CHALMERS

Definition of a dynamic source term module within RASTEP

Linking with a fast running deterministic code

Master's Thesis in the Master's Programme of Nuclear Engineering

FRANCESCO DI DEDDA

Department of Applied Physics
Division of Nuclear Engineering
CHALMERS UNIVERSITY OF TECHNOLOGY
Göteborg, Sweden 2013
CTH-NT-270

CTH-NT-270

Definition of a dynamic source term module within RASTEP

Linking with a fast running deterministic code

Master's Thesis
Master's Programme in Nuclear Engineering
CHALMERS UNIVERSITY OF TECHNOLOGY

FRANCESCO DI DEDDA

Department Applied Physics
Division of Nuclear Engineering
CHALMERS UNIVERSITY OF TECHNOLOGY
Göteborg, Sweden 2013

Definition of a dynamic Source Term module within RASTEP
Linking with a fast running deterministic code
FRANCESCO DI DEDDA

Master's Thesis
CTH-NT-270
ISSN 1653-4662
Department of Applied Physics
Division of Nuclear Engineering
Chalmers University of Technology
SE-412 96 Göteborg
Sweden
Telephone + 46 (0)31-772 1000

Chalmers Reproservice
Göteborg, Sweden 2013

Definition of a dynamic source term module within RASTEP
Linking with a fast running deterministic code
Master's Thesis in the Nuclear Engineering Master's programme
FRANCESCO DI DEDDA
Department of Applied Physics
Division of Nuclear Engineering
Chalmers University of Technology

ABSTRACT

RASTEP (RApid Source TErm Prediction) is a computerized tool developed by Scandpower AB for use in the fast online diagnosis of accidents at nuclear power plants. The tool is based on a Bayesian Belief Network (BBN) that is used to determine the most likely plant state, which is associated with pre-calculated source terms. In its current design, the source term predictions are not flexible enough. A previous study evaluated different methods for enhancing the source term module of RASTEP. This thesis work follows that approach and explores the integration of a fast running deterministic code within RASTEP in order to make the predictions more realistic. The MARS software, developed by Fauske & Associates, has been chosen as best candidate for this purpose. Literature studies, along with interviews with experts, and conceptual reasoning have been carried out in order to identify the best linking process. Two modes for coupling the BBN with the deterministic code MARS (integrated and iterative) have been proposed and evaluated. In both approaches, the strength of the probabilistic predictions is kept and the capabilities of MARS are proven also in a linked configuration. It is concluded from the study that this method can be used to enhance RASTEP and it is feasible for implementation in the short term.

Keywords: RASTEP; Source Term Predictions; Bayesian Belief Networks; Real-time analyses.

Contents

List of figures

List of tables

Acknowledgments

This report presents the Thesis work performed for Scandpower AB during the period of September 2012 - February 2013 and represents the graduation requirement for the Chalmers University of Technology Nuclear Engineering Master Program.

I would like to express my deep gratitude to my supervisor at Scandpower Vidar Hedtjärn Swaling, for his patient guidance and useful critiques of this work. I would also like to thank my examiner and supervisor at Chalmers Dr. Christophe Demazière and Dr. Paolo Vinai for their advice and assistance in keeping my progress on schedule. I wish to thank James C. Raines of Fauske & Associates and Jose Ramon Alonso Escos of CSN for their invaluable support on the MARS code. My special thanks are extended to the Scandpower staff for their assistance, for offering me a great opportunity of being in contact with experts in the nuclear field, and for making my time in the office very enjoyable.

Finally, I wish to thank my girlfriend Rebecca, my family and my friends for their support and encouragement throughout my study.

Francesco Di Dedda,
Stockholm Sweden, February 2013

Executive summary

This Master Thesis work represents a step forward in the process of making the source term module of RASTEP more dynamic and giving more realistic predictions. A review of state-of-the-art methods and approaches, and a feasibility study on linking RASTEP to a fast running deterministic code have been done during a previous project. The MARS code has been chosen as a best candidate for the linking process. The aim of this work is to define the conceptual and, to some extent, the technical integration of the MARS code into RASTEP. This means connecting two different worlds – a probabilistic one with a deterministic one.

The experience from the use of MARS at CSN (Spain) together with the information gathered from communications with experts at Fauske & Associates LLC has proven the consistency of the probabilistic module of RASTEP. During the current work, two new approaches of connecting the Bayesian Belief Network with MARS have been proposed: integrated and iterative mode. The strength of the probabilistic predictions of the BBN will still be valid.

Although this project confirms the possibility to integrate on-the-fly deterministic calculations within RASTEP, it also leads to new aspects to be explored. The current definition of the release categories is tightly connected to the PSA framework, predominant in the BBN model of RASTEP. In order to match the more dynamic predicted scenarios, a reclassification of the release categories and consequent modification of the BBN model could be needed.

The presence of MAAP models in all Swedish NPPs suggests a realistic opportunity to adopt MARS as part of the emergency preparedness system. More accurate and reliable predictions during accidental scenarios at nuclear power plants could result in a better off-site emergency planning. The same approach adopted for MARS could be extended to other codes (e.g., MELCOR) if the models and accuracy of the source term information will be validated.

Acronyms and abbreviations

ADAM	Accident Diagnosis Analysis And Management
AI	Artificial Intelligence
APRI	Accident Phenomena of Risk Importance (Swedish Research Programme)
BBN	Bayesian Belief Network
BWR	Boiling Water Reactor
CANDU	CANada Deuterium Uranium
DBN	Dynamic Bayesian Network
DDET	Discrete Dynamic Event Tree
DPSA	Dynamic Probabilistic Safety Assessment
DSA	Deterministic Safety Assessment
ENSI	Swiss Federal Nuclear Safety Inspectorate
EOP	Emergency Operating Procedure
ERI	Energy Resources International Inc.
ET	Event Tree
FAI	Fauske and Associates
FMEA	Failure mode and effect analysis
FT	Fault Tree
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit GmbH
LOCA	Loss Of Coolant Accident
MAAP	Modular Accident Analysis Program
MARS	Modular Accident Response System
MC	Monte Carlo
MCS	Minimal Cut Set
MCDET	Monte Carlo Dynamic Event Tree
NPP	Nuclear Power Plant
NRC	U.S. Nuclear Regulatory Commission
PSA	Probabilistic Safety Assessment
RASTEP	Rapid Source Term Prediction
RCPB	Reactor Coolant Pressure Boundary
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
SA	Severe Accident
SAMG	Severe Accident Management Guideline
SPRINT	System For The Probabilistic Inference of Nuclear Power Plant Transients
SSM	Strålsäkerhetsmyndigheten (Swedish Radiation Safety Authority)
STERPS	Source Term Prediction Based On Plant Status
VVER	*Water-Water Power Reactor (Vodo-Vodyanoi Energetichesky)*

1 Introduction

The desire for development of tools for use in the fast, online diagnosis of severe accident events and the subsequent radiological source term prediction at nuclear power plants has continuously increased during the last decades. Such analytical tools would enhance the efficiency in preparing rapid accident response. Online implementations would further assist in quickly predicting likely off-site consequences in order to accomplish an appropriate emergency response.

Large uncertainties are inevitable when dealing with severe accidents in nuclear power plants. Severe accident models are typically based on deterministic and probabilistic approaches, both necessary because complementary. Probabilistic safety assessment (PSA) is capable to provide insights into the safety status of the plant, something that cannot be gained solely by a deterministic approach.

The common practice is to distinguish three levels of a PSA. During level 1 PSA the frequency of core damage is evaluated. A level 2 PSA, using the results from level 1, investigates the probability of containment failure and the radionuclide releases into the environment after that failure. Level 3 PSA uses the results from both the previous analyses and provides knowledge on concentration of relevant radionuclides, doses, associated health effects together with economic impacts.

PSA studies are used to identify accidental sequences and, in the particular case of level 2 PSA, physical phenomena and related plant response must be analysed along the process by the use of deterministic models (DSA). The connection between accident scenarios identification and determination of the subsequent accident progression and radioactive source term releases is the premise of level 2 PSA. The amalgamation of numerous and complicated physical effects has to be taken into account for this kind of study [1].

The most common used method of performing PSA on any level is the event tree and fault tree analysis (ET/FT). Some critical sequences are initially identified and deterministic simulations are used to reveal dynamic behaviours related to them. In fact, these sequences are studied with DSA tools determining the plant response; usually severe accident integral codes (MAAP, MELCOR) are used for this purpose. Later on, given the necessary boundary conditions (e.g., time windows, redundancy of certain systems, etc.), these sequences are analysed probabilistically using EF/FT, where cut sets may reveal new sequences. The process is thus iterative, leading to a point where all the relevant sequences are identified. Success criteria are determined, and the size of radioactive release is also computed as output of accidental sequences.

Aiming at improving off-site emergency management, Scandpower AB is currently developing a computerized tool called RASTEP (RApid Source TErm Prediction). This tool is meant to be used during an actual accidental event, where predictions of the source term are made in real-time. The Swedish Radiation Safety Authority (SSM) and the Nordic Nuclear Safety Research (NKS) are the main contributors in this process. RASTEP consists of two modules, each with a fundamental purpose:

- One module via a *Bayesian Belief Network* (BBN) that predicts the states in the nuclear power plant and the probability of various accidental sequences. The more information provided to the network, the better the prediction will be.

- One module to define and characterize the source terms (amount of emissions/release, chemical composition, height and timing of the predicted radioactive release).

An interface called SPRINT (System for the Probabilistic Inference of Nuclear Power Plant Transients) is currently used to relate predefined source terms to the corresponding end states in the BBN. SPRINT was developed within the European project STERPS (Source Term Predictions Based on Plant Status) and will eventually be replaced by RASTEP. The SPRINT interface poses questions to the user giving the opportunity to introduce findings into the network.

The source terms, containing the release information, are generated by the deterministic code MAAP (Modular Accident Analysis Program), and are currently the same as those used in the PSA studies. They are stored in a repository file for use within the SPRINT software. The lack in determining source terms in real-time makes this approach non as realistic as desired and so the reason for the need of improvement.

1.1 Purpose

This report aims to show the current state of development of RASTEP and investigate new actions for improvement of the source term predictions.

A previous thesis diploma work identified one approach for enhancing the source term [2], and the main object for this thesis work is to follow that approach and explore the integration of a fast running deterministic source term module within RASTEP, making the tool more dynamic and the predictions more realistic.

The most feasible alternatives for the source term module will be analysed in this report and conceptual and technical issues will be discussed.

The purpose of this thesis is to:

- Address the challenge of combining DSA and PSA with particular focus on the use of these approaches in Severe Accident Management and give an overview on the rule of the IDPSA (Integrated Deterministic-Probabilistic Safety Assessment) for the purpose of developing tools for the use in emergency preparedness.

- Address the issue of implementing a fast running deterministic code with RASTEP.

1.2 Methodology

The research process within this thesis work can be summarised as follows:

- Literature study. Understanding of the actual use and capabilities of RASTEP. Deep understanding of PSA and DSA studies and their interactions with RASTEP.

- Closer look at the severe accident codes. Focus on feasible alternatives for the upcoming future. Making use of personal communications with experienced people in order to better understand the advantages and disadvantages of these methods.

- Communications with people from CSN (Consejo de Seguridad Nuclear), FAI (Fauske & Associates LLC) and OKG (Oskarshamn Nuclear Power Plant) to retrieve information and material on specific software tools.

- Research on how RASTEP can be connected to a deterministic code. After reasoning on a conceptual level, technical issues are addressed. People involved in software development became helpful during the process.

1.3 Limitations

This thesis aims at identifying possible implementations of the RASTEP tool with selected deterministic codes. Due to limited access to specifications of such codes, the real technical aspects may result difficult to clarify and discuss in an extensive way.

1.4 Disposition of the report

Chapter 2, *Background*, gives a description of the field of nuclear safety analyses (DSA, PSA) and severe accident studies. SAMGs, EOPs at nuclear power plants are also described. Chapter 3, *RASTEP*, provides an overview of the RASTEP tool, its usability, the mapping of source terms, the Bayesian network and the SPRINT user interface. The *Problem Statement* is found in chapter 4. Chapter 5, *Tools for source term assessment*, offers an overview of the current design of severe accident codes such as MAAP and MELCOR. In chapter 6, *On-line accident diagnostic tools* are described, and, in particular, those codes which can be implemented with RASTEP are deeply analysed. In chapter 7, *Analysis*, the description of the challenges and problems that affect the integration of DSA and PSA is offered. Suggestions for the connection of a fast running deterministic code with RASTEP are also described. Chapter 8, *Discussion and continued research*, offers a critique on the current development of RASTEP, unanswered questions and a review of possible improvements. The *Conclusions* are provided in chapter 9.

2 Background

There is a variety of potential severe accident scenarios and sequences for light water reactors. In general, accidents start from different initiating events that may lead directly or through additional failures to severe core degradation. The range of the potential plant states include operation at nominal power, plant heat-up, plant cool-down, and plant shutdown conditions.

Once an accident starts, loss of coolant inventory is followed by oxidation of the Zircaloy cladding, and eventually core damage, reactor pressure vessel failure, and a multitude of physical phenomena potentially challenging the containment integrity. The further the accident progresses into the severe accident regime, the more difficult it becomes to manage the accident following the Emergency Operating Procedures (EOPs). Therefore, many utilities tend to develop or have already developed Severe Accident Management Guidelines (SAMG) with a structure that is more appropriate for severe accident situations.

This section offers an overview of a variety of safety analyses currently adopted by the nuclear industry.

2.1 Safety analyses

Safety analysis in the nuclear field is an essential element of a safety assessment at nuclear power plants. It is an analytical study adopted to demonstrate how safety requirements are met for a wide range of operating conditions and initiating events. Safety analysis comprehends deterministic (DSA) and probabilistic analyses (PSA). These studies are in support of the design, commissioning, operation and decommissioning of a nuclear power plant. The requirements for the safety assessment are provided in regulatory standards [3].

2.1.1 Deterministic safety analyses

Deterministic safety analyses for nuclear power plants predict the response to a set of postulated initiating events. Specific rules together with success criteria are always applied. Typically, these criteria focus on issues such as neutronic, thermo-hydraulic, thermo-mechanical and structural aspects, which are analysed with various computational tools. Radiological studies are never neglected and are included in the analysis. The computations are carried out for selected operating modes and scenarios. These events include also postulated accidents, transients, beyond design basis accidents. Severe accidents with core degradation are of major interest to effectively prove the safety of the emergency systems. The results of the simulations and computations contain spatial and time dependencies of the physical variables of interest (e.g., thermal power of the reactor; neutron flux; pressure, temperature). Radiation doses to workers or the public are of interest when assessing the radiological consequences [3].

2.1.2 Probabilistic safety analyses

Probabilistic safety assessment (PSA) has been performed for the majority of the nuclear power plants (NPP) worldwide. PSA has primarily been carried out by regulatory bodies to acquire risk insights or compliance with regulatory requests, to identify and understand better key plant vulnerabilities, and to analyse the impact of proposed design or operational changes. Nowadays, PSA is also used for support in designing new nuclear power plants.

Due to the considerable resources invested in the development of PSA, there is a desire of both licensees and regulators to use the information derived from it to enhance plant safety while operating the nuclear plants in the most efficient way. Plant management, in fact, is one of the main objectives in using this kind of tools.

The PSA of a nuclear power plant analyses the risk associated with operating the plant. Usually, the risk is expressed in terms of various indicators related to levels of damage to the plant or the environment. The analysis is performed by the use of a logical and systematic approach based on realistic assessments.

When performing deterministic analyses, the adoption of conservative assumptions related to plant and system performance is an accepted approach to address uncertainty. In DSA combinations of limited number of faults and a conservative approach to the analysis of each fault are usually considered. This approach can produce inappropriate insights and might not always be the appropriate way for reducing risks at NPPs. On the other end, PSA considers the system inter-dependencies, offering a better understanding of the plant as a whole. Using realistic criteria for the plant performance, PSA is also capable to offer more valuable risk informed decisions. Compared to the deterministic methods, probabilistic safety assessment has the potential to determine the risks of operating the plant over a wider range of conditions. Therefore, PSA results to be a very useful tool for safety management and a perfect complement to the deterministic methods [4].

The PSA modelling is based on the use of fault trees and event trees. Event trees are logic models for identifying the accident sequences developing within the plant, involving the protection and safety systems, which can generate from a single initiating event. After the initiating events are defined, all the possible sequences related to a particular initiating event must be described, and each sequence models a specific scenario in the event tree (success/failure). The sequences are usually identified by the use of functional events (Figure 1). Functional events are those events that model the safety barriers of the system. As mentioned above, they correspond to safety functions and systems that need to be identified for each initiating event. As shown in Figure 1, the end states of each sequence represent a consequence (OK /Not OK).

