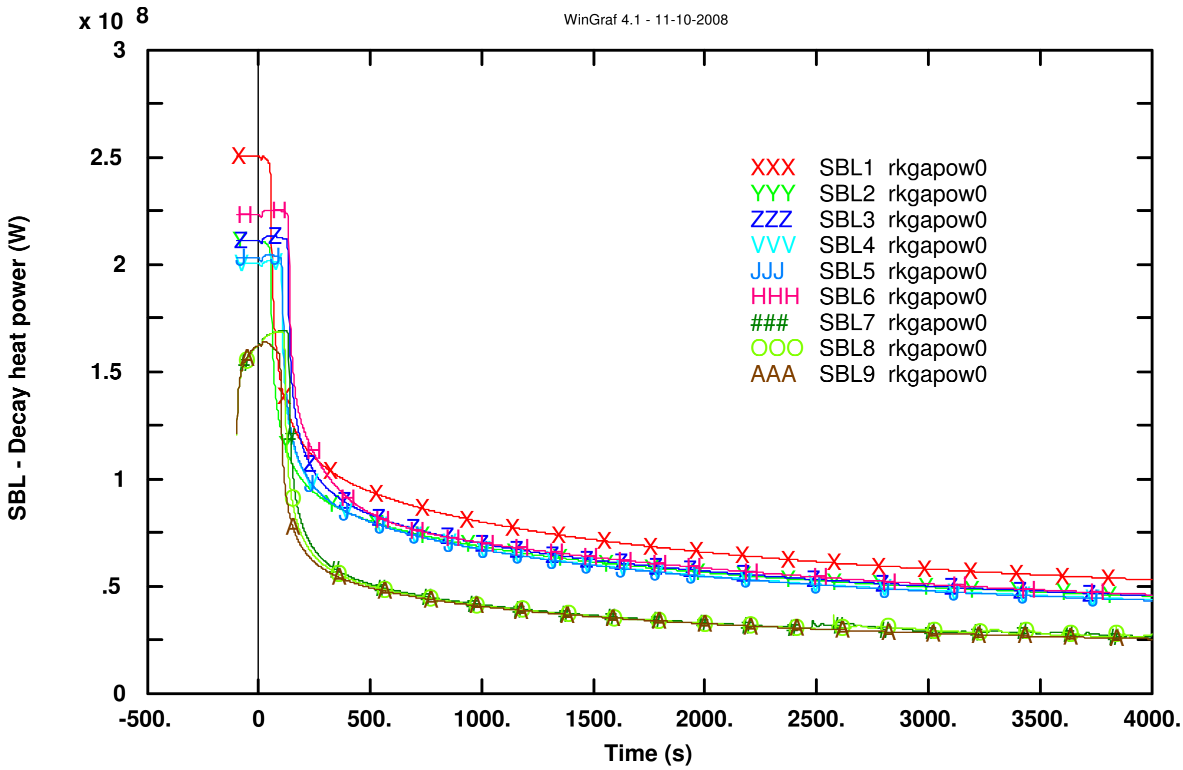


Fig. 32- Comparison of the total decay heat power produced in the transient 2, SBLOCA (up to 4000 s)

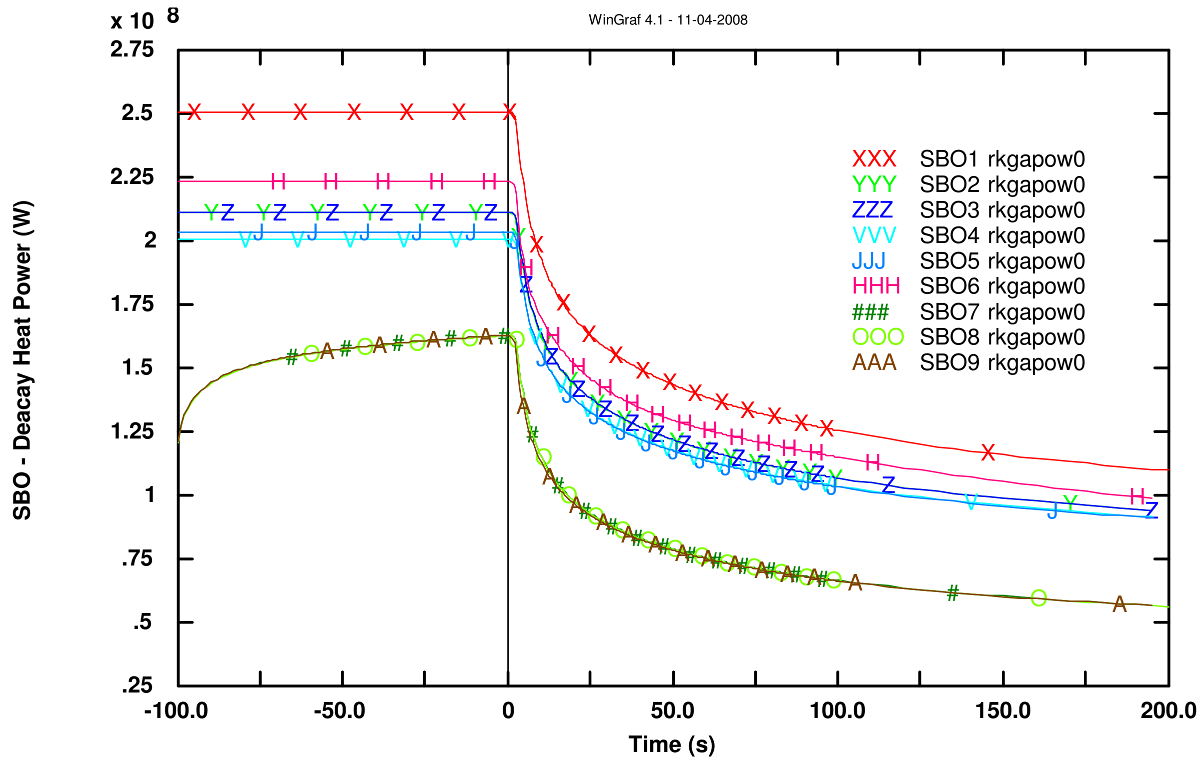


Fig. 33- Comparison of the total decay heat power produced in the transient 1, (up to 200 s)

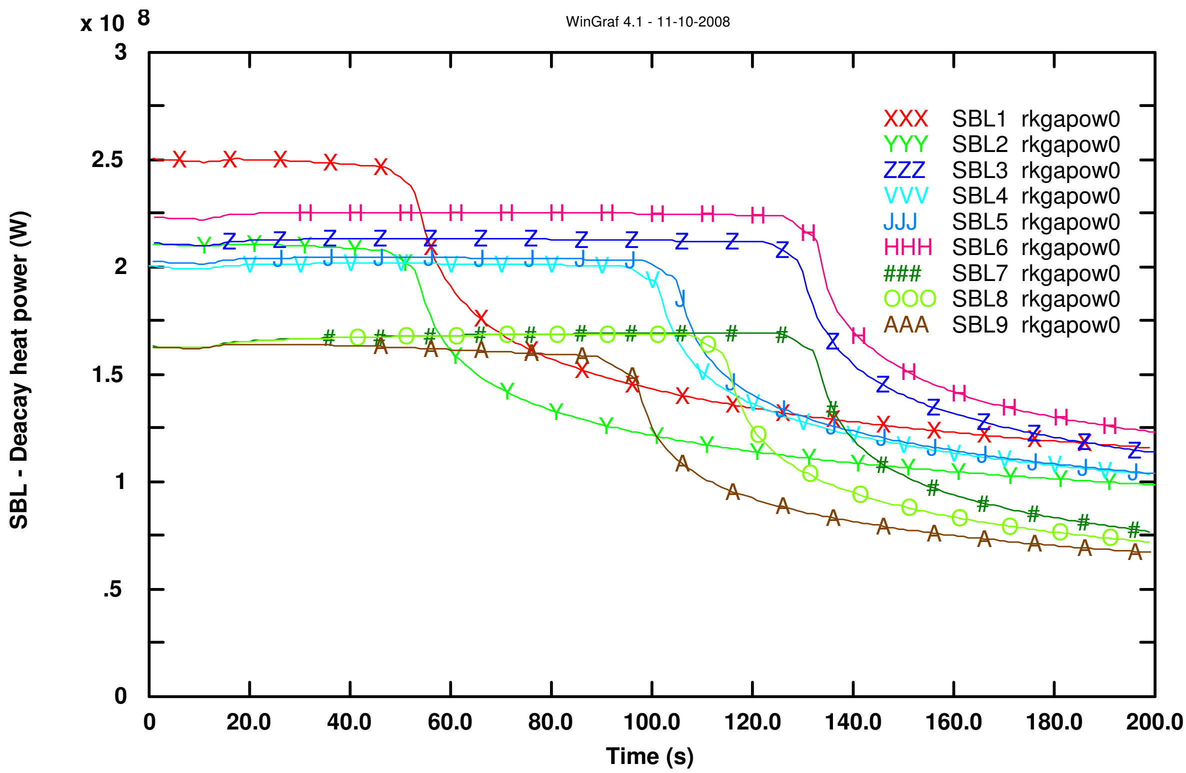


Fig. 34- Comparison of the total decay heat power produced in the transient 2, SBLOCA (up to 200 s with more detailed time step)

The figures Fig. *29* ÷ Fig. 34 compare the transients, SBO and SB-LOCA, and show that in both scenarios the curves have the same slopes and the same starting value after the steady state. In the SBO the curves maintain the same differences, due to the scram given by the SBO trip, in the SB-LOCA in the first 200 s they overpass each other, due to thermal-hydraulic phenomena of the SBLOCA.
It should be noted that the conservative calculation, SBL1 and SBO1, give the higher decay heat power and a delay of 30 s. From the other side the more realistic one, SBL8,9 and SBO8,9 give the lower power and a shorter time in decreasing.

4.2.4.1.5. *Hot rod temperature*

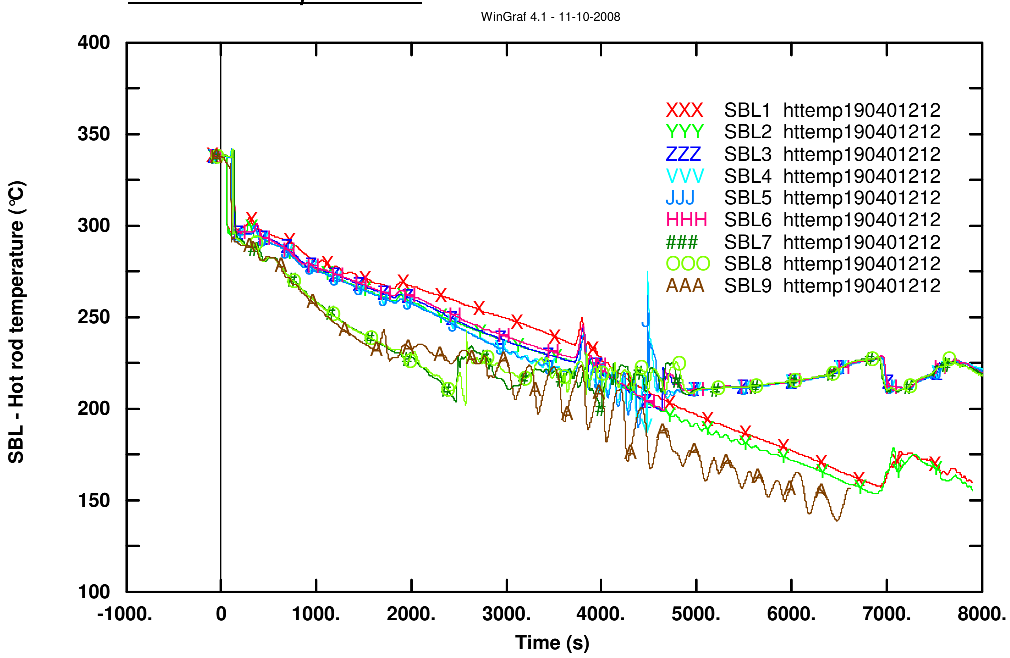


Fig. 35- Hot rod clad temperature in the transient 2 (up to 8000 s)

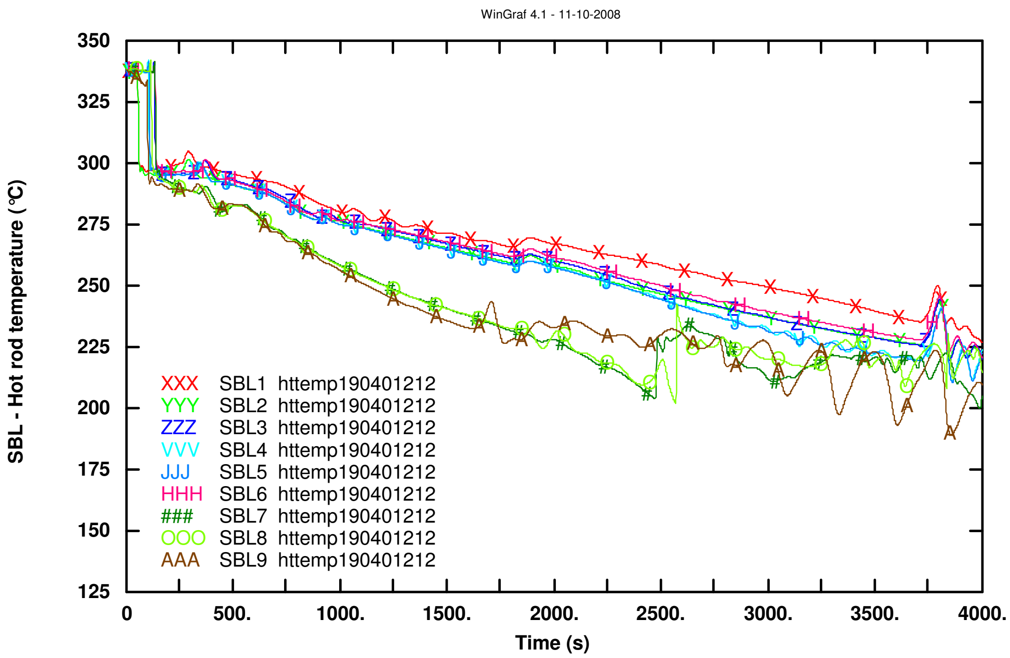


Fig. 36- Hot rod clad temperature in the transient 2 (up to 4000 s)

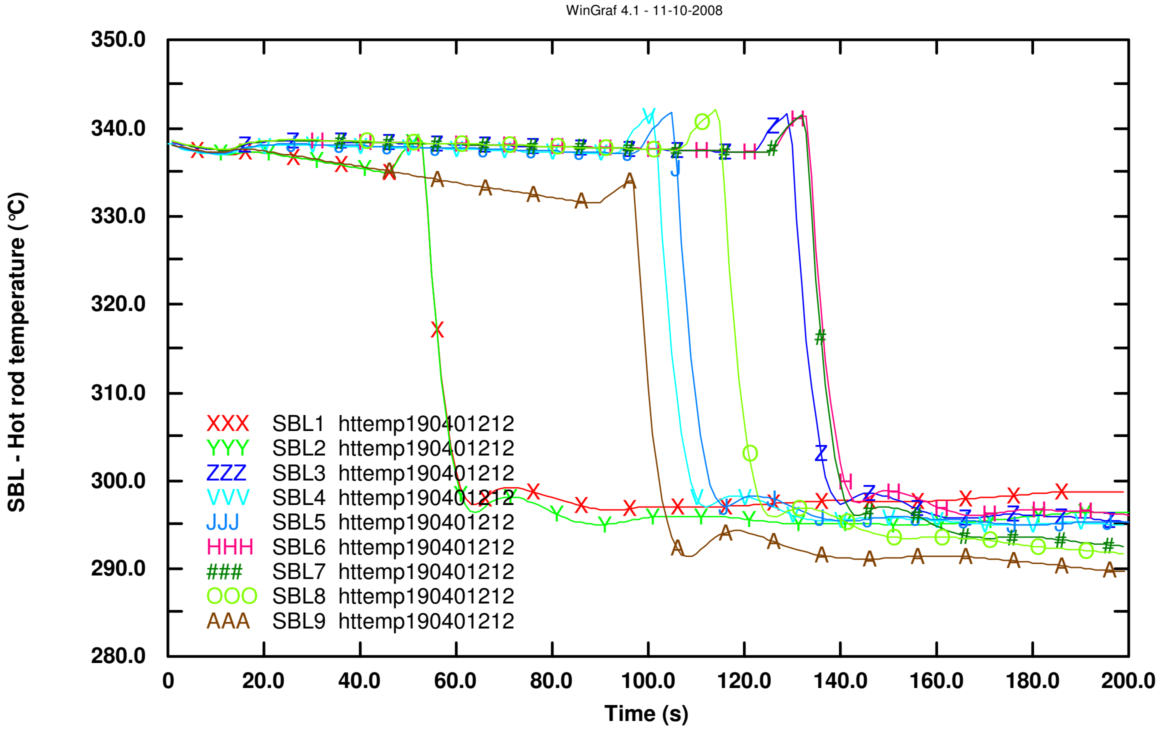


Fig. 37- Hot rod clad temperature in the transient 2

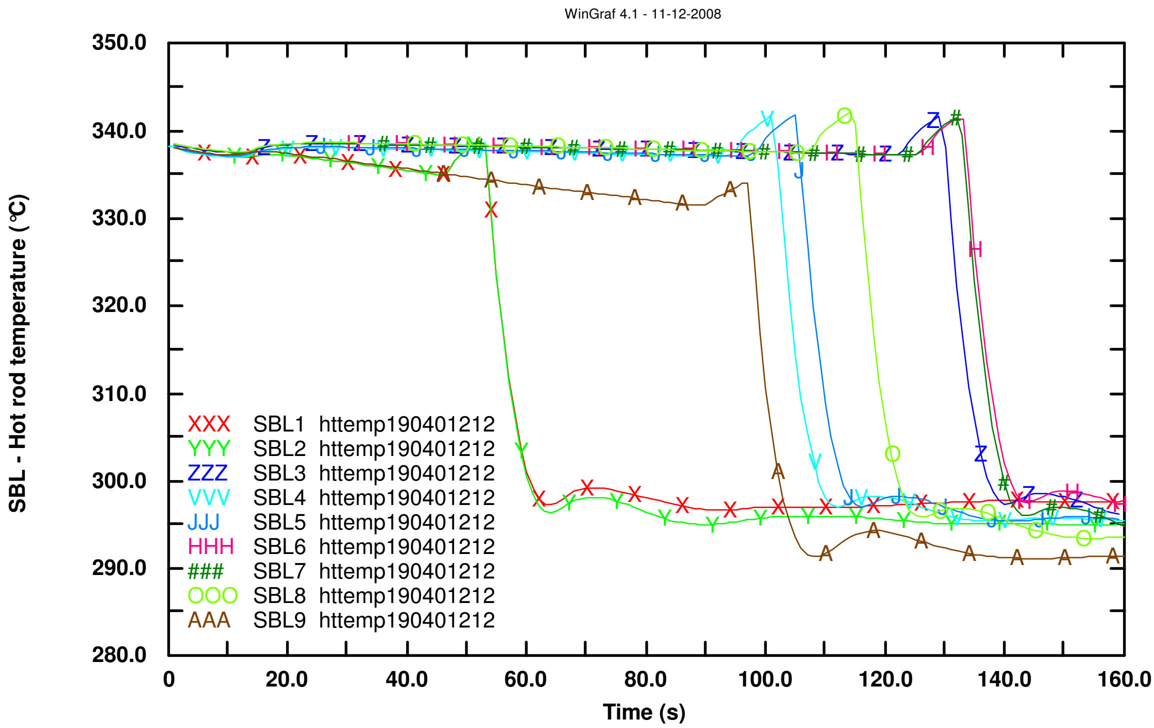


Fig. 38- Hot rod clad temperature in the transient 2 (up to 160 s)

The figures Fig. 35 ÷ Fig. 38 show the comparison of the decay heat standards with the temperature of the hot rod. The same considerations done above for the power can be repeated here: the ANS 79 and 94 give a shorter time for the cooling down of the temperature, about 30s; the ANS 73 give a value higher than the ANS79 conservative case, SBL 1. It should be noted that the power history can effect the behaviour of the temperature.

4.2.4.1.6. *Analysis of the results*

A comparison of the core power input resulting from the choice of the different ANS Standards is given in Tab. 6 and Tab. 7 for the first 10000 seconds after shutdown. The values shown in the tables are the result of a typical RELAP transient (in this case a Station Black Out transient and a Small Break LOCA) repeated using in each case a different option, as indicated in the tables.

TIME(s)	**ANS-73 [W]**	**ANS - 79-1 [W]**	**ANS-79-3 [W]**	**ANS-94-1 [W]**	**ANS 73/79.3**	**ANS 73/79.1**	**ANS 73/94.1**
100	223 400 000	211 130 000	200 610 000	203 120 000	1,0581	1,1136	1,0998
1000	69 820 000	69 430 000	67 434 000	67 141 000	1,0056	1,0354	1,0399
10000	35 528 000	34 973 000	33 872 000	33 412 000	1,015	1,0551	1,0633

Tab. 6 - Comparison of ANS-DECAY inputs in RELAP5/MOD3.2 SBO

TIME(s)	**ANS-73 [W]**	**ANS - 79-1 [W]**	**ANS-79-3 [W]**	**ANS-94-1 [W]**	**ANS 73/79.3**	**ANS 73/79.1**	**ANS 73/94.1**
100	224 880 000	211 840 000	193 680 000	201 450 000	1.1611	1.0616	1.1163
1000	70 301 000	69 657 000	67 393 000	67 178 000	1.0431	1.0092	1.0465
10000	52 151 000	50 852 000	52 007 000	49 439 000	1.0028	1.0255	1.0549

Tab. 7 - Comparison of ANS-DECAY inputs in RELAP5/MOD3.3 SBLOCA

The maximum ratio between the various values obtained (as can be seen in the last three columns of theTab. 6 and Tab. 7) amount to about 1.1.
It has also to be noted that the ANS-94-4 values practically coincide with the values of the International Standard ISO -10645 (Decay Heat of Nuclear Reactors).
The various ANS Standards Reports and the ISO Report give also guidance for uncertainty evaluations.

4.2.4.1.7. *Conclusion*

For Best Estimate Calculations it is here suggested to use, for Decay Power, the ANS-94-4 or the most recent ANS Standard (ANS 5.1 – 2005) or the equivalent ISO-10645 Standard; of course, the real power history before shutdown could also be used.
Three more issues have to be considered in order to use the correct core power during transient and accident studies in a BE environment:

- The full power time period before scram should, in principle, be considered, although in most cases it is not important for the calculation results;
- The fission power generated after scram by the delayed fission neutrons may be important in fast transients, where fuel overheating may occur in tens of

seconds after accident initiation; e.g. : large LOCAs; as a first approximation, it can be calculated (for ^{235}U, LWRs) by ref. [85] page 81:

$P/P_0 = 0.15\ e^{-0.1\ t}$ 11)

(P_0 is the power before the accident, t is time after scram in seconds)

4.2.4.2. Decay heat: gamma and beta contribution

The present section aims to summarize the activity performed to analyse the contribution of the decay gamma on the PCT [65].
A preliminary investigation was done by analytical consideration on the LOFT facility. Detailed evaluation was performed by the use of the MCNP 5 code either on LOFT facility and generic VVER – 1000.
Decay heat in fission reactors [32] is almost equally subdivided into two parts, one part due to beta rays and the other due to gamma photons[1]. This fact is not important for the overall thermal balance of the reactor, but it is important for the decay power distribution within the core and for the possible overheating of the so called "hot rod" during an accident. Beta rays are absorbed practically where they are generated, while gamma photons travel some distance in core before being absorbed. The decay power peaking factor is, in fact, decreased by this phenomenon of gamma decay heat redistribution. If a power peak exists during reactor operation, this power peak is attenuated during shutdown. This fact should be taken into account in transient and accident analyses (and particularly in best estimate analyses) since it can be very relevant for some interesting quantities and in particular for the calculated PCT (peak cladding temperature).
Also Appendix K to Part 50 of the U.S. C.F.R. states [11] that " The fraction of the locally generated gamma energy that is deposited in the fuel (including the cladding) may be different from 1.0; the value used shall be justified by a suitable calculation."

4.2.4.2.1. <u>Gamma decay heat source</u>

Gamma photons due to decay of fission products are usually grouped into seven energy groups Tab. 8 [27] shows these groups.
Tab. 9 shows the strength of the various groups at two times after fission, 100 and 10000 seconds, particularly relevant for accident studies, LOCA and SB-LOCA respectively.

---

1 *The heat load from decaying fission products in a fuel assembly is proportional to empirical emission rates of beta and gamma radiation. The rates per ^{235}U fission, and as a function of decay time (t_d in days), are:*

$\beta(t_d) = 1.50 \cdot 10^{-6} \cdot t_d^{-1.2}$ *MeV/s-f* *(1)*

$\gamma(t_d) = 1.67 \cdot 10^{-6} \cdot t_d^{-1.2}$ *MeV/s-f* *(2)*

These energy rates are roughly equal for 0.4 MeV mean energy beta particles and 0.7 MeV mean energy gamma-rays.

Group	Energy range [Mev]	Effective energy[Mev]
I	0.1 - 0.4	0.4
II	0.4 - 0.9	0.8
III	0.9 – 1.35	1.3
IV	1.35 - 1.8	1.7
V	1.8 – 2.2	2.2
VI	2.2 – 2.6	2.5
VII	> 2.6	2.8

Tab. 8 - Groups of decay γ photons and energies

Time after fission [s]	100 s			10000 s		
Group	[Mev/s fission]	Disintegr./s	% or probability	[Mev/s fission]	Disintegr./s	% or probability
I	6e-5	1.5e-4	8	2e-6	5e-6	19
II	2.4e-4	3e-4	16	1e-5	2.25e-5	48
III	1.2e-3	9.2e-4	49	2.5e-6	1.9e-6	7
IV	4.2e-4	2.5e-4	13.4	6e-6	3.5e-6	13
V	6e-5	2.7e-5	1.45	3.5e-6	1.6e-6	6
VI	2e-4	8e-5	4.3	3.5e-6	1.4e-6	5.4
VII	4.2e-4	1.5e-4	8	5e-7	1.8e-7	7

Tab. 9 - Spectrum of Fission Product Decay Gamma Rays at 100 and 10.000 s from the shut down

The average energy per disintegration is, for 100 s, 1.33 Mev/dis. while, for 10000 s, it is 1.07 Mev/dis.

4.2.4.2.2. <u>*Gamma peak factor for sinusoidal distribution*</u>

Inside the reactor, i.e. research and commercial reactors, hereafter, the gamma distribution is considered to have a sinusoidal distribution as for the neutron (1).

$$P_{Tot} = \int_V P_{\max} \cos\left(\frac{r}{R}\frac{\pi}{2}\right)\cos\left(\frac{2z-H}{H}\frac{\pi}{2}\right)2\pi r dr dz \qquad 1)$$

For the purpose the core is schematized as a cylinder with origin in the center of the axis z (2).

$$P_{Tot} = P_{\max}\int_{r_1}^{r_2}\cos\left(\frac{r}{R}\frac{\pi}{2}\right)2\pi r dr\int_0^{\frac{H}{2}}\cos\left(\frac{2z-H}{H}\frac{\pi}{2}\right)dz\,. \qquad 2)$$

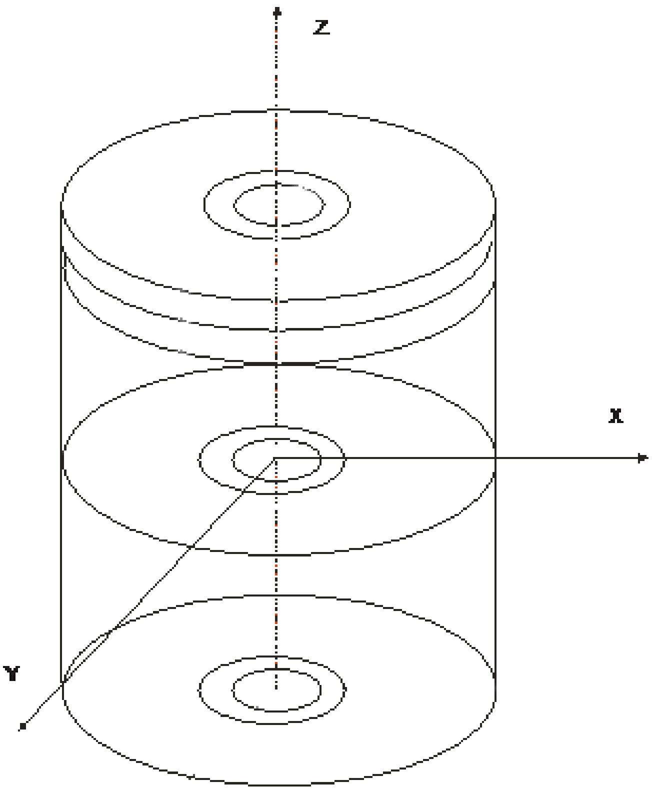


Fig. 39- Model of the LOFT and PWR core

Developing the integral along z and r (2), the total gamma power is given by (3).

$$P_{Tot} = P_{\max} 2\pi \left[r\left(\frac{2R}{\pi}\right)\sin\left(\frac{r}{R}\frac{\pi}{2}\right) + \left(\left(\frac{2R}{\pi}\right)^2 \cos\left(\frac{r}{R}\frac{\pi}{2}\right)\right)\right]_{r_1}^{r_2} 2\frac{H}{\pi} \qquad 3)$$

Developing the function (3) it is possible to see the relation between mean power $\overline{P}$ and P_{max} (4) and the ratio $\frac{P_{\max}}{\overline{P}}$ (5).

$$\overline{P} = \frac{P_{Tot}}{V} = \frac{0.2947 P_{\max} V}{V} = 0.2947 P_{\max} \qquad 4)$$

$$\frac{P_{\max}}{\overline{P}} = 3.393 = 1.8428^2 \qquad 5)$$

That this is the punctual peak factor of the core for linear sinusoidal distribution (axial and radial).

$$P_{Tot} = \overline{P}V = \frac{P_{\max}}{3.393} V \qquad 6)$$

Details of the calculations are reported in Annex IV.

4.2.4.2.3. *Model and calculation cases*

The simple model used for a core is a homogeneous cylinder subdivided into cylindrical rings and into slices normal to the axis, Fig. 39.

The density chosen is an average one (by the way, the result is dependent, but less than linearly, on density, because of the “build up effect” in γ attenuation). The normal operation power distribution has been simulated by a sinusoidal curve both in the radial and in the axial directions.
For the radial distribution, other shapes have been explored:
- the Hot Rod (simulated by a 1.4 factor energy peak in the central ring superimposed on a sinusoidal distribution for the whole core) and
- the Gaussian distributions for the whole core with σ^2 equal to 0.01(narrow), 0.05 (intermediate), 0.12 (wide) and 0.24 (extra wide).

The spectrum of γ photons has been taken into account.
The well known Monte Carlo code MCNP5 has been used for the calculation of the distribution in the core of the absorbed γ photons energy corresponding to the γ photons produced energy, distributed as above mentioned.

Core	Time after shut down [s]	Gamma distribution
LOFT	100	Sinusoidal
	10000	Sinusoidal
	100	Sinusoidal with hot rod
	100	Gaussian (σ^2=0.05)
	100	Gaussian (σ^2=0.12)
PWR - 1000	100	Sinusoidal
	100	Sinusoidal with hot rod
	100	Gaussian (σ^2=0.01)
	100	Gaussian (σ^2=0.12)

Tab. 10 – Gamma cases analysed

To run the MCNP5 code a geometrical model and probability distribution is needed. The LOFT (Loss Of Fluid Test, [28]) core is modelled in 35 radial rings and 66 slides. Considering the axial and radial cells, the probability distribution is defined with the relation (15) and (16) respectively, (17) for the radial Gaussian distribution. The PWR case was analysed with a coarse nodalization (35 radial rings and 66 slides) and with a fine one:
- a ring every centimetre and a slide every 2 centimetres to define the probability distribution
- a ring every 2 centimetres and a slide every 4 centimetres to define the geometry. This solution is chosen to fix the statistical error of the MCNP5 code.

4.2.4.2.4. <u>*Evaluation of the cell probability*</u>

To make the calculation of the power gamma decay heat absorbed in the fuel, it is needed to divide the core in cells. The dimension of a cell should allow the movement of the gamma from one cell to the other. For our purpose the dimension of the cell is an annular circular of 1 cm and thickness 2,5 cm.
The MNCP 5 requires the definition of the probability for each cell $P_c(r,z)$. It can be evaluated as fraction of the power of the cell and total power. Details of the calculations are reported in Annex IV.

4.2.4.2.5. Analysis of the results

The main results are the following:

- The maximum γ power absorbed in the core is significantly decreased when the γ photon redistribution is taken into account. Since the decay heat is due for one half to γ and for the other half to β rays, a certain reduction in absorbed γ power translates in a reduction of one half of it in γ + β (total) decay power. In particular:
    - *a)* For the small reactor without hot rod the reduction of absorbed γ power versus produced power is equal to about 10% at 100s and to about 15% at 10000 s after shutdown; the case of a large reactor with local neutron flux hills (due, for example, to specific control rod management strategies) can approximate the case of a small reactor;
    - *b)* For the small reactor with hot rod, the γ peak at the hot rod practically disappears and the overall (sine distribution plus hot rod) reduction in peak energy is equal to about 30% at 100 s (this is considered the most significant result since the γ redistribution, with corresponding γ+ β power decrease of 15%, may entail a calculated PCT reduction of the order of 100 – 150 K); for a large reactor, the peak energy is reduced by a 12% instead of 30%.
    - *c)* For the large reactor without hot rod the corresponding γ reduction is much lower (about 1% for a sine distribution at 100 s), with coarse nodalization, and of the same magnitude of LOFT for a fine nodalization.

Figures Fig. 40 to Fig. 43 show the distributions (radial and axial) of produced and absorbed energies for LOFT and for large reactor , coarse model, with sine distribution.

Fig. 44 to Fig. 47 show the same distributions for the case with hot rod.

Fig. 48 to Fig. 49 show the distributions (radial and axial) of produced and absorbed energies for large reactor , fine model, with sine distribution.

Fig. 50 to Fig. 51 show the same distributions for the case with hot rod.

Fig. 52 to Fig. 55 show the radial Gaussian distributions for the case σ^2=0.01 and σ^2=0.12.

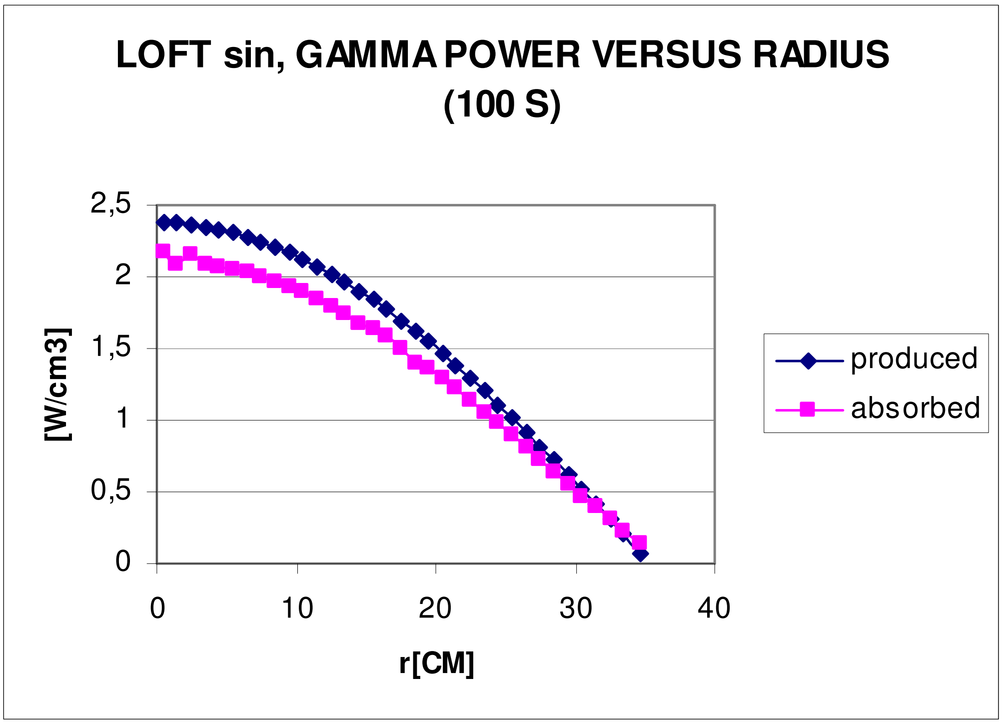


Fig. 40– LOFT sin, 100 s, produced and absorbed power versus radius

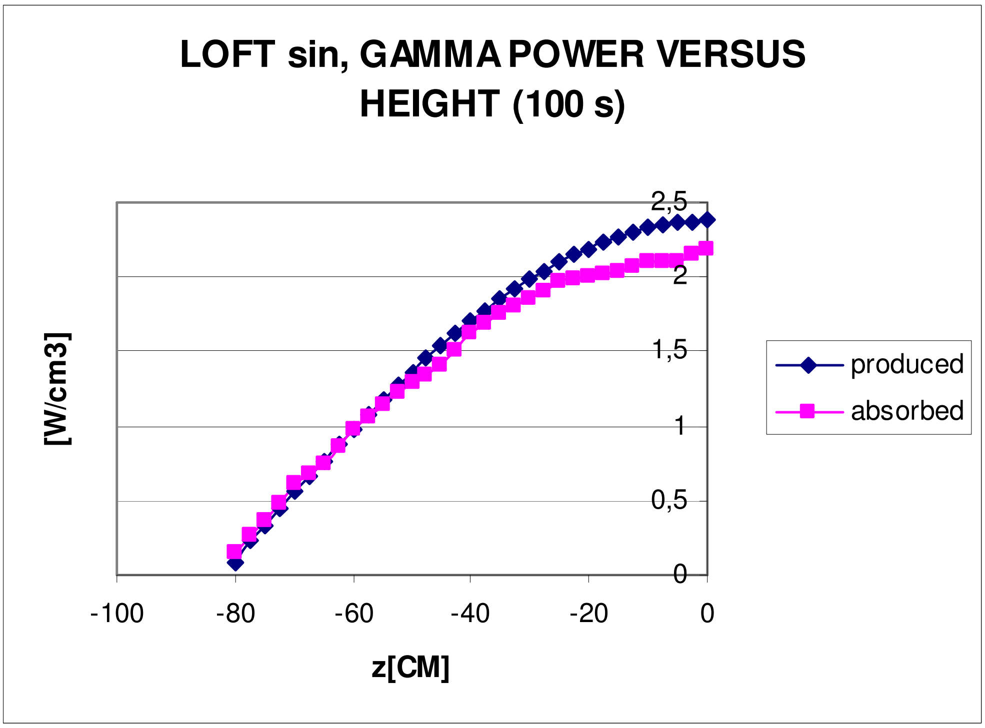


Fig. 41 – LOFT sin, 100 s, produced and absorbed power versus height

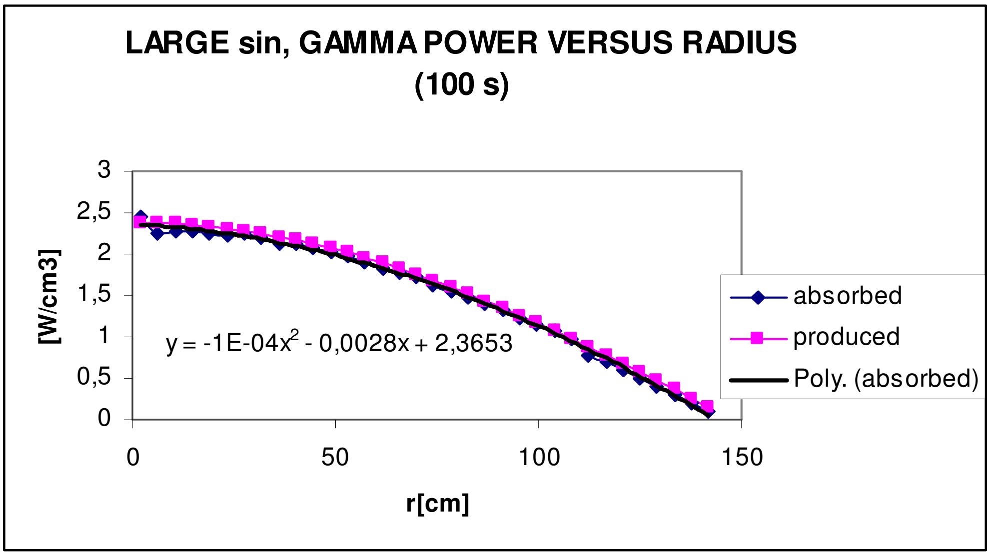


Fig. 42 – LARGE reactor sin, 100 s, produced and absorbed power versus radius

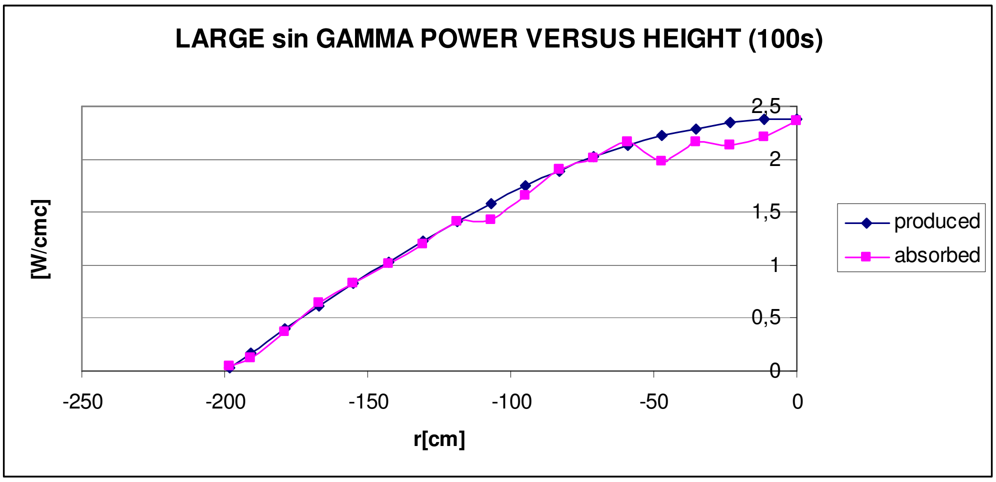


Fig. 43 – LARGE reactor sin, 100 s, produced and absorbed power versus height

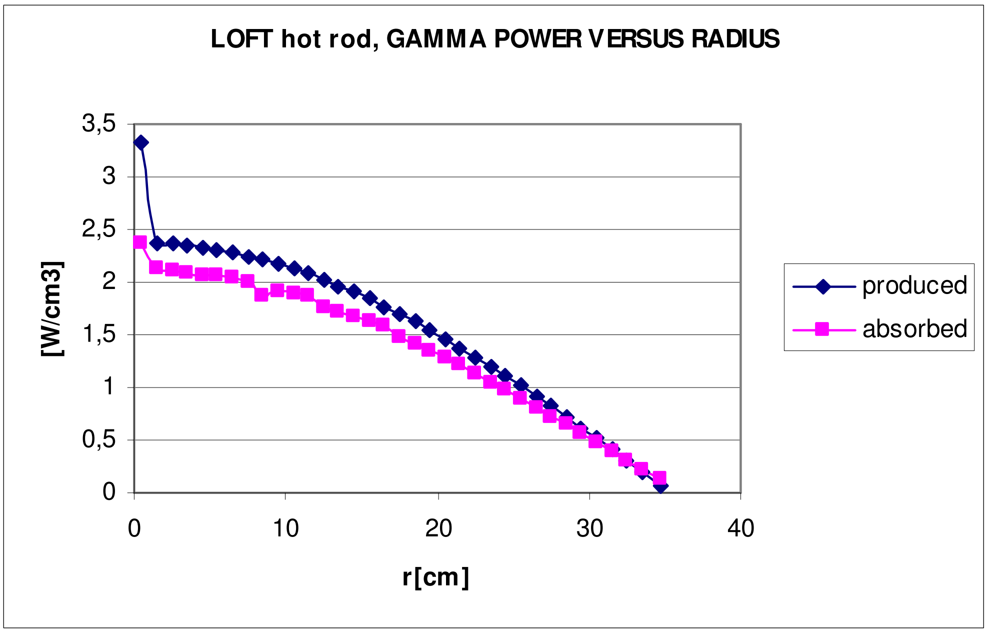


Fig. 44 – LOFT hot rod, 100 s, produced and absorbed power versus radius

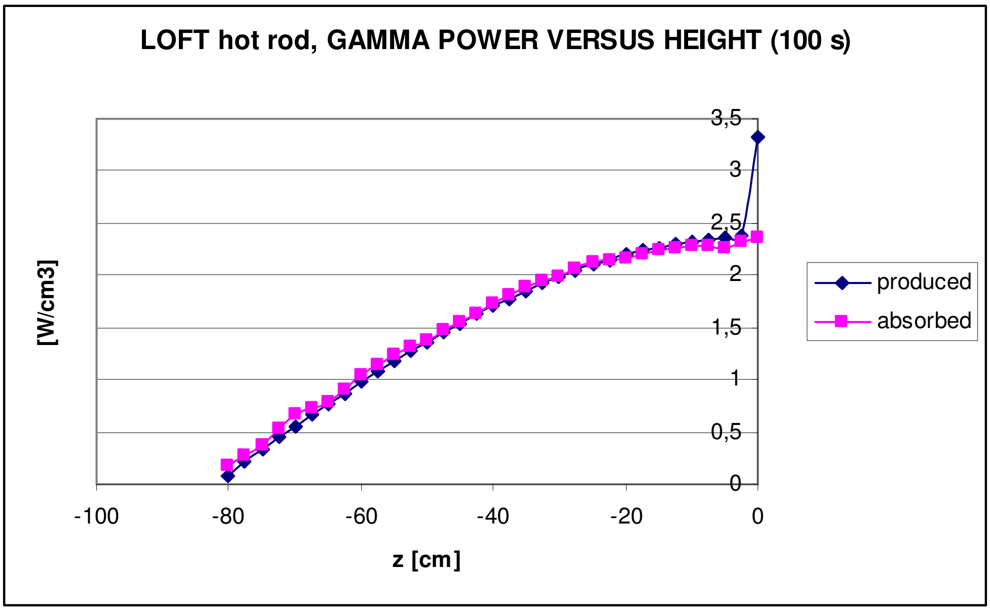


Fig. 45 – LOFT hot rod, 100 s, produced and absorbed power versus height

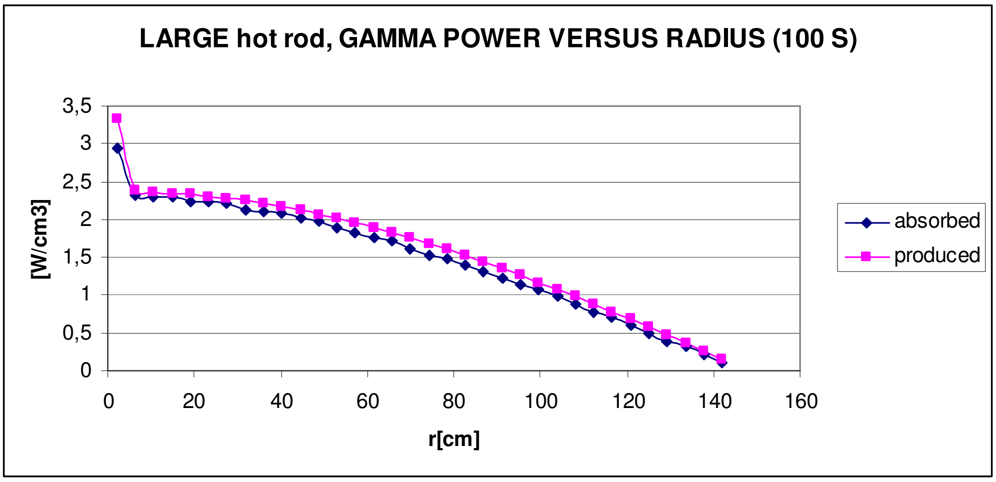


Fig. 46 – LARGE reactor hot rod, 100 s, produced and absorbed power versus radius