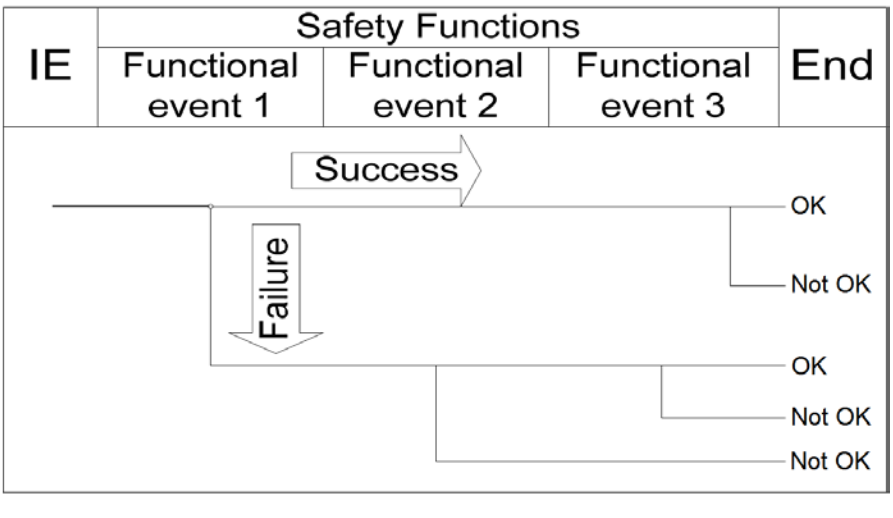

Figure 1 Example of an event tree

The sequence analysis performed using the event trees provides results used as input in the system analysis. The system analysis is a more expanded way of identifying the safety systems and is performed by using fault trees usually combined with the FMEA methodology.

Fault tree analysis is a technique that allows developing the casual relations leading to a given undesired event, focusing on single failure modes. This method provides qualitative and quantitative information on how and with what probability a particular event occurs and what consequences it leads to.

A fault tree is a graphical representation of relations between a system failure event, called top event, and other events that may cause the top event to occur. The first step in building a fault tree is identifying the system failure event of interest (top event), usually used as input to a function event in event trees, and all the contributing events that can cause it to occur (Figure 2).

As shown in Figure 2, the top event is connected to the other events by logical operators (AND-gate, OR-gate), and this procedure continues until all branches have been terminated in basic events or undeveloped events. Basic events are those events that do not require further breakdown into simpler failures. They already contain information about failure probability or frequency and are represented by a circle symbol.

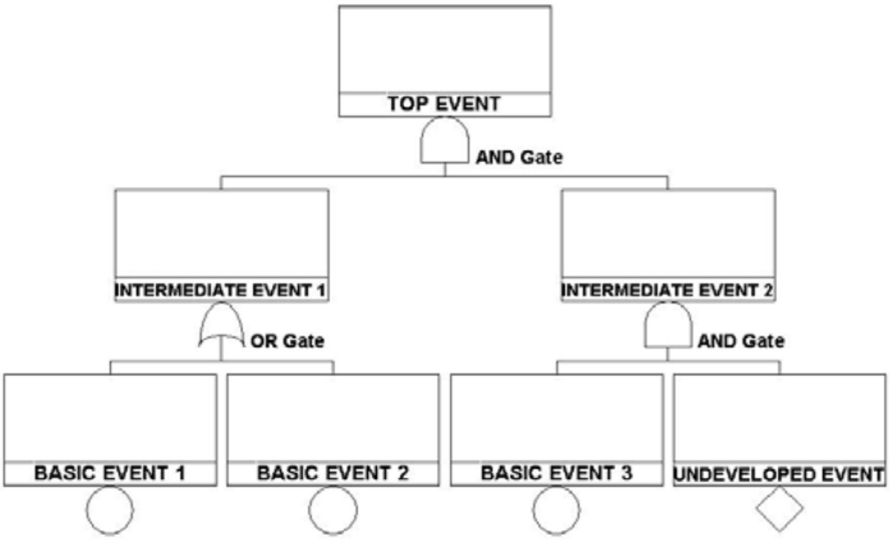

Figure 2 Example of a fault tree

A fault tree can be described by a set of Boolean algebraic equations, one for each gate of the tree, and utilizing the rules of Boolean algebra it is possible to solve these equations so that the top event can be expressed in terms of sets of primary events [5]. Using further this kind of expansion, the top event can be quantified in terms of Minimal Cut Sets (MCS). Cut sets are the unique combinations of component failures that can cause system failure (Figure 3). Specifically, a cut set is said to be a minimal cut set if, when any basic event is removed from the set, the remaining events collectively are no longer a cut set [6].

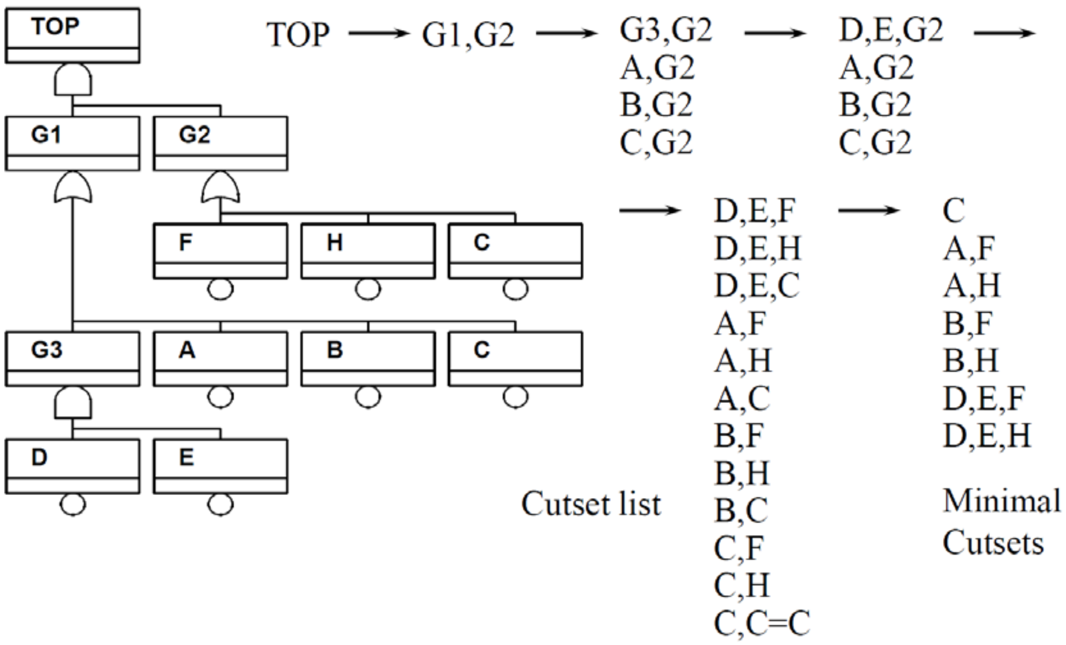

Figure 3 Example of minimal cut set generation

In practice, three levels of PSA are recognized:

- Level 1 PSA analyses the design and operation of the plant and the sequences of events that can lead to core damage. Estimations of the core damage frequency are also computed. The strengths and weaknesses of the safety systems and the procedures in place to prevent core damage are provided.

- Level 2 PSA analyses the chronological progression of core damage. The sequences identified in Level 1 PSA are used for this purpose and, a quantitative assessment of phenomena related to severe damage to reactor fuel is carried out. Level 2 PSA identifies the release paths of radioactive material from the fuel which can result in releases to the environment. Estimations of the frequency, magnitude and other characteristics of the releases are also computed. Furthermore, level 2 PSA provides insights into the importance of accident prevention along with measures to mitigate the release of radioactive material to the environment. The study of physical barriers (e.g., containment building) is included in this analysis.

- Level 3 PSA estimates the impact of the releases on public health and other societal consequences (e.g., contamination of land and food).

PSA analyses are sequential so that the results of each assessment typically serve as the basis for the PSA at the next level [7].

2.2 Bayesian Belief Networks

The Bayesian approach is applicable in all those situations where there is lack of exact knowledge of factors influencing the probability of a certain event. Bayes' theorem is the basis to calculate the probability of such events. The BBN approach is such that prior beliefs are stated in the beginning, and later on, when information on the progression of an event is available, these beliefs are modified and updated.

A Bayesian Belief Network (BBN) is a powerful tool consisting of a set of nodes and edges; the first representing variables with a finite number of states, the latter representing the probabilistic causal dependence among the variables. The nodes with edges directed into them are called *child* nodes and the nodes from which the edges depart are called *parent* nodes. There are other nodes without arches directed into them and these are called root nodes.

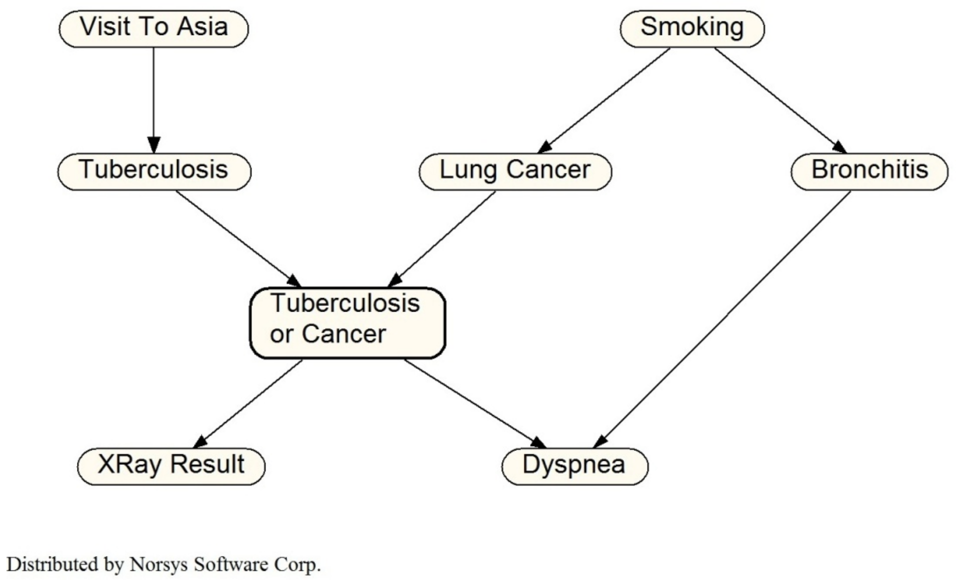

Figure 4 Example of Bayesian Belief Network

Figure 4 shows a simple example of a Bayesian Belief Network used for the diagnosis of chest diseases. In a BBN, the graphical structure represents the causal dependencies between nodes and gives the qualitative part of causal reasoning. The relations between variables and the corresponding states give the quantitative part, consisting of a Conditional Probability Tables (CPTs) related to each node with parents. Therefore, a Bayesian network is a representation of the joint distribution over all the variables represented in the graphical structure and the marginal and the conditional probabilities can be computed for each node of the network [8].

Bayesian Belief Networks became popular in reliability engineering mostly because of their wide area of applicability. In principle, BBN can be applied to all problems where there is the need to model systems' states describing, at the same time, the inherent probabilities. The fact that BBN is a graphical method is also another reason of its popularity. It is easy to understand and it is capable of capturing the causality in an intuitive way. In conclusion, Bayesian networks result to be a convenient language to cover a wide domain of problems remaining well defined, and much less demanding for computer processing compared to other methods.

A more comprehensive description of Bayesian Belief Networks can be found in the appendix at the very end of this report.

2.3 Emergency preparedness

In this section a brief overview of the Swedish emergency planning organisation is given. This is meant to offer a basic understanding of the context for use of tools like RASTEP.

In general, the emergency preparedness plan tries to link in-plant observations, mostly related to the status of safety barriers, to the potential release in the event of an accident. The need for countermeasures is evidently coupled with the release information availability. This implies that the systems are set to match a certain level of stand-by preparedness, at least in the early stage of an accident. By doing this, the observations from the plant are converted to notification levels in a limited number and, complementarily, data related to the status of the safety barriers and release will indicate the best actions to put in place. This is mostly referred to the emergency plan at the nuclear site [9].

The Swedish emergency preparedness organization is a network of authorities operating at different levels. These authorities have different tasks depending on their area of expertise. The county administrative boards (länsstyrelser in Swedish) are the main actors in the emergency preparedness organisation and represent the first line of protection for the population. These boards are in charge of all the rescue service teams in the area outside the NPP interested by the accident.

The Swedish Radiation Safety Authority (SSM) is in charge of making the communications to the concerned county administrative boards and facilitate the decision making process in the event of an accident. Strålsäkerhetsmyndigheten, this is the name of the safety authority in Swedish, provides also the necessary knowledge about the radiological impact to the population in complement to the information given by the nuclear power plant itself. For instance, information regarding the specific radioactive substances emitted and in what concentration together with meteorological data. These data are of crucial importance to mitigate the consequences and, of course, are included in the emergency planning [10]. The Swedish Meteorological and Hydrological Institute (SMHI) helps the Swedish Radiation Safety Authority with meteorological data. SHMI also provides SSM detailed updated calculations of the atmospheric dispersion of airborne radioactive substances [11].

2.4 EOP and SAMG

From the point of view of the operators of a nuclear power plant, it is common practice to distinguish between two procedure domains when addressing the issue of safety: a preventive domain and a mitigatory domain.

In the preventive domain, some descriptive steps are indicated as guidance by the regulatory boards and authorities. The plant status is described by the available data coming from the instrumentation, and all the actions to take in order to avoid certain consequences can be effectively determined in advance through appropriate analysis. The guidance for this domain is in the form of procedures. Usually, these are called emergency operating procedures (EOPs). EOPs are generally limited to the actions the operator takes before core damage occurs and they cover both design basis and beyond basis accidents [12].

EOPs are developed either as event oriented procedures or symptom based procedures. Both approaches are used in practice, but symptom oriented procedures are preferable due to the definition and prioritization of the major critical safety functions. This means that the decisions and measures that take place to respond to events are not predefined, but they are specified with respect to the symptoms while maintaining optimal operation of the plant [13].

The mitigatory domain includes all those situations when there are uncertainties regarding the plant status and the actions to put in place. This implies that the guidance for such domain is composed mainly by a range of possible mitigatory actions, including additional evaluation and alternative actions. For this particular kind of domain the guidance is generally called Severe Accident Management Guidelines (SAMGs).

The effectiveness of the SAMGs during the event of an accident is strongly dependent on training of the emergency response team. Moreover, during an accident, collaboration and interaction between the emergency system at the site and the emergency response centres is essential. In conclusion, the important idea is that communication with the general public has to be based on reliable information and accurate data.

3 RASTEP

3.1 Introduction

RASTEP is a computerized source term prediction tool currently under development at Scandpower AB. This software aims at providing a basis for the off-site emergency management at nuclear power plants. RASTEP will be using two different parts, a Bayesian Belief Network (BBN) to model severe accident progression, to predict plants states and release paths, and a source term module that will define the height, amount, composition and timing of the release.

At this stage of development a user interface called SPRINT is used for the RASTEP model. This interface is where the user answers questions about the plant observables so that the conditional probabilities in the BBN can be directly updated. On the other hand, the release information is extracted from pre-calculated source terms based on MAAP calculations for level 2 PSA studies.

The sections below describe in more detail how RASTEP is built and how it is meant to be used in the future [14].

3.2 Desired functionality

The users of RASTEP are assumed to be SSM, part of the emergency preparedness organisation, and possibly the very operators at the nuclear facilities. The aim of this tool is to provide the users (mainly SSM) with an independent view of the accident progression and possible off-site consequences. SSM is meant to interact with the plant operators and the external emergency preparedness organisation when using RASTEP. Another obvious usage of this tool is for training.

The RASTEP BBNs have been modelled for use only in power operation mode until now. The start-up, shut-down, and cold shut-down states are not considered in the current configuration. The model considers all the initiating event categories as in the level 2 PSA. However, in order to derive a sufficiently detailed structure some grouping of events has been adopted. For instance, LOCA events are divided into large or small break LOCA, based on system requirements from the emergency core cooling and auxiliary feedwater systems. Another example is that the BBN considers the loss of external power both as an initiating event and as a grid level consequence of other initiating events. Many transients are treated as one common category and that is the reason of using observables along the predictions. The containment in the Swedish BWRs is inerted during power operation, but the possibility of the containment to be air-filled is considered in the network.

It is important to consider at what stage in an accident sequence SSM might start using RASTEP. For now the starting point has been set to the time of the failure of the *first line of defence*. For instance, the failure of one or more of the systems for fission and pressure control, core cooling and residual heat removal can be typical signals that start RASTEP.

The current use of RASTEP can be seen as both static and dynamic. The static use includes the level 2 PSA information and the information gathered from deterministic calculations (e.g., MAAP or MELCOR) modelled in the Bayesian network (BBN). The dynamic use includes all those tools (e.g., LENA or ARGOS) for off-site consequence analysis and the data transmission with the plant when RASTEP is actually used. However, it is important to remember that the main objective in the short term is to give more dynamics to the source terms analysis.

3.3 User interface

RASTEP is a computer based tool intended for use in decision support during severe accident events at nuclear power plants. One of the important features required in this tool is a user interface easy to interact with and as functional as possible when coupled to other tools. This includes both the input/output interface within the RASTEP model and the interface with off-site consequence analysis tools like LENA and ARGOS.

As stated in section 3.1, the RASTEP model includes two different parts: the BBN model used to predicts plant states and release paths, and the source term module that characterises the releases. The SPRINT user interface developed within the STERPS EU project is used to describe the actual model. This interface will possibly be replaced with a new in-house software called RASTEP, where the functionality and features will be added.

Figure 5 shows the SPRINT user interface where the user answers questions about plant observables. The answers given by the user correspond to node states that are updated into the corresponding network node as findings. Netica (developed by Norsys Software Corp.) is the BBN engine that finds beliefs for all the other variables in the network.

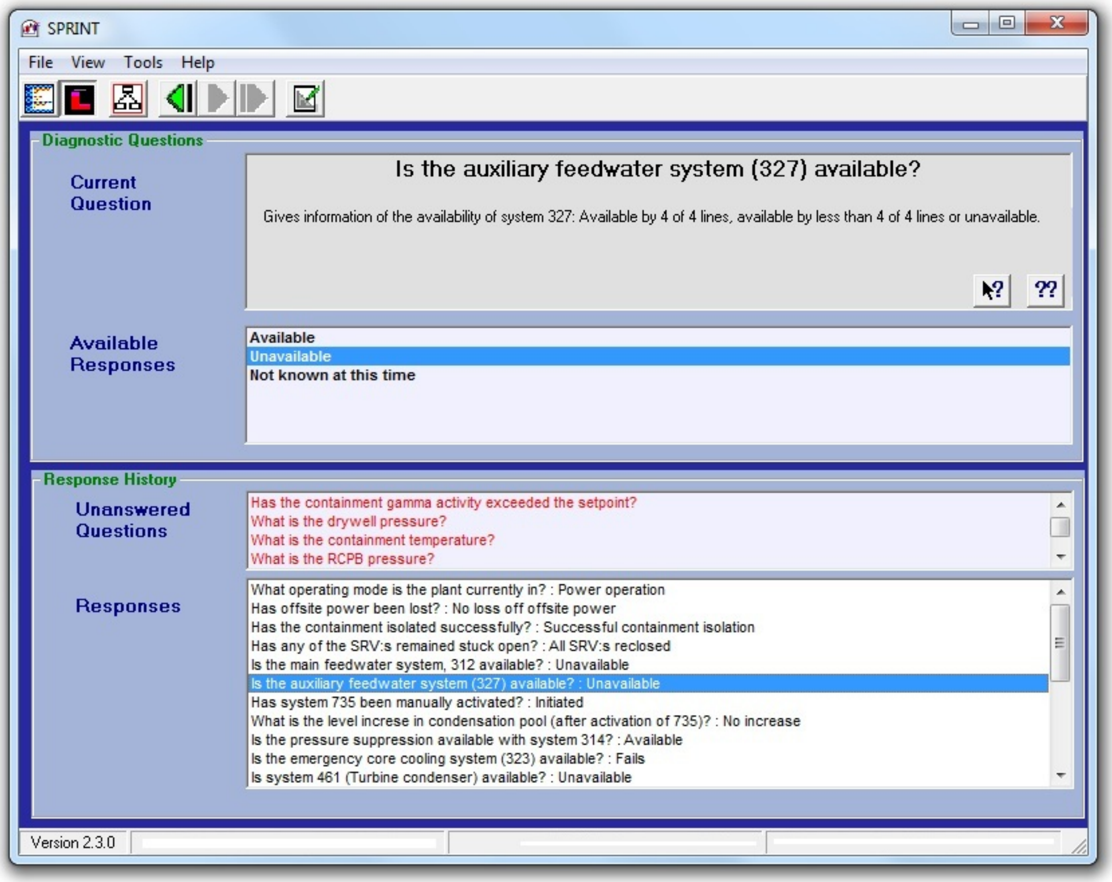

Figure 5 SPRINT user interface

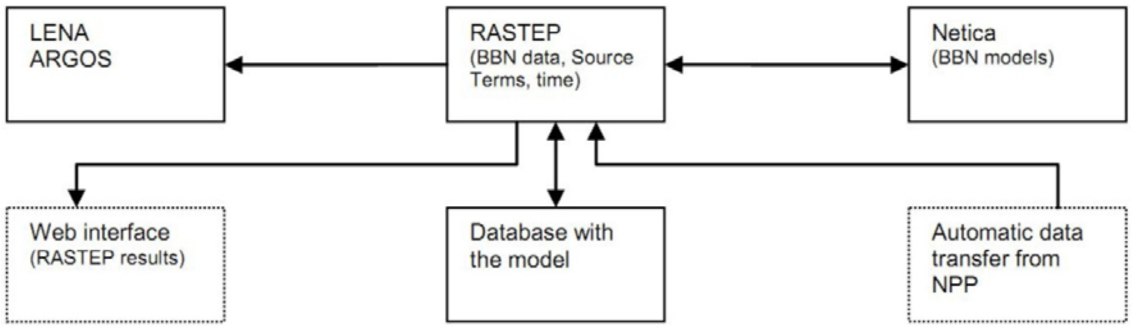

Figure 6 Schematic of RASTEP

Figure 6 shows the actual configuration of RASTEP where a web interface and an automatic transfer of data from NPPs are also included to present an example of desired features.

3.4 Mapping of plant characteristics

In order to create a RASTEP model for a specific plant, the mapping of plant characteristics is required. This mapping strategy includes a good knowledge and understanding of plant design features as well as all those systems aiming at mitigating severe accidents.

The plant parameters to be included in the model are identified considering the fission product transport and retention in the plant compartments. The description of plant systems and compartments are extracted from severe accident studies and severe accident strategy plans. The mapping activity also includes and has to be consistent with the modelling performed during level 2 PSA.

Similarly to the mapping of plant characteristics, the observables are also identified and considered in the model. The plant observables are variables describing the status of the plant during an accidental event. They are basically measurements such as pressure, temperature, water levels in specific regions of the plant, and all those signals available from instrument readings.

3.5 Physical phenomena

The RASTEP model has been created considering a large series of physical phenomena occurring during severe accidents at NPPs. These phenomena have generally the potential to threaten the integrity and the tightness of the reactor containment and have to be accurately modelled. A comprehensive description of such phenomena is indicated in the IAEA safety guides for level 2 PSA and in the APRI project (Accident Phenomena of Risk Importance) [15] [16].

Table 1 List of relevant physical phenomena

Phenomena	IAEA	APRI	RASTEP	Comment
Hydrogen production	X		X	The most important consequence of large hydrogen production is hydrogen contribution to the containment pressure.
Hydrogen combustion	X		X	Hydrogen combustion is an important phenomenon that may threat the integrity of the containment in accidents occurring during start-up and shut down of the reactor. It may change the accident sequence abruptly and this possibility should be considered in the BBN.
In-vessel steam explosion	X			Large steam explosion in the RPV that threatens the integrity of the containment is generally considered to have a low probability.
Ex-vessel steam explosion	X	X	X	Steam explosion in the containment is an important phenomenon that is considered in the BBN.
In-vessel corium coolability		X		It is assumed that there is water in the lower head and the corium remains covered with water.
Ex-vessel corium coolability		X	X	Corium coolability in the lower drywell and MCCI can be handled together as the consequences are the same, destroyed penetrations in the lower drywell floor. Represented with a singular node in the BBN.
Molten core concrete interaction (MCCI)	X		X	
Direct containment heating (DCH)	X		X	Even though direct containment heating generally has a low probability in Swedish BWRs, it is relevant to consider it in a node in the BBN.
Recriticality			X	The possibility of recriticality should be considered during an accident. It is thus relevant to introduce a node in the BBN.
Vessel thrust forces	X			Vessel thrust forces are not considered a threat to the containment integrity, and may therefore be left out of the BBN.

Table 1 shows the relevant physical phenomena considered for the RASTEP model of one of the reactors at Oskarshamn. The same list and selection is considered viable for all the Swedish BWRs. Each cross denotes the physical phenomenon included in the safety guides (IAEA and APRI) and in the RASTEP model respectively.

It is worth mentioning that some of the phenomena considered in the RASTEP model do not have a parallel description in the PSA level 2. This denotes that some of the conservative assumptions in the PSA have to be expanded and quantified when dealing with a tool for real-time predictions.

3.6 Source terms

The outcome of RASTEP is defined as a set of possible plant states ranked according to probability. Each state is associated with an environmental source term and represents the quantity, characteristics and timing of the release of radioactivity to the environment. The source terms are also associated to the corresponding available release paths.

As already stated, the source terms are currently based on pre-calculated plant specific source terms. A graphical presentation of the analysis results, both in terms of node probabilities and characteristics of radioactive releases is included within the SPRINT user interface.

Figure 7 shows an example of how the releases are displayed in the current user interface. In addition, the source terms are given in terms of INES categories (International Nuclear and Radiological Event Scale) to visualize the level of danger for the population and promptly enable communication of safety information in case of accidents.

Aspects related to the time of the releases are based on the default timing given by the MAAP calculations. Future adjustments and modification of RASTEP will imply that the releases will be presented according to the actual progression of the accident.

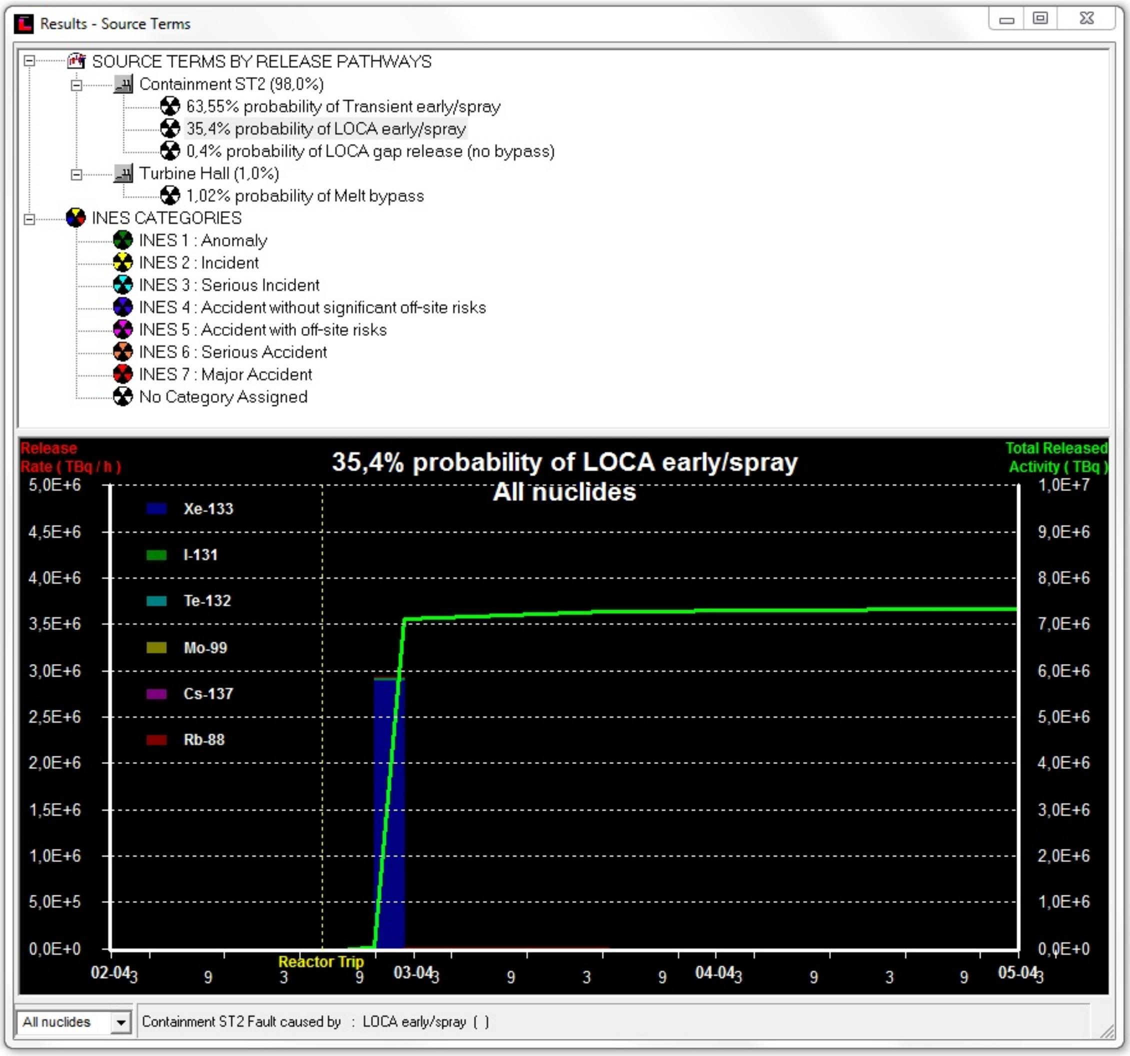

Figure 7 Graphical representation of the source terms

3.7 Network structure

The Bayesian Belief Network (BBN) is divided into sub-networks in order to create a structured system able to reflect the accident progression. The sub-networks currently defined in the RASTEP model are:

- Initiating event
- Core cooling
- RHR (Residual Heat Removal)
- Fuel status
- RPV/RCPB status
- Containment status
- Reactor building status
- Turbine building status
- Source terms

In the complete network, each of the sub-networks above includes a number of nodes. Figure 8 offers a simplified overview of the BBN model for a generic Swedish BWR, where the main blocks and the important interrelations are described.

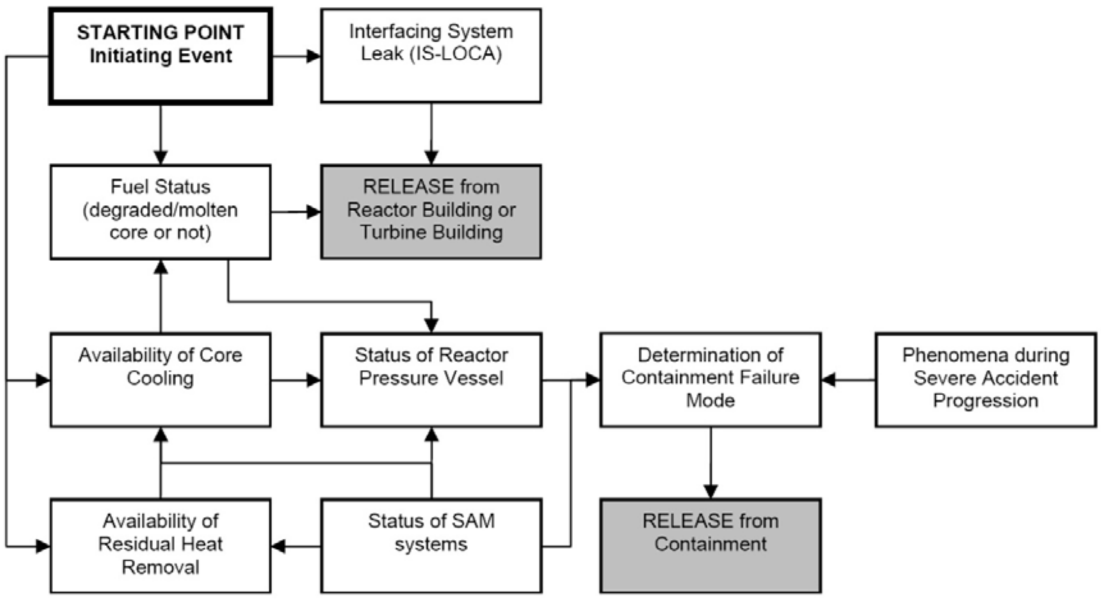

Figure 8 BBN structure for Swedish BWRs

As shown in Figure 8, the starting point of the entire network is the identification of the initiating event of the accidental scenario. Later on, the probability of the set of release paths is estimated based on the status of some fundamental sub-networks (fuel status, status of important safety systems, containment status, etc.). The success or failure of severe accident mitigation systems and of severe accident management actions also plays an important role in the definition of the probability of release paths.

The nodes in the BBN are either related to observables or hidden nodes. The first can be updated by the user, while hidden nodes are updated solely through the interactions with other nodes. The nodes are also categorised into seven sub-categories depending on which type of information they provide and process in the network. The list of sub-categories, the corresponding probability information and type of node is presented in Table 2.

Table 2 Sub-categories in the BBN

Sub-category	Type of node	Type of information
Boundary condition node	hidden/observable	Boolean
Initiating event node	hidden	Frequency data from PSA
Source term node	hidden	Boolean
System/Function-performance node	hidden/observable	State probability
Parent state determining node	hidden	Boolean
Outcome determining node	hidden/observable	Probability
Measurement node	observable	Unavailability data

A good strategy when modelling a Bayesian network is to minimise the number of states associated with each node. By doing this it is possible to reduce the size of the associated conditional probability tables. All the phenomena as well as manual actions referring to safety procedures are modelled in separate nodes to achieve the goal of reducing complexity.

3.8 Conditional probability tables (CPTs)

In view of the above and considering the definition of the BBN itself, Conditional probability tables are defined for each node in the network. CPTs contain the probabilities that defined the influence of parent node(s) on child node(s), and are calculated by the use of the Bayes' theorem.

Table 3 Definition of conditional probabilities in RASTEP

Nodes	Probability
Boundary condition node	The states of this type of nodes are either 100% or 0%, which is set by the modeller or user.
Initiating event node	State probabilities are derived from initiating event frequencies in the PSA model.
Source term node	Source term nodes have a probability set to 1 or 0 defining the release category for all possible plant states respectively. The corresponding source term is derived from level 2 PSA.
System/Function-performance node	State probabilities are derived from fault tree analysis cases in the PSA model.
Parent state determining node	If the parent state is TRUE then this is confirmed by a 100% probability of the parent state determining node and a 0% probability of negated state. The opposite applies if the parent state is FALSE.
Outcome determining node	The determination of state probabilities in outcome determining nodes is quite complex, and may rely on a mixture of engineering judgement, general considerations and calculations within the PSA. The latter applies to, e.g., phenomena probabilities.
Measurement node	State probabilities need to consider both correct and incorrect indications. Prior assumptions regarding correct indications can be based on component unavailability data. Regarding misleading incorrect indication, this will need to be evaluated case by case based on failure impact.