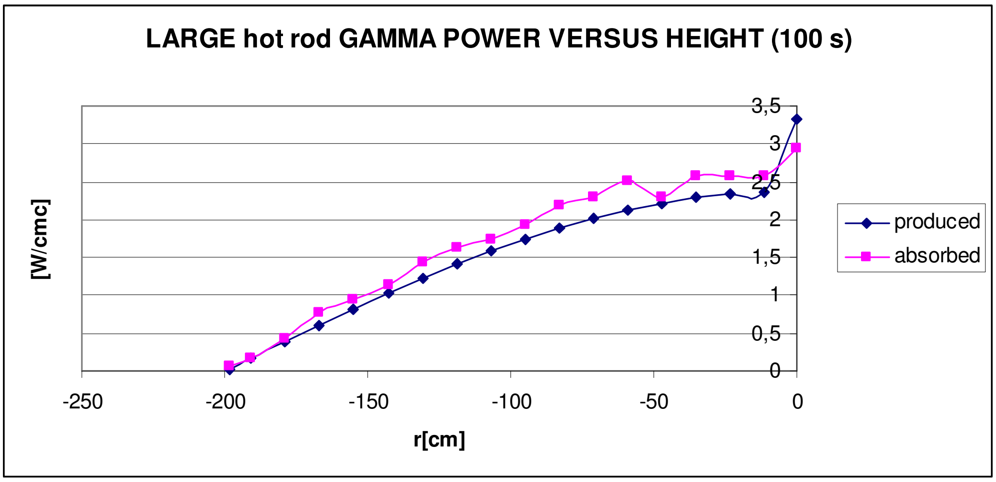


Fig. 47 – LARGE reactor hot rod, 100 s, produced and absorbed power versus height

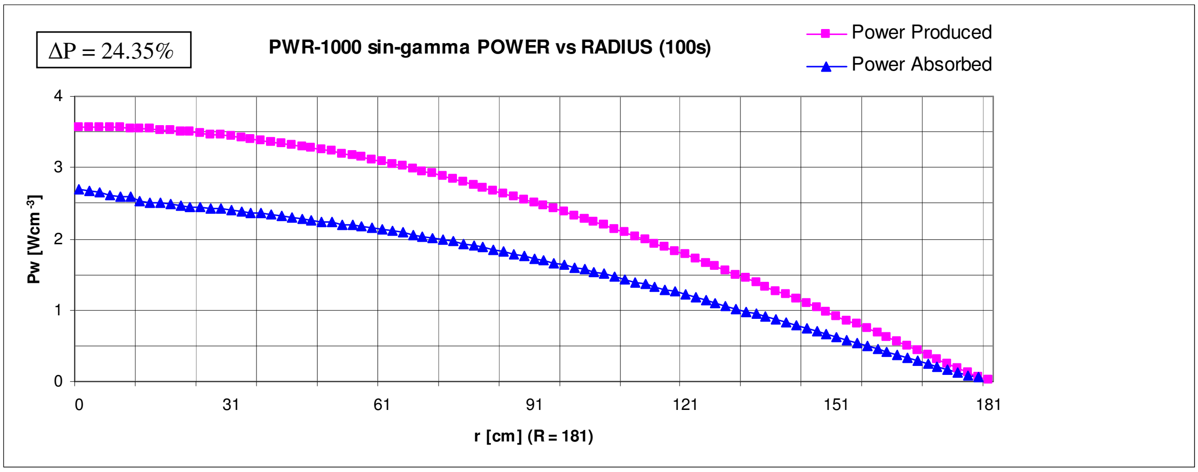


Fig. 48 – LARGE reactor sin, 100 s, produced and absorbed power versus radius fine model

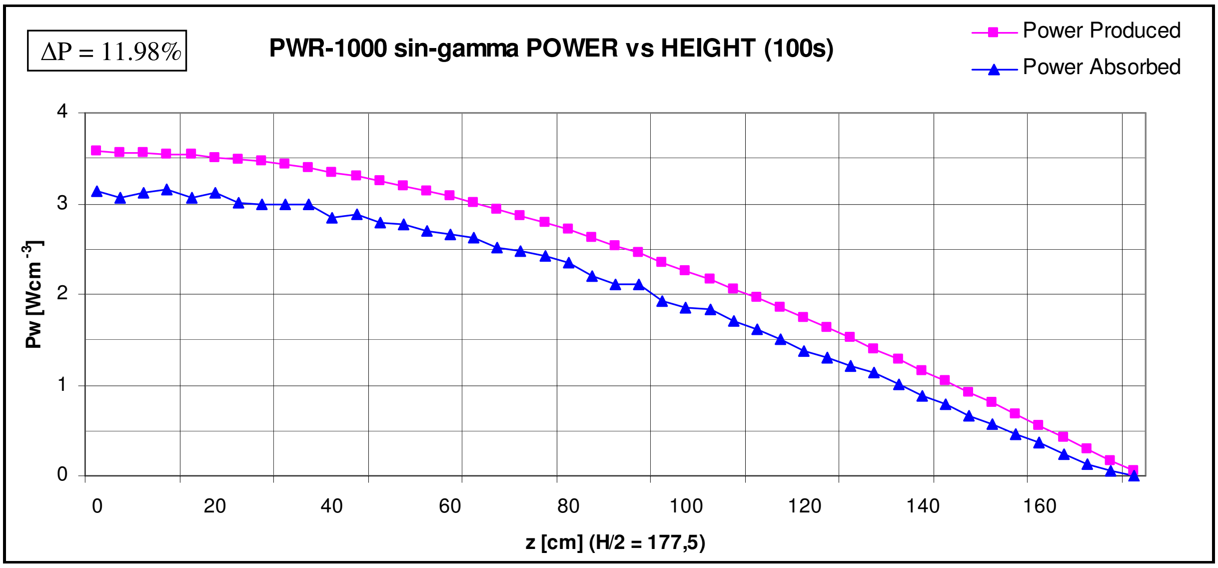


Fig. 49 – LARGE reactor sin, 100 s, produced and absorbed power versus height fine model

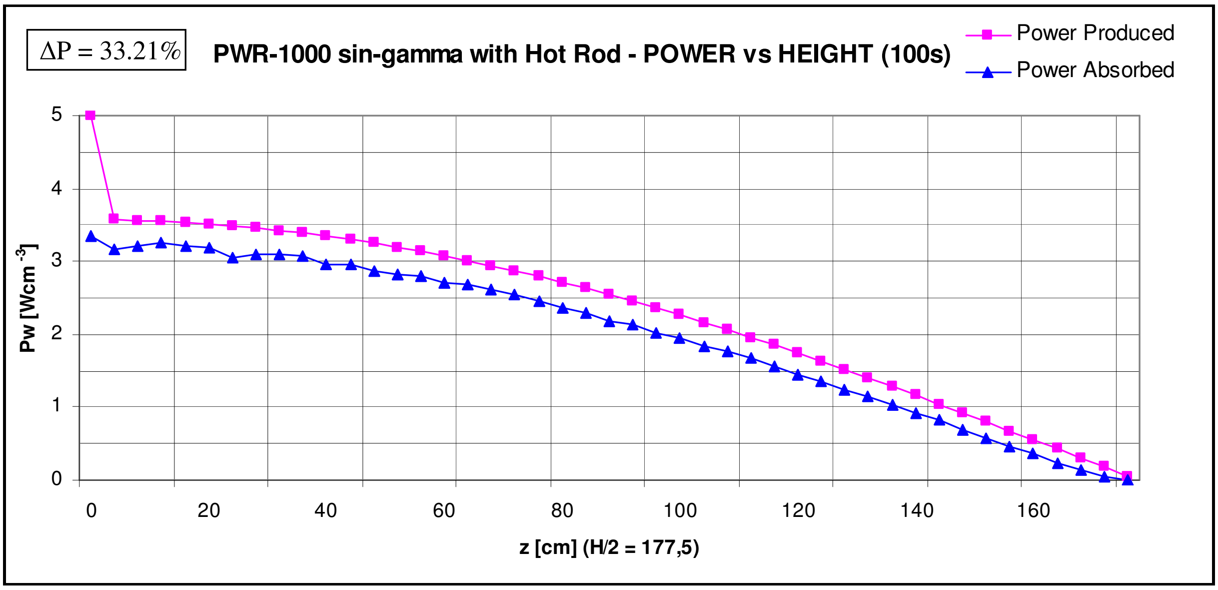


Fig. 50 – LARGE reactor hot rod, 100 s, produced and absorbed power versus radius fine model

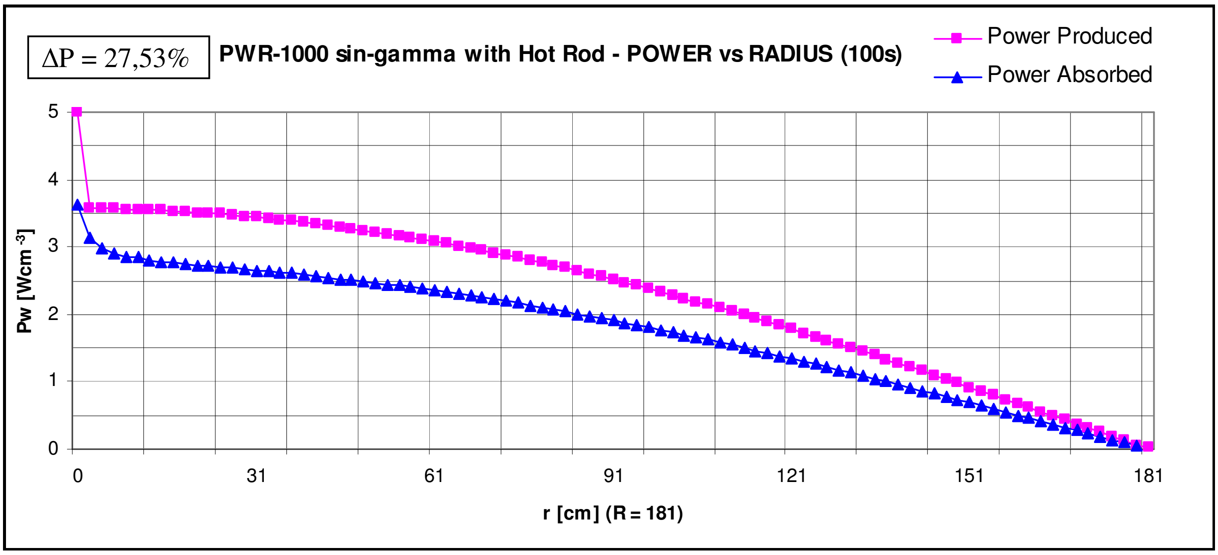


Fig. 51 – LARGE reactor hot rod, 100 s, produced and absorbed power versus height fine model

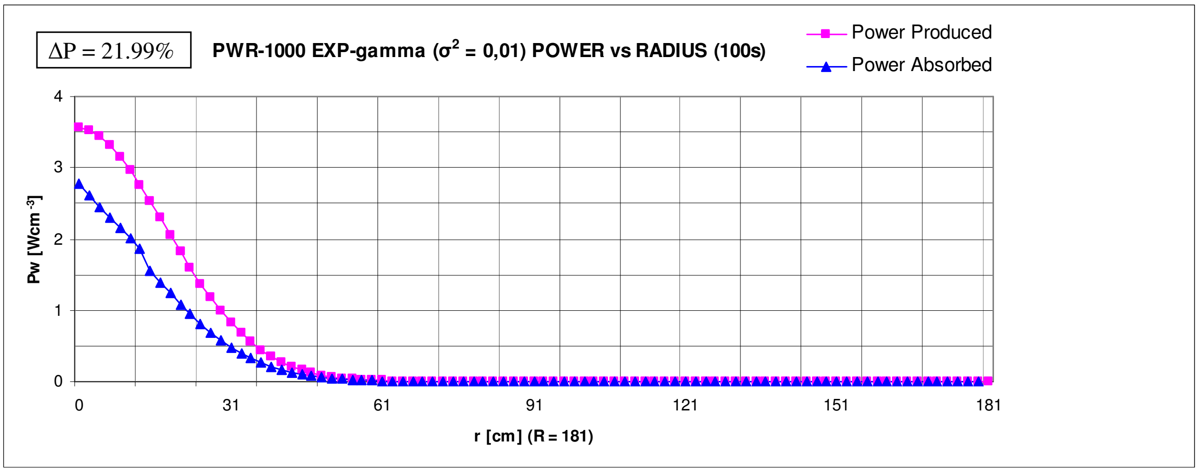


Fig. 52 – LARGE reactor sin, 100 s, produced and absorbed power versus radius fine model

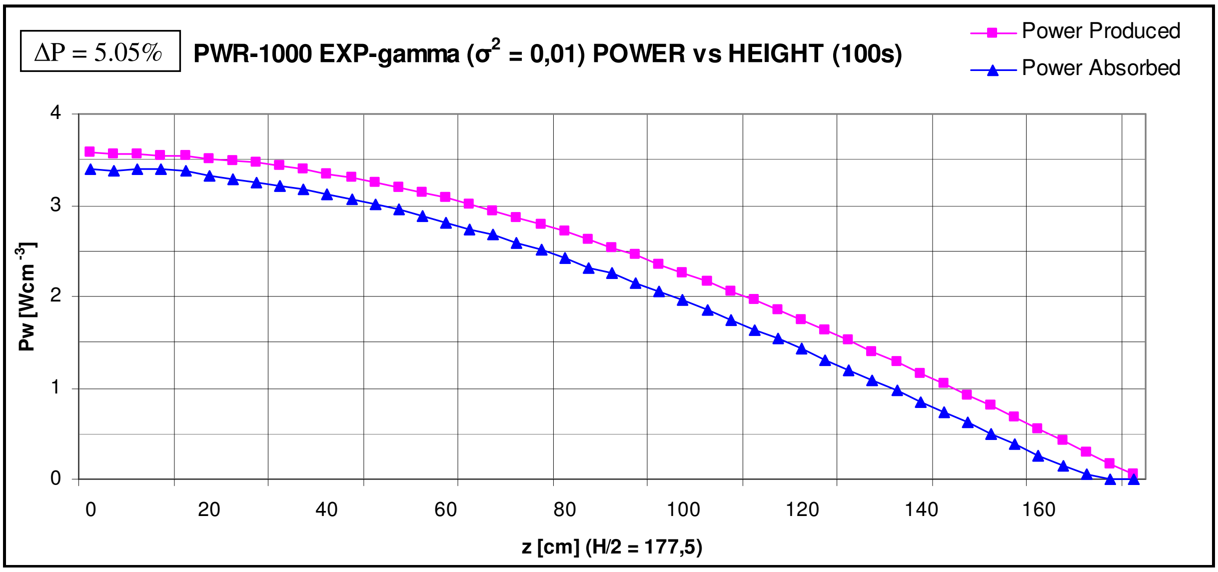


Fig. 53 – LARGE reactor sin, 100 s, produced and absorbed power versus height fine model

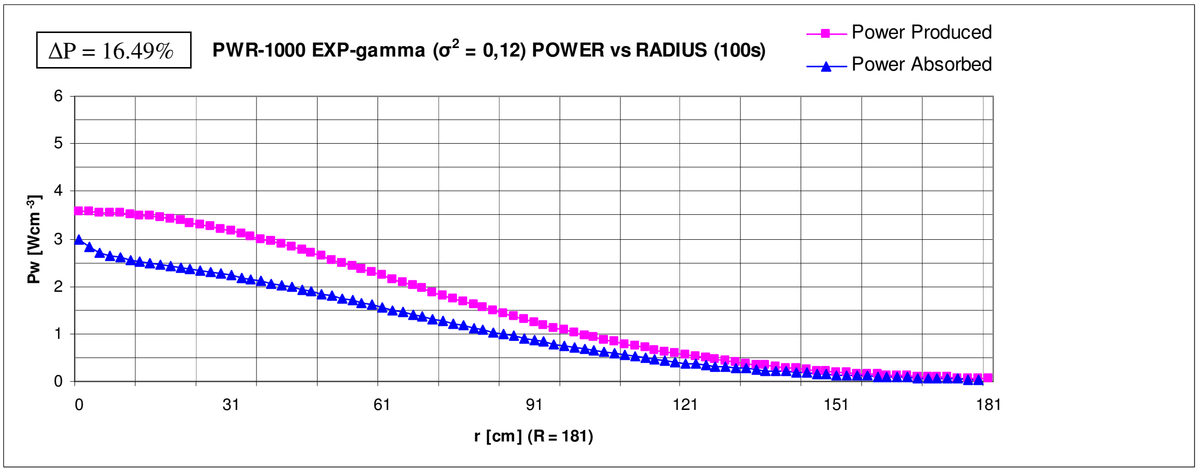


Fig. 54 – LARGE reactor sin, 100 s, produced and absorbed power versus radius fine model

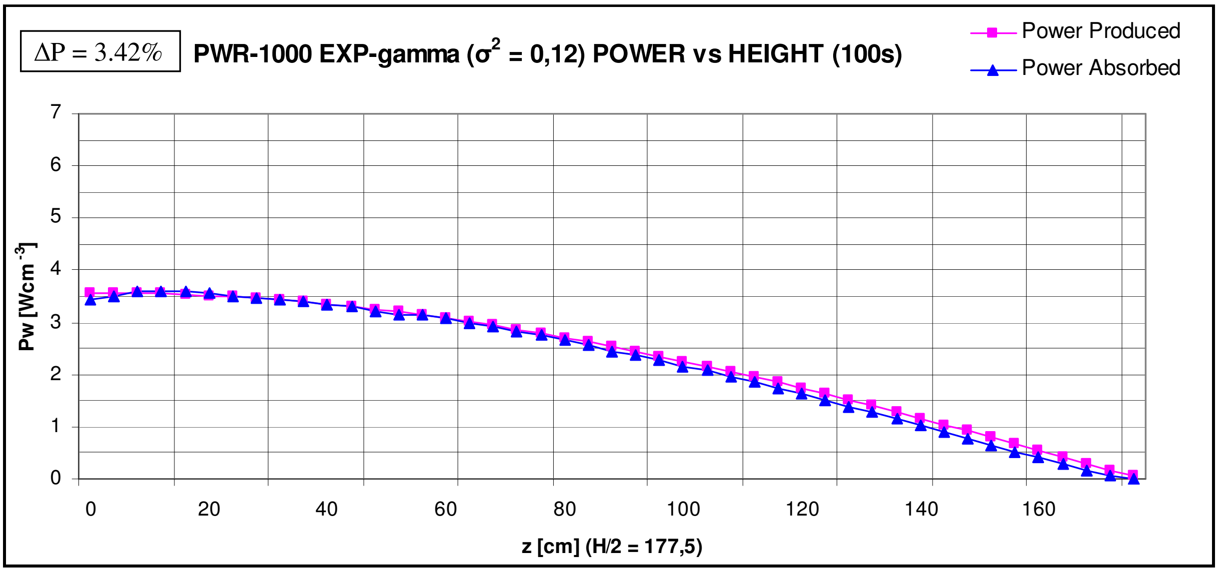


Fig. 55 – LARGE reactor sin, 100 s, produced and absorbed power versus height fine model

4.2.5. Main sources of uncertainty for evaluating the PCT

The main source of uncertainty can be identify as following, where the ☺ gives the positive contribution, i.e. an increasing of the parameter considered increases the value of the PCT, and ☹ gives negative contribution to the PCT, i.e. an increasing of the parameter considered decreases the value of the PCT: [42] [50]

– For blow-down
  - ☺ Fuel rod gap width for low burn-up
  - ☺ Reactor initial power
  - ☹ Fuel heat conductivity
  - ☹ Minimum film boiling temperature

- ☹ Model for critical heat flux
- ☹ Extensive experimental and operational data are needed to reference applied values
- ☹ 2-phase multiplier in horizontal pipe (higher steam content to break location>lower break flow>higher water content in core> lower clad temperature)

– For re-flood:
- ☹ Fuel heat conductivity
- ☺ Fuel rod gap width for low burn-up
- ☹ Model for 1-phase convection to steam (Mc Eligot correlation tends to cause higher clad temperatures than Dittus-Boelter II)
- ☹ Number of droplets (number higher>higher evaporation>lower PCT)
- ☹ Steam droplet cooling (higher cooling tends to result in lower PCT)

Such parameters could be subject to further analyses.

4.2.6. Metal-water reaction rate

As quoted in RG 1.157, the rate of energy release, hydrogen generation and cladding oxidation from the reaction of zircaloy cladding with steam should be calculated in a BE way. In fact, these quantities are required for transient calculations and for the demonstration of compliance with 10 CFR 50 App. K.
Here the same recommendation of RG 1.157 is made: Ref. [75] should be followed.
The subject is not developed because goes over the present analysis. However, it should be considered for the general safety of the system.

4.2.7. Heat transfer from reactor internals

RELAP code “Heat Transfer Package” includes a number of heat transfer correlations, which are used when the heat structure is connected to a hydrodynamic volume. Various heat transfer modes are taken into account ([7] Vol.2, 3.2): e.g. convection to non-condensable water mixture, sub-cooled nucleate boiling, condensation, etc.
Further heat transfer values chosen by the user can also be used.
It is here recommended to use, to the extent possible, the RELAP Heat Transfer Package correlations: no special input is, therefore, required in this connection.

4.2.8. Primary to secondary heat transfer in Steam Generators

The subject is not developed because goes over the present analysis. However, it should be considered for the general safety of the system.
The RELAP does not have any difference between the steam generator pipes and any other heat-transmitting pipe.

4.2.9. Thermal parameters for swelling and rupture of the cladding and fuel rods

The models included in Ref. [76] are here suggested for any calculation of cladding swelling and rupture and channel flow area reduction due to swelling. Consideration of these phenomena should be made according to 10 CFR 50 App. K.
The subject is not developed because goes over the present analysis. However, it should be considered for the general safety of the system.

4.2.10. Identification of hottest cladding point in core

The hottest cladding point not always is identified with the hot rod. Its position is depending on different parameters like: transient, burn-up, temperature, insertion of the control rod, etc.

The method of using statistically identified core power peaking factors and the method of calculating, by a 3D calculation, the likely core power distribution should be taken into consideration.

The most usual method uses core peaking factors. The RELAP Code input structure is very suitable for this traditional procedure.

Here, it is suggested to use such method together with suitable quasi-static or different peaking factors if no need for very accurate core temperature predictions exists. A core hot-rod and an average core rod can be defined.

However, if the transient is such to entail strong neutron flux distortion, then a very accurate BE calculation is opportune and a 3D coupled calculation is suggested. As examples of these cases, the rod ejection accident in a PWR, a rod drop accident in a BWR and transients entailing spatial flux oscillations in BWRs (including RBMKs) can be mentioned.

An application of such analysis is available at Ref. [77]. It shows a procedure for such more precise calculations.

4.2.11. Assumption of communication between adjacent channels

The assumption of closed fuel element channels has to be considered, together with the one of open, communicating, adjacent channels; this last situation more closely reflects the real PWR geometry, but can rise to a number of drawbacks. In particular, stability issues can arise.

Analysis has been performed on the model of the cross flow by RELAP between adjacent fuel bundles, unless prevailing, unavoidable and unrealistic stability problems arise.

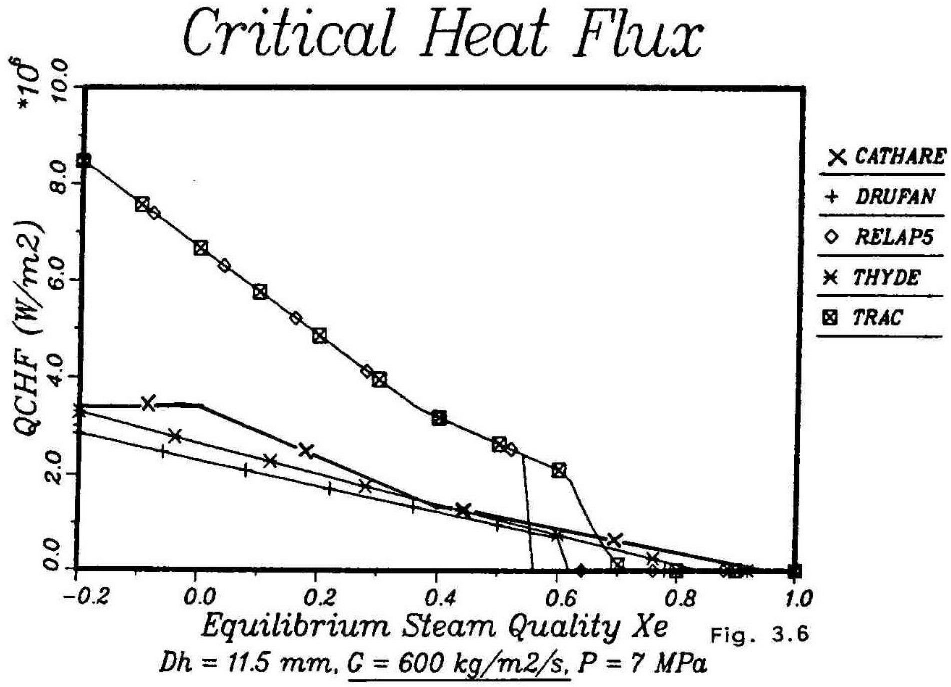
The results are reported in following paragraph.

4.2.12. Critical Heat Flux and Flow-rate

At § 3.8, the RG-1.157 suggests that BE models developed from appropriate steady-state or transient experimental data should be used in calculating critical heat flux (CHF) during LOCA. Suitable checks should be performed to ensure that the range of conditions over which these correlations are used are within those intended. Research has shown that CHF is highly dependent on the fuel rod geometry, local heat flux, and fluid conditions. After CHF is predicted at an axial fuel rod location, the calculation may use nucleate boiling heat transfer correlations if the calculated local fluid and surface conditions justify the reestablishment of nucleate boiling.

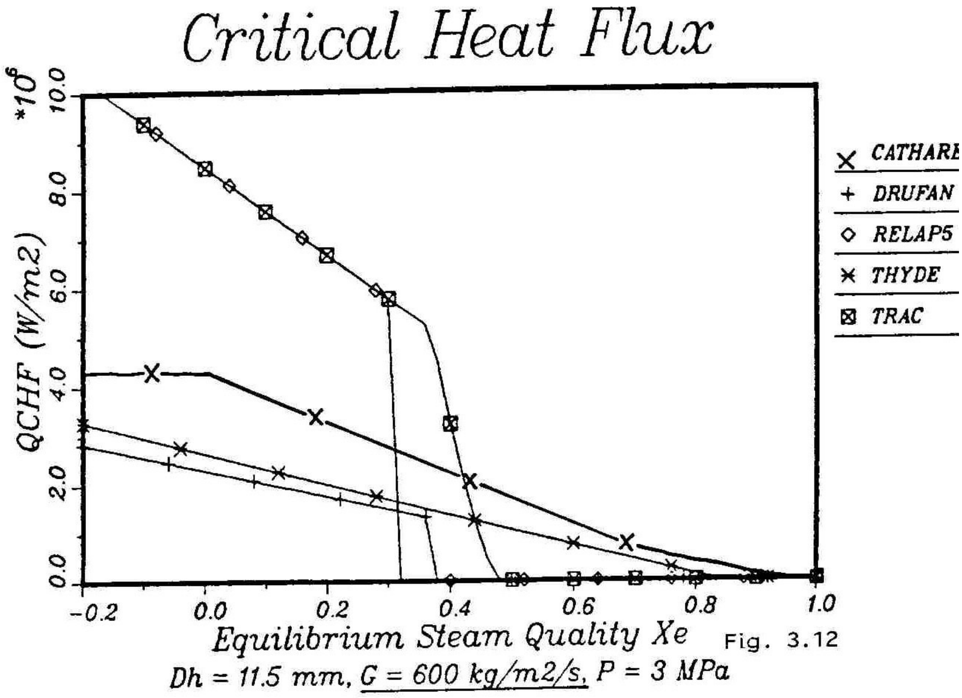
Critical heat flux

The Fig. 56 and Fig. 57 show comparisons between various T/H Codes for the calculation of CHF heat rate for RELAP5/Mod2:



Critical heat flux at P = 7 MPa

Fig. 56 – CHF comparison, 7 MPa (RELAP5/Mod 2)



Critical heat flux at P = 3 MPa

Fig. 57 – CHF Comparison 3 MPa (RELAP5/Mod2)

As it can be seen from the two Figures the differences between various codes can be important.

The CHF correlations have been improved in RELAP5/Mod3.3 ([7]-Vol. IV):

The heat transfer coefficients are determined in one of five subroutines: DITTUS, PREDNB, PREBUN, PSTDNB, and CONDEN. Subroutine CONDEN calculates the coefficients when the wall temperature is below the saturation temperature based on the partial pressure of steam.

- DITTUS is called for single-phase liquid or vapour conditions.
- PREDNB contains the nucleate boiling correlations for all surfaces except horizontal bundles and
- PREBUN is used for the outer surface of horizontal bundles of rods or tubes. PSTDNB has the transition and film boiling correlations.
- CHFKUT calculates CHF for horizontal bundles and
- CHFCAL determines the CHF for all other surfaces using a table lookup method.
- RELAP5/3 users may activate a new set of CHF correlations which were developed by the Nuclear Research Institute Rez in the Czech Republic (4.3-1,4.3-2): PG-CHF. These correlations replace the “CHF Table Lookup” method

There are four different formulations of the correlations (basic, flux, geometry, and power) with three different internal coefficient sets which are chosen by the user:

- The “basic” form uses the local equilibrium quality and the local heat flux.
- The “flux” form uses the local heat flux and the heated length including the axial power peaking factor.
- The “geometry” form uses the local equilibrium quality and the heated length including the axial power peaking factor.
- The “power” form comes from a heat balance method and can be used to calculate the critical power ratio (CPR).

When the first three forms are used, the resulting ratio represents the departure from nucleate boiling ratio (DNBR).

There are at least three distinct type of hydraulic models used to model reactor cores. The modelling terminology needs to be addressed to help readers understand the following paragraphs dealing with how to best use the PG-CHF correlations:

- Isolated sub-channel model - Code users are using an “isolated sub-channel model” when they use one heat structure connected to a hydraulic flow channel with no cross-flow. The contiguous stack of hydraulic volumes could represent a heated pipe or annulus, a fuel rod sub-channel, a rod bundle, or a complete core. Local coolant parameters in the “isolated sub-channel model” are determined in RELAP5 by applying conservation equations in an isolated (radially closed) stack of coolant cells.
- Bundle mean parameters model - This model has multiple heat structures connected to each hydraulic cell but, again the cells do not allow cross-flow. Use of the word “mean” is appropriate because the hydraulic conditions are the result of the integral of the heat flux from all the heat structures connected to a cell. This is the method applied in the present analysis.
- Sub-channel mixing model - This model uses mixing coefficients among adjacent coolant cells to determine local coolant parameters in every rod cell. The model is used in sub-channel codes (COBRA, VIPRE, etc.). Determined

local parameters depend on mixing coefficient values. If the mixing coefficient is zero the model transforms into the isolated sub-channel model and if the mixing coefficient is infinite the model transforms into the bundle mean parameters model.

Normally, users would choose the basic form of the correlation for the heated channel representing a tube, an internally heated annulus, or a rod bundle. However, depending on the nodalization used to model the heated channel, the choice of the flux form can be recommended.

Here is an example. When modelling the core region, the modeling practice is to place the hydraulic node boundaries at the position of grid spacers. The user may still need more detailed axial nodalization of the heat structure representing a fuel rod, e.g., two or more axial segments over one axial hydraulic node. If the basic form of the correlation is used in this case, local information for the bottom node is lost to some extent, because the code calculates volume averaged thermodynamic quality. If the flux form of the correlation is used in this case, local information is retained, because the heated length including the axial peaking factor is used instead of the thermodynamic equilibrium quality.

When modeling rod bundles, the flux form of the correlation can be used only if the isolated sub-channel thermal-hydraulic model is applied. The geometry form of the correlation may be of interest if the user prefers its combination of local parameters. Again, when modeling rod bundles, the geometry form of the correlation can be used only if the isolated sub-channel thermal-hydraulic model is applied.

The power form of the correlation would be chosen if the thermal-hydraulic analysis is performed to calculate the critical power ratio. For example, if a heated channel is operated in steady-state, the maximum power to avoid boiling crisis can be determined in a single RELAP5 run. Note that a series of trial and error runs would be needed if the other forms of correlations are used to solve this problem.

Again, when modeling rod bundles, the power form of the correlation can be used only if the isolated sub-channel thermal-hydraulic model is applied.

In the suggested procedure, the choice of one core pipe only with more recent Rez correlations is indicated, since the error in the Biasi correlation is excessive and non-conservative, Fig. 56 and Fig. 57.

4.2.12.1. Analysis performed

The analysis has been performed on LOFT and on a PWR-1000.

For both the studies the core is modelled in bundle each one with the heat structure. Each bundle volume is connected to the other bundle volume through a simple junction to simulate the cross flow.

4.2.12.1.1. <u>*LOFT analysis*</u>

The core is modelled with three bundle heat structures: central (8 rods), middle (203) and peripherical (1096) and three correspondent hydraulic volumes by a pipe with six volumes.

The analysis is performed on a LOCA for 120 s.

The Fig. 58 shows the RELAP model.

The Fig. 59 and Fig. 60 show respectively the core temperature and mass flow.

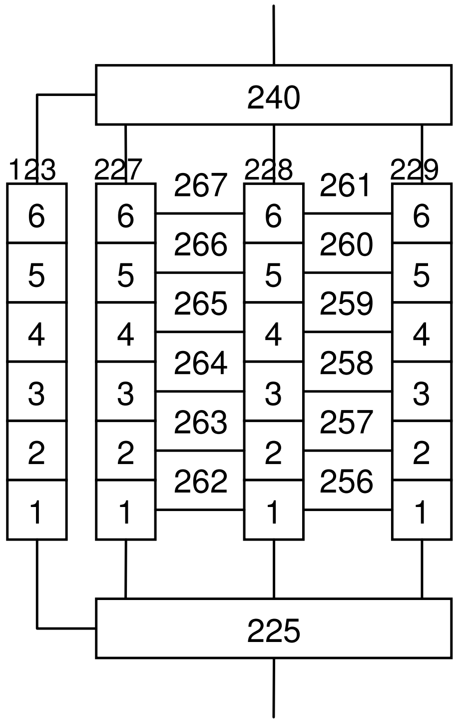


Fig. 58 – LOFT: core cross flow model

The curves in the Fig. 59 describe the mesh point temperature of the heat structure, 123 in the Fig. 60, related to the volume 4: the httemp 2384, in green, is related to the hot rod channel; the httemp 2374, in red, is related to the average temperature in the hot rod channel; the httemp 2364, in blue, is related to the average temperature in the average channel.

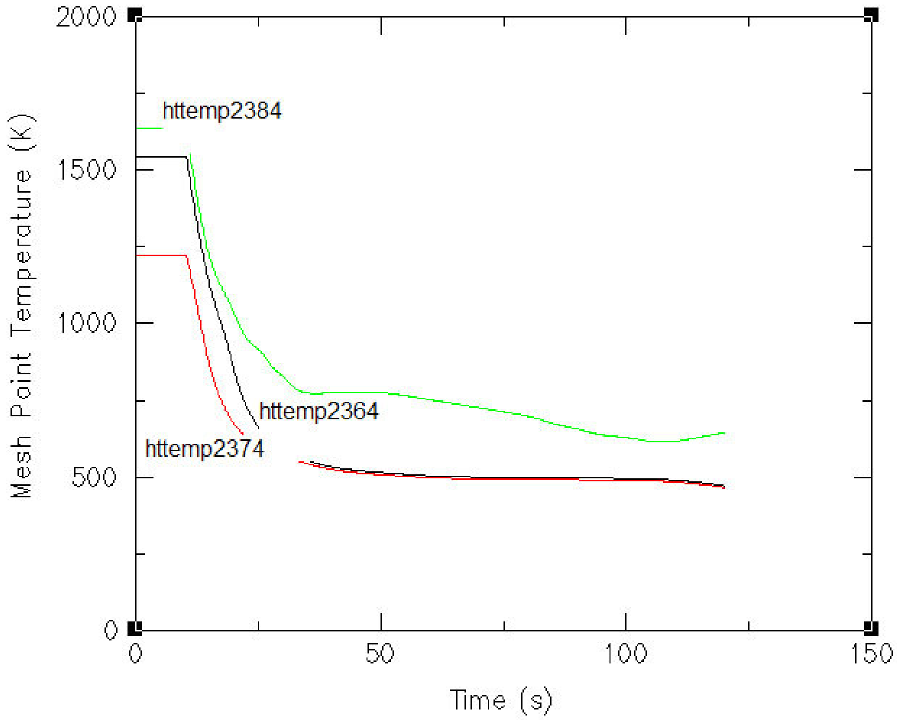


Fig. 59 – LOFT: core temperature

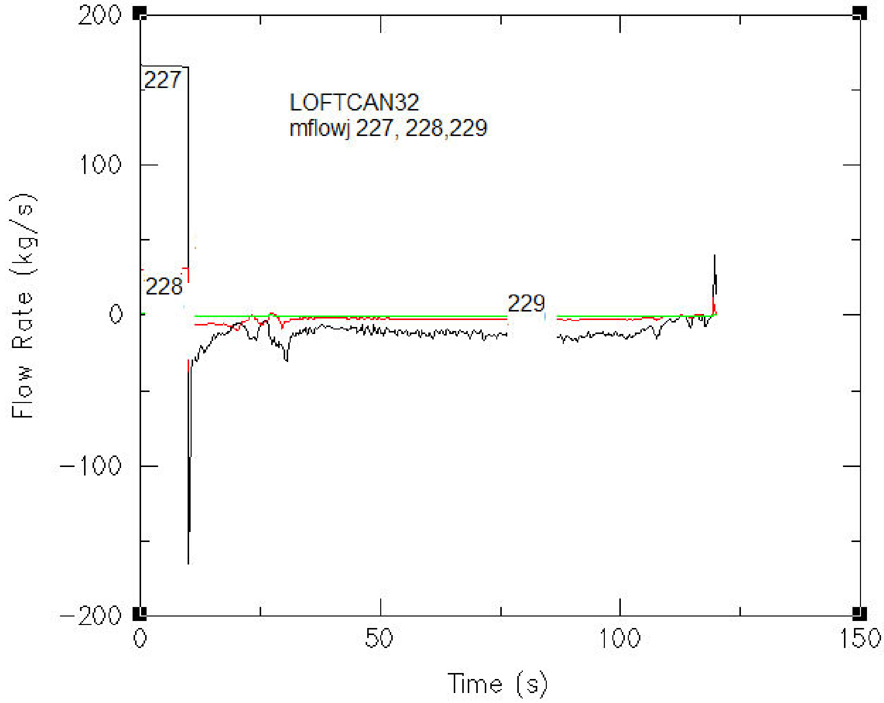


Fig. 60 – LOFT: core mass flow

Preliminary results do not show evident differences. Fig. 60 show a small cross flow in the hot channel (227), line in blue.

4.2.12.1.2. *PWR-1000 analysis*

The core is modelled with three bundle heat structures: central (16951 rods), middle (16952) and peripherical (16952) and three correspondents hydraulic volume by a pipe with 13 volumes and three pipes with single volume for the top of the core..

The analysis is performed on a LB-LOCA and SB-LOCA.

The Fig. 61 shows the RELAP model.

The Fig. 62 and Fig. 63 show respectively the core temperature and mass flow.

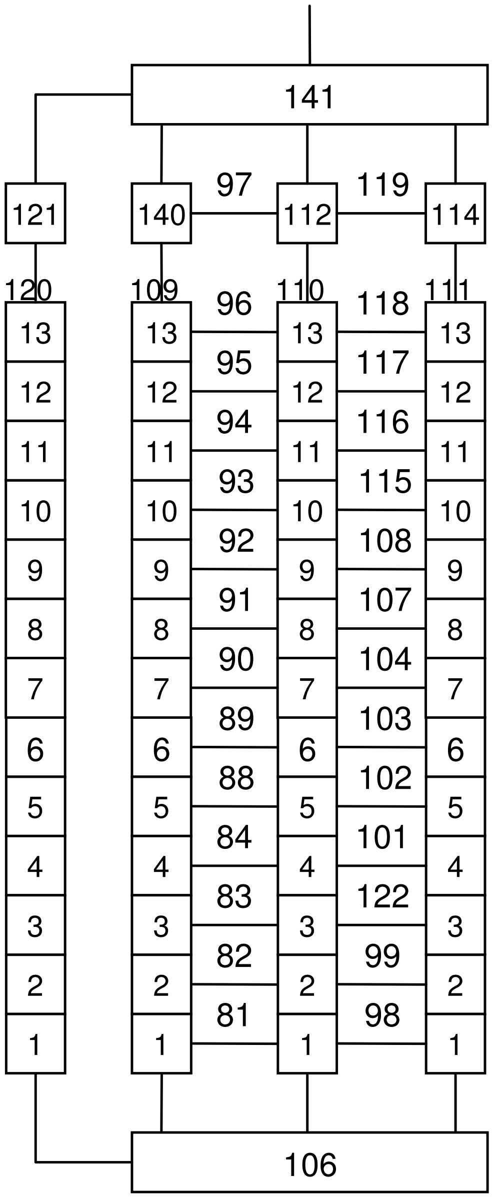


Fig. 61 – PWR-1000: core cross flow model

Fig. 62 shows the trend of the hot rod temperature at the level 12. It evident the effect of the cross flow model: the temperature is lower and goes done earlier than the temperature modelled without cross flow.

Fig. 63 shows the mass flow rate in the cross flow junction at the level 12 of the hot rod channel. The violet line is related to the calculation where it is supposed that central channel of the core is partially closed by “garbage” present in the core.

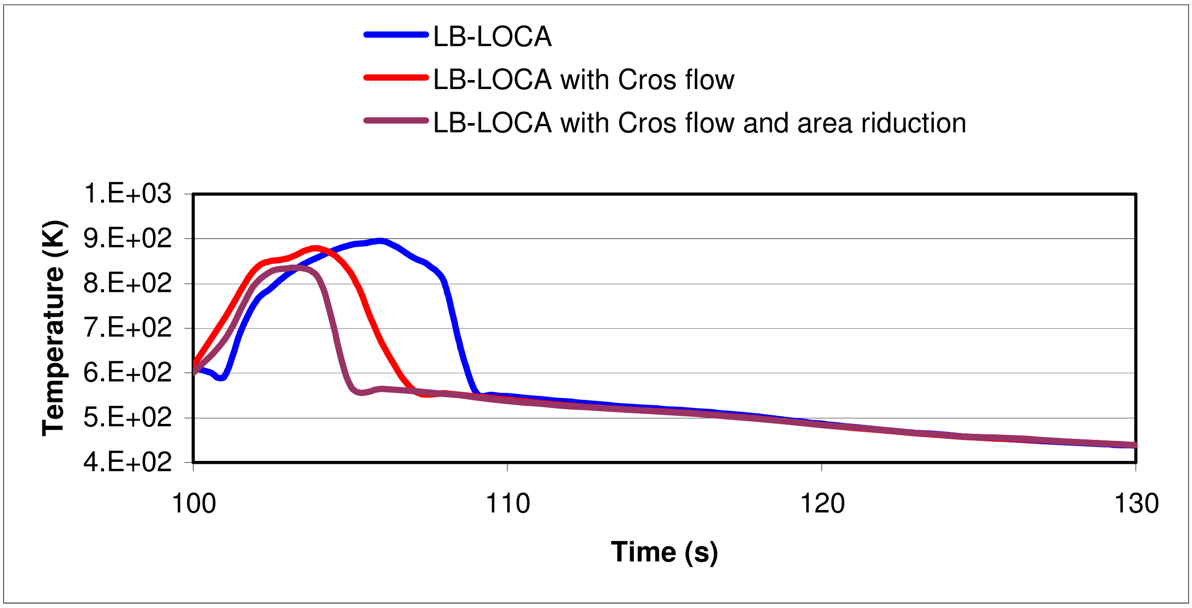


Fig. 62 – PWR-1000: core temperature comparison

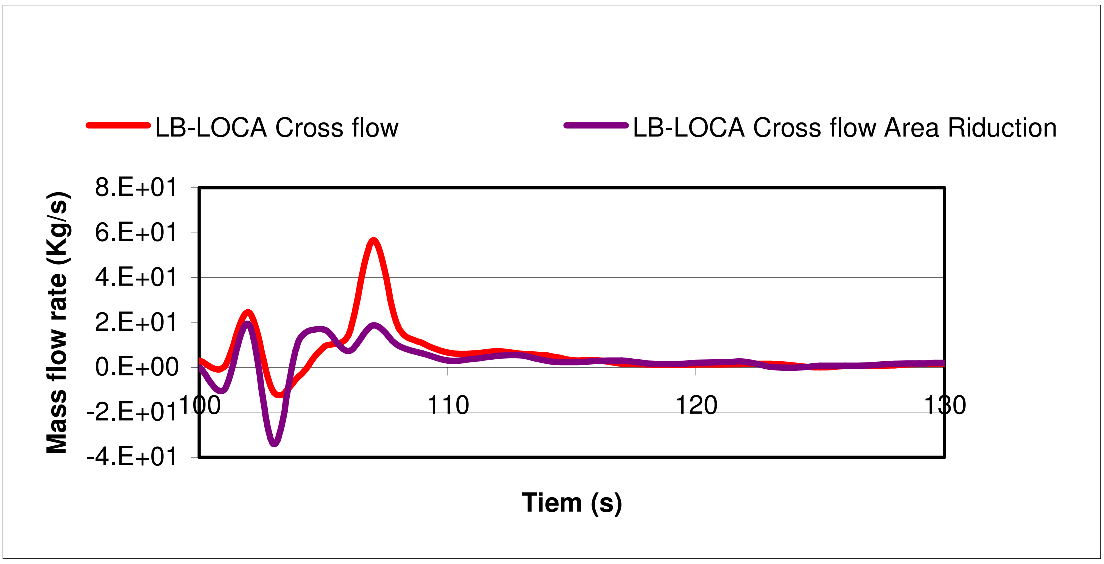


Fig. 63 – PWR-1000: core mass flow rate comparison

4.2.12.1.3. *Analysis results*

See also [29] [38]

The results achieved in the BEMUSE IV program [29] [38] provided the following achievements:

a) No difference for pressure time trends
b) No significant differences in predicting mass flow rates and inventories
c) Very similar PCT (few degrees of difference) and Time of PCT
d) About 80s difference for core quenching

The results, checked by analysis performed with LOFT, and a PWR 1000 follow BEMUSE results with difference on the quenching temperature, < 10s.