Table 3 shows the definition of the basic principles for CPT determination for each of the node categories stated in the previous section (3.7).

CPTs are based on a variety of sources depending on which type of probability has to be calculated. The PSA gives the basis for the computation of all those probabilities related to initiating events, hardware functions (systems and components and their availabilities) or human errors. They can be found in the plant component data base, or in the fault tree analyses of systems in the PSA.

Due to the complexity of certain accidental phenomena, expert judgement is needed when it comes to computing probabilities where the uncertainties are relevant (e.g., the probability of hydrogen explosion in the containment).

3.9 QPRO

QPRO (Quelltermsprognose) is a user interface designed by GRS (Gesellschaft für Anlagen- und Reaktorsicherheit GmbH) as an advancement to the SPRINT interface [17]. With this new tool GRS addresses some of the limitations of SPRINT, especially those regarding the source term definition, the user input and the validation of the tool. The new features introduced in QPRO can be summarised as follows:

- Removal of contradictory user questioning in the model;
- Addition of warning for the user;

- Traceability of user's answer and results history;
- Input data directly forwarded to the source terms spreadsheet file in order to modify the timing in the release output;
- PSA results showed when emergency condition is declared and no user input;
- Automatic export of source term data and graphical features improved;
- Use of the tool in conditions other than normal operation.

4 Problem statement

Probabilistic methods currently used for safety analyses are all based on static logic models. On the contrary, deterministic models are capable of resolving time dependent interactions between physical phenomena. Integrated Deterministic-Probabilistic Safety Assessment (IDPSA) is one of the proposed acronyms used to group new approaches on combined DSA and PSA. These methods aim at making safety studies more flexible and realistic with the goal of meeting increasingly stringent safety requirements, licensing rules and open the way for new reactor designs [18].

Static methods have usually the great advantage of being easy to understand, computationally simple and the representation schemes are easily handled. In the PSA community, the use of dynamic PSA methods has always been avoided in industrial practice, due to the mathematical complexity and computational demands of such approaches. Nevertheless, a brand new effort in understanding the dynamics behind the accident progression is justified when dealing with severe accident mitigation [19].

The BBN model of RASTEP is based on logical relationships between various key plant parameters, represented by a graphical network. The network is plant specific and is derived from the system description and plant documentation. The PSA functional structure is also included in the network as well as thermal-hydraulic considerations when it comes to specific systems. In addition to the network structure itself, Conditional Probability Tables (CPTs) are defined for each node in the network. These probability tables define the influence of each parent node on the child node. The nodes represent observables and the default values for these observables are usually determined by expert judgment.

The outcome of RASTEP is typically defined by various plant end-states ranked according to probability. Each plant state is then associated to environmental source terms which give the quantity, characteristic and timing of the radioactive release. As already mentioned, the source terms are currently determined by MAAP calculations during PSA level 2 studies.

Accident modelling in PSA analyses is performed in a quite conservative manner. Even though expert PSA practitioners make intelligent simplifications, being able to capture a portion of the system dynamics, this is not the same as describing real scenarios. The representation in a static Event Tree/Fault Tree framework is not capable of reproducing the intrinsic vulnerability of certain systems even though the identification of potential worst case scenarios is not omitted due to the generally conservative approach.

The analysis performed with RASTEP aims at addressing the current accidental scenario. This differs from the way the analysis is performed during PSA studies, where a set of representative scenarios are investigated. Taking this distinctive difference into consideration, the need of updating the source term information as the accidental scenario progresses would give the opportunity to have more realistic results for the environmental release in the very current situation.

The objective of this thesis is to evaluate how the source term predictions can be updated, and understand what the requirements are to implement this modification. A deeper understanding of the concepts behind the integration of deterministic and probabilistic methods is the starting point to achieve this goal.

The ambition is, then, to make RASTEP a more dynamic tool for source term prediction in a way that it will be more effective and useful for emergency preparedness and response. To enhance the predictive abilities of RASTEP a conceptual basis need to be provided before going to further development.

5 Tools for source term assessment

The tools that model the phenomena of severe accidents can be divided into three main types according to their use and capabilities:

- Mechanistic codes
- PSA codes
- Parametric codes

Mechanistic codes model the physical phenomena in a very detailed manner. Typically these codes need a very long computational time (i.e., some days), and are used in severe accident research providing valuable data for use within benchmark activities. Most of these codes are based on finite element for structural dynamics modelling and are able to handle non-linearities.

On the other hand, PSA codes are used for routine application in PSA studies. These codes are designed to run faster and calculate a large amount of sequences. To achieve run times several order of magnitude shorter than the mechanistic codes the models are generally simpler.

As last, there are also the simple parametric codes which are intended for very specific PSA applications. An example of application for this type of codes is the source term estimation. These codes are based on simple parametric models where interpolations between fixed points are made to determine the values of parameters. The fixed points used for the interpolations are previously calculated with a more complicated code. Parametric codes generally need a lot of effort in initial testing and calibration by the use of more detailed calculations or experimental data.

The general idea is to avoid using a single code to treat the whole range of phenomena in order to determine the threats to the containment and the fission product source terms. Multiple codes, each dealing with a specific aspect of the accident behaviour, are sometimes coupled. It is desirable to have an automatic transfer of information between one code and another. However, this is not always the case when it comes to practical use of such tools. Manual transfer is still the most common way of using coupled codes nowadays, making the introduction of errors sometimes unavoidable. It is worth mentioning that, using a manual transfer of data, the user can examine the outputs of each code during intermediate phases of the calculation process to check that the results are meaningful [1].

A more modular and integrated approach has been adopted in the newer generation of severe accident codes (e.g., MAAP and MELCOR).

5.1 MAAP

MAAP (Modular Accident Analysis Program) is an integrated computer code owned and licensed by EPRI and developed by FAI that simulates the response of light water and heavy water nuclear power plants. Several parallel versions of this code are specifically made for other types of reactors (e.g., CANDU, FUGEN and VVER reactors). MAAP simulates Loss-Of-Coolant Accident (LOCA) and non-LOCA transients mainly for PSA applications. It can also simulate severe accident sequences, including actions taken as part of the Severe Accident Management Guidelines

(SAMGs). When a set of initiating events and operator actions are given to the code, it predicts the plant's response as the accident progresses. MAAP, and specifically version 4 of the code, is used for the following:

- Predict the timing of key events (e.g., core damage, core uncover, core relocation to the lower plenum, and vessel failure);
- Evaluate the influence of mitigation systems, including the timing and the impact of operation of such systems;
- Evaluate the impact of operator actions;
- Predict the magnitude and timing of fission product releases (source terms);
- Investigate uncertainties in severe accident phenomena.

MAAP requires two files as input: a parameter file which contains plant-specific information, specifications of the output and user-controlled phenomenological parameters; and sequence input file which specifies the accident initiators, operator actions, and sequence control times. After the information in the two files has been processed, the code is able to predict the sequence of events and the corresponding plant conditions generating a number of output files. The output also includes a summary of the sequences and events, tables of time-dependent results, and tabulated results. These results provide all the details of the plant's status for the selected times and are suitable for plotting.

As mentioned above, MAAP results are primarily used for the determination of level 1 and 2 PSA success criteria and accident timing for human reliability analyses. These results are also used for equipment qualification applications, the determination of fission product release frequencies, emergency planning and training, simulator verification, analyses to support plant modifications and generic plant issue assessments.

MAAP, as integrated code, treats the full spectrum of important phenomena that could occur during an accident and, at the same time, create the model for thermal-hydraulics and fission products assessment. It is also capable of modelling the primary system and the containment and reactor/auxiliary building.

The parallel versions of MAAP for reactors other than LWR contain the same core model, containment model, fission product model, and input and output interfaces. In addition, they have distinct models meant to describe the primary system and the engineered safeguards. The passive features of advanced LWR designs are also implemented in the code [20].

5.2 MELCOR

MELCOR is a fully integrated computer code that models the progression of severe accidents in LWR nuclear power plants. MELCOR is developed at Sandia National Laboratories under contract from the U.S. Nuclear Regulatory Commission (NRC). A large series of severe accident phenomena is treated in MELCOR in a unified framework (e.g., thermal-hydraulic response in the reactor coolant system (RCS), cavity, containment, and buildings; core heat up, degradation, and relocation; etc.). The current uses of MELCOR also include estimation of severe accident source terms with sensitivity and uncertainty analysis [21].

MELCOR is capable to model all phases of severe accident progression. Thermodynamic state properties are treated in the spatial geometry and volumes of an NPP. Two-phase models and non-equilibrium thermodynamics are also implemented (e.g., different temperatures and phase conditions) as well as the opportunity to consider the volumes occupied by other material.

The code comprises a driver module and a number of model packages which are executed depending on the problem that has to be solved. The use of MELCOR in combination with the model packages offers the same types of assessments included in MAAP (see list in section 5.1).

MELCOR calculations are carried out in two separate steps, called MELGEN and MELCOR. MELGEN is the processor where the input is written, checked, accepted and then transferred to a file for use in MELCOR. Next, the code only needs a short input where time-steps and execution parameters are described.

Conversely to MAAP, MELCOR produces a number of output files which include a diagnostic file displaying errors and warnings to assist the user in debugging the input, an output text file, a plot file and a message file. Another code package also includes a graphics processor which needs a special input. However, it results to be more manageable to plot results using the plot file by means of the general graphics codes. A different package computes off-site consequences of radioactive releases based on source terms calculated previously [22].

6 On-line accident diagnostic tools

In this chapter, two on-line accident diagnostic tools, MARS and ADAM, are presented. Both these codes were indicated in a previous M.Sc. thesis work [2] as good candidates for use within RASTEP. Especially MARS (Modular Accident Response System) has been chosen within this thesis as best candidate software to be used as source term module connected to the BBN of RASTEP. According to this choice, particular focus will be given to the MARS software and its features.

Tools other than MARS or ADAM are already available (e.g., ASTRID, SABINE, SESAME, CRISALIDE, CAMS, etc.). However these tools do not look feasible for implementation with the Bayesian Network and the calculation speeds are not compatible with the usability that RASTEP is intended to offer.

6.1 MARS

6.1.1 Introduction and main features

MARS is an integrated software suite that provides a complete engineering simulation of an operating nuclear power plant based upon actual conditions within the plant. MARS models the plant response during many modes of operation such as shutdown, refueling, normal and abnormal, and accident conditions. The MARS suite includes:

- A simulation of the nuclear power plant using the Modular Accident Analysis Program (MAAP) and other SA codes;
- A user-friendly Graphical User Interface (GUI) that represents the nuclear power plant;
- The ability to use live on-line plant data for engineering simulations;
- Modules for performing alternative and/or redundant instrumentation readings;
- Models for emergency response activities (e.g., on- & off-site radiation dose assessment).

The MARS software suite uses the MAAP along with customized and industry standard software to support the MARS applications. As described in section 5.1, MAAP is able to simulate the thermal-hydraulic and fission product plant response of the nuclear power plant during normal, accident and severe accident sequences, including actions taken as part of the normal and emergency procedures.

The MAAP code is used extensively throughout the world to perform Level 2 PSAs, support plant activities, accident management assessments and is regularly used to address emergent issues. The MAAP models are tuned to specifically represent the nuclear power plant specific configuration and emergency systems. The MARS software suite's Graphical User Interface (GUI) provides the user with an easy-to-use method for interacting with the simulation providing the user with information that is easy to understand and as familiar as possible. The GUI features include:

- A dynamic graphical representation of the nuclear power plant that shows the vessel, containment, reactor building and systems;
- The ability to control the simulation via use of the mouse;
- Panels of dynamic widgets and tables that show the numeric values of plant signals and graphical representations of output instrumentation;
- Panels of easy to use X-Y Plotting of trended variables, with the ability to look at multiple variables, switch between variables and adjust the plotted variables;
- Panels of emergency procedures are graphically displayed and can be activated to automatically perform operator actions based upon the evolving simulation. The modelled operator actions include human factor times;
- Multiple monitors can be used to display the various GUI screens;
- Multiple users are able to simultaneously access the engineering simulation via separate networked PCs;
- Easy to use automatic report generation, that grabs the user requested information and automatically creates a Microsoft Word™ document that can be saved, printed, or Emailed to user selected recipients.

By using dynamic sets of on-line data, MARS is able to obtain insights into the current and potential future status of the plant during many modes of operation. The MARS' *tracker* function uses a limited set of dynamic plant signals to initialize the MAAP code and then follow the evolving plant response. The MARS *instrumentation* module provides an alternative means to validate and/or obtain plant instrumentation readings. The MARS *predictor* module provides the user with the ability to perform much faster-than-real-time evaluations, including modelling operator actions. For example, the MARS user can model the loss of a given safety system to determine its effect on the overall accident progression. The *predictors* provide ways to look into the future of the plant state.

6.1.2 Input of MARS

MARS uses, normally, around 75 signals as input data. These data come from the plant computer or by manual entry and generate the information that MAAP needs to start accident simulations [23]. In the case MARS is employed for training purposes, the plant data could be replaced by data from other tools which simulate the plant response, i.e. a control room simulator or other calculation tools. The incoming plant data (real or simulated) will go through two steps of processing:

- Verification
- Conversion

As shown in Figure 9, the MARS software processes the data either when they come directly from the control room computer of an NPP, or when they are simulated with other tools. In the verification process, the signals of good quality are accepted and those that are determined to be of poor quality are discounted. In this case, the user will be notified, and another method for determining a representative value will be used. For instance, if some parameter of the primary system needed by MARS is not included in the input, the remaining parameters of the primary system will be used in order to estimate a representative value. The user has also the ability to manually input a value based on information that is available off-line or derived from some other sources (e.g., a BBN in the case of RASTEP).

In the conversion process, the signals from the plant are converted into the correct format needed by MARS. For instance, variables that have been transmitted in the form of percentage numbers are converted back to actual values (i.e. flow rates, water levels, power etc.). When the conversion process is completed the processed plant data are passed on to the MARS initialization routines.

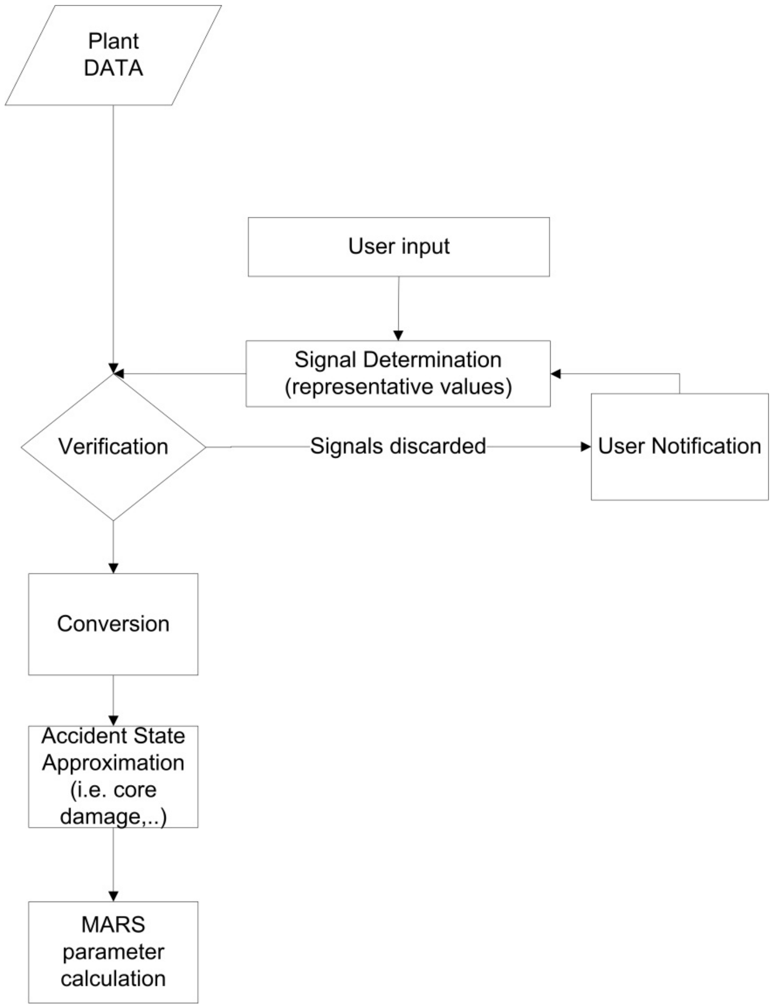

Figure 9 MARS processing

In the initialization routine, the processed plant data are used to estimate the approximate accident state (e.g., core uncovered, core been damaged, vessel failure, etc.). The identification of the time intervals for such events is very important. The readings from the available instrumentation can be used for modifying the initialization routine. This is a fundamental feature that gives the opportunity to follow the accident progression in its initial phases. For instance, the estimation of the fuel parameters can be very different if core damage is already detected compared to the case in which there is no core uncover.

When the accident state has been approximated, several thousand parameters that are required by MARS to be fully initiated are computed based on the plant data.