The use of the cross flow model is suggested in the BE applications

4.2.13. Heat transfer from uncovered rod bundle

The RELAP 5 default model is suggested here.

4.2.14. Break characteristics and flow

The most important situation involves efflux from a break when the fluid system is at high-medium pressure. Usually in this case the flow is choked and a special model is incorporated in RELAP.

A supplemental input, concerning a discharge coefficient K, has to be given to the model in order to account for geometry-specific two dimensional effects (the model is one-dimensional). The use of a discharge coefficient is required to account for multi-dimensional effects due to the break geometry being modelled. It is the code user task, then, to determine the necessary discharge coefficient values for the specific geometry. RELAP manual mentions a discharge coefficient of 0.85 for sub-cooled flows and a 0.82 coefficient for saturated steam flows. These suggestions are originated from the comparison between the computational model and tests (Marviken).

Here it is suggested to follow the suggestions of the RELAP Manual, unless special geometry features are present. Similarly, for the thermal non equilibrium constant ([7]Vol2, A, A.9.16) the default values of the code are suggested.

4.2.15. ECCS bypass

At § 3.4.2, the RG-1.157 states that the dominant processes governing ECC bypass are multidimensional; single-dimensional approximations justified through sufficient analysis and data may be acceptable. Cooling water that is not expelled, but remains in piping or is stored in parts of the vessel, should be calculated in a BE manner based on applicable experimental data.

The ECC bypass is modelled by a junction.

4.2.16. Noding near break and ECCS injection point

This subject is not implemented in the present work.

According to the RG-1.157, sensitivity studies should be performed on the noding and other important parameters to ensure that the calculations provide realistic results.

4.2.17. Frictional pressure drop

The models incorporated in RELAP are suggested here.

4.2.18. Pump modelling

It is suggested to use the RELAP model for pumps.

However, attention should be given to the heat generated by the pump and transmitted to the primary fluid, in particular for cases where its contribution may be important in the energy balance of the primary system.

The critical input values for that are: torque t, speed w (omega) and h (head) (equation 3-276 manual RELAP or 3.5-62 for the more recent edition, Vol.1). If these three terms are perfectly balanced (efficiency 1 of the pump) there is no increase in fluid temperature for this effect, otherwise yes.

4.2.19. Core flow distribution during blow-down and post-blow-down thermal hydraulics of a PWR.

The following considerations and requirements from RG 1.157 are incorporated in the suggested procedure.

For the purpose of calculation of the flow in the hottest region of the core during blow-down, the hottest region of the core should not be greater than the size of one fuel assembly.

Calculations of the flow in the hot region should take into account any cross flow between regions and any flow blockage calculated to occur during the blow-down as a result of clad swelling or rupture.

The numerical scheme should ensure that unrealistic oscillations of the calculated flow do not result.

The same considerations are valid for post-blow-down thermal hydraulics with further attention to the BE modelling of:

- resistance offered by the pumps
- carryover taking into account cross flow and core fluid distribution
- effect of compressed gas in accumulators
- effects of cladding swelling and ruptures.

4.2.20. General Options selection for RELAP5

The present paragraph analyses how the RELAP5 model the pipe.

The volume-related options are selected by the volume control flags that are required input for each hydrodynamic volume. The volume control flags are input as a packed word of the format “tlpvbfe.” The default options, obtained by entering 0000000, are generally recommended for use [7].

The t flag specifies whether or not the thermal front tracking model is active. The t = 0 option indicates it is not to be used, the t = 1 option indicates it is to be used. The thermal front tracking model provides a capability for eliminating numerical diffusion effects which artificially alter temperature profiles (such as hot liquid over cold liquid) in vertical regions. It is generally recommended that the t = 0 option be used. An exception is the simulation of pressurizer during in surge events, for which t = 1 is recommended in order to prevent over prediction of inter-phase condensation (which leads to a non-physical complete refilling of the pressurizer). The t = 0 option is recommended for pressurizer during simulation of out-surge events. The t = 1 option also may facilitate simulating refill of a steam region (such as the reactor vessel upper head) with cold water; calculation difficulties related to over prediction of condensation are sometimes encountered in this situation

The l flag specifies whether the level model is operative. The l = 0 option indicates the level model is not to be used, and l = 1 indicates it is to be used. This is a new option in RELAP5/MOD3. It is recommended that the l = 1 option be used in vertical pipes and tanks where a sharp level is desired (steam over liquid water).

The p flag specifies whether the water packing scheme is operative. The p = 0 option indicates water packing is to be used, and p = 1 indicates it is not to be used. This is a new option in RELAP5/MOD3; previously, the user did not have the option of deactivating the water packing scheme. It is recommended that the p = 0 option generally be used and the p = 1 option be reserved for situations where calculation difficulties are caused by repeated water packing occurrences. For

TMDPVOL, SEPARATR, JETMIXER, TURBINE, PUMP, and ACCUM components, the p flag is not used and 0 should be entered.
The v flag specifies whether the vertical stratification model is to be used. The v = 0 option indicates the vertical stratification model is to be used, and v = 1 indicates it is not to be used. This is another new option in RELAP5/MOD3; previously, the user did not have the option of deactivating the vertical stratification model. It is recommended that the v = 0 option generally be used. The v = 1 option is reserved for situations where the calculated vertical stratification behaviour is not desired. For TMDPVOL, SEPARATR, JETMIXER, ECCMIX, TURBINE, PUMP, and ACCUM components, the v flag is not used and 0 should be entered.
The b flag specifies the inter-phase friction model to be used. The b = 0 option indicates that the normal pipe inter-phase friction model is to be used. The b = 1 option indicates that the rod bundle inter-phase friction model is to be used. This is a third new option in RELAP5/MOD3; previously, the user did not have a choice of inter-phase friction models. The b = 0 option is generally recommended. For model regions with bundle geometries, such as steam generator secondary boiler regions and reactor core regions, the b = 1 option is recommended. For SEPARATR, JETMIXER, ECCMIX, TURBINE, PUMP, and ACCUM components, the b flag is not used and 0 should be entered.
The f flag indicates if wall friction is to be calculated. The f = 0 option specifies that wall friction is to be calculated, and the f = 1 flag indicates wall friction is not to be calculated. The f = 0 option is generally recommended. The f = 1 option is reserved for special situations where wall friction is undesirable. This situation might arise when a simplified model is constructed of a complex fluid region. In such situations, the input cell length (or that implied from volume and area) may be much longer than is prototypical. The f = 1 option could be used in this case to eliminate the excessive wall friction resulting from the long apparent cell length. For SEPARATR and PUMP components, the f flag is automatically set to 1, regardless of the value set by the user.
The e flag indicates whether phase non equilibrium or equilibrium options are to be used. In this terminology, “non-equilibrium” implies that the liquid and vapour phases may be at different temperatures. Conversely, “equilibrium” implies that the phases are constrained to be at the same temperature. The e = 0 flag indicates non-equilibrium assumptions are to be used; e = 1 indicates equilibrium assumptions are to be used. The e = 0 option is generally recommended. The e = 1 option is reserved for special situations where the non-equilibrium assumption causes difficulty in obtaining a reasonable solution because of insufficient thermal mixing between the phases. An example of the equilibrium option aiding a simulation is the down-comer of a once-through steam generator. Insufficient inter-phase condensation may prevent flow of sufficient steam through the aspirator; changing to the equilibrium option may enhance the condensation and improve the aspirator flow. Another example is the upper pressurizer dome region when spray is operating and the pressurizer level is high. In this situation, insufficient inter-phase condensation may be calculated and changing to the equilibrium option may improve the simulation. For ACCUM components, the e flag must be set to 0.
It is suggested that the recommendations of the RELAP manual should be followed in the proposed procedure.

4.2.21. Best estimate nodalization of systems: typical problems

The problem of the core nodalization (i.e cross flow or not, etc) has already been mentioned.

On passing, it is considered useful by experts to subdivide vertically, during nodalization, the various plant components by the same set of horizontal planes (horizontal slicing of the plant).

Although this situation is not usually foreseen, it may happen that some input data are uncertain for not satisfactory knowledge of plant details and for other reasons. This situation should be explicitly noted in the input or input description and its influence on important output parameters evaluated.

4.2.21.1. Spatial convergence

The spatial convergence is treated in ref. [80].

As it has been quite often misunderstood, a continuous refinement of the spatial resolution (e.g. a reduction of the cell sizes) does not automatically improve the accuracy of the prediction. There are two major reasons for this behaviour:

(1) The large number of empirical constitutive relations used in the codes have been developed on the basis of a fixed (in general coarse) nodalization.

(2) The numerical schemes used in the codes generally include a sufficient amount of artificial viscosity which is needed to provide stable numerical results.

A reduction of the cell sizes below a certain threshold value might result in severe non-physical instabilities.

From this it can be concluded that there exists no a priori optimal approach for the nodalization scheme. Even in relatively small scale integral test facilities, there exists multi-dimensional effects, especially with respect to flow splitting and flow merging processes: e.g. for the connection of the main coolant pipe to the pressure vessel.

The problem may become even more complicated due to the presence of additional bypass flows and a large re-distribution of flow during the transient. It is left to the code user to determine how to map these flow conditions within the frame of a predominately one-dimensional code, using the existing elements like branch components, multiple junction connections or cross-flow junctions.

These two examples show how the limitations in the physical modelling and the numerical method in the codes need to be compensated by an "engineering judgement" of the code user which, at best, is based on results of detailed sensitivity of assessment studies.

However, in many cases, due to lack of time or lack of appropriate experimental data, the user is forced to make ad-hoc decisions.

RELAP5 and ATHLET codes, with roughly 150 nodes, on the other hand the CATHARE code, with more than 300 nodes, and the TRAC code, with 250 nodes, can be distinguished. This choice is only partially due to the user. The code numerical structure also plays an important role for establishing the degree of detail of the nodalization model. As an example, RELAP5 code, owing to the Courant limit, needs nodes having length larger than few tens of centimeters, while CATHARE code does not have such a constraint and it allows a greater degree of freedom.

In principle, the best results for a physical simulation should be given by a nodalization with a number of nodes as large as possible, however, this is not strictly true for the current system codes. Otherwise, an optimal number of nodes can be recognized for each code for a given simulation problem. Directions for the selection of this number are not available in any code manual. Only the user experience can achieve this parameter, considering:

- the phenomena to be analyzed, in line with the available resources (CPU time needed, computers, etc.) and
- the goals of the study (e.g. sensitivity analyses aiming at the interpretation of physical phenomena, licensing calculations, etc.).

It should be noted that a large amount of sensitivity analyses can bring substantial improvements of nodalization parameters. In this way a coarse nodalization, with few elements, can produce better results than a fine nodalization with much larger number of elements.

Concerning the overall number of heat structure mesh points, the dominant influence of this parameter on the heat transfer mechanism must be stressed. In particular, the heat release from structures is strongly affected by the number of meshes.

4.2.21.2. Specification of state and transport property data

The calculation of state and transport properties is usually done implicitly by the code. However, in some cases, for example in RELAP5, the code user can define the range of reference points for property tables and, therefore, can influence the accuracy of the prediction. This might be of importance especially in more "difficult" regions, e.g. close to the critical point or at conditions near atmospheric pressure.

Another example is in relation to the fuel materials property data. The specification of fuel rod gap conductance (and thickness) is an important parameter, affecting core dry-out and rewet occurrences, that must be selected by the user. Usually this assumption, also connected with the actual fuel burn-up, is not reported as a user assumption.

4.2.21.3. Selection of parameters determining time step sizes

All the existing codes are using automatic procedures for the selection of time step sizes in order to provide convergence and accuracy of the prediction.

Experience shows, however, that these procedures do not always guarantee stable numerical results and, therefore, the user might often force the code to take very small time steps to pass through trouble spots. In some cases, if this action is not taken, very large numerical errors can be introduced in the evolution of any transient scenario and are not always checked by the code user.

4.2.21.4. Code input errors

To prepare a complete input data deck for a large system, the code user must provide a huge number of parameters (approximately 15 to 20 thousand values) which must be typed one by one. Even if all the codes provide consistency checks, the probability for code input errors is relatively high and can be reduced only by extreme care following clear quality assurance guidelines.

- With reference to a PWR typical plant, the choice of the hydraulic channel numbers in the generator and in the core (for a BWR plant same problem may occur in relation to the jet pumps and again to the number of channels [80]). A nodalization with only few channels can preserve the overall thermal energy, however cannot represent a non-uniform flow behaviour of the various channels (e.g. non-uniform flow distribution in the steam generator U or in the

lower plenum of reactor vessel or steam generator plenum, channel to channel oscillations, etc.);
- Passive structures of the plant. The consideration of all the material structures including vessel, piping and internal wall, as well as flanges, valves pump casings is almost impossible owing to limitations of computer memory. Approximations are needed and are usually done.

Finally, with regard to the form of loss coefficient items, the following remarks can be made:
- there is no theoretical model suitable to calculate this parameter in the wide variety of configurations encountered in modelling a typical nuclear plant or simulator; no relationship gives the dependency of these factors upon Reynolds number and local void fraction;
- experimental uncertainties are often connected to this parameter, usually derived from pressure drop measurements;
- loss coefficient values must account for the three dimensional effects of a one-dimensional code;
- some of the thermal-hydraulic model deficiencies can be adjusted by use of loss coefficients in an artificial way.

The data reported [80] give an idea of typical variation ranges associated with assumed values of these parameters (typically up to one order of magnitude). It should be noted that "all" the above values lead to "reasonable agreement" with experimental data, at least as far as the initial steady state is concerned .

In one specific case, ISP – 22, extensive post-test analysis and discussions at a specific workshop demonstrated that one of the main sources of discrepancies of interest also for plant calculations was: wrong calculation of initial mass of steam generators secondary side, notwithstanding a known code limitation in calculating the total mass inside a boiler, such as the secondary side of a steam generator.

For the post-test analysis, this limitation could be accommodated through proper adjustments of user selected parameters in the nodalization.

The following issues were important in the discrepancies between experiments and various calculations in ISP-26:
- The modeling of core and steam generator up-flow side as well as the different choices for the break flow calculation contributed to the results of the calculations significantly .
- The convergence of the solutions with respect to optimization of the noding was not assured. A "well balanced" nodalization, with a relatively fine noding in steam generators and core, as far as practical, may produce better results than an unbalanced nodalization where, as an example, a large number of nodes are used for the core but a coarse noding is used for steam generators.
- The calculated results are much affected not only by physical options but also by numerical options, i.e., convergence criteria and numerical scheme.

These options are selected by each user based on their own criterion. Thus, it would be desirable that the appropriate guidelines and information are provided to users for selecting input options
- The dead end volumes and the fluid temperatures inside dead ends may affect the overall energy and mass balance during the transient. These were not completely specified by experimentalists.

The following point, important also for real plants, was evidenced by ISP-27:

- achieving true steady state and overall system balance was specially important for this long transient.

This point was overlooked by most of the code users.

From what has been mentioned above it is quite clear that, at least for the presently existing codes and their limitations with respect to physical modelling and numerical techniques, there is no chance to completely avoid the influence of the code user on the predicted system behaviour. However, several methods by which the magnitude of the user effect might be reduced are indicated in the following paragraphs.

4.2.21.5. User training

As it has been described above, a large responsibility is imposed on the code user to provide an adequate description of the facility and to prepare the corresponding input data. This task can be fulfilled only if the user is fully aware of the physical modelling and the limitations of the codes, if he has a sufficient knowledge of the facility to be described and if he also has a good understanding of the major phenomena expected to occur during the transient. Therefore, user instruction and training might be the easiest way to improve the quality of the code predictions.

Unfortunately, there has been in the past a tendency to use ISPs as a type of "fast course" in training new code users without giving importance to above mentioned aspects. The rather poor results produced in these cases have largely contributed to the confusion on the user effect and the evaluation of the code capabilities. Based on ISP results, it is evident that a policy should be adopted to require the qualification of code users. In addition, it is also possible, as in the other industry branches, to require that only qualified users should be performing studies having some consequences, e.g. safety analysis.

An example of training is given by the recent 3D S.UN.COP (Scaling, Uncertainty, and 3D COuPled code calculations) seminars-trainings whose aim is to transfer competence, knowledge, and experience from about 30 recognized international experts coming from more than 10 different countries and institutions to analysts with a suitable background in nuclear technology. The program of the 3D S.UN.COP offers each year about 60 presentations and 100 hours of parallel code hands-on training subdivided in three weeks and covering the following topics: (a) system codes: evaluation, application, modeling and scaling; (b) international standard problems; (c) best-estimate in system code applications and uncertainty evaluation; (d) qualification procedures; (e) methods for sensitivity and uncertainty analysis; (f) relevant topics in best-estimate licensing approach; (g) industrial applications of the best-estimate plus-uncertainty methodology; (h) coupling methodologies and applications; (i) computational fluid dynamics codes. From the other side, the parallel hands-on training sessions on numerical codes (such as CATHARE, CATHENA, RELAP5, TRACE, and PARCS) allow the participants to achieve the capability to set up, run, and evaluate the results of a numerical tool through the application of the proposed qualitative and quantitative accuracy evaluation procedures. Finally, the 3D S.UN.COP seminars provides a forum for exchanges of ideas through scientific presentations and dialogue among representatives of the worlds of academy, research laboratories, industry, regulatory authorities, and international institutions.

4.2.21.6. *Improved user guidelines*

There has been in the past a continuous demand for user guidance which should be based on the results of a systematic code assessment programme. More detailed user guidelines are certainly a way to improve the quality of code prediction and to avoid larger mistakes and, in this sense, may also reduce the user effect. However, to be more realistic, such guidelines cannot give detailed recipes for all existing conditions and, therefore, cannot substitute for a trained and experienced code user.

4.2.21.7. *User discipline*

Even the best user guideline will not serve a useful purpose if the code user, as often found, is keen to invent "tricks" to drive the code prediction towards an experimental result or towards what the user expect to achieve. This does not contest the value of sensitivity studies which are often the only way to better understand code deficiencies.

What is meant here is a type of "tuning" of the results by a selection of completely unrealistic input values for physical models related parameters or boundary conditions, or by the extensive use of parallel channel and cross flow junctions to produce some "multi-dimensional" flow calculation that is beyond the code's capability. The result of this "tuning" is, at best, a compensation of errors which only contributes to confusion on the true prediction capability and, therefore, on the clear identification of code deficiencies and limitations.

4.2.21.8. *Quality assurance*

The preparation and testing of an input deck for a reactor or a related integral test facility is a tedious work which requires, even for a competent code user, an effort of about one man-year.

A reliable input deck can only be achieved if a clear quality assurance strategy is followed. Often this effort is not allocated (e.g. due to lack of time, money, or competence) and, consequently, incomplete or error-ridden decks are used.

A bad habit has been also that, in order to save time, existing decks are shared between different users who then introduce only minor modifications without a complete checking of the major part of the input data and without an understanding for what purpose they have been developed.

4.2.22. Long term cooling for Large Break LOCA

Long term cooling of the core for a large break LOCA is usually entrusted to the ECCS recirculation system. Since the duration in time of the cooling to be taken into consideration is certainly of several days, i.e. several hundred thousand seconds, before alternate cooling means are restored and put in line and since no system code suitable for the first period of the accident, less than 20-40000 s, can endure without instabilities and large errors for such a long time, then either a specially devised long time code has to be used or some code simplification has to be introduced for the long duration and a switch between two different codes or nodalizations has to be made at a suitable point in time.

On the other side, the need to extend the calculations for days after the accident arises from the need to take into account events whose probability is significant in the long time: passive components failure, partial clogging of sump filters, etc.

It is suggested that any BE code has a version or an extension suitable for long transient durations in cases, like the large LOCA, where this expanded calculation is opportune with suitable assumptions on component availabilities.

4.2.23. Interactions between primary/secondary systems and containment

The following subjects should be dealt with: sump low level, recirculation pumps cavitations, screen filter plugging.

4.2.24. Fission gases in the fuel gap and number of fissured rods in transients

The amount of fission gases can be calculated with methods already mentioned in 4.2.10. It is specific for each fuel history plant.

The number of fissured rods in transients is an important figure since it is directly connected with the amount of gap fission products released to the primary system and, after some reduction due to various phenomena, to the environment [7, Chp.3].

On a statistical basis, the amount of volatile fission products contained, during normal operation, in the gap between fuel and cladding (therefore released upon clad fissuring) is normally assumed to be comprised between 1 and 10 %. Ref. [50] suggests a value of 5% and this figure is suggested here too.

The amount of fissured rods during a large DBA is usually assumed in the range between 10% and 100%. The fissured gaps in normal operation are in the range between 1% and 2%.

Even in a best estimate calculation it is considered advisable to use rather conservative figures: 50% fissured rods is then assumed for large DBAs.

Volatile fission products except noble gases undergo a certain reduction for deposition in the primary system and in the containment.

4.2.25. General System assumptions in Best Estimate

The issue of Single Failure Assumption should be examined in the first place.

For the present work, it is suggested that the assumption of a single failure is retained in any BE calculation.

In fact, experience indicates that this event is sufficiently frequent to deserve special consideration.

Another useful check of any calculation is its sensitivity to intermittent or discontinuous operation of safety systems, which may also occur, see TMI case, and may introduce peculiar and confusing system situations.

From RG.1.157 the following consideration can be extracted: no list of BE code features could be all–inclusive because the important features may vary according to the transient being calculated and the required accuracy of the calculation. The search for other important issues and features should therefore continue.

5. DISCUSSION OF THE RESULTS

This chapter aim to summarize and discuss the main achievements of the present work. Starting from the status of the BE applications and licensing needs, the results on decay heat power, gamma and beta decay heat distribution and cross flow analyses are discussed. A summary close the chapter together with a table.

5.1. Status of the best estimation applications

Use of BE computer codes either combined with conservative input data or with realistic input data, but associated with evaluation of uncertainty of results are two acceptable options for demonstrating that the safety is ensured with sufficient margin, offered by the recently made advances [41] Tab. 4.
Considerable interest on use of BE tools is available from research organizations, utilities and regulators but with different objectives [42]:

- Regulator is interested that acceptance criteria are fulfilled with high confidence
- Utility is interested in "useful" results aiming at reduce conservatisms
- Research wishes to improve practicability of methods

Different techniques for the uncertainty propagation in the thermal-hydraulic code calculations are identified [66,93].
The review showed that initially the BEPU plant applications were mostly limited to LB-LOCA. Applications are also done to SBLOCA and non-LOCA transients.
The CSNI results on UMS [114] and the recent IAEA safety report [66] concluded that the existing qualified BEPU methods seem mature enough for application, while the future research will most likely focus on the codes with internal assessment of the uncertainty [93].
While the choice of an uncertainty method depends on what the end users will accept and in what features they are interested [114], it is necessary to incorporate into the licensing process the regulatory basis for the use of a realistic calculation method in the safety analysis. As a first step, trial applications of a realistic calculation are useful for the regulator to assess the feasibility of moving towards a more methodical use of realistic calculations in licensing. Such methods can become an established part of the licensing framework and guidelines. Apart from the trial licensing application, the industry can take a leading role in developing the methodology for its use in safety analysis and to validate it. The regulatory guideline should describe a process for the development and assessment of the evaluation models that can be used to analyse transient and accident behaviour. It should also provide guidance on realistic accident analyses, thus affording a more reliable framework for estimating the uncertainty when investigating transient and accident behaviour [66].

5.2. Decay heat power

The analyses of the decay heat power on the transients considered, SBO and SB-LOCA, show that in both scenarios the curves have the same slopes and the same starting value after the steady state. In the SBO the curves maintain the same differences, in the SB-LOCA in the first 200 s they overpass each other and then have the same behaviour of the SBO.

It should be noted that the conservative calculation, DH1 and SBO1, give an higher decay heat power and a delay of 30 s. From the other side the more realistic one, DH8,9 and SBO8,9 give the lower power and a shorter time in decreasing [§4.2.4.1].
The maximum ratio between the ANS 73 and various ANS standards, as reported in Tab. 6 and Tab. 7, amount to about 1.1.
It has also to be noted that the ANS-94-4 values practically coincide with the values of the International Standard ISO -10645, decay heat of nuclear reactors.
The various ANS Standards Reports and the ISO Report give also guidance for uncertainty evaluations.
In the conclusion of the analysis of the results [§ 4.2.4.1.7], it is suggested to use in a BE analysis the ANS 94-4 or the ISO -10645. Such option is proposed because it is available in RELAP5 library. The more recent ANSI/ANS-5.1-2005 is a valuable alternative, even if it was not compared in the present work.
The results obtained give also suggestion to apply the real power history. Tools are already available to be applied and to perform such BE analyses.
This recommendation is against to the AREVA experience in its licensing calculation [118, § A.5.7] for the LB-LOCA:
"*The decay heating described by the standard can be used for many types of calculations including LOCA analysis. However, considerations for LOCA are somewhat different from other applications. LOCA is a hypothetical event which must be analyzed prior to reactor operation. Thus, the operating history and the concentration of fissionable isotopes will not be known prior to a LOCA. Fortunately, simplifying assumptions can be made which allow calculation of a realistic but slightly conservative decay heat curve as a function of time using the 1979 standard. The decay heat calculated with these assumptions bounds the more detailed decay heat curves that would result if the conditions at the initiation of LOCA were known. The assumptions are:*
- *infinite operating time at full power.*
- *All fissions assumed from U 235*
- *200 MeV / fission (conservatively low)*
- *One standard deviation total decay heat*"

However, within the framework of the present work, consideration about the use of the real power history is suggested.

5.3. Gamma redistribution maximum energy reduction

Tab. 11 shows all the results obtained for a time after shutdown of 100 s for LOFT and large reactor coarse model.
In the table the following symbols have been adopted
$\Delta_{r,1/2}$= mid height width of the produced energy peak [cm]
E_p= maximum produced specific energy [w/cm^3] of photons
E_a= maximum absorbed specific energy [w/cm^3] of photons
σ^2=square of the standard deviation of Gaussian distributions of produced energy
LOFT, for LOFT core cases with final letter h for "hot rod" case, sin for sine distribution and a number for σ^2 used
LG, for large reactor with the same meaning of the final letters as for LOFT.

Tab. 11 – Results for time after shutdown of 100 s

$\Delta_{r,1/2}$[cm]	E_p[w/cm^3]	E_a[w/cm^3]	(E_p- E_a)/ E_p , %	E_a / E_p	Case
2	3.33	2.34	30	0.7	LOFTh
8.48	3.33	2.94	12	0.88	LGh
18	2.38	1.89	21	0.79	LOFT 0.05
19.5	2.38	2.1	12	0.88	LOFT 0.12
46	2.38	2.16	9	0.91	LOFTsin
67.5	2.38	1.85	22	0.78	LG 0.01
118.4	2.38	2.046	14	0.86	LG 0.24
192.4	2.38	2.36	0.8	0.99	LG sin

An attempt to correlate the maximum energy reduction due to gamma redistribution in core has been made: the ratio between maximum absorbed and produced gamma energies has been correlated with the mid-height width of the produced energy peak. Fig. 64 has been obtained.

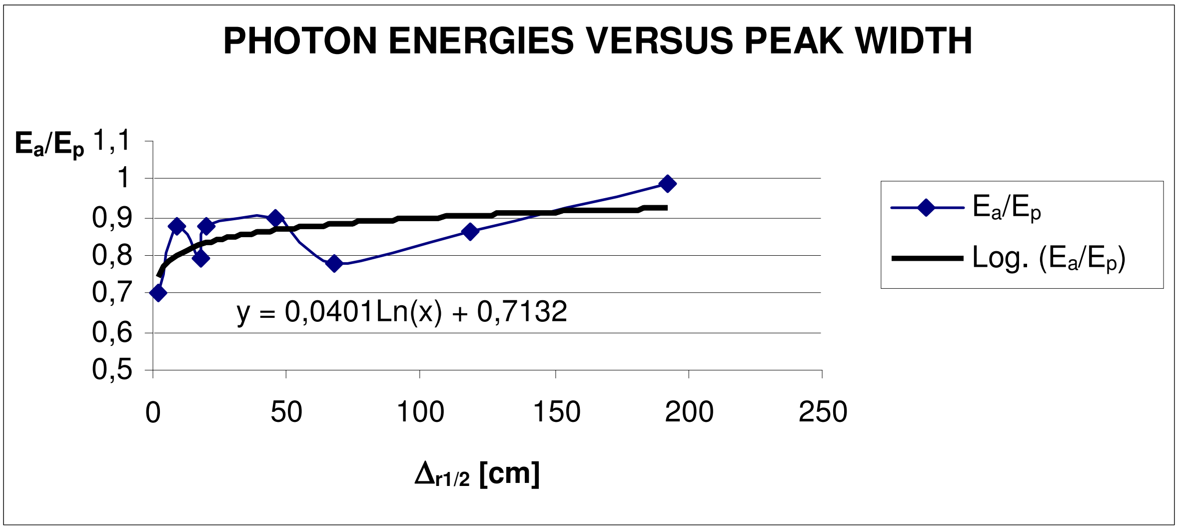


Fig. 64 – Correlation between absorbed and produced energies ratio with mid-height peak width

The calculated cases, listed in Table 10, have been used.
As a first approximation, it can be said that

– the average energy reduction ratio of 0.82 for gamma (18% peak reduction) can be used with tolerable error up to a mid-height width of 120 cm and at 100 s after shutdown;
– at 10000 s after shutdown a figure of 77% can be used instead of 82%.

Above this value of mid-height width, no peak energy reduction is warranted.

5.4. Cross flow model

The results of the present analysis provide the same achievements of the BEMUSE IV program [29] [38] [§ 4.2.12]:

a) No difference for pressure time trends
b) No significant differences in predicting mass flow rates and inventories
c) Very similar PCT (few degrees of difference) and Time of PCT

d) About 80 sec difference for core quenching.
The results, checked by analysis performed with LOFT, and a PWR 1000 follow BEMUSE results with difference on the quenching temperature, < 10s.
The use of the cross flow model is suggested in the BE applications

5.5. Summary of the results

The analyses performed in the present work show the efforts implemented to develop new licensing BEPU methods, from one side, and the trend focusing on the codes with internal assessment of the uncertainty, from the other side.
Could be natural to consider that the real question is: how convince a regulator to adopt BE evaluations? Perhaps it is already accepted and it simply happens that proponents, in some cases or countries, are still not using it.
The BE methods, from one side, look at to provide a real description of the plant and of the phenomena involved in an accident or operational transient, to give confidence that the safety requirements are fulfilled. From the other side, they show that technology and the understanding of the phenomena are improved since 1954, when the 10 CFR 50 was developed.
Up to the date, many steps have been realised to better regulate the safety assessment and safety analysis (e.g. RG-1.157 and 1.203, the IAEA safety standards, the application of BEPU in some licensing applications, etc.).
Now a new step can be implemented that come from a consideration on the safety margins.
Fixed the safety limits and the acceptance criteria, to make conservative a parameter, with reference to those, i.e. Fig. 6, it is needed to increase the margins to the acceptance criteria. In the same way to make realistic the same parameter, it is needed to decrease the safety margins with reference to the acceptance criteria, i.e a realistic calculated result of a safety parameter may allow to decrease the margin to acceptance criteria, for example by power up-rates.
The analysis, reported in Chap.4, is summarized in Tab. 12. It suggests a way, a “protocol”, how to reach a definition of a *less conservative safety margin* with respect to the existing ones, for each related value and for the code used by a qualified user.
This is a preliminary results subject to a discussion with the international scientific community. It is expected that the methodology applied in the present work can be discussed by the regulatory body and the scientific community in order to define some standard BE procedures or “protocols” to be used with a specific, widely used code (e.g. RELAP5/MOD3, CATHARE, etc.) and with a set of defined safety (uncertainty) margins.

Tab. 12 – Summary table

	Issue	Recommendation	Notes
1	General probabilistic reference for BE	The MODE (most probable value or situation) is suggested in general, unless other considerations prevail	
2	UO_2 Heat conductivity	INSC values	Initial stored energy in fuel
3	UO_2 Heat capacity	INSC values	Same as above
4	Fuel-cladding gap conductance	Lanning model [2, 11] , incorporated in RELAP Code models	Same as above
5	Cladding thermal conductivity and heat capacity	INSC	
6	Decay Heat	ANS Standard (1994-4) or the equivalent ISO-10645 Standard	
7	Metal-Water reaction rate	(ORNL/NUREG 17)	
8	Heat transfer in the reactor and in SGs	RELAP Heat Transfer Package correlations	
9	Rupture and swelling of cladding of fuel rods	NUREG 630	
10	Hottest cladding point in core	Peaking factors unless the transient entails strong flux distortions (3D-NK [77])	
11	CHF and flow in hottest channel	Table lookup method and flow from modelling of hot channel and of other typical channels with cross flow (unless unavoidable instabilities arise)	
12	Break characteristics and flow	Suggestions of the RELAP manual	
13	ECCS bypass	Ordinary junction	
14	Noding near break and injection point	RG 1.157 recommendations	
15	Friction pressure drop	RELAP models	

Tab. 12 – Summary table

	Issue	Recommendation	Notes
16	Pump	RELAP model	Attention given to heat generated by pump and transmitted to fluid, when important for heat balance
17	Flow distribution in core during blow-down and post blow-down	RG 1.157 recommendations and item CHF above	
18	General RELAP thermal hydraulic options	RELAP manual recommendations (water packing and vertical stratify model yes, others no)	
19	Plant nodalization	Many recommendations in §4.2.21	
20	Long term cooling for large LOCA	It is suggested that any BE code has a version or an extension suitable for long transient durations	
21	Interactions with containment	Consider containment pressure, sump level, filter plugging	
22	Fission gases in fuel gap and percentage of fissured rods	5% and 50%	Large DBAs
23	General System assumptions	Single Failure Assumption	Also passive components failure for long transients

6. CONCLUSIONS

From the analysis performed and the results achieved it is possible to reach the following conclusion:

Application of BE methods into the licensing process:

- The framework to introduce the BE analysis into the licensing process is still an open issue
- BE analysis with evaluation of uncertainties is the only way to quantify existing safety margins
- General trend in licensing calculations, from fully conservative analysis to BE analysis with evaluation of uncertainties
- At present, many organisations prefer using best BE codes with conservative parameter values, initial and boundary conditions: fully acceptable way from IAEA standards point of view
- Best estimate analysis offers useful results from the point of view of a utility wanting to reduce conservatisms and utilize safety margins
- Uncertainty evaluations is considered in many cases as time consuming: need to improve practicability of methods.

Numerical results:

- Decay Heat
    - *a)* The comparison between the conservative decay heat standard curve and the more realistic one gives a ratio of 1.1.
- Gamma distribution
    - *a)* For the small reactor without hot rod the reduction of absorbed γ power versus produced power is equal to about 10% at 100s and to about 15% at 10000 s after shutdown; the case of a large reactor with local neutron flux hills (due, for example, to specific control rod management strategies) can approximate the case of a small reactor;
    - *b)* For the small reactor with hot rod, the γ peak at the hot rod practically disappears and the overall (sine distribution plus hot rod) reduction in peak energy is equal to about 30% at 100 s (this is considered the most significant result since the γ redistribution, with corresponding γ+ β power decrease of 15%, may entail a calculated PCT reduction of the order of 100 – 150 K); for a large reactor, the peak energy is reduced by 12% instead of 30%.
    - *c)* For the large reactor without hot rod the corresponding γ reduction is much lower (about 1% for a sine distribution at 100 s), with coarse nodalization, and of the same magnitude of LOFT for a fine nodalization.
    - *d)* An attempt to correlate the maximum energy reduction due to gamma redistribution in core has been made: the ratio between maximum absorbed and produced gamma energies has been correlated with the mid-height width of the produced energy peak. As a first approximation, it can be said that the average energy reduction ratio of 0.82 for gamma (18% peak reduction) can be used with tolerable error up to a mid-height width of 120 cm and at 100 s after shutdown; at 10000 s after shutdown a figure of 77% can be used instead of 82%.

Above this value of mid-height width, no peak energy reduction is warranted.

- Cross flow
    a) No difference for pressure time trends
    b) No significant differences in predicting mass flow rates and Inventories
    c) Very similar PCT (few degrees of difference) and time of PCT
    d) About 80 s difference for core quenching
    e) The use of the cross flow model is suggested in the BE applications

Main achievement:

- A definition of BE evaluation has been proposed
- A proposal for a BE protocol to be applied into the licensing process has been developed
- Such protocol proposal is expected to be discussed in the international scientific community.

REFERENCES

1. IAEA, "Defense in depth", *IAEA INSAG series No. 10*, IAEA, Vienna (1996)
2. IAEA, "Uncertainty Evaluation in Best Estimate Safety Analysis for Nuclear Power Plants", *IAEA TECDOC Series* (Draft version), IAEA, Vienna (2003).
3. CSNI "Uncertainty Methods Study (UMS), Volume I of the Report of the Uncertainty Methods Study for Advanced Best Estimate Thermal Hydraulic Code Applications", NEA/CSNI/R(97)35, 1998
4. US Nuclear Regulatory Commissions, "Compendium of ECCS Research for Realistic LOCA Analysis", NUREG-1230, 1987
5. Kline, S.J., "**Similitude and Approximation Theory"**, Springer Verlag, New York, NY 1986
6. Lerchl, G., Austregesilo, H., Glaeser, H., Hrubisko, M., Luther, W.: "ATHLET Mod 2.0 Cycle A Validation", GRS-P-1/Vol.3, GRS, October 2003
7. "RELAP5/MOD3.3 Code Manual Volume III: Developmental Assessment Problems", *NUREG/CR-5535*, ISL, December 2001
8. 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light water cooled nuclear power reactors," Appendix K to 10 CFR 50, "ECCS Evaluation Models," *U.S. Federal Register*, Vol. 39, No. 3, January 1974
9. CSNI "Status Summary on Utilization of Best-Estimate Methodology in Safety Analysis and Licensing"; *NEA/CSNI/R(96)19*, October 1996
10. IAEA, "Safety of Nuclear Power Plants: Design", *Safety Standards Series No. NS-R-1*, IAEA, Vienna (2000).
11. US Nuclear Regulatory Commissions, 10 CFR 50 "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).
12. H.C. Simpson: "Some Thermo-hydraulic Problems associated with the Safety of Water Cooled Nuclear Reactors"; 8th International Heat Transfer Conference, San Francisco, U.S.A., 1986
13. US Nuclear Regulatory Commissions, RG 1.157 "Best-Estimate Calculations of Emergency Core Cooling System Performance", May 1989.
14. **BEMUSE Phase II Report**, NEA/CSNI/R(2006)2, 2006
15. Pourgol-Mohamad, M., Mosleh, A., Modarres, M.: "Treatment of Uncertainties; Output Updating in Complex TH Computational Tools"; 14th International Conference on Nuclear Engineering ICONE, Miami, U.S.A., 2006
16. CSNI Report "User Effect on the Transient System Code Calculations", *NEA/CSNI/R(94)35*, January 1995
17. Ashley, R., El-Shanawany, M., Eltawila, F., D'Auria, F.: "Good Practices for User Effect Reduction", *NEA/CSNI/R(98)22*, November 1998
18. Boyack, B.E., Catton I., Duffey, R.B., Griffith, P., Katsma, K.R., Lellouche, G.S., Levy, S., Rogatgi, U.S., Wilson, G.E., Wulff, W., Zuber, N.: "An Overview of the Code Scaling, Applicability, and Uncertainty Evaluation Methodology", *Nuclear Engineering and Design* 119 (1990)

19. Wilks, S.S., “Determination of Sample Sizes for Setting Tolerance Limits”; *Ann. Math. Statistic.*, 12 (1941), 91-96
20. Wilks, S.S., “Statistical prediction with special reference to the problem of tolerance limits”, Ann. Math. Statist., 13 (1942), 400-409
21. CSNI ”Integral Test Facility Validation Matrix for the Assessment of Thermal-Hydraulic Codes for LWR LOCA and Transients”, *NEA/CSNI/R(96)17*, July 1996
22. Wilson, G.E., Boyack, B.E. “The Role of the PIRT Process in Experiments, Code Development and Code Applications Associated with Reactor Safety Analysis”, *Nuclear Engineering and Design*, 186 (1998).
23. Petruzzi, A., D’Auria, F.: “Accuracy Quantification: Description of the Fast Fourier Transform Based Method (FFTBM), University of Pisa Technical Report, DIMNP NT 556(05), Pisa, May 2005
24. Bonuccelli, M., D’Auria, F., Debrecin, N., Galassi, G.M.: “A Methodology for the Qualification of Thermal-hydraulic Code Nodalization“, 6th NURETH Conference, Grenoble, France, 1993
25. Crécy, A., Bazin, P.: ”Quantification of the Uncertainties of the Physical Models of CATHARE 2“. BE2004 – International Meeting on Updates in Best Estimate Methods in Nuclear Installation Safety Analysis, Proceedings, pp.111, Washington D.C., USA, November 2004
26. Kloos M., Hofer E., “SUSA-PC, Personal Computer Version of the Program System for Uncertainty and Sensitivity Analysis of Results from Computer Models, Version 3.5, User’s Guide and Tutorial”, GRS (2002)
27. J.F.Perkins, R.W. King, Energy Release form the Decay of Fission Products, Nuclear Science and Engineering 3, 726-746 (1958)
28. P. D. Bayless, “Experiment Data Report for LOFT Large Break Loss-of-Coolant Experiment L2-5”, NUREG/CR-2826, EGG-2210, (pp 317), 1982.
29. **BEMUSE Phase IV Report**, DEGB LB-LOCA in NPP, NEA/CSNI/R(2007)2, 2007
30. U.S.NRC,“A Short History of Nuclear Regulation, 1946-1999”, NUREG/BR-0175, Rev. 1
31. IAEA, “The Safety of Nuclear Installations”, *Safety Series No. 110*, IAEA, Vienna (1993)
32. G. Petrangeli, **Nuclear Safety**, Elsevier, Oxford, 2006
33. IAEA, “Accident Analysis for Nuclear Power Plant”, *Safety Reports Series No. 23*, IAEA, Vienna (2002).
34. IAEA, “Incorporation of advanced accident analysis methodology into safety analysis report”, *IAEA-TECDOC-1351*, IAEA, Vienna (2003)
35. IAEA, “Safety of Nuclear Power Plants: Design”, *Safety Standards Series No. NS-R-1*, IAEA, Vienna (2000)
36. IAEA, “Best Estimate Approach in Licensing with Evaluation of Uncertainties”, Summary Report, J4-TM-26904, Technical Meeting on Use of a Best Estimate Approach in Licensing with Evaluation of Uncertainties, The University of Pisa, Pisa, Italy, 12 – 16 September 2005
37. F. D'auria, A. Bousbia-Salah, A. Petruzzi, A. Del Nevo, “State of the Art in Using Best Estimate Calculation Tools in Nuclear Technology”, Nuclear Engineering And Technology, Vol.38 No.1 February 2006
38. **BEMUSE Phase V Report**, CIAU-RELAP uncertainty Analysis of DEGB LB-LOCA in Zion NPP, NEA/CSNI/R(2008)2, 2008

39. 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light water cooled nuclear power reactors,” Appendix K to 10 CFR 50, “ECCS Evaluation Models,” *U.S. Federal Register, Vol. 39, No. 3*, January 1974
40. Statute of the IAEA, 28 December 1989
41. IAEA,” Safety Margins of Operating Reactors Analysis of Uncertainties and Implications for Decision Making”, *TECDOC-1332, 01*/ 2003
42. H. Glaeser, “Best-estimate approach in German licensing,”Lecture 28A, Seminar and Training on Scaling, Uncertainty and 3D Coupled Calculations in Nuclear Technology (3D S.UN. COP 2007), January-February, College Station, Texas, USA.
43. E. Hofer, et al., „Uncertainty and sensitivity analysis of accident consequence submodels”, Proc. Int. ANS/ENS Topical Mtg. on Probabilistic Safety Methods and Applications, San Francisco, CA 1985, Vol. 2, Electric Power Research Institute, Palo Alto, CA (1985) 130
44. Prošek, B. Mavko, “Review of Best Estimate Plus Uncertainty Methods of Thermal-Hydraulic Safety Analysis”, International Conference Nuclear Energy for Central Europe 2003 Portorož, Slovenia, September 8-11, 2003
45. M. R. Galetti Camargo, F. D´Auria, “Licensing Experience with BE in Brazil: Regulatory Audit Analysis”, J4-TM-26904, Technical Meeting on Use of a Best Estimate Approach in Licensing with Evaluation of Uncertainties, The University of Pisa, Pisa, Italy, 12 – 16 September 2005
46. M. R. Galetti Camargo, F. D´Auria, “Technical and Regulatory Concerns in the Use of Best Estimate Methodologies in an LBLOCA Analysis Licensing Process”, ANS Annual Meeting, San Diego, CA, June 2005
47. M. R. Galetti Camargo, ” Regulatory Scenario for the Acceptance of Uncertainty Analysis Methodologies for the LB-LOCA and the Brazilian Approach”, Hindawi Publishing Corporation Science and Technology of Nuclear Installations, Vol. 2008
48. E. Uspuras, S. Rimkevicius, A. Kaliatka,” Best-Estimate Approach for Ignalina Npp Licensing Process”, ICONE 13, Beijing, China, May 16-20, 2005
49. J. Macek, R. Meca, L. Denk, J. Hadek,” Utilisation of Best Estimate System Codes and Best Estimate Methods in Safety Analyses of VVER Reactors in the Czech Republic”, J4-TM-26904, Technical Meeting on Use of a Best Estimate Approach in Licensing with Evaluation of Uncertainties, The University of Pisa, Pisa, Italy, 12 – 16 September 2005
50. H. Glaeser, “Best estimate approach in German licensing”, J4-TM-26904, Technical Meeting on Use of a Best Estimate Approach in Licensing with Evaluation of Uncertainties, The University of Pisa, Pisa, Italy, 12 – 16 September 2005
51. N.K. Popov, A. Abdul-Razzak, R.B. Duffey, and H.E. Sills, “Overview of Development and Application of Best Estimate Analysis and Uncertainty Methodology at AECL”, J4-TM-26904, Technical Meeting on Use of a Best Estimate Approach in Licensing with Evaluation of Uncertainties, The University of Pisa, Pisa, Italy, 12 – 16 September 2005
52. O. Sandervåg, “Licensing of Power Uprates in Sweden”, J4-TM-26904, Technical Meeting on Use of a Best Estimate Approach in Licensing with