6.1.3 Tracking

The MARS *tracking* function is a code module used to check that the simulation follows the real plant behaviour. The *tracking* function is the basis for the *predictor* function (6.1.4) to be actuated.

Initially, the *tracking* function performs an assessment of the first symptoms of the accident to determine the plant status and the types of accident initiators. Once the plant status and the accident initiators are identified, and the evolving set of plant data is available, the *tracker* performs the calculations to guide the MAAP code towards the simulation of the plant behaviour.

A comparison of the plant data available with the *tracker* information is performed over multiple time intervals in order to assess how the simulation follows the real plant status. If differences between these two are found, and certain parameters are not in the range specified by the user, the *tracker* corrector logic is applied. This implies that the *tracker* simulation is modified considering the differences between the simulated values and the actual plant data. These modifications can vary from stopping the simulation for a single system, in the case that no absolute information is known about that particular system, to stopping and re-initializing the whole simulation.

The *tracking* module is also capable of evaluating the potential root causes of the accident. For instance, if the *tracker* identifies a LOCA, the information generated by the *tracker* about the break size (of the pipe) and the elevation may be used to determine possible locations of the break.

6.1.4 Predictors

The *predictor* module, also included in the MARS software suite, provides indications and predictions of future plant states when only the actual scenario is available. The information generated by the *tracker* is used by the *predictor* to perform the calculations necessary to provide relevant values for the plan state analysis. For instance, the *predictors* can be used to estimate the time before a major change in the accident scenario occurs (e.g., core uncover, core damage, reactor pressure vessel or containment failure) and the efficiency of accident management strategies.

When the *tracker* has gathered enough information on the plant status, the *predictors* are initialized automatically or manually. The calculations of the *predictors* can be performed assuming a wide variety of actions by the operators of the plant. Usually, in the first execution, the assumption that the operators follow entirely the emergency operating procedures (EOP) is made. The second *predictor* analyses the behaviour of the plant in both the short and long term assuming no additional actions by the operators. The rest of the *predictor* simulations analyse the effects when following the accident management guidelines (SAMGs) or other unexpected operator actions.

The *predictors* provide information such as:

- Minimum injection flow rates for successful mitigation of the accident
- Timing of vessel and containment failures
- Effects of the operator actions
- Source term assessment
- Future accessibility to all plant areas based in predicted radiation levels

6.1.5 Instrumentation

The availability of signals and parameters coming from the plant instrumentation is crucial when it comes to manage with an accident scenario. One important requirement is to obtain, at least, approximate values for the relevant parameters in the plant. It is also fundamental to discount all those instrumentation values which appear to be of poor quality in the simulations that can be provided by actual readings, and MARS provides means of addressing this issue. The MARS *tracker* and *predictor* functions can approximate values and trends on the basis of the current plant configuration and simulated future plant states. The capability to analyse the available plant signals and determine the validity of the simulated data is a key feature. Confidence levels are also calculated for some variables. As discussed above, in some situations, the user is notified and countermeasures will be taken by MARS to replace those data with representative values. Figure 10 shows a screenshot of the graphical user interface of MARS, and specifically, this is the case of the Spanish version of MARS (MARS-CSN) [24].

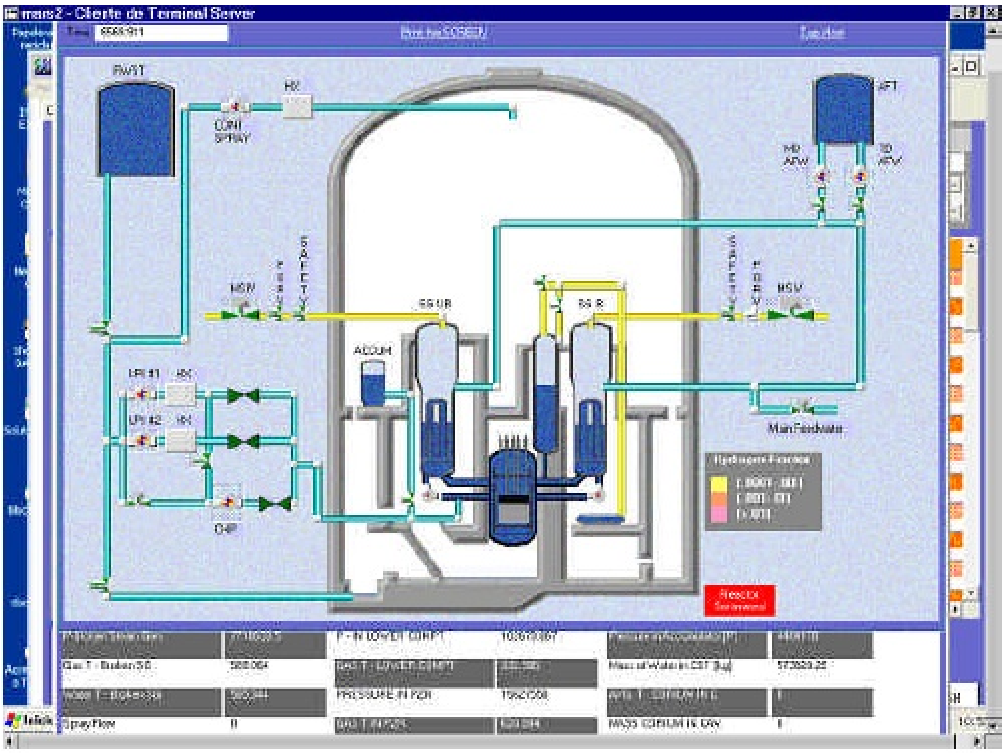

Figure 10 Example of MARS Graphical User Interface (Courtesy of CSN and FAI)

6.1.6 Operator actions

The users of MARS have the ability to change the predictor simulations once they have been already initialized. This feature enables the user to adjust the calculation during the accidental scenario by means of modifications in the operator actions (e.g., recovery actions) and procedures. For instance, the user could be interested in modelling the loss of a specific safety system in order to determine the effect on the overall accident progression.

6.1.7 Use of MARS at CSN

This paragraph is mainly based on personal communications with Mr José Ramón Alonso, one of the nuclear installations coordinators at CSN (Consejo de Seguridad Nuclear) in Spain. He has been responsible of the developments carried out at the CSN to enhance the capabilities of the *Operating Analysis Group* of the CSN's Emergency Room for more than 12 years. Since December 2010 he moved to another position within CSN.

The "MARS' story" at CSN is that, when they started to think about fast prediction during severe accidents at NPPs, they considered two possibilities feasible. The first one was to follow an approach very similar to RASTEP where the *likely plant states* were based merely on off-line studies. The second one was essentially the use of MARS as a fully real-time tool. At the time they started with this project MARS was just a prototypical tool but they opted for the second option anyway.

MARS-CSN is a tool based on the Severe Accident code MAAP-4 (original version was based on MAAP-3.0B). MARS-CSN needed a lot of effort to be rightly adapted to a specific plant or site, first of all to adequately reproduce the characteristics of the plant and, secondly, for validation aspects. These are never easy issues to deal with, mainly due to the lack of reference validated data (they have used data generated with RELAP5, TRAC and RETRAN, MELCOR but essentially from the own MAAP) and the discrepancies one obtains when using several different codes to simulate the same sequences.

The use of MARS-CSN requires the execution of the *tracker* module which capture the most significant information from the real-time data (30 second time frame) received from a specific plant in the case of an accident, and then to initialize MAAP simulations. Once the user considers that the *tracker* is executing a good approximation of the sequence that is actually occurring at the plant, he (the user) can launch the module *predictor*. The procedure of accepting the *tracker*'s approximation was something that initially FAI wanted to automatically implement by using some internal *acceptance criteria*, but it was proved to be not the best option in the case of the CSN system.

The *predictor* module executes automatically 5 different MAAP-4 specifically adjusted runs. These simulations are performed much faster than real-time when enough computer CPU capacity is available. The user can decide at any time to simulate a leak in the containment barrier with the due assumptions about size and location of the breach. The code will calculate the mass flow rate going out of the containment which logically incorporates the source term calculated by the MARS-*predictor* at the selected time in the *donor cell*. The *donor cell* is the indication of a particular chosen volume in the gas space of the containment. This volume is usually manually chosen by the user, but an automatic selection can also be made.

Unfortunately, all the MARS' development & validation activities are now currently "frozen" at the CSN. This occurred mainly because of lack of human resources which were relocated to other high priority activities they started around 2004. The decision of concluding this activity was taken, by chance, immediately after the finalization of the last development phase of the project they had signed with FAI. In practice, the CSN has not been able to profit of the results of this task; a task that included the development of plant specific models for all the Spanish NPPs, complete migration to MAAP-4, migration to a Windows™-base environment and other minor improvements [25].

6.2 ADAM

6.2.1 Introduction

The Accident Diagnostic, Analysis and Management (ADAM) computer code is a tool for online accident diagnostics, management and training developed by ERI (Energy Resources International, Inc.) and financed mainly by the Swiss Federal Nuclear Safety Inspectorate (ENSI).

The capabilities of ADAM incorporate a balance of mechanistic and phenomenological models with a simple parametric approach. All the typical analyses and characteristics typical of severe accident codes (like MAAP and MELCOR) are also included in the ADAM model.

6.2.2 Main features

The model in ADAM is defined by a coarse spatial nodalization of the reactor coolant system and containment, where the definition of timing is explicit. This model configuration enables this tool to give results much faster than real-time (i.e., 100 to 1000 times faster than real-time on a PC). Figure 11 shows a screenshot of the graphical user interface of the ADAM code.

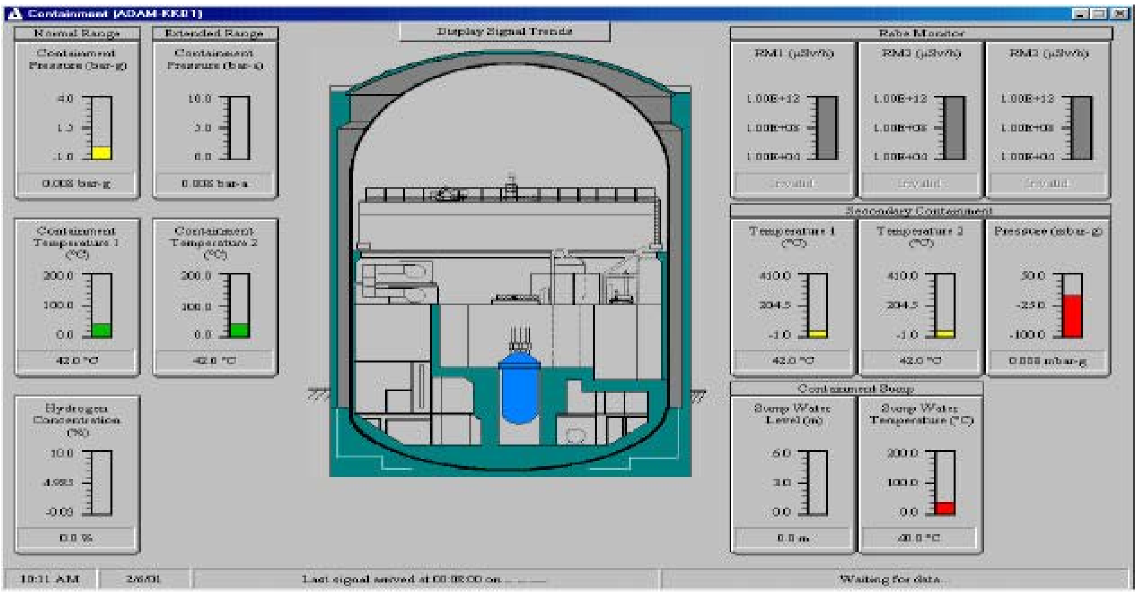

Figure 11 Example of ADAM Graphical User Interface [26]

ADAM is designed to operate in two modes referring to two different modules:

(1) **On-Line Diagnostics mode (D)**

In this mode, the parameters available from measurements at the plant are used as input for ADAM. These values are fed to ADAM at specified time intervals (e.g., every 2 minutes) and used to assess safety margins. A number of alarms denote these safety margins in the graphical interface (e.g., margin to containment failure, margin to venting system actuation, etc.). Furthermore, the state of the reactor, the containment, and all the auxiliary buildings are monitored and a deterministic logic is the basis for the calculation of the provisioned plant states.

(2) **Accident Management and Analysis mode (A)**

In this mode, ADAM is used to simulate multiple accident scenarios and determine the potential consequences and implications of a series of Severe Accident Management (SAM) actions. Consequently, ADAM offers efficient means for training, drills, accident analysis and emergency planning. The source term assessment is also included in this mode of operation as well as the evaluation of PSA success criteria.

Another mode is included only in the ADAM version used at the Swiss nuclear power plants (Pikett Ingenieur (PI) mode). This mode provides additional graphical information on the plant using simple diagnostic criteria for the calculations.

The ADAM system is designed to provide support and meet the needs of the experts at the accident response centre (regulator) where only limited on-line information about the plant status is usually available. The philosophy of this tool is to provide valuable support for the emergency preparedness and, at the same time, avoid complicated calculations in order to have faster than real-time predictions [27].

7 Analysis

7.1 Introduction

This chapter is meant to be the core of this thesis project. The discovery from literature research, the findings from communication with experts in this field, together with the information gained from people involved in the use of the software codes of interest will be described.

The study case taken into account is the one that involves the MARS code and the Bayesian Belief Network in RASTEP. No reasoning or attempts on the linking with codes other than MARS will be analysed for this purpose.

7.2 Linking the BBN to a fast running deterministic code

7.2.1 Overview

A good understanding of the functionality of the Bayesian Belief Network (BBN) and the information that the network uses for the calculations is the basis for the evaluation of the connection with the fast running source term code. The physical quantities and, in general, the observables play a major role in directing the BBN towards the final most probable outcome.

An important issue to address is when one is supposed to start the deterministic code simulation after the BBN has been initiated and how many reading/findings it needs to be started. The data that can be extracted merely from the output of the BBN are surely not enough to initiate a code like MAAP or another commonly used Severe Accident (SA) code. This information, as it will be described in the next paragraphs, can be used as input for the simulations with the MARS software instead.

7.2.2 MARS limited input

According to CSN, MARS is designed to use less than 100 parameters as input to give ideal simulations of reality. At the same time, in the event of lack of information, the MARS' *tracker* uses a limited set of dynamic plant signals to initialize the MAAP code and then to follow the evolving plant response. The ability of creating an input model for a deterministic code, using less on-line signals than commonly needed, represents an interesting feature in MARS. Even if MARS could serve solely for this purpose that would be a very effective feature to add into RASTEP.

Early versions of MARS were provided with a very limited set of plant data (around 15 parameters) and the results in the early age of this software tool were already satisfactory according to the operators experience at CSN [25]. More features have been included in the latest version of the MARS software so that, even if no on-line data are available, one can run MARS when the user manually sets some crucial parameters.

The MARS's *tracker* gets initialized using plant data (or manually inputting data – if necessary) and tracks the evolving plant behaviour. While MARS is tracking the evolving plant response, either after automatically or manually input, the user can initiate a set of *Predictors* to perform much faster than clock time simulations of how the plant could “look” in the future.

The faster-than-real-time *Predictors* can also run "What-If" evaluations in order to extrapolate deviations from normal operation in the plant and bound the uncertainty of not having solid sets of plant data [28]. The *Predictors* do include source term information which is typically coupled to off-site radiological codes such as MAAP4-DOSE, LENA, and ARGOS etc. Therefore, the MARS software performs mechanistic thermal-hydraulic and source term evaluations. One of the big advantages of the MARS *Predictors* is that it is possible to initiate a set of calculations to explore many possible plant states [25] [28].

Once the MARS input has been created, many scenarios can be simulated referring each to a different initial plant state. The set of plant states used for the analysis is ordinarily selected by the user, who will manually choose the most interesting scenarios to analyse according to other parameters and readings from the plant.

7.3 Preparing the input for MARS

To address the challenge of connecting the Bayesian network (i.e. plant states) to the input deck of MARS, a description of the available set of BBN nodes has to be given. An example of the available information is shown in Table 4. Specifically, the table shows some of the BBN nodes for a generic Swedish BWR.

Table 4 Example of nodes in the BBN (case of a BWR)

AFW availability
Containment depressurization line closure (rupture disc)
Containment sprays
Containment long-term pressure development
Containment isolation
AFW sufficiency
Core cooling sufficient
Corium coolability in lower drywell
Ex-vessel melt coolability
Containment hydrogen combustion
Containment status
Status of venting
Containment threat
ECCS availability
Mode of fuel release
Containment rupture due to phenomena
Direct containment heating (DCH)
Recriticality
Steam explosion
Reactor building exhaust system filtration
Reactor Building mode
Need for rupture disc depressurization
Independent RHR system available
Residual Heat Removal sufficient
RPV failure mode
Status of bypass to turbine building
LDW filling system successful
Manual activation of independent RHR system
Main feedwater availability
Status of the pressure suppression function
Rupture disc function
Restart of core cooling
Availability of filtered venting system
Turbine condenser
SRV LOCA (unclosed valves)
Availability of depressurization system

As one can see, the results from the BBN refer mainly to system availabilities and plant states. Each of the rows in Table 4 corresponds to a node in the network. After the BBN infers the solutions, each state in the node will end up in a certain state of occurrence with its own probability. This information together with plant data transmitted from the control room PC or manually introduced into an input deck is what the deterministic code (MARS) will possibly use to start the simulations.

As mentioned above, MARS has to be fed with characteristic parameters from the plant. This means that an accurate selection of parameters has to be made to take into account either the importance of findings from the BBN with related probabilities, or the use of the live plant data from the control room. It is not straightforward if one has to prefer the use of predicted events (e.g., steam explosion, release from fuel) rather than signals from the measurement instruments at the plant to start the deterministic code. It is certain that the outcomes from the BBN can provide, at least, insights and support for the preparation of the deterministic code input deck. This would perfectly fit the scope of RASTEP and its usability before validation studies and tests will assess good performances in other configurations.

7.4 Interface between RASTEP and MARS

Two alternatives for connecting the fast running deterministic code to the BBN will be described in this section. An integrated use of the deterministic code within RASTEP will be showing the case in which the MARS code is fed either with data from the plant or with relevant status information predicted by the BBN. Then, an iterative use will be showing the case in which the MARS code is initially fed with plant data and information from the BBN, however the scenarios simulated by the deterministic code will subsequently be used to drive new predictions of the BBN and then iteratively fed to MARS until some criteria are fulfilled in order to reach the end of the analysis.

7.4.1 Integrated use

As shown in Figure 12, an integrated use of the fast running deterministic code MARS is proposed.

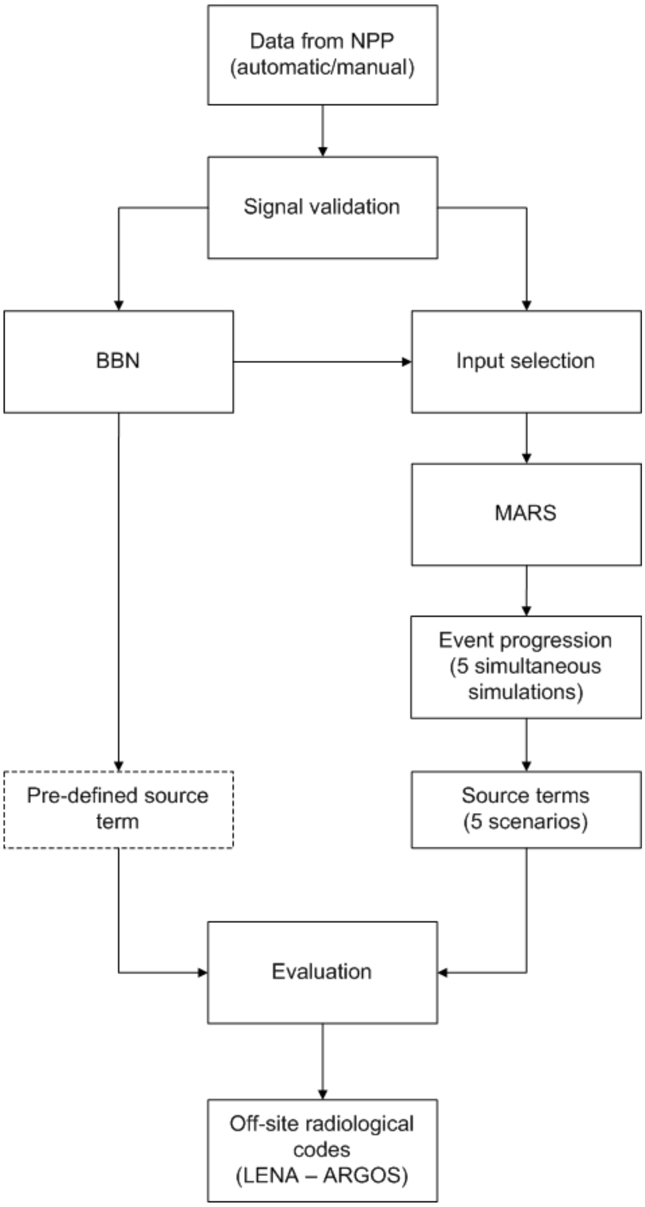

Figure 12 MARS linked to the BBN – Integrated mode

According to the block diagram, the input data for both the BBN and the deterministic code could come either directly from the nuclear power plant in an automatic way or entered manually by the user. These data are in the form of physical parameters and systems' statuses. The signals acquired must be analysed and validated in order to verify that they are between the margins for the input of the successive modules. Thus, the validation procedure has to be able to discriminate between signals which are in the range of acceptability both of the BBN input and of the MARS software. Specifically for the BBN, they need to be of the same category of signals accepted by the model currently in use.

Consequently, the input data which have been verified and validated are fed to the BBN, and the solution for the future plant status is inferred. The results from the BBN are, then, fed to the fast running deterministic code MARS. This is a crucial step in the proposed RASTEP process because the selection of the final states of the BBN has to be done carefully and in accordance with the MARS input specifications.

According to the scheme, it is in the block called *input selection* in Figure 12 where the choice between data from the plant physical parameters and from the output of the BBN (plant status and related probability) has to be made. A good strategy on how to select these data has to be studied and evaluated in order to prepare a smart input deck and drive the deterministic simulation towards credible scenarios. The ambition is to depend on plant data as less as possible, so the input selection ought to be feasible even without signals corresponding to what is inferred by the BBN. This means that, if only data for observables are available, no input selection needs to be performed since the only information needed is already available from the BBN inferred results.

In accordance with the information provided by the experts from Fauske and Associates [28] and CSN [25], the outcomes from the BBN is consistent with the MARS input definition. In fact, MARS is capable of accepting continuous parameter values, system availabilities and discrete values based on set points as input data.

As mentioned above, multiple MARS' simulations can be performed, up to a maximum number of 5, each related to 5 different plant statuses. This implies that, once the source terms for the predicted accidental scenarios are calculated, an accurate choice between the various alternatives has to be made. For this purpose, the pre-calculated source terms can be used to make comparisons and, according to the release categories defined in the BBN model, selected in order to give the best estimation.

To make the new source terms as realistic as possible the issue of indicating precisely less coarse release categories is probably necessary. This important conceptual issue will be analysed and described further in section 7.5.

7.4.2 Iterative use

In this paragraph an alternative way of connecting the BBN to the fast running deterministic code is proposed. The differences between the previous approach and this one will also be discussed. The reasoning on the way of getting information from the event progression analyses from the deterministic module and feed the BBN with it will become more important in this analysis.

Looking at Figure 13, the data from the plant are fed to a block called *signal validation*. The use of this block serves as screening for either the data coming directly from the plant or the data manually inserted by the operator/user in the BBN. In addition to what was already described for the integrated use, here the information is captured also from the event progression analysis of MARS and its predicted plant states.

Similarly to the integrated use described in 7.4.1, the data just selected and validated are fed to the BBN that will infer the solution of the most probable accidental end state. Up to 5 different plants states can be selected from the BBN outcomes which will subsequent feed the MARS code to run 5 simultaneous simulations for different accidental scenarios.

The parameters from the BBN together with the plant data are then selected to prepare the input for the deterministic code. As mentioned in the previous section, the plant data have to be accurately merged with the probabilistic outcomes in order to provide 5 different input decks which are consistent and exhaustive for the MARS input.

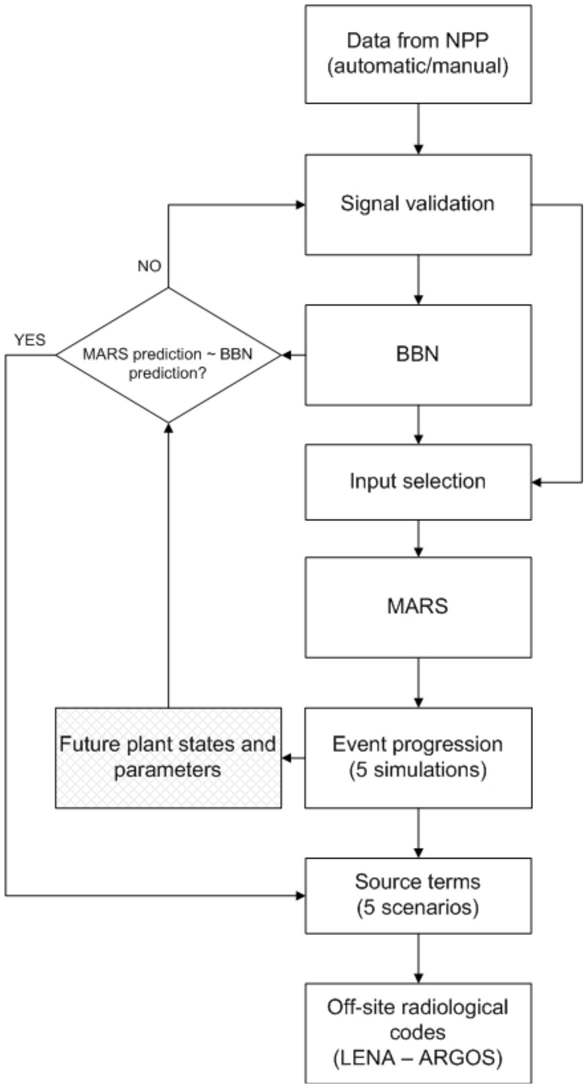

Figure 13 MARS linked to the BBN – Iterative mode

After MARS has computed the event progression simulations for the different scenarios, the predicted plant states are sent back to the *signal validation* through a *decision block*. However, the new plant states have to be extracted from the MAAP simulations and, according to CSN [25], there is still to clarify by which means the selection of data to extract has to be made. The decision on how to make this extraction and which type of information to take depends mainly on the quality of the predictions and the difference between the parameters previously selected in the BBN. Furthermore, the available parameters of the plant, at the time of the iteration, also play an important role.

The *decision block*, a diamond block in Figure 13, is where the choice to iterate the process has to be made. At this stage in the process, the meaningful data previously extracted from the deterministic analysis have to be compared with the outcomes of the BBN. If the most probable end states predicted by the probabilistic module are in accordance or, at least, comparable to those simulated by MARS, the iteration can be directly stopped. If this is the case the source term referring to the last MARS simulation, contained in the MAAP output, can be accepted and selected for the next step in the scheme. On the contrary, if the MARS predicted plant states are very different from those predicted by the BBN then the information coming from the deterministic simulation is redirected to the *signal validation* again.

In the *signal validation* block, the information is collected directly from the plant (automatic/manual) or via the *decision block*. These data are then chosen by means to define and sent again to the BBN to start the probabilistic prediction. In the section 7.5 below, the problems related to this integration mode and other questioning issues will be treated and discussed in a more detailed way.

7.5 Managing the conceptual framework

In this paragraph the approaches discussed in the sections above will be analysed further offering a more comprehensive picture of the conceptual reasoning behind the integration process. The issues concerning the connection of probabilistic methods with deterministic tools will be emphasized. It is important to remember that RASTEP is a tool that should provide insights during an early stage of an accident and, at the same time, has to remain functional throughout the accidental scenario. Consequently, the argumentation will follow taking into account this important feature.

As described for the iterative mode in 7.4, one of the first issues to address when linking the BBN to MARS regards the selection of the plant data that have to be fed both to the probabilistic predictor and the deterministic code. In this perspective, it is important to comprehend how the probabilistic prediction in the BBN is meant to drive the deterministic analysis. Nevertheless, it is important to decide which information has to be kept and what probability data are relevant in the BBN outcomes that, later on in the coupled configuration, will serve to initiate MARS.

When the deterministic code is driven by the parameters selected from the BBN end states, the results from the simulation need to be ranked in accordance with the initial predicted scenario (probabilistically inferred). This is the way in which RASTEP is supposed to address the prediction of an accidental scenario as described in 7.4.2. The deterministic results are in the form of predicted future plant states and physical parameters. The idea is to extract the values corresponding to the physical parameters at specific times of the simulated event progression. Again, in order to perfectly couple the probabilistic prediction with the deterministic one, these values have to match certain set points. This is something that only during future developments of the tool could be addressed with precision.

The approach to apply when it comes to select parameters and rank the output of the deterministic code is still a big challenge. However, it is normal practice when doing MAAP calculations to indicate time intervals for the simulation. The time factor in the deterministic simulations plays an important role both for the analysis of the evolution of single parameters and for the time intervals in which the source terms have to be evaluated. This appears in total accordance with the way in which the selection of the parameter should be made.

It appears to be probable that characterizing the solutions of MARS also in terms of system damage and quantity of substances in specific volumes of the containment could help the selection of parameters (see section 6.1.7). This would help when it is more complicated to rank the output of the deterministic code. Applying this approach would enable the user to make an easier choice for the comparison with the output of the BBN.

The source term data are still the most important information one should retrieve by using RASTEP. In both modes defined in 7.4, the relevant source terms come from the fast running deterministic code.

Another problem arising from this approach is the matching between the source terms extracted from on-the-fly deterministic simulations and those currently available in RASTEP. A way to solve this problem is to identify less coarse release categories both in the BBN model and its pre-calculated source terms and in the output of the deterministic code. This will facilitate the comparisons of the release data for both the codes avoiding wrong interpretations.

Moreover, the predictions of the BBN should be used as main pillar in the structure of the entire analysis. If there was no probabilistic usage in RASTEP such as the BBN inferring the solutions, this would merely result in using MARS as a “standalone” code. The power of the BBN model has to be kept and its capabilities better understood with the ambition of creating a tool that is reliable and fast enough to give insights truly reliable during accidental events at nuclear power plants.

7.6 Release categories

This paragraph is meant to give a short description of how the release categories are currently integrated in the source terms model of RASTEP, and propose a different approach in order to have a more accurate definition of releases as final output.

The interest in enhancing the source term module of RASTEP is clearly justifiable when looking at how the source term prediction is presently carried out. The basis for the RASTEP source term module is the PSA level 2 and related MAAP calculations. Severe accident sequences have been modelled in accordance with state-of-the-art practice to assess the releases connected with such sequences. In the PSA study, release categories are defined and subsequently associated with plant end-states.

Specifically, in PSA level 2, each release category is related to a set of sequences having similar features with respect to event progression, release path, initiating events, and actions put in place to mitigate the consequences. However, the sequences selected from PSA do not necessarily refer to the same plant damage state (PDS) in the actual scenario. For instance, *immediate* and *early* failures occurring in the event of an accident are considered to be the same with respect to the associated releases. These releases are grouped, or better “binned”, together either for transients or LOCAs. That is one of the reasons why release categories are generally assessed conservatively. This is the modelling strategy which in turn gives a static behaviour to RASTEP [29].

Typically, the number of release categories considered in a level 2 PSA study is around 15-20. As one can see in Table 5, 14 release categories have been selected to address the needs and specifications of the source term analysis for a Swedish BWR. In general, the identification of 14 release categories is common practice in Sweden when performing PSA studies. All the release categories are related to specific release paths in the containment volume, the reactor building and in the turbine hall. Each of them refers to specific accidental modes of the plant. These modes are closely connected to the PSA sequence analysis.

The release categories are currently arranged in two groups: one group referring to those releases which appears during an early stage of the accident event, and another group accounting for those releases occurring in a late stage of the accident. Apart from these two main groups there is the release category related to the diffuse leakage in the containment and additional categories taking into account the reactor building and the turbine building (BWR specific). This is a brief description on the way the mapping between the BBN model (node states) and the release information from the PSA level 2 sequence analyses has been carried out until now. In Table 5 the second column (Release Category) shows an example of naming the release categories in the PSA model. It has to be clarified that the abbreviations (e.g., OTSL, OTL-, etc.) are related specifically to the level 2 PSA model developed by Scandpower, and it is not relevant for the purpose of this thesis to define the meaning of that nomenclature.

Table 5 Source terms mapping for a generic Swedish BWR

Mode	Release Category	Building
Transient late/spray	OTLS	Containment
Transient late/no spray	OTL-	Containment
Transient early/no spray	OTI-	Containment
Transient early/spray	OTES	Containment
LOCA late/spray	OLLS	Containment
LOCA late/no spray	OLL-	Containment
LOCA early/no spray	OLI-	Containment
LOCA early/spray	OLES	Containment
Diffuse leakage	L--X	Containment
LOCA filtered venting/spray	F-ES	Containment
Transient filtered venting/spray	F-ES	Containment
LOCA filtered venting/no spray	F-E-	Containment
Transient filtered venting/no spray	F-E-	Containment
Template for RB and TB source terms (gap)	BYP-GAP	Containment
Template for RB and TB source terms (melt)	BYP-	Containment
No abnormal release	-	-
Melt bypass (filtered)	BYP-RB-F	Reactor Building
Gap bypass (filtered)	BYP-GAP-RB-F	Reactor Building
No bypass to RB	-	-
Melt bypass	BYP-TB	Turbine Building
Gap bypass	BYP-GAP-TB	Turbine Building
No bypass to TB	-	-

7.6.1 A new approach for the source terms definition in the RASTEP model

The predefined source terms currently modelled in the Bayesian network are not fully accurate and timing aspects of the release are not realistic enough. In this subsection a new approach for the definition of the source terms in RASTEP will be presented.

The way the release categories are bonded to the PSA model suggests the intrinsic static behaviour of the tool. That is the reason why the current approach is most probably not the best fit for the purpose of RASTEP. The ambition, in fact, is to have source terms which are more related to the accident scenarios in terms of timing and specific volumes of the plants. As suggested in the previous sections of this report, the capabilities of the fast running deterministic code are available in a way that the information can match more detailed release data in the sub-volumes of the plant.