Evaluation of Uncertainties, The University of Pisa, Pisa, Italy, 12 – 16 September 2005
53. Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance", U.S. Nuclear Regulatory Commission, May 1989
54. C. Frepoli," An Overview of Westinghouse Realistic Large Break LOCA Evaluation Model", Hindawi Publishing Corporation Science and Technology of Nuclear Installations, Vol. 2008
55. U. S. Nuclear Regulatory Commission, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Acceptability of the Topical Report WCAP-12945 (P) 'Westinghouse Code Qualification Document for Best-Estimate Loss of Coolant Accident Analysis' for referencing in PWR Licensing Applications,Westinghouse Electric Corporation"
56. M. Y. Young, S. M. Bajorek, M. E. Nissley, and L. E. Hochreiter, "Application of Code Scaling Applicability And Uncertainty Methodology to the Large Break Loss of Coolant," Nuclear Engineering and Design, vol. 186, no. 1-2, pp. 39–52, 1998
57. K. Takeuchi and M. Y. Young, "Assessment of flooding in a best estimate thermal hydraulic code (WCOBRA/TRAC)," Nuclear Engineering and Design, vol. 186, no. 1-2, pp. 225–255, 1998
58. K. Takeuchi and M. E. Nissley, "Best Estimate Loss-of-Coolant Accident Licensing Methodology Based on WCOBRA/TRAC Code," in Proceedings of International Meeting on Updates in Best Estimate Methods in Nuclear Installation Safety Analysis (BE '00),Washington, DC, USA, November 2000
59. C. Frepoli, M. E. Nissley, and K. Ohkawa, "Assessment of the Emergency Core Cooling System (ECCS) Performance for Current and Advanced Plant Designs Using WCOBRA/TRAC," in Proceedings of International Congress on Advances in Nuclear Power Plants (ICAPP '03), Cordoba, Spain, May 2003
60. R. P. Martin, L. D. O'Dell," AREVA's Realistic Large Break LOCA Analysis Methodology", Nuclear Engineering and Design 235 (2005) 1713–1725
61. M.Ishii, I.Kataoka, "Mechanism and correlation of droplet entrainment and deposition in annular two-phase flow", NUREG/CR-2885, July 1982, Argonne National Laboratory 82-44,
62. C. M. Allison et al., "**SCDAP/RELAP5/MOD3.1 Code Manual Volume IV: MATPRO** -- A Library of Materials Properties for Light-Water-Reactor Accident Analysis", NUREG/CR-6150-EGG-2720-Volume IV, November 1993, Idaho National Engineering Laboratory
63. J. F. Perkins, R.W. King, "Energy Release form the Decay of Fission Products", *Nuclear Science and Engineering* 3, 726-746 (1958)
64. Various, "Re-Analysis of the ISP-13 Exercise, Post Test Analysis of the LOFT L2-5 Test Calculation", BEMUSE Phase II final report, a project founded by the European Commission.
65. G. Petrangeli, C. Sollima "Gamma Decay Heat Distribution in Core: a Known Issue Revisited", *Hindawi Publishing Corporation Science and Technology of Nuclear Installations*, Vol. 2008
66. IAEA, "Best estimate safety analysis for nuclear power plants: uncertainty evaluation", *Safety series report N°52*, Vienna 2008

67. Regulatory Guide 1.203, “Transient and Accident Analysis Methods”, U.S. Nuclear Regulatory Commission, December 2005
68. A. Prošek, B. Mavko “Theory of the best estimate plus uncertainty methods”, J4-TM-26904, Technical Meeting on Use of a Best Estimate Approach in Licensing with Evaluation of Uncertainties, The University of Pisa, Pisa, Italy, 12 – 16 September 2005
69. NEA/CSNI/R(99)10, “Best Estimate Methods in Thermal Hydraulic Safety Analysis” Seminar Proceedings, Ankara, Turkey 29 June - 1 July 1998
70. D. Serghiuta, S. Shim, H. Tezel, “Regulatory Experience with Application of BEAU Methods in Canada,” International Meeting on Updates in Best Estimate Methods, Nuclear Society Embedded Topical, Washington D.C., USA, November 14-18, (2004).
71. L.E. Hochreiter, ”Guidelines for Safety Assessment of Applications Using Best-Estimate and Uncertainty Analysis Methods for CANDU Nuclear Power Plants”, CNSC Research Project, RSP-0192, May 2005.
72. R. Urbonas, A. Kaliatka, E. Uspuras, V. Vileiniskis, “RBMK-1500 Accident Analysis Using BE Approach”, BE 2004
73. “Compendium of ECCS Research for Realistic LOCA Anaiysis,” NUREG-1230, December 1988
74. D. Lanning and M. Cunningham, “Trends in Thermal Calculations for Light Water Reactor Fuel (1971-1981),” in *Ninth Water Reactor Safety Research Information Meeting,* USNRC, NUREG/CP-0024, Voi. 3, March 1982.
75. J.V.Cathcart, R.E. Pawel, R.A. McKee, R.E. Durscel, G.J. Yurek, J.J. Campbell, S.H. Jury, "Zirconium Metal- Water Oxidation Kinetics IV Reaction Rate Studies," Final Report **ORNL/NUREG-17**, Nuclear Regulatory Commission, August 1979
76. D. A. Powers, R.O. Meyer, “Cladding Swelling and Rupture Models for LOCA Analysis”. **NUREG-0630**, April 1980
77. Master degree thesis in Nuclear Engineering by Carlo Parisi, “Study of the VVER core behavior during a rod ejection accident by coupled three dimensional codes”, University of Pisa, DIMNP, 2004
78. A. Forge et al. “Comparison of thermal-hydraulic Safety Codes for PWR Systems”, CEE, Graham & Trotman, 1988
79. D. G. Cacuci, “On the use of adjoint operators for global optimization and data assimilation,” in Lectures Notes Workshop on Predictability, Observations, and Uncertainties in Geosciences, Florida State University, Tallahassee, Fla, USA, March 2006.
80. S.N. Aksan, F. D’Auria et al. User effects on the thermal hydraulic transient system code calculations, N.E.&D 145(1993)
81. D. G. Cacuci, *Sensitivity and Uncertainty Analysis, Theory, Volume I*, Chapman & Hall/CRC, Boca Raton, Fla, USA, 2003.
82. D. G. Cacuci, M. Ionescu-Bujor, and I. M. Navon, *Sensitivity and Uncertainty Analysis, Application to Large Scale Systems, Volume II*, Chapman &Hall/CRC, Boca Raton, Fla, USA, 2005.
83. G.A.Berna, C.E. Beyer, K.L. Davis, D.D.Lanning, FRAPCON 3: A computer Code for the calculation of Steady-State, Thermal – Mechanical Behavior of Oxide Fuel Rods for High Burnup, NUREG/CR-6534 (4 vol), 1997 (with updates 2005)

84. A. Petruzzi, F. D'Auria, University of Pisa, BEMUSE Phase II Report, 2007 (Re-analysis of the ISP-13 Exercise, Post Test Analysis of the LOFT L2-5 Test Calculation
85. L.S. Tong, J. Weisman, Thermal Analysis of pressurized water reactors, 3rd edition, ANS, La Grange Park, Illinois, USA,1996
86. NUREG/CR-6150-EGG-2720-Volume IV
87. A. Petruzzi, F. D'Auria, "Approaches, Relevant Topics, and Internal Method for Uncertainty Evaluation in Predictions of Thermal-Hydraulic System Codes", Hindawi Publishing Corporation Science and Technology of Nuclear Installations Volume 2008, Article ID 325071
88. A. Petruzzi, F. D'Auria, "Thermal-Hydraulic System Codes in Nuclear Reactor Safety and Qualification Procedures", Hindawi Publishing Corporation Science and Technology of Nuclear Installations Volume 2008, Article ID 460795
89. American Nuclear Society, "American National Standard for Decay Heat Power in Light Water Reactors," ANSI/ANS-5. 1-1979, August 1979. (ANS, *555* North Kensington Avenue, La Grange Park, Illinois 60525.)
90. ANSI/ANS-5.1-1994, "Decay heat power in light water reactors", American Nuclear Society, Il – USA, August 23, 1994.
91. C. Frepoli, A. Petruzzi, "Scaling, Uncertainty, and 3D Coupled Code Calculations in Nuclear Technology", Hindawi Publishing Corporation *Science and Technology of Nuclear Installations*, Volume 2008.
92. F. D'Auria, A. Bousbias-Salah, A. Petruzzi, A. Del Nevo, « State of the Art in Using Best Estimate Calculation Tools in Nuclear Technology", *Nuclear Engineering and Technology*, Vol. 38 No.1 February 2006.
93. A. Prošek, B. Mavko, "The state-of-the-art theory and applications of best-estimate plus uncertainty methods", Nuclear Engineering and Technology, Vol.158., No. 1, pp. 69-79, 2007.
94. F. D'auria,W. Giannotti, "Development of Code with Capability of Internal Assessment of Uncertainty", *Nuclear Engineering and Technology* Vol. 131 No. 1, pp. 159–196, 2000.
95. IAEA, "Assessment of defence in depth for nuclear power plants ", *Safety series report N°46*, Vienna 2005
96. IAEA, "Safety assessment and verification for nuclear power plants ", *Safety guide N°NS-G-1.2*, Vienna 2001
97. IAEA, "Best estimate safety analysis for nuclear power plants: uncertainty evaluation ", *Safety reports series N°52*, Vienna 2008
98. IAEA, "Review and assessment of nuclear facilities by regulatory body ", *Safety guide N°GS-G-1.2*, Vienna 2002
99. IAEA, "Format and content of safety analysis report or nuclear power plants ", *Safety guide N°GS-G-4.1*, Vienna 2004
100. IAEA, "Documentation for use in regulating nuclear facilities", *Safety guide N°GS-G-1.4*, Vienna 2002
101. IAEA, "Safety of nuclear power plants: design", *Safety standards series N0. NS-R-1*, Vienna 2000
102. IAEA, "Organization and staffing of the regulatory body for nuclear facilities", *Safety guide N°GS-G-1.1*, Vienna 2002
103. IAEA, "The operating organization for nuclear power plants ", Safety guide N°NS-G-2.4, Vienna 2001

104. B. E. Boyack, I. Catton, R. B. Duffey, P. Griffith, K. R. Katsma, G. S. Lellouche, S. Levy, U. S. Rohatgi, G. E. Wilson, W. Wulff, and N. Zuber, "Quantifying Reactor Safety Margin Parts 1 to 6," *Nuclear Engineering Design*, 119, 1, 1990
105. OECD, "Report of a CSNI Workshop on Uncertainty Analysis Methods, London 1– 4 March 1994," NEA/CSNI/R(1994)20, Vols. 1 and 2, 1994
106. OECD, "Report on the Uncertainty Methods Study," *NEA/CSNI/R(97)35*, Vols. 1 and 2, 1997
107. US-NRC, OECD, "Proceedings of the OECD0CSNI Workshop on Transient Thermal-hydraulic and Neutronic Codes Requirements Held in Annapolis," US NRC NUREG/CP-0159 and OECD/CSNI Report NEA/CSNI/R(97)4, 1997
108. OECD, "Best-Estimate Methods in Thermal Hydraulic Safety Analysis: Summary and Conclusions of an OECD-CSNI Seminar," NEA/CSNI/R(99)22, OECD, 1999
109. OECD, "Advanced Thermal-Hydraulic and Neutronic Codes: Current and Future Applications," *NEA/CSNI/R(2001)9*, 2001
110. M. R. Galetti,, "Regulatory scenario for the acceptance of uncertainty analysis methodologies for LB-LOCA and the Brasilian approach", Hindawi Publishing Corporation *Science and Technology of Nuclear Installations* Volume 2008, Article ID 325071
111. ACRS Letter Report—Subject: Issues Associated with Industry-Developed Thermal-Hydraulic Codes, January 2001
112. A. *Kaliatka, S. Rimkevicius, E. Uspuras (LEI), "* BEPU approach in licensing of Ignalina NPP with RBMK-1500", Seminar on "Valorisation of Current Trends in Safety and Design of RBMK and VVER Reactors", Lecture IV-03, Pisa, 2008
113. Nuovo Colombo, "Manuale dell'Ingegnere", Hoepli, A-34, A-56, 1997
114. H. Glaser, "Results from the application of uncertainty methods in the CSNI uncertainty methods study (UMS)", Ticket 2008, Pisa, May 2008.
115. B. Boyack et al., "Quantifying Reactor Safety Margins, Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large Break, Loss-of-Coolant Accident," NUREG/CR-5249, USNRC, December 1989.
116. RSK-Leitlinien fuer Druckwasserreaktoren, 3. Ausgabe, Oktober 1981 mit Aenderungen vom 15.11.1996.
117. H. Glaeser, "GRS Method for Uncertainty and Sensitivity Evaluation of Code Results and Applications", Hindawi Publishing Corporation *Science and Technology of Nuclear Installations* Volume 2008, Article ID 798901
118. R. P.Martin, L. D. O'Dell, "Development Considerations of AREVA NP Inc.'s Realistic LBLOCA Analysis Methodology", Hindawi Publishing Corporation *Science and Technology of Nuclear Installations* Volume 2008, Article ID 239718
119. Framatome ANP Richland Report, "Realistic large break LOCA methodology for pressurized water reactors," EMF-2103 (proprietary), 2001
120. Safety Evaluation by the Office of Nuclear Reactor Regulation: Topical Report EMF-2103(P)Revision 0, "Realistic Large Break LOCA

Methodology for Pressurized Water Reactors", Project Number 693, TAC NO.MB7554, April 2003
121. D. Bessete, "Initial and boundary conditions to LOCA analysis an examination of the requirements of Appendix K", ICONE-8, Baltimore, USA, 2000
122. A. PROŠEK, M. LESKOVAR, B. MAVKO, "Quantitative assessment with improved fast Fourier transform based method by signal mirroring", Nucl. Eng. Des., 2008, vol. 238, no. 10, pp. 2668-2677
123. S. Y. LEE and C. H. BAN, "Code-Accuracy-Based Uncertainty Estimation (CABUE) Methodology for Large-Break Loss-of-Coolant Accidents," Nucl. Technol., 148, 335, 2004
124. IAEA, "integrated safety approach", www.ns-iaea.org

A.1. ANNEX I: BEST ESTIMATE PLUS UNCERTAINTY METHODS

A.1.1. Best estimate plus uncertainty methods

There are several methods for the application of the BEPU approach –, few of them are discussed in detail later in this chapter. Every uncertainty method has to identify and characterize the relevant uncertainty parameters as well as quantify the global influence of the combination of these uncertainties on calculated results.
There are two principal options how to treat these two basic items of the uncertainty analysis. In the first option the input uncertainty is propagated through the computer code. Uncertainty is derived following the identification of 'uncertain' input parameters with specified ranges or/and probability distributions (pdf) of these parameters. Multiple calculations with random variations of uncertain parameters are performed to derive the uncertainty (Fig. 65).

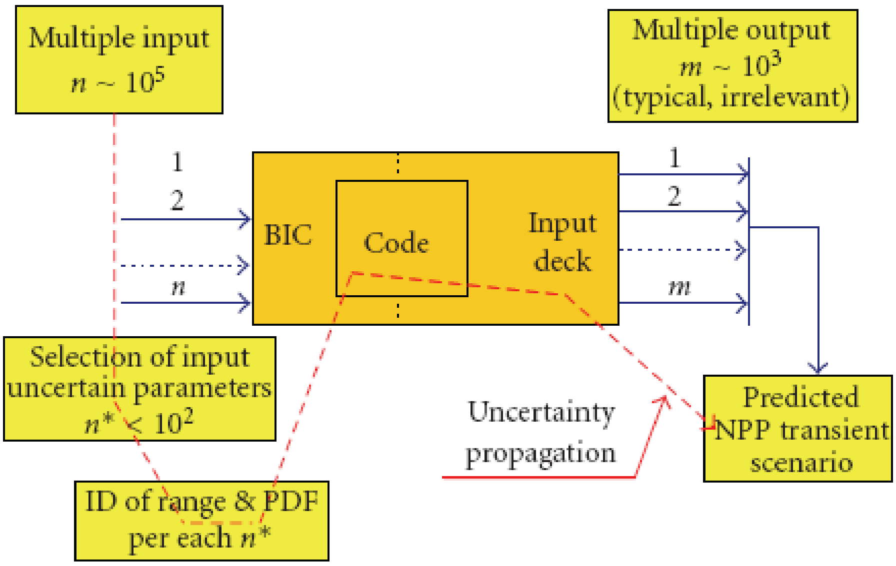


Fig. 65 – Uncertainty methods based upon propagation of input uncertainties [87]

Fig. 66 shows the other option referred to as the "extrapolation" of output errors. Uncertainty is derived from the accuracy between the calculation and relevant experimental data.
Recently, another trend is to combine the abovementioned two options and use the benefits of both (Fig. 67). The input uncertainty is propagated through code calculations and then the obtained output uncertainty is updated with relevant experimental data applying a Bayesian statistical method for output distribution correction based on error distribution [15]. It is based on adjoint sensitivity-analysis procedure (ASAP), global adjoint sensitivity-analysis procedure (GASAP) [81, 82], and data adjustment and assimilation (DAA) methodology [79] by which experimental and calculated data, including the computation of sensitivities (derived from ASAP), are mathematically combined for the prediction of the uncertainty scenarios [87].
The common feature is that each option highly depends upon the extensive experimental database, from which uncertainties can be derived.

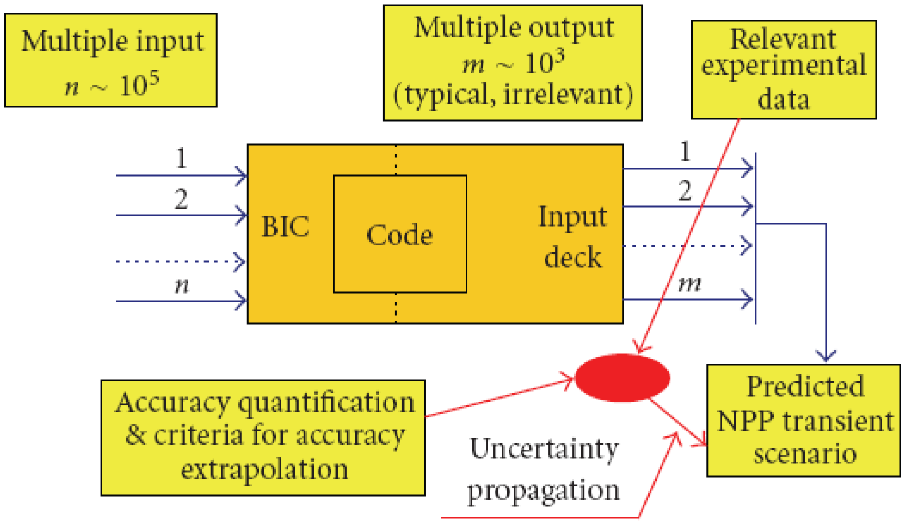


Fig. 66 – Uncertainty methods based upon propagation of output uncertainties [87]

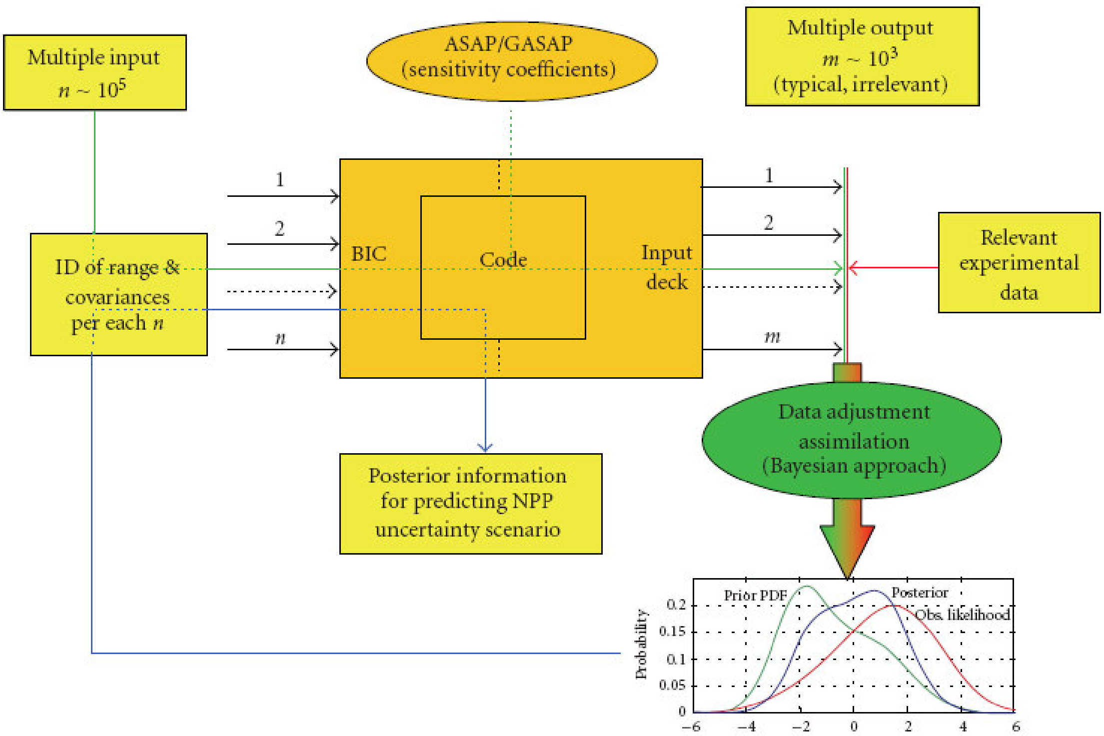


Fig. 67 – Uncertainty methodology based on adjoint sensitivity analysis procedure and data adjustment/assimilation [87]

A.1.1.1. Sources of uncertainty

Very essential step of the BEPU method is the identification and characterization of uncertainty. This is connected with the approximate nature of the codes and of the process of code applications. In other words, 'sources of uncertainty' affect predictions by best-estimate codes and must be taken into account. The major sources of uncertainty in the area of safety analysis are represented by the uncertainty of the code (associated with the code models and correlations, solution scheme, model options, data libraries, deficiencies of the code, simplifying assumptions and approximations), representation uncertainties (accuracy of the complex facility geometry, 3D effects, scaling, control and system simplifications) and plant data uncertainties (unavailability of some plant parameters, instrument errors and uncertainty in instrument response) (Fig. 68).

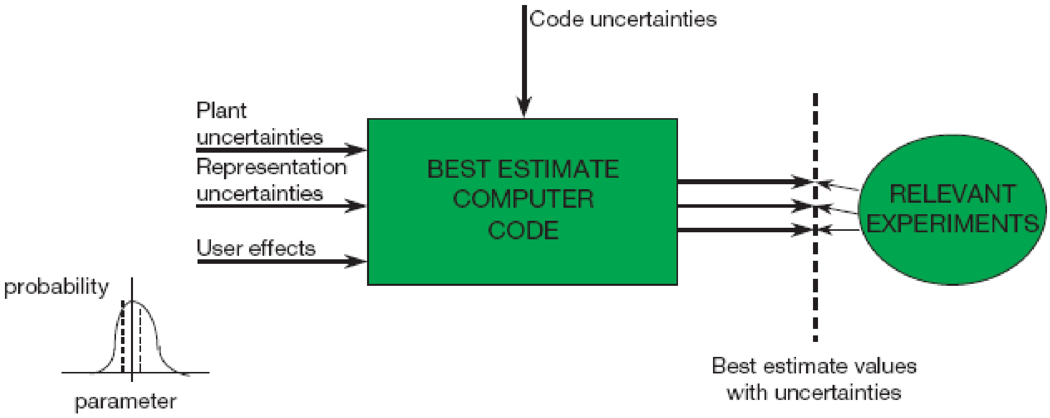


Fig. 68 – Evaluation process and main sources of uncertainties [97]

A more detailed list of uncertainty sources can be found in ref. [3], where an attempt has been made to distinguish 'independent' sources of 'basic' uncertainty.
All the above sources of uncertainty are quite well understood by the technical and scientific community that deals with system code development and application. Complex interactions among the basic uncertainty sources are expected and justify (in advance) the complex structure of an uncertainty method.
Comprehensive research programs have been completed or are in progress aimed at thermal-hydraulic system code assessment and improvement to reduce the influence of the basic uncertainties upon the results.

A.1.1.1.1. Code uncertainty

A system thermal-hydraulic code is a computational tool that typically includes three different sets of balance equations (of energy, mass and momentum), closure or constitutive equations, material and state properties, special process or component models and a numerical solution method.
Balance equations are not sufficiently sophisticated for application in special components or for the simulation of special processes. Examples for those components are the pumps and the steam separators and examples for those special processes are the Counter Current Flow Limiting (CCFL) condition and the two-phase critical flow, though this is not true for all the codes. In such cases, Empirical models 'substitute' the balance equations in such cases.

A.1.1.1.2. Representation uncertainty

The representation uncertainty deals with the process of setting up the nodalization (idealization). The nodalization constitutes the connection between the code and the 'physical reality' that is the objective of the simulation. The process for setting up the nodalization is a brainstorming activity carried out by the group of code users that aims at transferring the information from the real system (e.g. the NPP), including the related boundary and initial conditions, into a form understandable to the code. Limitation in available resources (in terms of man-months), lack of data, target of the code application, capabilities/power of the available computational tools, expertise of the users, have a role in this process. The result of the process may heavily affect the response of the code.

A.1.1.1.3. Scaling

The behavior of complex two-phase flow systems can be predicted only after extensive testing has been performed in separate effects, component, and integral test facilities. The test data are applicable to the prototype system if the test facilities as well as the initial and boundary conditions of the experiments are scaled properly. The scaling methodology employed needs to be defined for each facility and evaluated to ascertain that no test distortions are present that can affect the important physical processes occurring during the scenarios of interest.

Several different scaling techniques have been employed in the various test facilities used for reactor safety research including linear scaling, volume scaling and single- and two-phase scaling criteria developed by Ishii and Kataoka [61]. Various scaling techniques are briefly characterized in US NRC ECCS compendium [4]. Each one of them has certain advantages and disadvantages. Generally, an exact similitude cannot be achieved [5]. In such a circumstance the test facility design is optimized to reproduce the processes of greatest interest even if this may lead to distortions of other processes of lesser importance. Consequently, a single facility generally cannot simulate all phases of a plant transient equally well, and some phases of the simulation may be distorted relative to expected plant behavior. Such problems are more pronounced as scale size is reduced.

Due to abovementioned reasons impact of the scaling also has to be included in the evaluation of uncertainty.

A.1.1.1.4. Plant uncertainty

Uncertainty or limited knowledge of boundary and initial conditions and related values for an assigned NPP are reported as plant uncertainty. Typical examples are the pressurizer level at the start of the assigned transient, the thickness of the gap of the fuel rod, the conductivity of the UO_2, as well as the gap itself.

It might be noted that quantities like gap conductivity and thickness are relevant for the prediction of safety parameters (e.g. peak cladding temperature) and are affected by other parameters like burn-up whose knowledge is not as much detailed (e.g. each layer of a fuel element that may be part of the nodalization) as required. Thus such a source of error in the class of ‘plant uncertainty’ cannot be avoided and should be accounted for by the uncertainty method.

A.1.1.1.5. User effect

Complex computer codes used for safety analysis have many degrees of freedom that allow misapplication and errors by users. In addition, it was numerously demonstrated that even two competent users having the same qualified information and even using the same computer code come up with different results [16]. The cumulative effect of user community members to produce a range of answers for a well-defined problem with rigorously specified boundary and initial conditions is referred to as “user effect”.

The impact of user effect on BEPU analysis may be different depending upon the selected uncertainty method. For methods extrapolating the output error where calculations are a priori connected to experimental data user effect is significantly minimized. For input uncertainty propagation methods where determination of uncertain parameters are to a large extend based on expert judgment user effect might not be negligible.

User effect is not implicitly treated as uncertain parameter. While performing the BEPU analysis the intention is to reduce the user effect as much as possible. Recommended practices for user effect reduction can be found in [17]. Among others they include the comprehensive computer code documentation, appropriate quality assurance program, user training and user qualification and independent peer-review of the calculated results.

A.1.2. Overview of the uncertainty methods

Several uncertainty methods have been developed over past three decades. The reliability, quality and practicability of these methods were already evaluated by the technical community, e.g. in the frame of the NEA/CSNI Uncertainty methods study [3] or more recently within the NEA/CSNI BEMUSE project [14]. Another similar comparison of practical application of various BEPU methods is in preparation has been issue under the IAEA patronage [66].

The principal conclusion from these studies is that the BEPU methods are already mature enough to come up with acceptable results. However their practicability is not yet sufficiently established to support general use and acceptance by industry and safety authorities.

Within the uncertainty methods considered, uncertainties are evaluated using either (a) propagation of input uncertainties or (b) extrapolation of output uncertainties.

The propagation of input uncertainties can be performed by either deterministic or probabilistic methods.

The propagation of input uncertainties

Deterministic methods

The deterministic methods include the Atomic Energy Authority Winfrith (AEAW) and the Electricité de France (EDF) – Framatome method (deterministic realistic Method, DRM).

The deterministic methods have the following features in common with probabilistic methods:

(a) The code, nuclear power plant and transient are identified;
(b) Uncertainties (initial and boundary conditions, modelling, plant, fuel) are identified.

The difference with probabilistic methods is in quantifying the input parameter uncertainties. No probability distributions are used; instead, reasonable uncertainty ranges or bounding values are specified that encompass, for example, available relevant experimental data. The statements of the uncertainty of code results are deterministic, not probabilistic.

Probabilistic methods

Probabilistic methods include: CSAU, GRS, IPSN, ENUSA, GSUAM and BEAU. They have the following common features:

(a) The nuclear power plant, the code and the transient to be analysed are identified;
(b) Uncertainties (plant initial and boundary conditions, fuel parameters, modelling) are identified;

(c) The methods restrict the number of input uncertainties to be included in the calculations.

The selected input uncertainties are ranged using relevant separate effects data. The state of knowledge of each uncertain input parameter within its range is expressed by a probability distribution. Sometimes ‘state of knowledge uncertainty’ is referred to as ‘subjective uncertainty’ to distinguish it from uncertainty due to stochastic variability. Dependencies between uncertain input parameters should be identified and quantified provided that these dependencies are relevant.

The extrapolation of output uncertainties

Uncertainty methodology based on accuracy extrapolation

The UMAE method focuses not on the evaluation of individual parameter uncertainties but on direct scaling of data from an available database, calculating the final uncertainty by extrapolating the accuracy evaluated from relevant integral experiments to full scale nuclear power plants [66].

Availability of a method for the internal assessment of uncertainty

It is based on the code with the capability of internal assessment of uncertainty (CIAU) developed by the University of Pisa [94].

The basic idea of the CIAU can be summarized in two parts:

a) Consideration of plant state: each state is characterized by the value of six relevant quantities (i.e. a hypercube) and by the value of the time since the transient start.

b) Association of an uncertainty to each plant state.

Below three principal methods are described in a detail, other methods currently in use worldwide are also briefly mentioned [66,91,92,93].

A.1.2.1. CSAU Method

The pioneering work in the area of the BEPU methods was done by US NRC and its contractors and consultants while revising the acceptance rules on ECCS [11]. The revised rule states that an alternate ECCS performance analysis, based on best-estimate methods, may be used to provide more realistic estimates of the plant safety margins if the licensee quantifies the uncertainty of the estimates and includes that uncertainty when comparing the calculated results with prescribed acceptance limits. To support the revised ECCS rule a method called the Code Scaling, Applicability and Uncertainty (CSAU) was developed.

The method is intended to investigate the uncertainty of safety-related output parameters. Prior to that, a procedure is used to evaluate the code applicability to a selected plant scenario. Experts identify phenomena and rank their importance to plant safety by examining experimental data and code predictions of the scenario under investigation. Ranking is accomplished by expert judgment using the Phenomena Identification and Ranking Table (PIRT) procedure. The PIRT and code documentation are evaluated, and it is decided if the code is applicable to the plant scenario.

All necessary calculations are performed by using an optimized nodalization to capture the important physical phenomena based on experience obtained by analyzing separate effects tests and integral experiments. No particular method or criteria are prescribed to accomplish this task.
Only parameters important for the high ranked phenomena are selected to be considered as uncertain input parameters. The selection is based on the judgment of their influence on the output parameters. Additional output biases are introduced to consider the uncertainty of other parameters not included in the sensitivity calculations.
Information from manufacturing of NPP components, experiments, and previous calculations performed is used when defining the mean value and probability distribution or standard deviation of uncertain parameters. Additional biases can be introduced to the output uncertainties.
No statistical method for uncertainty evaluation has been formally proposed in the CSAU. A response surface approach has been used in the applications performed up to date [66]. The response surface fits the code predictions obtained for selected parameters, and is further used instead of the original computer code. Such an approach then implies the use of a limited number of uncertain parameters, in order to reduce the number of code runs and the cost of analysis. However, within the CSAU frame the response surface approach is not prescribed, and other methods may be applied.
Scaling is considered by CSAU, identifying several issues based on test facilities and on code assessment. The effect of scale distortions on main processes, the applicability of the existing database to the given NPP, the scale-up capability of closure relationships and their applicability to the NPP range is evaluated at a qualitative level. Biases are introduced if the scaling capability is not provided.
The CSAU methodology is described in detail in [18].

A.1.2.2. GRS Method

The GRS method is a probabilistic method based on the concept of propagating the input uncertainties. All relevant uncertain parameters including the code, representation and plant uncertainties are identified, any dependencies between uncertain parameters are quantified and ranges and/or probabilistic distribution functions (PDFs) for each uncertain parameter are determined. Expert judgment and experience from code applications to separate and integral test and full plant application are principal sources of information for uncertain parameters identification and quantification.
The uncertainty input parameters are randomly sampled taking into account PDFs. Code calculations are performed substituting identified uncertain parameters with sampled sets. The number of code calculations depends on two parameters – fractile α and confidence β. The fractile indicates the probability content of the probability distributions of the code results (e.g. $\alpha = 95\%$ means that PCT is below the tolerance limit with at least $\alpha = 95\%$ probability). One can be β % confident that at least α% of the combined influence of all characterized uncertainties are below the tolerance limit. The confidence level is specified because the probability is not analytically determined. It accounts for possible influence of the sampling error due to the fact that the statements are obtained from a random sample of limited size.

The minimum number n of code runs to be performed is given by the Wilks' formula [19] and [20]:

$$(1-\alpha)^n \geq \beta \qquad 1)$$

where n represents the size of a random sample (a number of calculations) such that the maximum calculated value in the sample is an upper statistical tolerance limit.
For two-sided statistical tolerance intervals (investigating the output parameter distribution within an interval) the formula is:

$$1-\alpha^n - n(1-\alpha)\alpha^{n-1} \geq \beta \qquad 2)$$

The minimum number n of calculations can be found in Tab. 13.
This method has no limit for the number of uncertain parameters to be considered in the analysis. As a consequence of Wilks' formula, the number n of code runs is independent of the number of selected input uncertain parameters, only depending on the fractile and confidence level.
Sensitivity measures by using regression or correlation techniques from the sets of input parameters and from the corresponding output values allow the ranking of the uncertain input parameters in relation to their contribution to output uncertainty. Therefore, the ranking of parameters is a result of the analysis, not of prior expert judgment. The 95% fractile, 95% confidence limit and sensitivity measures for continuous-valued output parameters are provided. The number of code runs for deriving sensitivity measures is also independent of the number of parameters.

-	One-sided statistical tolerance limits			Two-sided statistical tolerance limits		
β/α	**0.90**	**0.95**	**0.99**	**0.90**	**0.95**	**0.99**
0.90	22	45	230	38	77	388
0.95	29	59	299	46	93	473

Tab. 13 – Number of minimum calculations [19] [20]

For regulatory purposes where the margin to licensing criteria is of primary interest, the one-sided tolerance limit may be applied, i.e. for a $95^{th}/95^{th}$ percentile 59 calculations should be performed.

A.1.2.3. CIAU method

The method is based on the principle of extrapolating the output error calculating the final uncertainty by extrapolating the accuracy evaluated from relevant integral experiments to full scale NPPs.
Considering integral test facilities of a reference LWR, and qualified computer codes based on advanced models, the method relies on code capability, qualified by application to facilities of increasing scale. Direct data extrapolation from small scale experiments to reactor scale is difficult due to the imperfect scaling criteria adopted in the design of each scaled down facility. So, only the accuracy (i.e. the difference between measured and calculated quantities) is extrapolated. Experimental and calculated data in differently scaled facilities are used to demonstrate that physical phenomena and code predictive capabilities of important phenomena do not change when increasing the dimensions of the facilities;

however, available integral effects of test facility scales are far away from reactor scale.
Other basic assumptions are that phenomena and transient scenarios in larger scale facilities are close enough to plant conditions. The influence of user and nodalization upon the output uncertainty is minimized in the methodology. However, user and nodalization inadequacies affect the comparison between measured and calculated trends; the error due to this contribution is considered in the extrapolation process and gives a contribution to the overall uncertainty.
The method relies on a database from similar tests and counterpart tests performed in integral test facilities that are representative of plant conditions. The quantification of code accuracy is carried out by using a procedure based on the Fast Fourier Transform characterizing the discrepancies between code calculations and experimental data in the frequency domain, and defining figures of merit for the accuracy of each calculation. Different requirements have to be fulfilled in order to extrapolate the accuracy as discussed later.
Calculations of both ITFs and plant transients are used to attain uncertainty from accuracy. Discretized models and nodalizations are set up and qualified against on experimental data by an iterative procedure to, satisfy requiring that a reasonable level of accuracy is satisfied. Similar criteria are adopted in developing plant nodalization and in performing plant transient calculations. The demonstration of the similarity of the phenomena exhibited in test facilities and in plant calculations, accounting for scaling laws considerations, leads to qualified nodalization of the plant.
No limitation on the number of input uncertain parameters is considered in the application of the method. The related input parameter variation ranges are reflected in the output parameter variation ranges; it is not possible to establish a correspondence between each input and each output parameter without performing additional specific calculations. The process starts from the experimental and calculated database. Following the identification of the physical phenomena (e.g. from CSNI validation matrix [21]), involved in the selected transient scenario, relevant thermal-hydraulic aspects (RTA) are used to evaluate the acceptability of code calculations, the similarity among experimental data, and the similarity between plant calculation results and available data. Statistical treatment is pursued in order to process accuracy values calculated for the various test facilities and to get uncertainty ranges with 95% confidence level.
The scaling of both experimental and calculated data is explicitly assessed in the frame of the analysis. In fact, the demonstration of phenomena scalability is necessary for the application of the method and for the evaluation of the uncertainty associated with the prediction of the NPP scenario.

A.1.2.4. Other methods

There are another BEPU methods developed by various institutions and organizations around the world, e.g. by AEA Technology (Great Britain), ENUSA (Spain), IPSN (France), KAERI (Korea), Westinghouse (USA), Framatome-ANP (France), AECL (Canada) etc. In principle in every of these methods uncertain parameters have to be identified and various techniques are employed to propagate the uncertainty, like Monte Carlo analysis, response surface method, tolerance limit method or Wilks' formula. Some of these methods depend highly on expert judgment. However, most of these methods and techniques are being abandoned recently stepping down to the application of Wilks' formula similar to GRS method.

A.1.3. Supportive methods and software

Performing the BEPU evaluation constitutes much more complex simulation comparing to conservative approach. There are needs for identification of uncertain parameters and quantification of their uncertainty ranges and distributions, in some cases, reduction to manageable number of most important parameters is required, acceptability of the BE nodalization has to be demonstrated and so on. All these additional requirements lead to the development of various supportive methods and software as discussed briefly below.

A.1.3.1. PIRT

Uncertainty methods propagating the input uncertainty and not relying on Wilks' formula (e.g. CSAU method) need to reduce the number of input uncertain parameters in order to keep the number of required calculations to be performed on achievable level: (number of calculations increases with the power of number of parameters in order to cover the whole parameter space combining all identified uncertain parameter values).
For this purpose, as a part of CSAU development, a process, called phenomena identification and ranking table (PIRT), as a part of CSAU development was formulated. PIRT principle lies in identification of all relevant phenomena for specific plant design and scenario and their relative ranking. Only parameters with high rank are taking into account for uncertainty evaluation. The whole process heavily depends on expert judgment and in principle collective expertise of a team (experts on experimental programs, code development, code application, plant operation and PIRT methodology) is needed. Details about PIRT can be found in [18] and [22].

A.1.3.2. FFTBM

Fast Fourier Transform Based Method (FFTBM) is a tool to quantify the accuracy between the best-estimate simulation and experimental data and in this way to qualify the acceptability of the BE simulation. The simplest formulation about the accuracy of a given code calculation, with reference to the experimental measured trend, is obtained by the error function:

$$\Delta F(t) = F_{calc}(t) - F_{\exp}(t) \qquad 3)$$

To analyze the information contained in error function $\Delta F(t)$, Fourier transform is used to translate the time function into the frequency domain. To numerically solve this transformation an algorithm of fast Fourier transform (FFT) is applied. The method developed to quantify the accuracy of code calculations is based on the amplitude of the FFT of the experimental signal and of the difference between this one and the calculated trend. In particular, with reference to the error function $\Delta F(t)$, the method defines two values characterizing each calculation:

1. The Dimensionless Average Amplitude, AA:

$$AA = \frac{\sum_{n=0}^{2^m} \left|\tilde{\Delta}F(f_n)\right|}{\sum_{n=0}^{2^m} \left|\tilde{F}_{exp}(f_n)\right|} \qquad 4)$$

2. The Weighted Frequency, WF:

$$WF = \frac{\sum_{n=0}^{2^m} \left|\tilde{\Delta}F(f_n)\right| \cdot f_n}{\sum_{n=0}^{2^m} \left|\tilde{\Delta}F(f_n)\right|} \qquad 5)$$

The AA factor can be considered a sort of "*average fractional error*" of the addressed calculation, whereas the weighted frequency WF gives an idea of the frequencies related with to the inaccuracy.
The accuracy of a code calculation can be evaluated through these those values, by representing the discrepancies of the addressed calculation, with respect reference to the experimental data, with a point in the WF-AA plane. The most significant information is given by AA, which represents the relative magnitude of these such discrepancies; . WF supplies different information to better identify the character of accuracy. In fact, depending on the transient and on the parameter considered, low frequency errors can be more important than high frequency ones, or vice versa.
For overall picture of the accuracy of a given calculation, it is required to combine the information obtained for the single parameters into average indexes of performance.
This is obtained by defining the following quantities:

$$(AA)_{tot} = \sum_{i=1}^{N_{var}} (AA)_i (w_f)_i \qquad 6)$$

$$(WF)_{tot} = \sum_{i=1}^{N_{var}} (WF)_i (w_f)_i \qquad 7)$$

with:

$$\sum_{i=1}^{N_{var}} (w_f)_i = 1 \qquad 8)$$

where
N_{var} is the number of selected parameters;
$(w_f)_i$ are weighting factors introduced for each parameter, to take into account their importance from the viewpoint of safety analyses.

This introduces some degree of engineering judgment that has been fixed by a proper and unique definition of the weighting factors, necessary to account for the different relevance, from the point of view of safety and reliability of the measurement, of the various addressed quantities.
With reference to the accuracy of a given calculation, it has been defined the following acceptability criterion:

$$(AA)_{tot} < K \qquad 9)$$

where K is an acceptability factor valid for the whole transient. The lower the $(AA)_{tot}$ value is, the better is the accuracy of the analyzed calculation, (i.e., the code prediction capability and acceptability is higher). On the other hand, $(AA)_{tot}$ should not exceed unity in any part of the transient (AA = 1 means a calculation affected by a 100% error). Because of this requirement, the accuracy evaluation should be performed at different steps during the transient.
With reference to the experience gathered from previous application of this methodology, K = 0.4 has been chosen as the reference threshold value identifying good accuracy of a code calculation. In addition, in the case of upper plenum pressure, the acceptable threshold is given by K = 0.1. Some doubt is expressed on such value when pressure is a no dropping value [122].
Detailed description of this method can be found in ref. [23] and in [122].

A.1.3.3. Nodalization qualification

Nodalization qualification [24] represents the methodology to qualitatively and quantitatively qualify the applicability of the computer code and code nodalization for the relevant PIE. The nodalization can be considered as qualified when it (a) has a geometrical fidelity with the simulated system, (b) reproduces the nominal steady-state conditions of the system and (c) shows a satisfactory behavior in time-dependent conditions. To evaluate all these conditions the following steps have to be performed:

- Comparison of the computer code nodalization with the facility geometry and reproduction of the nominal facility parameters;
- Qualitative evaluation of experimental and calculated time trends and comparison of the timing of key events;
- Qualitative evaluation of the computer code suitability based on the phenomena identified in the CSNI validation matrix [21];
- Qualitative evaluation of calculated accuracy based on the identification of the relevant thermo-hydraulic aspects (RTA);
- Quantitative evaluation of calculated accuracy given by the application of the FFTBM.

For each step of the methodology qualitative and/or quantitative criteria to be fulfilled are defined. In principle, experimental data are needed for the comparison. This is not always true, especially for the full plant applications. In such case, the experimental data from the relevant ITF test can be used by performing so-called k_v-scaled calculation: for the full plant application, (initial and boundary conditions for the full plant application are defined in the way to reflect the scaling ratio between the plant and ITF).

A.1.3.4. Estimation of uncertainties

Principal part of the uncertainty evaluation is the selection of the uncertain parameters and definition of their uncertainty ranges and/or probabilistic distribution functions. Recently, this process is heavily dependent on expert judgment, giving to the uncertainty evaluation an high degree of subjectivity. However statistical tools are already being been developed, e.g. such as CABUE [123] and CIRCÉ [25], which is a statistical tool aimed at estimating the uncertainties of the physical models of CATHARE 2 computer code. CIRCÉ gives an estimation of the mean value (bias) and the standard deviation of each code model considered based on the comparison of the code model results to the experimental data. Similarly, in the first step of the code accuracy based uncertainty estimation (CABUE) technique, the code uncertainty parameters and their distributions are determined from separate effect tests. The Wilks formula is then used to show that the highest ordered prediction value is higher than the maximum value of the integral effects test code predictions (covering check calculation) [93]

Despite the fact that tools like CABUE and CIRCÉ are still in developing phase, they highlight the proper way in reducing the contribution of user effect in the safety analysis using BEPU approach.

A.1.3.5. SUSA

System for Uncertainty and Sensitivity Analysis (SUSA) developed by GRS is a software package to support GRS uncertainty method [26]. It assists the user in performing the uncertainty evaluation including the preparation of the data, random sampling, running the code calculation, evaluation of calculated results and generation of uncertainty bands and performing the sensitivity measures. Most of the performed steps are automated so the user typing errors are minimized. For sensitivity measures SUSA offers a set of ordinary and partial correlations and standardized partial regression coefficients to identify the most influence input uncertain parameters.

A.2. ANNEX II: ASSIGNAMENT OF SAFETY PRINCIPLES TO INDIVIDUAL LEVELS OF DEFENCE IN DEPTH

The present annex gives examples of the assignment of safety principles to levels of defence in depth [195]. They can reflect differences in current national practices. For instance, in some countries, normal operating procedures (SP (288)) cover both normal and abnormal operational regimes. In other countries, abnormal operational regimes are covered by emergency operating procedures (EOPs) 8 (SP (290)); the same EOPs are also applicable for accidents within the design basis and to some extent (before significant core degradation) also for beyond design basis accidents (BDBAs).

Phases of plant life	No. of SP	Safety principle (SP)	Level of defence				
			1	2	3	4	5
Siting	136	External factors affecting the plant	o				
	138	Radiological impact on the public and the local environment	o	o	o	o	o
	140	Feasibility of emergency plans					o
	142	Ultimate heat sink provisions	o	o	o	o	
Design	150	Design management	o	o	o	o	
	154	Proven technology	o	o	o	o	
	158	General basis for design	o	o	o	o	
	164	Plant process control systems	o	o			
	168	Automatic safety systems			o		
	174	Reliability targets			o		
	177	Dependent failures			o		
	182	Equipment qualification			o		
	186	Inspectability of safety equipment	o	o	o	o	
	188	Radiation protection in design	o				
	192	Protection against power transient accidents	o	o	o		
	195	Reactor core integrity	o	o	o		
	200	Automatic shutdown systems			o	o	
	203	Normal heat removal	o	o			
	205	Startup, shutdown and low power operation	o	o	o	o	
	207	Emergency heat removal			o	o	
	209	Reactor coolant system integrity	o	o			
	217	Confinement of radioactive material			o	o	
	221	Protection of confinement structure			o	o	
	227	Monitoring of plant safety status	o	o	o	o	
	230	Preservation of control capability	o	o	o	o	
	233	Station blackout			o	o	
	237	Control of accidents within the design basis			o		
	240	New and spent fuel storage	o	o			
	242	Physical protection of plant	o	o			

Phases of plant life	No. of SP	Safety principle (SP)	Level of defence				
			1	2	3	4	5
Manufacture and construction	246	Safety evaluation of design	o	o	o	o	
	249	Achievement of quality	o	o	o	o	
Commissioning	255	Verification of design and construction	o	o	o	o	
	258	Validation of operating and functional test procedures	o	o	o	o	
	260	Collection of baseline data	o	o	o	o	
	262	Pre-operational adjustment of plant	o	o	o	o	
Operation	265	Organization, responsibilities and staffing	o	o	o	o	o
	269	Safety review procedures	o	o	o	o	
	272	Conduct of operations	o				
	278	Training	o	o	o		
	284	Operational limits and conditions	o	o	o		
	288	Normal operating procedures	o				
	290	Emergency operating procedures	o	o	o	o	
	292	Radiation protection procedures	o	o	o	o	
	296	Engineering and technical support of operations	o	o	o	o	o
	299	Feedback of operating experience	o	o	o	o	
	305	Maintenance, testing and inspection	o	o	o	o	
	312	Quality assurance in operation	o	o	o	o	
Accident management	318	Strategy for accident management				o	
	323	Training and procedures for accident management				o	
	326	Engineered features for accident management				o	
Emergency preparedness	333	Emergency plans				o	o
	336	Emergency response facilities				o	o
	339	Assessment of accident consequences and radiological monitoring			o	o	o

A.3. ANNEX III: EXAMPLES OF APPLICATIONS OF BEST ESTIMATE METHODS IN LICENSING IN SELECTED COUNTRIES

The present appendix gives some examples of application of BE methods in the licensing applications. A state of art and a summary of the application is available at ref. [93]

A.3.1. Brazil

The CNEN, reviewed and assessed the Angra 2 NPP LB-LOCA analysis, performed with a realistic evaluation methodology.

It was submitted, based on the CSAU methodology [115] to evaluate the uncertainty. Aiming at performing a consistent safety assessment of this analysis, the Brazilian regulatory body relied upon two international consultants, GRS and University of Pisa. The LB-LOCA analysis, presented in the FSAR, was reviewed by CNEN staff taking into account those two independent reviews which led to a request for additional information, with a total of 27 questions to the applicant, each one classified according to their significance to safety.

Together with CNEN staff, the University of Pisa, as consultant, performed an independent calculation. This includes the independent LB-LOCA calculation with Relap5/Mod3.2.2 Gamma code and the independent uncertainty evaluation with the CIAU method. Based on its conclusions, three requests for additional information were issued to the applicant.

The main results are summarized in Fig. A.1, where PCT and related uncertainty bands obtained by the CIAU and by the computational tools adopted by applicant, are given. The following comments apply:

- The CIAU (and the applicant) analysis has been carried out as best-estimate analysis: however, current rules for such analysis might not be free of undue conservatism and the use of peak factors for linear power is the most visible example.
- The conservatism included in the reference input deck constitutes the main reason for getting the 'PCT licensing' from the CIAU application above the acceptability limit of 1200 °C.
- The amplitude of the uncertainty bands is quite similar from CIAU and applicant. Discrepancies in the evaluation of 'PCT licensing' outcome from the way of considering the 'center' of the uncertainty bands. In the case of CIAU, the 'center' of the uncertainty bands is represented by the phenomenological result for PCT obtained by the reference calculation (1100 °C in Fig. A.1). In the case of applicant the 'center' of the uncertainty bands is a statistical value obtained from a process where the reference calculation has a role (796 °C in Fig. A.2).
- The results of the CIAU method are supported by a number of 'finalized' sensitivity studies as large as about 150 (i.e. about 150 LBLOCA calculation have been performed to confirm the CIAU uncertainty results).
- The reference best estimate PCT calculated by the applicant (result on the left of the Fig. A.2) plus the calculated uncertainty is lower than the allowed licensing limit of 1473 K.
- The reference best estimate PCT calculated by CIAU (central result in the Fig. A.2) is higher than the PCT 'proposed' by the applicant and the upper limit for

the rod surface temperature even overpasses the allowed licensing limit of 1473 K, thus triggering licensing issues.

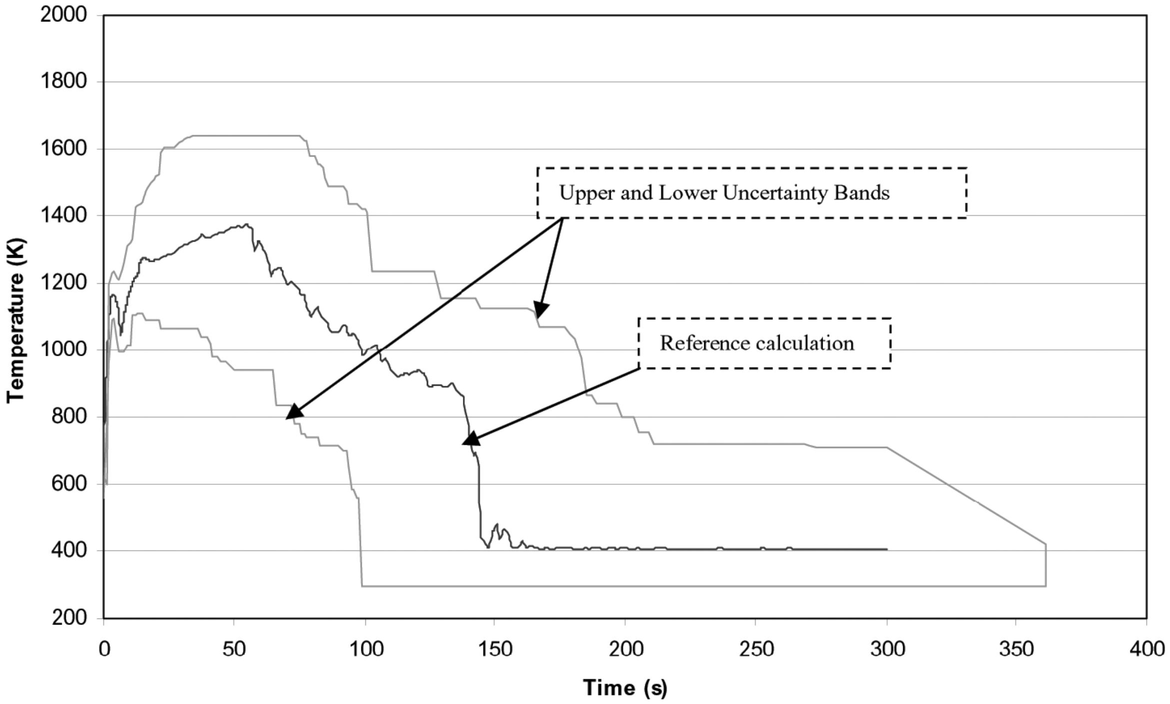


Fig. A.1 – Result of CIAU Application to Angra-2 LBLOCA Analysis: Uncertainty Bands for Rod Surface Temperature at'Axial Level 9' of the Hot Rod Realistic, Obtained by the Reference Run [97] [115]

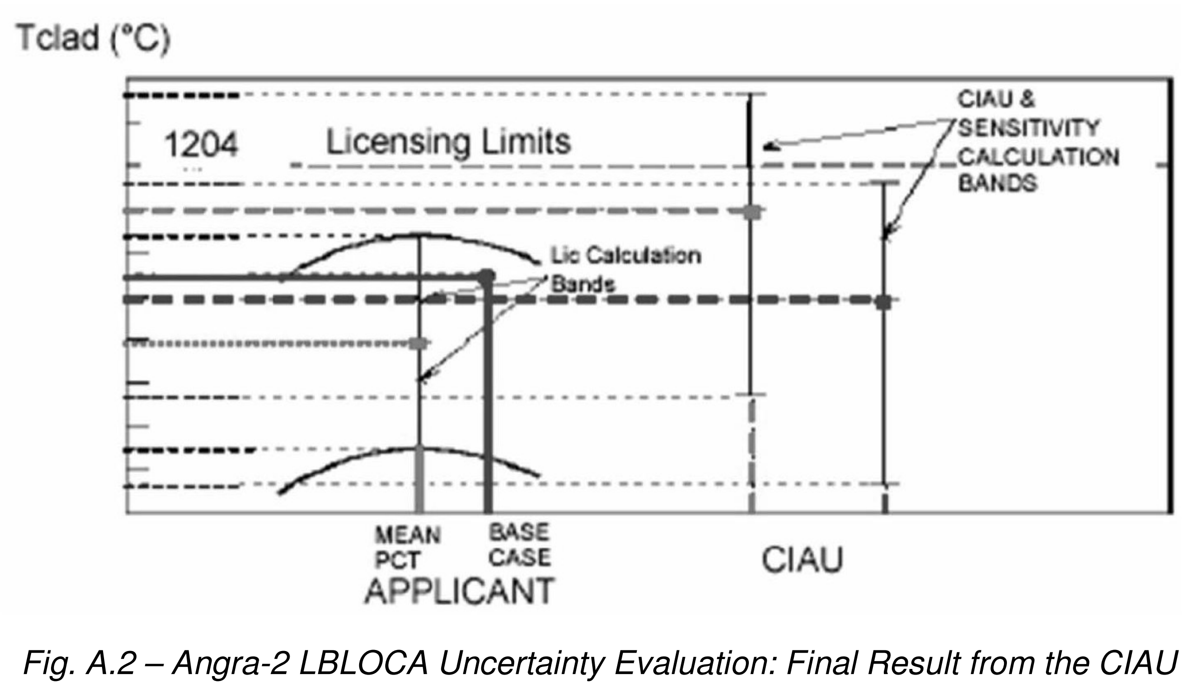


Fig. A.2 – Angra-2 LBLOCA Uncertainty Evaluation: Final Result from the CIAU Study and Comparison with Results of the Applicant [97] [115]

The most relevant outcomes derived from this work are [45]:

1) Results from a best-estimate code prediction are largely affected by the nodalization features.
2) The impact of relevant boundary and initial conditions upon the results is largely affected by the nodalization.
3) The comparison of the PCT's from the 'base-case' and the 'reference calculation' performed by ETN-FSAR and University of Pisa, respectively, indicates a discrepancy, with a higher value observed in the Pisa result. In the case of University of Pisa, it is shown that removal of conservatism of assuming no cross-flow to the hot channel, substantially lowers the reported value.
4) Although the uncertainty bands obtained by both ETN analysis and the independent evaluation are comparable in width, significant discrepancies were observed on how to apply them to obtain the PCT value to be compared with the acceptance criterion. Uncertainty bands in the Pisa methodology have been applied to the 'reference calculation'
5) The acquired experience dealing with the review of the LBLOCA realistic analysis was restricted to the PCT criterion.

As future applications, the Brazilian regulatory body has already been informed by the utility ETN of its intention to uprate 6% the Angra 2 power together with a change in the fuel design, replacing it to a high thermal performance fuel with M5 fuel cladding. This will require the reanalysis of the LB-LOCA with uncertainty quantification. Furthermore, for Angra 1 NPP steam-generators replacement, the utility will submit a realistic evaluation model for the LB-LOCA, using the Westinghouse methodology that encompasses the WCOBRA/TRAC code with the ASTRUM methodology for uncertainty calculation. Additionally, the power will be uprated 5% and a new fuel design will be used [47].

A.3.2. Czech Republic

Czech Republic operates 4 VVER-440 units and 2 VVER-1000 units. Their Safety Analyses are performed with advanced best-estimate computer codes of RELAP, ATHLET, CATHARE type which were developed for western PWRs. Up to now, these codes, while applied for licensing purposes, were used with conservative boundary and initial conditions which required a number of sensitivity analyses.

Under preparation is a proposal of the methodical procedure to be applied for thermo hydraulic analyses of some selected initiating events for VVER-440/213 and VVER-1000/320 reactors, which takes into account the mentioned trends and especially OECD recommendations. Considered is, for instance, application of this method for the evaluation of such events as "leak on the secondary side", SB and LB LOCA, MSLB, using uncertainty analysis of the input data and computer code models (GRS and IRSN method). These analyses will be carried out with the objective of demonstrate safety of nuclear power plants with VVER reactors, complying with the following criteria:

- Fuel integrity preserved.
- Primary circuit integrity preserved.
- Radiological consequences evaluated.

Hereafter, it is given an example of the GRS method application for acceptance criteria evaluation for VVER-440/213 confinement [49].

Results of set of calculations of response of hermetically sealed compartments and response of bubbler condenser to the accident „guillotine rupture of cold leg of primary circuit with diameter 2x 496 mm close to the reactor“ together with subsequent statistical analysis of results are given taking into account uncertainties. These calculations were carried out using the COCOSYS code V2.0 and the detail model of hermetically sealed compartments with the bubbler condenser in Dukovany NPP and these results relate to previous calculations of primary and secondary circuits in case of accident carried out using the RELAP5 code. The input data preparation taking into account uncertainties for hermetically sealed compartments as well as final statistical analysis of results was carried out using the SUSA tool.
Both monitored acceptance criteria AC14 and AC15 were fulfilled in all repeated calculations during the analyzed time interval:

- AC14**:** The maximum overpressure of 113,6 kPa was reached in computed variants what is value smaller than the recommended limit value of 150 kPa.
- AC15: The maximum pressure difference acting to bubbler condenser walls Δp = 18,75 kPa was reached in the computed variants what is lower value than the recommended limit value of 30 kPa.

The peaks of the upper tolerance limit for the maximum pressure (Fig. A.3) and for the pressure difference on BC walls (Fig. A.4) are far enough from the limit values expected by the design and by the NPP safety documentation.

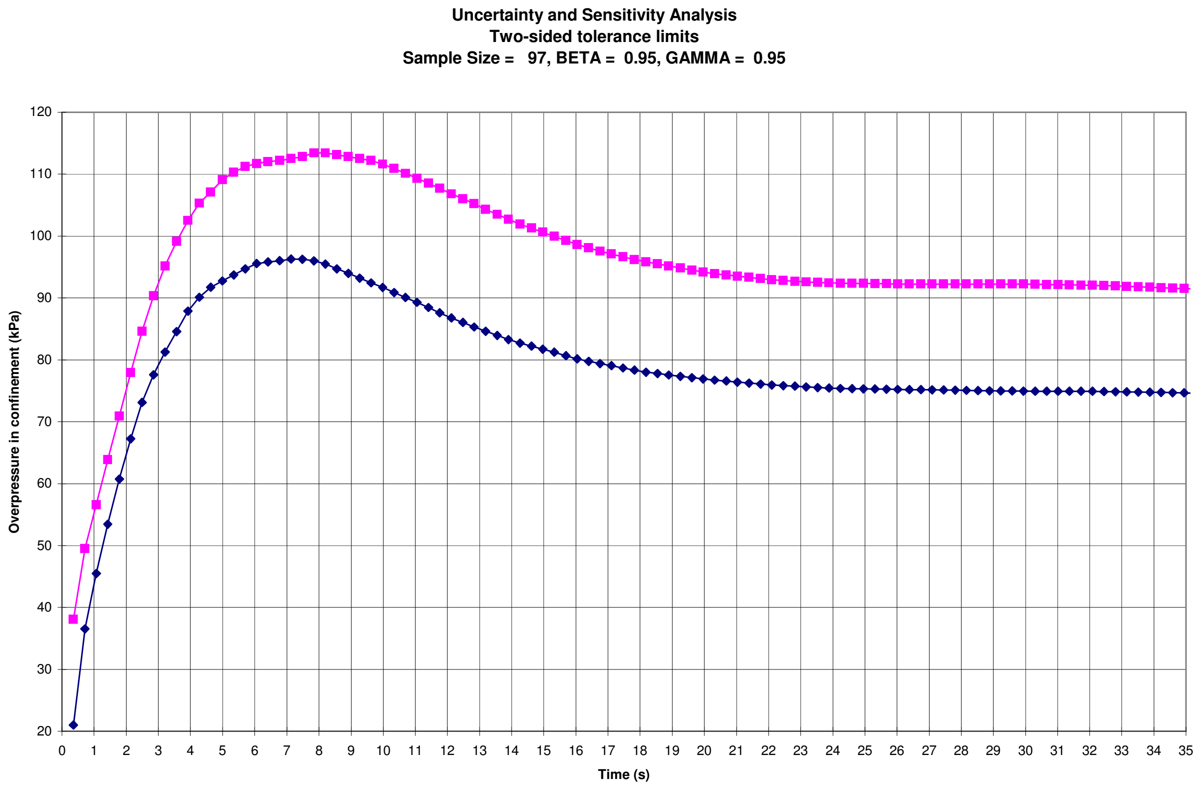


Fig. A.3 – Course of the maximum overpressure in hermetically sealed compartments, double-sided tolerance limits, 0 – 35 s [49]

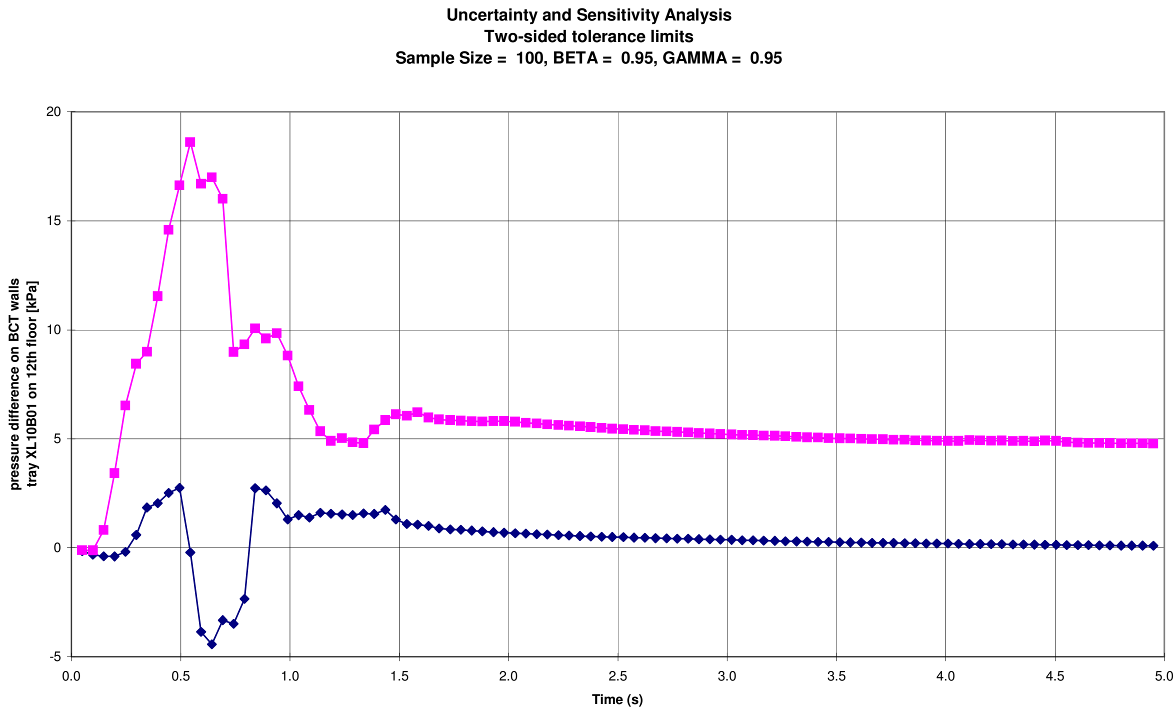


Fig. A.4 – Pressure difference on the wall of the most loaded BC tray, double-sided tolerance margins, 0 – 5 s [49]

A.3.3. Germany

There are applications of plant modifications to operating plants, like power-up-rates, increased enrichment and extended burn-up. 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 up-rate, 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 up-rate.

The analyses with regard to demonstrate the performance of the ECCS licensing requirements after power up-rate which are presented by the applicants are also performed by the assessors, mostly using the thermal-hydraulic code ATHLET and the fuel rod code TESPA to compare with their results. ATHLET and TESPA are developed by GRS; the developments are sponsored by the German Ministry of Economy and Labor (BMWA). 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. 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 behavior. It could be demonstrated that the licensing requirements are fulfilled.

Uncertainties are introduced to the calculation both through the computer code models and through input data for the code. Two different categories of conservative input data are distinguished.

The first one considers data related to assumptions on availability of plant systems: normal operation systems, control systems, safety systems. 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.

The second kind of conservative assumption is applied to cover insufficient knowledge with respect to all other NPP initial and boundary conditions.

German licensing practice is to use a best estimate code, conservative assumptions on availability of plant systems and conservative initial and boundary conditions. Since the knowledge base is growing continuously, the corresponding recommendations of code assumptions and physical models in the RSK (German Reactor Safety Commission) guidelines [116] allow the latitude towards the application of best-estimate models and assumptions. Some examples of such best estimate code models are:

- Critical discharge rate model validated against a variety of experiments, e.g. Super-Moby-Dick and Marviken instead of using the Moody model
- Experimentally validated heat transfer correlations (selected by quality and temperature difference) instead of adiabatic core heat-up prior to reflooding
- Experimentally confirmed quench front propagation and heat transfer model instead of modified Dougal-Rohsenow correlation for heat transfer coefficients during quenching.

Summarizing the influence of safety rules and guidelines it can be stated that the deterministic thermal-hydraulic analyses were performed under conservative and best estimate conditions in all licensing processes in Germany because of the absence of a request in the rules and guidelines to apply an evaluation code with frozen conservative assumptions. These rules and guidelines allowed to flexibly follow the advances in safety technology and to transfer reliable results of research and development into code models and assumptions. Within this flexible kind of procedure unnecessary conservatism could be more and more replaced by sound knowledge. This has been considered as acceptable up to now. At present the procedure is still typically used for safety analysis in many countries, is reasonably established and its use is straight forward. Just one calculation is claimed to be sufficient to demonstrate safe conditions. It is also suggested by the existing IAEA Safety Guide [96], paragraph 4.89, however, a “sufficient” evaluation of the uncertainties of the results should be performed.

In this scenario The GRS developed a method for the uncertainty and sensitivity evaluation of code results applicable for different codes to investigate the combined influence of all potentially important uncertainties on the calculation results. Several applications have been performed in GRS to investigate loss of coolant from the primary and secondary coolant systems of pressurised water reactors, as well as related experiments. For these analyses, it is used the thermal-hydraulic computer code ATHLET. Another uncertainty and sensitivity analysis is performed calculating an experiment simulating containment behaviour using the computer code COCOSYS [42].

Here are listed the uncertainty and sensitivity analyses performed by GRS using the thermal-hydraulic computer code ATHLET simulating breaks of the primary and secondary side cooling systems of pressurised water reactors [117]:

1) separate effects experiment OMEGA heater rod bundle Test 9
2) integral experiment LSTF-CL-18, 5% cold leg break, accumulator injection into cold legs

3) PWR 5% cold leg break, accumulator injection into hot legs (Siemens/ KWU reactor)
4) integral experiment LOFT L2-5, 2 × 100% cold leg break, accumulator injection into cold legs
5) PWR2×100% cold leg break, combined ECC injection into cold and hot legs
6) PWR 10% steam line break
7) PSB-VVER 11% upper plenum break experiment, UP-11-08 (OECD PSB-VVER Test1).

A.3.3.1. Application to a German PWR reference reactor, 2 × 100% cold leg break

An example of the results of an uncertainty analysis by GRS is reported [50]. It was funded by the German Ministry for Economy and Labor (BMWA) under contract RS1142. A double ended cold leg offset shear break design basis accident of a German PWR of 1300 MW electric power is investigated. The fuel rod peak linear heat generation rate is 530 W/cm. Loss of off-site power at turbine trip 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 for ECC injection (accumulator, high and low pressure system), and one hot leg accumulator is unavailable due to preventive maintenance.

The uncertainty analysis considered 56 uncertain input parameters. These consist of 37 model parameters, 4 parameters to select different model correlations for heat transfer and friction, 2 for bypass flow cross sections in the reactor vessel, 1 for temperature of accumulator water, 1 for core power, 1 for decay heat, 1 for radial power distribution in the core, 1 for hot channel factor, 5 for gap width (5 burn-up classes), 1 for fuel thermal conductivity and 2 for convergence criteria.

The calculations are performed using the code ATHLET Mod 1.2, cycle D. A total number of 100 calculations were performed.

Maximum Clad Temperature

Fig. A.5 shows at any point of time, at least 95% of the combined influence of all considered uncertainties on the calculated clad temperatures is below the presented uncertainty limit (one-sided tolerance limit), at a confidence level of at least 95%. A “conservative” calculation result is shown for comparison, 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.

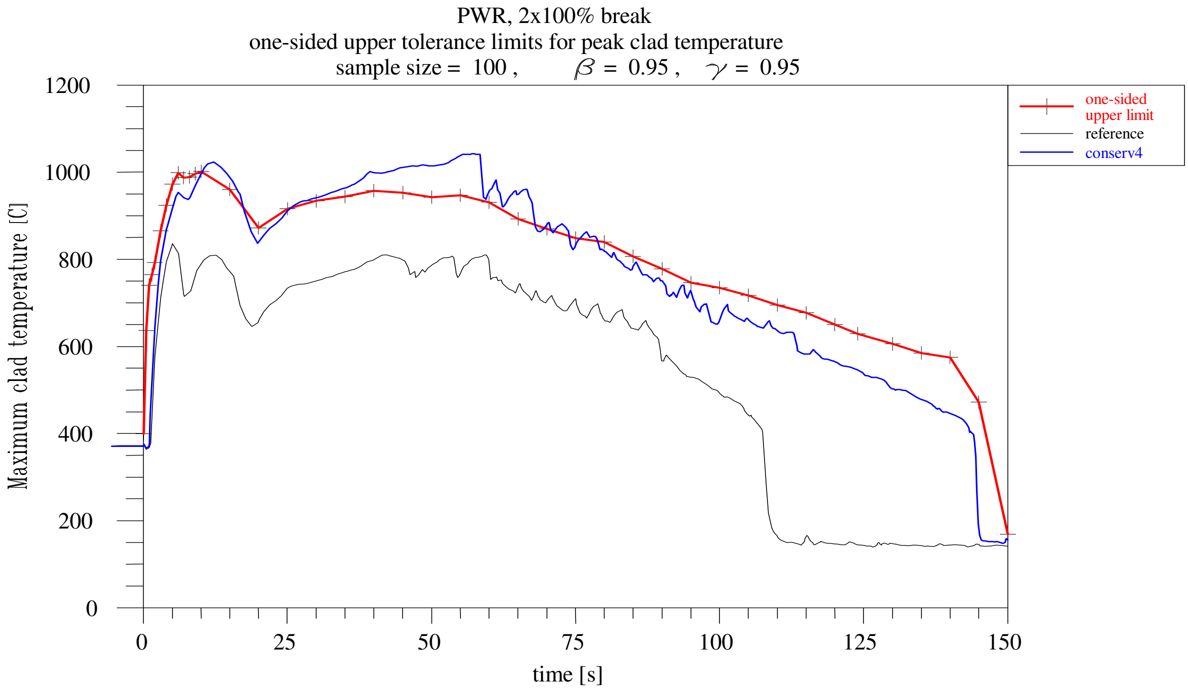


Fig. A.5 – Calculated one-sided 95%/95% uncertainty limit and best estimate reference calculation compared with a "conservative" calculation of rod clad temperature for a reference reactor during a postulated double ended offset shear cold leg break [50] [97] [114]

The "conservative" calculation is representative for the use of best estimate computer codes plus conservative initial and boundary conditions. The uncertainty of code models is not taken into account. Such an evaluation is possible in the licensing procedure of several countries, but not in the USA. The selection of conservative initial and boundary conditions shall bound these model uncertainties. That is obviously not the case for the whole transient in the present example. An uncertainty analysis quantifies uncertain initial and boundary conditions as well as model uncertainties. The peak clad temperatures, however, are bounded due to cumulating conservative values of the highly sensitive parameters gap width and pellet thermal conductivity. It is obvious that the results are dependent on the extent of conservatism implemented in the conservative calculations. Therefore, the US Code of Federal Regulation [11] 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.
The confidence level 95% denominates that the 95th percentile is overestimated conservatively by 95% probability providing a (95%, 95%) statement. That is the reason why some experts claim that a coverage of a (95%, 95%) statement by a conservative calculation is not needed. GRS requires coverage unless other suitable methods for comparison and quantification of "conservatisms" are presented. This could be achieved by an additional statistical test proving that the conservative calculation bounds the 95th percentile.

Sensitivity Measures

The most important parameter uncertainties, out of 56 identified potentially important parameters, with respect to the blowdown peak clad temperature uncertainty are:

- fuel rod gap width for low burn-up (positive sign)
- fuel heat conductivity (negative sign)
- minimum film boiling temperature (negative sign)
- model for critical heat flux (negative sign: Biasi correlation causes lower clad temperatures due to a later change from nucleate to transition boiling compared to the Hench - Levy correlation)
- reactor initial power (positive sign)
- 2-phase multiplier in horizontal pipe (negative sign: Higher water content to break location => lower break flow => higher water content in core => lower clad temperature).

The most important parameters for the reflood peak clad temperature uncertainty are:

- fuel heat conductivity (negative sign)
- fuel rod gap width for low burn-up (positive sign)
- model for 1-phase convection to steam (Mc Eligot correlation tends to cause higher clad temperatures than Dittus-Boelter II)
- number of droplets (negative sign: Number of droplets higher => higher evaporation => lower PCT)
- steam-droplet cooling (negative sign: Higher cooling tends to result in lower PCT).

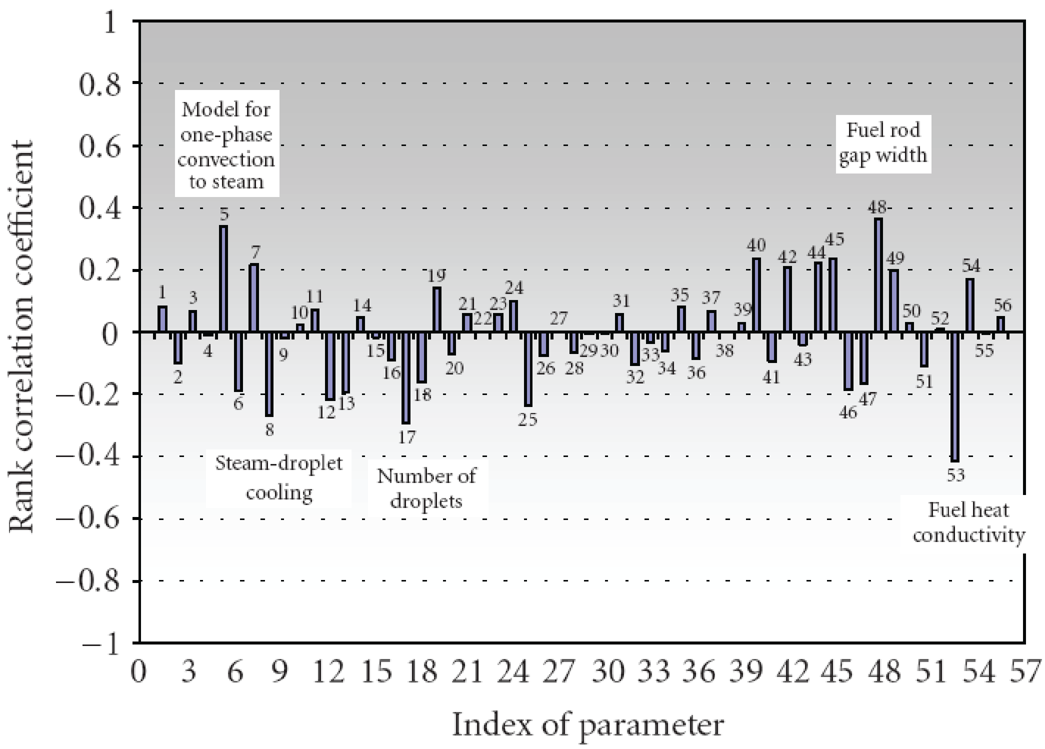


Fig. A.6 – Sensitivity measures of the reflood PCT with respect to the selected 56 uncertain input parameters (rank correlation coefficient) for the reference reactor large break [50], [97] [114]

A.3.4. Lithuania

In Lithuania in 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 are applied, point 1 ÷ 3 of Tab. 4. The BE methodology has been a performed in three accident analysis, which cover worst postulated RBMK-1500 accidents and transient, in Ignalina NPP licensing [72]:

- Main Circulation Pump (MCP) pressure header break with failure to close check valve of one Group Distribution Header (GDH), maximum design basis accident at RBMK-1500: its consequences cover all LOCA type accidents
- analysis of the postulated blocking of coolant flow rate in GDH event:
- partial GDH break: it can produce the flow stagnation in the group of channels, its consequences are similar to GDH blocking case

For all cases two types of analysis were performed: “best-estimate” and “partially-conservative”.

Thermal hydraulic analyses were performed by RELAP5/MOD3.2 for analysis of reactor cavity response. For the uncertainty analysis the GRS method was applied with the SUSA 3.2 tool.

The performed comparison between two approaches shows that in general “partially-conservative” approach provides higher peak temperatures. It is necessary to point that for the “partially-conservative” approach proper boundary and initial conditions should be selected.

A.3.4.1. MCP pressure header break

As it is shown in Fig. A.7, fuel cladding temperatures band does not exceed the acceptance criterion of 700 ℃.

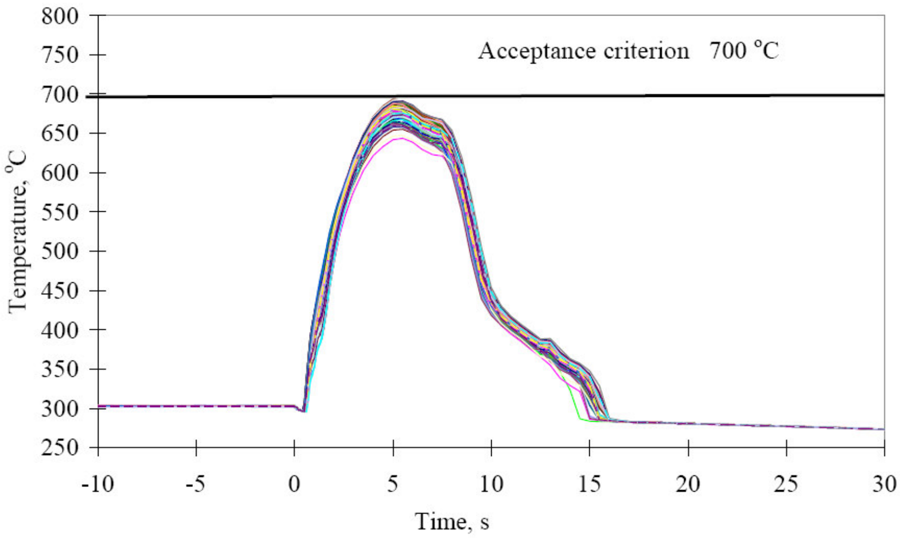


Fig. A.7 – Fuel cladding temperatures in 3.0 MW power FC at the location of 2.75 m from the core bottom obtained using SUSA generated runs from RELAP5 calculations [72]

Acceptance criterion is exceeded in the fuel channels with power higher than 3.0 MW and, thus, fuel cladding integrity in these FCs can be violated. For the evaluation of number of affected fuel channels, the real distribution of FC power in the GDH according Ignalina NPP data was analysed. Fig. A.8 shows a histogram of the reference channel power distribution in one GDH at the maximum permissible thermal operating power, i.e., 4200 MW. As it is seen in Fig. A.8 , there is a group of 12 fuel channels, power of which exceeds the 3.0 MW, therefore the integrity of fuel claddings in remaining 31 FC will be not violated with 95% of probability and 95% of confidence. This information about the number of FC, with possible affected fuel claddings, further was used in the analysis of radiological consequences. Since the number of possible affected fuel rods is small, that does not have any considerable impact to the radiological consequences.

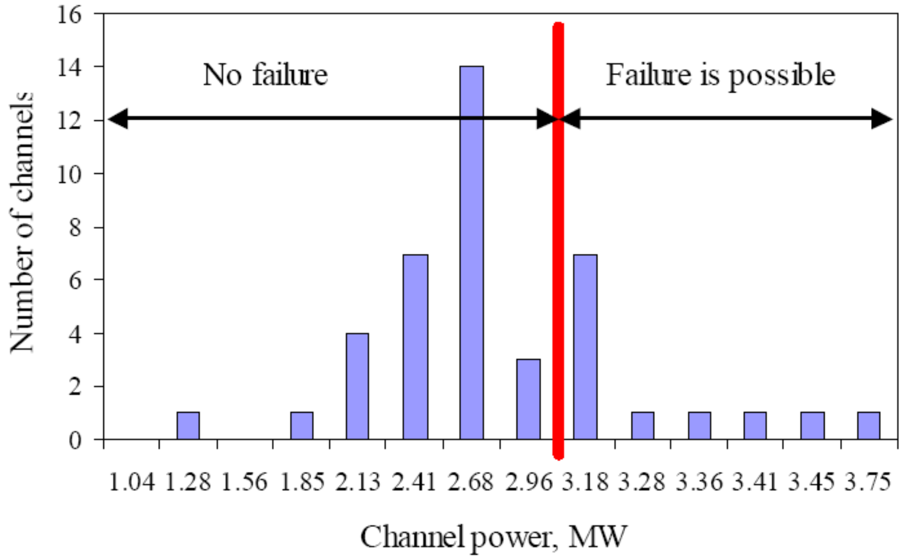


Fig. A.8 – Real distribution of FC power in the most loaded GDH at 4200 MW power level [72]

A.3.4.2. GDH BLOCKAGE

The blocking of coolant flow rate in group distribution header leads to the considerable coolant flow rate decrease in a group of 38-42 fuel channels connected to the affected GDH.

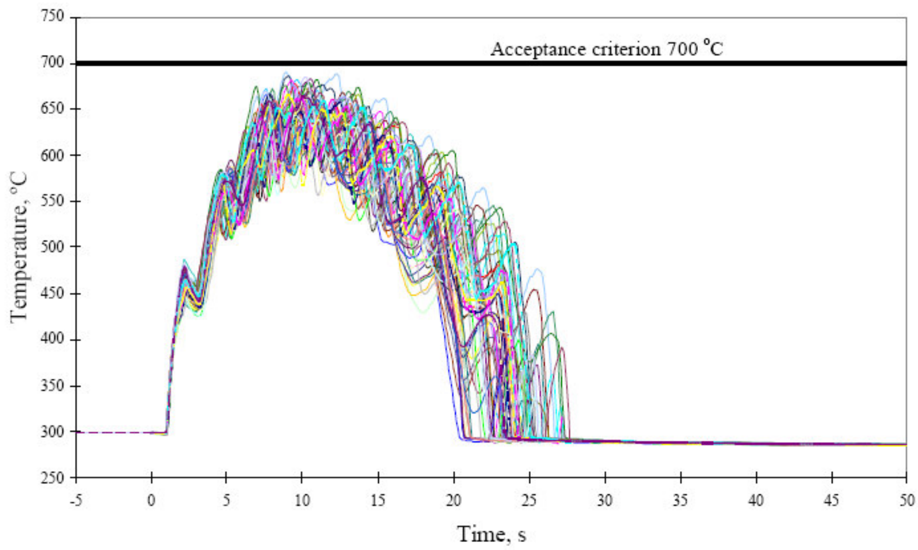


Fig. A.9 – Fuel cladding temperatures in maximum loaded FC, obtained using SUSA generated runs from RELAP5 calculations [72]

In calculations it is assumed that the manual valve in pipeline connecting MCP pressure header and one GDH is closed by mistake.
Fuel channel temperature is shown in Fig. A.9. The analysis showed that the selection of the mixture level tracking, mixture and thermal front tracking model selection and reactor thermal power had the highest impact to the fuel cladding temperatures.

A.3.4.3. PARTIAL GDH BREAK

Partial breaks in large diameter piping were not investigated during the design stage of the RBMK-1500 reactor at Ignalina NPP simply because they were not considered credible. They are now included into consideration for consistency with the present world trends in accident analysis.
Analysis was performed to substantiate RBMK-1500 safety, therefore only the upper limit of peak fuel cladding temperature in the FC, one of technological parameters, connected to the affected GDH is of importance, thus, in the sensitivity and uncertainty analysis, upper one-sided tolerance limit was used. 60 calculations were performed.
All calculated peak fuel cladding temperatures of the maximum loaded fuel channel connected to affected GDH are presented in Fig. A.10. As it is shown in the Fig. A.10, the acceptance criterion for the fuel cladding (700 °C) is not violated (there is a 50 °C margin). It can be mentioned that for fuel cladding temperature a broad scatter of peak temperatures in terms of time can be observed.
Since in the basic calculation fuel channel tube wall temperature increased insignificantly, the sensitivity and uncertainty analysis for FC wall temperature was not performed.

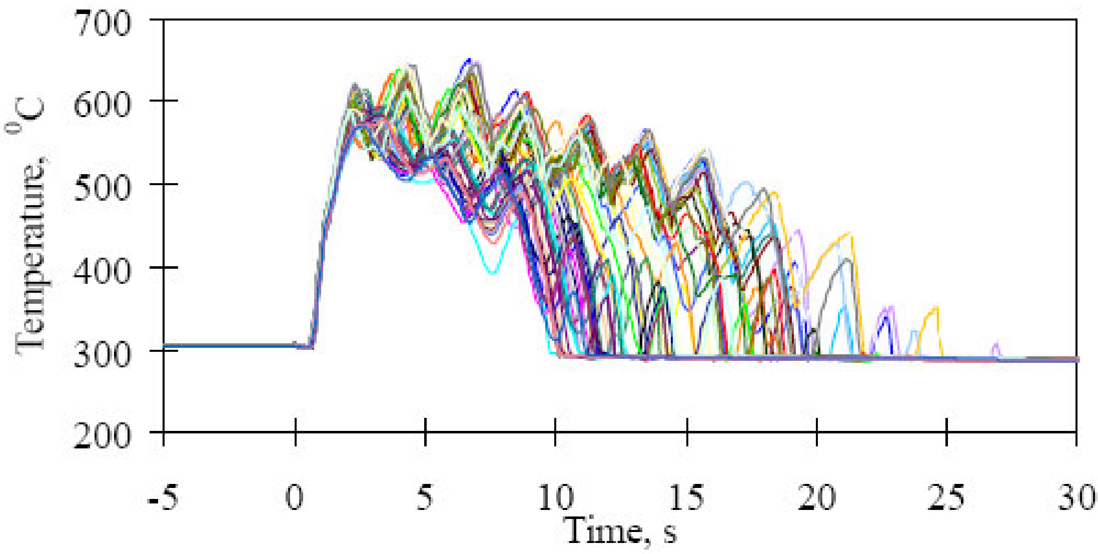


Fig. A.10 – Fuel cladding peak temperatures in maximum loaded FC of the affected GDH calculated using SUSA generated runs [72]

A.3.5. Sweden

Best estimate method by Westinghouse has been used by Ringhals 2 to increase power factors

Parameters considered at power increases are:

- *a)* Steam and feed-water flow
- *b)* Reactor response and control
- *c)* Primary water inventory
- *d)* Steam line vibrations
- *e)* Core shroud vibrations
- *f)* Pressure relief capacity
- *g)* Residual heat removal
- *h)* Wet well cooling
- *i)* Scram system
- *j)* Pump inertia
- *k)* ATWS
- *l)* Stability

The licensing process applied is shown in Fig. A.11 and Fig. A.12.