According to experts at CSN [28], more detailed release data in specific volumes of the containment can be assessed using MARS. This means that, if the RASTEP model is not ready yet to integrate this kind of information, adjustments in the BBN model may be necessary. The representative nodes (source terms nodes) in the BBN model are currently associated to the release categories which have been indicated and analysed during PSA level 2 studies. In order to understand how to capture the new release information given by the linked deterministic code it is necessary to consider some available techniques that may be used as a complement. It has to be mentioned since now that the approaches described in the following subsections are meant to be applied together in order to have the best advantages.

7.6.1.1 PSA sequence analysis

Again, the pre-calculated source terms containing the release information in the current model of RASTEP are based on level 2 PSA and the MAAP calculations for the relevant sequences. But the PSA model has more capabilities, and more attention can be given to those undiscovered scenarios which have not been considered yet [30].

The PSA for a nuclear power plant is able to catch all the vulnerability on specific systems and the over-all logical model is capable of representing all the critical sequences leading to unacceptable releases from the plant. The ambition is to investigate those sequences which are not necessarily leading to major releases, but that can still give information on minor releases in specific regions and volumes of the containment. The set of sequences can be extended and possibly integrate more scenarios related to those consequences which were considered of minor importance until now. The problem that the analyst will face when attempting this approach is that such sequences are not already available according to the ranking and selection normally adopted. This is an issue connected to the IDPSA framework, and in this context, many methods have been developed and are currently available to address the investigation of the missing sequences (e.g., MC methods and DDET method). Nevertheless, the PSA model offers the opportunity to make this adjustment and it can include as many sequences as desired, incorporating more specific scenarios [30].

The choice of the additional scenarios to include is strictly dependent on how RASTEP is modelled. This implies a modification in the probabilistic network (BBN) either in the case of the integrated or iterative use (section 7.4.1 and 7.4.2). A redefinition of the predefined source terms would also give the ability to do better comparisons with the source terms provided by the fast running deterministic code. Many advantages could derive from this approach even if no fast running deterministic code is linked to RASTEP.

Another important factor is the time in the release data. The source terms currently extracted from the MAAP calculations behave statically. Introducing new sequences will also help in capturing more relevant parameters and time intervals which were not considered before.

According to this description, more MAAP calculations are needed for evaluating cases previously neglected. The computational time required to run all the MAAP simulations would not impact the RASTEP usability because the deterministic calculations are always performed in advance.

7.6.1.2 MARS' source terms

According to the linking procedure introduced in section 7.2, the MARS software should be able to provide on-the-fly source terms and event progression analysis related to the most probable plant states indicated by the probabilistic module of RASTEP. At this point, the problem is about the way the source terms data have to be included into specific release categories and, later on, how to back-track this information within the linked configuration.

As a first consideration, the information contained in the MARS output has to be referred to specific volumes of the plant. These volumes have to be previously indicated and evaluated by means of other studies. As studied at CSN [25], during the validation tests of the MARS software for each Spanish NPP, such volumes and the relative source term data were extensively investigated. By doing this, more detailed information on specific regions of the containment becomes available. The problem of how to interpret this information and track it back to the BBN in RASTEP still remains.

The detailed deterministic analysis in MARS can meet the use of new release categories for very specific volumes of the plant. For instance, if one is interested in evaluating the leakage from the containment in the event of a LOCA, by restricting the analysis to a few regions close to the containment wall and indicating a release category for that specific area of the containment, could let the source terms be more accurate in many cases [25].

As mentioned above, in the case the new release categories would be introduced when MARS is linked, back-tracking this information would still be a major concern. The end-states in the BBN, in fact, are not currently modelled to match a new release grouping. This means that new techniques to interpret this deterministic information are needed. In the next chapter this topic will be analysed and advantages and disadvantages will be evaluated.

7.6.1.3 Remodelling the BBN: new nodes

This subsection is meant to briefly describe the problem arising when new release information is available and, consequently, propose some modifications in the BBN model.

As described in section 3.4, the BBN mapping captures all the relevant key plant parameters (release paths, system availabilities, etc.) making the network a solid logic model capable of directing the inferred solution towards the most probable end states.

In order to include new source terms data, the BBN model has to be reorganized and remodelled in accordance to the grouping of the release categories. The release information could come either from new sequences in the PSA (see section 7.6.1.1) or from a fast running deterministic code (see section 7.6.1.2). The nodes currently modelled in the BBN are possibly not in a sufficient number to accommodate all the information needed.

When source terms accounting for volumes in the plant not previously considered in the network are present, a new node and/or a new set of nodes should be created in the model. The end-states in the BBN need to match the release data coming from the deterministic code and, at the same time, give the correct outcome with the related probability. As described in chapter 0, the nodes related to the source terms have a probability between 0 and 1, and the prior probabilities are assigned based on the Level 2 PSA.

If MARS or any other deterministic code will be linked to RASTEP, the data extracted from the deterministic simulations have to be integrated in the model assuring that the proper information is contained in the parent nodes, all the connections are established for all the updated source terms and CPTs well described.

In the case new sequences are evaluated in the PSA studies, they have to be mapped into the BBN model and possibly specific new nodes for other systems in the plant have to be created. However, it is very probable that only the CPTs have to be redefined and no additional nodes have to be created in the network.

7.7 Lesson learned introducing MARS at CSN

This section offers an overview on the lessons learned at the Consejo de Seguridad Nuclear (CSN) during the implementation of the MARS software.

First, creating a systematic environment for using MARS at CSN was very time and cost demanding. It required around 12 years until they were able to make a full migration to MAAP in all the Spanish NPPs and developed the tools for validating the fast predictions obtained by MARS. FAI expertise resulted to be very helpful along the way. After some years from the beginning of the project all the MAAP models, specific for each plant, were created and the real-time plant data delivery system to CSN's emergency room was implemented. Unfortunately, soon after they signed the last development contract with FAI, and all the system was on the final test phase, giving already good results, the CSN decided to prioritize other projects. The main reason for the decline of this project was the lack in human resources [25].

It is worth mentioning that CSN decided, after some testing, to avoid the on-line import of parameters from the NPPs. They opted for a user-driven questioning and input model more than an automatic one. The advantages in usability were such that they preferred this way of employing the tool.

8 Discussion and continued research

8.1 Review of the approaches

The review activity together with the linking strategy performed within this thesis work has generated a set of criteria associated with the definition of a dynamic source term module in RASTEP. The new source term module should be:

1. *Realistic* – i.e. conservatism needs to be reduced
2. *Accurate* – i.e. live plant data as basis for the predictions
3. *Adaptable* – i.e. use probabilistic predictions as basis to perform new calculations

The Swedish Radiation Safety Authority (SSM) or the operators at the nuclear facilities are meant to be the final user of RASTEP. The SSM's emergency preparedness organization has to fulfil its responsibilities during a nuclear accident or other relevant events. Regulations specify that the operators of the nuclear power plants shall deliver a source term to SSM during an early stage of the accident. SSM is also responsible for the assessment of the source term in an independent way. Subsequently, this source term must be used for the calculation of the radiological consequences [31].

The use of RASTEP in its current configuration (i.e., without any integration with a fast running deterministic code) has demonstrated that the predicted most probable plant statuses are usually indicating quite accurately the future accidental scenarios. This is mainly due to the strength of the PSA modelling, the accurate mapping of the key plant parameters, and the systems interrelations modelled in the BBN. However, the issue of having predefined source terms make the current approach not realistic enough. In particular, the overall logic of the tool is still based on static plant models which constitute the foundation for the calculations. The first criterion in the list above is, then, not fulfilled by using the actual tool if no modifications are made in the model.

On the other hand, using a fast running deterministic code such as MARS could provide great advantages in performing on-the-fly calculations. Furthermore, the same live plant data necessary for the probabilistic module can be used with this software. Furthermore, the ability of MARS to perform what-if-analyses (e.g., when a major loss of signals from the control room occurs) is also highly attractive.

If RASTEP will be linked with MARS, the same end states currently resulting from the probabilistic inference appear to be a good fit for the input of a deterministic code. This demonstrates a good adaptability of the software to the actual model. Even if the end states of the BBN are grouped too coarsely to be used as input parameters, it is still remarkable that a code like MARS can adapt to use only those data and possibly give good results [28]. However, the information that can be captured from the BBN is not necessarily the final states of the predictions. More valuable information can be extracted from the hidden nodes within the probabilistic model giving better understanding of the accidental progression in the deterministic simulation.

As described in section 7.3, understanding which approach is favourable for generating the input for MARS is still a big challenge. The easiest way would surely be feeding the deterministic software directly with live plant data, but there is no transmission of the plant parameters outside the nuclear facility at the moment. Thorough assessments of the situation and analyses for predicting the accident progression are obviously a need for the Emergency Response Centre at SSM. In fact, SSM and the nuclear power plants' operators are developing a system for electronic transmission of data from the NPPs to the Emergency Response Centre, so that independent assessments can be done outside the Swedish nuclear facilities [31].

Moreover, the use of the BBN to infer the most likely plant state is a very powerful feature and, even in the case systems for live plant data transmission are implemented, it could still be very beneficial to use the probabilistic module to feed MARS. The boundary conditions related to the accidental scenario could be assessed rapidly in an early phase of the accident.

When it comes to addressing the feasibility of the implementation of MARS in the RASTEP tool, it can be seen that MARS has the considerable advantage of being based on MAAP models. MAAP models for all Swedish nuclear power plants already exist and, therefore, a large body of the modelling competence is available [32]. The developers of MAAP (FAI) visit the Swedish nuclear sites on a regular basis so that, if needed, feedback on the MARS software would be available. In contrast, other deterministic tools such as ADAM would need that plant specific models have to be created from scratch.

As described in the introduction section of this report, licensing agreements presume no evaluation purposes for MARS or ADAM. The modelling details were not available for both of them. The good relations of Scandpower and the Swedish nuclear facilities with FAI and CSN gave the opportunity for personal communication with James C. Raines (manager at FAI) and with José Ramón Alonso (Nuclear Installations coordinator at CSN).

Similarly to the features introduced in QPRO (section 3.9), a method for adjusting the available source terms stored in the spread sheet (from MAAP calculations) is probably the most easily accessible approach at the moment. Although, the suggestion is that Scandpower AB should start addressing this modification in order to verify the real potential. It is clear from now, according to the experience of GRS [17], that is not a method that can turn RASTEP into a totally accurate tool and the conservatism in the pre-calculated source terms is not avoidable putting only some effort in Excel programming.

New sequences can be discovered and more deterministic simulations can be run on MAAP. To tackle the conservatism in the predefined source terms one would also need to investigate the MAAP parameter files and re-perform simulations by refining some of the parameters related to specific systems in the plant. This would help in evaluating how the new deterministic results differ when changing the input more drastically. This task is also interesting when considering linking RASTEP to MARS, since much importance has to be given to the preparation of the input for MARS/MAAP calculations. However, reviewing MAAP parameter files and change them would require a big effort.

As a result, MARS seem to be the better choice especially due to the features introduced in the latest years in this tool. Again, the fact that the MAAP models are used in all Swedish NPPs also speaks in favour of MARS.

8.2 Questions to be answered

Two ways of using a fast running deterministic code (i.e., MARS) with RASTEP were indicated and analysed:

- Integrated use
- Iterative use

In both of them, the BBN will be giving insights on the plant status during an early stage of the accident and the deterministic code will be functioning better, later on, when more reliable information is available.

In the case of iterative use the strength of the BBN will be used more distinctly. In fact, the use of the probabilistic prediction will be important when the progression of the accidental scenario resulting from MARS is assessed and a cross comparison of plant parameters has to be made (section 7.4.2).

The iterative use seems to be more adequate and functional than the one where no iteration is included. The possibility to verify that the on-the-fly predictions given by the deterministic code are in line with those of the BBN represents an advantage in order to assess that the simulations follow the real accidental scenarios. This approach, of course, brings many difficulties with it.

As mentioned previously in this report, up to 5 different source terms could be calculated in MARS, enabling the user to manually choose the best fit according to the predicted status of the plant from the BBN. The number of 5 source terms is meant to be more as an indication of the capabilities of MARS than to depict the best way in which it should be used. In fact, it looks more reasonable to select only the most probable scenario from the probabilistic module and then run the deterministic simulation according to that information.

If one is interested only in adjusting the pre-calculated source terms, before attempting the actual implementation of deterministic code into RASTEP, studies should be carried out to determine:

- What characteristic times the user should investigate in the MAAP output table?
- How to perform the required Microsoft Excel programming tasks for this purpose?

In view of the above, the fast running deterministic code MARS should be acquired and tests should be performed. Moreover, some key questions have to be necessarily answered during the preparation of such a coupled system:

- At what stage in the accident scenario should the MARS' source term be considered reliable?
- Will it still be useful to use the probabilistic previsions in the BBN if data acquired directly from the plant are used as input for MARS?
- How to combine the information (live plant data + BBN plant status) in order to generate a reliable source term?
- What are the most important parameters in the BBN predictions to be used in MARS?
- How to technically redefine the release categories definition both in the PSA and BBN model?
- How many source terms are needed in order to have realistic previsions?
- Is the automatic selection of the source term the best way to retrieve the results?

- What are the relevant accidental sequences to be considered if a benchmarking of the coupled system has to be performed?
- How to technically link the MARS software with RASTEP?
- Does it really make sense to have more detailed and realistic source term data in all situations?
- How to determine set points for the deterministic code to start its calculations (before/after the BBN suggests a release path)?
- Can the information from MARS totally replace the predefined source terms or not?
- Is it reasonable to consider a “standalone” use of MARS? (CSN experience)

Moreover, the mathematical framework as well as the software infrastructure necessary for execution of the linking needs to be specified and tested. A new Master Thesis project started at Scandpower will address important mathematical issues of the BBN modelling.

8.3 Continued research

This section gives a brief example of developments within the field of Integrated Deterministic-Probabilistic Safety Assessment (IDPSA). Only one method, more feasible in the short term, will be described. It is worth mentioning from now that other innovative approaches are nowadays available for implementations, and more literature on the topic has become achievable.

IDPSA is a family of methods meant to support risk informed decision making. The focus of IDPSA is that safety must be based on the match of deterministic (consequences) and probabilistic (frequency) considerations. In order to address the mutual interactions between stochastic events and deterministic response of nuclear power plants, relevant applications have to be indicated.

Dynamic Bayesian Network (DNB) techniques like the one described in [33] can support in modifying the BBN model accurately and improving the RASTEP usability. This is true only if it is restricted to the analysis of sub-networks in the BBN. The introduction of "surrogate" deterministic calculations and the use of algorithms to define timing in the accident progressions seem to be of great value in order to give dynamics to the network.

A Dynamic Bayesian Network risk model could be useful for monitoring the risk of the reactor in real-time, due to its ability to update both the conditional probability tables and the parent node probabilities with real data. All the control room instrumentation readings could be fed into the network in order to continuously update reactor risk factors [33].

Many different methods were studied and implemented during the last two/three decades to render the fully static PSA analyses into fully dynamic by the use of proved algorithms (e.g., MCDET, EDS, etc.). Unless these methods look very promising and they are actually employed for very specific analyses, the industry continues to avoid them [19].

A feasibility study (performed for Beznau NPP), which interfaces a belief network for scenario selection with a fast running deterministic code, has been carried out in parallel during the

STERPS project. The aim of this work was to model accident progression and to calculate source terms in real-time [33].

9 Conclusions

The results of the thesis show that there is one feasible way to enhance the source term module in RASTEP. This is the linking with a deterministic code capable of simulating the dynamic response of a nuclear power plant in the event of an accident. MARS seems to be the best choice as a fast running deterministic code to link with RASTEP.

Other methods, such as adjusting the existing source terms based on accident progression and the use of new approaches within the IDPSA, put more difficult challenges and need to be further investigated in order to be implemented in the RASTEP tool. The complexity associated with the implementation of these innovative tools makes this alternative less suitable.

An investigation on how to address the actual linking should be performed as soon as possible to evaluate the real possibilities. Ideas from advanced new methods might be useful in terms of complementing the linking procedure.

Due to the fact that, MARS is based on MAAP and all Swedish NPPs already use MAAP models for deterministic analyses, makes this approach even more realistic. The demand in terms of costs and time for the implementation of such a system is still unknown. An evaluation of organisational costs appears reasonable before starting with the software development. According to the CSN experience in Spain, more resources were dedicated in creating the MAAP models at the nuclear sites rather than for the validation and tests of the MARS system itself.

The timing in operator actions during an accident scenario has been proven to be a very important aspect. In many cases the decisions can be made with some delays due to slow event progressions, but, this is not always the case. Sometimes the operator actions, meant to mitigate consequences, can actually lead to consequences totally unexpected. That is one of the reasons why the procedures (EOP) adopted in such situations is not always the best way of dealing with accidents. Prediction tools like RASTEP can be a very efficient way of integrating EOPs and SAMGs with on-line event progression analysis.

The evidence is that after the Fukushima accident the nuclear safety community and regulators (e.g., OECD-NES) are promoting different initiatives to take into account the lessons learnt from that particular accident. The committees concerned with this topic (mainly CRPPH and CSNI) have recently decided to initiate a project to interchange information about tools like RASTEP. The aim is to gain insights about the existing real capacities to make accurate fast estimations of the emissions to the environment in case of a severe accident.