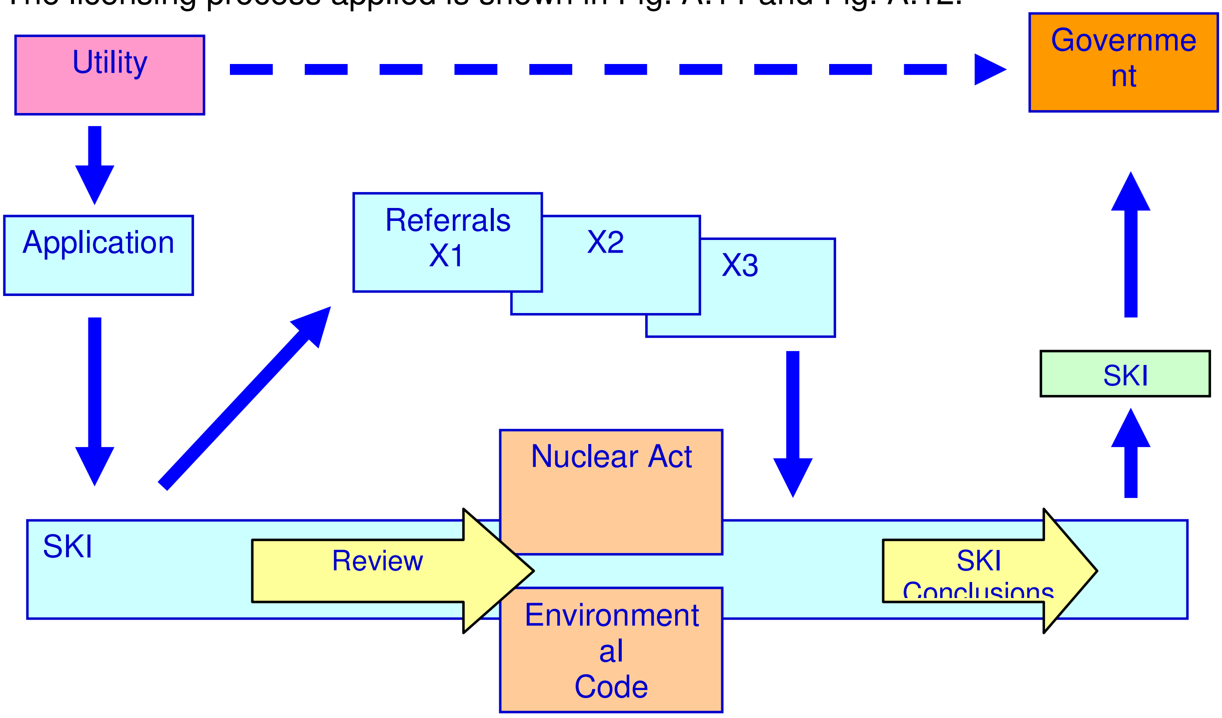


Fig. A.11 – Schematic processes in the Swedish reactor licensing program

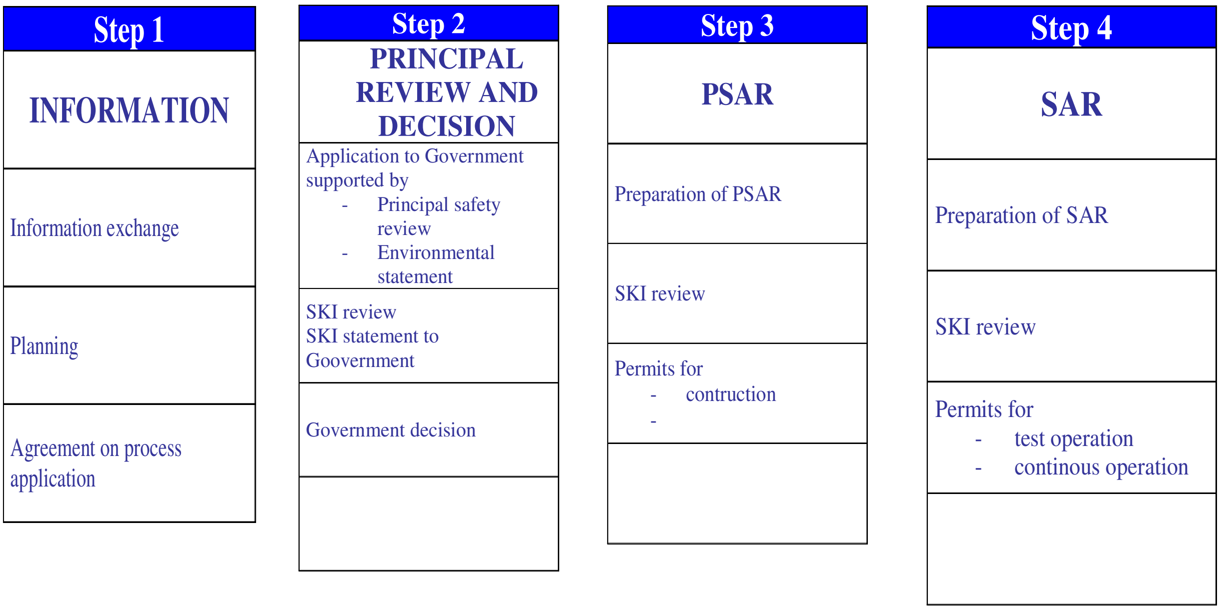


Fig. A.12 – Schematic of steps in the Swedish reactor licensing program

A.3.6. USA Approach to LB-LOCA

In Rulemaking Issue Notation Vote SECY-01-0133 dealing with Status Report on Study of Risk Informed changes to the Technical Requirements of 10 CFR 50 and recommendations on Risk Informed Changes to 10 CFR 50.46 * ECCS acceptance Criteria, the NRC staff recommends:

– modification of the existing 50.46 to change the ECCS acceptance criteria and the Appendix K ECCS evaluation model , and
– development of a voluntary risk-informed alternative to 5 0.46 Appendix K and General Design Criterion (GDC) 35 that will change ECCS reliability requirements. Presently the ECCS reliability is designed to meet specified functional requirements with an assumed single failure and an assumed loss of offsite power simultaneously with the LOCA. The NRC staff believes that the ECCS reliability resulting from the current technical requirements is not commensurate with the risk significance of the various LOCA sizes and that unnecessary conservatisms exist in the requirements. Conclusions by the staff include the statement that "Current ECCS reliability requirements may be overly conservative for large break LOCAs".

Other proposed revisions include:

– Replacing the current 1971 ANS decay heat curve with a model based on the 1994 ANS standard
– Replacing the current decay heat multiplier of 1.2 with an NRC prescribed uncertainty treatment
– Deleting the limitation on PWR reflood steam cooling for small reflood rates
– Replacing the Baker-Just zirconium model with the Cathcart-Pawel zirconium steam oxidation model for heat generation
– Deleting the prohibition on return to nucleate boiling during blow down.
– Within the development of risk informed alternative to 10 CFR 50.46, in place of the simultaneous loss of offsite power requirement and single failure criterion , two options would be offered to accomplish ECCS system reliability.

One of these options proposes that the ECCS design must be such that the core damage frequency (CDF) associated with a specified set of LOCAs is less than an NRC specified CDF threshold, with due consideration of uncertainties. [SECY-0 1-0133, page 6].

- For example, if a set of LOCAs can be demonstrated to have a collective mean frequency of occurrence of below 10^{-4} per year (/yr), some regulatory relief may be appropriate in terms of the level of conservatism and redundancy required in the design.
- If a set of LOCAs can be demonstrated to have a collective mean frequency of occurrence of below 10^{-5}/yr, it may be appropriate to remove these LOCAs from the plant design basis, as long as some mitigate capability remains in the plant, e.g. there is an expectation of success under accident management.
- Lastly, if a set of LOCAs can be demonstrated to have a collective mean frequency of occurrence of below 10^{-6}/yr, it may be appropriate to remove these LOCAs from the plant design basis. [SECY-01-0133, page 7].

The NRC staff confirms that the goal of protection of public health and safety can be achieved with a safety goal surrogate of limiting the Large Early Release Frequency (LERF) to values below 1 10^{-5}/yr. The intent of the guidelines is that the combined LERF for all initiators should remain less than 10^{-5}/yr. In this context, the frequency of LB-LOCA and the requirements for ECCS are discussed.

A.3.7. France: AREVA NP RLBLOCA methodology

The AREVA NP RLBLOCA methodology [118, 119] is a CSAU based methodology for performing best-estimate large-break LOCA analysis. The methodology addresses all of the expressed steps of the CSAU process. The key challenge to this process has been the defense of declared engineering judgment and the demonstration of the methodologies range of applicability. This was accomplished by careful characterization of dominant LOCA parameters and emphasis on validation through sensitivity studies and the statistical nature of the methodology.

The generic AREVA NP RLBLOCA methodology was approved by the USNRC in April 2003 and is now being applied to several nuclear power plants serviced by AREVA NP Inc. While the CSAU methodology represents a significant departure from traditional deterministic methods, the AREVA NP methodology applying nonparametric statistics retains an economical viability on par with existing methodologies. Throughout the 40 staff / years of development effort at AREVA NP, the CSAU process has withstood the technical questions and challenges to its foundation. The key benefits realized by AREVA during this development are

- The move to a realistic LOCA methodology brings a new clarity of understanding of the LBLOCA problem to the industry by demonstrating contrast to the very conservative 10 CFR 50 Appendix K methodologies.
- Through use of statistically-based methods, there is improved characterization of the conditions in which individual LBLOCA uncertainty contributors influence LBLOCA response.
- The reliance on experimental data has revived the importance of the many test programs that have long since been decommissioned.

These rewards alone have validated the CSAU approach.

A.4. APPENDIX IV: DECAY HEAT ANALYSIS

The present annex give additional information on the results achieved by the analyses performed on the sources of heat during a LOCA accident and other accidents.

A.4.1. Schema and nodalization of the generic PWR

The nodalization has been realized taking into account as reference plant a "generic" VVER 1000 plant; it means that all the main and relevant characteristics and systems of the VVER 1000 have been considered and modelled in the nodalization; the largest amount of data derived by information on typical VVER 1000 NPP.

A.4.1.1. References for the data of the NPP nodalization

Hereafter, the sources of the data to update the nodalization are reported. Geometrical data for the primary and secondary system are derived by:

- "Nuclear safety and the environment – Main characteristics of nuclear plants in European Union and candidate countries" – EUR 20056 EN – October 2001.
- "Posar – Chapter 4 – Table 4.1-1 comparison of fuel design", January 2001

Other geometrical data (generic VVER1000/320) are obtained by:

- "VVER1000 coolant transient benchmark" NEA/NSC/DOC(2000)6.

		VVER 1000/320 Nodalization
		(m)
1.	Total internal height	12.67
2.	RPV top(1)	29.6
3.	RPV bottom	16.93
4.	Bottom of hydraulic core	18.75
5.	Bottom of active core	18.75
6.	Top of active core	22.30
7.	Top of hydraulic core	22.28
8.	CL axis	23.90
9.	Top of DC(1)	24.95
10.	HL axis	27.50
11.	UP top(1)	28.3
12.	UH top(1)	29.60

Tab. 14 – Reactor Pressure Vessel elevations and main dimensions

VVER 1000 NPP steady state data are obtained by:

- "Nuclear safety and the environment – Main characteristics of nuclear plants in European Union and candidate countries" – EUR 20056 EN – October 2001

Information about valves and systems and their logic is derived by:

- "Dynamic calculation results due to steam and water hammer and water overfill overview", L. Pecinka, Workshop on High energy piping at 28.8m Level, Prague Nov. 2002

- “Pipe break probability calculation according to the PRAISE methodology”, L. Pecinka, Workshop on High energy piping at 28.8m Level, Prague Nov. 2002
- “Qualification of steam generator safety and relief valves” J. Fridrich, R. Josífko, A. Král, V. Maxa, Workshop on High energy piping at 28.8m Level, Prague Nov. 2002.

		VVER 1000/320 Nodalization (m)
1.	CL axis	23.90
2.	SG tubes top	30.25
3.	Loop seal Axis	20.64
4.	HL axis	27.50
5.	RPV bottom	16.93
6.	RPV top	29.6
7.	PRZ bottom	21.95
8.	PRZ top	33.12
9.	SG top	31.82
10.	SG bottom	28.1
11.	Steam header axis	34.90
12.	SIT 1 top	30.14
13.	SIT 1 bottom	21.34
14.	SIT 3 top	36.05
15.	SIT 3 bottom	27.25
16.	HL length	8.66
17.	CL length	25.06
18.	CL internal diameter	.85
19.	HL internal diameter	.85
20.	PRZ diameter	3

Tab. 15 – NPP System elevations and main dimensions

			VVER 1000/320 Nodalization
		Unit	
1.	Active length	m	3.55
2.	Bottom of active fuel	m	18.75
3.	Top of active fuel	m	22.30
4.	Number of Active rods	-	50856
5.	Fuel assemblies with control rod	-	61
6.	Fuel assembly flow area	m^2	0.025
7.	Total core flow area	m^2	4.172
8.	Bypass flow area	%	5
9.	Pellet diameter	mm	7.53
10	Center void diameter	mm	1.4
11	Clad internal diameter	mm	7.8
12	Clad external diameter	mm	9.1

Tab. 16 – Core elevations and main dimensions

			VVER 1000/320 Nodalization
		Unit	
1.	Total tubes length	m	121922 (hydraulic) 122100 (thermal)
2.	Tube number	-	10984 (hydraulic) 11000 (thermal)
3.	Tube ID	mm	13
4.	Tube OD	mm	16
5.	SG collector volume	m^3	2.18
6.	SG collector height	m	3.72
7.	PS total volume	m^3	20.54
8.	PS tubes volume	m^3	16.18

Tab. 17 – Steam Generator elevations and main dimensions

A.4.1.2. Steady state calculation

The steady state calculation has been performed in the framework of the nodalization quality assessment. The obtained results are shown in the Tab. 18 and in the Fig. A.13 to Fig. A.15. No differences are notable between the reference VVER NPP data and the calculated results. It is an expected result, because the importance of the changes in the nodalization is small with reference to steady state calculation.

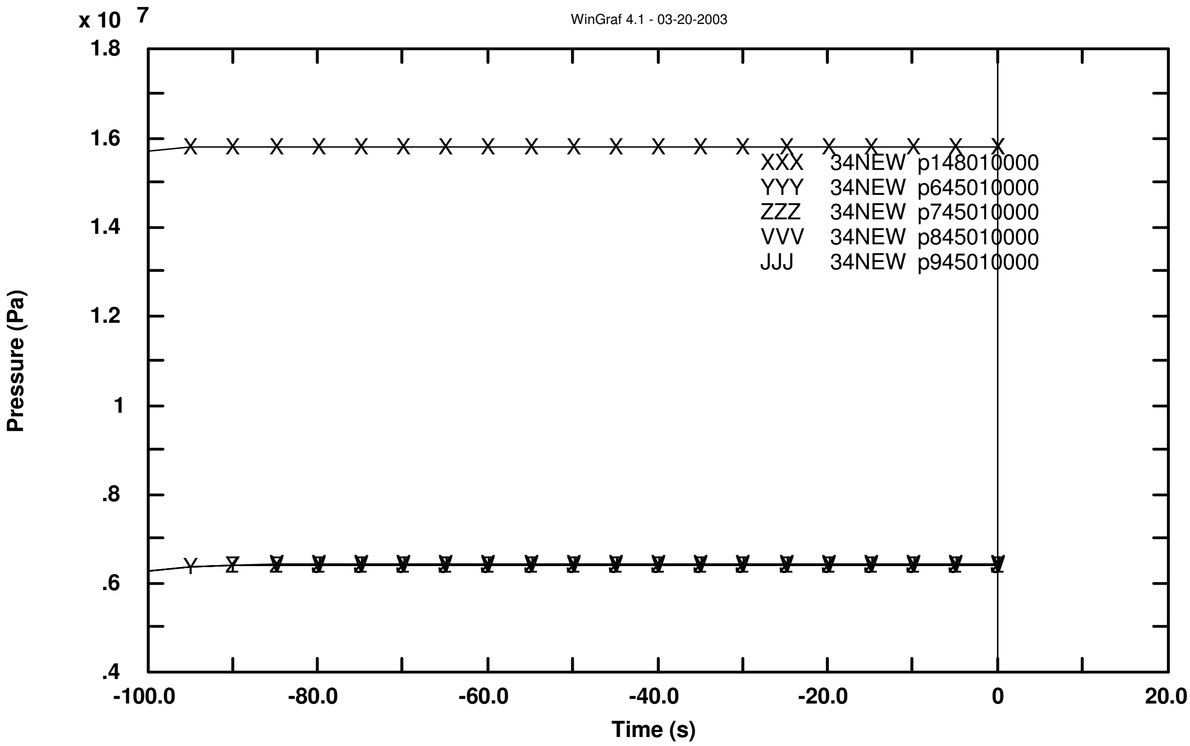


Fig. A.13 – VVER 1000, Steady State: Up and SGs (1 to 4) pressure

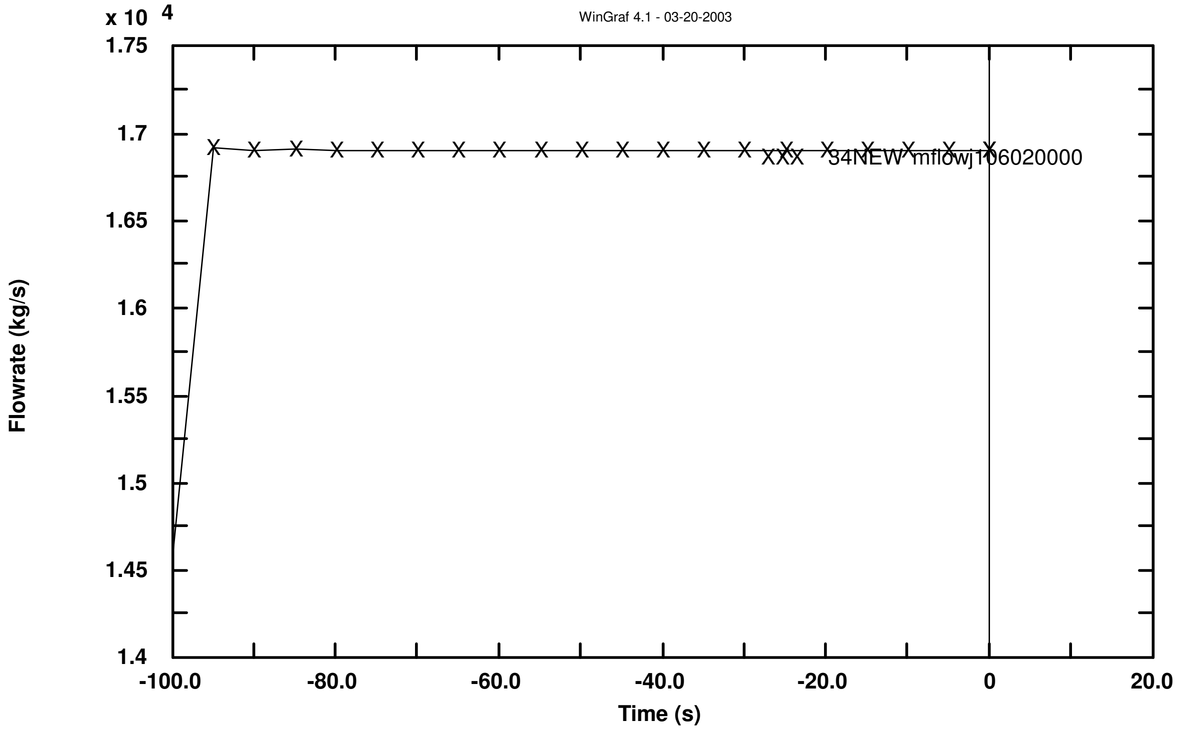


Fig. A.14 – VVER 1000, Steady State: Core inlet mass flowrate

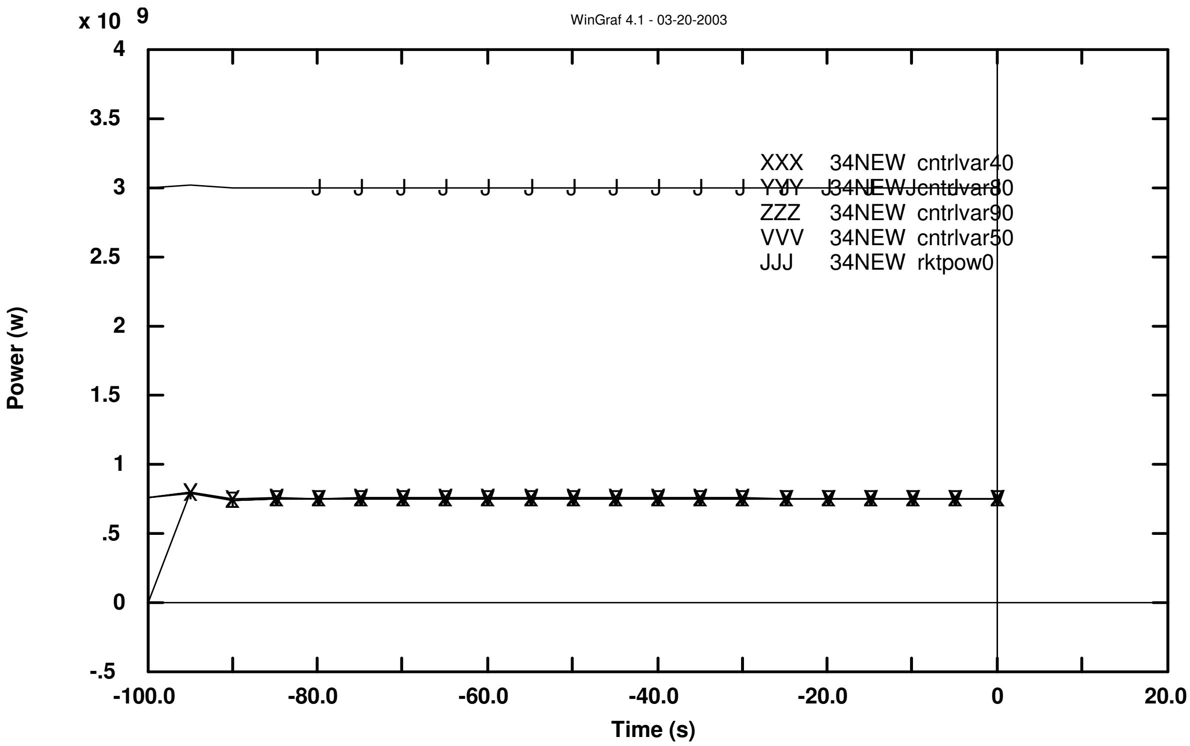


Fig. A.15 – VVER 1000, Steady State: Core and SG exchanged power

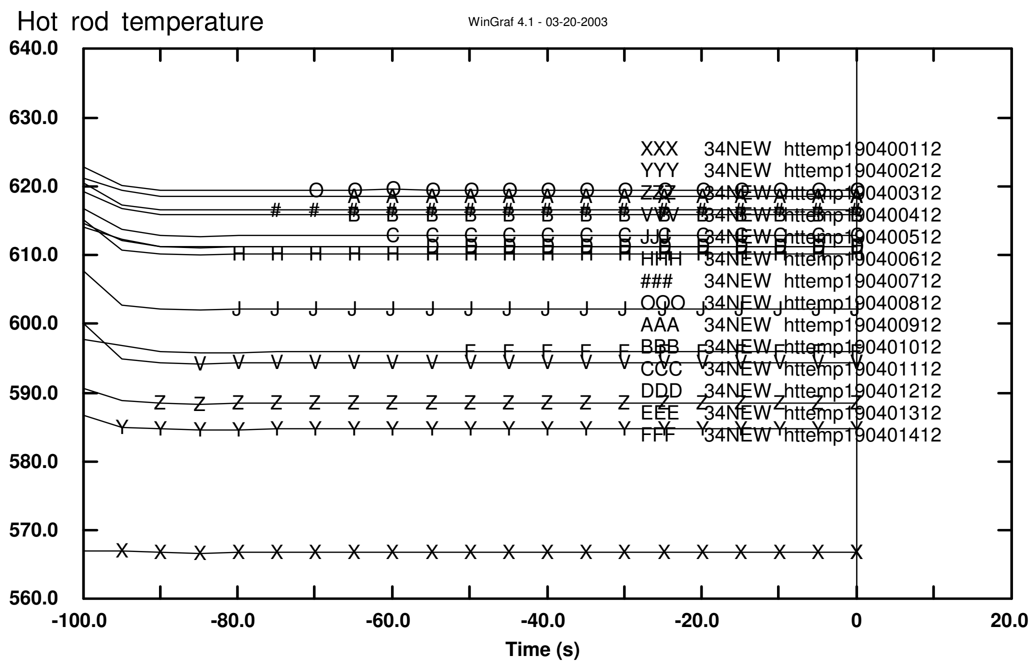


Fig. A.16 – VVER 1000, Steady State: Surface temperature of hot rod (all elevations)

N.	Quantity	Temelin NPP	Calculated
1	Power	3000	3000
2	Power (Mw)	3000	3000
3	Core inlet temperature (K)	562	562
4	Core ΔTemperature (K)	30	30
5	Core outlet temperature (K)	592	592
6	Coolant pressure outlet core (MPa)	15.7	15.8
7	One loop flow rate (kg/s)	(21200 m^3/h)	4526
8	Core flow rate	(84800 m^3/h)	Core flow rate 16901 Bypass flow rate 1147 Total flow rate 18048
9	Pump rotation speed (rad/s)	(995 rev/min)	104.2
10	PRZ pressure (MPa)	15.7	15.7
11	PRZ temperature (K)	619	619
12	Pump flow rate	(21200 m^3/h)	4526
13	SG exchanged power (Mw)	750	751
14	SG steam production (kg/s)	408	407
15	SG pressure (MPa)	6.3	6.4
16	SG steam temperature (K)	551	552
17	FW temperature (K)	493	493

Tab. 18 – Steady state values of some relevant parameters

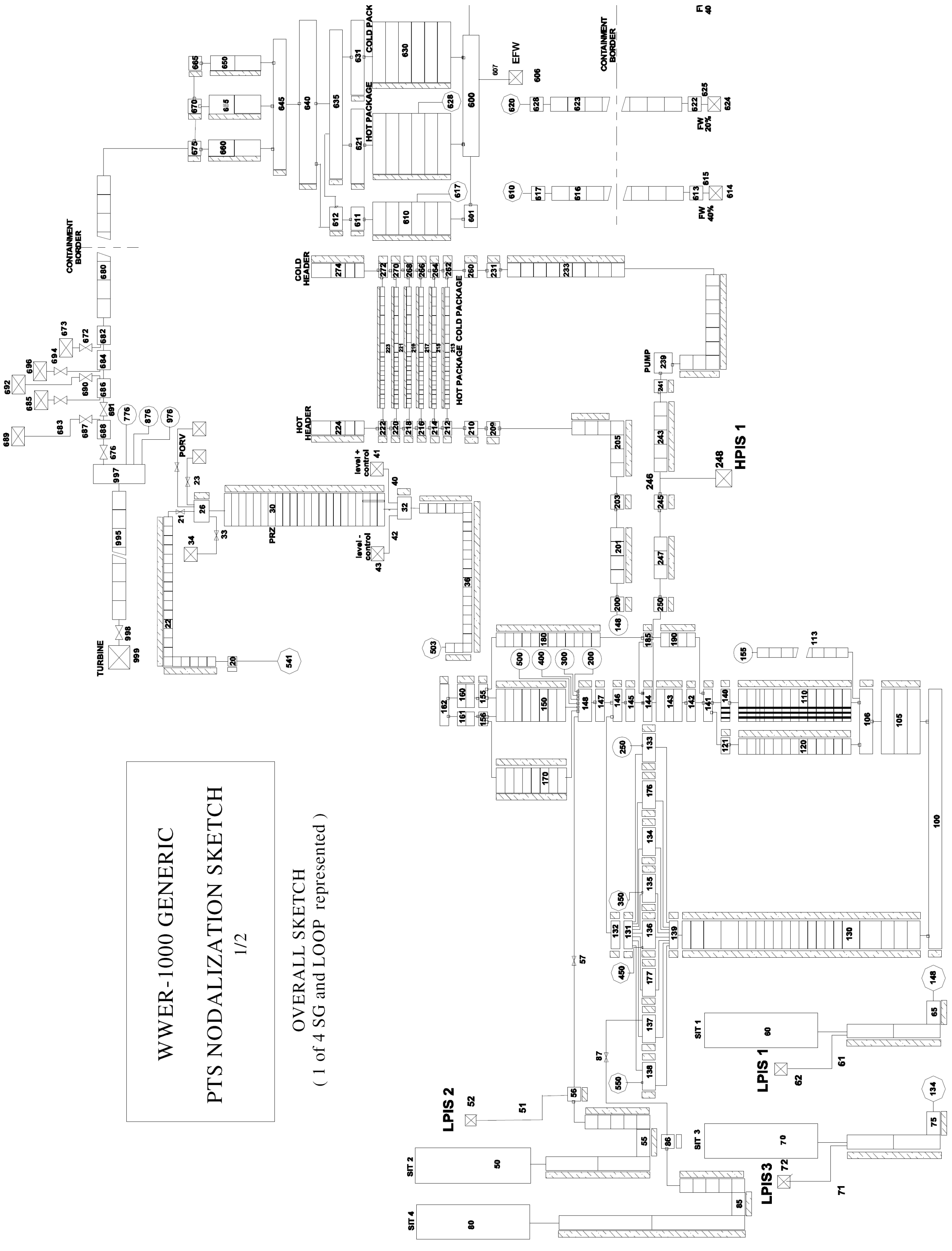


Fig. A.17 – Relap5 VVER1000 NPP nodalization

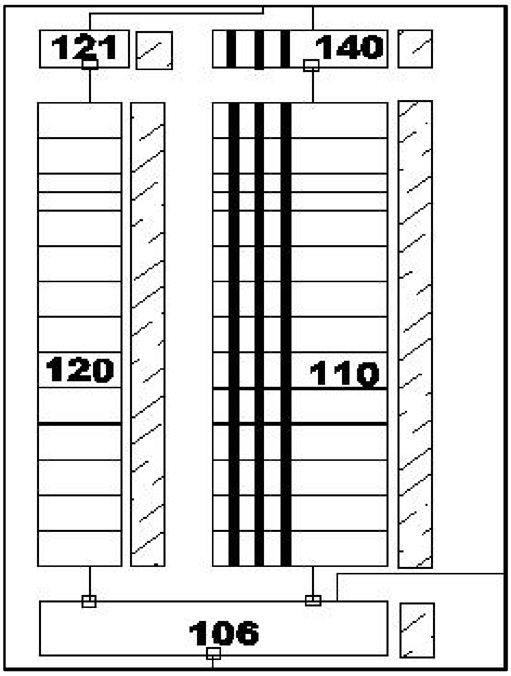


Fig. A.18 – Relap5 VVER1000 NPP nodalization of the core region

A.4.2. Results on decay heat power

The following section provide additional results on the analysis of the decay heat power.

A.4.2.1. Integral of the decay heat power

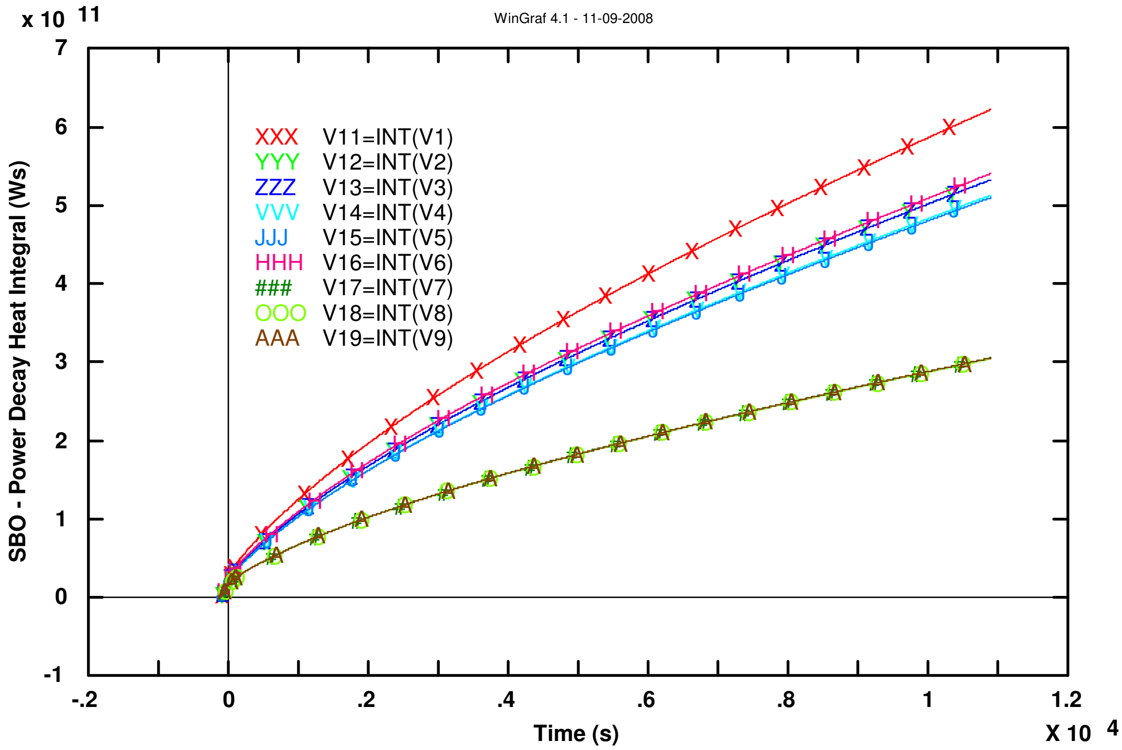


Fig. A.19 – Comparison of the integral of the total decay heat power produced in the transient 1 (up to 11000 s)

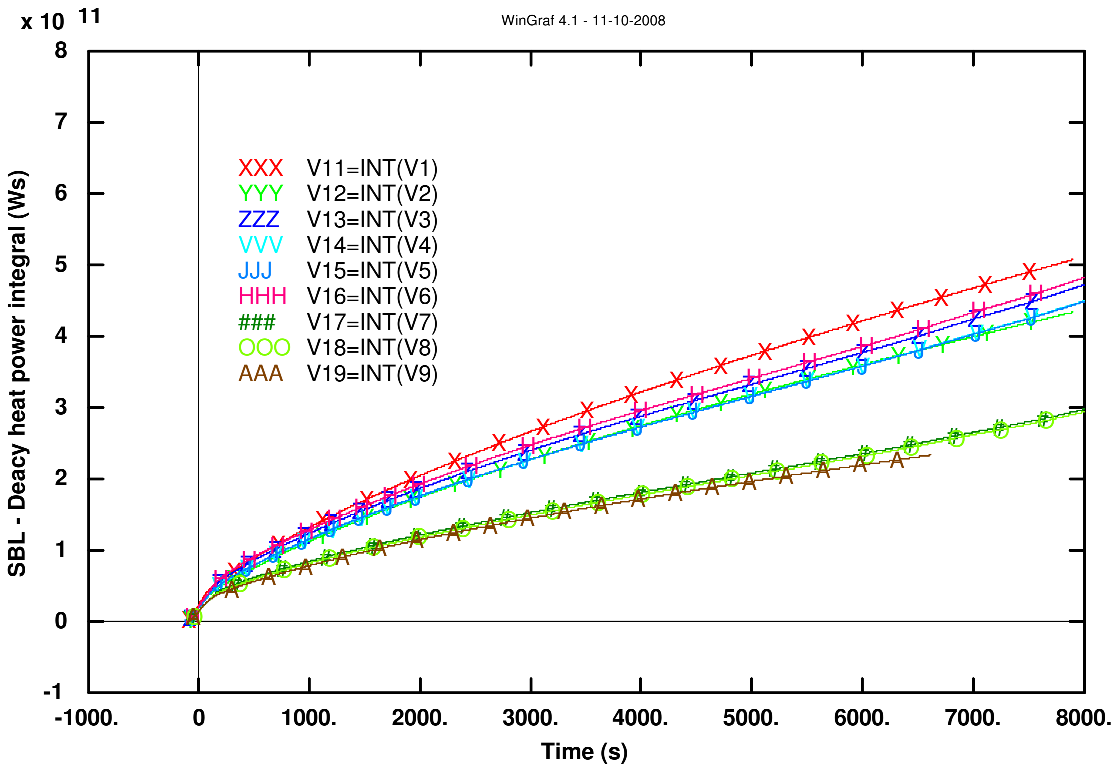


Fig. A.20 – Comparison of the integral of the total decay heat power produced in the transient 2 (up to 8000 s)

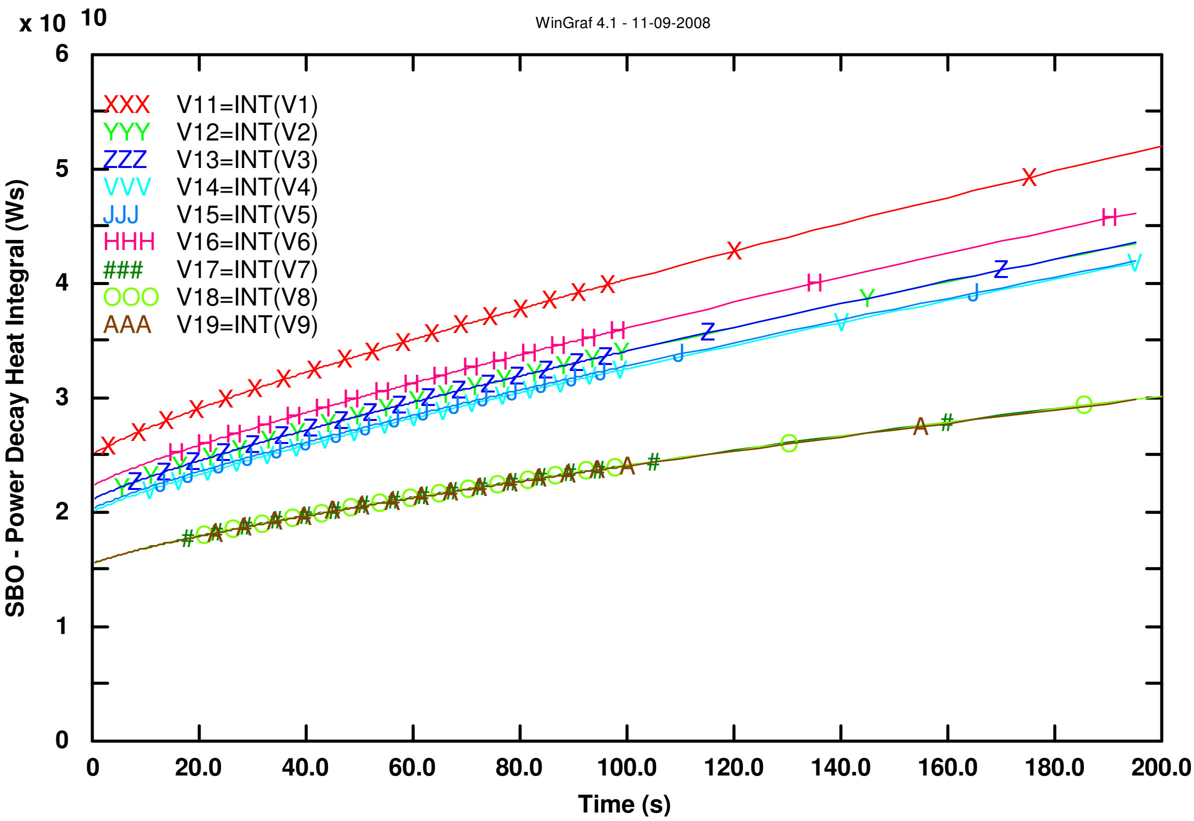


Fig. A.21 – Comparison of the integral of the total decay heat power produced in the transient 1 Magnification (up to 200 s with more detailed time step)

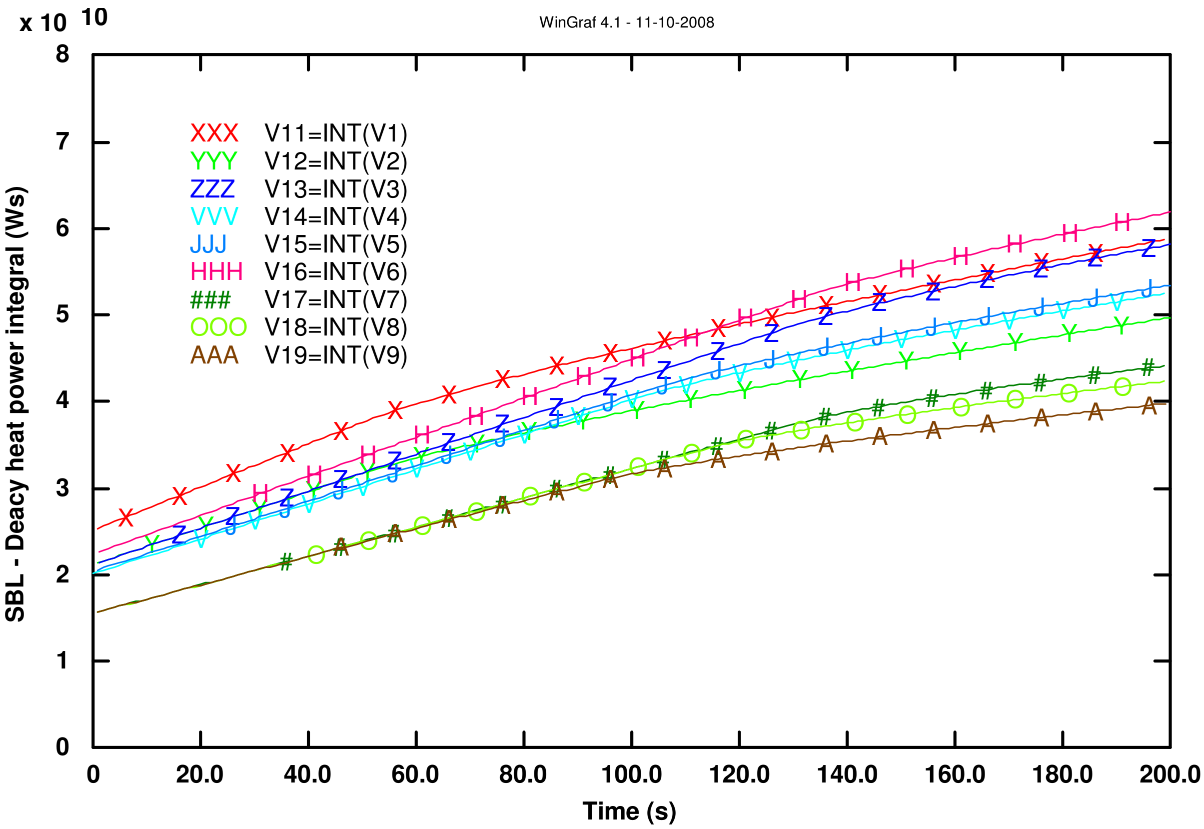


Fig. A.22 – Comparison of the integral of the total decay heat power produced in the transient 2 Magnification (up to 200 s with more detailed time step)

The figures Fig. A.19 ÷ Fig. A.25 show the amount of energy for each case analyzed. The licensing calculation, V11 and 16, has more energy than the BE one, V14,15,17÷18.

A.4.2.2. Derivate of the decay heat power

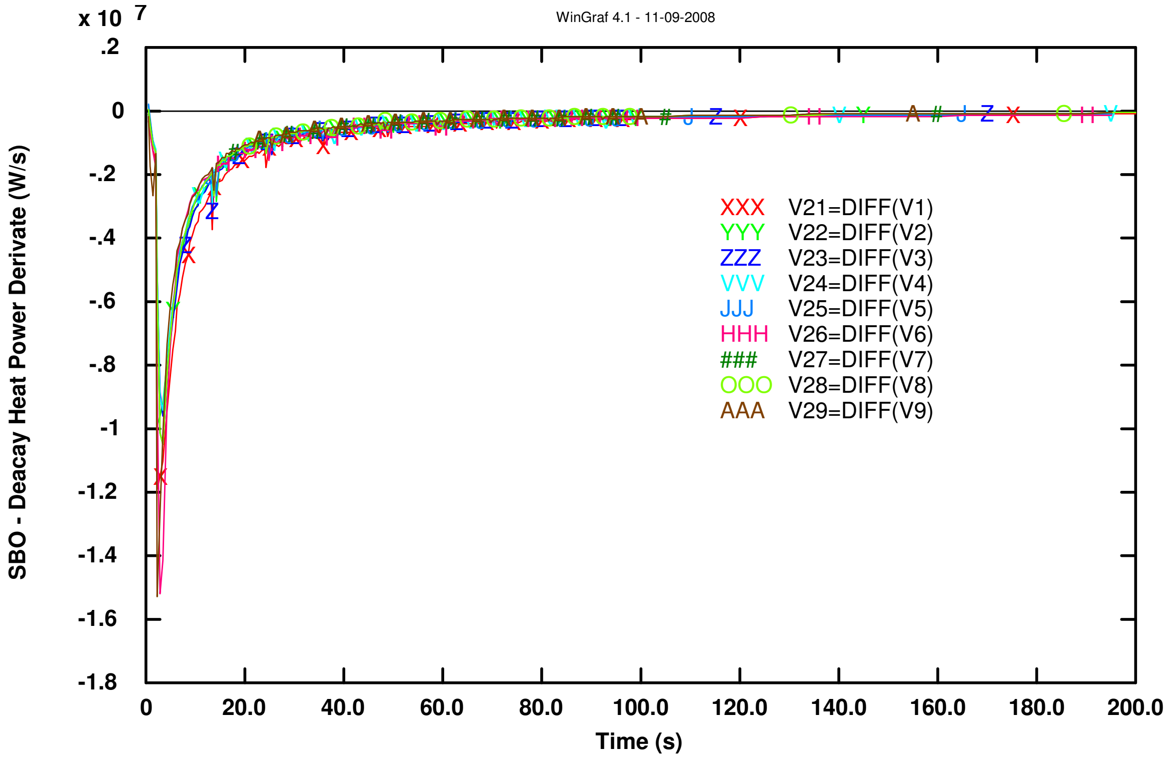


Fig. A.23 – Comparison of the derivate of the total decay heat power produced in the transient 1 (up to 200 s)

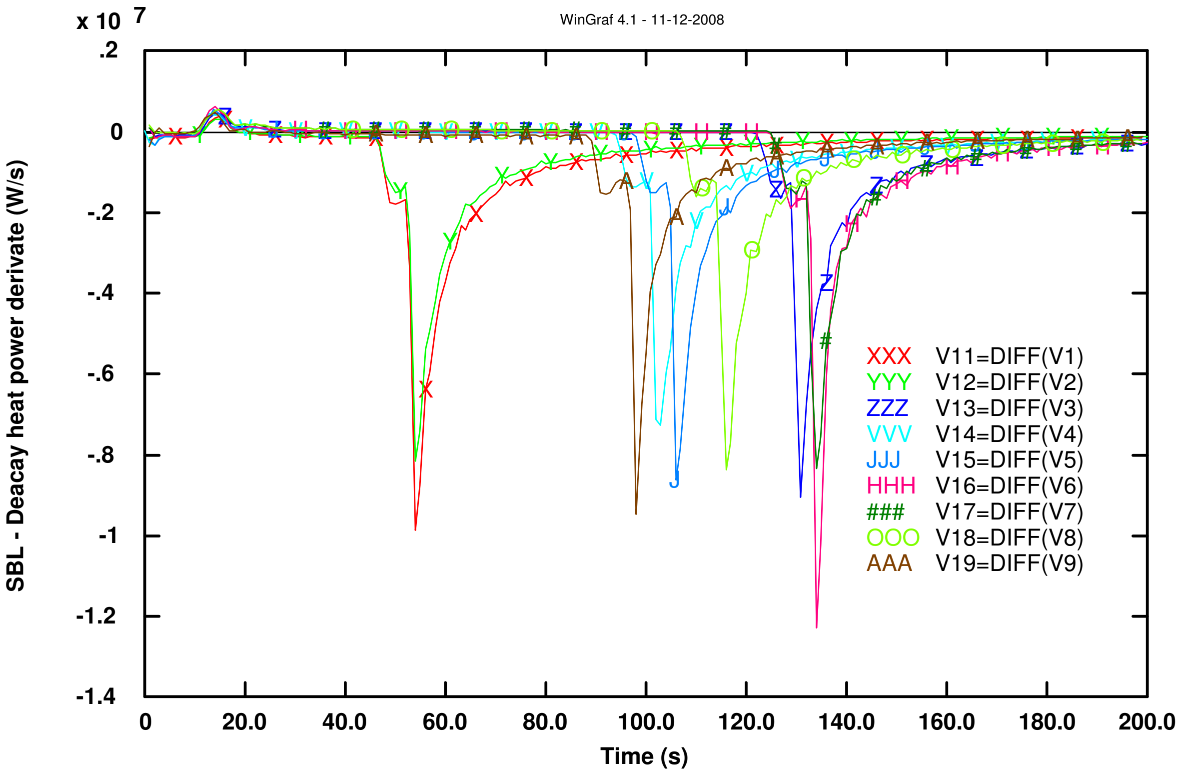


Fig. A.24 – Comparison of the derivate of the total decay heat power produced in the transient 2 (up to 200 s)

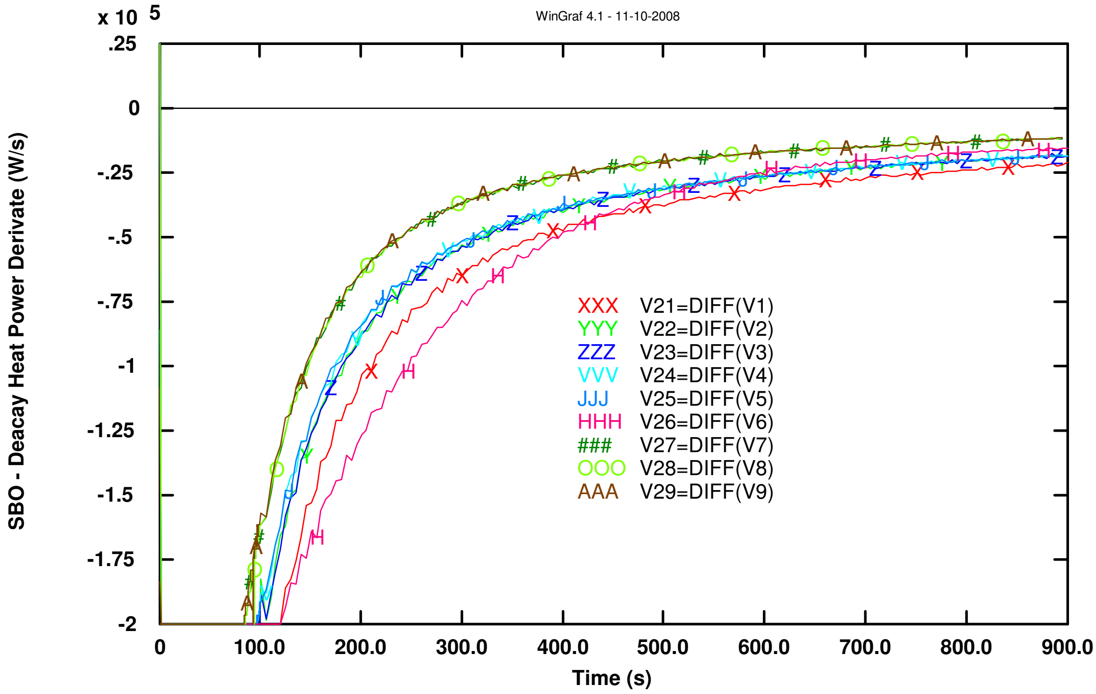


Fig. A.25 – Comparison of the derivate of the total decay heat power produced in the transient 1 (between 100s and 900s)

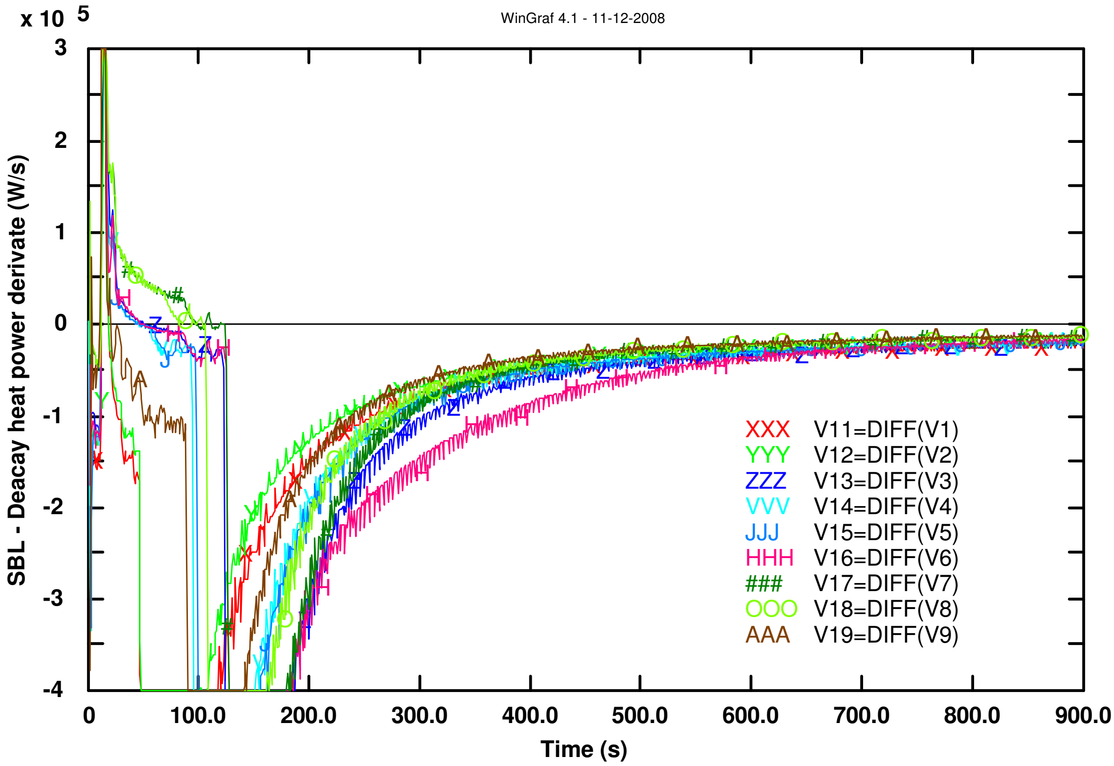


Fig. A.26 – Comparison of the derivate of the total decay heat power produced in the transient 2 (between 100s and 900s)

The figures Fig. A.23 ÷ Fig. A.26 give the derivate of the decay heat power for the cases analyzed, SBO and SBL. It is seen that the conservative assumption , V11 and 16, are faster than the BE one, V14,15,17÷18.

A.4.2.3. Heat flux

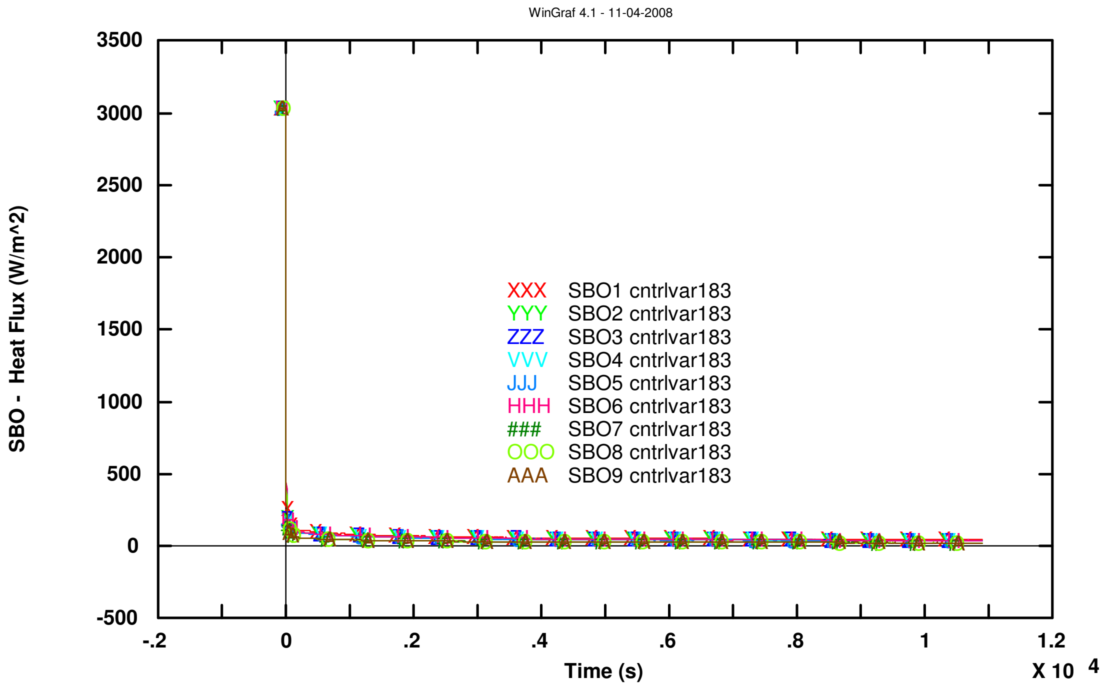


Fig. A.27 – Heat flux in the transient 1 (up to 11000s)

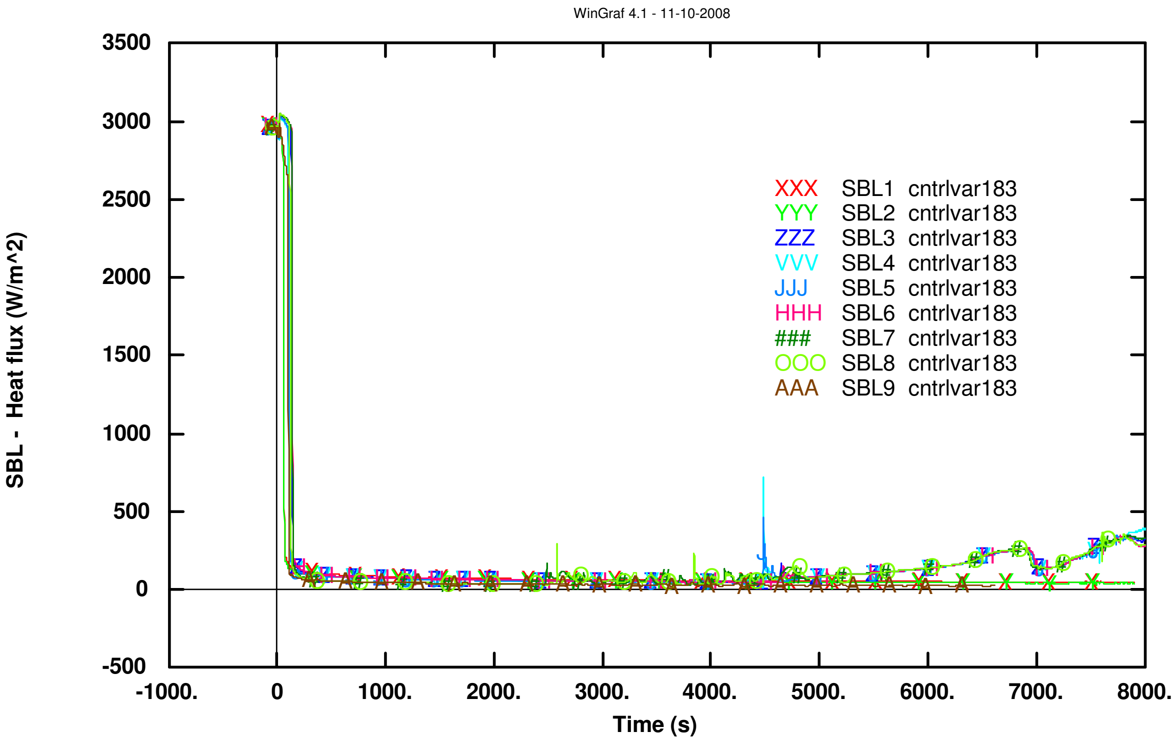


Fig. A.28 – Heat flux in the transient 2 (up to 8000s)

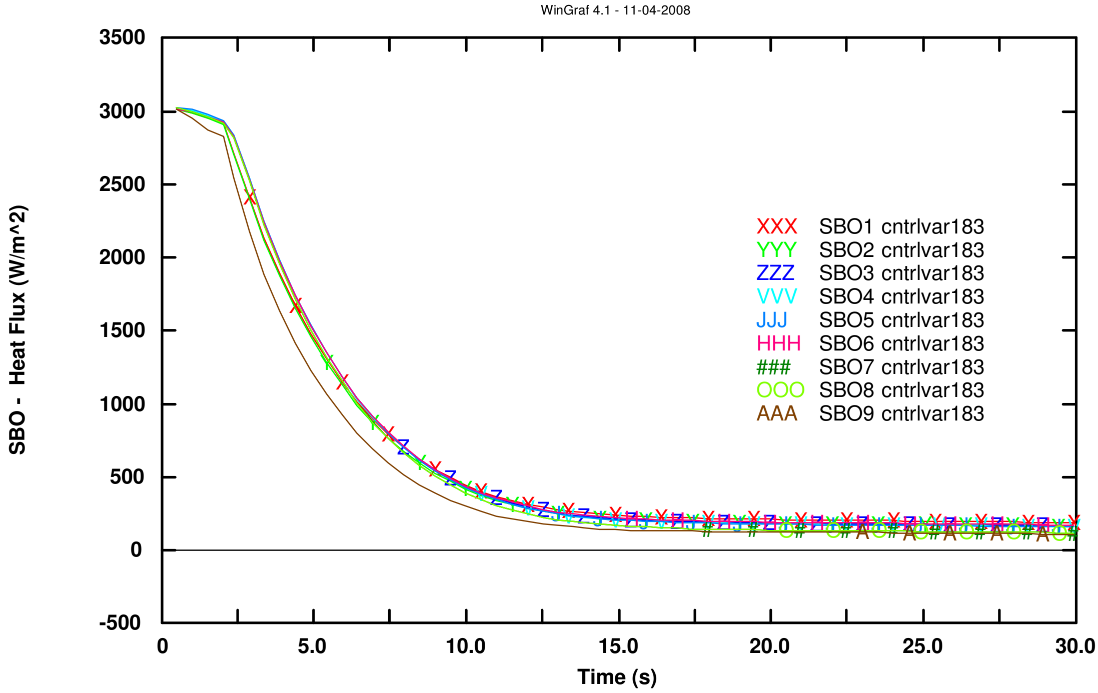


Fig. A.29 – Heat flux in the transient 1 (up to 30 s with more detailed time step)

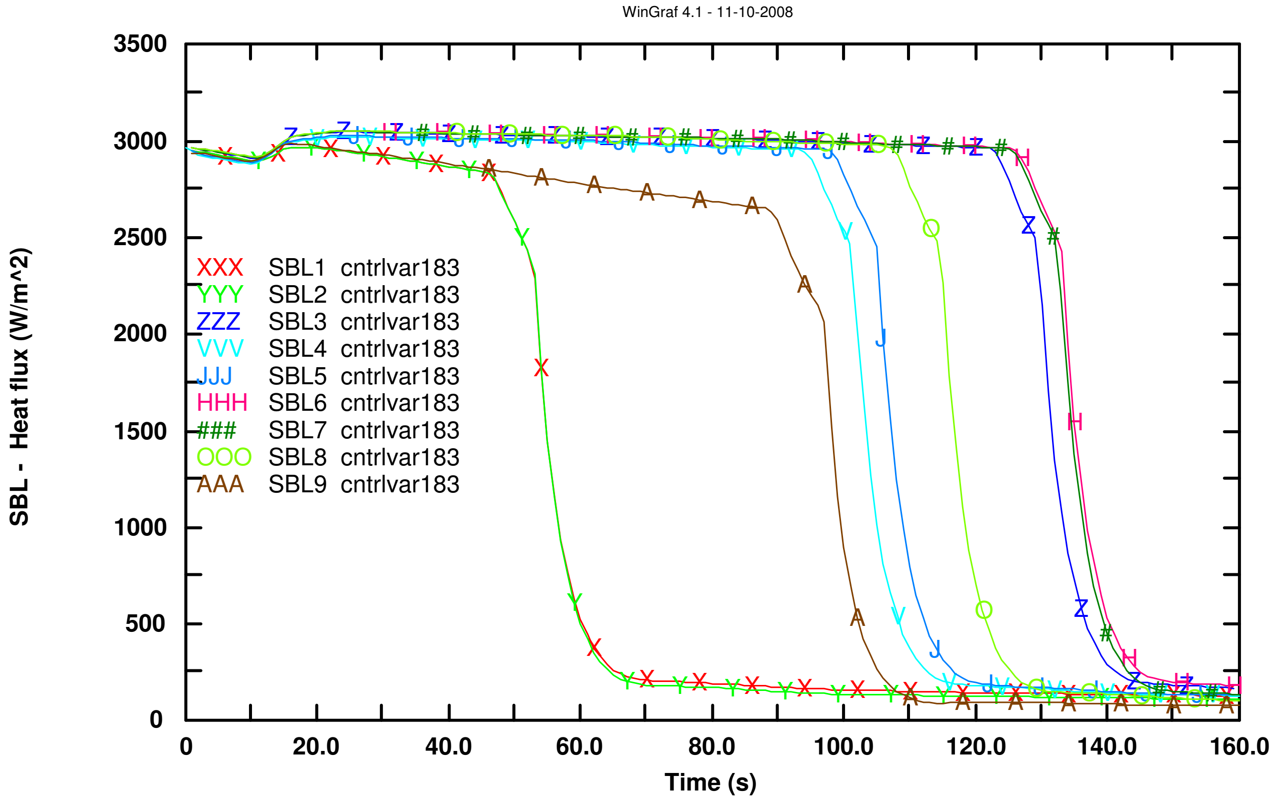


Fig. A.30 – Heat flux in the transient 2 (up to 30 s with more detailed time step)

A.4.2.4. Reactivity

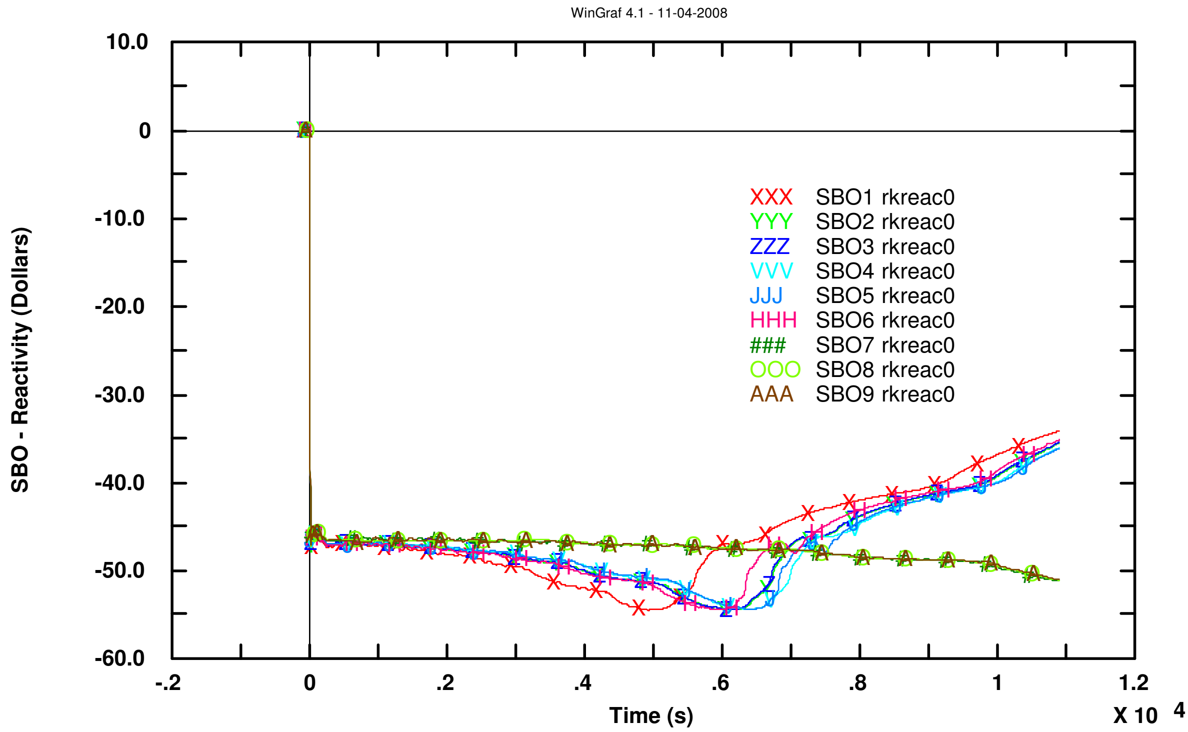


Fig. A.31 – Reactivity in the transient 1 (up to 11000s)

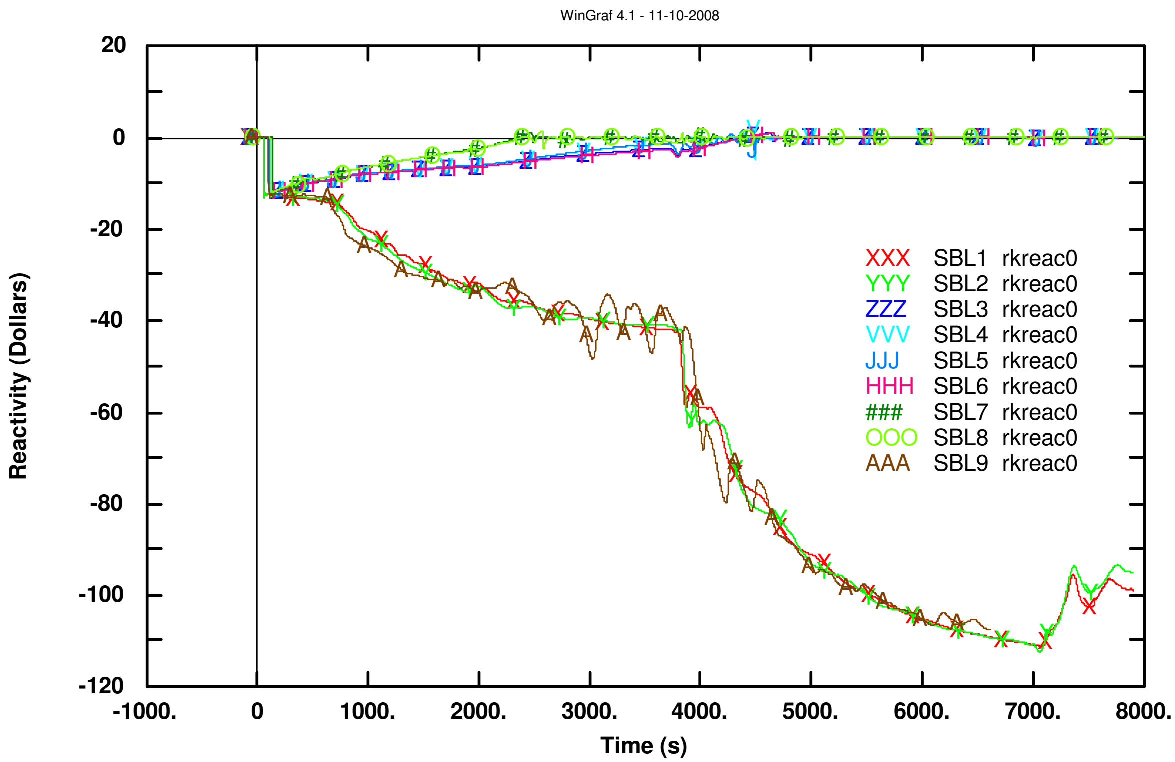


Fig. A.32 – Reactivity in the transient 2 (up to 8000s)

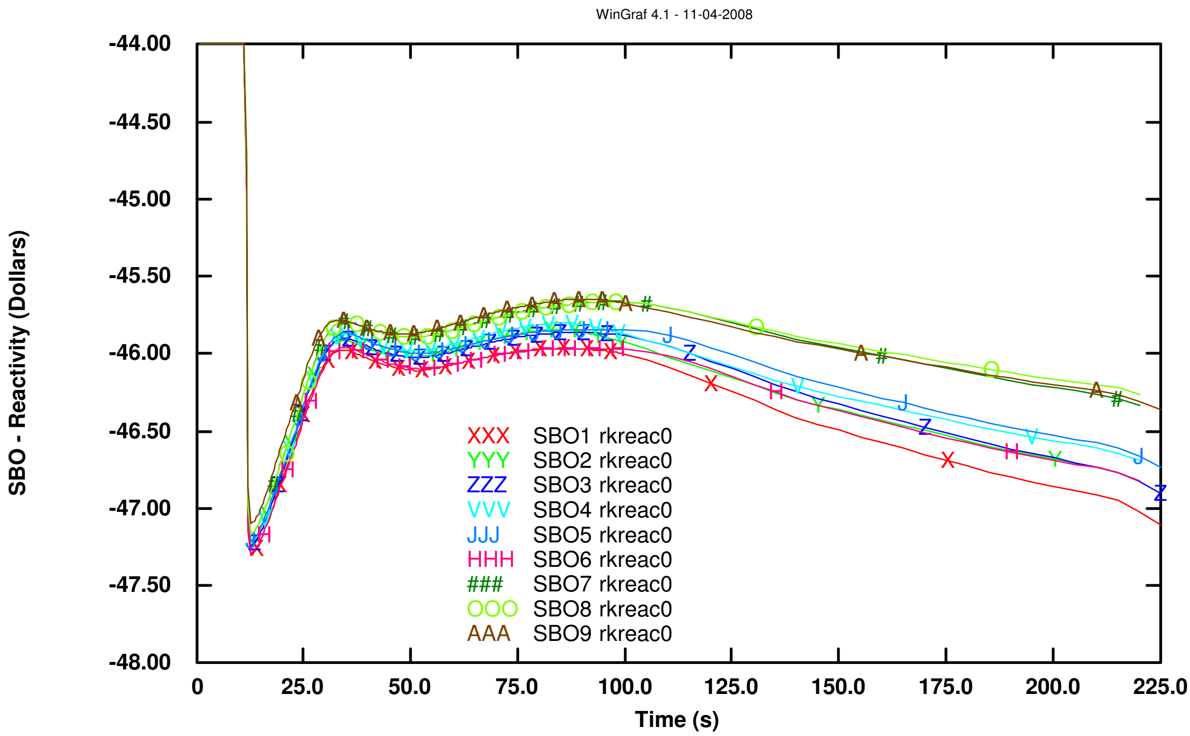


Fig. A.33 – Reactivity in the transient 1 (up to 200s)

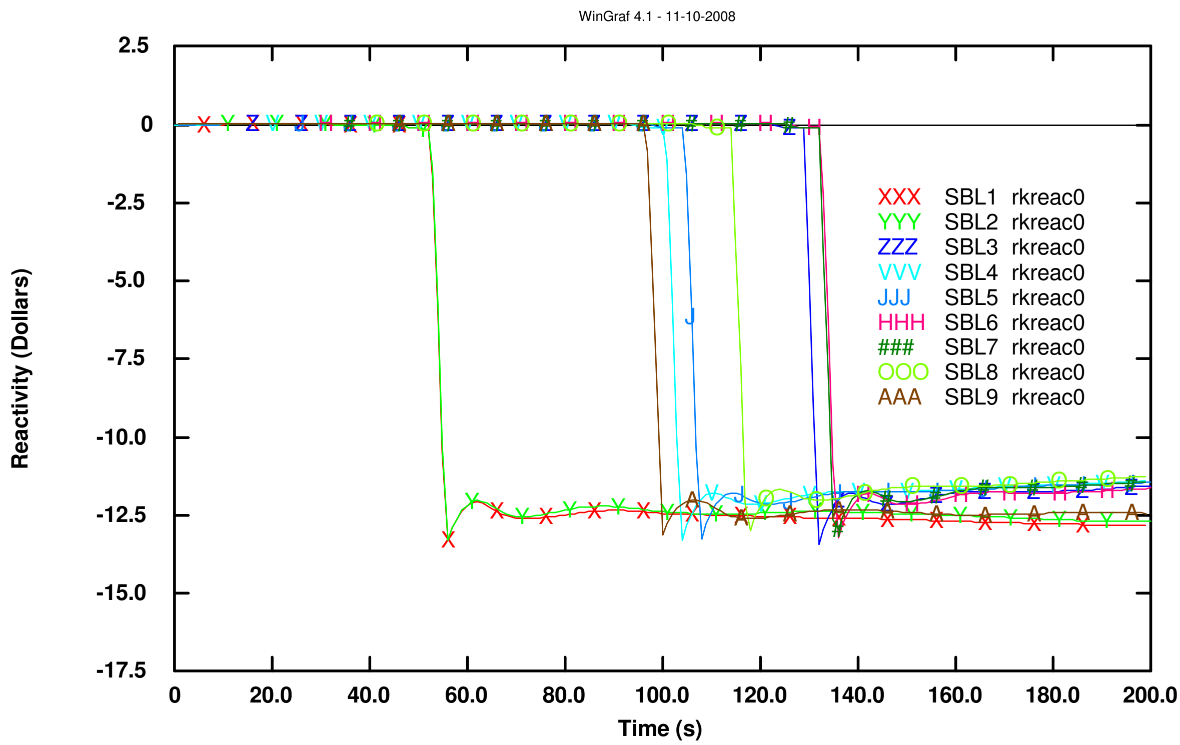


Fig. A.34 – Reactivity in the transient 2 (up to 200 s)

A.4.2.5. Temperature of the water at the outlet from the core

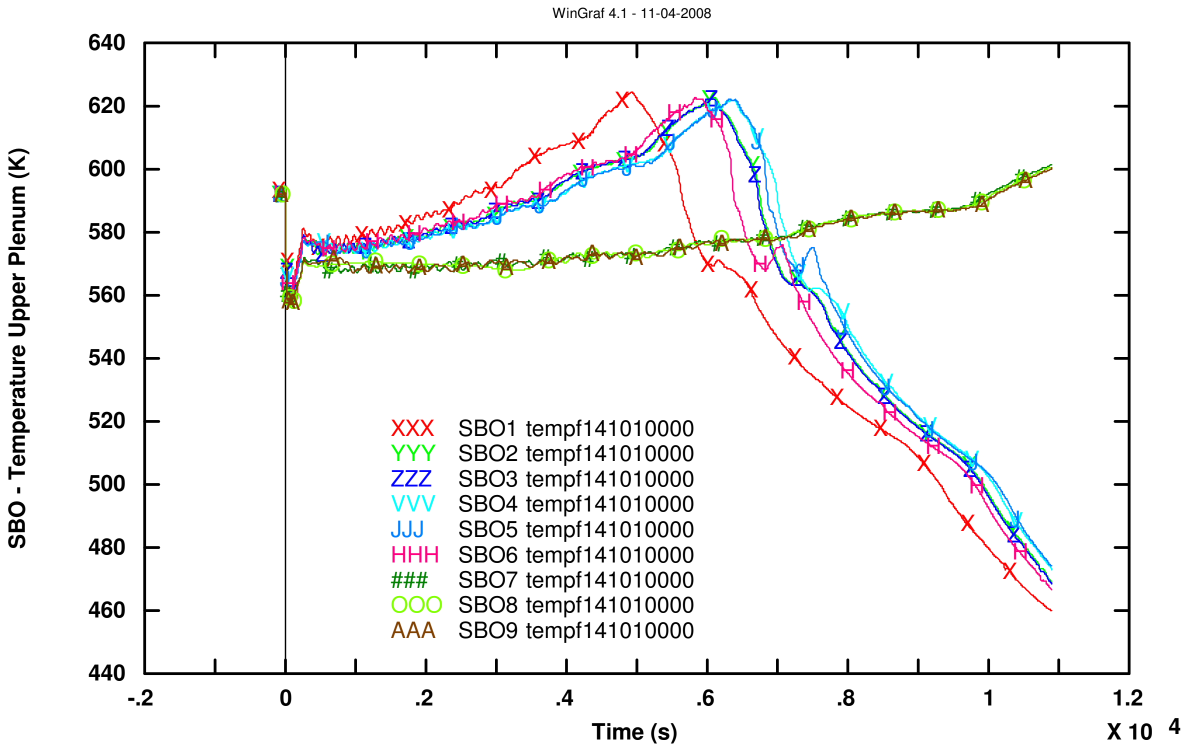


Fig. A.35 – Temperature of the water in the upper plenum in the transient 1 (up to 11000s)

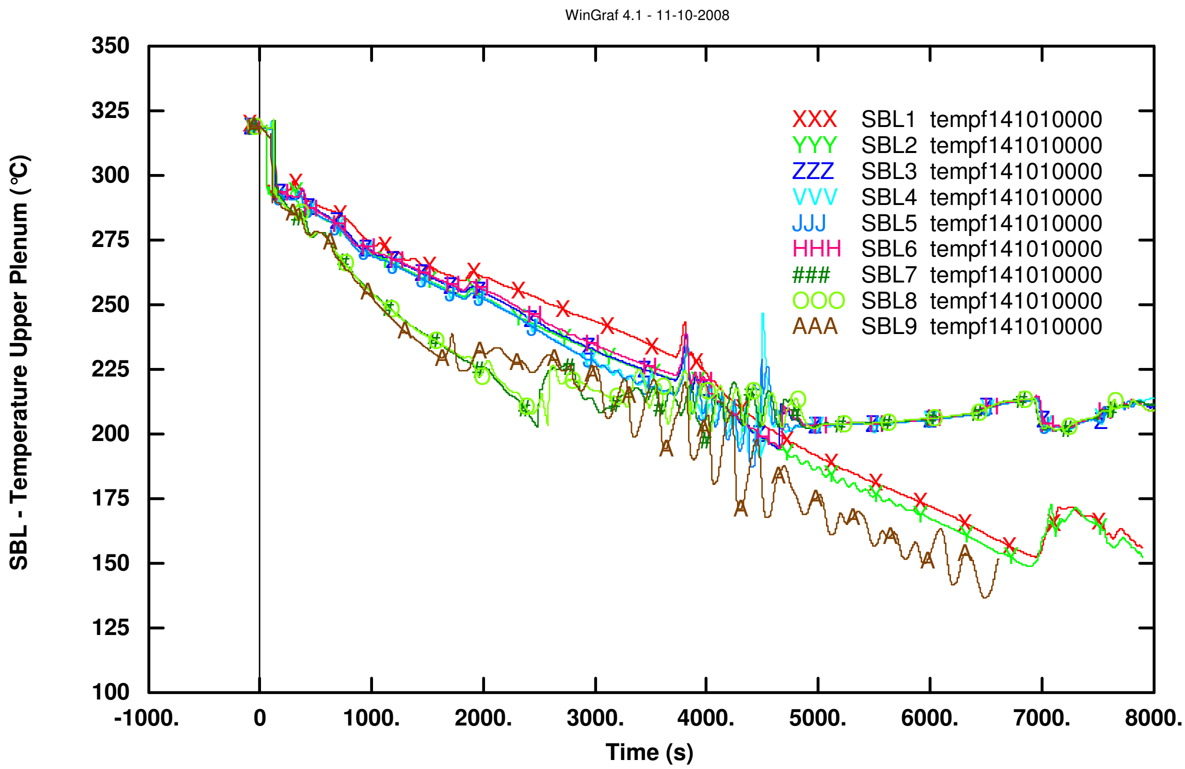


Fig. A.36 – Temperature of the water in the upper plenum in the transient 2 (up to 8000s)

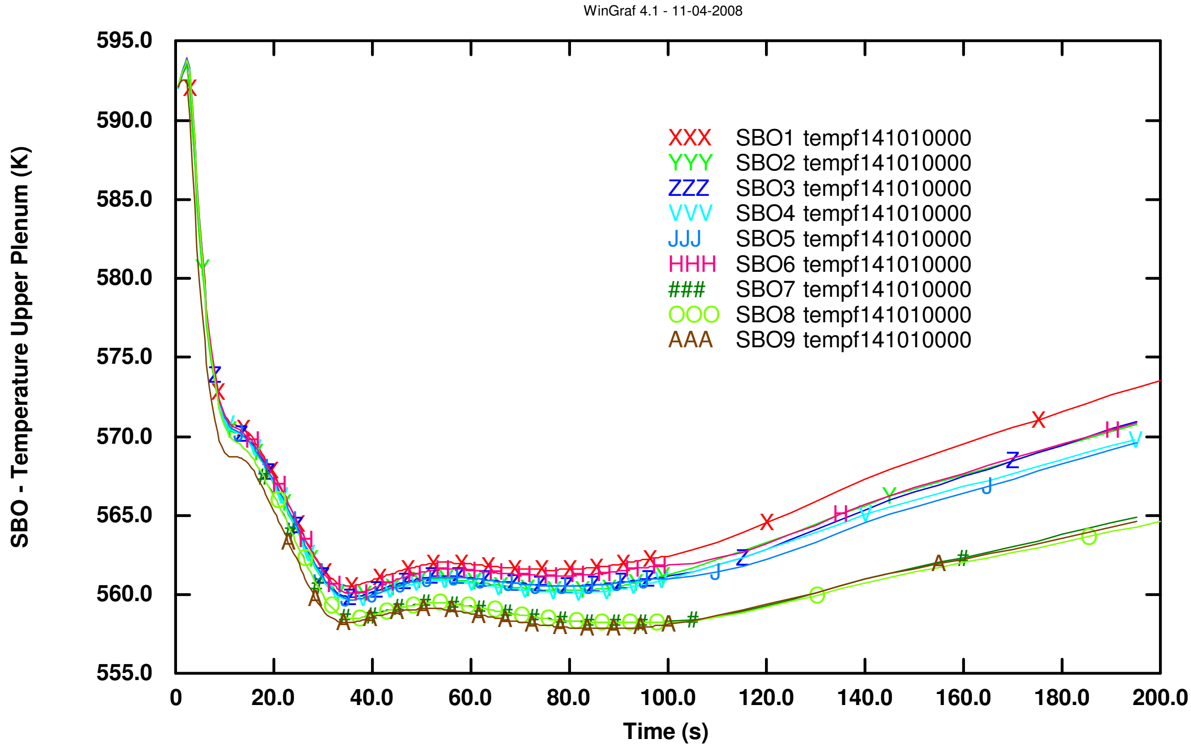


Fig. A.37 – Temperature of the water in the upper plenum in the transient 1(up to 200s)

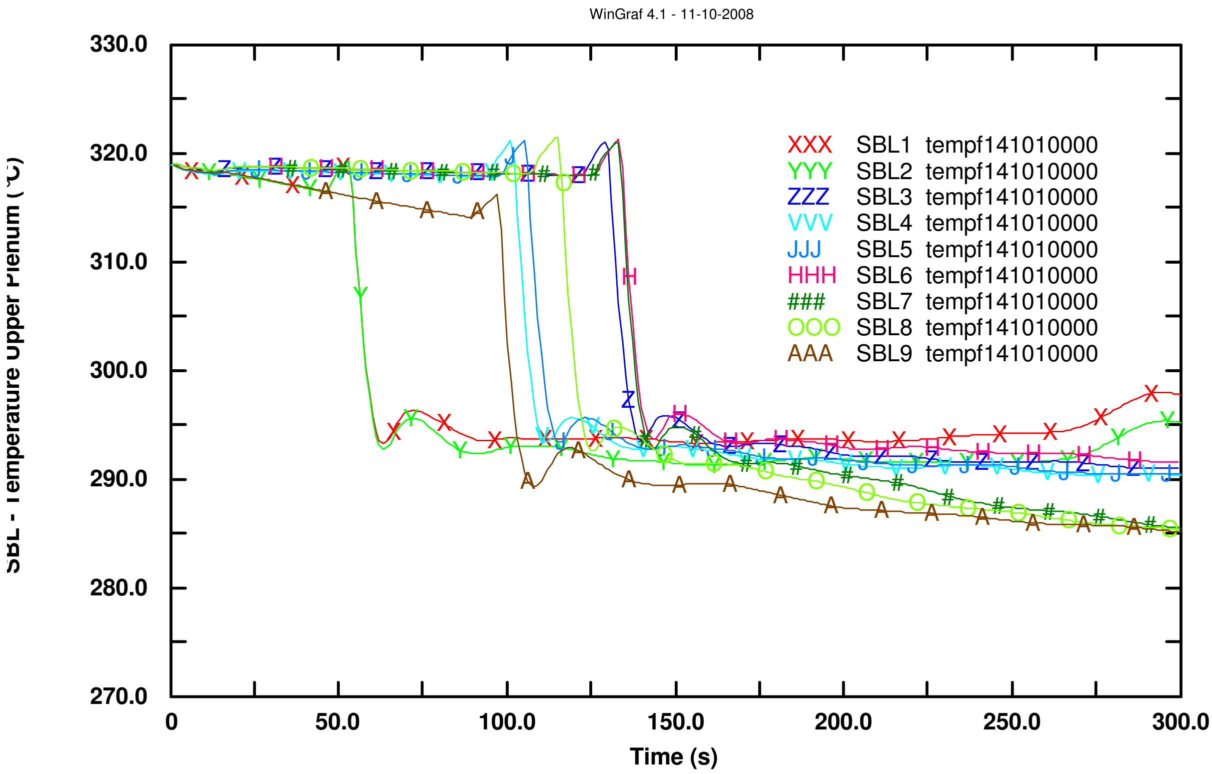


Fig. A.38 – Temperature of the water in the upper plenum in the transient 2 (up to 200s)

A.4.2.6. Mass flow-rate

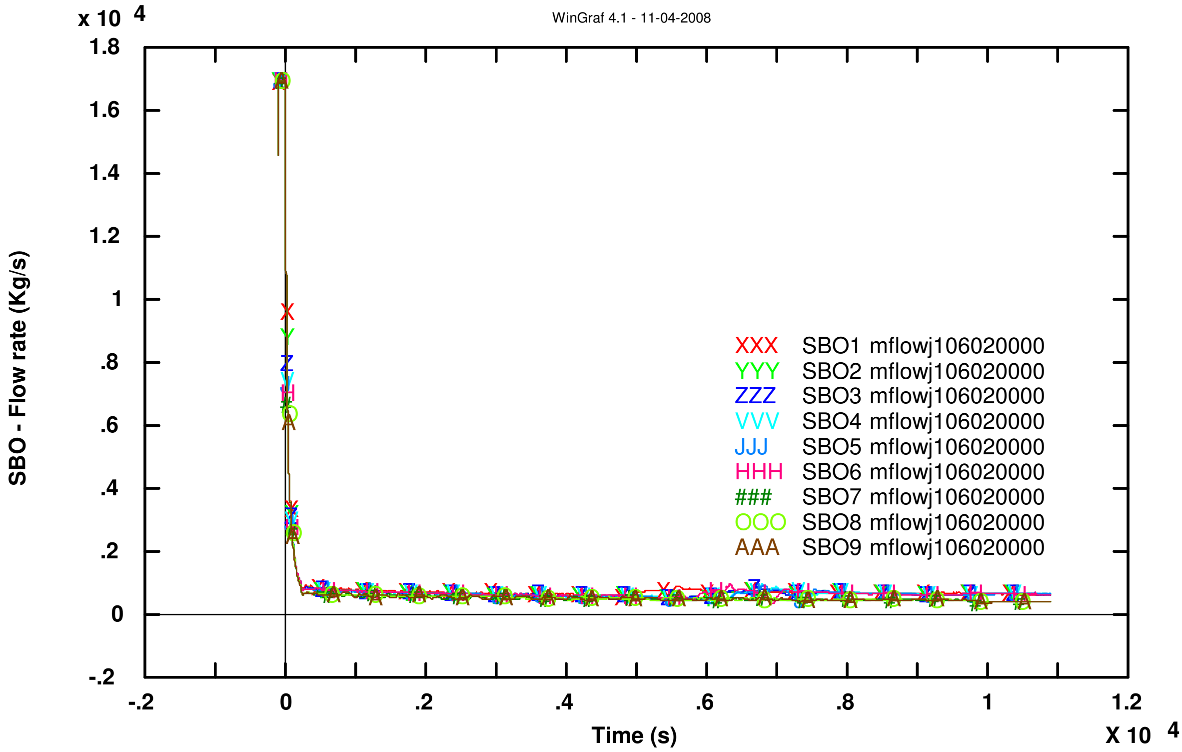


Fig. A.39 – Mass flow-rate in the inlet active core in the transient 1 (up to 11000s)