References

[1] Internation Atomic Energy Agency, "Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level 2)," Vienna, Safety Series No. 50-P-8, 1995.

[2] P. Alfheim, "Definition and evaluation of a dynamic source term module for use within RASTEP: A feasibility study," Uppsala Universitet, MSc Thesis 2012.

[3] International Atomic Energy Agency, "Deterministic Safety Analysis For Nuclear Power Plants," Vienna, 2009.

[4] International Atomic Energy Agency, "Applications of Probabilistic Safety Assessment (PSA) For Nuclear Power Plants," Vienna, 2001.

[5] E. Zio, *An Introduction to the Basics of Reliability and Risk Analysis*, Reliability and Engineering Statistics ed.: World Scientific Publishing Co., 2006, vol. 13.

[6] D. Kececioglu, *Reliability engineering handbook*.: Prentice-Hall, 1991, vol. 2.

[7] International Atomic Energy Agency, "Development And Application of Level 1 Probabilistic Safety Assessment For Nuclear Power Plants," Vienna, 2012.

[8] P. Trucco, E. Cagno, F. Ruggeri, et al., "A Bayesian Belief Network modelling of organisational factors in risk analysis: A case study in maritime transportation," *Reliability Engineering and System Safety*, vol. 93, pp. 823-834, 2008.

[9] P. Govaerts, "Using Plant Status Data for Off-Site Emergency Management: Technical and Political Aspects," *Radiation Protection Dosimetry*, vol. 109, no. 1-2, pp. 31-33, 2004.

[10] Strålsäkerhetsmyndigheten. (2011, June) Myndighetens arbete vid en kärnteknisk olycka eller allvarlig händelse. [Online]. http://www.stralsakerhetsmyndigheten.se/start/Karnkraft/Beredskap/Myndighetens-arbete-vid-en-karnteknisk-olycka-eller-allvarlig-handelse/

[11] Krisinformation.se. (2011, July) Samhällets beredskap. [Online]. http://www.krisinformation.se/web/Pages/Page____31318.aspx

[12] International Atomic Energy Agency, "Severe Accident Management Programmes for Nuclear Power Plants," Vienna, Safety Standard Series 2009.

[13] International Atomic Energy Agency, "Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants," Vienna, Safety Standard Series 2000.

[14] M. Knochenhauer, C. Lundtofte, L. Spanier, et al., "RASTEP Phase 1 - BBNmodel for Oskarshamn 3 and basis for generic BWR model," Scandpower AB, 2011.

[15] International Atomic Energy Agency, "Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level 2)," Vienna, Safety Series No. 50-P-8, 1995.

[16] Strålsäkerhetsmyndigheten, "APRI-6 Accident Phenomena of Risk Importance," Swedish Radiation Safety Authority, 2009.

[17] H. Löffler, F. Cester, M. Sonnenkalb, W. Klein-Hessling, et al., "Ressortforschungsberichte zur kerntechnischen Sicherheit und zum Strahlenschutz," Köln, 2009.

[18] Y. Adolfsson, J. E. Holmberg, G. Hultqvist, et al., "Proceedings of the deterministic/probabilistic safety analysis workshop," , Espoo, 2011.

[19] S. Swaminathan and C. Smidts, "The Event Sequence Diagram framework for dynamic Probabilistic Risk Assessment," *Reliability Engineering and System Safety*, vol. 63, no. 1, pp. 73-90,

1999.

[20] EPRI, "MAAP4 Applications Guidance," Palo Alto, CA, 1013500, 2006.

[21] Sandia National Laboratories, "MELCOR Computer Code Manuals - Vol. 3: Demonstration Problems," 2001.

[22] L. Nilsson, "Development of an Input Model to MELCOR 1.8.5 for Oskarshamn 3 BWR," Swedish Nuclear Power Inspectorate (SKI), Research 2006.

[23] J. Raines, R. Hammersley, R. Henry, et al., "MARS - An accident management tool," in *Specialist meeting on operator aids for severe accidents management and training*, Halden, 1993.

[24] J. R. Alonso, S. Aleza, M. Alonso, et al., "Consejo de Seguridad Nuclear use and experience with the MARS software," in *The 11th International Topical Meeting on Nuclear Reactor Thermal-Hydraulics (NURETH-11)*, Avignon, 2005.

[25] J. R. Alonso, *Nuclear Installations Coordinator, Consejo de Seguridad Nuclear (CSN), Personal communication*, October-December, 2012.

[26] M. Khatib-Rahbar, M. Zavisca, H. Esmaili, et al., "Accident Diagnostic, Analysis and Management (ADAM) System Applications to Severe Accident Management," in *OECD Workshop on operator training for Severe Accident Management and Instrumentation Capabilities during Severe Accidents*, Lyon, 12-14 March 2001.

[27] M. Zavisca, M. Khatib-Rahbar, and H. Esmaili, "ADAM: An Accident Diagnostic, Analysis and Management System - Application to Severe Accident Simulation and Management," in *10th International Conference on Nuclear Engineering*, Arlington, VA, USA, 2002.

[28] J. Raines, *Manager, Fauske & Associates LLC, Personal communication*, October-December, 2012.

[29] M. Knochenhauer and V. S. Hedtjärn, "Using Bayesian Belief Network (BBN) Modelling For Rapid Source Term Prediction – RASTEP Phase 1," Scandpower AB, Internal report 2012.

[30] P. Krčál, *Software Developer,Scandpower AB, Personal communication*, January 2013.

[31] Swedish Radiation Safety Authority, "Swedish action plan for nuclear power plants - Response to ENSREG's request within the European Stress Tests," 2012.

[32] A. Enerholm, "Principal consultant, Scandpower AB, Personal communication," January, 2013.

[33] A. Varuttamaseni, "Bayesian network representing system dynamics in risk analysis of nuclear systems," Ann Arbor, 2011.

[34] V. Mihajlovic and M. Petkovic, "Dynamic Bayesian networks: A state of the art," Twente, 2001.

[35] Charles River Analytics, Inc., "About Bayesian Belief Networks," Cambridge, Manual 2008.

[36] G. Heo et al., "Advisory system for the diagnosis of lost electric output in nuclear power plants," *Expert Systems with Applications*, vol. 29, pp. 747-756, 2005.

[37] Norsys Software Corp. (2013) Tutorial on Bayesia Networks with Netica. [Online]. http://www.norsys.com/tutorials/netica/secA/tut_A1.htm

[38] S. L. Lauritzen and D. J. Spiegelhalter, "Local Computations with Probabilities on Graphical Structures and Their Application to Expert Systems," *Journal of the Royal Statistical Society*, vol. 50, no. 2, pp. 157-224, 1988.

[39] Norsys Software Corp. (2013) Tutorial on Bayesian Networks with Netica. [Online]. http://www.norsys.com/tutorials/netica/secA/tut_A1.htm

APPENDIX A - An Introduction to Bayesian Belief Networks

A.1 Overview

Bayesian belief networks (BBNs) are an established method to represent problems involving uncertain relations among random variables [34]. They are tools used for modelling causes and effects for many domains and applications. Furthermore, they are able to capture the probabilistic relationships between variables and the historical information related to these relationships.

Bayesian belief networks are extremely effective in situations where some information is available and incoming data are partially unavailable and/or the level of uncertainty is high. An intuitive graphical interface defines the representation of cause and effect mechanisms and likelihoods in the network. For those domains where expert judgement is crucial, BBNs help in avoiding ineffective reasoning. Because of these capabilities, BBNs are being increasingly used especially where automated reasoning is needed.

Again, it is important to denote that Bayesian belief networks are not dependent on the knowledge of the historical information or current data. In fact, they are capable of producing very convincing results even when the data stored in the conditional probability tables and the current information is inexact. This is a great advantage in those situations where it is difficult or impossible to provide specific estimates. These features and its robustness make Bayesian networks a valuable alternative to other AI formalisms [35].

Each of the variables is represented by nodes in a Bayesian network and to each node correspond a state. The causal relationships between such nodes are represented by edges drawn between the nodes. These edges are denoted by arrows indicating the direction of influence from one node to another. In Figure 14 a simple Bayesian network is shown.

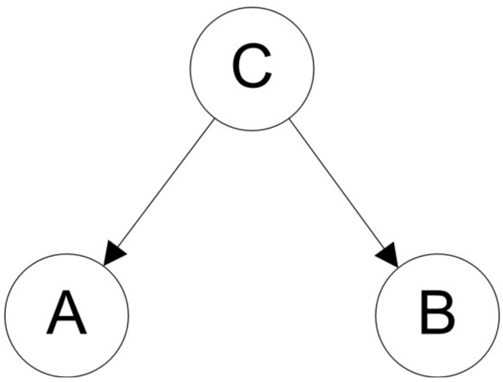

Figure 14 Simple Bayesian Network

In a Bayesian network, a scalar variable refers to each node, which may be discrete, continuous or propositional (true/false). Moreover, all the nodes in the network are associated with probability distribution functions. Every node also has a conditional probability table (CPT) associated with it. Conditional probabilities represent likelihoods based on prior information or past experience. Namely, nodes that are connected are *parents* and *children*. In Figure 14 one can see that the nodes B and C are children of A. In this particular case, the edges between A and B denote that:

- C causes A, or
- C partially causes or predisposes A, or
- C is an imperfect observation of A, or
- C and A are functionally related, or
- C and A are statistically correlated

According to probability theory:

$$0 \leq P(A) \leq 1$$

$P(A) = 1$ if and only if A is certain.
If A and B are mutually exclusive ($P(A \cap B) = 0$), then $P(A \cup B) = P(A) + P(B)$

And the fundamental rule of probability calculus:

$P(A, B) = P(A|B)P(B)$ Equation 1

In equation 1, $P(A, B)$ is the joint probability of the event $A \cap B$.

The Bayes' rule is defined as:

$P(A|B) = \frac{P(B|A)P(A)}{P(B)}$ Equation 2

Where the posterior probability ($P(A|B)$) is computed given the prior probability $P(A)$ and the likelihood ($P(B|A)$) that B will be realized if A is true.

In the example shown in Figure 14, the node A represents the hypothesis and the node B represents the evidence. If the node A and B are conditionally independent between each other one can determine that:

$P(A|C, B) = P(A|C)$ Equation 3

The knowledge base using a Bayesian network is made through both qualitative modelling and quantitative modelling. Qualitative modelling identifies cause and consequence relations. Instead, quantitative modelling allocates the necessary probabilities [36]. In addition, the graphical representation offers an easy way of describing the joint probability distribution for the variables.

A.2 Example of Bayesian Network

There are several different modelling tools to create Bayesian network models, both open source and proprietary. Netica (developed by Norsys Software Corp.) is currently used for the RASTEP model. In this section an example of the modelling in Netica will be presented [37].

Figure 15 shows a simple representation of the Bayesian network for a medical diagnosis for chest pathologies. This network is also known as "Asia", and is a popular example for introducing users to the BBN modelling in Netica [37] [38]. It is a simple network that could be useful to diagnose patients arriving at a clinic. Each node in the network corresponds to some conditions of the patient. In order to diagnose a patient, values are entered for those nodes where the information is available. The software then calculates the probabilities for all the other nodes considering the relationships between them.

The two top nodes represent the predispositions which in turn influence the likelihood of the diseases. The diseases are shown in the second row and the symptoms are in the bottom row. The edges in the network correspond to causation. This is a very common structure for networks when the purpose is to diagnose diseases or other *failure states*. Usually, the predispositions are in the top nodes, linked to nodes representing internal conditions and, finally, to failure states. Many layers of nodes for the representation of internal conditions are often present in the network of this type.

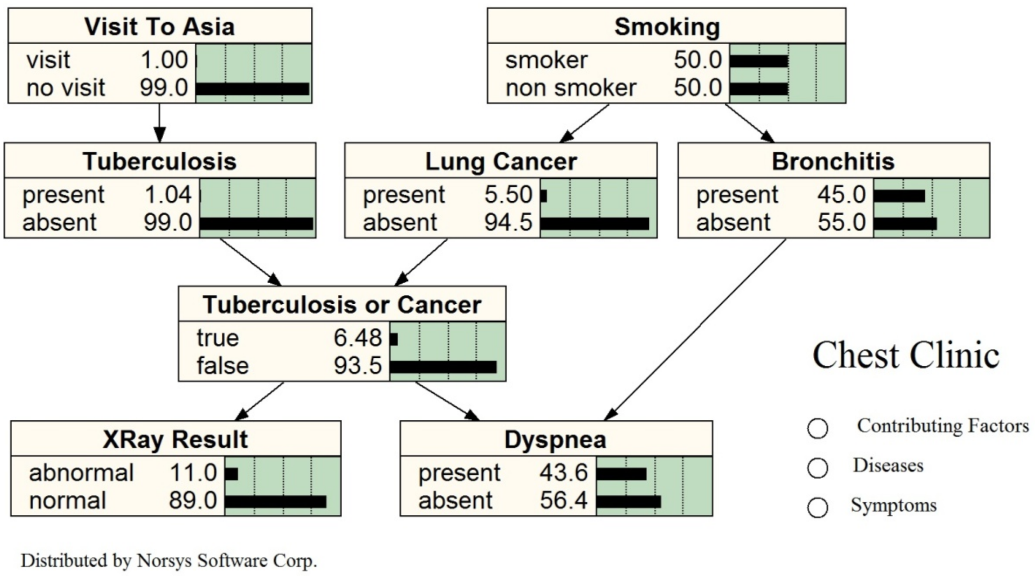

Figure 15 Example of BBN for a medical diagnosis

Conditional probability tables (CPTs) summarize the information about the states for each node of the network. Default probabilities contained in the CPTs can be based either on statistics or on the status of the parent nodes. For example, the probability of *Lung Cancer* depends on the verification that the patient is a smoker. Table 6 shows the conditional probability table for the node *Lung Cancer*.

Table 6 Conditional probability table

Parent node	Child node	
Smoking	Lung Cancer	No Lung Cancer
Smoker	10%	90%
Non Smoker	1%	99%

As shown in Figure 15, the probabilities are displayed as percentages in the network nodes. Initially, when no observation is available, the probabilities are calculated considering only prior beliefs contained in the nodes.

When more information is available, the findings are entered in network for each of the observable nodes. Figure 16 shows the specific case when the patient is not a smoker, has not visited Asia and whose medical examinations shows normal X-ray results with suspected presence of Dyspnea. Based on this information, after the beliefs are updated in the network, a high probability of bronchitis prevails.

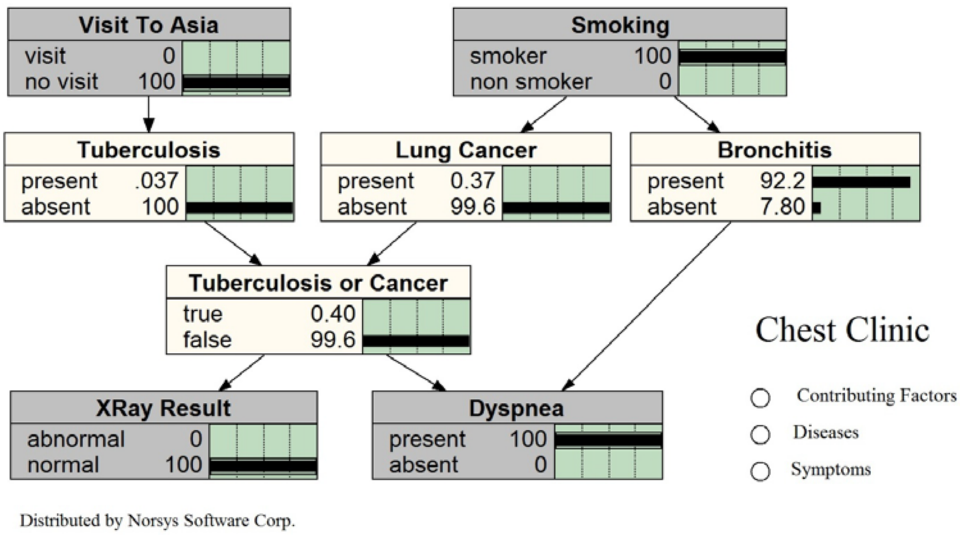

Figure 16 Example of BBN for a medical diagnosis – with observations

APPENDIX B - Review of the software codes

In this appendix a summary of the codes mentioned for the purpose of implementation with RASTEP is presented.

RASTEP (Rapid Source Term Prediction) is a code not yet available which Scandpower AB is developing and will replace the user interface used so far. The current user interface is called SPRINT (System for the Probabilistic Inference of Nuclear Power Plant Transients) and was developed within the STERPS EU project. Together with the SPRINT interface, the Bayesian Belief Network is modelled in software called NETICA (Norsys Software Corp.).

The source terms which are used right now in SPRINT are extracted from MAAP calculations. MAAP (Modular Accident Analysis Program) is an integrated computer code developed by FAI that simulates the response of light water and heavy water nuclear power plants.

The purpose of this thesis was to link a deterministic code to the Bayesian Network in order to give more reliable and realistic source term information and provide suggestions for implementation in the future software RASTEP. The deterministic code chosen in this work is MARS (Modular Accident Response System). Such software is developed by FAI and is already being used at CSN (Spain) within their emergency system.