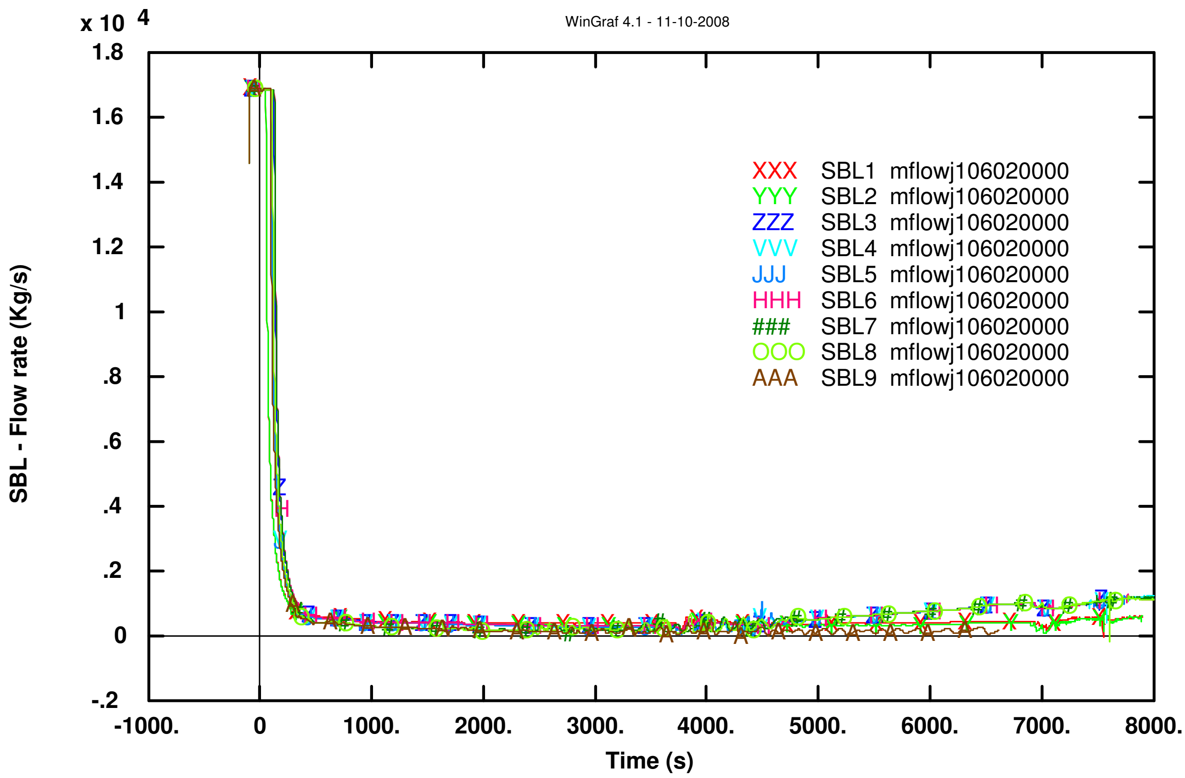


Fig. A.40 – Mass flow-rate in the inlet active core in the transient 2 (up to 8000s)

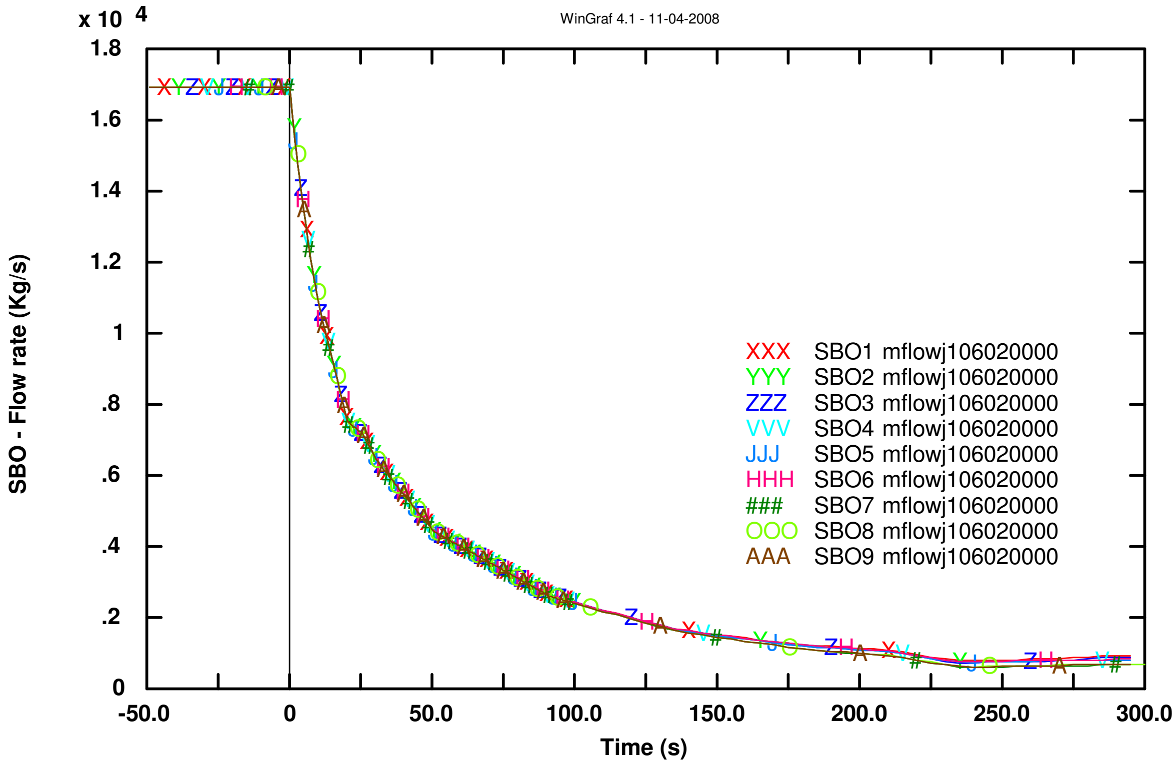


Fig. A.41 – Mass flow-rate in the inlet active core in the transient 1 (up to 300s)

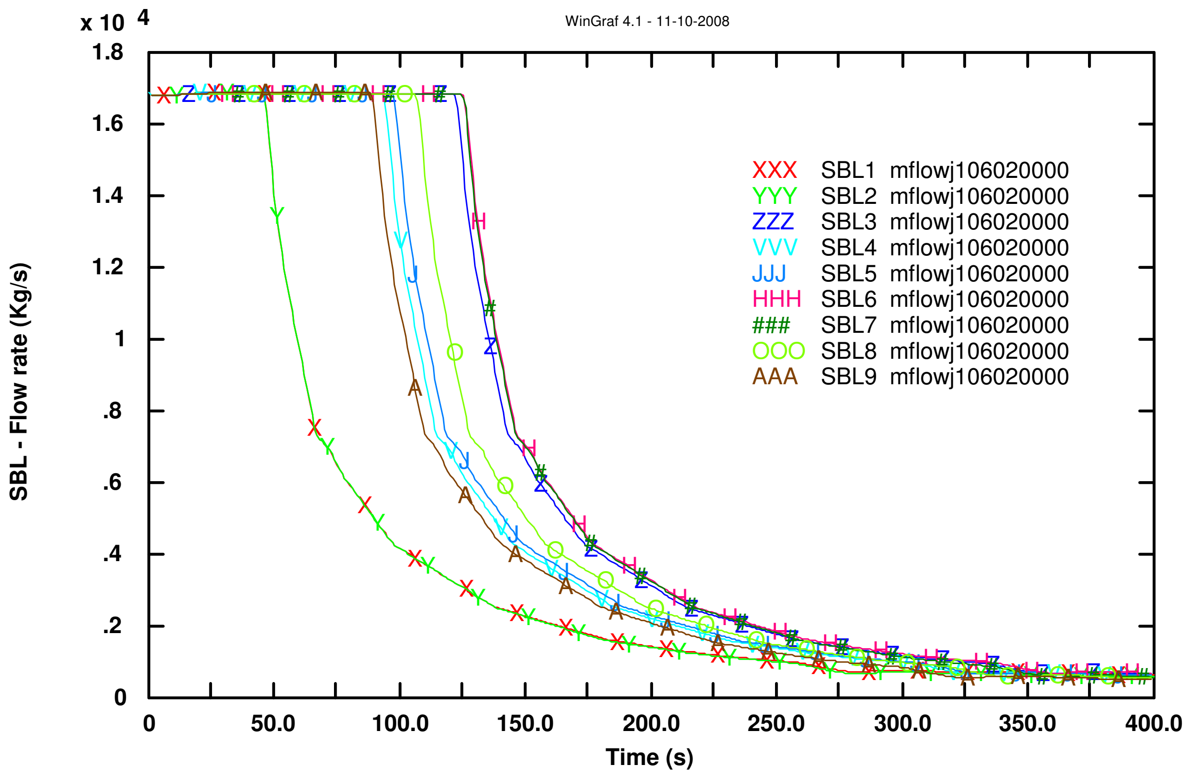


Fig. A.42 – Mass flow-rate in the inlet active core in the transient 2 (up to 400s)

A.4.2.7. Pressure in the upper plenum

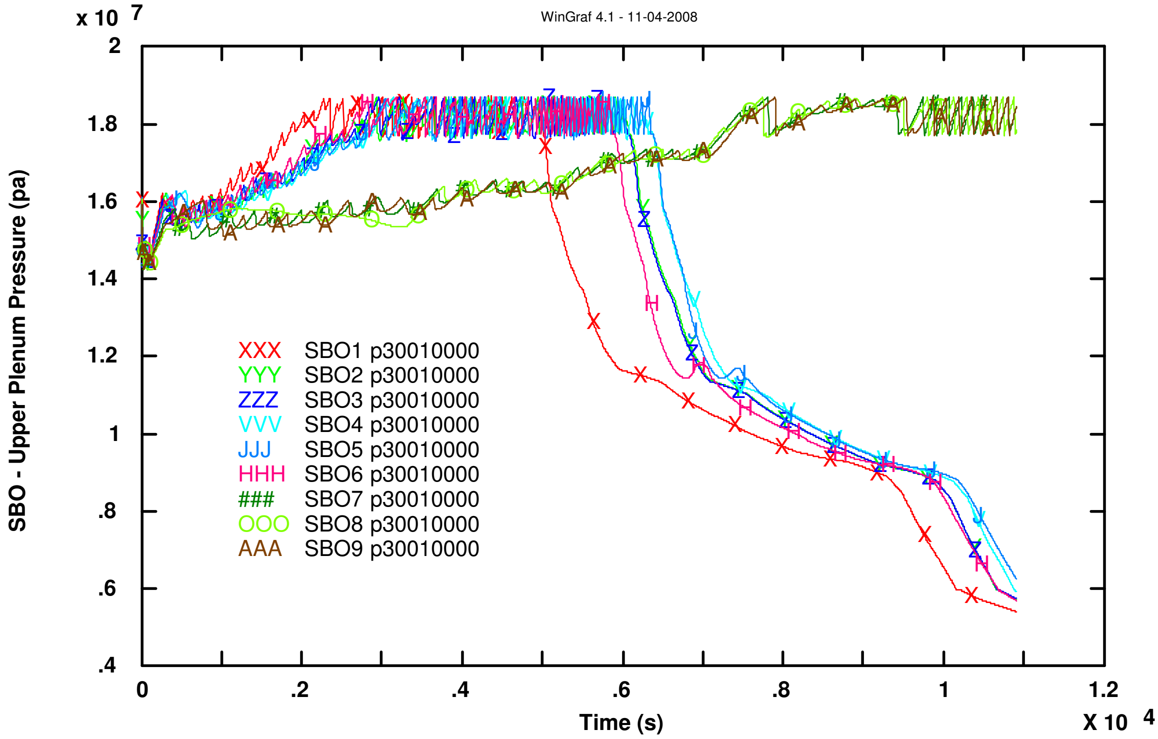


Fig. A.43 – Pressure in the upper plenum in the transient 1 (up to 11000s)

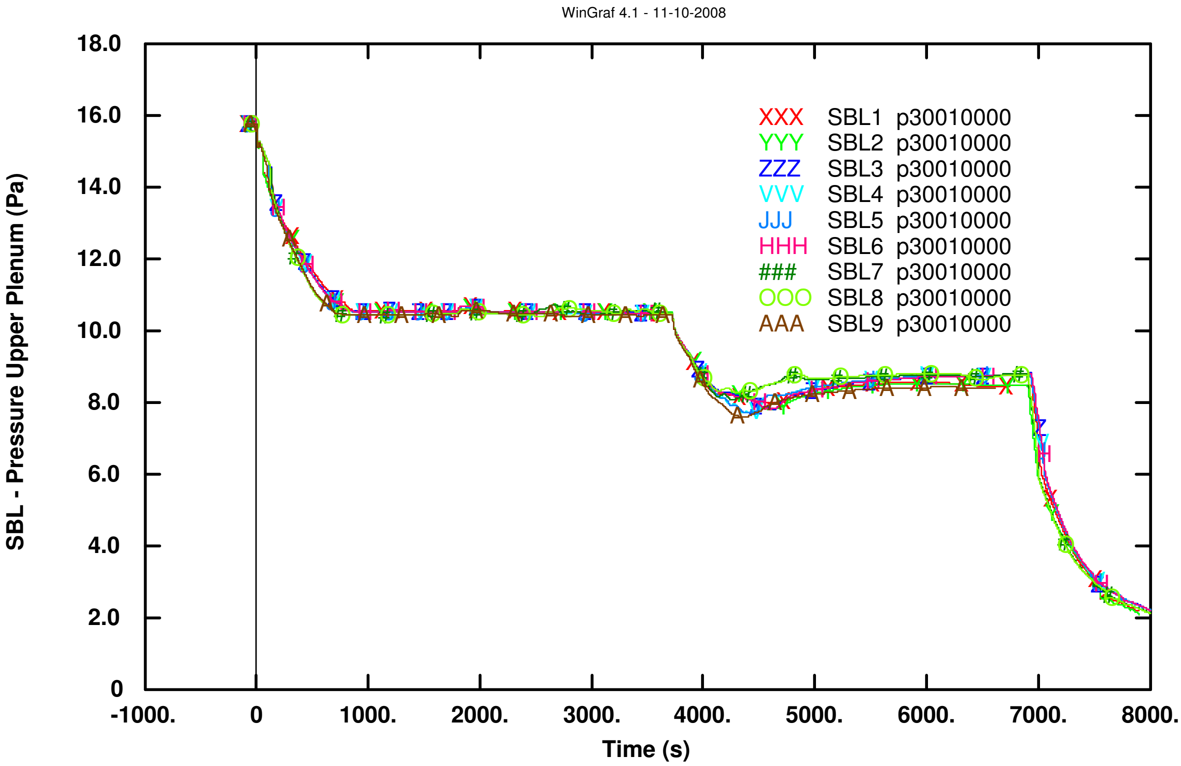


Fig. A.44 – Pressure in the upper plenum in the transient 2 (up to 8000s)

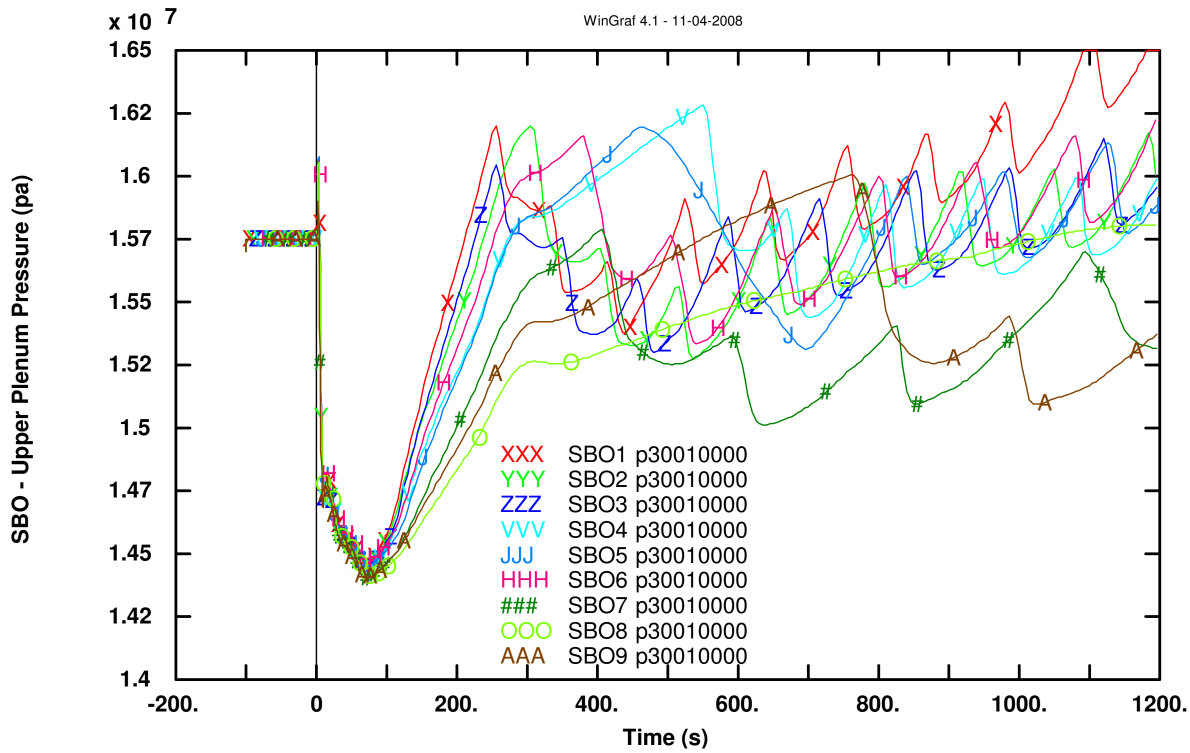


Fig. A.45 – Pressure in the upper plenum in the transient 1 (up to 1200s)

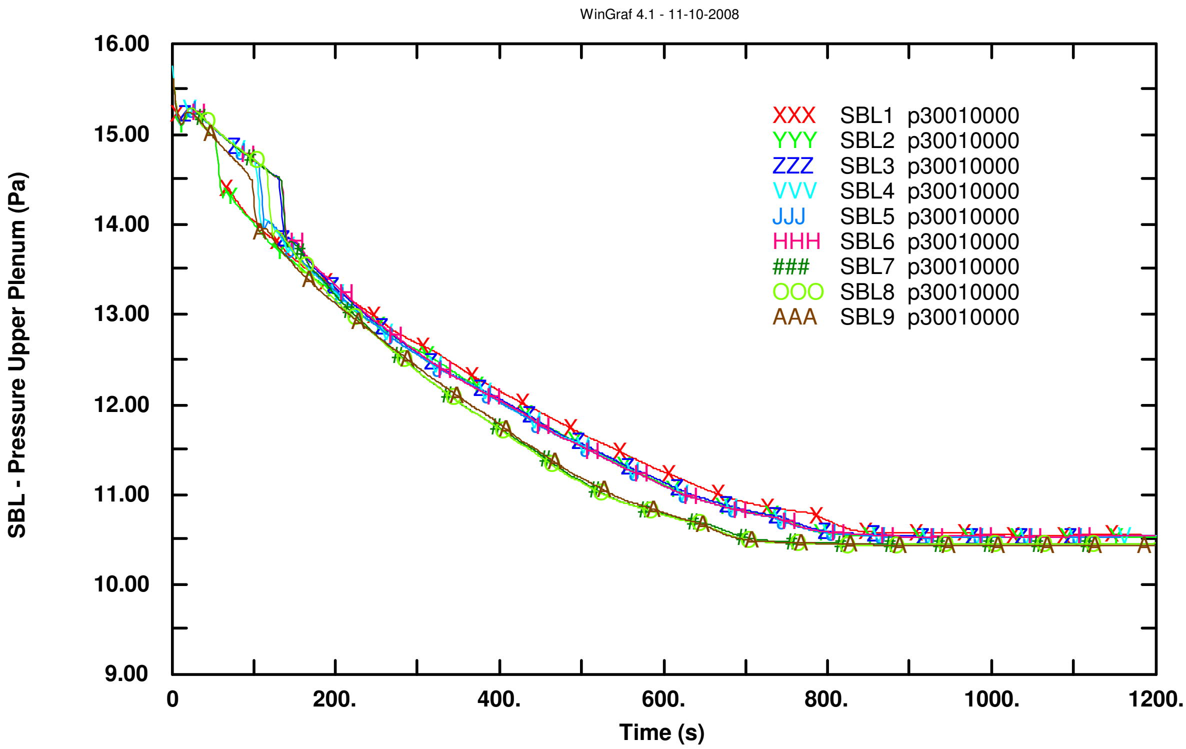


Fig. A.46 – Pressure in the upper plenum in the transient 2 (up to 1200s)

The figure Fig. A.43 show that in the SBO the overpressure in the conservative case, SBO1, is earlier than in the BE, SBO4 and 5: about 500s. Furthermore, if the real power history, SBO7 ÷ 9 is considered, the overpressure is 7000s later.

A.4.3. Results on decay heat: gamma and beta contribution

The section provide the calculation implemented to perform the Monte Carlo analyses.

A.4.3.1. Gamma peak factor for sinusoidal distribution

Inside the reactor, i.e. research and commercial reactors, hereafter, the gamma distribution is considered to have a sinusoidal distribution as for the neutron (1).

$$P_{Tot} = \int_V P_{\max} \cos\left(\frac{r}{R}\frac{\pi}{2}\right)\cos\left(\frac{2z-H}{H}\frac{\pi}{2}\right)2\pi r dr dz \qquad 1)$$

For the purpose the core is schematized as a cylinder with origin in the center of the axis z (2).

$$P_{Tot} = P_{\max}\int_{r_1}^{r_2}\cos\left(\frac{r}{R}\frac{\pi}{2}\right)2\pi r dr \int_0^{\frac{H}{2}}\cos\left(\frac{2z-H}{H}\frac{\pi}{2}\right)dz \qquad 2)$$

Considering that[2]:

$$\cos\left(\frac{2z-H}{H}\frac{\pi}{2}\right) = \cos-\left(\frac{\pi}{2}-\frac{z\pi}{H}\right) = \cos\left(\frac{\pi}{2}-\frac{z\pi}{H}\right) = \sin\left(\frac{\pi}{H}z\right)$$

Developing the integral along z and r (3)

$$P_{Tot}(z) = \int_{h_1}^{h_2}\sin\left(\frac{\pi}{H}z\right)dz = -\frac{H}{\pi}\left|\cos\left(\frac{\pi}{H}z\right)\right|_{h_1}^{h_2} = -\frac{H}{\pi}\left[\cos\left(\frac{\pi}{H}h_2\right)-\cos\left(\frac{\pi}{H}h_1\right)\right] \qquad 3)$$

$$P_{Tot}(H) = \int_0^H \sin\left(\frac{\pi}{H}z\right)dz = -\frac{H}{\pi}\left|\cos\left(\frac{\pi}{H}z\right)\right|_0^H = -\frac{H}{\pi}\left[\cos(\pi)-\cos(0)\right] = 2\frac{H}{\pi} \qquad 4)$$

---

[2] The calculation is done using the results of the tabulated transcendent functions [113]:

$$\int \sin^n axdx = -\frac{1}{na}\sin^{n-1}ax\cos ax + \frac{n-1}{n}\int \sin^{n-2}axdx$$

$$\int x^n \cos axdx = \frac{1}{a}x^n \sin ax - \frac{n}{a}\int x^{n-1}\sin axdx$$

and considering that: sin 90°+/-a = + cos a;
cos 90°+/-a =-/+sin a;
cos+/-a = cos a;
sin+/-a = -/+ sina

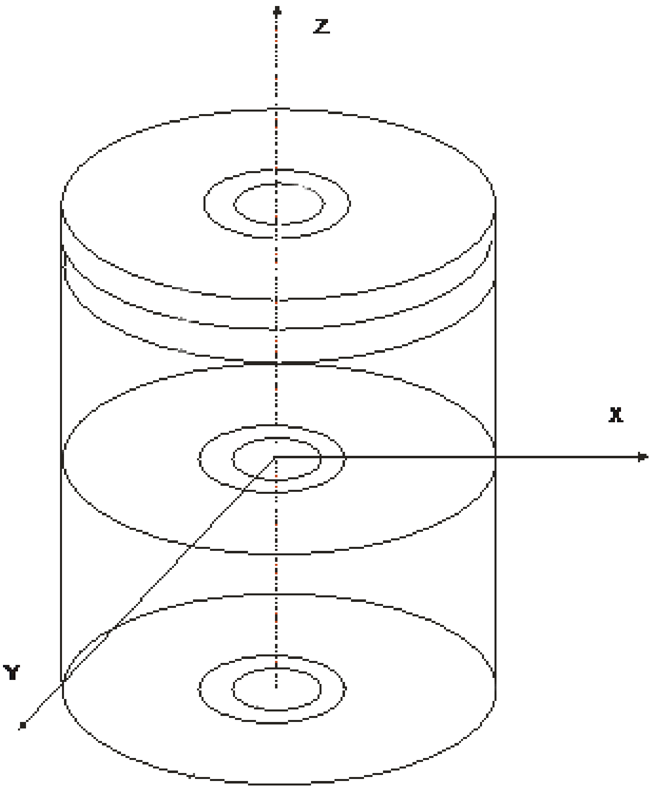


Fig. A.47 – Model of the LOFT and PWR core

$$P_{Tot}(r) = 2\pi\int_{r_1}^{r_2} r\cos\left(\frac{r}{R}\frac{\pi}{2}\right)dr = 2\pi\left[r\left(\frac{2R}{\pi}\right)\sin\left(\frac{r}{R}\frac{\pi}{2}\right) - \frac{2R}{\pi}\int_{r_1}^{r_2}\sin\left(\frac{r}{R}\frac{\pi}{2}\right)dr\right]_{r_1}^{r_2} =$$

$$= 2\pi\left[r\left(\frac{2R}{\pi}\right)\sin\left(\frac{r}{R}\frac{\pi}{2}\right) - \frac{2R}{\pi}\left(-\frac{2R}{\pi}\cos\left(\frac{r}{R}\frac{\pi}{2}\right)\right)\right]_{r_1}^{r_2} =$$

$$= 4R\left[\left(r_2\sin\left(\frac{r_2}{R}\frac{\pi}{2}\right) + \frac{2R}{\pi}\cos\left(\frac{r_2}{R}\frac{\pi}{2}\right)\right) - \left(r_1\sin\left(\frac{r_1}{R}\frac{\pi}{2}\right) + \frac{2R}{\pi}\cos\left(\frac{r_1}{R}\frac{\pi}{2}\right)\right)\right] \quad 5)$$

$$P_{Tot}(R) = 2\pi\int_0^R r\cos\left(\frac{r}{R}\frac{\pi}{2}\right)dr =$$

$$= 4R\left[\left(R\sin\left(\frac{R}{R}\frac{\pi}{2}\right) + \frac{2R}{\pi}\cos\left(\frac{R}{R}\frac{\pi}{2}\right)\right) - \left(0\sin\left(\frac{0}{R}\frac{\pi}{2}\right) + \frac{2R}{\pi}\cos\left(\frac{0}{R}\frac{\pi}{2}\right)\right)\right] = \quad 6)$$

$$= 4R\left[R\left(\sin\left(\frac{\pi}{2}\right) + \frac{2}{\pi}\cos\left(\frac{\pi}{2}\right)\right) - R\left(\frac{2}{\pi}\right)\right] = 4R\left[R - \frac{2R}{\pi}\right] = \frac{4R^2}{\pi}(\pi - 2)$$

The total gamma power is given by (7).

$$P_{Tot} = P_{max}2\pi\left[r\left(\frac{2R}{\pi}\right)\sin\left(\frac{r}{R}\frac{\pi}{2}\right) + \left(\left(\frac{2R}{\pi}\right)^2\cos\left(\frac{r}{R}\frac{\pi}{2}\right)\right)\right]_{r_1}^{r_2} 2\frac{H}{\pi} \quad 7)$$

Developing the function (7) it is possible to see the relation between mean power $\overline{P}$ and P_{max} (8) and the ratio $\frac{P_{max}}{\overline{P}}$ (9).

$$P_{Tot} = P_{\max} 2\pi \left[r\left(\frac{2R}{\pi}\right)\sin\left(\frac{r}{R}\frac{\pi}{2}\right) - \left(-\left(\frac{2R}{\pi}\right)^2 \cos\left(\frac{r}{R}\frac{\pi}{2}\right)\right)\right]_0^R 2\frac{H}{\pi} =$$

$$= P_{\max} 2\pi \left\{ \left[R\left(\frac{2R}{\pi}\right)\sin\left(\frac{R}{R}\frac{\pi}{2}\right) - \left(-\left(\frac{2R}{\pi}\right)^2 \cos\left(\frac{R}{R}\frac{\pi}{2}\right)\right)\right] - \left[0 - \left(-\left(\frac{2R}{\pi}\right)^2 \cos\left(\frac{0}{R}\frac{\pi}{2}\right)\right)\right]\right\} 2\frac{H}{\pi} =$$

$$= P_{\max} 2\pi \left\{ \left[R\left(\frac{2R}{\pi}\right)1 - 0\right] - \left[0 - \left(-\left(\frac{2R}{\pi}\right)^2 1\right)\right]\right\} 2\frac{H}{\pi} = P_{\max} 2\pi \left\{\frac{2R^2}{\pi} + \frac{4R^2}{\pi^2}\right\} 2\frac{H}{\pi} = P_{\max} 2\pi R^2 \left\{2 - \frac{4}{\pi}\right\} 2\frac{H}{\pi^2} =$$

$$= P_{\max} 2\left(\pi R^2 H\right)\left\{2 - \frac{4}{\pi}\right\} 2\frac{H}{\pi^2} = \left(\pi R^2 H\right)2\left\{2 - \frac{4}{\pi}\right\}\frac{2}{\pi^2} P_{\max} = \left(V\right)1.453\frac{2}{\pi^2} P_{\max} = 0.2947 P_{\max} V$$

$$\overline{P} = \frac{P_{Tot}}{V} = \frac{0.2947 P_{\max} V}{V} = 0.2947 P_{\max} \tag{8}$$

$$\frac{P_{\max}}{\overline{P}} = 3.393 = 1.8428^2 \tag{9}$$

That this is the punctual peak factor of the core for linear sinusoidal distribution (axial and radial).

$$P_{Tot} = \overline{P}V = \frac{P_{\max}}{3.393} V \tag{10}$$

A.4.3.2. Evaluation of the cell probability

To make the calculation of the power gamma decay heat absorbed in the fuel, it is needed to divide the core in cells. The dimension of a cell should allow the movement of the gamma from one cell to the other. For our purpose the dimension of the cell is a annular circular of 1 cm and thickness 2,5 cm.

The MNCP 5 requires the definition of the probability for each cell $P_c(r,z)$. It can be evaluated as fraction of the power of the cell and total power:

$$P_c = \frac{P_{wc}}{P_{wTot}} \tag{11}$$

Where, from (1),

$$P_{wc} = P_{\max} V_c \cos\left(\frac{r}{R}\frac{\pi}{2}\right)\cos\left(\frac{2z}{H}\frac{\pi}{2}\right) \tag{12}$$

and from (5) and (6),

$$P_{wTot} = \frac{P_{\max}}{3,393} V_{Tot} \tag{13}$$

The probability is defined as following:

$$P_c = \frac{P_{\max} \cos\left(\frac{r}{R}\frac{\pi}{2}\right)\cos\left(\frac{z}{\frac{H}{2}}\frac{\pi}{2}\right)(2\pi r D_r D_z)}{\frac{P_{\max}}{f_p} V_{Tot}} \qquad 14)$$

radial sinusoidal distribution

$$P_c(r) = \frac{P_{\max}\left(2\pi r D_r H\right)}{\left(\frac{P_{\max}}{f_p} V_{Tot}\right)} \cos\left(\frac{r}{R}\frac{\pi}{2}\right) \qquad 15)$$

axial sinusoidal distribution

$$P_c(z) = \frac{D_z}{H} \cos\left(\frac{z}{\frac{H}{2}}\frac{\pi}{2}\right) \qquad 16)$$

radial Gaussian distribution

$$P_c(r) = \frac{P_{\max}\left(2\pi r D_r H\right)}{\left(\frac{P_{\max}}{f_p} V_{Tot}\right)} e^{-\left[\left(\frac{r}{R}\right)^2 \frac{1}{2\sigma^2}\right]} \qquad 17)$$

radial power production

$$P_{wc\,\mathrm{Pr}od}(r) = \frac{P_c(r)}{\left(2\pi r D_r H\right)} \left(\frac{P_{\max}}{f_p} V_{Tot}\right) [\mathrm{Wcm^{-3}}] \qquad 18)$$

radial power absorbed

$$P_{wcAbs}(r) = \delta\left(P_{MCNP5}\right)_c (r) \overline{\delta}_{uo_2} n_\gamma [\mathrm{Wcm^{-3}}] \qquad 19)$$

axial power production

$$P_{wc\,\mathrm{Pr}od}(z) = \frac{P_c(z)}{\left(\frac{D_z}{H}\right)} P_{\max} \; [\mathrm{Wcm^{-3}}] \qquad 20)$$

axial power absorbed

$$P_{wcAbs}(z) = \delta\left(P_{MCNP5}\right)_c (z) \overline{\delta}_{uo_2} n_\gamma [\mathrm{Wcm^{-3}}] \qquad 21)$$

A.5. APPENDIX V: PEER REVIEW OF THE ICONE 16 PAPER

A paper summarizing the results achieved on the decay heat and gamma distribution of decay heat was sent and accepted to ICONE 16, May 2008, Orlando.
It also contains a draft of the idea developed in the present work.
It was send to some colleagues for a review:

- Michael Bykov and Alexander Moskalev (Gidropress, Russia)
- Juan Carlos Ferreri (Regulatory body, Argentina)
- Horst Glaeser (GRS, Germany)
- Enno Hicken (*Forschungszentrum Jülich, Germany)*
- Antonio Madonna (ITER Consult, Italia)
- Borut Mavko (Head, Reactor Engineering Department, Jozef Stefan Institute, Slovenia)
- Rizwan Uddin (University of Illinois Urbana – Champaign, USA)
- Evgenius Uspuras (Head of the institute) and Algirdas Kaliatka (Lithuanian Energy Institute, Lithuania)

Here after the abstract is provided and the comments received.

A.5.1. Abstract

New capabilities of computational tools, recent demands from the industry, including nuclear power plant power up-rating, and advancement in the understanding of safety concepts may require innovative solutions and approaches for safety analyses. . In the framework of such activities, the purpose of the present paper is to streamline the results related to Decay Heat and the gamma – decay heat distribution, two of the relevant issues.
Concerning the DH to be introduced in the calculations, a study is presented on the opportunities offered by present knowledge and RELAP 5 options; suggestions are also made on the assumptions to be used in the RELAP5 analysis .
The second issue, that is dealt with in this paper, is the decay heat gamma distribution. The short study presents the effect of considering the real absorbed γ distribution in the core. The importance of taking into account this effect is apparent when the influence on important calculation results is considered: e.g. Peak Clad Temperature could be affected by more than 100 K.

A.5.2. Michael Bykov and Alexander Moskalev (Gidropress, Russia)

The report was appreciated. A request was done to receive all the manuscript of the thesis in order to provide an official analysis by Gidropress.

A.5.3. Juan Carlos Ferreri (Regulatory body, Argentina)

The work was appreciated and a request to comment on the final work of the thesis was done.

A.5.4. Horst Glaeser (GRS, Germany)

Comments were provided on the subject of the paper. (Some of them are also included in the present work)

A.5.5. Enno Hicken

Request to comment on the final work of the thesis was done

A.5.6. Antonio Madonna (ITER Consult, Italy)

'The study performed addresses systematic and comprehensive considerations on the use of BE analysis for licensing process.
It is clearly identified the need to have approved methods and also regulatory guidance in order to get benefit from BE analysis in licensing process.
The regulatory needs in terms of methodology and acceptance criteria of safety assessment submitted for licensing purpose versus the current state of the art in quantifying code uncertainties and use of margins in safety assessment are well presented and considered.'

A.5.7. Borut Mavko, Andrej Prosek (Jozef Stefan Institute, Slovenia)

The work is very appreciate.

Comments on paper DECAY HEAT ISSUES FOR BEST ESTIMATE MODELS

Comment 1. Regarding review of best estimate analysis in licensing extensive review on applications of best estimate plus uncertainty methods is available in the enclosed Ref. 1.

Comment 2. In Chapter 3 decay heat power and gamma decay heat distribution are discussed. In Chapter 3.1 the RELAP5 was used for station blackout analysis and SBLOCA. Both cases are different from the LBLOCA for which the review was done in Chapter 2. For LBLOCA the decay heat plays its role just in the initial few hundreds of seconds. The RELAP5 analyses showed that the maximum differences were about 10% for the decay heat. This is in accordance with the uncertainty ranges for decay heat in the literature for the LBLOCA peak cladding temperature uncertainty evaluation. For example, in the Ref. 2 the uncertainty was ± 6.6% (two sigma, normal distribution). It is expected that the later ANS94-4 standard would provide more accurate results. It would be very valuable information on uncertainty of the ANS94-4 decay heat to better image how much the results would be improved with more accurate standard.

Comment 3. Chapter 3.1: Based on MCNP5 calculations for LOFT it is stated that 20% power decrease main entail a calculated PCT reduction of the order of 100 – 150 K. It may be such result based on sensitivity study. However, the AREVA LBLOCA application (see Table 4 of Ref. 3) showed that the sensitivity of 54 ˚F contributes only 4 ˚F to the final PCT (when several uncertainties are considered uncertainty both increase and decrease PCT). From this perspective care must be taken before it is stated that more accurate decay heat model will significantly reduce the calculated PCT. Also, in the AREVA study the decay heat is considered medium sensitive parameter to some other.

Comment 4. Reference 30 in your paper was not available on Hindawi Publishing Corporation home page (http://www.hindawi.com/journals/stni/), therefore comment 3 was based only on the summary information provided in the paper. (Reply: the paper is under publication process).

Comment 5. Conclusions: It is recommended ANS 1994-4 standard. However, this standard was already superseded with ANSI/ANS-5.1-2005 decay heat power. What is the reason to recommend older standard?

Comment 6. Conclusions: it is stated that real power history should before shutdown should be used. However, AREVA in his licensing calculation stated [Ref. 4]:

"The decay heating described by the standard can be used for many types of calculations including LOCA analysis. However, considerations for LOCA are somewhat different from other applications. LOCA is a hypothetical event which must be analyzed prior to reactor operation. Thus, the operating history and the concentration of fissionable isotopes will not be known prior to a LOCA. Fortunately, simplifying assumptions can be made which allow calculation of a realistic but slightly conservative decay heat curve as a function of time using the 1979 standard. The decay heat calculated with these assumptions bounds the more detailed decay heat curves that would result if the conditions at the initiation of LOCA were known. The assumptions are:

• infinite operating time at full power.
• All fissions assumed from ^{235}U
• 200 MeV / fission (conservatively low)
• One standard deviation total decay heat"

This example for LBLOCA is against what is recommended in the conclusions.

[Ref. 1] PROŠEK, Andrej, MAVKO, Borut. The state-of-the-art theory and applications of best-estimate plus uncertainty methods. Nucl. technol., 2007, vol. 158, no. 1, pp. 69-79.

[Ref. 2] C. H. BAN, S. Y. LEE, C. K. SUNG, "Development and Application of KEPRI Realistic Evaluation Methodology (KREM) for LB-LOCA", Best Estimate-2004: International Meeting on Updates in Best Estimate Methods in Nuclear Installations Safety Analysis, Washington, DC, November 14-18, 2004.

[Ref. 3] R. Martin, B. Dunn, Application and Licensing Requirements of the Framatome ANP RLBLOCA Methodology, BE-2004: International Meeting on Updates in Best Estimate Methods in Nuclear Installations Safety Analysis, Washington, DC, Nov. 14-18, 2004.

[Ref. 4] Realistic Large Break LOCA Methodology for Pressurized Water Reactors, August 2001, EMF-2103(NP), Revision 0.

A.5.8. Rizwan Uddin (University of Illinois Urbana – Champaign, USA)

The paper does need "some" editing for English. As such, it looks fine.
The subject looks exactly like the kind of thing I was looking for a PhD topic.
I think it will make a good thesis. May be you should add a practical application to demonstrate the licensing procedure you develop.
I would be happy to comment on your PhD proposal.

A.5.9. Egenijus Ušpuras and Algirdas Kaliatka (Lithuanian Energy Institute, Lithuania)

PAPER “DECAY HEAT ISSUES FOR BEST ESTIMATE MODELS”
AUTHORS Calogero Sollima and Gianni Petrangeli

Place a cross in the boxes which, in your opinion, best describe the following features of the manuscript:

Poor □ Marginal □ Acceptable ■ Good □

Please rate the paper on a scale of 1 (worst) to 5 (best) for the following characteristics:

Originality of Work:	1	2	**3**	4	5
Engineering Relevance:..	1	2	3	4	**5**
Scientific Relevance:	1	2	**3**	4	5
Completeness of the Reported Work:	1	2	**3**	4	5
Acknowledgment of the Work of Others by References:...	1	2	3	**4**	5
Organization of the Work:...	1	2	3	4	**5**
Clarity in Writing, Tables, Graphs, and Illustrations:	1	2	3	4	**5**

Is the technical treatment plausible and free of technical errors?	Yes ■	No □
Have you checked the equations?	Yes ■	No □
Are you aware of prior publication or presentation of this work?	Yes ■	No □
Is the work free of commercialism?	Yes ■	No □
Is the paper too long (more than 12 pages)?	Yes □	No ■

The work part of whish is presented is very actual. Also the practical importance of this work is unquestionable. I am thinking about the novelty of this work. The presented methods for decay heat calculations are well-known. Maybe the novelty is comparison of different methods and presented recommendations, related to applicability of BE methodology for accident analysis.

Some general comments for the manuscript:

The INTRODUCTION and second chapter (BEST ESTIMATE ANALYSIS IN LICENSING PROCESS) covers very vide problem. The decay heat issues are only one small part of problems mentioned in the first two chapters. In my opinion the chapter 2 is to long for such manuscript. Of course such chapter is very suitable for your PhD thesis.
The real recommendations related to the applicability of BE methodology in licensing process (calculation of reactor decay heat during accident analysis) are presented in chapter 4 CONCLUSIONS.

A.6. APPENDIX VI: LIST OF THE PUBLICATIONS

Related to the subject of the thesis

C. Sollima, G. Petrangeli, J. Misak, F. D'Auria "Framework and strategies for the introduction of best estimate models into the licensing process", ICONE17, July 12-16, 2009, Brussels, Belgium, EU

C. Sollima, G. Petrangeli, "Decay Heat issues for Best Estimate Models", ICONE16, May 11-15, 2008, Orlando, Florida, USA

G. Petrangeli, C. Sollima, "Gamma Decay Heat Distribution in Core: a Known Issue Revisited", Hindawi Publishing Corporation Science and Technology of Nuclear Installations, Vol. 2008

Further pubblications

2008

C. Sollima, " Paper Collection on RBMK Safety Technology", GRNSPG – DIMNP – Univerità di Pisa, June 2008,

F. D'Auria, S. Soloviev, D. Mazzini, and C. Sollima, "Deterministic Safety Technology for RBMK Reactors", Science and Technology of Nuclear Installations Volume 2008 (2008), Article ID 781824, 7 pagesdoi:10.1155/2008/781824Project Report

2007

November 2007:

L. Caillat, R. Garaguso, R: Garaguso, Y. Papazoglou, G. Pilone, C. Sollima, "CDR of the Quality Assurance and Risk assessment for the KM3NET, under sea water neutrino detector", VI Framework Program, INFN, Italy

2006

F. D'Auria, D. Mazzini, S. Soloviev, C. Sollima, "Deterministic Safety Technology For RBMK Reactors", The 15th Pacific Basin Nuclear Conference • Sydney • 15 – 20 October 2006

M. Mazzini, C. Sollima, "Execution of the Experiments", TACIS Project 30303, Università di Pisa

TACIS Project 30303 Part A: Development of accident management procedures on the test facility 'PSB-VVER 1000' at Electrogorsk, Università di Pisa

M. Mazzini, C. Sollima, "Computational Analysis of the Experimental Results", TACIS Project 30303 Part A: Development of accident management procedures on the test facility 'PSB-VVER 1000' at Electrogorsk, Università di Pisa

C. Sollima, "Final Technical Report", TACIS Project 30303 "Software development for accident analysis of VVER and RBMK reactors in Russia", Università di Pisa

Cherubini M., D’Auria F., Malofeev V., Moretti F., Moskalev A., Novoselsky O., Parisi C. , Pierro F., Radkevitch V., Sollima C., Soloviev S., Ušpuras E., “Final report Part B: Development of accident management procedures on the test facility 'PSB-VVER 1000' at Electrogorsk”, TACIS Project 30303 “Software development for accident analysis of VVER and RBMK reactors in Russia”, Università di Pisa

Contributor per: F. D’Auria, O. Melikhov, V. Melikhov, I. Elkin, A. Suslov, M. Bykov, A. Del Nevo, D. Araneo, N. Muellner, M. Cherubini, W. Giannotti “Final report Part A: Development of accident management procedures on the test facility 'PSB-VVER 1000' at Electrogorsk”, TACIS Project 30303 “Software development for accident analysis of VVER and RBMK reactors in Russia”, Università di Pisa

2005

K. Slavcheva , M. Mori , N. d’Amico C. Sollima, “Safety Culture and Organizational Issues During Transition from Operation to Decommissioning of NPPS’’, International Conference Nuclear Energy for New Europe 2005, Bled - Slovenia - September 5-8, 2005

C. Sollima , D. Mazzini, L. Giannini, F. Raffaelli, K. Slavcheva “Quality Assurance and Risk Assessment in the Framework of the Research Projects”, VLVnT Workshop, Catania, 09 Novembre 2005

Contributor per: F. D’Auria, O. Novoselsky, A. Moskalev, V. Radkevitch, V. Malofeev, C. Parisi, M. Cherubini, F. Pierro, F. Moretti, “Final report Part B: “Development of a code system for severe accident analysis in RBMK reactors”, TACIS Project 30303“Software development for accident analysis of VVER and RBMK reactors in Russia”, Università di Pisa

2004

European Commission Inception Report of Tacis Project 30303, Pisa, March 2004.

2003

Rivista Italiana della Saldatura Palette di Turbine a Vapore Rivestite mediante "Laser Welding": Problematiche di Tensocorrosione, , Genova, 2003 (Master thesis)